NUREG-0304
Vol. 9, No. 2

Regulatory and Technical Reports

Compilation for
Second Quarter 1984
April - June

U.S. Nuclear Regulatory Commission

Office of Administration

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Date Published: August 1984

Division of Technical Information and Document Control
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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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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Washington, D.C. 20555

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

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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 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 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.

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 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, V08, N03 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-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.

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 January 1984 through March 1984. Also included in this issue are the results of certain inspections performed prior to January 1984 that were not included in previous issues of NUREG-0040.

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

This compilation lists all NRC regulatory and technical reports published under the NUREG series during the first quarter of 1984.

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

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-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.

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 V06 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-29, 1984. * Division of Technical Information & Document Control. April 1984. 669pp. 84052200?2. 24553:001.

See NUREG-0540, V06, N01 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, V06, N01 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, V06, N01 abstract.

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.

Provides an overview of the status of the progress and plans for resolution of the generic tasks addressing "Unresolved Safety Issues" as reported to Congress.

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-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-0748 V04 N02: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of February 29, 1984. (Orange Book) * Management Information Branch. April 1984. 150pp. 8404240178. 24190:001.

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

NUREG-0748 V04 N03: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of March 31, 1984. (Orange Book) * Management Information Branch. May 1984. 355pp. 8405210574. 24530:001.

See NUREG-0748, V04, N02 abstract.

NUREG-0748 V04 N04: OPERATING REACTORS LICENSING ACTIONS SUMMARY. Data As Of April 30, 1984. (Orange Book) * Management Information Branch. June 1984. 384pp. 8406210444. 25099:073.

See NUREG-0748, V04, N02 abstract.

NUREG-0750 V17: NUCLEAR REGULATORY COMMISSION ISSUANCES. January-June 1983. Pages 1-1,196. * Division of Technical Information & Document Control. June 1983. 1,268pp. 8406200559. 25059:019.

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

NUREG-0750 V18 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-December 1983. * Division of Technical Information & Document Control. December 1983. 131pp. 8406280209. 25209:001.

See NUREG-0750, V17 abstract.

NUREG-0750 V18 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES. December 1983. Pages 1,303-1,482. * Division of Technical Information & Document Control. December 1983. 179pp. 8405220259. 24603:012.
See NUREG-0750,V17 abstract.

NUREG-0750 V19 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES. January 1984. Pp 1-485. * Division of Technical Information & Document Control. January 1984. 487pp. 8407130479. 25582:311.
See NUREG-0750,V17 abstract.

NUREG-0750 V19 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES. February 1984. Pp 487-554. * Division of Technical Information & Document Control. February 1984. 75pp. 8407130393. 25575:242.
See NUREG-0750,V17 abstract.

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 198-. 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-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 06.3 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 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-0826: 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-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-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-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-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-0936 V03 N01: NRC REGULATORY AGENDA. Quarterly Report, January-March 1984. * Division of Rules and Records. April 1984. 182pp. 8405020032. 24287:128.

The NRC Regulatory Agenda is a compilation of all rules on which the NRC has proposed or is considering action and all petitions for rulemaking which have been received by the Commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued each quarter. The Agendas for April and October are published in their entirety in the Federal Register while a notice of availability is published in the Federal Register for the January and July Agendas.

NUREG-0940 V03 N01: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Quarterly Progress Report (January - March 1984). * Director's Office, Office of Inspection and Enforcement. 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-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-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-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-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-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-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-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-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. * 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-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-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. * 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-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-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-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-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) * Office of Nuclear Reactor Regulation, Director. 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-1077: ENVIRONMENTAL IMPACT APPRAISAL 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-109[illegible] U.S. NUCLEAR REGULATORY COMMISSION 1983 ANNUAL REPORT.
MAHER, W[illegible] Office of Resource Management, Directo.. 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/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/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.

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, Instruction 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 form is initiated; the keywords are assigned by the computer using correlation tables from the Sequence and Search System.

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, V03, N3 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,V03,N3 abstract.

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-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-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 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-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-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-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-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-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-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-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 with 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-3295 V01: LIGHT WATER REACTOR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM: Notch Ductility & Fracture Toughness Degradation of A302-B & 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 spe mens 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) ith 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 & Tensile Strength Determinations For PSF Simulated Surveillance & 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-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-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-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-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-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-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-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-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-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 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-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. 8406230213. 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-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. KENDORSKI, 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-3504: TURBULENCE MODELING IN THE COMMIX COMPUTER CODE. CHEN, 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-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-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-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-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-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-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-9744-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-3572: DETERMINATION OF METABOLIC DATA APPROPRIATE FOR HLW DOSIMETRY (ICRP-30), I. ECKERMAN, K.F.; LEGGETT, R.W.; MEYER, R.; 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-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-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-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-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 determined 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-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. 24893:004.

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-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-3608: RELAP5 ASSESEMENT: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-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-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-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-3627: FRANTIC II APPLICATIONS TO STANDBY SAFETY SYSTEMS. GINZBURG, 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.; 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. SAND83-2652. 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-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 implementation 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-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 downcomer 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-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-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-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-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-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-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-9038. 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 la, e. 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-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-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 form: -(dp/dx)= av + bv(2), where a and b are constant coefficients defined by 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. 8405210582. 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-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-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 human 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-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 CLAVERT 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 flow 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-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-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-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-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-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.; FAIGENBLUM, 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-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 full 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-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 flau 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-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-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-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.; RODRIQUES, 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-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-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-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-3781 DRFT: PCI-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-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-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-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-3305: 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-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-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-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-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, Univ. of, Norman, 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-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.

Contractor Report Number Index

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

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

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OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 4/05/81)

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DIVISION OF RISK ANALYSIS & OPERATIONS (POST 840429)
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NUREG-0020 V08 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As of March 31, 1984. (Grey Book)
NUREG-0020 V08 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT. Data As Of April 30, 1984. (Grey Book)

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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.

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DIVISION OF SAFETY TECHNOLOGY

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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...."

NRC Contract Sponsor Index (Contractor Reports)

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

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

DIRECTOR'S OFFICE
NUREG/CR-2000 V03 N3: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of March 1984.
NUREG/CR-2000 V03 N4: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of April 1984.
NUREG/CR-2000 V03 N5: LICENSEE EVENT REPORT (LER) COMPILATION: For Month Of May 1984.

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DIVISION OF EMERGENCY PREPAREDNESS & ENGINEERING RESPONSE (POST 830103)
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NUREG/CR-3754: FAILURE EVALUATION OF GENERAL ELECTRIC SB-1 AND SB-9 REACTOR MODE SWITCHES.

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

DIVISION OF WASTE MANAGEMENT
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NUREG/CR-2614: IDENTIFICATION OF CHARACTERISTICS WHICH INFLUENCE REPOSITORY DESIGN - TUFF.
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NUREG/CR-3316: VERIFICATION AND FIELD COMPARISON OF THE SANDIA WASTE-ISOLATION FLOW AND TRANSPORT MODEL (SWIFT).
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NUREG/CR-3572: DETERMINATION OF METABOLIC DATA APPROPRIATE FOR HLW DOSIMETRY (ICRP-30), I.
NUREG/CR-3774 V01: ALTERNATIVE METHODS FOR DISPOSAL OF LOW-LEVEL

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RADIOACTIVE WASTES. Task 1: Description of Methods And Assessment Of Criteria.

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OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR

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NUREG/CR-3781 DRFT: PCT-RELATED CLADDING FAILURES DURING OFF-NORMAL EVENTS-DRAFT: Draft Report Of The USNRC PCI Review Group.

DIVISION OF ACCIDENT EVALUATION

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NUREG/CR-2921: CHEMICAL INTERACTIONS OF TELLURIUM VAPORS WITH REACTOR MATERIALS.

NUREG/CR-2940: REALISTIC SIMULATION OF SEVERE ACCIDENTS IN BWRS-COMPUTER MODELING REQUIREMENTS.

NUREG/CR-3023: MOLTEN THERMITE TEEMING INTO AN IRON OXIDE PARTICLE BED.

NUREG/CR-3307 V03: REACTOR SAFETY RESEARCH PROGRAMS. Quarterly Report July-September 1983.

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NUREG/CR-3329 V04: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM. Quarterly Report October-December 1983.

NUREG/CR-3335: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-3.

NUREG/CR-3350: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM: Postirradiation Examination Results For The Third Materials Experiment (MT-3).

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NUREG/CR-3366: HIGH TEMPERATURE MELT ATTACK ON STEEL AND URANIA-COATED STEEL.

NUREG/CR-3379: SLAM - A SODIUM-LIMESTONE CONCRETE ABLATION MODEL.

NUREG/CR-3410: CHMONE: A ONE-DIMENSIONAL COMPUTER CODE FOR SIMULATING TEMPERATURE, FLOW AND CHEMICAL CONCENTRATIONS IN WATER BODIES.

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NUREG/CR-3600: DATA SUMMARY REPORT FOR FISSION PRODUCT RELEASE TEST HI-4.

NUREG/CR-3603: MINET VALIDATION SURVEY USING EBB-II TEST DATA.

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NUREG/CR-3608: RELAP5 ASSESSEMENT: LOFT Large Break L2-5.
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DIVISION OF HEALTH, SITING & WASTE MANAGEMENT

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DIVISION OF RISK ANALYSIS & OPERATIONS (POST 840429)
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NRCM 1102
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U.S. NUCLEAR REGULATORY COMMISSION

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1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)

NUREG-0304, Vol. 9, No. 2

2. TITLE AND SUBTITLE

Regulatory and Technical Reports
Compilation for Second Quarter 1984
April - June

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR

5. AUTHOR(S)

6. DATE REPORT ISSUED

MONTH	YEAR
August	1984

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Technical Information and Document Control
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Same as 7, above.

11a. TYPE OF REPORT

Quarterly

b. PERIOD COVERED (Inclusive dates)

April - June 1984

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This compilation lists all NRC regulatory and technical reports published under the NUREG series during the second quarter of 1984.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

abstract
index

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT

Unlimited

16. SECURITY CLASSIFICATION

(This page)
Unclassified

(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC
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PERMIT No G67

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WASHINGTON DC 20555

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