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Vol. 10, No. 2

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
Second Quarter 1985
April - June

U.S. Nuclear Regulatory Commission

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Policy and Publications Management Branch
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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).

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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 staff of the Publishing and Translations Section of the NRC Division of Technical Information and Document Control.

Main Citations and Abstracts

The report listings in this compilation are arranged by report number, where NUREG-XXXX is an NRC staff-originated report, NUREG/CP-XXXX is an NRC-sponsored conference report, and NUREG/CR-XXXX is an NRC contractor-prepared report. The bibliographic information (see Preface for details) is followed by a brief abstract of this report.

NUREG-0017 R01: CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM PRESSURIZED WATER REACTORS (PWR-GALE CODE). CHANDRASEKARAN; LEE,J.Y.; WILLIS,C.A. Division of Systems Integration (post 811005). April 1985. 208pp. 8505280361. 30603:161.

This report revises the original issuances of NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Pressurized Water Reactors (PWR-GALE Code)" (April 1976), to incorporate more recent operating data now available as well as the results of a number of in-plant measurement programs at operating pressurized water reactors. The PWR-GALE Code is a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents (i.e., the gaseous and liquid source terms). The U.S. Nuclear Regulatory Commission uses the PWR-GALE Code to determine conformance with the requirements of Appendix I to 10 CFR Part 50.

NUREG-0020 V09 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of Febuary 28,1985.(Gray Book I). * Division of Budget & Analysis. April 1985. 403pp. 8505100053. 30269:082.

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 V09 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of March 31,1985.(Gray Book I) ROSS,P.A.; BEEBE,M.R. Division of Budget & Analysis. May 1985. 440pp. 8506170349. 30959:072.

See NUREG-0020,V09,N03 abstract.

NUREG-0020 V09 N05: LICENSING OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of April 30, 1985.(Gray Book I) * Division of Budget & Analysis. June 1985. 441pp. 8507080197. 31396:267.

See NUREG-0020,V09,N03 abstract.

NUREG-0040 V09 N01: LICENSEE CONTRACTOR AND VENDOR STATUS REPORT.Quarterly Report,January-March 1985.(White Book) * Division of QA, Vendor & Technical Training Center Programs (Post 850212). May 1985. 219pp. 8506030069. 30707:001.

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 January 1985 through March 1985. Also included in this issue are the results of certain inspections performed prior to January 1985 that were not included in previous issues of NUREG-0040.

NUREG-0090 V07 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1984. * AEOD, Director's Office. April 1985. 70pp. 8505160182. 30456:325.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period July 1 to September 30, 1984. During the report period, there were four abnormal occurrences at the nuclear power plants licensed to operate. These involved degraded isolation valves in emergency core cooling systems, degraded shutdown systems, a loss of offsite and onsite AC electrical power, and a refueling cavity water seal failure, respectively. There was one abnormal occurrence at a fuel cycle facility; the event involved degraded material access area barriers. There were four abnormal occurrences at the other NRC licensees. One involved contaminated radiopharmaceuticals used in several diagnostic administrations. Two involved therapeutic medical misadministrations. The other involved significant internal exposure to Iodine-125 to a hospital employee. There was one abnormal occurrence reported by an Agreement State; the event involved contaminated radiopharmaceuticals used in several diagnostic administrations. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V07 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1984. * AEOD, Director's Office. May 1985. 40pp. 8506180402. 30986:013.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health and safety and requires a quarterly report of such events to be made to Congress. This report covers the period October 1 to December 31, 1984. During the report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. One involved four control rods failing to insert during testing and the other involved degraded upper head injection system accumulator isolation valves. There was one abnormal occurrence at a fuel cycle facility; the event involved buildup of uranium in a ventilation system. There was one abnormal occurrence reported by an Agreement State; the event involved an overexposure of a radiographer trainee. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0304 V10 N01: REGULATORY AND TECHNICAL REPORTS.Compilation For First Quarter 1985. * Division of Technical Information & Document Control. April 1985. 129pp. 8505240207. 30564:195.

This journal lists all formal reports in the NUREG series prepared by the NRC staff and contractors, as well as proceedings of conferences and workshops. The entries in the compilation are indexed for access by title and abstract, contractor report number, personal author, subject, NRC organization, contractor, and licensed facility.

NUREG-0430 V05 N01: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.January 1984 - June 1984.(Gray Book II) * Director's Office, Office of Inspection and Enforcement. April 1985. 18pp. 8504290008. 30056:348.

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

NUREG-0525 R10: SAFEGUARDS SUMMARY EVENT LIST (SSEL),REVISION 10. * Licensing Policy & Programs Branch (Pre 850707). May 1985. 59pp. 8506140072. 30907:130.

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

NUREG-0540 V07 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-28,1985. * Division of Technical Information & Document Control. April 1985. 699pp. 8505160179. 30457:296.

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 V07 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31,1985. * Division of Technical Information & Document Control. April 1985. 430pp. 8505210419. 30522:018.

See NUREG-0540,V07,N02 abstract.

NUREG-0540 V07 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1985. * Division of Technical Information & Document Control. June 1985. 500pp. 8507080219. 31395:143.

See NUREG-0540,V07,N02 abstract.

NUREG-0606 V07 N02: UNRESOLVED SAFETY ISSUES SUMMARY.Data As Of May 17, 1985. (Aqua Book) * Division of Engineering Technology. June 1985. 61pp. 8507080200. 31390:169.

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

NUREG-0675 S28: 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. April 1985. 635pp. 8505100069. 30286:156.

Supplement No. 28 to the Safety Evaluation Report for the application by the Pacific Gas and Electric Company (PG&E) to operate the 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 status of the staff's investigation, inspection and evaluation of those allegations or concerns that have been identified to the NRC as of March 1, 1985.

NUREG-0675 S30: 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. April 1985. 137pp. 8504220336. 29943:202.

Supplement No. 30 to the Safety Evaluation Report for the application by the Pacific Gas and Electric Company (PG&E) to operate the 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 technical review and evaluation of the design and analysis of piping systems and pipe supports for Unit 2.

NUREG-0675 S31: 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. April 1985. 171pp. 8505240223. 30564:022.

Supplement 31 to the Safety Evaluation Report for the application by Pacific Gas and Electric Company for licenses to operate Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-275/323) has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses a number of matters related to the issuance of an operating license for Diablo Canyon Unit 2, in particular those issues identified by the NRC staff in earlier supplements, commitments made by the applicant, and certain license conditions included in Facility Operating License No. DPR-81 for Diablo Canyon Unit 2.

NUREG-0675 S31: 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. April 1985. 171pp. 8505240223. 30564:022.

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NUREG-0725 R05: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Division of Safeguards. June 1985. 55pp. 8507080177. 31394:330.

This circular has been prepared in response to numerous requests for information regarding routes used for the shipment of irradiated reactor (spent) fuel subject to regulation by the Nuclear Regulatory Commission (NRC), and to meet the requirements of Public Law 96-295. The NRC staff must approve such routes prior to their first use in accordance with the regulatory provisions of Section 73.37 of 10 CFR Part 73. The information included reflects NRC staff knowledge as of June 1, 1985. Spent fuel shipment routes, primarily for road transportation, but also including one rail route, are indicated on reproductions of DOT road maps. Also included are the amounts of material shipped during the approximate three year period that safeguards regulations for spent fuel shipments have been effective. In addition, the Commission has chosen to provide information in this document regarding the NRC's safety and safeguards regulations for spent fuel shipments as well as safeguards incidents regarding spent fuel shipments (of which none have been reported to date). This additional information is furnished by the Commission in order to convey to the public a more complete picture of

NRC regulatory practices concerning the shipment of spent fuel than could be obtained by the publication of the shipment routes and quantities alone.

NUREG-0748 V05 N02: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of February 28,1985.(Orange Book) * Management Support Branch. April 1985. 335pp. 8505070582. 30207:295.

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

NUREG-0748 V05 N03: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of March 31,1985. (Orange Book) * Management Support Branch. May 1985. 342pp. 8506030192. 30688:001.

See NUREG-0748,V05,N02 abstract.

NUREG-0748 V05 N04: OPERATING REACTORS LICENSING ACTIONS SUMMARY.Data As Of April 30,1985.(Orange Book) * Management Support Branch. June 1985. 400pp. 8507020440. 31310:077.

See NUREG-0748,V05,N02 abstract.

NUREG-0750 V21 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1985. * Division of Technical Information & Document Control. June 1985. 59pp. 8507080203. 31380:272.

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

NUREG-0750 V21 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1985. Pages 275-469. * Division of Technical Information & Document Control. April 1985. 203pp. 8504290238. 30056:001.

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

NUREG-0750 V21 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1985. Pages 471-559. * Division of Technical Information & Document Control. May 1985. 78pp. 8505280104. 30606:279.

See NUREG-0750,V21,N02 abstract.

NUREG-0750 V21 N04: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1985. Pages 561-1,041. * Division of Technical Information & Document Control. June 1985. 490pp. 8507080179. 31380:323.

See NUREG-0750,V21,N02 abstract.

NUREG-0797 S10: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION,UNITS 1 AND 2.Docket Nos. 50-445 And 50-446.(Texas Utilities Electric Company) * Division of Licensing. April 1985. 326pp. 8506030061. 30706:008.

Supplement No. 10 to the Safety Evaluation Report for the Texas Utilities Electric Company application for a license to operate Comanche Peak Steam Electric Station, Units 1 and 2 (Docket Nos. 50-445 and 50-446), located in Somervell County, Texas, has been jointly prepared by the Office of Nuclear Reactor Regulation and the Comanche Peak Technical Review Team of the U.S. Nuclear Regulatory Commission. This supplement provides the results of the staff's evaluation and resolution of approximately 400 technical concerns and allegations in the mechanical and piping area regarding construction practices at the Comanche Peak facility. This report does not address the Walsh/Doyle allegations regarding deficiencies in the pipe support design process. Issues raised by the Walsh/Doyle allegations as well as issues raised during recent Atomic Safety and Licensing Board hearings will be dealt with in future supplements to the Safety Evaluation Report as needed.

NUREG-0797 S11: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION,UNITS 1 AND 2.Docket Nos. 50-445 And 50-446.(Texas Utilities Generating Company, et al) * Division of Licensing. May 1985. 349pp. 8506190054. 31018:014.

Supplement No. 11 to the Safety Evaluation Report for the Texas Utilities Electric Company application for a license to operate the Comanche Peak Steam Electric Station, Units 1 and 2 (Docket Nos. 50-445 and 50-446), located in Somervell County, Texas, has been jointly prepared by the Office of Nuclear Reactor Regulation and the Comanche Peak Technical Review Team of the U.S. Nuclear Regulatory Commission (NRC) and is in two parts. Part 1 (Appendix O) of this supplement provides the results of the TRT's evaluation of approximately 125 concerns and allegations relating specifically to quality assurance and quality control (QA/QC) issues regarding construction practices at the Comanche Peak facility. Part 2 (Appendix P) contains overall summary and conclusion of the QA/QC aspects of the NRC Technical Review Team efforts as reported in Safety Evaluation Report (SER) Supplements 7, 8, 9 and 10. Issues raised during recent Atomic Safety and Licensing Board hearings will be dealt with in future supplements to the SER as needed.

NUREG-0797 S11: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF COMANCHE PEAK STEAM ELECTRIC STATION,UNITS 1 AND 2.Docket Nos. 50-445 And 50-446.(Texas Utilities Generating Company, et al) * Division of Licensing. May 1985. 349pp. 8506190054. 31018:014.

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NUREG-0829 DRFT: INTEGRATED PLANT SAFETY ASSESSMENT REPORT,SYSTEMATIC EVALUATION PROGRAM - SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1.Docket No. 50-206.(Southern California Edison Company) * Division of Licensing. April 1985. 558pp. 8505?40058. 30565:001.

The Systematic Evaluation Program was initiated in February 1977 by the U.S. Nuclear Regulatory Commission to review the designs of older operating nuclear reactor plants to confirm and document their safety. The review provides (1) an assessment of how these plants compare with current licensing safety requirements relating to selected issues, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety. This report documents the review of San Onofre Nuclear Generating Station, Unit 1, operated by Southern California Edison Company. The San Onofre 1 facility is one of 10 plants reviewed under Phase II of this program. This report indicates how 137 topics selected for review under Phase I of the program were addressed. Equipment and procedural changes have been identified as a result of the review.

NUREG-0844 DRFT FC: NRC INTEGRATED PROGRAM FOR RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3,A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY.Draft Report For Comment. MURPHY,E. Division of Licensing. April 1985. 166pp. 8505310663. 30666:007.

This report presents the results of the NRC integrated program for the resolution of Unresolved Safety Issues A-3, A-4, and A-5 regarding steam generator tube integrity. The report addresses issues within the areas of steam generator integrity, plant systems response, human factors, radiological consequences and the response of various organizations to a steam generator tube rupture. A generic risk assessment is provided and indicates that risk from steam generator tube rupture events is not a significant contributor to total risk at a given site, nor to the total risk to which the general public is routinely exposed. However, the report identifies a number of actions which the staff finds as a group would be effective in significantly reducing the incidence of steam generator tube degradation, the frequency of tube ruptures and the corresponding potential for significant non-core melt radiological releases, and occupational radiological exposures and which would be effective in mitigating the consequences of SGTR events. The actions would also further reduce risk and have been designated as "staff recommended actions." Final publication of the report herein, following a 90-day period for public comment, will constitute technical resolution of Unresolved Safety Issues A-3, A-4, and A-5.

NUREG-0857 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PALO VERDE NUCLEAR GENERATING STATION,UNITS 1,2 AND 3.Docket Nos. 50-528,50-529 And 50-530.(Arizona Public Service Company, et al) * Division of Licensing. May 1985. 37pp. 8506240081. 31177:312.

Supplement No. 8 to the Safety Evaluation Report for the application filed by Arizona Public Service Company, et al, for licenses to operate the Palo Verde Nuclear Generating Station, Units 1, 2 and 3 (Docket Nos. STN 50-528/529/530) located in Maricopa County, Arizona, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing an evaluation of (1) additional information submitted by the applicants since Supplement No. 7 was issued and (2) matters that the staff had under review when Supplement No. 7 was issued, specifically those issues which required resolution prior to plant operation of Unit 1 above 5% full power.

NUREG-0881 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WOLF CREEK GENERATING STATION,UNIT 1.Docket No. 50-482.(Kansas Gas And Electric Company,et al) * Division of Licensing. June 1985. 22pp. 8506240149. 31152:311.

Supplement No. 6 to the Safety Evaluation Report related to the operation of the Wolf Creek Generating Station, Unit No. 1 updates the information contained in the Safety Evaluation Report, dated April 1982 and Supplements 1, 2, 3, 4, and 5, dated August 1982, June 1983, August 1983, December 1983, and March 1985 respectively. Supplement No. 6 concludes that the facility can be operated by the licensee at power levels greater than 5% without endangering the health and safety of the public. The Safety Evaluation and its supplements pertains to the application for a license to operate the Wolf Creek Generating Station, Unit No. 1 filed by Kansas Gas and Electric Company on February 18, 1980. The Construction Permit No. CPPR-147 was issued on May 17, 1977 and a low power 5% license issued on March 11, 1985. The facility is located in Coffey County, Kansas.

NUREG-0887 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF PERRY NUCLEAR POWER PLANT,UNITS 1 AND 2.Docket Nos. 50-440 And 50-441.(Cleveland Electric Illuminating Company) * Division of Licensing. April 1985. 150pp. 8505010117. 30114:204.

Safety Evaluation Report, NUREG-0887, pertains to the application filed by the Cleveland Electric Illuminating Company on behalf of itself and as agent for the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company, and the Toledo Edison Company (the Central Area Power Coordination Group or CAPCO), as applicants and owners, for a license to operate the Perry Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-440 and 50-441). The report has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lake County, Ohio, approximately 35 miles northeast of Cleveland, Ohio. This Supplement, No. 6 addresses the remaining unresolved Atomic Safety and Licensing Board contention issues; TDI diesel generator reliability in Section 9.6.3.1; hydrogen control system design per the new hydrogen rule in Section 6.2.7; and several issues related to Emergency Plans in Section 13.3.

NUREG-0910 R01 S03: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Technical Information & Document Control. April 1985. 22pp. 8505100062. 30286:119.

In compliance with statutory requirements set forth in Title 44 U.S. Code, "Public Printing and Documents," and in the applicable regulations cited in Title 41 Code of Federal Regulations, "Public Contracts and Property Management," Chapter 101, Subchapter B, "Archives and Records," the U.S. Nuclear Regulatory Commission has published and maintains "NRC Comprehensive Records Disposition Schedule," (NUREG-0910) for records created or maintained by the NRC. Supplement 3 forwards changes to the General Records Schedules as made by the National Archives & Records Administration (NARA) and General Schedule 20 for inclusion.

NUREG-0910 R01 S03: NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Division of Technical Information & Document Control. April 1985. 22pp. 8505100062. 30286:119.

In compliance with statutory requirements set forth in Title 44 U.S. Code, "Public Printing and Documents," and in the applicable regulations cited in Title 41 Code of Federal Regulations, "Public Contracts and Property Management," Chapter 101, Subchapter B, "Archives and Records," the U.S. Nuclear Regulatory Commission has published and maintains "NRC Comprehensive Records Disposition Schedule," (NUREG-0910) for records created or maintained by the NRC. Supplement 3 forwards changes to the General Records Schedules as made by the National Archives & Records Administration (NARA) and General Schedule 20 for inclusion.

NUREG-0936 V04 N01: NRC REGULATORY AGENDA.Quarterly Report,January-March 1985. * Division of Rules and Records. May 1985. 201pp. 8505310669. 30666:173.

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 V04 N01: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1985. * Enforcement Staff. April 1985. 541pp. 8505100072. 30288:071.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1985) 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-0970: PROCEDURES FOR MEETING NRC ANTITRUST RESPONSIBILITIES. TOALSTON,A.L.; MESSIER,M.E.; LAMBE,W.M.; et al. Site Analysis Branch. May 1985. 29pp. 8506070360. 30798:013.

This report describes the procedures used by NRC staff to implement the antitrust review and enforcement prescribed in Sections 105 and 186 of the Atomic Energy Act of 1954, as amended (the Act), as covered largely by the Commission's Rules and Regulations in 10 CFR Parts 2.101, 2.102, 2.200, 50.33a, 50.80, and 50.90. These procedures set forth the steps and criteria the staff applies in the antitrust review of construction permit and operating license applications and the amendments to those applications that deal with changes in ownership. In addition, the procedures describe how the staff enforces compliance by licensees when antitrust conditions have been appended to construction permits and operating licenses.

NUREG-0975 V03: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING TECHNOLOGY.Annual Report For FY 1984. * Division of Engineering Technology. April 1985. 400pp. 8506040260. 30711:262.

This report presents summaries of the research work performed during Fiscal Year 1984 by laboratories and organizations under contracts administered by the NRC's Materials Engineering Branch, Office of Nuclear Regulatory Research. Each contractor has written a more complete and detailed annual report of their work which can be obtained by writing to NRC; however, we believe it is useful to have a summary of each contractor's efforts for the year combined into one volume.

NUREG-0991 S04: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF LIMERICK GENERATING STATION,UNITS 1 AND 2.Docket Nos. 50-352 And 50-353.(Philadelphia Electric Company) * Division of Licensing. May 1985. 51pp. 8506060596. 30776:256.

In August 1983 the NRC Staff issued its Safety Evaluation Report regarding the application for licenses to operate the Limerick Generating Station, Units 1 & 2 located on a site in Montgomery and Chester Counties, Pennsylvania. Supplement 1 was issued in December 1983 and addressed several outstanding issues. It also contains the comments made by the Advisory Committee on Reactor Safeguards in its interim report dated October 18, 1983. Supplement 2 was issued in October 1984. Supplement 3 was issued in October 1984 and addressed issues that required resolution before issuance of the operating license for Unit 1. On October 26, 1984 a license (NPF-27) for Unit 1 was issued which was restricted to a five percent power level and contained conditions which required resolution prior to proceeding beyond the five percent power level. This Supplement 4 addresses some of those technical issues and their associated license conditions which require resolution prior to proceeding beyond the five percent power level. The remaining issues will be addressed in a later supplement to this report. This Supplement 4 also contains the comments made by the Advisory Committee on Reactor Safeguards in its report dated November 6, 1984, regarding full power operation of Limerick Unit 1.

NUREG-1032 DRFT FC: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-44.Draft Report For Comment. BARANOWSKI,P.W. Office of Nuclear Regulatory Research, Director. * Office of Nuclear Reactor Regulation, Director. May 1985. 200pp. 8506250217. 31212:301.

"Station Blackout," which is the complete loss of alternating current (AC) electrical power in a nuclear power plant, has been designated as Unresolved Safety Issue A-44. Because many safety systems required for reactor core decay heat removal and containment heat removal depend on AC power, the consequences of a station blackout could be severe. This report documents the findings of technical studies performed as part of the program to resolve this issue. The important factors analyzed include: the frequency of loss of offsite power; the probability that emergency or onsite AC power supplies would be unavailable; the capability and reliability of decay heat removal systems independent of AC power; and the liklihood that offsite power would be restored before systems that cannot operate for extended periods without AC power fail, thus resulting in core damage. This report also addresses effects of different designs, locations, and operational features on the estimated frequency of core damage resulting from station blackout events.

NUREG-1033: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF WPPSS NUCLEAR PROJECT NO. 3.Docket No. 50-508.(Washington Public Power Supply System) * Division of Licensing. May 1985. 247pp. 8505310065. 30671:001.

The Final Environmental Statement related to the operation of Washington Nuclear Project No. 3 by Washington Public Power Supply System, et al (Docket No. 50-508), located in Grays Harbor County, Washington, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This statement reports on the staff's review of the impact of operation of the plant. Also included are comments of state and federal governments, local agencies and members of the public on the Draft Environmental Statement for this project and staff responses to these comments. The NRC staff has concluded, based on a weighing of environmental, technical and other factors, that an operating license could be granted.

NUREG-1037 DRFT FC: CONTAINMENT PERFORMANCE WORKING GROUP REPORT.Draft Report For Comment. * Division of Engineering. May 1985. 322pp. 8506140588. 30935:257.

Containment buildings for power reactors have been studied to estimate their leak rate as a function of increasing internal pressure and temperature associated with severe accident sequences involving significant core damage. Potential leak paths through containment penetration assemblies (such as equipment hatches, airlocks, purge and vent valves, and electrical penetrations) have been identified and their contributions to leak area for the containment are incorporated into containment leak rate and pressure temperature response as a function of time. Because of lack of reliable experimental data on the leakage behavior of containment penetrations and isolation barriers at pressure beyond their design conditions, an analytical approach has been used to estimate the leakage behavior of components found in specific reference plants that approximately characterize the various containment types.

NUREG-1038 S02: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF SHEARON HARRIS NUCLEAR POWER PLANT,UNIT 1.Docket No. 50-400. (Carolina Power And Light Company And North Carolina Eastern Municipal Power Agency) * Division of Licensing. June 1985. 65pp. 8506270137. 31258:277.

Supplement No. 2 to the Safety Evaluation Report for the application filed by Carolina Power and Light Company and North Carolina Eastern Municipal Power Agency for a license to operate the Shearon Harris Nuclear Power Plant, Unit 1 (Docket No. 50-400), located in Wake and Chatham Counties, North Carolina, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides more recent information regarding resolution of some of the open items identified in the Safety Evalutation Report and in Supplement No. 1. It also addresses one of the recommendations of the Advisory Committee on Reactor Safeguards in its report on the Shearon Harris Plant, dated January 16, 1984, which was inadvertently omitted in Supplement No. 1.

NUREG-1047 S01: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO. 2.Docket No. 50-410. (Niagara Mohawk Power Corporation) * Division of Licensing. June 1985. 35pp. 8507050411. 31372:268.

This report supplements the Safety Evaluation Report (NUREG-1047, February 1985) for the application filed by Niagara Mohawk Power Corporation, as applicant and co-owner, for a license to operate the Nine Mile Point Nuclear Station Unit 2

(Docket No. 50-410). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located near Oswego, New York. Subject to favorable resolution of the items discussed in this report, the NRC Staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

NUREG-1061 V02: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2:Evaluation Of Seismic Designs - A Review Of Seismic Design Requirements For Nuclear Power Plant Piping. * Piping Review Committee. April 1985. 213pp. 8505090016. 30254:043.

This document reports the position and recommendations of the NRC Piping Review Committee, Task Group on Seismic Design. The Task Group considered overlapping conservatism in the various steps of seismic design, the effects of using two levels of earthquake as a design criterion, and current industry practices. Issues such as damping values, spectra modification, multiple response spectra methods, nozzle and support design, design margins, inelastic piping response, and the use of snubbers are addressed. Effects of current regulatory requirements for piping design are evaluated, and recommendations for immediate licensing action, changes in existing requirements, and research programs are presented. Additional background information and suggestions given by consultants are also presented.

NUREG-1061 V05: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 5:Summary - Piping Review Committee Conclusions and Recommendations. * Piping Review Committee. April 1985. 55pp. 8505070580. 30209:240.

This document summarizes a comprehensive review of NRC requirements for Nuclear Piping by the U.S. NRC Piping Review Committee. Four topical areas, addressed in greater detail in Volumes 1 through 4 of this report, are included: (1) Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants, (2) Evaluation of Seismic Design, (3) Evaluation of Potential for Pipe Breaks, and (4) Evaluation of Other Dynamic Loads and Load Combinations. This volume summarizes the major issues, reviews the interfaces, and presents the Committee's conclusions and recommendations for updating NRC requirments on these issues. This report also suggests research or other work that may be required to respond to issues not amendable to resolution at this time.

NUREG-1061 VO2 ADD: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2 Addendum:Summary And Evaluation Of Historical Strong-Motion Earthquake Seismic Response And Damage To Aboveground Industrial Piping. * Piping Review Committee. * Stevenson & Associates. April 1985. 211pp. 8505100057. 30268:005.

Earthquake experience data for industrial piping has been summarized in this report. Conclusions and recommendations for improving the design of nuclear plant piping are made by the author. Input from R. L. Cloud, P. Yaner, and H. Shibata has been included. The material in this report served as background information for the NRC Piping Review Committee Seismic Design Task Group (and their consultants) in the development of the positions given in NUREG-1061 Volume 2.

NUREG-1065 R01: ACCEPTANCE CRITERIA FOR THE LOW ENRICHED URANIUM REFORM AMENDMENTS. EMEIGH,C.W.; GUNDERSEN,G.E.; WITHEE,C.J. Division of Safeguards. April 1985. 53pp. 8504240693. 29988:183.

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

NUREG-1085: FINAL ENVIRONMENTAL STATEMENT RELATED TO THE OPERATION OF NINE MILE POINT NUCLEAR STATION,UNIT NO. 2.Docket No. 50-410.(Niagara Mohawk Power Corporation,et al) * Division of Licensing. May 1985. 373pp. 8505230685. 30548:001.

This Final 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-1095: EVALUATION OF RESPONSES TO IE BULLETIN 82-02.Degradation Of Threaded Fasteners In Reactor Coolant Pressure Boundary Of Pressurized Water-Reactor Plants. ANDERSON,W.; STERNER,P. Division of Emergency Preparedness & Engineering Response (Post 830103). May 1985. 75pp. 8506240221. IEB-82-02. 31177:219.

IE Bulletin 82-02 was issued by the NRC on June 2, 1982 to notify licensees about incidents of severe degradation of threaded fasteners. Responses to the Bulletin from 41 PWR licensees included data from recent regular inspections of reactor coolant pressure boundary components connections of six-inch size and larger. Statistical analysis is used to determine significant factors related to frequency of leakage incidents in connections, occurrence of degradation of bolts and studs, and the need for bolt replacement. Factors examined include the age of the plant, types of components, use of lubricants and sealants, and differences between plants. The compiled data indicate that, on the average, 10% of the bolted connections which were inspected show evidence of leaking and 80% of those undergo some degradation of the bolting. A significant decrease in the occurrence of bolting degradation events as the age of the plant increases is observed. Valves appear to be less subject to bolting corrosion. A group of 5 of the 41 plants accounted for about one-half of the reported leakage and corrosion events. The common characteristic found for 4 of these 5 plants was the lubricant used. The use of nickel-graphite based lubricants appears to offer a significantly reduced incidence of leakage and corrosion; while use of molybdenum disulfide-based lubricants and graphite-based lubricants appears to result in a significantly increased incidence of leakage and corrosion.

NUREG-1116: A REVIEW OF THE CURRENT UNDERSTANDING OF THE POTENTIAL FOR CONTAINMENT FAILURE FROM IN-VESSEL STEAM EXPLOSIONS. * Steam Explosion Review Group. June 1985. 400PP. 8507030716. 31335:001.

A group of experts was convened to review the current understanding of the potential for containment failure from in-vessel steam explosions during core meltdown accidents in LWRs. The Steam Explosion Review Group (SERG) was requested to provide assessments of: (i) the conditional probability of containment failure due to a steam explosion, (ii) a Sandia National Laboratory (SNL) report entitled "An Uncertainty Study of PWR Steam Explosions," NUREG/CR-3369, (iii) a SNL proposed steam explosion research program. This report summarizes the results of the deliberations of the review group. It also presents the detailed response of each individual member to each of the issues. The consensus of the SERG is that the occurrence of a steam explosion of sufficient energetics which could lead to alpha-mode containment failure has a low probability. The SERG members disagreed with the methodology used in NUREG/CR-3369 for the purpose of establishing the uncertainty in the probability of containment failure by a steam explosion. A consensus was reached among SERG members on the need for a continuing steam explosion research program which would improve our understanding of certain aspects of steam explosion phenomenology.

NUREG-1118: ENVIRONMENTAL ASSESSMENT FOR RENEWAL OF SPECIAL NUCLEAR MATERIAL LICENSE NO.SNM-1107.Docket No. 70-1151. (Westinghouse Electric Corporation) * Division of Fuel Cycle & Material Safety. May 1985. 140pp. 8505240050. 30566:199.

This Environmental Assessment is issued by the U.S. Nuclear Regulatory Commission (NRC) in response to an application by the Westinghouse Electric Corporation for the renewal of Special Nuclear Material License No. SNM-1107 which covers the operations of the Columbia plant.

NUREG-1119: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE CAVALIER TRAINING REACTOR AT THE UNIVERSITY OF VIRGINIA.Docket No. 50-396.(University Of Virginia) * Division of Licensing. May 1985. 62pp. 8506060716. 30780:214.

This Safety Evaluation Report for the application filed by the University of Virginia for a renewal of operating license number R-123 to continue to operate a training and research reactor (CAVALIER) has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned and operated by the University of Virginia and is located on the campus in Charlottesville, Virginia. Based on its technical review, the staff concludes that the reactor facility can continue to be operated by the university without endangering the health and safety of the public or endangering the environment.

NUREG-1125 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 1,Part 1:ACRS Reports On Project Reviews (A-F). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 658pp. 8504220393. 29956:167.

This six-volume compilation contains over 1000 reports prepared by the Advisory Committee on Reactor Safeguards from September 1957 through December 1984. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

NUREG-1125 V02: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 2,Part 1:ACRS Reports On Project Reviews (G-P). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 720pp. 8504220344. 29944:001.

See NUREG-1125,V01 abstract.

NUREG-1125 V03: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 3,Part 1:ACRS Reports On Project Reviews (Q-Z). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 563pp. 8504220389. 29954:329.

See NUREG-1125,V01 abstract.

NUREG-1125 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 4,Part 2:ACRS Reports On Generic Subjects (Accident Analysis - Generic Items). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 627pp. 8504220406. 29961:225.

See NUREG-1125,V01 abstract.

NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 630pp. 8504220396. 29958:102.

See NUREG-1125,V01 abstract.

NUREG-1125 V06: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 6,Part 2:ACRS Reports On Generic Subjects (RPA - Appendix C). * ACRS - Advisory Committee on Reactor Safeguards. April 1985. 567pp. 8504220402. 29960:015.

See NUREG-1125,V01 abstract.

NUREG-1127: RADIATION PROTECTION TRAINING AT URANIUM HEXAFLUORIDE AND FUEL FABRICATION PLANTS. BRODSKY,A.; SOONG,A.L.; BELL,J. Division of Radiation Programs & Earth Sciences (post 840429). May 1985. 33pp. 8506190048. 31017:173.

This report provides general information and references useful for establishing or operating radiation safety training programs in plants that manufacture nuclear fuels or process uranium compounds that are used in the manufacture of nuclear fuels. In addition to a brief summary of the principles of effective management of radiation safety training, the report also contains an appendix that provides a comprehensive checklist of scientific, safety, and management topics, from which appropriate topics may be selected in preparing training outlines for various job categories or tasks pertaining to the uranium nuclear fuels industry. The report is designed for use by radiation safety training professionals who have the experience to utilize the report to not only select the appropriate topics, but also to tailor the specific details and depth of coverage of each training session to match both employee and management needs of a particular industrial operation.

NUREG-1128: TRIAL EVALUATIONS IN COMPARISON WITH THE 1983 SAFETY GOALS. RIGGS,R.; SEGE,G. Division of Safety Technology. June 1985. 200pp. 8507080209. 31402:041.

This report provides retrospective comparisons of selected generic regulatory actions to the 1983 NRC safety goals, which had been issued for evaluation during a two-year period. The issues covered are those analyzed by the Office of Nuclear Reactor Regulation (NRR) (assisted in some cases by the Battelle Pacific Northwest Laboratory). The issues include auxiliary feedwater reliability, pressurized thermal shock, power-operated relief valve isolation, asymmetric blowdown loads on PWR primary systems, pool dynamic loads for BWR containments, and steam generator tube rupture. Calculated core-melt frequencies, mortality risks, and cost-benefit ratios are compared with the corresponding safety-goal quantitative design objectives. Considerations that should influence interpretation of the comparisons are discussed. Comments are included on whether and how the safety goals may help in the regulatory decision process and on problems encountered.

NUREG-1131: FINANCIAL ANALYSIS OF POTENTIAL RETROSPECTIVE PREMIUM ASSESSMENTS UNDER THE PRICE-ANDERSON SYSTEM. WOOD,R.S. Office of State Programs, Director. April 1985. 17pp. 8505080348. 30218:171.

Ten representative nuclear utilities have been analyzed over the period 1981-1983 to evaluate the effects of three levels of retrospective premiums on various financial indicators. This analysis continues and expands on earlier analyses prepared as background for deliberations by the U.S. Congress for possible extension or modification of the Price-Anderson Act.

NUREG-1132: TECHNICAL SPECIFICATIONS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2.Docket No. 50-323.(Pacific Gas and Electric Company) * Division of Licensing. April 1985. 466pp. 8505280011. 30605:007.

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

NUREG-1133: TECHNICAL SPECIFICATIONS FOR PALO VERDE NUCLEAR GENERATING STATION,UNIT 1.Docket No. 50-528. (Arizona Public Service Company) * Division of Licensing. May 1985. 515pp. 8506240646. 31151:001.

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

NUREG-1134: RADIATION PROTECTION TRAINING FOR PERSONNEL EMPLOYED IN MEDICAL FACILITIES. MCELROY,N.L.; BRODSKY,A. Division of Radiation Programs & Earth Sciences (post 840429). May 1985. 61pp. 8506130363. 30868:116.

This report provides information useful for planning and conducting radiation safety training in medical facilities to keep exposures as low as reasonably achievable, and to meet other regulatory, safety and loss prevention requirements in today's hospitals. A brief discussion of the elements and basic considerations of radiation safety training programs is followed by a short bibliography of selected references and sample lecture (or session) outlines for various job categories. This information is intended for use by a professional who is thoroughly acquainted with the science and practice of radiation protection as well as the specific procedures and circumstances of the particular hospital's operations. Topics can be added or subtracted, amplified or condensed as appropriate. This document does not set forth specific training program requirements for any particular hospital or type of medical institution or group of employees.

NUREG-1135: SAFETY EVALUATION REPORT RELATED TO THE CONSTRUCTION PERMIT AND OPERATING LICENSE FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF TEXAS.Docket No. 50-602. (University of Texas) * Division of Licensing. May 1985. 88pp. 8506240665. 31152:223.

This Safety Evaluation Report for the application filed by the University of Texas for a construction permit and operating license to construct and operate a 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 University of Texas and is located at the Univeristy's Balcones Research Center, about 7 miles (11.6 kilometers) north of the main campus in Austin, Texas. The staff concludes that the TRIGA reactor facility can be constructed and operated by the University of Texas without endangering the health and safety of the public.

NUREG-1136: TECHNICAL SPECIFICATIONS FOR WOLF CREEK GENERATING STATION, UNIT 1.Docket No. 50-482.(Kansas Gas And Electric Company) * Division of Licensing. June 1985. 498pp. 8506270251. 31257:142.

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

NUREG-1137: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF VOGTLE ELECTRIC GENERATING PLANT,UNITS 1 AND 2.Docket Nos. 50-424 And 50-425.(Georgia Power Company,et al) * Division of Licensing. June 1985. 650pp. 8507030707. 31314:265.

The Safety Evaluation Report for the application filed by Georgia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and City of Dalton, Georgia, as applicants and owners, for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Burke County, Georgia, approximately 41.5 km (26 mi) south-southeast of Augusta, and on the Savannah River. Subject to favorable resolution of the items discussed in this report, the staff concludes that the applicant can operate the facility without endangering the health and safety of the public.

NUREG-1140 DRFT FC: A REGULATORY ANALYSIS ON EMERGENCY PREPAREDNESS FOR FUEL CYCLE AND OTHER RADIOACTIVE MATERIAL LICENSEES.Draft Report For Comment. MCGUIRE,S.A. Division of Risk Analysis & Operations (post 840429). June 1985. 125pp. 8507020410. 31309:033.

Potential accidents for 15 types of fuel cycle and other radioactive material licensees were analyzed. The most potentially hazardous accident, by a large margin, was determined to be the sudden rupture of a heated multi-ton cylinder of UF6. Acute fatalities offsite are probably not credible. Acute permanent injuries may be possible for many hundreds of meters, and clinically observable transient effects of unknown long term consequences may be possible for distances up to a few miles. These effects would be caused by the chemical toxicity of the UF6. Radiation doses would not be significant. The most potentially hazardous accident due to radiation exposure was determined to be a large fire at certain facilities handling large quantities of alpha-emitting radionuclides (i.e., Po-210, Pu-238, Pu-239, Am-241, Cm-242, Cm-244) or radioiodines (I-125 and I-131). However, acute fatalities or injuries to people offsite due to accidental releases of these materials do not seem plausible. The only other significant accident was identified as a long-term pulsating criticality at fuel cycle facilities handling high-enriched uranium or plutonium. An important feature of the most serious accidents is that releases are likely to start without prior warning. The releases would usually end within about half an hour. Thus protection actions would have to be taken quickly to be effective. There is not likely to be enough time for dose projections, complicated decisionmaking during the accident, or the participation of personnel not in the immediate vicinity of the site. The appropriate response by the facility is to immediately notify local fire, police, and other emergency personnel and give them a brief predetermined message recommending protective actions. Emergency personnel are generally well qualified to respond effectively to small accidents of these types.

NUREG-1145 V01: U.S. NUCLEAR REGULATORY COMMISSION 1984 ANNUAL REPORT. * Office of Resource Management, Director. June 1985. 234pp. 8506260386. 31246:057.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1984 within the NRC or involving the NRC.

NUREG-1147: SEISMIC SAFETY RESEARCH PROGRAM PLAN. * Division of Engineering Technology. June 1985. 230pp. 8507080215. 31392:150.

This plan describes the safety issues, regulatory needs, and the research necessary to address these needs. The plan also discusses the relationship between current and proposed research within the NRC and research sponsored by other government agencies, universities, industry groups, professional societies, and foreign sources.

NUREG/CP-0059 V01: PROCEEDINGS OF THE MITI-NRC SEISMIC INFORMATION EXCHANGE MEETING,VOLUME I. WEISS,A.J. Brookhaven National Laboratory. April 1985. 423pp. 8506070372. BNL-NUREG-51821. 30796:001.

The first Japan Ministry of International Trade and Industry (MITI) - U.S. Nuclear Regulatory Commission (NRC) Seismic Information Exchange Meeting (SIEM) was held July 18-20, 1984 in Palo Alto, California. The purpose of SIEM was to provide technical information on seismic research being conducted under MITI and NRC sponsorships to the participants. The aim was to improve understanding of the seismic research in progress in Japan and the United States for possible identification of areas of mutual interest which could be the basis for future cooperation. Approximately 40 Japanese and U.S. technical specialists in seismic research participated in the meeting. These proceedings represent the compilation of the papers presented at the meeting.

NUREG/CP-0062: PROCEEDINGS OF THE CONFERENCE ON THE APPLICATION OF GEOCHEMICAL MODELS TO HIGH-LEVEL NUCLEAR WASTE REPOSITORY ASSESSMENT. JACOBS,G.K.; WHATLEY,S.K. Oak Ridge National Laboratory. May 1985. 130pp. 8506130505. ORNL/TM-9585. 30892:227.

A conference on the application of geochemical models in the assessment of high-level nuclear waste repositories was held to discuss the current status of geochemical code development, thermodynamic data bases, reaction kinetics, and coupled-process models as applied to site characterization and performance assessment activities. These proceedings include extended abstracts of the technical presentations given at the conference, a discussion of the role of geochemical modeling in predicting the performance of repositories, and a set of recommendations that identify the key developments needed in order for geochemical models to become more applicable for quantitative evaluations of repositories. Detailed recommendations relevant to the following subjects are discussed: (1) improved simulation of repository performance through inclusion of additional important geochemical processes and parameters into current geochemical models, (2) more careful attention to uncertainties associated with geochemical model calculations, (3) assigning priorities to (through sensitivity studies and critical evaluations) and then improving and/or obtaining important thermodynamic data, and (4) addressing the importance of kinetics in simulating repository behavior.

NUREG/CP-0065: TRANSACTIONS OF THE 8TH INTERNATIONAL CONFERENCE ON STRUCTURE MECHANICS IN REACTOR TECHNOLOGY.Panel Session J-K: Status of Research In Structural And Mechanical Engineering For Nuclear Power Plants. BROWZIN,B.S. Division of Engineering Technology. June 1985. 266pp. 8507080187. 31393:277.

These transactions of the J-K/panel session include preprints of papers or abstracts which are listed in Volume A, "Introduction, General Contents, Authors' Index," Proceedings of the 8th International Conference on Structural Mechanics in Reactor Technology. These papers represent the body of the J-K/panel session, "Status of Research in Structural and Mechanical Engineering for Nuclear Power Plants," sponsored by the U.S. Nuclear Regulatory Commission.

NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite. MOORE,E.B. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1985. 41pp. 8505070571. 30209:200.

The purpose of this addendum is to examine the impacts of an interim inability to carryout offsite disposal of radioactive wastes and spent fuel on the decommissioning of multiple-reactor power station. The example selected for study is a four-PWR station in which each PWR is prepared for safe storage at two-year intervals, held in safe storage for 100 year intervals. BWRs are neglected for simplicity and in the expectation that the results would be similar to those for PWRs. Only SAFSTOR is considered because DECON and ENTOMB are unsuitable by definition for interim storage of radioactive wastes and/or spent fuel. It is assumed that all radioactive wastes and spent fuel are shipped offsite by the end of decommissiong.

NUREG/CR-2000 V04 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1985. * Oak Ridge National Laboratory. April 1985. 78pp. 8505070557. ORNL/NSIC-200. 30210:092.

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System.

NUREG/CR-2000 V04 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1985. * Oak Ridge National Laboratory. May 1985. 87pp. 8506130364. ORNL/NSIC-200. 30867:262.

See NUREG/CR-2000,V04,N03 abstract.

NUREG/CR-2000 V04 N5: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of May 1985. * Oak Ridge National Laboratory. June 1985. 111pp. 8507030669. ORNL/NSIC-200. 31314:155.

See NUREG/CR-2000,V04,N03 abstract.

NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984. WEISS,A.J. Brookhaven National Laboratory. May 1985. 117pp. 8506060147. BNL-NUREG-51454. 30781:002.

This progress report will describe current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the Division of Accident Evaluation, Division of Engineering Technology, and Division of Risk Analysis & Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The projects reported are the following: High Temperature Reactor Research, SSC Development, Validation and Application, Generic Balance of Plant Modeling, Thermal-Hydraulic Reactor Safety Experiments, Development of Plant Analayzer, 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; Probability Based Load Combinations for Design of Category I Structures, Identification of Age-Related Failure Modes; Analysis of Human Error Data for Nuclear Power Plant Safety Related Events, Human Factors Aspects of Safety/Safeguards Interactions, Emergency Action Levels, and Protective Action Decision Making.

NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984. WEISS,A.J. Brookhaven National Laboratory. May 1985. 139pp. 8507050378. BNL-NUREG-51454. 31373:001.

This progress report will describe current activities and technical progress in the programs at Brookhaven National Laboratory sponsored by the Division of Accident Evaluation, Division of Engineering Technology, and Division of Risk Analysis & Operations of the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The projects reported are the following: High Temperature Reactor Research, SSC/MINET Development, Validation and Application, Thermal-Hydraulic Reactor Safety Experiments, Plant Analyzer, Code Assessment and Application, Code Maintenance (RAMONA-3B), Calculational Quality Assurance in Support of PTS; Stress Corrosion Cracking of PWR Steam Generator Tubing, Probability Based Load Combinations for Design of Category I Structures, Soil-Structure Interaction Evaluations, Identification of Age-Related Failure Modes; Application of HRA/PRA Results to Resolve Human Reliability and Human Factors Safety Issues, PRA Technology Transfer Program, Emergency Action Levels, and Protective Action Decisionmaking.

NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK. HARDY,H.A.; LAATS,E.T. EG&G, Inc. May 1985. 95pp. 8505170007. EGG-2164. 30483:079.

The United States Nuclear Regulatory Commission (USNRC) has established the NRC/Division of Accident Evaluation (DAE) Data Bank Program to collect, store, and make available data from the many domestic and foreign water reactor safety research programs. Local direction of the program is provided by EG&G Idaho,Inc., prime contractor for the Department of Energy (DOE) at the Idaho National Engineering Laboratory (INEL). The NRC/DAE Data Bank Program provides a central computer storage mechanism and access software for data that is to be used by code development and assessment groups in meeting the code correlation needs of the nuclear industry. The administrative portion of the program provides data entry, documentaton, training, and advisory services to users and the USNRC. The NRC/DAE Data Bank and the capabilities of the data access software are described in this document.

NUREG/CR-2663 V01: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Main Report. * Ertec Western, Inc., May 1985. 574pp. 8506270457. 31261:001.

Evaluation of the geologic isolation of radioactive materials from the biosphere requires an intimate knowledge of site geologic conditions, which is gained through precharacterization and site characterization studies. This report presents the results of an intensive literature review, analysis and compilation to delineate the information needs, applicable techniques and evaluation criteria for programs to adequately characterize a site in six geologic media. These media, in order of presentation, are: granite, shale, basalt, tuff, bedded salt, and domed salt. Guidelines are presented to assess the efficacy (application, effectiveness, and resolution) of currently used to exploratory and testing techniques for precharacterization or characterization of a site. These guidelines include the reliability, accuracy, and resolution of techniques deemed acceptable, as well as cost estimate of various field and laboratory techniques used to obtain the necessary information. Guidelines presented do not assess the relative suitability of media. This report consists of two volumes: main report and appendices.

NUREG/CR-2663 V02: INFORMATION NEEDS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE REPOSITORY SITES IN SIX GEOLOGIC MEDIA.Appendices. * Ertec Western, Inc., May 1985. 700pp. 8506270247. 31259:001.

See NUREG/CR-2663,V01 abstract.

NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II:Parametric Studies. NELSON,L.S.; DUDA,P.M. Sandia National Laboratories. April 1985. 154pp. 8506140047. SAND82-1105. 30908:219.

The steam explosion experiments performed with single drops of molten iron oxide melted with a CO(2) laser, described in Part I of this report, were extended here. The following major parameters were varied: ambient pressure, water temperature and subcooling, melt temperature, and melt composition. Also, a few scoping experiments were performed to explore the effects of changing the nature of the coolant, and the viscosity of the melt. As each of the four major parameters was varied, thresholds could be located beyond which explosions were suppressed. However, in general, the explosions could be reinitiated by increasing the magnitude of the triggering pulse. The effects of increasing the ambient pressure up to 1.12 MPa were faster and finer melt fragmentation, and faster and more complete transfer of heat from melt to water. Moreover, triggering became easier over the range of ambient pressure between about 0.15 MPa and approximately 0.7 MPa.

NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris. OTTINGER,C.A.; MITCHELL,G.W.; LIPINSKI,R.J.; et al. Sandia National Laboratories. June 1985. 74pp. 8507050431. SAND84-1838. 31371:101.

The D9 experiment investigated the coolability of a shallow (77 mm), stratified urania bed in sodium. The bed was fission heated in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories to simulate the effects of radioactive decay heating. It was the first stratified debris bed experiment to use an extended UO2 particle size distribution (0.038 to 4.0 mm). Dryout occurred at powers ranging from 0.10 to 0.58 W/g, which was close to the incipient boiling power and before channels penetrated the subcooled zone in the bed, even with subcoolings as low as 80 degrees centigrade. Channel penetration was observed after dryout began, but the bed became only moderately more coolable. All these observations agree with current models.

NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS. NALEZNY,C.L. EG&G, Inc. April 1985. 212pp. 8507050424. EGG-2231. 31372:001.

This document is a summary of the main research results of the Loss-of-Fluid Test (LOFT) Program relative to code assessment, code development, licensing, rulemaking, safety technology, and reactor operations. The LOFT facility is a 50 MW(t) pressurized water reactor (PWR) system with instruments that measure and provide data on the system thermal-hydraulic and nuclear conditions. The transient response of the LOFT system to accident events is similar to large [1000 MW(e)] commercial PWRs. The main objectives of the LOFT Experimental Program were to qualify the engineered safety systems used in commercial PWRs and to verify the computer codes used in safety analyses. The LOFT Program contributed to the improvement of computer codes used to predict the response of commercial PWRs, demonstrated the adequacy of engineered safety systems, and contributed to improved understanding of PWR accident phenomena, particularly those associated with the evaluation model in Appendix K to 10 CFR 50 (the "ECCS rule").

NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984. JAIN,H.; VEAKIS,E.; SOO,P. Brookhaven National Laboratory. June 1985. 29pp. 8507050398. BNL-NUREG-51630. 31373:314.

The current study is part of an ongoing task to specify tests that may be used to verify that engineered waste package/repository systems comply with NRC radionuclide containment and controlled release performance objectives. Work covered in this report includes tuff packing material for use in a high level waste tuff repository. Ranges of repository conditions relevant to its testing and other factors important for its performance are discussed.

NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984. JAIN,H.; VEAKIS,E.; SOO,P. Brookhaven National Laboratory. June 1985. 34pp. 8507050402. BNL-NUREG-51630. 31372:302.

This ongoing study is part of a task to specify tests that may be used to verify that engineered waste packages/repository systems comply with NRC radionuclide containment and controlled release performance objectives. Work covered in this report includes crushed tuff packing material for use in a high level waste tuff repository. A review of available tests to quantify packing performance is given together with recommendations for future testing work.

NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983. HINZE,W.J.; BRAILE,L.W. Purdue Univ., West Lafayette, IN. KELLER,G.R.; et al. Texas, Univ. of, El Paso, TX. April 1985. 60pp. 8504220438. 29946:263.

Recent geophysical investigations have shown that the seismicity of the New Madrid, Missouri, seismogenic region correlates with an ancient rift complex suggesting that the anomalous seismicity is the result of the localization of the regional compressive stress pattern by basement structures. An integrated geophysical/gelogical research program is being conducted to evaluate the rift complex hypothesis, to refine our knowledge of the structure and physical properties of the rift complex, and to investigate the possible northern extensions of the New Madrid Fault zone, especially the possible northeastern connection to the Anna, Ohio, seismic region. Investigation of the northeast extension has focused upon the acquisition and preparation of arrays of gravity and magnetic data sets. During 1983, special emphasis was placed upon integration of these data with basement lithologic and seismicity information which has revealed several major lithologic/structural features in the crust of the Anna area. Current seismicity in this region appears to be related to an ancient rift structure (the Fort Wayne rift) and possibly its contact with a low density pluton. Minor seismicity may be caused by stress concentration associated with local basement inhomogeneities.

NUREG/CR-3178: STRUCTURAL AND TECTONIC STUDIES IN NEW YORK STATE.Final Report,July 1981 - June 1982. ISACHSEN,Y.W. New York, State Univ. of, Albany, NY. * Boston College, Chestnut Hill, MA. April 1985. 84pp. 8505100048. 30270:214.

Subjects treated in this report include the distribution, trends, exposure characteristics, aeromagnetic signatures, and detailed geometries of fracture systems, as well as tentative inferences concerning relative ages and causes of reactivation. Stress indicators are discussed, and a beginning is made at working out regional paleostress directions using the attitudes of dated mafic dikes. Attempts at defining Holecene and recent crustal movements using geological, geodetic, and seismological methods are reviewed, as well as attempts to relate projected focal mechanism solutions to ground geology. Finally, the distribution of earthquakes and their relationships to geology is reviewed.

NUREG/CR-3193: FORCED CONVECTIVE,NONEQUILIBRIUM,POST-CHF HEAT TRANSFER EXPERIMENT DATA AND CORRELATION COMPARISON REPORT. GOTTULA,R.C.; CONDIE,K.G.; SUNDARUM,R.K.; et al. EG&G, Inc. April 1985. 562pp. 8504160110. EGG-2245. 29833:001.

Forced convective postcritical-heat-flux heat transfer experiments with water flowing upward in a vertical tube have been conducted at the Idaho National Engineering Laboratory. Thermodynamic nonequilibrium in the form of superheated vapor temperatures was measured at a maximum of three different axial levels. Steady-state experiments were conducted at pressures of 0.2 to 0.7 MPa, mass fluxes of 12 to 24 kg/m(2).s, heat fluxes of 7.7 to 27.5 kW/m(2), and test section inlet qualities of 38 to 64%. Quasi-steady-state (slow moving quench front) experiments were conducted at pressures of 0.4 to 7 MPa, mass fluxes of 12 to 70 kg/m(2).s, heat fluxes of 8 to 225 kW/m(2), and test section inlet qualities of -7 to 47%. The multiple probe data and the data taken above 0.4 MPa are new data in parameter ranges not previously obtained. Comparison of the data with current vapor generation models and wall heat transfer models yielded unsatisfactory results. This is attributed to the effects of nonequilibrium, quench front quality, and distance from the quench front, which are factors not included in the current models compared.

NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions. ELRICK,R.M.; SALLACH,R.A.; OUELLETTE,A.L.; et al. Sandia National Laboratories. June 1985. 156pp. 8507020369. SAND83-0395. 31307:175.

Laboratory scale scoping studies, using chemical simulants, are examining physical and chemical processes that could occur between fission products and other primary system materials in a steam and hydrogen environment. The chemical systems studied were cesium hydroxide vapor reactions in steam and hydrogen at 970K in a 304 Stainless steel system, at 1120K in a 304SS system, at 1000K with iodine vapor in an alumina system and at 1000K with hydrogen iodine vapor in an alumina system. Major observations and conclusions are that: cesium in the CsOH reacts with the silicon dioxide in the inner oxide formed on stainless steel to produce a cesium silicate; the availability of SiO(2) may therefore control the extent of reaction of CsOH with 304SS in steam; the oxidation rate of 304SS is enhanced by the exposure to CsOH vapor; the reaction of CsOH with Iconel 600 is slow in steam and seems to react with the silica content in the oxide layer.

NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT. KNIGHT,T.D.; METZGER,V. Los Alamos Scientific Laboratory. April 1985. 371pp. 8504160087. LA-9700-MS. 29835:001.

This report describes the final results of the development assessment analyses conducted during the later stages of the TRAC-PD2 development. The calculations discussed in this report used the released version of TRAC-PD2 and cover separate-effects blowdown, heat transfer, and downcomer penetration tests together with integral tests from the Loss-of-Fluid Test and Semiscale facilities. Although these calculations are not an exhaustive test of the code, they demonstrate its capabilities, including automatic steady-state initialization and the complete transient from blowdown through refill and reflood. The results show good agreement between the calculated parameters and the data and indicate that the code is applicable to large-break loss-of-coolant accident analyses.

NUREG/CR-3228 V03: STRUCTURAL INTEGRITY OF WATER REACTOR PRESSURE BOUNDARY COMPONENTS.Annual Report For 1984. LOSS,F.J. Materials Engineering Associates, Inc. June 1985. 171pp. 8506260518. MEA-2075. 31245:026.

This program consists of research and engineering relating to 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 environmental cracking behavior under conditions of a nuclear environment, so investigation of irradiated materials is a key element of each task. 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. Experimental studies are supported by analytical models and investigations 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. Current work is organized into three major tasks: (1) fracture mechanics investigations, (2) environmentally-assisted crack growth in high temperature, primary reactor water and (3) radiation sensitivity and postirradiation properties recovery. Research progress in these three tasks for 1984 is summarized here.

NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report. ELDER,H.K. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 327pp. 8506140337. 30932:308.

Technical requirements, costs and safety are conceptually evaluated for the post-accident cleanup and decommissioning of fuel cycle and non-fuel cycle facilities that have experienced a significant accident. Accident cleanup is postulated to include 1) initial decontamination of building surfaces to reduce the subsequent occupational dose to cleanup and decommissioning workers and 2) management of the resulting wastes. Decommissioning is assumed to follow accident cleanup. In order to ensure that worker doses are ALARA, despite higher radiation exposure to workers during post-accident operations, careful planning and rehearsal of cleanup operations and the use of remote and semi-remote cleaning techniques are required to reduce occupancy times in high-radiation areas and to minimize occupational exposures during accident cleanup. The public safety impacts of post-accident cleanup and decommissioning are also evaluated; these are below permissible radiation dose levels in unrestricted areas and well within the range of annual radiation doses from normal background.

NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices. ELDER,H.K. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 288pp. 8506170550. 30979:001.

This volume contains the appendices concerning the technical requirements, costs and safety aspects conceptually evaluated for post-accident cleanup and decommissioning of fuel cycle and non-fuel cycle facilities that have experienced a significant accident. Accident cleanup is postulated to include 1) initial decontamination of building surfaces to reduce the subsequent occupational dose to cleanup and decommissioning workers and 2) management of the resulting wastes. Decommissioning is assumed to follow accident cleanup. In order to ensure that worker doses are ALARA, despite higher radiation exposure to workers during post-accident operations, careful planning and rehearsal of cleanup operations and the use of remote and semi-remote cleaning techniques are required to reduce occupancy times in high-radiation areas and to minimize occupational exposures during cleanup. The public safety impacts of post-accident cleanup and decommissioning are also evaluated; these are below permissible radiation dose levels in unrestricted areas and well within the range of annual radiation doses from normal background.

NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE. OWCZARSKI,P.C.; POSTMA,A.K.; SCHRECK,R.I. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 68pp. 8506240650. PNL-4742. 31152:156.

The Pacific Northwest Laboratory has developed a prototype version of a Suppression Pool Aerosol Removal Code (SPARC). This code was written to calculate the capture of aerosol particles in the pressure suppression pool (wet well) of a boiling water reactor under hypothetical accident conditions. The code incorporates five aerosol scrubbing models and two thermal-hydraulic models. The scrubbing models describe 1) steam condensation, 2) soluble particle growth in a humid atmosphere, 3) gravitational settling, 4) inertial deposition, 5) diffusional deposition. Mechanical entrainment of pool liquid by breaking of bubbles at the surface was also considered. An optional model for equilibrium pool temperature and a model for steam evaporation are the two thermal-hydraulic models used in the code. Steam evaporation was found to significantly retard deposition processes in pools near the boiling point. The code user supplies the values of several controlling variables in the code input. The SPARC output can include the decontamination factors (DF) of twenty different particle size groups, an overall DF for the whole particle distribution, particle log normal distribution parameters, and mass flow rates of particles (wet and dry) leaving the pool.

NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA. HUCHTON,R.L.; TKACHYK,J.W.; TAYLOR,J.T.; et al. Westinghouse Electric Corp. April 1985. 80pp. 8505230585. WINCO-1024. 30546:254.

A comparison was made among various silver impregnated adsorption media to determine their iodine, krypton, and xenon retention efficiencies. The program consisted of three components. First, laboratory measurements of the noble gas retention efficiencies of commercially available adsorption media were determined as a function of relative humidity, sample duration, test cartridge geometry, and ambient air purge. Second, a literature survey was performed to evaluate the iodine species retention efficiencies of the selected media. Third, data associated with a media previously proposed for an emergency response air sampler were incorporated to enlarge the data base.

NUREG/CR-3469 V02: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS. Annotated Bibliography Of Selected Readings In Radiation Protection And ALARA. BAUM,J.W.; WEILANDICS,C. Brookhaven National Laboratory. June 1985. 150PP. 8507020380. BNL-NUREG-51708. 31306:251.

This is the second volume of abstracts dealing with occupational dose, dose control, dose reduction and application of the ALARA (as low as reasonably achievable) principle at nuclear power plants. This volume contains abstracts selected from APPLIED HEALTH PHYSICS ABSTRACTS AND NOTES, Volumes 1, No. 1, 1975 through Volume 5, No. 4, October 1979, and from recent publications known to the authors. Author and subject indexes are included. The subject index in this volume covers abstracts in both Volumes 1 and 2. This volume contains abstract Numbers 252 through 549.

NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions. TOTH,L.M.; DODSON,K.E. Oak Ridge National Laboratory. April 1985. 22pp. 8506100496. ORNL/TM-8664/V2. 30830:083.

Redox reactions that might alter the volatility of aqueous iodine solutions have been examined experimentally using absorption spectrophotometry. Oxygen and hydrogen atmospheres had no effect on the iodine chemistry at temperatures up to 150 degrees centigrade. However, irradiation of aqueous solutions with a (60)Co source, 0.8 x 10(6) R/h, produced radiolysis products that either oxdized iodine ion or reduced 10(3)- in the pH range 6-9 and generated significant amounts of volatile iodine. The amount of iodine volatilized varied from a few percent for solute concentrations of 10(-4) M to as much as 10 to 19% for 10(-6) M CsI or KI0(3) solutes. Silver metal has been shown to provide an effective gettering route for I- in solution if these ions are first oxidized by OH radicals generated during the radiolysis of the solutions.

NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS. GUYMON,R.H.; CASTO,W.R.; COMPERE,E.L. Oak Ridge National Laboratory. May 1985. 82pp. 8506140622. ORNL/NOAC-214. 30934:031.

Storage and handling of compressed gases at nuclear power plants were studied to identify any potential safety hazards. Gases investigated were air, acetylene, carbon dioxide, chlorine, Halon, hydrogen, nitrogen, oxygen, propane, and sulfur hexafluoride. Physical properties of the gases were reviewed as were applicable industrial codes and standards. Incidents involving pressurized gases in general industry and in the nuclear industry were studied. In this report general hazards such as missiles from ruptures, rocketing of cylinders, pipe whipping, asphyxiation, and toxicity are discussed. Even though some serious injuries and deaths over the years have occurred in industries handling and using pressurized gases, the industrial codes, standards, practices, and procedures are very comprehensive. The most important safety consideration in handling gases is the serious enforcement of these well-known and established methods. Recommendations are made concerning compressed gas cylinder missiles, hydrogen line ruptures or leaks, and identification of lines and equipment.

NUREG/CR-3558: HANDBOOK OF NUCLEAR POWER PLANT SEISMIC FRAGILITIES. Seismic Safety Margins Research Program. COVER,L.E.; BOHN,M.P.; CAMPBELL,R.D.; et al. Lawrence Livermore National Laboratory. June 1985. 300pp. 8507080210. UCRL-53455. 31402:238.

The Seismic Safety Margins Research Program (SSMRP) is an NRC-funded, multiyear program conducted by Lawrence Livermore National Laboratory (LLNL). Its goal is to develop a complete and fully-coupled analysis procedure, including methods and computer codes, for estimating the risk of earthquake-induced radioactive release from a commercial nuclear power plant. As part of this program, calculations of the seismic risk from a typical commercial nuclear reactor were made. These calculations required a knowledge of the probability of failure (fragility) of safety-related components in the reactor system that actively participate in the hypothesized accident scenarios. This report describes the development of the required fragility relations and the data sources and data reduction techniques upon which they are based. Both building and component fragilities are covered. The building fragilities are for the Zion Unit 1 reactor, the specific plant used for development of methodology in the program. Some of the component fragilities are site-specific, but most would be usable for other sites as well.

NUREG/CR-3611: RADIOACTIVE MATERIAL (RAM) ACCIDENT/INCIDENT DATA ANALYSIS PROGRAM. EMERSON,E.L.; MCCLURE,J.D. Sandia National Laboratories. April 1985. 40pp. 8504220385. SAND82-2156. 29946:323.

This report describes the development of the Radioactive Materials Transportation Accident/Incident Data Base (RAM-AIDB), which contains information on the occurrences of transportation accidents and incidents, for radioactive materials (RAM) that are involved in the process of transportation, loading and unloading operations, or temporary storage. These transportation operations are in support of the nuclear fuel cycle for electrical energy generations of RAM. This study analyzes in some detail basic accident/incident statistical data, RAM packaging accident response data, and the health effects associated with RAM transport accidents/incidents. This report presents a summary of U.S. RAM transport accident/incident experience for the period 1971 through December 1981. In addition, a sample annual summary of accident/incident experience is presented for the calendar year 1981.

NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report for 1984. ATTERIDGE,D.G.; BRUEMMER,S.M.; PAGE,R.E. Battelle Memorial Institute. Pacific Northwest Laboratories. June 1985. 63pp. 8506270333. PNL-4971. 31262:215.

Pacific Northwest Laboratory (PNL), under a program sponsored by the Division of Engineering Technology of the U.S. Nuclear Regulatory Commission (NRC), is conducting a program to determine a method for evaluating the acceptance of welded and repair-welded stainless steel (SS) piping for light-water reactor (LWR) service. Validated models, based on experimental data, will be developed to predict the degree of sensitization (DOS) and the intergranular stress corrosion cracking (IGSCC) susceptibility in the heat affected zone (HAZ) of the SS weldments. IGSCC is caused by a combination of a sensitized microstructure, an aggressive environment, and tensile stress. Control of any of these three factors can eliminate IGSCC in most practical situations. This program will measure and model the development of a sensitized microstructure as it pertains to welded and repair-welded SS pipe. An empirical correlation between a material's DOS and its susceptibility to IGSCC will be determined using constant extension rate tests (CERTs). The successful completion of these tasks will result in a method for assessing the effects of welding/repairing parameters on the IGSCC susceptibility of component-specific nuclear reactor welds/repairs.

NUREG/CR-3626 V02: MAINTENANCE PERSONNEL PERFORMANCE SIMULATION (MAPPS) MODEL: DESCRIPTION OF MODEL CONTENT,STRUCTURE,AND SENSITIVITY TESTING. SIEGEL,A.I.; BARTTER,W.D.; WOLF,J.J.; et al. Oak Ridge National Laboratory. April 1985. 322pp. 8504170234. ORNL/TM-9041/V2. 29902:002.

This volume of NUREG/CR-3626 presents details of the content, structure, and sensitivity testing of the Maintenance Personnel Performance Simulation (MAPPS) model that was described in summary in volume one of this report. The MAPPS model is a generalized stochastic computer simulation model developed to simulate the performance of maintenance personnel in nuclear power plants. The MAPPS model considers workplace, maintenance technician, motivation, human factors, and task oriented variables to yield predictive information about the effects of these variables on successful maintenace task performance. All major model variables are discussed in detail and their implementation and interactive effects are outlined. The model was examined for disqualifying defects from a number of viewpoints, including sensitivity testing. This examination led to the identification of some minor recalibration efforts which were carried out. These positive results indicate that MAPPS is ready for initial and controlled applications which are in conformity with its purposes.

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NUREG/CR-3647: DESIGN AND FABRICATION OF A 1/8-SCALE STEEL CONTAINMENT MODEL. REESE,R.T.; HORSCHEL,D.S. Sandia National Laboratories. April 1985. 131pp. 8504170693. SAND84-0048. 29907:022.

A 1/8-scale steel model containment building was designed and fabricated in support of the Containment Safety Margins Program. This program is directed to determine the margin of safety of containments in severe accident conditions. It is planned to internally pressurize the model to failure. In this testing program, failure modes of the pressure vessel and scaled penetrations will be examined in detail. The model was designed according to Section III of ASME Code for Class MC containment vessels with the exception that no code stamp was required since no nuclear materials would be housed within the model. All the general requirements (subsection NCA) and specific requirements (subsection NE) of Section III of the ASME Code were met. The majority of the model was fabricated from 3/16-in. SA516 Grade 70 steel plate in the form of a right circular cylinder capped with a hemispherical dome. Eleven penetrations and two lifting trunnions were included in the model. The cylinder/dome section was joinded to a 2:1 ellipsoidal base (test fixture) composed of thicker (1 1/8-in. and 1 1/2-in.) plate material. The model was supported on six legs to permit access for personnel, instrumentation, data acquisition, power, and pressure piping. The model was fabricated in April through October 1983 by Chicago Bridge and Iron and erected at the test site in Albuquerque, New Mexico, in November 1983.

NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES. HARDY,J.E.; HERSKOVITZ,M.B. Oak Ridge National Laboratory. April 1985. 56pp. 8506070366. ORNL/TM-9067. 30798:265.

Instrumentation for making two-phase measurements in experimental refill-reflood test facilities was developed by Oak Ridge National Laboratory (ORNL) through the Advanced Instrumentation for Reflood Studies (AIRS) program. These unique instrumentation systems were designed to survive the severe in-vessel environmental conditions that exist during a simulated pressurized water reactor loss-of-coolant accident (LOCA). The measurements include two-phase flow velocity, void fraction, and film thickness and velocity, and are required for better understanding of reactor behavior during LOCAs. The adequacy (survivability and data quality) of the instrumentation systems installed in four experimental reflood test facilities is assessed. Signal conditioning electronics and sensor thermocouples functioned extremely well. For the first time, two-phase flow measurements were made in-core during a simulated LOCA. Because of the harsh environment and geometrical constraints, some sensor failures were considered likely; the number actually failing in service was within expectations. An exception to this record occurred in the Slab Core Test Facility -- Core 1. A chloride-ion stress corrosion problem destroyed signal cables at the vessel seal for most sensors. This problem was corrected by changing the sealant material at the vessel penetration in the subsequent facilities. Overall, the performance of the instrumentation was very satisfactory yielding valuable data during simulated LOCAs in refill-reflood test facilities.

NUREG/CR-3657: PRELIMINARY SCREENING OF FUEL CYCLE AND BY-PRODUCT MATERIAL LICENSES FOR EMERGENCY PLANNING. BENNETT,D.E.; RUNKLE,G.E.; ALPERT,D.J.; et al. Sandia National Laboratories. April 1985. 137pp. 8506060385. SAND84-0186. 30775:062.

This report summarizes work done for the U.S. Nuclear Regulatory Commission as part of a program considering the need for and appropriate level of emergency response planning at fuel cycle and by-product material facilities. The purpose is to (1) provide a base of technical information for identifying and ranking those facilities for which the need for emergency response planning and preparedness should be further considered, and (2) perform an initial screening of licenses issued by NRC. A data base containing the radionuclide possession limits for each license was developed. Dose estimates for a unit (1 curie) release of each of the radionuclides in the data base were calculated. To account for the variability in weather, distributions of doses were estimated for a full range of meteorological conditions. As requested by NRC, doses at the 99th percentile of the distribution were used. An initial screening analysis was performed for the approximately 9400+ licenses by comparing the estimated 99th percentile dose for a postulated release of a fraction of the licensed possession limit to the dose levels suggested in the Environmental Protection Agency's Protective Action Guides. Using relatively conservative assumptions in the screening analysis, all but at most a few hundred licenses were found to have estimated doses below the Protective Action Guide levels. The few hundred identified in this initial screening should be further evaluated using realistic assumptions and site-specific information to establish the need for, appropriate level and extent of, and potential effectiveness of emergency response planning and preparedness beyond that currently required.

NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY. ROHATGI,U.S.; PU,J.; SAHA,P.; et al. Brookhaven National Laboratory. April 1985. 98pp. 8505070497. BNL-NUREG-51750. 30211:005.

Several Oconee-1 overcooling transients that were computed by LANL and INEL using the latest versions of TRAC-PF1 and RELAP5/MOD1.5 codes have been reviewed by BNL. Three of these transients were selectd for detailed review as they either had the potential of challenging the integrity of the pressure vessel or highlighted the effect of code differences. These are (1) Main Steam Line Break (MSLB), (2) All Turbine Bypass Valves Stuck Open, and (3) 2-Inch Small Break LOCA.

NUREG/CR-3721 V01: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Hydrogen-Air Combustion Test Series 1 And 2 In The FITS Tank. ROLLER,S.F. Sandia National Laboratories. April 1985. 59pp. 8504170005. SAND83-2621/1. 29904:307.

Hydrogen combustion tests were performed in the Fully Instrumented Test Site (FITS) tank under the Hydrogen Behavior Program performed by Sandia National Laboratories under contract with the US Nuclear Regulatory Commission. Test series 1 and 2 examined the effects of a number of parameters on hydrogen-air combustion: the initial temperature and pressure of the gases, the effect of added steam or carbon dioxide as diluents, and the percent hydrogen in air. For tests in the range of 20% to 40% hydrogen in air, recorded peak pressures were equal to adiabatic, isochroic, complete combustion (AICC) values within an experimental error of 15%. This was contrary to the results of tests at a number of other facilities. The pre-ignition temperature had a strong effect on the peak pressure, while pre-ignition pressure in the range examined had no effect on combustion pressure ratios. Calculations showed that, although the effect of dynamic head on the peak pressure was a few percent or less, interactions of the wave preceding the flame front with the flame and with the vessel walls may be apparent in the experimental records.

NUREG/CR-3746 V02: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.Semiannual Progress Report,April 1984 - September 1984. LIPPINCOTT,E.P.; MCELROY,W.N. Hanford Engineering Development Laboratory. April 1985. 220pp. 8505070562. HEDL-TME 84-21. 30209:020.

This report describes progress made in the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) during FY84. The primary concern of this program is to improve, test, verify, and standardize the physics-dosimetry-metallurgy and associated reactor and damage analysis procedures and data used for predicting the integrated effects of neutron exposure to LWR-PVs and their support structures. These procedures and data are being recommended in a new and updated set of ASTM standards being prepared, tested, and verified by program participants. These standards, together with parts of the US Code of Federal Regulations and ASME codes, are needed and used for the assessment and control of the condition of LWR-PVs and their support structures during the 30- to 60-year lifetime of a nuclear power plant.

NUREG/CR-3746 V03: LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM.1984 Annual Report,October 1,1983 - September 30, 1984. MCELROY,W.N. Hanford Engineering Development Laboratory. April 1985. 110pp. 8505070543. HEDL-TME 84-31. 30208:270.

See NUREG/CR-3746,V02 abstract.

NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS. FOLEY,M.G.; KIMBALL,C.S.; MYERS,D.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 119pp. 8506140396. PNL-5064. 30933:275.

Under contract to the U.S. Nuclear Regulatory Commission, Pacific Northwest Laboratory has developed an approach for selecting and testing rock for its suitability and durability as armor for protecting decommissioned uranium mill tailings piles. A preliminary survey of the literature determined that existing techniques for testing rock durability were inadequate for evaluating long-term (100 years) applications. Suites of rock samples with common lithologies and documented durations of exposure to weathering were then collected and submitted to three-axis ultrasonic testing in an attempt to develop a more reliable testing technique. We found little correlation between the duration of weathering and ultrasound velocity or attenuation in the rock. Through further study, we determined that the best screening approach incorporates common geomorphologic field collection techniques and laboratory tests. Suites of samples with known durations of exposure to weathering can be subjected to wet abrasion and wetting-drying tests to screen local rock types and select those with the greatest potential durability. Furthermore, the expected decrease of rock mass with environmental stresses (e.g., flood impingement and diurnal wetting-drying cycles) can be estimated using this approach.

NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL. MCARTHUR,D.A.; MAST,P.K. Sandia National Laboratories. April 1985. 72pp. 8505160176. SAND84-0814. 30457:029.

The TRAN B-Series of experiments is being conducted at Sandia National Laboratories to investigate the characteristics of fuel removal and freezing through the upper axial blankets of an LMFBR during the transition phase of a hypothetical core disruptive accident. The second experiment in this series, TRAN B-2, was performed in July 1983. This experiment involved the injection of a mixture of 95% UO(2) and 5% stainless steel into a simple thick-walled steel cylindrical flow channel. The initial temperature of the steel channel was low, such that melting of the walls upon contact with the hot melt was not expected. Previous experiments under similar conditions but using pure UO(2) melts had shown stable crust growth and fairly long penetration distances. This experiment was intended to investigate whether those results were also applicable for the case of UO(2)/steel mixtures. The results of the TRAN B-2 experiment, consisting of data from online instrumentation and post-irradiation examination, suggest that low steel content fuel/steel mixtures behave very similarly to pure UO(2) melts.

NUREG/CR-3803: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY. LOYOLA,V.M.; WOMELSDUFF,J.E. Sandia National Laboratories. May 1985. 50pp. 8505230538. SAND84-0806. 30549:014.

We have studied the oxidation of zinc in a zinc-rich coating used in the nuclear power industry and have measured the rates of hydrogen generation from these coatings due to zinc oxidation at temperatures of up to 175 degrees centigrade. The results suggest that the real-time rates of hydrogen generation are considerably higher than previously believed. The higher rates measured in this study are probably due to differences in experimental methodologies between this and previous studies. In this study, the measurements were real-time measurements, as opposed to time-averaged values which are typically obtained. The results suggest, as have the results of other investigators, that the measured rates and reaction parameters may not be those of any specific reaction, but are instead the "effective" values of a series of complex systems operating together. However, the total quantity of hydrogen generated by this mechanism is significantly less than can be produced from other sources, e.g., steam: zirconium.

NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984. * Argonne National Laboratory. April 1985. 20pp. 8504250268. ANL-84-35 V04. 30032:091.

This quarterly progress report summarizes work done during the months of October-December 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 at 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.

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NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984. EDLER,S.K. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 31pp. 8506140415. PNL-5106-4. 30908:188.

This document summarizes work performed by Pacific Northwest Laboratory from October 1 through December 31, 1984, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Results from an instrumented fuel assembly irradiation program being performed at Halden, Norway, are reported. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic computer programs are providing best-estimate analyses for a variety of safety issues in light-water reactors. Severe fuel damage tests are being conducted in the NRU Reactor, Chalk River, Canada.

NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984. * Sandia National Laboratories. April 1985. 211pp. 8504160554. SAND84-1072. 29825:153.

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

NUREG/CR-3820 V03: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,July-September 1984. THOMPSON,S.L. Sandia National Laboratories. April 1985. 85pp. 8505160172. SAND84-1025. 30451:130.

The TRAC-PF1/MOD1 independent assessment program at Sandia National Laboratories is part of a multi-faceted effort sponsored by the Nuclear Regulatory Commission 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 TRAC-PF1/MOD1 code will be assessed against data from various integral and separate effects experimental test facilities, and the calculated results will also be compared with results from our previous RELAP5/MOD1 independent assessment analyses whenever possible.

NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT. SHACKELFORD,M. Argonne National Laboratory. * Sandia National Laboratories. April 1985. 361pp. 8505060534. SAND84-7139. 30192:018.

This report summarizes the survey work conducted by Argonne National Laboratory on the design and details of major penetrations in 48 nuclear power plants. The survey includes all containment types and materials in current use. It also includes details of all types of penetrations (except for electrical penetration assemblies and valves) and the seals and gaskets used in them. The report provides a test matrix for testing major penetrations and for testing seals and gaskets in order to evaluate their leakage potential under severe accident conditions.

NUREG/CR-3862: DEVELOPMENT OF TRANSIENT INITIATING EVENT FREQUENCIES FOR USE IN PROBABILISTIC RISK ASSESSMENTS. MACKOWIAK,D.P.; GENTILLON,C.D.; SMITH,K.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.),. May 1985. 278pp. 8506240069. EGG-2323. 31150:002.

Transient initiating event frequencies are an essential input to the analysis process of a nuclear power plant probabilistic risk assessment. These frequencies describe events causing or requiring scrams. This report documents an effort to validate and update from other sources a computer-based data file developed by the Electric Power Research Institute (EPRI) describing such events at 52 United States commercial nuclear power plants. Operating information from the United States Nuclear Regulatory Commission on 24 additional plants from their date of commercial operation has been combined with the EPRI data, and the entire data base has been updated to add 1980 through 1983 events for all 76 plants. The validity of the EPRI data and data analysis methodology and the adequacy of the EPRI transient categories are examined. New transient initiating event frequencies are derived from the expanded data base using the EPRI transient categories and data display methods. Upper bounds for these frequencies are also provided. Additional analyses explore changes in the dominant transients, changes in transient outage times and their impact on plant operation, and the effects of power level and scheduled scrams on transient event frequencies. A more rigorous data analysis methodology is developed to encourage further refinement of the transient initiating event frequencies derived herein.

NUREG/CR-3863: ASSESSMENT OF CLASS 1E PRESSURE TRANSMITTER RESPONSE WHEN SUBJECTED TO HARSH ENVIRONMENT SCREENING TESTS. FURGAL,D.T.; CRAFT,C.M.; SALAZAR,E.A. Sandia National Laboratories. April 1985. 194pp. 8506140052. SAND84-1264. 30907:289.

An experimental investigation into the performance of Class 1E electronic pressure transmitters exposed to environments within and beyond the design basis was conducted. Emphasis was placed on determining the instruments' failure and degradation modes in separate and simultaneous environmental exposures. Five unaged ITT Barton Model 763 pressure transmitters were tested and exposed to a unique environment. The response of the transmitters showed that temperature was the primary environmental stress affecting the tested transmitters' performance. Initial large errors that decrease with time-at-temperature were observed. The source of these errors is believed to be a common mode design weakness in the transmitters' calibration potentiometers. This weakness results from a dependency of material dielectric properties on temperature. The modification recommended by the manufacturer, although palliative in nature, did reduce this temperature-induced effect after the first few minutes of accident exposure. A potential second common failure mode which activates slowly with time-at-temperature was also identified. The operation of this failure mechanism is believed to be catalyzed by the presence of a lubricant used in the production of some potentiometers. The design of this transmitter proved to be exceptionally hard to radiation effects and there appeared to be no significant synergistic effects between radiation and temperature. The observed responses of the transmitters offer support for the position of IEEE 381-1977 which recommends that electronic modules aged to varying degrees of advanced life should be tested.

NUREG/CR-3872: DATA ACQUISITION AND CONTROL OF THE HSST SERIES V IRRADIATION EXPERIMENT AT THE ORR. MILLER,L.F.; HOBBS,R.W. Oak Ridge National Laboratory. April 1985. 97pp. 8505230574. ORNL/TM-9253. 30547:232.

Documentation relative to data acquisition and control for support of the HSST Series V Irradiation Experiment at the Oak Ridge Research (ORR) is included in this report. Part A describes the computer system hardware and real-time application support software, and Part B describes the temperature control methodology. Software that acquires data from analog input provides this information to the control algorithm software. Results from the control algorithm are, in turn, utilized by software which controls digital output hardware. Time intervals of execution, as well as sequencing of software modules, are controlled through commands to the operating system. Temperature data are recorded at one-hour intervals on computer printouts for documentation and immediate analysis and one magnetic media for permanent storage and subsequent analyses. Results from processing of data files show that the average temperatures at the 1/4T and 3/4T positions are maintained within 2.6 degrees centigrade of 288 degrees centigrade with associated standard deviations of less than 3 degrees centigrade. Average temperatures of the other thermocouples are maintained within 288 degrees centigrade plus or minus 12 degrees centigrade with standard deviations less than 3 degrees centigrade.

NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE. BOEGEL,A.J.; CHOCKIE,A.D.; HUENEFELD,J.C.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 120pp. 8507080183. PNL-5160. 31392:345.

This report presents the results of a project designed to compare and contrast Japanese and United States nuclear power plant operating experience, preventive maintenance/surveillance requirements, and organization and management practices relating to maintenance. Findings are based on information obtained on the November-December 1983 and November 1984 visits to Japan by the NRC and representatives of Battelle's Pacific Northwest Division, and on various documents obtained from the Japanese (primarily the Ministry of International Trade and Industry--MITI) during and subsequent to the visits. U.S. data sources included NUREG-0020 (Greybook) and plant technical specifications. The study shows that Japanese plants experienced far fewer manual shutdowns, manual scrams, automatic scrams, and reduced loads than U.S. plants and that their mean-time-between-event (MTBE), even when adjusted for differences in average plant availability, was approximately 10 times greater than the U.S. MTBE. The report also points out significant differences in the Japanese approach to preventive maintenance, and in the Japanese regulatory approach to maintenance, their management and organizational context for maintenance, and other socioeconomic factors that may affect the performance of maintenance.

NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1 - September 30,1984. BALL,S.J.; CLEVELAND,J.C.; HARRINGTON,R.M.; et al. Oak Ridge National Laboratory. April 1985. 28pp. 8505230519. ORNL/TM-9267/V3. 30549:063.

Modeling and code development work on the modular High-Temperature Gas-Cooled Reactor (HTGR) were continued. The longer-term heatup accident scenario in which cavity wall cooling is lost was also modeled. Sensitivity studies were run for a variety of parameter variations. Fission-product (FP) release and transport experiments were completed for several additional elements.

NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE. KOH,B.R.; KIM,S.H.; TALEYARKHAN,R.; et al. Oak Ridge National Laboratory. May 1985. 279pp. 8505160635. 30444:230.

This report summarizes improvements and modifications made in the MELRPI computer code. A major difference between this new, updated version of the code, called MELRPI.MOD2, and the one reported previously, concerns the inclusion of a model for the BWR emergency core cooling systems (ECCS). This model and its computer implementation, the ECCRPI subroutine, account for various emergency injection modes, for both intact and rubblized geometries. Other changes to MELRPI deal with an improved model for canister wall oxidation, rubble bed modeling, and numerical integration of system equations. A complete documentation of the entire MELRPI.MOD2 code is also given, including an input guide, list of subroutines, sample input/output and program listing.

NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,October-December 1984. STAHL,D.; MILLER,N.E. Battelle Memorial Institute, Columbus Laboratories. May 1985. 88pp. 8506060746. 30776:173.

Experiments for evaluating the glass-dissolution model are underway, and the procedure developed last quarter for dispersing RuO(2) in MCC 76-68 glass has been tested and proved to produce appropriate particle concentrations. Acetic and humic acids have been chosen to test the effect of natural organic acids on waste glass performance. In the overpack-corrosion effort, potentiodynamic polarization tests indicate that of the 15 chemical species tested, all but perchloate and hydrogen may effect stress-corrosion cracking behavior of carbon steel; several synergistic effects were also indicated. In slow strain rate studies, specimens tested in 0.0005 M FeCl(3) (a much lower chloride concentration than expected in groundwater) exhibited significant cracking over the temperature range 250-315 degrees centigrade. Pits were found to propagate readily, but slowly, in 1018 carbon steel exposed to aerated basalt groundwater at 90 degrees centigrade. The general-corrosion correlation was changed to incorporate a finite rate of film growth. Integral experiments are being prepared to provide information on combined-effects processes that may influence the long-term performance of the waste package.

NUREG/CR-3904: A COMPARISON OF UNCERTAINTY AND SENSITIVITY ANALYSIS TECHNIQUES FOR COMPUTER MODELS. IMAN,R.L.; HELTON,J.C. Sandia National Laboratories. May 1985. 118pp. 8506190020. SAND84-1461. 31019:001.

Uncertainty analysis and sensitivity analysis are important elements in the development and implementation of computer models for complex processes. Typically, there are many uncertainties associated with both the development and the application of such models. Understanding of these uncertainties and their causes is required to effectively interpret model behavior. Many different techniques have been proposed for performing uncertainty and sensitivity analyses. The objective of the present study is to compare several widely used techniques on three models having large uncertainties and varying degrees of complexity in order to highlight some of the problem areas that must be addressed in actual applications. The following approaches to uncertainty and sensitivity analysis are considered: (1) response surface methodology based on input determined from a fractional factorial design, (2) Latin hypercube sampling with and without regression analysis, and (3) differential analysis. These techniques are compared on the basis of (1) ease of implementation, (2) flexibility, (3) estimation of the cumulative distribution function of the output, and (4) adaptability to different methods of sensitivity analysis. With respect to these criteria, the technique using Latin hypercube sampling and regression analysis gives the best results overall. The models used in the comparisons are well documented, thus making it possible for researchers to make comparisons of other techniques with the results in this study.

NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide. GREEN,N.M.; MAYS,G.T.; JOHNSON,M.P.; et al. Oak Ridge National Laboratory. April 1985. 164pp. 8505070182. ORNL/NSIC-223. 30216:139.

Operating experience data from nuclear power plants are essential for safety and reliability analyses, especially analyses of trends and patterns. The licensee event reports (LERs) that are submitted to the Nuclear Regulatory Commission (NRC) by the nuclear power plant utilities contain much of this data. The NRC's Office of Analysis and Evaluation of Operational Data (AEOD) has developed, under contract with NOAC, a system for codifying the events reported in the LERs. The primary objective of the Sequence Coding and Search System (SCSS) is to reduce the descriptive text of the LERs to coded sequences that are 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. This four volume report documents and describes SCSS in detail. Volume 1 is a User's Guide for searching the SCSS database. Chapter 2 of this guide is a tutorial on retrieving, displaying, and analyzing LERs and provides hands-on experience in executing basic commands. Volume 2 contains all valid and acceptable codes used for searching and encoding the LER data. Volumes 3 and 4 provide a technical processor, new to SCSS, the information and methodology necessary to capture descriptive data from the LER and to codify that data into a structured format and serve as reference material for the more experienced technical processor, and contains information that is essential for the more advanced user who needs to be familiar with the intricate coding techniques in order to retrieve specific details in a sequence.

NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings. GALLAHER,R.B.; GUYMON,R.H.; MAYS,G.T.; et al. Oak Ridge National Laboratory. April 1985. 271pp. 8505070170. ORNL/NSIC-223. 30214:022.

See NUREG/CR-3905,V01 abstract.

NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings. GALLAHER,R.B.; GUYMON,R.H.; MAYS,G.T.; et al. Oak Ridge National Laboratory. April 1985. 271pp. 8505070170. ORNL/NSIC-223. 30214:022.

See NUREG/CR-3905,V01 abstract.

NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual. GALLAHER,R.B.; GUYMON,R.H.; MAYS,G.T.; et al. Oak Ridge National Laboratory. April 1985. 381pp. 8505070006. ORNL/NSIC-223. 30213:001.

See NUREG/CR-3905,V01 abstract.

NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual. GALLAHER,R.B.; GUYMON,R.H.; MAYS,G.T.; et al. Oak Ridge National Laboratory. April 1985. 347pp. 8505070184. ORNL/NSIC-223. 30212:001.

See NUREG/CR-3905,V01 abstract.

NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY. OPITZ,B.E.; DODSON,M.E.; SERNE,R.J. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 77pp. 8506190026. PNL-5179. 31019:119.

Laboratory experiments were performed to compare the effectiveness of limestone and hydrated lime for improving waste water quality through the neutralization of acidic uranium mill tailings liquor. The experiments were designed to assess the effects of three proposed mechanisms --- carbonate complexation, elevated pH and colloidal particle adsorption --- on the solubility of toxic contaminants found in a typical uranium mill waste solution. Of special interest were the effects of each of these possible mechanisms on the solution concentrations of trace metals such as Cd, Co, Mo, Zn and U after neutralization. Acidic untreated solid tailings from two mill sites and tailings neutralized with lime were leached with a laboratory-prepared ground water for several pore displacement volumes. Analyses performed on the column effluents indicate that prior neutralization results in a significant reduction in the concentration of all pH dependent constituents in the column effluents. In contrast, relatively high concentrations of several trace metals and macro ions were found in effluent solution from the untreated tailings columns.

NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL. CAMP,A.L.; WESTER,M.J.; DINGMAN,S.E. Sandia National Laboratories. April 1985. 325pp. 8504160098. SAND84-1522. 29832:026.

This report describes the features and use of HECTR Version 1.0. HECTR is a relatively fast-running, lumped-volume containment analysis computer program that is most useful for performing parametric studies. The main purpose of HECTR is to analyze nuclear reactor accidents involving the transport and combustion of hydrogen, but HECTR can also function as an experiment analysis tool and can solve a limited set of other types of containment problems. HECTR Version 1.0 has been particularly tailored to analyze accidents in ice-condenser PWR and Mark III BWR containments. HECTR is designed for flexibility and provides for user control of many important parameters, particularly those related to hydrogen combustion. Built-in correlations and default values of key parameters are also provided.

NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM. COLLINS,J.L.; OSBORNE,M.F.; LORENZ,R.A.; et al. Oak Ridge National Laboratory. April 1985. 73pp. 8504170679. ORNL/TM-9316. 29907:154.

Two control tests were conducted to study the behavior of CsI, CsOH, and Te in the experimental apparatus used to conduct fission product release tests with highly irradiated LWR fuel at ORNL. In this report the control tests are described, and the results are compared with those obtained for cesium, iodine, and tellurium in 26 tests of irradiated fuel and other tests using tracers. In good agreement with the LWR fuel tests, the CsI behavior in the control tests was similar to that observed for iodine in the fuel tests; iodine was released primarily as CsI rather than highly volatile molecular iodine. Cesium (not associated with CsI) behaved like CsOH in the LWR fuel tests. In both LWR fuel tests and the control tests, cesium hydroxide was observed to react with and be retained by zirconia ceramic surfaces in the temperature range 800 to 1200 degrees centigrade, probably forming cesium metazirconate (Cs(2)ZrO(3). In one of the control tests, cesium hydroxide reacted with tellurium in the gas phase and was collected as CsTe. Although the results are limited at this time, the indicated collected behavior of teelurium in the LWR fuel tests has been that of a telluride.

NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL. MCARTHUR,D.A.; MAST,P.K. Sandia National Laboratories. April 1985. 61pp. 8505030229. SAND84-1646. 30160:302.

The TRAN B series of experiments is being conducted at Sandia National Laboratories to investigate the characteristics of fuel removal/freezing through the upper axial blankets of an LMFBR during the transition phase of a hypothetical core disruptive accident. The third experiment in this series, TRAN B-3, was performed in February 1984, and the results are reported herein. This experiment involved the injection of molten UO(2) into an annular flow channel. Unlike the similar TRAN B-1 experiment, the initial steel wall temperature in B-3 was sufficiently high that instantaneous steel melting would occur upon contact with molten fuel. The earlier TRAN B-1 results had shown that fuel crusts are initially stable, both on the inside of a steel tube as well as on the outside of a steel rod, when no steel melting occurred. TRAN B-3 was designed to investigate this question of crust stability on surfaces of opposite curvature when surface melting did occur.

NUREG/CR-3953: THE USE OF MAG-1 SPECTACLES WITH POSITIVE AND NEGATIVE-PRESSURE RESPIRATORS. REED,K.A.; MOORE,T.O. Los Alamos Scientific Laboratory. May 1985. 41pp. 8506060798. LA-10229-MS. 30775:198.

Results of testing conducted at Los Alamos National Laboratory, Personnel Protection Studies Section, using MAG-1 spectacles in conjunction with positive- and negative-pressure full-facepiece respirators, are reported. The purpose of the three-phase study was to determine if the specially constructed strap of the MAG-1s affected the protection factors (PFs) of the respirators or the cylinder life of selected self-contained breathing apparatus (SCBA). The following respirators were tested with the MAG-1s: a) Phases I and II, positive-pressure full facepiece: Presur-Pak II SCBA (pressure-demand) Scottoramic facepiece, MSA 401 Air Mask Ultravue facepiece (medium), Survivair pressure-demand SCBA/silicone full facepiece, MSA powered air-purifying respirator/Ultravue facepiece (medium); b) Phase III, negative-pressure full facepiece: MSA Ultravue (small, medium, large), MSA Ultra-twin (small, medium, large), Norton Series 7600 (one size only). Statistical analysis and review of the test data from Phases I and II indicated little, if any, variation with and without the MAG-1s with most protection factors greater than 10,000. Test data also indicated little, if any, difference in the cylinder life with and without the MAG-1s, except the Scott Presur-Pak II SCBA used with the Scottoramic facepiece. Statistical analysis of the quantitative fit test data indicated no difference in PFs for the negative-pressure devices for the Ultravue negative-pressure respirator, but a significance at the 0.05 and 0.01 levels for the Ultra-twin and Norton full facepieces, respectively.

NUREG/CR-3977: RELAP5 THERMAL-HYDRAULIC ANALYSES OF PRESSURIZED THERMAL SHOCK SEQUENCES FOR H.B. ROBINSON UNIT 2 PRESSURIZED WATER REACTOR. FLETCHER,C.D.; BOLANDER,M.A.; WATERMAN,M.E.; et al. EG&G, Inc. April 1985. 233pp. 8506140624. EGG-2341. 30934:352.

Thermal-hydraulic analyses of fourteen hypothetical pressurized thermal shock (PTS) scenarios for the H. B. Robinson, Unit 2 pressurized water reactor were performed at the Idaho National Engineering Laboratory (INEL) using the RELAP5 computer code. The scenarios, which were developed at Oak Ridge National Laboratory (ORNL), contain significant conservatisms concerning equipment failures, operator actions, or both. The results of the thermal-hydraulic analyses presented here, along with additional analyses of multidimensional and fracture mechanics effects, will be utilized by ORNL, integrator of the PTS study, to assist the U.S. Nuclear Regulatory Commission in resolving the pressurized thermal shock unresolved safety issue.

NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984. REST,J. Argonne National Laboratory. May 1985. 51pp. 8507050429. ANL-84-61. 31371:306.

This progress report summarizes the Argonne National Laboratory work performed during July, August, and September 1984 on water reactor safety problems related to fuel and fuel cladding materials. The research and development areas covered are Transient Fuel Response and Fission Product Release and Clad Properties for Code Verification.

NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS. RANKIN,W.L.; RIDEOUT,T.B.; TRIGGS,T.J.; et al. Battelle Human Affairs Research Centers. June 1985. 102pp. 8507050435. PNL-5158. 31371:001.

This report presents the design philosophy and associated functional criteria and design principles for developing advanced computerized annunciator systems for use in the control rooms of nuclear power plants. The scope of the work includes advanced system recommendations that could be incorporated into operating nuclear power plants. The information contained in this report was obtained from a review of the revelant computer and visual display terminal literature, from site visits to advanced control rooms in the nuclear power and related industries, and from a study of technical reports on computerized control rooms. This report should assist the staff in development of a regulatory position regarding the design of computerized control room annunciator systems. The guidance in this report is consistent with that provided in NUREG-0700.

NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984. SHACK,W.J. Argonne National Laboratory. April 1985. 92pp. 8504220361. ANL-84-60. 29946:069.

This progress report summarizes the Argonne National Laboratory work performed during April, May, and June 1984 on water reactor safety problems related to out-of-core materials. The research and development areas covered are Environmentally Assisted Cracking in Light Water Reactors, Long-Term Embrittlement of Cast Duplex Stainless Steels in LWR Systems, and Nondestructive Evaluation and Leak Detection.

NUREG/CR-4003: CLOSEOUT OF IE BULLETIN 79-04:INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. June 1985. 29pp. 8507030702. IE-143. 31314:067.

IE Bulletin 79-04 was issued March 30, 1979 as a result of reports from three facilities that Velan Engineering Corporation had provided incorrect weights for swing check valves. The reason for concern was the possibility that these incorrect weights had been used in analyses of Seismic Category I piping systems at a large number of plants. Evaluation of utility responses and NRC/IE inspection reports shows that the bulletin can be closed out for 117 (92%) of the 127 current facilities on the basis of specific criteria. Followup items for the remaining 10 current facilities are proposed for use by NRC/IE. Incorrect weights reported for valves other than Velan swing check valves are identified as Remaining Areas of Concern. This bulletin has served its purpose and can be closed out. A final check of valve weights will be made per later IE Bulletin 79-14 on seismic analyses for as-built safety-related piping systems.

NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. April 1985. 44pp. 8505010088. IEB-79-25. 30112:260.

Robinson 2 submitted LER 78-29 December 19, 1978 to report sticking of a normally energized Westinghouse BFD relay. After reviewing this problem, Westinghouse issued Service Letter TS-E-412 to recommend that BFD relays in safety-related systems be replaced with later NBFD relays. During installation and testing of the new NBFD relays, Robinson 2 found some with marginal or unsatisfactory armature overtravel. Because of this new problem, Westinghouse issued Technical Bulletin NSD-TB-79-05 to recommend prompt checking of certain models of NBFD relays and returning those with inadequate overtravel for rework or replacement. IE Bulletin 79-25, with extracts of the Westinghouse service letter and technical bulletin enclosed, was issued November 2, 1979 to require responses and specific actions by all licensees and holders of construction permits with respect to BFD and NBFD relays in safety-related systems. Evaluation of utility responses and NRC/IE inspection reports indicates that the bulletin can be closed out for 121 (94%) of the 129 current facilities on the basis of specific criteria. Proposed followup items for the remaining 8 facilities are presented in Appendix C for use by NRC/IE. Because followup of corrective action is ensured, IE Bulletin 79-25 is considered closed.

NUREG/CR-4005: CLOSEOUT OF IE BULLETIN 80-12:DECAY HEAT REMOVAL SYSTEM OPERABILITY. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. June 1985. 59pp. 8507030675. IE-146. 31314:098.

On April 19, 1980, decay heat removal (DHR) capability was lost at Davis-Besse 1 for approximately two and one-half hours in a refueling mode. Typically for that mode, many systems and components were out of service for maintenance and testing or were deactivated to preclude inadvertent actuation. IE Bulletin 80-12 was issued May 8, 1980 for action by licensees of operating pressurized water reactors (PWRs); it was issued for information to nuclear power facilities other than operating PWRs. The intent of the bulletin was to improve nuclear plant safety by reducing the likelihood of losing DHR capability in PWRs, especially when some DHR components are unavailable because of maintenance activities during refueling and cold shutdown modes of operation. A related NRR Generic Letter was issued June 11, 1980 to licensees of operating PWRs, requesting amendment of technical specifications to ensure long-term maintenance of DHR capability. Evaluation of utility responses and NRC/IE inspection reports indicates that the bulletin can be closed out per specific criteria for 33 (75%) of the 44 affected facilities.

NUREG/CR-4009: HUMAN RELIABILITY DATA BANK.Evaluation Results. COMER,M.K.; DONOVAN,M.D.; GADDY,C.D.; et al. General Physics Corp. April 1985. 75pp. 8505070505. SAND85-7150. 30210:295.

The U.S. Nuclear Regulatory Commission and Sandia National Laboratories conducted a three-year research program to develop a human reliability data bank specifically tailored to support human reliability analysis segments of probabilistic risk assessments for nuclear power plants. Previous efforts of the program include a review of existing human performance data banks (NUREG/CR-2744, Vol. 1) and a concept and system description (NUREG/CR-2744, Vol. 2). Subsequent to the system description, a detailed specification for implementing the data bank was developed. An evaluation of this specification was conducted and is described in this report. The evaluation consisted primarily of an Operability Demonstration and Evaluation of the data processing procedures and personnel required, and a Data Review Demonstration and Evaluation involving members of the potential user population. The conclusions of this study were used to modify and improve the detailed implementation specification. The revised specification is published as NUREG/CR-4010, and it describes all the necessary materials, personnel, procedures, definitions, and data taxonomies to implement the data bank.

NUREG/CR-4010: SPECIFICATION OF A HUMAN RELIABILITY DATA BANK FOR CONDUCTING HRA SEGMENTS OF PRAS FOR NUCLEAR POWER PLANTS. COMER,M.K.; DONOVAN,M.D. General Physics Corp. * Sandia National Laboratories. April 1985. 400pp. 8505060499. SAND85-7151. 30190:331.

This document specifies the personnel, resources, policies, and procedures for implementing and operating a human reliability data bank specifically tailored to support human reliability analysis (HRA) segments of probabilistic risk assessments (PRAs) for nuclear power plants. The report concludes a three-year research program conducted by the U.S. Nuclear Regulatory Commission and Sandia National Laboratories. Previous efforts of the program include a review of existing human performance data banks (NUREG/CR-2744, Vol. 1), a concept and system description (NUREG/CR-2744, Vol. 2), and a peer evaluation study (NUREG/CR-4009). This report specifies the administrative organization of the data bank functional groups and their proposed interaction. Detailed procedures are included that specify how to process submitted data, how to classify and store the data, and how to combine similar data when appropriate. Included within the report is the skeleton data manual, which is a prototype, hardcopy, data manual that would be used to disseminate data to the user population. It describes the data taxonomy, procedures for retrieving data of interest, and presents several sample data retrieval problems. Definitions are supplied for all technical and behavioral terms used in the taxonomic structure. As its name implies, the skeleton data manual embodies the data manual in structure, but is void of emperical data.

NUREG/CR-4015: EFFECT OF STAINLESS STEEL WELD OVERLAY CLADDING ON THE STRUCTURAL INTEGRITY OF FLAWED STEEL PLATES IN BENDING SERIES 1. CORWIN,W.R.; ROBINSON,G.C.; NANSTAD,R.K.; et al. Oak Ridge National Laboratory. April 1985. 103pp. 8505230524. ORNL/TM-9390. 30549:090.

The HSST stainless steel cladding evaluations were initiated to study the interaction of stainless cladding with flaws initiated in and propagating in base metal of reactor pressure vessels. A complicating factor in understanding the role of stainless cladding in this setting is its toughness as a function of radiation dose and fabrication process. The initial phase of this study addressed this question by testing the response of specimens clad with single-wire submerged-arc weld overlay in varying toughness levels. The tests completed under the initial phase of this study indicate that cladding of moderate Charpy toughness has only limited capabilities to stop running cracks. This was a limited set of experiments, and the upper and lower bounds of cracks arrest capabilities are not yet determined. The fabrication techniques employed for this first series of tests have resulted in conditions that have prevented close control of the stress state at pop-in of the hydrogen-charged EB welds. Consequently, the arrest toughness of the stainless cladding was not closely bounded. General modifications are proposed for incorporation in a second series of tests to provide more comprehensive conditions of testing and materials of interest, to eliminate some undesirable test conditions that existed in the first series, and to provide an improved geometry for analytical interpretations.

NUREG/CR-4031 V02: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutronics Calculations." WILLIAMS,L.; REMEC,I.; KAM,F.B. Oak Ridge National Laboratory. May 1985. 42pp. 8505170010. ORNL/TM-9423/V2. 30484:001.

A series of calculations has been completed to compute dosimeter activation in the Oak Ridge Research Reactor (ORR) HSST Simulator Experiment. A comparison of calculated and experimental results shows that calculations underpredict dosimeter activities on the average of about 15%. The C/E values indicated the now familiar tendency to become lower as more iron is penetrated. The dosimeters in front of the simulator (and behind the thermal shield) typically have C/E values of 0.9-1.0, while those at the back of the simulator have values of 0.7-0.85. The calculations also show shifted axial distribution relative to the measurements: the C/E values near the bottom of the core are about 15% to 20% higher than those near the top. This is probably due to a discrepancy in the axial power distribution computed with VIPOR/VENTURE. The axial distribution of fuel obtained from the correlation in VIPOR could possibly be causing the power shift, although this speculation has not been verified. There is also, perhaps, a slight tendency for the C/E values to increase along the x (the coordinate parallel to the reactor face) traverse from the reactor centerline to a point near the core boundary; however, this variation is much less than in the axial direction. The results obtained with different dosimeters appear generally to be reasonably consistent, except that the (46)Ti C/E values seem to be consistently lower than for the other dosimeters. The systematic nature of the discrepancies in these calculations will be adjusted by the least squares procedure to produce an accurate representation of the flux distribution.

NUREG/CR-4031 V03: NEUTRON SPECTRAL CHARACTERIZATION FOR THE FIFTH HEAVY SECTION STEEL TECHNOLOGY (HSST) IRRADIATION SERIES. "Neutron Exposure Parameters." REMEC,I.; STALLMANN,F.W.; KAM,F.B. Oak Ridge National Laboratory. May 1985. 31pp. 8505160647. ORNL/TM-9423/V3. 30457:181.

This is the third volume of a three-volume report which describes the simulator experiments of the fifth series of HSST irradiation experiments which are sponsored by the U.S. Nuclear Regulatory Commission (NRC). The purpose of these three volumes is to document, in detail, the experimental and calculational methodology which will be used in determining the neutron-exposure parameters for the fifth and subsequent series of HSST irradiation experiments at ORNL. The methodolgy was also used in the fourth series of HSST irradiation experiments and represents the current state-of-the-art procedures developed in the Light Water Reactor Pressure Vessel Simulation Project which is a part of NRC's Surveillance Dosimetry Improvement Program. The neutron-exposure data from the fifth and subsequent series will be documented in a loose-leaf NUREG/CR report as the data become available. In this volume, the best estimates for the values and spatial distribution of fluence rate (0) (E 1.0 MeV), fluence rate (0) (E 0.1 MeV), and displacements per atom per second (dpa/s) are determined using LSL-M2, a least squares logarithmic spectrum adjustment procedure with input values taken from dosimetry data from Vol. 1 and neutronics calculations from Vol. 2. These estimates have an overall uncertainty of less than 20% relative standard deviation. This volume is essential to the metallurgist for defining the irradiation strategy to meet his objective(s).

NUREG/CR-4033: THE ROLE OF PERSONAL AIR SAMPLING IN RADIATION SAFETY PROGRAMS AND RESULTS OF A LABORATORY EVALUATION OF PERSONAL AIR-SAMPLING EQUIPMENT. RITTER,P.D.; HUNTSMAN,B.L.; NOVICK,V.J.; et al. EG&G, Inc. May 1985. 80pp. 8505230534. EGG-2352. 30549:256.

Recommended applications for personal air sampling in NRC licensee radiation protecting programs are presented. The recommendations are based on performance tests of currently available samplers, a review of research and regulatory literature, and a survey of current licensee air-sampling programs. The performance tests show that personal air samplers are available which can provide a reliable, convenient means for breathing-zone sampling of workers in practically any work environment which might be encountered in the licensee industries. The research literature emphasized that estimates of an individual's exposure may be greatly underestimated if based on general area air samples, as is common practice in current licensee programs, due to the unpredictable variability of airborne-activity concentrations in the worksite. A conclusion which may be drawn from the literature and from experimental results is that in most situations, personal air sampling (or more generally, true breathing-zone sampling) is the only means to reliably estimate the airborne activity to which a worker has been exposed (MPC h). Research concerning the applicability of air-sampling measurements for estimating intake, uptake, and internal dose was also reviewed.

NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983. MOHR,P.B.; MONT,M.E.; SCHWARTZ,M.W. Lawrence Livermore National Laboratory. April 1985. 52pp. 8505070560. UCRL-53587. 30210:169.

A rear-end collision occurred between a passenger automobile and a luggage trailer carrying 84 packages, 76 of which contained radiopharmaceuticals, on U.S. Highway 84 near Brookhaven, Mississippi on the afternoon of December 3, 1983. The purpose of this report is to document the mechanical circumstances of the accident, confirm the nature and quantity of radioactive materials involved, and assess the nature of the physical environment to which the packages were exposed and the response of the packages. The report consists of three major sections. The first deals with the nature and circumstances of the accident and findings of fact. The second gives an accounting and description of the materials involved and the consequences of their exposure. The third gives an assessment and analysis of the mechanisms of damage and the conclusions which may be drawn from the investigation.

NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS. GERTMAN,D.I.; JENKINS,J.P.; HANEY,L.N.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.),. May 1985. 68pp. 8507020109. EGG-2387. 31313:236.

An extensive review of literature on individual and group performance and decisionmaking under psychological stress was conducted and summarized. Specific stress-related variables relevant to reactor operation were pinpointed and incorporated in an experiment to assess the performance of reactor operators under psychological stress. The decisionmaking performance of 24 reactor operators under differing levels of workload, conflicting information, and detail of available written procedures was assessed in terms of selecting immediate, subsequent, and nonapplicable actions in response to 12 emergency scenarios resulting from a severe seismic event at a pressurized water reactor. Specific personality characteristics of the operators suggested by the literature to be related to performance under stress were assessed and correlated to decisionmaking under stress.

NUREG/CR-4044: TRAC-PF1 LOCA CALCULATIONS USING FINE-NODE AND COARSE-NODE INPUT MODELS. DOBRANICH,D.; BUXTON,L.D.; WONG,C.N. Sandia National Laboratories. May 1985. 86pp. 8506190042. SAND84-2305. 31015:116.

TRAC-PF1 calculations of a 200% cold-leg break LOCA have been completed for a UHI plant using both fine-node (with 776 mesh cells) and coarse-node (with 320 mesh cells) input models. This study was performed to determine the effect of noding on predicted results and on computer running time. It was found that the overall sequence of events and the important trends of the transient were predicted to be nearly the same with both the fine-node and coarse-node models. There were differences in the time-dependent behavior of the cold-leg accumulator injection, and the predicted PCT for the coarse-node calculation was about 75 K less than that of the fine-node calculation. The higher PCT of the fine-node calculation is attributed primarily to three-dimensional flow effects in the core. The complete (steady state plus transient) coarse-node calculation required 13.5 hours of CYBER 76 computer time compared to 68.3 hours for the fine node calculation, yielding an overall factor of five decrease in running time. Thus, we conclude that for any large break LOCA analyses in which only the overall trends are of concern, the loss of accuracy resulting from use of such a coarse-node model will normally be inconsequential compared to the savings in resources that are realized. However, if the objective of the analyses is the investigation of the effects of multi-dimensional flows on clad temperatures, then a detailed model is required.

NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS. SAARI,L.M.; MELBER,B.D.; WHITE,A.S.; et al. Battelle Human Affairs Research Centers. April 1985. 77pp. 8505010277. PNL-5303. 30114:352.

This report identifies job-related educational qualifications for the nuclear power plant licensed operator positions of reactor operator (RO), senior reactor operator (SRO), and shift supervisor (SS). The extent to which college engineering curriculum covers job-related academic knowledge was assessed. The approach used involved systematically comparing college engineering programs to knowledge needed on the job by having subject matter experts in the filed of general and nuclear engineering curriculum: (1) assess the coverage of specific academic knowledge identified by a job analysis as necessary for licensed operators in existing college engineering degree programs, and (2) make judgments concerning levels of formal engineering education necessary for application of knowledge on the job, based on job samples from a job analysis of activities under selected normal and emergency operating sequences. The major conclusions of the report are: a substantial amount (approximately 2/3) of job-related academic knowledge is covered in college engineering curriculum; college engineering curriculum provides considerable material beyond that identified as necessary for licensed operators; higher level operator positions (SS relative to SRO, SRO relative to RO) were judged as needing higher levels of education to perform the job.

NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS. KULAK,R.F. Argonne National Laboratory. * Sandia National Laboratories. April 1985. 109pp. 8505060514. ANL-84-41. 30193:019.

This report summarizes the analyses work performed by Argonne National Laboratory on three representative nuclear power plant penetrations for severe accident loads beyond the design basis conditions. These include analyses of an equipment hatch for a steel containment, a BWR-Mark II drywell head and a bellows connection. The objectives of the analyses were to identify the methodology required to simulate the response of the penetrations and determine their leakage potential under severe accident loads. This report provides the details of the analytical methodology used and the results obtained from the analyses.

NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR

POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling. HENAGER,C.H.; DALING,P.M.; JOHNSON,K.I. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1985. 61pp. 8504220375. PNL-5300. 29946:202.

This report describes the safety and economic consequences of bivalve fouling in raw-water systems at nuclear power plants. The report lists events that could cause a normal fouling situation to become more critical and describes scenarios in which bivalve fouling could cause unsafe or unwanted conditions such as transients and shutdowns. Several fouling events that have occurred at various nuclear plants are briefly reviewed, and recommendations are made to aid in the detection and control of bivalve fouling.

NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA. HESTER,O.V.; GENTILLON,C.D. EG&G, Inc. April 1985. 215pp. 8505280415. EGG-2362. 30601:072.

This report presents an overview of the 1981 Sequence Coding and Search System (SCSS) data base that contains nuclear power plant operational data derived from Licensee Event Reports (LERs) submitted to the United States Nuclear Regulatory Commission. Both overall event reporting and events related to specific components, subsystems, systems, and personnel are discussed. At all of these levels of information, software is used to generate count data for contingency tables. Contingency table analysis is the main tool for the trend and pattern analysis. The tables primarily focus on faults associated with various components and other items of interest across different plants. The abstracts and other SCSS information on the LERs accounting for unusual counts in the tables were examined to gain insights from the events.

NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES. A Review. BEEDLOW,P.A.; PARKER,G.B. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 29pp. 8505280093. PNL-5323. 30604:247.

This report reviews design considerations for protective covers for uranium mill tailings impoundments. The role of protective covers in tailings containment systems is discussed. Factors affecting the long-term stabilization of tailings (erosion, biotic intrusion, and soil moisture) are summarized. Basic elements to be considered in design of all uranium tailings covers are presented, and then quantitative techniques for designing site-specific covers are reviewed.

NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION. BEEDLOW,P.A.; CLINE,J.F.; FREEMAN,H.D.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 48pp. 8507030712. PNL-5324. 31318:250.

This report provides methods and procedures that can be used to verify compliance with Environmental Protection Agency (EPA) engineering standards for uranium mill tailings disposal sites. EPA standards for radon emissions, long-term isolation, and protection of water quality are discussed. Tailings isolation technologies are reviewed. Information the licensee needs to provide for the regulating agency to determine compliance is presented, as is the actual compliance criteria.

NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT. KITTMER,C.A.; WENSEL,R.G.; RHODES,D.B.; et al. Atomic Energy of Canada, Ltd. April 1985. 93pp. 8506170667. EGG-2365. 30979:290.

A testing program designed to provide fundamental information pertaining to the behavior of reactor coolant pump (RCP) shaft seals during a postulated nuclear power plant station blackout has been completed. The test plan was developed by EG&G Idaho personnel at the Idaho National Engineering Laboratory (INEL) and performed at the Chalk River Nuclear Laboratory, Ontario, Canada, under auspices of the U.S. Nuclear Regulatory Commission (NRC). One seal assembly, utilizing both hydrodynamic and hydrostatic types of seals, was modeled and tested. Extrusion tests were conducted to determine if seal materials could withstand predicted temperatures and pressures. A taper-face seal model was tested for seal stability under conditions when leaking water flashes to steam across the seal face. Test information was then used as the basis for a station blackout analysis. Test results indicate a potential problem with an elastomer material used for O-rings by a pump vendor; that vendor is considering a change in material specification. Test results also indicate a need for further research on the generic issue of RCP seal integrity and its possible consideration for designation as an unresolved safety issue.

NUREG/CR-4079: ANALYTIC STUDIES PERTAINING TO STEAM GENERATOR TUBE RUPTURE ACCIDENTS. KASHIWA,B.A.; MJOLSNESS,R.C. Los Alamos Scientific Laboratory. April 1985. 88pp. 8506060372. LA-10307-MS. 30773:271.

A study of thermal-hydraulic phenomena of possible steam generator tube rupture (SGTR) accidents leads to the conclusions that (1) flashing will not occur upstream of the tube rupture, so that the flow will be resistance limited rather than choked, (2) there is considerable potential for discharging the primary fluid in the form of micron-sized droplets, particularly when the fluid discharges into a vapor cavity surrounding the tube rupture, and (3) that the surrounding of the rupture site by water rather than vapor may be a means for preventing the formation of micron-sized droplets. The presence or absence of micron-sized droplets is considered to be a key issue for the damage assessment of SGTR accidents because they are currently thought to be the most likely route for radioactive iodine to be released to the atmosphere.

NUREG/CR-4084: DRY SPENT FUEL STORAGE TEST PLAN FOR DESTRUCTIVE FUEL ROD EXAMINATIONS. OLSEN,C.S. EG&G, Inc. April 1985. 41pp. 8504220369. EGG-2367. 29946:164.

A testing program using eight commercial pressurized water reactor and boiling water reactor spent fuel rods was conducted to investigate their long-term stability under a variety of possible dry storage conditions. The objective of this report is to provide the Nuclear Regulatory Commission with information to confirm or establish dry spent fuel storage licensing positions for long-term, low-temperature (250 degrees centigrade) spent fuel rod behavior during dry storage and for radioactive contamination that might occur with spallation of cladding crud. Six of the eight commercial fuel rods will be destructively examined. This report presents the test plan for the destructive examinations.

NUREG/CR-4086: TENSILE PROPERTIES OF IRRADIATED NUCLEAR GRADE PRESSURE VESSEL WELDS FOR THE THIRD HSST IRRADIATION SERIES. MCGOWAN,J.J. Oak Ridge National Laboratory. May 1985. 23pp. 8505160629. ORNL/TM-9477. 30439:288.

The Heavy Section Steel Technology (HSST) Program conducted a series of experiments to investigate the effect of neutron irradiation on the fracture toughness of nuclear pressure vessel materials. Four welds of A 508 class 2 steel were examined in this Third HSST Irradiation Series. The welds were fabricated according to "early" (pre-1972) lightwater reactor weld practice (i.e., copper-coated electrodes). As part of this study, tensile properties were measured after irradiation to 2 to 10 x 10(22) neutrons/m(2) (E 1 MeV) at temperatures between 250 and 290 degrees centigrade. Strength properties of all four welds increased with exposure to irradiation. Yield strength was more sensitive to irradiation than was ultimate strength. Tensile ductility was not affected significantly by exposure to irradiation.

NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS. HARTLEY,J.N.; GLISSMEYER,J.A.; HILL,O.F. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 69pp. 8507080192. PNL-5338. 31393:099.

Pacific Northwest Laboratory, under contract to the U.S. Nuclear Regulatory Commission (NRC), identified and evaluated methods for estimating radioactive and toxic particulate and gaseous airborne releases from uranium milling operations. Such methods need to be standardized so that all uranium mills can provide adequate data for NRC evaluation of potential environmental impacts and of compliance with 10 CFR 20, 40 CFR 190, and the National Environmental Policy Act. The general method for calculating source terms is to multiply together a normalized emission rate, contaminant content, emission control factor, and processing rate for each process being evaluated. This report describes the sources of airborne releases (ore storage area, ore crushing and grinding, ore processing, yellowcake production, and tailings impoundment) and the calculational procedures for estimating radioactive and toxic source terms. Example calculations are provided.

NUREG/CR-4091: THE EFFECT OF ALTERNATIVE AGING AND ACCIDENT SIMULATIONS ON POLYMER PROPERTIES. BUSTARD,L.D.; CHENION,J.; CARLIN,F.; et al. Sandia National Laboratories. June 1985. 177pp. 8507050391. SAND84-2291. 31373:139.

The influence of accident irradiation, steam, and chemical spray exposures on the behavior of twenty-three age-preconditioned polymer sample sets (twenty-one different materials) has been investigated. The test program varied the following conditions: 1. Accident simulations of irradiation and thermodynamic (steam and chemical spray) conditions were performed both sequentially and simultaneously; 2. Accident thermodynamic (steam and chemical spray) exposures were performed both with and without air present during the exposures; 3. Sequential accident irradiations were performed both at 28 degrees centigrade and 70 degrees centigrade; 4. Age preconditioning was performed both sequentially and simultaneously; 5. Sequential aging irradiations were performed both at 27 degrees centigrade and 70 degrees centigrade; and 6. Sequential aging exposures were performed using two sequences: (1) thermal followed by irradiation and (2) irradiation followed by thermal. This report presents both general trends applicable to a majority of the tested materials as well as specific results for each polymer. The data base consists of ultimate tensile properties at the completion of the accident exposure for three XLPO and XLPE, five EPR and EPDM, two CSPE (HYPALON), one CPE, one VAMAC, one polydiallylphtalate, and one PPS material. Report bend test results at completion of the accident exposures for two TEFZEL materials and permanent set after compression results for three EPR, one VAMAC, one BUNA N, one SILICONE, and one VITON material are also presented.

NUREG/CR-4092: ORNL CHARACTERIZATION OF HEAVY-SECTION STEEL TECHNOLOGY PROGRAM PLATES 01,02,AND 03. STELZMAN,W.J.; BERGGREN,R.G.; JONES,T.N. Oak Ridge National Laboratory. April 1985. 176pp. 8506060376. ORNL/TM-9491. 30773:101.

Charpy V-notch impact, tensile, and drop-weight data are presented for three 305-mm-thick (12-in.) A 533 grade B class 1 steel plates. The effects of specimen size and orientation were examined as well as the variation of properties between different plate locations and depths. Some observations based on data obtained from an instrumented Charpy testing machine are also presented.

NUREG/CR-4093: SAFETY/SAFEGUARDS INTERACTIONS DURING SAFETY-RELATED EMERGENCIES AT NUCLEAR POWER REACTOR FACILITIES. MOUL,D.A.; PILGRIM,M.K.; SCHWEIZER,R.L.; et al. Brookhaven National Laboratory. June 1985. 400pp. 8507020094. BNL-NUREG-51848. 31308:138.

This report contains an analysis of the safety/safeguards interactions that could occur during safety-related emergencies at licensed nuclear power reactors, and the extent to which these interactions are addressed in existing or proposed NRC guidance. The safety/safeguards interaction during a series of postulated emergencies was systematically examined to identify any potential performance deficiencies or conflicts between the Operations (safety) and Security (safeguards) organizations. This examination included the impacts of coordination with offsite emergency response personnel. Duties, responsibilities, optimal methods, and procedural actions inherent in these interactions were explored.

NUREG/CR-4095: TEST SERIES 2:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E EXIDE FHC-19 BATTERY CELLS. BONZON,L.L.; HENTE,D.B. Sandia National Laboratories. April 1985. 190pp. 8507020389. 31307:332.

This report, the second in a test series of an extensive seismic research program, covers the testing of 10-year old lead-calcium Exide FHC-19 cells from the Calvert Cliffs Nuclear Power Station operated by the Baltimore Gas and Electric Company. The Exide cells were tested in two configurations using a triaxial shake table: single-cell tests, both rigidly and loosely mounted; and multicell (three-cell) tests, mounted in a typical battery rack. A total of six electrically active cells was used in the two different cell configurations. None of the six cells failed in the first stage tests during the actual seismic test up to the 1.5 g ZPAs imposed. Subsequent discharge capacity tests showed, however, that only two of the cells could deliver the accepted standard of 80% of their rated electrical capacity for 3 hours. When two of the same cells were exposed to the second stage, higher g-level tests, both cells again provided instantaneous uninterrupted power. Subsequent capacity tests showed both of these cells to have capacities well below the accepted standard of 80%. Four of the cells were disassembled for examination and metallurgical analyses. The examination showed the active material on the positive plates was hard and cracked and that the positive bus bar material was corroded and brittle.

NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS. BONZON,L.L.; HENTE,D.B. Sandia National Laboratories. April 1985. 170pp. 8507020413. SAND84-2629. 31309:159.

This report, the third in a test series of an extensive seismic research program, covers the testing of 10-year old lead-calcium C&D LCU-13 cells from the North Anna Nuclear Power Station operated by the Virginia Electric and Power Company. The C&D cells were tested in two configurations using a triaxial shake table: single-cell tests, both rigidly and loosely mounted; and multicell (three-cell) tests, mounted in a typical battery rack. A total of seven electrically active cells was used in the two different cell configurations. None of the seven cells failed in the first stage tests during the actual seismic test up to the 1.5 g ZPAS imposed. Subsequent discharge capacity tests showed that while these cells suffered some loss of discharge capacity, all cells could deliver the accepted standard of 80% of their rated electrical capacity for 3 hours. When two of the same cells were exposed to the second stage, higher g-level tests, both cells again provided instantaneous uninterrupted power. Subsequent capacity tests showed both of these cells to have capacities well below the accepted standard of 80%. Four of the cells were disassembled for examination and metallurgical analyses. The examination showed that all plates and separators were in very good condition.

NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS. BONZON,L.L.; HENTE,D.B. Sandia National Laboratories. April 1985. 119pp. 8507020430. SAND84-2630. 31309:321.

This report, the fourth in a test series of an extensive seismic research program, covers the testing of 27-year old lead antimony Exide EMP-13 cells from the recently decommissioned Shippingport Atomic Power Station. The Exide cells were tested in two configurations using a trixial shake table: single-cell tests, rigidly mounted; and multicell (five-cell) tests, mounted in a typical battery rack. A total of nine electrically active cells was used in the two different cell configurations. None of the nine cells failed during the actual seismic tests when a range of ZPAs up to 1.5 g was imposed. Subsequent discharge capacity tests of five of the cells showed, however, that none of the cells could deliver the accepted standard of 80% of their rated electrical capacity for 3 hours. In fact, none of the 5 cells could deliver more than 33% capacity. Two of the seismically tested cells and one untested, low capacity well were disassembled for examination and metallurgical analyses. The inspection showed the cells to be in poor condition. The negative plates in the vicinity of the bus connections were extremely weak, the positive buses were corroded and brittle, negative and positive active material utilization was extremely uneven, and corrosion products littered the cells.

NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS. CLINE,J.E.; NOYCE,J.R.; COE,L.J.; et al. Science Applications International Corp., (formerly Science Applications Inc.). April 1985. 615pp. 8505100034. 30271:001.

The 10 CFR Part 61 waste classification system includes several nuclides which are difficult to assay without expensive radiochemical methods. In order for waste generators to classify wastes practically, NRC Staff has recommended the use of correlation factors to scale the difficult-to-measure nuclides with nuclides which can be measured more easily (i.e., gamma emitters such as (60)Co or (137)Cs). In this study, Science Applications International Corporation (SAIC) performed complete radiochemical assays for all the 10 CFR Part 61 waste classification nuclides on over 100 samples. These data, along with almost 800 other samples in the SAIC data base, were used to assess the validity of correlation factors suggested for use for nuclear power plant wastes. Specific generic correlation factors are recommended with other approaches to correlate nuclides for which generic scaling factors are not defensible.

NUREG/CR-4105: AN ASSESSMENT OF THERMAL GRADIENT TUBE RESULTS FROM THE HI SERIES OF FISSION PRODUCT RELEASE TESTS. NORWOOD,K.S. Oak Ridge National Laboratory. May 1985. 64pp. 8505230531. ORNL/TM-9506. 30549:192.

A thermal gradient tube was used to analyze fission product vapors released from fuel heated in the HI test series. Complete deposition profiles were obtained for Cs, I, Ag, and Sb. The cesium profiles were complex and probably were dominated by Cs-S-O compounds formed by release of sulfur from furnace ceramics. The iodine profiles were simple, indicating that more than 99.5% of the released iodine behaved as a single nonvolatile species, probably CsI. Mass transfer coefficients for this species onto platinum were estimated to be 1.9 to 5.8 cm/s. Silver was probably released in elemental form, condensed to an aerosol, and captured by filters. Antimony was released as the element and reacted rapidly with platinum (or gold) as it deposited. Antimony profiles were calculated a priori with some success. A method was developed for isolating tellurium platinum and mixed fission products in a form suitable for neutron activation analysis. The platinum samples were completely dissolved in acid (HC1/HNO(3), and the tellurium was precipitated on selenium carrier by reduction. Finally, tellurium was loaded onto Dowex 1X-4 ion-exchange resin for activation and analysis. Tellurium recovery was 88%, and the theoretical sensitivity was 3 ng.

NUREG/CR-4106: PRESSURIZED-THERMAL-SHOCK TEST OF 6-IN.-THICK PRESSURE VESSELS.PTSE-1:Investigation Of Warm Prestressing And Upper-Shelf Arrest. BRYAN,R.H.; BASS,B.R.; BOLT,S.E.; et al. Oak Ridge National Laboratory. April 1985. 288pp. 8506060815. ORNL-6135. 30770:024.

The first pressurized-thermal-shock test of a 148-mm-thick steel pressure vessel with a 1-m-long flaw was performed to investigate fracture behavior of a vessel under conditions relevant to a flawed nuclear reactor pressure vessel during an overcooling accident. The objectives were to observe crack arrest and stability on the ductile upper shelf and effects of warm prestressing on crack initiation. Three coordinated pressure and thermal transients were imposed on the vessel, which was preheated to 290 degrees centigrade. Two episodes of crack propagation and arrest occurred. The thermal transients were induced by coolant at -29 degrees centigrade to 15 degrees centigrade. Pressure transients were as high as 94.4 MPa. The experimental objectives were attained. The inhibiting effects of warm prestressing were definitely demonstrated. Crack propagation was nearly pure cleavage, and arrest at 30K above the onset of the Charpy upper shelf was experienced in a positive K(I) gradient and with K(I) 300 MPa square root m. Fracture-mechanics analysis of brittle fracture based on small-specimen toughness measurements was reasonably accurate. Flaw evaluation by procedures of the ASME Boiler and Pressure Vessel Code conservatively predicted vessel failure, which did not occur.

NUREG/CR-4109: TRAC-PF1 ANALYSES OF POTENTIAL PRESSURIZED-THERMAL SHOCK TRANSIENTS AT CALVERT CLIFFS/UNIT 1,A Combustion Engineering PWR. SPRIGGS,G.D.; KOENIG,J.E.; SMITH,R.C. Los Alamos Scientific Laboratory. April 1985. 355pp. 8504220382. LA-10321-MS. 29953:331.

Los Alamos National Laboratory participated in a program to assess the risk of a pressurized thermal shock (PTS) to the reactor vessel during a postulated overcooling transient in a pressurized water reactor (PWR). We provided the thermal-hydraulic analyses of three general accident categories: steamline breaks, runaway-feedwater transients, and small-break loss-of-coolant accidents. These postulated accidents included multiple operator and equipment failures. Results were provided to Oak Ridge National Laboratory (ORNL) who plan to determine the probability of vessel failure and accident occurrence for an overall assessment of PTS risk. Our study was performed for a Combustion Engineering PWR, Calvert Cliffs/Unit 1, using the Transient Reactor Analysis Code (TRAC-PF1). We found the results of the analyses to be very sensitive to the initial conditions of the plant. If the plant was initially at hot-zero power (compared to full power), the decay heat was much less, which made it possible for the same accident initiator to produce significantly lower downcomer temperatures. However, routine operator actions may reduce the consequences of any of these simulated accidents if the prescribed pressure-temperature relationships are followed.

NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE. KERSCHNER,H.F.; ETTINGER,H.J.; DEFIELD,J.D.; et al. Los Alamos Scientific Laboratory. April 1985. 62pp. 8504300121. LA-9985-MS. 30070:220.

Respirators fitted with high-efficiency particulate (HEPA) cartridge filters are designed to remove dust, fumes, mists, and airborne particulate radionuclides. If these filters are to be reused, a Quality Assurance (QA) program must be established to ensure that filter efficiency remains greater than 99.97 per cent. The standard method for performing QA testing is to challenge the filter with a thermally generated aerosol of 0.3-m-diam di-2-ethylhexyl phthalate (DEHP). Because of potential toxicological and other problems associated with the use of monodisperse DEHP, an investigation to study measured filter efficiencies on an HEPA respirator filter population, using several recommended replacement aerosols, has been conducted. Aerosols compared in this study were thermally generated di-2-ethylexyl sebecate (DEHS), thermally generated DEHP, air-jet-generated DEHS, and air-jet-generated salt (NaC1). The study also focused on determining compatibility for parallel use of aerosols generated for respirator-fit testing for use in QA filter testing. Results indicate that a polydisperse air-jet-generated aerosol of DEHS can substitute for thermally DEHP as a method of providing QA testing of HEPA respirator filters and that equipment used in the study designed for respirator quantitative-fit testing can easily be modified to perform this function.

NUREG/CR-4111: A COMPARATIVE STUDY OF HEPA FILTER EFFICIENCIES WHEN CHALLENGED WITH THERMAL- AND AIR-JET-GENERATED DI-2-ETHYLHEXYL SEBECATE,DI-2-ETHYLHEXYL PHTHALATE,AND SODIUM CHLORIDE. KERSCHNER,H.F.; ETTINGER,H.J.; DEFIELD,J.D.; et al. Los Alamos Scientific Laboratory. April 1985. 62pp. 8504300121. LA-9985-MS. 30070:220.

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NUREG/CR-4114: VALENCE EFFECTS ON THE SORPTION OF NUCLIDES ON ROCKS AND MINERALS.II. MEYER,R.E.; ARNOLD,W.D.; CASE,F.I. Oak Ridge National Laboratory. May 1985. 53pp. 8505210576. ORNL-6137. 30521:328.

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

NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES. DENHAM,D.H.; RATHBUN,L.A.; BARNES,M.G.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 31pp. 8507030713. PNL-5361. 31314:036.

Decommissioning of a uranium processing site requires radiological surveys to: 1) identify buildings, equipment, and open land areas that require cleanup; 2) verify that cleanup operations have been successful; and 3) provide a record of the radiological condition of the site following cleanup. This report describes the instruments, measurements, quality assurance, and statistical procedures that can be used to perform pre-and post-cleanup surveys. The procedures described include: 1) gamma-radiation exposure-rate measurements using micro-R-meters, 2) beta-gamma measurements using Geiger-Mueller tubes, 3) wipe tests for surface contamination, and 4) soil analyses for (226)Ra and other (238)U daughters. Locations likely to have (226)Ra concentrations that exceed standards can be identified by gamma-radiation exposure rate measurements. Samples of soil or other material from location showing elevated exposure rates can then be analyzed for (226)Ra to determine the boundaries of areas that exceed standards. Beta-gamma measurements and wipe sample analyses can be used to determine whether uranium concentrations exceed standards for either fixed or removable contamination.

NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984. KUPPERMAN,D.S.; CLAYTOR,T.N.; PRINE,D.W. Argonne National Laboratory. April 1985. 39pp. 8505060509. ANL-85-5. 30190:287.

This progress report summarizes work performed by the Argonne National Laboratory and GARD, Inc. (Division of Chamberlain Mfg. Corp.) as subcontractor on NDE of stainless steel and on-line monitoring of LWRs during the twelve months from October 1983 to September 1984.

NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984. KUPPERMAN,D.S.; CLAYTOR,T.N.; PRINE,D.W. Argonne National Laboratory. April 1985. 39pp. 8505060509. ANL-85-5. 30190:287.

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NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION. CAMERON,J.R.; DEWERD,L.A.; GOETSCH,S.J.; et al. Wisconsin, Univ. of, Madison, WI. May 1985. 38pp. 8506030101. 30707:220.

The research described in this report was performed to investigate the feasibility of a mailed dosimetry system as an alternative method of achieving the goals of the present U.S. Nuclear Regulatory Commission requirement that ionization chambers used for calibration of cobalt-60 teletherapy units be calibrated every two years. Both thermoluminescent dosimeters (TLD's) and a diode detector unit was used in this study. A total of 20 hospitals in the states of Illinois, Iowa and Wisconsin participated in a program in which this dosimetry package was sent to each institution on three separate occasions. The physicist, physician or chief technologist was asked to deliver 1.50 Gray (150 rads) to the device, assuming the device was equivalent in radiation adsorption characteristics to human tissue. A treatment field size of 10cm by 10cm was chosen and the institution was requested to use their clinical source-to-surface distance. The accuracy of the beam localization as indicated by the coincidence of the light field with the radiation field was measured as well. The criterion for accuracy of dose delivery was plus minus 3.0mm. Only two hospitals during the course of the study had both a disagreement of more than 3mm between the light field and the radiation field. It is recommended that such a mailed dosimetry package be considered as an alternative to the present NRC requirement for biennial calibration of ionization chambers used to calibrate cobalt-60 teletherapy sources.

NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES. CLAIRBORNE,H.C.; CROFF,A.G.; GRIESS,J.C.; et al. Oak Ridge National Laboratory. May 1985. 130pp. 8506130358. ORNL/TM-9522. 30867:350.

This document provides specifications for a model/methodology and approach that could be employed in determining post-closure repository environmental parameters relevant to high-level waste package performance for the Basalt Waste Isolation Project (BWIP). Guidance is provided on (1) the identity of the relevant repository environmental parameters (groundwater characteristics, temperature, radiation, and pressure), (2) the models/methodologies employed to determine the parameters, and (3) the input data base for the model/methodologies. Supporting studies included are (1) an analysis of potential waste package failure modes leading to identification of the relevant repository environmental parameters, (2) an evaluation of the credible range of the repository environmental parameters for the BWIP situation, and (3) a summary review of existing models/methodologies currently employed in determining repository environmental parameters relevant to waste package performance.

NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL. MCGUIRE,M.V.; WALSH,M.E.; MORISSEAU,D.S.; et al. Battelle Human Affairs Research Centers. May 1985. 87pp. 8505100041. PNL-5381. 30270:125.

This report describes a mailed survey of operations personnel at a sample of commercial nuclear power plants. The survey was conducted for the U.S. Nuclear Regulatory Commission (NRC) as part of the Operator Feedback Project. The survey sought to collect information on topics of concern to the NRC and to assess the feasibility of a mailed survey on an information collection mechanism. Participants in the survey were 520 personnel at 26 nuclear power plants representing all five NRC regions. The individual participants completed and returned by mail a ten-page questionnaire. This contained questions about operations crew practices, including work and shift schedules, operations shift crew staffing, the shift technical advisor position, respondents' own backgrounds, the questionnaire, and other information-collection techniques. Results of the survey offer some insight on operations crew practices at the plants participating in the survey. Survey results also suggest that the mailed survey is an information-collection technique that can be used effectively to obtain feedback for the NRC from operations personnel.

NUREG/CR-4140: DOMINANT ACCIDENT SEQUENCES IN OCONEE-1 PRESSURIZED WATER REACTOR. DEARING,J.F.; HENNINGER,R.J.; NASSERSHARIF,B.; et al. Los Alamos Scientific Laboratory. April 1985. 112pp. 8506240647. LA-10351-MS. 31149:230.

A set of dominant accident sequences in the Oconee-1 pressurized water reactor was selected using probabilistic risk analysis methods. Because some accident scenarios were similar, a subset of four accident sequences was selected to be analyzed with the Transient Reactor Analysis Code (TRAC) to further our insights into similar types of accidents. The sequences selected were loss-of-feedwater, small-small break loss-of-coolant, loss-of-feedwater-initiated transient without scram, and interfacing systems loss-of-coolant accidents. The normal plant response and the impact of equipment availability and potential operator actions were also examined. Strategies were developed for operator actions not covered in existing emergency operator actions not covered in existing emergency operator guidelines and were tested using TRAC simulations to evaluate their effectiveness in preventing core uncovery and maintaining core cooling.

NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS. DAVIS,T.; SHAFAGHI,A.; KURTZ,R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1985. 69pp. 8504220341. PNL-5389. 29946:001.

This report presents a method for focusing additional research on aging phenomena that affects nuclear power plant components. Specifically, the method ranks components using a risk aging sensitivity measure that describes the change in risk due to changes in component failure rate. Describing the aging phenomena and the resulting time-dependent component failure rate changes is beyond the scope of this study. The applications use average components unavailability equations currently employed in PRAs to calculate the risk aging sensitivity. A more exact calculation is possible by using unavailability equations that include the time-dependent characteristics of component failure rates; however, these time-dependent characteristics are not well-known. The risk aging sensitivity measure presented here is, therefore, segregated from these time-dependent effects and addresses only the time-independent portion of aging phenomena. The results identify the component types that show the most potential for risk change due to aging phenomena. Future research on the time-dependent portion of aging phenomena for these component types is needed to completely describe the risk impact due to component aging.

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NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER. BUCKALEW,W.H.; WYANT,F.J. Sandia National Laboratories. May 1985. 92pp. 8506240313. SAND85-0209. 31157:001.

Dow Corning Sylgard 170 Silicone Elastomer has been investigated to characterize its response to accelerated thermal aging, radiation exposure, and its behavior under applied compressive forces. Sylgard 170 response to accelerated thermal aging suggests the material properties are not particularly age dependent. Radiation exposures, however, produce significant, monotonic changes in both elongation and hardness with increasing absorbed radiation dose. Elastomer response to an applied compressive force was strongly dependent on environment temperature and degree of material confinement. Variations in temperature produced large changes in compressive forces applied to confined samples. Attempts to mitigate force fluctuations by means of pressure relief paths resulted in total loss of the applied compressive force. Thus, seal applications employing this elastomer in Class 1E equipment required to function during or following an accident should consider the potential loss of compressive force from long-term aging and potential LOCA-temperature transient conditions.

NUREG/CR-4149: ULTIMATE PRESSURE CAPACITY OF REINFORCED AND PRESTRESSED CONCRETE CONTAINMENT. SHARMA,S.; WANG,Y.K.; REICH,M. Brookhaven National Laboratory. May 1985. 95pp. 8506130467. BNL-NUREG-51857. 30901:328.

This report presents the results of an in-depth evaluation of current modeling techniques and analysis procedures for determining ultimate pressure capacity of reinforced and prestressed concrete containments. The material models used for describing the nonlinear material behavior of concrete and steel are reviewed in detail. Special attention is focused on post-cracking behavior of concrete which controls one of the containment failure modes, i.e., the shear failure. Various finite element idealizations utilized for containment analysis are reviewed. The effects of major assumptions pertaining to containment geometry, basemat restraint, finite element mesh, rebar locations and orientations, are evaluated. Finally, failure analyses of two selected reinforced and prestressed concrete containments are performed and results are compared with those presented in the literature.

NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS. DOBRANICH,D. Sandia National Laboratories. April 1985. 42pp. 8505060503. SAND85-0172. 30190:241.

The TRAC-PF1/MOD1 independent assessment project at Sandia is part of an overall effort funded by the NRC to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. As part of this effort, calculations for some of the Northwestern University perforated-plate CCFL tests have been performed. Two input models were constructed to represent the rectangular test channel: a two-dimensional model and a faster-running one-dimensional model. The results of both models indicate that for high water flow rates, with the water injected vertically above a perforated plate, TRAC overpredicts the steam flow rate necessary for complete weeping (CCFL). However, for flow conditions more typical of PWR transients, TRAC provides a reasonable prediction of weeping.

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NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING. LEWELLEN,W.S.; VARMA,A.K.; SHENG,Y.P. Aeronautical Research Associates of Princeton. April 1985. 79pp. 8504250168. ARAP NO. 504. 30032:015.

The current generation of dispersion models contains very simple parameterizations of deposition processes. The analysis here looks at the physical mechanisims governing these processes in an attempt to see if more valid parameterizations are available and what level of uncertainty is involved in either these simple parameterizations or any more advanced parameterization. The report is composed of three parts. The first, on dry deposition model sensitivity, provides an estimate of the uncertainty existing in current estimates of the deposition velocity due to uncertainties in independent variables such as meteorological stability, particle size, surface chemical reactivity and canopy structure. The range of uncertainty estimated for an appropriate dry deposition velocity for a plume generated by a nuclear power plant accident is three orders of magnitude. The second part discusses the uncertainties involved in precipitation scavenging rates for effluents resulting from a nuclear reactor accident. The conclusion is that major uncertainties are involved both as a result of the natural variability of the atmospheric precipitation process and due to our incomplete understanding of the underlying process. The third part involves a review of the important problems associated with modeling the interaction between the atmosphere and a forest. It gives an indication of the magnitude of the problem involved in modeling dry deposition in such environments.

NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS. LEWELLEN,W.S.; SYKES,R.I.; PARKER,S.F. Aeronautical Research Associates of Princeton. May 1985. 202pp. 8505310421. ARAP NO. 505. 30670:001.

Results from simulations by 12 different dispersion models are compared with observations from an extensive field experiment at the Idaho National Engineering Laboratory in July, 1981. Comparisons were made based on hourly ground-level SF(6) samples, out to approximately 10 km from the 46 m release tower, both during and following 7 different 8-hour releases. Comparisons are also made for total integrated doses collected out to approximately 40 km. Within the limited range appropriate for Class A models this data comparison shows that neither the puff models or the transport and diffusion models agree with the data any better then the simple Gaussian plume models. The puff and transport and diffusion models do show a slight edge in performance in comparison with the total dose over the extended range appropriate for Class B models. The best model results for the hourly samples show approximately 40% calculated within a factor of two when a 15 degree uncertainty in plume position is permitted, and it is assumed that higher data samples may occur at stations between the actual sample sites. This is increased to 60% for the 12 hour integrated dose and 70% for the total integrated dose. None of the models reproduce the observed patchy dose patterns. This patchiness appears to be consistent with the inherent uncertainty associated with time averaged plume observations.

NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS. MOELLER,M.P.; STOETZEL,G.A.; MUNSON,L.H. Battelle Memorial Institute, Pacific Northwest Laboratories. April 1985. 65pp. 8505070439. PNL-5404. 30214:293.

This report organizes existing data on occupational radiation exposure experience for consideration in the safety goal evaluation program. The report includes a review of occupational radiation exposures incurred by workers at commercial U.S. nuclear power plants. In addition, occupational radiation exposure information is presented for work performed at commercial U.S. nuclear power plants to meet regulatory actions and required backfits. This information identifies specific operations performed as part of these requirements. Where possible, actual radiation exposure histories are provided. A brief history of radiation dose limits and a review of the biological and health effects attributable to radiation exposure is included to provide a perspective on the development of radiation protection regulations.

NUREG/CR-4161 V01: CRITICAL PARAMETERS FOR A HIGH-LEVEL WASTE REPOSITORY. Volume 1:Basalt. BINNALL,E.P.; WOLLENBERG,H.A.; BENSON,S.M.; et al. Lawrence Berkeley Laboratory. May 1985. 95PP. 8506060810. UCID-20092. 30769:292.

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

NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES. WILLIFORD,R.E.; LANNING,D.D.; BEYER,C.E. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 277pp. 8506060567. PNL-5354. 30771:002.

This report describes the development, benchmarking and results of a computer code designed to permit comparison of BWR and PWR fuel rod failure behaviors during postulated reactor off-normal events such as control rod withdrawal errors. The code is called GT2F, and was developed from the GAPCON-THERMAL-2 code by the addition of new models for calculating transient temperatures, fission gas release, mechanical interaction between fuel and cladding, and Zircaloy cladding fracture behavior. Results indicate that for the conservatively severe overpower transient scenarios assumed, a full length commercial BWR fuel rod has a failure probability between 1% and 4.5% at 27 MWd/kgM when the transient begins from high operating power. A full length commercial PWR fuel rod has a failure probability between 2% and 11% at 28 MWd/kgM when the transient begins from low power. Failure probabilities are substantially smaller at lower burnups and for less extreme transient conditions.

NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM. OSTMEYER,R.M. Sandia National Laboratories. June 1985. 33pp. 8507050426. SAND84-1404. 31371:175.

A new code system is being developed for use in assessment of nuclear reactor accident risks. The code system, termed MELCOR, will treat thermal-hydraulic and fission product behavior jointly. A part of its treatment of thermal-hydraulic processes, the code system will evaluate decay heating from fission product inventories contained within the reactor core debris and compartments that are defined for the reactor system and containment. A simple approach to treating radionuclide decay heating is proposed for use in MELCOR. The proposed approach uses a table-lookup to estimate element decay powers as a function of time after reactor shutdown (start of accident). Decay power for each element in a compartment of the reactor system is found by multiplying the mass of the element in the compartment by the element's decay-heat rate per unit mass which is a function of time after reactor scram. The approach assumes that daughter products are transported along with the parent radionuclide during the accident. The validity of this assumption is discussed. In addition, methods for apportioning the decay energy between the walls and the gases in a compartment are also discussed. The proposed approach is based on SANDIA-ORIGEN calculations for a 3412 MWt PWR and a 3578 MWt BWR.

NUREG/CR-4176: EMISSION CONTROL TECHNOLOGY AND QUALITY ASSURANCE NEEDS AT URANIUM MILLING FACILITIES.Includes Supporting Methods For Testing,Operating,And Maintaining Air Pollution Control Devices. LUDWICK,J.D. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 55pp. 8507030684. PNL-5386. 31318:311.

Pacific Northwest Laboratory, under contract to the U.S. Nuclear Regulatory Commission, conducted an investigation of particulate emission control devices for application to process exhausts at uranium milling facilities. The scope of this investigation included devices now in use, as well as those devices that have potential application for milling sites. This report presents the results of the study. Emission control devices are categorized and described, including high-efficiency and moderate-efficiency devices as well as other (some novel) devices useful in specific situations. Preoperational considerations discussed include selecting devices, instrumentation, and testing programs. Operational and maintenance considerations related to dry and wet removal processes are described. Quality assurance documents and topics are also discussed.

NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants. DISALVO,R.; LEONARD,M.; MANAHAN,M.; et al. Battelle Memorial Institute, Columbus Laboratories. May 1985. 105pp. 8506130369. BMI-2123. 30868:176.

Accident management is examined from several related perspectives. The emphasis is on the role of the operating crew and the technical support provided to them before, during, and after an accident. The relationship among accident management, risk management and emergency management is examined. The roles played by industry, regulation, and research are reviewed. Finally, the results of viewing accident management from these various perspectives are reflected in the articulation of issues and some proposals for their resolution.

NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime. WREATHALL,J.; LEONARD,M.; DISALVO,R. Battelle Memorial Institute, Columbus Laboratories. May 1985. 75pp. 8506130132. BMI-2123. 30867:108.

This study examines the feasibility and value/impact of extending emergency operating procedures into the severe accident regime. It reviews the types of knowledge needed to develop such procedures and the applicability of existing regulatory review criteria. A method is developed and illustrated in two cases. This study concludes that it is feasible to develop procedures for operators to mitigate the consequences of accidents progressing past the onset of core damage. A preliminary value/impact assessment indicates a significant likelihood of there being an overall net positive benefit of developing mitigative procedures. A phased program has been proposed. First a pilot study should develop the application of the methods used in this feasibility study and provide more precise information for a detailed value-impact assessment. Based on the results of the pilot study, extension to a greater population of plants may be justified.

NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS. JAROSS,R.A.; MULCAHEY,T.P.; KOEHL,E.R. Argonne National Laboratory. April 1985. 118pp. 8504180201. ANL-84-102. 29921:264.

The state-of-the-art of solid-state motor controllers (SSMCs) is assessed in terms of use, probability of Class 1E qualification, failure rate experience, and reliability prediction. Surveys of commercial availability, nuclear and nonnuclear electric utility experience, and architect-engineering use were made relative to the suitability of SSMCs for nuclear service. Reasons for the limited use of SSMCs in nuclear plants are given. Available failure rate data are meager, and are augmented by data on other solid-state power electronic devices that are shown to have subcomponents similar to those found in SSMCs. In addition to large nonnuclear solid-state adjustable-speed motor drives, the reliability of nuclear plant inverter systems and high-voltage solid-state DC transmission line converters is assessed. Class 1E environmental qualification experience with nuclear plant converter/inverters and battery chargers in shown to be directly applicable to SSMCs. No problems are expected in qualifying them. Actual reliability predictions of two typical commercial SSMCs are given, together with predictions of improvements possible with use of high-quality parts and manufacturing procedures.

NUREG/CR-4181: LEACHABILITY OF RADIONUCLIDES FROM CEMENT SOLIDIFIED WASTE FORMS PRODUCED AT OPERATING NUCLEAR POWER REACTORS. CRONEY,S.T. EG&G Idaho, Inc. (subs. of EG&G, Inc.),. April 1985. 130pp. 8505070173. EGG-2355. 30217:161.

Different sized samples of cement-solidified liquid wastes were collected from two nuclear power plants, a pressurized water reactor (PWR) and a boiling water reactor (BWR), to correlate radionuclide leaching from small and full sized waste forms. Diffusion-based model analysis of measured radionuclide leach data from small samples and full size samples indicated that leach data from small samples could be used to determine leachability indexes for full size waste form. The leachability indexes for cesium, strontium, and cobalt isotopes were determined for waste samples from both nuclear plants according to models used in ANS 16.1. The leachability indexes for the PWR samples were 6.4 for cesium, 7.1 for strontium, and 10.4 for cobalt. The leachability indexes for BWR samples were 6.5, 8.6, and 11.1 for cesium, strontium, and cobalt, respectively.

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NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL. WONG,K.S.; BROWN,J. California, State of. May 1985. 22pp. 8506060807. 30770:316.

This report is an account of offshore radioactive material activities and was prepared to provide information about their safe use in the marine environments beyond California's jurisdiction. The report supplies the essential information called for and (a) identifies licensees with radioactive nuclide utilization programs, (b) describes the licensees' work stations, (c) identifies and/or describes radionuclide, quantities and their applications, and (d) describes the radiation safety concerns and existing methods for their resolution. Finally, three offshore sites were inspected in a typical compliance manner and the findings reported. Enclosed photographs of the work stations, during source and equipment use, illustrate conditions and the licensees' operations. It is concluded from observations during onsite visits to these unusual work environments, that periodic onsite compliance inspections are necessary to assure radiation protection for all concerned.

NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES. BOLAND,J.F.; BROOKSHIRE,R.L.; DANIELSON,W.F.; et al. Argonne National Laboratory. April 1985. 225pp. 8505100194. ANL-85-13. 30268:213.

This document presents the results of a survey of Licensee control-room-habitability practices. The survey is part of a comprehensive program plan instituted in August 1983 by the NRC to respond to ongoing questions from the Advisory Committtee on Reactor Safeguards (ACRS). The emphasis of this survey was to determine by field review the control-room habitability practices at three different plants, one of which is still under construction and scheduled to receive an operating license in 1986. The other two plants are currently operating, having received operating licenses in the mid-1970's and early 1980's. The major finding of this survey is that despite the fact that the latest control-room-habitability systems have become large and more complex than earlier systems surveyed, the latest systems do not appear to be functionally superior. The major recommendation of this report is to consolidate into a single NRC document, based upon a comprehensive systems engineering approach, the pertinent criteria for control-room-habitability design.

NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES. FAYER,M.J.; CONBERE,W. Battelle Memorial Institute, Pacific Northwest Laboratories. May 1985. 56pp. 8506190031. PNL-5421. 31017:324.

The computer code TRUNC was used to analyze three aspects of uranium mill tailings dewatering: the coupling of consolidation and fluid flow, drainage design, and cover load. One-dimensional simulations of the effects of consolidation on fluid flow within a tailings pile of either slimes or a sand/slimes mix showed that drainage flux was greater for a consolidating system early in the simulation. After days 1,400 and 160 of the simulations for the slimes and sand/slimes mix, respectively, however, the fluxes for the nonconsolidating systems were greater. In the sand/slimes mix, the nonconsolidating system had a cumulative flux by day 5,000 that was 93% of that of the consolidating system. At the same time, in the slimes tailings piles the nonconsolidating system had a cumultive flux of only 34% of that of the consolidating system, which indicates that consolidation and fluid flow should not be decoupled for the slimes. Two-dimensional simulations of an actual tailings pile drainage design showed that a sand blanket drain increased the rate of drainage and settlement. The sand blanket drain also significantly reduced differential settlement across the pile. This indicates that the use of a sand blanket drain could enable earlier placement of the cover system after tailings emplacement. In simulations of covered and uncovered tailings piles, nearly the same quantity of water was removed from each, but drainage occurred much more slowly without the cover; hence, surface settlement was slower when the tailings pile was not covered.

NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984. MCCRAY,J.G.; NOWATZKI,E.A.; ARMSTRONG,G.; et al. Arizona, Univ. of, Tucson, AZ. May 1985. 219pp. 8506240003. 31149:012.

This is the final report on a three year study to evaluate trench cap designs, trench construction and trench loading by accelerating the creation of void space to simulate waste degradation in order to apply stress conditions on the trench in a relative short period of time. Eight trenches were initially constructed and instrumented, four in a semi-arid region and four in a more humid mountainous region. After the first year the semi-arid site was abandoned due to cap failures. A new trench incorporating an improved soil slab design with a wick was constructed at the humid site. Conclusions from these experiments are: 1. Controlled compaction is not sufficient to mitigate long term surface subsidence. 2. Single sheet geotextile reinforcement is not adequate trench cap reinforcement. 3. Geotextile wrapped soil slab attenuates surface subsidence and surface water infiltration. 4. A steel-reinforced soil-cement slab appears to meet the requirements necessary for long term stability. 5. If the crown and cap remain stable so does the trench. 6. Aliphatic tracers performed well and dye type of tracers poorly. 7. Tracers are feasible and effective as a trench monitoring tool. 8. Narrow designed trenches improve trench cap stability. This report recommends a design for enhanced isolation disposal trench providing improved monitoring capabilities.

NUREG/CR-4196: OVERVIEW OF TRAC-BD1 (VERSION 12) ASSESSMENT STUDIES. KULLBERG,C.M. EG&G Idaho, Inc. (subs. of EG&G, Inc.),. April 1985. 55pp. 8506060796. EGG-2382. 30771:279.

A series of simulations were performed at Idaho National Engineering Laboratory to continue the advancement of Boiling Water Reactor (BWR) safety research, with the TRAC-BD1 (Version 12) computer code. The principal motivation for performing these simulations was to assess the code's capability to calculate Loss-of-Coolant Accident (LOCA) related phenomena. The results of a number of TRAC-BD1 (Version 12) simulations, which cover a broad range of conditions during different types of LOCA scenarios, are summarized in this document. Selected comparisons between calculated and measured results are presented. Conclusions derived from those comparisons are given.

NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES. BURKE,R.P.; BLOND,R.M. Sandia National Laboratories. June 1985. 50pp. 8507020415. SAND85-0634. 31313:301.

This study presents the results of analyses performed as part of the two-year evaluation program for the NRC safety goals. Analyses are performed to demonstrate the sensitivities of the quantitative design objective calculations to changes in input parameters and assumptions. Results are presented which show the influence of parameter changes on the health risk quantitative design objectives and on cost-benefit calculations. The alternative design objective risk measures are compared with alternative measures of the health impacts of LWR accidents. The results of this study provide background information and input to be used in the NRC staff evaluation of the safety goals and quantitative design objectives.

NUREG/CR-4198: FRACTURE IN GLASS/HIGH LEVEL WASTE CANISTERS. MARTIN,D.M. Iowa State Univ., Ames, IA. April 1985. 81pp. 8504170534. 29906:243.

The release rate of radionuclides from a vitrified waste form due to aqueous leaching by ground water will depend, among other factors, on the waste form's surface area. Large castings of glass will almost certainly be used as the waste form for high level nuclear wastes and such castings tend to fracture as a result of transient and residual stresses induced by the casting process; such fractures increase the surface area available for aqueous leaching of radionuclides from the HLW glass. The primary focus of this study was on achieving an understanding of the dependence of fracture surface area on glass properties and processing variables for both in-can melts and castings. The maximum fracture surface area per unit volume of glass observed in this study was about 7.1/cm (an equivalent spherical particle diameter of 0.85 cm) for a water quenched in-can melt. The processing parameter which appears to most strongly affect the extent of fracture surface area for both castings and in-can melts is the dimensionless Biot modulus (thermal film coefficient x radius/waste form thermal conductivity).

NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2. ALPERT,D.J.; IMAN,R.L.; HELTON,J.C.; et al. Sandia National Laboratories. June 1985. 59pp. 8507050415. SAND84-1824. 31372:210.

A demonstration uncertain/sensitivity analysis was performed for the reactor accident consequence model CRAC2 using techniques compiled as part of the NRC-sponsored MELCOR program. The principal objectives of the study were: 1) to demonstrate the use of the uncertain/sensitivity analysis techniques, 2) to test the computer models that implement the techniques 3) to identify possible difficulties in performing such an analysis, and 4) to explore alternative means of analyzing, displaying, and describing the results. Seventeen CRAC2 input variables thought to contribute significantly to uncertainty in estimated consequences were selected for analysis; subjective estimates of ranges, distributions, and correlations for these variables were made. Latin hypercube sampling, a modified Monte Carlo technique, was used to generate two multivariate samples of size 50 from the distributions assigned to the 17 input variables. A total of 100 CRAC2 runs, 50 for each sample, was performed. The results of the two samples were similar. A regression analysis was performed to estimate the contribution of each variable to uncertainty in estimated consequences. The study was first performed with the magnitude of the source term as one of the 17 variables. A second analysis was performed with a fixed source term. Only one sample of size 50 was generated in the second analysis. The uncertainty/sensitivity analysis techniques compiled for MELCOR appear well suited for use with a health and economic consequence model. Alternative methods for displaying and describing the results are presented. The insights gained from performing the analysis are reviewed and major conclusions summarized. A comparison of the results of this study with current point estimates of health and economic consequences is presented.

NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS. PICIULO,P.L.; SHEA,C.E.; BARLETTA,R.E. Brookhaven National Laboratory. May 1985. 46pp. 8506140593. BNL-NUREG-51868. 30936:219.

The NRC Technical Position on Waste Form (TP) specifies that waste should be resistant to biodegradation. The methods recommended in the TP for testing resistance to fungi, ASTM G21, and for testing resistance to bacteria, ASTM G22, were carried out on several types of solidified simulated wastes, and the effect of microbial activity on the mechanical strength of the materials tested was examined. The tests are believed to be sufficient for distinguishing between materials that are susceptible to biodegradation and those that are not. However, it is concluded that failure of these tests should not be regarded of itself as an indication that the waste form will biodegrade to an extent that the form does not meet the stability requirements of 10 CFR Part 61. In the case of failure of ASTM G21 or ASTM G22 or both, it is recommended that additional data be supplied by the waste generator to demonstrate the resistance of the waste form to microbial degradation.

NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS. PICIULO,P.L.; CHAN,S.F. Brookhaven National Laboratory. May 1985. 48pp. 8506060814. BNL-NUREG-51869. 30769:247.

The NRC Technical Position (TP) on Waste Form specifies that waste forms should be resistant to thermal degradation. The thermal cycle testing procedure outlined in the TP on Waste Form was carried out and is believed adequate for demonstrating the thermal stability of solidified waste forms. The inclusion of control samples and the monitoring of sample temperature are recommended additions to the test. An outline for reporting thermal cycling test results is given. To produce a data base on the applicability of the thermal cycling test, the following simulated laboratory-scale waste forms were prepared and tested: boric acid and sodium sulfate evaporator bottoms, mixed bed bead resins, and powdered resins each solidified in asphalt, cement and vinyl ester-styrene.

NUREG/CR-4203: A CALCULATIONAL METHOD FOR DETERMINING BIOLOGICAL DOSE RATES FROM IRRADIATED RESEARCH REACTOR FUEL. SCHNITZLER,B.G. EG&G Idaho, Inc. (subs. of EG&G, Inc.),. April 1985. 65pp. 8506060811. EGG-2383. 30775:237.

This report describes a calculational method for the determination of biological dose rate from irradiated research reactor fuels. The calculational method is implemented in a computer program for quick and convenient assessment of multigroup gamma and beta dose rates resulting from an arbitrary (user-supplied) irradiation history. The FUELDR program calculates dose rates at a fixed dose point using built-in fission product impulse source functions and precalculated gamma and beta transport factors. The fixed dose point is located on the axial mid-plane at a distance of 91.44 cm (3 ft) from the fuel element.

Transport factors are included for sixteen unique (235)U fuel types in use at thirteen nonpower reactor facilities.

NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS:Annual Report,October 1983 - September 1984. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. April 1985. 33pp. 8508140599. ANL-85-20. 30935:223.

This progress report summarizes work performed by Argonne National Laboratory during the twelve months from October 1983 to September 1984 on long-term embrittlement of cast duplex stainless steels used in light-water reactors.

NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL. JORDAN,H.; KUHLMAN,M.R. Battelle Memorial Institute, Columbus Laboratories. May 1985. 74pp. 8506190036. BMI-2124. 31017:100.

The TRAP-MELT 2 code is a development of the previously issued TRAP-MELT code which simulates the transport and deposition of aerosol particles and certain vapors in the reactor coolant system under hypothetical accident conditions in a light water reactor. This manual contains a brief description of the models of the processes treated in the code and of the code organization. The input to the code for a sample run are presented and output from a run are presented as well.

NUREG/CR-4206: A SELECT REVIEW OF THE RECENT (1979-1983) BEHAVIORAL RESEARCH LITERATURE ON TRAINING SIMULATORS. LAUGHERY,K.R. Oak Ridge National Laboratory. May 1985. 51pp. 8506130489. ORNL/TM-9445. 30901:249.

Report summarizes some selected reports of behavioral research performed in years 1979-1983 on training simulator application technology, and discusses findings related to nuclear power plant operators' simulator training. Findings are organized as related to the design, testing, and use of training simulators. Topics include Simulator Fidelity vs. Training Effectiveness, Operator Performance Measurement, Measuring Simulator Effectiveness, and Simulator Utilization Practices. Reviews 89 references.

NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium. BHATTACHARYYA; LARSEN,R.P.; OLDHAM,R.D.; et al. Argonne National Laboratory. May 1985. 99pp. 8507050425. ANL-85-21. 31371:207.

The gastrointestinal (GI) absorption of plutonium was measured in mice, rats, and dogs under conditions relevant to setting drinking water standards. The fractional GI absorption of Pu (VI) in adult mice was 2 x 10(-4) (0.02%) in fed mice and 2 x 10(-3) (0.02%) in fasted mice. The GI absorption of plutonium was independent of plutonium oxidation state, administration medium, and plutonium concentration; absorption was dependent upon animal species, state of animal fasting, state of Pu(IV) hydrolysis, and age of the animal. Fractional GI absorption values ranged from 3 x 10(-5) (0.003%) for hydrolyzed Pu(IV) administered to fed adult mice to 7 x 10(-3) (0.7%) for Pu(VI) administered to fed neonatal rats. From analysis of our data, we suggested values of f(1) (the fraction transferred from gut to blood in humans) for use in establishment of oral limits of exposure to plutonium. For an acute exposure in the occupational setting, we proposed one value of f(1) for fed (2 x10(-4) and one for fasted (2 x 10(-3) individuals. For the environmental setting, we developed two approaches to obtaining values of f(1); suggested values were 6 x 10(-4) and 4 x 10(-3), respectively. Both approaches took into account effects of animal age and fasting. We discussed uncertainties in proposed values of f(1) and made recommendations for further research.

NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS. BAYBUTT,P.; RAGHURAM,S.; AVCI,H.I. Battelle Memorial Institute, Columbus Laboratories. April 1985. 62pp. 8505080375. BMI-2125. 30218:271.

A new computer code called MATADOR (Methods for the Analysis of Transport And Deposition Of Radionuclides) has been developed to replace the CORRAL computer code which was written for the Reactor Safety Study (WASH-1400). This report contains a detailed description of the models used in MATADOR. A companion report provides a User's Manual for the code. MATADOR is intended for use in system risk studies to analyze radionuclide transport and deposition in reactor containments. The principal output of the code is information on the timing and magnitude of radionuclide releases to the environment as a result of severely degraded core accidents. MATADOR considers the transport of radionuclides through the containment and their removal by natural deposition and the operation of engineered safety systems such as sprays. The code requires input data on the source term from the primary system, the geometry of the containment, and the thermal-hydraulic conditions in the containment.

NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL. AVCI,H.I.; RAGHURAM,S.; BAYBUTT,P. Battelle Memorial Institute, Columbus Laboratories. April 1985. 75pp. 8505080373. BMI-2126. 30218:192.

A new computer code called MATADOR (Methods for the Analysis of Transport And Deposition of Radionuclides) has been developed to replace the CORRAL-2 computer code which was written for the Reactor Safety Study (WASH-1400). This report is a User's Manual for MATADOR. A companion report describes in detail the models used in the code. MATADOR is intended for use in system risk studies to analyze radionuclide transport and deposition in reactor containments. The principal output of the code is information on the timing and magnitude of radionuclide releases to the environment as a result of severely degraded core accidents. MATADOR considers the transport of radionuclides through the containment and their removal by natural deposition and by engineered safety systems such as sprays. It is capable of analyzing the behavior of radionuclides existing either as vapors or aerosols in the containment. The code requires input data on the source terms into the containment, the geometry of the containment, and thermal-hydraulic conditions in the containment.

NUREG/CR-4212: IN-PLACE THERMAL ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS. SERVER,W.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). April 1985. 250pp. 8505070548. EGG-MS-6708. 30211:101.

Radiation embrittlement of ferritic pressure vessel steels changes the toughness properties. A thermal anneal cycle well above the normal operating temperature of the vessel can restore most of the original properties. The Army SM-1A test reactor vessel was wet annealed in 1967, and wet annealing of the Belgian BR-3 reactor vessel has recently taken place. An industry survey indicates that dry annealing of a reactor vessel in-place is feasible, but solvable engineering problems exist. Limited toughness data available for five high copper content welds were reviewed. The review suggested that significant recovery results from annealing at 454 degrees centigrade (850 degrees fahrenheit) for one week, but scatter in the data makes assessment of recovery and reembrittlement response difficult to quantify. A thermal and structural analysis of a reactor vessel undergoing an annealing treatment found no problems with the reactor vessel itself, but did indicate a rotation at the nozzle region of the vessel which would plastically deform the attached primary piping. Further analytical studies attempted to solve this problem, but they were not successful. An American Society for Testing and Materials (ASTM) task group is upgrading and revising guide ASTM E 509-74 with emphasis on the materials and surveillance aspects of annealing.

NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA. MACKENZIE,D.R.; VASLOW,F.; DOUGHERTY,D.R.; et al. Brookhaven National Laboratory. May 1985. 77pp. 8506140405. BNL-NUREG-51873. 30908:116.

This report provides technical support to NRC in connection with the regulation 10 CFR Part 61 and NRC's Technical Position (TP) on waste form. Six specific areas are addressed, namely: the technical basis for limiting containers of radioactive gases to atmospheric pressure and 100 curies; the requirements to demonstrate that a stable waste would be recognizable for 300 or 500 years; the feasibility of achieving less than 5% deformation in buried wastes; the adequacy of ASTM tests G21 and G22 for testing for biodegradability; the adequacy of ASTM test B553 for testing for thermal degradation; and the basis for determining if a waste is explosive or pyrophoric. The principal conclusions of the report follow. A maximum pressure of 1.5 atmospheres for radioactive gases is acceptable, but the radioactivity limit should depend on the isotope, the quality of the container and the properties of the site. Site and package qualities and a wet/dry cycling test are suggested that appreciably increase the probability of indicating whether a waste would have long-term recognizability. Achieving deformation of buried waste of 5% would not be feasible using current solidification methods with either metal or polyethylene containers. ASTM tests G21 and G22, with modifications are suitable for biodegradability testing. A modified form of ASTM B553 is adequate for thermal testing. Required information on pyrophoric and explosive materials is provided.

NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign. HABERMAN,J.H. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 62pp. 8506260395. PNL-5433. 31244:326.

A series of in-reactor experiments were conducted using full-length 32-rod pressurized water reactor (PWR) fuel bundles as part of the Loss-of-Coolant Accident (LOCA) Simulation Program by Pacific Northwest Laboratory (PNL). The third materials test (MT-3) was the sixth experiment in a series of thermal-hydraulic and materials deformation/rupture experiments conducted in the National Research Universal (NRU) Reactor, Chalk River, Ontario, Canada. The MT-3 experiment was jointly funded by the U.S. Nuclear Regulatory Commission (NRC) and the United Kingdom Atomic Energy Authority (UKAEA) with the main objective of evaluation ballooning and rupture during active two-phase cooling at elevated tempertures. All 12 test rods in the center of the 32-rod bundle failed with an average peak strain of 55.4%. At the request of the UKAEA, a destructive postirradiation examination (PIE) was performed on 7 of the 12 test rods. The results of this examination were presented in a previous report. Subsequently, and at the request of UKAEA, PIE was performed on three additional rods along with further examination of one of the previously examined rods. Information obtained from the PIE included cladding thickness measurements, cladding metallography, and particle size analysis of the fractured fuel pellets. This report describes the additional PIE work performed and presents the results of the examinations.

NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS. PELTO,P.J.; GALLUCCI,R.H.; AMES,K.R. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 222pp. 8506260362. PNL-5432. 31245:199.

This report summarizes the results of the Reliability Analysis of Containment Isolation System Project. Work was performed in five basic areas: design review, operating experience review, related research review, generic analysis and plant specific analysis. Licensee Event Reports (LERs) and Integrated Leak Rate (ILRT) Test reports provided the major sources of containment performance information used in this study. Data extracted from the LERs were assembled into a computer data base. Qualitative and quantitative information developed for containment performance under normal operating conditions and design basis accidents indicate that there is room for improvement. A crude estimate for overall containment unavailability for relatively small leaks which violate plant technical specifications is 0.3. An estimate of containment unavailability due to large leakage events is in the range of 0.001 to 0.01. These estimates are dependent on several assumptions (particularly on event duration times) which are documented in the report.

NUREG/CR-4221: AN EVALUATION OF STRESS CORROSION CRACK GROWTH IN BWR PIPING SYSTEMS. KASSIR,M.; SHARMA,S.; REICH,M.; et al. Brookhaven National Laboratory. May 1985. 80pp. 8506130175. BNL-NUREG-51874. 30867:183.

This report presents the results of a study conducted to evaluate the effects of stress intensity factor and environment on the growth behavior of integranular stress corrosion cracks in type 304 stainless steel piping systems. Most of the detected cracks are known to be circumferential in shape, and initially start at the inside surface in the heat affected zone near girth welds. These cracks grow both radially in-depth and circumferentially in length and, in extreme cases, may cause leakage in the installation. The propagation of the crack is essentially due to the influence of the following simultaneous factors: (1) The action of applied and residual stress, (2) Sensitization of the base metal in the affected zone adjacent to firth weld and (3) The continuous exposure of the material to an aggressive environment of high temperature water containing dissolved oxygen and some levels of impurities. Each of these factors and their effects on the piping systems is discussed in detail in text of the report. The report also evaluates the time required for hypothetical cracks in BWR pipes to propagate to their critical size. The pertinent times are computed and displayed graphically. Finally, parametric study is performed in order to assess the relative influence and sensitivity of the various input parameters (residual stress, crack growth law, diameter of pipe, initial size of defect, etc.) which have bearing on the growth behavior of the intergranular stress corrosion cracks in type 304 stainless steel. Cracks in large-diameter as well as in small-diameter pipes are considered and analyzed.

NUREG/CR-4225: SUMMARY OF EFFICIENCY TESTING OF STANDARD AND HIGH-CAPACITY HIGH-EFFICIENCY PARTICULATE AIR FILTERS SUBJECTED TO SIMULATED TORNADO DEPRESSURIZATION AND EXPLOSIVE SHOCK WAVES. SMITH,P.R.; GREGORY,W.S. Los Alamos Scientific Laboratory. April 1985. 25pp. 8507020407. LA-10401-MS. 31309:007.

Pressure transients in nuclear facility air cleaning systems can originate from natural phenomena such as tornadoes or from accident-induced explosive blast waves. This study was concerned with the effective efficiency of high-efficiency particulate air (HEPA) filters during pressure surges resulting from simulated tornado and explosion transients. The primary objective of the study was to examine filter efficiencies at pressure levels below the point of structural failure. Both standard and high-capacity 0.61-m by 0.61-m HEPA filters were evaluated, as were several 0.2-m by 0.2-m HEPA filters. For a particular manufacturer, the material release when subjected to tornado transients is the same (per unit area) for both the 0.2-m by 0.2-m and the 0.61-m by 0.61m filters. For tornado transients, the material was on the order of micrograms per square meter. When subjecting clean HEPA filters to simulated tornado transients with aerosol entrained in the pressure pulse, all filters tested showed a degradation of filter efficiency. For explosive transients, the material release from preloaded high-capacity filters was as much as 340 g. When preloaded high-capacity filters were subjected to shock waves approximately 50% of the structural limit level, 1 to 2 mg of particulate was released.

NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983. BUSCHBACH,T.C. St. Louis Univ., St. Louis, MO. April 1985. 153pp. 8505070552. 30209:297.

The purpose of the New Madrid Seismotectonic Study is to identify the earthquake mechanisms within a 200-mile radius of New Madrid, Missouri. During 1983 there was more awareness of the significance of current regional stress patterns and the local concentration of stresses by basement structures and inhomogeneities. The program continued to concentrate on defining boundaries of a proposed rift complex in the area, as well as establishing the relationships of the east-west trending fault systems with the northeast-trending faults of the Wabash Valley and New Madrid areas. There were 204 earthquakes located by the Saint Louis University microearthquake network in 1983. In addition, the earthquake swarm in north-central Arkansas continued throughout the year, and 45,000 earthquakes have been recorded there since January, 1982. Trenching data from Late Cenozoic terrace deposits along the Kentucky River Fault System suggest that there was post-terrace deformation along some of the faults. Thermal and chemical data from groundwaters in the Mississippi Embayment appear to be useful in localizing deep faults that cut through the aquifers. Early indications from studies of jointing in Indiana are that the direction of major joint sets will be useful in determining regional stress directions. No Quaternary faulting was found in the Indiana or Illinois fault studies.

NUREG/CR-4229: EVALUATION OF CURRENT METHODOLOGY EMPLOYED IN PROBABILISTIC RISK ASSESSMENT (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS. RUGER,C.; BOCCIO,J.L.; AZARM,M.A. Brookhaven National Laboratory. May 1985. 47pp. 8506190103. BNL-NUREG-51877. 31017:277.

The report presents a general evaluation of the current methodology used by industry for the probabilistic assessment of fire events in nuclear power plants. The basis for this evaluation, in which the strengths and weaknesses of the methods are identified, stem from reviews of several, industry-sponsored, full-scope Probabilistic Risk Assessments (PRAs) and various deterministic/probabilistic approaches used by industry to judge their compliance with or used to seek exemptions from the fire-protection requirements enumerated in Appendix R to 10 CFR 50. In performing this evaluation of the current methodologies, state-of-the-art literature on the modeling of propagation/detection/suppression, input parameters, and modeling uncertainties are utilized. Areas are identified where recently-developed, more accurate and complete techniques can be implemented to reduce the state-of-knowledge uncertainties that presently exist. Recommendations are also made which could be the basis for a more suitable and complete fire-risk methodology.

NUREG/CR-4230: PROBABILITY-BASED EVALUATION OF SELECTED FIRE PROTECTION FEATURES IN NUCLEAR POWER PLANTS. AZARM,M.A.; BOCCIO,J.L. Brookhaven National Laboratory. May 1985. 93pp. 8506180415. BNL-NUREG-51878. 30985:281.

A probabilistic approach for the evaluation of major fire protection measures in nuclear power plants is described. The methods developed are applied to two representative fire areas -- one similar to a cable routing room and the other typical of a diesel generator room. The fire areas chosen for application, the fire scenarios described, and the various fire-damage states specified in the two illustrative examples are used to evaluate those fire-protection guidelines which deal with automatic/manual fire detection and suppression systems, rated barriers, divisional separation, drainage systems, dampers, and fire rating of electrical cables. Tabular results are presented, which reflect the relative merits of these systems/features in terms of conditional probabilities of achieving various room-damage states. The conclusions drawn and the lessons learned through the course of this study are discussed, and the areas that may need further investigation are identified.

NUREG/CR-4231: EVALUATION OF AVAILABLE DATA FOR PROBABILISTIC RISK ASSESSMENTS (PRA) OF FIRE EVENTS AT NUCLEAR POWER PLANTS. SAMANTA,P.K.; BOCCIO,J.L. Brookhaven National Laboratory. KRASNER,L.M.; et al. Factory Mutual Research Corp. May 1985. 71pp. 8506190077. BNL-NUREG-51879. 31017:205.

Several crucial parameters are needed in the assessment of fire risk in nuclear power plants. Among those that need to be developed from a data base are: (1) fire frequency, (2) fire detection time, and (3) fire suppression time. Currently, that data base for nuclear power plants is not large enough to develop these parameters, considering fuel location, fuel geometry, combustion properties, enclosure geometry, etc. This study attempts to augment the nuclear data base by investigating the usefulness of other nonnuclear data bases which contain fire incident loss experience of occupancy classes having somewhat similar physical features and fire protection engineering systems normally found in nuclear power plants. This study has found that indeed some useful information can be gleaned from nonnuclear sources; in particular, detection and suppression times. However, other fire-risk data needs such as fire frequency and fire size would require other forms of data searches and data analyses that at this stage can only be conceptualized.

NUREG/CR-4237: MOBILITY OF RADIONUCLIDES IN HIGH CHLORIDE ENVIRONMENTS. SIMPSON,H.J.; HERCZEG,A.L.; ANDERSON,R.F.; et al. Columbia Univ., New York, NY. April 1985. 77pp. 8505070484. 30210:219.

Concentrations of naturally occurring isotopes of uranium, thorium, radium and radon were measured in freshwaters and in sodium-chloride brines near the site of the Waste Isolation Pilot Plant (WIPP) located in southeastern New Mexico. Supplemental water chemistry analyses (chloride, alkalinity, P(CO2), CO(2), Fe, Mn, H(2)S) were made to aid in interpreting the data for natural radionuclides. Three features of radionuclide mobility are evident from the results: 1) There is a slight tendency for U and Ra concentrations to correlate with the chloride content of the water samples. Whether this tendency results from complexation by Cl- ions or cation exchange competition for adsorption sites cannot be resolved with the available information. 2) Much more dramatic than the correlation with Cl- concentration is the effect of the redox state of the waters on U and Ra concentrations. Chemically reducing groundwaters contain much lower U concentrations and much higher Ra concentrations than were measured in oxic and suboxic samples. Calculated retardation factors of 1 for Ra indicate that it can migrate freely in anoxic brines. 3) Low chemical recoveries of Th, and to a lesser extent U were observed for methods that work well with seawater samples. These elements may be present in a mobile, unreactive dissolved or colloidal complex with organic matter.

NUREG/CR-4245: IN-PLANT SOURCE TERM MEASUREMENTS AT BRUNSWICK STEAM ELECTRIC STATION. DUCE,S.W.; CRONEY,S.T.; AKERS,D.W.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.),. June 1985. 800pp. 8507020396. EGG-2392. 31311:061.

This report presents data obtained at Brunswick as part of the In-Plant Source Term Measurement Program in operating light water reactors (LWRs). The work was conducted for the Office of Nuclear Regulatory Research (RES) in support of the Meteorology and Effluent Treatment Branch (METB) of the Office of Nuclear Reactor Regulation (NRR). The primary objective of this program is to provide the Nuclear Regulatory Commission (NRC) with operational data that can be used in evaluation of plant designs for liquid and gaseous radwaste treatment systems. Data presented were obtained at the Brunswick Nuclear Generating Station, operated by Carolina Power and Light, located at Southport, North Carolina. In-plant measuremens were conducted during the time period from March 1982 to November 1982. This plant is the sixth in a series of operating LWRs to be studied and the first boiling water reactor (BWR) in the series.

NUREG/CR-4262 V01: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR.Main Report. BRUSKE,S.J.; BAXTER,D.E.; RANSOM,C.B.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.),. May 1985. 81pp. 8506240180. EGG-2394. 31150:280.

This report documents the evaluation of the effects of nonsafety grade control system failures on a typical boiling water reactor plant. The methods utilized in this evaluation include a system level failure modes and effects analysis, deterministic computer analysis, a review of 3 years of recorded plant occurrences, a probability analysis and a review of applicable NRC criteria pertaining to control systems. This study identified three system failures that could cause transients leading to a reactor vessel overfill and of these three failures, two could also lead to a reactor coolant cooldown of greater than 100 degrees fahrenheit per hour. This study concluded that the existing NRC criteria, concerning control systems, adequately address the potential problem areas that were identified during this evaluation. Based on the results of this study, it was recommended that the consequences and risk associated with overfill and overcool transients be further investigated.

NUREG/CR-4262 V02: EFFECTS OF CONTROL SYSTEM FAILURES ON TRANSIENTS AND ACCIDENTS AT A GENERAL ELECTRIC BOILING WATER REACTOR. Appendices. BRUSKE,S.J.; BAXTER,D.E.; RANSOM,C.B.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.),. May 1985. 341pp. 8506240689. EGG-2394. 31179:036.

Safety Implications of Control Systems (A-47) was approved as an Unresolved Safety Issue (USI) by the Nuclear Regulatory Commission (NRC) in December of 1980. USI A-47 concerns the potential for transients or accidents being made more severe as a result of control system failures. This report describes the work performed on the effects of control system failures on transients and accidents at a General Electric boiling water reactor. This work was conducted for the U.S. Nuclear Regulatory Commission, Division of Safety Technology by EG&G Idaho, Inc. and is based on the Browns Ferry Nuclear Plant. This report is contained in two volumes; a main report and five appendices. The main report describes the study methodology, the major areas of work performed, and the results and conclusions. The appendices contain detailed information consisting of failure mode and effects analysis tables, a detailed description of the computer analyses and significant transient excerpts.

NUREG/CR-4263: RELIABILITY ANALYSIS OF STIFF VERSUS FLEXIBLE PIPING FINAL PROJECT REPORT. LU,S.C.; CHOU,C.K. Lawrence Livermore National Laboratory. May 1985. 78pp. 8505280086. UCRL-20410. 30604:169.

This research project is to develop a technical basis for flexible piping designs which will improve piping reliability and minimize the use of pipe supports, snubbers, and pipe whip restraints. This study indicated that piping design can be made more reliable by some reduction of rigid supports and/or snubbers. This study also confirmed that the malfunction of pipe whip restraints introduced higher thermal stresses and tended to reduce the overall piping reliability. Finally, our results indicated that supports in a flexible piping design may need to be re-evaluated and that the elimination of pipe supports which are close to components should be done with care in order to minimize the impact on the component reliability.

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NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS. FENTON,D.L.; GREGORY,W.S.; et al. Los Alamos Scientific Laboratory. GUNAJI,M.V. New Mexico State Univ., Las Cruces, NM. May 1985. 32pp. 8507050422. LA-10436-MS. 31375:021.

Experiments were conducted to investigate high-efficiency particulate air (HEPA) filter plugging by combustion aerosols. These tests were done to obtain empirical data to improve our modeling of filter plugging phenomena using the Los Alamos National Laboratory fire accident analysis code FIRAC. Commercially available 0.61-m by 0.61-m square filters were tested in a specially designed facility to determine how airflow resistance varies with increased filter loading by combustion aerosols. Two organic fuels normally found in nuclear fuel cycle facilities, polystyrene (PS) and polymethylmethacrylate (PMMA), were burned under varied conditions to generate combustion aerosols. The test facility included a combustor, a 23-m-long duct, and a specially designed gravimetric balance for determing the aerosol mass gain of the filters. Test results include correlations of HEPA filter resistance ratios (actual resistance/initial resistance) with aerosol mass gain. The mass gain of plugged HEPA filters was found to correlate with the airborne mass concentration of material in the size range greater that approximately 2.0 m. Also, the fuel with a smaller soot fraction, PMMA, produced filter plugging at lower accumulated aerosol mass deposits on or within the filter.

NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM. BLAND,W.M. GeeB's, Inc. June 1985. 113pp. 8507080205. 31393:164.

Aerospace SRQA and management techniques, principally those developed and used by the NASA Lyndon B. Johnson Space Center on the manned space flight programs, have been assessed for possible application by the DOE and the DOE-contractors to the high level radioactive waste repository program that results from the implementation of the NWPA of 1982. Those techniques believed to have the greatest potential for usefulness to the DOE and the DOE-contractors have been discussed in detail and are recommended to the DOE for adoption; discussion is provided for the manner in which this transfer of technology can be implemented. Six SRQA techniques and two management techniques are recommended for adoption by the DOE; included with the management techniques is a recommendation for the DOE to include a licensing interface with the NRC in the application of the milestone review technique. These other techniques are recommended for study by the DOE for possible adaption to the DOE program.

NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT. ENDERLIN,W.I.; BAUGH,J.W. Battelle Memorial Institute, Pacific Northwest Laboratories. June 1985. 38pp. 8507030714. PNL-5477. 31322:273.

The Pacific Northwest Laboratory is studying the effects of increased tube/tube-support clearances in pressurized water reactor steam generators following chemical cleaning. The project purpose is to provide NRC with criteria for evaluating licensees' specific proposals for chemical cleaning of steam generators. This report describes the test and data analysis plans and procedures for the flow and accelerated wear tests to be performed in a scale-model steam generator. The flow tests will establish the forcing boundary conditions, using clearances representing various conditions following chemical cleaning. The accelerated wear tests will determine the potential wear rates possible, based on the vibrations characterized in the flow tests. The overall project status, including work completed to date and tasks planned for the remainder of FY85, is also documented.

NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY. DE JARLAIS,G.; ISHII,M. Argonne National Laboratory. April 1985. 115pp. 8507050406. ANL-85-31. 31338:074.

Steady-state inverted annular flow of Freon 113 in up flow was established in a transparent test section. Using a special inlet configuration consisting of long aspect-ratio liquid nozzles coaxially centered within a heated quartz tube, idealized inverted annular flow initial geometry (cylindrical liquid core surrounded by coaxial annulus of gas) could be established. Inlet liquid and gas flowrates, liquid subcooling, and gas density (using various gas species) were measured and varied systematically. The hydrodynamic behavior of the liquid core, and the subsequent downstream break-in of this core into slugs, ligaments and/or droplets of various sizes, was observed. In general, for low inlet liquid velocities it was observed that after the initial formation of roll waves on the liquid core surface, an agitated region of high surface area, with attendant high momemtum and energy transfers, occurs. This agitated region appears to propagate downstream in a quasi-periodic pattern. Increased inlet liquid flow rates, and high gas annulus flow rates tend to diminish the significance of this agitated region. Observed inverted annular flow (and subsequent downstream flow pattern) hydrodynamic behavior is reported, and comparisions are drawn to data generated by previous experimenters studying post-CHF flow.

NUREG/CR-4283: STUDY OF THE EFFECTS OF ELASTIC UNLOADINGS ON THE JI-R CURVES FROM COMPACT SPECIMENS. SUTTON,G.E.; VASSILAROS,M.G. David W. Taylor Naval Research & Development Center. June 1985. 50pp. 8506260731. 31227:118.

An investigation was performed to evaluate the efforts of elastic unloadings on the J-Integral Resistance Curves of ASTM A106 Class C steel and 3-Ni steel. Compact specimens (1T) were tested using a multi-specimen technique, direct current potential drop technique and the elastic unloading compliance technique with unloading ranging from 10 to 90%. The two former techniques were 0% unloading procedures used to generate the reference J-R curves for comparison to the elastic unloading J-R curves for the two steels. The results of the investigations of these materials indicate that there was no significant difference in the J-R curves that resulted from the elastic unloading compliance technique.

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NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.

Reactor Safeguards
NUREG-1125 V01: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 1,Part 1:ACRS Reports On Project Reviews (A-F).
NUREG-1125 V02: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 2,Part 1:ACRS Reports On Project Reviews (G-P).
NUREG-1125 V03: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 3,Part 1:ACRS Reports On Project Reviews (Q-Z).
NUREG-1125 V04: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 4,Part 2:ACRS Reports On Generic Subjects (Accident Analysis - Generic Items).
NUREG-1125 V05: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 5,Part 2:ACRS Reports On Generic Subjects (HTGR - Regulatory Guides).
NUREG-1125 V06: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1957-1984.Volume 6,Part 2:ACRS Reports On Generic Subjects (RPA - Appendix C).

Reactor Safety
NUREG/CR-3651: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.
NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1 - September 30,1984.

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NRC Originating Organization Index (Staff Reports)

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

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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 OF ADMINISTRATION
DIVISION OF RULES AND RECORDS
NUREG/CR-4040: OPERATIONAL DECISIONMAKING AND ACTION SELECTION UNDER PSYCHOLOGICAL STRESS IN NUCLEAR POWER PLANTS.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA
AEOD, DIRECTOR'S OFFICE
NUREG/CR-2000 V04 N3: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of March 1985.
NUREG/CR-2000 V04 N4: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of April 1985.
NUREG/CR-2000 V04 N5: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of May 1985.
NUREG/CR-3551: SAFETY IMPLICATIONS ASSOCIATED WITH IN-PLANT PRESSURIZED GAS STORAGE AND DISTRIBUTION SYSTEMS IN NUCLEAR POWER PLANTS.
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-3905 V02: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Code Listings.
NUREG/CR-3905 V03: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-3905 V04: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.Coder's Manual.
NUREG/CR-4071: EXPLORATORY TREND AND PATTERN ANALYSIS FOR 1981 LICENSEE EVENT REPORT DATA.

OFFICE OF INSPECTION & ENFORCEMENT (POST 12/11/80)
DIVISION OF EMERGENCY PREPAREDNESS & ENGINEERING RESPONSE (POST 830103)
NUREG/CR-4003: CLOSEOUT OF IE BULLETIN 79-04:INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION.
NUREG/CR-4004: CLOSEOUT OF IE BULLETIN 79-25:FAILURES OF WESTINGHOUSE BFD RELAYS IN SAFETY-RELATED SYSTEMS.
NUREG/CR-4005: CLOSEOUT OF IE BULLETIN 80-12:DECAY HEAT REMOVAL SYSTEM OPERABILITY.
DIVISION OF INSPECTION PROGRAMS (POST 850212)
NUREG/CR-4190: CALIFORNIA OFFSHORE SURVEY OF LICENSEES USING RADIOACTIVE MATERIAL.

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
DIVISION OF FUEL CYCLE & MATERIAL SAFETY
NUREG/CR-4035: A HIGHWAY ACCIDENT INVOLVING RADIOPHARMACEUTICALS NEAR BROOKHAVEN,MISSISSIPPI ON DECEMBER 3,1983.
DIVISION OF WASTE MANAGEMENT
NUREG/CR-3091 V04: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period October 1983 - March 1984.
NUREG/CR-3091 V05: REVIEW OF WASTE PACKAGE VERIFICATION TESTS.Semiannual Report Covering The Period April 1984 - September 1984.
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.
NUREG/CR-4134: REPOSITORY ENVIRONMENTAL PARAMETERS RELEVANT TO ASSESSING THE PERFORMANCE OF HIGH-LEVEL WASTE PACKAGES.
NUREG/CR-4200: BIODEGRADATION TESTING OF SOLIDIFIED LOW-LEVEL WASTE STREAMS.
NUREG/CR-4201: THERMAL STABILITY TESTING OF LOW-LEVEL WASTE FORMS.
NUREG/CR-4215: TECHNICAL FACTORS AFFECTING LOW-LEVEL WASTE FORM ACCEPTANCE CRITERIA.
NUREG/CR-4271: RECOMMENDED SAFETY,RELIABILITY,QUALITY ASSURANCE AND MANAGEMENT AEROSPACE TECHNIQUES WITH POSSIBLE APPLICATION BY THE DOE TO THE HIGH LEVEL RADIOACTIVE WASTE REPOSITORY PROGRAM.

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 4/05/81)
DIVISION OF ACCIDENT EVALUATION
NUREG/CR-2331 V04 N3: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report,July 1 -September 30,1984.
NUREG/CR-2331 V04 N4: SAFETY RESEARCH PROGRAMS SPONSORED BY OFFICE OF NUCLEAR REGULATORY RESEARCH.Quarterly Progress Report, October 1 - December 31, 1984.
NUREG/CR-2531 R03: INTRODUCTORY USER'S MANUAL FOR THE U.S. NUCLEAR REGULATORY COMMISSION REACTOR SAFETY RESEARCH DATA BANK.
NUREG/CR-2718: STEAM EXPLOSION EXPERIMENTS WITH SINGLE DROPS OF IRON OXIDE MELTED WITH A CO2 LASER.Part II.Parametric Studies.
NUREG/CR-2951: THE D9 EXPERIMENT.Heat Removal From Stratified UO2 Debris.
NUREG/CR-3005: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS.
NUREG/CR-3197 V01: REACTION BETWEEN SOME CESIUM-IODINE COMPOUNDS AND THE REACTOR MATERIALS 304 STAINLESS STEEL,INCONEL 600 & SILVER. Volume I:Cesium Hydroxide Reactions.
NUREG/CR-3208: TRAC-PD2 DEVELOPMENTAL ASSESSMENT.
NUREG/CR-3514 V02: THE CHEMICAL BEHAVIOR OF IODINE IN AQUEOUS SOLUTIONS UP TO 150 C.II.Radiation-Redox Conditions.
NUREG/CR-[illegible]51: ASSESSMENT OF THE ADEQUACY OF ORNL INSTRUMENTATION IN REFLOOD TEST FACILITIES.
NUREG/CR-3703: ASSESSMENT OF SELECTED TRAC AND RELAP5 CALCULATIONS FOR OCONEE-1 PRESSURIZED THERMAL SHOCK STUDY.
NUREG/CR-3721 V01: PRESSURE MEASUREMENTS IN A HYDROGEN COMBUSTION ENVIRONMENT. Hydrogen-Air Combustion Test Series 1 And 2 In The FITS Tank.
NUREG/CR-3757: TRAN B-2:THE EFFECT OF LOW STEEL CONTENT ON FUEL PENETRATION IN A NON-MELTING CYLINDRICAL FLOW CHANNEL.
NUREG/CR-[illegible]03: THE EFFECTS OF POST-LOCA CONDITIONS ON A PROTECTIVE COATING (PAINT) FOR THE NUCLEAR POWER INDUSTRY.
NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-3816 V02: REACTOR SAFETY RESEARCH.Quarterly Report,April-June 1984.
NUREG/CR-3820 V03: THERMAL/HYDRAULIC ANALYSIS RESEARCH PROGRAM.Quarterly Report,July-September 1984.
NUREG/CR-3885 V03: HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION.Quarterly Progress Report, July 1 - September 30,1984.
NUREG/CR-3889: THE MODELING OF BWR CORE MELTDOWN ACCIDENTS - FOR APPLICATION IN THE MELRPI.MOD2 COMPUTER CODE.
NUREG/CR-3913: HECTR VERSION 1.0 USER'S MANUAL.
NUREG/CR-3930: OBSERVED BEHAVIOR OF CESIUM,IODINE,AND TELLURIUM IN THE ORNL FISSION PRODUCT RELEASE PROGRAM.
NUREG/CR-3944: TRAN B-3:EXPERIMENTAL INVESTIGATION OF FUEL CRUST STABILITY ON MELTING SURFACES OF AN ANNULAR FLOW CHANNEL.

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Contractor Index

This index lists, in alphabetical order, the contractors that prepared the NUREG/CR reports listed in this compilation. Listed below each contractor are the NUREG/CR numbers and titles of their reports. If further information is needed, refer to the main citation by the NUREG/CR number.

AERONAUTICAL RESEARCH ASSOCIATES OF PRINCETON

NUREG/CR-4158: A COMPILATION OF INFORMATION ON UNCERTAINTIES INVOLVED IN DEPOSITION MODELING.
NUREG/CR-4159: COMPARISON OF THE 1981 INEL DISPERSION DATA WITH RESULTS FROM A NUMBER OF DIFFERENT MODELS.

ARGONNE NATIONAL LABORATORY

NUREG/CR-3804 V04: PHYSICS OF REACTOR SAFETY.Quarterly Report,October-December 1984.
NUREG/CR-3855: CHARACTERIZATION OF NUCLEAR REACTOR CONTAINMENT PENETRATION - FINAL REPORT.
NUREG/CR-3980 V03: LIGHT-WATER-REACTOR SAFETY FUEL SYSTEMS RESEARCH PROGRAMS. Quarterly Progress Report,July-September 1984.
NUREG/CR-3998 V02: LIGHT-WATER-REACTOR SAFETY MATERIALS ENGINEERING RESEARCH PROGRAMS.Quarterly Progress Report,April-June 1984.
NUREG/CR-4064: STRUCTURAL RESPONSE OF LARGE PENETRATIONS AND CLOSURES FOR CONTAINMENT VESSELS SUBJECTED TO LOADINGS BEYOND DESIGN BASIS.
NUREG/CR-4124: NDE OF STAINLESS STEEL AND ON-LINE LEAK MONITORING OF LWRS. Annual Report,October 1983 - September 1984.
NUREG/CR-4180: STATE-OF-THE-ART OF SOLID-STATE MOTOR CONTROLLERS.
NUREG/CR-4191: SURVEY OF LICENSEE CONTROL ROOM HABITABILITY PRACTICES.
NUREG/CR-4204: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS:Annual Report,October 1983 - September 1984.
NUREG/CR-4208: GASTROINTESTINAL ABSORPTION OF PLUTONIUM IN MICE,RATS, AND DOGS.Application To Establishing Values Of f1 For Soluble Plutonium.
NUREG/CR-4277: INVERTED ANNUAL FLOW EXPERIMENTAL STUDY.

ARIZONA, UNIV. OF, TUCSON, AZ

NUREG/CR-4194: LOW-LEVEL NUCLEAR WASTE SHALLOW LAND BURIAL TRENCH ISOLATION.Final Report,October 1981 - September 1984.

ATOMIC ENERGY OF CANADA, LTD.

NUREG/CR-4077: REACTOR COOLANT PUMP SHAFT SEAL BEHAVIOR DURING STATION BLACKOUT.

BATTELLE HUMAN AFFAIRS RESEARCH CENTERS

NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.

BATTELLE MEMORIAL INSTITUTE, COLUMBUS LABORATORIES

NUREG/CR-3900 V03: LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING.Quarterly Report,October-December 1984.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4177 V01: MANAGEMENT OF SEVERE ACCIDENTS.Perspectives On Managing Severe Accidents In Commercial Nuclear Power Plants.
NUREG/CR-4177 V02: MANAGEMENT OF SEVERE ACCIDENTS.Extending Plant Operating Procedures Into The Severe Accident Regime.
NUREG/CR-4205: TRAP-MELT2 USER'S MANUAL.
NUREG/CR-4210: MATADOR:A COMPUTER CODE FOR THE ANALYSIS OF RADIONUCLIDE BEHAVIOR DURING DEGRADED CORE ACCIDENTS IN LIGHT WATER REACTORS.
NUREG/CR-4211: MATADOR (METHODS FOR THE ANALYSIS OF TRANSPORT AND DEPOSITION OF RADIONUCLIDES) CODE DESCRIPTION AND USER'S MANUAL.

BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORIES

NUREG/CR-1755 ADD01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING NUCLEAR REACTORS AT MULTIPLE-REACTOR STATIONS.Effects On Decommissioning Of Interim Inability To Dispose Of Wastes Offsite.
NUREG/CR-3293 V01: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Main Report.
NUREG/CR-3293 V02: TECHNOLOGY,SAFETY AND COSTS OF DECOMMISSIONING REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES FOLLOWING POSTULATED ACCIDENTS.Appendices.
NUREG/CR-3317: TECHNICAL BASES AND USER'S MANUAL FOR THE PROTOTYPE OF SPARC - A SUPPRESSION POOL AEROSOL REMOVAL CODE.
NUREG/CR-3613 V02: EVALUATION OF WELDED AND REPAIR-WELDED STAINLESS STEEL FOR LWR SERVICE.Annual Report for 1984.
NUREG/CR-3747: THE SELECTION AND TESTING OF ROCK FOR ARMORING URANIUM TAILINGS IMPOUNDMENTS.
NUREG/CR-3810 V04: REACTOR SAFETY RESEARCH PROGRAMS.Quarterly Report,October-December 1984.
NUREG/CR-3883: ANALYSIS OF JAPANESE-U.S. NUCLEAR POWER PLANT MAINTENANCE.
NUREG/CR-3906: URANIUM MILL TAILINGS NEUTRALIZATION:CONTAMINANT COMPLEXATION AND TAILINGS LEACHING STUDY.
NUREG/CR-3987: COMPUTERIZED ANNUNCIATOR SYSTEMS.
NUREG/CR-4051: ASSESSMENT OF JOB-RELATED EDUCATIONAL QUALIFICATIONS FOR NUCLEAR POWER PLANT OPERATORS.
NUREG/CR-4070 V03: BIVALVE FOULING OF NUCLEAR POWER PLANT SERVICE-WATER SYSTEMS.Factors That May Intensify The Safety Consequences Of Biofouling.
NUREG/CR-4075: DESIGNING PROTECTIVE COVERS FOR URANIUM MILL TAILINGS PILES. A Review.
NUREG/CR-4076: DETERMINATION OF COMPLIANCE WITH CRITERIA FOR FINAL TAILINGS DISPOSAL SITE RECLAMATION.
NUREG/CR-4088: METHODS FOR ESTIMATING RADIOACTIVE AND TOXIC AIRBORNE SOURCE TERMS FOR URANIUM MILLING OPERATIONS.
NUREG/CR-4118: MONITORING METHODS FOR DETERMINATION COMPLIANCE WITH DECOMMISSIONING CLEANUP CRITERIA AT URANIUM RECOVERY SITES.
NUREG/CR-4139: THE MAILED SURVEY:A TECHNIQUE FOR OBTAINING FEEDBACK FROM OPERATIONS PERSONNEL.
NUREG/CR-4144: IMPORTANCE RANKING BASED ON AGING CONSIDERATIONS OF COMPONENTS INCLUDED IN PROBABILISTIC RISK ASSESSMENTS.
NUREG/CR-4160: HISTORICAL SUMMARY OF OCCUPATIONAL RADIATION EXPOSURE EXPERIENCE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS.
NUREG/CR-4168: GT2F:A COMPUTER CODE FOR ESTIMATING LIGHT WATER REACTOR FUEL ROD FAILURES.
NUREG/CR-4176: EMISSION CONTROL TECHNOLOGY AND QUALITY ASSURANCE NEEDS AT URANIUM MILLING FACILITIES.Includes Supporting Methods For Testing,Operating,And Maintaining Air Pollution Control Devices.
NUREG/CR-4192: THE ANALYSIS OF DRAINAGE AND CONSOLIDATION AT TYPICAL URANIUM MILL TAILINGS SITES.
NUREG/CR-4218: LOCA SIMULATION IN THE NATIONAL RESEARCH UNIVERSAL REACTOR PROGRAM.Postirradiation Examination Results For The Third Materials Test (MT-3) - Second Campaign.
NUREG/CR-4220: RELIABILITY ANALYSIS OF CONTAINMENT ISOLATION SYSTEMS.
NUREG/CR-4276: VIBRATION AND WEAR IN STEAM GENERATOR TUBES FOLLOWING CHEMICAL CLEANING - SEMIANNUAL REPORT.

NUREG/CR-4095: TEST SERIES 2:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E EXIDE FHC-19 BATTERY CELLS.
NUREG/CR-4096: TEST SERIES 3:SEISMIC FRAGILITY TESTS OF NATURALLY-AGED CLASS 1E C&D LCU-13 BATTERY CELLS.
NUREG/CR-4097: TEST SERIES 4:SEISMIC-FRAGILITY TESTS OF NATURALLY-AGED EXIDE EMP-13 BATTERY CELLS.
NUREG/CR-4147: THE EFFECT OF ENVIRONMENTAL STRESS ON SYLGARD 70 SILICONE ELASTOMER.
NUREG/CR-4155: TRAC-PF1/MOD1 INDEPENDENT ASSESSMENT:NORTHWESTERN UNIVERSITY PERFORATED-PLATE CCFL TESTS.
NUREG/CR-4169: AN APPROACH TO TREATING RADIONUCLIDE DECAY HEATING FOR USE IN THE MELCOR CODE SYSTEM.
NUREG/CR-4197: SAFETY GOAL SENSITIVITY STUDIES.
NUREG/CR-4199: A DEMONSTRATION UNCERTAINTY/SENSITIVITY ANALYSIS USING THE HEALTH AND ECONOMIC CONSEQUENCE MODEL CRAC2.
NUREG/CR-4264: INVESTIGATION ON HIGH-EFFICIENCY PARTICULATE AIR FILTER PLUGGING BY COMBUSTION AEROSOLS.

SCIENCE APPLICATIONS INTERNATIONAL CORP., (FORMERLY SCIENCE APPLICATIONS
NUREG/CR-3905 V01 R1: SEQUENCE CODING AND SEARCH SYSTEM FOR LICENSEE EVENT REPORTS.User's Guide.
NUREG/CR-4101: ASSAY OF LONG-LIVED RADIONUCLIDES IN LOW-LEVEL WASTES FROM POWER REACTORS.

ST. LOUIS UNIV., ST. LOUIS, MO
NUREG/CR-4226: NEW MADRID SEISMOTECTONIC STUDY.Activities During Fiscal Year 1983.

STEVENSON & ASSOCIATES
NUREG-1061 V02 ADD: REPORT OF THE U.S. NUCLEAR REGULATORY COMMISSION PIPING REVIEW COMMITTEE.Volume 2 Addendum:Summary And Evaluation Of Historical Strong-Motion Earthquake Seismic Response And Damage To Aboveground Industrial Piping.

TEXAS, UNIV. OF, EL PASO, TX
NUREG/CR-3174 V02: GEOPHYSICAL-GEOLOGICAL STUDIES OF POSSIBLE EXTENSIONS OF THE NEW MADRID FAULT ZONE.Annual Report For 1983.

WESTINGHOUSE ELECTRIC CORP.
NUREG/CR-3455: A COMPARISON OF IODINE,KRYPTON,AND XENON RETENTION EFFICIENCIES FOR VARIOUS SILVER LOADED ADSORPTION MEDIA.

WISCONSIN, UNIV. OF, MADISON, WI
NUREG/CR-4131: INVESTIGATION OF ALTERNATIVE MEANS TO ACCOMPLISH THE GOALS OF BIENNIAL ION CHAMBER CALIBRATION.

Licensed Facility Index

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

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