NUREG-0304
Vol. 9, No. 3

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
Third Quarter 1984
July - September

U.S. Nuclear Regulatory Commission

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PREFACE

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

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The main citations and abstracts in this compilation are listed in NUREG number order: NUREG-XXXX, NUREG/CP-XXXX, 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 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPT.Data As Of May 31,1984,(Grey Book) * Division of Budget & Analysis. July 1984. 394pp. 8408160191. 26123:001.

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 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of June 30,1984,(Grey Book) * Division of Budget & Analysis. August 1984. 400pp. 8409200282. 26605:001.

See NUREG-0020,V08,N06 abstract.

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

See NUREG-0020,V08,N06 abstract.

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

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

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

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period January 1 to March 31, 1984.

During the report period, there were three abnormal occurrences at the nuclear power plants licensed by the NRC to operate. The first involved an inoperable containment spray system; the second involved a through wall crack in a vent header inside a BWR containment torus; and the third involved a serious degradation of a reactor depressurization system. There were two abnormal occurrences for the other NRC licensees. The first involved an overexposure to a member of the public; and the second involved a therapeutic medical misadministration. There was one abnormal occurrence reported by the Agreement States; the event involved an overexposure of a radiographer and assistant.

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

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

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

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

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

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

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

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

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

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

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

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

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

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 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE,JUNE 1-30, 1984. * Division of Technical Information & Document Control. August 1984. 710pp. 8409200392. 26603:001.

See NUREG-0540,V06,N05 abstract.

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

See NUREG-0540,V06,N05 abstract.

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

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

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

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

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

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

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

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

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

"Order Modifying License" issued by the Office of Nuclear Reactor Regulation on April 18, 1984.

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

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

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

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

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

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

NUREG-0748 V04 N05: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of May 31,1984.(Orange Book) * Office of Resource Management, Director. July 1984. 200pp. 8407170557. 25630:001.

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 operating power and nonpower reactors.

NUREG-0748 V04 N06: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of June 30,1984. (Orange Book) * Office of Resource Management, Director. July 1984. 400pp. 8408080363. 25981:129.

See NUREG-0748,V04,N05 abstract.

NUREG-0748 V04 N07: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of July 31,1984.(Orange Book) * Office of Resource Management, Director. August 1984. 150pp. 8409170276. 26497:001.

See NUREG-0748,V04,N05 abstract.

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

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

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

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

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

See NUREG-0750,V19,N03 abstract.

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

See NUREG-0750,V19,N03 abstract.

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

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

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

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

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

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

NUREG-0831 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION,UNITS 1 AND 2.Docket Nos. 50-416 And 50-417.(Mississippi Power And Light Company,Middle South Energy,Inc And South Mississippi Electric Power Association) * Division of Licensing. August 1984. 122pp. 8409270154. 26719:164.

Supplement 5 to the Safety Evaluation Report for Mississippi Power & Light Company, et al, joint application for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2, located on the east bank of the Mississippi River near Port Gibson in Claiborne County, Mississippi, has been prepared by the Office of Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the status on the resolution of those issues that require further evaluation before authorizing operation of Unit 1 above 5% of rated power.

NUREG-0831 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION,UNITS 1 AND 2.Docket Nos. 50-416 And 50-417. (Mississippi Power And Light Company) * Division of Licensing. August 1984. 204pp. 8409260656. 26701:181.

Supplement No. 6 to the Safety Evaluation Report for Mississippi Power & Light Company et al joint application for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2, located on the east bank of the Mississippi River near Port Gibson in Claiborne County, Mississippi, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The supplement reports the NRC staff's evaluation of open items from previous supplements and Technical Specification changes required before authorizing operation of Unit 1 above 5% of rated power.

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

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

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

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

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

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

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

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 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June 1984. * Enforcement Staff. July 1984. 363pp. 8408220308. 26202:001.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1984) and includes copies of letters, notices, and orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions and the licensees' responses. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, in the interest of promoting public health and safety as well as common defense and security.

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

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

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

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

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

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

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

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

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

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

Appendix (A) Program Element Schedules.

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

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

NUREG-1031: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT NO. 3.Docket No. 50-423.(Northeast Nuclear Energy Company) * Division of Licensing. July 1984. 591pp. 8408160262. 26121:001.

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

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

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

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

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

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

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

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

NUREG-1064: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT 3.DOCKET NO. 50-423. (NORTHEAST NUCLEAR ENERGY COMPANY,et al) * Division of Licensing. July 1984. 335pp. 8408010148. 25871:012.

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

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

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

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

This Safety Evaluation Report for the application filed by the General Electric Corporation for a renewal of operating license R-33 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned and operated by the General Electric Corporation and is located in Pleasanton, California. The staff concludes that the reactor facility can continue to be operated by GE without endangering the health and safety of the public.

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

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

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

This Draft Environmental Statement contains the second assessment of the environmental impact associated with the operation of River Bend Station, pursuant to the National Environmental Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51, as amended, of the Nuclear Regulatory Commission regulations. This statement examines the environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs. Comments on this statement should be filed no later than 45 days after the date on which the Environmental Protection Agency notice of availability of this statement is published in the Federal Register.

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

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

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

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

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

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

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

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

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

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

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

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

This Draft Environmental Statement contains the assessment of the environmental impact associated with the operation of the Nine Mile Point Nuclear Station, Unit 2, pursuant to the National Environmental Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51, as amended, of the Nuclear Regulatory Commission regulations. This statement examines the environment, environmental consequences and mitigating actions, and environmental and economic benefits and costs.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of this document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting are described in detail in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system keywords, and component vendor indexes follow the summaries. The components, systems, and vendors are those identified by the utility when the LER form is initiated; the keywords are assigned by the computer using correlation tables from the Sequence Coding and Search System.

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

See NUREG/CR-2000,V03,N6 abstract.

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

See NUREG/CR-2000,V03,N6 abstract.

NUREG/CR-2015 V08: PHASE I FINAL REPORT - SYSTEMS ANALYSIS (PROJECT VII). Seismic Safety Margins Research Program. WELLS,J.E.; GEORGE,L.L.; CUMMINGS,G.E. Lawrence Livermore National Laboratory. September 1984. 188pp. 8409280110. UCRL-53021 V08. 26764:001.

This document reports on the Phase 1 efforts of the Systems

Analysis project to develop the tools and methods for computing the probability of radioactive release from a commercial nuclear power plant in the event of an earthquake.

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

The Advanced and Water Reactor Safety Research Programs Quarterly Progress Reports have been combined and are included in this report entitled, "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research - Quarterly Progress Report." This progress report will describe current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the Division of Accident Evaluation, Division of Engineering Technology, and Division of Facility Operations of the U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

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

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

The Advanced and Water Reactor Safety Research Programs Quarterly progress reports have been combined and are included in this report entitled, "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research - Quarterly Progress Report." This progress report will describe current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the Division of Accident Evaluation, Division of Engineering Technology, and Division of Facility Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

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

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

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

NUREG/CR-2499: REVIEW OF EMERGENCY RADIOLOGICAL INSTRUMENTATION AND ANALYTICAL METHODS AT NMSS-LICENSEE SITES. HERRINGTON,W.N.; KATHREN,R.L.; KENOYER,J.L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 60pp. 8409200301. PNL-4163. 26609:001.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

PLUGM models the flow and freezing of molten material in a nonmelting channel. PLUGM is being developed for applications in Sandia's Ex-Vessel Core Retention Materials Assessment Program and in Sandia's LMFBR Transition-Phase Program. PLUGM models time-dependent flow from a reservoir, through a channel and possibly into a catch tank. Three user-specified geometry options enable realistic modeling of melt flow and freezing in tubes, thin slits, and particle beds. Axial variation of relevant channel parameters is possible. Sample problems, pertaining to ex-vessel core retention and LMFBR transition phase, illustrate features and capabilities of the code.

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

The objective of this research program is to characterize materials behavior in relation to structural safety and reliability of pressure boundary components for light water reactors. Specific objectives include developing an understanding of elastic-plastic fracture and environmentally-assisted crack propagation phenomena in terms of continuum mechanics, metallurgical variables, and neutron irradiation. Emphasis is placed on identifying metallurgical factors responsible for radiation embrittlement of steels and on developing procedures for embrittlement relief, including guidelines for radiation-resistant steels. The underlying goal is the interpretation of material properties performance to establish engineering criteria for structural reliability and long-term operation. Current work is organized into three major tasks: (1) fracture mechanics investigations, (2) environmentally-assisted crack growth in high temperature, primary reactor water and (3) radiation sensitivity and postirradiation properties recovery. Research progress in these tasks for 1983 is summarized here.

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

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

NUREG/CR-3318: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM:PCA Experiments,Blind Test,And Physics-Dosimetry Support For The PSF Experiments. MCELROY,W.N. Hanford Engineering Development Laboratory. September 1984. 150pp. 8410190172. HEDL-TME 84-1. 27059:001.

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

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

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

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

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

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

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

Twenty-four terminal blocks were tested in simulated Design Basis Event (DBE), Loss of Coolant Accident (LOCA) environments. The terminal blocks were powered at voltages of 4 Vdc, 45 Vdc, and 125 Vdc. Resulting currents associated with these voltage levels were 1.8 mA, 20 mA, and 1 A, respectively. Terminal-to-terminal and terminal-to-ground leakage currents were monitored on a discrete time basis throughout the test. Based on these measurements, insulation resistance were calculated. During exposure to the LOCA steam environment insulation resistance was observed to decrease from initial values of 10(8) to 10(10) ohms ot 10(2) to 10(5) ohms. These decreases in IR are interpreted as being caused by conduction in surface moisture films rather than bulk conduction through the insulation material. Insulation resistance for all applied voltage levels appear to be approximately the same. Sporadic breakdowns lasting from fractions of a second to several minutes were observed.

Further, rapid increases in applied voltage caused large decreases in insulation resistance. The measured IR was also dependent upon temperature. Subsequent to the test, terminal block insulation resistance returned to acceptable levels (10(6) to 10(8) ohms), though not to pre-test levels. The comparison of spray and no-spray results shows that no discernable difference in IRs existed between the periods with and without chemical spray.

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

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

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

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

NUREG/CR-3469 V01: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. BAUM,J.W.; SCHULT,D.A. Brookhaven National Laboratory. September 1984. 125pp. 8410120021. BNL-NUREG-51708. 26984:216.

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

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

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

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

The purpose of this program was to assess the problems posed to reactor decommissioning by long-lived activation products in reactor construction materials. Samples of stainless steel, vessel steel, concrete, and concrete ingredients were analyzed for up to 52 elements in order to develop a data base of activatable major, minor, and trace elements. Large compositional variations were noted for some elements. A thorough evaluation was made of all possible nuclear reactions that could lead to long lived activation products. It was concluded that all major activation products have been satisfactorily accounted for in decommissioning planning studies completed to date. A comparison is made between calculated activation levels and regulatory guidelines for shallow land disposal according to 10 CFR 61. Most of the massive components were found to qualify as either Class A or Class B waste with the exception of PWR and BWR shroud material which clearly exceeds Class C limits. Selected samples of activated steel and concrete were subjected to a limited radiochemical analysis program as a verification of the computer model. Reasonably good agreement with the calculations was obtained where comparison was possible. In particular, the presence of 94Nb in activated stainless steel at or somewhat above expected levels was confirmed.

NUREG/CR-3480: VALUE/IMPACT ASSESSMENT FOR SEISMIC DESIGN CRITERIA USI A-40. COATS,D.W.; LAPPA,D.A. Lawrence Livermore National Laboratory. August 1984. 138pp. 8409200279. UCRL-53489. 26606:053.

In October, 1981, the Nuclear Regulatory Commission approved a reorganization that resulted in the establishment of the Committee to Review Generic Requirements (CRGR). The charter for the CRGR requires that written justification accompany all proposed new regulatory requirements submitted to the CRGR for review. At the request of the Nuclear Regulatory Commission's Generic Issues Branch, Lawrence Livermore National Laboratory has provided the required written justification to accompany proposed new requirements to SRP Sections 3.7.1., 3.7.2, and 3.7.3. These proposed new requirements are the result of technical studies performed, as part of the Unresolved Safety Issues (USI) A-40 program, by LLNL and others. NUREG/CR-1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria," by LLNL, provided the technical resolution to USI A-40 and was the basis for the proposed new recommendations. The report contained herein present a technical evaluation and value/impact assessment of the proposed new requirements.

NUREG/CR-3492 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, October-December 1983. BALL,S.J.; CLEVELAND,J.C.; HARRINGTON,R.M.; et al. Oak Ridge National Laboratory. July 1984. 27pp. 8408130001. ORNL/TM-8921/V4. 26011:225.

Development work continued on models and codes for predicting source terms in both the Fort St. Vrain (FSV) and 2240-MW(t) lead plant reactors. Experimental work on fission-product vapor pressures and diffusion rates through graphite continued on temperatures up to 2775 K, and a mathematical model of the experimental system was developed to aid analysis of the results and to guide improvements in the system and experiment design. Benchmarking of the BLAST steam generator code continued using FSV data, and more support work was done for proposed FSV core bypass flow model verification. Progress was made in setting up cooperative high-temperature gas-cooled reactor (HTGR) safety research with the Federal Republic of Germany. A review of a FSV technical specification on limiting maximum core temperature was begun.

NUREG/CR-3493: A REVIEW OF THE LIMERICK GENERATING STATION SEVERE ACCIDENT RISK ASSESSMENT.Review of Core Melt Frequency. AZARM,M.A.; BARI,R.A.,; BOCCIO,J.L.; et al. Brookhaven National Laboratory. July 1984. 192pp. 8408220333. BNL-NUREG-51711. 26200:205.

A limited review is performed of the Severe Accident Risk Assessment for the Limerick Generating Station. The review considers the impact on the core-melt frequency of seismic- and fire-initiating events. An evaluation is performed of methodologies used for determining the event frequencies and their impacts on the plant components and structures. Particular attention is given to uncertainties and critical assumptions. Limited requantification is performed for selected core-melt accident sequences in order to illustrate sensitivities of the results to the underlying assumptions.

NUREG/CR-3513: MECHANICAL RELIABILITY EVALUATION OF ALTERNATE MOTORS FOR USE IN A RADIOIODINE AIR SAMPLER. BIRD,S.K.; HUCHTON,R.L.; MOTES,B.G.; et al. EG&G, Inc. July 1984. 41pp. 8409110069. WINCO-1006. 26437:284.

The purpose of the study was to evaluate the mechanical reliability of motors for use in a prototype system designed for post accident collection and measurement of radioiodine in the environs of a nuclear reactor. The two types of motors were tested for lifetimes and operational performance characteristics under extremes of temperature, relative humidity and/or in dusty air and rainfall. The 12 volt direct current motors exhibited satisfactory performance under all environmental conditions and demonstrated lifetimes of 47 hours, 97 hours and 188 hours. The 12 volt direct current voltage and satisfactory operation on alternating current voltage; at failure the AC/DC voltage motors demonstrated lifetimes of nominally 6 hours, 3 hours and 2 hours. The direct current voltage only motors are the better candidates for incorporation into the air sampler.

NUREG/CR-3518 V01: SLIM-MAUD:AN APPROACH TO ASSESSING HUMAN ERROR PROBABILITIES USING STRUCTURED EXPERT JUDGEMENT.Volume I:Overview of SLIM-MAUD. EMBREY,D.E.; HUMPHREYS,P.; ROSA,E.A.; et al. Brookhaven National Laboratory. July 1984. 36pp. 8408010166. BNL-NUREG-51716. 25867:253.

This two-volume report presents the procedures and analyses in developing an approach for structuring expert judgments to estimate human error probabilities. Volume I presents an overview of work performed in developing the approach: SLIM-MAUD (Success Likelihood Index Methodology, implemented through the use of an interactive computer program called MAUD--Multi-Attribute Utility Decomposition). Volume II provides a more detailed analysis of the technical issues underlying the approach.

NUREG/CR-3520 V01: LONG-TERM RESEARCH PLAN FOR HUMAN FACTORS AFFECTING SAFEGUARDS AT NUCLEAR POWER PLANTS.Volume I:Summary And Users Guide. O'BRIEN,J.N.; FAINBERG,A. Brookhaven National Laboratory. August 1984. 42pp. 8409280071. BNL-NUREG-51718. 26750:228.

The first task was to identify and rank human factors affecting the quality of nuclear power plant safeguards in terms of their importance. The opinions of over 85 experts were solicited and 28 responses were received. These responses were rigorously analyzed to ascertain what human factors could be considered important to power plant safeguards. In addition, the Safeguards Summary List (NUREG-0525) was systematically analyzed for human factors influences. Also, relevant government and industry literature was reviewed. These data sources were then aggregated and an overall importance ranking of human factors issues was developed. This part of the research effort is fully documented and described in Chapter 2 of Volume II.

The second part of this effort involved determining the feasibility of conducting research in the areas found to be important to power plant safeguards. A determination of research feasibility was based on the practicality, usefulness, and acceptability of conducting research and using the results in a regulatory context. This part of the effort is fully documented in Chapter 3 of Volume II.

Research efforts addressing human factors in safeguards were then developed and prioritized according to the importance of human factors areas derived in the first part of the study and the feasibility of research determined in the second part. Research was also grouped to take advantage of common research approaches and data sources where appropriate. Chapter 4 of Volume II details the development of methodological groupings for optimizing resource use.

NUREG/CR-3520 V02: LONG-TERM RESEARCH PLAN FOR HUMAN FACTORS AFFECTING SAFEGUARDS AT NUCLEAR POWER PLANTS.Volume II:Development Of Detailed Analyses. O'BRIEN,J.N.; FAINBERG,A. Brookhaven National Laboratory. August 1984. 204pp. 8409280062. BNL-NUREG-51718. 26762:001.
See NUREG/CR-3520,V01 abstract.

NUREG/CR-3524: ORGANIZATIONAL INTERFACE IN REACTOR EMERGENCY PLANNING AND RESPONSE. SORENSEN,J.H.; COPENHAVER,E.D.; MILETI,D.S.; et al. Oak Ridge National Laboratory. July 1984. 52pp. 8409110100. ORNL-6010. 26446:236.

The purpose of this research was to determine if existing regulations have led to effective interfaces between utilities and offsite organizations in emergency planning and response. Findings suggest that regulations have provided the necessary framework for achieving adequate interfaces. That interface has been achieved is demonstrated by comprehensive response plans and good cohesiveness among organizations involved in emergency response. Interface problems identified in the research can be reduced by better implementation of existing regulations rather than by revision of existing ones.

NUREG/CR-3544: BETA PARTICLE MEASUREMENT AND DOSIMETRY AT NRC-LICENSED FACILITIES. RATHBUN,L.A.; ENDRES,G.W.; FOX,R.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 58pp. 8409070234. PNL-4886. 26419:203.

Researchers from Pacific Northwest Laboratory (PNL) have conducted beta radiation measurements under laboratory and field conditions to assess the degree of the measurement problem and offer suggestions for possible remedies. The primary measurement systems selected for use in this study were the silicon (Si) surface barrier spectrometer system and the multielement beta dosimeter. Three boiling water reactors (BWRs), two pressurized water reactors (PWRs), and one fuel fabrication facility were visited during the course of the study. Although beta fields from cobalt-60 were the most common type found at locations associated with spent fuel handling, liquid radioactive waste, and BWR turbine components. Commercially-available dosimeters and survey instruments were used to measure the same laboratory and licensee facility beta fields characterized with PNL's active and passive spectrometers. A prototype survey meter was also used in the laboratory measurements. The commercial instruments and dosimeters used in this study typically responded low to the beta fields measured, especially where maximum beta energies were less than approximately 500 kev. A single calibration factor is usually not adequate for either beta dosimeters or instruments. There is a need for more refinement in beta measurement devices and training for the users of such devices.

NUREG/CR-3569: SPECIAL AND DOSIMETRIC MEASUREMENTS OF PHOTON FIELDS AT COMMERCIAL NUCLEAR SITES. ROBERSON,P.L.; FOX,R.A.; HOLBROOK,K.L.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 182pp. 8408290171. PNL-4915. 26299:119.

Spectral and dosimetric measurements of photon fields were performed at seven commercial nuclear reactor sites. Revisions to 10 CFR 20 that specify exposure-to-dose conversion factors (Cx) much greater than unity for photons between 40 keV and 200 keV could impact personnel monitoring practices. Monitoring at effective depths of 1 cm of tissue and shallower could underestimate doses received from high-energy photon fields (>3 MeV).

No locations with large C(x) factors (approximately 1.5 rad/R) were found. The most significant production of low-energy photons was found to be due to photon scattering. The scatter continuum has an effective Cx factor of approximately 1.2 rad/R. One location was found with a nearly pure scatter spectrum. Other locations contained significant contributions from medium-energy photons due primarily to radioactive decay of cobalt and cesium isotopes. Monitoring requirements at 0.007-cm and 1.0-cm depths in tissue were found to be adequate for estimating dose received in radiation fields containing high-energy photons. Enhanced surfaced doses attributed to high-energy knock-on electrons were measured in all locations monitored. Personnel monitoring techniques may provide inaccurate results in high-energy fields.

NUREG/CR-3589 V01: REACTOR SAFETY RESEARCH QUARTERLY REPORT, January-March 1983. * Sandia Laboratories. July 1984. 179pp. 8409180335. SAND83-2425. 26588:001.

Sandia National Laboratories is conducting phenomenological research related to the safety of commercial nuclear power reactors. The overall objective of this work is to provide NRC a comprehensive data base essential to (1) defining key safety issues, (2)understanding risk-significant accident sequences, (3) developing and verifying models used in safety assessments, and (4) assuring the public that power reactor systems will not be licensed and placed in commercial service in the United States without appropriate consideration being given to their effects on health safety. This report describes progress in a number of activities dealing with current safety issues relevant to both light water and breeder reactors. The work includes a broad range of experiment to simulate accidental conditions to provide the data base required to understand important accident sequences and to serve as a basis for development and verification of the complex computer simulation models and codes used in accident analysis and licensing reviews. Current major emphasis is focused on providing information to NRC relevant to (1) its deliberations and decisions dealing with severe LWR accidents and (2) its safety evaluation of the proposed Clinch River Breeder Reactor.

NUREG/CR-3589 V02: REACTOR SAFETY RESEARCH QUARTERLY REPORT.April-June 1983. * Sandia Laboratories. July 1984. 166pp. 8409180443. SAND83-2425. 26588:180.

See NUREG/CR-3589,V01 abstract.

NUREG/CR-3590: EVALUATION OF ISOTOPE DILUTION MASS SPECTROMETRY FOR BIOASSAY MEASUREMENT OF URANIUM,PLUTONIUM,AND THORIUM IN URINE. DYER,F.F.; MAY,M.P.; WALKER,R.L.; et al. Oak Ridge National Laboratory. August 1984. 79pp. 8408300272. ORNL/TM-9006. 26330:224.

A study was made to evaluate the sensitivity, precision and accuracy, and practicality of isotope dilution mass spectrometry (IDMS) for bioassay of uranium, plutonium, and thorium in human urine. The study showed that uranium at a concentration of 0.06 mg/L (0.04 pCi/L natural uranium), plutonium at 3 pg/L (0.2 p Ci/L Pu-239, and thorium at 0.1 mg/L (0.01 pCi/L Th-232) could be measured with an uncertainty (RSD) of ten percent using 10 ml samples. The lower limits of detection for uranium and thorium were set by background contamination, whereas the detection limit for plutonium was determined by chemical yield and intrinsic instrumental sensitivity factors. Precision and accuracy is excellent (~1-3%, RSD) at concentration levels where background contamination is insignificant and instrumental sensitivity is adequate.

Comparison of IDMS with other methods shows the technique is more sensitive than conventional fluorometric methods but is similar in sensitivity to alpha-radioactivity measurement methods that utilize large sample volumes (1 L). Costs for urine analysis by IDMS ($60-$100 per sample) are estimated to be considerably higher than cost for fluorometric analysis and approximately the same as the cost for alpha-radioactivity methods. Other methods that have been used or are currently under development are discussed.

NUREG/CR-3591 V01: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1980-1981 A Status Report. COTTRELL,W.B.; MINARICK,J.W.; AUSTIN,P.N.; et al. Oak Ridge National Laboratory. July 1984. 262pp. 8408290190. 26312:001.

Descriptions of fifty-eight operational events reported as Licensee Event Reports, which occurred at commercial light-water reactors during 1980-1981 and which are considered to be precursors to potential severe core damage, are presented, along with associated event trees, categorization, and subsequent analyses. This study is a continuation of the work presented in NUREG/CR-2497 which somewhat similarly evaluated the 1969-1979 events. The current study incorporates improvements which evolved from an assessment of the comments on the earlier report and applies these in the assessment of the LERs which occurred during 1980 and 1981. The report sequentially discusses (1) the general rationale for this study, (2) the program methods for LER review and documentation, (3) the calculation of function failure probabilities and initiating event frequencies based upon precursor data, (4) the use of the conditional probability of subsequent severe core damage estimates to rank precursor events and estimate an average industrywide risk of severe core damage, and (5) the conduct of sensitivity analyses on these results. There was some apparent decrease in most initiating event frequencies and function failure probabilities in the 1980 and 1981 period, as compared to the earlier report. Although it was not possible to conclude that these decreases were statistically significant, they did result in a reduction in the industry average estimated severe core damage frequency for 1980-1981 as compared to the 1969-1979 period.

NUREG/CR-3591 V02: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS: 1980-1981 A Status Report. COTTRELL,W.B.; MINARICK,J.W.; AUSTIN,P.N.; et al. Oak Ridge National Laboratory. July 1984. 266pp. 8408290198. 26311:001.

See NUREG/CR-3591,V01 abstract.

NUREG/CR-3593 V01: SYSTEMS INTERACTION RESULTS FROM THE DIGRAPH MATRIX ANALYSIS OF A NUCLEAR POWER PLANT'S HIGH PRESSURE SAFETY INJECTION SYSTEM. SACKS,I.J.; ASHMORE,B.C.; ALESSO,H.P. Lawrence Livermore National Laboratory. July 1984. 184pp. 8409280090. UCRL-53467.

26749:181.

The report describes the demonstration of the Digraph-Matrix Analysis on a Nuclear Power Plant's High Pressure Safety Injection System. The demonstration work was beyond the scope of both the methods and the criteria used by the NRC to license nuclear power plants. The analysis discovered components whose failure could jeopardize the High Pressure Injection System given the postulated accident. All these components had been previously considered both in the safety analysis and in the licensing review. The results demonstrate the capability of Digraph-Matrix Analysis to model an accident sequence (including front-line systems, support systems, and operator actions) as a continuously integrated model to discover functional systems interactions. Also, the method is scrutable and can be used on a complex system which contains both a large number of components and dependent loops. Volume 1 is the main report and the description of the method. Volume 2 contains the digraphs, adjacency listings, and data base.

NUREG/CR-3593 V02: SYSTEMS INTERACTION RESULTS FROM THE DIGRAPH MATRIX ANALYSIS OF A NUCLEAR POWER PLANT'S HIGH PRESSURE SAFETY INJECTION SYSTEM.Volume 2. SACKS,I.J.; ASHMORE,B.C.; ALESSO,H.P. Lawrence Livermore National Laboratory. July 1984. 165pp. 8409280123. UCRL-53467. 26735:096.

See NUREG/CR-3593,V01 abstract.

NUREG/CR-3599: SOURCES OF UNCERTAINTY IN THE CALCULATIONS OF LOADS ON SUPPORTS OF PIPING SYSTEMS. * Oak Ridge National Laboratory. RODABAUGH,E.C. E.C. Rodabaugh Associates, Inc. July 1984. 73pp. 8408130041. 26040:166.

Loads on piping systems are obtained from an analysis of the piping system. The piping system analysis involves uncertainties from various sources. These sources of uncertainties are discussed and ranges of uncertainties are illustrated by simple examples. The sources of uncertainties are summarized and assigned a judgmental ranking of the typical relative significance of the uncertainty.

NUREG/CR-3610: NEUTRON DOSIMETRY AT COMMERCIAL NUCLEAR PLANTS:Final Report Of Subtask C: 3He Neutron Spectrometer. BRACKENBUSH,L.; REECE,W.D.; TANNER,J.E. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1984. 120pp. 8410120008. PNL-4943. 26983:001.

In commercial nuclear power plants, personnel routinely enter containment for maintenance and inspections while the reactor is operating and can be exposed to intense neutron fields. The low-energy neutron fields found in reactor containment cause problems in proper interpretation of TLD-albedo dosimeters and survey instrument readings. Described is a technique that can aid plant health physicists to improve the accuracy of personnel neutron dosimetry programs. A (3)He neutron spectrometer can be used to measure neutron energy spectra and determine dose equivalent rates at work locations inside containment. Energy correction factors for TLD-albedo dosimeters can be determined from the measured spectra if the dosimeter energy response is known, or from direct measurements with dosimeters placed on phantoms at locations where the dose equivalent rate has been measured. This report describes how to assemble a spectrometer system using only commercially available components, how to use it for reactor energy spectrum measurements, and how to analyze the data and interpret the results. Both (3)He and multisphere spectrometers were used to measure neutron energy spectra and dose equivalent at three PWRs and one BWR. In general, the (3)He spectrometer measures higher dose equivalent rates than the multisphere spectrometer. In the energy range from 10 keV to 1 MeV, the dose equivalents measured by the (3)He spectrometer and multisphere spectrometer agree within about 35% for the spectra measured.

NUREG/CR-3617: NOBLE GAS,IODINE,AND CESIUM TRANSPORT IN A POSTULATED LOSS OF DECAY HEAT REMOVAL ACCIDENT AT BROWNS FERRY. WICHNER,R.P.; WEBER,C.F.; WRIGHT,A.L.; et al. Oak Ridge National Laboratory. September 1984. 199pp. 8410120013. ORNL/TM-9028. 26978:001.

This report presents an analysis of the movement of noble gas, iodine, and cesium fission products within the Mark-I containment BWR reactor system represented by Browns Ferry Unit 1 during a postulated accident sequence initiated by a loss of decay heat removal capability following a scram. This accident could be brought under control by various means, but the sequence with no operator action ultimately leads to failure followed by loss of water from the reactor vessel, core degradation due to overheating, and reactor vessel failure with attendant movement of core debris onto the drywell floor. The fission product transport analysis is based on the no-operator-action sequence and provides an estimate of fission product inventories, as a function of time, within 14 control volumes outside the core, with the atmosphere considered as the final control volume in the transport sequence. We find small barrier for noble gas ejection to air, these gases being effectively purged from the drywell and reactor building by steam and concrete degradation gases. In contrast, large degrees of holdup for iodine and cesium are projected due to the chemical reactivity of these elements. Only about 2 x 10(-4%) of the initial iodine and cesium activity are predicted to be released to the atmosphere. Principal barriers for release are deposition on reactor vessel and containment walls.

NUREG/CR-3618: OCA-P,A DETERMINISTIC AND PROBABILISTIC FRACTURE-MECHANICS CODE FOR APPLICATION TO PRESSURE VESSELS. CHEVERTON,R.D.; BALL,D.G. Oak Ridge National Laboratory. July 1984. 105pp. 8408080415. ORNL-5991. 25977:338.

OCA-P is a probabilistic fracture-mechanics code that was prepared specifically for the purpose of evaluating the integrity of PWR pressure vessels when subjected to overcooling-accident loading conditions. The code has two-dimensional and some three-dimensional-flaw capability; it is based on linear elastic fracture mechanics; and it can treat cladding as a discrete region. Both deterministic and probabilistic analyses can be performed, and for the former analysis it is possible to conduct a search for critical values of the fluence and the nil ductility reference temperature corresponding to incipient initiation of the initial flaw. The probabilistic portion of OCA-P is based on Monte Carlo techniques, and simulated parameters include fluence, flaw depth, fracture toughness, nil ductility reference temperature, and concentrations of copper, nickel and phosphorous. Plotting capabilities include the construction of critical-crack-depth diagrams (determinstic analysis) and various histograms (probabilistic analysis).

NUREG/CR-3643: HETEROGENEOUS OXIDATIVE DEGRADATION IN IRRADIATED POLYMERS. CLOUGH,R.L.; GILLER,K.T.; QUINTANA,C.A. Sandia Laboratories. July 1984. 43pp. 8408080359. SAND83-2493. 25978:153.

When polymeric materials are irradiated in the presence of air, oxygen-diffusion effects can, depending upon dose rate, lead to oxidative degradation which occurs only near the edges. This report describes the use of several recently developed techniques which are of general use for studying heterogeneous degradation in commercial polymeric materials. The techniques discussed are: optical evaluation of cross-sectioned, polished samples; cross-sectional profiling of changes in relative hardness; and profiling of density changes. Oxidation penetration depths are given for a number of major polymer types as a function of dose rate. A detailed example is given graphically illustrating the effects of differing oxidative penetration depths on the radiation-degradation behavior of a Viton O-ring material; this particular material becomes hard and brittle when irradiated at high dose rate, but soft and stretchable when irradiated at low dose rates.

NUREG/CR-3654: PWR FLECHT SEASET SYSTEMS EFFECTS NATURAL CIRCULATION AND REFLUX CONDENSATION.Data Evaluation and Analysis Report NRC/EPRI/Westinghouse Report No. 14. HOCHREITER,L.E.; RUPPRECHT,S.D.; DEDERER,J.T.; et al. Westinghouse Electric Corp. September 1984. 584pp. 8410100382. EPRI NP-3497. 26899:001.

A series of natural circulation tests were conducted at a FLECHT SEASET facility which is scaled 1/307 by volume to a full size PWR. The purpose of these tests was to identify hydraulic and heat transfer phenomena during natural circulation cooling modes. The resulting data, evaluation, and analysis are to be used for PWR codes and model assessments as well as to provide a comparison to similar experiments in other scaled systems. Steady-state single-phase, two-phase, and reflux condensation modes of natural circulation cooling were established in the FLECHT SEASET systems effects facility and the flow and heat transfer characteristics of the different cooling modes were identified. This report presents the test data; data reduction, analysis, and evaluation; and resulting model development and analysis. The models which have been developed include a reflux tube condensation model as well as a single- and two-phase model for the overall system.

NUREG/CR-3655: A METHOD FOR ANALYTICAL EVALUATION OF COMPUTER-BASED DECISION AIDS. ROUSE,W.B.; FREY,P.R.; et al. Search Technology, Inc. KISNER,R.A. Oak Ridge National Laboratory. July 1984. 202pp. 8409110215. ORNL/TM-9068. 26445:001.

This report presents a proposed methodology that involves a two-stage process of classification and analytical evaluation of decision aids for nuclear power plant operators. The classification scheme relates any particular aid to one or more general decision-making tasks. Evaluation proceeds using a normative top-down design process based on the classification scheme and involves determining how various design issues associated with this process were resolved by the designer. The result is an assessment of the "understandability" of the aid as well as the identification of training and display requirements necessary to ensure understandability. The methodology is illustrated by applying it to the evaluation of an aid designed to support operators in recovery of critical safety functions at a pressurized-water reactor.

Two appendices are included. Appendix A contains information collected from manufacturers, developers, and users of operational aid systems. Appendix B is a review of NRC documents and guidelines that might apply to operational aids.

NUREG/CR-3660 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF WESTINGHOUSE PWR PLANTS.Volume 2:Pipe Failure Induced By Crack Growth. WOO,H.H.; MENSING,R.W.; BENDA,B.J. Lawrence Livermore National Laboratory. August 1984. 71pp. 8409200445. UCID-19988. 26628:132.

This report assesses the probability of reactor coolant loop (RCL) piping failures resulting from a crack growth mechanism. The Westinghouse pressurized water reactor (PWR) plants in the United States east of the Rocky Mountains are considered. After the introduction (Section 1), the assessment is presented in five parts (Sections 2-6). Section 2 describes the characteristics of RCL piping in these Westinghouse PWR plants. Section 3 describes the methodology used in the analysis. Sections 4 and 5 present the best-estimate and uncertainty analyses, respectively. Our conclusions are presented in Section 6, along with recommended items for consideration in future licensing regulations.

NUREG/CR-3662: FUEL-DISRUPTION EXPERIMENTS UNDER HIGH-RAMP-RATE HEATING CONDITIONS. WRIGHT,S.A.; WORLEDGE,D.H.; CANO,G.L.; et al. Sandia Laboratories. August 1984. 86pp. 8409280084. SAND81-0413. 26764:340.

This topical report presents the preliminary results and analysis of the High Ramp Rate fuel-disruption experiment series. These experiments were performed in the Annular Core Research Reactor at Sandia National Laboratories to investigate the timing and mode of fuel disruption during the prompt-burst phase of a loss-of-flow accident. High-speed cinematography was used to observe the timing and mode of the fuel disruption in a stack of five fuel pellets. Of the four experiments discussed, one used fresh mixed-oxide fuel, and three used irradiated mixed-oxide fuel.

Analysis of the experiments indicates that in all cases, the observed disruption occurred well before fuel-vapor pressure was high enough to cause the disruption. The disruption appeared as a rapid spray-like expansion and occurred near the onset of fuel melting in the irradiated-fuel experiments and near the time of complete fuel melting in the fresh-fuel experiment. This early occurrence of fuel disruption is significant because it can potentially lower the work-energy release resulting from a prompt-burst disassembly accident.

NUREG/CR-3663 V02: PROBABILITY OF PIPE FAILURE IN THE REACTOR COOLANT LOOPS OF COMBUSTION ENGINEERING PWR PLANTS.Vol 2:Pipe Failure Induced by Crack Growth. LO,T.Y.; MENSING,R.W.; WOO,H.H.; et al. Lawrence Livermore National Laboratory. September 1984. 94PP. 8410120004. UCRL-53500. 26985:177.

The U.S. Nuclear Regulatory Commission (NRC) contracted with the Lawrence Livermore National Laboratory (LLNL) to conduct a study to determine if the probability of occurrence of a double-ended guillotine break (DEGB) in the primary coolant piping warrants the current design requirements that safeguards against the effect of DEGB. This report describes the results of an assessment of reactor coolant loop piping systems designed by Combustion Engineering, Inc.

A probabilistic fracture mechanics approach was used to estimate the crack growth and to assess the crack stability in the piping throughout the lifetime of the plant. The results of the assessment indicate that the probability of occurrence of DEGB due to crack growth and instability is extremely small, which supports the argument that the postulation of DEGB in design should be eliminated and replaced with more reasonable criteria.

NUREG/CR-3665: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS.Executive Summary. COHEN,J.J. Science Applications, Inc. September 1984. 7pp. 8410100787. SAIC-84/1317. 26903:248.

An area of growing concern in recent years has been the apparent increase in levels of collective radiation dose to workers at nuclear power plants in the USA. U.S. Nuclear Regulatory Commission (NRC) decisions and rulings related to in-service inspection, retrofits, and plant upgrades have been primarily intended to reduce the risk of public radiation exposure resulting from either routine release of radioactivity or potential accident situations. However, implementation of the required control measures and procedures can often result in increased levels of occupational radiation exposure. Recognizing the need to incorporate occupational dose into probabilistic risk assessments (PRA), value-impact, and cost-benefit analyses, the NRC has sponsored this study with the objective of developing an appropriate methodology to factor potential worker exposures into safety assessments. This report on the study is presented in three volumes. The following are subtitles for Volumes 1-3: Volume 1, "A Review of Occupational Dose Assessment Considerations in Current Probabilistic Risk Assessments and Cost-Benefit Analyses, " Volume 2, " Considerations in Factoring Occupational Dose Into Value-Impact and Cost-Benefit Analyses, " and Volume 3, " A Calculation Method."

NUREG/CR-3665 V01: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS.A Review Of Occupational Dose Assessment Considerations In Current Probabilistic Risk Assessment And Cost-Benefit Analyses. LOBNER,P. Science Applications, Inc. September 1984. 55pp. 8410100656. SAI-83/1125. 26900:232.

This report reviews current value-impact analysis and probabilistic risk assessment methods and discusses the manner and degree to which these methods consider occupational radiation exposure that may form a variety of in-plant activities, including: (a) normal operation and maintenance, (b) repair, (c) retrofit, (d) minor incidents and cleanup, (e) major accidents, and (f) decommissioning. Value-impact analysis methods which include occupational exposure as an element of the value-impact equation have been developed, however, no standard approach to such analysis has been adopted. Comparison of the results of value-impact analysis must, therefore, be done with caution because different value-laden assumptions made by the analyst can have strong effects on the outcome. Such assumptions include the monetray equivalent of a person-rem, and the relative valus of occupational and public exposure. Probabilistic methods have been used in value-impact evaluations to quantify incremental or averted occupational exposure from reactor accidents, however, occupational exposure has not been addressed in probabilistic risk assessments (PRAs) of nuclear power plants. Consideration of occupational exposure in a PRA would greatly increase the complexity of the plant model and the benefits from such an analysis are uncertain. In lieu of expanding the scope of PRAs to address occupational risk, the separate, limited-scope probabilistic evaluations developed for value-impact analysis should provide a more practical analytical capability to support the evaluation and optimization of occupational and public radiation exposure.

NUREG/CR-3665 V02: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS.Considerations In Factoring Occupational Dose Into Value-Impact And Cost-Benefit Analyses. COHEN,J.J. Science Applications, Inc. September 1984. 46pp. 8410100785. SAI-84/1010 V02. 26903:200.

Many NRC decisions intended for the improvement of public health and safety involve concomitant increases in occupational radiation exposure. Previous study (Volume 1) indicates that occupational dose consequences generally have not been considered in cost-benefit and value-impact analyses supporting decisions related to public safety. Such consideration, however, would be consistent with ALARA guidance. This study derives a methodology for factoring occupational vs. public radiation exposure, stochastic vs. non-stochastic effects, probabilistic risk considerations, uncertainty, and de minimus levels.

NUREG/CR-3665 V03: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS.A Calculation Method. HORTON,W.H. Science Applications, Inc. September 1984. 87pp. 8410100755. SAI-84/3037 V03. 26902:001.

The methodology presented in this report formulates an approach for the optimization of benefits resulting from NRC decision making processes. Recent increases in occupational exposures in nuclear power plants resulting from NRC regulatory practices have led to the questioning by NRC of the overall benefit of specific regulations. The optimization methodology in this report provides a tool for the determination of the cost-benefit of proposed NRC regulations. Detailed methods are presented for the modeling of plant safety systems undergoing inspection, testing, and/or repair. This methodology utilizes dynamic Markov modeling techniques with extensive additional model development associated with operator errors involved in the inspection, test, and repair activities of the plant. Closed form solutions to the Markov models are provided. The report appendix presents the Markov model solution process in detail sufficient for model verification. Other methods necessary for the optimization process are discussed in lesser detail. An application of the methodology dealing with steam generator inspection frequency and steam generator tube rupture events is presented. The example determines the steam generator inspection intervals which minimize expected costs and total expected occupational and public dose.

NUREG/CR-3671: ASSESSMENT OF RADIATION EFFECTS RELATING TO REACTOR PRESSURE VESSEL CLADDING. CORWIN,W.R. Oak Ridge National Laboratory. July 1984. 70pp. 8408080419. ORNL-6047. 25977:271.

Because the weld overlay cladding on the interior of light-water reactor pressure vessels was applied for corrosion resistance and not for structure, little attention has been given to the potential of mechanical property degradation due to radiation exposure. In light of the concerns recently raised regarding overcooling transients in nuclear power reactors, it has been suggested that any such degradation could adversely affect the serviceability and/or integrity of the vessel.

A literature survey assesses the current knowledge regarding the effects of neutron radiation on the mechanical fracture properties of stainless steel weld overlay cladding under conditions relevant to light-water reactor operation. In particular, effects on the material's microstructure and tensile, fatigue, impact, and fracture properties are examined. Although information is lacking on the specific materials under the exact irradiation conditions of interest, a wealth of information is available on irradiated stainless steel weldments in general, from which basic behavioral trends can be obtained.

Some irradiation embrittlement apparently does occur in stainless steel weldments at the relatively low temperatures and fluences typical of light-water reactors. Tensile strength increases and ductility decreases. Low-cycle fatigue behavior is degraded somewhat, but high-cycle fatigue and fatigue crack growth seem largely unaffected.

Effects of ferrite on fracture resistance are small in both irradiated and unirradiated materials. Notch impact and fracture toughness are both reduced by irradiation, and a dependence of toughness on testing rate, not seen in wrought material, is indicated.

NUREG/CR-3678: ESTIMATION METHODS FOR PROCESS HOLDUP OF SPECIAL NUCLEAR MATERIALS. PILLAY,K.S.; PICARD,R.R.; MARSHALL,R.S. Los Alamos Scientific Laboratory. July 1984. 121pp. 8408080409. LA-10038. 25980:028.

Los Alamos National Laboratory studied the use of statistical estimation methods for materials holdup at highly enriched uranium (HEU)-processing facilities. Use of historical holdup data from processing facilities and selected holdup measurements at two operating facilities confirm the need for high-quality data and reasonable control over process parameters in developing these models. Large-scale experiments were conducted to demonstrate the value of the models from good-quality experimental data. Using these data, we developed statistical models to estimate residual inventories of uranium in large process equipment and facilities. Some important findings are the following:

Holdup in some equipment at HEU-processing facilities, such as air filters, ductwork, calciners, dissolvers, pumps, pipes, and pipefittings can be readily modeled.

Holdup profiles of process equipment such as glove boxes, precipitators, and rotary drum filters can change with time, necessitating several measurements at the time of inventory.

Reasonable estimation of hidden inventories of holdup to meet regulatory requirements can be accomplished through good measurements and statistical modeling.

NUREG/CR-3679: CALIBRATION AND QUALIFICATION OF THE LOS ALAMOS FAILURE MODEL (LAFM). BAARS,R.E. Los Alamos Scientific Laboratory. July 1984. 64pp. 8408080357. LA-10041-MS. 25980:329.

The analysis procedure is described in detail for use of the LAFM computer code to predict LMFBR fuel pin performance under transient overpower conditions; also, 5 tests for calibration and 13 tests for qualification are analyzed. The times of cladding breach (molten fuel expulsion) were predicted with an average relative error of 5 per cent. An enthalpy of 1112 kj/kg correlated the peak fuel enthalpies at the time of failure with a standard deviation of 98 kj/kg. We conclude with a discussion that many varied tests must be analyzed for adequate evaluation of a fuel pin performance code.

NUREG/CR-3689 V01: MATERIAL SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM:Quarterly Progress Report, January-March 1983. SHACK,W.J. Argonne National Laboratory. July 1984. 169pp. 8408100151. ANL-83-85. 25998:313.

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

NUREG/CR-3689 V02: MATERIALS SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER REACTOR SAFETY RESEARCH PROGRAM.Quarterly Progress Report, April-June 1983. SHACK,W.J. Argonne National Laboratory. July 1984. 141pp. 8408310083. ANL-83-85. 26348:215.

See NUREG/CR-3689,V01 abstract.

NUREG/CR-3689 V03: MATERIALS SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER REACTOR SAFETY RESEARCH PROGRAM.Quarterly Progress Report, July-September 1983. REST,J. Argonne National Laboratory. July 1984. 40pp. 8408310077. ANL-83-85. 26347:277.

See NUREG/CR-3689,V01 abstract.

NUREG/CR-3690: RELAP5 ASSESSMENT:SEMISCALE NATURAL CIRCULATION TESTS S-NC-3,S-NC-4,AND S-NC-8. WONG,C.C.; KMETY,K.L. Sandia Laboratories. July 1984. 115pp. 8408310089. SAND-0402. 26348:099.

The RELAP5/MOD1 code is being assessed against test data from a number of integral and separate effects test facilities. As part of this assessment matrix, we have analyzed a number of natural circulation tests performed at the Semiscale facility. Our results for the single-loop and two-loop steady state basecase tests S-NC-2 and S-NC-7 have previously been documented; this report gives the results of calculations for two single-loop degraded heat transfer tests, S-NC-3 and S-NC-4, and for the two-loop ultra-small break transient test S-NC-8. For tests S-NC-3 and S-NC-4, our analyses show that RELAP5/MOD1 describes correctly the qualitative influence of steam generator secondary side heat transfer degradation on both two-phase and reflux natural circulation. The agreement between calculated and measured two-phase mass flow rates in test S-NC-3 is better with a primary mass inventory of 85% (where the peak two-phase mass flow rate is calculated to occur) instead of 92% (where the peak mass flow rate occurred in S-NC-2). Flow oscillations are calculated for both tests, and were seen during S-NC-3, but were not reported in the S-NC-4 experiment. Some of these predicted oscillations are real, but others are nonphysical and can be inhibited by reducing the time step being used (indicating problems in the time step control algorithm). The results for test S-NC-8, an ultra-small (0.4%) cold leg break, also compare reasonably well with the outcome of that experiment. The overall conclusions and their possible relevance to future RELAP5 code application and development are discussed.

NUREG/CR-3692: POSSIBLE MODES OF STEAM GENERATOR OVERFILL RESULTING FROM CONTROL SYSTEM MALFUNCTIONS AT OCONEE-1 NUCLEAR PLANT. CLARK,F.H.; CLAPP,N.E. Oak Ridge National Laboratory. BROADWATER,R. Tennessee Tech. Univ., Cookeville, TN. July 1984. 50pp.

8409180488. ORNL/TM-9061. 26589:213.

A study has been made of control system failures which might lead to overfill of the steam generator in Babcock and Wilcox nuclear plants. The steam generator and its control system are described. Only one sequence has been found in which a single failure would lead to overfill, and in that case the final stages of the overfill would proceed rather slowly. Because of high level protective features all other failure sequences we have examined require at least two failures to produce overfill beyond the point of high level protection. Several sequences are described in which high level protection features can be placed in an undetected failed state by a control system failure; a subsequent additional failure, occurring prior to the detection and correction of the first failure, could then produce system overfill. Mechanical damage is identified which might be consequent upon steam generator overfill and water entry into the main steam line. Several ways of reducing the probability of steam generator overfill are suggested. No assessment has been made of the probability of occurrence of any of the sequence.

NUREG/CR-3708: LWR SPENT FUEL DRY STORAGE BEHAVIOR AT 229 C. EINZIGER,R.E. Westinghouse Electric Corp. COOK,J.A. EG&G, Inc. July 1984. 134pp. 8408240379. 20255:001.

A whole rod test was conducted at 229 degrees centigrade to investigate the long-term stability of spent fuel rods under a variety of possible dry storage conditions. All combinations of BWR or PWR rods, inert or air atmospheres, and intact or defected rods were tested. After 2235 hours, visual observations, diametral measurements and radiograhic smears were used to assess the degree of cladding deformation and particulate release. The same examinations plus metallography and x-ray analysis were conducted after 5962 hours.

None of the intact rods, the rods tested in inert atmosphere, or the defected PWR rod tested in unlimited air showed any measurable change from the pretest condition. The upper defect on the BWR rod tested in unlimited air had split open ~0.5 in. after 2235 hours and had ~10% cladding deformation. The crack grew to ~2.5 in. after 7962 hours. X-ray analysis indicated that the UO(2) had oxidized to U(3)O(8). The difference in behavior of the upper and lower defects is attributed to the air's accessibility to the fuel because of the deflect's position with respect to the pellet-pellet interface.

The oxidized fuel appeared to form a powdery compact that remained for the most part in the split cladding. Only a fraction of the fuel out of the cladding became airborne. Some crud spalled from the rods but appeared to have no airborne particulate in the 2- to 15-mean respirable range. This report discusses the details and meaning of the data from this test.

NUREG/CR-3711: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE I TEST RESULTS. HWANG,W.S.; ALAMGIR; SUTHERLAND,W.A.; et al. General Electric Co. September 1984. 300pp. 8410100085. EPRI NP-3602. 26901:001.

A new full height BWR system simulator has been built under the Full Integral Simulation Test (FIST) program to investigate the system responses to various transients. The test program consists of two test phases. This report provides a summary, discussions, highlights, and conclusions of the FIST Phase I Tests. Eight matrix tests were conducted in the FIST Phase I. These tests have investigated the large break, small break and steamline break LOCAs, as well as natural circulation and power transients. Results and government phenomena of each test have been evaluated and discussed in detail in this report. Two of these tests tie back to tests in the earlier TLTA facility. Comparisons between the FIST and TLTA tests have been made. The similarities and differences between counterpart tests are identified. Effects of the facility scaling compromises on the test results are identified. One of the FIST program objectives is to assess the TRAC code by comparisons with test data. Two pretest predictions made with TRACB02 are presented and compared with test data in this report. These predictions agree very well with the test results. TRAC's capability to correctly predict the system responses during the transient is demonstrated.

NUREG/CR-3714: ON THE DEVELOPMENT OF ENVIRONMENTAL RADIATION STANDARDS FOR GEOLOGIC DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES. KOCHER,D.C. Oak Ridge National Laboratory. July 1984. 80pp. 8408080422. ORNL-6006. 25977:192.

This report discusses the different technical issues that must be considered in developing an environmental standard for geologic disposal of high-level radioactive wastes. These issues include (1) defining acceptable risk, (2) specifying acceptable risk in the standard, (3) formulating the standard so that reasonable demonstrations of compliance can be obtained, (4) applying the standard to protection of individuals or the population, (5) applying the standard to expected occurrences only or to unexpected processes as well, (6) determining a time limit for the standard, and (7) specifying conditions to be assumed for demonstrating compliance. It is concluded that many issues are not resolvable on technical grounds alone, but that an effective standard will allow flexibility and the exercise of subjective scientific judgements is reaching licensing decisions.

NUREG/CR-3724: ULTIMATE STRENGTH ANALYSES OF THE WATTS BAR,MAINE YANKEE,AND BELLEFONTE CONTAINMENTS. JUNG,J. Sandia Laboratories. July 1984. 82pp. 8409260631. SAND84-0660. 26699:240.

As part of Sandia National Laboratories' Severe Accident Sequence Analysis (SASA) Program, structural analyses of the Watts Bar, Maine Yankee, and Bellefonte containment structures were performed with the objective of obtaining realistic estimates of their ultimate static pressure capacities. The Watts Bar investigation included analyses of the containment shell, equipment hatch, anchorage systems, and personnel lock. The ultimate pressure capability is estimated to be between 120 and 137 psig, corresponding to shell yielding and equipment hatch buckling, respectively. The Main Yankee investigation consisted of an analysis of the containment shell and estimated its failure pressure to be between 96 and 118 psig. For the Bellefonte containment, analyses of the containment shell and equipment hatch were performed. The pressure capacity of the Bellefonte containment is estimated to be between 130 and 139 psig, corresponding to dome tendon yielding and cylinder wall tendon yielding, respectively.

NUREG/CR-3734: LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM.Semiannual Report,October 1982 - March 1983. BERMAN,M. Sandia Laboratories. July 1984. 276pp. 8408100152. SAND84-0688. 25996:001.

This report describes the investigations and analyses conducted at Sandia National Laboratories, Albuquerque, in support of the Light Water Reactor Safety Research Program from October 1982 through March 1983. The Molten Fuel/Concrete Interactions (MFCI) Study investigates the mechanism of concrete erosion by molten core materials, the nature and rate of generation of evolved gases, and the effects of fission-product release. The Core Melt/Coolant Interactions (CMCI) Study investigates the characteristics of explosive and nonexplosive interactions between molten core materials and concrete, and the probabilities and consequences of such interactions. In the Hydrogen Program, the HECTR code for modelling hydrogen deflagration is being developed, experiments (including those in the FITS facility) are being conducted, and the Grand Gulf Hydrogen Igniter System II is being reviewed. All activities are continuing.

NUREG/CR-3735: ACCIDENT-INDUCED FLOW AND MATERIAL TRANSPORT IN NUCLEAR FACILITIES--A LITERATURE REVIEW. BOLSTAD,J.W.; GREGORY,W.S.; MARTIN,R.A.; et al. Los Alamos Scientific Laboratory. July 1984. 45pp. 8408160146. LA-100079-MS. 26122:266.

The reported investigation is part of a program that was established for deriving radiological source terms at a nuclear facility's atmospheric boundaries under postulated accident conditions. The overall program consists of three parts: (1) accident delineation and survey, (2) internal source term characterization and release, and (3) induced flow and material transport. This report is an outline of pertinent induced-flow and material transport literature. Our objectives are to develop analytical techniques and data that will permit prediction of accident-induced transport of airborne material to a plant's atmospheric boundaries.

Prediction of material transport requires investigation of the area of flow dynamics and reentrainment/deposition. A review of material transport, fluid dynamics, and reentrainment/deposition literature is discussed. In particular, those references dealing with model development are discussed with special emphasis on application to a facility's interconnected ventilation system.

NUREG/CR-3739: THE OPERATOR FEEDBACK WORKSHOP:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL. MCGUIRE,M.V.; WALSH,M.E.; BOEGEL,A.J. Battelle Human Affairs Research Centers. September 1984. 246pp. 8409280100. PNL-5214. 26750:001.

This report presents the results of three workshops that were designed, conducted, and assessed for the Nuclear Regulatory Commission. The purposes of the workshops were to (1) examine the effectiveness of workshops and other techniques as mechanisms for obtaining feedback from utility personnel, including comparison of several different workshop procedures; and (2) obtain feedback for the NRC on topics of interest and concern. The workshops were held in NRC Regions I, II, and III between December 1981 and May 1982. A total of 60 utility personnel attended the workshops and offered comments and suggestions concerning staffing, engineering support in the control room, training tools, training programs, and licensing examinations. Workshop participants and observers evaluated the workshops favorably. Further assessment of the workshop process and content suggested that the workshops were effective in obtaining useful feedback for the NRC.

NUREG/CR-3742: BUCKLING OF STEEL CONTAINMENT SHELLS UNDER TIME-DEPENDENT LOADING. BABCOCK,C.D.; BAKER,W.E.; FLY,J.; et al. Los Alamos Scientific Laboratory. July 1984. 38pp. 8408100153. LA-10087-MS. 25997:268.

The problem of dynamic effects for steel containment shells subjected to time-dependent loadings that produce large compressive membrane stresses in the shell wall is considered. Loadings on typical containment structures are reviewed, along with a description of the complete dynamic-buckling problem. Simplifications and the assumptions that are currently used are critically examined and reviewed with respect to buckling analysis. Based on these reviews, three program objectives are defined and the tasks that can accomplish these objectives within a 2-year effort at level funding are outlined in detail.

NUREG/CR-3744 V01: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1983 - MARCH 1984. PUGH,C.E. Oak Ridge National Laboratory. July 1984. 200pp. 8408080355. ORNL/TM-9154/V1. 25979:146.

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

NUREG/CR-3750: JOB ANALYSIS OF NUCLEAR POWER REACTOR HEALTH PHYSICS TECHNICIANS. DAVIS,L.T.; MAZOUR,T.J.; CLARK,P.V.; et al. Analysis & Technology, Inc. August 1984. 200pp. 8410120030. BNL-NUREG-51769. 26980:001.

This report describes a project, an industry-wide Job Analysis of Nuclear Power Reactor Health Physics Technicians (HPTs), sponsored by the Nuclear Regulatory Commission and conducted by Brookhaven National Laboratory and Analysis & Technology, Inc., to provide the industry with job-performance data that can be used in systematically defining training programs in terms of required job functions, responsibilities, and performance standards. The job-analysis methodology is consistent with that used by the Institute of Nuclear Power Operations (INPO) in similar industry-wide projects and includes administration of over 850 job task questionnaires to utility and contractor Health Physics Technicians throughout the country. Data collected includes task performance (difficulty, importance, and frequency) and industry-wide demographics (job levels, experience, education, and training). The results of this project discussed herein include model job descriptions for HPT positions, summaries of HPT experience, education, and training, industry-wide listings with task-performance characteristics, and recommendations of selected tasks as a basis for HPT training development. Finally, potential future applications of the data base by utility and contractor organizations in training program development and evaluation and personnel qualifications are discussed.

NUREG/CR-3751: EFFECTS OF ROCK RIPRAP DESIGN PARAMETERS ON FLOOD PROTECTION COSTS FOR URANIUM TAILINGS IMPOUNDMENTS. ECKER,R.M. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1984. 98pp. 8408130043. PNL-5068. 26011:254.

This report examines the costs of rock riprap flood protection for design flood events at two uranium tailings impoundments in western Colorado. The two sites are the Grand Junction impoundment located along the Colorado River and the Slickrock impoundment located along the Dolores River. The sensitivity of rock type, embankment side slope, and various safety factors is evaluated for six design flood events at Grand Junction and one flood event at Slickrock. The safety factor method of riprap design is used for the cost comparison.

NUREG/CR-3758: CROSSHOLE GEOPHYSICAL METHODS USED TO INVESTIGATE THE NEAR VICINITY OF HIGH LEVEL WASTE REPOSITORIES. RAMIREZ,A.L.; LYTLE,R.J.; HARBEN,P. Lawrence Livermore National Laboratory. August 1984. 76pp. 8409110090. UCID-20060. 26443:290.

An evaluation is given of remote-probing geophysical techniques likely to be used to investigate the near vicinity of geologic repositories for nuclear waste. The sensors to be used would be placed inside the boreholes, shafts and tunnels of the repository to provide high resolution information of the rock near the repository. The geophysical methods evaluated are known as active methods because they make use of artificial seismic, electric or electromagnetic fields to probe rock mass. Techniques involving through transmission measurements are emphasized. These techniques show merit for remote detection of geological heterogeneities such as fracture zones which influence the containment capacity of repository sites. The report discusses the results obtained with exploration methods used at a site near Oracle, Arizona.

NUREG/CR-3761: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR THE OCONEE-1 PRESSURIZED WATER REACTOR. FLETCHER,C.D.; BOLANDER,M.A.; STITT,B.D.; et al. EG&G, Inc. July 1984. 165pp. 8408160248. EGG-2310. 26125:001.

Using the RELAP5 computer code, engineers at the Idaho National Engineering Laboratory (INEL) performed thermal-hydraulic analyses of pressurized thermal shock sequences for the Oconee-1 pressurized water reactor. This report summarizes the results of previously reported calculations and presents the results of more recently completed calculations. Comparisons of two counterpart calculations performed, using the RELAP5 code at the INEL and the TRAC code at Los Alamos National Laboratory (LANL), are included as appendices. The results of these thermal-hydraulic analyses will serve as boundary conditions for fracture-mechanics calculations which are to be performed at Oak Ridge National Laboratory.

NUREG/CR-3763: REVIEW AND ASSESSMENT OF RADIONUCLIDE SORPTION INFORMATION FOR THE BASALT WASTE ISOLATION PROJECT SITE (1979 Through May,1983). KELMERS,A.D. Oak Ridge National Laboratory. September 1984. 47pp. 8410180132. ORNL/TM-9157. 27045:041.

This document presents a scientific review and technological assessment of the radionuclide sorption information reported by the Basalt Waste Isolation Project (BWIP) for the candidate high-level waste repository in the Columbia River basalt flows in the Hanford Reservation. Quantified radionuclide sorption data are necessary for repository performance assessment to model expected radioactivity release rates in groundwater-intrusion-groundwater-migration scenarios. Three key BWIP reports were identified which contain most of the sorption information for a number of radionuclides with basalt, secondary minerals, or interbed materials. An extended review of these data is presented in this document. The technological assessment identified seven potentially significant deficiencies in the radionuclide sorption information published by BWIP that could lead to questionable or nonconservative radioactivity release calculations. These deficiencies are discussed in the document in detail. Specific additional information needs were also defined and reported.

NUREG/CR-3765: MINET SIMULATION OF A HELICAL COIL SODIUM/WATER STEAM GENERATOR,INCLUDING STRUCTURAL EFFECTS. VAN TUYLE,G.J. Brookhaven National Laboratory. September 1984. 27pp. 8410120010. BNL-NUREG-51766. 26985:338.

A test transient performed at a helical coil sodium-to-water steam generator test facility was simulated using the MINET code. It was determined that correct calculation of the sodium outlet temperature requires representation of heat capacitance of the structure.

NUREG/CR-3766: TESTING OF NUCLEAR GRADE LUBRICANTS AND THEIR EFFECT ON A540 AND A193 B7 BOLTING MATERIALS. CZAJKOWSKI,C.J. Brookhaven National Laboratory. September 1984. 83pp. 8409260637. BNL-NUREG-51767. 26699:322.

An investigation was performed on eleven commonly used lubricants by the nuclear power industry which included EDS analysis of the lubricants, notched-tensile constant extension rate testing of bolting materials with the lubricants, frictional testing of the lubricants and weight loss testing of a bonded solid film lubricant. The report concludes that there is a significant amount of variance in the mechanical properties of common bolting materials, that MoS2 can hydrolyze to form H2S at 100 degrees C and cause stress corrosion cracking (SCC) of bolting materials. One of the most significant findings of this report is the observation that both A193 B7 and A540 B24 bolting materials are susceptible to transgranular stress corrosion cracking in demineralized H2O at 280 degrees C in notched tensile tests.

NUREG/CR-3776: TESTING OF SAFETY-RELATED NUCLEAR POWER PLANT EQUIPMENT AT THE CENTRAL RECEIVER TEST FACILITY. DANDINI,V.J.; ARAGON,J.J. Sandia Laboratories. July 1984. 86pp. 8409110074. SAND83-1960. 26437:198.

The use of a solar energy facility for simulating the thermal environment (heat flux) produced as a result of hydrogen burns in a full-scale reactor containment building is described. Using a heat flux profile developed from calculations performed by the HECTR computer code, the Central Receiver Test Facility simulated the multiple burn thermal environment which HECTR predicted would result from the deliberate ignition of hydrogen generated by an S2D accident. Functioning specimens of reactor monitoring and safety system equipment were exposed to this environment. Results of the equipment performance and temperature response are presented.

NUREG/CR-3777: CAPABILITIES AND DIAGNOSTICS OF THE SANDIA PELLETRON-RASTER SYSTEM. BUCKALEW,W.H.; LOCKWOOD,G.J.; LUKER,S.M.; et al. Sandia Laboratories. July 1984. 61pp. 8408080367. SAND84-0912. 25978:090.

The radiation capabilities of the PELLETRON Electron Beam Accelerator have been expanded to include a controllable, variable dimension, beam diffusion option. This rastered beam option has been studied in detail. Beam characteristics have been determined as a function of incident electron beam energy, current, and deflection system parameters. The beam diagnostics required to define any given diffuse beam pattern are accurate and predictable. Recently, utility of this added PELLETRON capability was demonstrated by simulating the effects of complex nuclear reactor accident electron environments on electrical insulation materials similar to those used in nuclear power plants.

NUREG/CR-3786: A REVIEW OF REGULATORY REQUIREMENTS GOVERNING CONTROL ROOM HABITABILITY SYSTEMS. JACOBUS,M.J. Sandia Laboratories. August 1984. 63pp. 8410120011. SAND84-0978. 26978:201.

This report reviews applicable guides, standards, and codes which govern the design, manufacture, selection, installation, and surveillance practices for components and systems important to control room habitability. It covers the fundamental guidance contained in General Design Criteria, Regulatory Guides, and applicable sections of the Standard Review Plan, as well as numerous documents referenced by this guidance.

Instances are cited where the present guidance is misleading, contradictory, or vague. In some cases, the problems in the guidance result from inadequate technical bases; in other cases, the problems result from several documents which are not completely consistent.

To independently assess the suitability of the regulatory guide which covers accidental chlorine releases, a computer program was developed to calculate chlorine concentrations in the control room following chlorine release. Although problems with the assumptions used to develop the guide were found, the conservative nature of the chlorine calculations appears to adequately compensate for these problems.

NUREG/CR-3787: EFFECTIVENESS OF ENGINEERED SAFETY FEATURE (ESF) SYSTEMS IN RETAINING FISSION PRODUCTS.Background Information. MISHIMA,J.; BLAHNIK,D.E.; HALVERSON,M.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 115pp. 8409170417. PNL-5101. 26499:206.

The Pacific Northwest Laboratory has compiled and reviewed base line data on the effectiveness of Engineered Safety Feature (ESF) systems in the retention of fission products and particulate material resulting from a nuclear reactor accident. This work is part of an NRC project to provide the best estimates of the consequences of severe reactor accidents.

The resulting report describes the ESF systems (containment spray, secondary containment filter, containment recirculating filter, pressure suppression pool, ice condenser, and main steam line isolation valve leakage control systems). Also described are the anticipated atmospheres in which the ESFs must operate, the experimental studies of ESF system effectiveness, and the models currently available for assessing the performance of the various ESF systems. The information gaps identified as a result of this review have resulted in recommendations for additional work in the areas of: 1) performance data and models of containment chiller/coolers; 2) continued development and experimental verification of the ice condenser model; 3) continued development of the pressure suppression pool model; and 4) continued investigations of the behavior of filtration devices.

NUREG/CR-3788 V01: STRUCTURAL INTEGRITY OF LIGHT WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Four-Year Plan 1984-1988. * Materials Engineering Associates, Inc. September 1984. 111pp. 8410180235. MEA-2047. 27045:088.

This document is the first in a series intended to provide an up-to-date statement of the four-year plan for the program, "Structural Integrity of Light Water Reactor Pressure Boundary Components," which is being conducted by Materials Engineering Associates, Inc. (MEA). This program consists of engineering and research in fracture, fatigue, and radiation sensitivity of nuclear structural steels and weldments and addresses many of the key uncertainties in the margin of safety in operating nuclear plants. All tasks are integrated to focus on structural integrity of LWR pressure boundary components. The approach centers on an experimental characterization of nuclear grade steels and an assessment of fracture and fatigue behavior under conditions of a nuclear environment, so investigation of irradiated materials is a key element of each task. Experimental studies are supported by analytical models and investigation of the mechanisms responsible for the observed behavior. Data developed in the program will provide the basis for recommendations for the ASME Boiler and Pressure Vessel Code and ASTM test methods, and revisions to NRC Guides.

NUREG/CR-3792: CLOSEOUT OF IE BULLETIN 79-11:FAULTY OVERCURRENT TRIP DEVICE IN CIRCUIT BREAKERS FOR ENGINEERED SAFETY SYSTEMS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. August 1984. 34pp. 8408310084. IEB-79-11. 26348:057.

IE Bulletin 79-11 was issued May 22, 1979 as a result of information received in April 1979 from Westinghouse and an NRC licensee relating to the potential failure of a circuit breaker in an engieered safety system of a nuclear power plant. The defect of concern was a small hairline crack in the dashpot end cap of one of the three overcurrent trip devices of a Type DB-75 breaker. The Bulletin was also applicable to Type DB-50 breakers, because they use the same type of dashpot end cap. The defective end cap had been installed in 1973 as a replacement, in compliance with IE Bulletin 73-1. Westinghouse Technical Bulletin NSD-TB-79-02 was issued April 17, 1979 to alert utilities to the potential problem, to provide background information, to recommend review of calibration test data and retesting of erratic breakers, to advise visual examination of end caps for cracks and to call for replacement of cracked end caps. Evaluation of utility responses and NRC/IE inspection reports shows that 114 of the 129 current facilities do not use the affected breakers in safety-related systems. Followup items for the five facilities with open status are proposed. The Bulletin has been closed out for the remaining ten facilities with safety-related Westinghouse DB-50 and DB-75 breakers having dashpots, on the basis of acceptable utility responses and NRC/IE regional inspection reports. Erratic performance of three DB-50 breakers with worn seals at one facility is identified as a Remaining Area of Concern because the worn seals had essentially the same effect on performance as cracked end caps. The recommendation is made that preventive mainte[illegible]e programs of licensees be reviewed to make sure that breakers are kept clean to avoid plugging dashpot orifices. The Bulletin has served its purpose by resulting in identification of the potential problem at a limited number (15) of facilities and of the need for corrective actions at only five facilities.

NUREG/CR-3795: CLOSEOUT OF IE BULLETIN 82-04:DEFICIENCIES IN PRIMARY CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES. FOLEY,W.J.; HENNICK,A. Parameter, Inc. July 1984. 53pp. 8408090272. IEB-82-04. 25983:001.

IE Information Notice 82-40 was issued September 22, 1982 as an early notification of a potentially significant problem pertaining to electrical penetration assemblies (EPAs) supplied by the Bunker Ramo Corporation (BRC) of Chatsworth, California. All deficiencies described in the Notice were identified as existing in BRC EPAs with a hard epoxy module design. Utility personnel were asked to review the Notice and take appropriate actions, but were not required to respond or take any specific action. After further study, NRC concluded that there were potential generic safety implications at a limited number of plants. Accordingly, IE Bulletin 82-04 was issued December 3, 1982 to require responses and specific actions by all licensees and holders of construction permits. Evaluation of utility responses, deficiency reports and NRC/IE inspection reports has resulted in Bulletin closeout for 124 of 129 current facilities. Deficiencies described in the Bulletin were identified at all facilities, of which two are operating and nine under construction. Followup of corrective actions and verification of inspection procedures are proposed in Appendix C for the five facilities with affected assemblies are summarized in Table B.6. Completion by NRC/IE of all the followup items identified in Appendix C is expected to resolve fully the specific problem of Bunker Ramo electrical penetrations that utilized a hard epoxy design.

NUREG/CR-3796: EMERGENCY PREPAREDNESS SOURCE TERM DEVELOPMENT FOR THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS LICENSED FACILITIES. SUTTER,S.L.; MISHIMA,J.; BALLINGER,M.Y.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 352pp. 8409040447. PNL-5081. 26363:049.

To establish requirements for emergency preparedness plans at facilities licensed by the Office of Nuclear Materials Safety and Safeguards, the Nuclear Regulatory Commission (NRC) needs to develop source terms (the amount of material made airborne) for accidents. They are used to estimate potential public doses from the events, which will be used to guide whether emergency preparedness plans are needed. Pacific Northwest Laboratory is providing the NRC with source terms by developing accident scenarios for fuel cycle and by-product operations. Several scenarios are developed for each operation, leading to the identification of the maximum release considered for emergency preparedness planning (MREPP) scenario. Fire was the MREPP at oxide fuel fabrication, UF(6) production, radiopharmaceutical manufacturing, radiopharmacy, sealed source manufacturing, waste warehousing, and university research and development facilities. Tornadoes were MREPP events for uranium mills and plutonium contaminated facilities, and criticalities were significant at nonoxide fuel fabrication and nuclear research and development facilities. Techniques for adjusting the MREPP release to different facilities are also described.

NUREG/CR-3798: CHARACTERIZATION OF CEMENT AND BITUMEN WASTE FORMS CONTAINING SIMULATED LOW-LEVEL WASTE INCINERATOR ASH. WESTSIK,J.H.; BUSCHBOM,R.L.; DIVINE,J.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 100pp. 8408310075. PNL-5153. 26347:319.

Incinerator ash from the combustion of general trash and ion exchange resins were immobilized in cement and bitumen. Tests were conducted on the resulting waste forms to provide a data base for the acceptability of actual low-level waste forms. The testing was done in accordance with the Technical Position on Waste Form. Bitumen had a measured compressive strength of 120 psi and a leashability index of 13 as measured with the ANS 16.1 leach test procedure. Cement demonstrated a compressive strength of 1400 psi and a leachability index of 7. Both waste forms easily exceed the minimum compressive strength of 50 psi and leachability index of 6 specified in the Technical Position. Irradiation of 10(8) RAD and exposure to thirty +60 degrees to -30 degrees centigrade thermal cycles did not significantly impact these properties. Neither waste form supported bacterial or fungal growth as measured with ASTM G21 and G22 procedures. Neither bitumen nor cement containing incinerator ash caused any corrosion or degradation of potential container materials including steel, polyethlyene and fiberglass. However, moist ash did cause corrosion of the steel.

NUREG/CR-3804 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report January - March 1984. * Argonne National Laboratory. July 1984. 1p. 8408080374. ANL-84-35. 25979:356.

This quarterly progress report summarizes work done during the months of January-March 1984 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research in the U. S. Nuclear Regulatory Commission. The work in the Applied Physics Division includes reports on reactor safety modeling and assessment by members of the Reactor Safety Appraisals Section. Work on reactor core thermal-hydraulics is performed in ANL's Components Technology Division, emphasizing 3-dimensional code development for LMFBR accidents under natural convection conditions. An executive summary is provided including a statement of the findings and recommendations of the report.

NUREG/CR-3806: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS: Annual Report,October 1982 - September 1983. SHACK,W.J.; KASSNAR,T.F.; KUPPERMAN,D.; et al. Argonne National Laboratory. August 1984. 143pp. 8410030340. ANL-84-36. 26820:086.

This progress report summarizes work on environmentally assisted cracking in lightwater reactors during the twelve months from October 1982 - September 1983. The objective of this program is to develop an independent capability for prediction, detection, and control of intergranular stress corrosion cracking (IGSCC) in lightwater reactor (LWR) systems. The program is primarily directed at IGSCC problems in existing plants, but also includes the development of recommendations for plants under construction and future plants. The scope includes the following: (1) development of the means to evaluate acoustic leak detection systems objectively and quantitatively; (2) evaluation of the influence of metallurgical variables, stress, and the environment on IGSCC susceptibility, including the influence of plant operations on these variables; and (3) examination of practical limits for these variables to effectively control IGSCC in LWR systems. The initial experimental work concentrates primarily on problems related to pipe cracking in BWR systems. However, ongoing research work on other environmentally assisted cracking problems involving pressure vessels, nozzles, and turbines will be monitored and assessed, and where unanswered technical questions are identified, experimental programs to obtain the necessary information will be developed to the extent that available resources permit.

NUREG/CR-3812: ASSESSMENT OF IRRADIATION EFFECTS IN RADWASTE CONTAINING ORGANIC ION-EXCHANGE MEDIA. SWYLER,K.J.; DODGE,C.J.; DAYAL,R. Brookhaven National Laboratory. September 1984. 82pp. 8410030353. BNL-NUREG-51774. 26820:001.

Recently, regulatory consideration has been devoted to the effects of self-irradiation on radwaste containing organic ion exchange media. This consideration was prompted by decontamination operations at TMI-II, and by the development of technical positions in support of NRC regulation 10 CFR 60. This report addresses the effects of high radiation dose on the storage and disposal of radwaste ion-exchange media, and the validity of laboratory test procedures for predicting field performance. Our work shows that accelerated testing of ion-exchange media using high-dose-rate external gamma irradiation appears to be a valid procedure for assessing certain aspects of field behavior--i.e., radiolytic scission of the resin functional group, radiolytic gas generation of free liquids and resin agglomeration, provided both the test data and the field conditions refer to storage in a closed environment. Certain resin decomposition processes appear to depend largely on resin moisture content, and may not be particularly sensitive to resin loading. One practical consequence of radiolytic acidity is to promote the corrosion of mild steel in irradiated resin. However, the corrosion process is very complex. Case-specific, long-term (i.e., low radiation dose) evaluations might be necessary if rigorous guidelines to protect radwaste containers against corrosion are required.

NUREG/CR-3813: MINET VALIDATION STUDY USING STEAM GENERATOR TRANSIENT DATA. VAN TUYLE,G.J. Brookhaven National Laboratory. September 1984. 39pp. 8410120057. BNL-NUREG-51775. 26986:100.

Three steam generator transient test cases, that were simulated using the MINET computer code, are described, with computed results compared against experimental data. The MINET calculations closely agreed with the experiment for both the once-through and the U-tube steam generator test cases. The effort is part of an ongoing effort to validate the MINET computer code for thermal-hydraulic plant systems transient analysis, and strongly supports the validity of the MINET models.

NUREG/CR-3815: STATISTICAL EVALUATION OF THE METALLURGICAL TEST DATA IN THE ORR-PSF-PVS IRRADIATION EXPERIMENT. STALLMAN,F.W. Oak Ridge National Laboratory. August 1984. 34pp. 8409170414. ORNL/TM-9207.

26498:164.

A statistical analysis of Charpy test results of the two-year Pressure Vessel Simulation metallurgical irradiation experiment was performed. determination of transition temperature and upper shelf energy derived from computer fits compare well with eyeball fits. Uncertainties for all results can be obtained with computer fits. The results were compared with predictions in Regulatory Guide 1.99 and other irradiation damage models.

NUREG/CR-3818: REPORT OF RESULTS OF NUCLEAR POWER PLANT AGING WORKSHOPS. CLARK,N.H.; BERRY,D.L. Sandia Laboratories. August 1984. 64pp. 8408240320. SAND-84-0374. 26237:284.

Two workshops were conducted to identify whether there is any evidence of component or structural aging problems in nuclear power plants, and, if so, what problems are of greatest importance. Fifteen representatives from national laboratories, architect/engineers, nuclear steam supply system vendors, research firms, and a university participated in the workshops. Based on completed questionnaires and group discussions which screened over 112 components believed to be susceptible to excessive aging, pressure/temperature sensors, valve operators, and snubbers emerged by consensus as the most important aging issues. Potential aging problems related to off-normal common mode effects or aging problems which are just now developing were found to be outside the scope of the workshops, because little or no first hand experience is available for these off-normal or yet to develop circumstances. Recommendations are made for a systematic approach to rate components in terms of overall safety and for a cooperative effort between industry research groups and regulatory research groups to resolve known aging problems and to identify off-normal or yet to develop aging issues.

NUREG/CR-3820 V01: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM,Quarterly Report,January-March 1984. THOMPSON,S.L. Sandia Laboratories. July 1984. 62pp. 8408100147. SAND84-1025/1. 25996:280.

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

The first quarter of FY84 marked the beginning of the TRAC-PF1/MOD1 independent assessment project at SNLA. The code was obtained from Los Alamos National Laboratory (LANL) in October, and brought up on both our Cyber-76 and Cray-1S computers. The assessment matrix was formalized, several TRAC nodalizations for the various facilities required were developed, and limited calculations were begun, all described in the last quarterly. During this quarter, more nodalizations were developed and calculations begun, and the first PF1/MOD1 assessment analysis was completed.

NUREG/CR-3821: EVALUATION OF CRACK PLANE EQUILIBRIUM MODEL FOR PREDICTING PLASTIC FRACTURE. BUTLER,T.A.; SMITH,F.W. Los Alamos Scientific Laboratory. July 1984. 23pp. 8409110107. LA-10129-MS. 26437:328.

A simple model for predicting the initiation of crack growth during plastic fracture is evaluated. The model is based on requiring equilibrium between applied loads and an assumed stress distribution in the uncracked ligament near the crack. The fracture parameters required are the material's ultimate tensile strength and a process-zone size at the crack tip that is determined from simple fracture tests. The Crack Plane Equilibrium model predicts crack-growth initiation for the crack geometries studied with sufficient accuracy to warrant extending it for investigating other geometries and for predicting stable crack growth and the onset of unstable crack growth.

NUREG/CR-3822: SOLA-PTS: A Transient,Three-Dimensional Algorithm For Fluid-Thermal Mixing And Wall Heat Transfer In Complex Geometrics. DALY,B.J.; TORREY,M.D. Los Alamos Scientific Laboratory. July 1984. 103pp. 8409110081. LA-10132-MS. 26445:203.

The SOLA-PTS computer code has been developed to analyze fluid-thermal mixing in the cold legs and downcomer of pressurized water reactors in support of the pressurized thermal shock study. SOLA-PTS is a transient, three-demensional code with the capability of resolving complex geometries using variable cell noding in the three coordinate directions. The computational procedure is second-order accurate and utilizes a state-of-the-art iteration method that allows rapid convergence to an accurate solution for the pressure field. Two different turbulence models are used in the code, a two-equation k-e model that is used in the cold leg pipe away from the HPI inlet and a three-equation k-e-T' model for use near the HPI inlet and in the downcomer.

The physical modeling and the numerical procedure used in SOLA-PTS are described in this report. Applications of the method to two Creare 1/5th-scale experiments are also presented. Two appendices are included. Appendix A provides a comparison of the two- and three-equation turbulence models, while Appendix B provides instructions for setting up and running a problem with SOLA-PTS.

NUREG/CR-3824: CONTING PROGRAM GUIDE. HENRY,E.B.; GENTILLON,C.D.; STEVERSON,J.A. EG&G, Inc. September 1984. 140pp. 8410030079. EGG-2315. 26821:001.

CONTING is an interactive computer program for automated trends and pattern analysis of data. The data are License Events Reports, which are coded into a computer-readable, searchable Sequence Coding and Search System (SCSS) format developed by the United States Nuclear Regulatory Commission. In the SCSS, the data are broken down into occurrences or steps, which are described by categorical variables such as system, component, and cause. CONTING searches the steps to obtain counts for contingency tables (hence, its name). The rows and columns of these tables correspond to user specified conditions for the variables. The pattern of counts appearing in such a table can provide insights concerning the operational experience at nuclear power plants. In addition, CONTING supports trend analysis since the counts can be grouped by the associated event dates. A statistical package formats the contingency tables; this facilitates the use of log-linear statistical program may include exposure times for use in normalizing the counts to obtain occurrence rates. CONTING has many features to aid the user in performing this analysis. CONTING was developed at the Idaho National Engineering Laboratory (INEL) on the CYBER 176, using FORTRAN 77. It operates on the SCSS data base located at the INEL.

NUREG/CR-3826: RECOMMENDATIONS FOR PROTECTING AGAINST FAILURE BY BRITTLE FRACTURE IN FERRITIC STEEL SHIPPING CONTAINERS GREATER THAN FOUR INCHES THICK. SCHWARTZ,M.W. Lawrence Livermore National Laboratory. July 1984. 131pp. 8408010155. UCRL-53538. 25872:087.

Various criteria for protecting against brittle fracture in spent-fuel shipping containers made from ferritic steel forgings greater than four inches thick are evaluated. A fracture initiation criterion based upon yield stress levels and allowable flaw sizes specified in Section XI of the ASME Code is recommended. This recommendation is based upon a value evaluation taking into account its effect upon industry and the risk of brittle fracture.

NUREG/CR-3830 V01: AEROSOL RELEASE AND TRANSPORT PROGRAM,SEMIANNUAL PROGRESS REPORT FOR OCTOBER 1983 - MARCH 1984. ADAMS,R.E.; TOBIAS,M.L. Oak Ridge National Laboratory. July 1984. 79pp. 8409170442. ORNL/TM-9217/V1. 26498:199.

This report summarizes progress for the Aerosol Release and Transport Program sponsored by the Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Accident Evaluation, for the period October 1983-March 1984. Topics discussed include (1) the experimental program in the Fuel Aerosol Simulant Test (FAST) facility, (2) NSPP experiments involving mixtures of aerosols of iron oxide and uranium in steam and dry atmospheres, (3) support work for the DEMONS (West Germany) and Marviken (Sweden) projects, (4) analysis of core melt experiments involving boric oxide volatility, (5) initial operation of the new 250-kW induction generator, (6) comparisons of NAUA results with experiments, and (7) tests and improvements in the UVABUBL-II code.

NUREG/CR-3832: UNCERTAINTIES IN LONG-TERM REPOSITORY PERFORMANCE DUE TO THE EFFECTS OF FUTURE GEOLOGIC PROCESSES. SJOREEN,A.L.; KOCHER,D.C. Oak Ridge National Laboratory. August 1984. 41pp. 8409110084. ORNL-6049. 26447:099.

This report discusses uncertainties in predicting the long-term performance of geologic repositories for high-level waste that result from the effects of future geologic processes. This type of uncertainty arises from uncertainties in determining current rates of geologic processes, predicting process rates over long time periods in the future, and predicting the effects of future geologic processes on performance. This report emphasizes the qualitative and judgmental nature of predictions of future geologic processes and their effects on repository performance. However, significant changes generally occur over time periods of 100,000 years or more. Thus, at sites chosen for their stability, geologic processes should not have significant effects on repository performance over a period of 10,000 years.

NUREG/CR-3833: BEHAVIOR OF SUBCRITICAL AND SLOW-STABLE CRACK GROWTH FOLLOWING A POST-IRRADIATION THERMAL ANNEAL CYCLE. CULLEN,W.H.; HISER,A.L. Materials Engineering Associates, Inc. August 1984. 41pp. 8409200295. MEA-2048. 26609:061.

This report presents the experimental results of Phase I of a Small Business Innovation Research Program which investigated the response of environmentally-assisted monotonic and cyclic crack growth following a simulated anneal of a reactor-pressure vessel weld. Unirradiated steels were used in this (initial) Phase I of the program. Fatigue cracks were grown in several specimens of a submerged arc weld deposit in pressurized, high-temperature reactor-grade water. The specimens were removed from the environment, and annealed for one week at either 399 degree C or 454 degree C. Fatigue crack growth in high-temperature water was resumed on several annealed specimens and unannealed controls. No effect of the anneal was noted on the fatigue crack growth rates, which continued with about the same degree of environmental assistance as exhibited before the anneal. An elastic-plastic fracture specimen, tested in 93 degree C air at a very slow loading rate, showed that neither annealing nor the slow rate had a significant effect on the J-R curve characteristics. However, conducting the tests at a slow loading rate in 93 degree C PWR water resulted in a 25% to 30% decrease in JIc and a small decrease in T avg. Examination of the oxides on the fatigue fracture surfaces showed that magnetite (formed during the crack growth in pressurized, high-temperature water) was the predominant oxide specie.

NUREG/CR-3834: ON THE THRESHOLD SULFUR AND LITHIUM TO SULFUR RATIO IN STRESS CORROSION CRACKING OF SENSITIZED ALLOY 600 IN BORATED THIOSULFATE SOLUTION. BANDY,R.; KELLY,K. Brookhaven National Laboratory. July 1984. 35pp. 8408070014. BNL-NUREG-51785. 25954:296.

The stress corrosion cracking (SCC) of sensitized Alloy 600 was investigated in aerated solutions of sodium thiosulfate containing 1.3% boric acid, using U-bends, constant load, and slow strain rate tests. The aim of the investigation, among others, was to determine the existence, if any, of a threshold level of sulfur, and Li to S ratio governing the SCC. For U-bends, 5 ppm Li as LiOH in the presence of 7 ppm S as thiosulfate prevented occurrence of SCC. However, in slow strain rate tests, significant SCC occurred at a S level of 30 ppb in the presence of 0.7 ppm of Li. For a specimen held under constant load, a propagating crack continued to grow until fracture during controlled progressive dilution of the bulk solution, leading to final Li concentration of 1.5 ppm and S concentration of 9.6 ppb respectively. The implications of the results to initiation and propagation of SCC in aerated thiosulfate solutions, and their relevance to future operation of the steam generators at Three Mile Island Unit 1 (TMI-1) are discussed.

NUREG/CR-3835: SIMULATION OF FLAME PROPAGATION THROUGH VORTICITY REGIONS USING THE DISCRETE VORTEX METHOD. BARR,P.K. Sandia Laboratories. September 1984. 19pp. 8410170076. SAND84-8715. 27041:045.

The interaction of a freely propagating premixed flame with regions of high vorticity in the flow is investigated using a computer model. These vorticity regions are formed due to the flame-generated volume expansion that pushes gas past obstacles ahead of the flame. In the computer model the discrete vortex dynamics method is used to simulate the time development of the vorticity regions downstream of each obstacle. The flame front is modeled as a wrinkled laminar flame interface that propagates normal to itself at the laminar burning velocity, separating the two different density fluids: burned and unburned. Two different obstacle configurations are discussed in this paper. In the first case, a flame causes unburned gas to exhaust out of a planar duct, and when the flame reaches the duct exit it interacts with the vorticity which was formed at the exit. Two versions of this configuration are considered: sharp and square edge exit. The second case involves a series of obstacles in a channel.

Here, the repeated obstacles in the channel leads to acceleration of the flame as indicated by the dramatic increase in fuel consumption.

NUREG/CR-3840: COST ANALYSIS FOR POTENTIAL MODIFICATIONS TO ENHANCE THE ABILITY OF A NUCLEAR PLANT TO ENDURE STATION BLACKOUT. CLARK,R.A.; THOMAS,W.R.; et al. Science & Engineering Associates, Inc. RIORDON,B.J. MATHTECH, Inc. July 1984. 167pp. 8408080472. 25980:160.

Cost estimates were required to serve as partial bases for decisions on four potential nuclear reactor facility modifications being considered in the resolution of USI A-44, Station Blackout. The modification constituting the four subtasks in this report are (1) increasing battery capacity, (2) adding an AC-independent charging pump for reactor coolant seal injection, (3) increasing condensate storage tank capacity, and (4) increasing compressed air supply for instrument air.

The cost estimates contained in this report include those for the following: (1) engineering and design, (2) equipment, materials, and structures, (3) installation, and (4) present worth of the annual operation and maintenance over the remaining useful life of the reactor.

In addition to providing engineering requirements for the four modifications, the report evaluates the potential for synergistic solutions. It was found that some modifications to provide for reactor coolant seal injection would effectively satisfy the DC system augmentation requirements, with the costs for solving both problems being competitive with that of additional batteries alone. The report also identifies an innovative potential solution to the DC system capacity problem through the use of high energy density primary batteries which would be far more cost effective than the addition of traditional lead acid batteries for mitigating extended station blackout effects.

NUREG/CR-3842: STEAM GENERATOR GROUP PROJECT TASK 8 - SELECTIVE TUBE UNPLUGGING. WHEELER,K.R.; DOCTOR,P.G.; FETROW,L.K.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1984. 200pp. 8410170217. PNL-4876. 27029:034.

The Steam Generator Group Project utilizes a retired from service pressurized water reactor steam generator as a test bed and source of specimens for research. Program objectives emphasize validation of the ability to nondestructively characterize the condition of the steam generator tubing in service.

During operation 748 of the 3388 tubes in the research generator were removed from service by explosive plugging on both ends. Tubes were plugged due to defect indications, inspectability limits caused by denting, and as a preventative measure. The plugged tubes contained a substantial portion of the defects necessary for the research program. This report summarizes activities conducted during a campaign removal of 969 explosive tube plugs. The report provides detailed descriptions of the planning, training, supplies, equipment, and operations that led to the successful completion of the unplugging in 20 days of operation. Also presented is information on problems encountered and observations that could aid future unplugging operations.

NUREG/CR-3843: STEAM GENERATOR GROUP PROJECT TASK 10 - SECONDARY SIDE EXAMINATION. SCHWENK,E.B.; WHEELER,K.R. Battelle Memorial Institute, Pacific Northwest Laboratories. July 1984. 69pp. 8410120034. PNL-5033. 26985:269.

The Steam Generator Group Project utilizes a retired from service pressurized water reactor steam generator as a test bed and source of specimens for research. Program objectives emphasize validation of the ability to nondestructively characterize the condition of steam generator tubing in service. Remaining integrity of tubing with service induced defects is studied through burst and leak rate tests. Other program objectives seek to characterize overall generator condition, including secondary side structure, and provide realistic samples for development of primary side decontamination, secondary side cleaning, and nondestructive examination technology.

This report provides information on secondary side characterization efforts. The methods and equipment used are discussed, along with comparisons of benefits offered by various techniques. Details of secondary side steam generator conditions are then presented, emphasizing support plate and U-bend regions.

NUREG/CR-3844: CHARACTERIZATION OF THE RADIOACTIVE WASTE PACKAGES OF THE MINNESOTA MINING AND MANUFACTURING COMPANY. KEMPF,C.R.; SISKIND,B.; BARLETTA,R.E.; et al. Brookhaven National Laboratory. July 1984. 93pp. 8408010151. BNL-NUREG-51787. 25872:001.

An evaluation of the low-level waste packages generated by Minnesota Mining and Manufacturing Co. (3M) was made on the basis of 10 CFR Part 61 criteria and on the Technical Position on Waste Form and Waste Classification (TP). This evaluation was the result of a study initiated by the U. S. Nuclear Regulatory Commission (NRC), in which 3M participated.

3M produces a variety of radioactive products and wastes. The dominant radioisotopes are Po-210 and Cs-137. The Po-210 packages are generally Class A and meet the requirements in 10 CFR Part 61. The Cs-137 and Sr-90 packages fall into all three waste classifications (A, B, and C). These wastes are packaged by 3M in 30-gallon or 55-gallon carbon steel drums (Class A) or 30-gallon lined drums (Class B and C). The Class B and greater lead- and concrete-lined packages have been evaluated with respect to meeting the stability requirements for waste disposed of in a high integrity container. When so evaluated, eleven areas of concern were identified with respect to the regulations and recommendations in the TP.

NUREG/CR-3845: PREDICTION OF NONLINEAR STRUCTURAL RESPONSE IN LMFBR ELEVATED-TEMPERATURE PIPING. FARRAR,C. Los Alamos Scientific Laboratory. July 1984. 28pp. 8409170447. LA-10090-MS. 26498:295.

The development of structural analysis capabilities to investigate possible accident initiations caused by structural degradation o liquid metal fast breeder reactor (LMFBR) piping is summarized. The ABAQUS finite element code is used to perform a non-linear analysis of a bench mark problem proposed by the Pressure Vessel Research Committee. The problem is representative both in geometry and loading of an LMFBR elevated-temperature piping system, and published analytical results are available for comparison. Results show the system to be most sensitive to large, radial, thermal gradients that occur when the system experiences certain thermal transients. Repeated cycles of these transients will lead to thermal ratcheting, causing progressive deformation and strain accumulation in the system. Future work will verify the accuracy of the finite element model and quantify damage accumulated during the lifetime of an LMFBR elevated-temperature piping system.

NUREG/CR-3851 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for October-December 1983. KELMERS,A.D.; KESSLER,J.H.; ARNOLD,W.D.; et al. Oak Ridge National Laboratory. August 1984. 48pp. 8408300282. ORNL/TM-9191/V1. 26331:224.

Oak Ridge National Laboratory (ORNL) is conducting an experimental investigation of geochemical information for the Nuclear Regulatory Commission (NRC). During this quarter, the project evaluated both radionuclide solubility data and retardation parameters reported by the Basalt Waste Isolation Project (BWIP), and the methodologies used to develop those values. Under oxic conditions, neptunium had a sorption ratio of 1.7 L/kg for McCoy Canyon basalt and synthetic groundwater GR-2, which is lower than the "conservative best estimate" value recommended by BWIP. Under anoxic conditions, the basalt showed little or no ability to remove technetium (VII) from GR-2 by sorption or precipitation. Several important concerns may make it impossible to assert that the addition of hydrazine to groundwater is modeling the repository redox condition. These are: (1) its reaction with any reducible solute is undefined, (2) its dissociation to release hydroxide ions probably dominates the groundwater pH, (3) it could react with bicarbonate to form the carbamate ion, (4) it is corrosive to polycarbonate or polypropylene test tubes, (5) it may alter or disaggregate clay mineral structure, and (6) uncertainty exists as to the solid phase or solution species formed by reaction with pertechnetate ion. Thus, BWIP data obtained in the presence of hydrazine may be nonconservative for use in assessment studies.

NUREG/CR-3852: INSIGHT INTO PRA METHODOLOGIES. GALLAGHER,D. Science Applications, Inc. August 1984. 121pp. 8409200290. 26607:001.

This report describes the results of a survey of six probabilistic risk assessments to determine the impact of different aspects of the methodology on dominant sequence ordering and core-melt likelihood. The results indicate that effort should be given to human error analysis, system dependency analysis, and modeling of AC power systems.

NUREG/CR-3856: AN ULTRASONIC LEVEL AND TEMPERATURE SENSOR FOR POWER REACTOR APPLICATIONS. DRESS,W.B.; MILLER,G.N. Oak Ridge National Laboratory. August 1984. 30pp. 8409180280. ORNL/TM-9236. 26589:001.

An ultrasonic waveguide employing torsional and extensional acoustic waves has been developed for use as a level and temperature sensor in pressurized and boiling water nuclear power reactors. Features of the device include continuous measurement of level, density, and temperature producing a realtime profile of these parameters along a chosen path through the reactor vessel.

NUREG/CR-3867: DATA SUMMARIES OF LICENSEE EVENT REPORTS OF INVERTERS AT U.S. COMMERCIAL NUCLEAR POWER PLANTS,JANUARY 1,1976 TO DECEMBER 31,1982. BROWN,S.R.; TROJOVSKY,M. EG&G, Inc. August 1984. 131pp. 8409280077. EGG-2324. 26762:206.

This report describes a computer-based data file developed from License Event Reports (LERs) of inverters in U.S. commercial nuclear power plants for the period January 1, 1976 to December 31, 1982. In addition to the creation of the file, summaries of data contained in the file were made to obtain data for risk assessment and statistical purposes. Gross constant failure rates were estimated for inverters found in selected systems. Explanations, figures, and summary tables of the results are provided.

NUREG/CR-3868: CONTAINMENT BUILDING ATMOSPHERE RESPONSES DUE TO REACTOR GAS BURNING UNDER SEVERE ACCIDENT CONDITIONS. KRUEGER,P.G. Brookhaven National Laboratory. July 1984. 42pp. 8410030363. BNL-NUREG-51793. 26817:315.

The formation of combustible atmospheres during unrestricted core heatup accidents in High Temperature Gas-Cooled Reactors is being investigated, considering the effects of only partially mixed atmospheres. It is found that the previously used assumption of complete mixing presents the more severe limit in most cases. In the few cases where higher loads were obtained, these were still below the invocation of even more remote failure scenarios. A qualitative discussion applying the above results to comparable accident at Fort St. Vrain is included.

NUREG/CR-3869: ANALYSIS OF THE IMPACT OF INSERVICE INSPECTION USING A PIPING RELIABILITY MODEL. SIMONEN,F.A.; WOO,H.H. Battelle Memorial Institute, Pacific Northwest Laboratories. August 1984. 55pp. 8408220370. PNL-5149. 26199:249.

This report presents the results of a study of the impact of inservice (ISI) programs on the reliability of specific nuclear piping systems that have actually failed in service. Two major factors are considered in the ISI programs: one is the capability of detecting flaws; the other is the frequency of performing ISI. A probabilistic fracture mechanics model issued to estimate the reliability of two nuclear piping lines over the plant life as functions of the ISI programs. Examples chosen for the study are the PWR feedwater steam generator nozzle cracking incident and the BWR recirculation reactor vessel nozzle safe-end cracking incident. The results show that an effective inservice inspection requires a suitable combination of flaw detection capability and inspection schedule. An augmented inspection schedule is required for piping with fast-growing flaws to ensure that the inspection is done before the flaws reach critical sizes. Also, the elimination of "poor" inspection teams through training and qualification testing can produce significant benefits to ISI effectiveness.

NUREG/CR-3870: RADIATION DOSE ESTIMATES AND HAZARD EVALUATIONS FOR INHALED AIRBORNE RADIONUCLIDES.Annual Progress Rept July 1982 -June 1983. MEWHINNEY,J.A. Innalation Toxicology Research Institute. July 1984. 38pp. 8408160125. LMF-109. 26122:313.

The objective of this project is to conduct confirmatory research on aerosol characteristics and the resulting radiation dose distribution in animals after inhalation and to provide prediction of health consequences in humans from airborne radioactivity that might be released in normal operations or under accident conditions during production of nuclear fuel composed of mixed oxides of uranium and plutonium. Two research reports summarize the progress of current research. The first paper details results from the completed radiation dose distribution studies in which dogs, monkeys, and rats were exposed to either UO(2) + PuO(2) treated at 750 degrees centigrade, (U, Pu)O(2) treated at 175 degrees centigrade, or PuO(2) treated at 850 degrees centigrade. This paper focuses on analysis of the data from the last animals sacrificed in the study and updates earlier analyses of lung retention, tissue distribution, and excretion. The second paper details preliminary analyses of the lung retention in Fischer-344 rats exposed to either (U, Pu)O(2) or to PuO(2) at one of three levels of projected dose to lung for each aerosol. This paper presents the methods and the application of a rigorous statistical procedure allowing detection of similarities and differences in the lung retention of rats at different dose levels and for different aerosols.

NUREG/CR-3871: AN OVERVIEW OF THE UNIFIED TRANSPORT APPROACH. ERASLAN,A.H.; WITTEN,A.J. Oak Ridge National Laboratory. August 1984. 123pp. 8409200399. ORNL-TM-9249. 26607:128.

The Unified Transport Approach (UTA) consists of a set of nine complementary models developed for assessing the environmental impacts associated with nuclear power plant discharges to receiving water bodies. This set of models has the capability to simulate natural and plant-induced flow, temperature, salinity, sediment transport, radionuclide transport, and chemical species concentrations. While these UTA models were developed for predicting impacts associated with the operation of nuclear power plants, they are quite general and can be applied to a variety of situations. The UTA models have been used to simulate the impacts associated with the operation of many industrial and energy production technologies, as well as to simulate laboratory and naturally occurring conditions. In all cases where data have been available for validation, the UTA model results have compared favorably. The purpose of this report is to provide an overview of the UTA as a whole, highlighting the important features and unique capabilities of this approach.

NUREG/CR-3874: NEAR-GROUND TORNADO WIND FIELDS. MCDONALD,J.R. Texas Tech Univ., Lubbock, TX. July 1984. 164pp. 8408220327. 26198:197.

This report is written as a general treatise on near-ground tornado wind fields. In Section II an engineering perspective on tornadoes is stated. Section III describes the data available for the study of near-ground tornado wind fields. Section IV discusses tornado wind speeds and briefly describes a new method for making more rational estimates of tornado wind speeds from damaged structures. Section V describes the damage indicators that are present in the wake of a tornado event and discusses other factors that affect the appearance of damage. A perspective on tornado-generated missiles is presented in Section VI. Conclusions and recommendations for further study are contained in the last section of the report.

NUREG/CR-3878: MODELING CONSIDERATIONS FOR THE PRIMARY SYSTEM OF THE EXPERIMENTAL BREEDER REACTOR-II. MADNI,I.K. Brookhaven National Laboratory. September 1984. 45pp. 8410120016. BNL-NUREG-51797. 26978:271.

This report describes the additional heat transfer and coolant dynamic models for components and processes, that are needed for simulation of the primary system of the Experimental Breeder Reactor-II (EBR-II). This work forms part of the Super System Code (SSC) application efforts to provide predictions of EBR-II overall plant behavior.

NUREG/CR-3884: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS PROGRAM - THREE MILE ISLAND UNIT 2 POLAR CRANE RECOVERY. DOERGE,D.H.;

MILLER,R.L. United Nuclear Corp. August 1984. 63pp. 8408290185. 26309:267.

This document summarizes information concerning restoration of the Three Mile Island-unit 2 Polar Crane to a fully operational condition following the loss of coolant accident experienced on March 28, 1979.

The data collected from activity reports, reactor containment entry records and other sources were placed in a computerized information retrieval/manipulation system which permits extraction/manipulation of specific data which could be utilized in planning for recovery activities should a similar accident occur in a nuclear generating plant. The information is presented in both computer output form and a manually assembled summarization.

This report contains only manpower requirements and radiation exposures actually incurred during recovery operations within the reactor containment and does not include support activities or costs.

NUREG/CR-3888: ANALYSIS OF THE VENUS PWR ENGINEERING MOCKUP EXPERIMENT -PHASE I: SOURCE DISTRIBUTION. MORAKINYO,P.O.; WILLIAMS,M.L.; KAM,B.K. Oak Ridge National Laboratory. August 1984. 81pp. 8410030356. ORNL/TM-9238. 26833:162.

The neutron fission source distribution in the core of the VENUS PWR Mockup Experiment is computed and compared to experimental measurements. Of particular concern is the accuracy of the source calculation near the core-baffle interface, which is the important region for contributing to RPV fluence.

Results indicate that the calculated neutron source distribution within the VENUS core agrees with the experimentally measured values with an average error of less than 3%. At the important core-baffle interface, the agreement is within 3% error, except at the baffle corner, where the error is about 6%. Better accuracy in the calculations can be obtained by applying a detailed space dependent cross-section weighting procedure to the core-baffle interface region. Using this cross-section weighting, the maximum error introduced into the predicted RPV fluence due to source errors should be on the order of 5%. However, in power reactor analysis, additional complexities (such as the time-dependent core composition and the use of few group diffusion theory) could affect this uncertainty value.

NUREG/CR-3892: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANICAL EQUIPMENT.Summary Report. KANA,D.D. Southwest Research Institute. August 1984. 43pp. 8409070235. 26418:289.

This document constitutes the Summary for the indicated research contract on equipment seismic qualification methodology. Although the program was conducted by Southwest Research Institute, the results were periodically reviewed by a Peer Review Panel of ten members from various segments of the nuclear industry, and by various members of the NRC staff. In addition, a continuing communication with the IEEE 344 (Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations) revision committee was maintained throughout the program to ensure that the results were disseminated to the industry. Thus, although the results are principally the findings of SWRI, acknowledgement of input from various other sources is recognized.

The program has spanned a period of three years and resulted in seven technical summary reports, each of which covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes. This volume is to summarize the entire program from an overall philosophical point of view.

Volume 1 includes Task 1 Summary Reports parts 1, 2, and 3, which describe evaluations of various aspects of equipment qualifications methodology. Volumes 2, 3, and 4 include the summary reports for Tasks 2, 3, and 4, which are concerned with correlation of methodologies, recommendations for improvements, and evaluation of fragility methodology.

NUREG/CR-3892 V01: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANCIAL EQUIPMENT.Task 1 - Survey Of Methods For Equipment And Components:Evaluation Of Methodology;Qualification And Methodology.... KANA,D.D.; POLCH,E.Z.; POMERENING,D.J.; et al. Southwest Research Institute. August 1984. 393pp. 8409070233. 26413:001.

The Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment has spanned a period of three years and resulted in seven technical summary reports, each of which covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes.

Volume 1 comprises three parts. Part I reviews the methods currently utilized for seismic qualification of nuclear plant equipment with emphasis on qualification by testing. In this review various anomalies that are associated with qualification are identified. Part II provides an in-depth evaluation of the technical issues/anomalies previously identified. Part III provides an evaluation of the method applicable to line mounted items; e.g., valves.

NUREG/CR-3892 V02: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANICAL EQUIPMENT.Task 2-Correlation Of Methodologies For Seismic Qualification Tests Of Nuclear Plant Equipment. KANA,D.D.; POMERENING,D.J. Southwest Research Institute. August 1984. 105pp. 8409070269. 26409:160.

The Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment has spanned a period of three years and resulted in seven technical summary reports, each of which covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes.

Volume 2 presents a general method for correlating the severity of one seismic qualification motion of given dynamic characteristics to another motion, possibly of different dynamic characteristics. The method provides a method of measuring relative damage severity of two different motions in terms of a relative damage severity ratio.

NUREG/CR-3892 V03: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANICAL EQUIPMENT.Task 3-Recommendations For Improvement Of Equipment Qualification Methodology And Criteria. KANA,D.D.; POMERENING,D.J. Southwest Research Institute. August 1984. 74pp. 8409070272. 26409:047.

The Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment has spanned a period of three years and resulted in seven technical summary reports, each of which covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes.

Volume 3 presents recommendations for improvement of equipment qualification methodology and criteria. These recommendations are grouped into categories: standardization of procedures, demonstration of adequate methodology, a new methodology and procedural clarification/modification. The fifth category identifies issues where adequate information does not exist to allow a recommendation to be made.

NUREG/CR-3892 V04: A RESEARCH PROGRAM FOR SEISMIC QUALIFICATION OF NUCLEAR PLANT ELECTRICAL AND MECHANICAL EQUIPMENT.Task 4 - The Use Of Fragility In Design Of Nuclear Plant Equipment. KANA,D.D.; POMERENING,D.J. Southwest Research Institute. August 1984. 44PP. 8409070302. 26409:119.

The Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment has spanned a period of three years and resulted in seven technical summary reports, each of which have covered in detail the findings of different tasks and subtasks, and have been combined into five NUREG/CR volumes.

Volume 4 presents a study of the use of fragility concepts in the design of nuclear plant equipment and compares the results of state-of-the-art proof testing with fragility testing.

NUREG/CR-3893: LABORATORY STUDIES:DYNAMIC RESPONSE OF PROTOTYPICAL PIPING SYSTEMS. HOWARD,G.E.; WALTON,W.B.; JOHNSON,B.A. ANCO Engineers, Inc. August 1984. 101pp. 8409070292. 26409:258.

This report presents details of the test methods, specimens and a preliminary assessment of results. Two test configurations will be used to achieve the project objectives. Both were three dimensional configurations; the second configuration had branch pipes. The piping systems sustained no apparent damage after being subjected to an earthquake approximately four times greater than the SSE. Additionally, one of the piping systems resisted five OBE's, nine SSE's and nearly thirty shocks.

NUREG/CR-3894: ULTRASONIC AND METALLURGICAL EXAMINATION OF A CRACKED TYPE 304 STAINLESS STEEL BWR PIPE WELDMENT. PARK,J.Y.; KUPPERMAN,D. Argonne National Laboratory. July 1984. 22pp. 8408240322. ANL-84-1. 26255:169.

An ultrasonic in-service inspection (ISI) indicated that a crack had developed in a 22-inch-diameter Type 304 stainless steel pipe manifold endcap weldment of the Hatch-2 boiling water reactor. A section of the weldment was sent to Argonne National Laboratory (ANL) for further examination. The ANL effort included ultrasonic examinations, destructive crack-depth measurements, metallography, degree of sensitization (DOS) measurements, and chemical analyses of material. The results showed that the extent of the cracking was significantly less than indicated by the ISI.

NUREG/CR-3895: INVESTIGATION OF COLD LEG WATER HAMMER IN A PWR DUE TO THE ADMISSION OF ECC DURING A SMALL BREAK LOCA. JACKOBEK,A.B.; GRIFFITH,P. Massachusetts Institute of Technology, Cambridge, MA. September 1984. 60pp. 8410120001. 26986:141.

Experimental studies using a prototypical flow model of a pressurized water reactor (PWR) demonstrate water hammer in the cold legs due to the admission of emergency core cooling (ECC). Such water hammer can occur in an actual PWR during reflood provided there exists a stratified flow of steam and water in the cold legs. The hydraulic are postulated in this report. Calculations, based on a published criterion for water hammer initiation, show that the amount of ECC administered by the high pressure safety injection (HPSI) system, is not great enough to produce liquid depths in the cold leg which can lead to slug formation and subsequent steam bubble collapse water hammer. However, a few water hammers can occur during ECC as the cold leg is being refilled.

A simple analysis developed in this report calculates the water hammer pressures possible under these postulated flow conditions. Potentially dangerous water hammer pressures are predicted during reflood at high system operating pressures characteristic of a small break loss-of-coolant accident (SB-LOCA). Similar calculations done for the geometry of the experimental apparatus were compared to measurements taken during water hammer.

NUREG/CR-3896: SIMULATION EXPERIMENTS COMPARING ALTERNATIVE PROCESS FORMULATIONS USING A FACTORIAL DESIGN. KALUZNY,S.P.; SWARTZMAN,G.L. Washington, Univ. of, Seattle, WA. July 1984. 29pp. 8408060386. 25937:282.

This paper reviews methods for exploring the differences between alternative equations in complex ecosystem models. A factorial design is proposed as a method for exposing possible interactions between equation forms in their effect on model output as well as to clarify differences between the main candidate equations. A number of display methods arising from statistical analysis are used including normal Q-Q plots, linear rank plots, and interaction diagrams. The methods were illustrated using a complex ecosystem model of Lake Ontario. We found the methods effective at illustrating major differences between equations although several difficulties arose due to the complexity of the models and the diffuse nature of the data supporting model validation. Questions of the method for standardization of equation forms so that the compared equations are in some way analogous are important. These methods are probably most useful in cases where the data are of sufficient quality to indicate not only how different equations effect model output but also which forms are to be preferred.

NUREG/CR-3897: EVALUATION OF ECOSYSTEM SIMULATION MODELS AS TOOLS FOR ASSESSMENT OF POWER PLANT IMPACTS ON FISH POPULATIONS. Final Rept. SWARTZMAN,G.L. Washington, Univ. of, Seattle, WA. July 1984. 10pp. 8408010158. 25867:292.

This two-volume report presents the procedures and analyses in developing an approach for structuring expert judgments to estimate human error probabilities. Volume I presents an overview of work performed in developing the approach: SLIM-MAUD (Success Likelihood Index Methodology, implemented through the use of an interactive computer program called MAUD--Multi-Attribute Utility Decomposition). Volume II provides a more detailed analysis of the technical issues underlying the approach.

NUREG/CR-3899: UTILITY FINANCIAL STABILITY AND THE AVAILABILITY OF FUNDS FOR DECOMMISSIONING. SIEGEL,J.J Engineering & Economics Research, Inc. September 1984. 28pp. 8410030368. 26817:289.

The NRC is currently developing rulemaking in the area of decommissioning nuclear facilities. A part of that rulemaking effort is assuring that funds will be available at the time of decommissioning of power reactors. Previous NRC reports have examined this issue by studying various funding methods. This report provides an update by analyzing the relative level of assurance of funding methods, considering the present utility financial situation. In its analysis the report makes use of specific case situations. The report concludes that the various funding methods studied in the earlier reports including the internal reserve method provide assurance of the availability of funds for decommissioning.

NUREG/CR-3900 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Three,April-June 1984. STAHL,D.; MILLER,N.E. Battelle Memorial Institute, Columbus Laboratories. September 1984. 111pp. 8410120024. 26984:346.

Devitrification severity of glass waste forms is being studied in terms of volume fraction of crystallization and crystal grain size. Glass-water contact during the heating and cooling periods of glass leaching experiments is being evaluated for its effect on the overall results of the isothermal period. Modeling efforts included the study of possible colloid formation and the change of water chemistry during glass dissolution. The electrochemical properties of container steels were found to be only slightly affected by the groundwater-species concentration, the presence of basalt rock, or the steels' cleanliness or microstructure. Hydrogen-embrittlement susceptibility may increase at expected repository temperatures. Results of the corrosion-modeling effort suggest that radiolysis may significantly affect general-corrosion kinetics. The water-radiolysis model was extended to account for more groundwater species and was used to predict the concentrations of two species in aqueous iron sulfate; results were compared with experimental data. A method was selected for performing uncertainty analyses of waste-package models. Integral experiments have been designed to address the combined effects of repository conditions on the waste package.

NUREG/CR-3905: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSE EVENT REPORTS. Users Guide. GREENE,N.M.; MAYS,G.T. Oak Ridge National Laboratory. JOHNSON,M.P. JBF Associates. August 1984. 160pp. 8409270117. ORNL/NSIC-223. 26716:177.

The Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data has developed, through the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory (ORNL), a system to aid in the evaluation of the Licensee Event Reports (LERs) submitted by the nuclear power plant utilities. The primary objective of the Sequence Coding and Search System (SCSS) is to reduce the descriptive text of the incident reports to a coded sequence that is both computer-readable and computer-searchable. This system provides a structured format for detailed coding of component, system, and unit effects, as well as personnel errors. The database contains all current LERs submitted by the nuclear power plant utilities after January 1, 1981, and is updated on a continual basis with new LERs, as they are submitted. The database is maintained by NOAC on the IBM-3033 computer system at ORNL. Following a description of SCSS and structure of the database, a tutorial section is provided to acquaint the first-time user with logon procedures and the necessary commands to retrieve, display, and analyze LERs. Each command is subsequently discussed in detail in the fundamental and advanced command sections.

NUREG/CR-3907: GT2R2:AN UPDATED VERSION OF GAPCON-THERMAL-2. CUNNINGHAM,M.E.; BEYER,C.E. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1984. 70pp. 8410100121. PNL-5178. 26902:223.

The GAPCON-THERMAL-2 code is used by the U.S. Nuclear Regulatory Commission for audit calculations of nuclear fuel thermal performance computer codes. Since the code was originally written, errors and needed updates have been identified. Revision 2 of GAPCON-THERMAL-2 contains a number of coding corrections and updates, and now conforms with the American National Standards Institute FORTRAN-77 standard. The changes to the code are presented in detail. Benchmarking calculations, concentrating on fuel temperatures and fission gas release, were performed to qualify the effect of model changes on the performance of GAPCON-THERMAL-2, Revision 2. It was concluded that use of the old fuel relocation model combined with the modified ANS 5.4 fission gas release model provides the best overall comparison with the thermal performance and fission gas release data used for the benchmarking exercise. The use of the new fuel relocation model combined with the Beyer-Hann fission gas release model provided the best comparisons of thermal behavior but significantly underpredicted fission gas release.

NUREG/CR-3921: DRY SPENT FUEL STORAGE TEST PLAN FOR FINAL NONDESTRUCTIVE FUEL ROD EXAMINATION. OLSEN,C.S. EG&G Inc. July 1984. 14pp. 8409180283. EGG-2328. 26589:120.

A test plan for the third and final nondestructive examination of eight fuel rods used in a low-temperature, long-term, dry fuel storage program is presented. This examination is part of a long-range project to evaluate the behavior of spent fuel during dry storage conditions. The objective of this project is to provide the Nuclear Regulatory Commission with the information to confirm or establish spent fuel dry storage licensing positions for long-term, low-temperature (<523 K), spent fuel rod behavior during dry storage and for radioactive contamination arising from spallation of cladding crud. This examination consists of visual and photographic examinations, dimensional measurements, and gamma scanning of eight fuel rods.

NUREG/CR-3929: LOSS-OF-BENEFITS ANALYSIS FOR NUCLEAR POWER PLANT SHUTDOWNS. Methodology And Illustrative Case Study. PEERENBOOM,J.P.; BUEHRING,W.A.; GUZIEL,K.A. Argonne National Laboratory. September 1984. 73pp. 8409270130. ANL/AA-29. 26719:287.

A framework for loss-of-benefits analysis and a taxonomy for identifying and categorizing the effects of nuclear power plant shutdowns or accidents are presented. The framework consists of three fundamental steps: (1) characterizing the shutdown; (2) identifying benefits cost as a result of the shutdown; and (3) quantifying effects. A decision analysis approach to regulatory decision making is presented that explicitly considers the loss of benefits. A case study of a hypothetical reactor shutdown illustrates one key loss of benefits: net replacement energy costs (i.e., change in production costs). Sensitivity studies investigate the responsiveness of case study results to changes in nuclear capacity factor, load growth, fuel price escalation, and discount rate. The effects of multiple reactor shutdowns on production costs are also described.

NUREG/CR-3932: BENCHMARK DESCRIPTION OF CURRENT REGULATORY REQUIREMENTS AND PRACTICES IN NUCLEAR SAFETY AND RELIABILITY ASSURANCE. HALVERSON,S.L.; BEZELLA,W.A.; CHARAK,I.; et al. Argonne National Laboratory. August 1984. 115pp. 8410030347. ANL-84-34. 26819:097.

The objectives of this work are to evaluate and benchmark the current safety and reliability assurance-related practices employed by the NRC. This effort represents an initial phase of a program whose overall purpose is to develop a reliability program (RP). A review of NRC regulations relevant to reliability assurance was made for a boiling water reactor using two representative safety systems; the reactor protection system, and the residual heat removal system. The primary sources of information were the Standard Review Plan and Title 10 of the Code of Federal Regulations, especially Part 50. In addition, relevant regulatory guides, NRC branch technical positions and industry consensus standard were identified and catalogued for the two reference safety systems over the plant's life cycle. The identified standards and criteria were then organized into a RP element matrix of current regulatory requirements organized by life cycle phase, top level assurance function, and items directly auditable by the NRC. A brief review of the licensing process was also undertaken to indicate the effectiveness of NRC implementation of a RP. The results of this work showed that within the NRC regulations a framework already exists in which to integrate, not add, a reliability assurance program.

NUREG/CR-3933: RISK RELATED RELIABILITY REQUIREMENTS FOR BWR SAFETY -IMPORTANT SYSTEMS WITH EMPHASIS ON THE RESIDUAL HEAT REMOVAL SYSTEM. TZANOS,C.P.; BEZELLA,W.A. Argonne National Laboratory. August 1984. 140pp. 8410030385. ANL-84-52. 26819:218.

The objective of this study was to identify and evaluate the major safety risk parameters of typical reactor safety systems for use in developing a reliability program. This effort was part of a larger research project aiming to evaluate the feasibility and effectiveness of introducing elements of proven reliability programs from other high technology industries into the nuclear industry. As a reference safety system, the Residual Heat Removal (RHR) system of a Boiling Water Reactor (BWR) was selected. A scoping evaluation was also made for BWR reactor protection system (RPS). Plant information, existing PRA and other relevant analyses, as well as Licensee Event Reports were used as base material for this study. The results of this evaluation indicate that: (1) recovery of faults can have a very significant impact on the reliability requirements, (2) there exists an obvious need for an adequate reliability data base, (3) reliability analyses must be supported by detailed analyses of the plant's response to accident sequences, and (4) the development of effective emergency operating instructions and proper operator training must be one of the major elements of a Reliability Program.

NUREG/CR-3939: WATER HAMMER,FLOW INDUCED VIBRATION AND SAFETY/RELIEF VALVE LOADS. UFFER,R.A.; VALANDANI,P.; SEXTON,D. Quadrex Corp. September 1984. 86pp. 8410120003. EGG-2340. 26985:096.

This report presents the results of an evaluation performed to determine current and recommended practices regarding the consideration of water hammer flow-induced vibration and safety-relief valve loads in the design of nuclear power plant piping systems. Current practices were determined by a survey of industry experts. Recommended practices were determined by evaluating factors such as load magnitude and frequency content, system susceptibility to loads, frequency of load occurrence and safety effects of postulated piping damage.

This report was prepared for use by the NRC staff in developing positions regarding consideration of dynamic piping loads for use by the NRC's Piping Review Committee.

NUREG/CR-3940: FIELD EXPERIMENT DETERMINATIONS OF DISTRIBUTION COEFFICIENTS OF ACTINIDE ELEMENTS IN ALKALINE LAKE ENVIRONMENTS. SIMPSON,H.J.; TRIER,R.M.; LI,Y.H.; et al. Columbia Univ., New York, NY. August 1984. 124pp. 8409260650. 26702:037.

Measurements of the radioisotope concentrations of a number of elements (Am, Pu, U, Pa, Th, Ac, Ra, Pb, Cs, and Sr) in the water and sediments of a group of alkeline (pH = 9-10), saline lakes demonstrate greatly enhanced soluble-phase concentrations of elements with oxidation states of (III)-(VI) as the result of complexing by carbonate ion. Ratios of soluble radionuclide concentrations in Mono Lake to those in seawater ([CO3 (2-)] in Mono Lake = 200 times that of seawater) were: Pu(=10), (238)U(=150), (231)Pa, (238)Th, (230)Th(=10(3), and (232)Th(=10(5). Effective distribution coefficients of these radionuclides in high CO(3)(2-) environments are several orders of magnitude lower (i.e., less particle reactive) than in most other natural waters. The importance of CO(3)(2-) ion on effective K(d) values was also strongly suggested by laboratory experiments in which most of the dissolved actinide elements became adsorbed to particles after a water sample normally at a pH of 10 was acidified, stripped of all CO(2) and then returned to pH 10 by adding NH(4)OH. Furthermore, the effect complexation by organic ligands is of secondary importance in the presence of appreciable carbonate ion concentration.

Neither pure phase solubility calculations nor laboratory scale K(d) determinations accurately predicted the measured natural system concentrations. Therefore, measurements of the distribution of radionuclides in natural systems are essential for assessment of the likely fate of potential releases from high level waste repositories to groundwater.

NUREG/CR-3951: INTRODUCTION TO BIBELOT:A BIBLIOGRAPHIC FINDING AND RETRIEVAL SYSTEM. COCHRAN,M.I. Battelle Memorial Institute, Pacific Northwest Laboratories. September 1984. 46pp. 8410100776. PNL-5202. 26903:260.

The BIBELOT System of COBOL and Datatrieve programs for bibliographic storage and retrieval is described. The storage scheme is also briefly described. The use of unique citation numbers and user defined keywords is illustrated by many retrieval examples. Finally, typical questions about the use of BIBELOT are answered.

NUREG/CR-3988: MARCH 2 (MELTDOWN ACCIDENT RESPONSE CHARACTERISTICS) CODE DESCRIPTION AND USERS MANUAL. WOOTEN,R.O.; CYBULSKIS,P.; QUAYLE,S.F. Battelle Memorial Institute, Columbus Laboratories. September 1984. 400pp. 8410170214. BMI-2115. 27030:001.

MARCH 2 describes the response of water-cooled reactor systems to severe accidents, particularly those leading to core meltdown. The code performs the calculations from the time of accident initiation through the stages of coolant blowdown and boiloff, core heat up and meltdown, pressure vessel bottom head melting and failure, and debris-water and debris-concrete interactions in the reactor cavity.

Both the primary system and the building are modeled. Mass and energy additions to the containment building are evaluated and the pressure-temperature response of the containment with or without engineered safety features is calculated. A maximum of eight containment sub-volumes may be modeled. Engineered safety features modeled include emergency core cooling systems, containment sprays, building coolers and fans, suppression pool and ice condenser containments, and emergency core cooling and spray heat exchangers. Effects of metal-water reactions, combustion of hydrogen and carbon monoxide, heat losses to containment structures, and redistribution of the decay heat due to loss of volatile fission products from the core are considered. MARCH 2 is intended to replace the earlier MARCH 1 code. It is written in FORTRAN 77 to improve transportability.

NUREG/CR-4001: CONTEMPT4/MOD5:AN IMPROVEMENT TO CONTEMPT4/MOD4 MULTICOMPARTMENT CONTAINMENT SYSTEM ANALYSIS PROGRAM FOR ICE CONTAINMENT ANALYSIS. LIN,C.C. Brookhaven National Laboratory. September 1984. 40pp. 8410180123. BNL-NUREG-51824. 27045:289.

CONTEMPT4 is a digital computer program for multicompartment containment system analysis. Previous version of the CONTEMPT4 code, MOD4, consists of an implicit algorithm to computer junction flow when numerically induced flow oscillations are encountered. This document presents analytical model and UPDATE statements that are required to extend the capability of the MOD4 implicit routine for ice containment analysis. A sample problem is analyzed both with and without the use of the implicit routine to demonstrate the effectiveness and the need of an implicit algoritnm for such problems.

NUREG/CR-4007: LOWER LIMIT OF DETECTION:DEFINITION AND ELABORATION OF A PROPOSED POSITION FOR RADIOLOGICAL EFFLUENT AND ENVIRONMENTAL MEASUREMENTS. CURRIE,L.A. Commerce, Dept. of, National Bureau of Standards. September 1984. 153pp. 8410170308. 27031:060.

A manual is provided to define and illustrate a proposed use of the Lower Limit of Detection (LLD) for Radiological Effluent and Environmental Measurements. The manual contains a review of information regarding LLD practices gained from site visits; a review of the literature and a summary of basic principles underlying the concept of detection in Nuclear and Analytical Chemistry; a detailed presentation of the application of LLD principles to a range of problem categories (simple counting to multinuclide spectroscopy), including derivations, equations, and numerical examples; and a brief examination of related issues such as reference samples, numerical quality control, and instrumental limitations. An appendix contains a summary of notation and terminology, a bibliography, and worked-out examples.

NUREG/CR-4011: THE 21/55 DATA BASE USER'S MANUAL. SILVER,E.G. Oak Ridge National Laboratory. September 1984. 317pp. 8410120007. ORNL/NSIC-221. 26982:001.

The Nuclear Regulatory Commission's Office for the Analysis and Evaluation of Operational Data has developed, through the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory (ORNL), a data base for storing and organizing information obtained from the reports on construction deficiencies (CDRs) submitted to NRC under the requirements of 10 CFR 21 and 10 CFR 50.55(e) by holders of construction permits for nuclear facilities. The computerized data base stores coded and textual information about the reports issued and the events to which they refer, including such data as dates of events and reports, affected systems and components, source of information, manufacturers and vendors of affected components and the like. There is also provision for direct access to the data base by NRC Headquarters and Field Office staff both for accessing the information in the data base, and for entry of specific data concerning assignments of NRC follow-up staff and resolution actions taken. The document includes a tutorial guide for novice users of the data base. A system of access control to assure the integrity of the NRC-input data was developed and is described.

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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Process Equation Comparison

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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TRAC-PF1/MOD1
NUREG/CR-3820 V01: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,January-March 1984.

Task Action Plans
NUREG-0649 R01: TASK ACTION PLANS FOR UNRESOLVED SAFETY ISSUES RELATED TO NUCLEAR POWER PLANTS.

Technical Specification
NUREG-1072: TECHNICAL SPECIFICATIONS FOR CATAWBA NUCLEAR STATION,Unit 1. Docket No. 50-413.

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NUREG/CR-3815: STATISTICAL EVALUATION OF THE METALLURGICAL TEST DATA IN THE ORR-PSF-PVS IRRADIATION EXPERIMENT.
NUREG/CR-3856: AN ULTRASONIC LEVEL AND TEMPERATURE SENSOR FOR POWER REACTOR APPLICATIONS.
NUREG/CR-3856: AN ULTRASONIC LEVEL AND TEMPERATURE SENSOR FOR POWER REACTOR APPLICATIONS.

Terminal Block
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Thermal-Hydraulic
NUREG/CR-2331 V03 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July-September 1983.
NUREG/CR-2331 V03 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,October 1 -December 31,1983.
NUREG/CR-3761: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR THE OCONEE-1 PRESSURIZED WATER REACTOR.
NUREG/CR-3765: MINET SIMULATION OF A HELICAL COIL SODIUM/WATER STEAM GENERATOR,INCLUDING STRUCTURAL EFFECTS.
NUREG/CR-3820 V01: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,January-March 1984.
NUREG/CR-3878: MODELING CONSIDERATIONS FOR THE PRIMARY SYSTEM OF THE EXPERIMENTAL BREEDER REACTOR-II.
Title List
NUREG-0540 V06 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1984.
NUREG-0540 V06 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE July 1-31, 1984.
Training
NUREG/CR-3520 V01: LONG-TERM RESEARCH PLAN FOR HUMAN FACTORS AFFECTING SAFEGUARDS AT NUCLEAR POWER PLANTS.Volume I:Summary And Users Guide.
NUREG/CR-3520 V02: LONG-TERM RESEARCH PLAN FOR HUMAN FACTORS AFFECTING SAFEGUARDS AT NUCLEAR POWER PLANTS.Volume II:Development Of Detailed Analyses.
NUREG/CR-3739: THE OPERATOR FEEDBACK WORKSHOP:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.
Transient
NUREG/CR-3888: ANALYSIS OF THE VENUS PWR ENGINEERING MOCKUP EXPERIMENT -PHASE I: SOURCE DISTRIBUTION.
Tube Plugging
NUREG/CR-3842: STEAM GENERATOR GROUP PROJECT TASK 8 - SELECTIVE TUBE UNPLUGGING.
Tube Rupture
NUREG/CR-3665 V03: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS.A Calculation Method.
Tubes
NUREG/CR-3459: EXPERIMENT DATA REPORT FOR MULTIROD BURST TEST (MRBT) BUNDLE B-5.
NUREG/CR-3460: EXPERIMENT DATA REPORT FOR MULTIROD BURST TEST (MRBT) BUNDLE B-6.
Type 304 Stainless Steel Pipe
NUREG/CR-3894: ULTRASONIC AND METALLURGICAL EXAMINATION OF A CRACKED TYPE 304 STAINLESS STEEL BWR PIPE WELDMENT.
U-Bends
NUREG/CR-3834: ON THE THRESHOLD SULFUR AND LITHIUM TO SULFUR RATIO IN STRESS CORROSION CRACKING OF SENSITIZED ALLOY 600 IN BORATED THIOSULFATE SOLUTION.
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Uncertainty Analysis
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Uranium Tailings
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Urinalyses
NUREG/CR-3346: BIOASSAY DATA AND A RETENTION-EXCRETION MODEL FOR SYSTEMIC PLUTONIUM.
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NUREG/CR-4011: THE 21/55 DATA BASE USER'S MANUAL.
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NUREG/CR-3888: ANALYSIS OF THE VENUS PWR ENGINEERING MOCKUP EXPERIMENT -PHASE I: SOURCE DISTRIBUTION.
VGES
NUREG/CR-3273: COMBUSTION OF HYDROGEN:AIR MIXTURES IN THE VGES CYLINDRICAL TANK.
Value/Impact
NUREG/CR-3665: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS.Executive Summary.
NUREG/CR-3665 V01: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS.A Review Of Occupational Dose Assessment Considerations In Current Probabilistic Risk Assessment And Cost-Benefit Analyses.
NUREG/CR-3665 V02: OPTIMIZATION OF PUBLIC AND OCCUPATIONAL RADIATION PROTECTION AT NUCLEAR POWER PLANTS.Considerations In Factoring Occupational Dose Into Value-Impact And Cost-Benefit Analyses.
Variable Geometry Experimental System
NUREG/CR-3273: COMBUSTION OF HYDROGEN:AIR MIXTURES IN THE VGES CYLINDRICAL TANK.
Vibration
NUREG/CR-3939: WATER HAMMER,FLOW INDUCED VIBRATION AND SAFETY/RELIEF VALVE LOADS.
Waste Package
NUREG/CR-2482 V05: REVIEW OF DOE WASTE PACKAGE PROGRAM.Subtask 1.1 - National Waste Package Program,April 1983 - September 1983.
NUREG/CR-3900 V01: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.First Quarterly Report,Year Three,April-June 1984.
Waste Repositories
NUREG/CR-3758: CROSSHOLE GEOPHYSICAL METHODS USED TO INVESTIGATE THE NEAR VICINITY OF HIGH LEVEL WASTE REPOSITORIES.
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NUREG/CR-3689 V01: MATERIAL SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM:Quarterly Progress Report, January-March 1983.
Water Hammer
NUREG/CR-3895: INVESTIGATION OF COLD LEG WATER HAMMER IN A PWR DUE TO THE ADMISSION OF ECC DURING A SMALL BREAK LOCA.
NUREG/CR-3939: WATER HAMMER,FLOW INDUCED VIBRATION AND SAFETY/RELIEF VALVE LOADS.
Weldment
NUREG/CR-3788 V01: STRUCTURAL INTEGRITY OF LIGHT WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Four-Year Plan 1984-1988.
NUREG/CR-3894: ULTRASONIC AND METALLURGICAL EXAMINATION OF A CRACKED TYPE 304 STAINLESS STEEL BWR PIPE WELDMENT.
Whole Rod Testing
NUREG/CR-3708: LWR SPENT FUEL DRY STORAGE BEHAVIOR AT 229 C.
Workshop
NUREG/CR-3739: THE OPERATOR FEEDBACK WORKSHOP:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.
Zircaloy
NUREG/CR-3459: EXPERIMENT DATA REPORT FOR MULTIROD BURST TEST (MRBT)

BUNDLE B-5.
NUREG/CR-3460: EXPERIMENT DATA REPORT FOR MULTIROD BURST TEST (MRBT)
BUNDLE B-6.

NRC Originating Organization Index (Staff Reports)

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

EDO - OFFICE OF ADMINISTRATION

DIVISION OF TECHNICAL INFORMATION & DOCUMENT CONTROL
NUREG-0304 V09 N02: REGULATORY AND TECHNICAL REPORTS.Compilation For Second Quarter 1984.
NUREG-0540 V06 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1984.
NUREG-0540 V06 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.JUNE 1-30, 1984.
NUREG-0540 V06 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE July 1-31, 1984.
NUREG-0750 V19 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY-MARCH 1984.
NUREG-0750 V19 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1984. Pages 555-936.
NUREG-0750 V19 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1984.Pages 937-1,149.
NUREG-0750 V19 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1984.Pages 1,151-1,321.

DIVISION OF RULES AND RECORDS
NUREG-0936 V03 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1984.

EDO - OFFICE OF EXECUTIVE LEGAL DIRECTOR

OFFICE OF THE EXECUTIVE LEGAL DIRECTOR
NUREG-0386 D03: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

DIRECTOR'S OFFICE
NUREG-0090 V07 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1984.

OFFICE OF INSPECTION & ENFORCEMENT (POST 12/11/80)

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DIRECTOR'S OFFICE, OFFICE OF INSPECTION AND ENFORCEMENT
NUREG-0430 V04 N02: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data,July 1983-December 1983.(Buff book)

ENFORCEMENT STAFF
NUREG-0940 V03 N02: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,April-June 1984.

DIVISION OF QA, SAFEGUARDS & INSPECTION PROGRAMS (POST 830103)
NUREG-0040 V08 N02: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1984.(white book).

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

OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR
NUREG-1080 V01: LONG-RANGE RESEARCH PLAN FY 1985-1989.

DIVISION OF RISK ANALYSIS & OPERATIONS (POST 840429)
NUREG-0935: ACOUSTIC WAVE PROPAGATION IN FLUIDS WITH COUPLED CHEMICAL REACTIONS.
NUREG-1050: PROBABILISTIC RISK ASSESSMENT (PRA) REFERENCE DOCUMENT.Final Rept.

DIVISION OF RADIATION PROGRAMS & EARTH SCIENCES (POST 840429)
NUREG-1029: A COMPUTER CODE FOR GENERAL ANALYSIS OF RADON RISKS (GARR).

DIVISION OF ENGINEERING TECHNOLOGY
NUREG-1092: ENVIRONMENTAL ASSESSMENT FOR 10 CFR PART 72, "LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT FUEL AND HIGH-LEVEL RADIOACTIVE WASTE."
NUREG/CP-0051: PROCEEDINGS OF THE CSNI SPECIALIST MEETING ON LEAK-BEFORE-BREAK IN NUCLEAR REACTOR PIPING.

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INTRA-AGENCY COMMITTEES, REVIEW GROUPS, ETC.

PIPING REVIEW COMMITTEE
NUREG-1061 V01: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 1:Investigation And Evaluation Of Stress Corrosion Cracking In Piping Of Boiling Water Reactor Plants.

EDO-RESOURCE MANAGEMENT

OFFICE OF RESOURCE MANAGEMENT, DIRECTOR
NUREG-0748 V04 N05: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of May 31,1984.(Orange Book)
NUREG-0748 V04 N06: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of June 30,1984. (Orange Book)
NUREG-0748 V04 N07: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of July 31,1984.(Orange Book)

DIVISION OF BUDGET & ANALYSIS
NUREG-0020 V08 N06: LICENSED OPERATING REACTORS STATUS SUMMARY REPT,Data As Of May 31,1984.(Grey Book)
NUREG-0020 V08 N07: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT,Data As Of June 30,1984.(Grey Book)

NUREG-0020 V08 N08: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Date As Of July 31,1984.(Grey Book)

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

OFFICE OF NUCLEAR REACTOR REGULATION, DIRECTOR

NUREG-0787 S07: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATERFORD NUCLEAR POWER PLANT,UNIT 3.Docket No. 50-382. (Louisiana Power & Light Company)

NUREG-0800 06.2.1 R6: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS,LWR Edition Revision 6 to Section 6.2.1.1.C, "Pressure-Suppression Type BWR Containments."

DIVISION OF ENGINEERING

NUREG-1054: SIMPLIFIED ANALYSIS FOR LIQUID PATHWAY STUDIES.

NUREG-1081: POST-ACCIDENT GAS GENERATION FROM RADIALYSIS OF ORGANIC MATERIALS.

DIVISION OF HUMAN FACTORS SAFETY

NUREG-0985 R01: U.S. NUCLEAR REGULATORY COMMISSION HUMAN FACTORS PROGRAM PLAN.

DIVISION OF SYSTEMS INTEGRATION (POST 811005)

NUREG-0978: MARK III LOCA-RELATED HYDRODYNAMIC LOAD DEFINITION.Generic Technical Activity B-10.Final Report.

DIVISION OF LICENSING

NUREG-0420 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION,UNIT NO. 1.Docket No. 50-322. (Long Island Lighting Company)

NUREG-0420 S07: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHOREHAM NUCLEAR POWER STATION UNIT NO.1.Docket No. 50-322. (Long Island Lighting Company)

NUREG-0675 S24: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 & 2.Docket Nos. 50-275 & 50-323.(Pacific Gas & Electric Company)

NUREG-0675 S25: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company)

NUREG-0675 S26: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.

NUREG-0675 S27: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-275 And 50-323.(Pacific Gas And Electric Company)

NUREG-0680 S05: TMI-1 RESTART.An Evaluation Of The Licensee's Management Integrity As It Affects Restart Of Three Mile Island Unit 1 Docket 50-289.

NUREG-0798 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF ENRICO FERMI ATOMIC POWER PLANT,UNIT NO. 2.Docket No. 50-341. (Detroit Edison Company)

NUREG-0831 S05: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION,UNITS 1 AND 2.Docket Nos. 50-416 And 50-417.(Mississippi Power And Light Company,Middle South Energy,Inc And South Mississippi Electric Power Association)

NUREG-0831 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF GRAND GULF NUCLEAR STATION,UNITS 1 AND 2.Docket Nos. 50-416 And 50-417. (Mississippi Power And Light Company)

NUREG-0926 R01: TECHNICAL SPECIFICATIONS FOR GRAND GULF NUCLEAR STATION, UNIT 1.Docket No. 50-416. (Mississippi Power And Light Company)

NUREG-0954 S03: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF

CATAWBA NUCLEAR STATION,UNITS 1 AND 2,DOCKET Nos. 50-413 And 50-414.(Duke Power Company,et al)

NUREG-1031: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION,UNIT NO. 3.Docket No. 50-423.(Northeast Nuclear Energy Company)

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

NUREG-1069: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE GENERAL ELECTRIC-NUCLEAR TEST REACTOR(GE-NTR),DOCKET NO. 50-73.(General Electric Company)

NUREG-1072: TECHNICAL SPECIFICATIONS FOR CATAWBA NUCLEAR STATION,Unit 1. Docket No. 50-413.

NUREG-1073: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF RIVER BEND STATION.Docket No. 50-458.(Gulf States Utilities Company & Cajun Electric Power Cooperative)

NUREG-1075: DECENTRALIZATION OF OPERATING REACTOR LICENSING REVIEWS,NRR Pilot Program.

NUREG-1083: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE WESTINGHOUSE RESEARCH REACTOR AT ZION,ILLINOIS.DOCKET NO. 50-87.(Westinghouse Electric Company)

NUREG-1084: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT MICHIGAN STATE UNIVERSITY.Docket No. 50-294.

NUREG-1085: DRAFT ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO.2.Docket No. 50-410. (Niagara Mohawk Power Corporation,Rochester Gas & Electric Corporation And Central Hudson Gas & Electric Corporation)

DIVISION OF SAFETY TECHNOLOGY

NUREG-0606 V06 N03: UNRESOLVED SAFETY ISSUES SUMMARY.Data As Of August 17,1984. (Aqua Book)

NUREG-0649 R01: TASK ACTION PLANS FOR UNRESOLVED SAFETY ISSUES RELATED TO NUCLEAR POWER PLANTS.

NUREG-0933 S01: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

NUREG-1068: REVIEW INSIGHTS ON THE PROBABILISTIC RISK ASSESSMENTS FOR THE LIMERICK GENERATING STATION,UNIT 1 AND 2.

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 N6: LICENSEE EVENT REPORT (LER) COMPILATION:For Month of June 1984.
NUREG/CR-2000 V03 N7: LICENSEE EVENT REPORT (LER):Compilation For Month of July 1984.
NUREG/CR-2000 V03 N8: LICENSEE EVENT REPORT (LER):Compilation For Month Of August 1984.
NUREG/CR-3824: CONTING PROGRAM GUIDE.
NUREG/CR-3905: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSE EVENT REPORTS, Users Guide.
NUREG/CR-4011: THE 21/55 DATA BASE USER'S MANUAL.

OFFICE OF INSPECTION & ENFORCEMENT (POST 12/11/80)

DIVISION OF EMERGENCY PREPAREDNESS & ENGINEERING RESPONSE (POST 830103)
NUREG/CR-3053: CLOSEOUT OF IE BULLETIN 80-08:EXAMINATION OF CONTAINMENT LINER PENETRATION WELDS.
NUREG/CR-3792: CLOSEOUT OF IE BULLETIN 79-11:FAULTY OVERCURRENT TRIP DEVICE IN CIRCUIT BREAKERS FOR ENGINEERED SAFETY SYSTEMS.
NUREG/CR-3795: CLOSEOUT OF IE BULLETIN 82-04:DEFICIENCIES IN PRIMARY CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES.

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS

DIVISION OF FUEL CYCLE & MATERIAL SAFETY
NUREG/CR-2499: REVIEW OF EMERGENCY RADIOLOGICAL INSTRUMENTATION AND ANALYTICAL METHODS AT NMSS-LICENSEE SITES.

URANIUM FUEL LICENSING BRANCH
NUREG/CR-3796: EMERGENCY PREPAREDNESS SOURCE TERM DEVELOPMENT FOR THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS LICENSED FACILITIES.

DIVISION OF WASTE MANAGEMENT
NUREG/CR-2482 V05: REVIEW OF DOE WASTE PACKAGE PROGRAM,Subtask 1.1 - National Waste Package Program,April 1983 - September 1983.

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NUREG/CR-3763: REVIEW AND ASSESSMENT OF RADIONUCLIDE SORPTION INFORMATION FOR THE BASALT WASTE ISOLATION PROJECT SITE (1979 Through May,1983).
NUREG/CR-3844: CHARACTERIZATION OF THE RADIOACTIVE WASTE PACKAGES OF THE MINNESOTA MINING AND MANUFACTURING COMPANY.
NUREG/CR-3851 V01: PROGRESS IN EVALUATION OF RADIONUCLIDE GEOCHEMICAL INFORMATION DEVELOPED BY DOE HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE PROJECTS.Report for October-December 1983.

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

OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR
NUREG/CR-3469 V01: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.
NUREG/CR-3689 V01: MATERIAL SCIENCE AND TECHNOLOGY DIVISION LIGHT-WATER-REACTOR SAFETY RESEARCH PROGRAM:Quarterly Progress Report, January-March 1983.

ACCIDENT SOURCE TERM PROGRAM OFFICE
NUREG/CR-3929: LOSS-OF-BENEFITS ANALYSIS FOR NUCLEAR POWER PLANT SHUTDOWNS. Methodology And Illustrative Case Study.

DIVISION OF ACCIDENT EVALUATION (PRE 840101)
NUREG/CR-3821: EVALUATION OF CRACK PLANE EQUILIBRIUM MODEL FOR PREDICTING PLASTIC FRACTURE.

DIVISION OF ACCIDENT EVALUATION
NUREG/CR-0169 V17: LOFT EXPERIMENTAL MEASUREMENTS UNCERTAINTY ANALYSIS.Volume XVII Process Instruments Recorded On DAVDS.
NUREG/CR-2331 V03 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July-September 1983.
NUREG/CR-2576: BWR FULL INTEGRAL SIMULATION TEST (FIST)--Facility Description Report.
NUREG/CR-3169: SUPER SYSTEM CODE (SSC,REV. 0).AN ADVANCED THERMOHYDRAULIC SIMULATION FOR TRANSIENTS IN LMFBRS.
NUREG/CR-3190: PLUGM: A COUPLED THERMAL-HYDRAULIC COMPUTER MODEL FOR FREEZING MELT FLOW IN A CHANNEL.
NUREG/CR-3273: COMBUSTION OF HYDROGEN:AIR MIXTURES IN THE VGES CYLINDRICAL TANK.
NUREG/CR-3369: AN UNCERTAINTY STUDY OF PWR STEAM EXPLOSIONS.
NUREG/CR-3459: EXPERIMENT DATA REPORT FOR MULTIROD BURST TEST (MRBT) BUNDLE B-5.
NUREG/CR-3460: EXPERIMENT DATA REPORT FOR MULTIROD BURST TEST (MRBT) BUNDLE B-6.
NUREG/CR-3470: ATWS AT BROWNS FERRY UNIT ONE - ACCIDENT SEQUENCE ANALYSIS.
NUREG/CR-3492 V04: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, October-December 1983.
NUREG/CR-3589 V01: REACTOR SAFETY RESEARCH QUARTERLY REPORT.January-March 1983.
NUREG/CR-3589 V02: REACTOR SAFETY RESEARCH QUARTERLY REPORT.April-June 1983.
NUREG/CR-3617: NOBLE GAS,IODINE,AND CESIUM TRANSPORT IN A POSTULATED LOSS OF DECAY HEAT REMOVAL ACCIDENT AT BROWNS FERRY.
NUREG/CR-3654: PWR FLECHT SEASET SYSTEMS EFFECTS NATURAL CIRCULATION AND REFLUX CONDENSATION.Data Evaluation and Analysis Report NRC/EPRI/Westinghouse Report No. 14.
NUREG/CR-3662: FUEL-DISRUPTION EXPERIMENTS UNDER HIGH-RAMP-RATE

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HEATING CONDITIONS.

NUREG/CR-3672: CALIBRATION AND QUALIFICATION OF THE LOS ALAMOS FAILURE MODEL (LAFM).

NUREG/CR-3690: RELAP5 ASSESSMENT:SEMISCALE NATURAL CIRCULATION TESTS S-NC-3,S-NC-4,AND S-NC-8.

NUREG/CR-3711: BWR FULL INTEGRAL SIMULATION TEST (FIST) PHASE I TEST RESULTS.

NUREG/CR-3724: ULTIMATE STRENGTH ANALYSES OF THE WATTS BAR,MAINE YANKEE,AND BELLEFONTE CONTAINMENTS.

NUREG/CR-3734: LIGHT WATER REACTOR SAFETY RESEARCH PROGRAM.Semiannual Report,October 1982 - March 1983.

NUREG/CR-3761: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR THE OCONEE-1 PRESSURIZED WATER REACTOR.

NUREG/CR-3765: MINET SIMULATION OF A HELICAL COIL SODIUM/WATER STEAM GENERATOR,INCLUDING STRUCTURAL EFFECTS.

NUREG/CR-3776: TESTING OF SAFETY-RELATED NUCLEAR POWER PLANT EQUIPMENT AT THE CENTRAL RECEIVER TEST FACILITY.

NUREG/CR-3804 V01: PHYSICS OF REACTOR SAFETY.Quarterly Report January - March 1984.

NUREG/CR-3813: MINET VALIDATION STUDY USING STEAM GENERATOR TRANSIENT DATA.

NUREG/CR-3820 V01: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,January-March 1984.

NUREG/CR-3822: SOLA-PTS: A Transient,Three-Dimensional Algorithm For Fluid-Thermal Mixing And Wall Heat Transfer In Complex Geometrics.

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