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Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1984

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PREFACE

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

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

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

A detailed explanation of the entries precedes each index.

The bibliographic elements of the main citations are the following:

Staff Report

NUREG-0508: 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).

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 NRC-GPO Sales Office or from the National Technical Information Service, Springfield, Virginia 22161. To purchase documents from the NRC-GPO Sales Office send a check or money order, payable to the Superintendent of Documents, to the following address:

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NRC Report Codes

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

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

All these report codes are controlled and assigned by the staff of the Publishing and Translations Section of the NRC Division of Technical Information and Document Control.

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, and NUREG/CR-XXXX is an NRC contractor-prepared report. The bibliographic information (see Preface for details) is followed by a brief abstract of the report.

NUREG-0020 V07 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of October 31, 1983. (Grey Book) * Division of Budget & Analysis. January 1984. 408pp. 8402060291. 22105:267.

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 V07 N12: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of November 30, 1983. (Grey Book) * Division of Budget & Analysis. February 1984. 379pp. 8402210078. 22320:313.

See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N01: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of December 31, 1983. (Grey Book) * Division of Budget & Analysis. February 1984. 396pp. 8403150058. 22647:001.

See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N02: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of January 31, 1984. (Grey Book) * Division of Budget & Analysis. March 1984. 22pp. 8404160205. 24063:341.

See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of February 29, 1984. (Grey Book) * Division of Budget & Analysis. April 1984. 386pp. 8405220049. 24563:007.

See NUREG-0020, V07, N11 abstract.

1 NUREG-0020 V08 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of March 31, 1984. (Grey Book) * Division of Budget & Analysis. May 1984. 407pp. 8406120532. 24916:063.
See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of April 30, 1984. (Grey Book) * Division of Budget & Analysis. June 1984. 372pp. 8407180022. 25653:001.
See NUREG-0020, V08, N03 abstract.

NUREG-0020 V08 N06: LICENSED OPERATINC REACTORS STATUS SUMMARY REPT. Data As Of May 31, 1984. (Grey Book) * Division of Budget & Analysis. July 1984. 394pp. 8408160191. 26123:001.
See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N07: LICENSED OPERATINC REACTORS STATUS SUMMARY REPORT. Data As Of June 30, 1984. (Grey Book) * Division of Budget & Analysis. August 1984. 400pp. 8409200282. 26605:001.
See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N08: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Date As Of July 31, 1984. (Grey Book) * Division of Budget & Analysis. September 1984. 410pp. 8410180141. 27181:143.
See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N09: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of August 31, 1983. (Grey Book) * Division of Budget & Analysis. October 1984. 416pp. 8411200011. 27587:001.
See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N10: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of September 30, 1984. (Grey Book) * Division of Budget & Analysis. November 1984. 412pp. 8412170449. 27971:001.
See NUREG-0020, V07, N11 abstract.

NUREG-0020 V08 N11: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As of October 31, 1984. (Grey Book) * Division of Budget & Analysis. December 1984. 424pp. 8501030274. 28193:001.
See NUREG-0020, V07, N11 abstract.

NUREG-0040 V07 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, October 1983 - December 1983. (White Book) * Region 4, Office of Director. January 1984. 190pp. 8402170320. 22303:001.
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 1983 through December 1983. Also included in this issue are the results of certain

inspections performed prior to October 1983 that were not included in previous issues of NUREG-0040.

NUREG-0040 V08 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, January 1984 - March 1984. (White Book) * Region 4, Office of Director. April 1984. 309pp. 8405020039. 24297:126.

See NUREG-0040, V07, N04 abstract.

NUREG-0040 V08 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, April-June 1984. (White Book) * Division of QA, Safeguards & Inspection Programs (Post 830103). July 1984. 433pp. 8408130186. 26037:001.

See NUREG-0040, V07, N04 abstract.

NUREG-0040 V08 N03: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report, July-September 1984. (White Book) * Division of QA, Safeguards & Inspection Programs (Post 830103). October 1984. 182pp. 8411080453. 27404:194.

See NUREG-0040, V07, N04 abstract.

NUREG-0090 V06 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. July-September 1983. * Director's Office. April 1984. 57pp. 8405220091. 24601:296.

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 or safety and requires a quarterly report of such events to be made to Congress. This report covers the period July 1 to September 30, 1983.

During the report period, there were three abnormal occurrences at the nuclear power plants licensed by the NRC to operate. The first involved large diameter pipe cracking in boiling water reactors; the second involved an uncontrolled leakage of reactor coolant outside primary containment; and the third involved improper control rod manipulations. There were seven abnormal occurrences for the other NRC licensees. Three involved overexposures; two involved medical misadministrations; one involved widespread radiological contamination; and one involved willful violation of license and a material false statement to the NRC. There were no abnormal occurrences reported by the Agreement States.

The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V06 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. October-December 1983. * Director's Office. May 1984. 29pp. 8406190041. 25025:220.

See NUREG-0090, V06, N03 abstract.

NUREG-0090 V07 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. January-March 1984. * Director's Office. July 1984. 52pp. 8408290353. 26299:305.

See NUREG-0090, V06, N03 abstract.

NUREG-0090 V07 N02: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. April-June 1984. * Director's Office. October 1984. 48pp. 8411210153. 27618:070.

See NUREG-0090, V06, N03 abstract.

NUREG-0304 V08 N04: REGULATORY AND TECHNICAL REPORT. Annual Compilation For 1983. * Division of Technical Information & Document Control. February 1984. 569pp. 8403070401. 22554:217.

This compilation lists all NRC regulatory and technical reports published under the NUREG series during 1983.

NUREG-0304 V09 N01: REGULATORY AND TECHNICAL REPORTS. Compilation For First Quarter 1984. * Division of Technical Information & Document Control. May 1984. 146pp. 8407110023. 25544:227.

See NUREG-0304, V08, N04 abstract.

NUREG-0304 V09 N02: REGULATORY AND TECHNICAL REPORTS. Compilation For Second Quarter 1984. * Division of Technical Information & Document Control. August 1984. 180pp. 8408300279. 26331:346.

See NUREG-0304, V08, N04 abstract.

NUREG-0304 V09 N03: REGULATORY AND TECHNICAL REPORTS. Compilation For Third Quarter 1984. * Division of Technical Information & Document Control. November 1984. 149pp. 8412050837. 27798:247.

See NUREG-0304, V08, N04 abstract.

NUREG-0325 R06: U. S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS. BARRY, L. W. Office of Resource Management, Director. January 1984. 63pp. 8401310461. 22042:349.

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

NUREG-0383 V01 R07: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Summary Report Of NRC Approved Packages. * Division of Fuel Cycle & Material Safety. November 1984. 446pp. 8412040283. 27774:030.

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 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 R07: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Certificates Of Compliance. * Division of Fuel Cycle & Material Safety. November 1984. 641pp. 8412040271. 27775:300.

See NUREG-0383,V01,R07 abstract.

NUREG-0383 V03 R04: DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES. Summary Report Of NRC Approved Quality Assurance Programs For Radioactive Material Packages. * Division of Fuel Cycle & Material Safety. November 1984. 129pp. 8412040276. 27758:165.

See NUREG-0383,V01,R07 abstract.

NUREG-0386 D03: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST. * Office of the Executive Legal Director. * Aspen Systems, Inc. July 1984. 569pp. 8408230125. 26232:033.

This third edition of the NRC Staff Practice and Procedure Digest, prepared by Aspen Systems Corporation under contract with the NRC, contains digests of board decisions issued during the period from July 1, 1972 to December 31, 1981 interpreting the NRC's rules of practice in 10 CFR Part 2. This third edition replaces the second edition and its three supplements and contains additional material on decisions issued through the end of 1981. This third edition also contains multiple indices not included in previous editions of the digest.

The third edition of the digest will be supplemented periodically with updated replacement page supplements.

NUREG-0390 V07 N02: TOPICAL REPORT REVIEW STATUS. Data As of October 15,1984. (Blue Book) * Management Information Branch. November 1984. 212pp. 8412070075. 27837:315.

The primary purpose of this document is to provide periodic progress reports of on-going topical report reviews, to identify those topical reports for which the Nuclear Regulatory Commission (NRC) staff review has been completed, and to the extent practicable, to provide NRC management with sufficient information regarding the conduct of the topical report program to permit taking whatever actions deemed necessary or appropriate.

NUREG-0420 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION,UNIT NO. 1. Docket No. 50-322. (Long Island Lighting Company) * Division of Licensing. April 1984. 36pp. 8405220021. 24556:341.

Supplement No. 5 (SSER 5) to the Safety Evaluation Report on Long Island Lighting Company's application for a license to operate the Shoreham Nuclear Power Station, Unit 1, located in Suffolk County, New York, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses several items that have been reviewed by the staff since the previous supplement was issued.

NUREG-0420 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION, UNIT NO. 1. Docket No. 50-322. (Long Island Lighting Company) * Division of Licensing. July 1984. 33pp. 8408220107. 26199:304.

Supplement No. 6 (SSER 6) to the Safety Evaluation Report on Long Island Lighting Company's application for a license to operate the Shoreham Nuclear Power Station, Unit 1, located in Suffolk County, New York, has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. This supplement addresses several items that have been reviewed by the staff since the previous supplement was issued.

NUREG-0420 S07: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION UNIT NO. 1. Docket No. 50-322. (Long Island Lighting Company) * Division of Licensing. September 1984. 161pp. 8410100158. 26904:073.

Supplement 7 (SSER 7) to the Safety Evaluation Report on Long Island Lighting Company's Unit 1, located in Suffolk County, New York, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses several items that have been reviewed by the staff since the previous supplement was issued.

NUREG-0420 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION, UNIT 1. Docket No. 50-322 (Long Island Lighting Company) * Division of Licensing. December 1984. 101pp. 8412310394. 28148:180.

Supplement 8 (SSET 8) to the Safety Evaluation Report on Long Island Lighting Company's application for a license to operate the Shoreham Nuclear Power Station, Unit 1, located in Suffolk County, New York, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses several items that have been reviewed by the staff since the previous supplement was issued.

NUREG-0430 V04 N01: LICENSED FUEL FACILITY STATUS REPORT. Inventory Difference Data. January 1983 - June 1983. (Grey Book) * Director's Office, Office of Inspection and Enforcement. March 1984. 14pp. 8403220189. 22721:097.

NRC is committed to the periodic publication for 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, plutonium, or Uranium-235.

NUREG-0430 V04 N02: LICENSED FUEL FACILITY STATUS REPORT. Inventory Difference Data, July 1983-December 1983. (Grey Book) * Director's Office, Office of Inspection and Enforcement. August 1984. 18pp. 8409170410. 26498:278.

See NUREG-0430, V04, N01 abstract.

NUREG-0485 V05 N11: SYSTEMATIC EVALUATION PROGRAM, STATUS SUMMARY REPORT. Data As Of November 30, 1983. (Buff Book) * Office of Resource Management, Director. January 1984. 75pp. 8401200362. 21872:003.

The Systematic Evaluation Program is intended to examine many safety-related aspects of 11 of the older light water reactors. This document provides the existing status of the review process including individual topic and overall completion status.

NUREG-0519 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LA SALLE COUNTY STATION, UNITS 1 AND 2. Docket Nos. 50-373 And 50-374. (Commomwealth Edison Company) * Division of Licensing. March 1984. 30pp. 8404090036. 22947:282.

Supplement No. 8 to the Safety Evaluation Report of Commonwealth Edison Company's application for a license to operate its La Salle County Station, Unit 2, located in Brookfield Township, La Salle County, Illinois, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement is to update our evaluations on Unit 2 issues identified in the previous Safety Evaluation Report and Supplements that need resolution prior to issuance of the full power operating license for Unit 2.

NUREG-0525 R08: SAFEGUARDS SUMMARY EVENT LIST (SSEL). * Licensing Policy & Programs Branch. March 1984. 72pp. 8404050493. 22915:177.

The Safeguards Summary Event List (SSEL) provides brief summaries of several hundred safeguards-related events involving nuclear material or facilities regulated by the U.S. Nuclear Regulatory Commission (NRC). Events are described under the categories of bomb-related, intrusion, missing/allegedly stolen, transportation, tampering/vandalism, arson, firearms-related, radiological sabotage and miscellaneous. The information contained in the event descriptions is derived primarily from official NRC reporting channels.

NUREG-0525 R09: SAFEGUARDS SUMMARY EVENT LIST (SSEL). * Licensing Policy & Programs Branch. June 1984. 53pp. 8407180039. 25654:272.

See NUREG-0525, R09 abstract.

NUREG-0540 V05 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30, 1983. * Division of Technical Information & Document Control. January 1984. 587pp. 8402210016. 22324:001.

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 V05 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1983. * Division of Technical Information & Document Control. March 1984. 538pp. 8403260289. 22762:001.

See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1984. * Division of Technical Information & Document Control. April 1984. 599pp. 8404250005. 24210:184.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-29, 1984. * Division of Technical Information & Document Control. April 1984. 669pp. 8405220092. 24553:001.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE March 1-31, 1984. * Division of Technical Information & Document Control. May 1984. 632pp. 8406190044. 25026:001.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1984. * Division of Technical Information & Document Control. June 1984. 644pp. 8407170553. 25628:001.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. May 1-31, 1984. * Division of Technical Information & Document Control. July 1984. 649pp. 8408130049. 26035:001.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1984. * Division of Technical Information & Document Control. August 1984. 710pp. 8409200392. 26603:001.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. July 1-31, 1984. * Division of Technical Information & Document Control. September 1984. 600pp. 8410110527. 26943:001.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N08: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. August 1-31, 1984. * Division of Technical Information & Document Control. October 1984. 637pp. 8411080549. 27402:277.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N09: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. September 1-30, 1984. * Division of Technical Information & Document Control. November 1984. 518pp. 8412200527. 28037:263.
See NUREG-0540, V05, N11 abstract.

NUREG-0540 V06 N10: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. October 1-31, 1984. * Division of Technical Information & Document Control. December 1984. 450pp. 8501080456. 28260:015.
See NUREG-0540, V05, N11 abstract.

NUREG-0580 V12 N12: REGULATORY LICENSING STATUS SUMMARY REPORT. Data As Of December 31, 1983. (Blue Book) * Management Information Branch. January 1984. 60pp. 8402100442. 22213:049.

Provides a review of the status of the progress of the licensing reviews for all construction permits, operating licenses, special projects and non-power reactor renewals under review, as reported to Congress.

NUREG-0606 V06 N01: UNRESOLVED SAFETY ISSUES SUMMARY. Data As Of February 17, 1984. (Aqua Book) * Management Information Branch. February 1984. 55pp. 8403230212. 22743:294.

Provides an overview of the status of the generic tasks addressing "Unresolved Safety Issues" as reported to Congress.

NUREG-0606 V06 N02: UNRESOLVED SAFETY ISSUES SUMMARY. Data As Of May 18, 1984. (Aqua Book) * Division of Safety Technology. May 1984. 57pp. 8406120260. 24910:236.

See NUREG-0606, V06, N01 abstract.

NUREG-0606 V06 N03: UNRESOLVED SAFETY ISSUES SUMMARY. Data As Of August 17, 1984. (Aqua Book) * Division of Safety Technology. August 1984. 57pp. 8409260639. 26730:201.

See NUREG-0606, V06, N01 abstract.

NUREG-0606 V06 N04: UNRESOLVED SAFETY ISSUES SUMMARY. Data As Of November 16, 1984. (Aqua Book) * Division of Safety Technology. November 1984. 53pp. 8501030031. 28195:308.

See NUREG-0606, V06, N01 abstract.

NUREG-0647: SAFETY EVALUATION AND ENVIRONMENTAL ASSESSMENT, THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 2. Docket No. 50-320. (Metropolitan Edison Company, Jersey Central Power And Light Company And Pennsylvania Electric Company) * Office of Nuclear Reactor Regulation, Director. February 1980. 24pp. 8401270427. 22002:246.

This report contains an order for the Three Mile Island Nuclear Station, Unit 2, issued by the NRC. The order requires that effective immediately, the facility be maintained in accordance with the requirements of the attached proposed Technical Specifications and (2) proposes to formally amend the Facility Operating License to include the proposed Tecnnical Specifications, taking into account the present account of the plant systems, so as to ensure that the unit will remain in a safe posture during the Recovery Mode.

NUREG-0649 R01: TASK ACTION PLANS FOR UNRESOLVED SAFETY ISSUES RELATED TO NUCLEAR POWER PLANTS. * Division of Safety Technology. September 1984. 200pp. 8410170313. 27026:081.

This document contains Task Action Plans for generic tasks addressing Unresolved Safety Issues (USIs) related to nuclear power plants. Progress on USIs is reported to Congress each year in the NRC Annual Report pursuant to the requirements of Section 210 of the Energy Reorganization Act of 1974, as amended. In addition, the NRR issues NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book" on a quarterly basis; this report provides current schedule information for each USI.

The Task Action Plans in this document include a description of the issue, a description of the NRC staff's approach to resolving the issue, a general discussion of the basis for continued operation and licensing pending resolution of the issue, a discussion of the technical organizations involved in the task, the requirements of manpower and program support funding, interactions with outside organizations and potential problems. This document does not include Task Action Plans for generic tasks addressing USIs for which reports providing the NRC staff resolution of the issue have been published. Those tasks for which reports have been published are identified and the reports are referenced.

The Task Action Plans for active USIs are revised on a yearly basis. This report contains the 1984 revisions to the Task Action Plans.

NUREG-0675 S17: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-275 And 50-323. (Pacific Gas And Electric Company) * Division of Licensing. February 1984. 22pp. 8403070104. 22561:214.

Supplement No. 17 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate the Diablo Canyon Nuclear Power Plant (Docket Nos. 50-275 and 50-323) located in San Luis Obispo County, California has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement updates the Safety Evaluation Report by providing additional information on the breakwater issue.

NUREG-0675 S22: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-275 And 50-323. (Pacific Gas And Electric Company) * Division of Licensing. March 1984. 400pp. 8403300300. 22843:001.

Supplement No. 22 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate the Diablo Canyon Nuclear Power Plant (Docket Nos. 50-275 and 50-323), located in San Luis Obispo County, California, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides information on the Commission's review of allegations and concerns about the design, construction and operation of Diablo Canyon identified to the NRC as of March 9, 1984. It includes the criteria that were used by the NRC to determine which of the allegations that have been evaluated thus far must be resolved prior to Unit 1 achieving criticality and operating at power levels up to 5 percent of rated power (i.e., low power operation).

NUREG-0675 S23: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-275 And 50-323. (Pacific Gas And Electric Company) * Division of Licensing. June 1984. 46pp. 8407110014. 25546:280.

Supplement No. 23 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate the Diablo Canyon Nuclear Power Plants (Docket Nos. 50-275 and 50-323), located in San Luis Obispo County, California, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses the licensee's requests for deviations from Section III.G in Appendix R (related to fire protection) of Title 10 of the Code of Federal Regulations Part

50, presents the staff's evaluation and conclusion regarding each request, and summarizes the staff's review of the licensee's requests.

NUREG-0675 S24: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-275 And 50-323. (Pacific Gas And Electric Company) * Division of Licensing. July 1984. 16pp. 8408130002. 26038:295.

Supplement 24 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-275 and 50-323), has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. This supplement reports on the independent design verification program (IDVP) for Diablo Canyon Unit 1 that was performed between November 1981 and May 1984 in response to Commission Order CLI-81-30 and an NRC letter and its application by PG&E in the Internal Technical Program (ITP). Specifically, Supplement 25 presents the final resolution of the remaining issues and other matters identified in Supplements 18, 19 and 20. This SER Supplement applies only to Diablo Canyon Unit 1.

NUREG-0675 S25: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-275 And 50-323. (Pacific Gas And Electric Company) * Division of Licensing. July 1984. 122pp. 8408160080. 26124:033.

Supplement 25 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-275 and 50-323) has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. This supplement reports on the staff's inspection and evaluation efforts on the matter of piping and piping supports as reflected by the seven technical license conditions in our "Order Modifying License" issued by the Office of Nuclear Reactor Regulation on April 18, 1984.

NUREG-0675 S26: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2. * Division of Licensing. July 1984. 204pp. 8408220346. 26200:001.

Supplement 26 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate Diablo Canyon Nuclear Power Plants, Units 1 and 2 (Docket Nos. 50-275 and 50-323), has been prepared jointly by the Office of Nuclear Reactor Regulation and the Region V Office of the U. S. Nuclear Regulatory Commission. This supplement reports on the status of the staff's investigation, inspection and evaluation of those allegations or concerns that have been identified to the NRC as of July 1, 1984. The report specifically addresses those allegations which the staff determined must be satisfactorily resolved prior to full power operation of Diablo Canyon Unit 1.

NUREG-0675 S27: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-275 And 50-323. (Pacific Gas And Electric Company) * Division of Licensing. July 1984. 31pp. 8408160250. 26122:235.

Supplement No. 27 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Dockets 50-275 and 50-323), has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. This supplement reports on the independent design verification program (IDVP) for Diablo Canyon and conditions contained in Amendment No. 10 to the Operating License.

NUREG-0680 S05: TMI-1 RESTART. An Evaluation Of The Licensee's Management Integrity As It Affects Restart Of Three Mile Island Unit 1, Docket 50-289. * Division of Licensing. July 1984. 179pp. 8408060413. 25937:102.

Supplement 5 to the Safety Evaluation Report (SER) on TMI-1 Restart documents the review by the Nuclear Regulatory Commission (NRC) staff of nine investigations conducted by the NRC Office of Investigations into matters identified as relevant and material to an evaluation of the licensee's "management integrity." The staff has included, as part of its evaluation, materials from its review of the GPU v. B&W lausuit record (NUREG-1020LD, "GPU v. B&W Lawsuit Review and Its Effect on TMI-1") as well as other relevant materials developed since the close of the record in the TMI-1 Restart proceeding. In developing its position on General Public Utilities Nuclear Corporation's character (i.e., management integrity), the staff evaluated matters that cast doubt on the licensee's character, individually and collectively; considered the remedial actions taken by the licensee; and balanced past improper conduct of the licensee against its subsequent record of remedial actions and performance and record of current senior management of the licensee. The staff concluded that, while the past improper conduct was grave, the remedial actions taken, the subsequent record of performance, and the record of current senior management support of finding that GPUN can and will operate TMI-1 without undue risk to the health and safety of the public.

NUREG-0683 S01: PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT RELATED TO DECONTAMINATION AND DISPOSAL OF RADIOACTIVE WASTES RESULTING FROM MARCH 28, 1979 ACCIDENT, THREE MILE ISLAND NUCLEAR STATION, UNIT 2. Docket No 50-320 (GPU Nuclear, Incorporated) * TMI Program Office. October 1984. 175pp. 8411010080. 27260:118.

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 has been supplemented. The supplement was required because current information indicates that cleanup may entail substantially more occupational radiation dose to the cleanup work force than originally anticipated. Cleanup was originally estimated to result in from 2000 to 8000 person-rem of occupational radiation dose. Although only 2000 person-rem have resulted from cleanup operations performed up to now, current estimates now indicate that between 13,000 and 46,000 person-rem are expected to be required. Alternative cleanup methods considered in the supplement either did not result in appreciable dose savings or were not known to be technically feasible.

NUREG-0698 R02: NRC PLAN FOR CLEANUP OPERATIONS AT THREE MILE ISLAND UNIT 2. MASNIK, M. T.; SNYDER, B. J. TMI Program Office. March 1984. 40pp. 8404100141. 22995:027.

This report updates a plan that defines NRC's role in cleanup operations at Three Mile Island Unit 2 (TMI-2) and outlines NRC's regulatory responsibilities in fulfilling this role.

Since the initial issuance of this NRC Plan in July 1980, this office has issued the Final NRC Programmatic Evironmental Impact Statement (PEIS) related to the entire TMI-2 cleanup and a draft Supplement to the PEIS related to occupational radiation exposure. Additionally, a number of developments have occurred which will have an impact on the course of cleanup operations. This revision provides a discussion of these developments, specifically in the areas of the functional role of the NRC in cleanup operations, the cleanup schedule, and the current status of the cleanup. The plan also discusses NRC's perceived role in future cleanup activities. Because of major uncertainties in the funding of the cleanup, portions of this plan, including the estimated schedule, are likely to require further changes as availability of funding and other factors affect the pace of the cleanup.

NUREG-0725 R04: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Division of Safeguards. June 1984. 51pp. 8407190487. 25693:177.

This circular has been prepared in response to numerous requests for information regarding routes used for the shipment of irradiated reactor (spent fuel) subject to regulation by the Nuclear Regulatory Commission (NRC), and to meet the requirements of Public Law 96-295. The NRC staff must approve such routes prior to their first use in accordance with the regulatory provisions of Section 73.37 of 10 CFR Part 73. The information included reflects NRC staff knowledge as of June 1, 1984. Spent fuel shipment routes, primarily for road transportation, but also including one rail route, are indicated on reproductions of DOT maps. Also included are the amounts of material shipped during the approximate three year period that safeguards regulations for spent fuel shipments have been effective. In addition, the Commission has chosen to provide information in this document regarding the NRC's safety and safeguards regulations for spent fuel shipments as well as safeguards incidents regarding spent fuel shipments (of which none have been reported to date). This additional information is furnished by the Commission in order to convey to the public a more complete picture of NRC regulatory practices concerning the shipment of spent fuel than could be obtained by the publication of the shipment routes and quantities alone.

NUREG-0732 R01: ANSWERS TO FREQUENTLY ASKED QUESTIONS ABOUT CLEANUP ACTIVITIES AT THREE MILE ISLAND, UNIT 2. * TMI Program Office. March 1984. 55pp. 8404130056. 24049:001.

This question-and-answer report provides answers in nontechnical language to frequently asked questions about the status of cleanup activities at Three Mile Island, Unit 2. This revision updates answers first prepared in 1981, shortly after the cleanup got under way.

NUREG-0748 V03 N11: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of November 30, 1983. (Orange Book) * Division of Budget & Analysis. January 1984. 336pp. 8401260119. 21968:312.

The Operating Reactors Licensing Actions Summary is a monthly publication designed to provide the Management of the Nuclear Regulatory Commission (NRC) with an overview of licensing actions dealing with operating power and nonpower reactors.

NUREG-0748 V03 N12: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of December 31, 1983. (Orange Book) * Division of Budget & Analysis. January 1984. 335pp. 8402060497. 22108:039.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N01: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of January 31, 1984. (Orange Book) * Division of Budget & Analysis. March 1984. 332pp. 8403270269. 22788:001.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N02: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of February 29, 1984. (Orange Book) * Division of Budget & Analysis. April 1984. 150pp. 8404240178. 24190:001.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N03: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of March 31, 1984. (Orange Book) * Division of Budget & Analysis. May 1984. 355pp. 8405210574. 24530:001.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N04: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of April 30, 1984. (Orange Book) * Division of Budget & Analysis. June 1984. 384pp. 8406210444. 25099:073.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N05: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of May 31, 1984. (Orange Book) * Division of Budget & Analysis. July 1984. 200pp. 8407170557. 25630:001.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N06: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of June 30, 1984. (Orange Book) * Division of Budget & Analysis. July 1984. 400pp. 8408080363. 25981:129.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N07: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of July 31, 1984. (Orange Book) * Division of Budget & Analysis. August 1984. 150pp. 8409170276. 26497:001.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N08: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As of August 31, 1984. (Orange Book) * Division of Budget & Analysis. October 1984. 400pp. 8410230630. 27099:001.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N09: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As of September 30, 1984. (Orange Book) * Division of Budget & Analysis. November 1984. 416pp. 8411200447. 27589:001.
See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N10: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of October 31, 1984. (Orange Book) * Division of Budget & Analysis. November 1984. 310pp. 8412260139. 28070:001.

See NUREG-0748, V03, N11 abstract.

NUREG-0748 V04 N11: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of November 30, 1984. (Orange Book) * Division of Budget & Analysis. December 1984. 300pp. 8501080647. 28259:070.

See NUREG-0748, V03, N11 abstract.

NUREG-0750 V16 B01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-SEPTEMBER 1982. Pages 1-1,218. * Division of Technical Information & Document Control. September 1982. 1220. 8402170121. 22306:001.

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

NUREG-0750 V16 B02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER-DECEMBER 1982. Pages 1,219-2,140. * Division of Technical Information & Document Control. December 1982. 960pp. 8402170060. 22309:264.

See NUREG-0750, V16, B01 abstract.

NUREG-0750 V18 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY-SEPTEMBER 1983. * Division of Technical Information & Document Control. September 1983. 85pp. 8404110317. 24007:101.

See NUREG-0750, V16, B01 abstract.

NUREG-0750 V18 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1983. Pages 299-742. * Division of Technical Information & Document Control. September 1983. 451pp. 8402280257. 22503:301.

See NUREG-0750, V16, B01 abstract.

NUREG-0750 V18 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1983. Pages 743-1,137. * Division of Technical Information & Document Control. October 1983. 400pp. 8403230183. 22813:040.

See NUREG-0750, V16, B01 abstract.

NUREG-0750 V18 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1983. Pages 1,139-1,301. * Division of Technical Information & Document Control. November 1983. 167pp. 8404130221. 24036:175.

See NUREG-0750, V16, B01 abstract.

NUREG-0750 V19 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY MARCH 1984. * Division of Technical Information & Document Control. September 1984. 73pp. 8410100166. 26906:247.

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

NUREG-0750 V19 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY-JUNE 1984. * Division of Technical Information & Document Control. November 1984. 115pp. 8412100351. 27870:190.
See NUREG-0750,V19,I01 abstract.

NUREG-0750 V19 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1984. Pages 1-485. * Division of Technical Information & Document Control. January 1984. 487pp. 8407130479. 25582:311.
See NUREG-0750,V16,B01 abstract.

NUREG-0750 V19 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1984. Pages 487-554. * Division of Technical Information & Document Control. February 1984. 75pp. 8407130393. 25575:242.
See NUREG-0750,V16,B01 abstract.

NUREG-0750 V19 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1984. Pages 555-936. * Division of Technical Information & Document Control. August 1984. 386pp. 8408240187. 26250:001.
See NUREG-0750,V16,B01 abstract.

NUREG-0750 V19 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1984. Pages 937-1,149. * Division of Technical Information & Document Control. August 1984. 200pp. 8409260625. 26705:114.
See NUREG-0750,V16,B01 abstract.

NUREG-0750 V19 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1984. Pages 1,151-1,321. * Division of Technical Information & Document Control. September 1984. 180pp. 8410150096. 26999:144.
See NUREG-0750,V16,B01 abstract.

NUREG-0750 V19 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1984. Pages 1,323-1,606. * Division of Technical Information & Document Control. October 1984. 294pp. 8411080545. 27407:115.
See NUREG-0750,V16,B01 abstract.

NUREG-0750 V20 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1984. Pages 1-247. * Division of Technical Information & Document Control. November 1984. 254pp. 8412100269. 27871:001.
See NUREG-0750,V16,B01 abstract.

NUREG-0750 V20 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1984. Pages 249-798. * Division of Technical Information & Document Control. December 1984. 550pp. 8501030273. 28197:001.
See NUREG-0750,V16,B01 abstract.

NUREG-0750 V20 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1984. Pages 799-1,054. * Division of Technical Information & Document Control. December 1984. 260pp. 8501070409. 28219:082.
See NUREG-0750,V16,B01 abstract.

NUREG-0775: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NOS. 1 AND 2. Docket Nos. 50-445 And 50-446. (Texas Utilities Generating Company) * Office of Nuclear Reactor Regulation, Director. September 1981. 160pp. 8401270428. 22001:143.

The information in this Final Environmental Statement is the second assessment of the environmental impact associated with the construction and operation of the Comanche Peak Steam Electric Station, Units 1 and 2, located at a site on Squaw Creek Reservoir in Somervell County, Texas. The Station will be operated by the Texas Utilities Generating Company. The Draft Environmental Statement related to the operation of the station was issued May 14, 1981. The first assessment was the Final Environmental Statement related to the proposed station and was issued in June 1974, prior to issuance of the construction permits in December 1974. The present assessment is the result of the NRC staff review of the activities associated with the proposed operation of the station, and includes the staff response to comments on the Draft Environmental Statement.

NUREG-0776 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2. Docket Nos. 50-387 And 50-388. (Pennsylvania Power And Light Company; Allegheny Electric Cooperative, Incorporated) * Division of Licensing. March 1984. 100pp. 8404110036. 24002:123.

In April 1981, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0776) regarding the application of the Pennsylvania Power and Light Company (applicant or licensee) and the Allegheny Electric Cooperative, Inc. (co-applicant) for licenses to operate the Susquehanna Steam Electric Station, Units 1 and 2, located on a site in Luzerne County, Pennsylvania.

Supplements 1 and 2 were issued in June 1981 and September 1981, respectively and addressed several outstanding issues. Supplement No. 2 also contains NRC staff responses to the comments made by the Advisory Committee on Reactor Safeguards in its report, dated August 11, 1981. Supplement 3 was issued in July 1982 and addressed five items that remained open and closed them out. On July 17, 1982, Operating License NPF-14 was issued to allow Unit 1 operation at power levels not to exceed 5% of rated power. Supplement 4 was issued November 1982 and discusses the resolution of several license conditions. On November 12, 1982, Operating License NPF-14 was amended to remove the 5% power restriction, thereby permitting full-power operation of Unit 1. Supplement 5 and this Supplement, No. 6 addresses several issues that require resolution before licensing operation of Unit 2.

NUREG-0776 S07: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2. Docket Nos. 50-387 And 50-388. (Pennsylvania Power And Light Company, Allegheny Electric Cooperative, Incorporated) * Division of Licensing. May 1984. 19pp. 8406190069. 25025:253.

In April 1981, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0776) regarding the application of the Pennsylvania Power & Light Company (the applicant and/or licensee) and the Allegheny Electric Cooperative, Inc. (co-applicant) for licenses to operate the Susquehanna Steam Electric Station, Units 1 and 2, located on a site in Luzerne County, Pennsylvania.

Supplements 1 and 2 were issued in June 1981 and September 1981, respectively. Supplement No. 2 also contains NRC staff responses to the comments made by the Advisory Committee on Reactor Safeguards in its report, dated August 11, 1981. Supplement No. 3 was issued in July 1982 and closed out 5 remaining items. On July 17, 1982, Operating License NPF-14 was issued to Unit 1 to allow operation at 5% of rated power. Supplement No. 4 was issued in November 1982 and discusses the resolution of several license conditions. On November 12, 1982, Operating License NPF-14 was amended to remove the 5% power restriction, thereby permitting full-power operation of Unit 1. Supplement 5 was issued March 1983, Supplement 6 was issued in March 1984 and both addressed remaining issues that required resolution prior to operating Unit 2. On March 23, 1984 Operating License NPF-22 was issued to allow Unit 2 operation not to exceed 5% of rated power. This Supplement addresses those issues which require resolution prior to allowing Unit 2 operation at power levels exceeding 5% rated power.

NUREG-0787 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD STEAM ELECTRIC STATION, UNIT 3. Docket No. 50-382. (Louisiana Power And Light Company) * Division of Licensing. June 1984. 168pp. 8407110007. 25545:008.

Supplement 6 to the Safety Evaluation Report for the application filed by Louisiana Power & Light Company for a license to operate the Waterford Steam Electric Station, Unit 3 (Docket No. 50-382), located in St. Charles Parish, Louisiana, 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 the staff's evaluation of information submitted by the applicant since the Safety Evaluation Report and its five previous supplements were issued.

NUREG-0787 S07: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD NUCLEAR POWER PLANT, UNIT 3. Docket No. 50-382. (Louisiana Power And Light Company) * Office of Nuclear Reactor Regulation, Director. September 1984. 300pp. 8410110526. 26941:001.

Supplement 7 to the Safety Evaluation Report for Louisiana Power & Light application for a license to operate Waterford Steam Electric Station, Unit 3 (Docket No. 50-382), located in St. Charles Parish, Louisiana, has been jointly prepared by thr Office of Reactor Regulation and the Region IV Office of the U.S. Nuclear Regulatory Commission. This supplement provides the results of the staff's evaluation of approximately 350 allegations and concerns of poor construction practices at the Waterford 3 facility.

NUREG-0787 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD STEAM ELECTRIC STATION, UNIT NO. 3. Docket No. 50-382 (Louisiana Power & Light Company) * Division of Licensing. December 1984. 100pp. 8412260149. 28069:255.

Supplement 8 to the Safety Evaluation Report for the application filed by Louisiana Power & Light Company for a license to operate the Waterford Steam Electric Station, Unit 3 (Docket No. 50-382), located in St. Charles Parish, Louisiana, 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 the staff's evaluation of information submitted by the applicant since the Safety Evaluation Report and its seven previous supplements were issued.

NUREG-0787 S09: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD STEAM ELECTRIC STATION, UNIT NO. 3. Docket No. 50-382 (Louisiana Power & Light Company) * Division of Licensing. December 1984. 400pp. 8501080457. 28272:001.

Supplement No. 9 to the Safety Evaluation Report for Louisiana Power & Light's application for a license to operate Waterford Steam Electric Station, Unit 3 (Docket No. 50-382), located in St. Charles Parish, Louisiana, has been jointly prepared by the Office of Nuclear Reactor Regulation and the Region IV Office of the U.S. Nuclear Regulatory Commission. This supplement provides the results of the staff's completion of its evaluation of approximately 350 allegations and concerns of poor construction practices at the Waterford 3 facility.

NUREG-0797 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2. Docket Nos. 50-445 And 50-446. (Texas Utilities Generating Company, et al) * Division of Licensing. November 1984. 250pp. 8412050040. 27808:018.

Supplement No. 6 to the Safety Evaluation Report (SER) related to the operation of the Comanche Peak Steam Electric Station, Units 1 and 2 has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Somervell County, Texas. Subject to favorable resolution of the items identified in this supplement, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public. This supplement addresses open items and issues concerning the issuance of a five percent low power license. Supplement No. 6 is being issued out of sequence. Supplement No. 5 addresses the NRC staff's evaluation of the Independent Assessment Program for Comanche Peak Steam Electric Station prepared for the applicant by Cygna Energy Services. This report will be issued later during 1984.

NUREG-0798 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF ENRICO FERMI ATOMIC POWER PLANT, UNIT NO. 2. Docket No. 50-341. (Detroit Edison Company) * Division of Licensing. September 1984. 91pp. 8410120046. 26983:121.

Supplement No. 4 to the Safety Evaluation Report related to the operation of the Enrico Fermi Atomic Power Plant, Unit 2, provides the staff's evaluation of additional information submitted by the application regarding outstanding review issues identified in Supplement No. 3 to the Safety Evaluation Report, dated January 1983.

NUREG-0800 03.9.3 R1: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 1 To Section 3.9.3, Appendix A. SERKIZ, A.W. Division of Safety Technology. April 1984. 11pp. 8404170399. 24068:311.

Revision No. 1 to Appendix A of Standard Review Plan Section 3.9.3 incorporates changes that have been developed since the original issuance in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer".

NUREG-0800 03.9.4 R2: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 2 To Section 3.9.4, "Control Rod Drive Systems." SERKIZ, A.W. Division of Safety Technology. April 1984. 9pp. 8404170381. 24068:322.

Revision No. 2 to Standard Review Plan Section 3.9.4 incorporates changes that have been developed since the issuance of Revision 1 in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer".

NUREG-0800 05.4.6 R3: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 3 To Section 5.4.6, "Reactor Core Isolation Cooling System (BWR)." SERKIZ, A.W. Division of Safety Technology. April 1984. 11pp. 8404170467. 24091:227.

Revision No. 3 to Standard Review Plan Section 5.4.6. incorporates changes that have been developed since the issuance of Revision 2 in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer".

NUREG-0800 05.4.7 R3: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 3 To Section 5.4.7, "Residual Heat Removal (RHR) System." SERKIZ, A.W. Division of Safety Technology. April 1984. 20pp. 8404170350. 24069:253.

Revision No. 3 to Standard Review Plan Section 5.4.7 incorporates changes that have been developed since the issuance of Revision 2 in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer".

NUREG-0800 06.2.1 R6: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 6 to Section 6.2.1.1.C, "Pressure-Suppression Type BWR Containments." * Office of Nuclear Reactor Regulation, Director. August 1984. 9pp. 8410020474. 26791:275.

Revision 6 to SRP Section 6.2.1.1.C of the Standard Review Plan incorporates the resolution of Generic Issue B-10, "Behavior of BWR/Mark III Containments."

NUREG-0800 06.3 R2: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 2 To Section 6.3, "Emergency Core Cooling System." SERKIZ, A.W. Division of Safety Technology. April 1984. 16pp. 8404170375. 24068:331.

Revision No. 2 to Standard Review Plan Section 6.3 incorporates changes that have been developed since the issuance of Revision 1 in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer". BTP RSB 6-1 is also included with revised page numbers--no other changes were made to the BTP.

NUREG-0800 07.1 R3: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision No. 3 To Section 7.1, Revision 1 To Appendix A. * Office of Nuclear Reactor Regulation, Director. March 1984. 24pp. 8404180395. 24109:060.

Revision No. 3 to Section 7.1 of the Standard Review Plan incorporates changes that have been developed since the original issuance in July 1981. This revision incorporates the resolution of Generic Issues 45, "Inoperability of Instrumentation Due to Extreme Cold Weather."

NUREG-0800 07.5 R3: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision No. 3 To Section 7.5. * Office of Nuclear Reactor Regulation, Director. March 1984. 8pp. 8404180397. 24109:052.

Revision 3 to Section 7.5 of the Standard Review Plan incorporates changes that have been developed since the original issuance in July 1981. This revision incorporates the resolution of Generic Issue 45, "Inoperability of Instrumentation Due to Extreme Cold Weather."

NUREG-0800 07.7 R3: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision No. 3 To Section 7.7. * Office of Nuclear Reactor Regulation, Director. March 1984. 7pp. 8404180399. 24109:045.

Revision No. 3 to Section 7.7 of the Standard Review Plan incorporates changes that have been developed since the original issuance in July 1981. This revision incorporates the resolution of Generic Issue 45, "Inoperability of Instrumentation Due to Extreme Cold Weather."

NUREG-0800 09.2.1 R3: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision No. 3 To Section 9.2.1, "Station Service Water System." SERKIZ, A. W. Division of Safety Technology. April 1984. 10pp. 8404170057. 24091:257.

Revision No. 3 to Standard Review Plan Section 9.2.1 incorporates changes that have been developed since the issuance of Revision 2 in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer".

NUREG-0800 09.2.2 R2: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 2 To Section 9.2.2, "Reactor Auxiliary Cooling Water Systems." SERKIZ, A. W. Division of Safety Technology. April 1984. 12pp. 8404170042. 24091:283.

Revision No. 2 to Standard Review Plan Section 9.2.2 incorporates changes that have been developed since the issuance of Revision 1 in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer".

NUREG-0800 10.3 R3: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision No. 3 To Section 10.3, "Main Steam Supply System." SERKIZ, A. W. Division of Safety Technology. April 1984. 12pp. 8404170062. 24069:241.

Revision No. 3 to Standard Review Plan Section 10.3 incorporates changes that have been developed since the issuance of Revision 2 in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer".

NUREG-0800 10.4.7 R3: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 3 To Section 10.4.7, "Condensate And Feedwater System" And BTP ASB 10-2, "Design Guidelines For Avoiding Water Hammer...." SERKIZ, A. W. Division of Safety Technology. April 1984. 11pp. 8404170353. 24068:347.

Revision No. 3 to Standard Review Plan Section 10.4.7 and BTP ASB 10-2 incorporates changes that have been developed since the issuance of Revision 2 in July 1981. This revision incorporates the resolution of Unresolved Safety Issue A-1, "Water Hammer".

NUREG-0800 18.0 R1: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 1 to SRP Section 18.0, "Human Factors Engineering-Introduction." * Office of Nuclear Reactor Regulation, Director. October 1984. 5pp. 8411070294. 27388:085.

Revision 1 to SRP Section 18.0 of the Standard Review Plan incorporates the guidelines of Task Action Plan Item I.D.1 of NUREG-0660 as clarified in Supplement 1 of NUREG-0737.

NUREG-0800 18.1 R0: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS. LWR Edition. Revision 0 to SRP Section 18.1, "Control Room," and Appendix A, Rev 0, "Evaluation Criteria for Detailed Control Room Design...." * Office of Nuclear Reactor Regulation, Director. October 1984. 42pp. 8411070296. 27388:007.

Revision 0 to SRP Section 18.1 of the Standard Review Plan incorporates the guidelines of Task Action Plan Item I.D.1 of NUREG-0660 as clarified in Supplement 1 of NUREG-0737. Appendix A to this section was formerly draft NUREG-0801.

NUREG-0828: INTEGRATED PLANT SAFETY ASSESSMENT REPORT, SYSTEMATIC EVALUATION PROGRAM. Big Rock Point Plant. Docket No. 50-155. (Consumers Power Company) * Division of Licensing. May 1984. 800pp. 8406120255. 24917:115.

The Systematic Evaluation Program was initiated in February 1977 by the U.S. Nuclear Regulatory Commission to review the designs of older operating nuclear reactor plants to confirm and document their safety. The review provides (1) an assessment of how these plants compare with current licensing safety requirements relating to selected issues, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

This report documents the review of the Big Rock Point Plant, operated by Consumers Power Company located in Charlevoix, Michigan. Big Rock Point is one of ten plants reviewed under Phase II of this program. This report indicates how 137 topics selected for review under Phase I of the program were addressed. It also addresses a majority of the pending licensing actions for Big Rock Point, which include TMI Action Plan requirements and implementation criteria for resolved generic issues. Equipment and procedural changes have been identified as a result of the review.

NUREG-0830 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CALLAWAY PLANT, UNIT NO. 1. Docket No. 50-483. (Union Electric Company) * Division of Licensing. May 1984. 194pp. 8405290428. 24695:074.

Supplement No. 3 to the Safety Evaluation Report related to the operation of the Callaway Plant, Unit No. 1 resolves open items and updates information contained in the Safety Evaluation, dated October 1981. Supplements 1 and 2, dated January 1982 and June 1983, respectively also updates the information contained in the Safety Evaluation Report. Supplement No. 1 contained the ACRS Report issued

on November 17, 1981.
The Safety Evaluation Report and its supplements pertain to the application for a license to operate the Callaway Plant filed by the Union Electric Company on October 19, 1979.

NUREG-0830 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CALLAWAY PLANT, UNIT ONE. Docket No. 50-483. (Union Electric Company) * Division of Licensing. October 1984. 80pp. 8411070331. 27386:180.
Supplement No. 4 to the Safety Evaluation Report related to the operation of the Callaway Plant Unit No. 1 resolves open items and updates information contained in the Safety Evaluation Report and Supplements 1, 2 and 3. On June 11, 1984 a fuel load and low power testing (5%) license (NPF-25) was issued to the Union Electric Company. This Supplement pertains to those items which must be resolved prior to issuance of the full power license.

NUREG-0831 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION, UNITS 1 AND 2. Docket Nos. 50-416 And 50-417. (Mississippi Power And Light Company, Middle South Energy, Inc And South Mississippi Electric Power Association) * Division of Licensing. August 1984. 122pp. 8409270154. 26719:164.
Supplement 5 to the Safety Evaluation Report for Mississippi Power & Light Company, et al, joint application for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2, located on the east bank of the Mississippi River near Port Gibson in Claiborne County, Mississippi, has been prepared by the Office of Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the status on the resolution of those issues that require further evaluation before authorizing operation of Unit 1 above 5% of rated power.

NUREG-0831 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION, UNITS 1 AND 2. Docket Nos. 50-416 And 50-417. (Mississippi Power And Light Company) * Division of Licensing. August 1984. 204pp. 8409260656. 26701:181.
Supplement No. 6 to the Safety Evaluation Report for Mississippi Power & Light Company et al joint application for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2, located on the east bank of the Mississippi River near Port Gibson in Claiborne County, Mississippi, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The supplement reports the NRC staff's evaluation of open items from previous supplements and Technical Specification changes required before authorizing operation of Unit 1 above 5% of rated power.

NUREG-0831 S07: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION, UNITS 1 And 2. Docket Nos. 50-416 And 50-417. (Mississippi Power And Light Company) * Division of Licensing. October 1984. 16pp. 8411210112. 27618:114.
This report supplements the Safety Evaluation Report (NUREG-0831) issued in September 1981 by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by Mississippi Power & Light (MP&L) Company, Middle South Energy, Inc. and South Mississippi Electric Power Association as applicants and owners, for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2 (Docket Nos. 50-416 and 50-417,

respectively). The facility is located on the east bank of the Mississippi River near Port Gibson in Clairborne County, Mississippi. This supplement provides information on the NRC staff's evaluation of the requests for exemptions to NRC regulations pursuant to the Commission's direction in CLI-84-19, dated October 25, 1984.

NUREG-0837 V03 N03: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report, July-September 1983. COSTELLO, F.; THOMPSON, T.; COHEN, L.; et al. Region 1, Office of Director. March 1984. 138pp. 8404020212. 22877:245.

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 1983.

NUREG-0837 V03 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report September-December 1983. COSTELLO, F.; THOMPSON, T.; COHEN, L.; et al. Region 1, Office of Director. May 1984. 247pp. 8406060392. 24741:148.

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 1983.

NUREG-0837 V04 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report January-March 1984. COSTELLO, F.; KRAMARIC, M.; COHEN, L. Region 1, Office of Director. October 1984. 147pp. 8411260458. 27646:001.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity of NRC licensed facility sites throughout the country for the first quarter of 1984.

NUREG-0837 V04 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK. Progress Report, April-June 1984. JANG, J.; KRAMARIC, M.; COHEN, L. Region 1, Office of Director. November 1984. 149pp. 8501020018. 28156:230.

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 second quarter of 1984.

NUREG-0847 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2. Docket Nos. 50-390 And 50-391. (Tennessee Valley Authority) * Division of Licensing. January 1984. 39pp. 8402100502. 22213:109.

This report supplements the Safety Evaluation Report, NUREG-0847 (June 1982) and Supplement No. 1 (September 1982), issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the

Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the open and confirmatory items and license conditions identified in the Safety Evaluation Report.

NUREG-0853 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CLINTON POWER STATION, UNIT NO. 1. Docket No. 50-461. (Illinois Power Company, et al) * Division of Licensing. May 1984. 40pp. 8406190045. 25025:274.

Supplement No. 3 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 certain items that had not been resolved at the time of publication of the Safety Evaluation Report and Supplements No. 1 and No. 2.

NUREG-0857 S06: 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) * Division of Licensing. October 1984. 78pp. 8411210379. 27617:354.

Supplement No. 6 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 applicants since Supplement No. 5 was issued and (2) matters that the staff had under review when Supplement No. 5 was issued.

NUREG-0857 S07: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION, UNIT NOS. 1, 2 AND 3. Docket Nos. 50-528, 50-529 & 50-530. (Arizona Public Service Company) * Division of Licensing. December 1984. 300pp. 8501080275. 28269:004.

Supplement No. 7 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 applicants since Supplement No. 6 was issued and (2) matters that the staff had under review when Supplement 6 was issued.

NUREG-0871 V03 N01: SUMMARY INFORMATION REPORT. Data As Of December 31, 1983. (Brown Book) * Management Information Branch. June 1984. 52pp. 8406250269. 25138:239.

Provides summary data concerning NRC and its licensees for general use by the Chairman, other Commissioners and Commission staff offices, the Executive Director for Operations, and the Office Directors.

NUREG-0876 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF THE BYRON STATION, UNITS 1 AND 2. Docket Nos. STN 50-454 And STN 50-455. (Commonwealth Edison Company) * Division of Licensing. May 1984. 32pp. 8406060010. 24847:254.

Supplement No. 4 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, Ogle County, Illinois, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report and Supplements 1 through 3.

NUREG-0876 S05: 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 Licensing. October 1984. 178pp. 8411200507. 27588:057.

Supplement No. 5 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, Ogle County, Illinois, has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. This supplement addresses open items that had not been resolved at the time of publication of Supplement No. 4.

NUREG-0885 I03: US NUCLEAR REGULATORY COMMISSION POLICY AND PLANNING GUIDANCE 1984. * Commissioners. January 1984. 23pp. 8403150211. 22662:139.

The purpose of the Policy and Planning Guidance document are to state, clearly and succinctly, the major policies and planning objectives of the Commission so that all employees will know where the Agency is headed; to provide a common basis within NRC for the development of programs, the establishment of priorities, and the allocation of resources; to furnish guidance that can be used to develop Agency budget requests; and to help fulfill the requirement that NRC's annual report to the President for submission to Congress contain a clear statement of the short-range and long-range goals, priorities, and plans of the Commission as they relate to the benefits, costs, and risks of commercial nuclear power.

NUREG-0887 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2. Docket Nos. 50-440 And 50-441. (Cleveland Electric Illuminating Company) * Division of Licensing. February 1984. 142pp. 8402240381. 22381:118.

Supplement No. 4 to the Safety Evaluation Report on the application filed by the Cleveland Electric Illuminating Company on behalf of itself and as agent for the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company and the Toledo Edison Company (the Central Area Power Coordination Group or CAPCO), as applicants and owners, for a license to operate the Perry Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-440 and 50-441), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lake County, Ohio. This supplement, No. 4 further updates the SER and Supplements 1 through 3 by providing the results of the staff's review of information submitted by the applicant by letter addressing some of the issues listed in Sections 1.9, 1.10, and 1.11 of the SER that were unresolved at the time Supplement 3 was issued.

NUREG-0892 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WPPSS NUCLEAR PROJECT NO. 2 Docket No. 50-397.(Washington Public Power Supply System) * Division of Licensing. April 1984. 41pp. 8404240005. 24189:087.

Supplement No. 5 to the Safety Evaluation Report on the application filed by Washington Public Power Supply System for a license to operate the WPPSS Nuclear Project No. 2, located in Richland, Washington, has been prepared by the Division of Licensing, Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement is to update our evaluations on issues identified in the previous Safety Evaluation Report and Supplements that need resolution prior to issuance of the full power operating license.

NUREG-0926 R01: TECHNICAL SPECIFICATIONS FOR GRAND GULF NUCLEAR STATION, UNIT 1.Docket No. 50-416. (Mississippi Power And Light Company) HOFFMAN,D.R. Division of Licensing. August 1984. 540pp. 8409260633. 26691:001.

The Grand Gulf Nuclear Station, 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 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-0927 R01: EVALUATION OF WATER HAMMER OCCURRENCE IN NUCLEAR POWER PLANTS.Technical Findings Relevant To USI A-1. SERKIZ,A.W. Division of Safety Technology. March 1984. 79pp. 8404090032. 22947:198.

This report, which includes responses to public comments, summarizes key technical findings relevant to the Unresolved Safety Issue A-1, Water Hammer. These findings were derived from studies of reported water hammer occurrences and underlying causes and provide key insights into means to minimize or eliminate further water hammer occurrences. It should also be noted that this report does not represent a substitute for current rules and regulations.

NUREG-0933 S01: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; MINNERS,W.; VANDER MOLEN,H.; et al. Division of Safety Technology. July 1984. 120pp. 8408220265. 26212:062.

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-0934: TECHNICAL SPECIFICATIONS FOR GRAND GULF NUCLEAR STATION, UNIT 1. DOCKET No. 50-416. (Mississippi Power And Light Company) * Division of Licensing. October 1984. 532pp. 8411210148. 27619:243.

The Grand Gulf Nuclear Station, 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 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-0935: ACOUSTIC WAVE PROPAGATION IN FLUIDS WITH COUPLED CHEMICAL REACTIONS. MARGULIES, T. S.; SCHWARZ, W. H. Division of Risk Analysis & Operations (post 840429). August 1984. 44pp. 8409200389. 26608:310.

This report presents a hydroacoustic theory which accounts for sound absorption and dispersion in a multicomponent mixture of reacting fluids (assuming a set of first-order acoustic equations without diffusion) such that several coupled reactions can occur simultaneously. General results are obtained in the form of a biquadratic characteristic equation (called the Kirchhoff-Langevin equation) for the complex propagation variable chi = - (alpha + w/c) in which alpha is the attenuation coefficient, c is the phase speed of the progressive wave and w is the angular frequency. Computer simulations of sound absorption spectra have been made for three different chemical systems each comprised of two-step chemical reactions using physico-chemical data available in the literature. The relative chemical reaction and classical viscothermal contributions to the sound absorption are also presented. Several discrepancies that can arise when interprating ultrasonic measurements for estimating thermodynamic data (chemical reaction heats or volume changes) for multistep chemcial reaction systems are discussed.

NUREG-0936 V02 N04: NRC REGULATORY AGENDA. Quarterly Report, October-December 1983. * Division of Rules and Records. February 1984. 180pp. 8403020165. 22482:212.

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 V03 N01: NRC REGULATORY AGENDA. Quarterly Report, January-March 1984. * Division of Rules and Records. April 1984. 182pp. 8405020032. 24287:128.

See NUREG-0936, V02, N04 abstract.

NUREG-0936 V03 N02: NRC REGULATORY AGENDA. Quarterly Report, April-June 1984. * Division of Rules and Records. July 1984. 201pp. 8408100149. 25998:112.

See NUREG-0936, V02, N04 abstract.

NUREG-0936 V03 N03: NRC REGULATORY AGENDA. Quarterly Rept, July-September 1984. * Division of Rules and Records. October 1984. 200pp. 8412190371. 28027:001.

See NUREG-0936, V02, N04 abstract.

NUREG-0940 V02 N04: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report, October-December 1983. * Enforcement Staff. January 1984. 437pp. 8402210028. 22341:286.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1983) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' 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 V03 N01: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report, January-March 1984. * Enforcement Staff. April 1984. 347pp. 8405220263. 24595:008.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March) 1984 and includes copies of letters, notices, and orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' 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 V03 N02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report, April-June 1984. * Enforcement Staff. July 1984. 363pp. 8408220308. 26202:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1984) and includes copies of letters, notices, and orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' 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 V03 N03: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report, July-September 1984. * Enforcement Staff. October 1984. 288pp. 8411160087. 27561:035.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July - September 1984) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' 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-0944 S01: SAFETY EVALUATION REPORT RELATED TO THE FULL-TERM OPERATING LICENSE FOR R.E. GINNA NUCLEAR POWER PLANT. Docket No. 50-244. (Rochester Gas And Electric Corporation) * Division of Licensing. October 1984. 40pp. 8411080517. 27402:237.

The Safety Evaluation Report for the full-term operating license application filed by Rochester Gas and Electric Corporation for the R.E. Ginna Nuclear Power Plant was prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission and issued in October 1983. This supplement includes the ACRS review and updates appropriate sections of the SER as required. The facility is located in Wayne County, Rochester, New York. Pending the favorable resolution of the items discussed in this report, the staff concludes that the facility can continue to be operated without endangering the health and safety of the public.

NUREG-0954 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CATAWBA NUCLEAR STATION, UNITS 1 AND 2. Docket Nos. 50-413 And 50-414. (Duke Power Company, et al) * Division of Licensing. June 1984. 134pp. 8407130504. 25578:156.

This report supplements the Safety Evaluation Report (NUREG-0954) issued in February 1983 and Supplement 1 issued in April 1983 by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by Duke Power Company, North Carolina Municipal Power Agency Number 1, North Carolina Membership Corporation, and Saluda River Electric Cooperative, Inc., as applicants and owners, for licenses to operate the Catawba Nuclear Station, Units 1 and 2 (Docket Nos. 50-413 and 50-414, respectively). The facility is located in York County, South Carolina, approximately 9.6 km (6 mi) north of Rock Hill and adjacent to Lake Wylie. This supplement provides more recent information regarding resolution or updating of some of the open and confirmatory issues and license conditions identified in the Safety Evaluation Report.

NUREG-0954 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CATAWBA NUCLEAR STATION, UNITS 1 AND 2. Docket Nos. 50-413 And 50-414. (Duke Power Company, et al) * Division of Licensing. July 1984. 140pp. 8408070009. 25954:334.

The report supplements the Safety Evaluation Report (NUREG-0954) issued in February 1983 by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission with respect to the application filed by Duke Power Company, North Carolina Municipal Power Agency Number 1, North Carolina Membership Corporation, and Saluda River Electric Cooperative, Inc., as applicants and owners, for licenses to operate the Catawba Nuclear Station, Units 1 and 2 (Docket Nos. 50-413 and 50-414, respectively). The facility is located in York County, South Carolina, approximately 9.6 km (6 mi) north of Rock Hill and adjacent to Lake Wylie. This supplement provides additional information supporting the license for fuel loading and precriticality testing for Unit 1.

NUREG-0954 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF CATAWBA NUCLEAR STATION, UNITS 1 AND 2. Docket Nos. 50-413 And 50-414. (Duke Power Company, et al) * Division of Licensing. December 1984. 220pp. 8412260120. 28071:039.

This report supplements the Safety Evaluation Report (NUREG-0954) issued in February 1983 by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by Duke Power Company, North Carolina Municipal Power Agency Number 1, North Carolina Membership Corporation, and Saluda River Electric Cooperative, Inc., as applicants and owners, for licenses to operate the Catawba Nuclear Station, Units 1 and 2 (Docket Nos. 50-413 and 50-414, respectively). The facility is located in York County, South Carolina, approximately 9.6 km (6 mi) north of Rock Hill and adjacent to Lake Wylie. This supplement provides additional information supporting the license for initial criticality and power ascension to full-power operation for Unit 1.

NUREG-0973: TECHNICAL SPECIFICATIONS FOR WATERFORD STEAM ELECTRIC STATION, UNIT NO. 3. Docket No. 50-382. (Louisiana Power And Light Company) HOFFMAN, D.R. Office of Nuclear Reactor Regulation, Director. December 1984. 483pp. 8501030334. 28199:001.

The Waterford, 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-0974: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION, UNITS 1 AND 2. Docket Nos. 50-352 And 50-353. (Philadelphia Electric Company) * Division of Licensing. April 1984. 320pp. 8404170288. 24089:096.

The information in this Final Environmental Statement is the second assessment of the environmental impact associated with the construction and operation of the Limerick Generating Station, Units 1 and 2. The first assessment was the Final Environmental Statement related to the construction of the facilities. The present assessment is the result of the NRC Staff review of the activities associated with the proposed operation of the station.

NUREG-0975 V02: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH DIVISION OF ENGINEERING TECHNOLOGY. Annual Report For FY 1983. * Division of Engineering Technology. March 1984. 312pp. 8404170012. 24092:183.

This report presents summaries of the research work performed during Fiscal Year 1983 by laboratories and organizations under contracts administered by the NRC's Materials Engineering 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-0976: REGULATION OF NATURALLY OCCURRING AND ACCELERATOR-PRODUCED RADIOACTIVE MATERIALS. An Update. BOLLING, L.A.; LUBENAU, J.O.; NUSSBAUMER, D.A. Office of State Programs, Director. October 1984. 19pp. 8410240480. 27147:330.

In 1977, NRC published a report (NUREG-0301) of a task force review of the need for, and feasibility of, the Federal government regulating naturally occurring and accelerator-produced radioactive materials (NARM). Since that time, the Federal regulatory role has not significantly changed but State calls for increased Federal involvement have continued. In 1983, a National Governor's Association report on the NRC Agreement State program recommended amendment of the Atomic Energy Act to authorize NRC regulation of these materials. Based on that recommendation, and with the cooperation of the Conference of Radiation Control Program Directors, Inc., NRC staff undertook a review of the current status of use and regulation of NARM. This report contains the results of that review.

NUREG-0978: MARK III LOCA-RELATED HYDRODYNAMIC LOAD DEFINITION. Generic Technical Activity B-10. Final Report. FIELDS, M. B.; KUDRICK, J. A. Division of Systems Integration (post 811005). August 1984. 100pp. 8409200450. 26628:031.

This report, prepared by the staff of the Office of Nuclear Reactor Regulation and its consultants at the Brookhaven National Laboratory, provides a discussion of LOCA-related suppression pool hydrodynamic loads in boiling water reactor (BWR) facilities with the Mark III pressure-suppression containment design. Its issuance completes NRC Generic Technical Activity B-10, "Behavior of BWR Mark III Containment."

On the basis of certain large-scale tests conducted between 1973 and 1979, the General Electric Company developed LOCA-related hydrodynamic load definitions for use in the design of the standard Mark III containment. The staff and its consultants have reviewed these load definitions and their bases and conclude that, with a few specified changes, the proposed load definitions provide conservative loading conditions.

The staff approved acceptance criteria for LOCA-related hydrodynamic loads are provided in Appendix C of this report.

NUREG-0978 FC: MARK III LOCA-RELATED HYDRODYNAMIC LOAD DEFINITION. Generic Technical Activity B-10. * Division of Systems Integration (post 811005). February 1984. 75pp. 8403220326. 22721:112.

See NUREG-0978 abstract.

NUREG-0979 S02: SAFETY EVALUATION REPORT RELATED TO THE FINAL DESIGN APPROVAL OF THE GESSAR II BWR/6 NUCLEAR ISLAND DESIGN. Docket No. 50-447. (General Electric Company) * Division of Licensing. November 1984. 131pp. 8412050032. 27808:272.

Supplement 2 to the Safety Evaluation Report (SER) for the application filed by General Electric Company for the final design approval for the GE BWR/6 nuclear island design (GESSAR II) has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. This report supplements the GESSAR II SER (NUREG-0979), issued in April 1983, summarizing the results of the staff's safety review of the GESSAR II BWR/6 nuclear island design. The review is carried out in accordance with the procedures for demonstrating the acceptability of the design for the severe-accident concerns described in draft NUREG-1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident issues in Nuclear Power Plant Regulation." Supplement 2 also provides more recent information regarding resolution or update of the confirmatory items and FDA-1 conditions identified in SSER 1. Subject to favorable resolution of the items discussed in this supplement, the staff concludes that the GESSAR II design satisfactorily address the severe accident concerns described in draft NUREG-1070.

NUREG-0980: NUCLEAR REGULATORY LEGISLATION. FOTIAS, A. Office of the Executive Legal Director. June 1984. 649pp. 8407130401. 25580:001.

NUREG-0980 is a compilation of nuclear regulatory legislation and other relevant material through the 97th 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. Sections of Title 5, United States Code, on Administrative Procedure are also included.

NUREG-0980 R01: NUCLEAR REGULATORY LEGISLATION. FOTIAS, A. Office of the Executive Legal Director. November 1984. 9pp. 8412110188. 27890:208.

Revision 1 to NUREG-0980 updates information on countries which have signed treaties and agreements on peaceful nuclear cooperation (through October, 1984). The U.S. Department of State has provided this current information on signatories to the Nuclear Non-Proliferation Treaty, the Treaty for the Prohibition of Nuclear Weapons in Latin America, and bilateral agreements between the United States and other countries for peaceful nuclear cooperation. NUREG-0980, a compilation of nuclear regulatory legislation (Atomic Energy Act of 1954, as amended; Energy Reorganization Act of 1974, as amended; Nuclear Waste Policy Act of 1982) and other relevant material (statutes and treaties on export licensing, nuclear non-proliferation, and environmental protection) through the 97th Congress, 2nd Session, has been prepared for use as a resource document, which the NRC intends to update regularly at the end of every Congress by inserting or deleting of material in the compilation.

NUREG-0985 R01: U.S. NUCLEAR REGULATORY COMMISSION HUMAN FACTORS PROGRAM PLAN. * Division of Human Factors Safety. September 1984. 62pp. 8409280095. 26735:253.

This document is the First Annual Revision to the NRC Human Factors Program Plan originally published August 1983.

The purpose of this document is to ensure that proper consideration is given to human factors in the planning, design, construction, operation and maintenance of nuclear facilities. The plan represents a systematic and comprehensive approach for addressing human factors concerns important to nuclear power plant safety in the FY-84 through FY-86 time frame.

The plan addresses the planning of seven major program elements: 1.0 Staffing and Qualifications, 2.0 Training, 3.0 Licensing Examinations, 4.0 Procedures, 5.0 Man-Machine Interface, 6.0 Management and Organization, and 7.0 Human Reliability.

Appendix (A) Program Element Schedules.

NUREG-0989: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION. Docket No. 50-458. (Gulf States Utilities Company, Cajun Electric Power Cooperative) * Division of Licensing. May 1984. 597PP. 8405310124. 24735:001.

The Safety Evaluation Report for the application filed by the Gulf States Utilities Company, as applicant and owner, for a license to operate the River Bend Station (Docket No. 50-458) has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near St. Francisville, Louisiana. Subject to favorable resolution of the items discussed in this report, the NRC staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-0989 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF RIVER BEND STATION. Docket No. 50-458. (Gulf States Utilities Company, Cajun Electric Power Cooperative) * Division of Licensing. October 1984. 49pp. 8411200024. 27586:250.

Supplement No. 1 to the Safety Evaluation Report for the application filed by Gulf States Utilities Company as applicant and for itself and Cajun Electric Power Cooperative, as owners, for a license to operate River Bend Station has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in West Felicina Parish, near St. Francisville, Louisiana. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

NUREG-0991 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION, UNITS 1 & 2. DOCKET Nos. 50-352 And 50-353. (Philadelphia Electric Company) * Division of Licensing. October 1984. 128pp. 8411090445. 27423:001.

In August 1983 the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0991) regarding the application of the Philadelphia Electric Company (the applicant) for licenses to operate the Limerick Generating Station, Units 1 and 2, located on a site in Montgomery and Chester Counties, Pennsylvania. Supplement 1 to NUREG-0991 was issued in December 1983 and addressed several outstanding issues. Supplement 1 also included the interim report of the Advisory Committee on Reactor Safeguards and the staff's initial response to the comments made in the report. This supplement to NUREG-0991 addresses further issues that require resolution and closes them out.

NUREG-0991 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION, UNITS 1 And 2. Docket Nos. 50-352 And 50-353. (Philadelphia Electric Company) * Division of Licensing. October 1984. 156pp. 8411210188. 27619:087.

In August 1983 the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0991) regarding the application of the Philadelphia Electric Company (the applicant) for licenses to operate the Limerick Generating Station, Units 1 and 2 located on a site in Montgomery and Chester Counties, Pennsylvania. Supplement No. 1 to NUREG-0991 was issued in December 1983 and addressed several outstanding issues. Supplement No. 1 also contains the comments made by the Advisory Committee on Reactor Safeguards in its report dated October 18, 1983. Supplement No. 2 was issued in

October 1984 and addressed fourteen outstanding and fifty-three confirmatory issues and closed them out. This Supplement No. 3 to NUREG-0991 addresses the remaining issues that require resolution before issuance of the operating license for Unit 1 and closes them out.

NUREG-0993 R01: REGULATORY ANALYSIS FOR USI A-1, "WATER HAMMER." SERKIZ, A.W. Division of Safety Technology. March 1984. 30pp. 8404090143. 22947:309.

NUREG-0993, Revision 1 is the staff's regulatory analysis dealing with the resolution of the Unresolved Safety Issue A-1, Water Hammer. This report contains the value-impact analysis for this issue, public comments received, and staff response, or action taken, in response to those comments. The staff's technical findings regarding water hammer in nuclear power plants are contained in NUREG-0927.

NUREG-1007 S01: SAFETY EVALUATION REPORT RELATED TO THE LICENSE RENEWAL AND POWER INCREASE FOR THE NATIONAL BUREAU OF STANDARDS REACTOR. Docket No. 50-184. (National Bureau Of Standards) * Division of Licensing. March 1984. 18pp. 8404120323. 24035:133.

Supplement 1 to the Safety Evaluation Report (SER) related to the renewal of the operating license and for a power increase (10 MWt to 20 MWt) for the research reactor at the National Bureau of Standards (NBS) facility has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The reactor facility is located in Montgomery County, Maryland. This supplement reports on the review of the licensee's emergency plan, which had not been reviewed at the time the Safety Evaluation Report (NUREG-1007) was published, and the review of the NBS application by the Advisory Committee on Reactor Safeguards, which was completed subsequent to the publication of the SER.

NUREG-1012: TECHNICAL SPECIFICATIONS FOR SHOREHAM NUCLEAR POWER STATION, UNIT No. 1. Docket No. 50-322. (Long Island Lighting Company) HOFFMAN, D.R. Division of Licensing. December 1984. 400pp. 8412310128. 28151:082.

The Shoreham, Unit 1, Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to set forth the 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-1015: STATE SURVEILLANCE OF RADIOACTIVE MATERIAL TRANSPORTATION. Final Report. SALOMON, S.N. Office of State Programs, Director. February 1984. 72pp. 8403230190. 22759:048.

The main objective of this final report on the State surveillance of the transportation of radioactive material (RAM) is to suggest the most cost-effective inspection areas where enforcement actions might be taken by States during their participation in the State Hazardous Materials Enforcement Development (SHMED) Program of the U.S. Department of Transportation (DOT). On the basis of the lessons learned from the surveillance program, these actions are enforcement at low-level radioactive burial sites by means of civil penalties and site use suspension; enforcement at airports and at terminals that forward freight; and enforcement of courier companies. More effective and efficient enforcement can be achieved through instrumented police patrol cars and remote surveillance because they require the least amount of time of enforcement personnel. Also, there is a strong relationship between effective emergency response and enforcement because the appropriate shipping papers, placarding and knowledge of appropriate emergency response procedures lead to improved emergency response. These lessons originate from a ten-State surveillance program from 1977 through 1981 jointly sponsored by the U.S. Nuclear Regulatory Commission (NRC) and DOT. States give recommendations in the categories of education, training expanded surveillance, coordination and enforcement. Topics of special interest covered include low-level radioactive waste disposal sites, airports, cargo terminals, highways, ports, and accidents and incidents. The relationship to other studies, the status of the SHMED Program, a synopsis of State RAM surveillance reports, and NRC/DOT expenditures are given. Also, relevant laws and regulations and a selected bibliography are included.

NUREG-1020LD V01: GPU V. B&W LAWSUIT REVIEW AND ITS EFFECT ON TMI-1.General Public Utilities Corporation,et al. v. The Babcock & Wilcox Company,et al.Three Mile Island Nuclear Station,Unit 1, Docket 50-289. * Office of Nuclear Reactor Regulation, Director. June 1984. 152pp. 8407130502. 25579:089.

This report documents a review by the Nuclear Regulatory Commission (NRC) staff of the General Public Utilities v. Babcock & Wilcox lawsuit record to assess whether any of the staff's previous conclusions or their principal bases presented at the Three Mile Island Unit 1 (TMI-1) restart hearing, supporting restart of TMI-1, should be amended in light of the information contained in the lawsuit record. Details of the lawsuit record are provided in the appendices contained in Volume 2 of this report.

NUREG-1020LD V02: GPU V. B&W LAWSUIT REVIEW AND ITS EFFECT ON TMI-1.General Public Utilities Corporation,et al. v. The Babcock & Wilcox Company,et al.Three Mile Island Nuclear Station,Unit 1, Docket 50-289. * Office of Nuclear Reactor Regulation, Director. June 1984. 875pp. 8407130415. 25576:001.

See NUREG-1020LD,V01 abstract.

NUREG-1022 S01: LICENSEE EVENT REPORT SYSTEM.Description Of System And Guidelines For Reporting. HEBDON,F.J. Director's Office. February 1984. 58pp. 8403070121. 22561:308.

On July 26, 1983, the Commission published in the Federal Register a final rule (10 CFR 50.73) that modified and codified the Licensee Event Report (LER) system. The rule became effective on January 1, 1984. In September 1983, the NRC published NUREG-1022 which provides supporting information and guidance that is of interest to persons responsible for the preparation and review of LERs. The information contained in NUREG-1022 includes: (1) a brief description of how LERs are analyzed by the NRC, (2) a restatement of the guidance contained in the Statement of Consideration that accompanied publication of the LER rule, (3) a set of examples of potentially reportable events with staff comments on the actual reportability of each event, (4) guidance on how to prepare an LER, including the LER forms, and (5) guidance on submittal of LERs. Subsequently, during the period from October 25, 1983 to November 16, 1983, the NRC staff held five regional meetings to discuss the scope and content of the LER rule with utility and NRC regional representatives. During these meetings, numerous questions arose and were answered. This supplement to NUREG-1022 contains a summary of the questions asked and the answers given.

NUREG-1026: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF BRAIDWOOD STATION, UNITS 1 AND 2 Docket Nos. STN 50-456 And STN 50-457. (Commonwealth Edison Company) * Division of Licensing. June 1984. 276pp. 8407180017. 25682:126.

The information in this statement is the second assessment of the environmental impact associated with the construction and operation of the Braidwood Station, Units 1 and 2, located in northeastern Illinois within Reed Township, Will County, Illinois. The first assessment was the Final Environmental Statement related to construction issued in July 1974 prior to issuance of the Braidwood Construction Permits. The present assessment is the result of the NRC staff review of the activities associated with the proposed operation of the plant.

NUREG-1028: RUPTURED CESIUM-137 WELL-LOGGING SOURCE AT SHELWELL SERVICES, INC., HEBRON, OHIO. AXELSON, W. Division of Radiological & Materials Safety Programs. April 1984. 135pp. 8405220266. 24601:162.

This U.S. Nuclear Regulatory Commission report documents the circumstances surrounding the September 13, 1983, cesium-137 sealed source rupture incident at Shelwell Services, Inc., facility in Hebron, Ohio. It focuses on the period from approximately 4:00 p.m. (EDT) on September 13, 1983, when the source ruptured, to October 5, 1983, when the radiological emergency response aspects of the event were concluded. Information outside these periods is recounted as necessary. The incident resulted in radiation doses to two licensee employees that exceeded the regulatory limits for whole-body and extremity exposures, and contamination of the licensee's employees, families, and friends. The emergency response required the combined efforts of NRC, the U.S. Department of Energy, and state personnel. The report describes the factual information and significant findings associated with the event and, thereby, provides a data base for subsequent detailed analyses and recommendations by various NRC offices.

NUREG-1029: A COMPUTER CODE FOR GENERAL ANALYSIS OF RADON RISKS (GARR). GINEVAN, M. Division of Radiation Programs & Earth Sciences (post 840429). September 1984. 96pp. 8410120006. 26986:006.

Evaluating the level of lung cancer risk associated with a given level of radon-daughter exposure is a complex matter. There is the question of whether one's risk assessment should apply absolute risk models or relative risk models and, even when a general model form has been selected, there are decisions as to the exact form of risk projection, the appropriate method of accounting exposure over time, and how much a personal habit such as smoking can modify risk. This document presents a computer model for general analysis of radon risks that allows the user to specify a large number of possible models with a small number of simple commands. The model is written in a version of BASIC which conforms closely to the American National Standards Institute (ANSI) definition for minimal BASIC and thus is readily modified for use on a wide variety of computers and, in particular, microcomputers.

NUREG-1031: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3. Docket No. 50-423. (Northeast Nuclear Energy Company) * Division of Licensing. July 1984. 391pp. 8408160262. 26121:001.

This report provides the results of the NRC staff review of Northeast Nuclear Energy Company's application for a license to operate the Millstone Nuclear Power Plant, Unit No. 3. The facility is located in Waterford Township, New London County, Connecticut. Subject to favorable resolution of the items discussed in the Safety Evaluation Report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-1038 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1. Docket No. STN 50-400. (Carolina Power And Light Company, North Carolina Eastern Municipal Power Agency) * Division of Licensing. June 1984. 52pp. 8407180053. 25665:341.

Supplement No. 1 to the Safety Evaluation Report for the application filed by Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency for a license to operate the Shearon Harris Nuclear Power Plant, Unit 1 (Docket No. 50-400), located in Wake and Chatham Counties, North Carolina, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides more recent information regarding resolution of some of the open items identified in the Safety Evaluation Report. This supplement also provides and discusses the recommendations of the Advisory Committee on Reactor Safeguards in its report on Shearon Harris, dated January 16, 1984.

NUREG-1039: REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION SAFETY RESEARCH PROGRAM FOR FISCAL YEAR 1985. * ACRS - Advisory Committee on Reactor Safeguards. February 1984. 74pp. 8403070385. 22561:234.

Public Law 95-209 includes a requirement that the Advisory Committee on Reactor Safeguards submit an annual report to Congress on the safety research program of the Nuclear Regulatory Commission. This report presents the results of the ACRS review and evaluation of the NRC safety research program for Fiscal Year 1985. The report contains a number of comments and recommendations.

NUREG-1040: FY 1985 BUDGET ESTIMATES. * Division of Budget & Analysis. January 1984. 68pp. 8402210188. 22346:072.

This report contains the fiscal year budget justifications to Congress. The budget estimates for salaries and expenses for fiscal year 1985 provide for obligations of $468,200,000 to be funded in total by a new appropriation.

NUREG-1042: TECHNICAL SPECIFICATIONS FOR SUSQUEHANNA STEAM ELECTRIC STATION, UNIT NO. 2. Docket No. 50-388. (Pennsylvania Power And Light Company) HOFFMAN, D. R. Division of Licensing. March 1984. 519pp. 8404100519. 22978:001.

Susquehanna Steam Electric 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-1043: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE TRAINING AND RESEARCH REACTOR AT THE UNIVERSITY OF MARYLAND. Docket No. 50-166. (University Of Maryland) * Division of Licensing. March 1984. 73pp. 8404160225. 24065:294.

This Safety Evaluation Report for the application filed by the University of Maryland, (UMD) for a renewal of operating license R-70 to continue to operate a training and research reactor facility 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 Maryland and is located at a site in College Park, Prince Georges County, Maryland. The staff concludes that this training reactor facility can continue to be operated by UMD without endangering the health and safety of the public.

NUREG-1044: EVALUATION OF THE NEED FOR A RAPID DEPRESSURIZATION CAPABILITY FOR CE PLANT. MARSH,L.; LIANG,C. Division of Systems Integration (post 811005). December 1984. 137pp. 8501070403. 28225:020.

This report documents the NRC staff evaluation of the need for providing a rapid primary system depressurization capability, in particular by using a power-operated relief valve(s) (PORVs), in the current 3410-MWt and 3800-MWt classes of plants designed by Combustion Engineering (CE). The staff reviewed the responses of licensees, applicants, and vendors to staff questions, supplemented by independent analyses by the staff and its contractors. The staff review led to the conclusion that, on the basis of risk reduction and cost/benefit considerations, no overwhelming benefit would result from requiring the installation of PORVs in CE plants that currently do not have them. However, when other unquantifiable considerations regarding the potential benefits of a PORV are factored into the evaluation, it appears that more substantial benefits could be realized. Given the more comprehensive studies currently under way to resolve the generic unresolved safety issue, USI A-45, Decay Heat Removal Reliability, the staff concludes that the decision regarding PORVs for these CE plants should be deferred and incorporated into the technical resolution of USI A-45. Resolution of USI A-45 will also include the effects of residual risks due to fires, floods, earthquakes, and sabotage.

NUREG-1045: GUIDANCE ON THE APPLICATION OF COMPENSATORY SAFEGUARDS MEASURES FOR POWER REACTOR LICENSEES. BLUMENTHAL,H.S. Division of Safeguards. January 1984. 10pp. 8402170517. 22320:024.

This NUREG provides criteria and examples to be used by power reactor licensees and the NRC staff in determining the acceptability of compensatory safeguards measures.

NUREG-1046 DRFT: DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES IN THE UNSATURATED ZONE: TECHNICAL CONSIDERATIONS. Draft Report For Comment. OSTROWSKI,N.; NICHOLSON,T.J.; ALEXANDER,D.H.; et al. Division of Health, Siting & Waste Management. February 1984. 43pp. 8402240373. 22380:264.

The Nuclear Regulatory Commission (NRC) recently published final regulations related to the disposal of high-level radioactive wastes in geologic repositories (46 FR 13971 and 48 FR 28194). These regulations were limited to geologic repositories in the saturated zone since earlier Department of Energy program plans emphasized fully saturated geologic media. However, the Department of Energy recently has initiated preliminary studies in unsaturated geologic media, and requested that NRC reexamine and modify 10 CFR Part 60 so that it will apply to all geologic media. This request also was made by several commenters on the proposed 10 CFR Part 60 technical criteria. NRC has considered this request and has proposed amendments to ensure that the provisions of 10 CFR Part 60 are equally applicable to waste disposal in either the saturated or unsaturated zone. In reaching this decision, NRC conducted a detailed study of the arguments presented by the public commenters, the issues involved in disposal within the unsaturated zone, and the relative attributes and concerns associated with disposal in the unsaturated zone. These points are discussed in this document. The NRC staff has concluded that disposal within the unsaturated zone is possible, provided that the site and the design of the geologic repository are capable of meeting the performance objectives of 10 CFR Part 60.

NUREG-1048: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION. Docket No. 50-354. (Public Service Electric and Gas Company) * Division of Licensing. October 1984. 704pp. 8411210181. 27621:055.

The Safety Evaluation Report for the application filed by Public Service Electric and Gas Company, as applicant, for a license to operate the Hope Creek Generating Station (Docket No. 50-354), has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Salem County, New Jersey. Subject to favorable resolution of the items discussed in this report, the NRC staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-1049: ENVIRONMENTAL IMPACT APPRAISAL FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-42. Docket No. 70-27. (Babcock & Wilcox Company, Naval Nuclear Fuel Division) * Division of Fuel Cycle & Material Safety. March 1984. 87pp. 8403270267. 22787:235.

This Environmental Impact Appraisal is issued by the U.S. Nuclear Regulatory Commission in response to an application submitted by Babcock & Wilcox Company, Naval Nuclear Fuel Division, for renewal of Special Nuclear Material License No. SNM-42.

NUREG-1050: PROBABILISTIC RISK ASSESSMENT (PRA) REFERENCE DOCUMENT. Final Rept. * Division of Risk Analysis & Operations (post 840429). September 1984. 230pp. 8410120005. 26984:001.

The Commission's Safety Goal Policy Statement in NUREG-0880, Rev. 1, directs the staff "...to collect available information on PRA studies and prepare a reference document that describes the current status of knowledge concerning the risk of plants licensed in the U.S." The document discusses the purpose and content of a PRA and identifies the PRAs and other probabilistic studies performed to date. It then discusses the level of development and uncertainties associated with the various elements of PRA methodology as well as those generic insights derived from studies performed. Finally, potential uses of PRA in regulation are evaluated.

NUREG-1050 DRFT: PROBABILISTIC RISK ASSESSMENT (PRA): STATUS REPORT AND GUIDANCE FOR REGULATORY APPLICATION. Draft Report For Comment. * Division of Risk Analysis & Operations (post 840429). February 1984. 337pp. 8402130612. 22215:001.

See NUREG-1050 abstract.

NUREG-1051: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF KANSAS. Docket No. 50-148. (University Of Kansas) * Division of Licensing. May 1984. 68pp. 8406060419. 24847:182.

This Safety Evaluation Report for the application filed by the University of Kansas (KU) for a renewal of Operating License R-78 to continue to operate the KU 250-kw open-pool training 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 Kansas and is located on the KU campus in Lawrence, Douglas County, Kansas. The staff concludes that the reactor facility can continue to be operated by KU without endangering the health and safety of the public.

NUREG-1052: FEDERAL/STATE COOPERATION IN THE LICENSING OF A NUCLEAR POWER PROJECT. A Joint Process Between The U.S. Nuclear Regulatory Commission And The Washington State Energy Facility Site Evaluation Council. * Office of Nuclear Reactor Regulation, Director. May 1984. 53pp. 8406230318. 25131:265.

This report summarizes and documents a joint environmental review and licensing process established between the U.S. Nuclear Regulatory Commission (NRC) and the Washington State Energy Facility Site Evaluation Council (EFSEC) in 1980-83 for the Skagit/Hanford Nuclear project (S/HNP). It documents the agreements made between the agencies to prepare a joint environmental impact statement responsive to the requirements of the National Environmental Policy Act 1969 (NEPA) and the Washington State Environmental Policy Act. These agreements also established protocol to conduct joint public evidentiary hearings on matters of mutual jurisdiction, thereby reducing the duplication of effort and increasing the efficiency of the resources of Federal and State governments and other entities involved in the process. This report may provide guidance and rationale to licensing bodies that may wish to adopt some of the procedures discussed in the report in the event that they become involved in the licensing of a nuclear power plant project. The history of the S/HNP and of the agreement processes are discussed. Discussions are provided on implementing the joint review process. A separate section is included which presents independent evaluations of the process by the applicant, NRC, and EFSEC. Cooperating Federal agencies in the environmental review included the U.S. Department of Energy, the Bonneville Power Administration, and the Bureau of Reclamation.

NUREG-1054: SIMPLIFIED ANALYSIS FOR LIQUID PATHWAY STUDIES. CODELL, R.B. Division of Engineering. August 1984. 106pp. 8408300292. 26331:001.

The analysis of the potential contamination of surface water via groundwater contamination from severe nuclear accidents is routinely calculated during licensing reviews. This analysis is facilitated by the methods described in this report, which is codified into a BASIC language computer program, SCREENLP. This program performs simplified calculates population doses to potential users of the contaminated water irrespective of possible mitigation methods. The results are then compared to similar analyses performed using data for the generic sites in NUREG-0400, "Liquid Pathway Generic Study", to determine if the site being investigated would pose any unusual liquid pathway hazards.

NUREG-1055: IMPROVING QUALITY AND THE ASSURANCE OF QUALITY IN THE DESIGN AND CONSTRUCTION OF COMMERCIAL NUCLEAR POWER PLANTS. A Report To Congress. ALTMAN, W. ; ANKRUM, T. ; BRACH, W. QA Branch. May 1984. 524pp. 8406010533. 24763:001.

At the request of Congress, NRC conducted a study of existing and alternative programs for improving quality and the assurance of quality in the design and construction of commercial nuclear power plants. A primary focus of the study was to determine the underlying causes of major quality-related problems in the construction of some nuclear power plants and the untimely detection and correction of these problems. The study concluded that the root cause for major quality-related problems was the failure or inability of some utility managements to effectively implement a management system that ensured adequate control over all aspects of the project. These management shortcomings arose in part from inexperience on the part of some project teams in the construction of nuclear power plants. As a corollary, NRC's past licensing and inspection practices did not adequately screen construction permit applicants for overall capability to manage or provide effective management oversight over the construction project. The study recommends a number of improvements in industry and NRC programs.

NUREG-1056: REPORT ON U.S.-JAPAN 1983 MEETINGS ON STEAM GENERATORS. * Office of Nuclear Reactor Regulation, Director. April 1984. 124pp. 8404240014. 24189:131.

This is a report on a trip to Japan by personnel of the U.S. Nuclear Regulatory Commission in 1983 to exchange information on steam generators of nuclear power plants. Steam generators of Japanese pressurized water reactors have experienced nearly all of the forms of degradations that have been experienced in U.S. recirculating-type steam generators, except for denting and pitting. More tubes have been plugged per year of reactor operation in Japanese than in U.S. steam generators, but much of the Japanese tube plugging is preventative rather than the result of leaks experienced. The number of leaks per reactor year is much smaller for Japanese than for U.S. steam generators. No steam generators have been replaced in Japan while several have replaced in the U.S. The Japanese experience may be related to their very stringent inspection and maintenance programs for steam generators.

NUREG-1058: TECHNICAL SPECIFICATIONS FOR CALLAWAY PLANT, UNIT NO. 1. Docket No. STN 50-483. (Union Electric Company) ANDERSON, F. D. Division of Licensing June 1984. 490pp. 8407020225. 25230:206.

The Calloway 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 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-1058 R01: TECHNICAL SPECIFICATIONS FOR CALLAWAY PLANT, UNIT NO. 1. Docket No. 50-483. (Union Electric Company) ANDERSON, F. D. Division of Licensing. October 1984. 491pp. 8411200386. 27598:252.

See NUREG-1058 abstract.

NUREG-1059: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE UNION CARBIDE SUBSIDIARY B, INC. RESEARCH REACTOR. Docket No. 50-54. (Union Carbide Corporation) * Division of Licensing. June 1984. 98pp. 8407180046. 25683:041.

This Safety Evaluation Report for the application filed by the Union Carbide Subsidiary B, Inc. (UNC) for a renewal of operating license R-81 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 Union Carbide Subsidiary B, Inc. and is located in the City of Tuxedo, Orange County, New York. The staff concludes that the reactor facility can continue to be operated by UNC without endangering the health and safety of the public.

NUREG-1061 V01: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE. Volume 1: Investigation And Evaluation Of Stress Corrosion Cracking In Piping Of Boiling Water Reactor Plants. * Piping Review Committee. August 1984. 400pp. 8409260636. 26700:045.

Severe intergranular stress corrosion cracking (IGSCC) of the recirculation piping system in several boiling water reactors occurred during 1982-1983. A Task Group on Pipe Cracks was established by the U.S. Nuclear Regulatory Commission with the broad charter of developing an integrated program to deal with the entirety of the stress corrosion cracking problem. This report presents specific conclusions and recommendations that are tied closely to relevant regulatory documents so that necessary changes can be implemented.

This report covers aspects such as the causes and descriptions of IGSCC phenomena; current status of pipe cracking in BWR's; nondestructive evaluations of piping welds; inspection of piping for IGSCC; decisions and criteria for replacement, review of continued operation without repair; risks related to the presence of IGSCC; and a value-impact assessment of IGSCC.

NUREG-1061 V03: REPORT TO U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE. Volume 3: Evaluation Of Potential For Pipe Breaks. * Piping Review Committee. November 1984. 200pp. 8412100188. 27849:092.

The Executive Director for Operations of the U. S. Nuclear Regulatory Commission (NRC) requested that a comprehensive review be made of NRC requirements in the area of nuclear power plant piping. In response to this request an NRC Piping Review Committee was formed. The activities of this review committee were divided into four tasks handled by appropriate task groups, namely: Pipe Crack Task Group, Seismic Design Task Group, Pipe Break Task Group, and Dynamic Load/Load Combination Task Group. This report was prepared by the Pipe Break Task Group and deals with the potential for pipe breaks and recommends modifications to the existing position. Specifically, this report contains the Task Group's recommendations for application of the leak-before-break (LBB) approach in the NRC licensing process. The LBB approach means the application of fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience doubled-ended ruptures or their equivalent as longitudinal or diagonal splits.

NUREG-1061 V04: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE. Volume 4: Evaluation Of Other Loads And Load Combinations. * Piping Review Committee. December 1984. 387pp. 8501070405. 28224:001.

This report deals with six topical areas: Event Combinations, Response Combinations, Stress Limits and Dynamic Allowables, Water Hammer Loadings, Relief Valve Opening and Closing Loads and Piping Vibration Loads. Recommendations prepared by the staff were based on consultant position papers and industry comments and treat revisions to present NRC requirements and directions for future research. Foreign information was obtained from sources in Belgium, Canada, France, Italy, Japan, Sweden and the Federal Republic of Germany. In addition, the report contains qualitative value impacts for the proposed recommendations. This report was developed over a period of approximately one year, and partially fulfills and complies with the requirements of the July 13, 1983 memorandum from the Directors of the Offices of Nuclear Reactor Regulation and Nuclear Regulatory Research to NRC's Executive Director for Operations.

NUREG-1062: DOSE CALCULATIONS FOR SEVERE LWR ACCIDENT SCENARIOS. MARGULIES, T.S.; MARTIN, J.A. Division of Risk Analysis & Operations (post 840429). May 1984. 227pp. 8406230205. 25132:045.

This report presents a set of precalculated doses based on a set of postulated accident releases and intended for use in emergency planning and emergency response. Doses were calculated for the PWR (Pressurized Water Reactor) accident categories of the Reactor Safety Study (WASH-1400) using the CRAC (Calculations of Reactor Accident Consequences) code. Whole body and thyroid doses are presented for a selected set of weather cases. For each weather case these calculations were performed for various times and distances including three different dose pathways--cloud (plume) shine, ground shine and inhalation. During an emergency this information can be useful since it is immediately available for projecting offsite radiological doses based on reactor accident sequence information in the absence of plant measurements of emission rates (source terms). It can be used for emergency drill scenario development as well.

NUREG-1063: STEAM GENERATOR OPERATING EXPERIENCE UPDATE 1982-1983. FRANK, L. Division of Engineering. June 1984. 50pp. 8406270122. 25173:231.

This report is a continuation of earlier reports by the staff addressing pressurized water reactor steam generator operating experience. NUREG-0886, "Steam Generator Tube Experience," published in February 1982 summarized experience in domestic and foreign plants through December 1981. This report summarizes steam generator operating experience in domestic plants for the years 1982 and 1983. Included are new problems encountered with secondary-side loose parts, sulfur-induced stress-assisted corrosion cracking, and flow-induced vibrational wear in the new preheater design steam generators. The status of Unresolved Safety Issues A3, A4, and A5 is also discussed.

NUREG-1064: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION, UNIT 3. DOCKET NO. 50-423. (NORTHEAST

NUCLEAR ENERGY COMPANY, et al) * Division of Licensing. July 1984. 335pp. 8408010148. 25871:012.

The information in this statement is the second assessment of the environmental impact associated with the construction and operation of the Millstone Nuclear Power Station, Unit No. 3, located in Waterford Township, New London County, Connecticut. The first assessment was the Final Environmental Statement related to construction issued in February 1974 prior to issuance of the Millstone Construction Permit. The present assessment is the result of the NRC staff review of the activities associated with the proposed operation of the plant.

NUREG-1064: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3. Docket No 50-423. (Northeast Nuclear Energy Company) * Division of Licensing. December 1984. 350pp. 8501080462. 28270:001.

The information in this Final Environmental Statement is the second assessment of the environmental impact associated with the construction and operation of the Millstone Nuclear Power Station, Unit No. 3, located in Waterford Township, New London County, Connecticut. The first assessment was the Final Environmental Statement related to construction issued in February 1974 prior to issuance of the Millstone Unit 3 Construction Permit. The Draft Environmental Statement related to operation was issued in July 1984. The present assessment is the result of the NRC staff's review of the activities associated with the proposed operation of the plant, and includes the staff response to comments on the Draft Environmental Statement.

NUREG-1065: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS. EMEIGH, C. W.; GUNDERSEN, G. E.; WITHEE, C. J. Division of Safeguards. May 1984. 49pp. 8406080305. 24877:126.

This report documents a standard format suggested by the NRC for use in preparing fundamental nuclear material control plans as required by the Low Enriched Uranium Reform Amendments (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 Amendments. 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 equal level of completeness and detail may use their own format.

NUREG-1066: COMPARISON OF IMPLEMENTATION OF SELECTED TMI ACTION PLAN REQUIREMENTS ON OPERATING PLANTS DESIGNED BY BABCOCK AND WILCOX. THOMA, J. O.; HERNAN, R.; KADAMBI, N. P.; et al. Division of Licensing. May 1984. 186pp. 8406020464. 24800:001.

This report provides the results of a study conducted by the U.S. Nuclear Regulatory Commission staff to compare the degree to which eight Babcock and Wilcox (B&W) designed licensed nuclear power plants have complied with the requirements in NUREG-0737, "Clarification of TMI Action Plan Requirements". The eight licensed operating plants examined are as follows: Arkansas Nuclear One Unit 1 (ANO-1), Crystal River Unit 3, Davis Besse, Oconee Units 1, 2 and 3, Rancho Seco, and

Three Mile Island Unit 1 (TMI-1). The purpose of this audit was to establish the progress of the TMI-1 licensee, General Public Utilities (GPU) Nuclear Corporation, in completing the long-term requirements in NUREG-0737 relative to the other B&W licensees examined.

NUREG-1068: REVIEW INSIGHTS ON THE PROBABILISTIC RISK ASSESSMENTS FOR THE LIMERICK GENERATING STATION, UNIT 1 AND 2. CHELLIAH, E. Division of Safety Technology. August 1984. 126pp. 8408290158. 26309:151.

In recognition of the high population density around the Limerick Generating Station site and the proposed power level, the Philadelphia Electric Company, in response to NRC staff requests, conducted and submitted between March 1981 and November 1983 a probabilistic risk assessment (PRA) on internal event contributors and a severe accident risk assessment on external event contributors to assess risks posed by operation of the plant. The applicant has developed perspectives using PRA models on the risk profile of the Limerick plant and has altered the plant design to reduce accident vulnerabilities identified in these PRAs. The staff's review of the Limerick PRA has particularly emphasized the dominant accident sequences and the resulting insights into demonstration of compliance with regulatory requirements, unique design features and major plant vulnerabilities to assess the need for any additional measures to further improve the safety of the LGS. The staff's review insights and PRA safety review conclusions are presented in this report.

NUREG-1069: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE GENERAL ELECTRIC-NUCLEAR TEST REACTOR(GE-NTR). DOCKET NO. 50-73. (General Electric Company) * Division of Licensing. September 1984. 89pp. 8410180222. 27045:201.

This Safety Evaluation Report for the application filed by the General Electric Corporation for a renewal of operating license R-33 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 General Electric Corporation and is located in Pleasanton, California. The staff concludes that the reactor facility can continue to be operated by GE without endangering the health and safety of the public.

NUREG-1071: ENVIRONMENTAL IMPACT APPRAISAL FOR RENEWAL OF SOURCE MATERIAL LICENSE NO. SUB-526. Docket No. 40-3392. (Allied Chemical Company, UF6 Conversion Plant) * Division of Fuel Cycle & Material Safety. May 1984. 110pp. 8405310034. 24737:072.

This Environmental Impact Appraisal is issued by the U.S. Nuclear Regulatory Commission in response to an application by Allied Chemical Company for renewal of Source Material License No. SUB-526.

NUREG-1072: TECHNICAL SPECIFICATIONS FOR CATAWBA NUCLEAR STATION, UNIT 1. Docket No. 50-413. (Duke Power Company) ANDERSON, F. D. Division of Licensing. July 1984. 525pp. 8408130011. 26039:001.

The Catawba Nuclear Station, 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 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-1073: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF RIVER BEND STATION. Docket No. 50-458. (Gulf States Utilities Company & Cajun Electric Power Cooperative) * Division of Licensing. July 1984. 249pp. 8408220325. 26199:001.

This Draft Environmental Statement contains the second assessment of the environmental impact associated with the operation of River Bend Station, 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. This statement examines the environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs. Comments on this statement should be filed no later than 45 days after the date on which the Environmental Protection Agency notice of availability of this statement is published in the Federal Register.

NUREG-1074: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION. Docket No. 50-354. (Public Service Electric And Gas Co And Atlantic City Electric Co) * Division of Licensing. June 1984. 227pp. 8407110001. 25544:001.

The Draft Environmental Statement related to the operation of Hope Creek Generating Station, located in Salem County, New Jersey, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The statement reports on staff's review of the environmental and socio-economic impacts of plant operation. Comments received on this document will be included and addressed in the Final Environmental Statement.

NUREG-1074: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF HOPE CREEK GENERATING STATION. Docket No. 50-354. (Public Service Electric And Gas Company And Atlantic City Electric Company) * Division of Licensing. December 1984. 277pp. 8412310301. 28150:166.

This Final Environmental Statement contains the second assessment of the environmental impact associated with the operation of the Hope Creek Generating Station 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. This statement examines the environmental impacts, environmental consequences and mitigating actions, and environmental and economic benefits and costs associated with station operation. Land use and terrestrial and aquatic ecological impacts will be small. No operational impacts to historic and archeological sites are anticipated. The effects of routine operations, energy transmission, and periodic maintenance of rights-of-way and transmission facilities should not jeopardize any populations of endangered or threatened species. No significant impacts are anticipated from normal operational releases of radioactivity. The risk of radiation exposure associated with accidental release of radioactivity is very low. Socio-economic impacts of the project are anticipated to be minimal. The action called for is the issuance of an operating license for Hope Creek Generating Station.

NUREG-1075: DECENTRALIZATION OF OPERATING REACTOR LICENSING REVIEWS. NRR Pilot Program. HANNON, J.L. Division of Licensing. July 1984. 26pp. 8408080370. 25981:102.

This report, which has incorporated comments received from the Commission and ACRS, describes the program for decentralization of selected operating reactor licensing technical review activities. The 2-year pilot program will be reviewed to verify that safety is enhanced as anticipated by the incorporation of prescribed management techniques and application of resources. If the program fails to operate as designed, it will be terminated.

The 2-year pilot program will be limited to two operating power plants in each of three regions and will be implemented to: (1) test the method of selecting licensing actions for technical review in the regions, (2) evaluate predicted improvements in the effectiveness of licensing and inspection programs, and (3) verify that safety is enhanced (as anticipated) by incorporating prescribed management techniques and applying regional resources to this technical review function.

NUREG-1077: ENVIRONMENTAL IMPACT APPPRAISAL FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-21.Docket No. 70-25. (Energy Systems Group,Rockwell International Corporation) * Division of Fuel Cycle & Material Safety. June 1984. 121pp. 8406280455. 25195:047.

This Environmental Impact Appraisal is issued by the U.S. Nuclear Regulatory Commission in response to an application by Energy Systems Group, Rockwell International Corporation, for renewal of Special Nuclear Material (SNM) License No. SNM-21.

NUREG-1078: ENVIRONMENTAL IMPACT APPRAISAL FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO. SNM-1097.Docket No. 70-1113. (General Electric Company,Wilmington Manufacturing Department) * Division of Fuel Cycle & Material Safety. June 1984. 84pp. 8407020195. 25275:094.

This Environmental Impact Appraisal is issued by the U.S. Nuclear Regulatory Commission in response to an application by General Electric Company, Wilmington, NC, for renewal of Special Nuclear Material (SNM) License No. SNM-1097.

NUREG-1080 V01: LONG-RANGE RESEARCH PLAN FY 1985-1989. * Office of Nuclear Regulatory Research, Director. September 1984. 199pp. 8410100090. 26903:001.

The Long-Range Research Plan (LRRP) was prepared by the Office of Nuclear Regulatory Research (RES) to assist the NRC in coordinating its long-range research planning with the short-range budget cycles. The LRRP lays out programmatic approaches for research to help resolve regulatory issues. The plan will be updated annually.

NUREG-1081: POST-ACCIDENT GAS GENERATION FROM RADIALYSIS OF ORGANIC MATERIALS. WING,J. Division of Engineering. September 1984. 40pp. 8410190332. 27083:157.

This report presents a methodology for estimating the gas generation rates resulting from radiolysis of organic materials in paints and electrical cable insulation inside a nuclear reactor containment building under design basis accident conditions. The methodology was based on absorption of the radiation energies from the post-accident fission products and the assumed gas yields of the irradiated materials. A sample calculation was made using conservative assumptions, plant-specific data of a nuclear power plant, and a radiation source term which took into account the time-dependent release and physico-chemical behavior of the fission products.

NUREG-1083: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE WESTINGHOUSE RESEARCH REACTOR AT ZION, ILLINOIS. DOCKET NO. 50-87. (Westinghouse Electric Company) * Division of Licensing. September 1984. 74pp. 8410170290. 27029:208.

This Safety Evaluation Report, prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission, is for an application filed by the Westinghouse Electric Corporation (WEC) for renewal of operating license R-119. The facility is owned and operated by the Westinghouse Electric Corporation and is located in the City of Zion, Illinois. The staff concludes that the reactor facility can continue to be operated by WEC without endangering the health and safety of the public.

NUREG-1084: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT MICHIGAN STATE UNIVERSITY. Docket No. 50-294. (Michigan State University) * Division of Licensing. August 1984. 89pp. 8409260645. 26701:086.

This Safety Evaluation Report for the application filed by the Michigan State University (MSU) for a renewal of operating license number R-114 to continue to operate the TRIGA 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 Michigan State University and is located on the campus of Michigan State University in East Lansing, Ingham County, Michigan. The staff concludes that the TRIGA reactor facility can continue to be operated by MSU without endangering the health and safety of the public.

NUREG-1085: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION, UNIT NO. 2. Docket No. 50-410. (Niagara Mohawk Power Corporation, Rochester Gas & Electric Corporation And Central Hudson Gas & Electric Corporation) * Division of Licensing. July 1984. 313pp. 8408220318. 26201:037.

This Draft Environmental Statement contains the assessment of the environmental impact associated with the operation of the Nine Mile Point Nuclear Station, Unit 2, 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. This statement examines the environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs.

NUREG-1086: SAFETY EVALUATION RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF MISSOURI - ROLLA. Docket No. 50-123. (University Of Missouri, Rolla) * Division of Licensing. December 1984. 77pp. 8501030030. 28196:138.

This Safety Evaluation Report for the application filed by the University of Missouri-Rolla for a renewal of Operating License R-79 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 by the University of Missouri and is located on the campus in Rolla, Missouri. On the basis of its technical review, the staff concludes that the reactor facility can continue to be operated by the university without endangering the health and safety of the public or the environment.

NUREG-1087: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2. Docket Nos. 50-424 And 50-425. (Georgia Power Company) * Division of Licensing. October 1984. 314pp. 8411210118. 27618:133.

This Draft Environmental Statement contains an assessment of the environmental impact associated with the operation of the Vogtle Electric Generating Plant, Units 1 and 2, pursuant to the National Environmental Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51 (10 CFR 51), as amended, of the Nuclear Regulatory Commission regulations. This statement examines the environmental impacts, environmental consequences and mitigating actions, and environmental and economic benefits and costs associated with station operation.

NUREG-1088: TECHNICAL SPECIFICATIONS FOR LIMERICK GENERATING STATION, UNIT No. 1. Docket No. 50-352. (Philadelphia Electric Company) HOFFMAN, D.R. Division of Licensing. October 1984. 500pp. 8411130680. 27470:058.

The Limerick Generating Station, 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 Part 50 for the protection of the health and safety of the public.

NUREG-1090: U.S. NUCLEAR REGULATORY COMMISSION 1983 ANNUAL REPORT. MAHER, W. Office of Resource Management, Director. June 1984. 206pp. 8406250266. 25138:001.

This report addresses all NRC activities, policies, and decisions made during the reporting period, complete with illustrations, charts, and treatment of technical material in lay language for consumption by the lay public.

NUREG-1092: ENVIRONMENTAL ASSESSMENT FOR 10 CFR PART 72, "LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT FUEL AND HIGH-LEVEL RADIOACTIVE WASTE." SCHULTEN, C.S. Division of Engineering Technology. August 1984. 75pp. 8409200397. 26607:253.

The Nuclear Waste Policy Act of 1982 (NWPA) addresses the need for development of monitored retrievable storage for spent fuel and high-level radioactive waste. The Commission has examined its regulations and determined that much of existing 10 CFR Part 72 regulations can be used during initial design development for a monitored retrievable storage installation (MRS), however, changes are needed to 10 CFR Part 72 to clarify specific issues which have been raised by the NWPA. The proposed revisions to 10 CFR Part 72 establish licensing requirements for a monitored retrievable storage installation. However, unless Congress authorizes construction of an MRS promulgation of these requirements would not result in construction or operation of such an installation. The issues identified as requiring resolution by the proposal amendments are (1) establishing license criteria for the long-term storage of spent fuel and high-level radioactive waste in an MRS, (2) inclusion of license requirements for the long-term storage of spent fuel and high-level radioactive waste in an MRS under 10 CFR Part 72, and (3) elimination of the current restrictions placed on fuel cladding integrity in the present Part 72 which require the fuel cladding be protected against degradation and gross ruptures, and substitution of restrictions on radioactive releases to the environment.

NUREG-1093: RELIABILITY AND RISK ANALYSIS METHODS RESEARCH PLAN. * Division of Risk Analysis & Operations (post 840429). October 1984. 101pp. 8411090520. 27435:081.

This document presents a plan for reliability and risk analysis methods research to be performed mainly by the Reactor Risk Branch (RRB), Division of Risk Analysis and Operations (DRAO), Office of Nuclear Regulatory Research. It includes those activities of other DRAO branches which are very closely related to those of the RRB. Related or interfacing programs of other divisions, offices and organizations are merely indicated. The primary use of this document is envisioned as an NRC working document, covering about a 3-year period, to foster better coordination in reliability and risk analysis methods development between the offices of Nuclear Regulatory Research and Nuclear Reactor Regulation. It will also serve as an information source for contractors and others to more clearly understand the objectives, needs programmatic activities and interfaces together with the overall logical structure of the program.

NUREG-1094: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF BEAVER VALLEY POWER STATION, UNIT NO. 2. Docket No. 50-412. (Duquesne Light Company) * Division of Licensing. December 1984. 225pp. 8501080458. 28264:036.

The information in this statement is the second assessment of the environmental impact associated with the construction and operation of the Beaver Valley Power Station, Unit 2, located in Beaver Country, Pennsylvania. The first assessment was the Final Environmental Statement related to construction, issued in October 1973 prior to issuance of the construction permit for Beaver Valley Unit 2. The Beaver Valley Unit 2 plant construction is now 82% complete and fuel loAd is scheduled for June 1986. The present assessment is the result of the NRC staff review of the activities associated with the proposed operation of the plant.

NUREG-1097: TECHNICAL SPECIFICATIONS FOR BYRON STATION UNIT 1. Docket No. 50-454. (Commonwealth Edison Co) MOON, C. W. Division of Licensing. October 1984. 485pp. 8411160595. 27562:001.

The Byron Station, 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 Part 50 for the protection of the health and safety of the public.

NUREG-1099: TECHNICAL SPECIFICATIONS FOR CATAWBA NUCLEAR STATION UNIT 1. Docket No. 50-413. (Duke Power Company) * Division of Licensing. December 1984. 250pp. 8412310101. 28149:001.

The Catawba Nuclear Station, 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 reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the health and safety of the public. A limited power license.

NUREG-1102: TECHNICAL SPECIFICATIONS FOR DIABLO CANYON NUCLEAR POWER PLANT UNIT NO. 1. Docket No. 50-275. (Pacific Gas and Electric Company) * Division of Licensing. November 1984. 446pp. 8411290164. 27692:324.

The Diablo Canyon Nuclear Power 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 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-1107: RCSLK9: REACTOR COOLANT SYSTEM LEAK RATE DETERMINATION FOR PWRs. User's Guide. HOLLAND, R. A.; KIRKPATRICK, D.; WOODRUFF, R. W. Division of Emergency Preparedness & Engineering Response (Post 830103). December 1984. 98pp. 8501070494. 28223:004.

RCSLK9 is a computer program that was developed to analyzed the leak tightness of the primary coolant system for any pressurized water reactor. From system conditions, water levels in tanks, and certain system design parameters, RCSLK9 calculates the loss of water from the cooling system and the increase of water in the leakage collection system during an arbitrary time interval. The program determines the system leak rates and displays or prints a report of the results. For initial application of the program at a reactor, RCSLK9 creates a file of system parameters and stores it for future use. RCLK9 is written for use on the IBM PC.

NUREG-1111: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1. Docket No. 50-528. (Arizona Public Service Company) * Division of Licensing. December 1984. 430pp. 8501070420. 28239:001.

The Palo Verde, 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 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/CP-0048 V01: PROCEEDINGS OF THE ELEVENTH WATER REACTOR SAFETY INFORMATION MEETING. SZAWLEWICZ, S. A. Office of Nuclear Regulatory Research, Director. January 1984. 721pp. 8401310150. 22030:001.

The papers published in this report were presented at the Eleventh Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 24-28, 1983. The papers describe the results and program plans in nuclear reactor safety research conducted in this country and abroad. International participation in the exchange of nuclear safety research information is consistently encouraged by the Nuclear Regulatory Commission and was exemplified at this meeting by the relatively large number of papers (23) given by representatives of seven different European countries, Japan and Taiwan.

Volume 1 reports information presented at the a. Plenary Session, b. Integral Systems Experiments, c. Separate Effects, d. Foreign Programs in Thermal Hydraulics, and e. The EPRI Safety Research Session.

NUREG/CP-0048 V02: PROCEEDINGS OF THE ELEVENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ, S. A. Office of Nuclear

Regulatory Research, Director. January 1984. 370pp. 8401260097. 21988:001.

The papers published in this report were presented at the Eleventh Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 24-28, 1983. The papers describe the results and program plans in nuclear reactor safety research conducted in this country and abroad. International participation in the exchange of nuclear safety research information is consistently encouraged by the Nuclear Regulatory Commission and was exemplified at this meeting by the relatively large number of papers (23) given by representatives of seven different European countries, Japan and Taiwan.

Volume 2 reports information presented on a. Pressurized Thermal Shock; b. Code Assessment and Improvement; c. 2D/3D Research Program and d. Nuclear Plant Analyzer Program.

NUREG/CP-0048 V03: PROCEEDINGS OF THE ELEVENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ, S. A. Office of Nuclear Regulatory Research, Director. January 1984. 705pp. 8401260096. 21986:001.

The papers published in this report were presented at the Eleventh Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 24-28, 1983. The papers describe the results and program plans in nuclear reactor safety research conducted in this country and abroad. International participation in the exchange of nuclear safety research information is consistently encouraged by the Nuclear Regulatory Commission and was exemplified at this meeting by the relatively large number of papers (23) given by representatives of seven different European countries, Japan and Taiwan.

Volume 3 reports information presented on a. Containment Systems Research; b. Status of Source Term Reassessment; c. Fuel Systems Research Program, and d. Risk Analysis.

NUREG/CP-0048 V04: PROCEEDINGS OF THE ELEVENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ, S. A. Office of Nuclear Regulatory Research, Director. January 1984. 470pp. 8401260108. 21969:327.

The papers published in this report were presented at the Eleventh Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 24-28, 1983. The papers describe the results and program plans in nuclear reactor safety research conducted in this country and abroad. International participation in the exchange of nuclear safety research information is consistently encouraged by the Nuclear Regulatory Commission and was exemplified at this meeting by the relatively large number of papers (23) given by representatives of seven different European countries, Japan and Taiwan.

Volume 4 covers the sessions on Materials Engineering Research.

NUREG/CP-0048 V05: PROCEEDINGS OF THE ELEVENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ, S. A. Office of Nuclear Regulatory Research, Director. January 1984. 353pp. 8401250162. 21950:254.

The papers published in this report were presented at the Eleventh Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 24-28,

1983. The papers describe the results and program plans in nuclear reactor safety research conducted in this country and abroad. International participation in the exchange of nuclear safety research information is consistently encouraged by the Nuclear Regulatory Commission and was exemplified at this meeting by the relatively large number of papers (23) given by representatives of seven different European countries, Japan and Taiwan.

Volume 5 reports information presented on a. Mechanical Engineering; b. Structural Engineering; c. Seismic Research Program; d. Instrumentation and Control Program, and e. Research on Equipment Survival in Accidents.

NUREG/CP-0048 V06: PROCEEDINGS OF THE ELEVENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. SZAWLEWICZ, S. A. Office of Nuclear Regulatory Research, Director. January 1984. 206pp. 8401250125. 21950:046.

The papers published in this report were presented at the Eleventh Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 24-28, 1983. The papers describe the results and program plans in nuclear reactor safety research conducted in this country and abroad. International participation in the exchange of nuclear safety research information is consistently encouraged by the Nuclear Regulatory Commission and was exemplified at this meeting by the relatively large number of papers (23) given by representatives of seven different European countries, Japan and Taiwan.

Volume 6 reports information presented on a. Human Factors Research; b. Safeguards Research; c. Emergency Preparedness; d. Process Control, and e. Occupational Radiation Protection.

NUREG/CP-0049: PROCEEDINGS OF THE WORKSHOP ON SPENT FUEL/CLADDING REACTION DURING DRY STORAGE. REISENWEAVER, D. Division of Engineering Technology. March 1984. 499pp. 8404130225. 24047:069.

This document presents the papers that were presented at the workshop on spent fuel dry storage research which was held in Gaithersburg, MD on August 17-18, 1983.

NUREG/CP-0050: PROCEEDINGS OF THE INTERNATIONAL BETA DOSIMETRY SYMPOSIUM. Held at Washington, DC, February 15-18, 1983. * Office of Nuclear Regulatory Research, Director. * Energy, Dept. of. * Health Physics Society. January 1984. 670pp. 8402100499. 22209:001.

At the International Beta Dosimetry Symposium, 1982, invited lecturers presented introductory summaries for assigned topics and chaired both technical sessions and related workshops. These proceedings contain the technical papers that were presented, summaries of each of the workshop discussion sessions, and the final summary.

1. Review the current state-of-the-art in beta dosimetry and the applied problems throughout the nuclear industry;
2. Review ongoing research and development and new technology.
3. Stimulate vigorous interchange on an international basis to encourage new ideas and technology transfer;
4. Review biological effects and explore the need for better-defined protection standards
5. Provide a public record of the above information; and
6. Charge the industry and advisory bodies to provide improved

technology and applied techniques as well as more clear guidance for worker protection.

NUREG/CP-0051: PROCEEDINGS OF THE CSNI SPECIALIST MEETING ON LEAK-BEFORE-BREAK IN NUCLEAR REACTOR PIPING. * Division of Engineering Technology. August 1984. 561pp. 8409070237. 82. 26410:001.

On September 1 and 2, 1983, the CSNI Subcommittee on Primary System Integrity held a special meeting in Monterey, California, on the subject of Leak-Before-Break in Nuclear Reactor Piping Systems. The purpose of the meeting was to provide an international forum for the exchange of ideas, positions, and research results; to identify areas requiring additional research and development; and to determine the general attitude toward acceptance of the leak-before-break concept. This report documents the presentations made at the meeting in the areas of (1) application of piping fracture mechanics to leak-before-break; (2) leak rate and leak detection; (3) leak-before-break studies, methods, and results; and (4) current and proposed positions on leak-before-break.

NUREG/CP-0052: NRC NUCLEAR WASTE MANAGEMENT GEOCHEMISTRY '83. ALEXANDER, D. H.; BIRCHARD, G. F. Division of Health, Siting & Waste Management. May 1984. 541pp. 8406060366. 24846:001.

This document summarizes papers and panel discussions presented at the Office of Nuclear Regulatory Research sponsored conference on "Nuclear Waste Management Research on Geochemistry of HLW Disposal" The conference was held at the United States Geological Federal Center in Reston, Virginia on August 30-31, 1983. The purpose of the meeting was to present results from NRC sponsored research and to identify regulatory research issues which need to be addressed prior to licensing a high level waste repository. Important summaries of technical issues and recommendations are included with each paper. The issues reflect areas of technical uncertainty addressed by the NRC Research program in geochemistry. The objectives of the NRC Research Program in geochemistry are to provide a technical basis for waste management rulemaking, to provide the NRC Waste Management Licensing Office with information that can be used to support sound licensing decisions, and to identify investigations that need to be conducted by DOE to support a license application.

NUREG/CP-0053: PROCEEDINGS OF THE NINTH ANNUAL STATISTICS SYMPOSIUM ON NATIONAL ENERGY ISSUES, October 19-21, 1983. BRYSON, M. C. Los Alamos Scientific Laboratory. August 1984. 200pp. 8408240249. LA-10127-C. 26251:027.

The Ninth Annual Statistics Symposium on National Energy Issues was held in Rockville, Maryland, at the Holiday Inn Crowne Plaza, October 19-21, 1983, under the joint sponsorship of Los Alamos National Laboratory and the Nuclear Regulatory Commission. Sessions included two contributed-paper sessions, two tutorial sessions, and one discussion group. Included in these proceedings are those papers for which final copy was provided by the authors, together with a list of papers presented and a list of attendees.

NUREG/CP-0055: PROCEDURES OF THE STATE WORKSHOP ON SHALLOW LAND BURIAL AND ALTERNATIVE DISPOSAL CONCEPTS. Held At Bethesda, Maryland, May 2-3, 1984. * Division of Waste Management. * Division of Fuel Cycle

& Material Safety. * Office of State Programs, Director. October 1984. 383pp. 8411160139. 27564:001.

Three of the major conclusions reached by State participants were the following: (1) Significant data gaps and information needs have to be addressed before timely State decisionmaking can be accomplished. State participants felt a generic cost/risk/benefit analysis for all viable alternatives would be useful and might best be performed by the Federal government on behalf of the states. (2) Recognizing the imprecision in summarizing overall attitudes of the workshop participants, alternative disposal concepts that appear to be the most favorably perceived when rank ordered by "critical" factors are augered holes with liners, belowground vaults, earth mounded concrete bunkers, aboveground vaults and mined cavities. (3) The public appears to place greater confidence in disposal methods that incorporate man-made engineered barriers because of some past problems at closed shallow land burial facilities. Concern was expressed by workshop participants that the public may not consider the perceived risks associated with shallow land burial to be acceptable. In addition to the four 10 CFR Part 61 Subpart C performance objectives, public acceptance of risk was considered to be a critical factor by State officials in selecting a disposal technology. The States should take the lead in pursuing development-oriented analyses, such as detailed concept engineering and economic feasibility studies. It is not within the purview of NRC responsibility to undertake such studies.

NUREG/CP-0056: PROCEEDINGS OF THE SECOND WORKSHOP ON CONTAINMENT INTEGRITY. * Sandia Laboratories. November 1984. 659pp. 8411210170. SAND84-1514. 27623:039.

The Second Workshop on Containment Integrity was held in Crystal City, Virginia, on June 13-15, 1984. The workshop provided a forum for exchanging information on the integrity of containments at nuclear power plants. The behavior of containments during severe accidents was of primary interest to over 130 participants. Forty-three oral presentations were made at the workshop. Written contributions that correspond to each of the presentations make up the body of this report. The workshop was hosted by Sandia National Laboratories under the sponsorship of the U.S. Nuclear Regulatory Commission. Principal organizers for the workshop were T. E. Blejwas and W. A. von Riesemann of Sandia, T. D. Molina of Technadyne, and J. F. Costello of the U. S. Nuclear Regulatory Commission.

NUREG/CP-0057: TRANSACTIONS OF THE TWELFTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING. * Office of Nuclear Regulatory Research, Director. October 1984. 350pp. 8410120049. 26979:001.

This report contains summaries of papers on reactor safety research to be presented at the 12th Water Reactor Safety Research Information Meeting. The meeting will be held at the National Bureau of Standards in Gaithersburg, Maryland, October 22-26, 1984. 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 are also included. They cover the highlights of reactor safety research conducted by the electric utilities through the Electric Power Research Institute, 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.

NUREG/CP-0060: THE FIRST INTERNATIONAL WORKSHOP ON FUNDAMENTAL ASPECTS OF POST-DRYOUT HEAT TRANSFER. Held At Salt Lake City, Utah, April 2-4, 1984. LEE, R. Division of Accident Evaluation. December 1984. 710pp. 8412280495. 28129:001.

This report contains papers presented at the First International Workshop on Fundamental Aspects of Post-Dryout Heat Transfer that was held in Salt Lake City on April 2-4, 1984. The purpose of the workshop is to review recent development and the state-of-the-art in the field of post-dryout heat transfer. It centered on interchanging ideas, reviewing current research results, and defining future research needs. The five sessions dealing with the fundamental aspects of post-dryout heat transfer were: computer code modeling and flow phenomena, quenching phenomena, low-void heat transfer, dispersed-flow heat transfer and effects of grids and blockages.

NUREG/CR-0130 ADD03: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING A REFERENCE PRESSURIZED WATER REACTOR POWER STATION. MURPHY, E. S. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1984. 50pp. 8410170293. 27026:290.

The radioactive wastes expected to result from decommisioning of the reference pressurized water reactor power station are reviewed and classified in accordance with 10 CFR 61. The 17,885 cubic meters of waste from DECON are classified as follows: Class A, 98.0%; Class B, 1.2%; Class C, 0.1%. About 0.7% (133 cubic meters) of the waste would be generally unacceptable for disposal using near-surface disposal methods.

NUREG/CR-0169 V17: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSIS. Volume XVII Process Instruments Recorded On DAVDS. EVANS, R. P.; MCKNIGHT, K. D. EG&G, Inc. September 1984. 55pp. 8410120037. EGG-2037. 26977:159.

Uncertainty analyses are presented to quantify the uncertainty bounds for the Loss-of-Fluid Test (LOFT) process measurements. The process instruments are those used to control the plant operation safety. The uncertainties presented are of two types: objective uncertainties (basically random) which can be duplicated in the laboratory and for which data are available, and subjective uncertainties (basically systematic) for which no specific data are available.

NUREG/CR-0169 V22: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSIS Volume XXII. Fission Product Detection System Instruments Recorded On The DAVDS. EVANS, R. P. EG&G, Inc. December 1984. 31pp. 8501030043. EGG-2037. 28196:214.

An uncertainty analysis was performed for the Loss-of-Fluid Test Fission Product Instruments recorded on the Data Acquisition and Visual Display System in order to document the accuracy of these channels under steady state operating conditions. In addition, an uncertainty analysis was performed for certain temperature and pressure measurement channels excluding their recording system.

NUREG/CR-0200 V1-3R3: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. PARKS, C. V. Oak Ridge National Laboratory. December 1984. 400pp. 8501070411. 28220:001.

SCALE (Standardized Computer Analyses for Licensing Evaluation), a multi-faceted computational system, has been developed to provide a standard analysis tool for use by the NRC staff and licensees in evaluation of nuclear fuel facility and package designs. Revision 3 describes recent additions and modifications made to the SCALE computational system. Combining this material with previously issued material will update the SCALE Manual to be consistent with the latest publicly released version of SCALE, called SCALE-3.

NUREG/CR-0672 ADD02: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING A REFERENCE BOILING WATER REACTOR POWER STATION. Classification Of Decommissioning Wastes. MURPHY, E. S. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1984. 50pp. 8410170289. 27029:283.

The radioactive wastes expected to result from decommissioning of the reference boiling water reactor power station are reviewed and classified in accordance with 10 CFR 61. The 18,949 cubic meters of waste from DECON are classified as follows: Class A, 97.5%; Class B, 2.0%; Class C, 0.3%. About 0.2% (47 cubic meters) of the waste would be generally unacceptable for disposal using near-surface methods.

NUREG/CR-1740 R01: DATA SUMMARIES OF LICENSEE EVENT REPORTS OF SELECTED INSTRUMENTATION AND CONTROL COMPONENTS AT U.S. COMMERCIAL NUCLEAR POWER PLANTS JANUARY 1, 1976 TO DECEMBER 31, 1981. TROJOVSKY, M.; BROWN, S. R. EG&G, Inc. July 1984. 344pp. 8408240382. EGG-2307. 26249:001.

This report describes a computer-based data file developed from Licensee Event Reports (LERs) of instrumentation and control (I&C) components in United States commercial nuclear power plants for the period January 1, 1976, to December 31, 1981. In addition to the creation of the file, summaries of data contained in the file were made to obtain data for risk assessment and statistical purposes. Gross constant fault (failure and command fault) rates were estimated for major components and channels that provide a direct reactor trip. Explanations, figures, and summary tables of the results are provided. This report updates and supersedes the original May 1981 edition of NUREG/CR-1740.

NUREG/CR-1755: TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS. WITTENBROCK, N. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1982. 339pp. 8401230677. 21901:083.

Safety and cost information is developed for the conceptual decommissioning of large (1175-MWe) pressurized water reactors (PWR) and large (1155-MWe) boiling water reactors (BWR) at multiple-reactor stations. Three decommissioning alternatives are studied: DECON (immediate decontamination), SAFSTOR (safe storage followed by deferred decontamination), and ENTOMB (entombment). Safety and costs of decommissioning are estimated by determining the impact of probable features of multiple-reactor-station operation that are considered to be unavailable at a single-reactor station, and applying these estimated impacts to the decommissioning costs and radiation doses estimated in previous PWR and BWR decommissioning studies. The multiple-reactor-station features analyzed are: the use of interim onsite nuclear waste storage with later removal to an offsite waste disposal facility, the use of permanent onsite nuclear waste disposal, the dedication of the site to nuclear power generation, and the provision of centralized services.

NUREG/CR-2000 V02N12: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of December 1983. * Oak Ridge National Laboratory. January 1984. 169pp. 8402060529. ORNL/NSIC-200. 22092:143.

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 this 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 are described in detail in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keywords, and component vendor indexes follow the summaries. The components, systems, and vendors are those identified by the utility when the LER is initiated; the keywords are assigned by the computer using correlation tables from the Sequence Coding and Search System.

NUREG/CR-2000 V03 N1: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of January 1984. * Oak Ridge National Laboratory. February 1984. 123pp. 8403070392. ORNL/NSIC-200. 22554:092.

See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03 N2: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of February 1984. * Oak Ridge National Laboratory. March 1984. 167pp. 8404020225. ORNL/NSIC-200. 22876:160.

See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03 N3: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of March 1984. * Oak Ridge National Laboratory. April 1984. 175pp. 8405010064. ORNL/NSIC-200. 24257:064.

See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03 N4: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of April 1984. * Oak Ridge National Laboratory. May 1984. 180pp. 8406040026. ORNL/NSIC-200. 24805:078.

See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03 N5: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of May 1984. * Oak Ridge National Laboratory. June 1984. 129pp. 8407160280. ORNL/NSIC-200. 25625:070.

See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03 N6: LICENSEE EVENT REPORT (LER) COMPILATION: For Month of June 1984. * Oak Ridge National Laboratory. July 1984. 87pp. 8408070007. ORNL/NSIC-200. 25955:112.

See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03 N7: LICENSEE EVENT REPORT (LER) COMPILATION: For Month of July 1984. * Oak Ridge National Laboratory. August 1984. 101pp. 8408300287. ORNL/NSIC-200. 26331:123.

See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03 N8: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of August 1984. * Oak Ridge National Laboratory. September 1984. 68pp. 8410100171. ORNL/NSIC-200. 26904:001.
See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03 N9: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of September 1984. * Oak Ridge National Laboratory. October 1984. 106pp. 8411070344. ORNL/NSIC-200. 27386:257.
See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03N10: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of October 1984. * Oak Ridge National Laboratory. November 1984. 69pp. 8412070087. ORNL/NSIC-200. 27839:122.
See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2000 V03N11: LICENSEE EVENT REPORT (LER) COMPILATION: For Month of November 1984. * Oak Ridge National Laboratory. December 1984. 63pp. 8501030024. ORNL/NSIC-200. 28194:249.
See NUREG/CR-2000,V02,N12 abstract.

NUREG/CR-2015 V08: PHASE I FINAL REPORT - SYSTEMS ANALYSIS (PROJECT VII). Seismic Safety Margins Research Program. WELLS,J.E.; GEORGE,L.L.; CUMMINGS,G.E. Lawrence Livermore National Laboratory. September 1984. 188pp. 8409280110. UCRL-53021 V08. 26764:001.
This document reports on the Phase 1 efforts of the Systems Analysis project to develop the tools and methods for computing the probability of radioactive release from a commercial nuclear power plant in the event of an earthquake.

NUREG/CR-2210: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS. LUDWICK,J.D.; MOORE,E.B. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 7pp. 8402060341. 22105:001.
Safety and Cost information is developed for the conceptual decommissioning of five representative independent spent fuel storage installations. This information is presented by analyzing major facility components and then developing safety and cost information for the reference installations made up of these components. Three decommissioning alternatives are studied to obtain comparisons between costs (in 1981 dollars), occupational radiation doses, potential radiation dose to the public, and other safety impacts. The alternatives considered are: DECON (immediate decontamination), SAFSTOR (safe storage followed by deferred decontamination), and ENTOMB (entombment).

NUREG/CR-2267: EVALUATION OF PORTABLE RADIOLOGICAL INSTRUMENTS FOR EMERGENCY RESPONSE MEASUREMENTS OF RADIOIODINE. KRUPA,J.F. BIRD,S.K.; MOTES,B.G. Idaho National Engineering Laboratory. October 1984. 100pp. 8411130669. WINCO-1003. 27472:117.
In the event radionuclides are released to the environment following an accident at a commercial nuclear power facility,

measurement methods are required to assess the potential radiological hazards to the surrounding populace. Reported are the results of a study to evaluate the relative ability of selected portable radiological instruments to measure: (131)I in milk, on barren and grass covered ground, and in human and bovine thyroid phantoms. The study of the three GM and three scintillation, (NA(TI), instruments entailed evaluations of their sensitivities, accuracies, energy responses and performance characteristics in simulated environmental conditions of temperature and relative humidity and upon mechanical shock and vibration.

NUREG/CR-2331 V03 N2: SAFETY RESEARCH PROGRAMS SPONSORED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH. Quarterly Progress Report, April-June 1983. BARI, R. A.; CERBONE, R. J.; GINSBERG, T.; et al. Brookhaven National Laboratory. March 1984. 159pp. 8404020011. BNL-NUREG-51454. 22876:001.

The Advanced and Water Reactor Safety Research Programs Quarterly Progress Reports have been combined and are included in this report entitled, "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research - Quarterly Progress Report." The projects reported are the following: HTGR Safety Evaluation, SSC Development, Validation and Application, CRBR Balance of Plant Modeling, Thermal-Hydraulic Reactor Safety Experiments, LWR Plant Analyzer Development, LWR Code Assessment and Application, Thermal Reactor Code Development (RAMONA-3B); Stress Corrosion Cracking of PWR Steam Generator Tubing, Bolting Failure Analysis, Probability Based Load Combinations for Design of Category I Structures, Mechanical Piping Benchmark Problems; Human Error Data for Nuclear Power Plant Safety Related Events, Criteria for Human Engineering Regulatory Guides and Human Factors in Nuclear Power Plant Safeguards.

NUREG/CR-2331 V03 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH. Quarterly Progress Report, July-September 1983. WEISS, A. J. Brookhaven National Laboratory. July 1984. 154pp. 8408010179. BNL-NUREG-51454. 25870:181.

The Advanced and Water Reactor Safety Research Programs Quarterly Progress Reports have been combined and are included in this report entitled, "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research - Quarterly Progress Report." 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 Facility Operations of the U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

The projects reported are the following: HTGR Safety Evaluation, SSC Development, Validation and Application, CRBR Balance of Plant Modeling, Thermal-Hydraulic Reactor Safety Experiments, LWR Plant Analyzer Development, LWR Code Assessment and Application, Thermal Reactor Code Development (RAMONA-3B); Stress Corrosion Cracking of PWR Steam Generator Tubing, Bolting Failure Analysis, Probability Based Load Combinations for Design of Category I Structures, Mechanical Piping Benchmark Problems; Human Error Data for Nuclear Power Plant Safety-Related Events, and Human Factors in Nuclear Power Plant Safeguards. The previous reports have covered the period October 1, 1976 through June 30, 1983.

NUREG/CR-2331 V03 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH. Quarterly Progress Report, October 1 -December 31, 1983. WEISS, A. J. Brookhaven National Laboratory. September 1984. 129pp. 8410120059. BNL-NUREG-51454. 26983:212.

The Advanced and Water Reactor Safety Research Programs Quarterly progress reports have been combined and are included in this report entitled, "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research - Quarterly Progress Report." 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 Facility Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

The projects reported are the following: High Temperature Reactor Research, SSC Development, Validation and Application, CRBR Balance of Plant Modeling, Thermal-Hydraulic Reactor Safety Experiments, Development of Plant Analyzer, Code Assessment and Application (Transient and LOCA Analyses), Thermal Reactor Code Development (RAMONA-3B), Calculational Quality Assurance in Support of PTS; Stress Corrosion Cracking of PWR Steam Generator Tubing, Bolting Failure Analysis, Probability Based Load Combinations for Design of Category I Structures, Mechanical Piping Benchmarking Problems, Identification of Age-Related Failure Modes; Analysis of Human Error Data for Nuclear Power Plant Safety-Related Events, Human Factors in Nuclear Power Plant Safeguards, Emergency Action Levels, and Protective Action Decision Making. The previous reports have covered the period October 1, 1976 through September 30, 1983.

NUREG/CR-2331 V03 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH. Quarterly Progress Report, October 1 -December 31, 1983. WEISS, A. J. Brookhaven National Laboratory. September 1984. 129pp. 8410120059. BNL-NUREG-51454. 26983:212.

The Advanced and Water Reactor Safety Research Programs Quarterly progress reports have been combined and are included in this report entitled, "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research - Quarterly Progress Report." 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 Facility Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

The projects reported are the following: High Temperature Reactor Research, SSC Development, Validation and Application, CRBR Balance of Plant Modeling, Thermal-Hydraulic Reactor Safety Experiments, Development of Plant Analyzer, Code Assessment and Application (Transient and LOCA Analyses), Thermal Reactor Code Development (RAMONA-3B), Calculational Quality Assurance in Support of PTS; Stress Corrosion Cracking of PWR Steam Generator Tubing, Bolting Failure Analysis, Probability Based Load Combinations for Design of Category I Structures, Mechanical Piping Benchmarking Problems, Identification of Age-Related Failure Modes; Analysis of Human Error Data for Nuclear Power Plant Safety-Related Events, Human Factors in Nuclear Power Plant Safeguards, Emergency Action Levels, and Protective Action Decision Making. The previous reports have covered the period October 1, 1976 through September 30, 1983.

NUREG/CR-2331 V04 N1: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH. Quarterly Progress Report, January 1

-March 31, 1984. WEISS, A. J. Brookhaven National Laboratory. November 1984. 162pp. 8412240029. BNL-NUREG-51454. 28059:128.

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 Facility Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The projects reported are the following: High Temperature Reactor Research, SSC Development, Validation and Application, CRBR Balance of Plant Modeling, Thermal-Hydraulic Reactor Safety Experiments, Development of Plant Analyzer, Code Assessment and Application (Transient and LOCA Analyses), Thermal Reactor Code Development (RAMONA-3B), Calculational Quality Assurance in Support of PTS; Stress Corrosion Cracking of PWR Steam Generator Tubing, Bolting Failure Analysis, Probability Based Load Combinations for Design of Category I Structures, Mechanical Piping Benchmark Problems, Identification of Age-Related Events, Human Factors Aspects of Safety/Safeguards Interactions During Routine Reactor Operations and Off-Normal Conditions, Emergency Action Levels, and Protective Action Decision Making.

NUREG/CR-2335: RESULTS OF THE SEMISCALE MOD-2A NATURAL CIRCULATION EXPERIMENTS. LOOMIS, G. G.; SODA, K. EG&G, Inc. October 1982. 74pp. 8307210265. EGG-2200. 19688:093.

A series of experiments was conducted in a scaled model of a pressurized water reactor (Semiscale Mod-2A) to investigate natural circulation heat rejection under normal and abnormal operating conditions. The effects on natural circulation of diminished primary and secondary coolant inventory, as well as the presence of non-condensible gas in the primary, were determined. Three distinct modes of natural circulation were found to occur as a function of primary coolant inventory: single-phase, two-phase (liquid continuous), and reflux condensation. The primary coolant inventory limit for adequate heat rejection was found to be the amount of coolant necessary to keep the core covered. The presence of nitrogen gas in plausible quantities altered natural circulation behavior, but did not preclude adequate heat rejection.

NUREG/CR-2397: FUEL INVENTORY AND AFTERHEAT POWER STUDIES OF URANIUM-FUELED PRESSURIZED WATER REACTOR FUEL ASSEMBLIES USING THE SAS2 AND ORIGEN-S MODULES OF SCALE WITH AN ENDF/B-V UPDATED CROSS SECTION LIBRARY. RYMAN, J. C.; HERMANN, O. W.; WEBSTER, C. C.; et al. Oak Ridge National Laboratory. October 1982. 144pp. 8307250159. ORNL/CSD-90. 19738:280.

The SAS2 control module and ORIGEN-2 code of the SCALE code system have been used with the standard ORIGEN-S data libraries and the SCALE 27-group ENDF/B-V cross-section library to predict afterheat power and fuel inventories in uranium-fueled pressurized water reactor (PWR) fuel assemblies. Based on present comparisons with measured data and on previous experience, it is concluded that this combination of codes and data bases is properly qualified for the calculation of afterheat power in uranium-fueled PWR fuel assemblies. The prediction of fission product fuel inventory data for fuel samples from three PWR fuel assemblies appears to be adequate, but the prediction of actinide inventory data is seen to be quite conservative with respect to measured data. Additional investigation of the differences between calculated and measured inventory parameters, improvements in the SAS2 cross section treatment, and acquisition of additional experimental data appear to be needed to qualify the SAS2, ORIGEN-2, and data base combination for uranium-fueled PWR fuel inventory calculations. It was determined, however, that predicted values of afterheat power for three uranium-fueled PWR fuel assemblies were in good agreement (about 5% conservative) with measured values.

NUREG/CR-2424 V01: MATHEMATICAL SIMULATION OF SEDIMENT AND RADIONUCLIDE TRANSPORT IN COASTAL WATERS. Vol 1: Testing Of The Sediment/ Radionuclide Transport Model FETRA. ONISHI, Y.; THOMPSON, F. L. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 112pp. 8406230270. PNL-5088-1. 25132:269.

The finite element model, FETRA, is an unsteady, two-dimensional (longitudinal and lateral) model for simulating the transport of sediment and contaminants (e.g., radionuclides, heavy metals, pesticides) in coastal waters. FETRA includes major transport and fate mechanisms explicitly, including sediment/contaminant interactions. The model was tested by applying it to the Irish Sea to simulate wind-generated waves and the migration of sediment and (137)Cs. The model predicted distributions of suspended sand; suspended silt; suspended clay; (137)Cs sorbed by each of the three sizes of suspended sediments; dissolved (137)Cs; bed sediment size fractions; and (137)Cs sorbed by bed sand, bed silt, and bed clay over a two-month period in 1974. FETRA predicted that approximately 82%, 0.002%, and 18% of the total (137)Cs remaining in this study area were dissolved, suspended sediment-sorbed, and bed-sediment-sorbed radionuclides, respectively.

NUREG/CR-2424 V02: MATHEMATICAL SIMULATION OF SEDIMENT AND RADIONUCLIDE TRANSPORT IN COASTAL WATERS. V 2 User's M CP Listing for FETRA. ONISHI, Y.; THOMPSON, F. L. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 89pp. 8407110172. PNL-5088. 25542:134.

FETRA is a finite element model for simulating the sediment and containment transport to surface water. The model was applied to a test site in the Irish Sea and modified to account for wave mechanisms that affect sediment suspension. Volume 2 of this report presents a very brief users guide for FETRA and a computer program listing of the model.

NUREG/CR-2482 V05: REVIEW OF DOE WASTE PACKAGE PROGRAM. Subtask 1.1 - National Waste Package Program, April 1983 - September 1983. SOO, P. Brookhaven National Laboratory. August 1984. 122pp. 8409170270. BNL-NUREG-51494. 26498:045.

This report addresses part of an ongoing task to review the national high-level waste package effort. It includes evaluations of reference waste form, container and packing material components with respect to determining how they may contribute to the containment and controlled release of radionuclides after waste packages have been emplaced in salt, basalt, and tuff repositories. A section on carbon steel container corrosion is included to complement prior work on TiCode-12 and Type 34 stainless steel. Use of crushed tuff as a packing material is discussed, and waste package component interaction test data are included. Licensing data requirements are specified.

NUREG/CR-2499: REVIEW OF EMERGENCY RADIOLOGICAL INSTRUMENTATION AND ANALYTICAL METHODS AT NMSS-LICENSEE SITES. HERRINGTON, W. N.;

KATHREN, R. L. ; KENOYER, J. L. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 60pp. 8409200301. PNL-4163. 26609:001.

This report provides a brief review of emergency radiological monitoring instrumentation capabilities based on visits to Nuclear Material Safety and Safeguards (NMSS) licensees and on a review of the open literature. Recommendations based on findings are made with regard to instrument design and operation, training, calibration, testing, analytical methods, sampling procedures, and quality assurance. An assessment of currently available instrumentation is made with respect to types of instruments, instrument specifications, and the future needs of NRC/NMSS licensees as seen by instrument manufacturers and to what extent those needs will be met.

NUREG/CR-2531 R02: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK. SCOFIELD, N. R. ; HARDY, H. A. ; LAATS, E. T. EG&G, Inc. April 1984. 102pp. 8405220080. EGG-2164. 24556:173.

The United States Nuclear Regulatory Commission (NRC) has established the NRC/Division of Accident Evaluation (DAE) Data Bank Program to collect, store, and make available data from the many domestic and foreign water reactor safety research programs. The NRC/DAE Data Bank Program provides a central computer storage mechanism and access software for data that is to be used by code development and assessment groups in meeting the code and correlation needs of the nuclear industry. The administration portion of the program provides data entry, documentation, training, and advisory services to users and the NRC. The NRC/DAE Data Bank and the capabilities of the data access software are described in this document.

NUREG/CR-2552: CRAC2 MODEL DESCRIPTION. RITCHIE, L. T. ; ALPERT, D. J. ; BURKE, R. P. et al. Sandia Laboratories. April 1984. 95PP. 8405220186. SAND82-0342. 24602:188.

The CRAC2 computer code is a revised version of CRAC (Calculation of Reactor Accident Consequences) which was developed for the Reactor Safety Study. This document provides an overview of the CRAC2 code and a description of each of the models used. Significant improvements incorporated into CRAC2 include an improved weather sequence sampling technique, a new evacuation model, and new output capabilities. In addition, refinements have been made to the atmospheric transport and deposition model. Details of the modeling differences between CRAC2 and CRAC are emphasized in the model descriptions.

NUREG/CR-2576: BWR FULL INTEGRAL SIMULATION TEST (FIST)--Facility Description Report. STEPHENS, A. G. General Electric Co. September 1984. 267pp. 8410120002. GEAP-22054. 26955:100.

A new boiling water reactor safety test facility (FIST, Full Integral Simulation Test) is described. It will be used to investigate small breaks and operational transients and to tie results from such tests to earlier large break test results determined in the TLTA. The new facility's full height and prototypical components constitute a major scaling improvement over earlier test facilities. A heated feedwater system, permitting steady state operation, and a large increase in the number of measurements are other significant improvements. Program background is outlined and program objectives defined. Design basis is presented together with a detailed, complete description of the facility and measurements to be made. An extensive component scaling analysis and prediction of performance are presented. The report is intended to serve as a reference document for those needing detailed information about the facility.

NUREG/CR-2613: IDENTIFICATION OF CHARACTERISTICS WHICH INFLUENCE REPOSITORY DESIGN - DOMAL SALT. RAWLINGS, G. ; ANTONNEN, G. ; CHAMNESS, M. ; et al. Golder Associates. April 1984. 171pp. 8405220085. 24594:100.

The purpose of the complete project is to provide NRC with technical assistance to enable the focused, adequate review by NRC of the aspects related to design and construction of an undergound test facility and final geologic repository as presented by the Department of Energy (DOE).

The study presented in this report covers the identification of characteristics which influence design and construction of a geologic repository in domal salt. This report has identified five key issues, i.e., constructibility, thermal response, mechanical response, hydrologic response, and geochemical response. This report involves both short-term (up to closure) and long-term (post closure) effects.

The characteristics of domal salt and its environment are described under the headings of stragraphic/structural, tectonic, mechanical, thermal and hydrologic. Characteristics are separated into parameters (quantified and measured) and factors (qualitative). The characteristics are then subjectively ranked by their influence on the key issues. This takes into account the availability and suitability of conservative design/construction techniques, uncertainty in model and model sensitivity to the characteristic.

NUREG/CR-2614: IDENTIFICATION OF CHARACTERISTICS WHICH INFLUENCE REPOSITORY DESIGN - TUFF. RAWLINGS, G. ; ANTONNEN, G. ; FINDLEY, D. ; et al. Golder Associates. April 1984. 156pp. 8405220065. 813-1162C. 24564:216.

The purpose of the complete project is to provide NRC with technical assistance to enable the focused, adequate review by NRC of the aspects related to design and construction of an underground test facility and final geologic repository as presented by the Department of Energy (DOE). The study presented in this report covers the identification of characteristics which influence design and construction of a geologic repository in tuff at the Nevada Test Site (NTS). This report has identified five key issues, i.e., constructibility, thermal response, mechanical response, hydrological response, and geochemical response. This report involves both short-term (up to closure) and long-term (post closure) effects. The characteristics of tuff and its environment are described under the headings of stratigraphic/structual tectonic, mechanical, thermal and hydrologic. Characteristics are separated into parameters (quantified and measured) and factors (qualitative). The characteristics are then subjectively ranked by their influence on the key issues. This ranking took into account availability and suitabiity of conservative design/construction techniques, uncertainty in model and the model sensitivity to characteristics.

NUREG/CR-2675 V04: RELEVANCE OF BIOTIC PATHWAYS TO THE LONG-TERM REGULATION OF NUCLEAR WASTE DISPOSAL: Phase I Final Report. MCKENZIE, D. H. ; CADWELL, L. L. ; EBERHARDT, L. E. ; et al. Battelle

Memorial Institute, Pacific Northwest Laboratories. May 1984. 49pp. 8406230239. PNL-4241. 25130:267.

Licensing and regulation of commercial low-level waste (CLLW) burial facilities require that anticipated risks associated with burial sites be evaluated for the life of the facility. This work reviewed the existing capability to evaluate dose to man resulting from the potential redistribution of buried radionuclides by plants and animals. Through biotic transport, radionuclides can be moved to locations where they can enter exposure pathways to man. We found that predictive models currently in use did not address the long-term risks resulting from the cumulative transport of radionuclides. Although reports in the literature confirm that biotic transport phenomena are common, assessments routinely ignore the associated risks or dismiss them as insignificant. To determine the potential impacts of biotic transport, we made order-of-magnitude estimates of the dose to man for biotic transport processes at reference arid and humid CLLW disposal sites. Estimated doses to site residents after assumed loss of institutional control were comparable to dose estimates for the intruder-agricultural scenario defined in the DEIS for 10 CFR 61 (NRC). The reported lack of potential importance of biotic transport at low-level waste sites in earlier assessment studies is not confirmed by order of magnitude estimates presented in this study.

NUREG/CR-2679 V03: ADVANCED REACTOR SAFETY RESEARCH QUARTERLY REPORT, JULY-SEPTEMBER 1982. * Sandia Laboratories. March 1984. 244pp. 8404020211. SAND82-0904. 22877:001.

Sandia National Laboratories is conducting, under USNRC's sponsorship, phenomenological research related to the safety of commercial nuclear power reactors.

The overall objective of this work is to provide NRC a comprehensive data base essential to (1) defining key safety issues, (2) understanding risk-significant accident sequences, (3) developing and verifying models used in safety assessments, and (4) assuring the public that power reactor systems will not be licensed and placed in commercial service in the United States without appropriate consideration being given to their effects on health and safety.

Together with other programs, the Sandia effort is directed at assuring the soundness of the technology base upon which licensing decisions are made.

This report describes progress in a number of activities dealing with current safety issues relevant to both light water and breeder reactors. The work includes a broad range of experiments to simulate accidental conditions to provide the required data base to understand important accident sequences and to serve as a basis for development and verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Such a program must include the development of analytical models, verified by experiment, which can be used to predict reactor and safety system performance under a broad variety of abnormal conditions.

Current major emphasis is focused on providing information to NRC relevant to (1) its deliberations and decisions dealing with sever LWR accidents, and (2) its safety evaluation of the proposed Clinch River Breeder Reactor.

NUREG/CR-2679 V04: ADVANCED REACTOR SAFETY RESEARCH, QUARTERLY REPORT, OCTOBER-DECEMBER 1982. * Sandia Laboratories. April 1984. 207pp. 8406210433. SAND82-0904. 25100:097.

This report describes progress in a number of activities dealing with current safety issues relevant to both light water reactors (LWRs) and breeder reactors. The work includes a broad range of experiments to simulate accidental conditions to provide the required data base to understand important accident sequences and to serve as a basis for development and verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Such a program must include the development of analytical models, verified by experiment, which can be used to predict reactor and safety system performance under a broad variety of abnormal conditions.

Current major emphasis is focused on providing information to NRC relevant to (1) its deliberations and decisions dealing with severe LWR accidents and (2) its safety evaluation of the proposed Clinch River Breeder Reactor.

NUREG/CR-2691: EFFECTS OF CLADDING SURFACE THERMOCOUPLES AND ELECTRICAL HEATER ROD DESIGN ON QUENCH BEHAVIOR. GOTTULA, R. C. EG&G, Inc. April 1984. 105pp. 8405220051. EGG-2186. 24551:225.

A separate effects experiment program was conducted on a bundle of nine electrical heater rods in the Loss-Of-Fluid Test (LOFT) Test Support Facility (LTSF). The objective of the experiment program were to (a) evaluate the effect of cladding external thermocouples on the quench (cooling) behavior of a cartridge-type nuclear fuel rod simulator, (b) determine how accurately cladding external thermocouples measure cladding temperature during a high pressure quench, (c) provide a functional and reliability test for cladding-embedded thermocouples that are prototypes of a design to be used in the LOFT fuel rods, and (d) compare the quench behavior of a cartridge-type heater rod (which simulates a fuel pellet-cladding gap) with that of a solid-type heater rod (without a pellet-cladding gap) under thermal-hydraulic conditions that could occur during the blowdown phase (0 to 10 s) of a large-break loss-of-coolant accident in a pressurized water reactor. The prototype cladding-embedded thermocouples did not function correctly during the experiment; however, useful data were obtained such that the objectives of the experiment program could be met.

NUREG/CR-2721: SCOPING STUDY OF THE ALTERNATIVES FOR MANAGING WASTE CONTAINING CHELATING DECONTAMINATION CHEMICALS. PREMUZIC, E. T.; MANAKTOLA, H. K. Brookhaven National Laboratory. February 1984. 56pp. 8402240378. BNL-NUREG-51593. 22380:309.

Selected trench waters from several low-level waste disposal sites have been analyzed for the presence of chelating decontamination chemicals. Several methods for decomposition of chelating agents used in decontamination processes are discussed. Specifically, nitriloacetic acid (NTA), ethylenediamine tetra acetic acid (EDTA), and diethylene triamine pentaacetic acid (DTPA) decomposition properties under thermal, biological, oxidative, and photochemical conditions are reviewed and discussed in the light of currently available information. Based on the analysis of the information, it is concluded that combustion and oxyphotolysis may be worth further exploration as possible methodologies for the degradation of chelating decontamination chemicals.

NUREG/CR-2803: IMPROVED FIELD EXPERIMENTAL DESIGNS AND QUANTITATIVE EVALUATION OF AQUATIC ECOSYSTEMS. MCKENZIE, D. H.; THOMAS, J. M.

Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 31pp. 8405210607. PNL-4138 24534:242.

We used the paired-station concept and a log transformed analysis of variance methods to evaluate zooplankton density data collected during five years at an electrical generation station on Lake Michigan. To discuss the example and the field design necessary for a valid statistical analysis, we provide considerable background on the questions of selecting 1) sampling station pairs, 2) experimentwise error rates for multi-species analyses, 3) levels of Type I and II error rates, 4) procedures for conducting the field monitoring program, and 5) a discussion of the consequences of violating statistical assumptions. We include details for estimating sample sizes necessary to detect changes of a specified magnitude.

Both statistical and biological problems with monitoring programs (as now conducted) are addressed; serial correlation of successive observations in the time series obtained was identified as one principal statistical difficulty. Our procedure reduces this problem to a level where statistical methods can be used confidently.

NUREG/CR-2810: VARIATIONS IN ZIRCALOY-4 CLADDING DEFORMATION IN REPLICATE LOCA SIMULATION TESTS. LONGEST, A. W. ; CROWLEY, J. L. ; CHAPMAN, R. H. Oak Ridge National Laboratory. October 1982. 54pp. 8307210260. ORNL/TM-8413. 19688:167.

Five single-rod, heated-shroud replicate burst tests were conducted to study statistical variations in Zircaloy cladding deformation under simulated loss-of-coolant accident conditions. The test conditions used (low steam coolant flow and a heating rate of ~10 K/s to tube failure at ~775 C) were conducive to large deformation and matched those used in two of the Multirod Burst Test Program bundle tests so that the results could be used to aid in interpretation of differences observed for individual rods in bundle tests.

The results established estimates of variations that can be expected for freely deforming tubes under these test conditions. The data also indicated a potential for rod-to-rod mechanical interactions in a large bundle.

NUREG/CR-2812: THE RELATIVE IMPORTANCE OF TEMPERATURE, PH AND BORIC ACID CONCENTRATION ON RATES OF H2 PRODUCTION FROM GALVANIZED STEEL CORROSION. LOYOLA, V. M. ; WOMELSDUFF, J. E. Sandia Laboratories. January 1984. 31pp. 8402230378. SAND82-1179. 22376:177.

The corrosion of galvanized steel, to produce hydrogen gas, will occur if sprays operate during a Loss-of-Coolant Accident in a Light Water Reactor. The rates of hydrogen generation, however, are variable and dependent on accident and post-accident conditions. This report describes a study designed to identify the important parameters (temperature, pH, and boric acid concentration) in determining the rates of hydrogen generation from Light Water Reactor containment building spray solutions. The data are gathered over a wide range of temperature, pH, and boric acid concentration, and are used in a two-level, three-factor factorial experiment to determine the relative importance of the three parameters to the hydrogen generation process. A statistical treatment of the data gives an indication of the relative importance of the parameters (temperature, pH, boric acid concentration) and of their interactions. It attempts to fit the data to a relatively simple equation to model the interactions of the various parameters.

NUREG/CR-2815: PROBABILISTIC SAFETY ANALYSIS PROCEDURES GUIDE. BARI, R. A. ; PAPAZOGLOU, I. A. ; BUSLIK, A. J. ; et al. Brookhaven National Laboratory. January 1984. 234pp. 8402060484. BNL-NUREG-51559. 22109:014.

A procedures guide for the performance of probabilistic safety assessment has been prepared for interim use in the Nuclear Regulatory Commission programs. It will be revised as comments are received, and as experience is gained from its use. The probabilistic safety assessment studies performed are intended to produce probabilistic predictive models that can be used and extended by the utilities and by NRC to sharpen the focus of inquiries into a range of issues affecting reactor safety. This guide addresses the determination of the probability (per year) of core damage resulting from accident initiators internal to the plant, and from loss of offsite electric power. The scope includes analyses of problem-solving (cognitive) human errors, a determination of importance of the various core damage accident sequences, and an explicit treatment and display of uncertainties for the key accident sequences. Ultimately, the guide will be augmented to include the plant-specific analysis of in-plant processes (i.e., containment performance) and the risk associated with external accident initiators, as consensus is developed regarding suitable methodologies in these areas. This guide provides the structure of a probabilistic safety study to be performed, and indicates what products of the study are essential for regulatory decision making. Methodology is treated in the guide only to the extent necessary to indicate the range of methods which is acceptable; ample reference is given to alternative methodologies which may be utilized in the performance of the study.

NUREG/CR-2824 V01: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1982. DODD, C. V. ; DEEDS, W. E. ; MCCLUNG, R. W. Oak Ridge National Laboratory. October 1982. 9pp. 8307210268. ORNL/TM-8418/V1. 19672:304.

Eddy-current methods provide the best in-service inspection of steam generator tubing, and these techniques can produce ambiguity because of the many independent variables that affect the signals. This development program has used mathematical models and has developed or modified computer programs to design optimum probes, instrumentation, and techniques for multifrequency multiproperty examinations. Interactive calculations and experimental measurements have been made with modular eddy-current instrumentation and a minicomputer. These establish the coefficients for the complex equations that define the values of the desired properties (and the attainable accuracy) despite changes in other significant variables. The computer programs for calculating the accuracy with which various properties can be measured indicate that the tubing wall thickness and the defect size can be measured much more accurately than is required, even when other calculations show that an array of small pancake coils pressed against the inner wall of the tubing can detect and locate small flaws on the outer wall of the tubing with much greater accuracy and reliability than can the usual large circumferential coils. We are continuing to investigate such arrays.

NUREG/CR-2869 R01: DIRECTORY AND PROFILE OF LICENSED URANIUM-RECOVERY FACILITIES. WHITE, W. S. Argonne National Laboratory. March 1984. 115pp. 8404130132. ANL/ES-128 R01. 24048:221.

The DIRECTORY AND PROFILE OF LICENSED URANIUM-RECOVERY FACILITIES presents facts, data, and information about conventional mills, in-situ mining facilities, heap leach operations, and other operations which process and produce marketable quantities of yellowcake. In the United States, such facilities are found in Agreement States (Arizona, Colorado, Florida, Louisiana, New Mexico, Texas, and Washington) and in Non-Agreement States (South Dakota, Utah and Wyoming).

Each facility is described on a case-by-case basis. Reporting of information on the conventional uranium mills begins with a brief narrative description that outlines general and specific characteristics about the site. Data sheets summarize the principal operating characteristics of the facility by listing the following information: location/ownership, licensing data, processing of uranium, characteristics of effluent releases and/or tailings, and radiological parameters. For in-situ and heap leach facilities, only data sheets are included.

NUREG/CR-2896 V01: COMMIX-1A: A THREE-DIMENSIONAL TRANSIENT SINGLE-PHASE COMPUTER PROGRAM FOR THERMAL HYDRAULIC ANALYSIS OF SINGLE AND MULTICOMPONENT SYSTEMS. Volume I: User's Manual. DOMANUS, H. M. ; SCHMITT, R. C. ; SHA, W. T. ; et al. Argonne National Laboratory. March 1984. 327pp. 8403300261. ANL-82-25 V01. 22839:217.

The COMMIX-1A computer program, the improved version of COMMIX-1, is designed to analyze steady-state/transient, single-phase, three-dimensional fluid flow with heat transfer in reactor components and multicomponent systems. The concepts of volume porosity, directional surface permeability, distributed resistance, and distributed heat source or sink is used to model a flow domain with stationary structures. The new porous-media formulation permits simulation of either a single component or a multicomponent system. The conservation equations of mass, momemtum, and energy based on the new porous-media formulation are solved as a boundary-value problem in space and an initial-value problem in time.

This report (Volume I) describes in detail the basic equations, formulations, solution procedures, flow charts, rebalancing scheme for faster convergency, options available to users, models to describe the auxiliary phenomena, input instructions, and two sample problems. Volume II assembles and summarizes the results of many simulations that have been performed with the COMMIX-1A computer program.

NUREG/CR-2896 V02: COMMIX-1A: A THREE-DIMENSIONAL TRANSIENT SINGLE-PHASE COMPUTER PROGRAM FOR THERMAL HYDRAULIC ANALYSIS OF SINGLE AND MULTICOMPONENT SYSTEMS. Volume II: Assessment And Verification. DOMANUS, H. M. ; SCHMITT, R. C. ; SHA, W. T. Argonne National Laboratory. February 1984. 95pp. 8403300285. ANL-82-25 V02. 22840:182.

This report assembles and summarizes the results of many simulations that have been performed with COMMIX-1A computer program since the beginning of its development in 1976. The COMMIX-1A is a three-dimensional, time-dependent, single-phase computer program for thermal-hydraulic analysis of component/multicomponent systems under normal/off-normal operating conditions. This report compiles most simulations for which comparisons with experimental measurements or analytical solutions have been made.

NUREG/CR-2907 V02: RADIOACTIVE MATERIALS RELEASED FROM NUCLEAR POWER PLANTS. Annual Report 1981. TICHLER, J. ; BENKOVITZ, C. Brookhaven National Laboratory. June 1984. 213pp. 8407170576.

BNL-NUREG-51581. 25631:033.

Releases of radioactive materials in airborne and liquid effluents from commercial light water reactors during 1981 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 1981 release data are compared with previous years' releases in tabular form. Data covering specific radionuclides are summarized.

NUREG/CR-2921: CHEMICAL INTERACTIONS OF TELLURIUM VAPORS WITH REACTOR MATERIALS. SALLACH, R. A. ; GREENHOLT, C. J. ; TAIG, A. R. Sandia Laboratories. April 1984. 70pp. 8405220180. SAND82-1145. 24602:115.

The reaction of tellurium vapor with 304 stainless steel and Inconel-600 alloys in an as-received state and in a preoxidized state was studied for the temperature range 500C to 800C. Most reaction products were identified. The reaction is fast and appears largely limited by tellurium transport through the surrounding gas phase.

Also studied are the reactions of tellurium vapor with silver Zircaloy-2. Tellurium desorption rates from solid solutions of tellurium in nickel and 304 stainless steel were measured. The FLATDEP model for calculating tellurium deposition profiles is presented.

NUREG/CR-2926: SIMS AND ESCA STUDIES OF POSSIBLE SODIUM URANATE PRECURSORS AS RELATED TO AEROSOL CHARACTERIZATION FROM A SIMULATED HCDA. ZANOTELLI, W. A. ; MILLER, G. D. ; CRAVEN, S. M. Mound Facility/Monsanto Research Corp. October 1982. 18pp. 8307210255. MLM-2983. 19688:297.

During the main thrust of the HCDA studies, it was found that sodium uranates, especially (3)NaU(4)O, were formed when the Na-U-O system was subjected to high temperatures approximating those of the HCDA. Mechanisms through which these rather complicated compounds are formed remain unknown. The purpose of these SIMS and ESCA studies was to detect the formation of any precursor ion species to the sodium uranates. The main species detected from the Ar+ excited positive SIMS analyses of oxides in salts of uranium were (2)UO+, UO+, and U+. The main ion detected from the Ar+ excited SIMS analyses of a (2)Na(2)U(7)O or of a uranium dioxide - disodium oxide pellet or from a sodium film deposited on a uranium metal foil. ESC analyses show peak shapes and binding energies for the (2)Na(2)U(7)O pellet that are different from those for the U-Na foil samples and the uranium dioxide - disodium oxidee pressed pellet. The ESCA results agree with theory and support the presence of (2)U(7)O = in (2)Na(2)U(7)O; however, SIMS analyses show no evidence of possible uranate precursor formation in an Ar+ sputtered ion beam.

NUREG/CR-2932 V02: EQUIPMENT QUALIFICATION RESEARCH TEST OF ELECTRIC CABLE WITH FACTORY SPLICES AND INSULATION REWORK TEST NO. 2, REPORT NO. 2. MINOR, E. E. ; FURGAL, D. T. Sandia Laboratories. November 1982. 65pp. 8307210257. SAND81-2027. 19688:222.

Electric cables with fire-retardant chemically crosslinked polyolefin extruded insulation containing factory-made center-conductor splices and insulation repairs manufactured by General Electric Company were used in a methodology test of the IEEE Standard 383-2974. This standard is concerned with the ability of cables to function during and following exposure to aging and LOCA/MSLB environments. Cable specimens were radiation aged at a low-dose rate and then thermally aged to simulate a 40-year containment exposure. After aging, the specimens were subjected to LOCA radiation and a 33-day steam and chemical spray exposure. The cables were electrically loaded and functioned without failure during and after LOCA steam and chemical spray exposure. Insulation resistance measurements were taken during the exposure sequence. Subsequent to the exposures, hipot and mandrel bend tests were conducted. Test results indicate that the methods given in IEEE 383-1974 are adequate to show that cables can function and support control operations during and after a LOCA/MSLB of the severity simulated by the test. Further, the presence of center-conductor splices and insulation repairs did not appear to degrade cable performance.

NUREG/CR-2940: REALISTIC SIMULATION OF SEVERE ACCIDENTS IN BWRS-COMPUTER MODELING REQUIREMENTS. GREENE, S. R. Oak Ridge National Laboratory. April 1984. 237pp. 8405220029. ORNL/TM-8517. 24557:017.

This report documents the results of an assessment performed at Oak Ridge National Laboratory to determine the reactor and containment hardware, systems, and phenomena which must be modeled in realistic boiling water reactor severe accident analysis computer codes. The scope of the assessment is limited to BWR-4, 5, and 6 reactors and Mark I, II, and III containment systems. The report presents a concise review of the subject reactor and containment designs, together with a description of the reactor and containment systems which have the capacity to impact the outcome of severe accidents. the results of recent BWR probabilistic risk assessments are briefly discussed, and a detailed visualization of a BWR core melt accident is presented. Recommendations are made regarding the type of phenomena which should be modeled and the level of modeling sophistication required form various stages of the core melt accident. Finally, the current availability of the necessary models is discussed along with the associated model development priorities.

NUREG/CR-2955: ANALYSIS OF URANIUM URINALYSIS AND IN VIVO MEASUREMENT RESULTS FROM ELEVEN PARTICIPATING URANIUM MILLS. SPITZ, H. B. SIMPSON, J. C. ALDRIDGE, T. L. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 50pp. 8405310117. PNL-4550. 24736:241.

Uranium urinalysis and in vivo examination results obtained from workers at eleven uranium mills between 1978 and 1980 were evaluated by Pacific Northwest Laboratory at the request of the U.S. Nuclear Regulatory Commission. The main purpose of this evaluation was to determine the degree of the mills' compliance with bioassay monitoring recommendations given in the draft NRC Regulatory Guide 8.22. The effect of anticipated changes in the draft guidance, as expressed to PNL in May 1982, was also studied. Statistical analyses of the data showed that the bioassay results did not reliably meet the limited performance criteria given in the draft regulatory guide. Furthermore, quality control measurements of uranium in urine indicated that detection limits at alpha=beta= 0.05 ranged from 13 miligrams/ to 29 miligrams/, whereas the draft regulatory guidance suggests 5 miligrams/ as the detection limit. Recommendations for monitoring frequencies given in the draft guide were not followed consistently from mill to mill. The results of these statistical analyses indicate a need to include performance criteria for accuracy, precision, and confidence in revisions of the draft regulatory guide. Revised guidance should also emphasize the need for each mill to continually test the laboratory performing urinalysis by submitting quality control samples to insure that the performance criteria are being met.

NUREG/CR-2970 V04: MATERIALS SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM: QUARTERLY PROGRESS REPORT OCTOBER-DECEMBER 1982. SHACK, W. J.; REST, J.; KASSNER, T. F.; et al. Argonne National Laboratory. January 1984. 125pp. 8401160443. ANL-82-41. 21824:001.

This progress report summarizes the Argonne National Laboratory work performed during October, November, and December 1983 on water reactor safety problems. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Transient Fuel Response and Fission Product Release, Clad Properties for Code Verification and Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems.

NUREG/CR-2996: SENSITIVITY OF DETECTING IN-CORE VIBRATIONS AND BOILING IN PRESSURIZED WATER REACTORS USING EX-CORE NEUTRON NOISE. SWEENEY, F. J.; RENIER, J. P. Oak Ridge National Laboratory. July 1984. 98pp. 8409110086. ORNL/TM-8549. 26447:003.

Neutron transport and diffusion theory space- and energy-dependent reactor kinetics calculations were performed in the frequency domain to determine the sensitivity of an ex-core neutron detector to in-core vibrations and coolant boiling in a pressurized water reactor (PWR). The results of these calculations indicate that the ex-core detectors are sensitive to neutron sources, to vibrations, and to boiling occurring over large regions of the core. Calculations were also performed to determine the effects of fuel burnup, boron concentration, and xenon poisoning on the spatial sensitivity of the ex-core detector. These calculated results show that fuel assembly vibrations would be expected to produce ~60% greater ex-core detector response at the end of the first fuel cycle at Sequoyah-1 compared to the beginning of the fuel cycle for a constant amplitude of vibration. The results were compared with experimental ex-core neutron noise data obtained from Sequoyah-1 during the first fuel cycle. The predicted increase in ex-core neutron noise was experimentally observed in the 2.5- to 4.0-Hz frequency range (the range of frequencies associated with fuel assembly vibration), indicating that the vibrational amplitude of the fuel assemblies did not increase significantly during the first fuel cycle.

NUREG/CR-3023: MOLTEN THERMITE TEEMING INTO AN IRON OXIDE PARTICLE BED. TARBELL, W. W.; BLOSE, R. E.; ARELLANO, F. E. Sandia Laboratories. April 1984. 80pp. 8405220033. SAND82-2475. 24552:280.

The two particle bed tests employed 10-kg thermite melts (2700 degree K) teemed into a bed of iron oxide particles. The objective of the experiments was to investigate bed penetration, particle floatation and fracture, and heat flux partitioning. The results show that the hydraulic forces exerted by the melt did not immediately displace the bed. Bed penetration was by melting and absorbing of the particles with the major portion of the displaced iron oxide terminating in the alumina phase of the melt. The movement of the penetration front suggests the movement to be a series of melt/freeze/remelt processes. The large grain structure of the iron phase indicates that the cooling was slow and continuous. A coherent 1-cm-thick layer of iron oxide in contact with the melt was created by sintering of the particles. The particle size of the unaffected portions of the bed showed very little fracturing due to thermal stress and slightly over 7% particle growth due to sintering. The calculated heat flux values to the surrounding crucible structure suggest that the bed is effective in delaying and reducing the magnitude of the peak heat flux values.

NUREG/CR-3025: HIGH-PRESSURE MELT STREAMING (HIPS) PROGRAM PLAN. TARBELL, W. W.; BROCKMANN, J. E.; PILCH, M. Sandia Laboratories. December 1984. 209pp. 8501040253. 28216:001.

The Zion Probabilistic Safety Study (ZPSS) envisions accident sequences that could lead to failure of the reactor vessel while the primary system is pressurized. The resulting ejection of molten core material into the reactor cavity followed by the blowdown of steam and hydrogen is shown to cause the debris to enter into the containment region. The High Pressure Melt Streaming (HIPS) program has been developed to provide an experimental and analytical investigation of the scenario described above. One-tenth linear scale models of the Zion cavity region will be used to investigate the debris dispersal phenomena. Smaller-scale experiments (SPIT-tests) are also used to study high-velocity jets, jet-water interactions, and 1/20th scale cavity geometries. Both matrices are developed using a factorial design approach. The document describes certain aspects of the ZPSS ex-vessel phenomena, the experimental matrices, test equipment, and instrumentation, and the program's analytical efforts. Preliminary data from SPIT testing are included.

NUREG/CR-3052: CLOSEOUT OF IE BULLETIN 80-07: BWR JET PUMP ASSEMBLY FAILURE. DEAN, R. S.; MILLS, W. R.; FOLEY, W. J.; et al. Parameter, Inc. November 1984. 59pp. 8412070166. IEB-80-07. 27839:062.

In February 1980, disassembly of a jet pump at Dresden 3 was diagnosed from changes in operating parameters. After prompt shutdown, it was found that a broken hold-down beam had caused the failure and that six other beams had small cracks. In March 1980, one cracked beam at Quad Cities 2 and three at Pilgrim 1 were discovered, and an earlier pump failure at a foreign facility was found to be like that at Dresden 3. IE Bulletin 80-07 was issued April 4, 1980 to licensees of all General Electric BWR/3 and BWR/4 operating facilities to require daily operability surveillance of jet pumps and nondestructive examinations every refueling outage. The bulletin was issued for information to holders of construction permits for General Electric facilities; later, 13 of these facilities were selected for written responses. Extensive studies led to the conclusion that failures were caused by very slowly progressing stress corrosion cracking, and resulted in manufacture of improved beams. Bulletin status is determined by applying closeout criteria. Closeout of bulletin Item B.2 requiring operability surveillance is based on the short-term action of implementing an acceptable method and the long-term action of continuing that method until satisfactory corrective action has been completed. Followup items are suggested for all 20 operating facilities to ensure compliance with bulletin requirements and intent. The safety significance of a jet pump failure is that flow distribution would be affected during normal operation and the water level in the core region would decrease during a coincidental loss of cooling accident.

NUREG/CR-3053: CLOSEOUT OF IE BULLETIN 80-08: EXAMINATION OF CONTAINMENT LINER PENETRATION WELDS. DEAN, R. S. ; FOLEY, W. J. ; HENNICK, A. Parameter, Inc. July 1984. 31pp. 8408130265. IEB-80-08. 26038:074.

During an NRC inspection at Nine Mile Point 2, examination by radiography of primary containment liner penetration sleeve-to-process pipe (flued head fitting) welds revealed rejectable defects not originally found by ultrasonic examination. Apparently, ultrasonic signals from the weld backing bar masked signals from defects. Further investigation found similar problems at Beaver Valley 2 and North Anna 3 and 4. IE Bulletin 80-08 was issued to acquire information from all facilities to determine the generic nature of the problem. It was found that, because of evolution of the ASME Nuclear Code, plants under construction designed to that Code since about 1974 are required to volumetrically examine these welds, and so, in general, do not have the problem. Operating plants, built to earlier codes not requiring such design and examination for the containment welds, present a concern for the quality of this type of weld and for the integrity of the primary containment boundary. Bulletin status is closed for all but 11 facilities. Recommendations are made for resolution of the problem for these facilities. These include meaningful radiographic examination of welds of concern, if possible, and if not, licensee justification for not making a radiographic examination.

NUREG/CR-3054: CLOSEOUT OF IE BULLETIN 81-03: FLOW BLOCKAGE OF COOLING WATER TO SAFETY SYSTEM COMPONENTS BY CORBICULA SP. (ASIATIC CLAM) AND MYTILUS SP. (MUSSEL). RAINS, J. H. ; FOLEY, W. J. ; HENNICK, A. Parameter, Inc. June 1984. 59pp. 8406270113. IEB-81-03. 25173:282.

On April 10, 1981, the Office of Inspection and Enforcement (IE) of the U.S. Nuclear Regulatory Commission (NRC) issued Bulletin 81-03 requiring all nuclear generating unit licensees to assess the potential for biofouling of safety-related system components as a result of Asiatic clams (Corbicula sp.) and marine mussels (Mytilus sp.). Issuance of the Bulletin was prompted by the shutdown of Arkansas Nuclear One, Unit 2 on September 3, 1980, as a result of flow blockage of safety systems by Asiatic clams. Licensee responses to Bulletin 81-03 have been compiled and evaluated to determine the magnitude of existing biofouling problems and potential for future problems. An assessment of the real extent of Asiatic clam and marine mussel infestation has been made along with an evaluation of detection and control procedures currently in use by licensees. Recommendations are provided with regard to adequacy of detection, inspection and prevention practices currently in use, biocidal treatment programs, and additional areas of concern. Safety implications and licensee responsibilities are discussed. Of 79 facilities licensed to operate, 17 have reported biofouling problems, 21 are judged to have high biofouling potential, 17 are judged to have low or future potential, and 24 are judged to have little or no potential. For 49 facilities under construction, the number of units for matching conditions of biofouling are 3, 25, 15, and 6 in the same decreasing order of severity. The Bulletin has been closed out for 85 of 129 current facilities. Followup needed to close out the Bulletin for 21 operating facilities and 23 facilities under construction is proposed in Appendix C.

NUREG/CR-3056: POPULATION DISTRIBUTION ANALYSIS FOR NUCLEAR POWER PLANT SITING. DURFEE, R. C. ; COLEMAN, P. R. Oak Ridge National Laboratory.

December 1983. 104pp. 8402170439. ORNL/CSD/TM-197. 22302:166.

The Nuclear Regulatory Commission is in the process of reviewing guidelines and regulations associated with population distribution criteria around nuclear power plant sites. The purpose of this paper is to describe the methodology for calculating population distribution in the U.S. and then evaluating specific population criteria and their effect upon the selection of future nuclear power plant sites. Through the use of computer systems, different alternatives may be evaluated for individual sites or for major regions of the country to determine their restrictiveness on siting nuclear plants. Two types of criteria were used. They involved the analysis of population distributions radially out from each possible site and the study of angular distributions around each site. Results are presented in both tabular and graphic form using national, regional, and site-level computer maps.

NUREG/CR-3057: ANALYSIS OF AVAILABILITY OF PREVIOUSLY IDENTIFIED SITES UNDER ALTERNATIVE DEMOGRAPHIC CRITERIA. KELLY, M. J.; RUSH, R. M.; OTT, W. R.; et al. Oak Ridge National Laboratory. January 1984. 44pp. 8402170417. ORNL-5936. 22320:041.

This study evaluates the effect of various alternative population criteria on the availability of suitable sites for new nuclear power plants. The population criteria investigated include population density limits of 500 and 750 persons per square mile within 30 miles of the site and three different limits on the total population in adjacent sectors. The site availability in the states with higher population density was investigated--using power pools, regional aggregations, and specific sites as appropriate. The results of this study indicate that the application of any of the alternative population criteria studied will not rule out the use of the nuclear option in the northeast quadrant of the United States. The analysis demonstrates that viable sites exist even in the states and service areas with the largest population densities. These results, together with a cursory examination of the rest of the United States indicating that both real and potential viable sites are available, confirm that the application of any of the alternative population criteria studies will not preclude the identification of suitable nuclear sites in any region.

NUREG/CR-3067: CALIBRATION SOURCES FOR THE G-M COUNTER USED WITH THE BNL AIR SAMPLER. HUCHTON, R. L.; BIRD, S. K.; TKACHYK, J. W.; et al. Exxon Nuclear Co. Inc. (subs. of Exxon Corp.). February 1984. 52pp. 8403060533. ENICO-1125. 22522:064.

Three calibration sources were designed, developed, and fabricated for a CDV-700 ratemeter equipped with a specially-shielded 6306 G-M detector. The CDV-700/6306 has been proposed for use with the BNL Air Sampler designed for radioiodine monitoring upon a nuclear reactor accident.

Specifically, the three sources were constructed in a geometry identical to the BNL air sampler radioiodine absorption canister, which is a silver silica gel filled 2.75-inch diameter right circular cylinder with a 1.0-inch diameter annulus, into which the 6306 G-M detector is inserted. As fabricated, each source consisted of an outer stainless steel housing, an inner (133)Ba impregnated polyester liner, 4-weight percent silver silica gel media, and a laser welded hermetically-sealed stainless steel lid. Respectively, the levels of (133)Ba, an (131)I simulant, were varied in the three sources to yield nominal CDV-700/6306 instrument responses of 200 cpm, 2,000 cpm and 20,000 cpm.

NUREG/CR-3070: MEASURED IN-REACTOR DATA AND POSTIRRADIATION OBSERVATIONS FOR IFA-527. CUNNINGHAM, M. E. ; LANNING, D. D. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 36pp. 8401310449. PNL-4542. 22036:234.

Preirradiation characterization data and irradiation data are summarized and postirradiation examination (PIE) data are presented for the six-rod instrumented fuel assembly IFA-527. This assembly was irradiated in the Halden Boiling Water Reactor, Halden, Norway, as part of the U.S. Nuclear Regulatory Commission-sponsored Experimental Support and Verification of Single-Rod Fuel Codes Program. The assembly was irradiated from July 1, 1980, to April 8, 1981, during which time rod-average burnups of approximately 1.0 MWd/kgM were obtained. All six rods had pressure boundary failures prior to removal from the reactor. The PIE was conducted at the Harwell laboratory of the United Kingdom's Atomic Energy Research Establishment. Examinations were oriented toward discovering the site and cause of rod failure (visual, neutron radiography, leak checking) and the effects of operating an assembly with failed rods (micrography of fuel and cladding). It was concluded that rod failures occurred in the end fittings and not in the cladding. No significant fuel or cladding microstructural changes were observed.

NUREG/CR-3071: IRRADIATION HISTORY AND INTERIM POSTIRRADIATION DATA FOR IFA-432. LANNING, D. D. ; BRADLEY, E. R. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1984. 189pp. 8404160201. PNL-4543. 24065:042.

Preirradiation characterization data and measured in-reactor data are summarized for the six-rod instrumented Halden fuel assembly IFA-432. Postirradiation data are presented on three of the rods--Rods 1, 6, and 8--from this assembly. Rods 1 and 6 operated from December 1975 to June 1981 and accumulated peak burnups of 34 MWd/kgM. Fuel temperatures, gas pressures, and rod elongation were measured throughout the life of these rods, together with the corresponding power histories. Postirradiation observations on the rods are correlated with their design features and operating history. End-of-life fission gas release data demonstrate the influence of both as-fabricated grain size and the operating history.

Rods 1 and 6 experienced classic thermal feedback between fuel temperatures and gas release but had different rates of response, such that they attained lifetime peak temperatures at different burnups. Detailed measurements of the distribution of retained gas were performed on these two rods, and the results are compared with measured and calculated temperatures and with computer code predictions.

NUREG/CR-3091 V03: REVIEW OF WASTE PACKAGE VERIFICATION TESTS. Semiannual Report Covering The Period April-September 1983. SOO, P. Brookhaven National Laboratory. February 1984. 164pp. 8403070115. BNL-NUREG-51630. 22562:176.

This report is part of an ongoing task to specify tasks that may be used to verify that engineered waste package/repository systems meet the containment and controlled release performance objectives of 10 CFR Part 60. This report analyzes verification tests form carbon steel container corrosion and the testing methodologies that are appropriate for evaluating interaction effects between adjacent components in waste packages.

NUREG/CR-3094: SECONDARY SIDE PHOTOGRAPHIC TECHNIQUES USED IN CHARACTERIZATION OF SURRY STEAM GENERATOR. SINCLAIR, R. B. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1984. 45pp. 8411130636. PNL-5053. 27477:112.

Characterization of the generator's secondary side prior to destructive removal of tubing presents a significant challenge. Information must be obtained in a radioactive field (up to 15 R/hr) throughout the tightly spaced bundle of steam generator tubes. This report discusses the various techniques employed, along with their respective advantages and disadvantages. The most successful approach to nondestructive secondary side characterization and documentation was through use of in-house developed pinhole cameras. These devices provided accurate photographic documentation of generator condition. They could be fabricated in geometries allowing access to all parts of the generator. Semi-remote operation coupled with large area coverage per investigation and short at-location times resulted in significant personnel exposure advantages. The fabrication and use of pinhole cameras for remote inspection is discussed in detail.

NUREG/CR-3104: AQUIFER RESTORATION TECHNIQUES FOR IN-SITU LEACH URANIUM MINES. DEUTSCH, W. J.; BELL, N. E.; MERCER, B. W. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1984. 58pp. 8403230177. PNL-4583. 22759:121.

In-situ leach uranium mines and pilot-scale test facilities are currently operating in the states of Wyoming, Texas, New Mexico and Colorado. This report summarizes the technical considerations involved in restoring a leached ore zone and its aquifer to the required level. Background information is provided on the geology and geochemistry of mineralized roll-front deposits and on the leaching techniques used to extract the uranium.

NUREG/CR-3127: PROBABILISTIC SEISMIC RESISTANCE OF STEEL CONTAINMENTS. GREIMANN, L.; FANOUS, F.; KETELAAR, D.; et al. Iowa State Univ., Ames, IA. January 1984. 132pp. 8402270207. 22397:225.

A probabilistic description of the seismic resistance of six containment vessels was developed. A random vibration approach and an advanced first order second moment reliability method were combined to predict the cumulative distribution of the resistance. Strain ductility was used as the failure criteria. BOSOR4 was used to determine linear vibration modes. Extreme Value Type I distributions were used to characterize the maximum responses (stress resultants). BOSOR5 was implemented to predict the proportional increase of these maximums which would cause buckling. Imperfections were included in the shell models. Only shell failure modes were considered.

NUREG/CR-3134: A SETS USER'S MANUAL FOR VITAL AREA ANALYSIS. STACK, D. W.; HILL, M. S. Sandia Laboratories. June 1984. 108pp. 8407170560. SAND83-0074. 25634:037.

This manual describes the use of the Set Equation Transformation System (SETS) for vital area analysis. Various techniques are presented for using SETS to solve vital area analysis fault trees. Depending on the input to SETS, the solution to the vital area analysis fault tree can be in terms of vital areas or primary events of the vital area analysis fault tree. The techniques presented are also suitable and efficient for other kinds of common cause analysis.

NUREG/CR-3139: SCENARIOS AND ANALYTICAL METHODS FOR UF6 RELEASES AT NRC-LICENSED FUEL CYCLE FACILITIES. SIMAN-TOV, M. ; DYKSTRA, J. ; HOLT, D. D. ; et al. Oak Ridge National Laboratory. July 1984. 97pp. 8408130007. ORNL/ENG/TM-25. 26038:203.

This report identifies and discusses potential scenarios for the accidental release of UF(6) at NRC-licensed UF(6) production and fuel fabrication facilities based on a literature review, site visits, and DOE enrichment plant experience. Calculational methods needed for analyzing such releases are also reviewed. Accident scenarios are presented under the headings of cylinder failures, process system failures, criticality events, and operator errors and are categorized by location, release source, UF(6) phase prior to release, release flow characteristics, release causes, initiating events, and UF(6) inventory at risk. Releases identified for further examination include: (1) a release from a cylinder outdoors, (2) a release from a pigtail or cylinder in a steam chest, and (3) an indoor release from either (a) a pigtail or cylinder or (b) other indoor source depending on facility design and operating procedures. Indoor release phenomena may be analyzed using a time-dependent homogeneous compartment model or a more complex hydrodynamic model if time-dependent, spatial variations in concentrations, temperature, and pressure are important. Analytical tools for modeling directed jets and explosive releases are discussed as well as some of the complex phenomena to be considered in analyzing UF(6) releases both indoors and outdoors.

NUREG/CR-3145 V02: GEOPHYSICAL INVESTIGATION OF THE WESTERN OHIO - INDIANA REGION. Annual Report, October 1982 - September 1983. POLLACK, H. N. ; CHRISTENSEN, D. Michigan, Univ. of, Ann Arbor, MI. March 30, 1984. 64pp. 8403230131. 22739:232.

Earthquake activity in the Western Ohio-Indiana region is monitored with a precision seismograph network which consists of nine stations located in west central Ohio and four stations sited in Indiana. Five local and near regional earthquakes have been recorded during this report period. Three events were located outside the array, near the cities of Kalamazoo, Michigan, Cleveland, Ohio and Cincinnati, Ohio. Two events occurred in the center of the Ohio array. A focal mechanism was calculated for the larger of these two events. This focal mechanism shows mainly strike slip motion on steeply dipping nodal planes, striking at N35 degrees-45 degrees E and N50 degrees-70 degrees W. Both of these planes correspond to local structures.

NUREG/CR-3153: CALCULATION OF FLUID CIRCULATION PATTERNS IN THE VICINITY OF SUBMERGED JETS USING ORSMAC. PARK, J. E. ; CROSS, K. E. Oak Ridge National Laboratory. December 1983. 130pp. 8403260286. ORNL/TM-8653. 22761:176.

As the world demand for electricity is met by large coal- or nuclear-fueled central generating stations, the effluent streams from these plants will have an increasingly important impact on the local environment. The Nuclear Regulatory Commission has a responsibility to assess the impact of proposed and operating nuclear power plants.

To support this NRC mission, a numerical algorithm and associated computer program have been developed to predict the temperatures occurring in the immediate vicinity (the near field) of a hot water discharge from a power plant. The algorithm is a natural extension of the classic Marker-and-Cell (MAC) technique developed by F.H. Harlow at the Los Alamos Scientific Laboratory. ORSMAC (Oak Ridge Simplified Marker and Cell), adds the logic for simple turbulence modeling, energy conservation and buoyancy effects to the MAC model. Modern numerical techniques have been used wherever practical.

In this report, the MAC and SMAC (Simplified MAC) algorithms are reviewed, and the ORSMAC algorithm is described. The finite difference analogs are given and discussed. Solutions for several sample problems are presented which illustrate the features of the ORSMAC algorithm. A complete FORTRAN listing is included with input and sample output.

NUREG/CR-3169: SUPER SYSTEM CODE (SSC, REV. 0). AN ADVANCED THERMOHYDRAULIC SIMULATION FOR TRANSIENTS IN LMFBRS. GUPPY, J.G. Brookhaven National Laboratory. September 1984. 460pp. 8409260630. BNL-NUREG-51650. 26702:163.

The Super System Code (SSC) calculates the response of nuclear reactor systems during operational, incidental and accidental transients, especially natural circulation events. Modules simulated and parameters calculated include: core flow rates and temperatures, loop flow rates and temperatures, pump performance, and heat exchanger operation. Additionally, all plant protection systems and plant control systems are accounted for. All calculations are done in SI units.

SSC is a general system transient code. It is highly flexible, with complete variable dimensioning, allowing any number of user specified loops, pipes and nodes. Single phase and two phase thermal hydraulics are used in a multi-channel core representation. Interassembly flow redistribution is accounted for; a detailed fuel pin model is used. The heat transport system geometry is user specified. The code has both transient and steady state options. Restart capability is provided.

SSC is available in either a CDC UPDATE format or as FORTRAN source. The customary transmittal package also includes the input files for the three standard benchmark problems, as well as 48x microfiche which contain the SSC support documentation and sample output for each of the benchmark problems. SSC is currently available as a draft release from Brookhaven National Laboratory with NRC consent.

NUREG/CR-3171: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-2. OSBORNE, M.F.; LORENZ, R.A.; TRAVIS, J.R.; et al. Oak Ridge National Laboratory. March 1984. 68pp. 8404020277. 22873:223.

The second in a series of high-temperature fission product release tests was conducted for 20 min at about 1700 degrees C in flowing steam. The test specimen, a 20-cm-long section of a H.B. Robinson fuel rod that had been irradiated to a burnup of 28,000 MWd/t, was heated in an induction furnace mounted in a hot cell.

Posttest analyses of the furnace, the thermal gradient tube, filters, and other components of the experimental apparatus showed that about 50% of the (85)Kr, (137)Cs, and (129)I were released from the specimen during the test. In addition, approximately 2% of the (110m)Ag and (125)Sb along with smaller fractions of several other radionuclides were measured by gamma spectrometry. Spark-source mass spectrometric data from a limited number of samples showed significant releases of fission product tellurium and molybdenum, as well as structural (zirconium and tin) and furnace (primarily tungsten) materials. Metallographic examination of the fuel specimen revealed extensive fractures in the cladding, essentially complete oxidation to ZrO(2), and evidence of fuel-cladding interaction.

NUREG/CR-3172: FLOWER: A COMPUTER CODE FOR SIMULATING THREE-DIMENSIONAL FLOW, TEMPERATURE AND SALINITY CONDITIONS IN RIVERS, ESTUARIES AND COASTAL REGIONS. ERASLAN, A.; LIN, W.; SHARP, R.D. Oak Ridge National Laboratory. December 1983. 381pp. 8404030564. ORNL/TM-8401. 22878:176.

FLOWER is a three-dimensional computer code for simulating fast-transient, free-surface flow, temperature, and salinity conditions in rivers, estuaries, and coastal regions. The model also includes rotational effects (Coriolis force) and is capable of accomodating wind-stress coupling, a capability which enables the model to be applied to large water bodies with significant wind-driven currents, such as the Great Lakes. The mathematical formulation utilizes the integral form of the governing equations of the discrete-element method. In this method, interior flow regions are represented as rectangular elements of variable size, while impermeable boundary elements are constructed from truncated rectangles, thus allowing accurate representations of complex shorelines.

NUREG/CR-3181: QUANTITY AND NATURE OF LWR AEROSOLS PRODUCED IN THE PRESSURE VESSEL DURING CORE HEATUP ACCIDENTS - A CHEMICAL EQUILIBRIUM ESTIMATE. WICHNER, R.P.; SPENCE, R.D. Oak Ridge National Laboratory. March 1984. 43pp. 8404020263. ORNL/TM-8683. 22873:181.

The degree of vaporization of LWR core materials was estimated using a highly idealized procedure involving (1) specification of the phases that are present for both structural and fuel material; (2) estimation of the vapor pressures exerted by the individual components of each phase; and (3) assuming a degree of vaporization of each phase constituent, to allow equilibration between gaseous and condensed species within the assumed pressure vessel volume. Obviously, the degree of vaporization of many core materials is limited by other complex factors such as local mass transport conditions and solid-phase diffusion rates; however, the results obtained in this study serve to illustrate the types of driving forces for vaporization that are present and to give some indication of the composition of the vaporized material. Some comparisons are provided with estimated degrees of core vaporization from other sources.

In addition to estimated degrees of vaporization, information is included regarding the projected chemical forms of the condensed material for the expected range of oxygen potentials in the reactor vessel.

NUREG/CR-3190: PLUGM: A COUPLED THERMAL-HYDRAULIC COMPUTER MODEL FOR FREEZING MELT FLOW IN A CHANNEL. PILCH, M. Sandia Laboratories. MAST, P.K. Science Applications, Inc. September 1984. 140pp. 8410120042. SAND82-1580. 26977:001.

PLUGM models the flow and freezing of molten material in a nonmelting channel. PLUGM is being developed for applications in Sandia's Ex-Vessel Core Retention Materials Assessment Program and in Sandia's LMFBR Transition-Phase Program. PLUGM models time-dependent flow from a reservoir, through a channel and possibly into a catch tank. Three user-specified geometry options enable realistic modeling of melt flow and freezing in tubes, thin slits, and particle beds.

Axial variation of relevant channel parameters is possible. Sample problems, pertaining to ex-vessel core retention and LMFBR transition phase, illustrate features and capabilities of the code.

NUREG/CR-3200 V03: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1983. DODD, C. V. ; DEEDS, W. E. ; SMITH, J. H. ; et al. Oak Ridge National Laboratory. March 1984. 22pp. 8404090145. ORNL/TM-8796/V3. 22971:262.

Eddy-current inspection is the most suitable method for rapid boreside evaluation of steam generator tubing. However, small flaws can be masked by the effects of harmless variables such as tube supports. To identify the critical properties accurately and reliably in the presence of extraneous signals caused by variations of unimportant properties, sufficient information is needed to identify harmful variations and to reject harmless ones. For this reason we have developed instrumentation capable of measuring both the amplitude and phase of the eddy-current signal at several different frequencies as well as computer equipment capable of processing the data quickly and reliably. Our most recent computer-optimized probe design uses an array of small flat "pancake" coils pressed against the inside wall of the steam generator tubing. Data have been taken with such coils on tubes with various combinations of abnormalities. Data were also taken for machined defects in tubes and flat plates to verify our basic flaw theory and to check our inversion theory for characterizing flaw properties from scans across the defect.

NUREG/CR-3200 V04: EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING PROGRAM ANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1983. DODD, C. V. ; DEEDS, W. E. ; SMITH, J. H. ; et al. Oak Ridge National Laboratory. May 1984. 18pp. 8406210100. ORNL/TM-8796/V4. 25114:298.

Eddy-current inspection is the most suitable method for rapid boreside evaluation of steam generator tubing. However, small flaws can be masked by the effects of harmless variables, such as tube supports. To identify the critical properties accurately and reliably in the presence of extraneous signals caused by variations of unimportant properties, sufficient information is needed to identify harmful variations and reject harmless ones. For this reason we have been developing instrumentation capable of measuring both the amplitude and phase of the eddy-current signal at several different frequencies, as well as computer equipment capable of processing the data quickly and reliably. Our probes and test conditions are also computer-optimized. The most recent probe design embodies an array of small flat "pancake" coils and improves the detection of small flaws and the rejection of tube support signals. We have also experimentally verified the accuracy of our computer programs for calculating the signals produced by defects in tubing and are adapting our new IBM System 9000 computer to take and process the larger amounts of data required by additional variables, such as copper coating and intergranular attack.

NUREG/CR-3218: EVALUATION OF ENGINEERING ASPECTS OF BACKFILL PLACEMENT FOR HIGH LEVEL NUCLEAR WASTE (HLW) DEEP GEOLOGIC REPOSITORIES. Final Report (Task 5) June 1981 - February 1983. ROBERDS, W. ; KLEPPE, J. ; GONANO, L. Golder Associates. April 1984. 469pp. 8405220037. 813-1166. 24652:291.

This report includes the identification and subjective evaluation of alternative schemes for backfilling around waste packages and within emplacement rooms. The aspects of backfilling specifically considered in this study include construction and testing; costs have not been considered. However, because construction and testing are simply implementation and verification of design, a design basis for backfill is required. A generic basis has been developed for this study by first identifying qualitative performance objectives for backfill and then weighting each with respect to its potential influence on achieving the repository system performance objectives. Alternative backfill materials and additives have been identified and evaluated with respect to the perceived extent to which each combination can be expected to achieve the backfill design basis. Several distinctly different combinations of materials and additives which are perceived to have the highest potential for achieving the backfill design basis have been selected for further study. These combinations include zeolite/clinoptilolite, bentonite, muck and muck mixed witn bentonite. Feasible alternative construction and testing procedures for each selected combination have been discussed. Recommendations have been made regarding appropriate backfill scheme for hard rock (i.e., domal salt on the Gulf Coast and generic bedded salt).

NUREG/CR-3224: AN ASSESSMENT OF CRBR CORE DISRUPTIVE ACCIDENT ENERGETICS. THEOFANOUS,T.G.; BELL,C.R. Los Alamos Scientific Laboratory. March 1984. 436pp. 8403230216. LA-9716-MS. 22740:001.

The results of an independent assessment of core disruptive accident energetics for the Clinch River Breeder Reactor are presented in this document. This assessment was performed for the Nuclear Regulatory Commission under the direction of the CRBR Program Office within the Office of Nuclear Reactor Regulation. It considered in detail the accident behavior for three accident initiators that are representative of three different classes of events; unprotected loss of flow, unprotected reactivity insertion, and protected loss of heat sink. The primary system's energetics accommodation capability was determined in terms of core events. This accommodation capability was found to be equivalent to a ramp rate of about 200 $/s applied to a classical two-phase disassembly. This accommodation capability was contrasted to the potential for energetic behavior which was shown to arise only in the advanced core disruption states (gravity driven recriticalities). The accident behavior was found to be dominated by neutronic activity that was bounded conservatively by 100 $/s events. Based on a qualitative probabilistic approach, we concluded that massive failure of the reactor head with associated early challenge to the containment building is physically unreasonable.

NUREG/CR-3228 V02: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS. Annual Report For 1983. * Materials Engineering Associates, Inc. September 1984. 135pp. 8410170224. MEP-2051. 27031:213.

The objective of this research program is to characterize materials behavior in relation to structural safety and reliability of pressure boundary components for light water reactors. Specific objectives include developing an understanding of elastic-plastic fracture and environmentally-assisted crack propagation phenomena in terms of continuum mechanics, metallurgical variables, and neutron irradiation. Emphasis is placed on identifying metallurgical factors responsible for radiation embrittlement of steels and on developing procedures for embrittlement relief, including guidelines for radiation-resistant steels. The underlying goal is the interpretation of material properties performance to establish engineering criteria for structural reliability and long-term operation. 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 tasks for 1983 is summarized here.

NUREG/CR-3242: THE LOS ALAMOS NATIONAL LABORATORY/NEW MEXICO STATE UNIVERSITY FILTER PLUGGING TEST FACILITY DESCRIPTION AND PRELIMINARY TEST RESULTS. FENTON, D. L.; DALLMAN, D. J.; SMITH, P. R.; et al. Los Alamos Scientific Laboratory. January 1984. 13pp. 8402010111. LA-9929-MS. 22058:028.

A facility to test the plugging effects of combustion products on high-efficiency particulate air filters has been constructed. This facility can supply experimental data to support pressure-drop models of filter plugging under fire accident conditions, which are needed for use in an existing fire accident analysis computer code. The test facility includes a specially designed null-balance filter-weighing system. The resolution of this system is approximately 2 to 3 g out of 14 kg for a commercial 0.6- by 0.6-m filter. Using this system, commercial filters can be tested to provide data with which to correlate pressure drop and smoke aerosol mass accumulation and flow rate. Some recently accomplished tests and future test plans are discussed.

NUREG/CR-3251: THE ROLE OF SECURITY DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER PLANTS. CARDWELL, R. G.; MOUL, D. A.; MCBRIDE, J. A.; et al. Union Carbide Corp. March 1984. 83pp. 8404170575. Y/DS-178. 24093:272.

This report provides an analysis of the literature and on-site data gathering relating to the actions of security forces at licensed nuclear power plants during safety-related emergencies. Literature search findings and results of on-site data gathering are furnished and subjected to analysis. Taking into account the analysis provided, appropriate recommendations are presented. Recommendations are keyed as to how improvements can be made in the regulatory approach and licensee planning and procedures as they relate to the subject matter under examination. In addition, certain technological problems and issues are examined within the context of the study. Appendices provide the results of the literature search, an annotated bibliography, the Data Collection Guide used, and additional details regarding certain aspects of the study that are relevant for further explication of the body of the report.

NUREG/CR-3273: COMBUSTION OF HYDROGEN:AIR MIXTURES IN THE VGES CYLINDRICAL TANK. BENEDICK, W. B.; CUMMINGS, J. C.; PRASSINOS, P. G. Sandia Laboratories. July 1984. 165pp. 8408100155. SAND83-1022. 25997:308.

Sandia National Laboratories is currently involved in a number of experimental projects to provide data that will help quantify the threat of hydrogen combustion during nuclear plant accidents. Several experimental facilities are part of the Variable Geometry Experimental System (VGES). The purpose of this report is to document the experimental results from the first round of combustion tests performed at one of these facilities: a 5-m(3) cylindrical tank. The data provided by tests at this facility can be used to guide further testing and for the development and assessment of analytical models to predict hydrogen combustion behavior.

NUREG/CR-3275: JOB ANALYSIS OF THE ELECTRICIAN POSITION FOR THE NUCLEAR POWER PLANT MAINTENANCE PERSONNEL RELIABILITY MODEL. FEDERMAN, P. J.; BARTTER, W. D.; SIEGEL, A. I.; et al. Applied Science Associates, Inc. February 1984. 191pp. 8403300325. ORNL/TM-8755. 22842:007.

The job analysis of the electrician position is part of the work being done within a program that is developing and will evaluate a computer simulation model that will generate maintenance performance reliability data. This report is the fourth and last in a series of job analysis studies which characterize maintenance positions in nuclear power plants (NPPs). This characterization takes the form of detailed information about: (1) frequency of task performance, (2) time required for task performance, (3) the training required for adequate task performance, and (4) the perceived consequences of inadequate performance. Additionally, information is also presented about the mental and perceptual-motor loading imposed by various work functions.

A list of 199 tasks were compiled and verified through a number of visits to NPPs. Two formal questionnaires concerning these tasks were distributed to 24 NPPs and resulted in a 61% return rate. Results from the data received from the questionnaires formed the basis of this job analysis.

A statistically significant positive correlation was found between electrician training requirements and the perceived severity of adverse consequences following inadequate task performance. This and other information from this job analysis report will have direct influence on the development of the computer simulation model.

NUREG/CR-3285: PRE-TEST VISUAL EXAMINATION AND CRUD CHARACTERIZATION OF LWR RODS USED IN THE LONG-TERM, LOW-TEMPERATURE WHOLE ROD TEST. EINSIGER, R. E.; COOK, J. A. Hanford Engineering Development Laboratory. March 1984. 82pp. 8404110020. HEDL-TME 83-9. 24002:223.

Westinghouse Hanford Company (WHC) and EG&G-Idaho are jointly conducting a testing program to provide information that the Nuclear Regulatory Commission (NRC) can use to establish a licensing position relative to long-term, low-temperature performance of spent fuel rods in dry storage. The tests investigate the performance of intact and defected light water reactor (LWR) spent fuel rods in inert gas and unlimited air environments. The tests consist of placing four H. B. Robinson Unit 2 pressurized water reactor (PWR) and four Peach Bottom-II boiling water reactor (BWR) spent fuel rods in a furnace at 230 degrees C for a maximum of 50 months. Interim and final examinations are planned to assess behavior during the test. To establish the initial condition of the test rods, visual examinations of the test rods and crud examinations of companion rods were conducted. In addition, an open literature evaluation of crud characteristics was compiled.

NUREG/CR-3295 V01: LIGHT WATER REACTOR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM: Notch Ductility And Fracture Toughness Degradation Of A302-B And A533-B Reference Plates From PSF Simulated Surveillance & Through-Wall Irradiation Capsules. HAWTHORNE, J. R.;

MENKE, B. H. ; HISER, A. L. ; et al. Materials Engineering Associates, Inc. April 1984. 104pp. 8405220006. MEA-2017. 24560:208.

The NRC's Light Water Reactor-Pressure Vessel Surveillance Dosimetry Program has irradiated Charpy-V (C(v), compact tension (CT) and tension test specimens of selected steels at 288 degrees centigrade in a pressure vessel wall/thermal shield mock-up known as the Pool Side Facility. Objectives include the study of through-wall toughness gradients produced by irradiation, the relative irradiation effect at surveillance capsule vs. in-wall locations and the correspondence of C(v) vs. CT fracture toughness test methods in their independent descriptions of radiation-induced embrittlement. This report presents properties data developed for two steels: the ASTM A302-B reference plate and the HSST Program A533-B Plate 03.

Irradiation at the simulated surveillance location reproduced reasonably well the irradiation degradation developed at the vessel inner surface nd quarter wall thickness locations. The radiation-induced toughness gradient was small; the difference between transition temperatures at the inner surface vs. mil-wall locations was 31 degrees centigrade or less, independent of the test method. The temperature elevation of the C(v) curve (41 J level) with irradiation was generally less than that defined by fracture toughness tests (100 MPa square root of m level) but greater than defined by "Beta (Ic)-corrected" data.

NUREG/CR-3295 V02: LIGHT WATER REACTOR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM: Postirradiation Notch Ductility And Tensile Strength Determinations For PSF Simulated Surveillance And Through-Wall Specimen Capsules. HAWTHORNE, J. R. ; MENKE, B. H. Materials Engineering Associates, Inc. * ENSA, Inc. April 1984. 133pp. 8405220025. MEA-2017. 24561:011.

The NRC's Light Water Reactor-Pressure Vessel Surveillance Dosimetry Improvement Program has irradiated Charpy-V (C(v) and tension test specimens of selected steels at 288 degrees centigrade in a pressure vessel wall/thermal shield mock-up known as the Pool Side Facility. Objectives include the study of through-wall toughness gradients produced by neutron irradiation and the relative irradiation effect at surveillance capsule vs. in-wall locations. This report presents properties data developed for six steels: the ASTM A 302-B reference plate, the HSST Program A 533-B Plate 03, 508-3 and 22NiMo-Cr37 forgings, and two submerged arc weld deposits.

The radiation-induced toughness gradient between inner surface vs. mid-wall locations was small (31 degrees centigrade or less) for five of the six indications. Simulated surveillance capsule irradiations reproduced well the embrittlement observed for vessel inner surface and quarter wall thickness locations in almost all cases. The primary exceptions to both trends were provided by a 0.23% Cu, 1.58% Ni weld deposit which showed the highest embrittlement sensitivity. Material irradiation sensitivity levels are in accord with predictions based on copper and nickel contents.

NUREG/CR-3300 V01: REVIEW AND EVALUATION OF THE ZION PROBABILISTIC SAFETY STUDY: Plant Analysis. BERRY, D. L. ; BRISBIN, N. L. ; CARLSON, D. D. ; et al. Sandia Laboratories. May 1984. 479pp. 8406070143. SAND83-1118. 24859:165.

This report describes the review of the internal and external event plant analyses of the Zion Probabilistic Safety Study (ZPSS). The review was conducted by Sandia National Laboratories. The purpose of the review was to search for areas in the ZPSS where omissions and critical judgments were made which could impact the quantitative results. The review identified several of these areas.

NUREG/CR-3303: USE OF NEUTRON NOISE FOR DIAGNOSIS OF IN-VESSEL ANOMALIES IN LIGHT-WATER REACTORS. FRY, D. N.; MARCH-LEUBA, J.; SWEENEY, F. J. Oak Ridge National Laboratory. May 1984. 100pp. 8405290438. ORNL/TM-8774. 24696:256.

The value of neutron noise analysis for diagnosis of in-vessel anomalies in light-water reactors (LWRs) was assessed by: (1) analyzing ex-core neutron noise from seven pressurized-water reactors (PWRs) to determine the degree of similarity in the noise signatures and the sources of ex-core neutron noise; (2) measuring changes in ex-core neutron noise over an entire fuel cycle at a commercial PWR; (3) applying PWR neutron noise analysis to diagnose a loose core barrel, to infer in-core coolant velocity, and to infer fuel assembly motion; and (4) applying BWR neutron noise analysis to diagnose in-core instrument tube vibrations and bypass coolant boiling, to infer in-core two-phase flow velocity and void fraction, and to infer stability associated with reactivity feedback.

This report summarizes these assessments and provides guidance for the acquisition and analysis of neutron noise in LWRs.

NUREG/CR-3305: COMPARISON OF BEACON AND COMPARE REACTOR CAVITY SUBCOMPARTMENT ANALYSES. BURKETT, M. W.; IDAR, E. S., GIDO, R. G.; et al. Los Alamos Scientific Laboratory. April 1984. 54pp. 8405220096. LA-9776-MS. 24594:313.

In this study, a more advanced "best-estimate" containment code, BEACON-MOD3A, was used to calculate force and moment loads resulting from a high-energy blowdown for two reactor cavity geometries previously analyzed with the licensing computer code COMPARE-MOD1A. The BEACON force and moment loads were compared with the COMPARE results to determine the safety margins provided by the COMPARE code. The forces and moments calculated by the codes were found to be different, although not in any consistent manner, for the two reactor cavity geometries studied. Therefore, generic summary statements regarding margins cannot be made because of the effects of the detailed physical configuration. However, differences in the BEACON and COMPARE calculated forces and moments can be attributed to differences in the modeling assumptions used in the codes and the analyses.

NUREG/CR-3307 V03: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report, July-September 1983. EDLER, S. K. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 72pp. 8405220055. PNL-4705-3. 24561:298.

This document summarizes work performed by Pacific Northwest Laboratory from July 1 through September 30, 1983, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Evaluations of nondestructive examination (NDE) techniques and instrumentation include demonstrating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, and examining NDE reliability and probabilistic fracture mechanics. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Experimental data validated models are being used to determine a method for evaluating the acceptance of welded or weld-repaired stainless steel piping. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of full scale reactor systems under postulated accident conditions. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada; and an instrumented fuel assembly irradiation program is being performed at Halden, Norway. Fuel assemblies and analytical support are being provided for experimental programs at other facilities, including the Super Sara Test Program, Ispra, Italy, and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

NUREG/CR-3307 V04: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report, October-December 1983. EDLER, S. K. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 38pp. 8406060432. PNL-4705-4. 24668:347.

This document summarizes work performed by Pacific Northwest Laboratory from October 1 through December 31, 1983, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Evaluations of nondestructive examination (NDE) techniques and instrumentation include investigating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems and examining NDE reliability and probabilistic fracture mechanics. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Experimental data and validated models are being used to determine a methods for evaluating the acceptance of welded or weld-repaired stainless steel piping. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of fullscale reactor systems under postulated accident conditions. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada; an instrumented fuel assembly irradiation program is being performed at Halden, Norway; and fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

NUREG/CR-3309: A SIMULATOR-BASED STUDY OF HUMAN ERRORS IN NUCLEAR POWER PLANT CONTROL ROOM TASKS. BEARE, A. N.; DORRIS, R. E.; BOVELL, C. R.; et al. General Physics Corp. January 1984. 190pp. 8403300342. SAND83-7095. 22844:070.

The purposes of this study were to empirically establish error rates for control selection and operation during the performance of proceduralized tasks in nuclear power plant control rooms during simulated casualties, and to compare the observed error rates with the human error probabilities in the Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278

NUREG/CR-3310: TESTING OF THE CONTAIN CODE. SCIACCA, F. W.; BERGERON, K. D.; MURATA, K. K.; et al. Sandia Laboratories. April 1984. 200pp. 8407020036. SAND83-1149. 25230:001.

CONTAIN is a large computer code intended for use in the analysis of severe nuclear power plant accidents. Many tests have been conducted on CONTAIN to assess its adequacy for dealing with nuclear-accident problems. This report describes the CONTAIN test program and summarizes the results obtained to date. These results are presented so that users may be aware of the features of CONTAIN that have been checked and of the areas where problems have been identified. In addition, this report provides information needed by users to repeat tests of interest in their specific work areas.

The test efforts have identified a substantial number of problems in the coding or logic of the CONTAIN code. Most of these problems have been corrected. These corrections have been included in the most recent versions of the code. CONTAIN can accurately treat most of the phenomena expected to occur in containment atmospheres. Some problems identified by the test program, involving pool-related phenomena, have prompted the development of a substantially new system of models for pool phenomena. When completed, this new system will be subjected to intense testing of the type described here.

NUREG/CR-3316: VERIFICATION AND FIELD COMPARISON OF THE SANDIA WASTE-ISOLATION FLOW AND TRANSPORT MODEL (SWIFT). WARDS, D. S.; REEVES, M.; DUDA, L. E. Sandia Laboratories. April 1984. 170pp. 8407060054. SAND83-1154. 25432:120.

The SWIFT Model has been developed and maintained by Sandia National Laboratories. The Nuclear Regulatory Commission has sponsored this work under the high-level nuclear waste program. SWIFT is a fully-coupled, transient, three-dimensional model. It is implemented by a finite-difference code which solves the equations for flow and transport in geologic media and is used to evaluate repository-site performance. This document represents an important part of the quality-assurance records for the code. Here the process simulators for flow, heat and radionuclide transport are examined using two different types of tests. The analytical verifications test SWIFT calculations against analytical solutions, and the field comparisons test SWIFT calculations against field data. Both types of tests yield good agreement between the SWIFT computations and the comparative data.

NUREG/CR-3318: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM: PCA Experiments, Blind Test, And Physics-Dosimetry Support For The PSF Experiments. MCELROY, W. M. Hanford Engineering Development Laboratory. September 1984. 297pp. 8410190172. HEDL-TME 84-1. 27059:001.

This report was prepared to: 1) Serve as a general reference document containing benchmarked experimental and theoretical data and information required to determine and certify the accuracy of the experimental and analytical methods and data that are recommended in a series of ASTM LWR pressure vessel surveillance standards; 2) Provide detailed experimental and theoretical results to determine the limiting accuracy of transport theory calculations for predicting dosimetry sensor reaction rates and derived values of neutron exposure parameters (total fluence, fluence greater than 0.1 and 1.0 MeV, and dpa) for LWR pressure vessel benchmark fields simulating steel-water configurations of commercial power reactors; 3) Assess the accuracy of the methodology used to translate measured pressure vessel steel damage and exposure data (and the corresponding uncertainties) obtained at surveillance locations to the pressure vessel beltline region; 4) Provide PCA 4/12 and 4/12 SSC configurations' experimental and theoretical physics-dosimetry results in support of the "PSF Experiments and Blind Test."

After an executive summary, a description of the PCA experimental test facility is provided in Section 1 followed by the presentation and discussion of experimental measurements and data in Sections 2, 3, and 4. The results of neutronic calculations by participants are given and referenced in Section 6. The comparison and evaluation of measured and derived data are considered in Section 7.

NUREG/CR-3329 V03: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM. Quarterly Report, July-September 1983. THOMPSON, S. L. Sandia Laboratories. February 1984. 84pp. 8402280123. SAND83-1171. 22430:123.

The RELAP5 independent assessment project at Sandia National Laboratories is part of a multi-faceted effort sponsored by the NRC to determine the ability of various system codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. The version used for the FY82 assessment project {1,2,3,4} was RELAP5/MOD1/CYCLE14 {5}, the latest publicly released version available at the time this project was begun.

Brief individual status reports on ongoing LOFT and Semiscale NC and UT calculation (all being done with RELAP5/MOD1), and the MOD1.5 BCL analyses, are given in the main body of this quarterly, as is the current status of the new UHI plant analyses (with both TRAC-PF1 and RELAP5). During this quarter, calculations were completed for Semiscale degraded heat transfer natural circulation tests S-NC-3 and S-NC-4, for Semiscale transient natural circulation test S-NC-8, and for LOFT large break test L2-5.

NUREG/CR-3329 V04: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM. Quarterly Report October-December 1983. THOMPSON, S. L. Sandia Laboratories. April 1984. 65pp. 8405220044. SAND83-1171. 24556:275.

The TRAC-PF1/MOD1 independent assessment program at Sandia National Laboratories (SNLA) is part of a multi-faceted effort sponsored by the Nuclear Regulatory Commission (NRC) to determine the ability of various system codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. This program is a successor to the RELAP5/MOD1 independent assessment project underway at Sandia for the last two years.

NUREG/CR-3334 V02: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL-JUNE 1983. PUGH, C. E. Oak Ridge National Laboratory. January 1984. 68pp. 8401230066. ORNL/TM-8787/V2. 21921:001.

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 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 seven tasks: (1) program administration and procurement, (2) fracturemechanics analyses and investigations, (3) investigations of irradiated materials, (4) thermal-shock investigations, (5) pressure vessel investigations, (6) stainless steel cladding investigations, and (7) environmentally assisted crack growth studies.

NUREG/CR-3334 V03: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM QUARTERLY PROGRESS REPORT FOR JULY-SEPTEMBER 1983. PUGH, C. E. Oak Ridge National Laboratory. March 1984. 120pp. 8403300317. ORNL/TM-8787/V3. 22826:067.

See NUREG/CR-3334, V02 abstract.

NUREG/CR-3335: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-3. OSBORNE, M. F.; LORENZ, R. A.; NORWOOD, K. S.; et al. Oak Ridge National Laboratory. May 1984. 69pp. 8405290450. ORNL/TM-8793. 24711:141.

The third in a series of high-temperature fission product release tests was conducted for 20 min at 2000 degrees centigrade in flowing steam. The test specimen, a 20-cm-long section of H.B. Robinson fuel rod that had been irradiated to ~25,200 MWd/t, was heated in an induction furnace in a hot cell.

Posttest examination showed that the Zircaloy cladding had melted, causing extensive disintegration of hte UO(2) fuel and formation of molten phases that appeared to be rich in uranium. Analyses of test components revealed very high fractional releases of (85)Kr (59.0%), (137)Cs (58.8%), and (129)I (35.4%). The releases if (125)Sb and (110m)Ag, however, were much less than those observed in test HI-2 at 1700 degrees centigrade, perhaps as a result of lower steam flow rate in test HI-3. The extent of aerosol formation, as evidenced by mass of material collected on filters, was similar in the two tests.

NUREG/CR-3346: BIOASSAY DATA AND A RETENTION-EXCRETION MODEL FOR SYSTEMIC PLUTONIUM. LEGGETT, R. W. Oak Ridge National Laboratory. July 1984. 96pp. 8408130009. ORNL/TM-8795. 26038:108.

The estimation of systemic burdens from urinalyses has been the most common and useful method of quantifying occupational exposures to plutonium. Problems arise in using this technique, however, because of inadequate modeling of human retention, translocation, and excretion of this element. Present methods for estimating the systemic burden from urinalyses were derived to a large extent from patterns observed in the first few months after exposure, but there is now evidence that these same patterns do not persist over long periods. In fact, recent comparisons of autopsy data with urinalyses suggest that extrapolation to extended periods based on these observed patterns usually leads to a large overestimate of the systemic burden at times more than a few years after exposure. In this report we collect and discuss human and animal data for Pu together with general physiological properties needed for the interpretation of bioassay results. This information is used to develop a mechanistic model of the movement, retention, and excretion of systemic Pu. This model appears to be a reasonably accurate predictor of excretion for times ranging from one day to several decades after contamination of blood.

NUREG/CR-3350: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM: Postirradiation Examination Results For The Third

Materials Experiment (MT-3). RAUSCH, W. N. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 71pp. 8404300179. PNL-4933. 24233:238.

A series of in-reactor experiments were conducted by Pacific Northwest Laboratory, using full-length 32-rod pressurized water reactor fuel bundles, as part of the Loss-of-Coolant Accident (LOCA) Simulation Program. The third materials experiment (MT-3) was the sixth in the series of thermal-hydraulic and materials deformation/rupture experiments conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario, Canada. MT-3 was jointly funded by the U.S. Nuclear Regulatory Commission and the United Kingdom Atomic Energy Authority. The experiment evaluated ballooning and rupture during active two-phase cooling in the temperature range from 1400 to 1500 fahrenheit. The 12 test rods in the center of the 32-rod bundle were initially pressurized to 550 psi to insure rupture in the correct temperature range. All 12 of the rods ruptured, with an average peak bundle strain of about 55%. A hot cell postirradiation examination (PIE) of several of the ruptured rods was also conducted. This report describes the work performed and presents the PIE results. Information obtained during the PIE analysis included cladding thickness measurements, metallography, and particle size analysis of the cracked and broken fuel pellets.

NUREG/CR-3351: SECURITY OFFICER TACTICAL TRAINING ISSUES INVOLVING ESS EQUIPMENT. ROUNTREE, S. L. Sandia Laboratories. January 1984. 44pp. 8402060336. SAND82-2933. 22107:272.

Security officer tactical training issues are discussed in relation to the possible implementation of the Tactical Improvement Package (TIP), utilizing the Engagement Simulation System (ESS) equipment, by nuclear power plant licensees for security officer tactical training. The ESS equipment provides the capability to simulate engagement conditions between adversaries armed with weapons which have harmless laser transmitters. A brief discussion of the TIP is presented, along with some concerns and considerations in the use of the TIP.

NUREG/CR-3359 V03: PHYSICS OF REACTOR SAFETY. Quarterly Report, July-September 1983. * Argonne National Laboratory. January 1984. 15pp. 8401230062. ANL-83-11 V03. 21931:075.

This quarterly progress report summarizes work done during the months of July-September 1983 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research of the U.S. Nuclear Regulatory Commission. The work in the Applied Physics Division includes reports on reactor safety modeling and assessment by members of the Reactor Safety Appraisals Section. Work on reactor core thermal-hydraulics is performed in ANL's Components Technology Division, emphasizing 3-dimensional code development for LMFBR accidents under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-3359 V04: PHYSICS OF REACTOR SAFETY. Quarterly Report, October-December 1983. * Argonne National Laboratory. February 1984. 22pp. 8404020217. ANL-83-11 V04. 22876:329.

This quarterly progress report summarizes work done during the months of October-December 1983 in Argonne National Laboratory's Applied Physics and Components Technology Division of Reactor Safety

Research in the U.S. Nuclear Regulatory Commission. The work in the Applied Physics Division includes reports on reactor safety modeling and assessment by members of the Reactor Safety Appraisals Section. Work on reactor core thermal-hydraulics is performed in ANL's Components Technology Division, emphasizing 3-dimensional code development for LMFBR accidents under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-3360: COMPUTER PROGRAM CDCID: AN AUTOMATED QUALITY CONTROL PROGRAM USING CDC UPDATE. SINGER, G. L.; AGUILAR, F. EG&G, Inc. April 1984. 70pp. 8405220028. EGG-2302. 24557:253.

A computer program, CDCID, has been developed in coordination with a quality control program to provide a highly automated method of documenting changes to computer programs at EG&G Idaho, Inc. The method uses the standard CDC UPDATE program in such a manner that updates and their associated documentation are easily made and retrieved in various formats. The method allows each card image of a source program to point to the document which describes it, who created the card, and when it was created.

The method described is applicable to the quality control of computer programs in general. The computer program described is executable only on CDC computing systems, but the program could be modified and applied to any computing system with an adequate updating program.

NUREG/CR-3366: HIGH TEMPERATURE MELT ATTACK ON STEEL AND URANIA-COATED STEEL. POWERS, D. A.; ARELLANO, F. E. Sandia Laboratories. April 1984. 95pp. 8406230297. SAND83-1350. 25128:155.

Corium and Thermitic melts were teemed at various velocities onto bare steel plates and steel plates coated with urania. An empirical correlation of the penetration data is developed.

NUREG/CR-3369: AN UNCERTAINTY STUDY OF PWR STEAM EXPLOSIONS. BERMAN, M.; SWENSON, D. V.; WICKETT, A. J. Sandia Laboratories. July 1984. 91pp. 8408130006. SAND83-1438. 26040:238.

Some previous assessments of the probability of containment failure caused by in-vessel steam explosions in a PWR have recognized large uncertainties and assigned broad ranges to the probability, while others have concluded that the probability is small or zero. In this report we study the uncertainty in the probability of containment failure by combining the uncertainties in the component physical processes using a Monte Carlo method. We conclude that, despite substantial research, the combined uncertainty is still large. Some areas are identified in which improvements in our understanding may lead to large reductions in the overall uncertainty.

NUREG/CR-3371 V03: TASK ANALYSIS OF NUCLEAR POWER PLANT CONTROL ROOM CREWS. Volume 3: Task Data Forms. BURGY, D.; LEMPGES, C.; MILLER, A.; et al. General Physics Corp. December 1984. 898pp. 8501080463. GP-R-221020. 28261:221.

A task analysis of nuclear power plant control room crews was performed by General Physics Corporation and BioTechnology, Inc. for the Office of Nuclear Regulatory Research, US NRC. The task analysis methodology used in the project is discussed and compared with traditional task analysis and job analysis methods. The objective of the project was to conduct a crew task analysis that would provide data for evaluating six areas: (1) human engineering design of control rooms, (2) the numbers and types of control room operators needed with requisite skills and knowledge, (3) operator qualification and training requirements, (4) normal, off-normal, and emergency operating procedures, (5) job performance aids, and (6) communications. A generic structural framework for assembling the large task data base was employed from observations and videotaping of crew behaviors during 44 operating sequences conducted at 8 power plant sites. The results of the data collection effort were compiled in a computerized task database (SEEK). Six demonstrations for verifying the suitability of the analytical approach and for suitability analysis of each of the 6 areas were performed and described. Volume 1 details the Project Approach and Methodology. Volume 2 provides the Data Results including a description of the computerized task analysis data format. Volumes 3 and 4 present the Task Data Forms that resulted from the project.

NUREG/CR-3371 V04: TASK ANALYSIS OF NUCLEAR POWER PLANT CONTROL ROOM CREWS. Volume 3: Task Data Forms. BURGY, D.; LEMPGES, C.; MILLER, A.; et al. General Physics Corp. December 1984. 832pp. 8501080460. GP-R-221020. 28265:001.

See NUREG/CR-3371, V03 abstract.

NUREG/CR-3378: VERIFICATION OF THE NETWORK FLOW AND TRANSPORT/DISTRIBUTED VELOCITY METHOD (NWFT/DVM) COMPUTER CODE. DUDA, L.E. Sandia Laboratories. May 1984. 50pp. 8406190081. SAND83-1466. 25029:188.

The Network Flow and Transport/Distributed Velocity Method (NWFT/DVM) computer code was developed to provide a computationally efficient ground-water flow and contaminant transport capability for use in risk analyses. It is a semi-analytic, quasi-two-dimensional network code that simulates ground-water flow and the transport of dissolved species (radionuclides) in saturated porous medium. This code development was funded by the U.S. Nuclear Regulatory Commission as part of a methodology for assessing the risk from disposal of radioactive wastes in geologic formations. A separate project was funded to ensure that the codes developed are as error-free as possible and include verification and validation tests to represent the processes for which it is intended. This document contains four verification problems for the NWFT/DVM computer code. Two of these problems are analytical verifications of NWFT/DVM where results are compared to analytical solutions. The other two are code-to-code verifications where results are compared to those of another computer code. The NWFT/DVM results showed good agreement with both the analytical solutions and the results from the other code.

NUREG/CR-3379: SLAM - A SODIUM-LIMESTONE CONCRETE ABLATION MODEL. SUO-ANTTILA, A. Sandia Laboratories. April 1984. 77pp. 8405220176. SAND83-7114. 24601:082.

The Sodium-Limestone Ablation Model (SLAM) is described in detail in this report. SLAM is a three-region model, containing a pool (sodium and reaction debris) region, a dry (boundary layer and dehydrated concrete) region, and a wet (hydrated concrete) region. The model includes a solution to the mass, momentum, and energy equations in each region. A chemical kinetics model is included to provide heat sources due to chemical reactions between the sodium and the concrete.

Both isolated model as well as integrated "whole code" evaluations have been made with good results. The chemical kinetics and water migration models were evaluated separately, with good results. Several small and large-scale sodium limestone concrete experiments were simulated with reasonable agreement between SLAM and the experimental results.

The SLAM code was applied to investigate the effects of mixing, pool temperature, pool depth and fluidization. All these phenomena were found to be of significance in the predicted response of the sodium concrete interaction. Pool fluidization is predicted to be the most important variable in large scale interactions.

NUREG/CR-3383: IRRADIATION EFFECTS ON THE STORAGE AND DISPOSAL OF RADWASTE CONTAINING ORGANIC ION-EXCHANGE MEDIA. SWYLER, K. J.; DODGE, C. J.; DAYAL, R. Brookhaven National Laboratory. April 1984. 86pp. 8405220086. BNL-NUREG-51691. 24602:288.

The effects of external irradiation on anion, cation, and mixed bed organic ion exchangers have been investigated under conditions relevant to radwaste storage and disposal. Two effects are emphasized: (1) radiolytically induced release of acids, radionuclides or chemically aggressive species, and (2) radiolytic generation/uptake of corrosive or combustible gases. For sulfonic acid cation resin, sulfate ion is produced in the radiolytic scission of the functional group. The insensitivity to external parameters may make the sulfate yield a convenient measure of radiation durability for regulatory considerations. The acidity which results from a given sulfate yield depends on the resin loading. Acidity is substantially reduced for loadings other than H(+). For heavy irradiation doses incorporating cation/anion resins in mixed bed form, the presence of anions resin did not protect against radiolytic acidity formation. The irradiated anion resin may also release substantial amounts of free liquid. Radiolytic hydrogen gas yield data support the validity of accelerated testing at high radiation dose rates. Oxygen gas is removed from the environment of irradiated resins by an efficient radiolytic oxidation process.

NUREG/CR-3389: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS. MEYER, R. E.; ARNOLD, W. D.; CASE, F. I. Oak Ridge National Laboratory. February 1984. 49pp. 8404020260. ORNL-5978. 22845:284.

Estimation of the rates of migration of nuclides from nuclear waste repositories requires knowledge of the interaction of these nuclides with the components of the geological formations in the path of the migration. These interactions will be dependent upon the valence state and speciation of the nuclide. Experiments designed to measure interaction of multivalent nuclides and minerals must therefore include some form of valence state control. An electrochemical method of valence state control was developed which makes use of a porous electrode in a flow system containing a column of the adsorbent. By use of this method and solvent extraction analyses of the valence states, a number of reactions of interest to HLW repositories were investigated. These include the reduction of Np(V) and Tc(VII) by crushed basalt and other minerals. For the reduction of Np(V) by basalt, the experiments indicate that the sorption of basalt increases with pH and that most of the Np is reduced to Np(IV) which is very difficult to remove from the basalt even if oxygenated tracer-free solution is added to the solution. For the experiments with Tc(VII), the results are considerably more complicated. The results of these experiments are used to assess some of the techniques and methods currently used in safety analyses of proposed HLW repositories.

NUREG/CR-3390: DOCUMENTATION AND USER'S GUIDE: UNSAT2 - VARIABLY SATURATED FLOW MODEL. (Including 4 Example Problems). DAVIS, L. A.; NEUMANN, S. P. Water, Waste & Land, Inc. December 1983. 218pp. 8403230197. WWL/TM-1791-1. 22742:001.

This report presents documentation and a user's guide for program UNSAT2. Mathematical equations and physical principles utilized to develop the code are presented in Section 2. The numerical approach used (Galerkin Finite Elements) is presented in Section 3. Section 4 presents an overview of how problems should be set up to properly use the code while detailed input instructions are presented in Section 5. Output produced by the code is discussed in Section 6. Four example problems, including sample input data sets and output data, are presented in Section 7. Program information and a complete listing of the program is provided in Section 8. This report was prepared as part of a project in which the NRC staff was presented a training course on how to properly use this computer program. Program UNSAT2 can be utilized to analyze flow in unsaturated, partially unsaturated, or fully saturated flow regions. It is anticipated that the NRC will use the model for checking information provided by a licensee, for evaluation alternative sites and designs for waste disposal, and for comparing their results with results from other methods of solution.

NUREG/CR-3391 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM. Quarterly Progress Report, April 1983 - June 1983. LIPPINCOTT, E. P.; MCELROY, W. M. Hanford Engineering Development Laboratory. April 1984. 113pp. 8404170026. HEDL-TME 83-22. 24092:068.

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) has been established by NRC to improve, test, verify, and standardize the physics-dosimetry-metallurgy, damage correlation, and the 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 US NRC-supported activities at HEDL, ORNL, NBS, and MEA-ENSA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Julich, Germany), and several UK 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.

NUREG/CR-3391 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM. Annual Report, October 1, 1982-September 30, 1983. MCELROY, W. M.; KAM, F. B.; GRUNDL, J. A.; et al. Hanford Engineering Development Laboratory. June 1984. 198pp. 8407180011. HEDL-TME 83-23. 25652:102.

See NUREG/CR-3391, V02 abstract.

NUREG/CR-3391 V04: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM. Quarterly Progress Report, October 1983-December 1983. LIPPINCOTT, E. P.; MCELROY, W. M. Hanford Engineering Development Laboratory. May 1984. 92pp. 8406080261. 24877:001.

See NUREG/CR-3391, V02 abstract.

NUREG/CR-3396: EXPERIENCE WITH THE SHIFT TECHNICAL ADVISOR POSITION. Interviews With Personnel From Nine Plants. MELBER, B. D.; OLSON, J.; SCHREIBER, R. E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1984. 61pp. 8404020220. PNL-3786. 22875:276.

The provisions of engineering expertise on shift at commercial nuclear power plants has mainly taken the form of the Shift Technical Advisor (STA). This person, acting in a capacity that is part engineer and part operator, is expected to advise the operations crew in the event of an emergency and review plant operating experience during normal circumstances. The position was mandated by the Nuclear Regulatory Commission following the incident at Three Mile Island. This report expands on a growing body of knowledge regarding the effectiveness of the STA. The new data presented here come from interviews with plant personnel and utility officials from nine sites. Researchers from the Pacific Northwest Laboratory (PNL) interviewed plant personnel, including the STA and immediate management, the shift supervisor and management, the training department, and ancillary staff, all of whom affect the intended performance of the STA. The conclusions of the report are that the design of the STA position results in limited contribution during an emergency; more comprehensive ways should be sought to provide the variety and specificity of engineering expertise needed during such times.

NUREG/CR-3410: CHMONE: A ONE-DIMENSIONAL COMPUTER CODE FOR SIMULATING TEMPERATURE, FLOW AND CHEMICAL CONCENTRATIONS IN WATER BODIES. FISCHER, S. K.; HETRICK, D. M.; LIETZKE, M. H.; et al. Oak Ridge National Laboratory. April 1984. 257pp. 8404180403. ORNL/TM-8786. 24107:098.

The computer code CHMONE simulates fast-transient, one dimensional hydrodynamic, thermal, and chemical-species-concentration conditions in controlled rivers and tidal estuaries. The code is particularly designed for applications to actual site-specific problems that require accurate predictions of the chemical species concentrations for preliminary studies of the aggregate chemical impact on a common waterbody caused by chlorination of the discharge water from multiple power plant operations.

The CHMONE code can continuously simulate the hydrodynamic and thermal conditions and concentrations of four chemical species for a 30-d period. Because only a small amount of CPU time is necessary, CHMONE can be readily utilized as a cost-effective tool in studying thermal and chemical impacts of power plant discharges in controlled rivers and tidal estuaries.

NUREG/CR-3412: CONTAINMENT INTEGRITY PROGRAM QUARTERLY REPORT JANUARY-MARCH 1983. BELJWAS, T. E.; HORSCHEL, D. S.; JUNG, J.; et al. Sandia Laboratories. February 1984. 44pp. 8402280014. SAND83-1482. 22429:308.

This report contains a description of work performed in the second quarter of FY83 on the Containment Integrity Program. Plans for future work are presented. The overall objective of the program is the qualification of methods for reliably predicting the capability of containment structures to function under loadings caused by severe accidents and extreme environments. Both analytical and experimental efforts are under way. The experiments are tests of entire containment for qualifying the analytical methods.

NUREG/CR-3418: SCREENING TESTS OF TERMINAL BLOCK PERFORMANCE IN A SIMULATED LOCA ENVIRONMENT. CRAFT, C. M. Sandia Laboratories. August 1984. 275pp. 8410120045. SAND83-1617. 26981:001.

Twenty-four terminal blocks were tested in simulated Design Basis Event (DBE), Loss of Coolant Accident (LOCA) environments. The terminal blocks were powered at voltages of 4 Vdc, 45 Vdc, and 125 Vdc. Resulting currents associated with these voltage levels were 1.8 mA, 20 mA, and 1 A, respectively. Terminal-to-terminal and terminal-to-ground leakage currents were monitored on a discrete time basis throughout the test. Based on these measurements, insulation resistance were calculated. During exposure to the LOCA steam environment insulation resistance was observed to decrease from initial values of 10(8) to 10(10) ohms ot 10(2) to 10(5) ohms. These decreases in IR are interpreted as being caused by conduction in surface moisture films rather than bulk conduction through the insulation material. Insulation resistance for all applied voltage levels appear to be approximately the same. Sporadic breakdowns lasting from fractions of a second to several minutes were observed. Further, rapid increases in applied voltage caused large decreases in insulation resistance. The measured IR was also dependent upon temperature. Subsequent to the test, terminal block insulation resistance returned to acceptable levels (10(6) to 10(8) ohms), though not to pre-test levels. The comparison of spray and no-spray results shows that no discernable difference in IRs existed between the periods with and without chemical spray.

NUREG/CR-3422 V02: AEROSOL RELEASE AND TRANSPORT PROGRAM. Quarterly Progress Report For April-June 1983. ADAMS, R. E.; TOBIAS, M. L. Oak Ridge National Laboratory. February 1984. 52pp. 8403150219. ORNL/TM-8849/V2. 22662:087.

This report summarizes progress for the Aerosol Release and Transport Program sponsored by the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, Division of Accident Evaluation, for April-June, 1983. Topics discussed include (1) several capacitor discharge vaporization experiments in the CRI-III and Fuel Aerosol Simulant Test facilities, (2) descriptions of dry atmosphere test 511 and steam atmosphere 505 with iron oxide aerosols in the NSPP facility, (3) technical support work for the aerosol test program at Marviken, Sweden, (4) US-German exchange experiment proposals concerning aerosols containing simulant fission products, (5) a core melt test to observe metal-interaction effects in cladding degradation, (6) progress in construction of a 10 kg core-melt induction furnace, (7) analytical work in support of the FAST/CRI-III experiments, (8) aerosol code implementation, (9) NAUA code validation studies, and (10) steam-only experiments in the NSPP.

NUREG/CR-3422 V03: AEROSOL RELEASE AND TRANSPORT PROGRAM. Quarterly Progress Report For July-September 1983. ADAMS, R. E.; TOBIAS, M. L. Oak Ridge National Laboratory. April 1984. 53pp. 8405290448. ORNL/TM-8849/V3. 24711:090.

This report summarizes progress for the Aerosol Release and Transport Program sponsored by the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, Division of Accident Evaluation, for July-September 1983. Topics discussed include (1) several capacitor discharge vaporization (CDV) experiments in the Fuel Aerosol Simulant Test Facility; (2) descriptions of mixed-aerosol experiments 611 and 612, which involved iron oxide and uranium oxide in steam; (3) technical support work for the aerosol test program at Marviken, Sweden; (4) core-melt experiment CM-35, in which tellurium and its oxide were used as additives; (5) progress in construction of a 10-kg core-melt induction furnace; (6) finite-difference calculations of energy deposition in CDV specimens; (7) a steam-only experiment in the NSPP; (8) code implementation activities; and (9) NAUA code validation studies.

NUREG/CR-3427 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING. Third Quarterly Report, October-December 1983. STAHL, D.; MILLER, N. E. Battelle Memorial Institute, Columbus Laboratories. March 1984. 90pp. 8403300294. 22826:203.

Data from the preliminary glass leaching experiments were analyzed, and methods for leachate analysis were developed. Testing conditions were determined for the leaching pilot experiment, and a flow chart for devitrification calculations was developed. Methods for evaluating spent fuel and radiation effects were reviewed. Experiments in the internal corrosion task were essentially completed, and data analysis continues. External corrosion studies indicated low general corrosion rates for the canister; however, some relatively deep pits were found on one specimen in the vapor phase. Further experiments will be conducted. Electrochemical studies indicate that steel can passivate and become susceptible to localized corrosion in basaltic groundwater. Fracture toughness tests of cast and wrought low carbon steels in nitrogen and hydrogen were initiated. Hydrogen reduced the tearing modulus of the steel. Hydrogen absorption in cast and wrought steel in basaltic environments is being evaluated. Results of computer simulations of groundwater radiolysis were compared with results from the general-corrosion correlation. Gamma energy deposition was calculated for commercial high-level waste. Efforts in correlation development focused on pitting corrosion and water chemistry with emphasis on pit-growth kinetics. A quality assurance program audit was initiated.

NUREG/CR-3427 V04: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING. Annual Report, April 1983 - April 1984. STAHL, D.; MILLER, N. E. Battelle Memorial Institute, Columbus Laboratories. June 1984. 282pp. 8407180206. BMI-2113. 25665:001.

The effects on glass waste-form dissolution of temperature, pressure, solution chemistry, and ratio of glass surface area to solution volume have been studied. The glass-dissolution correlation is ready to be evaluated by comparison with experiments. The devitrification correlation has been completed. In canister-corrosion studies, CF8 alloy was found less susceptible to glass attack than Type 304L stainless steel. Limited experiments revealed no corrosion mechanism which would indicate that cast steel could not be used as a container material; additional tests with cracking agents are planned. In hydrogen-uptake studies, cast steel was found to absorb more hydrogen than wrought steel. Parts of the general-corrosion correlation have been tested, and work continues on obtaining realistic experimental data as input for it. Gamma fluxes and dose rates in and near the waste package were calculated for CHLW and spent-fuel waste forms. The current water-radiolysis model was found adequate when tested against existing data, and preliminary calculations were performed with the current water-chemistry model; in both cases, additional chemical species are being incorporated.

NUREG/CR-3428: APPLICATION OF THE SSMRP METHODOLOGY TO THE SEISMIC RISK AT THE ZION NUCLEAR POWER PLANT. BOHN,M.P.; SHIEH,L.C.; WELLS,J.E.; et al. Lawrence Livermore National Laboratory. January 1984. 324pp. 8402210021. UCRL-53483. 22322:001.

The risk analysis included a detailed seismological evaluation of the region around Zion, Illinois which provided the earthquake hazard function and an appropriately randomized set of 180 time histories. These time histories were used as input to dynamic structural response calculations for four separate Zion buildings. Detailed finite element models of the buildings were used. Calculated time histories at piping support points were then used to determine moments throughout critical piping systems. Twenty-one separate piping systems were analyzed. Finally, the responses of piping and safety system components within the buildings were combined with probabilistic failure criteria and event tree/fault tree models of the plant safety systems to produce an estimate of the probability of core melt and radioactive release due to the occurrence of earthquakes.

The computed median probability of core melt was found to be 3E-5 per year. The upper (90%) bound on the core melt probability was computed to be 2E-5 per year, and lower (10%) bound was computed to be IE-7 per year.

NUREG/CR-3429: BOREHOLE LOGGING FOR RADIUM-226:Recommended Procedures And Equipment. OLSEN,K.B.; THOMAS,K.W. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1984. 42pp. 8411280136. PNL-5094. 27683:226.

Field investigations and a literature review were conducted to determine whether existing well-logging techniques are suitable for measuring (226)Ra at remedial action sites. These methods include passive gamma-ray measurement techniques using NaI(TI) and, occasionally, intrinsic germanium detectors. Parameters that must be considered when logging boreholes at remedial action sites include 1) casing material and thickness, 2) water in the borehole, 3) borehole diameter, 4) disequilibrium between uranium and its daughters when using scintillation detectors, and 5) spatial distribution of the tailings material. Information from the uranium exploration industry demonstrates that borehole logging is a better method for estimating radionuclide concentrations in subsurface soils than core and drill cutting analysis. Field measurements using NaI(TI) and IG detectors at Edgemont, South Dakota, have shown that NaI(TI) detectors log boreholes faster than IGs. However, if NaI(TI) detectors are used, additional time is required after logging to obtain representative samples of any anomalies found during logging, conform those samples to a constant geometry, and then count the sample using IG detectors to determine if the materials are tailings.

NUREG/CR-3435: A NEW IMPLICIT NUMERICAL SOLUTION SCHEME IN THE COMMIX-1A COMPUTER PROGRAM. DOMANUS,H.M.; SCHMITT,R.C.; SHA,W.T.; et al. Argonne National Laboratory. January 1984. 52pp. 8402060339. ANL-83-64. 22107:348.

The report describes, in detail, the new fully-implicit numerical solution procedure implemented in the COMMIX-1A computer code. This procedure, named SIMPLEST-ANL, is similar to the SIMPLE/SIMPLER procedure developed at Imperial College, London. SIMPLEST-ANL has been implemented as an additional option to the previously implemented semi-implicit procedure. The new procedure permits the use of larger time-step sizes without causing any instability in the solution of system equations. It is advantageous specifically for steady-state analysis of slowly varying long-transient problems. SIMPLEST-ANL requires less computer storage than the SIMPLER procedure with comparable or better computing efficiency.

NUREG/CR-3438: REPORT ON LARGE SCALE MOLTEN CORE/MAGNESIA INTERACTION TEST. CHU, T. Y.; BENTZ, J. H.; ARELLANO, F. E.; et al. Sandia Laboratories. October 1984. 5pp. 8411290150. SAND83-1692. 27691:227.

A molten core/material interaction experiment was performed at the Large-Scale Melt Facility at Sandia National Laboratories. The experiment involved the release of 230 kg of core melt, heated to 2923 degrees K, into a magnesia brick crucible. Descriptions of the facility, the melting technology, as well as results of the experiment, are presented. Preliminary evaluations of the results indicate that magnesia brick can be suitable material for core ladle construction.

NUREG/CR-3439 V01: A DESCRIPTION OF THE HARDWARE AND SOFTWARE OF THE POWER SPECTRAL DENSITY RECOGNITION (PSDREC) CONTINUOUS ON-LINE REACTOR SURVEILLANCE SYSTEM (CALIFORNIA DISTRIBUTION), VOLUME 1. SMITH, C. M. Oak Ridge National Laboratory. January 1984. 73pp. 8401230039. ORNL/TM-8862/V1. 21931:322.

This report documents the Power Spectral Density Recognition (PSDREC) Continuous On-Line Reactor Surveillance Program. Volume 1 of this manual is a description of the major concepts and philosophy of the PSDREC surveillance system. Volume 1 is of interest to readers who desire to understand how the system operates. Volume 2 is the appendices which contain detailed information useful only to a reader involved with the computer implementation of this system. Volume 1 has been given a general distribution. Volume 2 is available from the author upon request.

NUREG/CR-3439 V02: A DESCRIPTION OF THE HARDWARE AND SOFTWARE OF THE POWER SPECTRAL DENSITY RECOGNITION (PSDREC) CONTINUOUS ON-LINE REACTOR SURVEILLANCE SYSTEM (CALIFORNIA DISTRIBUTION), VOLUME 2. SMITH, C. M. Oak Ridge National Laboratory. January 1984. 139pp. 8401230690. ORNL/TM-8862/V2. 21900:300.

See NUREG/CR-3439, V01 abstract.

NUREG/CR-3440: IDENTIFICATION OF SEVERE ACCIDENT UNCERTAINTIES. RIVARD, J. B. Sandia Laboratories. December 1984. 300pp. 8501030262. SAND83-1689. 28195:001.

Understanding of severe accidents in light-water reactors is currently beset with uncertainty. The uncertainties that are present limit the capability to analyze the progression and possible consequences of such accidents, and therefore, restrict the technical basis for regulatory actions by the U.S. Nuclear Regulatory Commission (NRC) regarding severe accidents. A working group was formed to pool relevant knowledge and experience in assessing the uncertainties attending present (1983) knowledge of severe accidents. This, the initial report of the Severe Accident Uncertainty Analysis (SAUNA) working group, has as its main goal the identification of a consolidated list of uncertainties that affect in-plant processes and systems. Many uncertainties have been identified. A set of so-called "key" uncertainties summarizes many of the identified uncertainties. Quantification of the influence of these uncertainties, a necessary second step, is not attempted in the present report, although attempts are made qualitatively to demonstrate the relevance of the identified uncertainties.

NUREG/CR-3444 V01: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION, WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE. Annual Report. DAVIS, M.S. Brookhaven National Laboratory. January 1984. 124pp. 8402170494. BNL-NUREG-51699. 22304:202.

This report describes generic and specific aspects of hard and soft chemical decontaminations and considers the radiation and thermal stability of the reagents involved. Disposal options for LWR decontamination wastes are reviewed and advantages and disadvantages related to the options are discussed. Studies indicating the potential impact of these wastes on a shallow land burial ability to retain radionuclides are summarized. Processes being considered for the management of spent ion-exchange resins are reviewed. Problems associated with the state-of-the-art of incineration, and chemical digestion are evaluated. The solidification and disposal of decontamination wastes are considered with respect to criteria given in the Licensing Rule for Land Disposal of Radioactive Waste, 10 CFR 61. These are evaluated with respect to possible solidification in cement, bitumen, and plastics. The various options in mixing decontamination wastes with normal LWR resin waste are discussed with respect to their impacts on occupational exposure.

NUREG/CR-3448: URANIUM HOLDUP MODELING. PICARD, R.R.; MARSHALL, R.S. Los Alamos Scientific Laboratory. January 1984. 10pp. 8402170498. LA-9853-MS. 22320:001.

Statistical modeling of nuclear materials holdup in processing facilities can play an important role in operation, variability of process conditions, quality of measurements, and measurement standards impact the value of model-based estimates. Recognition of both the benefits and the limitations of model-based estimates and the periodic updating of such estimates are essential to maintaining a credible holdup estimation model.

NUREG/CR-3449: LABORATORY EVALUATION OF LIMESTONE AND LIME NEUTRALIZATION OF ACIDIC URANIUM MILL TAILINGS SOLUTION. Progress Report. OPITZ, B.E.; DODSON, M.E.; SERNE, R.J. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1984. 50pp. 8403070118. 22562:004.

Experiments were conducted to evaluate a two-step neutralization scheme for treatment of acidic uranium tailings solutions. Tailings solutions from Lucky Mc and Exxon Highland Mills were neutralized with limestone, CaCO(3), to an intermediate pH of 4.0 or 5.0, followed by lime, Ca(OH)(2), neutralization pH 7.2. The combination limestone/lime treatment methods, CaCO(3) neutralization of pH 4 followed by neutralization with Ca(OH)(2) to pH 7 resulted in the highest quality effluent solution with respect to EPA's water quality guidelines. The combination method is the most cost effective treatment procedure tested in our studies.

Experiments to evaluate the optimum solution pH for contaminant removal were performed on tailings solutions utilizing only Ca(OH)(2) as the neutralizing agent. The data indicates solution neutralization above pH 7.2 does not significantly increase removal of pH dependent contaminants from solution.

Column leaching experiments were performed on the neutralized sludge material (the precipitated material which forms as the acidic tailings solutions are neutralized). The sludges were contacted with laboratory prepared ground water and several effluent pore volumes were collected. Effluent solutions were analyzed for macro ions, trace metals and radionuclides to evaluate the effectiveness of attenuating contaminants in sludges formed during neutralization.

NUREG/CR-3457: VALIDATION OF METHODS FOR EVALUATING RADON-FLUX ATTENUATION THROUGH EARTHEN COVERS. KALKWARF, D. R.; FREEMAN, H. D.; HARTLEY, J. N. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1984. 48pp. 8411080461. PNL-5092. 27402:191.

Field and laboratory measurements were made to test the validity of methods for calculating radon-flux attenuation through earthen covers as described in A Handbook for the Determination of Radon Attenuation Through Cover Materials, NUREG/CR-3533. The validity of the diffusion equations presented in the handbook was established by the generally good agreement between the measured radon flux at six field sites and the flux predicted when measured properties of soil underlying these sites were used in these equations. When approximate values presented in the handbook for various soil properties were used in the diffusion equations, the predicted fluxes were larger than the measured values by factors of up to 31. However, investigation of the theoretical relationship between the radon flux from an earth-covered tailings pile and the thickness of that cover indicated that the latter would only be overestimated by a factor of up to 1.5 at field sites similar to those examined in this study.

NUREG/CR-3459: EXPERIMENT DATA REPORT FOR MULTIROD BURST TEST (MRBT) BUNDLE B-5. CHAPMAN, R. H.; CROWLEY, J. L.; LONGEST, A. W. Oak Ridge National Laboratory. August 1984. 173pp. 8409110096. ORNL/TM-8889. 26444:157.

B-5 test data are presented and interpreted to the extent necessary for understanding pertinent features of the 8 x 8 test. Objectives of the test were to investigate the effects of array size and rod-to-rod interactions on cladding deformation in the high-alpha-Zircaloy temperature range under conditions that simulated the adiabatic heatup (reheat) phase of a light-water-reactor loss-of-coolant accident. Test conditions, nominally the same as used in an earlier 4 x 4 (B-3) test, were conducive to large deformation. The fuel pin simulators were electrically heated (3.0 kW/m) and were slightly cooled with a very low flow (Re ~ 140) of low-pressure superheated steam. Cladding temperature increased at a rate of 9.8 degrees C/s. The simulators burst in a very narrow temperature range, with an average of 768 degrees C. Cladding burst strain ranged from 32 to 95%, with an average of 61%. Heated length volumetric expansion ranged from 35 to 79%, with an average of 52%. Average burst strain was slightly greater for the interior than for the exterior simulators; average volumetric expansion was significantly greater. Maximum coolant channel flow area reduction was 69% for the entire 8 x 8 array, 83% for the interior 6 x 6 array, and 91% for the central 4 x

4 array. The results show deformation was greater in the bundle interior and suggest rod-to-rod mechanical interactions caused axial propagation of the deformation.

NUREG/CR-3460: EXPERIMENT DATA REPORT FOR MULTIROD BURST TEST (MRBT) BUNDLE B-6. CHAPMAN, R. H.; LONGEST, A. W.; CROWLEY, J. L. Oak Ridge National Laboratory. July 1984. 157pp. 8409110104. ORNL/TM-8890. 26444:004.

A reference source of MRBT bundle B-6 test data is presented with minimum interpretation. The primary objective of this 8 x 8 multirod burst test was to investigate cladding deformation in the alpha-plus-beta-Zircaloy temperature range under simulated light-water-reactor (LWR) loss-of-coolant accident (LOCA) conditions. B-6 test conditions simulated the adiabatic heatup (reheat) phase of a LOCA and produced very uniform temperature distributions. The fuel pin simulators were electrically heated (average linear power generation of 1.42 kW/m) and were slightly cooled with a very low flow (Re ~ 140) of low-pressure superheated steam. The cladding temperature increased from the initial temperature (330 degrees C) to the burst temperature at a rate of 3.5 degrees C/s. The simulators burst in a very narrow temperature range, with an average of 930 degrees C. Cladding burst strain ranged from 21 to 56%, with an average of 31%. Volumetric expansion over the heated length of the cladding ranged from 16 to 32%, with an average of 23%. The average burst strain and the average volumetric expansion for the interior simulators were only slightly greater than the averages for the exterior simulators. The coolant channel flow area reduction was modest, with a maximum of 39% for the entire 8 x 8 array, 43% if based on the interior 6 x 6 array, and 45% if based on the central 4 x 4 array. As expected, no evidence of rod-to-rod mechanical interaction effects was observed.

NUREG/CR-3461: RESPONSE TREE EVALUATION - IMPLICATIONS FOR THE USE OF ARTIFICIAL INTELLIGENCE IN PROCESS CONTROL ROOMS. BRAY, M. A.; NELSON, W. R.; BLACKMAN, H. S.; et al. EG&G, Inc. January 1984. 34pp. 8402030099. EGG-2272. 22090:340.

An experiment was performed during 1983 that measured performance of nuclear plant operators with and without a computer-based operator aid. This report discusses the results of that experiment and their implications for design and regulation of advanced computer aids in nuclear control rooms. The aid tested is called a response tree and is intended to help operators properly align a piping system despite component or support system failures. In the experiment, 28 reactor operator subjects were required to align a system to inject coolant and stop a temperature excursion in a simulated reactor. The experiment did not show an improvement in operator performance when the response tree aid was used. More important, the experiment did produce several conclusions related to the design and evaluation of computer aids in nuclear control rooms.

NUREG/CR-3469 V01: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. BAUM, J. W.; SCHULT, D. A. Brookhaven National Laboratory. September 1984. 125pp. 8410120021. BNL-NUREG-51708. 26984:218.

This report contains selected abstracts on dose and dose reduction at nuclear power plants. Abstracts were derived primarily from APPLIED HEALTH PHYSICS ABSTRACTS AND NOTES, Volume 6, No. 1, 1980 through Volume 9, No. 1, January 1983. Subsequent reports will contain additional abstracts from earlier and more recent literature.

NUREG/CR-3470: ATWS AT BROWNS FERRY UNIT ONE - ACCIDENT SEQUENCE ANALYSIS. HARRINGTON, R. M.; HODGE, S. A. Oak Ridge National Laboratory. July 1984. 234pp. 8409110093. ORNL/TM-8902. 26446:001.

This study describes the predicted response of Unit One at Browns Ferry Nuclear Plant to a postulated complete failure to scram following a transient occurrence that has caused closure of all Main Steam Isolation Valves (MSIVs). This hypothetical event constitutes the most severe example of the type of accident classified as Anticipated Transient Without Scram (ATWS). Without the automatic control rod insertion provided by scram, the void coefficient of reactivity and the mechanisms by which voids are formed in the moderator/coolant play a dominant role in the progression of the accident. Actions taken by the operator greatly influence the quantity of voids in the coolant and the effect is analyzed in this report. The progression of the accident sequence under existing and under recommended procedures is discussed. For the extremely unlikely cases in which equipment failure and wrongful operator actions might lead to severe core damage, the sequence of emergency action levels and the associated timing of events are presented.

NUREG/CR-3471: PRESSURE SUPPRESSION POOL THERMAL MIXING. COOK, D. H. Oak Ridge National Laboratory. October 1984. 192pp. 8411270043. ORNL/TM-8906. 27675:093.

A model is developed and verified to describe the thermal mixing that occurs in the pressure suppression pool (PSP) of a commercial BWR. The model is designed specifically for a Mark-I containment and is intended for use in severe accident sequence analyses. The model developed in this work produces space and time dependent temperature results throughout the PSP and is useful for evaluating the bulk PSP thermal mixing, the condensation effectiveness of the PSP, and the long-term containment integrity. The model is designed to accommodate single or multiple discharging T-quenchers, a PSP circumferential circulation induced by the residue heat removal system discharge, and the thermal stratification of the pool that occurs immediately after the relief valves close. The PSP thermal mixing model is verified by comparing the model-predicted temperatures to experimental temperatures that were measured in an operating BWR suppression pool. The model is then used to investigate several PSP thermal mixing problems that include the time to saturate at full relief valve flow, the temperature response to a typical stuck open relief valve scenario, and the effect of operator rotation of the relief valve discharge point.

NUREG/CR-3474: LONG-LIVED ACTIVATION PRODUCTS IN REACTOR MATERIALS. EVANS, J. C.; LEPEL, E. L.; SANDERS, R. W.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 179pp. 8409200285. PNL-4824. 26606:190.

The purpose of this program was to assess the problems posed to reactor decommissioning by long-lived activation products in reactor construction materials. Samples of stainless steel, vessel steel, concrete, and concrete ingredients were analyzed for up to 52 elements in order to develop a data base of activatable major, minor, and trace elements. Large compositional variations were noted for some elements. A thorough evaluation was made of all possible nuclear reactions that could lead to long lived activation products. It was concluded that all major activation products have been satisfactorily accounted for in decommissioning planning studies completed to date. A comparison is made between calculated activation levels and regulatory guidelines for shallow land disposal according to 10 CFR 61. Most of the massive components were found to qualify as either Class A or Class B waste with the exception of PWR and BWR shroud material which clearly exceeds Class C limits. Selected samples of activated steel and concrete were subjected to a limited radiochemical analysis program as a verification of the computer model. Reasonably good agreement with the calculations was obtained where comparison was possible. In particular, the presence of 94Nb in activated stainless steel at or somewhat above expected levels was confirmed.

NUREG/CR-3476: CHEMICALS IN EFFLUENT WATERS FROM NUCLEAR POWER STATIONS: THE DISTRIBUTION, FATE AND EFFECTS OF COPPER. HARRISON, F. L. Lawrence Livermore National Laboratory. April 1984. 62pp. 8406230215. UCRL-53486. 25129:315.

This report provides a summary of research performed to determine the physicochemical forms and fate of copper in effluents from power stations adjacent to aquatic ecosystems with water that differs in salinity, pH, and concentrations of organic and inorganic constituents. In addition, research performed to evaluate responses of selected ecologically and economically important marine and freshwater organisms to increased concentrations of soluble copper is reviewed.

Copper concentration and speciation showed that the quantities of copper associated with particles, colloids, and organic and inorganic ligands differed with the site, season, and mode of operation of the station. Under normal operating conditions, the differences between influent and effluent waters were generally small, and most of the copper was in bound (complexed) species except when low pH water was circulated. However, copper was high in concentration and present in labile species during start-up of water circulation through some cooling systems and during changeover from open-cycle to closed-cycle operation.

The toxic response to copper differed with the species and life stage of the organism and with the chemical form of copper in the water. Our primary emphasis was on acute effects. However, sublethal effects of copper on a population of bluegills living in a power station cooling lake containing water of low pH and on a population exposed to increased soluble copper in the laboratory were also assessed.

NUREG/CR-3477: CONCENTRATIONS OF COPPER-BINDING PROTEINS IN LIVERS OF BLUEGILLS EXPOSED TO INCREASED CONCENTRATIONS OF SOLUBLE COPPER. HARRISON, F. L.; LAM, J. R. Lawrence Livermore National Laboratory. January 1984. 30pp. 8402060333. UCRL-53487. 22107:317.

We conducted experiments to determine the concentrations of copper in effluents and the durations of chronic exposure to sublethal levels of copper that bluegills can tolerate without adverse effects. Groups of young bluegills were exposed to water regulated to pH 5.5 that contained either no additional copper or 20, 40, 80 or 160 mg/L additional copper. In one tank, water was regulated to pH 7 and the copper to 80 mg/L.

Liver metalloproteins from the bluegills were separated into a low molecular-weight (LMW) protein fraction, which contains metallothioneins (MTs), and into intermediate molecular-weight (IMW) and high molecular-weight (HMW) fractions, which contain metalloenzymes (MEs), using high performance liquid chromatography. Large differences in quantities of copper associated with metalloproteins were found. Copper concentrations in the LMW, IMW, and HMW protein fractions and in the pellet (insoluble fraction) increased with increased exposure concentration and time. Bluegills maintained in 80 mg Cu/L water at pH 5.5 accumulated higher concentrations of copper in the HMW and IMW protein fractions and lower concentrations in the LMW protein fraction and pellet than those maintained in 80 mg Cu/L water at pH 7. Mortality was dependent on exposure concentration and duration and appeared to be related to the exceeding of the MT-detoxification capability.

NUREG/CR-3478: REVIEW OF IMPACT OF COPPER RELEASED INTO FRESHWATER ENVIRONS. HARRISON, F. L. Lawrence Livermore National Laboratory. January 1984. 98pp. 8402130117. UCRL-53488. 22213:198.

The concentrations of copper in the abiotic and biotic compartments of freshwater ecosystems, and the effects on biota of increased amounts of copper in the water and sediments are reviewed. Data compiled and discussed include the quantities and physicochemical forms of copper in the water column, the concentrations of copper in the bed-load sediments and interstitial waters, and the concentrations of copper in primary producers, annelid worms, molluscs, crustacea, aquatic insects, minor invertebrates, and fishes. In addition, the acute and sublethal effects of copper on the same groups of biota are presented, as well as data on copper concentration factors. This information can be used to: (1) determine for different types of ecosystems the ranges of copper concentrations that occur in nature, (2) identify ecosystems that are or may be impacted by copper released from industrial and urban sources, and (3) assess the effects on biota of the use of copper alloys in nuclear power station cooling systems.

NUREG/CR-3480: VALUE/IMPACT ASSESSMENT FOR SEISMIC DESIGN CRITERIA USI A-40. COATS, D. W.; LAPPA, D. A. Lawrence Livermore National Laboratory. August 1984. 138pp. 8409200279. UCRL-53489. 26606:053.

In October, 1981, the Nuclear Regulatory Commission approved a reorganization that resulted in the establishment of the Committee to Review Generic Requirements (CRGR). The charter for the CRGR requires that written justification accompany all proposed new regulatory requirements submitted to the CRGR for review. At the request of the Nuclear Regulatory Commission's Generic Issues Branch, Lawrence Livermore National Laboratory has provided the required written justification to accompany proposed new requirements to SRP Sections 3.7.1., 3.7.2, and 3.7.3. These proposed new requirements are the result of technical studies performed, as part of the Unresolved Safety Issues (USI) A-40 program, by LLNL and others. NUREG/CR-1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria," by LLNL, provided the technical resolution to USI A-40 and was the basis for the proposed new recommendations. The report contained herein present a technical evaluation and value/impact assessment of the proposed new requirements.

NUREG/CR-3481 V01: NUCLEAR POWER PLANT PERSONNEL QUALIFICATIONS AND TRAINING: TSORT--AN AUTOMATED TECHNIQUE TO ASSIGN TASKS TO TRAINING STRATEGIES. JORGENSEN, C. C. Oak Ridge National Laboratory. October 1984. 128pp. 8411290173. ORNL/TM-9308/V1. 27696:165.

This report discusses TSORT, a technique to assist the Nuclear

Regulatory Commission (NRC) in evaluating whether training program developers have allocated nuclear power plant tasks to appropriate training strategies. The TSORT structure is presented including training categories selected, dimensions of task information considered, measurement metrics used, and a guide to application. TSORT is implemented as an automated software tool for an IBM-PC. It uses full color graphics and interactive menu selection to provide NRC with a variety of evaluation options including: rank ordering of training strategies reasonable for each task, rank ordering of tasks within strategies, and a variety of special analyses. The program code is also presented along with a comprehensive example of 20 realistic tasks illustrating each of 17 options available.

NUREG/CR-3482: ANALYSIS OF FERRITE DATA FROM PRODUCTION STAINLESS STEEL PIPE WELDS. HEBBLE, T. L. ; CANONICO, D. A. ; EDMONDS, D. P. ; et al. Oak Ridge National Laboratory. January 1984. 16pp. 8402010115. ORNL-6024. 22058:007.

An American Society of Mechanical Engineers task group on stainless steel weld materials was organized to determine the need for ferrite measurements of production welds required by the U.S. Nuclear Regulatory Commission Regulatory Guide 1.31 (Rev. 1). The task group studied paired ferrite measurements [i.e., calculated and measured ferrite numbers (FNs) for the material qualifications versus measured ferrite numbers for corresponding production welds (PWs)]. Our purpose was to compare delta-ferrite content as measured in the filler metal weld qualification pad with that in the resultant PW. Welds made predominantly by three common processes (submerged arc, shielded metal arc, and gas tungsten arc) were included in the study. Weld metals investigated included types 308, 308L, 316, and 316L stainless steel. An initial evaluation of the paired ferrite measurements was made by the task group, and specific conclusions and recommendations were made. We describe the analysis of the data and the conclusions drawn.

The data base consisted of a heterogeneous collection of 1449 paired ferrite measurements for several forms and combinations of types 304 and 316 stainless steel pipe qualification pad and production welds. Qualification pad values ranged from 5 to 15 FN, and corresponding values for the PWs ranged from 2.3 to 17.5 FN. Only two PW ferrite numbers were less than 3. For qualification weld ferrite numbers less than 14, the median PW ferrite number was in reasonable agreement. However, the results show a wide seatter.

As a result of this analysis and the task group evaluation, we concluded that the requirements of Regulatory Guide 1.31 on the measurement of ferrite in PWs are not necessary and that a ferrite number of 5 in the qualification welds will, in most cases, result in PW ferrite contents greater than 3 FN.

NUREG/CR-3483: A STUDY OF THE REGULATORY POSITION ON POSTULATED PIPE RUPTURE LOCATION CRITERIA. WOO, H. H. Lawrence Livermore National Laboratory. January 1984. 44pp. 8402210019. UCRL-53490. 22323:115.

This report presents the results of studies on the current regulatory position on postulated pipe rupture location criteria and recommends future licensing regulations.

The report contains five parts. The first part highlights the current regulatory positions on criteria for postulated pipe rupture locations. The second part reviews data on nuclear piping failures related to the failure locations in the piping systems. The third part presents a probabilistic assessment of three nuclear piping lines under fatigue loadings. The fourth part recommends modifying the criteria and scopes future work. the fifth part, Appendix A, provides the validation results for the stress corrosion cracking model. The failure case chosen for comparison with the analytical result is a recirculation line safe-end cracking incident in a boiling water reactor.

NUREG/CR-3484: TRAN B-1: EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON SURFACES OF AN ANNULAR FLOW CHANNEL. MCARTHUR, D. A.; MAST, P. K. Sandia Laboratories. January 1984. 32pp. 8402210024. SAND83-1916. 22343:281.

The TRAN B-Series experiments are being conducted at Sandia National Laboratories to investigate the characteristics of fuel removal/freezing through the upper axial blankets of a liquid-metal fast-breeder reactor during the transition phase of a hypothetical core-disruptive accident. The first experiment in this series, TRAN B-1, was performed in February 1983 and the results are reported herein. This experiment involved the injection of molten UO(2) into an annular flow channel. Previous experiments had shown crusts to be stable on the inside of a cylindrical flow channel. This experiment was intended to investigate whether the conclusion of crust stability could be extended to freezing on the outside of cylindrical rods (the more prototypic geometry of the upper axial blankets).

The results of the TRAN B-1 experiment, consisting of data from online instrumentation and postirradiaton examination, indicate that the crusts on both inner and outer surfaces of the annular channel were stable during the duration of the fuel flow. Thus, a conduction-limiting model could be used to describe the fuel-flow/freezing process. However, the experiment data also indicated that the crusts were less stable on the outside of a rod than on the inside of a cylinder late in time.

NUREG/CR-3488 V02: IDAHO FIELD EXPERIMENT 1981. Vol 1: Measurement Data. START, E. E.; CATE, J. H.; DICKSON, C. R.; et al. Commerce, Dept. of, Natl. Oceanographic & Atmospheric Administration. April 1984. 944pp. 8405220082. 24549:001.

The 1981 Idaho Field Experiment was conducted in South East Idaho over the Upper Snake River Plain. Nine test-day case studies were measured between July 15 and 30, 1981. Eight-hour releases of SF(6) gaseous tracer were made from 46 m above ground. Tracer was sampled hourly, for 12 sequential hours at about 100 locations within an area 24 km square. Also, a single total integrated sample of about 30 hours duration was collected at approximately 100 sites within an area 48 by 72 km (using 6 km spacings). Extensive tower profiles of meteorology at the release point were collected. RAWINSONDES, RABALS and PIBALS were collected at 3 to 5 sites. Horizontal, low-altitude winds were monitored using the INEL MESONET. SF(6) tracer plumes were marked with co-located oil fog releases and bi-hourly sequential launches of tetroon pairs. Aerial LIDAR observations of the oil fog plume airborne samples of SF(6) were collected. High-altitude aerial photographs of daytime plumes were also collected. The Idaho Field Experiment is reported in three volumes, Volume II lists the data in tabular form or cites the special supplemental reports by other participating contractors. While the primary user file and the data achieve are maintained on 9 track/1600 cpi magnetic tapes, listings of the individual values are provided for the user who either cannot utilize the tapes or wishes to preview the data. The accuracies and quality of these data are described.

NUREG/CR-3489: ASSESSMENT OF RETRIEVAL ALTERNATIVES FOR THE GEOLOGIC DISPOSAL OF NUCLEAR WASTE. KENOORSKI, F. S. ; HAMBLEY, D. F. ; WILKEY, P. L. Engineers International, Inc. May 1984. 656pp. 8406210455. EI-1077. 25095:001.

Currently, the most feasible alternative for permanent disposal of high level nuclear waste is storage in deep underground repositories in geologic media. Uncertainties in investigation, design and construction necessitate maintaining the retrieval option until the isolation is proven likely. Investigations were limited to concepts in geologic media currently being investigated by DOE. Retrieval in most concepts is not a simple reversal of waste emplacement. This study identified several concerns. Technological concerns are associated with remining and monitoring radioactivity in backfilled storage rooms and retrieval of breached canisters. Retrieval systems currently incorporated into DOE designs were found inadequate for handling breached canisters or those bound in the storage holes. Short holes containing single canisters could be overcored but equipment must be developed to overcore large diameter holes. Safety concerns common to all repository concepts are protection of personnel from heat, traffic congestion, and deterioration of ground support. Concerns on radionuclide release were the radiation and radionuclides which would be released into the air and water present in a storage room if there were a canister breach. The confinement ventilation circuit air-flows provided in the DOE conceptual designs are just adequate for retrieval and are inadequate for retrieval from backfilled rooms.

NUREG/CR-3490: THE ROLE OF GEOCHEMICAL FACTORS IN THE ASSESSMENT AND REGULATION OF GEOLOGIC DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTE. O'KELLEY, G. D. ; MEYER, R. E. Oak Ridge National Laboratory. March 1984. 38pp. 8404100524. ORNL-5988. 22981:001.

Analyses of the performance of a high-level nuclear waste repository will require a detailed study of the chemical factors involved in the interaction of water-mobilized nuclides with the geologic formations of the host repository, including sorption phenomena, redox processes, hydrolysis, complexation, solubility, and formation of polymeric and colloidal forms of the nuclides. A discussion and review of these factors is given along with their pertinence to the migration of the nuclides and the development of computer codes for the prediction of this migration. Much of the chemistry of the nuclides of interest is very sensitive to pH and redox conditions and, in general, increase of acidity and oxidizing power of the groundwater could have serious consequences, additional concerns are the formation of negatively charged species, which tend to exhibit very low adsorption, and the formation of insoluble products through redox processes. Because of the great complexity of the chemistry involved, it will probably be necessary to develop techniques of prediction which do not take into account all reactions but only those which are thought to limit system performance.

NUREG/CR-3491: OCA-II, A CODE FOR CALCULATING THE BEHAVIOR OF 2-D AND 3-D SURFACE FLAWS IN A PRESSURE VESSEL SUBJECTED TO TEMPERATURE AND PRESSURE TRANSIENTS. BALL, D. G. ; CHEVERTON, R. D. ; DRAKE, J. B. ; et al. Oak Ridge National Laboratory. February 1984. 110pp. 8402210073.

ORNL-5934. 22323:189.

The OCA-II computer code, like its predecessor OCA-I, performs the thermal, stress, and linear elastic fracture-mechanics analysis for long flaws on the surface of a cylinder that is subjected to thermal and pressure transients. OCA-I represents a revised and expanded version of OCA-I and includes as new features (1) cladding as a discrete region, (2) a finite-element subroutine for calculating the stresses, and (3) the ability to calculate stress intensity factors for certain three-dimensional flaws, for two-dimensional circumferential flaws on the inner surface, and for both axial and circumferential flaws on the outer surface. OCA-I considered only inner-surface flaws. An option is included in OCA-II that permits a search for critical values of fluence or nil-ductility reference temperature corresponding to a specified failure criterion. These and other features of OCA-II are described in the report, which also includes user instructions for the code.

NUREG/CR-3492 V02: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1983. BALL, S. J.; CLEVELAND, J. C.; HARRINGTON, R. M.; et al. Oak Ridge National Laboratory. January 1984. 13pp. 8402060457. 22115:008.

Work on postulated severe accident sequence code development and application continued both for the Fort St. Vrain and 2240-MW(t) lead plant designs. Initial experiments on high-temperature gas-cooled reactor (HTGR) fission-product release and transport were run in an existing high-temperature (>2000 degree C) graphite-resistance furnace. Initial work was done on a cooperative study of HTGR safety research needs.

NUREG/CR-3492 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, JULY - SEPTEMBER, 1983. BALL, S. J.; CLEVELAND, J. C.; HARRINGTON, R. M.; et al. Oak Ridge National Laboratory. March 1984. 26pp. 8404100531. ORNL/TM-8921/V3. 22979:315.

Work continued on code development aimed at establishing capabilities for licensing and source-term calculations for design. Refinements were made in liner cooling system (LCS) models. Three runs were completed in the fission-product release and transport studies, indicating rapid diffusion of rare earths through graphite at high temperatures (2700 K). Benchmarking work continued for the BLAST steam generator code. Initial plans were developed for large-scale thermal-hydraulic tests for verifying LCS and core bypass flow models.

NUREG/CR-3492 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, October-December 1983. BALL, S. J.; CLEVELAND, J. C.; HARRINGTON, R. M.; et al. Oak Ridge National Laboratory. July 1984. 27pp. 8408130001. ORNL/TM-8921/V4. 26011:225.

Development work continued on models and codes for predicting source terms in both the Fort St. Vrain (FSV) and 2240-MW(t) lead plant reactors. Experimental work on fission-product vapor pressures and diffusion rates through graphite continued on temperatures up to 2775 K, and a mathematical model of the experimental system was developed to aid analysis of the results and to guide improvements in the system and experiment design. Benchmarking of the BLAST steam generator code continued using FSV data, and more support work was done for proposed FSV core bypass flow model verification. Progress was made in setting up cooperative high-temperature gas-cooled reactor (HTGR) safety research with the Federal Republic of Germany. A review of a FSV technical specification on limiting maximum core temperature was begun.

NUREG/CR-3493: A REVIEW OF THE LIMERICK GENERATING STATION SEVERE ACCIDENT RISK ASSESSMENT. Review of Core Melt Frequency. AZARM, M. A.; BARI, R. A.; BOCCIO, J. L.; et al. Brookhaven National Laboratory. July 1984. 192pp. 8408220333. BNL-NUREG-51711. 26200:205.

A limited review is performed of the Severe Accident Risk Assessment for the Limerick Generating Station. The review considers the impact on the core-melt frequency of seismic- and fire-initiating events. An evaluation is performed of methodologies used for determining the event frequencies and their impacts on the plant components and structures. Particular attention is given to uncertainties and critical assumptions. Limited requantification is performed for selected core-melt accident sequences in order to illustrate sensitivities of the results to the underlying assumptions.

NUREG/CR-3496: REVIEW OF A TEST PROGRAM FOR QUALIFYING THE SOLIDIFICATION OF EPICOR II RESINS WITH CEMENT. BARLETTA, R. E.; DAVIS, R. E. Brookhaven National Laboratory. January 1984. 32pp. 8402210026. BNL-NUREG-51712. 22323:001.

The results and recommendations of the resin solidification test program conducted by Metropolitan Edison Company are reviewed. The original purpose of this program was to recommend a formulation or range of formulations suitable for the cement solidification of first-stage Epicor-II liners generated during cleanup activities at Three Mile Island. This was to be accomplished through a systemmatic laboratory and full-scale testing program using ionexchange materials supplied by Epicor, Incorporated. Events, however, caused the truncation of the full-scale testing. Hence, a formulation was recommended based upon the results of laboratory scale testing. Failure to achieve satisfactory solidification in a single full-scale test using this formulation was observed. The unqualified conclusion that these tests demonstrate that the Epicor-II spent ion exchange media can be successfully solidified in cement appears to be unwarranted. Through a full-scale testing program, some of the deficiencies of the full-scale waste form may be resolved by simple technical modification or implementation of a process control program. Met-Ed/GPU had recognized the need for additional full-scale testing. Further, conflicting results of the screening and primary phases of the Met-Ed/GPU test program and the general conclusion of the Met-Ed/GPU study are noted in this report.

NUREG/CR-3501: THE EFFECTS OF DELAYING THE OPERATION OF A NUCLEAR POWER PLANT. HILL, L. J.; RAINEY, J. A.; TEPEL, R. C.; et al. Oak Ridge National Laboratory. January 1984. 47pp. 8403200358. ORNL/TM-8694. 22709:010.

The paper presents an analysis of an actual 24-month nuclear power plant licensing delay under alternate assumptions about regulatory practice sources of replacement power, and the cost of the plant. The analysis focuses on both the delay period and periods subsequent to the delay. The methodology utilized to simulate the impacts involved the recursive interaction of a generation costing program to estimate fuel replacement costs and a financial regulatory model to concomitantly determine the impact on the utility, its ratepayers, and security issues. The results indicate that a licensing delay has an adverse impact on the utility's internal generation of funds and financial indicators used to evaluate financial soundness. The direction of impact on electricity rates is contingent on the source of fuel used for replacement power.

NUREG/CR-3502: HIGH DRYOUT QUALITY FILM BOILING AND STEAM COOLING HEAT TRANSFER DATA FROM A ROD BUNDLE. YODER, G. L.; ANKLAN, T. M.; MORRIS, D. G.; et al. Oak Ridge National Laboratory. January 1984. 84pp. 8401300018. ORNL/TM-8794. 22027:334.

A series of eight steady-state rod bundle tests has been performed at the Oak Ridge National Laboratory in the Thermal Hydraulic Test Facility to gather data in both the low flow film boiling region and high flow steam cooling region. This test series includes experiments both with and without liquid entrainment above the dryout point.

Bundle fluid conditions were calculated using steady-state energy and mass conservation equations. The experimental heat transfer data have been compared to several film boiling heat transfer correlations and one vapor correlation. Results of these comparisons support the conclusions reached in the analysis of prior ORNL transient and steady state tests (3.03.6AR, 3.06.6B, 3.08.6C, series 3.07.9, series 3.02.10, and series 3.09.10). Results indicate that the Dougall-Rohsenou correlation often overpredicts the heat transfer coefficient, while the Groeneveld 5.7 and Condie-Bengston IV correlations tend to underpredict, however they are in better agreement with the data. The Groeneveld-Delorme correlation underpredicts heat fluxes. The Dittus-Boelter correlation was evaluated only when equilibrium qualities were greater than one, and tends to overpredict the heat transfer coefficient.

NUREG/CR-3504: TURBULENCE MODELING IN THE COMMIX COMPUTER CODE. CHER, F. F.; DOMANUS, H. M.; SHA, W. T.; et al. Argonne National Laboratory. May 1984. 53pp. 8407110019. ANL-83-65. 25546:001.

The report describes the three additional turbulence models [0-equation (mixing-length), 1-equation (k), and 2-equation (k-E)] recently implemented in the COMMIS-1B computer code. COMMIS-1B is a three-dimensional, steady-state/transient, single-phase computer code for thermal-hydraulic analysis of single/multicomponent systems under normal and off-normal operating conditions. All three turbulence models are provided as options, and a user can select the one that is most appropriate for his or her application.

To validate these turbulence models, we have performed several numerical simulations and compared the results with experimental data. Three of the simulations--turbulent flow in a pipe, flow in a circular duct with sudden expansion, and thermal and fluid mixing in the cold leg and downcomer of a PWR--are presented here along with their comparisons with experimental data. More analyses are needed for further validation. Incorporation of the three turbulence models has expanded the range of application of the COMMIX code.

NUREG/CR-3505: A VOLUME-WEIGHTED SKEW-UPWIND DIFFERENCE SCHEME IN COMMIX. MIAO, C. C.; LYCZKOWSKI, R. W.; LEAF, G. K.; et al. Argonne National Laboratory. May 1984. 92pp. 8407180028. ANL-83-66. 25683:140.

A numerical difference scheme, called volume-weighted skew-upwind difference (VWSUD), has been developed, and Raithby's two-dimensional skew-upwind difference (SUD) scheme has been extended to three dimensions. Both schemes have been implemented in the energy equation of the COMMIX-1B computer program. The VWSUD scheme has the following five major features: (1) it has the same order of accuracy as SUD, but eliminates all of the undershoots observed in SUD; (2) it retains the simplicity of SUD, without resorting to the artificial cut-offs needed in SUD; (3) it significantly reduces numerical diffusion; (4) a linear stability analysis shows that VWSUD is numerically stable; and (5) a coarser mesh than for the pure-upwind difference scheme can be used while obtaining results that are of the same order of accuracy.

The assessment of SUD and VWSUD are accomplished by comparing several multidimensional thermal mixing benchmark computations with analytical solutions. In addition, the analysis of two thermal mixing experiments shows that use of the VWSUD scheme substantially improves agreement with thermocouple response data in regions with highly angled flows.

NUREG/CR-3506: J-R CURVE CHARACTERIZATION OF IRRADIATED LOW UPPER SHELF WELDS. HISER, A. L. ; LOSS, F. J. ; MENKE, B. H. Materials Engineering Associates, Inc. April 1984. 616pp. 8405210598. MEA-2028. 24531:001.

This investigation provides a data base of J-R curve trends from irradiated A 508 and A 533-B weld metals exhibiting low upper shelf Charpy-V (C(v) energy. These welds were made with Linde 80 flux of the same lots used for vessels currently in service. These materials exhibited postirradiation C(v) upper shelf energies of 58 J to 80 J. Compact toughness (CT) specimens of four different sizes (0.5T- to 4T-CT) were characterized. These specimens were irradiated to a fluence of ~1 x 10 (19) n/cm(2) > 1 MeV as part of the NRC-sponsored HSST program.

The J-R curves exhibited a power-law behavior for small crack extensions (e.g., < 2 mm). Irradiation decreased the level of the R-curve significantly in most cases. The value of J-integral at the initiation of crack growth (J(Ic) decreased on average by ~25% at 200 degrees centigrade and by ~35% at 288 degrees centigrade. The average value of tearing modulus (T(avg) was a more discriminating indication of the degradation due to irradiation, as T (avg) decreased on average by ~54% at 200 degrees centigrade and by ~69% at 288 degrees centigrade. A modest size effect associated with large specimens was indicted for tests in the unirradiated condition, while no size effect was apparent for tests in the irradiated condition.

These data compare favorably with correltions between C(v) upper shelf energy and J-R curve parameters observed from prior studies with 1T-CT specimens. These correltions could enhance the significance of C(v) reactor surveillance data with respect to structural integrity.

NUREG/CR-3507: AN ANALYSIS OF THE NRC SAFETY GOALS FOR NUCLEAR POWER. FISCHHOFF, B. Decision Research, Inc. * Oak Ridge National Laboratory. April 1984. 48pp. 8404300067. ORNL/SUB-7576/2. 24230:196.

The document analyzes the proposed "safety goals" with the general theory of standard setting. The analysis discusses the concept of "acceptable risk" and the attempt to build policy instruments around it.

NUREG/CR-3508: STANDARD SETTING STANDARDS: A SYSTEMATIC APPROACH TO MANAGING PUBLIC HEALTH AND SAFETY RISKS. FISCHHOFF, B. Decision Research, Inc. * Oak Ridge National Laboratory. February 1984. 66pp. 8403190444. ORNL/SUB-7576/3. 22681:004.

Standards are an effective means of managing hazardous technologies. This guide presents a general framework for the design development and implementation of safety standards. Particular strategies along with inherent strengths and weaknesses are described.

NUREG/CR-3511 V01: INTERIM RELIABILITY EVALUATION PROGRAM: ANALYSIS OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT. Volume 1. Main Report. PAYNE, A. C. Sandia Laboratories. May 1984. 273pp. 8405220017. SAND83-2086. 24552:007.

This report presents the results of the Probabilistic Risk Assessment (PRA) of Calvert Cliffs Unit 1 Nuclear Power Plant. The analysis was performed as part of the Interim Reliability Evaluation Program (IREP). The analysis used fault tree and event tree models as the primary tools to evaluate the risk due to a core melt at Calvert Cliffs. Core melt sequences initiated by one of three break-size LOCAs or one of six categories of transients were evaluated, and the dominant (i.e., highest frequency) sequences were further analyzed to estimate the magnitude of radionuclide release. The accident sequences were then placed into the release categories defined in the Reactor Safety Study to estimate this magnitude. The most significant sequences contributing to the core melt frequency are (1) Anticipated Transients Without Scram (ATWS) (44% of the total core melt frequency), (2) Small-small LOCAs (i.e., 3" to 1.9" in diameter) with makeup system failure in the recirculation phase (19% of the total core melt frequency), and (3) the loss of a DC bus followed by failure of secondary heat removal (14% of the total core melt frequency). The estimated core melt frequency for Calvert Cliffs Unit 1 (CC-1) is similar to the values predicted by PRAs of other PWRs.

NUREG/CR-3511 V02: INTERIM RELIABILITY EVALUATION PROGRAM: ANALYSIS OF THE CALVERT CLIFFS UNIT 1 NUCLEAR POWER PLANT. Volume II, Appendices A, B And C. PAYNE, A. C. Sandia Laboratories. October 1984. 778pp. 8412100350. 27868:001.

See NUREG/CR-3511, V01 abstract.

NUREG/CR-3512: RAPID FIELD METHOD FOR THE CONCENTRATION OF RADIOIODINE FROM MILK. SOLBRIG, R. M.; HUCHTON, R. L.; MOTES, B. G. Exxon Nuclear Co., Inc. (subs. of Exxon Corp.). February 1984. 24pp. 8402280010. ENICO-1137. 22429:280.

The development and testing of a practical field method for the collection of radioiodine from milk is reported. The technique is simple, rapid and relatively inexpensive to perform and yields a good iodine collection efficiency. The batch method involves adding a 250-ml volume of Dowex 1-x8 anion exchange resin to 3.5 liters of milk, mixing at a rate of 30 rotations per minute for three minutes, separating the resin and milk by pouring the mixture through a wire-screened canister, and rinsing the resin with 200-ml of distilled water. Advantages of the procedure include: minimal equipment and personnel training; an inexpensive industrial-grade resin; a total of ten minutes to complete; and an adaptability of the wire-screened canister to different geometries for direct counting. Additionally, the total collection efficiency of 81 + or - 6% is relatively independent of temperature, iodide concentration, mixing rate and mixing time.

NUREG/CR-3513: MECHANICAL RELIABILITY EVALUATION OF ALTERNATE MOTORS FOR USE IN A RADIOIODINE AIR SAMPLER. BIRD, S. K. ; HUCHTON, R. L. ; MOTES, B. G. ; et al. EG&G, Inc. July 1984. 41pp. 8409110069. WINCO-1006. 26437:284.

The purpose of the study was to evaluate the mechanical reliability of motors for use in a prototype system designed for post accident collection and measurement of radioiodine in the environs of a nuclear reactor. The two types of motors were tested for lifetimes and operational performance characteristics under extremes of temperature, relative humidity and/or in dusty air and rainfall. The 12 volt direct current motors exhibited satisfactory performance under all environmental conditions and demonstrated lifetimes of 47 hours, 97 hours and 188 hours. The 12 volt direct current voltage and satisfactory operation on alternating current voltage; at failure the AC/DC voltage motors demonstrated lifetimes of nominally 6 hours, 3 hours and 2 hours. The direct current voltage only motors are the better candidates for incorporation into the air sampler.

NUREG/CR-3514: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C. An Experimental Study of Nonredox Conditions. TOTH, L. M. ; PANNELL, K. D. ; KIRKLAND, O. L. Oak Ridge National Laboratory. April 1984. 47pp. 8405290434. ORNL/TM-8664. 24711:001.

The chemical behavior of iodine, I2, in (pH = 6 to 10) aqueous solutions containing 2500 ppm boron as H3BO3 (0.231 M) was studied at temperatures up to 150C. Absorption spectrophotometry was used to identify and monitor the iodine species present. Three objectives were considered: (1) species identification, with special attention given to "HOI"; (2) the kinetics of reaction between iodine and water to produce iodide and iodate ions; and (3) partition coefficients between liquid and vapor phases for individual iodine species.

Kinetic rate constants for the disproportionation of the "HOI" intermediate were measured. A typical activation energy for this reaction was found to be 28.4 kJ/mol (6.8 kcal/mol). No absorption bands can be assigned to the "HOI" intermediate even though it has been shown, in some cases, to be present at concentrations of >1 x 10(-3) M. A very low molar absorptivity (<10 M-1 cm-1) is probably responsible for its undetectability. A partition coefficient of >1 x 10(4) has been estimated for "HOI".

NUREG/CR-3515: SAFETY-RELATED OPERATION ACTIONS: METHODOLOGY FOR DEVELOPING CRITERIA. KOZINSKY, E. J. ; GRAY, L. H. ; BEARE, A. N. ; et al. Oak Ridge National Laboratory. April 1984. 166pp. 8405210559. ORNL/TM-8942. 24527:123.

This report presents a methodology for developing criteria for design evaluation of safety-related actions by nuclear power plant reactor operators, and identifies a supporting data base. It is the eleventh and final NUREG/CR Report on the Safety-Related Operator Actions Program, conducted by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission. The operator performance data were developed from training simulator experiments involving operator responses to simulated scenarios of plant disturbances; from field data on events with similar scenarios; and from task analytic data. A conceptual model was run, using the SAINT modeling language. Proposed is a quantitative predictive model of operator performance, the

"Operator Personnel Performance Simulation (OPPS) Model," driven by task requirements, information presentation, and system dynamics. The model output, a probability distribution of predicted time to correctly complete safety-related operator actions, provides data for objective evaluation of quantitative design criteria.

NUREG/CR-3518 V01: SLIM-MAUD: AN APPROACH TO ASSESSING HUMAN ERROR PROBABILITIES USING STRUCTURED EXPERT JUDGEMENT. Volume I: Overview of SLIM-MAUD. EMBREY, D. E.; HUMPHREYS, P.; ROSA, E. A.; et al. Brookhaven National Laboratory. July 1984. 36pp. 8408010166. BNL-NUREG-51716. 25867:253.

This two-volume report presents the procedures and analyses in developing an approach for structuring expert judgments to estimate human error probabilities. Volume I presents an overview of work performed in developing the approach: SLIM-MAUD (Success Likelihood Index Methodology, implemented through the use of an interactive computer program called MAUD--Multi-Attribute Utility Decomposition). Volume II provides a more detailed analysis of the technical issues underlying the approach.

NUREG/CR-3518 V02: SLIM-MAUD: AN APPROACH TO ASSESSING HUMAN ERROR PROBABILITIES USING STRUCTURED EXPERT JUDGEMENT. Volume II: Detailed Analysis Of The Technical Issues. EMBREY, D. E.; HUMPHREYS, P.; ROSA, E. A.; et al. Brookhaven National Laboratory. December 1984. 164pp. 8501020484. 28164:058.

This two-volume report presents the procedures and analyses performed in developing an approach for structuring expert judgments to estimate human error probabilities. Volume I presents an overview of work performed in developing the approach: SLIM-MAUD (Success Likelihood Index Methodology, implemented through the use of an interactive computer program called MAUD--Multi-Attribute Utility Decomposition). Volume II provides a more detailed analysis of the technical issues underlying the approach.

NUREG/CR-3520 V01: LONG-TERM RESEARCH PLAN FOR HUMAN FACTORS AFFECTING SAFEGUARDS AT NUCLEAR POWER PLANTS. Volume I: Summary And Users Guide. O'BRIEN, J. N.; FAINBERG, A. Brookhaven National Laboratory. August 1984. 42pp. 8409280071. BNL-NUREG-51718. 26750:228.

The first task was to identify and rank human factors affecting the quality of nuclear power plant safeguards in terms of their importance. The opinions of over 85 experts were solicited and 28 responses were received. These responses were rigorously analyzed to ascertain what human factors could be considered important to power plant safeguards. In addition, the Safeguards Summary List (NUREG-0525) was systematically analyzed for human factors influences. Also, relevant government and industry literature was reviewed. These data sources were then aggregated and an overall importance ranking of human factors issues was developed. This part of the research effort is fully documented and described in Chapter 2 of Volume II.

The second part of this effort involved determining the feasibility of conducting research in the areas found to be important to power plant safeguards. A determination of research feasibility was based on the practicality, usefulness, and acceptability of conducting research and using the results in a regulatory context. This part of the effort is fully documented in Chapter 3 of Volume II.

Research efforts addressing human factors in safeguards were then developed and prioritized according to the importance of human factors areas derived in the first part of the study and the feasibility of research determined in the second part. Research was also grouped to take advantage of common research approaches and data sources where appropriate. Chapter 4 of Volume II details the development of methodological groupings for optimizing resource use.

NUREG/CR-3520 V02: LONG-TERM RESEARCH PLAN FOR HUMAN FACTORS AFFECTING SAFEGUARDS AT NUCLEAR POWER PLANTS. Volume II: Development Of Detailed Analyses. O'BRIEN, J. N.; FAINBERG, A. Brookhaven National Laboratory. August 1984. 204pp. 8409280062. BNL-NUREG-51718. 26762:001.

See NUREG/CR-3520, V01 abstract.

NUREG/CR-3521: HYDROGEN-BURN SURVIVAL EXPERIMENTS AT FULLY INSTRUMENTED TEST SITE (FITS). RICHARDS, E.; ARAGON, J. J. Sandia Laboratories. November 1984. 53pp. 8412190375. SAND83-1715. 28028:158.

A series of hydrogen-burn experiments conducted for the Hydrogen-Burn Survival Program is described. The experiments, executed at Sandia's Fully Instrumented Test Site (FITS) facility, provided data concerning the Hydrogen-burn thermal environment as it relates to equipment survivability in nuclear power plants. The test plan, instrumentation, and results are presented, along with a brief discussion of test volume (scale) considerations. Conclusions drawn from the results concern repeatability of the tests, the suitability of thermocouples for measuring gas temperatures, and the effects of initial hydrogen concentrations and fans on the responses of calorimeters and components. The effect of initial steam concentration on temperature response cannot be determined because of preignition pressure considerations.

NUREG/CR-3522 V01: REFERENCE MATERIALS FOR NONDESTRUCTIVE ASSAY OF SPECIAL NUCLEAR MATERIALS, VOL 1: Uranium Oxide Plus Graphite Powder. SPRINKLE, J. K.; LIKES, R. N.; PARKER, J. L.; et al. Los Alamos Scientific Laboratory. January 1984. 41pp. 8402010368. LA-9910-MS V01. 22061:001.

This manual describes the fabrication of reference materials for use in gamma-ray-based nondestructive assay of low-density uranium-bearing samples. The sample containers are 2 liter bottles. The reference materials consist of small amounts of UO(2) spread throughout a graphite matrix. The (235)U content ranges from 0 to 100 g. The manual also describes the far-field procedure used with low-resolution detectors.

NUREG/CR-3522 V02: REFERENCE MATERIALS FOR NONDESTRUCTIVE ASSAY OF SPECIAL NUCLEAR MATERIAL. Volume 2: Thin Metal Foils Of Highly Enriched Uranium. SPRINKLE, J. K.; LIKES, R. N.; SMITH, H. A. Los Alamos Scientific Laboratory. January 1984. 15pp. 8402010313. LA-9910-MS V02. 22046:210.

This manual describes the fabrication of reference materials for use in gamma-ray-based nondestructive assay of small high-density uranium samples. The sample containers are small Petri dishes. The reference materials consist of thin circular discs of highly enriched uranium metal foil. The 235U content ranges from 0.2 to 10 g. The manual also describes the assay procedure used with low-resolution detectors.

NUREG/CR-3523: A RANKING SCHEME FOR MAKING DECISIONS ON THE RELATIVE TRAINING IMPORTANCE OF POTENTIAL NUCLEAR POWER PLANT MALFUNCTIONS. SELBY, D. L.; HENSLEY, W. T. Oak Ridge National Laboratory. February 1984. 121pp. 8403300306. ORNL/TM-8950. 22842:190.

The research summarized in this report was conducted as part of a program entitled "Nuclear Power Plant Entry Level Qualification and Training." A process is developed to assist in the selection of plant malfunctions which should be specifically addressed as part of the training program, and further guidance is given for determining which of those malfunctions should be included in simulator training. Consequences (C), difficulty (D), and frequency (F) rating forms are developed to determine the relative importance for training of any system malfunction. Plant malfunctions were categorized for 46 plant systems. Thirteen of these malfunction categories were then used to demonstrate the C-D-F rating forms.

NUREG/CR-3524: ORGANIZATIONAL INTERFACE IN REACTOR EMERGENCY PLANNING AND RESPONSE. SORENSEN, J. H.; COPENHAVER, E. D.; MILETI, D. S.; et al. Oak Ridge National Laboratory. July 1984. 52pp. 8409110100. ORNL-6010. 26446:236.

The purpose of this research was to determine if existing regulations have led to effective interfaces between utilities and offsite organizations in emergency planning and response. Findings suggest that regulations have provided the necessary framework for achieving adequate interfaces. That interface has been achieved is demonstrated by comprehensive response plans and good cohesiveness among organizations involved in emergency response. Interface problems identified in the research can be reduced by better implementation of existing regulations rather than by revision of existing ones.

NUREG/CR-3525: MECHANISTIC CORE-WIDE MELTDOWN AND RELOCATION MODELING FOR BWR APPLICATIONS. PADOWSKI, M. Z.; TALEYARKHAN, R.; LAHEY, R. T. Oak Ridge National Laboratory. January 1984. 75pp. 8401260101. ORNL/SUB/81-908. 21971:085.

This report summarizes the results of developmental work at Rensselaer Polytechnic Institute (RPI) of methods of core modeling for use in the analyses of the progression of accidents that involve core damage in BWRs. Accomplishments include the development of an analytical model for channel box and control rod heatup, oxidation, and melting, and a mechanistic core-wide meltdown and relocation model. These are provided in the form of a FORTRAN subroutine denoted MELRPI that can be exercised with an existing version of the MARCH code. The modeling concept and the modular structure employed in MELRPI have been designed so that new models subsequently developed for specific phenomena can be rapidly incorporated when they become available.

NUREG/CR-3526: IMPACT OF CHANGES IN DAMPING AND SPECTRUM PEAK BROADENING ON THE SEISMIC RESPONSE OF PIPING SYSTEMS. CHUANG, T. Y.; LU, S. C.; BENDA, B. J.; et al. Lawrence Livermore National Laboratory. March 1984. 72pp. 8404120384. UCRL-53491. 24035:062.

The Technical Committee on Piping Systems of the Pressure Vessel Research Committee has proposed two modifications that affect seismic analysis of piping systems to regulatory guides. One modification would change damping values for piping systems specified in Regulatory Guide 1.122.

In this study we quantified the reduction in piping responses of three piping systems in the Zion nuclear power plant resulting from these two modifications, separately and in combination. We concluded that: The proposed damping values reduce piping response substantially; and the proposed alternative to peak broadening reduces piping response only marginally.

We calculate the seismic response of the three piping systems by two methods: Response spectrum analysis and multi-support time history analysis. We used the proposed modifications in the response spectrum analysis. The results of the response spectrum analysis were calibrated against those of time history analysis. We found that conservatism remains under the proposed modifications.

One of the three piping systems was used to show the potential benefit of the proposed modifications. We found that both snubbers and 7 of the 10 horizontal restraints could be removed without causing stresses in the piping system to exceed code allowables. Hence, the potential benefit of the proposals is very promising.

NUREG/CR-3529: REVIEW OF THE ARKANSAS NUCLEAR ONE GENERATING STATION UNIT NO. 1 EMERGENCY FEEDWATER SYSTEM RELIABILITY ANALYSIS. YOUNGBLOOD, R. W.; PAPAZOGLOU, I. A. Brookhaven National Laboratory. February 1984. 102pp. 8403230133. BNL-NUREG-51721. 22739:130.

The purposes of this report are: (1) to review the Emergency Feedwater System Upgrade Reliability Analysis for the Arkansas Nuclear One Nuclear Generating Station Unit No. 1, and (2) to estimate the probability that the Emergency Feedwater System will not perform its mission for each of three different initiators: (1) loss of main feedwater with offsite power available, (2) loss of offsite power, (3) loss of all 4160 VAC power. The scope, methodology, and failure data are prescribed by NUREG-0611, Appendix III.

NUREG/CR-3530: REVIEW OF THE DAVIS-BESSE UNIT NO. 1 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS. YOUNGBLOOD, R. W.; PAPAZOGLOU, I. A. Brookhaven National Laboratory. February 1984. 17pp. 8402170430. BNL-NUREG-51722. 22318:319.

The purpose of this report is to review the "Davis-Besse Unit No. 1 Auxiliary Feedwater System Reliability Analysis Final Report," and to provide an independent estimate of the Auxiliary Feedwater System Reliability. This report presents estimates of the probability that the Auxiliary Feedwater System will not perform its mission for each of three different initiators: (1) loss of main feedwater with offsite power available, (2) loss of offsite power, (3) loss of all 4160 VAC power. The scope, methodology, and failure data are prescribed by NUREG-0611, Appendix III.

NUREG/CR-3531: REVIEW OF SEABROOK UNITS 1 AND 2 AUXILIARY FEEDWATER SYSTEM RELIABILITY ANALYSIS. FRESCO, A.; YOUNGBLOOD, R. W.; PAPAZOGLOU, I. A. Brookhaven National Laboratory. February 1984. 194pp. 8402170503. BNL-NUREG-51723. 22304:320.

This report presents the results of a review of the Emergency Feedwater System Reliability Analysis for Seabrook Nuclear Station Units 1 and 2. The objective of this report is to estimate the probability that the Emergency Feedwater System will fail to perform its mission for each of three different initiators: (1) loss of main feedwater with offsite power available, (2) loss of offsite power, (3) loss of all AC power except vital instrumentation and control 125 VDC/120 VAC power. The scope, methodology, and failure data are prescribed by NUREG-0611, Appendix III. The results are compared with those obtained in NUREG-0611 for other Westinghouse plants.

NUREG/CR-3532: RESPONSE OF RUBBER INSULATION MATERIALS TO MONOENERGETIC ELECTRON IRRADIATIONS. BUCKALEW,W.H.; WYANT,F.J.; LOCKWOOD,G.J. Sandia Laboratories. January 1984. 60pp. 8401130105. SAND83-2098. 21821:040.

The papers published in this report were presented at the Eleventh Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 24-28, 1983. The papers describe the results and program plans in nuclear reactor safety research conducted in this country and abroad. International participation in the exchange of nuclear safety research information is consistently encouraged by the Nuclear Regulatory Commission and was exemplified at this meeting by the relatively large number of papers (23) given by representatives of seven different European countries, Japan and Taiwan.

NUREG/CR-3533: RADON ATTENUATION HANDBOOK FOR URANIUM-MILL TAILINGS COVER DESIGN. ROGERS,V.C.; NIELSON,K.K. Rogers & Associates Engineering Corp. KALKWARF,D.R. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 89pp. 8405210555. PNL-4878. 24529:119.

This handbook has been prepared to facilitate the design of earthen covers to control radon emission from uranium mill tailings. Radon emissions from bare and covered uranium mill tailings can be estimated by equations based on diffusion theory. Basic equations are presented for calculating surface radon fluxes from covered tailings, or alternatively, the cover thickness required to satisfy a given radon flux criterion. Procedures are also given for measuring diffusion coefficients for radon, or for estimating them from empirical correlations. Since long-term soil moisture content is a critical parameter in determining the value of the diffusion coefficient, methods are given for estimating the long-term moisture contents of soils. The effects of cover defects or advection are also discussed and guidelines are given for determining if they are significant. For most practical cases, advection and cover defect effects on radon flux can be neglected. Several examples are given to demonstrate cover design calculations, and an extensive list of references is included.

NUREG/CR-3535: AGE-DEPENDENT DOSE-CONVERSION FACTORS FOR SELECTED BONE-SEEKING RADIONUCLIDES. CRISTY,M.; LEGGETT,R.W.; DUNNING,D.E.; et al. Oak Ridge National Laboratory. May 1984. 79pp. 8405210611. ORNL/TM-8929. 24534:274.

The transuranic elements and the radiostrontiums are bone-seekers and are potentially important contributors to bone dose from releases from a breeder reactor such as the Clinch River Breeder Reactor. Currently available age-specific dose-conversion factors for these nuclides are based on methods of ICRP Publication 2, published in 1959. ICRP Publications 26 and 30, published in 1977 and 1979, outline methodology incorporating new models and new concepts of risk, including consideration of dose to endosteal surfaces and active bone marrow rather than dose to whole bone. This report gives dose-conversion factors for acute intake of a given radionuclide by ingestion or inhalation at various ages from birth to adulthood, using the methodology of ICRP 26 and 30, but modified and extended as appropriate to include age-dependence. Results for 32 isotopes of strontium, plutonium, americium, and curium are tabulated.

NUREG/CR-3536: SIMULATION OF LOADING CONDITIONS FOR A TYPE A PACKAGE CONTAINING AMERICIUM-241 INVOLVED IN AN AIRPLANE CRASH AT DETROIT METRO AIRPORT IN JANUARY 1983. MAXFIELD, B. W. ; WOO, H. H. Lawrence Livermore National Laboratory. January 1984. 19pp. 8402060452. UCRL-53492. 22107:167.

On January 11, 1983, a United Airlines DC-8F cargo aircraft crashed shortly after takeoff from Detroit Metro Airport. A lower rear cargo pit had a type A package containing 10,000 Americium-241 (241(Am) solid-form sources, each of 1.5-microcurie (mCi) strength, used in smoke detectors. Although burned and somewhat battered, the 1-gal metal can holding all these sources was recovered completely intact with no release of radioactive material to the environment or loss of any sources. This report describes Lawrence Livermore National Laboratory's attempt to reconstruct, as closely as practical, the mechanical and thermal environments experienced by this can during and immediately after the accident. Mechanical loading of the metal can in shipping carton was simulated by impacts from a 16-lb pendulum mass falling through vertical displacements to demolish internal plastic jars and to produce major deformation of the metal can. The thermal environment was best reproduced by the simple burning of the outer shipping carton.

NUREG/CR-3539: IMPACT OF CONTAINMENT BUILDING LEAKAGE ON LWR ACCIDENT RISK. HERMANN, O. W. ; BURNS, T. J. Oak Ridge National Laboratory. April 1984. 23pp. 8405210566. ORNL/TM-8964. 24526:257.

The consequences, or risks, from light-water reactor accidents have been evaluated as a function of containment building leakage rates. The analysis used the set of generic source terms and frequencies of occurrence developed as representative of the range of postulated types of accidents currently applied in reactor safety research, and the calculated result was the variable M(sp), defined as the accident-spectrum-weighted impact fraction rate from containment building leakage. Explicitly, M(sp) was formulated as the sum of fractional increases in consequences, due to the building leakage, for each type of accident weighted by its frequency of occurrence. The base case common to similar types of analyses was applied. The computed result was M(sp) less than or equal to 1.5 10(-3) fractional increase in the accident spectrum risk per %/day containment building leakage rate.

NUREG/CR-3543: SURVEY OF OPERATING EXPERIENCE FROM LERS TO IDENTIFY AGING TREND. Progress Report, September 1983. MURPHY, G. A. ; CASADA, M. L. HOY, H. C. Oak Ridge National Laboratory. January 1984. 44pp. 8402060478. ORNL/NSIC-216. 22110:230.

The results of a study utilizing the Oak Ridge National Laboratory (ORNL) Nuclear Operations Analysis Center (NOAC) computer files of operating experience reports [licensee event reports (LERs), abnormal occurrences, etc.] are summarized. In this study, specific time-related degradation mechanisms are identified as possible causes of a reportable occurrence. Data collected on domestic commercial nuclear power plants covering 1969 to 1982 yielded over 5800 events attributable to age-related failures. Of these events, 2795 were attributable to instrument drift, which are addressed separately in the report. The remaining events (3098) were reviewed, and data were collected for each event, identifying the specific system, component, and subpart; the information included age-related failure mechanism, severity of failure, and method of detection of the failure. About two-thirds of the failures were judged to be degraded, with one-third listed as catastrophic failures. No events were found to be incipient failures because an LER is prepared only on degraded or catastrophic failure conditions that place plant operation outside the Technical Specifications. The study found that information desired for evaluation of aging effects (equipment, age, service life, and environment) was seldom available from LERs. This reflects the intent of the LER system as a regulatory instrument, rather than an engineering data collection system. The study is a part of the overall Nuclear Regulatory Commission research on nuclear power plant aging effects.

NUREG/CR-3544: BETA PARTICLE MEASUREMENT AND DOSIMETRY AT NRC-LICENSED FACILITIES. RATHBUN,L.A.; ENDRES,G.W.; FOX,R.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 58pp. 8409070234. PNL-4886. 26419:203.

Researchers from Pacific Northwest Laboratory (PNL) have conducted beta radiation measurements under laboratory and field conditions to assess the degree of the measurement problem and offer suggestions for possible remedies. The primary measurement systems selected for use in this study were the silicon (Si) surface barrier spectrometer system and the multielement beta dosimeter. Three boiling water reactors (BWRs), two pressurized water reactors (PWRs), and one fuel fabrication facility were visited during the course of the study. Although beta fields from cobalt-60 were the most common type found at locations associated with spent fuel handling, liquid radioactive waste, and BWR turbine components. Commercially-available dosimeters and survey instruments were used to measure the same laboratory and licensee facility beta fields characterized with PNL's active and passive spectrometers. A prototype survey meter was also used in the laboratory measurements. The commercial instruments and dosimeters used in this study typically responded low to the beta fields measured, especially where maximum beta energies were less than approximately 500 keV. A single calibration factor is usually not adequate for either beta dosimeters or instruments. There is a need for more refinement in beta measurement devices and training for the users of such devices.

NUREG/CR-3546: THE TEMPERATURE DEPENDENCE OF FATIGUE CRACK GROWTH RATES OF A 351 CF8A CAST STAINLESS STEEL IN LWR ENVIRONMENT. CULLEN,W.H.; TAYLOR,R.E.; TORRONEN,K.; et al. Materials Engineering Associates, Inc. April 1984. 36pp. 8405220001. MEA-2030. 24551:323.

The fatigue crack growth rates for A 351-CF8A cast stainless steel were determined over a range of temperatures from 95 degrees centigrade to 338 degrees centigrade (200 degrees to 640 degrees fahrenheit). The waveform was 17 mHz sinusoidal and the load ratio was 0.2. The environment was borated and lithiated water with a dissolved oxygen content of ~ 1 ppb. The results show an easily measurable (factors of 2 to 8) increase in crack growth rates due to the environment. However, these rates are well within the known band of results for low-alloy pressure vessel and low-carbon piping steels in LWR environments. An extensive fractographic investigation shows fatigue fracture surfaces covered with brittle-like features. This morphology is similar to that resulting from the environmental assistance mechanism producing increased crack growth rates due to stress-corrosion cracking.

NUREG/CR-3547: A SETS USER'S MANUAL FOR ACCIDENT SEQUENCE ANALYSIS. STACK, D. W. Sandia Laboratories. January 1984. 165pp. 8404020272. SAND83-2238. 22875:109.

This manual describes the use of the Set Equation Transformation System (SETS) to perform the accident sequence analysis portion of a probabilistic risk assessment (PRA) for a nuclear power plant. Other tasks in a PRA provide the input to the accident sequence analysis task. The SETS computer program is used to process these inputs to identify the dominant failure modes and to compute an approximate frequency of occurrence for an accident sequence. The use of SETS for each step in an accident sequence analysis is described and an example SETS user program is provided for each step.

NUREG/CR-3549: EVALUATION OF CONTAINMENT LEAK RATE TESTING CRITERIA. DOUGAN, J. R. Oak Ridge National Laboratory. March 1984. 56pp. 8403300318. ORNL/TM-8909. 22826:303.

Revision of Appendix J, to reflect technological advances and testing experience, has been under consideration for years and has culminated in the issuance of a draft version of a proposed revision to Appendix J. To assist in the revision process, a review of 49 Type A test reports and 46 Type B and C test reports was accomplished. Exemption requests found in 25 reports and 100 License Event Reports were also reviewed. Two major findings of the data analysis were that Type A test duration of less than 24 hours are practical and that almost all Type A test failures and delays were caused by excessive leakage through Type B and C tested components. Excessive valve leakage represented 38% of the LERs and highlighted the need for improved maintenance, repair and testing of these components. Excessive airlock leakage was generally the result of worn, damaged, misaligned, or dirty door seals. The proposed revision to Appendix J appears to be very responsive to the results of test experience and technological changes. The introduction of a regulatory guide provides a vehicle for the NRC to specify any exceptions to the relevant industry standards and to resolve areas of conflict.

NUREG/CR-3550: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Annual Summary Report - Fiscal Year 1983. MILLER, R. L.; BAUMANN, B. L. United Nuclear Corp. (subs. of UNC Resources, Inc.). January 1984. 173pp. 8402170304. 22304:029.

This document summarizes work performed during the 1983 fiscal year for the Nuclear Regulatory Commission's Evaluation of Nuclear Facility Decommissioning Projects program. This report describes actual work performed during the reporting period and work planned for the future. Included as an appendix to this report is a draft of the Decommissioning Code Table/Indexes for BWR, PWR, Research and Test Reactors included in this study. Other appendices list current data from the TMI-2 recovery efforts and Shippingport Atomic Power Station decommissioning.

NUREG/CR-3553: AN EFFICIENT SIMULATION APPROACH FOR EVALUATING THE POTENTIAL EFFECTS OF NUCLEAR POWER PLANT SHUTDOWNS ON ELECTRICAL GENERATING SYSTEMS. VANKUIKEN, J. C. Argonne National Laboratory. January 1984. 37pp. 8402010274. ANL/EES-TM-233. 22048:061.

Production-cost and reliability studies of electrical utility systems have, in the past, been hindered by high computational costs and long lead times for preparing case studies. Computational costs have been especially high for simulating large systems or long periods and for conducting comprehensive sensitivity analyses. This report describes modeling developments that help alleviate previous constraints. A new simulation approach preserves an acceptable level of accuracy and detail from previous production-cost and reliability methods and, at the same time, significantly reduces the computational requirements. An automated data assembly package facilitates the collection and preparation of simulation inputs used to characterize utility systems. Several case studies are investigated to test and demonstrate the new procedures for reactor shutdown evaluations.

NUREG/CR-3554: RADIONUCLIDE MIGRATION IN GROUNDWATER. Annual Progress Report For 1982. ROBERTSON, D. E. ; TOSTE, A. P. ; ABEL, K. H. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 82pp. 8402210013. PNL-4773. 22323:031.

Research has continued at a low-level waste disposal facility to characterize the physiochemical species of radionuclides migrating in groundwater. This facility consists of an unlined basin and connecting trench which receives effluent water containing low levels of a wide variety of fission and activation products and trace amounts of transuranic radionuclides. The effluent water percolates through the soil and a small fraction of it emerges at seepage springs located some 260 meters from the trench. The disposal basin and trench are very efficient in retaining most of the radionuclides, but trace amounts of a number of radionuclides existing in mobile chemical forms migrate in the groundwater from the trench to the springs. This facility provides the opportunity for characterizing the rates and mechanisms of radionuclide migration in groundwaters, identifying retardation processes, and validating geochemical models.

NUREG/CR-3556: NONINTERACTIVE SIMULATION EVALUATION FOR CRT-GENERATED DISPLAYS. BLACKMAN, H. S. ; GERTMAN, D. I. ; GILMORE, W. E. ; et al. EG&G, Inc. January 1984. 36pp. 8401130092. EGG-2284. 21820:289.

The United States Nuclear Regulatory Commission (USNRC) is sponsoring an ongoing research program to develop methods of assessing various types of computer generated displays currently being implemented in nuclear power plant control rooms. The purpose of this report is to present a noninteractive simulation technique for the evaluation of computer generated displays. Four safety parameter display formats were evaluated in two separate experiments. Three formats were evaluated in Experiment I (STAR, BAR, METER). Two formats were evaluated in Experiment II (BAR, P-T map). All formats contained top-level safety parameters minimally necessary for the safe operation of a pressurized water reactor at the Loss-of-Fluid Test (LOFT) reactor. Subjects for the experiments were current or former operators at the Loss-of-Fluid Test (LOFT) reactor. The results of this experiment have indicated that the noninteractive technique can be used to evaluate the detection and recognition of transients in safety parameter display evaluation. In addition, the data suggest that, given a reliable set of parameters and good human engineering, that graphical format of the display has negligible impact of performance. The implications of these results are discussed in terms of future work and display design.

NUREG/CR-3560: EVALUATION METHODS FOR THE CONSEQUENCES OF BELOW WATER TABLE MINE DISPOSAL OF URANIUM MILL TAILINGS. MCKEON, T. J.; NELSON, R. W. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 90pp. 8403230173. PNL-4904. 22742:217.

A method has been developed at the Pacific Northwest Laboratory to evaluate the environment consequences of below water table disposal of uranium mill tailings in mine stopes. The method described uses analytical expressions for the velocity potential and examines the convection transport of tailings liquor and leachate through the aquifer and into a water supply well located down gradient from the mine stope. The arrival distribution of contaminant (mass flux versus time) and the concentration pumped from the well as a function of time are the final results of the analysis.

NUREG/CR-3561: EDDY CURRENT ROUND ROBIN TEST ON LABORATORY PRODUCED INTERGRANULAR STRESS CORROSION CRACKED INCONEL STEAM GENERATOR TUBES. BICKFORD, R. L.; CLARK, R. A.; DOCTOR, P. G.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 130pp. 8402270213. PNL-4695. 22395:263.

This report provides the results of an eddy current round robin test conducted on Inconel 600 steam generator tubing specimens with laboratory induced IGSCC. This test was an attempt to establish the best available nondestructive testing method for characterizing IGSCC in Inconel steam generator tubing. The participants were permitted to use any available eddy current method and were not limited to probes or methods that are or could be used in commercial primary side in-service inspection. Areas covered in this report include production of the specimens, defect characterizations by the participating teams, data from the subsequent destructive metallographic analysis of the specimens and a statistical evaluation of the results per team and between teams.

NUREG/CR-3562: STEAM GENERATOR TUBE INTEGRITY PROGRAM LEAK RATE TESTS-PROGRESS REPORT. CLARK, R. A.; BICKFORD, R. L. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 68pp. 8402010373. PNL-4629. 22061:090.

This interim report presents preliminary results on leak rate tests performed on through-wall defected Inconel 600 steam generator tubing. Tube defects included an EDM (electro-discharge machine) notch and IGSCC (intergranular stress corrosion cracks) of various lengths. Tests were conducted at PWR operating temperatures with leakage of hot water/steam into air. A number of IGSCC cracks were unstable under the experiment conditions of these initial tests, continuing to grow until system capacity limitations resulted in decreased pressure differential. However, initial testing also pointed to a need for reconfiguration of the test apparatus to sustain increased flow and, more importantly, alter the mode of control. The initial test configuration is based on flow control, with pressure differential across the specimen an independent variable. This often results in pressure increases too rapid to establish the initiation of crack instability. A reconfigured system based on pressure control with flow as an independent parameter is being recommended for future tests.

NUREG/CR-3563: COBRA/TRAC SIMULATION OF SEMISCALE S-UT-2 TEST. BIAN, S. H.; THURGOOD, M. J.; KELLY, J. M. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 63pp. 8402060343. PNL-4908. 22107:202.

The computer code COBRA/TRAC was used to simulate a Small-Break Loss-of-Coolant Accident (SBLOCA) test performed at the Semiscale MOD-2A Test Facility operated by the Idaho National Engineering Laboratory. The results of the simulation were compared with the results of the actual test. The comparison showed that the COBRA/TRAC calculation gave a reasonable match with the measured data and that the code has the capability to model the loop components in an integrated coolant systen for a pressurized water reactor (PWR).

NUREG/CR-3564: PRESSURIZED THERMAL SHOCK:TEMPEST COMPUTER CODE SIMULATION OF THERMAL MIXING IN THE DOWNCOMER OF A PRESSURIZED WATER REACTOR. EYLER,L.L.; TRENT,D.S. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 92pp. 8404300363. PNL-4909. 24233:144.

The TEMPEST computer program was used to simulate fluid and thermal mixing in the cold leg and downcomer of a pressurized water reactor under emergency core cooling high-pressure injection (HPI), which is of concern to the pressurized thermal shock (PTS) problem. Application of the code was made in performing an analysis simulation of a full-scale Westinghouse three-loop plant design cold leg and downcomer. Verification/assessment of the code was performed and analysis procedures developed using data from Creare 1/5-scale experimental tests. Results of three simulations are presented. The first is a no-loop-flow case with high-velocity, low-negative-bouyancy HPI in a 1/5-scale model of a cold leg and downcomer. The second is a no-loop-flow case with low-velocity, high-negative density (modeled with salt water) injection in a 1/5-scale model. Comparison of TEMPEST code predictions with experimental data for these two cases show good agreement. The third simulation is a three-dimensional model of a one loop of a full size Westinghouse three-loop plant design. Included in this latter simulation are loop components extending from the steam generator to the reactor vessel and a one-third sector of the vessel downcomer and lower plenum. No data were available for this case.

NUREG/CR-3566: SOCIOECONOMIC CONSEQUENCES OF NUCLEAR REACTOR ACCIDENTS. TAWILL,J.J.; CALLAWAY,J.W.; COLES,B.L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1984. 206pp. 8406270117. PNL-4911. 25171:138.

This report identifies and characterizes the off-site socioeconomic consequences that would likely result from a severe radiological accident at a nuclear power plant. The types of impacts that are addressed include economic impacts, health impacts, social/psychological impacts and institutional impacts. These impacts are identified for each of several phases of a reactor accident--from the warning phase through the post-resettlement phase. The relative importance of the impact during each accident phase and the degree to which the impact can be predicted are indicated. The report also examines the methods that are currently used for assessing nuclear reactor accidents, including development of accident scenarios and the estimating of socioeconomic accident consequences with various models. Finally, a critical evaluation is made regarding the use of impact analyses in estimating the contribution of socioeconomic consequences to nuclear accident reactor accident risk.

NUREG/CR-3567: TRAC-PF1: AN ADVANCED BEST-ESTIMATE COMPUTER PROGRAM FOR PRESSURIZED WATER REACTOR ANALYSIS. * Los Alamos Scientific Laboratory. April 1984. 60pp. 8405220073. LA-9944-MS. 24558:011.

The Transient Reactor Analysis Code (TRAC) is being developed at the Los Alamos National Laboratory to provide advanced best-estimate predictions of postulated accidents in light water reactors. The TRAC-PF1 program provides this capability for pressurized water reactors and for many thermal-hydraulic experimental facilities. The code features either a one-dimensional or a three-dimensional treatment of the pressure vessel and its associated internals; a two-phase, two-fluid nonequilibrium hydrodynamics model with a noncondensable gas field; flow-regime-dependent constitutive equation treatment; optional reflood tracking capability for both bottom flood and falling-film quench fronts; and consistent treatment of entire accident sequences including the generation of consistent initial conditions. A new numerical algorithm is used in the one-dimensional hydrodynamics that permit this portion of the fluid dynamics to violate the material Courant condition. This technique permits large time steps and, hence, reduced running time for slow transients.

This report describes the thermal-hydraulic models and the numerical solution methods used in the code. Detailed programming and user information also are provided. A second Los Alamos report, "TRAC-PF1 Developmental Assessment," presents the results of the developmental assessment calculations.

NUREG/CR-3569: SPECIAL AND DOSIMETRIC MEASUREMENTS OF PHOTON FIELDS AT COMMERCIAL NUCLEAR SITES. ROBERSON, P. L.; FOX, R. A.; HOLBROOK, K. L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 182pp. 8408290171. PNL-4915. 26299:119.

Spectral and dosimetric measurements of photon fields were performed at seven commercial nuclear reactor sites. Revisions to 10 CFR 20 that specify exposure-to-dose conversion factors (Cx) much greater than unity for photons between 40 KeV and 200 KeV could impact personnel monitoring practices. Monitoring at effective depths of 1 cm of tissue and shallower could underestimate doses received from high-energy photon fields (>3 MeV).

No locations with large C(x) factors (approximately 1.5 rad/R) were found. The most significant production of low-energy photons was found to be due to photon scattering. The scatter continuum has an effective Cx factor of approximately 1.2 rad/R. One location was found with a nearly pure scatter spectrum. Other locations contained significant contributions from medium-energy photons due primarily to radioactive decay of cobalt and cesium isotopes. Monitoring requirements at 0.007-cm and 1.0-cm depths in tissue were found to be adequate for estimating dose received in radiation fields containing high-energy photons. Enhanced surface doses attributed to high-energy knock-on electrons were measured in all locations monitored. Personnel monitoring techniques may provide inaccurate results in high-energy fields.

NUREG/CR-3572: DETERMINATION OF METABOLIC DATA APPROPRIATE FOR HLW DOSIMETRY (ICRP-30), I. ECKERMAN, K. F.; LEGGETT, R. W.; MEYER, R. E.; et al. Oak Ridge National Laboratory. May 1984. 78pp. 8405290437. ORNL/TM-8939. 24696:176.

This report provides an initial evaluation of the dependence on chemical forms of estimates of health effects from radionuclides in high-level waste (HLW). Discussion is limited mainly to a review of studies of plutonium, americium, neptunium, and strontium that may be useful in identifying (a) chemical forms of these radionuclides that are likely to reach humans after migration from a waste repository and (b) differences in metabolism and organ doses that result from intake of various chemical forms of these radionuclides; we also attempt to identify research needs in these two areas. In addition to providing a limited review of the literature, this report identifies some of the problems involved in determining speciation of these radionuclides in the environment and provides a general picture of the potential errors that may be involved in applying models assumed to be independent of chemical form to estimate metabolism and dose from exposure to different chemical species of a radionuclide.

NUREG/CR-3573: PERSONNEL EXPOSURE FROM RIGHT CYLINDRICAL SOURCES (PERCS). The Theory, The Code And Examples. REECE, W. D.; HADLEY, R. T.; HARDTY, R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 173pp. 8403190449. PNL-4923. 22680:170.

A new interactive point kernel shielding code for gamma rays named PERCS (Personnel Exposure From Right Cylindrical Surfaces) is presented with theory, a listing and examples. PERCS calculates doses from more complicated geometries faster than other known shielding codes. The code is graphics-oriented, interactive, menu-driven and easy to use. PERCS is especially suited for calculation of dose arising from activated corrosion products plated on primary piping in commercial power plants.

NUREG/CR-3577: THE MEASUREMENT OF COUNTERCURRENT PHASE SEPARATION AND DISTRIBUTION IN A TWO-DIMENSIONAL TEST SECTION. BUKHARI, M.; LAHEY, R. T. Rensselaer Polytechnic Institute, Troy, NY. January 1984. 220pp. 8401250134. 21949:028.

The degree of phase separation that occurs in the core of a pressurized water reactor (PWR) during various postulated accidents is an important consideration for studying the course of events during such accidents. The dependence of countercurrent phase separation and distribution phenomena on flow quality, mass flux and system geometry was studied experimentally in a two-dimentional (2-D) test section. A two-phase (air/water) mixture flowed upwards and single-phase water flowed downward along one side of the test section. This countercurrent flow configuration was intended to simulate the so-called "chimney effect" in the diabatic JAERI 2-D experiments in Japan.

A large air/water loop was used with a 3' x 3' x 0.5" (91.44 cm x 1.27 cm) test section to study phase separation and distribution effects. A traversing single beam gamma-densitometer was used to measure the chordal average void fractions at several elevations along the test section. Cross-plots between various low conditions and geometries were made. An error analysis giving the total error in the void fraction measurements was also performed.

High speed photographs were also made of the flow structure, to provide information on flow regimes. The photographic records and the void fraction and hydraulic inflow/outflow data are presented in a form suitable for the assessment of advanced generation computer codes (eg: TRAC).

NUREG/CR-3578: STEAM GENERATOR GROUP PROJECT PROGRESS REPORT. Task 3-Health Physics. REECE, W. D.; HOENES, G. R.; PARKHURST, M. A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. January

1984. 33pp. 8401260105. PNL-4711. 21969:288.

The gamma radiation fields in and around the retired Surry steam generator were measured extensively with thermoluminescent dosimeters (TLD's) and/other standard radiation instruments. The techniques of measurement and the results are described for locations outside the shell, inside the channel head, and inside the secondary side of the steam generator. The gamma fields ranged from more than 10 R/hr in the middle of the tube bundle on the secondary side to less than 5 mR/hr at the bottom of the outside of the shell below the channel head. Co-60 was the only detected gamma emitter. The results of the measurements were used in an analytical model which predicted the Co-60 inventory to be between 70 and 87 curies.

NUREG/CR-3579: STEAM GENERATOR GROUP PROJECT. Progress Report On Data Acquisition/Statistical Analysis. DOCTOR, P. G.; BUCHANAN, J. A.; MCINTYRE, J. M.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 38pp. 8402170311. PNL-3955. 22318:181.

A major task of the Steam Generator Group Project (SGGP) is to establish the reliability of the eddy current inservice inspections of PWR steam generator tubing, by comparing the eddy current data to the actual physical condition of the tubes via destructive analyses. This report describes the plans for the computer systems needed to acquire, store and analyze the diverse data to be collected during the project. The real-time acquisition of the baseline eddy current inspection data will be handled using a specially designed data acquisition computer system based on a Digital Equipment Corporation (DEC) PDP-11/44. The data will be archived in digital form for use after the project is completed. Data base management and statistical analyses will be done on a DEC VAX-11/780. Color graphics will be heavily used to summarize the data and the results of the analyses. The report describes the data that will be taken during the project and the statistical methods that will be used to analyze the data.

NUREG/CR-3580: STEAM GENERATOR GROUP PROJECT. Semiannual Progress Report, July-December 1982. CLARK, R. A.; LEWIS, M. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 34pp. 8402170308. PNL-4692. 22318:223.

The Steam Group Project (SGGP) is an NRC program joined by additional sponsors. The SGGP utilizes a steam generator removed from service at a nuclear plant as a vehicle for research on a variety of safety and reliability issues. This report is a semi-annual summary of progress of the program. Information is presented on positioning the generator into the Steam Generator Examination Facility, and examination of the secondary side to confirm pretransport generator condition. The report then presents radiological field mapping results and personnel exposure monitoring data. Radiation field reduction achieved in channel head decontamination efforts is reported. The results of a profilometry examination to determine the extent of denting are summarized. Plans for unplugging of selective explosively plugged tubes are discussed.

NUREG/CR-3581: STEAM GENERATOR GROUP PROJECT. Annual Report - 1982. CLARK, R. A.; LEWIS, M.; MUSCARA, J. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1984. 111pp. 8404200016. PNL-4693. 24604:026.

The Steam Generator Group Project (SGGP) is an NRC program joined by additional sponsors. The SGGP utilizes a steam generator removed from service at a nuclear plant as a vehicle for research on a variety of safety and reliability issues. This report is an annual summary of progress of the program for 1982. Information is presented on the Steam Generator Examination Facility (SGEF), especially designed and constructed for this research. Loading of the generator into the SGEF is then discussed. The report then presents radiological field mapping results and personnel exposure monitoring. This is followed by information on field reduction achieved by channel head decontaminations. The report then presents results of a secondary side examination through shell penetrations placed prior to transport, confirming no change in generator condition due to transport. Decontamination of the channel head is discussed followed by plans for eddy current testing and removal of tube plugs placed during service. Results of a preliminary profilometry examination are then provided.

NUREG/CR-3583: EVALUATION OF LOW-ALTITUDE REMOTE SENSING TECHNIQUES FOR OBTAINING SITE CHARACTERISTIC INFORMATION. ESTES, J. E. ; SCEPAN, J. ; RITTER, L. ; et al. California, Univ. of, Santa Barbara, CA. April 1984. 79pp. 8407110004. S-762-R. 25545.247.

The Nuclear Regulatory Commission contracted with EG&G/EM and the University of California, Santa Barbara to assess the potential of photographic remote sensing for demographic and environmental monitoring. Aerial infrared imagery and ground truth along with collateral data provided information on site area demographics and land use, land cover characteristics. The ability to determine transient populations from remotely sensed data was also evaluated. Both manual and machine-assisted techniques for extracting these data from reflectance infrared images were qualitatively assessed. The NASA Aircraft Programs 'U-2' acquired color infrared imagery at scales of 1:65,000 and 1:130,000, and Keystone Aerial Surveys (Philadelphia, Pennsylvania) using a Lear-Jet acquired color infrared imagery at scales of 1:36,000, 1:48,000, 1:60,000, and 1:80,000. Data on residence types and counts, industrial facilities types and location, transient facilities, transportation networks, and the location of water bodies were generated specifically for the study site surrounding the Limerick Power Station in Pottstown, Pennsylvania. Of the three techniques of population estimations examined, the "Dwelling Unit" method was evaluated for respective utility and accuracy within NRC guidelines. The level of spatial and classification accuracy of the derived products depended on both scale and image quality. Area weighed thematic accuracy from manual analysis was 96%, while by-category accuracies ranged from 71% to 100%.

NUREG/CR-3584: COMMONLY USED NUCLEAR MATERIAL MEASUREMENTS AND THEIR SOURCES OF ERROR. ROBERTS, F. P. ; BROUNS, R. J. ; BYERS, K. R. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. January 1984. 116pp. 8402240233. 22381:001.

In a study commissioned by the Nuclear Regulatory Commission, Battelle's Pacific Northwest Laboratories examined practices for calibrating and/or determining biases for the principal nuclear material measurement systems in use by the nuclear industry: uranium gravimetry, uranium determination by reduction-oxidation titrimetry, plutonium determination by amperometric titration, isotopic analysis by mass spectrometry, isotopic-dilution assay of iron and plutonium, uranium fluorometry, iron spectrophotometry, mass measurements, volume measurements, flow measurements, nondestructive assays, and holdup measurements. A number of frequently used methods are described in this report. The principle, procedures, apparatus, applications, and calibration for each are discussed. The sources of measurement bias are identified, also the expected magnitudes of errors associated with the methods.

NUREG/CR-3585: DE MINIMIS WASTE IMPACTS ANALYSIS METHODOLOGY. OZTUNALI, O. I. Dames & Moore. ROLES, G. W. NRC - No Detailed Affiliation Given. February 1984. 531pp. 8402280006. 22415:001.

A calculational methodology is presented which can be used to estimate impacts from disposal of radioactive waste by less restrictive means ("de minimis" waste disposal). The methodology consists of two computer codes: one which determines annual radiological impacts to individuals and to populations from release of de minimis waste into less restrictive disposal pathways, and another which determines limiting concentrations of radionuclides based upon comparison with a set of individual dose limitation criteria. Operational impacts are calculated for de minimis waste transportation, treatment by incineration, and disposal. Long-term impacts after disposal (e.g., groundwater migration) are also calculated as are possible impacts from recycle of metal or glass. Alternatives for waste treatment/disposal include on-site, off-site as a municipal waste, and off-site as a hazardous waste.

NUREG/CR-3588: THE EFFECT OF LOCA SIMULATION PROCEDURES ON CROSS-LINKED POLYOLEFIN CABLE'S PERFORMANCE. BUSTARD, L. D. Sandia Laboratories. April 1984. 100pp. 8407060059. SAND83-2406. 25441:190.

Electrical and mechanical properties of three commercial cross-linked polyolefin (XLPO) materials, typically used as electrical cable insulation, have been monitored during three simulations of nuclear power plant aging and accident stresses. For one XLPO cable we first performed accelerated thermal aging, then irradiated the samples to the combined aging and LOCA total dose. Finally, we applied a steam exposure. For a second and third set of XLPO cables we used simultaneous radiation and steam exposures to simulate a LOCA environment.

Our measurement parameters during these tests included: dc insulation resistance, ac leakage current, ultimate tensile strength, ultimate tensile elongation, percentage dimensional changes, and percentage moisture absorption. We present test results for three XLPO materials.

NUREG/CR-3589 V01: REACTOR SAFETY RESEARCH QUARTERLY REPORT. January-March 1983. * Sandia Laboratories. July 1984. 179pp. 8409180335. SAND83-2425. 26588:001.

Sandia National Laboratories is conducting phenomenological research related to the safety of commercial nuclear power reactors. The overall objective of this work is to provide NRC a comprehensive data base essential to (1) defining key safety issues, (2)understanding risk-significant accident sequences, (3) developing and verifying models used in safety assessments, and (4) assuring the public that power reactor systems will not be licensed and placed in commercial service in the United States without appropriate consideration being given to their effects on health safety. This report describes progress in a number of activities dealing with current safety issues relevant to both light water and breeder reactors. The work includes a broad range of experiment to simulate accidental conditions to provide the data base required to understand important accident sequences and to serve as a basis for development and verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Current major emphasis is focused on providing information to NRC relevant to (1) its deliberations and decisions dealing with severe LWR accidents and (2) its safety evaluation of the proposed Clinch River Breeder Reactor.

NUREG/CR-3589 V02: REACTOR SAFETY RESEARCH QUARTERLY REPORT. April-June 1983. * Sandia Laboratories. July 1984. 166pp. 8409180443. SAND83-2425. 26588:180.

See NUREG/CR-3589,V01 abstract.

NUREG/CR-3589 V03/4: REACTOR SAFETY RESEARCH QUARTERLY REPORT, JULY-SEPTEMBER 1983 AND OCTOBER-DECEMBER 1983. Volumes 27 And 28. * Sandia Laboratories. October 1984. 175pp. 8411130682. SAND83-2425. 27474:013.

This report describes progress in a number of activities dealing with current safety issues relevant to both light water reactors (LWRs) and breeder reactors. The work includes a broad range of experiments to simulate accidental conditions to provide the data base required to understand important accident sequences and to serve as a basis for development and verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Such a program must include the development of analytical models, verified by experiment, which can be used to predict reactor and safety system performance under a broad variety of abnormal conditions. Current major emphasis is focused on providing information to NRC relevant to (1) its deliberations and decisions dealing with severe LWR accidents and (2) its safety evaluation of the proposed Clinch River Breeder Reactor.

NUREG/CR-3590: EVALUATION OF ISOTOPE DILUTION MASS SPECTROMETRY FOR BIOASSAY MEASUREMENT OF URANIUM, PLUTONIUM, AND THORIUM IN URINE. DYER, F.F.; MAY, M.P.; WALKER, R.L.; et al. Oak Ridge National Laboratory. August 1984. 79pp. 8408300272. ORNL/TM-9006. 26330:224.

A study was made to evaluate the sensitivity, precision and accuracy, and practicality of isotope dilution mass spectrometry (IDMS) for bioassay of uranium, plutonium, and thorium in human urine. The study showed that uranium at a concentration of 0.06 mg/L (0.04 pCi/L natural uranium), plutonium at 3 pg/L (0.2 p Ci/L Pu-239, and thorium at 0.1 mg/L (0.01 pCi/L Th-232) could be measured with an uncertainty (RSD) of ten percent using 10 ml samples. The lower limits of detection for uranium and thorium were set by background contamination, whereas the detection limit for plutonium was determined by chemical yield and intrinsic instrumental sensitivity factors. Precision and accuracy is excellent (~1-3%, RSD) at concentration levels where background contamination is insignificant and instrumental sensitivity is adequate.

Comparison of IDMS with other methods shows the technique is more sensitive than conventional fluorometric methods but is similar in sensitivity to alpha-radioactivity measurement methods that utilize large sample volumes (1 L). Costs for urine analysis by IDMS ($60-$100 per sample) are estimated to be considerably higher than cost for fluorometric analysis and approximately the same as the cost for alpha-radioactivity methods. Other methods that have been used or are currently under development are discussed.

NUREG/CR-3591 V01: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1980-1981 A Status Report. COTTRELL, W. B. ; MINARICK, J. W. ; AUSTIN, P. N. ; et al. Oak Ridge National Laboratory. July 1984. 262pp. 8408290190. 26312:001.

Descriptions of fifty-eight operational events reported as Licensee Event Reports, which occurred at commercial light-water reactors during 1980-1981 and which are considered to be precursors to potential severe core damage, are presented, along with associated event trees, categorization, and subsequent analyses. This study is a continuation of the work presented in NUREG/CR-2497 which somewhat similarly evaluated the 1969-1979 events. The current study incorporates improvements which evolved from an assessment of the comments on the earlier report and applies these in the assessment of the LERs which occurred during 1980 and 1981. The report sequentially discusses (1) the general rationale for this study, (2) the program methods for LER review and documentation, (3) the calculation of function failure probabilities and initiating event frequencies based upon precursor data, (4) the use of the conditional probability of subsequent severe core damage estimates to rank precursor events and estimate an average industrywide risk of severe core damage, and (5) the conduct of sensitivity analyses on these results. There was some apparent decrease in most initiating event frequencies and function failure probabilities in the 1980 and 1981 period, as compared to the earlier report. Although it was not possible to conclude that these decreases were statistically significant, they did result in a reduction in the industry average estimated severe core damage frequency for 1980-1981 as compared to the 1969-1979 period.

NUREG/CR-3591 V02: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1980-1981 A Status Report. COTTRELL, W. B. ; MINARICK, J. W. ; AUSTIN, P. N. ; et al. Oak Ridge National Laboratory. July 1984. 266pp. 8408290198. 26311:001.

See NUREG/CR-3591, V01 abstract.

NUREG/CR-3593 V01: SYSTEMS INTERACTION RESULTS FROM THE DIGRAPH MATRIX ANALYSIS OF A NUCLEAR POWER PLANT'S HIGH PRESSURE SAFETY INJECTION SYSTEM. SACKS, I. J. ; ASHMORE, B. C. ; ALESSO, H. P. Lawrence Livermore National Laboratory. July 1984. 184pp. 8409280090. UCRL-53467. 26749:181.

The report describes the demonstration of the Digraph-Matrix Analysis on a Nuclear Pouer Plant's High Pressure Safety Injection System. The demonstration work was beyond the scope of both the methods and the criteria used by the NRC to license nuclear power plants. The analysis discovered components whose failure could jeopardize the High Pressure Injection System given the postulated accident. All these components had been previously considered both in the safety analysis and in the licensing review. The results demonstrate the capability of Digraph-Matrix Analysis to model an accident sequence (including front-line systems, support systems, and operator actions) as a continuously integrated model to discover functional systems interactions. Also, the method is scrutable and can be used on a complex system which contains both a large number of components and dependent loops. Volume 1 is the main report and the description of the method. Volume 2 contains the digraphs, adjacency listings, and data base.

NUREG/CR-3593 V02: SYSTEMS INTERACTION RESULTS FROM THE DIGRAPH MATRIX ANALYSIS OF A NUCLEAR POWER PLANT'S HIGH PRESSURE SAFETY INJECTION SYSTEM. SACKS, I. J.; ASHMORE, B. C.; ALESSO, H. P. Lawrence Livermore National Laboratory. July 1984. 165pp. 8409280123. UCRL-53467. 26735:096.

See NUREG/CR-3593,V01 abstract.

NUREG/CR-3595: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM - FIVE YEAR PLAN FY 1983-1987. * Oak Ridge National Laboratory. April 1984. 138pp. 8405220027. ORNL/TM-9008. 24603:202.

The first 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 Material Engineering Branch, Division of Engineering Technology, 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 nine technical tasks and a management function. A background statement and a plan-of-action is given for each. The nine technical tasks address fracture methodology and analysis, materials characterization, crack growth, crack arrest, irradiation effects, cladding evaluations, intermediate-vessel testing, thermal-shock testing, and pressurized thermal-shock experiments.

NUREG/CR-3596: SEVERE ACCIDENT SEQUENCE ANALYSIS (SASA) PROGRAM SEQUENCE EVENT TREE: BOILING WATER REACTOR ANTICIPATED TRANSIENT WITHOUT SCRAM. BRUSKE, S. Z.; WRIGHT, R. E. EG&G, Inc. April 1984. 22pp. 8405220031. EGG-2288. 24601:001.

The United States Nuclear Regulatory Commission is sponsoring an on-going safety research program to assess dominant risk events in boiling water reactors. As part of this program, a sequence event tree for a boiling water reactor anticipated transient without scram (ATWS) has been developed and quantified. The goal of the sequence event tree is to provide a logical representation of the systems that must respond to an ATWS, the required operator response to the event, operator actions that could be performed in response to multiple failures, and the phenomenological concerns. The purpose of the sequence event tree is to provide a basis upon which to perform additional deterministic thermal-hydraulic and core damage analyses in the most cost effective manner based on the most likely sequence of events that will lead to containment/core damage. The ATWS sequence event tree is based on the General Electric Owners Group emergency procedure guidelines and on preliminary deterministic thermal-hydraulic analyses performed by EG&G Idaho, Inc. personnel at the Idaho National engineering laboratory under direction of the Severe Accident Sequence Analysis Program.

NUREG/CR-3597: EQUIPMENT QUALIFICATION METHODOLOGY RESEARCH: TESTS OF RTDs. WYANT, E. J.; MINOR, E. E. Sandia Laboratories. October 1984. 77pp. 8411130737. SAND84-0938. 27472:248.

Ten resistance temperature detectors (RTDs), from three manufacturers were subjected to an abbreviated loss-of-coolant accident (LOCA) environment (saturated steam and chemical spray) simulation test as part of the NRC-sponsored Equipment Qualification Methodology Research Test Program (A-1355). The test was a "screening test" on unaged specimens that lasted about 24 hours and was of short duration to isolate any obvious problem areas. The LOCA environment caused functional failures and some physical damage in four of the RTDs tested. One RTD failed early in the test, two others of the same type produced erroneous temperature readings 7.5 hours into the test. Post-test investigations revealed that water leakage into the head areas of the three affected RTDs (as well as one other of the same model) may have contributed to the anomolous behavior.

NUREG/CR-3598: OCCUPATIONAL RADIOLOGICAL MONITORING AT URANIUM MILLS. SWAJA, R.E.; SIMS, C.S. Oak Ridge National Laboratory. February 1984. 97pp. 8402280114. ORNL-6023. 22416:199.

This document provides guidance and procedures for conducting an occupational radiological monitoring program at uranium mills. Included are a review of the objectives of an occupational monitoring program and a description of normal physical and radiological environments at uranium mills. Detailed monitoring procedures are presented for airborne particulates, radon and radon daughters, external radiation, and surface contamination. Although specifically written for uranium mills, some of the procedures contained in this document may be applied to other uranium recovery facilities with similar environments.

NUREG/CR-3599: SOURCES OF UNCERTAINTY IN THE CALCULATIONS OF LOADS ON SUPPORTS OF PIPING SYSTEMS. * Oak Ridge National Laboratory. RODABAUGH, E.C. E.C. Rodabaugh Associates, Inc. July 1984. 73pp. 8408130041. 26040:166.

Loads on piping systems are obtained from an analysis of the piping system. The piping system analysis involves uncertainties from various sources. These sources of uncertainties are discussed and ranges of uncertainties are illustrated by simple examples. The sources of uncertainties are summarized and assigned a judgmental ranking of the typical relative significance of the uncertainty.

NUREG/CR-3600: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-4. OSBORNE, M.F.; COLLINS, J.L.; LORENZ, R.A.; et al. Oak Ridge National Laboratory. June 1984. 70pp. 8407020156. ORNL/TM-9011. 25275:328.

The fourth in a series of high-temperature fission product release tests was conducted in which a 20.3-cm-long fuel specimen from the Peach Bottom-2 reactor was heated for 20 min at a maximum temperature of ~1850 degrees centigrade in a flowing steam-helium atmosphere. The test specimen was part of a fuel rod which was irradiated to ~10.10 MWd/kg.

Posttest metallographic examination of the fuel specimen revealed evidence of cladding melting at each of the transverse cuts that were made. Gas analysis during the test indicated that ~54% of the cladding was oxidized. Total oxidation did not occur because of the low steam flow which was used.

Gamma sepctrometry (GS) and neutron activation (NA) analyses of test components revealed the following releases: (1) GS - 21.1% (85)Kr, 31.7% (137)Cs; and (2) NA - 24.7% (129)I (percentages of the total calculated segment inventories). A value of 35.8% cesium release was determiner by counting the fuel rod segment before and after the test. If the pellet-clad gap fission gas inventory had also been available for release in the test, the (85)Kr release would have been 31.3%.

Significant releases of radiogenic Rb, Cd, Ag, and Br, as well as trace amounts of Te, La, Ba, Sr, and Eu, were detected by spark-source mass spectrometric analysis.

NUREG/CR-3601: MANAGEMENT AND ORGANIZATIONAL ASSESSMENTS: A REVIEW OF SELECTED ORGANIZATIONS. NADEL, M. V.; KERWIN, C. M. Battelle Human Affairs Research Centers. * Battelle Memorial Institute, Pacific Northwest Laboratories. February 1984. 42pp. 8403070109. PNL-4813. 22561:172.

This report reviews the processes and criteria used by organizations other than the NRC in conducting management and organization audits and evaluations. As part of a larger project assisting the NRC in establishing improved procedures and guidelines for assessing the managenent and organization of applicants for nuclear power plant operating licenses, this report provides a comparative perspective on organizational assessment. The organizations whose management audits are reviewed are state public utility commissions, the Comptroller of the Currency, the Department of Health and Human Services' Office of Health Maintenance Organizations, the Food and Drug Administration, the General Accounting Office, and a large commercial insurance company. This report examines the purposes, areas of emphases, and processes used in these reviews. These organizations conclude that management is the key performance of organizational functions.

NUREG/CR-3602: FUEL PERFORMANCE ANNUAL REPORT FOR 1982. BAILEY, W. J. Battelle Memorial Institute, Pacific Northwest Laboratories. TOKAR, M. NRC - No Detailed Affiliation Given. March 1984. 99pp. 8404110312. PNL-4817. 22997:135.

This annual report, the fifth in a series, provides a brief description of fuel performance during 1982 in commercial nuclear power plants. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience, 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-3603: MINET VALIDATION SURVEY USING EBR-II TEST DATA. VAN TUYLE, G. J. Brookhaven National Laboratory. May 1984. 39pp. 8405210571. BNL-NUREG-51733. 24529:311.

A natural circulation test transient performed at EBR-II facility is simulated using the MINET computer code, and calculated results are compared against data from the plant. The MINET EBR-II representation includes much of the intermediate loop and the steam generator system, and corresponds to the portion of the plant usually represented by MINET when it is executed with SSC, the Super System Code. MINET calculations agreed well with the plant transient data, with discrepancies well within uncertainties in thermocouple time constants and boundary conditions.

NUREG/CR-3604: BOLTING APPLICATIONS. CZAJKOWSKI, C. J. Brookhaven National Laboratory. May 1984. 303pp. 8406120535.

BNL-NUREG-51735. 24898:004.

An investigation of bolting practices specific to the nuclear industry was performed. The report covered a large spectrum of topics e.g. bolts embedded in concrete, specifications, inspection of bolting, both at receipt and inservice. Plots of preload versus yield strength for different bolting materials in different environments are presented as well as information relative to the stress corrosion cracking resistance of the more recent reactor internals bolting materials A286 and Inconel X-750. Part of the report contains input by Standard Pressed Steel Inc. (a bolting consultant) relative to bolting standards, cottering methods and potential areas for bolting improvement.

NUREG/CR-3605: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Summary Report - Plum Brook Reactor Facility. DOERGE, D. H.; MILLER, R. L. United Nuclear Corp. (subs. of UNC Resources, Inc.). February 1984. 54pp. 8403150214. 22649:052.

This document summarizes information concerning the decommissioning of the Plum Brook Reactor Facility, which was placed in a Nuclear Regulatory Commission (NRC) approved safe storage configuration. The data were placed in a computerized information retrieval/manipulation system which permits future utilization of this information in decommissioning of similar facilities. The information is presented both in computer output form and a manually assembled summarization.

Complete cost data were not readily available and decommissioning activities did not in all cases conform with current criteria for the SAFSTOR decommissioning mode, therefore no cost comparisons were made.

NUREG/CR-3606: NUCLEAR POWER PLANT CONTROL ROOM CREW TASK ANALYSIS DATABASE: SEEK SYSTEM. (Users Manual). BURGY, D.; SCHROEDER, L. General Physics Corp. May 1984. 134pp. 8406190517. GP-R-212106. 25029:052.

The Crew Task Analysis SEEK Users Manual was prepared for the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission. It is designed for use with the existing computerized Control Room Crew Task Analysis Database. The SEEK system consists of a PRIME computer with its associated peripherals and software augmented by General Physics Corporation SEEK database management software. The SEEK software programs provide the Crew Task Database user with rapid access to any number of records desired. The software uses English-like sentences to allow the user to construct logical sorts and outputs of the task data. Given the multiple-associative nature of the database, users can directly access the data at the plant, operating sequence, task, or element level - or any combination of these levels. A complete description of the crew task data contained in the database is presented in NUREG/CR-3371, "Task Analysis of Nuclear Power Plant Control Room Crews (Volumes 1 and 2)."

NUREG/CR-3607: RADIONUCLIDE DISTRIBUTIONS AND MIGRATION MECHANISMS AT SHALLOW LAND BURIAL SITES: 1982 Annual Report of Research Investigations On The Distribution, Migration And Containment Of Radionuclides At Maxey Flats, Kentucky. KIRBY, L. J. Battelle Memorial Institute, Pacific Northwest Laboratories. February 1984. 80pp. 8403070093. 22563:2?9.

Subsurface waters at Maxey Flats are anoxic, have a high alkaorganic carbon. The trench leachates are extremely variable in composition. Prominent radionuclides include (3)H, (60)Co, (90)Sr, (137)Cs, (238)(239)(240)Pu and (241)Am. A wide spectrum of dissolved organic compounds is present in the leachates, including EDTA, polar organics and decomposition products from the waste forms. Cobalt-60 and plutonium are present as EDTA complexes and (90)Sr and (137)Cs are associated with carboxylic acid type compounds. The chemistry of these waters changes drastically as they become oxic and plutonium becomes less mobile under these new conditions. Water enters the trenches by infiltration through the trench caps, though subsidence areas, and through interfaces between new landfill and the original soil. Lateral flow is very complex and slow and apparently occurs mainly by fracture flow. The plastic infiltration barrier installed in 1981-1982 has been effective in reducing soil moisture if cracks and leaks are eliminated. To date, no direct evidence of radionuclide transport to offsite locations by subsurface flow has been confirmed. The offsite distribution of radionuclides, except for tritium, is comparable to the ambient fallout from nuclear weapons testing. Tritium concentrations in water offsite are orders of magnitude below MPC levels.

NUREG/CR-3608: RELAP5 ASSESSEMENT: LOFT LARGE BREAK L2-5. THOMPSON, S. L.; KMETYK, L. N. Sandia Laboratories. April 1984. 115pp. 8405220255. SAND83-2549. 24602:001.

The RELAP5 independent assessment project at Sandia National Laboratories is part of an overall effort funded by the NRC to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accidents and off-normal conditions. The RELAP5 code is being assessed at SNLA against test data from various integral and separate effects test facilities. As part of this assessment matrix, a large break transient performed at the LOFT facility has been analyzed.

The results show that RELAP5/MOD1 correctly calculates many of the major system variables (i.e., pressure, break flows, peak clad temperature) early in a large break LOCA. The major problems encountered in the analyses were incorrect pump coastdown and loop seal clearing early in the calculation, excessive pump speedup later in the transient (probably due to too much condensation-induced pressure drop at the ECC injection point), and excess ECC bypass calculated throughout the later portions of the test; only the latter problem significantly affected the overall results. This excess ECC bypass through the downcomer and vessel-side break resulted in too-large late-time break flows and high system pressure due to prolonged choked flow conditions. It also resulted in a second core heatup being calculated after the accumulator emptied, since water was not being retained in the vessel. Analogous calculations with a split-downcomer nodalization delivered some ECC water to the lower plenum, which was then swept up the core and upper plenum and out the other (pump-side) break; thus no significant differences in long-term overall behavior were evident.

NUREG/CR-3610: NEUTRON DOSIMETRY AT COMMERCIAL NUCLEAR PLANTS: Final Report Of Subtask C: 3He Neutron Spectrometer. BRACKENBUSH, L.; REECE, W. D.; TANNER, J. E. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1984. 120pp. 8410120008. PNL-4943. 26983:001.

In commercial nuclear power plants, personnel routinely enter containment for maintenance and inspections while the reactor is operating and can be exposed to intense neutron fields. The low-energy neutron fields found in reactor containment cause problems in proper interpretation of TLD-albedo dosimeters and survey instrument readings. Described is a technique that can aid plant health physicists to improve the accuracy of personnel neutron dosimetry programs. A (3)He neutron spectrometer can be used to measure neutron energy spectra and determine dose equivalent rates at work locations inside containment. Energy correction factors for TLD-albedo dosimeters can be determined from the measured spectra if the dosimeter energy response is known, or from direct measurements with dosimeters placed on phantoms at locations where the dose equivalent rate has been measured. This report describes how to assemble a spectrometer system using only commercially available components, how to use it for reactor energy spectrum measurements, and how to analyze the data and interpret the results. Both (3)He and multisphere spectrometers were used to measure neutron energy spectra and dose equivalent at three PWRs and one BWR. In general, the (3)He spectrometer measures higher dose equivalent rates than the multisphere spectrometer. In the energy range from 10 keV to 1 MeV, the dose equivalents measured by the (3)He spectrometer and multisphere spectrometer agree within about 35% for the spectra measured.

NUREG/CR-3612: PREDICTION OF FAR-FIELD SUBSURFACE RADIONUCLIDE DISPERSION COEFFICIENTS FROM HYDRAULIC CONDUCTIVITY MEASUREMENTS. A Multidimensional Stochastic Theory With Application To Fractured Rocks. WINTER, C. L.; NEUMAN, S. P.; NEWMAN, C. M. Arizona, Univ. of, Tucson, AZ. March 1984. 66pp. 8404050505. 22912:091.

A multidimensional stochastic theory is presented for far-field dispersion due to the spatial variability of hydraulic conductivities. We use a second-order perturbation approach to relate the far-field velocity vector, V, and dispersion tensor, D, to the mean and covariance of the local seepage velocity vector, v, and the local dispersion tensor, d. We find that, in general, V is not necessarily equal to the ensemble mean of v, micron, and that D is a second-rank symmetric tensor. In the particular case where v x velocity vector = 0 (e.g., incompressible fluid in a rigid porous medium of uniform effective porosity), V becomes equal to micron, and our expressions for D simplify to those presented by Gelhar and Axness [1983]. We further extend a conclusion of these authors, that as the Peclet number, v, increases, D becomes asymptotically linear in micron, by showing that it holds for arbitrary velocity covariance functions. Finally, we derive expressions for D as a function of v for situations where the logarithm of hydraulic conductivity fits a spherical covariance or semivariogram function, as is often the case. These expressions are applied to log hydraulic conductivity data form packer tests conducted in seven boreholes penetrating fractured granites near Oracle, southern Arizona.

NUREG/CR-3613: EVALUATION AND ACCEPTANCE OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE. Annual Rept for 1983. ATTERIDGE, D. G.; BRUEMMER, S. M.; AGE, R. E. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1984. 55pp. 8406280214. PNL-4941. 25192:277.

Pacific Northwest Laboratory (PNL), under a program sponsored by the Division of Engineering Technology of the U.S. Nuclear Regulatory Commission (NRC), is conducting a program to determine a method for evaluating the acceptance of welded and repair-welded stainless steel (SS) piping for light-water reactor (LWR) service. Validated models, based on experimental data, will be developed to predict the degree of sensitization (DOS) and the intergranular stress corrosion cracking (IGSCC) susceptibility in the heat affected zone (HAZ) of the SS weldments. IGSCC is caused by a combination of a sensitized microstructure, an aggressive environment, and tensile stress. Control of any of these three factors can eliminate IGSCC in most practical situations.

This program will measure and model the development of a sensitized microstructure as it pertains to welded and repair-welded SS pipe. An empirical correlation between a material's DOS and its susceptibility to IGSCC will be determined using constant extension rate tests (CERTs). The successful completion of these tasks will result in a method for assessing the effects of weld/repairing parameters on the IGSCC susceptibility of component-specific nuclear reactor welds/repairs

NUREG/CR-3614: CONSTANT EXTENSION RATE TESTING OF SA302 GRADE B MATERIAL IN NEUTRAL AND CHLORIDE SOLUTIONS. CZAJKOWSKI, C. J. Brookhaven National Laboratory. February 1984. 48pp. 8402220099. BNL-NUREG-51736. 22364:096.

A test program was conducted on welded specimens (both stress relieved and non stress relieved) of SA302 Grade B materials. The specimens were tested in a constant extension rate apparatus in various environments in order to reproduce the transgranular cracking at Indian Point 3. The report concludes that SA302 Grade B material is susceptible to transgranular stress corrosion cracking in constant extension rate testing with as little as 1 ppm chloride (as $CuCl_2$) in 268 degrees C H_2O.

NUREG/CR-3615: HYDRODYNAMICS OF TWO PHASE FLOW THROUGH HOMOGENEOUS AND STRATIFIED POROUS LAYERS. CHU, W.; LEE, H.; DHIR, V. K.; et al. California, Univ. of, Los Angeles, CA. January 1984. 158pp. 8401250129. 21949:251.

An experimental investigation of two phase flow through porous layers formed of non-heated glass particles (nominal diameter 1-6 mm) has been made. Particulate bed depths of 30 cm and 70 cm were used. The effect of particle size distribution, bed porosity and bed stratification on void fraction and pressure drop through particulate beds formed in cylindrical and rectangular test sections has been investigated. The superficial velocity of liquid (water) is varied from 1.83-18.3 mm/s while the superficial velocity of gas (air) is varied from 0-68.4 mm/s. These superficial velocities were chosen so that pressure drop and void fraction measurement could be made for the porous layers in fixed and fluidized states. A model based on drift flux approach has been developed for the void fraction in homogeneous beds. Using the two phase friction pressure drop data, the relative permeabilities of the two phases have been concluded with void fraction.

The void fraction and two phase friction pressure gradient in beds composed of mixtures of spherical particles as well as shapes of different nominal sizes have also been examined. It is found that the models for single size particles are also applicable to mixtures of particles if a mean particle diameter for the mixture is defined.

The observations in stratified beds indicate depletion or build up of voids at the interface between high and low permeability regions. Blocking of the flow into one of the layers of laterally stratified beds caused the pressures at different horizontal locations at the same bed height to be different from each other.

NUREG/CR-3616: TRANSPORT AND SCREEN BLOCKAGE CHARACTERISTICS OF REFLECTIVE METALLIC INSULATION MATERIALS. BROCARD, D.N. Alden Research Laboratory. * Sandia Laboratories. January 1984. 48pp. 8402010365. ARL-124-83/M39F. 22061:040.

A loss-of-coolant-accident (LOCA) in a nuclear power plant could result in the formation of insulation debris which could transport to PWR sump screens (or BWR RHR suction intakes) and result in screen blockage. This report presents the transport and screen blockage characteristics of reflective metallic insulation materials and supplements previously acquired information on fibrous insulation materials (see NUREG/CR-2982).

These tests revealed that thin metallic foils (0.0025" and 0.004") could transport a low flow velocities, 0.2-0.5 ft/sec. Thicker foils (0.008") transported at higher velocities, 0.4-0.8 ft/sec, and "as fabricated" half cylinder insulation units required velocities in excess of 1.0 ft/sec for transport. These tests also provided information on screen blockage patterns that showed blockage could occur at the lower portion of the screen as foils readily flipped on the screen when reaching it. The tests also revealed that, although transport of foils occurred in a folding and tumbling mode, the foils did not become "water borne" and did not block the screen above their largest dimension. A maximum 80% blockage was observed in these tests.

NUREG/CR-3617: NOBLE GAS, IODINE, AND CESIUM TRANSPORT IN A POSTULATED LOSS OF DECAY HEAT REMOVAL ACCIDENT AT BROWNS FERRY. WICHNER, R.P.; WEBER, C.F.; WRIGHT, A.L.; et al. Oak Ridge National Laboratory. September 1984. 199pp. 8410120013. ORNL/TM-9028. 26978:001.

This report presents an analysis of the movement of noble gas, iodine, and cesium fission products within the Mark-I containment BWR reactor system represented by Browns Ferry Unit 1 during a postulated accident sequence initiated by a loss of decay heat removal capability following a scram. This accident could be brought under control by various means, but the sequence with no operator action ultimately leads to failure folloued by loss of water from the reactor vessel, core degradation due to overheating, and reactor vessel failure with attendant movement of core debris onto the drywell floor. The fission product transport analysis is based on the no-operator-action sequence and provides an estimate of fission product inventories, as a function of time, within 14 control volumes outside the core, with the atmosphere considered as the final control volume in the transport sequence. We find small barrier for noble gas ejection to air, these gases being effectively purged from the drywell and reactor building by steam and concrete degradation gases. In contrast, large degrees of holdup for iodine and cesium are projected due to the chemical reactivity of these elements. Only about 2 x 10(-4%) of the initial iodine and cesium activity are predicted to be released to the atmosphere. Principal barriers for release are deposition on reactor vessel and containment ualls.

NUREG/CR-3618: OCA-P, A DETERMINISTIC AND PROBABILISTIC FRACTURE-MECHANICS CODE FOR APPLICATION TO PRESSURE VESSELS. CHEVERTON, R.D.; BALL, D.G. Oak Ridge National Laboratory. July 1984. 105pp. 8408080415. ORNL-5991. 25977:338.

OCA-P is a probabilistic fracture-mechanics code that was prepared specifically for the purpose of evaluating the integrity of PWR pressure vessels when subjected to overcooling-accident loading conditions. The code has two-dimensional and some three-dimensional-flaw capability; it is based on linear elastic fracture mechanics; and it can treat cladding as a discrete region. Both deterministic and probabilistic analyses can be performed, and for the former analysis it is possible to conduct a search for critical values of the fluence and the nil ductility reference temperature corresponding to incipient initiation of the initial flaw. The probabilistic portion of OCA-P is based on Monte Carlo techniques, and simulated parameters include fluence, flaw depth, fracture toughness, nil ductility reference temperature, and concentrations of copper, nickel and phosphorous. Plotting capabilities include the construction of critical-crack-depth diagrams (determinstic analysis) and various histograms (probabilistic analysis).

NUREG/CR-3619: SURVEY OF COMMERCIAL NON-NUCLEAR SECURITY PROGRAMS. ISHIMOTO, W. Y. SAS of Texas, Ltd. March 1984. 49pp. 8404120389. 24035:152.

This study provides a limited, but current review of security techniques and practices used in analogous non-nuclear commercial industries to defend against external threats. Nine non-nuclear commercial industries which engage in high-value or high-risk operations were interviewed. Current security periodicals and books were also reviewed to determine whether there are any unique security practices, techniques, or procedures in use by non-nuclear commercial industries that may benefit the security posture of the NRC and its licensees. The study briefly reviews ten specific types of threat posed by the external adversary; basic reasons for adversarial success; and detection and prevention strategies. The "average" level of security, as evidenced in practice, and the most stringent level of security used by the interviewees are also examined.

NUREG/CR-3620: INTRUDER DOSE PATHWAY ANALYSIS FOR THE ONSITE DISPOSAL OF RADIOACTIVE WASTES: The ONSITE/MAXI1 Computer Program. NAPIER, B. A.; PELOQUIN, R. A.; KENNEDY, W. E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. October 1984. 130pp. 8412110041. PNL-4054. 27890:218.

The ONSITE/MAXI1 computer software package was developed for use by NRC in reviewing license applictions for onsite disposal of radioactive waste. The ONSITE/MAXI1 package (June 1984 version) permits dose pathway analysis for scenarios of an intruder into the disposed waste. The integrated software package consists of the ONSITE and MAXI1 computer programs, a data base of dose factors and radioactive decay information, two auxiliary computer programs (MASI2 and MAXI3) that allow modification of the data base, and system procedures that both control program execution and reduce user interaction with the computer operating system. The interactive program ONSITE is used to generate input for MAXI1, which is then used to calculate the maximum annual dose to an exposed individual. Five representative human-intrusion scenarios are presented in the ONSITE program. The ONSITE program assists the user in defining a scenario by including default values for the various parameters, logically presenting applicable parameters for user modification in "English-phrased" statements, and testing user input against allowable parameter ranges. ONSITE-generated files may then be directly submitted for MAXI1 execution. Exposure pathways that can be evaluated in MAXI1 include direct external exposure to contaminated soil or building surfaces, inhalation of resuspended material, and ingestion of drinking water or terrestrial or aquatic foods.

NUREG/CR-3621: SAFETY SYSTEM STATUS MONITORING. LEWIS, J. R.; MORGENSTERN, M.; RIDEOUT, T. B.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1984. 129pp. 8403230200. PNL-4832. 22741:077.

The Pacific Northwest Laboratory has studied the safety aspects of monitoring the preoperational status of safety systems in nuclear power plants. The goals of the study were to assess for the NRC the effectiveness of current monitoring systems and procedures, to develop acceptance criteria by which the adequacy of safety status monitoring systems can be evaluated, to develop near-term guidelines for reducing human errors associated with monitoring safety system status, and to recommend a regulatory position on this issue. A review of safety system status monitoring practices indicated that current systems and procedures do not adequately aid control room operators in monitoring safety system status. This is true even of some systems and procedures installed to meet existing regulatory guidelines (Regulatory Guide 1.47). In consequence, this report suggests acceptance criteria for meeting the functional requirements of an adequate system for monitoring safety system status. Also suggested are near-term guidelines that could reduce the likelihood of human errors in specific, high-priority status monitoring tasks. It is our recommendation that 1) Regulatory Guide 1.47 be revised to address these acceptance criteria, and 2) the revised Regulatory Guide 1.47 be applied to all plants, including those built since the issuance of the original Regulatory Guide.

NUREG/CR-3622: A PROBABILISTIC MODEL OF ANNULAR-DISPERSED FLOW IN A REACTOR SUBCHANNEL AS SEEN BY CYLINDRICAL GEOMETRY IMPEDANCE PROBES. ALLGOOD, G. O.; ROBERTS, M. J. Oak Ridge National Laboratory. February 1984. 65pp. 8403230057. ORNL/TM-8841. 22742:300.

A probabilistic model of annular-dispersed flow in a reactor sub-channel as seen by cylindrical geometry impedance probes is developed from a finite element model of these electrodes in an inhomogeneous, isotropic medium of water and steam. The model is based on a derived finite difference equation for the potential at the center of a cube in terms of the potentials of the adjacent cubes and their material properties (Ampere's circuital law, low-frequency case).

The probabilistic model returns admittance (or capacitance) signals for two sets of probes based on specified film and two-phase void fractures. These signals will have temporal variations based on the flow velocities, the probe separation distance, and the frequency content of signals. The model assumes one-dimensional flow.

NUREG/CR-3623: STATUS REPORT: CORRELATION OF ELECTRICAL CABLE FAILURE WITH MECHANICAL DEGRADATION. STUETZER, O. Sandia Laboratories. April 1984. 90pp. 8406250272. SAND83-2622. 25139:151.

An attempt is being made to assess complete electrical failure of signal and low-power cables typically used in nuclear power plant containments and to correlate failure modes with the mechanical deterioration of the elastomeric cable material. Work over the past 24 months, although limited to one cable configuration, has identified creep shortout and insulator cracking, both aggravated by mechanical stresses, as the phenomena most likely to cause electrical breakdown. Comprehensive tests have been run for six months and are continuing. Preliminary conclusions can be drawn and are reported.

NUREG/CR-3624: A FORTRAN 77 PROGRAM AND USER'S GUIDE FOR THE GENERATION OF LATIN HYPERCUBE AND RANDOM SAMPLES FOR USE WITH COMPUTER MODELS. IMAN, R. L. ; SHORTENCARIER Sandia Laboratories. June 1984. 67pp. 8407110012. SAND83-2365. 25545:178.

This document has been designed for users of the computer program developed by the authors at Sandia National Laboratories for the generation of either Latin hypercube or random multivariate samples. The Latin hypercube technique employs a constrained sampling scheme, whereas random sampling corresponds to a simple Monte Carlo technique. The generation of these samples is based on information supplied to the program by the user describing the variables or parameters used as input to the computer model. The actual sampled values are used to form vectors of variables commonly used as input to computer models for purposes of sensitivity and uncertainty analysis studies. The present program replaces the previous Latin hypercube sampling program developed at Sandia National Laboratories (Iman, Davenport, and Zeigler, 1980). The present version is written using FORTRAN 77 and greatly extends the program while making the program portable and user friendly.

NUREG/CR-3625: REVIEW AND DISCUSSION OF THE DEVELOPMENT OF SYNTHETIC APERTURE FOCUSING TECHNIQUE FOR ULTRASONIC TESTING (SAFT-UT). BUSSE, L. J. ; COLLINS, H. D. ; DOCTOR, S. R. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1984. 112pp. 8404110296. PNL-4957. 22997:233.

The development and capabilities of synthetic aperture focusing techniques for ultrasonic testing (SAFT-UT) are presented. The purpose of SAFT-UT is to produce high-resolution images of the interior of opaque objects. The goal of this work is to develop and implement methods which can be used to detect and to quantify the extent of defects and cracks in critical components of nuclear reactors (pressure vessels, primary piping systems, and nozzles). This report places particular emphasis upon the practical experimental results that have been obtained using SAFT-UT as well as the theoretical background that underlies synthetic aperture focusing. A discussion regarding high-speed and "real-time" implementations of two- and three-dimensional synthetic aperture focusing is also presented.

NUREG/CR-3626 V01: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: Summary Description. SIEGEL, A. I. ; BARTTER, W. D. ; WOLF, J. J. ; et al. Oak Ridge National Laboratory. May 1984. 52pp. 8407060056. ORNL/TM-9041/V1. 25452:250.

A summary description is presented of the rationale for and the content and structure of the Maintenance Personnel Performance Simulation (MAPPS) model. The MAPPS model is a generalized stochastic computer simulation model developed to simulate the performance of maintenance personnel in nuclear power plants. The MAPPS model considers workplace, maintenance technician, motivation, human factors, and task-oriented variables to yield predictive information about the effects of these variables on successful maintenance personnel requirements. The model, which is drawn from a firm research analytic base, was examined for disqualifying defects from a number of viewpoints and its sensitivity was extensively tested. The MAPPS model is believed to be ready for initial and controlled applications which are in conformity with its purposes.

NUREG/CR-3626 V01: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: SUMMARY DESCRIPTION. SIEGEL, A. I.; KNEE, H. E.; HAAS, P. M.; et al. Oak Ridge National Laboratory. May 1984. 52pp. 8407060056. ORNL/TM-9041/V1. 25452:250.

A summary description is presented of the rationale for and the content and structure of the Maintenance Personnel Performance Simulation (MAPPS) model. The MAPPS model is a generalized stochastic computer simulation model developed to simulate the performance of maintenance personnel in nuclear power plants. The MAPPS model considers workplace, maintenance technician, motivation, human factors, and task-oriented variables to yield predictive information about the effects of these variables on successful maintenance personnel requirements. The model, which is drawn from a firm research analytic base, was examined for disqualifying defects from a number of viewpoints and its sensitivity was extensively tested. The MAPPS model is believed to be ready for initial and controlled applications which are in conformity with its purposes.

NUREG/CR-3627: FRANTIC II APPLICATIONS TO STANDBY SAFETY SYSTEMS. GINSBERG, T.; BOCCIO, J. L.; HALL, R. E. Brookhaven National Laboratory. June 1984. 151pp. 8406270111. BNL-NUREG-51738. 25174:001.

This report deals with practical applications of the FRANTIC II code in analyzing the reliability of standby safety systems. Time-dependent unavailability models such as FRANTIC II have two important advantages over more simplistic time-independent models: (1) accountability for the "burn-in" and "wear out" effects in describing component failure distribution; and (2) distinguishability between two systems having the same average unavailability, but with different periods of high risk. Thus, studies can be performed to assess the percentage of time the system spends with unavailability above a prescribed threshold level.

This report demonstrates the capability of FRANTIC II to evaluate the standby safety system unavailability on a more realistic basis and perform a detailed examination of period testing policies. Once the requisite input parameters to FRANTIC have been described and interpreted, and estimates made from the available data, the code is applied to the three systems: Emergency Feedwater System (PWR); Automatic Depressurization System (BWR); and High Pressure Coolant Injection System (BWR).

The analysis includes system description, fault tree quantification, unavailability calculation, and error propagation evaluation. Suggestions are also made on how to optimize gathering plant reliability data.

NUREG/CR-3628: PROBABILITY BASED SAFETY CHECKING OF NUCLEAR PLANT STRUCTURES. ELLINGWOOD, B. Brookhaven National Laboratory. * Commerce, Dept. of, National Bureau of Standards. May 1984. 73pp. 8405210585. BNL-NUREG-51737. 24535:181.

This report describes the basis for the development of practical probability-based design criteria for nuclear plant structures. A brief critical review of existing criteria is provided to highlight desirable features of probability-based-safety checking. A specific deterministic design criteria format is then recommended. Finally, the selection of a set of structures to test the validity of the probability-based checking equations is described. Statistical data on structural loads are summarized in an appendix.

NUREG/CR-3629: THE EFFECT OF THERMAL AND IRRADIATION AGING SIMULATION PROCEDURES ON POLYMER PROPERITIES. BUSTARD, L. D.; MINOR, E. E.; CHENION, J.; et al. Sandia Laboratories. May 1984. 81pp. 8405210589. SAND83-2651. 24532:257.

Prior to initiating a qualification test on safety-related equipment, the testing sequence for thermal and irradiation aging exposure must by chosen. Likewise, the temperature during irradiation must be selected. Typically, U.S. qualification efforts employ ambient temperature irradiation while French qualification efforts employ 70 degree C irradiations. For several polymer materials, the influence of the thermal and irradiation aging sequence has been investigated in preparation for Loss-Of-Coolant Accident simulated tests.

Ultimate tensile properties at completion of aging are presented for three XLPO and XLPE, five EPR and EPDM, two CSPE (HYPALON), one CPE, one VAMAC, one polydiallyphtalate, and one PPS material.

Bend test results at completion of aging are presented for two TEFZEL materials.

Permanent set after compression results are presented for three EPR, one VAMAC, one BUNA N, one Silicone, and one Viton material.

NUREG/CR-3630: EQUIPMENT QUALIFICATION METHODOLOGY RESEARCH: TESTS OF PRESSURE SWITCHES. SALAZAR, E. A. Sandia Laboratories. April 1984. 180pp. 8406210432. SAN083-2652. 25097:223.

Pressure switches, two each of five different models from two manufacturers, were tested in baseline evaluation tests typical of IEEE-323 (1974) suggested profiles as part of the NRC-sponsored Equipment Qualification Methodology Research Test Program (A-1355). The tests incorporated generic seismic and loss-of-coolant accident (LOCA) environments to assess the functional capabilities of unaged equipment. During the baseline evaluation tests, the seismic environment did not affect the functionality of the pressure switches, but the LOCA environment caused numerous functional failures and extensive physical damage in four of five models tested. As a result, eight other switches of the same make and model as those used in the baseline evaluation tests were tested in a follow-up test. In the follow-up test (a discrete-step pressure ramp LOCA environment) erratic functional behavior or complete failure was observed in all the equipment early in the test.

NUREG/CR-3631: RESPONSE TREES AND EXPERT SYSTEMS FOR NUCLEAR REACTOR OPERATORS. NELSON, W. R. EG&G, Inc. March 1984. 26pp. 8404170031. EGG-2293. 24091:314.

The United States Nuclear Regulatory Commission is sponsoring a Project performed by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL) to evaluate different display concepts for use in nuclear reactor control rooms. Included in this project is the evaluation of the response tree computer based decision aid and its associated displays. This report serves as an overview of the response tree methodology and how it has been implemented as a computer based decision aid utilizing color graphic displays. A qualitative assessment of the applicability of the response tree aid is generalized to address a larger category of computer aids, those known as knowledge based expert systems. General characteristics of expert systems are discussed, as well as examples of their application in other domains. A survey of ongoing work on expert systems in the nuclear industry is presented and an assessment of their potential applicability is made. Finally, recommendations for the design and evaluation of computer based decision aids are presented.

NUREG/CR-3632: METHODS FOR IMPLEMENTING REVISIONS TO EMERGENCY OPERATING PROCEDURES. MYERS, L. B.; BELL, A. J. Battelle Memorial Institute, Columbus Laboratories. * Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 38pp. 8405210600. PNL-4927. 24534:200.

In response to the Three Mile Island (TMI) accident, the U.S. Nuclear Regulatory Commission (NRC) has published the TMI Action Plan. The TMI Action Plan Item I.C.1 called for the upgrading of Emergency Operating Procedures (EOPs) at nuclear power plants. The program developed from this Action Plan item has resulted in utility efforts to 1) revise EOPs, 2) train personnel in the use of the EOPs, and 3) implement the revised EOPs.

The NRC supported the study presented in this report to identify factors which influence the effectiveness of training and implementation of revised EOPs. The NRC's major concern was the possible effects of negative transfer of training. The report includes a summary of existing methods for implementing revisions to procedures based on interviews of plant personnel, a review of the training literature applicable to the effect of previously learned procedures on the learning of and performance with revised procedures (i.e., negative transfer) and recommendations of methods and schedules for implementing revised EOPs. While the study found that the concern over negative transfer of training was not as great as anticipated, several recommendations were made. These include (1) overtraining of operators to reduce the effect of observed negative transfer, and (2) implementation of the revised EOPs as soon as possible after training to minimize the time operators must rely upon the old EOPs after having been trained on the revised EOPs. The results of the study should be useful both to the utilities and the NRC in the development and review of EOP implenentation programs.

NUREG/CR-3633 V01: TRAC-BD1/MOD1: AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS. Volume 1: Model Description. TAYLOR, D. D.; MOHR, C. M. EG&G, Inc. April 1984. 231pp. 8405210576. EGG-2294. 24528:001.

The TRAC-BD1/MOD1 computer program provides a best-estimate analysis capability for the analysis of the full range of postulated accidents in Boiling Water Reactor (BWR) systems and related experimental facilities. The program is described in four volumes: Volume 1, Code Description; Volume 2, User's Guide; Volume 3, Code Structure and Programming Information; and Volume 4, Developmental Assessment. Volume 1 describes the thermal-hydraulic models, numerical methods, and component models available. Volume 2 describes the input and output of the TRAC-BD1/MOD1 code and provides guidelines for use of the code modeling of BWR systems. Volume 3 is designed for the programmer or model developer who needs to implement or modify the TRAC-BD1/MOD1 program. Volume 4 discusses the results of the development assessment calculations performed with TRAC-BD1/MOD1.

NUREG/CR-3633 V02: TRAC-BD1/MOD1: AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS. Volume 2: Users Guide. SCHUMWAY, R. W.; MOHR, C. M. EG&G, Inc. April 1984. 117pp. 8405210578. EGG-2294. 24528:232.

See NUREG/CR-3633, V01 abstract.

NUREG/CR-3633 V03: TRAC-BD1/MOD1: AN ADVANCED BEST ESTIMATE COMPUTER PROGRAM FOR BOILING WATER REACTOR TRANSIENT ANALYSIS. Volume 3: Code Structure and Programming Information. SINGER, G. L. ; MOHR, C. M. EG&G, Inc. April 1984. 110pp. 8405210579. EGG-2294. 24528:357.

See NUREG/CR-3633, V01 abstract.

NUREG/CR-3636: BENCHMARK PROBLEMS FOR REPOSITORY DESIGN MODELS. WART, R. J. ; SKIBA, E. L. ; CURTIS, R. H. ; et al. Acres American, Inc. February 1984. 191pp. 8402220358. 22364:145.

This report describes benchmark problems to test computer codes used in design of nuclear waste repositories. Problems with analytical solutions, hypothetical repository design problems, and problems simulating field experiments are used. Types of problems include: thermal conduction, geomechanical stress and coupled stress, ground water flow, and temperature problems. Specific phenomena addressed are thermal conduction, propagation, thermal expansion, and consolidation.

NUREG/CR-3637: THE APPLICATION OF STEIN AND RELATED PARAMETRIC EMPIRICAL BAYES ESTIMATORS TO THE NUCLEAR PLANT RELIABILITY DATA SYSTEM. HILL, J. R. ; HEGER, A. S. ; KOEN, B. V. ; et al. Texas, Univ. of, Austin, TX. April 1984. 42pp. 8405220179. EGG-2295. 24594:269.

This report is the result of a preliminary feasibility study of the applicability of Stein and related parametric empirical Bayes (PEB) estimators to the Nuclear Plant Reliability Data System (NPRDS). A new estimator is derived for the means of several independent Poisson distributions with different sampling times. This estimator is applied to data from NPRDS in an attempt to improve failure rate estimation. Theoretical and Monte Carlo results indicate that the new PEB estimator can perform significantly better than the standard maximum likelihood estimator if the estimation of the individual means can be combined through the loss function or through a parametric class of prior distributions.

NUREG/CR-3639: LARGE BREAK LOCA ANALYSES FOR TWO-LOOP PWRS WITH UPPER-PLENUM INJECTION. DOBRANICH, D. ; BUXTON, L. D. Sandia Laboratories. May 1984. 67pp. 8406040023. SAND84-0040. 24805:001.

A series of best-estimate thermal-hydraulic calculations was performed using TRAC-PF1 to simulate a hypothetical loss-of-coolant accident in Westinghouse two-loop pressurized water reactors. Those reactors are equipped for low-pressure injection of emergency coolant directly into the upper plenum of the reactor vessel. This type of injection is referred to as upper plenum injection (UPI). The calculations were performed to evaluate the effectiveness of UPI compared to injection into the vessel downcomer, referred to as downcomer injection (DI).

The TRAC results indicated that some channeling of upper plenum injected liquid down the core periphery occurred; however, a large percentage of that liquid was vaporized as it drained toward the lower plenum. This vaporization degraded the bottom-flood quench front compared to that seen in TRAC calculations in which downcomer injection was assumed. For the case of upper plenum injection, counter-current flow limiting conditions at the upper core support plate led to formation of a large subcooled liquid pool in the upper plenum; part of this subcooled liquid was entrained into the hot legs and steam generators. Only a small saturated liquid pool formed in the case of downconer injection. Overall, the calculations show that higher peak clad temperatures are produced when the low-pressure injection is into the upper plenum instead of the vessel downcomer.

NUREG/CR-3641: RELIABILITY ASSESSMENT OF INDIAN POINT UNIT 3 CONTAINMENT STRUCTURE. KAWAKAMI, J.; HWANG, H.; CHANG, M. T.; et al. Brookhaven National Laboratory. May 1984. 51pp. 8405310082. BNL-NUREG-51740. 24737:019.

In the current design criteria, the load combinations specified or design of concrete containment structures are in the deterministic formats. However, by applying the probability-based reliability method developed by BNL to the concrete containment structures designed according to the criteria, it is possible to evaluate the reliability levels implied in the current design criteria. For this purpose, the reliability analysis is applied to the Indian Point Unit No. 3 containment.

The details of the containment structure such as the geometries and the rebar arrangements, etc., are taken from the working drawings and the final safety analysis reports. Three kinds of loads are considered in the reliability analysis. They are, dead load accidental pressure due to a large LOCA (P), and earthquake ground acceleration (E). Reliability analysis of the containment subjected to all combinations of loads is performed. The results are presented in this report.

NUREG/CR-3642: A COBRA/TRAC BEST-ESTIMATE ANALYSIS OF A LARGE-BREAK ACCIDENT IN A PWR EQUIPPED WITH UPPER HEAD INJECTION. GUIDOTTI, T. E.; THURGOOD, M. J. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1984. 243pp. 8404100521. PNL-4971. 22977:039.

This report documents the simulation of a double-ended (200 percent), cold leg break, loss-of-coolant accident in a PWR equipped with an upper head injection system. The simulation was performed using the COBRA/TRAC thermal-hydraulic computer program developed at the Pacific Northwest Laboratories for the Nuclear Regulatory Commission to analyze PWR's with the upper head injection system. This analysis used best-estimate assumptions and a 556 cell multidimensional mesh in the vessel. Each of the four primary loops were modeled. Four cooling periods were predicted prior to the beginning of bottom reflood. The first cooling period was caused by the flashing of liquid in the lower plenum while the other three cooling periods were related to the hydrodynamic behavior in the upper head and the delivery of upper head cooling water to the core. The entire core was quenched during the first period of upper head water delivery to the core. The peak clad temperature during the transient was 1,155 degrees fahrenheit and occurred 8 sec after the initiation of blowdown. The peak temperature remained below 600 degrees fahrenheit for the remainder of the transient. The transient behavior of the reactor coolant system is presented in the form of plots of the key thermal hydraulic variables as a function of time. Major phenomena calculated during the transient (e.g., multidimensional effects, counter-current flow limiting, ECC bypass, etc.) are discussed in detail. These results are compared with a Westinghouse SATAN calculation.

NUREG/CR-3643: HETEROGENEOUS OXIDATIVE DEGRADATION IN IRRADIATED POLYMERS. CLOUGH, R. L.; GILLEN, K. T.; QUINTANA, C. A. Sandia

Laboratories. July 1984. 43pp. 8408080359. SAND83-2493. 25978:153.

When polymeric materials are irradiated in the presence of air, oxygen-diffusion effects can, depending upon dose rate, lead to oxidative degradation which occurs only near the edges. This report describes the use of several recently developed techniques which are of general use for studying heterogeneous degradation in commercial polymeric materials. The techniques discussed are: optical evaluation of cross-sectioned, polished samples; cross-sectional profiling of changes in relative hardness; and profiling of density changes. Oxidation penetration depths are given for a number of major polymer types as a function of dose rate. A detailed example is given graphically illustrating the effects of differing oxidative penetration depths on the radiation-degradation behavior of a Viton O-ring material; this particular material becomes hard and brittle when irradiated at high dose rate, but soft and stretchable when irradiated at low dose rates.

NUREG/CR-3644: REVIEW OF PROPOSED FAILURE CRITERIA FOR DUCTILE MATERIALS. JU,F.D.; BUTLER,T.A. Los Alamos Scientific Laboratory. April 1984. 36pp. 8405220015. LA-10007-MS. 24554:308.

In this report, failure criteria for structural components constituting the primary coolant-system boundary of a Liquid Metal Fast Breeder Reactor (LMFBR) are reviewed. Because the materials being considered, mild ferritic steel and austenitic stainless steels, are ductile, especially under LMFBR normal operating and accident conditions, only ductile criteria are considered. The ductile criteria must be used in combination with true stress and strain measures of deformation and internal load. Specific criteria reviewed include maximum stress and strain or plastic instability based on uniaxial tensile-test data and a hole-growth theory based on coalescence of neighboring voids under load. Criteria based only on maximum stress or strain are not recommended for general use because they are not appropriate under general multiaxial stress conditions. The plastic instability criterion, because it leaves a large unused toughness region before fracture, is recommended where considerable conservatism is warranted. The hole-growth criterion is recognized as being analytically sound; however, it has not been extended to general three-dimensional geometry and multiaxial stress conditions. The theory needs to be substantiated with experimental data for specific materials being considered.

NUREG/CR-3645: A GUIDE TO LITERATURE RELEVANT TO THE ORGANIZATION AND ADMINISTRATION OF NUCLEAR POWER PLANTS. SOMMERS,P. Battelle Human Affairs Research Centers. * Battelle Memorial Institute, Pacific Northwest Laboratories. February 1984. 656pp. 8403190055. PNL-4906. 22678:001.

The purpose of this report is to assist applicants for a nuclear power plant operating license through a structured review of the organization and administration literature. The Nuclear Regulatory Commission reviews an applicant's proposed organization and administration as part of the operating license review. NRC is developing draft Guidelines for a Utility Organization Plan and a proposed Workbook for Assessment of Organization and Administration for reviewing responses to the guidelines. It is the intention of the NRC to incorporate these documents when completed into Chapter 13 of the Standard Review Plan NUREG-0800. This report assists users of the Guidelines and Workbook by providing a guide to the academic literatures relevant to the concepts used in these two regulatory documents. Persons preparing responses to the Guidelines or reviewing these responses can locate literature relevant to a particular topic discussed in the Guidelines or Workbook through use of this report.

This report is organized as follows. The Introduction describes the purpose and scope of the review. Details of the search processes used to identify relevant literature are also covered. The next section documents government concern with nuclear plant organization and administration issues through a review of major reports and Congressional hearings. The third and fourth sections contain abstracts of selected items and a lengthy bibliography of the relevant items identified in our literature searches. The final section of the report is a keyword index which allows readers to identify relevant items in the abstract and bibliography sections through use of keywords. The keywords include the major concepts used in the Guidelines and Workbook to facilitate location of relevant items in the literature.

NUREG/CR-3649: DYNAMIC TESTING OF AS-BUILT CIVIL ENGINEERING STRUCTURES -A REVIEW AND EVALUATION. SRINIVASAN, M. G.; KOT, C. A.; HSIEH, B. J. Argonne National Laboratory. January 1984. 117pp. 8403070114. ANL-83-20. 22562:055.

The experience with dynamic testing of as-built large civil engineering structures other than nuclear power plant buildings is evaluated. A review of literature on the dynamic testing of a large number of structures formed the basis for this evaluation. Methods of excitation and data analysis for determining dynamic parameters from measured response are evaluated. Tests of as-built structures have enabled a partial verification of linear models, have revealed the inadequacies of some assumptions made in conventional analytical methods and have been the major source of data on damping. Tests before and after have been used to obtain a measure of the change in the integrity of a structure subjected to strong excitation. Tests on some structures at various excitation levels coupled with the development of analytical methods for determining nonlinear models from such tests is needed before testing can be used as a means of predicting response to strong excitation. However, recent improvements in experimental and analytical techniques associated with low-level testing of as-built structures enhance its utility for model verification and improvement and for the investigation of phenomena such as damping associated with soil-structure interaction.

NUREG/CR-3650: A STATISTICAL ANALYSIS OF NUCLEAR POWER PLANT PUMP FAILURE RATE VARIABILITY - Some Preliminary Results. MARTZ, H. F.; WHITEMAN, D. E. Los Alamos Scientific Laboratory. April 1984. 55pp. 8405220072. LA-10014-MS. 24557:315.

In-Plant Reliability Data System (IPRDS) pump failure data on over 60 selected pumps in four nuclear power plants are statistically analyzed using the Failure Rate Analysis Code (FRAC). A major purpose of the analysis is to determine which environmental, system, and operating factors adequately explain the variability in the failure data. Catastrophic, degraded, and incipient failure severity categories are considered for both demand-related and time-dependent failures.

For catastrophic demand-related pump failures, the variability is explained by the following factors listed in their order of importance: system application, pump driver, operating mode, reactor type, pump type, and unidentified plant-specific influences. Quantitative failure rate adjustments are provided for the effects of these factors.

In the case of catastrophic time-dependent pump failures, the failure rate variability is explained by three factors: reactor type, pump driver, and unidentified plant-specific influences.

Finally, point and confidence interval failure rate estimates are provided for each selected pump by considering the influential factors. Both types of estimates represent an improvement over the estimates computed exclusively from the data on each pump.

NUREG/CR-3652: EVALUATION OF INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING IN BOILING WATER REACTORS. LEWIN, T. Oak Ridge National Laboratory. April 1984. 31pp. 8405210602. ORNL/TM-9029. 24526:278.

This report is a review of the APPROACH TO INADEQUATE CORE COOLING issue in Boiling Water Reactors. The report consists of seven sections. The principal conclusion is that the condition of the reference leg, and operator awareness of that condition are of primary importance in level indication reliability for safety. An indication of reference leg level and temperature displayed to the operators would be a useful enhancement of reliability and a guide to further operator action in all circumstances. We conclude that the BWR practice of multiple, redundant coolant level measurements, with overlapping ranges, can be a reliable basis for indication of approach to an ICC condition, and, in correlation with the other control and safety systems of modern BWRs, will prevent unsafe conditions.

NUREG/CR-3653: CONTAINMENT ANALYSIS TECHNIQUES. A State-Of-The-Art Summary. GREIMANN, L.; FANOUS, F.; BLUHM, D. Ames Laboratory, Energy & Mineral Resources Research Institute. April 1984. 167pp. 8406210097. SAND83-7463. 25097:055.

The purpose of the work contained herein is to review the state-of-the-art for the analysis of LWR nuclear containments with uniform internal pressure. This includes:

(a) A review of calculated static failure pressure of various containments.
(b) A review of the different failure criteria used for predicting containment failure.
(c) Comments on possible uncertainties associated with analysis techniques, material and geometric models, and other analysis features.

A state-of-the-art containment analysis is a finite element solution of an axisymmetric model. Material and geometric nonlinearities are included. Nonsymmetric features may be analyzed on an individual basis but are omitted in the axisymmetric model. State-of-the-art models of the material constitutive relationships are used. Deformation predictions are generally regarded as reliable, assuming the containment configuration is accurately described, e.g., known geometry, material and loads. Predictions of leakage are much more uncertain. There is no general agreement on when and where leakage will occur.

NUREG/CR-3654: PWR FLECHT SEASET SYSTEMS EFFECTS NATURAL CIRCULATION AND REFLUX CONDENSATION. Data Evaluation and Analysis Report. NRC/EPRI/Westinghouse Report No. 14. HOCHREITER, L. E.; RUPPRECHT, S. D.; DEDERER, J. T.; et al. Westinghouse Electric Corp. September 1984.

584pp. 8410100382. EPRI NP-3497. 26899:001.

A series of natural circulation tests were conducted at a FLECHT SEASET facility which is scaled 1/307 by volume to a full size PWR. The purpose of these tests was to identify hydraulic and heat transfer phenomena during natural circulation cooling modes. The resulting data, evaluation, and analysis are to be used for PWR codes and model assessments as well as to provide a comparison to similar experiments in other scaled systems. Steady-state single-phase, two-phase, and reflux condensation modes of natural circulation cooling were established in the FLECHT SEASET systems effects facility and the flow and heat transfer characteristics of the different cooling modes were identified. This report presents the test data; data reduction, analysis, and evaluation; and resulting model development and analysis. The models which have been developed include a reflux tube condensation model as well as a single- and two-phase model for the overall system.

NUREG/CR-3655: A METHOD FOR ANALYTICAL EVALUATION OF COMPUTER-BASED DECISION AIDS. ROUSE, W. B.; FREY, P. R.; et al. Search Technology, Inc. KISNER, R. A. Oak Ridge National Laboratory. July 1984. 202pp. 8409110215. ORNL/TM-9068. 26445:001.

This report presents a proposed methodology that involves a two-stage process of classification and analytical evaluation of decision aids for nuclear power plant operators. The classification scheme relates any particular aid to one or more general decision-making tasks. Evaluation proceeds using a normative top-down design process based on the classification scheme and involves determining how various design issues associated with this process were resolved by the designer. The result is an assessment of the "understandability" of the aid as well as the identification of training and display requirements necessary to ensure understandability. The methodology is illustrated by applying it to the evaluation of an aid designed to support operators in recovery of critical safety functions at a pressurized-water reactor.

Two appendices are included. Appendix A contains information collected from manufacturers, developers, and users of operational aid systems. Appendix B is a review of NRC documents and guidelines that might apply to operational aids.

NUREG/CR-3658: CONSIDERATIONS RELEVANT TO THE DRY STORAGE OF LWR FUEL RODS CONTAINING WATER. WOODLEY, R. E. Hanford Engineering Development Laboratory. June 1984. 35pp. 8407190060. HEDL-TME 84-14. 25693:247.

The performance under dry storage conditions of LWR fuel rods containing water was analyzed to determine if radionuclide containment by the fuel rod cladding would be adversely affected. Fuel rod and storage canister pressurization as well as cladding and fuel oxidation were examined using "worst case" conditions. The results of this study are presented.

NUREG/CR-3660 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS. Volume 2: Pipe Failure Induced By Crack Growth. WOO, H. H.; MENSING, R. W.; BENDA, B. J. Lawrence Livermore National Laboratory. August 1984. 71pp. 8409200445. UCID-19988. 26628:132.

This report assesses the probability of reactor coolant loop (RCL) piping failures resulting from a crack growth mechanism. The

Westinghouse pressurized water reactor (PWR) plants in the United States east of the Rocky Mountains are considered. After the introduction (Section 1), the assessment is presented in five parts (Sections 2-6). Section 2 describes the characteristics of RCL piping in these Westinghouse PWR plants. Section 3 describes the methodology used in the analysis. Sections 4 and 5 present the best-estimate and uncertainty analyses, respectively. Our conclusions are presented in Section 6, along with recommended items for consideration in future licensing regulations.

NUREG/CR-3661: PROTOTYPICAL STEAM GENERATOR (MB-2) TRANSIENT TESTING PROGRAM. Task Plan/Scaling Analysis Report. YOUNG, M.; TAKEUCHI, K.; MENDLER, J.; et al. Westinghouse Electric Corp. March 1984. 198pp. 8403230208. EPRI NP-3494. 22743:096.

This report describes the Westinghouse MB-2 model boiler test facility and the test program currently planned (with Westinghouse/EPRI/NRC funding) to investigate various types of possible accidents which might occur in a PWR steam generator. The planned tests will simulate loss of feedwater (LOF) transients, various steam generator tube rupture (SGTR) scenarios, and steamline breaks (SLB).

The facility will be extensively modified to allow measurement of local wall and fluid temperatures, and to measure possible moisture carryover during the SLB and SGTR tests.

This report is divided into six sections. The first three sections describe the facility and the new components and instrumentation to be installed. The next section is a detailed scaling analysis of MB-2. Section 5 describes the analysis of the data which is planned.

NUREG/CR-3662: FUEL-DISRUPTION EXPERIMENTS UNDER HIGH-RAMP-RATE HEATING CONDITIONS. WRIGHT, S.A.; WORLEDGE, D.H.; CANO, G.L.; et al. Sandia Laboratories. August 1984. 86pp. 8409280084. SAND81-0413. 26764:340.

This topical report presents the preliminary results and analysis of the High Ramp Rate fuel-disruption experiment series. These experiments were performed in the Annular Core Research Reactor at Sandia National Laboratories to investigate the timing and mode of fuel disruption during the prompt-burst phase of a loss-of-flow accident. High-speed cinematography was used to observe the timing and mode of the fuel disruption in a stack of five fuel pellets. Of the four experiments discussed, one used fresh mixed-oxide fuel, and three used irradiated mixed-oxide fuel.

Analysis of the experiments indicates that in all cases, the observed disruption occurred well before fuel-vapor pressure was high enough to cause the disruption. The disruption appeared as a rapid spray-like expansion and occurred near the onset of fuel melting in the irradiated-fuel experiments and near the time of complete fuel melting in the fresh-fuel experiment. This early occurrence of fuel disruption is significant because it can potentially lower the work-energy release resulting from a prompt-burst disassembly accident.

NUREG/CR-3663 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS. Vol 2: Pipe Failure Induced by Crack Growth. LO, T.Y.; MENSING, R.W.; WOO, H.H.; et al. Lawrence Livermore National Laboratory. September 1984. 94PP. 8410120004.

UCRL-53500. 26985:17/.

The U.S. Nuclear Regulatory Commission (NRC) contracted with the Lawrence Livermore National Laboratory (LLNL) to conduct a study to determine if the probability of occurrence of a double-ended guillotine break (DEGB) in the primary coolant piping warrants the current design requirements that safeguards against the effect of DEGB. This report describes the results of an assessment of reactor coolant loop piping systems designed by Combustion Engineering, Inc. A probabilistic fracture mechanics approach was used to estimate the crack growth and to assess the crack stability in the piping throughout the lifetime of the plant. The results of the assessment indicate that the probability of occurrence of DEGB due to crack growth and instability is extremely small, which supports the argument that the postulation of DEGB in design should be eliminated and replaced with more reasonable criteria.

NUREG/CR-3664: A DESCRIPTION AND ASSESSMENT OF RAMONA-3B MOD.0 CYCLE 4: A COMPUTER CODE WITH THREE-DIMENSIONAL NEUTRON KINETICS FOR BWR SYSTEM TRANSIENTS. WULFF, W.; CHENG, H.S.; DIAMOND, D.J.; et al. Brookhaven National Laboratory. May 1984. 428pp. 8405210615. BNL-NUREG-51746. 24533:001.

This report documents the physical models and the numerical methods employed in the BWR systems code RAMONA-3B. The RAMONA-3B code simulates three-dimensional neutron kinetics and multichannel core hydraulics of nonhomogeneous, nonequilibrium two-phase flows. RAMONA-3B is programmed to calculate the steady and transient conditions in the main steam supply for normal and abnormal operational transients, including the performances of plant control and protection systems.

Presented are code capabilities and limitations, models and solution techniques, the results of developmental code assessment and suggestions for improving the code in the future.

NUREG/CR-3665: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS. Executive Summary. COHEN, J.J. Science Applications, Inc. September 1984. 7pp. 8410100787. SAIC-84/1317. 26903:248.

An area of growing concern in recent years has been the apparent increase in levels of collective radiation dose to workers at nuclear power plants in the USA. U.S. Nuclear Regulatory Commission (NRC) decisions and rulings related to in-service inspection, retrofits, and plant upgrades have been primarily intended to reduce the risk of public radiation exposure resulting from either routine release of radioactivity or potential accident situations. However, implementation of the required control measures and procedures can often result in increased levels of occupational radiation exposure. Recognizing the need to incorporate occupational dose into probabilistic risk assessments (PRA), value-impact, and cost-benefit analyses, the NRC has sponsored this study with the objective of developing an appropriate methodology to factor potential worker exposures into safety assessments. This report on the study is presented in three volumes. The following are subtitles for Volumes 1-3: Volume 1, "A Review of Occupational Dose Assessment Considerations in Current Probabilistic Risk Assessments and Cost-Benefit Analyses," Volume 2, " Considerations in Factoring Occupational Dose Into Value-Impact and Cost-Benefit Analyses," and Volume 3, " A Calculation Method."

NUREG/CR-3665 V01: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS. A Review Of Occupational Dose Assessment Considerations In Current Probabilistic Risk Assessment And Cost-Benefit Analyses. LOBNER, P. Science Applications, Inc. September 1984. 55pp. 8410100656. SAI-83/1125. 26900:232.

This report reviews current value-impact analysis and probabilistic risk assessment methods and discusses the manner and degree to which these methods consider occupational radiation exposure that may form a variety of in-plant activities, including: (a) normal operation and maintenance, (b) repair, (c) retrofit, (d) minor incidents and cleanup, (e) major accidents, and (f) decommissioning. Value-impact analysis methods which include occupational exposure as an element of the value-impact equation have been developed, however, no standard approach to such analysis has been adopted. Comparison of the results of value-impact analysis must, therefore, be done with caution because different value-laden assumptions made by the analyst can have strong effects on the outcome. Such assumptions include the monetray equivalent of a person-rem, and the relative value of occupational and public exposure. Probabilistic methods have been used in value-impact evaluations to quantify incremental or averted occupational exposure from reactor accidents, however, occupational exposure has not been addressed in probabilistic risk assessments (PRAs) of nuclear power plants. Consideration of occupational exposure in a PRA would greatly increase the complexity of the plant model and the benefits from such an analysis are uncertain. In lieu of expanding the scope of PRAs to address occupational risk, the separate, limited-scope probabilistic evaluations developed for value-impact analysis should provide a more practical analytical capability to support the evaluation and optimization of occupational and public radiation exposure.

NUREG/CR-3665 V02: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS. Considerations In Factoring Occupational Dose Into Value-Impact And Cost-Benefit Analyses. COHEN, J. J. Science Applications, Inc. September 1984. 46pp. 8410100785. SAI-84/1010 V02. 26903:200.

Many NRC decisions intended for the improvement of public health and safety involve concomitant increases in occupational radiation exposure. Previous study (Volume 1) indicates that occupational dose consequences generally have not been considered in cost-benefit and value-impact analyses supporting decisions related to public safety. Such consideration, however, would be consistent with ALARA guidance. This study derives a methodology for factoring occupational vs. public radiation exposure, stochastic vs. non-stochastic effects, probabilistic risk considerations, uncertainty, and de minimus levels.

NUREG/CR-3665 V03: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS. A Calculation Method. HORTON, W. H. Science Applications, Inc. September 1984. 87pp. 8410100755. SAI-84/3037 V03. 26902:001.

The methodology presented in this report formulates an approach for the optimization of benefits resulting from NRC decision making processes. Recent increases in occupational exposures in nuclear power plants resulting from NRC regulatory practices have led to the questioning by NRC of the overall benefit of specific regulations. The optimization methodology in this report provides a tool for the determination of the cost-benefit of proposed NRC regulations. Detailed methods are presented for the modeling of plant safety systems undergoing inspection, testing, and/or repair. This methodology utilizes dynamic Markov modeling techniques with extensive additional model development associated with operator errors involved in the inspection, test, and repair activities of the plant. Closed form solutions to the Markov models are provided. The report appendix presents the Markov model solution process in detail sufficient for model verification. Other methods necessary for the optimization process are discussed in lesser detail. An application of the methodology dealing with steam generator inspection frequency and steam generator tube rupture events is presented. The example determines the steam generator inspection intervals which minimize expected costs and total expected occupational and public dose.

NUREG/CR-3666: ASSESSMENT OF THE IMPLICATIONS OF CONVERSION OF UNIVERSITY RESEARCH AND TRAINING REACTORS TO LOW ENRICHMENT URANIUM FUEL. HARRIS,D.R.; BURN,R.R.; CLARK,L.; et al. Rensselaer Polytechnic Institute, Troy, NY. February 1984. 165pp. 8403190437. 22680:001.

The impacts of a proposed rule requiring conversion of source or all NRC-Licensed training and research reactors to low enrichment uranium fuel were determined. Consideration focused on: technical feasibility, functional impacts, licensing, and schedule. It was determined that the cost of conversion would range from $1,000,000 to $12,000,000, depending on rule criteria.

NUREG/CR-3667: A GUIDE FOR REVIEWING ESTIMATES OF PRODUCTION-COST INCREASES THAT RESULT FROM NUCLEAR POWER PLANT OUTAGES. PEERENBOOM,J.P.; BUEHRING,W.A. Argonne National Laboratory. March 1984. 66pp. 8404050473. ANL/EES-TM-241. 22912:023.

Shutdowns of nuclear power plants typically result in significant increases in operating costs (production costs) for the affected utilities. This report presents a framework that will help users evaluate the reasonableness of estimated production-cost changes resulting from reactor shutdowns. The framework consists of three basic steps: (1) preliminary evaluation and classification of the outage, (2) evaluation of the input data and assumptions, and (3) evaluation of production-cost results. Several simplified procedures for estimating changes in production costs are presented.

NUREG/CR-3669: PLUTONIUM RECYCLE TEST REACTOR (PRTR) ACCIDENT:A FINAL REPORT ON THE INVESTIGATION OF FISSION PRODUCT CHEMICAL FORMS. HENSLEY,W.K.; ROGERS,L.A. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 48pp. 8405020442. PNL-5003. 24298:075.

In September of 1965, an intentionally defective fuel rod failed in the Plutonium Recycle Test Reactor (PRTR), causing the rupture of the surrounding pressure tube and the release of superheated cooling water into a region of the reactor core. The Pacific Northwest Laboratory (PNL) has reviewed the PRTR incident to assemble and update all the available information regarding the incident. A principal goal of the review was to analyze any remaining clues that may indicate the stoichiometry or most probable chemical and physical forms of the released fission products. The review confirmed the role of water in limiting iodine release. About 97% of the iodine released during the accident was subsequently found in tanks containing the reactor/rupture-loop coolant. Although the chemical form of the released radioiodine cannot be stated unambiguously, the available evidence suggests that it was released in the form of cesium iodide. Most of the remaining 3% was found in the condensate collected from air cooling systems. The chemical form of this scrubbed iodine remains undefined.

NUREG/CR-3670: VIOLENT TORNADO CLIMATOGRAPHY, 1880-1982. GRAZULIS, T. P. Environmental Films, Inc. * Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 188pp. 8406040071. PNL-5006. 24806:001.

All known information sources, ranging from newspaper reports to the University of Chicago (DAPPL) and National Oceanic and Atmospheric Administration (NOAA/NSSFC) data lists, were utilized to produce a self-consistent compilation and description of violent tornado occurrences in the contiguous United States for the years 1880 through 1982. The 969 F-scale 4 and 5 tornadoes comprise the most complete and rational data base available for studies related to violent tornado risk assessment; the data provide improved bases for licensing decisions and development of standards in safety at nuclear facility sites. Reconciliation of the DAPPL and NSSFC data lists for violent tornadoes has been achieved. Analysis of the data shows geographical and temporal variability of tornado occurrences; suggestions are given to help account for the nonuniform distributions, and other suggestions are made for needed future research.

NUREG/CR-3671: ASSESSMENT OF RADIATION EFFECTS RELATING TO REACTOR PRESSURE VESSEL CLADDING. CORWIN, W. R. Oak Ridge National Laboratory. July 1984. 70pp. 8408080419. ORNL-6047. 25977:271.

Because the weld overlay cladding on the interior of light-water reactor pressure vessels was applied for corrosion resistance and not for structure, little attention has been given to the potential of mechanical property degradation due to radiation exposure. In light of the concerns recently raised regarding overcooling transients in nuclear power reactors, it has been suggested that any such degradation could adversely affect the serviceability and/or integrity of the vessel.

A literature survey assesses the current knowledge regarding the effects of neutron radiation on the mechanical fracture properties of stainless steel weld overlay cladding under conditions relevant to light-water reactor operation. In particular, effects on the material's microstructure and tensile, fatigue, impact, and fracture properties are examined. Although information is lacking on the specific materials under the exact irradiation conditions of interest, a wealth of information is available on irradiated stainless steel weldments in general, from which basic behavioral trends can be obtained.

Some irradiation embrittlement apparently does occur in stainless steel weldments at the relatively low temperatures and fluences typical of light-water reactors. Tensile strength increases and ductility decreases. Low-cycle fatigue behavior is degraded somewhat, but high-cycle fatigue and fatigue crack growth seem largely unaffected.

Effects of ferrite on fracture resistance are small in both irradiated and unirradiated materials. Notch impact and fracture toughness are both reduced by irradiation, and a dependence of toughness on testing rate, not seen in wrought material, is indicated.

NUREG/CR-3672: EXAMINATION OF THE SIZE EFFECTS AND DATA SCATTER OBSERVED IN SMALL SPECIMEN CLEAVAGE FRACTURE TOUGHNESS TESTING. MERKLE, J. G. Oak Ridge National Laboratory. April 1984. 87pp. 8405220026. ORNL/TM-9088. 24565:013.

In the transition range of temperature, the cleavage fracture toughness of steel rises steeply with temperature, often necessitating the use of elastic-plastic methods for calculating toughness values with small specimens. This usually leads to size effects, whereby measured toughness values increase with decreasing specimen size and data scatter becomes large. Existing literature pertaining to the physical aspects of the onset of cleavage fracture and the occurrence of size effects is examined, and it is concluded that the primary cause of size effects is loss of triaxial contstraint due to crack tip yielding and transverse contraction along the crack front. The implications of an existing semiemperical equation, known as the Irwin B(Ic) equation, for removing size effects from small specimen cleavage fracture toughness data are examined by developing the equation based on reasonable assumptions including the conditions specified by ASTM E399 for valid K(Ic) testing. The applicability of the Irwin B(Ic) adjustment equation to pressure vessel steels is evaluated by applying it to several sets of small specimen fracture toughness data, and it is found that the equation consistently eliminates apparent size effects and significantly reduces data scatter. The B(Ic) adjustment appears to be applicable to dynamic as well as to static initiation toughness data, but only to the cleavage fracture toughness and not to the ductile tearing resistance.

NUREG/CR-3673: ECONOMIC RISKS OF NUCLEAR POWER REACTOR ACCIDENTS. BURKE, R. P.; ALDRICH, D. C.; RASMUSSEN, N. C. Sandia Laboratories. May 1984. 250pp. 8406230081. SAND84-0178. 25131:005.

Models to be used for analyses of economic risks from events which occur during U.S. LWR plant operation are developed in this study. The models include capabilities to estimate both onsite and offsite costs of LWR events ranging from routine plant forced outages to severe core-melt accidents resulting in large releases of radioactive material to the environment. The models have been developed for potential use by both the nuclear power industry and regulatory agencies in cost/benefit analyses for decision-making purposes.

The newly developed economic consequence models are applied in an example to estimate the economic risks from operation of the Surry #2 plant. The analyses indicate that economic risks from LWR operation in constrast to public health risks, are dominated by relatively high-frequency forced outage events. The implications of this conclusion for U.S. nuclear power plant operation and regulation are discussed. The sensitivities and uncertainties in economic risk estimates are also addressed.

NUREG/CR-3674: DESIGNING VEGETATION COVERS FOR LONG-TERM STABILIZATION OF URANIUM MILL TAILINGS. BEEDLOW, P. A. Battelle Memorial Institute, Pacific Northwest Laboratories. March 1984. 103pp. 8403230207. PNL-4986. 22741:207.

The use of vegetation and vegetation-rock combinations for long-term stabilization of uranium mill tailings is discussed. Interactions between surface covers and the tailings containment are identified and used as the basis for designing protective covers. The role of vegetation in erosional processes is reviewed and the effectiveness of vegetation for controlling erosion in the western

U.S. is discussed. Principles of revegetation are presented. Environmental influences on vegetation are reviewed. The effects of surface covers on water dynamics within the containment system are presented. A systematic approach is given for designing protective covers using vegetation.

NUREG/CR-3675: IN-CORE FUEL FREEZING AND PLUGGING EXPERIMENTS:Preliminary Results Of Sandia TRAN Series I Experiments. MCARTHUR,D.A.; HAYDEN,N.K.; MAST,P.K. Sandia Laboratories. October 1984. 141pp. 8411290576. SAND81-1726. 27691:036.

The freezing and plugging behavior of molten reactor materials (UO(2) and 316 SS) is of primary importance in analyzing hypothetical accident scenarios of the Liquid Metal Fast Breeder Reactor (LMFBR). However, the reactor materials melt and vaporize at such high temperatures that it is very difficult to obtain data on the behavior of the pure materials in the laboratory. Because almost no data are available for the pure materials, several different theoretical models have been developed to predict freezing behavior. However, these models yield such a wide range of predictions that useful accident analyses are difficult to perform. Described in this report is a new experimental apparatus (the "TRAN" apparatus) in which a pulsed nuclear reactor (the Sandia Annular Core Research Reactor) is used to melt the reactor materials rapidly. After the melt if formed, it is forced under moderate pressure into a "freezing" channel that has a geometry representative of the LMFBR fuel pin structure. The flow and freezing behavior of the pure reactor materials is then inferred from the final distribution of the frozen materials, as well as from the transient behavior of the driving pressure.

NUREG/CR-3677: COMPARISON OF RADON FLUXES WITH GAMMA-RADIATION EXPOSURE RATES AND SOIL 266RA CONCENTRATIONS. YOUNG,J.A.; THOMAS,V.W. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 25pp. 8405020380. PNL-5016. 24287:321.

Radon fluxes and contact gamma-radiation exposures rates were measured at the grid points of rectangular grids on three properties in Edgemont, South Dakota that were known to have deposits of residual radioactivity relatively near to the surface. The coefficient of determination, r(2), between the radon fluxes and the contact gamma-radiation-exposure rates varied from 0.89 to 0.31 for the three properties. Correlations between fluxes and (226)Ra concentrations measured in boreholes that varied in depth from 60 to 195 cm were generally lower than those between fluxes and exposure rates, indicating that exposure rates are better than (226)Ra measurements for detecting elevated radon fluxes from near-surface deposits. Measurements made on one property at two different times indicated that if the average flux were determined from a large number (40) of measurements at one time, the average flux at a later time could be estimated from a few measurements using the assumption that the change in the flux at individual locations will be equal to the change in the average flux. Flux measurements around two buildings showing elevated indoor radon-daughter concentrations, but around which no residual radioactivity had been discovered by (226)Ra and gamma-radiation measurements, provided no clear indication of the presence of such material.

NUREG/CR-3678: ESTIMATION METHODS FOR PROCESS HOLDUP OF SPECIAL NUCLEAR MATERIALS. PILLAY,K.S.; PICARD,R.R.; MARSHALL,R.S. Los Alamos

Scientific Laboratory. July 1984. 121pp. 8408080409. LA-10038. 25980:028

Los Alamos National Laboratory studied the use of statistical estimation methods for materials holdup at highly enriched uranium (HEU)-processing facilities. Use of historical holdup data from processing facilities and selected holdup measurements at two operating facilities confirm the need for high-quality data and reasonable control over process parameters in developing these models. Large-scale experiments were conducted to demonstrate the value of the models from good-quality experimental data. Using these data, we developed statistical models to estimate residual inventories of uranium in large process equipment and facilities. Some important findings are the following:

Holdup in some equipment at HEU-processing facilities, such as air filters, ductwork, calciners, dissolvers, pumps, pipes, and pipefittings can be readily modeled.

Holdup profiles of process equipment such as glove boxes, precipitators, and rotary drum filters can change with time, necessitating several measurements at the time of inventory.

Reasonable estimation of hidden inventories of holdup to meet regulatory requirements can be accomplished through good measurements and statistical modeling.

NUREG/CR-3679: CALIBRATION AND QUALIFICATION OF THE LOS ALAMOS FAILURE MODEL (LAFM). BAARS, R.E. Los Alamos Scientific Laboratory. July 1984. 64pp. 8408080357. LA-10041-MS. 25980:329.

The analysis procedure is described in detail for use of the LAFM computer code to predict LMFBR fuel pin performance under transient overpower conditions; also, 5 tests for calibration and 13 tests for qualification are analyzed. The times of cladding breach (molten fuel expulsion) were predicted with an average relative error of 5 per cent. An enthalpy of 1112 kJ/kg correlated the peak fuel enthalpies at the time of failure with a standard deviation of 98 kJ/kg. We conclude with a discussion that many varied tests must be analyzed for adequate evaluation of a fuel pin performance code.

NUREG/CR-3680: RELATIONSHIP BETWEEN THE GAS CONDUCTIVITY AND GEOMETRY OF A NATURAL FRACTURE. SCHRAUF, T.W.; EVANS, D.D. Arizona, Univ. of, Tucson, AZ. April 1984. 140pp. 8405220089. 24561:156.

In recent years considerable interest in determining the relationship between the hydraulic conductivity of a rock fracture and its average aperture has developed. The present study involved both theoretical and experimental studies of the geometrical factors which influence gas conductivity of rock fractures. Theoretical analysis of parallel plate gas flow revealed that the gas conductivity of a fracture is the same as for incompressible fluids and can be expected to follow a cubic law relationship. Application of the cubic law to practical field test situations, however, was found to be limited by uncertainties in flow boundary conditions, nonlinearity of flow behavior, and effects of fracture surface roughness. Quantitative assessment of uncertainties in flow boundary conditions including elliptical injection boundaries, secondary intersecting fractures, and estimation of effective radius was performed. Nonlinear flow behavior was also analyzed and the results applied to measurements of gas flow rate through a single natural fracture. Evaluation of these results suggested a general flow equation of the f[illegible]/dx)= av + bv(2), where a and b are constant coefficients d[illegible] a fracture's average aperture and surface roughness.

NUREG/CR-3681: MITIGATIVE TECHNIQUES AND ANALYSIS OF GENERIC SITE CONDITIONS FOR GROUND-WATER CONTAMINATION ASSOCIATED WITH SEVERE ACCIDENTS. SHAFER, J. M. ; OBERLANDER, P. L. ; SKAGGS, R. L. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 367pp. 8405220069. PNL-5072. 24562:001.

The purpose of this study is to evaluate the feasibility of using ground-water contaminant mitigation techniques to control radionuclide migration following a severe commercial nuclear power reactor accident. The two types of severe commercial reactor accidents investigated are: 1) containment basemat penetration of core melt debris which slowly cools and leaches radionuclides to the subsurface environment, and 2) containment basemat penetration of sump water without full penetration of the core mass. Six generic hydrogeologic site classifications are developed from an evaluation of reported data pertaining to the hydrogeologic properties of all existing and proposed commercial reactor sites. One-dimensional radionuclide transport analyses are conducted on each of the individual reactor sites to determine the generic characteristics of a radionuclide discharge to an accessible environment. Ground-water contaminant mitigation techniques that may be suitable, depending on specific site and accident conditions, for severe power plant accidents are identified and evaluated. Feasible mitigative techniques and associated constraints on feasibility are determined for each of the six hydrogeologic site classifications. The first of three case studies is conducted on a site located on the Tesas Gulf Coastal Plain. Mitigative strategies are evaluated for their impact on containment transport and results show that the techniques evaluated significantly increased ground-water travel times.

NUREG/CR-3682: NUCLEAR FUEL CYCLE RISK ASSESSMENT: Review and Evaluation of Existing Methods. PELTO, P. J. ; RHOADS, R. E. ; VESELY, W. E. ; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 133pp. 8405210594. PNL-4990. 24534:069.

The U.S. Nuclear Regulatory Commission initiated the Fuel Cycle Risk Assessment Program to provide risk assessment methods for use in the regulatory process for nuclear fuel cycle facilities other than reactors. The first report from this program, NUREG/CR-2873, defined and described fuel cycle elements considered in the program. The second report, NUREG/CR-2933, described the survey and compilation of fuel cycle risk-related literature. This report presents a review of the state-of-the-art of risk assessment methods for nuclear fuel cycle facilities and an evaluation of the adequacy of these methods to meet NRC's needs for risk assessment information. The approach used to perform this work included: identification of potential uses of fuel cycle risk assessments at NRC; review of currently available fuel cycle risk assessment methods; and identification of potential methods development needs.

NUREG/CR-3683: NUCLEAR FUEL CYCLE RISK ASSESSMENT: Program Summary Through Fiscal Year 1983. GEFFEN, C. A. ; PELTO, P. J. ; RHOADS, R. E. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 52pp. 8405210562. PNL-4991. 24535:127.

The U.S. Nuclear Regulatory Commission initiated the Fuel Cycle Risk Assessment Program to provide risk assessment methods for use in the regulatory process for nuclear fuel cycle facilities other than reactors. This report presents a summary of the work completed in the Fuel Cycle Risk Assessment Program through fiscal year 1983. These efforts include descriptions of representative non-reactor facilities

(NUREG/CR-2873), a survey and computer compilation of risk-related literature (NUREG/CR-2933), a preliminary relative ranking of fuel cycle facilities on the basis of risk, and an assessment of the adequcy of existing risk assessment methods (NUREG/CR-3682). Further work in the program has been postponed at this point in time because of funding constraints and higher priorities for other ongoing programs within the NRC. This program summary document will serve as a reference for use in future fuel cycle risk assessment research.

NUREG/CR-3684: NUCLEAR POWER PLANT ALARM PRIORITIZATION (NPPAP) PROGRAM STATUS REPORT. January 1, 1983 to September 31, 1983. ROSCOE, B. J. Sandia Laboratories. April 1984. 75pp. 8405220042. SAND84-0140. 24540:292.

This report describes the status of a research project directed toward nuclear power plant alarm prioritization. Criteria for modified alarm activation are being developed and studied. Also being developed are measures to regulate the alarm rate at some desired level. The problem of alarm prioritization based upon maintenance of critical safety functions while maintaining complete alarm coverage of accidents is being addressed. The plant information needed to support the associated technical development areas is being compiled for a specific plant, categorized, and entered into a computer data base. Near term recommendations for regulatory action on plant annunciator systems are presented.

NUREG/CR-3685: TOXIC GAS ACCIDENT ANALYSIS CODE USER'S MANUAL. CHANIN, D. I.; SHIVER, A. W.; BENNETT, D. E. Sandia Laboratories. October 1984. 65pp. 8411130735. SAND84-0367. 27477:046.

One of the offsite hazards which could threaten the safety of a nuclear power plant is nearby transportation accidents involving releases of toxic gases or volatile liquids. Significant releases of such materials could endanger the plant through incapacitation of control room personnel. An interactive computer program has been developed to aid in the evaluation of control room habitability for these accidents. The first part of the program can be used to study the time history of toxic material concentrations in the control room under varying external conditions, all of which can be specified by the user. The second part estimates the annual probability of operator incapacitation at a particular plant due to nearby accidents on roads or rail lines, or at storage sites. A data base manager is provided so that all data (site and route layouts, plant characteristics, meteorological data, and chemical data) can be entered and maintained in a convenient format. The program was developed for use on CDC computers using the NOS timesharing system.

NUREG/CR-3686: WIPS--COMPUTER CODE FOR WHIP AND IMPACT ANALYSIS OF PIPING SYSTEMS. Summary Report. POWELL, G. H. California, Univ. of, Berkeley, CA. * Lawrence Livermore National Laboratory. June 1984. 12pp. 8407110060. UCRL-15597. 25542:050.

WIPS (Whip and Impact of Piping Systems) is a special purpose computer code for the structural analysis of pipe whip dynamic effects following a postulated pipe rupture. WIPS has been developed primarily to provide support for the pipe whip analysis procedures described in Section 3.6.2 of the U.S. Nuclear Regulatory Commission Standard Review Plan.

This report summarizes the purpose and scope of the WIPS development effort, identifying those clauses in the Standard Review

Plan which refer to pipe whip analysis, and indicating how the WIPS code can be used to provide supporting data. Detailed information on use of the code is contained in accompanying reports which cover (1) use instructions, (2) theory, (3) programming procedures, and (4) verification examples.

NUREG/CR-3686 V01: WIPS--COMPUTER CODE FOR WHIP AND IMPACT ANALYSIS OF PIPING SYSTEMS. Part A - User's Manual. POWELL, G. H. ; HOLLINGS, J. P. ; ROW, D. G. ; et al. California, Univ. of, Berkeley, CA. June 1984. 227pp. 8407110034. UCRL-15597. 25546:053.
See NUREG/CR-3686, Summary abstract.

NUREG/CR-3686 V02: WIPS--COMPUTER CODE FOR WHIP AND IMPACT ANALYSIS OF PIPING SYSTEMS. Part B - Theory Manual. POWELL, G. H. ; HOLLINGS, J. P. ; ROW, D. G. ; et al. California, Univ. of, Berkeley, CA. June 1984. 182pp. 8407110090. UCRL-15597. 25538:001.
See NUREG/CR-3686, Summary abstract.

NUREG/CR-3686 V03: WIPS--COMPUTER CODE FOR WHIP AND IMPACT ANALYSIS OF PIPING SYSTEMS. Part C - Programmer's Manual. POWELL, G. H. ; HOLLINGS, J. P. ; ROW, D. G. ; et al. California, Univ. of, Berkeley, CA. June 1984. 84pp. 8407110076. UCRL-15597. 25537:248.
See NUREG/CR-3686, Summary abstract.

NUREG/CR-3686 V04: WIPS--COMPUTER CODE FOR WHIP AND IMPACT ANALYSIS OF PIPING SYSTEMS. Part D - Verification Manual. POWELL, G. H. ; HU, F-C. California, Univ. of, Berkeley, CA. * Lawrence Livermore National Laboratory. June 1984. 247pp. 8407110052. UCRL-15597. 25537:001.
See NUREG/CR-3686, Summary abstract.

NUREG/CR-3687: LOOSE-PART MONITORING PROGRAMS AND RECENT OPERATIONAL EXPERIENCE IN SELECTED U. S. AND WESTERN EUROPEAN COMMERCIAL NUCLEAR POWER STATIONS. KRYTER, R. C. Oak Ridge National Laboratory. April 1984. 60pp. 8405290447. ORNL/TM-9107. 24711:030.

Technical personnel at thirteen nuclear power stations (ten in the U.S.A. and three in Western Europe) were interviewed to ascertain their collective experience with acoustic-based loose-part monitoring systems (LPMSs). Subjects receiving special attention were the number and location of accelerometers required to reliably detect and locate loose parts in both pressurized-and boiling-water reactor types; detection sensitivity to loose objects in both primary and secondary coolant loops; false alarm experience; calibration procedures; day-to-day monitoring system operation; premature failure of in-containment components of the LPMS; and overall success to date in detecting the presence of potentially damaging loose parts and in assessing their operational and safety implications. The individual utilities' responses to questions addressing these issues are provided, along with the author's summary and interpretation of what the information gathered means in a collective sense.

It is concluded that the technology of loose-part detection and assessment is moving slowly toward increased acceptance by the utility industry but, at the same time, the full potential benefits of loose-part monitoring systems are not presently being realized and, furthermore, probably will not be unless actions are taken in four recommended areas.

NUREG/CR-3689 V01: MATERIAL SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM. Quarterly Progress Report, January-March 1983. SHACK, W. J. Argonne National Laboratory. July 1984. 169pp. 8403100151. ANL-83-85. 25998:313.

This progress report summarizes the Argonne National Laboratory work performed during January, February, and March 1983 on water reactor safety problems. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Transient Fuel Response and Fission Produce Release, Clad Properties for Code Verification, and Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems.

NUREG/CR-3689 V02: MATERIALS SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER REACTOR SAFETY RESEARCH PROGRAM. Quarterly Progress Report, April-June 1983. SHACK, W. J. Argonne National Laboratory. July 1984. 141pp. 8408310083. ANL-83-85. 26348:215.

This progress report summarizes the Argonne National Laboratory work performed during April, May, and June 1983 on water reactor safety problems. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Transient Fuel Response and Fission Product Release, Clad Properties for Code Verification, and Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems.

NUREG/CR-3689 V03: MATERIALS SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER REACTOR SAFETY RESEARCH PROGRAM. Quarterly Progress Report, July-September 1983. REST, J. Argonne National Laboratory. July 1984. 40pp. 8408310077. ANL-83-85. 26347:277.

This progress report summarizes the Argonne National Laboratory work performed during July, August, and September 1983 on water safety problems. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Transient Fuel Response and Fission Product Release, Clad Properties for Code Verification, and Long-Term Embrittlement of Cast Duplex stainless Steels in LWR Systems.

NUREG/CR-3689 V04: MATERIALS SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM. Quarterly Progress Report, October-December 1983. SHACK, W. J. Argonne National Laboratory. October 1984. 170pp. 8411140028. ANL-83-85. 27496:137.

This progress report summarizes the Argonne National Laboratory work performed during October, November, and December 1983 on water reactor safety problems. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Transient Fuel Response and Fission Product Release, Clad Properties for Code Verification, and Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems.

NUREG/CR-3690: RELAP5 ASSESSMENT: SEMISCALE NATURAL CIRCULATION TESTS S-NC-3, S-NC-4, AND S-NC-8. WONG, C. C.; KMETY, K. L. Sandia Laboratories. July 1984. 115pp. 8408310089. SAND-0402. 26348:099.

The RELAP5/MOD1 code is being assessed against test data from a number of integral and separate effects test facilities. As part of this assessment matrix, we have analyzed a number of natural circulation tests performed at the Semiscale facility. Our results for the single-loop and two-loop steady state basecase tests S-NC-2 and S-NC-7 have previously been documented; this report gives the results of calculations for two single-loop degraded heat transfer tests, S-NC-3 and S-NC-4, and for the two-loop ultra-small break transient test S-NC-8. For tests S-NC-3 and S-NC-4, our analyses show that RELAP5/MOD1 describes correctly the qualitative influence of steam generator secondary side heat transfer degradation on both two-phase and reflux natural circulation. The agreement between calculated and measured two-phase mass flow rates in test S-NC-3 is better with a primary mass inventory of 85% (where the peak two-phase mass flow rate is calculated to occur) instead of 92% (where the peak mass flow rate occurred in S-NC-2). Flow oscillations are calculated for both tests, and were seen during S-NC-3, but were not reported in the S-NC-4 experiment. Some of these predicted oscillations are real, but others are nonphysical and can be inhibited by reducing the time step being used (indicating problems in the time step control algorithm). The results for test S-NC-8, an ultra-small (0.4%) cold leg break, also compare reasonably well with the outcome of that experiment. The overall conclusions and their possible relevance to future RELAP5 code application and development are discussed.

NUREG/CR-3691: AN ASSESSMENT OF TERMINAL BLOCKS IN THE NUCLEAR POWER INDUSTRY. CRAFT,C.M. Sandia Laboratories. October 1984. 139pp. 8411290131. SAND84-0422. 27690:307.

The primary application of terminal blocks in the nuclear power industry is instrumentation and control (I&C) circuits. The performance of these circuits can be degraded by low level leakage currents and low insulation resistance (IR) between conductors or to ground. Analyses of these circuits show that terminal blocks, when exposed to steam environments, experience leakage currents and low surface IR levels sufficient to affect some I&C applications. Since the mechanism reducing surface IR (conductive surface moisture films) is primarily controlled by external environmental factors, the degradation of terminal block performance is mostly independent of terminal block design. Testing shows that potential methods of reducing surface leakage currents will not reduce them sufficiently to prevent terminal blocks from affecting I&C circuits. Therefore, terminal blocks can cause erroneous indications or actions of the I&C circuits in which they are a component. Most of the present qualification test of terminal blocks do not address the issue of low level leakage currents, and hence do not demonstrate that terminal blocks will operate properly in I&C circuits.

NUREG/CR-3692: POSSIBLE MODES OF STEAM GENERATOR OVERFILL RESULTING FROM CONTROL SYSTEM MALFUNCTIONS AT OCONEE-1 NUCLEAR PLANT. CLARK,F.H.; CLAPP,N.E. Oak Ridge National Laboratory. BROADWATER,R. Tennessee Tech. Univ., Cookeville, TN. July 1984. 50pp. 8409180488. ORNL/TM-9031. 26589:213.

A study has been made of control system failures which might lead to overfill of the steam generator in Babcock and Wilcox nuclear plants. The steam generator and its control system are described. Only one sequence has been found in which a single failure would lead to overfill, and in that case the final stages of the overfill would proceed rather slowly. Because of high level protective features all other failure sequences we have examined require at least two failures to produce overfill beyond the point of high level protection. Several sequences are described in which high level protection features can be placed in an undetected failed state by a control system failure; a subsequent additional failure, occurring prior to the detection and correction of the first failure, could then produce system overfill. Mechanical damage is identified which might be consequent upon steam generator overfill and water entry into the main steam line. Several ways of reducing the probability of steam generator overfill are suggested. No assessment has been made of the probability of occurrence of any of the sequence.

NUREG/CR-3693: ACOUSTIC EMISSION MONITORING OF HOT FUNCTIONAL TESTING.Watts Bar Unit 1 Nuclear Reactor. HUTTON,P.H.; DAWSON,J.F.; FRIESEL,M.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1984. 52pp. 8407020165. PNL-5022. 25275:181.

Acoustic emission (AE) monitoring of selected pressure boundary areas at TVA's Watts Bar Unit 1 Nuclear Power Plant during hot functional preservice testing is described in this report. The report deals with background, methodology, and results. The work discussed here is a major milestone in a program supported by NRC to develop and demonstrate application of AE monitoring for continuous surveillance of reactor pressure boundaries to detect and evaluate growing flaws. The subject work demonstrated that anticipated problem areas can be overcome. Work is continuing toward AE monitoring during reactor operation.

NUREG/CR-3696: POTENTIAL HUMAN FACTORS DEFICIENCIES IN THE DESIGN OF LOCAL CONTROL STATIONS AND OPERATOR INTERFACES IN NUCLEAR POWER PLANTS. HARTLEY,C.S.; LEVY,I.S.; FECHT,B.A. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 134pp. 8405220088. PNL-4952. 24565:098.

The Pacific Northwest Laboratory has completed a project to identify human factors deficiencies in safety-significant control stations outside the control room of a nuclear power plant and to determine whether NUREG-0700, "Guidelines for Control Room Design Reviews," would be sufficient for reviewing those local control stations (LCSs). The project accomplished this task by first, reviewing existing data pertaining to human factors deficiencies in LCSs involved in significant safety actions; second, surveying LCSs environments and design features at several operating nuclear power plants; and third, assessing the results of that survey relative to the contents of NUREG-0700. The study's conclusions are 1) a definitive list of safety-significant local control stations cannot be specified because power plant designs vary significantly; 2) most, if not all, local control stations have design deficiencies that could be corrected by applying hunan factors engineering principles; and 3) NUREG-0700 is generally applicable to LCSs but that guidance is needed to address the design of manually operated valves and the design requirements of LCSs in extreme environment conditions. Finally, the study recommends an approach for improving present LCSs to reduce the likelihood that operator error will occur.

NUREG/CR-3697: LABORATORY TESTING OF CHEMICAL STABILIZERS FOR CONTROL OF FUGITIVE DUST EMISSIONS FROM URANIUM MILL TAILINGS. ELMORE,M.R.; HARTLEY,J.N. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 53pp. 8405220076. PNL-5025. 24560:324.

Pacific Northwest Laboratory, under contract to the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, is investigating techniques to control fugitive dust emissions from active uranium mill tailings piles. This report describes laboratory tests conducted to evaluate 45 commercially available chemical stabilizers. Tests were conducted in a wind tunnel to evaluate the effectiveness and durability of the stabilizers under similar conditions. The effects of application rate, temperature (freeze/thaw) cycling, wet/dry cycling, and wind speed were determined. In addition, tests were conducted to determine the effects of ultraviolet light and water erosion on the durability of the stabilizers. Permeability tests were also conducted to determine the potential effect of each stabilizer on the overall stability of the tailings pile. Results of these laboratory tests indicated that 16 of the stabilizers were equally effective and more durable than the others.

NUREG/CR-3698: AN AGE STRUCTURED STOCHASTIC RECRUITMENT MODEL FOR ASSESSMENT POWER PLANT IMPACT. SULLIVAN, P. J.; SWARTZMAN, G. L. Washington, Univ. of, Seattle, WA. March 1984. 29pp. 8404120390. 24034:306.

The dynamics of the Hudson River striped bass (Morone Saxatilis) stock were analyzed using a stochastic age structured model. The effect of river flow on recruitment was combined with the mortality due to fishing and power plant water uptake to obtain an overall effect of these variables on the fishery. Model equations and parameters were documented and their underlying assumptions presented. Preliminary model runs resulted in yields well below those actually observed. Calibration of model parameters brought these values closer to the observed yields, but stock values proved inexact. The influence of power plant mortality on fishery yield was evident, but the simulation results remain inconclusive.

NUREG/CR-3699: A SUMMARY OF CODES FOR WASTE PACKAGE PERFORMANCE ASSESSMENT. COFFMAN, W.; VOGT, D.; MILLS, M. CorSTAR Research, Inc. March 1984. 314pp. 8404020158. 22848:012.

This is the fourth in a series of five reports that will provide critical reviews and summaries of computer programs that can be used to analyze the potential performance of a high-level radioactive waste repository. The computer programs identified in this report address the performance of a waste package, including the areas of thermal analysis, structural analysis, and special purpose programs. The report provides a summary description of 19 computer programs. Fourteen of these computer programs are being used by the U.S. Department of Energy or the U.S. Nuclear Regulatory Commission to analyze various aspects of waste package performance. The remaining five codes can be used to analyze phenomena that may be important in waste package performance.

NUREG/CR-3700: DECAY OF BUOYANCY DRIVEN STRATIFIED LAYERS WITH APPLICATION TO PRESSURIZED THERMAL SHOCK (PTS). THEOFANOUS, T. G.; NOURBAKHSH, H. P.; GHERSON, P.; et al. Purdue Univ., West Lafayette, IN. May 1984. 214pp. 8406060373. 24669:022.

This report consists of two parts. In Part I physically based calculational models are proposed for predicting (a) conditions for stratification due to HPI in a circulating reactor loop (stratification model) and (b) cooldown transients due to HPI in a stagnated primary reactor fluid (thermal mixing model). The integral aspects of these models are confirmed by comparison to the CREARE 1/5-scale data. In Part II the thermal mixing model is assessed in an integral as well as in a local sense by comparison to the first round of data from Purdue's 1/2-scale facility. These data are the only available large-scale data at this time and they are an important complement to CREARE's 1/5-scale results in constructing a basis for scale-up to reactor conditions. Facility construction, instrumentation, data reduction techniques and detailed experimental results are also included in Part II.

NUREG/CR-3704: THREE-DIMENSIONAL CALCULATIONS OF TRANSIENT FLUID-THERMAL MIXING IN THE DOWNCOMER OF THE CALVERT CLIFFS-1 PLANT USING SOLA-PTS. DALY, B. J. Los Alamos Scientific Laboratory. April 1984. 86pp. 8406230301. LA-10039-MS. 25131:319.

The SOLA-PTS code has been used to analyze transient fluid-thermal mixing in a 180 degree sector of the downcomer and a cold leg of the Calvert Cliffs-1 plant for three assumed accident scenarios. The inlet boundary conditions for these calculations were obtained from mass flou rates and temperatures that were computed in systems code studies. The results of the three-dimensional SOLA-PTS calculations indicated that a pressurized thermal shock risk was mitigated for these accident scenarios as the result of the particular circulation patterns that developed in the downcomer.

NUREG/CR-3708: LWR SPENT FUEL DRY STORAGE BEHAVIOR AT 229 C. EINZIGER, R. E. Westinghouse Electric Corp. COOK, J. A. EG&G, Inc. July 1984. 134pp. 8408240379. 26255:001.

A whole rod test was conducted at 229 degrees centigrade to investigate the long-term stability of spent fuel rods under a variety of possible dry storage conditions. All combinations of BWR or PWR rods, inert or air atmospheres, and intact or defected rods were tested. After 2235 hours, visual observations, diametral measurements and radiograhic smears uere used to assess the degree of cladding deformation and particulate release. The same examinations plus metallography and x-ray analysis were conducted after 5962 hours.

None of the intact rods, the rods tested in inert atmosphere, or the defected PWR rod tested in unlimited air showed any measurable change from the pretest condition. The upper defect on the BWR rod tested in unlimited air had split open ~0.5 in. after 2235 hours and had ~10% cladding defornation. The crack grew to ~2.5 in. after 7962 hours. X-ray analysis indicated that the UO(2) had oxidized to U(3)O(8). The difference in behavior of the upper and lower defects is attributed to the air's accessibility to the fuel because of the deflect's position with respect to the pellet-pellet interface.

The oxidized fuel appeared to form a powdery compact that remained for the most part in the split cladding. Only a fraction of the fuel out of the cladding became airborne. Some crud spalled from the rods but appeared to have no airborne particulate in the 2- to 15-mean respirable range. This report discusses the details and meaning of the data from this test.

NUREG/CR-3711: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE I TEST RESULTS. HWANG, W. S.; ALAMGIR; SUTHERLAND, W. A.; et al. General Electric Co. September 1984. 300pp. 8410100085. EPRI NP-3602. 26901:001.

A new full height BWR system simulator has been built under the Full Integral Simulation Test (FIST) program to investigate the system responses to various transients. The test program consists of two test phases. This report provides a summary, discussions, highlights, and conclusions of the FIST Phase I Tests. Eight matrix tests were conducted in the FIST Phase I. These tests have investigated the large break, small break and steamline break LOCAs, as well as natural circulation and power transients. Results and government phenomena of each test have been evaluated and discussed in detail in this report. Two of these tests tie back to tests in the earlier TLTA facility. Comparisons between the FIST and TLTA tests have been made. The similarities and differences between counterpart tests are identified. Effects of the facility scaling compromises on the test results are identified. One of the FIST program objectives is to assess the TRAC code by comparisons with test data. Two pretest predictions made with TRACBO2 are presented and compared with test data in this report. These predictions agree very well with the test results. TRAC's capability to correctly predict the system responses during the transient is demonstrated.

NUREG/CR-3712: RADIONUCLIDE MIGRATION IN GROUNDWATER. Annual Report for FY 1983. FRUCHTER, J. S.; COWAN, C. E.; ROBERTSON, D. E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1984. 133pp. 8501070504. PNL-5040. 28238:129.

Staff at Pacific Northwest Laboratory for the past several years have been collecting the available data concerning radionuclide migration in the groundwater at a low-level disposal site and studying the physicochemical processes that control the mobility of radionuclides in the groundwater. This report summarizes the results obtained and progress made during FY83.

NUREG/CR-3713: GROUPING OF LIGHT WATER REACTORS FOR EVALUATION OF DECAY HEAT REMOVAL CAPABILITY. KAROL, R.; FRESCO, A.; PERKINS, K. R. Brookhaven National Laboratory. June 1984. 82pp. 8407110022. BNL-NUREG-51752. 25547:001.

This grouping report provides a compilation of decay heat removal systems (DHRS) data for operating commercial light water reactors. The reactors have been divided into 12 groups based on similarity of the DHRS and related systems as part of the NRC Task Action Plan on Shutdown Decay Heat Removal Requirements.

NUREG/CR-3714: ON THE DEVELOPMENT OF ENVIRONMENTAL RADIATION STANDARDS FOR GEOLOGIC DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES. KOCHER, D. C. Oak Ridge National Laboratory. July 1984. 80pp. 8408080422. ORNL-6006. 25977:192.

This report discusses the different technical issues that must be considered in developing an environmental standard for geologic disposal of high-level radioactive wastes. These issues include (1) defining acceptable risk, (2) specifying acceptable risk in the standard, (3) formulating the standard so that reasonable demonstrations of compliance can be obtained, (4) applying the standard to protection of individuals or the population, (5) applying the standard to expected occurrences only or to unexpected processes as well, (6) determining a time limit for the standard, and (7) specifying conditions to be assumed for demonstrating compliance. It is concluded that many issues are not resolvable on technical grounds alone, but that an effective standard will allow flexibility and the exercise of subjective scientific judgements is reaching licensing decisions.

NUREG/CR-3715: DESIGN, CONSTRUCTION AND TESTING OF A 2000 C FURNACE AND FISSION PRODUCT COLLECTION SYSTEM. OSBORNE, M. F.; COLLINS, J. L.; LORENZ, R. A.; et al. Oak Ridge National Laboratory. October 1984. 58pp. 8411290160. ORNL/TM-9135. 27700:314.

An induction furnace, capable of operation at 2000 degrees centigrade in steam, was developed to conduct product release tests. The test specimen and steam atmosphere are contained in a stabilized ZrO(2) furnace tube, which is heated by a concentric susceptor of either tungsten or graphite. A two-color optical pyrometer and high-temperature thermocouples are used for temperature measurement. The furnace has operated reliably for periods up to 30 min at test temperatures of 1400 to 2000 degrees centigrade with steam flowing at approximately 1 L/min. The apparatus for collecting the released fission products includes a TGT, an aerosol deposition sampler, a series of glass fiber filters, and heated charcoal. A steam condenser and cooled charcoal, for inert gas adsorption, are located further downstream. The principal analytical techniques used for fission product identification and measurement are (1) gamma spectrometry, for all radionuclides on all test components; (2) spark-source mass spectrometry, for all elements, primarily deposits on the thermal gradient tube and filters; and (3) neutron activation, for iodine on selected test components.

NUREG/CR-3716: CONTEMPT4/MOD4 A Multicomponent Containment System Analysis Program. LIN, C. C.; ECONOMOS, C.; LEHNER, J. R.; et al. Brookhaven National Laboratory. March 1984. 281pp. 8404110024. BNL-NUREG-51754. 24003:001.

CONTEMP4/MOD4 is a digital computer program that describes the response of multicompartment containment systems subjected to postulated loss-of-coolant accident (LOCA) conditions. The program is written in FORTRAN IV and can accommodate both pressurized water reactor (PWR) and boiling water reactor (BWR) containment systems. Also, both design basis accident (DBA) and degraded core type: LOCA conditions can be analyzed. The program calculates the time variation of compartment pressures, temperatures and mass and energy inventories due to intercompartment mass and energy exchange, LOCA source terms, containment fans and pumps, cooling sprays, heat conducting structures, sump drains, PWR ice condensers, BWR pressure suppression systems hydrogen combustion within compartments and energy transfer due to gas radiation. Dynamic storage allocation (DBA) is used to limit the amount of computer core used for each problem and to provide the multicompartment capability of up to 999 individual compartments. The program employs an implicit algorithm to compute junction flow when numerically induced flow oscillations are encountered. This capability provides significant reduction of computer run time relative to previous codes in the CONTEMPT series.

NUREG/CR-3717: EVALUATION OF ROBOTIC INSPECTION SYSTEMS AT NUCLEAR POWER PLANTS. WHITE, J. R. ; EBERSOLE, R. E. ; FARNSTROM, K. A. ; et al. Remote Technology Corp. March 1984. 100pp. 8404020268. 22875:001.

This report presents and demonstrates a cost-effective approach for robotics application (CARA) to surveillance and inspection and work in existing nuclear power plants.

The CARA was developed by the Remote Technology Corporation to systematically determine the specific surveillance/inspection tasks, worker hazards, and access or equipment placement restraints in each of the many individual rooms or areas at a power plant. Solutions for each area are based upon the modular arrangement of commercially-available sensors and other robotic components.

Techniques for maximizing the cost effectiveness of robotics are emphasized in the report including: selection of low-cost robotic components, minimal installation work in plant areas, portable systems for common use in different areas, and standardized robotic modules. Factors considered as benefits are reduced radiation exposure, man-hours, power outage, waste material, and worker safety concerns.

A partial demonstration of the CARA approach to a large BWR plant is provided in the report along with specific examples of robotic installations. Full utilization will require in-depth application at specific plants.

NUREG/CR-3718: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING - STATUS REPORT. LU, S. C. ; CHOU, C. K. Lawrence Livermore National Laboratory. April 1984. 44pp. 8404250006. UCID-19722. 24202:295.

A confirmatory piping reliability assessment for stiff versus flexible piping systems indicated that removing rigid supports tends, in general, to reduce thermal stress but to increase seismic stress in the pipe. As a result, piping design can be made more reliable by some reduction of rigid supports. We also observed that piping design using snubbers among support devices may not exhibit the intended reliability because snubbers often fail to perform the desired function. It was demonstrated that certain piping systems with snubbers removed actually exhibit higher reliability than do those of the original design.

The Steering and Technical Committees on Piping Systems established by the Pressure Vessel Research Committee (PVRC) have investigated changes to be implemented in Regulatory Guide (RG) 1.61 and RG 1.122 aimed at more flexible piping design. An independent impact assessment conducted by this project concluded that: (1) PVRC proposed changes substantially reduce calculated piping response; (2) proposed changes allow piping redesigns with significant reduction in number of supports and snubbers without violating ASME code requirements; and (3) the more flexible piping redesigns are capable of exhibiting reliability levels equal to or higher than the original stiffer design.

NUREG/CR-3719: DETONATION CALCULATIONS USING A MODIFIED VERSION OF CSQII: EXAMPLES FOR HYDROGEN-AIR MIXTURES. BYERS, R. K. Sandia Laboratories. October 1984. 53pp. 8411290061. SAND83-2624. 27692:269.

CSQ is a well-tested and versatile wave propagation computer program, a modified version of which has been used to perform a number of USNRC-supported analyses of detonations of hydrogen-air mixtures in nuclear reactor containment buildings. The modifications, from a user's viewpoint, are fairly minor, and this version of CSQ is being prepared for release to interested organizations. This report documents the use of CSQ in this form, as well as certain codes which aid in performing the detonation calculations.

NUREG/CR-3720: PREDICTION AND EXPERIMENT COMPARISONS FOR GERMAN STANDARD PROBLEM 4A:PIPING RESPONSE TO BLOWDOWN. HOWARD,G.E. ANCO Engineers, Inc. April 1984. 4pp. 8404110013. 22995:304.

This report consists of comparisons of prediction and experiment for German Standard Problem 4a, a blowdown experiment involving structural dynamic response. The comparisons presented herein are of the time histories of displacement, bending stress, and bending axis angle. The reasons for error in the predictions are discussed. The structural model is improved to obtain a better match with the experimental natural frequencies.

NUREG/CR-3721 V02: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Volume 2:An Evaluation of Three Pressure Transducers. MARSHALL,B.W.; RATZEL,A.C. Sandia Laboratories. October 1984. 69pp. 8411130663. SAN083-2621. 27477:235.

A series of hydrogen:air combustion tests was performed at the Fully Instrumented Test Site (FITS), located at Sandia National Laboratories in Albuquerque, NM, to evaluate the performance of three strain-gage-type pressure transducers in a combustion environment. Results of the sixty tests indicated that the three pressure transducers, when thermally shielded with felt metal, recorded peak combustion pressures that were generally within 5% of the statistical mean for each test. The pressure profiles and associated burn times obtained from all of the protected transducers were also comparable. The Precise Sensor model 111-1 gages, when unprotected, were affected significantly by the hot gases of combustion and must be thermally protected with felt metal to obtain accurate measurements. However, thermally unprotected Precise Sensor 141-1 gages recorded transient combustion pressure traces that compared well with those recorded by the thermally protected gages. The Kulite sensor was always used with thermal protection as recommended by the manufacturer. These tests also showed that the Brunswick 1101 felt metal serves as an effective thermal barrier without affecting the magnitudes of the peak pressures, the rise times, or the composite shape of the transient combustion pressure response.

NUREG/CR-3722: DAMPING TEST RESULTS FOR STRAIGHT SECTIONS OF 3-INCH AND 8-INCH UNPRESSURIZED PIPES. WARE,A.G.; THINNES,G.L. EG&G, Inc. May 1984. 68pp. 8406070132. EGG-2305. 24850:280.

EG&G Idaho is assisting the Nuclear Regulatory Commission and the Pressure Vessel Research Committee in supporting a final position on revised damping values for structural analyses of nuclear piping systems. As part of this program, a series of vibrational tests on unpressurized 3-in. and 8-in. Schedule 40 carbon steel piping was conducted to determine the changes in structural damping due to various parametric effects. The 33-ft straight sections of piping were supported at the ends. Additionally, intermediate supports comprising spring, rod, and constant-force hangers, as well as a sway brace and snubbers, were used. Excitation was provided by low-force-level hammer impacts, a hydraulic shaker, and a 50-ton overhead crane for snapback testing. Data was recorded using acceleration, strain, and displacement time histories. This report presents test results showing the effect of stress level and type of supports on structural damping in piping.

NUREG/CR-3724: ULTIMATE STRENGTH ANALYSES OF THE WATTS BAR, MAINE YANKEE, AND BELLEFONTE CONTAINMENTS. JUNG, J. Sandia Laboratories. July 1984. 82pp. 8409260631. SAND84-0660. 26699:240.

As part of Sandia National Laboratories' Severe Accident Sequence Analysis (SASA) Program, structural analyses of the Watts Bar, Maine Yankee, and Bellefonte containment structures were performed with the objective of obtaining realistic estimates of their ultimate static pressure capacities. The Watts Bar investigation included analyses of the containment shell, equipment hatch, anchorage systems, and personnel lock. The ultimate pressure capability is estimated to be between 120 and 137 psig, corresponding to shell yielding and equipment hatch buckling, respectively. The Main Yankee investigation consisted of an analysis of the containment shell and estimated its failure pressure to be between 96 and 118 psig. For the Bellefonte containment, analyses of the containment shell and equipment hatch were performed. The pressure capacity of the Bellefonte containment is estimated to be between 130 and 139 psig, corresponding to dome tendon yielding and cylinder wall tendon yielding, respectively.

NUREG/CR-3725: NUCLEAR POWER PLANT SIMULATORS FOR OPERATOR LICENSING AND TRAINING: Part I - The Need For Plant-Reference Simulators; Part II - The Use Of Plant-Reference Simulators. RANKIN, W. L.; BALTON, P. A.; SHIKIAR, R.; et al. Battelle Human Affairs Research Centers. May 1984. 126pp. 8405310090. PNL-5049. 24734:223.

Part I of this report presents technical justification for the use of plant-reference simulators in the licensing and training of nuclear power plant operators and examines alternatives to the use of plant-reference simulators. The technical rationale is based on research on the use of simulators in other industries, psychological learning and testing principles, expert opinion, and user opinion. Strong technical justification exists for requiring plant-reference simulators for operator licensing purposes. Technical justification for the use of plant-reference simulators for operator training is less well grounded empirically, although expert opinion is that plant-reference simulators, when properly used, result in the most effective training. Part II discusses the central considerations in using plant-reference simulators for licensing examination of nuclear power plant operators and for incorporating simulators into nuclear power plant training programs. Recommendations are presented for the administration of simulator examinations in operator licensing that reflect the goal of maximizing both reliability and validity in the examination process. A series of organizational tasks that promote the acceptance, use, and effectiveness of simulator training as part of the onsite training program is delineated.

NUREG/CR-3726: SIMULATOR FIDELITY AND TRAINING EFFECTIVENESS: A Comprehensive Bibliography With Selected Annotations. BOLTON, P. A.; FAICENBLUM, J. M.; HOPE, A. M.; et al. Battelle Human Affairs Research Centers. May 1984. 70pp. 8405300079. PNL-4765. 24713:302.

This document contains a comprehensive bibliography on the topic of simulator fidelity and training effectiveness, prepared during the preliminary phases of work on an NRC-sponsored project on the Role of Nuclear Power Plant Simulators in Operator Licensing and Training. Section A of the document is an annotated bibliography consisting of articles and reports with relevance to the psychological aspects of simulators in a variety of settings, including military. The annotated items are drawn from a more comprehensive bibliography, presented in Section B, listing documents treating the role of simulators in operator training both in the nuclear industry and elsewhere.

NUREG/CR-3727: FISSION PRODUCT REMOVAL IN ENGINEERED SAFETY FEATURE (ESF) SYSTEMS. Data Base Assessment And Suggested Experimental Program. ZALOUDEK,F.R.; POSTMA,A.K.; WINEGARDNER,W. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 49pp. 8405220035. PNL-5050. 24560:159.

The available data base of the fission product removal capabilities of nuclear reactor Engineered Safety Feature (ESF) systems was reviewed and assessed. The systems considered included pressure suppression pools, ice condenser systems, containment sprays, filter systems and containment air coolers. Based on this assessment, a research program was recommended to expand this data base to support the development of mechanistic models and computer codes for the prediction of ESF system fission product removal. This research program included experimental efforts to better define the performance of ice condenser systems, expand the range of data available on water spray systems and to investigate the behavior of containment air coolers, demisters and fans in the presence of aerosols typical of those expected following a severe accident.

NUREG/CR-3728: EFFECT OF TEMPERATURE ON THE STRESS-RELAXATION RESPONSE OF A PRESSURE VESSEL STEEL. GOODWIN,G.M.; NANSTAD,R.K. Oak Ridge National Laboratory. October 1984. 19pp. 8501030036. ORNL/TM-9149. 28200:270.

Extensive cracking in the steam generator shells at Indian Point Station Unit 3 led to questions as to the effectiveness of the postweld heat treatment (PWHT) used during fabrication. A literature review revealed an absence of stress-relaxation data for the steels of interest, SA-302, grade B, and SA-533, grade B. This investigation was undertaken to characterize the stress-relaxation response at various PWHT temperatures and to determine the correlation, if any, with other measurable properties, such as hardness. A novel technique utilizing a closed-loop thermal-mechanical simulator, the Gleeble 1500, was developed and used. After producing a microstructure typical of a portion of the heat-affected zone (HAZ) of a weldment, we tested stress relaxation at eight temperatures, 482, 510, 538, 566, 593, 621, 649, and 677 degrees centigrade (900, 950, 1000, 1050, 1100, 1150, 1200, and 1250 degrees farenheit). We determined that 20 min (1200 s) was an adequate test time and that the magnitude of stress measured agreed with limited literature data for comparable materials. Substantial stress relaxation was noted at 593 degrees centigrade (1100 degrees farenheit) and above. Post-test hardness correlated very well with relaxation data.

NUREG/CR-3730: EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS. KELMERS,A.D.; CLARK,R.J.; CUTSHALL,N.H.; et al. Oak Ridge National Laboratory. March 1984. 59pp. 8404160218. ORNL/TM-9109. 24065:234.

This report summarizes the results of the activities of the first year in a Nuclear Regulatory Commission (NRC) supported, experimentally oriented project to evaluate geochemical information (primarily radionuclide sorption and apparent concentration limit values). The geochemical information is being developed by the Department of Energy (DOE) in their high-level waste repository site projects for use in site performance assessment calculations. During this period, work was focused on the candidate site in basalt being characterized by the Basalt Waste Isolation Project (BWIP). Activities involved measuring sorption values for technetium and neptunium under geochemical parameters expected to be relevant to the site. Oxic (air saturated), reducing (hydrazine added), and anoxic (air excluded) redox conditions were employed in batch contact tests. Geochemical modeling calculations using available thermodynamic data and codes also were carried out to evaluate site groundwaters. The results are compared with values published by BWIP as "conservative best estimate" numbers in their Site Characterization Report and to other values and information in other available BWIP reports.

NUREG/CR-3734: LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM. Semiannual Report, October 1982 - March 1983. BERMAN, M. Sandia Laboratories. July 1984. 276pp. 8408100152. SAND84-0688. 25996:001.

This report describes the investigations and analyses conducted at Sandia National Laboratories, Albuquerque, in support of the Light Water Reactor Safety Research Program from October 1982 through March 1983. The Molten Fuel/Concrete Interactions (MFCI) Study investigates the mechanism of concrete erosion by molten core materials, the nature and rate of generation of evolved gases, and the effects of fission-product release. The Core Melt/Coolant Interactions (CMCI) Study investigates the characteristics of explosive and nonexplosive interactions between molten core materials and concrete, and the probabilities and consequences of such interactions. In the Hydrogen Program, the HECTR code for modelling hydrogen deflagration is being developed, experiments (including those in the FITS facility) are being conducted, and the Grand Gulf Hydrogen Igniter System II is being reviewed. All activities are continuing.

NUREG/CR-3735: ACCIDENT-INDUCED FLOW AND MATERIAL TRANSPORT IN NUCLEAR FACILITIES--A LITERATURE REVIEW. BOLSTAD, J. W.; GREGORY, W. S.; MARTIN, R. A.; et al. Los Alamos Scientific Laboratory. July 1984. 45pp. 8408160146. LA-100079-MS. 26122:266.

The reported investigation is part of a program that was established for deriving radiological source terms at a nuclear facility's atmospheric boundaries under postulated accident conditions. The overall program consists of three parts: (1) accident delineation and survey, (2) internal source term characterization and release, and (3) induced flow and material transport. This report is an outline of pertinent induced-flow and material transport literature. Our objectives are to develop analytical techniques and data that will permit prediction of accident-induced transport of airborne material to a plant's atmospheric boundaries.

Prediction of material transport requires investigation of the ares of flow dynamics and reentrainment/deposition. A review of material transport, fluid dynamics, and reentrainment/deposition literature is discussed. In particular, those references dealing with model development are discussed with special emphasis on application to a facility's interconnected ventilation system.

NUREG/CR-3735: ACCIDENT-INDUCED FLOW AND MATERIAL TRANSPORT IN NUCLEAR FACILITIES--A LITERATURE REVIEW. BOLSTAD, J. W.; TANG, P. K.; MERRYMAN, R. G.; et al. Los Alamos Scientific Laboratory. July 1984. 45pp. 8408160146. LA-100079-MS. 26122:266.

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NUREG/CR-3737: AN INITIAL EMPIRICAL ANALYSIS OF NUCLEAR POWER PLANT ORGANIZATION AND ITS EFFECT ON SAFETY PERFORMANCE. OLSON, J.; MCLAUGHLIN, S. D.; OSBORN, R. N.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. November 1984. 100pp. 8411260129. PNL-5102. 27646:151.

This report contains an analysis of the relationship between selectd aspects of organizational structure and the safety related performance of nuclear power plants. The report starts by identifying and operationalizing certain key dimensions of organizational structure that may be expected to be related to plant safety performance. Next, indicators of plant safety performance are created by combining existing performance measures into more reliable indicators. Finally, the indicators of organizational structure were related to the indicators of plant safety performance using correlational and discriminant analysis. The overall results show that plants with better developed coordination mechanisms, shorter vertical hierarchies, and a greater number of departments tend to perform more safely.

NUREG/CR-3739: THE OPERATOR FEEDBACK WORKSHOP: A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL. MCGUIRE, M. V.; WALSH, M. E.; BOEGEL, A. J. Battelle Human Affairs Research Centers. September 1984. 246pp. 8409280100. PNL-5214. 26750:001.

This report presents the results of three workshops that were designed, conducted, and assessed for the Nuclear Regulatory Commission. The purposes of the workshops were to (1) examine the effectiveness of workshops and other techniques as mechanisms for obtaining feedback from utility personnel, including comparison of several different workshop procedures; and (2) obtain feedback for the NRC on topics of interest and concern. The workshops were held in NRC Regions I, II, and III between December 1981 and May 1982. A total of 60 utility personnel attended the workshops and offered comments and suggestions concerning staffing, engineering support in the control room, training tools, training programs, and licensing examinations. Workshop participants and observers evaluated the workshops favorably. Further assessment of the workshop process and content suggested that the workshops were effective in obtaining useful feedback for the NRC.

NUREG/CR-3740: J-INTEGRAL TEARING INSTABILITY ANALYSIS FOR 8-INCH DIAMETER ASTM A106 STEEL PIPE. VASSILAROS, M. G. ; HAYS, R. A. ; GUDAS, J. P. ; et al. David W. Taylor Naval Research & Development Center. April 1984. 100pp. 8404300225. 24254:135.

An experimental investigation was performed to evaluate the applicability of using J-Integral tearing instability analysis to describe the fracture behavior of 8-inch (203 mm) diameter, nuclear grade, ASTM A106 steel pipe. Pipe sections measuring 48-inches (1219 mm) in length and 8.60 inches (219 mm) in diameter with circumferential fatigue precracks were loaded in four point bending using a variable compliance test arrangement. J-Integral tests were performed on 1/2 T, 1 T, and 2 T plan compact specimens machined from the pipe. These J-Integral resistance curves (J1-R curves) were compared to the J1-R curves from the pipe bend tests. Two different J-Integral analyses were used to describe fracture behavior. In one analysis, the material was modelled by assuming elastic-perfectly measurements of mechanical response of the loaded structure including hardening of the steel. The evaluation of the J-Integral tearing instability analysis was performed using J versus T plots of each test. The results of the investigation indicate that compact speciment JI-R curve test results appear to agree with the JI-R curves from ,ll size pipe bend tests. Further, J-Integral tearing instability analysis can accurately describe the ductile tearing behavior of 8-inch ASTM A106 steel pipe provided the actual load, displacement, crack length and hardening behavior is available. Additionally, the results indicated that such an analysis with assumed elastic fully plastic behavior appears to produce conservative results.

NUREG/CR-3741 V01: EVALUATION OF POWER REACTOR FUEL ROD ANALYSIS CAPABILITIES. Phase 2 Topical Report, Volume 1: Data Evaluation. COLEMAN, D. R. Control Data Corp. April 1984. 137pp. 8405220084. 24565:230.

The second phase of acquisition, review, analysis, and processing of power reactor fuel performance data resources is described in this report. These data resources are characterized to support subsequent evaluations of the NRC-sponsored fuel rod behavior code, FRAPCON.

Application of the Fuel Performance Data Base is shown to provide the basic data files which are sorted, processed, and restructured to establish key parameters of interest on an individual rod basis. The design, operational, and performance parameters are analyzed to determine the data populations and the representation of various fuel design types in the data sample. Also presented are the performance data distribution and trends relative to operational parameters such as power and burnup, and description of the data processing methods.

Significant amounts of power reactor fuel performance data are available to support high burnup code evaluation studies. The data clearly indicates the cumulative effects of rod deformation, fission gas release, and corrosion which tend to alter the as-built fuel rod thermal and mechanical conditions. The available data reflect the current status of commercial fuel utilization in that incumbent designs are gradually being replaced by high burnup designs, but the newer fuel types do not yet dominate the data sample.

NUREG/CR-3742: BUCKLING OF STEEL CONTAINMENT SHELLS UNDER TIME-DEPENDENT LOADING. BABCOCK, C. D. ; BAKER, W. E. ; FLY, J. ; et al. Los Alamos Scientific Laboratory. July 1984. 38pp. 8408100153. LA-10087-MS. 25997:268.

The problem of dynamic effects for steel containment shells subjected to time-dependent loadings that produce large compressive membrane stresses in the shell wall is considered. Loadings on typical containment structures are reviewed, along with a description of the complete dynamic-buckling problem. Simplifications and the assumptions that are currently used are critically examined and reviewed with respect to buckling analysis. Based on these reviews, three program objectives are defined and the tasks that can accomplish these objectives within a 2-year effort at level funding are outlined in detail.

NUREG/CR-3743: THE IMPACT OF NDE UNRELIABILITY ON PRESSURE VESSEL FRACTURE PREDICTIONS. SIMONEN, F.A. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 28pp. 8406060417. PNL-5062. 24669:236.

This report reviews the significant variables of flaw depth, length, location and orientation required for fracture mechanics evaluations of pressure vessel integrity. Results of calculations are presented which emphasize pressurized thermal shock (PTS) and the significance of flaws located at or near the inside surface of the vessel. For PTS conditions, previous studies have shown that vessel failure probability is relatively insensitive to flaw depth. In this study the impact of flaw length is also evaluated, indicating the importance of fully characterizing all flaw dimensions by NDE. Results of other evaluations are presented, showing the importance of accurately locating flaws by NDE. The influence of vessel cladding is emphasized, with the relative significance of flaws through the clad and at various depths below the clad being addressed.

NUREG/CR-3744 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1983 - MARCH 1984. PUGH, C.E. Oak Ridge National Laboratory. July 1984. 200pp. 8408080355. ORNL/TM-9154/V1. 25979:146.

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 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 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-3745: BIOLOGICAL CHARACTERIZATION OF RADIATION EXPOSURE AND DOSE ESTIMATES FOR INHALED URANIUM MILLING EFFLUENTS. Annual Progress Report, April 1, 1982 - March 31, 1983. EIDSON, A.F. Inhalation Toxicology Research Institute. * Lovelace Biomed & Environmental Research Institute. May 1984. 53pp. 8406010536. LMF-108. 24764:175.

A quantitative infrared absorption method for yellowcake allowed the fraction of ammonium diuranate in a mixture to be determined accurately within 7% and the U(3)O(8) fraction within 13%. The composition of yellowcake from six operating mills ranged from nearly pure ammonium diuranate to nearly pure U(3)O(8). A study of retention and translocation of uranium after subcutaneous implantation in rats was done. The results showed that 49% of the implanted yellowcake cleared from the body with a half-time in the body of 0.3 days, and the remainder was cleared with a half-time of 11 to 30 days. Twenty dogs exposed to a more soluble yellowcake form inhaled aerosols producing an estimated initial lung burden of 130 micrograms of U per kilogram of body weight. Aerosols inhaled by dogs exposed to a less soluble yellowcake form averaged an estimated initial lung burden of 140 micrograms of U per kilograms of body weight. Biochemical indicators of kidney dysfunction that appeared in blood and urine 4 to 8 days after exposure to the more soluble yellowcake showed significant changes in dogs, but levels returned to normal by 16 days after exposure. No biochemical evidence of kidney dysfunction was observed in dogs exposed to the less soluble yellowcake form.

NUREG/CR-3746 V01: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM. Semiannual Progress Report, October 1983 -March 1984. LIPPINCOTT, E. P.; MCELROY, W. M. Hanford Engineering Development Laboratory. November 1984. 168pp. 8412120266. HEDL-TME 84-20. 27913:128.

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) has been established by 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 US NRC-supported activities at HEDL, ORNL, NBS, and MEA-ENSA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Julich, Germany), and several UK 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 are responsible for the preparation of LWR surveillance standards.

NUREG/CR-3748: COBRA/TRAC SIMULATION OF SEMISCALE S-UT-5 TEST. BIAN, S. H.; THURGOOD, M. J. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 102pp. 8405210570. PNL-5065. 24529:208.

The computer code COBRA/TRAC was used to simulate a Small Break Loss-of-Coolant Accident (SBLOCA) test performed at the Semiscale MOD-2A Test Facility operated by the Idaho National Engineering Laboratory. The results of the simulation were compared with the results of the actual test. The comparison showed that the code has the capability to model small-break accidents in an integrated coolant system of a pressurized water reactor (PWR).

NUREG/CR-3749: COBRA-NC POST-TEST PREDICTIONS FOR HDR CONTAINMENT STEAM BLOWDOWN TEST V44 (INTERNATIONAL STANDARD PROBLEM 16). THURGOOD, M. J. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 127pp. 8405210604. PNL-5066. 24535:001.

COBRA-NC is a digital computer program written in FORTRAN IV that simulates the response of multicompartment light water reactor containment systems to postulated loss-of-coolant accidents. It has been used to perform post-test predictions of the response of the German project HDR containment system to a simulated steam line break blowdown transient. Predictions were made of compartment pressures, gas temperatures, structural temperatures, and differential pressures between rooms. In this report, these predictions are compared with the experimentally measured values. The agreement with the data is reasonable. Improvements in the prediction can be made by more carefully modeling the flow openings between rooms or by using a finer mesh.

NUREG/CR-3750: JOB ANALYSIS OF NUCLEAR POWER REACTOR HEALTH PHYSICS TECHNICIANS. DAVIS, L. T.; MAZOUR, T. J.; CLARK, P. V.; et al. Analysis & Technology, Inc. August 1984. 200pp. 8410120030. BNL-NUREG-51769. 26980:001.

This report describes a project, an industry-wide Job Analysis of Nuclear Power Reactor Health Physics Technicians (HPTs), sponsored by the Nuclear Regulatory Commission and conducted by Brookhaven National Laboratory and Analysis & Technology, Inc., to provide the industry with job-performance data that can be used in systematically defining training programs in terms of required job functions, responsibilities, and performance standards. The job-analysis methodology is consistent with that used by the Institute of Nuclear Power Operations (INPO) in similar industry-wide projects and includes administration of over 850 job task questionnaires to utility and contractor Health Physics Technicians throughout the country. Data collected includes task performance (difficulty, importance, and frequency) and industry-wide demographics (job levels, experience, education, and training). The results of this project discussed herein include model job descriptions for HPT positions, summaries of HPT experience, education, and training, industry-wide listings with task-performance characteristics, and recommendations of selected tasks as a basis for HPT training development. Finally, potential future applications of the data base by utility and contractor organizations in training program development and evaluation and personnel qualifications are discussed.

NUREG/CR-3751: EFFECTS OF ROCK RIPRAP DESIGN PARAMETERS ON FLOOD PROTECTION COSTS FOR URANIUM TAILINGS IMPOUNDMENTS. ECKER, R. M. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1984. 98pp. 8408130043. PNL-5068. 26011:254.

This report examines the costs of rock riprap flood protection for design flood events at two uranium tailings impoundments in western Colorado. The two sites are the Grand Junction impoundment located along the Colorado River and the Slickrock impoundment located along the Dolores River. The sensitivity of rock type, embankment side slope, and various safety factors is evaluated for six design flood events at Grand Junction and one flood event at Slickrock. The safety factor method of riprap design is used for the cost comparison.

NUREG/CR-3753: AN EVALUATION OF MANUAL ULTRASONIC INSPECTION OF CENTRIFUGALLY CAST STAINLESS STEEL PIPING. TAYLOR, T. T. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 22pp. 8406080265. PNL-5070. 24877:094.

This work was performed as a portion of a NRC research program entitled "Integration of Nondestructive Examination and Fracture Mechanics" (FIN. B2289). The NRC technical monitor is Dr. Joe

Muscara. Two studies have attempted to provide an answer to the degree of inspectability of Centrifugally Cast Stainless Steel (CCSS) pipe. One study was an NRC-sponsored Pipe Inspection Round Robin (PIRR) test conducted at Pacific Northwest Laboratory (PNL). Another study was conducted by Westinghouse. The PNL study reported that less than 30% detection was achieved on thermal fatigue cracks ranging from 5% to 50% through-wall. The Westinghouse study reported that 80% detection was achieved for 20% through-wall mechanical fatigue cracks. A cooperative program between PNL and Westinghouse was conducted to resolve the differences between the two studies. The program was designed as a limited round robin. The data reported here indicates that flaw type (thermal fatigue versus mechanical fatigue) was a significant factor in detection. Mechanical fatigue cracks were more easily detected than thermal fatigue cracks. The data conclusively shows that manual ultrasonic inspection cannot size flaws in cast stainless steel pipe be continued because some failure mechanisms (i.e., mechanical fatigue cracks) have proven to be detectable.

NUREG/CR-3754: FAILURE EVALUATION OF GENERAL ELECTRIC SB-1 AND SB-9 REACTOR MODE SWITCHES. BACANSKAS, V.P. Franklin Institute/Franklin Research Center. April 1984. 37pp. 8405220046. F-C5896-002. 24559:250.

As a result of reactor mode switch malfunctions at operating nuclear power plants (IE Information Notice 83-42), the NRC requested that the Franklin Research Center perform a failure evaluation of the GE SB-1 reactor mode switch.

The objectives of the program were to identify the failure mechanisms for the SB-1 switch and determine if the failure mechanisms were the result of age-related conditions, defects of a particular switch, or design. In addition, the vendor proposed SB-9 replacement switch was evaluated for susceptibility to similar failure mechanisms.

The SB-1 reactor mode switch that malfunctioned at Quad Cities Unit 1 was evaluated along with new SB-1 and SB-9 switches.

The SB-1 reactor mode switch malfunctions were most probably the result of the switch being placed in a false detent position (an intermediate switch position just prior to the actual deten position) which allowed several of the contacts required to be closed to remain open. The false detent noted in the SB-1 switch operation is a result of the indexer mechanism design and not age-related conditions or a defect of a particular switch. The indexer mechanism for the SB-9 switch is of a different design and is not susceptible to a similar failure mechanism.

NUREG/CR-3755: STRONG GROUND MOTION STUDIES FOR SOUTH CAROLINA EARTHQUAKES. NUTTLI, O.W.; RODRIGUES, R.; HERRMANN, R.B.; et al. St. Louis Univ., St. Louis, MO. April 1984. 96pp. 8405020377. UCRL-15594. 24297:026.

This report is concerned with estimating the strong ground motion that will result from damaging earthquakes that occur in South Carolina, varying in size from those that can produce only minor damage to those as large as the 1886 event. The report is divided into three parts. Part I discusses acceleration, velocity and displacement modeling, using available observational data (accelerograms and non-strong motion seismographic) and response spectra obtained from those data. Part II uses MM intensity data for estimating strong ground motion. Part III surface-wave focal mechanism studies of South Carolina earthquakes.

NUREG/CR-3756: SEISMIC HAZARD CHARACTERIZATION OF THE EASTERN UNITED STATES: Methodology And Interim Results For Ten Sites. BERNREUTER, D. L.; SAVY, J. B.; MENSING, R. W.; et al. Lawrence Livermore National Laboratory. April 1984. 542pp. 8405220095. UCRL-53527. 24554:354.

The EUS Seismic Hazard Characterization Project (SHC) is the outgrowth of an earlier study performed as part of the U.S. Nuclear Regulatory Commission's (NRC) Systematic Evaluation Program (SEP). The objectives of the SHC are: 1) to develop a seismic hazard characterization methodology for the region east of the Rocky Mountains; and 2) the application of the methodology to ten sites to assist the NRC staff in their assessment of the implications in the clarification of the U.S. Geological Survey (USGS) position on the Charleston earthquake.

As in the SEP, the fundamental characteristics of the methodology used in SHC consists in using expert opinions for all the input data. The most important improvement over the methodology used in the SEP led to an estimate of the distribution of the hazard rather than just point estimates. An important aspect of eliciting expert opinion consists in holding feedback meetings in order to fine tune the methodology and the input data. At this point, the feedback process has not been completed. Our methodology and preliminary input from the expert panels is presented. Estimates of the hazard (PGA and spectral velocity) at ten representative sites are discussed including a sensitivity analysis and a comparison with the SEP results at four sites.

NUREG/CR-3758: CROSSHOLE GEOPHYSICAL METHODS USED TO INVESTIGATE THE NEAR VICINITY OF HIGH LEVEL WASTE REPOSITORIES. RAMIREZ, A. L.; LYTLE, R. J.; HARBEN, P. Lawrence Livermore National Laboratory. August 1984. 76pp. 8409110090. UCID-20060. 26443:290.

An evaluation is given of remote-probing geophysical techniques likely to be used to investigate the near vicinity of geologic repositories for nuclear waste. The sensors to be used would be placed inside the boreholes, shafts and tunnels of the repository to provide high resolution information of the rock near the repository. The geophysical methods evaluated are known as active methods because they make use of artificial seismic, electric or electromagnetic fields to probe rock mass. Techniques involving through transmission measurements are emphasized. These techniques show merit for remote detection of geological heterogeneities such as fracture zones which influence the containment capacity of repository sites. The report discusses the results obtained with exploration methods used at a site near Oracle, Arizona.

NUREG/CR-3759: LIGHTNING STRIKE DENSITY FOR THE CONTIGUOUS UNITED STATES FROM THUNDERSTORM DURATION RECORDS. MACGORMAN, D. R.; MAIER, M. W.; RUST, W. D. Commerce, Dept. of, Natl. Oceanographic & Atmospheric Administration. May 1984. 52pp. 8406010535. 24764:227.

An improved lightning ground strike climatology has been obtained from thunderstorm duration data recorded by 450 air weather stations. From lightning strike location data collected in Florida and Oklahoma, it was found that strike density could be estimated from thunderstorm duration by the equation N(s) = 0.054H(1.1), where N(s) is the number of strikes per square kilometer and H is thunderstorm duration in hours. This relationship was applied to thunderstorm duration data from the aviation stations to obtain lightning strike density for the contiguous United States.

NUREG/CR-3761: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR THE OCONEE-1 PRESSURIZED WATER REACTOR. FLETCHER, C. D.; BOLANDER, M. A.; STITT, B. D.; et al. EG&G, Inc. July 1984. 165pp. 8408160248. EGG-2310. 26125:001.

Using the RELAP5 computer code, engineers at the Idaho National Engineering Laboratory (INEL) performed thermal-hydraulic analyses of pressurized thermal shock sequences for the Oconee-1 pressurized water reactor. This report summarizes the results of previously reported calculations and presents the results of more recently completed calculations. Comparisons of two counterpart calculations performed, using the RELAP5 code at the INEL and the TRAC code at Los Alamos National Laboratory (LANL), are included as appendices. The results of these thermal-hydraulic analyses will serve as boundary conditions for fracture-mechanics calculations which are to be performed at Oak Ridge National Laboratory.

NUREG/CR-3762: IDENTIFICATION OF EQUIPMENT AND COMPONENTS PREDICTED AS SIGNIFICANT CONTRIBUTORS TO SEVERE CORE DAMAGE. HEISELMANN, H. W. EG&G, Inc. May 1984. 60pp. 8405290430. EGG-2311. 24696:106.

The Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, sponsored the Equipment Qualification Research Program which performed a survey of applicable severe accident study reports to aid in focusing the program efforts. The objective of the survey was to identify, where possible, equipment and components that have been predicted to be significant contributors to high probability accident sequence resulting in severe core damage. A survey of the results of the survey is presented in Tables 1 and 2 of this report. Future updates of this report are anticipated as applicable risk study reports become available.

NUREG/CR-3763: REVIEW AND ASSESSMENT OF RADIONUCLIDE SORPTION INFORMATION FOR THE BASALT WASTE ISOLATION PROJECT SITE (1979 Through May, 1983). KELMERS, A. D. Oak Ridge National Laboratory. September 1984. 47pp. 8410180132. ORNL/TM-9157. 27045:041.

This document presents a scientific review and technological assessment of the radionuclide sorption information reported by the Basalt Waste Isolation Project (BWIP) for the candidate high-level waste repository in the Columbia River basalt flows in the Hanford Reservation. Quantified radionuclide sorption data are necessary for repository performance assessment to model expected radioactivity release rates in groundwater-intrusion-groundwater-migration scenarios. Three key BWIP reports were identified which contain most of the sorption information for a number of radionuclides with basalt, secondary minerals, or interbed materials. An extended review of these data is presented in this document. The technological assessment identified seven potentially significant deficiencies in the radionuclide sorption information published by BWIP that could lead to questionable or nonconservative radioactivity release calculations. These deficiencies are discussed in the document in detail. Specific additional information needs were also defined and reported.

NUREG/CR-3765: MINET SIMULATION OF A HELICAL COIL SODIUM/WATER STEAM GENERATOR, INCLUDING STRUCTURAL EFFECTS. VAN TUYLE, G. J. Brookhaven National Laboratory. September 1984. 27pp. 8410120010. BNL-NUREG-51766. 26985:338.

A test transient performed at a helical coil sodium-to-water steam generator test facility was simulated using the MINET code. It was determined that correct calculation of the sodium outlet temperature requires representation of heat capacitance of the structure.

NUREG/CR-3766: TESTING OF NUCLEAR GRADE LUBRICANTS AND THEIR EFFECT ON A540 AND A193 B7 BOLTING MATERIALS. CZAJKOWSKI, C. J. Brookhaven National Laboratory. September 1984. 83pp. 8409260637. BNL-NUREG-51767. 26699:322.

An investigation was performed on eleven commonly used lubricants by the nuclear power industry which included EDS analysis of the lubricants, notched-tensile constant extension rate testing of bolting materials with the lubricants, frictional testing of the lubricants and weight loss testing of a bonded solid film lubricant. The report concludes that there is a significant amount of variance in the mechanical properties of common bolting materials, that MoS_2 can hydrolyze to form H_2S at 100 degrees C and cause stress corrosion cracking (SCC) of bolting materials. One of the most significant findings of this report is the observation that both A193 B7 and A540 B24 bolting materials are susceptible to transgranular stress corrosion cracking in demineralized H_2O at 280 degrees C in notched tensile tests.

NUREG/CR-3768: NEW MADRID SEISMOTECTONIC STUDY: Activities During Fiscal Year 1982. BUSCHBACH, T. C. St. Louis Univ., St. Louis, MO. April 1984. 180pp. 8405220039. 24564:035.

The purpose of the New Madrid Seismotectonic Study is to identify the earthquake mechanisms within a 200 mile radius of New Madrid, Missouri.

Fiscal year 1982 marked the beginning of geological and studies aimed at better definition of the east-west trending fault systems -- the Rough Creek and Cottage Grove systems -- and the northwest-trending Ste. Genevieve faulting. A prime objective is to determine the nature and history of faulting and to establish the relationship with that faulting and the northeast-trending faults of the Wabash Valley and New Madrid areas. One question to be answered is whether or not the 38th-Parallel Lineament decouples the structural features to the north from those south of the lineament.

There were 222 earthquakes located by the Saint Louis University microearthquake network in 1982. In addition, an earthquake swarm occurred in north-central Arkansas, and more than 17,000 events were recorded there during the year.

A seismic surveying program in the Wabash Valley area was completed in 1982, and the acquired data are being processed. Early interpretations suggest that there is a trough filled with bedded units that are apparently pre-Mt. Simon sediments or volcanics.

Studies of recent fault movement suggest that there may have been some Post-Pleistocene movement along the Kentucky River Fault Zone but none along the Shawneetown, Illinois Fault Zone.

Researchers at Washington University postulate the existence of a Precambrian rift extending northwest-southeast through the state of Missouri -- and beyond -- based on subtle gravity anomaly patterns and digital image processing.

NUREG/CR-3769: DESCRIPTION AND SIGNIFICANCE OF THE GRAVITY FIELD IN THE REELFOOT LAKE REGION OF NORTHWEST TENNESSEE. STEARNS, R. G. ; TOWE, S. K. ; HAGEE, V. L. ; et al. Vanderbilt Univ., Nashville, TN. April 1984. 49pp. 8405020505. 24298:124.

Gravity surveys at various levels of detail have been made at approximately 1200 stations in the Reelfoot Lake region of northwest Tennessee and adjacent portions of Missouri and Arkansas. Individual features were surveyed in detail. At Reelfoot Scarp, six lines of stations having a 100-500 feet spacing with close elevation control were measured. Anomalies on these lines are caused by near-surface geology (faulting, clay-filled channel of abandoned course of Mississippi River). A survey of less accuracy discovered an anomaly along a fault at Henning in the Ripley South Quadrangle.

In the Reelfoot Lake Region the area of abundant earthquake occurrence is related to the gravity anomaly pattern. The earthquake area is sharply limited on the South by an abrupt change in anomaly trends, and the earthquakes diminish in number at a similar change in trend to the north. Some positive gravity anomalies appear to mark plutons where they coincide with positive magnetic anomalies.

Gravity is useful in the region as a main component in a combined geophysical search for faults. The search at Henning was successful, using gravity, combined with earth resistivity, as the main search technique.

NUREG/CR-3771: VESSEL V-7 AND V-8 REPAIR AND CHARACTERIZATION OF INSERT MATERIAL. DOMIAN, H. A. Babcock & Wilcox Co. * Oak Ridge National Laboratory. May 1984. 101pp. 8407020266. 25279:029.

Pieces of Type SA508-2 steel, specially tempered to produce a high-impact-transition temperature, were welded in the side walls of Intermediate Test Vessels V-7 and V-8. These vessels are to be tested by the Oak Ridge National Laboratory (ORNL) in the Pressurized-Thermal-Shock (PTS) Project of the Heavy-Section Steel Technology (HSST) Program.

A comparable piece of forging taken from the same source and heat treated with the vessels was characterized for its mechanical properties to provide data for use in the PTS tests.

NUREG/CR-3773: VARIATION OF PLANETARY BOUNDARY LAYER DISPERSION PROPERTIES WITH HEIGHT IN UNSTABLE CONDITIONS. HICKS, B. B. Commerce, Dept. of, Natl. Oceanographic & Atmospheric Administration. May 1984. 50pp. 8406190078. 25029:238.

Recent developments in surface boundary layer and planetary boundary layer meteorology are combined to evaluate the height dependency of the dispersion parameters standard deviation z and standard deviation y of the familiar Gaussian plume relationships. Recommendations are based on analyses of surface boundary layer data, such as are collected at industrial sites under existing NRC guidelines.

NUREG/CR-3774 V01: ALTERNATIVE METHODS FOR DISPOSAL OF LOW-LEVEL RADIOACTIVE WASTES. Task 1: Description of Methods And Assessment Of Criteria. BENNETT, R. D. ; MILLER, W. O. ; WARRINER, J. B. ; et al. Army, Dept. of, Army Engineer Waterways Experiment Station. April 1984. 82pp. 8405220068. 24559:289.

The study reported herein contains the results of Task 1 of a four-task study entitled "Criteria for Evaluating Engineered Facilities." The overall objective of this study is to ensure that the criteria needed to evaluate five alternative low-level radioacive waste (LLW) disposal methods are available to the Nuclear Regulatory Commission (NRC) and the Agreement States. The alternative methods considered are belowground vaults, aboveground vaults, earthmounded concrete bunkers, mined cavities, and augered holes. Each of these alternatives is either being used by other countries for low-level radioactive waste (LLW) disposal or is being considered by other countries or U.S. agencies.

In this report the performance requirements are listed, each alternative is described, the experience gained with its use is discussed, and the performance capabilities of each method are addressed. Next, the existing 10 CFR Part 61 Subpart D criteria with respect to paragraphs 61.50 through 61.53, pertaining to site suitability, design, operations and closure, and monitoring are assessed for applicability to evaluation of each alternative. Preliminary conclusions and recommendations are offered on each method's suitability as an LLW disposal alternative, the applicability of the criteria, and the need for supplemental or modified criteria.

NUREG/CR-3775: QUALITY ASSURANCE FOR MEASUREMENTS OF IONIZING RADIATION. EISENHOWER,E.H. Commerce, Dept. of, National Bureau of Standards. June 1984. 163pp. 8407170566. 25632:001.

This report describes results of a three-year program that will enable the Nuclear Regulatory Commission to improve, demonstrate, and document traceability of its measurements to the national physical measurement standards for ionizing radiation. The principal actions taken were: (a) characterization of the response of a thermoluminescence dosimetry system used for routine surveillance of nuclear facilities; (b) characterization of the response of six models of portable survey instruments; and (c) implementation of routine quality assurance services that will demonstrate that laboratories which calibrate survey instruments for the NRC are sufficiently consistent (in agreement) with national measurement standards. Tests of the TLD system were performed as specified in American National Standard N545-1975, plus several additional tests not contained in that document. Measurement assurance tests were conducted for the NRC Region-1 laboratory. The response of the survey instruments was determined for photon energies as high as 6.5 MeV, and for beta particles of various energies, including those emitted by (133)Xe gas. The basic principles under which the long-range interactive MQA program will operate were developed and documented, and the feasibility of the program was demonstrated.

NUREG/CR-3776: TESTING OF SAFETY-RELATED NUCLEAR POWER PLANT EQUIPMENT AT THE CENTRAL RECEIVER TEST FACILITY. DANDINI,V.J.; ARAGON,J.J. Sandia Laboratories. July 1984. 86pp. 8409110074. SAND83-1960. 26437:198.

The use of a solar energy facility for simulating the thermal environment (heat flux) produced as a result of hydrogen burns in a full-scale reactor containment building is described. Using a heat flux profile developed from calculations performed by the HECTR computer code, the Central Receiver Test Facility simulated the multiple burn thermal environment which HECTR predicted would result from the deliberate ignition of hydrogen generated by an S2D accident. Functioning specimens of reactor monitoring and safety system equipment were exposed to this environment. Results of the equipment performance and temperature response are presented.

NUREG/CR-3777: CAPABILITIES AND DIAGNOSTICS OF THE SANDIA PELLETRON-RASTER SYSTEM. BUCKALEW, W. H. ; LOCKWOOD, G. J. ; LUKER, S. M. ; et al. Sandia Laboratories. July 1984. 61pp. 8408080367. SAND84-0912. 25978:090.

The radiation capabilities of the PELLETRON Electron Beam Accelerator have been expanded to include a controllable, variable dimension, beam diffusion option. This rastered beam option has been studied in detail. Beam characteristics have been determined as a function of incident electron beam energy, current, and deflection system parameters. The beam diagnostics required to define any given diffuse beam pattern are accurate and predictable. Recently, utility of this added PELLETRON capability was demonstrated by simulating the effects of complex nuclear reactor accident electron environments on electrical insulation materials similar to those used in nuclear power plants.

NUREG/CR-3779 V01: THE HYDROGEN BURN -EQUIPMENT RESPONSE ALGORITHM (HYBER). Users Guide. KING, D. B. ; RATZEL, A. C. ; KEMPKA, S. N. ; et al. Sandia Laboratories. August 1984. 136pp. 8411160085. SAND84-0160. 27565:116

The users guide for the Hydrogen Burn Survival algorithm, HYBER, is provided in this report. HYBER (Hydrogen Burn - Equipment Response) is comprised of two self-contained computer programs, DATGEN and SOLVER. These computer codes may be used to estimate the thermal response of safety-related equipment exposed to hydrogen combustion in nuclear reactor containments. The state and composition gases are obtained by modeling single or multiple deflagrations in mixtures of hydrogen, steam, carbon monoxide, and air. This users manual describes the interactive generation of the input data files using DATGEN and discusses the execution of SOLVER. Special usage of the algorithm on IBM Personal Computer (PC) and IBM compatible computer systems is also described. Illustrative examples, including the input data files, are provided to demonstrate some of the capabilities of the algorithm.

NUREG/CR-3779 V02: THE HYDROGEN BURN - EQUIPMENT RESPONSE ALGORITHM (HYBER) Reference Manual. KEMPKA, S. N. ; RATZEL, A. C. Sandia Laboratories. August 1984. 89pp. 8411160083. SAND83-2579. 27565:024.

The reference guide for HYBER (Hydrogen Burn - Equipment Response), the Hydrogen Burn Survival algorithm, is provided in this report. HYBER is comprised of two self-contained computer programs, DATGEN and SOLVER, developed on VAX 11/780 and CRAY-1 computer systems for use on IBM Personal Computers. These computer codes model single or multiple hydrogen:carbon monoxide:air combustion processes in single-volume vessels, providing predictions for the environment gase state and composition. HYBER was developed to model combustion processes in nuclear reactor containments and to estimate the thermal response of safety-related equipment subject to the combustion and post-combustion environments. This reference guide discusses the combustion, heat, and mass transfer models included in the algorithm computer codes. The assumptions used in the codes and the execution procedure (i.e., computational framework) in SOLVER are provided. HYBER simulations of two experiments are compared with the data for two different vessel (5.6m(3) and 2084.m(3) total volume, respectively). These comparisons demonstrate the capabilities of modeling combustion processes with HYBER.

NUREG/CR-3780: FUEL DISRUPTION MECHANISMS DETERMINED IN-PILE IN THE ANNULAR CORE RESEARCH REACTOR (ACRR). WRIGHT, S. A.; FISCHER, E. A. Sandia Laboratories. October 1984. 50pp. 8501030021. SAND83-1750. 28200:142.

Over thirty in-pile experiments were performed to investigate fuel disruption behavior for LMFBR loss of flow (LOF) accidents. These experiments reproduced the heating transients for a variety of accidents ranging from slow LOF accidents to rapid LOF-driven-TOP accidents. In all experiments the timing and mode of the fuel disruption were observed with a high speed camera, enabling detailed comparisons with a fuel pin code, SANDFIN, which models transient intra- and inter-granular fission gas behavior to predict the macroscopic fuel behavior, such as fission gas induced swelling and frothing, cracking and breakup of solid fuel, and fuel vapor pressure driven dispersal. This report reviews the different modes of fuel disruption as seen in the experiments and then describes the mechanism responsible for the disruption. An analysis is presented that describes a set of conditions specifying the mode of fuel disruption and the heating conditions required to produce the disruption. The heating conditions are described in terms of heating rate (K/s), temperature gradient, and fuel temperature. A fuel disruption map is presented which plots heating rate as a function of fuel temperature to illustrate the different criteria for disruption. Although this approach to describing fuel disruption oversimplifies the fission gas processes modeled by SANDPIN, it does illustrate the criteria used to determine which fuel disruption mechanism is dominant and on what major fission gas parameters it depends.

NUREG/CR-3781 DRFT: PCT-RELATED CLADDING FAILURES DURING OFF-NORMAL EVENTS-DRAFT: Draft Report Of The USNRC PCI Review Group. MACDONALD, P. E. EG&G, Inc. TOKAR, M.; VAN HOUTEN, R. NRC - No Detailed Affiliation Given. June 1984. 112pp. 8407020360. EGG-2313. 25274:148.

Because fuel failure estimates are used as input to radiological dose calculations, the U.S. Nuclear Regulatory Commission has formed a task force of fuel behavior experts to study pellet-cladding interaction (PCI), due to concerns that existing rod overheating criteria might be inadequate for evaluating transient severity in this regard. This report includes preliminary findings for reactor events of the type addressed by Chapter 15 of the NRC Standard Review Plan. Specifically, the BWR turbine trip without bypass, PWR control rod withdrawal error, subcritical PWR control rod withdrawal error, BWR control blade withdrawal error, and the PWR steamline break are analyzed on the joint bases of peak rod power, power increase, ramp rate, and duration at elevated power. These Chapter 15 events are compared to numerous test reactor results and to other relevant investigations, and tentative conclusions on transient severity and data base adequacy are presented. Progress in developing computer codes for predicting PCI-induced fuel rod failure is also discussed.

NUREG/CR-3782: GEOLOGIC AND HYDROLOGIC RESEARCH AT WESTERN NEW YORK NUCLEAR SERVICE CENTER, WEST VALLEY, NEW YORK. Final Report, August 1982 - December 1983. ALBANESE, J. R.; ANDERSON, S. L.; FAKUNDINY, R. H.; et al. New York, State of. June 1984. 437pp. 8409260641. 26698:163.

This report is the last in a series by the New York State Geological Survey on studies funded by the U.S. Nuclear Regulatory Commission. The report covers five important aspects of the geology and hydrology of the Western New York Nuclear Service Center, near

West Valley, New York: geomorphogy, stratigraphy, sedimentology, surface water, and radionuclide analyses. We reviewed past research on these subjects and present new data obtained in the final phase of NYSGS research at the site. Also presented are up-to-date summaries of the present knowledge of geomorphology and stratigraphy. The report contains a significant bibliography of previous West Valley studies. Appendices include a report on the Fall 1983 Drilling Project and the procedures used, history and prognosis of Cattaraugus Creek and tributaries down cutting, and bar modification and landslide processes of Buttermilk Valley.

NUREG/CR-3784: LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM. Semiannual Report, April-September 1983. BERMAN, M. Sandia Laboratories. October 1984. 250pp. 8412190369. SAND84-0689. 28016:052.

This report describes the investigations and analyses conducted at Sandia National Laboratories, Albuquerque, in support of the Light Water Reactor Safety Research Program from April 1983 through September 1983. The Molten Fuel/Concrete Interactions (MFCI) Study investigates the mechanism of concrete erosion by molten core materials, the nature and rate of generation of evolved gases, and the effects of fission-product release. The Core Melt/Coolant Interactions (CMCI) Study investigates the characteristics of explosive and nonexplosive interactions between molten core materials and concrete, and the probabilities and consequences of such interactions. In the Hydrogen Program, the HECTR code for modelling hydrogen deflagration is being conducted, and the Grand Gulf Hydrogen Igniter System II is being reviewed. All activities are continuing.

NUREG/CR-3785: ALTERNATIVE APPROACHES TO PROVIDING ENGINEERING EXPERTISE ON SHIFT. OLSON, J.; SCHREIBER, R. E.; MELBER, B. D. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1984. 61pp. 8406080258. PNL-5087. 24865:181.

This report represents the conclusions of a project studying the role of engineering expertise on shift in nuclear power plants. Using the present shift technical advisor (STA) position as the base case, several alternatives were analyzed. On-shift alternatives include the STA, the shift supervisor (SS), and the shift engineer (SE). The SE is degreed, experienced, trained and licensed as a Senior Reactor Operator. Some non-shift alternatives were also studied. These included a cadre of on-call engineers and specialists within continual contact and easy reach of the plant, a technical system of phone and data lines linking the plant with a facility similar to an on-site technical support center, and a safety parameter display system (SPDS) to augment technical upgrading of operator aids presently available. Potential problems considered in the analysis of implementation of these alternatives included job content constraints, problems of crew acceptance, and problems of labor supply and retention. Of the considered alternatives, the SE and SS options appear superior to the current STA approach. The SE option appears the easiest to implement and the most effective under varied plant conditions. The SE may also serve as liaison to off-site support facilities.

NUREG/CR-3786: A REVIEW OF REGULATORY REQUIREMENTS GOVERNING CONTROL ROOM HABITABILITY SYSTEMS. JACOBUS, M. J. Sandia Laboratories. August 1984. 63pp. 8410120011. SAND84-0978. 26978:201.

This report reviews applicable guides, standards, and codes which govern the design, manufacture, selection, installation, and surveillance practices for components and systems important to control room habitability. It covers the fundamental guidance contained in General Design Criteria, Regulatory Guides, and applicable sections of the Standard Review Plan, as well as numerous documents referenced by this guidance.

Instances are cited where the present guidance is misleading, contradictory, or vague. In some cases, the problems in the guidance result from inadequate technical bases; in other cases, the problems result from several documents which are not completely consistent.

To independently assess the suitability of the regulatory guide which covers accidental chlorine releases, a computer program was developed to calculate chlorine concentrations in the control room following chlorine release. Although problems with the assumptions used to develop the guide were found, the conservative nature of the chlorine calculations appears to adequately compensate for these problems.

NUREG/CR-3787: EFFECTIVENESS OF ENGINEERED SAFETY FEATURE (ESF) SYSTEMS IN RETAINING FISSION PRODUCTS. Background Information. MISHIMA, J.; BLAHNIK, D. E.; HALVERSON, M. A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 115pp. 8409170417. PNL-5101. 26499:206.

The Pacific Northwest Laboratory has compiled and reviewed base line data on the effectiveness of Engineered Safety Feature (ESF) systems in the retention of fission products and particulate material resulting from a nuclear reactor accident. This work is part of an NRC project to provide the best estimates of the consequences of severe reactor accidents.

The resulting report describes the ESF systems (containment spray, secondary containment filter, containment recirculating filter, pressure suppression pool, ice condenser, and main steam line isolation valve leakage control systems). Also described are the anticipated atmospheres in which the ESFs must operate, the experimental studies of ESF system effectiveness, and the models currently available for assessing the performance of the various ESF systems. The information gaps identified as a result of this review have resulted in recommendations for additional work in the areas of: 1) performance data and models of containment chiller/coolers; 2) continued development and experimental verification of the ice condenser model; 3) continued development of the pressure suppression pool model; and 4) continued investigations of the behavior of filtration devices.

NUREG/CR-3788 V01: STRUCTURAL INTEGRITY OF LIGHT WATER REACTOR PRESSURE BOUNDARY COMPONENTS. Four-Year Plan 1984-1988. * Materials Engineering Associates, Inc. September 1984. 111pp. 8410180235. MEA-2047. 27045:088.

This document is the first in a series intended to provide an up-to-date statement of the four-year plan for the program, "Structural Integrity of Light Water Reactor Pressure Boundary Components," which is being conducted by Materials Engineering Associates, Inc. (MEA). This program consists of engineering and research in fracture, fatigue, and radiation sensitivity of nuclear structural steels and weldments and addresses many of the key uncertainties in the margin of safety in operating nuclear plants. All tasks are integrated to focus on structural integrity of LWR pressure boundary components. The approach centers on an experimental characterization of nuclear grade steels and an assessment of fracture and fatigue behavior under conditions of a nuclear environment, so investigation of irradiated materials is a key element of each task. Experimental studies are supported by analytical models and investigation of the mechanisms responsible for the observed behavior. Data developed in the program will provide the basis for recommendations for the ASME Boiler and Pressure Vessel Code and ASTM test methods, and revisions to NRC Guides.

NUREG/CR-3792: CLOSEOUT OF IE BULLETIN 79-11: FAULTY OVERCURRENT TRIP DEVICE IN CIRCUIT BREAKERS FOR ENGINEERED SAFETY SYSTEMS. FOLEY, W. J.; DEAN, R. S.; HENNICK, A. Parameter, Inc. August 1984. 34pp. 8408310084. IEB-79-11. 26348:057.

IE Bulletin 79-11 was issued May 22, 1979 as a result of information received in April 1979 from Westinghouse and an NRC licensee relating to the potential failure of a circuit breaker in an engieered safety system of a nuclear power plant. The defect of concern was a small hairline crack in the dashpot end cap of one of the three overcurrent trip devices of a Type DB-75 breaker. The Bulletin was also applicable to Type DB-50 breakers, because they use the same type of dashpot end cap. The defective end cap had been installed in 1973 as a replacement, in compliance with IE Bulletin 73-1. Westinghouse Technical Bulletin NSD-TB-79-02 was issued April 17, 1979 to alert utilities to the potential problem, to provide background information, to recommend review of calibration test data and retesting of erratic breakers, to advise visual examination of end caps for cracks and to call for replacement of cracked end caps. Evaluation of utility responses and NRC/IE inspection reports shows that 114 of the 129 current facilities do not use the affected breakers in safety-related systems. Followup items for the five facilities with open status are proposed. The Bulletin has been closed out for the remaining ten facilities with safety-related Westinghouse DB-50 and DB-75 breakers having dashpots, on the basis of acceptable utility responses and NRC/IE regional inspection reports. Erratic performance of three DB-50 breakers with worn seals at one facility is identified as a Remaining Area of Concern because the worn seals had essentially the same effect on performance as cracked end caps. The recommendation is made that preventive maintenance programs of licensees be reviewed to make sure that breakers are kept clean to avoid plugging dashpot orifices. The Bulletin has served its purpose by resulting in identification of the potential problem at a limited number (15) of facilities and of the need for corrective actions at only five facilities.

NUREG/CR-3795: CLOSEOUT OF IE BULLETIN 82-04: DEFICIENCIES IN PRIMARY CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES. FOLEY, W. J.; HENNICK, A. Parameter, Inc. July 1984. 53pp. 8408090272. IEB-82-04. 25983:001.

IE Information Notice 82-40 was issued September 22, 1982 as an early notification of a potentially significant problem pertaining to electrical penetration assemblies (EPAs) supplied by the Bunker Ramo Corporation (BRC) of Chatsworth, California. All deficiencies described in the Notice were identified as existing in BRC EPAs with a hard epoxy module design. Utility personnel were asked to review the Notice and take appropriate actions, but were not required to respond or take any specific action. After further study, NRC concluded that there were potential generic safety implications at a limited number of plants. Accordingly, IE Bulletin 82-04 was issued December 3, 1982 to require responses and specific actions by all licensees and holders of construction permits. Evaluation of utility responses, deficiency reports and NRC/IE inspection reports has resulted in Bulletin closeout for 124 of 129 current facilities. Deficiencies described in the Bulletin were identified at all facilities, of which two are operating and nine under construction. Followup of corrective actions and verification of inspection procedures are proposed in Appendix C for the five facilities with affected assemblies are summarized in Table B.6. Completion by NRC/IE of all the followup items identified in Appendix C is expected to resolve fully the specific problem of Bunker Ramo electrical penetrations that utilized a hard epoxy design.

NUREG/CR-3796: EMERGENCY PREPAREDNESS SOURCE TERM DEVELOPMENT FOR THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS LICENSED FACILITIES. SUTTER, S. L.; MISHIMA, J.; BALLINGER, M. Y.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 352pp. 8409040447. PNL-5081. 26363:049.

To establish requirements for emergency preparedness plans at facilities licensed by the Office of Nuclear Materials Safety and Safeguards, the Nuclear Regulatory Commission (NRC) needs to develop source terms (the amount of material made airborne) for accidents. They are used to estimate potential public doses from the events, which will be used to guide whether emergency preparedness plans are needed. Pacific Northwest Laboratory is providing the NRC with source terms by developing accident scenarios for fuel cycle and by-product operations. Several scenarios are developed for each operation, leading to the identification of the maximum release considered for emergency preparedness planning (MREPP) scenario. Fire was the MREPP at oxide fuel fabrication, UF(6) production, radiopharmaceutical manufacturing, radiopharmacy, sealed source manufacturing, waste warehousing, and university research and development facilities. Tornadoes were MREPP events for uranium mills and plutonium contaminated facilities, and criticalities were significant at nonoxide fuel fabrication and nuclear research and development facilities. Techniques for adjusting the MREPP release to different facilities are also described.

NUREG/CR-3797: DIGMAN: A COMPUTER PROGRAM TO ILLUSTRATE THE COMPLEXITIES IN SAMPLING COMMERCIAL LOW-LEVEL WASTE SITES FOR RADIONUCLIDE SPILLS OR MIGRATION. SIMMONS, M. A.; SKALSKI, J. R.; SWANNACK, R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1984. 38pp. 8406040139. PNL-5028. 24805:265.

DIGMAN is an interactive computer program which allows the user to sample a hypothetical waste site. Using sample results, the user is then required to determine the area contaminated by a waste spill or migration. The report contains instructions for running the program and a sample session to aid the novice user. DIGMAN is programmed for an Apple II computer with a minimum of 64K RAM and one disk drive. A disk containing a copy of the program is available from the authors.

NUREG/CR-3798: CHARACTERIZATION OF CEMENT AND BITUMEN WASTE FORMS CONTAINING SIMULATED LOW-LEVEL WASTE INCINERATOR ASH. WESTSIK, J. H.; BUSCHBOM, R. L.; DIVINE, J. R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 100pp. 8408310075. PNL-5153. 26347:319.

Incinerator ash from the combustion of general trash and ion exchange resins were immobilized in cement and bitumen. Tests were conducted on the r sulting waste forms to provide a data base for the acceptability of actual low-level waste forms. The testing was done in accordance with the Technical Position on Waste Form. Bitumen had a measured compressive strength of 120 psi and a leashability index of 13 as measured with the ANS 16.1 leach test procedure. Cement demonstrated a compressive strength of 1400 psi and a leachability index of 7. Both waste forms easily exceed the minimum compressive strength of 50 psi and leachability index of 6 specified in the Technical Position. Irradiation of 10(8) RAD and exposure to thirty +60 degrees to -30 degrees centigrade thermal cycles did not significantly impact these properties. Neither waste form supported bacterial or fungal grouth as measured with ASTM G21 and G22 procedures. Neither bitumen nor cement containing incinerator ash caused any corrosion or degradation of potential container materials including steel, polyethlyene and fiberglass. However, moist ash did cause corrosion of the steel.

NUREG/CR-3800: REFCO-83 USER'S MANUAL. DELENE, J. G. ; HERMANN, O. W. Oak Ridge National Laboratory. June 1984. 76pp. 8407110018. ORNL/TM-9186. 25547:087.

The computer code REFCO-83 utilizes a discounted cash flow (DCF) analysis procedure to calculate batch, cycle, and lifetime levelized nuclear fuel cycle costs. This code is an updated version of the REFCO computer code originally written in the early 1970s. The basic methodology and procedures were retained; however, extensive modifications were made to the input and data handling procedures. Several computational procedures were updated to make the code more versatile and to simulate recent events such as the provisions of the Nuclear Waste Policy Act of 1982.

This report is a user's guide for the revised REFCO code. It contains a description of the code methodology, a cost data base, a discussion of the general code structure, the code input instructions, and sample cases.

NUREG/CR-3804 V01: PHYSICS OF REACTOR SAFETY. Quarterly Report, January - March 1984. * Argonne National Laboratory. July 1984. 1p. 8408080374. ANL-84-35. 25979:356.

This quarterly progress report summarizes work done during the months of January-March 1984 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research in the U. S. Nuclear Regulatory Commission. The work in the Applied Physics Division includes reports on reactor safety modeling and assessment by members of the Reactor Safety Appraisals Section. Work on reactor core thermal-hydraulics is performed in ANL's Components Technology Division, emphasizing 3-dimensional code development for LMFBR accidents under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-3804 V02: PHYSICS OF REACTOR SAFETY. Quarterly Report, April-June 1984. * Argonne National Laboratory. October 1984. 32pp. 8411130562. ANL-84-35. 27476:293.

This quarterly progress report summarizes work done during the months of April-June 1984 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research in the U. S. Nuclear Regulatory Commission. The work in the Applied Physics Division includes reports on reactor safety modeling and assessment by members of the Reactor Safety Appraisals Section. Work on reactor core thermal-hydraulics is performed in ANL's Components Technology Division, emphasizing 3-dimensional code development for LMFBR accidents under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-3805: ENGINEERING CHARACTERIZATION OF GROUND MOTION. Task I: Effects Of Characteristics Of Free-Field Motion On Structural Response. KENNEDY, R. P.; SHORT, S. A.; MERZ, K. L.; et al. Structural Mechanics Associates. May 1984. 389pp. 8406210448. 25098:044.

This report presents the results of the first task of a two-task study on the engineering characterization of earthquake ground motion for nuclear power plant design. The overall objective of this study is to develop recommendations for methods for selecting design response spectra or acceleration time histories to be used to characterize motion at the foundation level of nuclear power plants. Task I of the study, presented herein, develops a basis for selecting design response spectra, taking into account the characteristics of free-field ground motion found to be significant in causing structural damage. Task II of the study, to be completed later in 1984, will provide recommendations for methods for selecting response spectra and time histories incorporating wave passage and soil-structure interaction effects and Task I results.

NUREG/CR-3806: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS: Annual Report, October 1982 - September 1983. SHACK, W. J.; KASSNER, T. F.; KUPPERMAN, D. S.; et al. Argonne National Laboratory. August 1984. 143pp. 8410030340. ANL-84-36. 26820:086.

This progress report summarizes work on environmentally assisted cracking in lightwater reactors during the twelve months from October 1982 - September 1983. The objective of this program is to develop an independent capability for prediction, detection, and control of intergranular stress corrosion cracking (IGSCC) in lightwater reactor (LWR) systems. The program is primarily directed at IGSCC problems in existing plants, but also includes the development of recommendations for plants under construction and future plants. The scope includes the following: (1) development of the means to evaluate acoustic leak detection systems objectively and quantitatively; (2) evaluation of the influence of metallurgical variables, stress, and the environment on IGSCC susceptibility, including the influence of plant operations on these variables; and (3) examination of practical limits for these variables to effectively control IGSCC in LWR systems. The initial experimental work concentrates primarily on problems related to pipe cracking in BWR systems. However, ongoing research work on other environmentally assisted cracking problems involving pressure vessels, nozzles, and turbines will be monitored and assessed, and where unanswered technical questions are identified, experimental programs to obtain the necessary information will be developed to the extent that available resources permit.

NUREG/CR-3808: AGING-SEISMIC CORRELATION STUDY ON CLASS 1E EQUIPMENT. SUGARMAN, A. C. NUTECH Engineers, Inc. * Sandia Laboratories. October 1984. 74pp. 8411130674. SAND84-7135. 27474:317.

This paper presents a new method of analysis for evaluating the effect of aging in electrical equipment on the seismic capacity. The method is based on the probability of mechanical failure of weak link materials which may be subjected to a load during an earthquake. It is shown that aging-seismic correlation is related to: number of age-degradable weak link components, rate of degradation in weak link components, the seismic stresses on the components and component material failure. Before conducting the Probabilistic Failure Analysis, preliminary screening for equipment with potential aging-seismic correlation is performed.

NUREG/CR-3809 V01: PERFORMANCE TESTING OF RADIOBIOASSAY LABORATORIES: IN-VITRO MEASUREMENTS, PILOT STUDY REPORT. ROBINSON, A. V.; FISHER, D. R.; HANDLEY, R. T. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1984. 81pp. 8501070416. DOE/NBM1701. 28222:242.

The research program at the Pacific Northwest Laboratory entitled "Technical Evaluation of Draft ANSI Standard N13.30, Performance Criteria for Radiobioassay" is jointly sponsored by the Nuclear Regulatory Commission and the Department of Energy. It is a nationwide, two-round bioassay intercomparison study to test the analytical performance of both in-vitro and in-vivo bioassay laboratories and determine their ability to meet the minimum performance criteria specified in the draft ANSI Standard. Round One is a Pilot Study involving a small number of voluntarily participating laboratories. Round Two will involve a larger number of laboratories and will expand on the results of Round One. This report, In-Vitro Measurements, is a review of the methodology and results of Round One. For this part of the research, test samples of artificial urine containing precisely known quantities of certain radionuclides were sent to 19 bioassay laboratories. Results show that some of the participating laboratories had difficulty meeting the performance criteria specified in the current draft ANSI Standard N13.30. Based on these results, specific recommendations were made to the working group preparing the draft Standard.

NUREG/CR-3810 V01: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report, January-March 1984. EDLER, S. K. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1984. 35pp. 8407180005. PNL-5106-1. 25665:287.

This document summarizes work performed by Pacific Northwest Laboratory from January 1 through March 31, 1984, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Results from an instrumental fuel assembly irradiation program being performed at Halden, Norway, are reported. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada.

NUREG/CR-3810 V02: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report, April-June 1984. EDLER, S. K. Battelle Memorial Institute, Pacific Northwest Laboratories. November 1984. 32pp. 8412120581. PNL-5106-2. 27913:332.

This document summarizes work performed by Pacific Northwest Laboratory from April 1 through June 30, 1984, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Results from an instrumented fuel assembly irradiation program being performed at Halden, Norway, are reported. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada.

NUREG/CR-3811: ALTERNATE PROCEDURES FOR THE SEISMIC ANALYSIS OF MULTIPLY SUPPORTED PIPING SYSTEMS. SUBUDHI, M.; BEZLER, P.; WANG, Y. K.; et al. Brookhaven National Laboratory. October 1984. 326pp. 8411130582. BNL-NUREG-51773. 27473:046.

Independent support motion methodologies have been used to analyze piping systems subjected to multiple support excitations. Methods to compute both the dynamic and pseudo-static components of response were investigated. In order to formulate a general procedure for predicting seismic response, a sample of six piping systems, two of which were subjected to thirty-three earthquakes, were analyzed. The dynamic component of response was evaluated considering fourteen variations of the combination sequence and procedure between modes, directions and support groups. The pseudo-static component of response was predicted using five different methods constituting nine different cases. In addition, a combination procedure between the two response components was developed in order to obtain the total seismic response. The study also provides a comparison of the above results with time history results as well as with results developed using the current SRP methodology. Recommendations concerning the use of independent support motion methods in the evaluation of piping response are included.

NUREG/CR-3812: ASSESSMENT OF IRRADIATION EFFECTS IN RADWASTE CONTAINING ORGANIC ION-EXCHANGE MEDIA. SWYLER, K. J.; DODGE, C. J.; DAYAL, R. Brookhaven National Laboratory. September 1984. 82pp. 8410030353. BNL-NUREG-51774. 26820:001.

Recently, regulatory consideration has been devoted to the effects of self-irradiation on radwaste containing organic ion exchange media. This consideration was prompted by decontamination operations at TMI-II, and by the development of technical positions in support of NRC regulation 10 CFR 60. This report addresses the effects of high radiation dose on the storage and disposal of radwaste ion-exchange media, and the validity of laboratory test procedures for predicting field performance. Our work shows that accelerated testing of ion-exchange media using high-dose-rate external gamma irradiation appears to be a valid procedure for assessing certain aspects of field behavior--i. e., radiolytic scission of the resin functional group, radiolytic gas generation of free liquids and resin agglomeration, provided both the test data and the field conditions refer to storage in a closed environment. Certain resin decomposition processes appear to depend largely on resin moisture content, and may not be particularly sensitive to resin loading. One practical consequence of radiolytic acidity is to promote the corrosion of mild steel in irradiated resin. However, the corrosion process is very complex. Case-specific, long-tern (i. e., low radiation dose) evaluations might be necessary if rigorous guidelines to protect radwaste containers against corrosion are required.

NUREG/CR-3813: MINET VALIDATION STUDY USING STEAM GENERATOR TRANSIENT DATA. VAN TUYLE, G. J. Brookhaven National Laboratory. September 1984. 39pp. 8410120057. BNL-NUREG-51775. 26986:100.

Three steam generator transient test cases, that were simulated using the MINET computer code, are described, with computed results compared against experimental data. The MINET calculations closely agreed with the experiment for both the once-through and the U-tube steam generator test cases. The effort is part of an ongoing effort to validate the MINET computer code for thermal-hydraulic plant systems transient analysis, and strongly supports the validity of the MINET models.

NUREG/CR-3814: DETERMINATION OF DAMAGE EXPOSURE PARAMETER VALUES IN THE PSF METALLURGICAL IRRADIATION EXPERIMENT. STALLMAN, F. W. Oak Ridge National Laboratory. October 1984. 40pp. 8411280229. ORNL/TM-9166. 27681:313.

Values for the damage exposure parameters fluence > 1.0 MeV, fluence > 0.1 MeV, and dpa were determined for all locations of metallurgical specimens in the test assembly of the ORR-PSF irradiation experiment. Determination is based on dosimetry measurements by HEDL and the fluence calculations by R. E. Maerker and B. A. Worley at ORNL. The LSL-M2 adjustment procedure was used. The space dependency of the damage parameter values can be presented as a cosine-exponential function. Uncertainties are between 5 and 10%.

NUREG/CR-3815: STATISTICAL EVALUATION OF THE METALLURGICAL TEST DATA IN THE ORR-PSF-PVS IRRADIATION EXPERIMENT. STALLMAN, F. W. Oak Ridge National Laboratory. August 1984. 34pp. 8409170414. ORNL/TM-9207. 26498:164.

A statistical analysis of Charpy test results of the two-year Pressure Vessel Simulation metallurgical irradiation experiment was performed. determination of transition temperature and upper shelf energy derived from computer fits compare well with eyeball fits. Uncertainties for all results can be obtained with computer fits. The results were compared with predictions in Regulatory Guide 1.99 and other irradiation damage models.

NUREG/CR-3818: REPORT OF RESULTS OF NUCLEAR POWER PLANT AGING WORKSHOPS. CLARK, N. H.; BERRY, D. L. Sandia Laboratories. August 1984. 64pp. 8408240320. SAND-84-0374. 26237:284.

Two workshops were conducted to identify whether there is any evidence of component or structural aging problems in nuclear power plants, and, if so, what problems are of greatest importance. Fifteen representatives from national laboratories, architect/engineers, nuclear steam supply system vendors, research firms, and a university participated in the workshops. Based on completed questionnaires and group discussions which screened over 112 components believed to be susceptible to excessive aging, pressure/temperature sensors, valve operators, and snubbers emerged by consensus as the most important aging issues. Potential aging problems related to off-normal common mode effects or aging problems which are just now developing were found to be outside the scope of the workshops, because little or no first hand experience is available for these off-normal or yet to develop circumstances. Recommendations are made for a systematic approach to rate components in terms of overall safety and for a cooperative effort between industry research groups and regulatory research groups to resolve known aging problems and to identify off-normal or yet to develop aging issues.

NUREG/CR-3820 V01: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM. Quarterly Report, January-March 1984. THOMPSON, S. L. Sandia Laboratories. July 1984. 62pp. 8408100147. SAND84-1025/1. 25996:280.

The TRAC-PF1/MOD1 independent assessment program is part of a multi-faceted effort sponsored by the Nuclear Regulatory Commission (NRC) to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. This program is a successor to the RELAP5/MOD1 independent assessment project underway at Sandia for the last two years.

The first quarter of FY84 marked the beginning of the TRAC-PF1/MOD1 independent assessment project at SNLA. The code was obtained from Los Alamos National Laboratory (LANL) in October, and brought up on both our Cyber-76 and Cray-1S computers. The assessment matrix was formalized, several TRAC nodalizations for the various facilities required were developed, and limited calculations were begun, all described in the last quarterly. During this quarter, more nodalizations were developed and calculations begun, and the first PF1/MOD1 assessment analysis was completed.

NUREG/CR-3820 V02: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM QUARTERLY REPORT, APRIL-JUNE 1984. THOMPSON, S. L. Sandia Laboratories. October 1984. 57pp. 8411140102. SAND84-1025/2. 27526:073.

The TRAC-PF1/MOD1 independent assessment program is part of a multi-faceted effort to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. This program is a successor to the RELAP5/MOD1 independent assessment project underway at Sandia for the last two years. During this quarter, more code errors were identified and corrected, both by Sandia staff and by the code developers at Los Alamos National Laboratory. Work continued on the PKL natural circulation tests. Analyses of the condensing horizontal stratified flow tests and a NEPTUNUS pressurizer test were completed. The nodalization and steady state calculations for LOBI intermediate break test B-R1M were completed, and both the A1-04R and R-R1M transients were run until the time accumulator injection began. Finally, the status of the steady state calculations for Semiscale test S-IB-3, and the nodalization development and steady state calculations for Semiscale feedline break test S-SF-3 are given. These Semiscale test models were used to develop and test guidelines for correctly imolementing the separator model in TRAC during steady state calculations.

NUREG/CR-3821: EVALUATION OF CRACK PLANE EQUILIBRIUM MODEL FOR PREDICTING PLASTIC FRACTURE. BUTLER, T.A.; SMITH, F.W. Los Alamos Scientific Laboratory. July 1984. 23pp. 8409110107. LA-10129-MS. 26437:328.

A simple model for predicting the initiation of crack growth during plastic fracture is evaluated. The model is based on requiring equilibrium between applied loads and an assumed stress distribution in the uncracked ligament near the crack. The fracture parameters required are the material's ultimate tensile strength and a process-zone size at the crack tip that is determined from simple fracture tests. The Crack Plane Equilibrium model predicts crack-growth initiation for the crack geometries studied with sufficient accuracy to warrant extending it for investigating other geometries and for predicting stable crack growth and the onset of unstable crack growth.

NUREG/CR-3822: SOLA-PTS: A Transient, Three-Dimensional Algorithm For Fluid-Thermal Mixing And Wall Heat Transfer In Complex Geometrics. DALY, B.J.; TORREY, M.D. Los Alamos Scientific Laboratory. July 1984. 103pp. 8409110081. LA-10132-MS. 26445:203.

The SOLA-PTS computer code has been developed to analyze fluid-thermal mixing in the cold legs and downcomer of pressurized water reactors in support of the pressurized thermal shock study. SOLA-PTS is a transient, three-demensional code with the capability of resolving complex geometries using variable cell noding in the three coordinate directions. The computational procedure is second-order accurate and utilizes a state-of-the-art iteration method that allows rapid convergence to an accurate solution for the pressure field. Two different turbulence models are used in the code, a two-equation k-e model that is used in the cold leg pipe away from the HPI inlet and a three-equation k-e-T' model for use near the HPI inlet and in the downcomer.

The physical modeling and the numerical procedure used in SOLA-PTS are described in this report. Applications of the method to two Creare 1/5th-scale experiments are also presented. Two appendices are included. Appendix A provides a comparison of the two- and three-equation turbulence models, while Appendix B provides instructions for setting up and running a problem with SOLA-PTS.

NUREG/CR-3824: CONTING PROGRAM GUIDE. HENRY, E.B.; GENTILLON, C.D.; STEVERSON, J.A. EG&G, Inc. September 1984. 140pp. 8410030079. EGG-2315. 26821:001.

CONTING is an interactive computer program for automated trends and pattern analysis of data. The data are License Events Reports, which are coded into a computer-readable, searchable Sequence Coding and Search System (SCSS) format developed by the United States Nuclear Regulatory Commission. In the SCSS, the data are broken down into occurrences or steps, which are described by categorical variables such as system, component, and cause. CONTING searches the steps to obtain counts for contingency tables (hence, its name). The rows and columns of these tables correspond to user specified conditions for the variables. The pattern of counts appearing in such a table can provide insights concerning the operational experience at nuclear power plants. In addition, CONTING supports trend analysis since the counts can be grouped by the associated event dates. A statistical package formats the contingency tables; this facilitates the use of log-linear statistical program may include exposure times for use in normalizing the counts to obtain occurrence rates. CONTING has many features to aid the user in performing this analysis. CONTING was developed at the Idaho National Engineering Laboratory (INEL) on the CYBER 176, using FORTRAN 77. It operates on the SCSS data base located at the INEL.

NUREG/CR-3825 V01-02: ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS. Quarterly Report: October 1983 - March 1984. Vols 1 & 2. HUTTON, P. H.; KURTZ, R. J. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1984. 47pp. 8407120539. PNL-5125. 25556:187.

This report describes technical progress on a program to apply acoustic emission for continuous integrity surveillance of nuclear reactor pressure boundaries. The period is October 1983-March 1984. Test data from the completed intermediate scale vessel (ZB-1) test is being analyzed to isolate AE from crack growth for the purpose of refining AE signal identification and AE interpretation methods. Fatigue crack growth in the ZB-1 vessel is being characterized by destructive examination. Acoustic data obtained from the No. 2 inlet nozzle during hot functional testing at Watts Bar Unit 1 reactor showed a source concentration. A cooperative effort between TVA and PNL is planned to evaluate the significance of the data. Identification of crack growth AE by pattern recognition is showing much improved results. Fatigue testing of A106B ferritic pipe material is showing mixed AE results related to previous relationships developed for A533B steel. Development of an ASTM Standard Practice for continuous AE monitoring of pressure boundaries has been initiated. A NUREG document on results from AE monitoring at Watts Bar, Unit 1 reactor during hot functional testing has been completed.

NUREG/CR-3826: RECOMMENDATIONS FOR PROTECTING AGAINST FAILURE BY BRITTLE FRACTURE IN FERRITIC STEEL SHIPPING CONTAINERS GREATER THAN FOUR INCHES THICK. SCHWARTZ, M. W. Lawrence Livermore National Laboratory. July 1984. 131pp. 8408010155. UCRL-53338. 25872:087.

Various criteria for protecting against brittle fracture in spent-fuel shipping containers made from ferritic steel forgings greater than four inches thick are evaluated. A fracture initiation criterion based upon yield stress levels and allowable flaw sizes specified in Section XI of the ASME Code is recommended. This recommendation is based upon a value evaluation taking into account its effect upon industry and the risk of brittle fracture.

NUREG/CR-3830 V01: AEROSOL RELEASE AND TRANSPORT PROGRAM. Semiannual Progress Report For October 1983 - March 1984. ADAMS, R. E.; TOBIAS, M. L. Oak Ridge National Laboratory. July 1984. 79pp. 8409170442. ORNL/TM-9217/V1. 26498:199.

This report summarizes progress for the Aerosol Release and Transport Program sponsored by the Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Accident Evaluation, for the period October 1983-March 1984. Topics discussed include (1) the experimental program in the Fuel Aerosol Simulant Test (FAST) facility, (2) NSPP experiments involving mixtures of aerosols of iron oxide and uranium in steam and dry atmospheres, (3) support work for the DEMONS (West Germany) and Marviken (Sweden) projects, (4) analysis of core melt experiments involving boric oxide volatility, (5) initial operation of the new 250-kW induction generator, (6) comparisons of NAUA results with experiments, and (7) tests and improvements in the UVABUBL-II code.

NUREG/CR-3832: UNCERTAINTIES IN LONG-TERM REPOSITORY PERFORMANCE DUE TO THE EFFECTS OF FUTURE GEOLOGIC PROCESSES. SJOREEN, A.L.; KOCHER, D.C. Oak Ridge National Laboratory. August 1984. 41pp. 8409110084. ORNL-6049. 26447:099.

This report discusses uncertainties in predicting the long-term performance of geologic repositories for high-level waste that result from the effects of future geologic processes. This type of uncertainty arises from uncertainties in determining current rates of geologic processes, predicting process rates over long time periods in the future, and predicting the effects of future geologic processes on performance. This report emphasizes the qualitative and judgmental nature of predictions of future geologic processes and their effects on repository performance. However, significant changes generally occur over time periods of 100,000 years or more. Thus, at sites chosen for their stability, geologic processes should not have significant effects on repository performance over a period of 10,000 years.

NUREG/CR-3833: BEHAVIOR OF SUBCRITICAL AND SLOW-STABLE CRACK GROWTH FOLLOWING A POST-IRRADIATION THERMAL ANNEAL CYCLE. CULLEN, W.H.; HISER, A.L. Materials Engineering Associates, Inc. August 1984. 41pp. 8409200295. MEA-2049. 26609:061.

This report presents the experimental results of Phase I of a Small Business Innovation Research Program which investigated the response of environmentally-assisted monotonic and cyclic crack growth following a simulated anneal of a reactor-pressure vessel weld. Unirradiated steels were used in this (initial) Phase I of the program. Fatigue cracks were grown in several specimens of a submerged arc weld deposit in pressurized, high-temperature reactor-grade water. The specimens were removed from the environment, and annealed for one week at either 399 degree C or 454 degree C. Fatigue crack growth in high-temperature water was resumed on several annealed specimens and unannealed controls. No effect of the anneal was noted on the fatigue crack growth rates, which continued with about the same degree of environmental assistance as exhibited before the anneal. An elastic-plastic fracture specimen, tested in 93 degree C air at a very slow loading rate, showed that neither annealing nor the slow rate had a significant effect on the J-R curve characteristics. However, conducting the tests at a slow loading rate in 93 degree C PWR water resulted in a 25% to 30% decrease in JIc and a small decrease in T avg. Examination of the oxides on the fatigue fracture surfaces showed that magnetite (formed during the crack growth in pressurized, high-temperature water) was the predominant oxide specie.

NUREG/CR-3834: ON THE THRESHOLD SULFUR AND LITHIUM TO SULFUR RATIO IN STRESS CORROSION CRACKING OF SENSITIZED ALLOY 600 IN BORATED THIOSULFATE SOLUTION. BANDY, R.; KELLY, K. Brookhaven National Laboratory. July 1984. 35pp. 8408070014. BNL-NUREG-51785. 25954:296.

The stress corrosion cracking (SCC) of sensitized Alloy 600 was investigated in aerated solutions of sodium thiosulfate containing

1.3% boric acid, using U-bends, constant load, and slow strain rate tests. The aim of the investigation, among others, was to determine the existence, if any, of a threshold level of sulfur, and Li to S ratio governing the SCC. For U-bends, 5 ppm Li as LiOH in the presence of 7 ppm S as thiosulfate prevented occurrence of SCC. However, in slow strain rate tests, significant SCC occurred at a S level of 30 ppb in the presence of 0.7 ppm of Li. For a specimen held under constant load, a propagating crack continued to grow until fracture during controlled progressive dilution of the bulk solution, leading to final Li concentration of 1.5 ppm and S concentration of 9.6 ppb respectively. The implications of the results to initiation and propagation of SCC in aerated thiosulfate solutions, and their relevance to future operation of the steam generators at Three Mile Island Unit 1 (TMI-1) are discussed.

NUREG/CR-3835: SIMULATION OF FLAME PROPAGATION THROUGH VORTICITY REGIONS USING THE DISCRETE VORTEX METHOD. BARR, K.P. Sandia Laboratories. September 1984. 19pp. 8410170076. SAND84-8715. 27041:045.

The interaction of a freely propagating premixed flame with regions of high vorticity in the flow is investigated using a computer model. These vorticity regions are formed due to the flame-generated volume expansion that pushes gas past obstacles ahead of the flame. In the computer model the discrete vortex dynamics method is used to simulate the time development of the vorticity regions downstream of each obstacle. The flame front is modeled as a wrinkled laminar flame interface that propagates normal to itself at the laminar burning velocity, separating the two different density fluids: burned and unburned. Two different obstacle configurations are discussed in this paper. In the first case, a flame causes unburned gas to exhaust out of a planar duct, and when the flame reaches the duct exit it interacts with the vorticity which was formed at the exit. Two versions of this configuration are considered: sharp and square edge exit. The second case involves a series of obstacles in a channel. Here, the repeated obstacles in the channel leads to acceleration of the flame as indicated by the dramatic increase in fuel consumption.

NUREG/CR-3838: AN INITIAL REVIEW OF SEVERAL METEOROLOGICAL MODELS SUITABLE FOR LOW-LEVEL WASTE DISPOSAL FACILITIES. CULKOWSKI, W.M. Commerce, Dept. of, Natl. Oceanographic & Atmospheric Administration. June 1984. 21pp. 8407110180. 25536:296.

Several mathematical models of the meteorological aspects of effluent releases have been examined for Dames and Moore, Inc., Science Applications, Inc., Argonne National Laboratory, and Oak Ridge National Laboratory, contain provisions for various combinations of wind erosion, area, and point source configurations as well as deposition and elevated releases. Methods employed by these models are compared for relevance, availability of supporting data and potential benefit versus cost.

NUREG/CR-3839: AN EMPIRICAL ASSESSMENT OF NEAR-SOURCE GROUND MOTION FOR A 6.6 MB (7.5 MS) EARTHQUAKE IN THE EASTERN UNITED STATES. CAMPBELL, K.W. Lawrence Livermore National Laboratory. June 1984. 66pp. 8407180329. UCID-20083. 25654:203.

To help assess the impact of the current U.S. Geological Survey position on the seismic safety of nuclear power plants in the Eastern United States (EUS), several techniques for estimating near-source strong ground motion for a Charleston size earthquake were evaluated. The techniques for estimating the near-source strong ground motion for a 6.6 mb (7.5 Ms) in the Eastern United States which were assessed are methods based on (1) site specific analyses, (2) semi-theoretical scaling techniques, and (3) intensity-based estimates. Each method differently approaches the problem of estimating near-source strong ground motions. The results and limitations of each technique are discussed and recommendations made to correct for bias in the methods. Suggestions for future work are also presented.

NUREG/CR-3840: COST ANALYSIS FOR POTENTIAL MODIFICATIONS TO ENHANCE THE ABILITY OF A NUCLEAR PLANT TO ENDURE STATION BLACKOUT. CLARK, R.A.; THOMAS, W.R.; et al. Science & Engineering Associates, Inc. RIORDON, B.J. MATHTECH, Inc. July 1984. 167pp. 8408080472. 25980:160.

Cost estimates were required to serve as partial bases for decisions on four potential nuclear reactor facility modifications being considered in the resolution of USI A-44, Station Blackout. The modification constituting the four subtasks in this report are (1) increasing battery capacity, (2) adding an AC-independent charging pump for reactor coolant seal injection, (3) increasing condensate storage tank capacity, and (4) increasing compressed air supply for instrument air.

The cost estimates contained in this report include those for the following: (1) engineering and design, (2) equipment, materials, and structures, (3) installation, and (4) present worth of the annual operation and maintenance over the remaining useful life of the reactor.

In addition to providing engineering requirements for the four modifications, the report evaluates the potential for synergistic solutions. It was found that some modifications to provide for reactor coolant seal injection would effectively satisfy the DC system augmentation requirements, with the costs for solving both problems being competitive with that of additional batteries alone. The report also identifies an innovative potential solution to the DC system capacity problem through the use of high energy density primary batteries which would be far more cost effective than the addition of traditional lead acid batteries for mitigating extended station blackout effects.

NUREG/CR-3841: STEAM GENERATOR GROUP PROJECT. Task 6 - Channel Head Decontamination. ALLEN, R.P.; CLARK, R.L.; REECE, W.D. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 364pp. 8411130643. PNL-4712. 27475:317.

The Steam Generator Group Project utilizes a retired from service pressurized water reactor steam generator as a test bed and source of specimens for research. Program objectives emphasize validation of the ability to nondestructively characterize the condition of steam generator tubing in service. Remaining integrity of tubing with service induced defects is studied through burst and leak rate tests. Other program objectives seek to characterize overall generator condition, including secondary side structure, and provide realistic samples for development of primary side decontamination, secondary side cleaning, and nondestructive examination technology. An important preparatory step to primary side research activities was reduction of the radiation field in the steam generator channel head. This task report describes the channel head decontamination activities. Though not a programmatic research objective, it was judged beneficial to explore the use of dilute reagent chemical decontamination techniques. These techniques presented potential for reduced personnel exposure and reduced secondary radwaste generation, over currently used abrasive blasting techniques. Two techniques with extensive laboratory research and vendors prepared to offer commercial application were tested, one on either side of the channel head. As indicated in the report, both techniques accomplished similar decontamination objectives. Neither technique damaged the generator channel head or tubing materials, as applied. This report provides details of the decontamination operations. Application system and operating conditions are described. Areas of improvement are suggested.

NUREG/CR-3842: STEAM GENERATOR GROUP PROJECT TASK 8 - SELECTIVE TUBE UNPLUGGING. WHEELER,K.R.; DOCTOR,P.G.; FETROW,L.K.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1984. 174pp. 8410170217. PNL-4876. 27029:034.

The Steam Generator Group Project utilizes a retired from service pressurized water reactor steam generator as a test bed and source of specimens for research. Program objectives emphasize validation of the ability to nondestructively characterize the condition of the steam generator tubing in service.

During operation 748 of the 3388 tubes in the research generator were removed from service by explosive plugging on both ends. Tubes were plugged due to defect indications, inspectability limits caused by denting, and as a preventative measure. The plugged tubes contained a substantial portion of the defects necessary for the research program. This report summarizes activities conducted during a campaign removal of 969 explosive tube plugs. The report provides detailed descriptions of the planning, training, supplies, equipment, and operations that led to the successful completion of the unplugging in 20 days of operation. Also presented is information on problems encountered and observations that could aid future unplugging operations.

NUREG/CR-3843: STEAM GENERATOR GROUP PROJECT TASK 10 - SECONDARY SIDE EXAMINATION. SCHWENK,E.B.; WHEELER,K.R. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1984. 69pp. 8410120034. PNL-5033. 26985:269.

The Steam Generator Group Project utilizes a retired from service pressurized water reactor steam generator as a test bed and source of specimens for research. Program objectives emphasize validation of the ability to nondestructively characterize the condition of steam generator tubing in service. Remaining integrity of tubing with service induced defects is studied through burst and leak rate tests. Other program objectives seek to characterize overall generator condition, including secondary side structure, and provide realistic samples for development of primary side decontamination, secondary side cleaning, and nondestructive examination technology.

This report provides information on secondary side characterization efforts. The methods and equipment used are discussed, along with comparisons of benefits offered by various techniques. Details of secondary side steam generator conditions are then presented, emphasizing support plate and U-bend regions.

NUREG/CR-3844: CHARACTERIZATION OF THE RADIOACTIVE WASTE PACKAGES OF THE MINNESOTA MINING AND MANUFACTURING COMPANY. KEMPF,C.R.;

SISKIND, B. ; BARLETTA, R. E. ; et al. Brookhaven National Laboratory. July 1984. 93pp. 8408010151. BNL-NUREG-51787. 25872:001.

An evaluation of the low-level waste packages generated by Minnesota Mining and Manufacturing Co. (3M) was made on the basis of 10 CFR Part 61 criteria and on the Technical Position on Waste Form and Waste Classification (TP). This evaluation was the result of a study initiated by the U. S. Nuclear Regulatory Commission (NRC), in which 3M participated.

3M produces a variety of radioactive products and wastes. The dominant radioisotopes are Po-210 and Cs-137. The Po-210 packages are generally Class A and meet the requirements in 10 CFR Part 61. The Cs-137 and Sr-90 packages fall into all three waste classifications (A, B, and C). These wastes are packaged by 3M in 30-gallon or 55-gallon carbon steel drums (Class A) or 30-gallon lined drums (Class B and C). The Class B and greater lead- and concrete-lined packages have been evaluated with respect to meeting the stability requirements for waste disposed of in a high integrity container. When so evaluated, eleven areas of concern were identified with respect to the regulations and recommendations in the TP.

NUREG/CR-3845: PREDICTION OF NONLINEAR STRUCTURAL RESPONSE IN LMFBR ELEVATED-TEMPERATURE PIPING. FARRAR, C. Los Alamos Scientific Laboratory. July 1984. 28pp. 8409170447. LA-10090-MS. 26498:295.

The development of structural analysis capabilities to investigate possible accident initiations caused by structural degradation o liquid metal fast breeder reactor (LMFBR) piping is summarized. The ABAQUS finite element code is used to perform a non-linear analysis of a bench mark problem proposed by the Pressure Vessel Research Committee. The problem is representative both in geometry and loading of an LMFBR elevated-temperature piping system, and published analytical results are available for comparison. Results show the system to be most sensitive to large, radial, thermal gradients that occur when the system experiences certain thermal transients. Repeated cycles of these transients will lead to thermal ratcheting, causing progressive deformation and strain accumulation in the system. Future work will verify the accuracy of the finite element model and quantify damage accumulated during the lifetime of an LMFBR elevated-temperature piping system.

NUREG/CR-3847: CLIMATIC CALIBRATION OF POLLEN DATA:A User's Guide For The Applicable Computer Programs In The Statistical Package For Social Scientists (SPSS). ARIGO, R. ; HOWE, S. E. ; WEBB, T. ; et al. Brown Univ., Providence, RI. June 1984. 39pp. 8407020174. 25275:235.

Radiocarbon-dated pollen records are a source of quantitative estimates for climatic variables for the past 9000 years. Multiple regression is the main method for calculation of these estimates and requires a series of steps to gain equations that meet the statistical assumptions of the analysis. This manual describes these steps which include (1) selection of the region for analysis, (2) selection of the pollen types for statistical analysis, (3) deletion of univariate outliners, (4) transformation to produce linear relationships, (5) selection of the regression equation, and (6) tests of the regression residuals. The input commands and the output from a series of SPSS (Statistical Package for Social Scientists) programs are illustrated and described, and, as an example, modern pollen and climatic data from lower Michigan are used to calculate a regression equation for July mean temperature.

NUREG/CR-3848: EXPERIMENTAL INVESTIGATION OF UNSTEADY TORNADIC WIND LOADS ON STRUCTURES. JISCHKE, M. C. ; MOSLEHI, F. Oklahoma Teaching Hosps., Oklahoma City, OK. June 1984. 34pp. 8407120632. 25556:233.

Ward's tornado simulator was used to model the effects of a tornado-like vortex on cylindrical model structure. The experiment was conducted at swirl angles of 0 and 45 degrees. Pressure coefficients were measured at different locations on the model for steady and unsteady cases, corresponding to situations where the relative velocity between the vortex and model is zero and nonzero. Results are presented in the forms of sectional pressure coefficient profiles, and sectional force coefficients. Pressure profiles show that there are significant differences between the steady and unsteady results. Translation of the model through the simulator produces a more symmetric pressure distribution, and also results in a more substantial pressure drop on the model.

It is observed that in a flow with swirl angle at 45 degrees, translation causes a significant increase in the horizontal sectional force coefficient. Outside of the core region, translation causes an increase in the axial sectional force coefficient. The formation of very low pressure regions over the top section of the structure leads to very strong axial force coefficients. This may cause the failure to first appear on the roof, and then propagate throughout the structure and cause total failure.

NUREG/CR-3849: TWO-PHASE 3X3 ROD BUNDLE TEST FACILITY FOR POST-CRITICAL HEAT FLUX BOILING. TUZLA, K. ; UNAL, C. ; BADR, O. A. ; et al. Lehigh Univ., Bethlehem, PA. June 1984. 57pp. 8407060340. TS-843. 25432:290.

This report describes the rod bundle post-CHF tests in progress and the test facility at Lehigh University. The mechanical and electrical design of the experimental facility and the iterative process used to arrive at the choices made for the design are described in detail. The test facility consists of a nine (3 x 3) rod bundle in a square shroud which form the test section together with the hot patches at the top and bottom ends. The rods and the hot patches are electrically heated while the shroud is radiatively heated. The test section includes instrumentation to measure the vapor superheat temperature and pressure drop upstream and downstream of a rod gap spacer. This is the first application of the hot patch technique for generating post-CHF conditions in a rod bundle and thus quasi-steady-state tests are being thought of as a backup procedure for conducting these post-CHF heat transfer tests.

The test section is part of a well instrumented recirculating loop to generate the desired post-CHF conditions. The other major components of the heat transfer loop include the surge tank, pumps, boiler, separation tank and condenser. The test facility also includes a versatile one hundred channel data acquisition system. The mechanical and electrical components in the facility have been chosen to have sufficient accuracy to yield meaningful results for the heat transfer coefficient in the rod bundle under various post-CHF conditions.

NUREG/CR-3850: DEBRIS BED QUENCHING UNDER BOTTOM FLOOD CONDITIONS (IN-VESSEL DEGRADED CORE COOLING PHENOMENOLOGY). TUTU, N. K. ; GINSBERG, T. ; KLEIN, J. ; et al. Brookhaven National Laboratory. October 1984. 80pp. 8411130622. BNL-NUREG-51788. 27477:158.

This report is directed towards development of an understanding of the transient quenching of in-vessel debris beds, located in the reactor core region, under conditions where the coolant is injected from below. Specifically, the objective is to develop and experimentally verify analytical models for the prediction of the temperature distribution and the steam generation rate during the transient quenching of superheated debris beds by cooling water supplied from the bottom of the debris bed. Experiments involving the quenching of superheated debris beds formed with 3.18 mm stainless steel spheres were performed. Water at saturation temperature was injected from below at a constant rate to initiate the quenching process. Measurements were made of the instantaneous heat flux, and thermocouple temperatures at various location within the bed. The experimental data suggest that for small liquid supply rate and low initial particle temperature the bed quench process is a one-dimensional frontal phenomenon. A quasi-steady one-dimensional model for the quenching process is developed for this "deep bed" regime. For large liquid supply rate and high initial bed temperature, the bed quench process is a complex, multi-dimensional phenomenon. A simplified transient model of coolant-debris interaction was developed to characterize this "shallow-bed" regime.

NUREG/CR-3851 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS Report for October-December 1983. KELMERS, A. D. ; KESSLER, J. H. ; ARNOLD, W. D. ; et al. Oak Ridge National Laboratory. August 1984. 48pp. 8408300282. ORNL/TM-9191/V1. 26331:224.

Oak Ridge National Laboratory (ORNL) is conducting an experimental investigation of geochemical information for the Nuclear Regulatory Commission (NRC). During this quarter, the project evaluated both radionuclide solubility data and retardation parameters reported by the Basalt Waste Isolation Project (BWIP), and the methodologies used to develop those values. Under oxic conditions, neptunium had a sorption ratio of 1.7 L/kg for McCoy Canyon basalt and synthetic groundwater GR-2, which is lower than the "conservative best estimate" value recommended by BWIP. Under anoxic conditions, the basalt showed little or no ability to remove technetium (VII) from GR-2 by sorption or precipitation. Several important concerns may make it impossible to assert that the addition of hydrazine to groundwater is modeling the repository redox condition. These are: (1) its reaction with any reducible solute is undefined, (2) its dissociation to release hydroxide ions probably dominates the groundwater pH, (3) it could react with bicarbonate to form the carbamate ion, (4) it is corrosive to polycarbonate or polypropylene test tubes, (5) it may alter or disaggregate clay mineral structure, and (6) uncertainty exists as to the solid phase or solution species formed by reaction with pertechnetate ion. Thus, BWIP data obtained in the presence of hydrazine may be nonconservative for use in assessment studies.

NUREG/CR-3851 V02: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS: Report For January-March 1984. KELMERS, A. D. ; KESSLER, J. H. ; ARNOLD, W. D. ; et al. Oak Ridge National Laboratory. November 1984. 63pp. 8412260116. ORNL/TM-9191/V2. 28069:194.

Geochemical information relevant to the retention of radionuclides by Department of Energy (DOE) candidate high-level waste repositories is being investigated by Oak Ridge National Laboratory (ORNL) for the Nuclear Regulatory Commission (NRC). The project has evaluated values that have been reported by the Basalt Waste Isolation Project (BWIP) and the methodologies used to develop those values. Neptunium sorption was dependent upon the basalt flow used in the test. Increasing the test temperature from 24 to 60 C increased the neptunium sorption ratio. Hydrazine did not reduce neptunium(V) to neptunium (IV) in solution. The Amicon filters used to separate the basalt and groundwater after contact have been shown to adsorb a low but significant fraction of the neptunium(V) in solution. Technetium(VII) removal by basalt from groundwater solutions was shown to be independent of the contact methodology but was quite sensitive to the solution composition. Sorption of uranium(VI) by basalt has yielded sorption ratio values higher than those reported by BWIP. Column chromatographic experiments have confirmed the technetium(VII) sorption ratio of about 0 L/kg. The McCoy Canyon basalt used in this experimental work was mineralogically characterized. Six phases were identified: plagioclase, mesostasis, pyroxene, magnetite, apatite, and pyrite. Geochemical modeling with PHREEQE and MINTEQ checked the calculated solubilities of 16 radionuclides reported by BWIP; the agreement was generally excellent.

NUREG/CR-3852: INSIGHT INTO PRA METHODOLOGIES. GALLAGHER, D. Science Applications, Inc. August 1984. 121pp. 8409200290. 26607:001.

This report describes the results of a survey of six probabilistic risk assessments to determine the impact of different aspects of the methodology on dominant sequence ordering and core-melt likelihood. The results indicate that effort should be given to human error analysis, system dependency analysis, and modeling of AC power systems.

NUREG/CR-3853: PRELOADING OF BOLTED CONNECTIONS IN NUCLEAR REACTOR COMPONENT SUPPORTS. YAHR, G. T. Oak Ridge National Laboratory. October 1984. 73pp. 8411210175. ORNL-6093. 27643:078.

A number of instances of failures of threaded fasteners in nuclear reactor component supports have been reported. Many of those failures were attributed to stress corrosion cracking. This report discusses how stress corrosion cracking can be avoided in bolting by controlling the maximum bolt preloads so that the sustained stresses in the bolts are below the level required to cause stress corrosion cracking. This is a basic departure from ordinary bolted joint design where the only limits on preload are on the minimum preload. The importance of detailed analysis to determine the acceptable range of preload and the selection of a method for measuring the preload that is sufficiently accurate to assure that the preload is actually within the acceptable range are stressed. Procedures for determining acceptable preload range are given and the accuracies of various methods of measuring preload are given.

NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATIONS - PRELIMINARY REPORT. BUMP, T. R. Argonne National Laboratory. * Sandia Laboratories. October 1984. 272pp. 8411130571. SAND84-7139. 27475:042.

This report summarizes the survey work conducted by Argonne National Laboratory on the design and details of major penetrations in 22 nuclear power plants. The survey includes all containment types and materials in current use. It also includes details of all types of penetrations (except for electrical penetration assemblies and valves) and the seals and gaskets used in them. The report provides a test matrix for testing major penetrations and for testing seals and gaskets in order to evaluate their leakage potential under severe accident conditions.

NUREG/CR-3856. AN ULTRASONIC LEVEL AND TEMPERATURE SENSOR FOR POWER REACTOR APPLICATIONS. DRESS, W. B.; MILLER, G. N. Oak Ridge National Laboratory. August 1984. 30pp. 8409180280. ORNL/TM-9236. 26589:001.

An ultrasonic waveguide employing torsional and extensional acoustic waves has been developed for use as a level and temperature sensor in pressurized and boiling water nuclear power reactors. Features of the device include continuous measurement of level, density, and temperature producing a realtime profile of these parameters along a chosen path through the reactor vessel.

NUREG/CR-3857: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS: Annual Report, October 1982 - September 1983. CHOPRA, O. K.; AYRAULT, G. Argonne National Laboratory. October 1984. 30pp. 8411130553. ANL-84-44. 27477:013.

This progress report summarizes work performed by the Argonne National Laboratory during the twelve months from October 1982 to September 1983 on long-term embrittlement of cast duplex stainless steels used in light-water reactors.

NUREG/CR-3864: CHARACTERIZATION OF THE LOW-LEVEL RADIOACTIVE WASTES AND WASTE PACKAGES OF GENERAL ELECTRIC VALLECITOS NUCLEAR CENTER. KEMPF, C. R.; MACKENZIE, D. R.; BOWERMAN, B. S.; et al. Brookhaven National Laboratory. November 1984. 96pp. 8412110033. BNL-NUREG-51791. 27896:139.

An evaluation of the low-level wastes and waste packages generated by General Electric Vallecitos Nuclear Center (GEVNC) was made on the basis of 10 CFR Part 61 criteria and on the Technical Position on Waste Form (TP). In addition, a review has been performed of the handling and storage methods used by GEVNC for their transuranic wastes. Several options have been discussed for management of these materials. This evaluation was the result of a study initiated by the U. S. Nuclear Regulatory Commission (NRC), in which GEVNC participated. GEVNC generates Class B or greater radioactive wastes in hot cell processes which include examination of reactor fuel and components, and production of sources and radiopharmaceuticals. The dominant contaminating radioisotopes are Cs-137 and Co-60. In addition, transuranic wastes which result from examination and burnup analyses of fuel are all currently stored on-site at GEVNC. Class B and greater wastes are packaged in 84-gallon extended 17H drums that are grouted with cement. Based on the evaluation, overall, the waste forms of these packages are expected to maintain their stability, but a few concerns are identified and testing should be performed by GEVNC to demonstrate waste form stability.

NUREG/CR-3867: DATA SUMMARIES OF LICENSEE EVENT REPORTS OF INVERTERS AT U. S. COMMERCIAL NUCLEAR POWER PLANTS, JANUARY 1, 1976 TO DECEMBER 31, 1982. BROWN S. R.; TROJOVSKY, M. EG&G, Inc. August 1984. 131pp. 8409280077. EGG-2324. 26762:206.

This report describes a computer-based data file developed from License Event Reports (LERs) of inverters in U. S. commercial nuclear power plants for the period January 1, 1976 to December 31, 1982. In addition to the creation of the file, summaries of data contained in the file were made to obtain data for risk assessment and statistical purposes. Gross constant failure rates were estimated for inverters found in selected systems. Explanations, figures, and summary tables of the results are provided.

NUREG/CR-3868: CONTAINMENT BUILDING ATMOSPHERE RESPONSES DUE TO REACTOR GAS BURNING UNDER SEVERE ACCIDENT CONDITIONS. KROEGER, P. G. Brookhaven National Laboratory. July 1984. 42pp. 8410030363. BNL-NUREG-51793. 26817:315.

The formation of combustible atmospheres during unrestricted core heatup accidents in High Temperature Gas-Cooled Reactors is being investigated, considering the effects of only partially mixed atmospheres. It is found that the previously used assumption of complete mixing presents the more severe limit in most cases. In the few cases where higher loads were obtained, these were still below the invocation of even more remote failure scenarios. A qualitative discussion applying the above results to comparable accident at Fort St. Vrain is included.

NUREG/CR-3869: ANALYSIS OF THE IMPACT OF INSERVICE INSPECTION USING A PIPING RELIABILITY MODEL. SIMONEN, F. A.; WOO, H. H. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 55pp. 8408220370. PNL-5149. 26199:249.

This report presents the results of a study of the impact of inservice (ISI) programs on the reliability of specific nuclear piping systems that have actually failed in service. Two major factors are considered in the ISI programs: one is the capability of detecting flaws; the other is the frequency of performing ISI. A probabilistic fracture mechanics model issued to estimate the reliability of two nuclear piping lines over the plant life as functions of the ISI programs. Examples chosen for the study are the PWR feedwater steam generator nozzle cracking incident and the BWR recirculation reactor vessel nozzle safe-end cracking incident. The results show that an effective inservice inspection requires a suitable combination of flaw detection capability and inspection schedule. An augmented inspection schedule is required for piping with fast-growing flaws to ensure that the inspection is done before the flaws reach critical sizes. Also, the elimination of "poor" inspection teams through training and qualification testing can produce significant benefits to ISI effectiveness.

NUREG/CR-3870: RADIATION DOSE ESTIMATES AND HAZARD EVALUATIONS FOR INHALED AIRBORNE RADIONUCLIDES. Annual Progress Rept July 1982 -June 1983. MEWHINNEY, J. A. Inhalation Toxicology Research Institute. July 1984. 38pp. 8408160125. LMF-109. 26122:313.

The objective of this project is to conduct confirmatory research on aerosol characteristics and the resulting radiation dose distribution in animals after inhalation and to provide prediction of health consequences in humans from airborne radioactivity that might be released in normal operations or under accident conditions during production of nuclear fuel composed of mixed oxides of uranium and plutonium. Two research reports summarize the progress of current research. The first paper details results from the completed radiation dose distribution studies in which dogs, monkeys, and rats were exposed to either UO(2) + PuO(2) treated at 750 degrees centigrade, (U, Pu)O(2) treated at 175 degrees centigrade, or PuO(2) treated at 850 degrees centigrade. This paper focuses on analysis of the data from the last animals sacrificed in the study and updates earlier analyses of lung retention, tissue distribution, and excretion. The second paper details preliminary analyses of the lung retention in Fischer-344 rats exposed to either (U, Pu)O(2) or to PuO(2) at one of three levels of projected dose to lung for each aerosol. This paper presents the methods and the application of a rigorous statistical procedure allowing detection of similarities and differences in the lung retention of rats at different dose levels and for different aerosols.

NUREG/CR-3871: AN OVERVIEW OF THE UNIFIED TRANSPORT APPROACH. ERASLAN, A. H.; WITTEN, A. J. Oak Ridge National Laboratory. August 1984. 123pp. 8409200399. ORNL-TM-9249. 26607:128.

The Unified Transport Approach (UTA) consists of a set of nine complementary models developed for assessing the environmental impacts associated with nuclear power plant discharges to receiving water bodies. This set of models has the capability to simulate natural and plant-induced flow, temperature, salinity, sediment transport, radionuclide transport, and chemical species concentrations. While these UTA models were developed for predicting impacts associated with the operation of nuclear power plants, they are quite general and can be applied to a variety of situations. The UTA models have been used to simulate the impacts associated with the operation of many industrial and energy production technologies, as well as to simulate laboratory and naturally occurring conditions. In all cases where data have been available for validation, the UTA model results have compared favorably. The purpose of this report is to provide an overview of the UTA as a whole, highlighting the important features and unique capabilities of this approach.

NUREG/CR-3874: NEAR-GROUND TORNADO WIND FIELDS. MCDONALD, J. R. Texas Tech Univ., Lubbock, TX. July 1984. 164pp. 8408220327. 26198:197.

This report is written as a general treatise on near-ground tornado wind fields. In Section II an engineering perspective on tornadoes is stated. Section III describes the data available for the study of near-ground tornado wind fields. Section IV discusses tornado wind speeds and briefly describes a new method for making more rational estimates of tornado wind speeds from damaged structures. Section V describes the damage indicators that are present in the wake of a tornado event and discusses other factors that affect the appearance of damage. A perspective on tornado-generated missiles is presented in Section VI. Conclusions and recommendations for further study are contained in the last section of the report.

NUREG/CR-3875: THE USE OF IN-SITU PROCEDURES FOR SEISMIC QUALIFICATION OF EQUIPMENT IN CURRENTLY OPERATING PLANTS. SADIK, S.; ARENDTS, J. G.; DIXON, B. W.; et al. EG&G, Inc. June 1984. 186pp. 8407180218. EGG-EA-6650. 25654:015.

This report supports the Nuclear Regulatory Commission (NRC) Unresolved Safety Issue A-46, "Seismic Qualification of Equipment in Operating Plants." The report is divided into four distinct sections. Part A identifies the basic technical approaches for using in-situ test procedures as a tool in alternate methods for the seismic qualification of equipment in operating plants. Part B includes the development of improved methods of developing structural models using the results of in-situ procedures, and predicting structural response during seismic events using methods of random vibrations. Thorough technical justification for these methods of analysis is provided to support the related guidance and acceptance criteria presented in Part C. Part D contains a cost estimate for using the various alternative methods for seismic qualification of equipment.

NUREG/CR-3878: MODELING CONSIDERATIONS FOR THE PRIMARY SYSTEM OF THE EXPERIMENTAL BREEDER REACTOR-II. MADNI, I. K. Brookhaven National Laboratory. September 1984. 45pp. 8410120016. BNL-NUREG-51797. 26978:271.

This report describes the additional heat transfer and coolant dynamic models for components and processes, that are needed for simulation of the primary system of the Experimental Breeder Reactor-II (EBR-II). This work forms part of the Super System Code (SSC) application efforts to provide predictions of EBR-II overall plant behavior.

NUREG/CR-3884: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS PROGRAM - THREE MILE ISLAND UNIT 2 POLAR CRANE RECOVERY. DOERGE, D. H.; MILLER, R. L. United Nuclear Corp. (subs. of UNC Resources, Inc.). August 1984. 63pp. 8408290185. 26309:267.

This document summarizes information concerning restoration of the Three Mile Island-Unit 2 Polar Crane to a fully operational condition following the loss of coolant accident experienced on March 28, 1979.

The data collected from activity reports, reactor containment entry records and other sources were placed in a computerized information retrieval/manipulation system which permits extraction/manipulation of specific data which could be utilized in planning for recovery activities should a similar accident occur in a nuclear generating plant. The information is presented in both computer output form and a manually assembled summarization.

This report contains only manpower requirements and radiation exposures actually incurred during recovery operations within the reactor containment and does not include support activities or costs.

NUREG/CR-3885 V01: HIGH TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, January 1 - March 31, 1984. BALL, S. J.; CLEVELAND, J. C.; HARRINGTON, R. M.; et al. Oak Ridge National Laboratory. October 1984. 21pp. 8411130696. ORNL/TM-9267/V1. 27471:214.

Modeling and code development work for predicting source terms for the Fort St. Vrain and 2240-MW(t) reactors continued and investigations and modeling work for small modular High-Temperature Gas-Cooled Reactor designs were begun. Fission-product transport experiments to determine coefficients for diffusion through graphite have included studies with Ag, Rh, and Pd. The review of an FSV technical specification (tech spec) on limiting maximum core temperature involved code development and FSV data analysis, leading to new proposed limiting conditions and validation tests.

NUREG/CR-3885 V02: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, April 1 - June 30, 1984. BALL, S. J.; CLEVELAND, J. C.; HARRINGTON, R. M.; et al. Oak Ridge National Laboratory. November 1984. 26pp.

8501030019. ORNL/TM-92. 28198:323.

Modeling, code development, and accident analysis work on the modular High-Temperature Gas-Cooled Reactor (HTGR) systems concentrated on predictions of core and other system temperature histories for postulated long-term loss of forced circulation accidents both with and without system depressurization. Fission-product (FP) release experiments to investigate vapor pressure and diffusion rates through graphite were continued. Experiments with additional elements were conducted.

NUREG/CR-3886: ACTIVITY AND FLUENCE CALCULATIONS FOR THE STARTUP AND TWO-YEAR IRRADIATION EXPERIMENTS PERFORMED AT THE POOLSIDE FACILITY. MAERKER, R.E.; WORLEY, B.A. Oak Ridge National Laboratory. October 1984. 46pp. 8411290586. ORNL/TM-4265. 27700:153.

This report is based on two separate intra-laboratory correspondences to F. B. K. Kam of the Operations Division from the authors, dated August 8 and October 21, 1983, detailing results of an analysis of two experiments performed at the Poolside Facility. It expands somewhat on the comparisons between measurement and calculation for the two-year experiment over those that were originally presented in the correspondence because of the recent availability of additional measured results. The two experiments analyzed in this report represents a scoping short-term dosimetry experiment and a long-term experiment using virtually the same geometry but incorporating metallurgical specimens as well as dosimetry. The long-term experiment is intended as an international metallurgical benchmark, and the fluence calculations described in this report have been made available to the international reactor dosimetry community to be used in their damage assessment studies and/or adjustment procedures.

NUREG/CR-3888: ANALYSIS OF THE VENUS PWR ENGINEERING MOCKUP EXPERIMENT -PHASE I: SOURCE DISTRIBUTION. MORAKINYO, P.O.; WILLIAMS, M.L.; KAM, B.K. Oak Ridge National Laboratory. August 1984. 81pp. 8410030356. ORNL/TM-9238. 26833:162.

The neutron fission source distribution in the core of the VENUS PWR Mockup Experiment is computed and compared to experimental measurements. Of particular concern is the accuracy of the source calculation near the core-baffle interface, which is the important region for contributing to RPV fluence.

Results indicate that the calculated neutron source distribution within the VENUS core agrees with the experimentally measured values with an average error of less than 3%. At the important core-baffle interface, the agreement is within 3% error, except at the baffle corner, where the error is about 6%. Better accuracy in the calculations can be obtained by applying a detailed space dependent cross-section weighting procedure to the core-baffle interface region. Using this cross-section weighting, the maximum error introduced into the predicted RPV fluence due to source errors should be on the order of 5%. However, in power reactor analysis, additional complexities (such as the time-dependent core composition and the use of few group diffusion theory) could affect this uncertainty value.

NUREG/CR-3890: TECHNOLOGY FOR URANIUM MILL PONDS USING GEOMEMBRANES. MITCHELL, D.H. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1984. 88pp. 8412210135. PNL-5164. 28049:286.

Pacific Northwest Laboratory has analyzed the performance of polymeric membrane-lined impoundments containing tailings and leachate at active uranium mills. The U.S. Nuclear Regulatory Commission has requested this information to support licensing of impoundments. Data on the performance of lined ponds in the U.S. uranium industry, mechanisms for damage of liners, and design, installation, and inspection practices are presented in this report. Design, construction, and inspection methods that are capable of minimizing failures are also identified. No cases of contaminated groundwater are attributed to uranium mill ponds lined with polymeric membranes (geomembranes) in the U.S. The leading causes of geomembrane problems for all industrial pond applications are faulty seams, puncture and errors during placement, improper connections to submerged structures, puncture by soils in contact with the geomembrane, and geotechnical problems due to liquids in the support soil. Although some instances of liner problems with potential for significant consequences have been identified, the concensus of mill operators and regulatory personnel is that performance of ponds with geomembranes in the U.S. uranium industry has been satisfactory.

NUREG/CR-3891: SAMCR: A TWO-DIMENSIONAL DYNAMIC FINITE ELEMENT CODE FOR THE STRESS ANALYSIS OF MOVING CRACKS. SCHWARTZ, C. W.; CHONA, R.; FOURNEY, W. L.; et al. Maryland, Univ. of, College Park, MD. November 1984. 277pp. 8412100263. 27872:001.

The mathematical formulation, program structure, and details of required input data are described for SAMCR, a two-dimensional dynamic finite element code for the Stress Analysis of Moving CRacks, which has been developed at the University of Maryland. The code has been shown, through an extensive series of verification analyses, to perform well in modeling dynamic behavior of both uncracked and cracked structures. In particular, the code has been demonstrated to provide useful information regarding run-arrest events in polymeric laboratory samples and large thermally shocked steel cylinders. Complete documentation of the code has been included so that this document can serve as both a technical manual and a user's manual for the code.

NUREG/CR-3892: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANICAL EQUIPMENT. Summary Report. KANA, D. D. Southwest Research Institute. August 1984. 43pp. 8409070235. 26418:289.

This document constitutes the Summary for the indicated research contract on equipment seismic qualification methodology. Although the program was conducted by Southwest Research Institute, the results were periodically reviewed by a Peer Review Panel of ten members from various segments of the nuclear industry, and by various members of the NRC staff. In addition, a continuing communication with the IEEE 344 (Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations) revision committee was maintained throughout the program to ensure that the results were disseminated to the industry. Thus, although the results are principally the findings of SWRI, acknowledgement of input from various other sources is recognized.

The program has spanned a period of three years and resulted in seven technical summary reports, each of which covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes. This volume is to summarize the entire program from an overall philosophical point of view.

Volume 1 includes Task 1 Summary Reports parts 1, 2, and 3, which describe evaluations of various aspects of equipment qualifications methodology. Volumes 2, 3, and 4 include the summary reports for Tasks 2, 3, and 4, which are concerned with correlation of methodologies, recommendations for improvements, and evaluation of fragility methodology.

NUREG/CR-3892 V01: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANCIAL EQUIPMENT. Task 1 - Survey Of Methods For Equipment And Components: Evaluation Of Methodology; Qualification And Methodology.... KANA, D. D. ; POLCH, E. Z. ; POMERENING, D. J. ; et al. Southwest Research Institute. August 1984. 393pp. 8409070233. 26413:001.

The Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment has spanned a period of three years and resulted in seven technical summary reports, each of which covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes.

Volume 1 comprises three parts. Part I reviews the methods currently utilized for seismic qualification of nuclear plant equipment with emphasis on qualification by testing. In this review various anomalies that are associated with qualification are identified. Part II provides an in-depth evaluation of the technical issues/anomalies previously identified. Part III provides an evaluation of the method applicable to line mounted items; e.g., valves.

NUREG/CR-3892 V02: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANICAL EQUIPMENT. Task 2-Correlation Of Methodologies For Seismic Qualification Tests Of Nuclear Plant Equipment. KANA, D. D. ; POMERENING, D. J. Southwest Research Institute. August 1984. 105pp. 8409070269. 26409:160.

The Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment has spanned a period of three years and resulted in seven technical summary reports, each of which covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes.

Volume 2 presents a general method for correlating the severity of one seismic qualification motion of given dynamic characteristics to another motion, possibly of different dynamic characteristics. The method provides a method of measuring relative damage severity of two different motions in terms of a relative damage severity ratio.

NUREG/CR-3892 V03: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANICAL EQUIPMENT. Task 3-Recommendations For Improvement Of Equipment Qualification Methodology And Criteria. KANA, D. D. ; POMERENING, D. J. Southwest Research Institute. August 1984. 74pp. 8409070272. 26409:047.

The Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment has spanned a period of three years and resulted in seven technical summary reports, each of which covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes.

Volume 3 presents recommendations for improvement of equipment qualification methodology and criteria. These recommendations are grouped into categories: standardization of procedures, demonstration of adequate methodology, a new methodology and procedural clarification/modification. The fifth category identifies issues where adequate information does not exist to allow a recommendation to be made.

NUREG/CR-3892 V04: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANICAL EQUIPMENT. Task 4 - The Use Of Fragility In Design Of Nuclear Plant Equipment. KANA, D. D.; POMERENING, D. J. Southwest Research Institute. August 1984. 44PP. 8409070302. 26409:119.

The Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment has spanned a period of three years and resulted in seven technical summary reports, each of which have covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes.

Volume 4 presents a study of the use of fragility concepts in the design of nuclear plant equipment and compares the results of state-of-the-art proof testing with fragility testing.

NUREG/CR-3893: LABORATORY STUDIES: DYNAMIC RESPONSE OF PROTOTYPICAL PIPING SYSTEMS. HOWARD, G. E.; WALTON, W. B.; JOHNSON, B. A. ANCO Engineers, Inc. August 1984. 101pp. 8409070292. 26409:258.

This report presents details of the test methods, specimens and a preliminary assessment of results. Two test configurations will be used to achieve the project objectives. Both were three dimensional configurations; the second configuration had branch pipes. The piping systems sustained no apparent damage after being subjected to an earthquake approximately four times greater than the SSE. Additionally, one of the piping systems resisted five OBE's, nine SSE's and nearly thirty shocks.

NUREG/CR-3894: ULTRASONIC AND METALLURGICAL EXAMINATION OF A CRACKED TYPE 304 STAINLESS STEEL BWR PIPE WELDMENT. PARK, J. Y.; KUPPERMAN, D. Argonne National Laboratory. July 1984. 22pp. 8408240322. ANL-84-1. 26255:169.

An ultrasonic in-service inspection (ISI) indicated that a crack had developed in a 22-inch-diameter Type 304 stainless steel pipe manifold endcap weldment of the Hatch-2 boiling water reactor. A section of the weldment was sent to Argonne National Laboratory (ANL) for further examination. The ANL effort included ultrasonic examinations, destructive crack-depth measurements, metallography, degree of sensitization (DOS) measurements, and chemical analyses of material. The results showed that the extent of the cracking was significantly less than indicated by the ISI.

NUREG/CR-3895: INVESTIGATION OF COLD LEG WATER HAMMER IN A PWR DUE TO THE ADMISSION OF ECC DURING A SMALL BREAK LOCA. JACKOBEK, A. B.; GRIFFITH, P. Massachusetts Institute of Technology, Cambridge, MA. September 1984. 60pp. 8410120001. 26986:141.

Experimental studies using a prototypical flow model of a pressurized water reactor (PWR) demonstrate water hammer in the cold legs due to the admission of emergency core cooling (ECC). Such water hammer can occur in an actual PWR during reflood provided there exists a stratified flow of steam and water in the cold legs. The hydraulic are postulated in this report. Calculations, based on a published criterion for water hammer initiation, show that the amount of ECC administered by the high pressure safety injection (HPSI) system, is not great enough to produce liquid depths in the cold leg which can lead to slug formation and subsequent steam bubble collapse water hammer. However, a few water hammers can occur during ECC as the cold leg is being refilled.

A simple analysis developed in this report calculates the water hammer pressures possible under these postulated flow conditions. Potentially dangerous water hammer pressures are predicted during reflood at high system operating pressures characteristic of a small break loss-of-coolant accident (SB-LOCA). Similar calculations done for the geometry of the experimental apparatus were compared to measurements taken during water hammer.

NUREG/CR-3896: SIMULATION EXPERIMENTS COMPARING ALTERNATIVE PROCESS FORMULATIONS USING A FACTORIAL DESIGN. KALUZNY, S. P.; SWARTZMAN, G. L. Washington, Univ. of, Seattle, WA. July 1984. 29pp. 8408060386. 25937:282.

This paper reviews methods for exploring the differences between alternative equations in complex ecosystem models. A factorial design is proposed as a method for exposing possible interactions between equation forms in their effect on model output as well as to clarify differences between the main candidate equations. A number of display methods arising from statistical analysis are used including normal Q-Q plots, linear rank plots, and interaction diagrams. The methods were illustrated using a complex ecosystem model of Lake Ontario. We found the methods effective at illustrating major differences between equations although several difficulties arose due to the complexity of the models and the diffuse nature of the data supporting model validation. Questions of the method for standardization of equation forms so that the compared equations are in some way analogous are important. These methods are probably most useful in cases where the data are of sufficient quality to indicate not only how different equations effect model output but also which forms are to be preferred.

NUREG/CR-3897: EVALUATION OF ECOSYSTEM SIMULATION MODELS AS TOOLS FOR ASSESSMENT OF POWER PLANT IMPACTS ON FISH POPULATIONS. Final Rept. SWARTZMAN, G. L. Washington, Univ. of, Seattle, WA. July 1984. 10pp. 8408010158. 25867:292.

This two-volume report presents the procedures and analyses in developing an approach for structuring expert judgments to estimate human error probabilities. Volume I presents an overview of work performed in developing the approach: SLIM-MAUD (Success Likelihood Index Methodology, implemented through the use of an interactive computer program called MAUD--Multi-Attribute Utility Decomposition). Volume II provides a more detailed analysis of the technical issues underlying the approach.

NUREG/CR-3898: AN EVALUATION OF EFFECTS OF GAMMA IRRADIATION ON THE MECHANICAL PROPERTIES OF HIGH DENSITY POLYETHYLENE. DOUGHERTY, D. R.; ADAMS, J. W.; BARLETTA, R. E. Brookhaven National Laboratory. December 1984. 102pp. 8501030016. BNL-NUREG-51802. 28187:277.

Mechanical tests following gamma irradiation and creep tests during irradiation have been conducted on high-density polyethylene (HDPE) to assess the adequacy of this material for use in high-integrity containers (HICs). Two types of HDPE, a highly cross-linked rotationally-molded material and a non-cross-linked blow molded material, were used in these tests. Gamma-ray irradiations were performed at several dose rates in environments of air, Barnwell and Hanford backfill soils, and ion-excnange resins. The results of tensile and bend testing on these materials following irradiation at 10-11 degrees centigrade showed no effects directly or solely attributable to radiation-induced oxidation. However, effects due to radiation-induced cross-linking, including an increase in yield strength and decreases in both elongation at yield and elongation at break, were observed. Irradiation at 60-63 degrees centigrade showed effects or radiation-induced oxidation including a decrease in yield strength. These effects were more marked in thinner test specimens. Creep testing during irradiation indicated that irradiation increases the creep rate but that the effect is really only significant at creep loads greater than about half the nominal yield strength under the conditions of these tests (10-11 degrees centigrade and 5 krad/h).

NUREG/CR-3899: UTILITY FINANCIAL STABILITY AND THE AVAILABILITY OF FUNDS FOR DECOMMISSIONING. SIEGEL, J. J. Engineering & Economics Research, Inc. September 1984. 28pp. 8410030368. 26817:289.

The NRC is currently developing rulemaking in the area of decommissioning nuclear facilities. A part of that rulemaking effort is assuring that funds will be available at the time of decommissioning of power reactors. Previous NRC reports have examined this issue by studying various funding methods. This report provides an update by analyzing the relative level of assurance of funding methods, considering the present utility financial situation. In its analysis the report makes use of specific case situations. The report concludes that the various funding methods studied in the earlier reports including the internal reserve method provide assurance of the availability of funds for decommissioning.

NUREG/CR-3900 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING. First Quarterly Report, Year Three, April-June 1984. STAHL, D.; MILLER, N. E. Battelle Memorial Institute, Columbus Laboratories. September 1984. 111pp. 8410120024. 26984:346.

Devitrification severity of glass waste forms is being studied in terms of volume fraction of crystallization and crystal grain size. Glass-water contact during the heating and cooling periods of glass leaching experiments is being evaluated for its effect on the overall results of the isothermal period. Modeling efforts included the study of possible colloid formation and the change of water chemistry during glass dissolution. The electrochemical properties of container steels were found to be only slightly affected by the groundwater-species concentration, the presence of basalt rock, or the steels' cleanliness or microstructure. Hydrogen-embrittlement susceptibility may increase at expected repository temperatures. Results of the corrosion-modeling effort suggest that radiolysis may significantly affect general-corrosion kinetics. The water-radiolysis model was extended to account for more groundwater species and was used to predict the concentrations of two species in aqueous iron sulfate; results were compared with experimental data. A method was selected for performing uncertainty analyses of waste-package models. Integral experiments have been designed to address the combined effects of repository conditions on the waste package.

NUREG/CR-3905: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSE EVENT REPORTS. Users Guide. GREENE, N. M.; MAYS, G. T. Oak Ridge National

Laboratory. JOHNSON, M. P. JBF Associates. August 1984. 160pp. 8409270117. ORNL/NSIC-223. 26716:177.

The Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data has developed, through the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory (ORNL), a system to aid in the evaluation of the Licensee Event Reports (LERs) submitted by the nuclear power plant utilities. The primary objective of the Sequence Coding and Search System (SCSS) is to reduce the descriptive text of the incident reports to a coded sequence that is both computer-readable and computer-searchable. This system provides a structured format for detailed coding of component, system, and unit effects, as well as personnel errors. The database contains all current LERs submitted by the nuclear power plant utilities after January 1, 1981, and is updated on a continual basis with new LERs, as they are submitted. The database is maintained by NOAC on the IBM-3033 computer system at ORNL. Following a description of SCSS and structure of the database, a tutorial section is provided to acquaint the first-time user with logon procedures and the necessary commands to retrieve, display, and analyze LERs. Each command is subsequently discussed in detail in the fundamental and advanced command sections.

NUREG/CR-3907: GT2R2: AN UPDATED VERSION OF GAPCON-THERMAL-2. CUNNINGHAM, M. E.; BEYER, C. E. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1984. 70pp. 8410100121. PNL-5178. 26902:223.

The GAPCON-THERMAL-2 code is used by the U.S. Nuclear Regulatory Commission for audit calculations of nuclear fuel thermal performance computer codes. Since the code was originally written, errors and needed updates have been identified. Revision 2 of GAPCON-THERMAL-2 contains a number of coding corrections and updates, and now conforms with the American National Standards Institute FORTRAN-77 standard. The changes to the code are presented in detail. Benchmarking calculations, concentrating on fuel temperatures and fission gas release, were performed to qualify the effect of model changes on the performance of GAPCON-THERMAL-2, Revision 2. It was concluded that use of the old fuel relocation model combined with the modified ANS 5.4 fission gas release model provides the best overall comparison with the thermal performance and fission gas release data used for the benchmarking exercise. The use of the new fuel relocation model combined with the Beyer-Hann fission gas release model provided the best comparisons of thermal behavior but significantly underpredicted fission gas release.

NUREG/CR-3909: SOLIDIFICATION AND LEACHING OF BORIC ACID AND RESIN LWR WASTES. ARRORA, H.; DAYAL, R. Brookhaven National Laboratory. October 1984. 47pp. 8411140096. BNL-NUREG-51805. 27525:214.

Leach testing was conducted on two types of reactor wastes (resin beads from a BWR and boric acid concentrate from a PWR) solidified in cement. In these wastes, Cs-isotopes were the most mobile constituents followed by Sr-90. Co-60 was found to be the least mobile. Effective diffusivities of these radionuclides were approximately equal to 10(-9) cm2/s for Cs-isotopes, approximately equal to 10(-11) cm2/s for Sr-90, and approximately equal to 10(-13) cm2/s for Co-60. A comparison of the release of Cs-137 from these wastes were solidified and leach under previously study in which simulants of these reactor wastes were solidified and leached under identical conditions shows a general correspondence in their release behavior, indicating the leach data derived from testing could be employed to evaluate and predict the release behavior for reactor wastes. Leachability index (LI) values were calculated for determining regulatory compliance of waste forms. The BNL release data meet the proposed NRC guidelines on leachability criteria (LI greater than or equal to 6.0). Also summarized are the limitations in the use of LI values for demonstrating regulatory compliance.

NUREG/CR-3910: DYNAMIC SIMULATION OF THE AIR-COOLED DECAY HEAT REMOVAL SYS OF THE GERMAN KNK-II EXPERIMENTAL BREEDER REACTOR. SCHUBERT, B. K. Brookhaven National Laboratory. November 1984. 56pp. 84.2190376. BNL-NUREG-51806. 28026:255.

A Dump Heat Exchanger and associated feedback control system models for decay heat removal in the German KNK-II experimental fast breeder reactor are presented. The purpose of the controller is to minimize temperature variations in the circuits and, hence, to prevent thermal shocks in the structures. The basic models for the DHX include the sodium-air thermodynamics and hydraulics, as well as a control system. Valve control models for the primary and intermediate sodium flow regulation during post shutdown conditions are also presented. These models have been interfaced with the SSC-L code. Typical results of sample transients are discussed.

NUREG/CR-3911 V01: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE: Quarterly Report, January-March 1984. ATTERIDGE, D. G.; BRUEMMER, S. M.; PAGE, R. E. Battelle Memorial Institute, Pacific Northwest Laboratories. November 1984. 34pp. 8412260135. PNL-5181. 28071:008.

The Division of Enigneering Technology, U.S. Nuclear Regulatory Commission, is sponsoring a program at Pacific Northwest Laboratory to evaluate welded and repair-welded stainless piping for light-water reactor (LWR) service. Stainless steels often become sensitized, or less resistant to stress corrosion cracking (SCC), after undergoing heating and cooling cycles such as those encountered in welding. The weld heat-affected zone (HAZ) is often the site of crack initiation. This program will therefore measure and model the development of a sensitized microstructure and its resultant resistance of SCC in welded and repair-welded stainless steel pipe. The result will be a method to assess the effects of welding variables on the SCC susceptibility of component-specific nuclear reactor welds/repairs. The progress achieved toward this objective during January - March 1984 is described in this report.

NUREG/CR-3918: COMPOSITIONAL EFFECTS ON THE SENSITIZATION OF AUSTENITIC STAINLESS STEELS. BRUEMMER, S. M.; CHARLOT, L. A.; ATTERIDGE, D. G. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1984. 77pp. 8501030041. PNL-5186. 28196:243.

Pacific Northwest Laboratory is conducting a program, sponsored by the U.S. Nuclear Regulatory Commission, to develop validated models for the prediction of stress corrosion cracking susceptibility in the heat-affected zone of stainless steel weldments. This report reviews the effects of alloying and impurity elements on the sensitization propensity of Types 304 and 316 stainless steel. As expected, carbon was found to be the dominant element controlling sensitization, with chromium, molybdenum, and nickel also important. Other alloying elements, such as manganese and silicon, have at most only a small effect on sensitization. However, strongly segregation elements, such as nitrogen, boron, and phosphorus, may have a significant effect on intergranular corrosion and stress corrosion cracking susceptibility.

NUREG/CR-3920: CORCON-MOD2: A COMPUTER PROGRAM FOR ANALYSIS OF MOLTEN-CORE CONCRETE INTERACTIONS. COLE, R. K.; KELLY, D. P.; ELLIS, M. P. Sandia Laboratories. October 1984. 195pp. 8411290068. SAND84-1246. 27690:110.

CORCON is a computer code for modelling the interactions between molten core materials and concrete, such as might occur following a core meltdown accident in a Light Water Reactor. It may also be applied to experiments which simulate such accident conditions. The code predicts the behavior of the system, including heat transfer, concrete ablation, cavity shape change, and gas generation. The first version, CORCON-MOD1, was released in 1981. This report is a complete users' manual and reference for the updated version, CORCON-MOD2. The major changes are the inclusion of models for solidification of the melt and for its (non-explosive) interactions with coolant water. In addition, a number of improvements have been made in response to experience with CORCON-MOD1. The new code remains compatible with the old in the sense that MOD2 will accept any input data set which was accepted by MOD1.

NUREG/CR-3921: DRY SPENT FUEL STORAGE TEST PLAN FOR FINAL NONDESTRUCTIVE FUEL ROD EXAMINATION. OLSEN, C. S. EG&G, Inc. July 1984. 14pp. 8409180283. EGG-2328. 26589:120.

A test plan for the third and final nondestructive examination of eight fuel rods used in a low-temperature, long-term, dry fuel storage program is presented. This examination is part of a long-range project to evaluate the behavior of spent fuel during dry storage conditions. The objective of this project is to provide the Nuclear Regulatory Commission with the information to confirm or establish spent fuel dry storage licensing positions for long-term, low-temperature (<523 K), spent fuel rod behavior during dry storage and for radioactive contamination arising from spallation of cladding crud. This examination consists of visual and photographic examinations, dimensional measurements, and gamma scanning of eight fuel rods.

NUREG/CR-3923: TEST SERIES 1: SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E GOULD NCX-2250 BATTERY CELLS. BONZON, L. L.; HENTE, D. B. Sandia Laboratories. October 1984. 232pp. 8412190357. SAND84-1737. 28026:021.

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 "end-of-life" of a battery, given a seismic event. This report covers the first test series of an extensive program using 12-year old, lead-calcium, Gould NCX-2250 cells, from the James A. Fitzpatrick Nuclear Power Station operated by the New York Authority. Seismic tests with three cell configurations were performed using a triaxial shake table: single-cell tests, rigidly mounted; multi-cell (three) tests, mounted in a typical battery rack; and single-cell tests specifically aimed towards examining propagation of pre-existing case cracks. In general the test philosophy was to monitor the electrical properties including discharge capacity of cells through a graduated series of g-level step increases until either the shake-table limits were reached or until electrical "failure" of the cells occurred. Of nine electrically active cells, six failed during seismic testing over a range of imposed g-level loads in excess of a 1-g ZPA. Post-test examination revealed a common failure mode, the cracking at the abnormally brittle, positive lead bus-bar/post interface; further examination showed that the failure zone was extremely coarse grained and extensively corroded. Presently accepted accelerated-aging methods for qualifying batteries, per IEEE Std. 535-1979, are based on plate growth, but these naturally-aged 12-year old cells showed no significant plate growth.

NUREG/CR-3926 V01: STRATEGIC ANALYSIS FOR SAFEGUARDS SYSTEMS: A FEASIBILITY STUDY. Main Report. SEAVER, D. A. ; GOLDMAN, A. J. ; IMMERMAN, W. H. ; et al. Maxima Corp. December 1984. 81pp. 8501070499. 28240:226.

Strategic analysis (game theory) is a formal method for modeling adversary situations that, when solved, yields an optimal strategy that maximizes the expected payoff to the player. As such, it appears to be potentially applicable in the nuclear material accounting context in which there is potential for an adversary attempting to divert special nuclear material. The NRC has previously supported research to develop preliminary strategic analysis models which has been considered to be only partially successful. This study reviewed previous efforts and other game theory research and assessed the feasibility of: (1) applying strategic analysis in a regulatory framework, (2) making strategic analysis understandable by licensees, and (3) assuring that strategic analysis can effectively be enforced. This report includes a discussion of the role of strategic analysis in material control and accounting, and of the mechanisms by which the NRC could implement strategic analysis. A set of feasibility criteria are described including both technical feasibility and organizational/implementation feasibility. Alternative strategic analysis model options are evaluated with respect to these criteria, as is the current material accounting practice. The assessment determined that the development of a payoff function that adequately represented the NRC's (and therefore the public's) values with respect to the consequences of diversion and the actions taken to prevent it is the most serious impediment to implementation.

NUREG/CR-3926 V02: STRATEGIC ANALYSIS FOR SAFEGUARDS SYSTEMS: A FEASIBILITY STUDY. Appendix. GOLDMAN, A. J. Maxima Corp. December 1984. 130pp. 8501070511. 28223:130.

This appendix provides detailed information regarding game theory (strategic analysis) and its potential role in safeguards to supplement the main body of this report. In particular, it includes an extensive, though not comprehensive review of literature on game theory and on other topics that relate to the formulation of game-theoretic model (e.g., the payoff functions). The appendix describes the basic form and components of game theory models, and the solvability of various models. It then discusses three basic issues related to the use of strategic analysis in material accounting: (1) its understandability; (2) its viability in regulatory settings; and (3) difficulties in the use of mixed strategies. Each of the components of game theoretic model are then discussed and related to the present context.

NUREG/CR-3927: CHARPY TOUGHNESS AND TENSILE PROPERTIES OF A NEUTRON-IRRADIATED STAINLESS STEEL SUBMERGED ARC WELD CLADDING OVERLAY. CORWIN, W. R.; BERGGREN, R. G.; NANSTAD, R. K. Oak Ridge National Laboratory. October 1984. 38pp. 8412100199. ORNL/TM-9309. 27867:270.

It has been proposed that the existence of a tough surface layer of weld-deposited stainless steel cladding on the interior of a reactor pressure vessel (RPV) can keep a short surface flaw from becoming long, either by impeding the initiation of extension of a static flaw and/or by arresting a running flaw. To obtain preliminary material properties typical of those needed to make such an evaluation for light-water reactors (LWRs), a program has been established to obtain data on the degradation (or lack thereof) of the fracture properties of stainless steel weld overlay cladding. A recent review of the literature has indicated that fracture properties of stainless steel weld metal can degrade significantly under irradiation conditions relevant to LWRs. To evaluate this potential degradation, tensile, Charpy V-notch, and precracked Charpy specimens of stainless steel weld overlay cladding were irradiated to about 2 x 10(23) neutrons/m(2) (>1 MeV) at 288 degrees centigrade. The results of tensile and Charpy V-notch tests are reported here and compared with the properties of unirradiated cladding.

NUREG/CR-3928: ASSESSMENT OF REACTOR COOLANT PUMP INSTRUMENTATION IN SUPPORT OF COOLANT INVENTORY TREND ANALYSIS. ARAVE, A. E. EG&G, Inc. October 1984. 46pp. 8411140035. EGG-2333. 27496:308.

Reactor coolant pump motor power and temperature measurements are used by Babcock & Wilcox (B&W) plant owners to calculate void fraction for trending ICC conditions while the pumps are running. This new measurements technology satisfies NUREG-0737, Item II.F.2, "...licensees shall provide ... additional instrumentation ... to supplement existing instrumentation in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling." In this report, the Nuclear Power Plant Instrument Evaluation (NPPIE) project compares system accuracy, capability, and limitations to measurement requirements using small-break test data and full-scale plant analytical studies. Small-break experimental data show that ICC void fraction calculations are conservative compared to gamma densitometer void fraction measurements in the pipe just upstream of the pumps and liquid level conductivity probes in the reactor vessel. Analytical studies verify that a measure of void fraction at the pumps in conservative relative to the desired coolant inventory trend conditions in the reactor vessel.

NUREG/CR-3929: LOSS-OF-BENEFITS ANALYSIS FOR NUCLEAR POWER PLANT SHUTDOWNS. Methodology And Illustrative Case Study. PEERENBOOM, J. P.; BUEHRING, W. A.; GUZIEL, K. A. Argonne National Laboratory. September 1984. 73pp. 8409270130. ANL/AA-29. 26719:287.

A framework for loss-of-benefits analysis and a taxonomy for identifying and categorizing the effects of nuclear power plant shutdowns or accidents are presented. The framework consists of three fundamental steps: (1) characterizing the shutdown; (2) identifying benefits cost as a result of the shutdown; and (3) quantifying effects. A decision analysis approach to regulatory decision making is presented that explicitly considers the loss of benefits. A case study of a hypothetical reactor shutdown illustrates one key loss of benefits: net replacement energy costs (i.e., change in production costs). Sensitivity studies investigate the responsiveness of case study results to changes in nuclear capacity factor, load growth, fuel price escalation, and discount rate. The effects of multiple reactor shutdowns on production costs are also described.

NUREG/CR-3932 BENCHMARK DESCRIPTION OF CURRENT REGULATORY REQUIREMENTS AND PRACTICES IN NUCLEAR SAFETY AND RELIABILITY ASSURANCE. HALVERSON, S. L. ; BEZELLA, W. A. ; CHARAK, I. ; et al. Argonne National Laboratory. August 1984. 115pp. 8410030347. ANL-84-34. 26819:097.

The objectives of this work are to evaluate and benchmark the current safety and reliability assurance-related practices employed by the NRC. This effort represents an initial phase of a program whose overall purpose is to develop a reliability program (RP). A review of NRC regulations relevant to reliability assurance was made for a boiling water reactor using two representative safety systems; the reactor protection system, and the residual heat removal system. The primary sources of information were the standard Review Plan and Title 10 of the Code of Federal Regulations, especially Part 50. In addition, relevant regulatory guides, NRC branch technical positions and industry consensus standard were identified and catalogued for the two reference safety systems over the plant's life cycle. The identified standards and criteria were then organized into a RP element matrix of current regulatory requirements organized by life cycle phase, top level assurance function, and items directly auditable by the NRC. A brief review of the licensing process was also undertaken to indicate the effectiveness of NRC implementation of a RP. The results of this work showed that within the NRC regulations a framework already exists in which to integrate, not add, a reliability assurance program.

NUREG/CR-3933: RISK RELATED RELIABILITY REQUIREMENTS FOR BWR SAFETY -IMPORTANT SYSTEMS WITH EMPHASIS ON THE RESIDUAL HEAT REMOVAL SYSTEM. TZANOS, C. P. ; BEZELLA, W. A. Argonne National Laboratory. August 1984. 140pp. 8410030385. ANL-84-52. 26819:218.

The objective of this study was to identify and evaluate the major safety risk parameters of typical reactor safety systems for use in developing a reliability program. This effort was part of a larger research project aiming to evaluate the feasibility and effectiveness of introducing elements of proven reliability programs from other high technology industries into the nuclear industry. As a reference safety system, the Residual Heat Removal (RHR) system of a Boiling Water Reactor (BWR) was selected. A scoping evaluation was also made for BWR reactor protection system (RPS). Plant information, existing PRA and other relevant analyses, as well as Licensee Event Reports were used as base material for this study. The results of this evaluation indicate that: (1) recovery of faults can have a very significant impact on the reliability requirements, (2) there exists an obvious need for an adequate reliability data base, (3) reliability analyses must be supported by detailed analyses of the plant's response to accident sequences, and (4) the development of effective emergency operating instructions and proper operator training must be one of the major elements of a Reliability Program.

NUREG/CR-3938: REACTIVITY INITIATED ACCIDENT TEST SERIES TEST RIA 1-4 FUEL BEHAVIOR REPORT. COOK, B. A. ; MARTINSON, Z. R. EG&G, Inc. October 1984. 49pp. 8411130557. EGG-2336. 27477:321.

This report presents and discusses results from the final test in the Reactivity Initiated Accident (RIA) Test Series, Test RIA 1-4, conducted in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory. Nine preirradiated fuel rods in a 3 x 3 bundle configuration were subjected to a power burst while at boiling water reactor hot-startup system conditions. The test resulted in estimated axial peak, radial average fuel enthalpies of 234 cal/g UO(2) on the center rod, 255 cal/g UO(2) on the side rods, and 277 cal/g UO(2) on the corner rods. Test RIA 1-4 was conducted to investigate fuel coolability and channel blockage within a bundle of preirradiated rods near the present enthalpy limit of 280 cal/g UO(2) established by the U. S. Nuclear Regulatory Commission. The test design and conduct are described, and the bundle and individual rod thermal and mechanical responses are evaluated. Conclusions from this final test and the entire PBF RIA Test Series are presented.

NUREG/CR-3939: WATER HAMMER, FLOW INDUCED VIBRATION AND SAFETY/RELIEF VALVE LOADS. UFFER, R. A.; VALANDANI, P.; SEXTON, D. Quadrex Corp. September 1984. 86pp. 8410120003. EGG-2340. 26985:096.

This report presents the results of an evaluation performed to determine current and recommended practices regarding the consideration of water hammer flow-induced vibration and safety-relief valve loads in the design of nuclear power plant piping systems. Current practices were determined by a survey of industry experts. Recommended practices were determined by evaluating factors such as load magnitude and frequency content, system susceptibility to loads, frequency of load occurrence and safety effects of postulated piping damage.

This report was prepared for use by the NRC staff in developing positions regarding consideration of dynamic piping loads for use by the NRC's Piping Review Committee.

NUREG/CR-3940: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN ALKALINE LAKE ENVIRONMENTS. SIMPSON, H. J.; TRIER, R. M.; LI, Y. H.; et al. Columbia Univ., New York, NY. August 1984. 124pp. 8409260650. 26702:037.

Measurements of the radioisotope concentrations of a number of elements (Am, Pu, U, Pa, Th, Ac, Ra, Pb, Cs, and Sr) in the water and sediments of a group of alkaline (pH = 9-10), saline lakes demonstrate greatly enhanced soluble-phase concentrations of elements with oxidation states of (III)-(VI) as the result of complexing by carbonate ion. Ratios of soluble radionuclide concentrations in Mono Lake to those in seawater ([CO3 (2-)] in Mono Lake = 200 times that of seawater) were: Pu(=10), (238)U(=150), (231)Pa, (238)Th, (230)Th(=10(3), and (232)Th(=10(5). Effective distribution coefficients of these radionuclides in high CO(3)(2-) environments are several orders of magnitude lower (i.e., less particle reactive) than in most other natural waters. The importance of CO(3)(2-) ion on effective K(d) values was also strongly suggested by laboratory experiments in which most of the dissolved actinide elements became adsorbed to particles after a water sample normally at a pH of 10 was acidified, stripped of all CO(2) and then returned to pH 10 by adding NH(4)OH. Furthermore, the effect complexation by organic ligands is of secondary importance in the presence of appreciable carbonate ion concentration.

Neither pure phase solubility calculations nor laboratory scale K(d) determinations accurately predicted the measured natural system concentrations. Therefore, measurements of the distribution of radionuclides in natural systems are essential for assessment of the likely fate of potential releases from high level waste repositories to groundwater.

NUREG/CR-3941 V01: RADIONUCLIDE MIGRATION AROUND URANIUM ORE BODIES -- ANALOGUE OF RADIOACTIVE WASTE REPOSITORIES. AIREY, P. L. Australia, Govt. of. October 1984. 155pp. 8411060506. AAEC/C40. 27383:300.

A number of uranium ore bodies in the Northern Territory of Australia have been evaluated as geochemical analogues of high-level radioactive waste repositories. The aim of the study is to contribute to the understanding of the scientific basis for the long-term prediction of the transport of radionuclides. Particular attention is being paid to investigations of (i) mechanisms of mobilization and subsequent retardation of uranium series nuclides following the weathering of metamorphic host rocks, (ii) the role of iron minerals in the retardation of uranium and thorium, (iii) the role of groundwater colloids in the transport of radionuclides, (iv) experimental methods for studying the time dependence of adsorption coefficients, and (v) conceptual methods for studying the effect of transport of uranium series nuclides through crystalline host rocks over geological time. The possibility of incorporating certain transuranic and fission product elements into the analogue is discussed.

NUREG/CR-3942: TESTS TO DETERMINE HOW SUPPORT TYPE AND EXCITATION SOURCE INFLUENCE PIPE DAMPING. ARENDTS, J. G.; WARE, A. G. EG&G, Inc. October 1984. 29pp. 8411130732. EGG-2337. 27472:218.

A series of vibration tests was performed on the second configuration of the NRC/EPRI/ANCO piping system at the ANCO Engineers test facility. Excitation was provided by a hydraulic shaker at three different locations/directions using both random and swept-sine-excitation methods. For random excitation, the frequency-response-function, complex-exponential-curve-fit method was used to compute damping values. For swept-sine test, half power-bandwith techniques were used for damping determination. Damping for the lowest three modes was 1 to 3% of critical damping and decreased as frequency increased. A Rayleigh damping curve fit approximated the data well. We conclude as a result of these investigations that type of excitation (random versus swept sine) and type of support (rigid strut, mechanical snubber, hydraulic snubber, rigid strut with gap) has little influence on damping.

NUREG/CR-3951: INTRODUCTION TO BIBELOT: A BIBLIOGRAPHIC FINDING AND RETRIEVAL SYSTEM. COCHRAN, M. I. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1984. 46pp. 8410100776. PNL-5202. 26903:260.

The BIBELOT System of COBOL and Datatrieve programs for bibliographic storage and retrieval is described. The storage scheme is also briefly described. The use of unique citation numbers and user defined keywords is illustrated by many retrieval examples. Finally, typical questions about the use of BIBELOT are answered.

NUREG/CR-3971: A HANDBOOK FOR COST ESTIMATING: A Method For Developing Estimates Of Costs For Generic Actions For Nuclear Power Plants. BALL, J. R. Argonne National Laboratory. COHEN, S. S. Cohen & Associates, Inc. ZIEGLER, E. Z. United Engineers & Constructors, Inc. (subs. of Raytheon Co.). October 1984. 187pp. 8411160028.

ANL/EES-TM-265. 27563:126.

This document provides overall guidance to assist the NRC in preparing the types of cost estimates required by the Regulatory Analysis Guidelines and to assist in the assignment of priorities in resolving generic safety issues. The Handbook presents an overall cost model that allows the cost analyst to develop a chronological series of activities needed to implement a specific regulatory requirement throughout all applicable commercial LWR power plants and to identify the significant cost elements for each activity. References to available cost data are provided along with rules of thumb and cost factors to assist in evaluating each cost element. A suitable code-of-accounts data base is presented to assist in organizing and aggregating costs. Rudimentary cost analysis methods are described to allow the analyst to produce a constant-dollar, lifetime cost for the requirement. A step-by-step example cost estimate is included to demonstrate the overall use of the Handbook.

NUREG/CR-3973: ALTERNATIVE CONTAINERS FOR LOW-LEVEL WASTES CONTAINING LARGE AMOUNTS OF TRITIUM. GAUSE, E. P.; LEE, B. S.; MACKENZIE, D. R.; et al. Brookhaven National Laboratory. November 1984. 151pp. 8412100247. BNL-NUREG-51814. 27864:080.

High-activity tritiated waste generated in the United States is mainly composed of tritium gas and tritium-contaminated organic solvents sorbed onto Speedi-Dri which are packaged in small glass bulbs. Low-activity waste consists of solidified and adsorbed liquids. In this report, current packages for high-activity gaseous and low-activity adsorbed liquid wastes are emphasized with regard to containment potential. Containers for low-level radioactive waste containing large amounts of tritium need to be developed. An integrity may be threatened by: physical degradation due to soil corrosion, gas pressure build-up (due to radiolysis and/or biodegradation), rapid permeation of tritium through the container, and corrosion from container contents. Literature available on these points is summarized in this report.

NUREG/CR-3975: IDENTIFICATION AND ASSESSMENT OF ANTICIPATED MAJOR CHANGES IN CONTROL ROOMS. FORD, R. E.; MEYER, O. R.; BLACKMAN, H. S.; et al. EG&G, Inc. October 1984. 81pp. 8411130734. EGG-2339. 27472:326.

This report fulfills the objective of the Advanced Control Room Concepts Project to identify the major changes and establish the appropriate categories in nuclear power plant control room designs so that a continuum of changes can be identified. A modified Delphi Technique was used for conferring with control room experts to identify the possible control room changes or concepts, and to identify the category into which these concepts belong. The results of the first Delphi conference round were then structured into a multilevel hierarchy. The top level, or focus, of this hierarchy is the control room man machine system. The second level is the identified categories; the third level consists of modifiers to the second level; and the bottom, or fourth, level is the concepts or changes. The second round of the Delphi conference asked the panel to pairwise compare groups of related concepts as to the likelihood of their being used in a backfit or future generation control room. The results of the second Delphi conference round ranks the possible changes from the most likely to the least likely for both the related group of changes and for the top level, the control room man-machine system.

NUREG/CR-3976: REGULATORY ANALYSES FOR SEVERE ACCIDENT ISSUES: AN EXAMPLE. BURKE, R. P.; STRIP, D. R.; ALDRICH, D. C. Sandia Laboratories. December 1984. 127pp. 8501030027. SAND84-1727. 28198:195.

This report presents the results of an effort to develop a regulatory analysis methodology and presentation format to provide information for regulatory decision-making related to severe accident issues. Insights and conclusions gained from an example analysis are presented. The example analysis draws upon information generated in several previous and current NRC research programs (the Severe Accident Risk Reduction Program (SARRP), Accident Sequence Evaluation Program (ASEP), Value-Impact Handbook, Economic Risk Analysis, and studies of vented containment Systems and Alternative Decay Heat Removal Systems) to perform preliminary value-impact analyses on the installation of either a vented containment system or an alternative decay heat removal systen at the Peach Bottom #2 plant. The results presented in this report are "first-cut" estimates, and are presented only for illustrative purposes in the context of this document. This study should serve to focus discussion on issues relating to the type of information, the appropriate level of detail, and the presentation format which would make a regulatory analysis most useful in the decisionmaking process.

NUREG/CR-3980 V01: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS: Quarterly Progress Report, January-March 1984. * Argonne National Laboratory. October 1984. 45pp. 8411130652. ANL-84-61. 27476:328.

This progress report summarizes the Argonne National Laboratory work performed during January, February, and March 1984 on water reactor safety problems related to fuel and fuel cladding materials. The research and development areas covered are Transient Fuel Response and Fission Product Release and Clad Properties for Code Verification.

NUREG/CR-3982: CASE STUDY OF THE PROPAGATION OF A SMALL FLAW UNDER PWR LOADING CONDITIONS AND COMPARISON WITH THE ASME CODE DESIGN LIFE - Comparison Of ASME Code Sections III And XI. YAHR, G. T.; GWALTNEY, R. C.; et al. Oak Ridge National Laboratory. RICHARDSON, A. K. EG&G, Inc. November 1984. 52pp. 8412190379. ORNL-6099. 28026:309.

A cooperative study was performed by EG&G Idaho, Inc. and Oak Ridge National Laboratory to investigate the degree of conservatism and consistency in the ASME Boiler and Pressure Vessel Code Sect. III fatigue evaluation procedure and Sect. XI flaw acceptance standards. A single, realistic, sample problem was analyzed to determine the significance of certain points of criticism of an earlier parametric study by staff members of the Division of Engineering Standards of the Nuclear Regulatory Commission. The problem was based on a semielliptical flaw located on the inside surface of the hot-leg piping at the reactor vessel safe-end weld for the Zion 1 pressurized-water reactor (PWR). Two main criteria were used in selecting the problem: first, it should be a straight pipe to minimize the computational expense; second, it should exhibit as high a cumulative usage factor as possible. Although the problem selected has one of the highest cumulative usage factors of any straight pipe in the primary system of PWRs, it is still very low. The Code Sect. III fatigue usage factor was only 0.00046, assuming it was in the as-welded condition, and fatigue crack-growth analyses predicted negligible crack growth during the 40-year design life. When the analyses were extended past the design life, the usage factor was less than 1.0 when the flaw had propagated to failure. The current study shows that the criticism of the earlier report should not detract from the conclusion that if a component experiences a high level of cyclic stress corresponding to a fatigue usage factor near 1.0, very small cracks can propagate to unacceptable sizes.

NUREG/CR-3985: ORGANIC COMPLEXANT-ENHANCED MOBILITY OF TOXIC ELEMENTS IN LOW-LEVEL WASTES: Annual Report -- July 1983 - June 1984. SWANSON, J.L. Battelle Memorial Institute, Pacific Northwest Laboratories. November 1984. 55pp. 8412070092. PNL-4965-4. 27839:006.

This report describes the initial results obtained in a project whose objective is to determine how and to what extent organic complexants affect the mobility of toxic elements in subsurface groundwaters at commercial low-level waste disposal sites. Generic soil components (e.g., hydrous oxides, silica, clays) are being employed so that the results will be broadly applicable. Organic complexants used in the nuclear industry are being emphasized. Data have been obtained with two radioactive (Ni and Pu) and one nonradioactive toxic element (Cd). Work with Ni has been emphasized; it was studied with five different generic soil components (hydrous ferric oxide, silica, titania, kaolinite, and montmorillonite) and five different complexants (EDTA, NTA, picolinate, citrate, and oxalate). EDTA was the complexant studied most extensively and hydrous ferric oxide was the most studied soil component. A wide diveristy of effects of organic complexants on toxic elements sorption was observed. The effects vary not only among complexants, but also among toxic elements and among soil components. In some systems the complexant results in increased toxic element sorption (decreased mobility) while in other systems the complexant results in decreased toxic element sorption (increased mobility).

NUREG/CR-3986: THERMAL-HYDRAULIC PROCESS MODELING IN RISK ANALYSIS: AN ASSESSMENT OF THE RELEVANT SYSTEMS, STRUCTURES AND PHENOMENA. WEIGAND, G.G. Sandia Laboratories. December 1984. 645pp. 8501030034. SAND84-1219. 28191:001.

The MELCOR project is developing a new generation of risk assessment computer programs for analysis of severe accidents at nuclear power plants. As part of this project, a three-part study was conducted to identify the relevant phenomena and models required for performing these PRA studies. Evaluations were performed in (1) thermal-hydraulics, (2) fission product behavior, and (3) health and environmental consequences areas. This document details the findings in the thermal-hydraulic areas for PWRs and areas common to both PWRs and BWRs. A separate BWR specific report has been published by Oak Ridge National Laboratory. Reports are being prepared for the other two topics. The study, performed by specialists from each of the various nuclear plant design or analysis areas, found that the current level of thermal-hydraulic modeling that exists for performing risk assessments is typified by the modeling in the MARCH code. This level of modeling of important phenomena, particularly invessel phenomena, systems, and structures. Pressurized water reactor modeling was found to be more complete than modeling for boiling water reactor designs, particularly in containment. Finally, although reactor cavity modeling was considered essential for risk assessments, the lack of adequate modeling found for the cavity was identified as a serious impediment to the development of second generation risk codes.

NUREG/CR-3988: MARCH 2 (MELTDOWN ACCIDENT RESPONSE CHARACTERISTICS) CODE DESCRIPTION AND USERS MANUAL. WOOTEN, R. O. ; CYBULSKIS, P. QUAYLE, S. F. Battelle Memorial Institute, Columbus Laboratories. September 1984. 419pp. 8410170214. BMI-2115. 27030:001.

MARCH 2 describes the response of water-cooled reactor systems to severe accidents, particularly those leading to core meltdown. The code performs the calculations from the time of accident initiation through the stages of coolant blowdown and boiloff, core heat up and meltdown, pressure vessel bottom head melting and failure, and debris-water and debris-concrete interactions in the reactor cavity. Both the primary system and the building are modeled. Mass and energy additions to the containment building are evaluated and the pressure-temperature response of the containment with or without engineered safety features is calculated. A maximum of eight containment sub-volumes may be modeled. Engineered safety features modeled include emergency core cooling systems, containment sprays, building coolers and fans, suppression pool and ice condenser containments, and emergency core cooling and spray heat exchangers. Effects of metal-water reactions, combustion of hydrogen and carbon monoxide, heat losses to containment structures, and redistribution of the decay heat due to loss of volatile fission products from the core are considered. MARCH 2 is intended to replace the earlier MARCH 1 code. It is written in FORTRAN 77 to improve transportability.

NUREG/CR-3993: GEOCHEMICAL INVESTIGATIONS AT MAXEY FLATS RADIOACTIVE WASTE DISPOSAL SITE. DAYAL, R. ; PIETRZAK, R. F. ; CLINTON, J. Brookhaven National Laboratory. October 1984. 143pp. 8411130577. BNL-NUREG-51820. 27474:175.

As part of the NRC efforts to develop a data base on source term characteristics for low level wastes, Brookhaven National Laboratory (BNL) has produced and analyzed a large amount of data on trench leachate chemistry at existing shallow land burial sites. In this report, we present the results of our investigations at the Maxey Flats, Kentucky disposal site. In particular, data on trench leachate chemistry are reviewed and discussed in terms of mechanisms and processes controlling the composition of trench solutes. Particular emphasis is placed on identifying both intra- and extra-trench factors and processes contributing to source term characteristics, modifications, and uncertainties. The problems associated with unsegregated, poorly packaged, and unstabilized wastes encountered at the Maxey Flats disposal site point to the need for waste segregation, improved stabilization, and proper packaging. Stabilized, packaged waste not only ensures trench stability but also decreases the rate and extent of leaching and microbial degradation of buried waste. In addition, the uncertainties in the source term are reduced.

NUREG/CR-3994: BURYIT/ANALYZ: A COMPUTER PACKAGE FOR ASSESSMENT OF RADIOLOGICAL RISK OF LOW-LEVEL RADIOACTIVE WASTE LAND DISPOSAL. FISHER, J. E. ; COX, N. D. ; ATWOOD, C. L. EG&G, Inc. November 1984. 133pp. 8501030327. EGG-2345. 28196:001.

This report is a user's manual for a partially completed code for risk assessment of a low-level waste shallow-land burial site, to be used in the licensing of burial sites. This code is intended as a tool to be used for considering nuclide transport mechanisms, including atmospheric, groundwater, erosion, and infiltration to an underlying aquifer. It also calculates doses to individuals and the population through direct exposure, inhalation, and ingestion. The methodology of the risk assessment is based primarily on the response surface method of uncertainty analysis. The parameters of a model for predicting dose commitment due to a release are treated as statistical variables in order to compute statistical distributions for various dose commitment contributions. The likelihood of a release is also accounted for by statistically evaluating the arithmetic product of the dose commitment distributions with the probability of release occurrence. An example is given using the atmospheric the atmospheric transport pathway as modeled by a code called BURYIT. The framework for using other release pathways is described in this manual. Information on parameter uncertainties, reference site characteristics, and probabilities of release events is included.

NUREG/CR-3995: HYDRODYNAMICS OF COUNTER-CURRENT TWO PHASE FLOW THROUGH POROUS MEDIA. MARSHALL, J. S. ; DHIR, V. K. California, Univ. of, Los Angeles, CA. December 1984. 223pp. 8412270181. 28116:044.

Understanding of the hydrodynamic characteristics and flow limitations of two phase flow through porous media is necessary to evaluate the coolability of a top flooded degraded nuclear reactor core. In the present work, an analytical and experimental investigation of counter-current two phase flow through 80-100 cm deep porous layers composed of both uniform size spherical particles (nominal diameters 1-19 mm), mixtures of spherical particles and nonspherical "sharps" has been performed. The porous layers were formed in a 20 cm diameter plexiglass tube. Water and air were used as the test fluids, with superficial velocities ranging from 0-19.5 mm/s and 0-163 mm/s, respectively. Bed porosity, mean particle diameter, flooding limits, and void fraction and pressure gradient at flooding were investigated. An analytical approach based upon geometrical models was used to derive expressions for porosity and mean particle diameter. An empirical correlation has been found for the flooding data which is slightly different than that found in the literature. The effect of coupling of the overlying liquid layer with the bed and of axial gas injection upon the flooding limit were also studied. The results of these hydrodynamic investigations were applied to obtain dryout heat flux in bottom and volume heated particulate beds.

NUREG/CR-3996: RESPONSE MARGINS OF THE DYNAMIC ANALYSIS OF PIPING SYSTEMS. JOHNSON, J. J. ; BENDA, B. J. ; CHUANG, T. Y. ; et al. Lawrence Livermore National Laboratory. October 1984. 75pp. 8411140107. UCID-20067. 27509:162.

The seismic response of piping systems is frequently separated into two parts -- the inertial response and the pseudostatic response. Various analysis procedures have been developed to calculate each portion of the response separately. The analysis procedures used are frequently simplified and, in so doing, introduce significant conservatism. Conservatism in the US NRC SRP response spectrum analysis methodology is quantified here as measured against a multi-support time history analysis procedure. Also, best estimate piping system responses are compared to design values which is valuable to seismic PRA applications.

NUREG/CR-3998 V01: LIGHT WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS: Quarterly Progress Report, January-March 1984. SHACK, W. J. Argonne National Laboratory. October 1984. 99pp. 8411140111. ANL-84-60 V01. 27509:237.

This progress report summarizes the Argonne National Laboratory work performed during January, February, and March 1984 on water reactor safety problems related to out-of-core materials. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems, and Nondestructive Evaluation and Leak Detection.

NUREG/CR-4001: CONTEMPT4/MOD5: AN IMPROVEMENT TO CONTEMPT4/MOD4 MULTICOMPARTMENT CONTAINMENT SYSTEM ANALYSIS PROGRAM FOR ICE CONTAINMENT ANALYSIS. LIN,C.C. Brookhaven National Laboratory. September 1984. 40pp. 8410180123. BNL-NUREG-51824. 27045:289.

CONTEMPT4 is a digital computer program for multicompartment containment system analysis. Previous version of the CONTEMPT4 code, MOD4, consists of an implicit algorithm to computer junction flow when numerically induced flow oscillations are encountered. This document presents analytical model and UPDATE statements that are required to extend the capability of the MOD4 implicit routine for ice containment analysis. A sample problem is analyzed both with and without the use of the implicit routine to demonstrate the effectiveness and the need of an implicit algorithm for such problems.

NUREG/CR-4002: DEVELOPMENT AND APPLICATION OF ECONOMETRIC DEMAND AND SUPPLY MODELS FOR SELECTED CHESAPEAKE BAY SEAFOOD PRODUCTS. NIEVES,L.A.; MOE,R.J. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1984. 180pp. 8412260052. PNL-5226. 28069:001.

Five models were developed to forecast future Chesapeake seafood product prices, harvest quantities, and resulting income. Annual econometric models are documented for oysters, hard and soft blue crabs, and hard and soft clams. To the degree that data permit, these models represent demand and supply at the retail, wholesale, and harvest levels. The resulting models have broad applications in environmental policy issues and regulatory analyses for the Chesapeake Bay.

NUREG/CR-4007: LOWER LIMIT OF DETECTION: DEFINITION AND ELABORATION OF A PROPOSED POSITION FOR RADIOLOGICAL EFFLUENT AND ENVIRONMENTAL MEASUREMENTS. CURRIE,L.A. Commerce, Dept. of, National Bureau of Standards. September 1984. 153pp. 8410170308. 27031:060.

A manual is provided to define and illustrate a proposed use of the Lower Limit of Detection (LLD) for Radiological Effluent and Environmental Measurements. The manual contains a review of information regarding LLD practices gained from site visits; a review of the literature and a summary of basic principles underlying the concept of detection in Nuclear and Analytical Chemistry; a detailed presentation of the application of LLD principles to a range of problem categories (simple counting to multinuclide spectroscopy), including derivations, equations, and numerical examples; and a brief examination of related issues such as reference samples, numerical quality control, and instrumental limitations. An appendix contains a summary of notation and terminology, a bibliography, and worked-out examples.

NUREG/CR-4011: THE 21/55 DATA BASE USER'S MANUAL. SILVER,E.G. Oak Ridge National Laboratory. September 1984. 317pp. 8410120007. ORNL/NSIC-221. 26982:001.

The Nuclear Regulatory Commission's Office for the Analysis and Evaluation of Operational Data has developed, through the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory (ORNL), a data base for storing and organizing information obtained from the reports on construction deficiencies (CDRs) submitted to NRC under the requirements of 10 CFR 21 and 10 CFR 50.55(e) by holders of construction permits for nuclear facilities. The computerized data base stores coded and textual information about the reports issued and the events to which they refer, including such data as dates of events and reports, affected systems and components, source of information, manufacturers and vendors of affected components and the like. There is also provision for direct access to the data base by NRC Headquarters and Field Office staff both for accessing the information in the data base, and for entry of specific data concerning assignments of NRC follow-up staff and resolution actions taken. The document includes a tutorial guide for novice users of the data base. A system of access control to assure the integrity of the NRC-input data was developed and is described.

NUREG/CR-4012: REPLACEMENT ENERGY COSTS FOR NUCLEAR ELECTRICITY-GENERATING UNITS IN THE UNITED STATES. VANKUIKEN, J. C.; BUEHRING, W. A.; GUZIEL, K. A. Argonne National Laboratory. October 1984. 219pp. 8411080457. ANL-AA-30. 27397:161.

Seasonal replacement energy costs are estimated for potential short-term shutdowns of 108 nuclear electricity-generating units. These estimates were developed to help the Nuclear Regulatory Commission 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 two-year period beginning with fall 1984 and ending with summer 1986.

NUREG/CR-4014: LABORATORY MEASUREMENTS OF PARAMETERS AFFECTING WET DEPOSITION METHYL IODIDE. VOILLEQUE, P. G.; MAECK, W. J.; HONKUS, R. J.; et al. Idaho National Engineering Laboratory. November 1984. 53pp. 8412190378. WINCO-1023. 28027:305.

The transfer of gaseous methyl iodide (CH(3)I) to raindrops and the initial retention by vegetation of CH(3)I in raindrops have been studied in a laboratory experimental program. The measured air-to-drop transfer parameters and initial retention factors both affect the wet deposition of methyl iodide onto vegetation. No large effects on the air-to-drop transfer due to methyl iodide concentration, temperature, acidity, or rain type were observed. Differences between laboratory measurements and theoretical values of the mass transfer coefficient were found. Grass, lettuce, and alfalfa were used to study initial retention of methyl iodide by vegetation. Only a small fraction of the incident CH(3)I in raindrops was held by any of the three vegetation types.

NUREG/CR-4017: INTERIM CRITERIA FOR THE USE OF PROGRAMMABLE DIGITAL DEVICES IN SAFETY AND CONTROL SYSTEMS. ADAMS, D. M.; SVOBODA, J. M. EG&G, Inc. December 1984. 38pp. 8501030040. EGG-2348. 28196:318.

Proposed criteria for the application of stored program, digital computers in commercial nuclear power plants is presented. This report emphasizes recommendations for the design of computer systems and recommends a method for the regulatory review of computer system designs. More restrictive requirements are made for protection systems than control systems or other plant computer systems. In making these recommendations, the study team reviewed current regulations, past Nuclear Regulatory Commission reviews of computer systems, the work done by other government agencies, and the work done by many other countries. The results of this study provide a classification of systems, a recommended design method, and a specification of design issues to be resolved during the design and development of digital computer systems. Also included is a recommendation of subject areas that need further research activity. This report is part of a larger program to research computer system design issues, to develop design criteria (hardware and software) for Safety Parameter Display Systems, to research software quality assurance, to provide a comparative risk assessment of digital technology, and to develop electrical isolation criteria.

NUREG/CR-4018: SSI SENSITIVITY STUDIES AND MODEL IMPROVEMENTS FOR THE US NRC SEISMIC SAFETY MARGINS RESEARCH PROGRAM. JOHNSON, J. L.; MASLENIKOV, O. R.; BENDA, B. J. Lawrence Livermore National Laboratory. November 1984. 134pp. 8412070084. UCID-20212. 27838:235.

The Seismic Safety Margins Research Program (SSMRP) is a US NRC-funded program conducted by Lawrence Livermore National Laboratory. Its goal is to develop a complete fully coupled analysis procedure for estimating the risk of an earthquake-induced radioacive release from a commercial nuclear power plant. In Phase II of the SSMRP, the methodology was applied to the Zion nuclear power plant. Three topics in the SSI analysis of Zion were investigated and reported here--flexible foundation modeling, structure-to-structure interaction, and basemat uplift. The results of these investigations were incorporated in the SSMRP seismic risk analysis.

NUREG/CR-4019: A NEW STEAM COOLED REACTOR. EDLUND, M. C.; SCHULTZ, M. Virginia Polytechnic Institute & State Univ., Blacksburg, VA. November 1984. 176pp. 8412210140. 28049:110.

A novel concept is described for a nuclear power plant that is ultra safe based on current knowledge of nuclear reactor safety. Both burner and breeder-type cores are studied. The concept utilizes steam-cooling during normal operation with automatic shutdown and heat removal by natural convection under off-normal conditions.

NUREG/CR-4021: VERIFICATION OF EXPERIMENTAL MODAL MODELING USING HDR (HEISSDAMPFREAKTOR) DYNAMIC TEST DATA. SRINIVASAN, M. G.; KOT, C. A.; HSIEH, B. J., et al. Argonne National Laboratory. October 1984. 107pp. 8412190380. ANL-84-25. 28027:199.

An attempt to verify the reliability of the experimental modal modeling code, MODAL-PLUS, is described in this report. MODAL-PLUS is capable of synthesizing a modal model of a structure using data from dynamic testing of a structure. The objective was to determine whether a modal model synthesized from one set of test data would be capable of correctly predicting response to a different form of excitation from a different set of data. Recorded test data from the shaker and rocket tests on the containment building of the HDR (Heissdampfreaktor) were used in the effort. The attempt verification was only partially successful in that only one modal model with a limited range of validity could be synthesized from the shaker test data. The goodness of fit in this limited range was adequate. The rocket test data could not be used to synthesize a modal model due to numerical difficulties. However, the effort was useful in showing the need for taking into account the possible use of the data, and the data analysis method to be employed, at an early stage when the tests are being designed.

NUREG/CR-4028: UNIFIED THEORY FOR PREDICTING MAXIMUM FLUID PARTICLE SIZE FOR DROPS AND BUBBLES. KOCAMUSTAGAOGUL; CHEN, I.Y.; ISHII, M. Argonne National Laboratory. October 1984. 47pp. 8411280518. ANL-84-67. 27682:108.

A simple model is developed based on a two-dimensional linearized Kelvin-Helmholtz stability theory to describe the breakup of drops and bubbles in fluid media. Breakup is predicted to occur if the growth of disturbances at the interface is faster than the rate at which disturbances propagate around the interface to the side of particle. Agreement between the model and experimental data indicates that the principle physical mechanisms involved are properly accounted for by the model. The same theory is applicable to drops in liquid, drops in gas, and bubbles in liquid. The present analysis gives the first unified theory for fluid particle breakups which has not been available previously.

NUREG/CR-4029: LOCAL FORMULATION OF INTERFACIAL AREA CONCENTRATION AND ITS MEASUREMENTS IN TWO-PHASE FLOW. KATAOKA, I.; ISHII, M.; SERIZAWA, A. Argonne National Laboratory. October 1984. 72pp. 8412100274. ANL-84-68. 27884:238.

The interfacial area concentration is one of the most important parameters in analyzing two-phase flow based on the two-fluid model. The local instantaneous formulation of the interfacial area concentration is introduced here. Based on this formulation, time and spatial averaged interfacial area concentrations are derived, and the local ergodic therorem (the equivalency of the time and spatial averaged values) is obtained for stationary developed two-phase flow. On the other hand, the global ergodic theorem is derived for general two-phase flow. Measurement methods are discussed in detail in relation to the present analysis. The three-probe method, with which local interfacial area concentration can be measured accurately, has been proposed. The one probe method under some statistical assumptions has also been proposed. In collaboration with the experimental data for the interfacial velocity, radial profiles of the local interfacial area concentration are obtained based on the one probe method. The result indicates that the local interfacial area concentration has a peak value near the tube wall in bubbly flow, while in slug flow it shows a higher value in the central region of the tube for that particular set of data.

NUREG/CR-4031: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Simulator Experiments." BALDWIN, C.A.; KAM, F.B.; STALLMAN, F.W. Oak Ridge National Laboratory. December 1984. 33pp. 8501030063. ORNL/TM-9423. 28190:278.

Three neutron dosimetry experiments were performed at the Oak Ridge Research Reactor Poolside Facility to study the feasibility of using the facility for the Fifth Nuclear Regulatory Commission Heavy Section Steel Technology Metallurgical Irradiations. The first two experiments revealed the original experimental configuration to be inadequate because the fluence rates estimated from the measured saturation activities were too low. In response to this, the core loading was changed and the entire experimental facility was moved closer to the core. A third experiment was performed and the resulting saturation activities and fluence rate estimates increased by approximately 40% at the points of interest. The latter fluence rate estimates were considered satisfactory, so no further changes were necessary. This report describes the three characterization experiments in detail and gives all measurement results. An analysis of the results with regard to consistency and measurement uncertainty is also presented. It is shown that the experimental results are consistent within uncertainty bounds.

NUREG/CR-4032: DETECTION OF STEAM GENERATOR TUBE LEAKS IN PRESSURIZED WATER REACTORS. ROACH, W.H. EG&G, Inc. November 1984. 22pp. 8501040073. EGG-2351. 28216:278.

This report addresses the early detection of small steam generator tube leaks in pressurized water reactors. It identifies physical parameters, establishes instrumentation performance goals, and specifies sensor types and locations. It presents a simple algorithm that yields the leak rate as a function of known or measurable quantities. Leak rates of less than one-tenth gram per second should be detectable with existing instrumentation.

NUREG/CR-4034: A STUDY OF DROPLET HYDRODYNAMICS ACROSS A GRID SPACER. LEE, S.L.; CHO, S.K.; SHEEN, H.J. New York, State Univ. of, Stony Brook, NY. November 1984. 65pp. 8412070080. 27838:167.

The results of measurement of droplet size and velocity upstream and downstream of the grid spacer are reported. All experiments were performed at ambient atmospheric pressure and room temperature. The Laser Doppler Anemometer (LDA) system was used for simultaneous measurement of size and velocity of a droplet, and at every measurement location two size measurement schemes were employed. These measurement results show that some of the thermally inactive large droplets (greater than 1mm) are intercepted by the grid spacer and broken down to thermally more active smaller droplets (less than 100 microns) after the grid spacer. From the obtained data, Sauter mean diameter and area density were calculated. The correlations on the Sauter mean diameter and area density were calculated. The correlations on the Sauter mean diameter of the large and small droplets downstream of the grid spacer are reported. In addition, correlations on the fractional volume of the small droplets and nozzle characteristics are reported respectively. These results show that after liquid droplets' breakage, the total surface area of the droplets is greatly increased. This phenomena will lead to enhanced cooling of fuel rods in nuclear reactor during reflood for a postulated large loss of coolant accident.

NUREG/CR-4058 V01: A STUDY OF SEISMICITY AND EARTHQUAKE HAZARD IN NORTHERN ALABAMA AND ADJACENT PARTS OF TENNESSEE AND GEORGIA, Vol. 1. DAINTY, A.M.; LONG, L.T.; LIOW, J. Georgia Institute of Technology, Atlanta, GA. December 1984. 44pp. 8501030013. 28194:204.

The Georgia Tech-Geological Survey of Alabama Seismic Network has been in operation in Alabama since April 1981 and in southeast Tennessee since April 1982. During this time 83 events have been located. In Alabama, the distribution of epicenters generally confirms a broad northeast-southwest trend and a north-south trend.

In southeast Tennessee, four trends running approximately east-west were noted. Few focal depths have been obtained; those that have lie between 9 and 15 km in the Greenville basement below the Pelozoic Valley Ridge sediments. Cumulative frequency-magnitude plots using m(b)(Lg) estimated from duration indicate a slope of 0.65 in southern Tennessee; this is consistent with the Alabama data. Events tend to be closely paired in space and time (within one day). A preliminary determination of crustal structure in Alabama is presented in Appendix 1. The average crustal velocities observed are 6.15 km/sec for P waves and 3.55 km/sec for S waves with an average crustal thickness of 35 km. Conclusive evidence of a high velocity layer in the crust has not yet been found.

NUREG/CR-4070 V01: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS. Volume 1: Correlation of Bivalve Biological Characteristics And Raw-Water System Design. NEITZEL, D. A.; JOHNSON, K. I.; PAGE, T. L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. December 1984. 149pp. 8501030330. PNL-5300. 28194:065.

Fouling of raw water systems in nuclear power plants in the United States can affect the safe operation of a power plant. This report describes correlations between the biology of bivalve organisms and the design and operation of power plants that allow bivalves to enter and reside in nuclear power plants. Discussions are focused on safety-related raw-water systems subject to fouling by the Asiatic clam (Corbicula fluminea), the blue mussel (Mytilus edulis), and the American oyster (Crassostrea virginica). Score sheets to rate fouling potential of power plant systems and components are provided.

Contractor Report Number Index

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