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Vol. 13, No. 2

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
Second Quarter 1988
April - June

U.S. Nuclear Regulatory Commission

Office of Administration and Resources Management

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Regulatory Publications Branch
Division of Freedom of Information and Publications Services
Office of Administration and Resources Management
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CONTENTS

PREFACE

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

Division of Publications Services
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Washington, D.C. 20555

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

Secondary Report Number Index
Personal Author Index
Subject Index
NRC Originating Organization Index (Staff Reports)
NRC Originating Organization Index (International Agreements)
NRC Contract Sponsor Index (Contractor Reports)
Contractor Index
International Organization 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-0808: MARK II CONTAINMENT PROGRAM EVALUATION AND ACCEPTANCE CRITERIA. ANDERSON, C.J. Division of Safety Technology. August 1981. 90 pp. 8109140048. 09570:200.

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

Conference Report

NUREG/CP-0017: EXECUTIVE SEMINAR ON THE FUTURE ROLE OF RISK ASSESSMENT AND RELIABILITY ENGINEERING IN NUCLEAR REGULATION. JAMERP, J.S. Argonne National Laboratory. May 1981. 141 pp. 8105280299. ANL-81-3. 08634:070.

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).

International Agreement Report

NUREG/IA-0001: ASSESSMENT OF TRAC-PD2 USING SUPER CANNON AND HDR EXPERIMENTAL DATA. NEUMANN, U. Kraftwerk Union. August 1986. 223 pp. 8608270424. 37659:138.

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

The following abbreviations are used to identify the document status of a report:

ADD - addendum
APP - appendix
DRFT - draft
ERR - errata
N - number
R - revision
S - supplement
V - volume

Availability of NRC Publications

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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 and NUREG/IA is used for international agreement reports.

All these report codes are controlled and assigned by the staff of the Publishing and Translations Section of the NRC Division of Publications Services.

Main Citations and Abstracts

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

NUREG-0020 V12 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of February 29,1988.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). April 1988. 505pp. 8805020159. 45340:145.

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 Administration and Resources Management from the Headquarters staff of NRC's Office of Enforcement (OE), 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, OE 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 V12 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of March 31,1988.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). May 1988. 513pp. 8805200063. 45550:091.

See NUREG-0020,V12,N03 abstract.

NUREG-0020 V12 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of April 30,1988.(Gray Book I) SCHWARTZ,I. Division of Computer & Telecommunications Services (Post 870413). June 1988. 500pp. 8806230155. 45899:010.

See NUREG-0020,V12,N03 abstract.

NUREG-0040 V12 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1988.(White Book) * Division of Reactor Inspection & Safeguards (Post 870413). May 1988. 136pp. 8806230188. 45897:313.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from January 1988 through March 1988. Also included in this issue are the results of certain inspections performed prior to January 1987 that were not included in previous issues of NUREG-0040.

NUREG-0090 V10 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1987. * Office for Analysis & Evaluation of Operational Data, Director. March 1988. 45pp. 8804280308. 45270:244.

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, 1987. During the report period, there were two abnormal occurrences at the nuclear power plants licensed to operate. The first involved a significant degradation of plant safety at Oyster Creek; and the second involved a steam generator tube rupture at North Anna Unit 1. There were four abnormal occurrences at the other NRC licensees. The first involved a therapeutic medical misadministration; the second involved a failure to report diagnostic medical misadministrations; the third involved the suspension of a well logging company's license; and the fourth involved the suspension of an industrial radiography company's license. There were two abnormal occurrences reported by an Agreement State (New York). The first involved a hospital contamination incident and the second involved therapeutic medical misadministrations. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0090 V10 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.October-December 1987. * Office for Analysis & Evaluation of Operational Data, Director. March 1988. 34pp. 8804280329. 45270:213.

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, 1987. During the report period, there was one abnormal occurrence at the NRC licensees; the item involved the suspension of license of an oil and gas well tracer company for noncompliance with NRC regulatory requirements. There were no abnormal occurrences reported by the Agreement States. The report also contains information updating some previously reported abnormal occurrences.

NUREG-0304 V13 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1988,January-March. * Division of Freedom of Information & Publication Services (Post 880515). June 1988. 51pp. 8807110503. 46085:073.

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

NUREG-0540 V10 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-29,1988. * Division of Publication Services (870413-880514). April 1988. 346pp. 8804280623. 45273:038.

This document is a monthly publication containing descriptions of information received and generated by the U.S. Nuclear Regulatory Commission (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: Per-

sonal Author, Corporate Source, Report Number, and Cross Reference to Principal Documents.

NUREG-0540 V10 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31,1988. * Division of Publication Services (870413-880514). May 1988. 400pp. 8806020034. 45706:303.

See NUREG-0540,V10,N02 abstract.

NUREG-0540 V10 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30,1988. * Division of Freedom of Information & Publication Services (Post 880515). June 1988. 354pp. 8807070426. 46012:018.

See NUREG-0540,V10,N02 abstract.

NUREG-0683 S03 DRFT: PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT RELATED TO DECONTAMINATION AND DISPOSAL OF RADIOACTIVE WASTES RESULTING FROM MARCH 28,1979 ACCIDENT,THREE MILE ISLAND NUCLEAR STATION,UNIT 2.Docket No. 50-320.(GPU Nuclear,Incorporated) * Office of Nuclear Reactor Regulation, Director (Post 870411). April 1988. 204pp. 8804280348. 45270:009.

In accordance with the National Environmental Policy Act, the Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste from March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2 (PEIS) has been supplemented. This draft supplement addresses potential environmental impacts associated with the licensee's (GPU Nuclear's) proposal to place the TMI-2 facility in a post-defueling monitored storage mode followed by the completion of cleanup. The NRC staff has concluded, based on this evaluation, that the licensee's proposed plan and the NRC staff-identified alternatives for completion of cleanup are within applicable regulatory limits and could be implemented without significant environmental impact. No alternative was found to be clearly preferable from an environmental impact perspective. The staff concluded that the benefits of cleanup action outweigh the small associated impacts.

NUREG-0713 V07: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES 1985.Eighteenth Annual Report. BROOKS,B. Division of Regulatory Applications (Post 870413). April 1988. 168pp. 8804280390. 45271:136.

This report summarizes the occupational radiation exposure information that has been reported to the NRC's Radiation Exposure Information Reporting System (REIRS) by nuclear power facilities and certain other categories of NRC licensees during the years 1969 through 1985. The bulk of the data presented in the report was obtained from annual radiation exposure reports submitted in accordance with the requirements of 10 CFR 20.407. Data on workers terminating their employment at certain NRC licensed facilities were obtained from reports submitted pursuant to 10 CFR 20.408. The 1985 annual reports submitted by about 500 licensees indicated that approximately 216,000 individuals were monitored, 94,000 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.22 rem (cSv) and an average measureable dose of 0.43 rem (cSv). Termination radiation exposure reports were analyzed to reveal that about 77,300 individuals completed their employment with one or more of the 500 covered licensees during 1984. Some 73,200 of these individuals terminated from power reactor facilities, and about 7,400 of them were considered to be transient workers who received an average dose of 1.05 rem (cSv).

NUREG-0725 R06: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Division of Safeguards & Transportation (Post 870413). April 1988. 95pp. 8805200083. 45560:346.

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 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 September 30, 1987. Spent fuel shipment routes, primarily for road transportation, but also including three rail routes, are indicated on reproductions of Department of Transportation road maps. Also included are the amounts of material shipped during the appropriate eight-year period that safeguards regulations for spent fuel shipments have been efective. 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-0750 V26 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1987. * Division of Publication Services (870413-880514). March 1988. 50pp. 8804290204. 45324:282.

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 V26 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-December 1987. * Division of Publication Services (870413-880514). April 1988. 65pp. 8805200092. 45550:026.

See NUREG-0750,V26,I01 abstract.

NUREG-0750 V27 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1988. * Division of Freedom of Information & Publication Services (Post 880515). June 1988. 46pp. 8807080350. 46077:107.

See NUREG-0750,V26,I01 abstract.

NUREG-0750 V27 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1988. Pages 41-255. * Division of Publication Services (870413-880514). April 1988. 223pp. 8805090108. 45413:073.

Legal issuances of the Commission, the Atomic Safety and Licensing Appeal Panel, the Atomic Safety and Licensing Board Panel, the Administrative Law Judge, and NRC program offices are presented.

NUREG-0750 V27 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1988. Pages 257-334. * Division of Publication Services (870413-880514). May 1988. 169pp. 8806070066. 45726:299.

See NUREG-0750,V27,N02 abstract.

NUREG-0837 V07 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, October-December 1987. STRUCKMEYER,R.; MCNAMARA,N.; COHEN,L. Region 1, Ofc of the Director. April 1988. 323pp. 8804290185. 45283:099.

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

NUREG-0837 V08 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report, January-March 1988. STRUCKMEYER,R.; MCNAMARA,N. Region 1, Ofc of the Director. June 1988. 224pp. 8807110535. 46078:214.

This report provides the status and results of the NRC Thermoluminescent Dosimeter (TLD) Direct Radiation Monitoring Network. It presents the radiation levels measured in the vicinity

of NRC licensed facility sites throughout the country for the first quarter of 1988.

NUREG-0933 S07: A PRIORITIZATION OF GENERIC SAFETY ISSUES. ENRIT,R.; RIGGS,R.; MILSTEAD,W.; et al. Division of Regulatory Applications (Post 870413). April 1988. 291pp. 8805060326. 45408:020.

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

NUREG-0940 V07 N01: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,January-March 1988. * Ofc of Enforcement (Post 870413). June 1988. 250pp. 8807070438. 46014:048.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January - March 1988) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

NUREG-0975 V06: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING.Annual Report For FY 1987. * Materials Engineering Branch. June 1988. 425pp. 8807110528. 46083:319.

This report presents summaries of the research work performed during fiscal year 1987 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-1002 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company) * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). June 1988. 29pp. 8807060087. 46002:338.

In November 1983, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1002) regarding the application filed by the Commonwealth Edison Company, as applicant and owner, for a license to operate Braidwood Station, Units 1 and 2 (Docket Nos. 50-456 and 50-457). The first supplement to NUREG-1002 was issued in September 1986; the second supplement was issued in October 1986; the third supplement was issued in May 1987; the fourth supplement was issued in July 1987 in support of the full-power license for Unit 1; the fifth supplement was issued in December 1987 in support of the low-power license for Unit 2. This sixth supplement to NUREG-1002 is in support of the full-power license for Unit 2 and provides the status of items that remained unresolved at the time Supplement 5 was published. The facility is located in Reed Township, Will County, Illinois.

NUREG-1032: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-44.Final Report. BARANOWSKY,P.W. Office of Nuclear Regulatory Research, Director (Post 860720). BARANOWSKY,P.W. Office of Nuclear Reactor Regulation, Director (Post 870411). June 1988. 170pp. 8806100152. 45785:188.

"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 likelihood 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-1109: REGULATORY/BACKFIT ANALYSIS FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUE A-44,STATION BLACKOUT. RUBIN,A.M. Office of Nuclear Regulatory Research, Director (Post 860720). RUBIN,A.M. Office of Nuclear Reactor Regulation, Director (Post 870411). June 1988. 98pp. 8806100170. 45785:118.

Station blackout is the complete loss of alternating current (ac) electric power to the essential and nonessential buses in a nuclear power plant; it results when both offsite power and the onsite emergency ac power systems are unavailable. 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. Because of the concern about the frequency of loss of offsite power, the number of failures of emergency diesel generators, and the potentially severe consequences of a loss of all ac power, "Station Blackout" was designated as Unresolved Safety Issue (USI) A-44. This report presents the regulatory/backfit analysis for USI A-44. It includes (1) a summary of the issue, (2) the recommended technical resolution, (3) alternative resolutions considered by the Nuclear Regulatory Commission (NRC) staff, (4) an assessment of the benefits and costs of the recommended resolution, (5) the decision rationale, (6) the relationship between USI A-44 and other NRC programs and requirements, and (7) a backfit analysis demonstrating that the resolution of USI A-44 complies with the backfit rule (10 CFR 50.109).

NUREG-1125 V09: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1987 Annual. * ACRS - Advisory Committee on Reactor Safeguards. April 1988. 137pp. 8805030096. 45342:247.

This compilation contains 47 ACRS reports submitted to the Commission or to the Executive Director for Operations during calendar year 1987. It also includes a report to the Congress on the NRC Safety Research Program for FY 1988. All reports have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

NUREG-1214 R03: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. PERSINKO,D. Division of Licensee Performance & Quality Evaluation (Post 870411). May 1988. 78pp. 8805260374. 45632:048.

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

NUREG-1217 DRFT FC: EVALUATION OF SAFETY IMPLICATIONS OF CONTROL SYSTEMS IN LWR NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-47.Draft Report For Comment. SZUKIEWICZ,A.J. Division of Engineering (Post 870413). April 1988. 73pp. 8805260301. 45634:085.

This report summarizes the work performed by the Nuclear Regulatory Commission staff and its contractors, Idaho National Engineering Laboratories (INEL), Oak Ridge National Laboratory (ORNL), and Pacific Northwest Laboratory (PNL), leading to the proposed resolution of Unresolved Safety Issue (USI) A-47, "Safety Implications of Control Systems." An in-depth evaluation was performed on non- safety-grade control systems (see Section 1) that are typically used during normal plant operation on four nuclear steam system (NSS) plants: a General Electric Company (GE) boiling-water reactor (BWR), a 3-loop Westinghouse (W) pressurized-water reactor (PWR) design, a once-through steam generator PWR designed by Babcock & Wilcox Co. (B&W), and a Combustion Engineering (CE) PWR design. This report describes the technical studies performed by the laboratories, the NRC staff assessment of the results, the generic applicability of the evaluations, and the technical findings resulting from these studies.

NUREG-1218 DRFT FC: REGULATORY ANALYSIS FOR PROPOSED RESOLUTION OF USI A-47. Safety Implications Of Control Systems.Draft Rept For Comment. SZUKIEWICZ,A.J. Division of Engineering (Post 870413). April 1988. 82pp. 8805260236. 45631:014.

This report presents a summary of the regulatory analysis conducted by the NRC staff to evaluate the value impact of alternatives for the resolution of Unresolved Safety Issue (USI) A-47, "Safety Implications of Control Systems." The NRC staff proposed resolution is based on these analyses and the technical findings and conclusions presented in NUREG-1217. The staff has concluded that certain actions should be taken to improve safety in light-water reactor (LWR) plants. The actions recommended that certain plants upgrade their control systems to preclude reactor vessel/steam generator overfill events and to prevent steam generator dryout, modify their technical specification to periodically verify operability of these systems, and modify selected emergency procedures to ensure plant safe shutdown following a small-break loss-of-coolant accident.

NUREG-1221: SUMMARY,ANALYSIS AND RESPONSE TO PUBLIC COMMENTS ON PROPOSED AMENDMENTS TO 10 CFR PARTS 30,40,50,51,70 AND 72: DECOMMISSIONING CRITERIA FOR NUCLEAR FACILITIES. * Division of Engineering (Post 870413). June 1988. 600pp. 8807060104. 46000:182.

In February, 1985, the NRC issued for public comment proposed rules on decommissioning of nuclear facilities. Comment letters were received from 143 organizations and individuals. This report provides a summary and analysis and response to the public comments received.

NUREG-1226: DEVELOPMENT AND UTILIZATION OF THE NRC POLICY STATEMENT ON THE REGULATION OF ADVANCED NUCLEAR POWER PLANTS. WILLIAMS,P.M.; KING,T.L. Office of Nuclear Regulatory Research Director (Post 860720). June 1988. 49pp. 8807070137. 46013:217.

On March 26, 1985, the U.S. Nuclear Regulatory Commission issued for public comment a "Proposed Policy for Regulation of Advanced Nuclear Power Plants" (50 FR 11884). This report presents and discusses the Commission's final version of that policy as titled and published on July 8, 1986 "Regulation of Advanced Nuclear Power Plants, Statement of Policy" (51 FR 24643). It provides an overview of comments received from the public, of the significant changes from the proposed Policy Statement to the final Policy Statement, and of the Commission's response to six questions contained in the proposed Policy Statement. The report also discusses the definition for advanced reactors, the establishment of an Advanced Reactors Group, the staff review approach and information needs, and the utilization of the Policy Statement in relation to other NRC programs, including the policies for safety goals, severe accidents and standardization. In addition, guidance for advanced reactors with respect to operating experience, technology development, foreign information and data, and prototype testing is provided. Finally, a discussion on the use of less prescriptive and nonprescriptive design criteria for advanced reactors, which the Policy Statement encourages, is presented.

NUREG-1232 V02: SAFETY EVALUATION REPORT ON TENNESSEE VALLEY AUTHORITY, Sequoyah Nuclear Performance Plan. * Ofc of Special Projects. May 1988. 362pp. 8805260367. 45659:001.

This Safety Evaluation (SER) on the information submitted by the Tennessee Valley Authority (TVA) in its Sequoyah Nuclear Performance Plan, through Revision 2, and supporting documents has been prepared by the U.S. Nuclear Regulatory Commission staff. The Plan addresses the plant-specific concerns requiring resolution before startup of either of the Sequoyah units. In particular, the SER addresses required actions for Unit 2 restart. In many cases, the programmatic aspects for Unit 1 are identical to those for Unit 2; the staff will conduct inspections of implementation for those programs. Where the Unit 1 program is different, the staff evaluation will be provided in a supplement to this SER. On the basis of its review, the staff concludes that Sequoyah-specific issues have been resolved to the extent that would support restart of Sequoyah Unit 2.

NUREG-1233 DRFT FC: REGULATORY ANALYSIS FOR USI A-40, "SEISMIC DESIGN CRITERIA." Draft Report For Comment. SHAUKAT,S.K.; CHOKSHI,N.C.; ANDERSON,N.R. Division of Engineering (Post 870413). April 1988. 53pp. 8806060019. 45726:031.

This report consists of a regulatory analysis for Unresolved Safety Issue (USI) A-40, "Seismic Design Criteria." The regulatory analysis discusses the impact of the proposed changes in the Standard Review Plan (SRP) Sections 2.5.2, 3.7.1, 3.7.2 and 3.7.3 (NUREG-0800).

NUREG-1266 V02: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - 1987. * Office of Nuclear Regulatory Research, Director (Post 860720). May 1988. 59pp. 8805200095. 45549:327.

This report, the third in a series of annual reports, was prepared in response to congressional inquiries concerning how nuclear regulatory research is used. It summarizes the accomplishments of the Office of Nuclear Regulatory Research during 1987. The goal of this office is to ensure that research provides the technical bases for rulemaking and for related decisions in support of NRC licensing and inspection activities. This report describes both the direct contributions to scientific and technical knowledge with regard to nuclear safety and their regulatory applications.

NUREG-1273: TECHNICAL FINDINGS AND REGULATORY ANALYSIS FOR GENERIC SAFETY ISSUE II.E.4.3, "CONTAINMENT INTEGRITY CHECK." SERKIZ,A.W. Division of Reactor & Plant Systems (870413-880716). April 1988. 170pp. 8805030120. 45343:024.

This report contains the technical findings and regulatory analysis for Generic Safety Issue II.E.4.3, "Containment Integrity Check." An evaluation of the containment isolation history from 1965 to 1983 reveals that (except for a small number of events) containment integrity has been maintained and that the majority of reported events have been events related to exceeding Technical Specification limits (or 0.6 times the allowable leakage level). In addition, more recent risk analyses have shown that allowable leakage rates even if increased by a factor of 10 would not significantly increase risk. Potential method of continuous monitoring are identified and evaluated. Therefore, these technical findings and risk evaluations support closure of Generic Issue II.E.4.3.

NUREG-1283: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT PURDUE UNIVERSITY. * Standardization & Non-Power Reactor Project Directorate. April 1988. 69pp. 8805090116. 45432:303.

This Safety Evaluation Report for the application filed by Purdue University for a renewal of Operating License R-87 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned by Purdue University and is located on the campus in West Lafayette, Indiana. On the basis of its technical review, the staff concludes that the reactor facility can continue to be operated by the University without endangering the health and safety of the public or the environment.

NUREG-1296: THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON SAFETY-RELATED MOTOR-OPERATED VALVES - GENERIC ISSUE II.E.6.1. ROTHBERG,O. Division of Engineering (Post 870413). June 1988. 51pp. 8807110511. 46083:122.

The NRC regulatory positions, as stated in Regulatory Guide 1.106, Revision 1, have been identified by the Office for Analysis and Evaluation of Operational Data (AEOD) as potential contributors to valve motor burnout. AEOD is particularly concerned about the allowed policy of bypassing thermal overload devices during normal conditions. Regulatory Guide 1.106 favors compromising the function of thermal overload devices in order to avoid interfering with the safety-related operation of motor-operated valves. This report describes thermal overload devices and their use. It is concluded that the policies stated in Regulatory Guide 1.106 are technically correct and allow sufficient flexibility to allow the use of thermal overload protection without interfering with safety-related functions of motor-operated valves. However, it appears that licensees are needlessly bypassing or otherwise compromising the use of thermal overload protection. Some licensees are using inadequate design practices to size thermal overload devices. The problem of valve motor burnout is related to a lack of standards and uniform guidance for the design, installation, maintenance, and testing of motor overload protective devices. The NRC's Office of Nuclear Regulatory Research will contact several nuclear standards organizations to suggest that detailed guidance for thermal overload protection of motor-operated valves be developed.

NUREG-1306: NRC SAFETY SIGNIFICANCE ASSESSMENT TEAM REPORT ON ALLEGATIONS RELATED TO THE SOUTH TEXAS PROJECT,UNITS 1 & 2. * Division of Reactor Projects - III,IV,V & Special Projects (Post 870411). March 1988. 151pp. 8804280527. 45274:151.

This report provides the results of a review by the Safety Significance Assessment Team (SSAT) of the Nuclear Regulatory Commission (NRC) of alleged construction irregularities at Houston Lighting and Power Company's South Texas Project (STP), Units 1 and 2 (Docket Nos. 50-498 and 50-499), located in Matagorda County, Texas. These allegations were provided to the NRC by the Government Accountability Project (GAP) which received them from approximately 35 current and former employees of STP, and covered a wide range of concerns with hardware and quality assurance and control, and issues of management, harassment and intimidation and wrongdoing. Only those concerns considered by the SSAT to be technically-oriented were selected for review based on their possible safety significance, generic implications, specificity to a particular plant component, system or structure, and to provide a multidiscipline overview of the implementation and effectiveness of the STP Quality Assurance Program. The SSAT review of GAP's allegations has identified no substantive safety issue that would warrant delay in the NRC's consideration of a full-power license for STP Unit 1.

NUREG-1308: RADIOACTIVE MATERIAL IN THE WEST LAKE LANDFILL.Summary Report. * Division of Industrial & Medical Nuclear Safety (Post 870729). April 1988. 22pp. 8805090165. 45412:328.

The West Lake Landfill is located near the city of St. Louis in Bridgeton, St. Louis County, Missouri. The site has been used since 1962 for disposing of municipal refuse, industrial solid and liquid wastes, and construction demolition debris. This report summarizes the circumstances of the radioactive material found in the West Lake Landfill. Primary emphasis is on the radiological environmental aspects as they relate to potential disposition of the material.

NUREG-1311: FUNDING THE NRC TRAINING PROGRAM FOR STATES. LUBENAU,J.O.; CORLEY,J.H.; KAMMERER,C. State, Local & Indian Tribe Programs. June 1988. 77pp. 8807070455. 46011:276.

On February 3, 1988, the Commission received a briefing on the State, Local and Indian Tribes Programs of the NRC Office of Governmental and Public Affairs. The briefing included discussion of the Agreement State program and, more particularly, the training provided to State personnel to help them maintain programs which are adequate to protect public health and safety and compatible with the Commission's program. The Commission endorsed the NRC State training program but questioned the long-standing practice of paying the travel and per diem of State personnel approved to attend the NRC sponsored training. The staff was requested by the Commission to provide a report on this aspect. This report is the staff's response. It includes an evaluation of the practice of funding State travel and per diem costs for personnel attending NRC courses and other options to make this program more cost effective including utilizing whenever possible minimal cost Federal and commercial training facilities.

NUREG-1318: TECHNICAL POSITION ON ITEMS AND ACTIVITIES IN THE HIGH-LEVEL WASTE GEOLOGIC REPOSITORY PROGRAM SUBJECT TO QUALITY ASSURANCE REQUIREMENTS. DUNCAN,A.B.; BILHORN,S.G.; KENNEDY,J.E. Division of High Level Waste Management (Post 870413). April 1988. 28pp. 8805200014. 45549:155.

This document provides guidance on how to identify items and activities subject to Quality Assurance in the high-level nuclear waste repository program for pre-closure and post-closure phases of the repository. In the pre-closure phase, structures, systems, and components essential to the prevention or mitigation of an accident that could result in an off-site radiation dose of 0.5 rem or greater are termed "important to safety." In the post-closure phase, the barriers which are relied on to meet the containment and isolation requirements are defined as "important to waste isolation." These structures, systems, components, and barriers, and the activities related to their characterization, design, construction and operation are required to meet quality assurance (QA) criteria to provide confidence in the performance of the geologic repository. The list of structures, sys-

tems, and components important to safety and engineered barriers important to waste isolation is referred to as the "Q-List" and lies within the scope of the QA program.

NUREG-1320: NUCLEAR FUEL CYCLE FACILITY ACCIDENT ANALYSIS HANDBOOK. AYER,J.E.; CLARK,A.T.; LOYSEN,P.; et al. Division of Industrial & Medical Nuclear Safety (Post 870729). May 1988. 300pp. 8806100179. 45788:258.

The Nuclear Regulatory Commission (NRC) has sponsored a research program to develop improved methods for realistically evaluating the consequences of major accidents in nuclear fuel cycle facilities. These methods, along with supporting information and illustrative examples are provided in this handbook. Four types of facilities: fuel manufacturing, fuel reprocessing, waste storage and solidification, and spent fuel storage; and six accident types: fires, explosions, spills, tornadoes, nuclear criticalities, and equipment failures, are covered. Both computer and hand calculations are presented to estimate the source term from potential accidents. This source term information is used as input to computer codes which calculate the transport and release of mass, energy, and material throughout the facility and to the environment.

NUREG-1325: DISPOSITION OF RECOMMENDATIONS OF THE NATIONAL RESEARCH COUNCIL IN THE REPORT "REVITALIZING NUCLEAR SAFETY RESEARCH." * Office of Nuclear Regulatory Research, Director (Post 860720). June 1988. 43pp. 8807060093. 46002:293.

On December 8, 1986, the Committee on Nuclear Safety Research of the National Research Council submitted its report, "Revitalizing Nuclear Safety Research," to the U.S. Nuclear Regulatory Commission (NRC). The Commission and its staff have carefully reviewed the Committee's report and have extensively examined the planning, implementation, and management of NRC research programs in order to respond most effectively to the Committee's recommendations. This report presents the Commission's view of the Committee's report and describes the actions that are under way in response to its recommendations.

NUREG/CP-0075: PROCEEDINGS OF CSNI/NRC WORKSHOP ON DUCTILE PIPING FRACTURE MECHANICS. KANNINEN,M.F. Southwest Research Institute. May 1988. 164pp. 8807110512. CSNI 97. 46082:318.

This report contains the papers presented at a workshop meeting that was conducted to compare the various different elastic plastic fracture mechanics analysis methods that can be applied to assess the margin of safety in cracked nuclear plant pipes. A specific problem - a circumferentially cracked Type 304 stainless steel pipe in combined axial tension and bending - was addressed. The applied bending moments at crack growth initiation and at fracture instability were sought. Seven estimation type solutions were performed along with a benchmark elastic-plastic finite element solution. It was learned that precise specification of the material stress-strain curve must be made to obtain meaningful results. But, when applied under controlled conditions, the different estimation method solutions do provide reasonably consistent results. These results appear to be conservative in comparison with an elastic-plastic finite element solution that was performed to provide a comparison with these results.

NUREG/CP-0093: PROCEEDINGS OF THE MEETING ON ULTRASENSITIVE TECHNIQUES FOR MEASUREMENT OF URANIUM IN BIOLOGICAL SAMPLES AND THE NEPHROTOXICITY OF URANIUM. KATHREN,R.L.; WEBER,J.R. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1988. 172pp. 8805260245. PNL-6511. 45632:120.

Edited transcripts are provided of two public meetings sponsored by the Occupational Radiation Protection Branch of the Division of Radiation Programs and Earth Sciences, Nuclear Regulatory Commission. The first meeting, held on December 3, 1985, included nine presentations covering ultrasensitive techniques for measurement of uranium in biological specimens. Topics include laser-spectrometric techniques for uranium bioassay, correlation of urinary uranium samples with air sampling results in industrial settings, delayed neutron counting, laser-kinetic phosphometry, isotope dilution mass spectrometry, resonance ionization spectroscopy, fission track analysis, laser-induced fluoresence, and costs of sampling and processing. The nine presentations of the second meeting, on December 4, 1985, dealt with the nephrotoxicity of uranium. Among the topics were the physiology of the kidney, the effects of heavy metals on the kidney, animal studies in uranium nephrotoxicity, comparisons of kidney histology in nine humans, renal effects in uranium mill workers, renal damage from different uranium isotopes, and Canadian studies on uranium toxicity. Discussions following the presentations are included in the edited transcripts.

NUREG/CR-2000 V07 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1988. * Oak Ridge National Laboratory. April 1988. 163pp. 8805090093. ORNL/NSIC-200. 45412:001.

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one-month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-1061, "Instructions for Preparation of Data Entry Sheets for Licensee Event Reports." For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, "Licensee Event Report System - Description of Systems and Guidelines for Reporting," provides supporting guidance and information on the revised LER rule. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System.

NUREG/CR-2000 V07 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1988. * Oak Ridge National Laboratory. May 1988. 132pp. 8806020043. ORNL/NSIC-200. 45706:089.

See NUREG/CR-2000,V07,N03 abstract.

NUREG/CR-2000 V07 N5: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of May 1988. * Oak Ridge National Laboratory. June 1988. 147pp. 8807110514. ORNL/NSIC-200. 46083:169.

See NUREG/CR-2000,V07,N03 abstract.

NUREG/CR-3509: POWER SPECTRAL DENSITY FUNCTIONS COMPATIBLE WITH NRC REGULATORY GUIDE 1.60 RESPONSE SPECTRA. SHINOZUKA,M.; MOCHIO,T.; SAMARAS,E.F. Columbia Univ., New York, NY. June 1988. 71pp. 8807080360. 46077:033.

Among other possibilities, the Kanai-Tajimi power spectral density enhanced at the higher frequency range is found to be useful for generating ground acceleration time histories that satisfy NRC RG 1.60 requirements. The values of the parameters involved in the spectral density function are recommended for this purpose. Also, a suggestion is made as to in what way the power speciral density requirements can be placed in combination with those of the current NRC RG 1.60 to ensure both the response and power spectrum requirements.

NUREG/CR-3899 S01: UTILITY FINANCIAL STABILITY AND THE AVAILABILITY OF FUNDS FOR DECOMMISSIONING.An Analysis Of Internal And External Funding. SIEGEL,J.J. Engineering & Economics Research, Inc. June 1988. 38pp. 8807110543. 46078:173.

The NRC is currently developing final rules in the area of decommissioning nuclear facilities. A part of that rulemaking effort is assuring that funds will be available at the time of decommissioning of power reactors. Previous NRC reports, including NUREG/CR-3899, published September, 1984, have examined this issue by studying various funding methods. This report provides an update by considering public comments received on the NRC's proposed rule on decommissioning (published February, 1985) and by analyzing the relative level of assurance of internal and external reserves. In its analysis, the report makes use of specific case utility financial situations. The report concludes that from a financial standpoint, with the exception of PSNH, internal reserves currently provide sufficient assurance of funds for decommissioning. The report also concludes that the NRC should recommend changes in bankruptcy laws, including decommissioning obligations in utility prospectuses, and conduct periodic financial reviews of nuclear utilities due to changing economic conditions.

NUREG/CR-4082 V06: DEGRADED PIPING PROGRAM - PHASE II.Sixth Program Report. October 1986 - September 1987. WILKOWSKI,G.M.; AHMAD,J.; BARNES,C.R.; et al. Battelle Memorial Institute, Columbus Laboratories. April 1988. 333pp. 8805060305. BMI-2120. 45431:012.

Presented herein is an Annual Report of the U.S. NRC's Degraded Piping Program - Phase II. This is the sixth program report on this program. Prior reports were semiannual reports. The intent of this program is to experimentally validate and enhance available analytical methods for evaluating the mechanical behavior of nuclear power plant piping containing circumferentially oriented defects. Fifty-seven pipe experiments have been conducted to date. These and approximately fifty additional pipe experiments from other programs have been analyzed. In the analytical effort, a screening criterion has been developed to show when the net-section-collapse analysis is valid. This shows that even tough materials such as stainless steel can fail at less than net-section-collapse loads if the pipe diameter is sufficiently large. Numerous predictive J-estimation schemes have been evaluated and modified. A finite length surface-cracked pipe estimation scheme has also been developed and incorporated into a computer code called NRCPIPE. This code provides a convenient way of analyzing cracked pipe with a number of currently accepted analytical methods. Supporting research efforts involve investigating geometry effects on J-R curves, as well as characterizing the material properties for each pipe tested. The significance of all of the efforts to date relative to pipe fracture analyses and flaw assessment criteria are discussed.

NUREG/CR-4219 V04 N2: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1987. CORWIN,W.R. Oak Ridge National Laboratory. April 1988. 336pp. 8806020048. ORNL/TM-9593. 45705:113.

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

NUREG/CR-4315 V09 R1: EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS. Summary Status Report,Three Mile Island Unit 2,Radioactive Waste And Laundry Shipments. DOERGE,D.H.; HAFFNER,D.R. Westinghouse Hanford Co. June 1988. 48pp. 8807110504. 46085:120.

This document summarizes information concerning radioactive waste and laundry shipments from the Three Mile Island Nuclear Station Unit 2 to radioactive waste disposal sites and to protective clothing decontamination facilities (laundries) since the loss of coolant accident experienced on March 28, 1979. Data were collected from radioactive shipment records, summarized, and placed in a computerized information retrieval/manipulation system which permits extraction of specific information as required. Information contained in this report includes: waste disposal site locations, dose rates, curie content, waste description, container type and number, volumes and weights.

NUREG/CR-4605: TRAINING MANUAL ON STATISTICAL METHODS FOR NUCLEAR MATERIAL MANAGEMENT. JAECH,J.L. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1988. 450pp. 8806230207. PNL-5855. 45896:128.

This training manual on statistical methods for nuclear material management is a companion publication to a reference book entitled "Statistical Methods for Nuclear Material Management" edited by W. Michael Bowen and Carl A. Bennett. The training manual follows the reference book in its terminology, notation and methodology. It is intended to be used with the reference book either as a guide for self-paced instruction or as a problem manual for formal courses in statistics on this subject. It consists of a number of example problems for each chapter, following the chapters in the reference book, and worked solutions to all problems. The material is presented in loose-leaf form with each problem and solution starting on a new page so that in connection with its usage with short courses the problems may be handed the class participants first and the solutions later.

NUREG/CR-4639 V01: NUCLEAR COMPUTERIZED LIBRARY FOR ASSESSING REACTOR RELIABILITY (NUCLARR).Volume I.Summary Description. GERTMAN,D.I.; GILMORE,W.E.; GALYEAN,W.J.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). February 1988. 30pp. 8805090146. EGG-2458. 45413:043.

A data management system has been implemented which supports a variety of risk-related analyses and provides a repository of hardware component failure and human error probability data to the risk analyst. The Nuclear Computerized Library for Assessing Reactor Reliability, NUCLARR, is an interactive, graphically oriented system which resides on a personal computer (PC) or PC-compatible environment. An overview of the data management system, including a description of data collection, specification, data structure, and taxonomies, is presented in Volume I of this report. Programming activities, procedures for processing data, user's guide, and hard copy data manual are presented in Volumes II through V.

NUREG/CR-4665: CLOSEOUT OF IE BULLETIN 83-08 ELECTRICAL CIRCUIT BREAKERS WITH AN UNDERVOLTAGE TRIP FEATURE IN USE IN SAFETY-RELATED APPLICATIONS OTHER THAN THE REACTOR TRIP SYSTEM. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. April 1988. 41pp. 8805090133. PARAMETER IE162. 45413:001.

The NRC/IE issued Bulletin 83-08 December 28, 1983 because of concern about circuit breaker deficiencies reported per previous bulletins 83-01 and 83-04. The object of IEB 83-08 was to assure proper operation of circuit breakers with undervoltage trip attachments (UVTAs) in all safety-related applications other than use as reactor trip breakers (RTBs). The bulletin was issued for action to all licensees and holders of construction permits of power reactors. Evaluation of utility responses and NRC/Region inspection reports shows that the bulletin can be closed out per specific criteria for 123 (99%) of

the 124 facilities to which it was issued. A followup item is proposed for use by the NRC to assure completion of required actions at the remaining facility. Circuit breakers with UVTAs were used in safety-related applications other than the reactor trip system in six (6) facilities. Malfunctions of the UTVAs were reported only for the facility for which followup is proposed.

NUREG/CR-4674 V05: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1986,A STATUS REPORT. MINARICK,J.W.; HARRIS,J.D.; AUSTIN,P.N.; et al. Oak Ridge National Laboratory. May 1988. 183pp. 8805260278. ORNL/NOAC-232. 45650:260.

Thirty-five operational events, reported in licensee event reports and occurring at commercial LWRs during 1986, are considered to be precursors to potential severe core damage. These are described along with associated significance estimates, categorization, and subsequent analyses. This study is a continuation of earlier work, which evaluated the 1969-1981 and 1984-1985 events. The report discusses (1) the general rationale for this study, (2) the selection and documentation of events as precursors, (3) the estimation and use of conditional probabilities of subsequent severe core damage to rank precursor events, and (4) the initial conclusions from the assessment of 1986 events and from the collective assessment of 1984-1986 events.

NUREG/CR-4674 V06: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1986,A STATUS REPORT. MINARICK,J.W.; HARRIS,J.D.; AUSTIN,P.N.; et al. Oak Ridge National Laboratory. May 1988. 315pp. 8805260286. ORNL/NOAC-232. 45631:093.

See NUREG/CR-4674,V05 abstract.

NUREG/CR-4728: EQUIPMENT QUALIFICATION RESEARCH TEST OF A HIGH-RANGE RADIATION MONITOR. RICHARDS,E.H.; JACOBUS,M.J.; DROZDA,P.M.; et al. Sandia National Laboratories. February 1988. 64pp. 8806230158. SAND86-1938. 45898:305.

A high-range radiation detector was tested in a simultaneous steam and radiation environment simulating a postulated loss-of- coolant accident (LOCA) to assess possible synergistic effects that may be important to its performance in an accident. The detector, manufactured by General Atomic, was simultaneously subjected to a simulated accident environment including 171 degree C (340 degree F) steam at 410 kPa gage (60 psig) and 4 Mrad/hr gamma radiation while its performance was monitored. Test results showed that the detector successfully operated at the high dose rate and temperature, without evidence of synergisms. However, at reduced radiation levels in a saturated steam environment, the detector signal at the readout module deteriorated in accuracy or ceased altogether. The cause of these anomalies is attributed to leakage currents and/or possible galvanic action in the coaxial connections and/or cables.

NUREG/CR-4735 V03: EVALUATION AND COMPILATION OF DOE WASTE PACKAGE TEST DATA. Biannual Report:February-July 1987. INTERRANTE,C.; ESCALANTE,E.; FRAKER,A.; et al. Commerce Dept. of, National Bureau of Standards. May 1988. 160pp. 8806070059. 45721:023.

This report summarizes evaluations by the National Bureau of Standards (NBS) of Department of Energy activities on waste packages designed for containment of high-level nuclear waste (HLW). The waste package is a proposed engineered barrier that is part of a permanent repository for HLW. Metal alloys are the principal barriers within the engineered barrier system. The corrosion of candidate container materials, particularly carbon steels, stainless steels and copper, is discussed. The level of understanding of several proposed container materials is questioned for the candidate repository in tuff. Three issues are addressed: (1) The possibility of stress-induced failure of Zircaloy, (2) possible corrosion of copper and copper alloys and (3) the lack of site-specific characterization data. The Basalt Waste Isolation Project (BWIP) section discusses localized corrosion and environmentally- assisted cracking of AISI 1020 steel at elevated temperatures (150 degree C). For the proposed salt site, the importance of the duration of corrosion tests and conditions that may preclude prompt initiation of long-term testing are discussed. NBS work related to vitrification of HLW borosilicate glass at the West Valley Demonstration Project (WVDP) and at the Defense Waste Processing Facility (DWPF) is discussed. Activities of the Materials Characterization Center (MCC) are presented. To stress the necessity of independent data interpretations, an illustration is given of a conclusion drawn from published data that is different from the conclusion reached by the investigator.

NUREG/CR-4763: SAFETY-RELATED EQUIPMENT SURVIVAL IN HYDROGEN BURNS IN LARGE DRY PWR CONTAINMENT BUILDINGS. KING,D.B.; NICOLETTE,J.F.; DANDINI,V.J.; et al. Sandia National Laboratories. March 1988. 299pp. 8804280588. SAND86-2280. 45272:099.

Analytical and experimental investigations of equipment survival in hydrogen burns in large dry PWR containment buildings have been conducted. Both atmospheric and subatmospheric containments were considered. Two sets of analytical studies were carried out for atmospheric large dry containments. One set analyzed the hydrogen burn that occurred as a result of the March 1979, accident at Three Mile Island. The other set considered a hybrid power plant consisting of the Zion reactor housed in the TMI-2 containment building. An analytical study of subatmospheric containments was also carried out using a model of the Surry nuclear power plant. To complement the analyses, a series of experiments simulating hydrogen burns in large dry containments was also conducted using the Sandia severe combined environment test chamber (SCETCh). The experiments investigated the survivability of thermally and radiation aged nuclear qualified Brand Rex power and control cable and a Barton 763 pressure transmitter in a simulated LOCA/hydrogen burn environment.

NUREG/CR-4777: STEAM OXIDATION OF ZIRCALOY CLADDING IN THE ORNL FISSION PRODUCT RELEASE TESTS. YAMASHITA,T. Oak Ridge National Laboratory. March 1988. 58pp. 8805200101. ORNL/TM-10272. 45560:051.

A simple model (ZIRCOX) has been developed to calculate the extent of oxidation of Zircaloy cladding in the vertical furnace tests conducted at ORNL. The model is based on the fact that the oxidation of Zircaloy is governed by the parabolic rate law. By introducing the equivalent time t*, the model is able to handle nonisothermal oxidation as well as isothermal. The degree of cladding oxidation and H(2) production were calculated using three sets of rate constants for the Zircaloy oxidation, and these results were compared with those measured experimentally. The calculated results showed excellent agreement with the experiments. The rate constant for Zircaloy oxidation that gave the best agreement was one based upon combined Pawel et al. and Prater and Courtright data. The temperature difference between the cladding and the zirconia furnace tube liner was estimated by the HEATING6 heat transfer program and was found to be about 50 K for the maximum oxidation rate. The heating atmosphere during test VI-1 was also evaluated in terms of the oxygen potential.

NUREG/CR-4778: PRELIMINARY STUDIES OF THE MORPHOLOGY OF THERMAL GRADIENT TUBE DEPOSITS FROM FISSION PRODUCT RELEASE EXPERIMENTS. WISBEY,S.J. Oak Ridge National Laboratory. March 1988. 57pp. 8805090156. ORNL/TM-10273. 45413:296.

Sections of thermal gradient tubes and deposits from filters used as collectors in several fission product release tests at ORNL have been examined by scanning electron microscopy with elemental identification by energy dispersive X-ray analysis. Shape, size, and composition of the deposits are reported; correlations with experimental conditions such as gas composition and temperature, as well as with independent analyses, have

been made where possible. A wide variety of shapes and structures, apparently dependent on the deposition temperature, were photographed and elemental analyses were recorded. Although some fission products (Cs, Ba, Ag) were detected, structural and impurity elements (Sn, Si, S, W, Pt) were predominant in most cases. Recommendations for analytical procedures and handling of similar samples are made for the future.

NUREG/CR-4836: APPROACHES TO UNCERTAINTY ANALYSIS IN PROBABILISTIC RISK ASSESSMENT. BOHN,M.P.; WHEELER,T.A.; PARRY,G.W. Sandia National Laboratories. January 1988. 108pp. 8805090083. SAND87-0871. 45449:001.

An integral part of any probabilistic risk assessment (PRA) is the performance of an uncertainty analysis to quantify the uncertainty in the point estimates of the risk measures considered. While a variety of classical methods of uncertainty analysis exist, application of these methods and developing new techniques consistent with existing PRA data bases and the need for expert (subjective) input has been an area of considerable interest since the pioneering Reactor Safety Study (WASH-1400) in 1975. This report presents the results of a critical review of existing methods for performing uncertainty analyses for PRAs, with special emphasis on identifying data base limitations on the various methods. Both classical and Baysian approaches have been examined. This work was funded by the U.S. Nuclear Regulatory Commission in support of its ongoing full- scope PRA of the LaSalle nuclear power station. Thus, in addition to the review, this report contains recommendations for a suitable uncertainty analysis methodology for the LaSalle PRA.

NUREG/CR-4864 V01: THERMODYNAMIC TABLES FOR NUCLEAR WASTE ISOLATION.Aqueous Solutions Database. PHILLIPS,S.L.; HALE,F.V.; SILVESTER,L.F.; et al. Lawrence Berkeley Laboratory. June 1988. 195pp. 8807110498. LBL-22860. 46082:118.

Tables of consistent thermodynamic property values for nuclear waste isolation are given. The tables include critically assessed values for Gibbs energy of formation, enthalpy of formation, entropy and heat capacity for generic minerals; solids; aqueous ions; ion pairs and complex ions of selected actinide and fission decay products at 25 degree C and zero ionic strength. These intrinsic data are used to calculate equilibrium constants and standard potentials which are compared with typical experimental measurements and other work. Recommendations for additional research are given.

NUREG/CR-4879 V01: DEMONSTRATION OF PERFORMANCE MODELING OF A LOW-LEVEL WASTE SHALLOW-LAND BURIAL SITE.A Comparison Of Predictive Radionuclide Transport Modeling Versus Field Observations At The Nitrate Disposal Pit Site,Chalk River Nuclear Labs. ROBERTSON,D.E.; BERGERON,M.P.; MYERS,D.A.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. November 1987. 137pp. 8805090112. PNL-6175. 45412:164.

Before a license can be obtained to construct a facility for the shallow-land burial of low-level waste, the U.S. Nuclear Regulatory Commission must be assured that the facilities will meet both performance objectives and prescriptive requirements set forth in 10CFR61, "Licensing Requirements for Land Disposal of Radioactive Waste." Subpart D, Section 61.50(a)(2) of 10CFR61 states that a "disposal site shall be capable of being characterized, modeled, analyzed and monitored." In order to test the concept of "site modelability," a 30-year old low-level radioactive waste disposal site at Chalk River Nuclear Laboratories (CRNL), Canada, was used as a field location for evaluating the process of site characterization and the subsequent modeling predictions of radionuclide transport from the site by groundwater. This evaluation was performed by comparing the actual measured radionuclide migration with predicted migration estimated from hydrologic/radionuclide transport models. This comparison has provided valuable insights into the applicability of transport modeling, and in determining what level of effort is needed in site characterization at locations similar to the Nitrate Disposal Pit to provide the desired degree of predictive capabilities.

NUREG/CR-4881: FISSION PRODUCT RELEASE CHARACTERISTICS INTO CONTAINMENT UNDER DESIGN BASIS AND SEVERE ACCIDENT CONDITIONS. NOURBAKHSH,H.P.; KHATIB-RAHBAR; DAVIS,R.E. Brookhaven National Laboratory. March 1988. 127pp. 8807070446. BNL-NUREG-52059. 46013:284.

A detailed review of the radiological release estimates for light water reactor accident sequences is presented as a basis for development of a simplified approach for prediction of characteristics of radiological releases into containments under design basis and severe accident conditions for both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). Resulting source term estimates are also compared with parallel results using the Source Term Code Package (STCP) methodology.

NUREG/CR-4932: CLOSEOUT OF IE BULLETIN 80-03:LOSS OF CHARCOAL FROM STANDARD TYPE II,TWO-INCH,TRAY ADSORBER CELLS. DEAN,R.S.; FOLEY,W.J.; HENNICK,A. Parameter, Inc. April 1988. 28pp. 8805200093. PARAMETER IE167. 45560:318.

Because of concern about defective charcoal tray adsorber cells found in certain ventilation systems at the Sequoyah Nuclear Plant, the NRC/IE issued IE Bulletin 80-03 on February 6, 1980. Some charcoal cells are used in ventilation systems associated with engineered safety features, which are provided for protection from abnormal events. Others are installed to control radioactive materials during expected operations. Licensees of operating power reactors and holders of permits for those under construction were required to take specific actions. Evaluation of utility responses and NRC/Region inspection reports shows that the bulletin can be closed out by means of specific criteria for 123 (99%) of the 124 facilities with operating licenses or construction permits. A followup item is proposed for the only facility with open bulletin status, for use by the NRC in ensuring satisfactory completion of corrective action. The cells with riveted screens which were identified at Sequoyah were not found at any other facility. Although cells with miscellaneous defects were found at nine facilities other than Sequoyah, there were no charcoal problems.

NUREG/CR-4933: CLOSEOUT OF IE BULLETIN 80-19:FAILURES OF MERCURY-WETTED MATRIX RELAYS IN REACTOR PROTECTIVE SYSTEMS OF OPERATING NUCLEAR POWER PLANTS DESIGNED BY COMBUSTION ENGINEERING. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. Parameter, Inc. April 1988. 26pp. 8805090125. PARAMETER IE168. 45412:301.

The NRC/IE issued IE Bulletin 80-19 initially on July 31, 1980, and issued Revision 1 of the bulletin on August 13, 1980. The bulletin was issued to all licensees and holders of construction permits of power reactors, because of numerous reports about single and multiple failures of C.P. Clare Model HG2X-1011 mercury-wetted matrix relays in reactor protective systems. The concern based on those reports was the build-up of coincident "failed-closed" failures of certain sets of relays could result in trip failures for off-normal events. Evaluation of utility responses and NRC/Region inspection reports shows that the bulletin can be closed out per specific criteria for 100% of the 123 power facilities with operating licenses or construction permits. In effect, all of the responses and inspection reports apply to Revision 1 of the bulletin. All except the three following facilities either do not have the subject relays in the reactor protective system, or have changed the relays to an acceptable type as a result of the bulletin. An inspection report closed the bulletin for Calvert Cliffs 1 & 2 based on licensee commitments to replace the mercury-wetted relays with the dry-contact type. For St. Lucie 1, the licensee elected to continue to use the subject relays and presented acceptable justification.

NUREG/CR-4971: RESULTS OF SEMISCALE MOD-2C FEEDWATER AND STEAM LINE BREAK (S-FS) EXPERIMENT SERIES.Main Steam Line Break Accident Experiments. BOUCHER,T.J. EG&G Idaho, Inc. (subs. of EG&G, Inc.). March 1988. 112pp. 8804280505. EGG-2516. 45274:302.

This report presents the results of two experiments conducted in the Semiscale Mod-2C facility which simulated main steam line break accidents at high pressure and temperature. Tests S-FS-1 and S-FS-2 simulated double-ended offset shears of the main steam line downstream and upstream, respectively, of the flow restrictor. Initial and boundary conditions were scaled from, and compounding failures and assumptions simulated, those conditions utilized for Final Safety Analysis Report (FSAR) calculations. Primary and secondary thermal- hydraulic responses are characterized, and the influence of the break size or location on the responses is discussed. The limiting of primary-to-secondary heat transfer by conduction heat transfer is shown to produce a trend of increased primary cooling with decreased break size, pointing out the need for further analysis for smaller break sizes. The degree of conservatism inherent in FSAR separator performance and break size and location assumptions is shown to be questionable, and the FSAR assumption of a loss of offsite power is shown to be nonconservative. The effectiveness of the recovery operations in regaining and maintaining control of the system is addressed; and main steam line break issues are discussed. Finally, conclusions are drawn and recommendations are made for further utilization of the data.

NUREG/CR-5000: METHODOLOGY FOR UNCERTAINTY ESTIMATION IN NUREG-1150 (DRAFT).Conclusions Of A Review Panel. KOUTS,H.; CORNELL,A.; FARMER,R.; et al. Brookhaven National Laboratory. December 1987. 23pp. 8806230139. BNL-NUREG-52119. 45896:102.

A review has been undertaken by a panel of experts, of the methodology for estimation of uncertainty in severe accident risk resulting from accidents to nuclear power plants as presented in the Draft NUREG-1150 report. This report provides detailed discussions and conclusions resulting from this review process.

NUREG/CR-5013: FATIGUE LIFE CHARACTERIZATION OF SMOOTH AND NOTCHED PIPING STEEL SPECIMENS IN 288 DEGREE C AIR ENVIRONMENTS. TERRELL,J.B. Materials Engineering Associates, Inc. May 1988. 77pp. 8805200115. MEA-2232. 45561:256.

Fatigue strain-life tests were conducted on ASME SA 106-B piping steel at 24 degree C (76 degree F) and at PWR operating temperature, 288 degree C (550 degree F), under completely reversed loading. Smooth specimens were tested at both temperatures whereas specimens containing notches of various acuities were tested at 288 degree C. Fatigue limits at 10(7) cycles were estimated to be 185 MPa (26.8 ksi) at 24 degree C and 232 MPa (33.7 ksi) at 288 degree C. The difference in fatigue strength observed at the PWR temperature is postulated to be due to dynamic strain aging processes. However, there is a reduction in low cycle fatigue strength at this temperature which results in a decrease in the intended safety factor of the ASME Section III design curve for carbon steels. Notch strain histories were estimated for the notched specimen tests using various interpretations of Neuber's rule. It was concluded that the use of the fatigue notch concentration factor (Kf) in the Neuber relation in conjunction with the uniaxial cyclic stress-strain curve provided the best correlation of notched specimen fatigue data with results obtained from smooth specimen tests. The notched specimen strain-life results derived from the application of Neuber's rule alone proved to be conservative when compared to smooth specimen test results to such an extent that Neuber-generated notch stress and strain amplitudes cannot be compared to the ASME Section III fatigue curves for carbon steels.

NUREG/CR-5015: IMPROVED RELIABILITY OF RESIDUAL HEAT REMOVAL CAPABILITY IN PWRS AS RELATED TO RESOLUTION OF GENERIC ISSUE 99. CHU,T.L.; FITZPATRICK,R.; YOON,W.H.; et al. Brookhaven National Laboratory. May 1988. 159pp. 8006070054. BNL-NUREG-52121. 45727:184.

This report summarizes a study performed by Brookhaven National Laboratory for the Reactor and Plant Safety Issues Branch, RES of the U.S. Nuclear Regulatory Commission in pursuit of the resolution of NRC Generic Issue 99. Generic Issue 99 focuses on the risk associated with loss of residual heat removal events at PWRs while shut down. Numerous loss of residual heat removal events have occurred at pressurized water reactors (PWRs) in the USA, which were terminated prior to damaging the reactor core. This study estimates the risk from loss of residual heat removal events and investigates ways of lowering this risk.

NUREG/CR-5016: COMPENDIUM AND COMPARISON OF INTERNATIONAL PRACTICE FOR PLUGGING,REPAIR AND INSPECTION OF STEAM GENERATOR TUBING. CLARK,R.A.; KURTZ,R.J. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1988. 71pp. 8805090114. PNL-6341. 45432:233.

The Committee on the Safety of Nuclear Installations (CSNI) of the Organization for Economic Cooperation and Development - Nuclear Energy Agency (OECD-NEA) is an international body of scientists and engineers with responsibilities for nuclear safety research and nuclear licensing. The CSNI fosters international cooperation in nuclear safety amongst OECD member countries. In July 1986, the Committee's Principal Working Group on Primary Circuit Integrity agreed it would be useful to those setting criteria and making decisions about plugging of degraded steam generator tubes to have a better appreciation of the criteria presently employed in other members countries and their technical bases. The United States Nuclear Regulatory Commission (U.S. NRC) offered to arrange the preparation of a comparative summary for CSNI based on responses to a questionnaire circulated by the OECD-NEA among member countries. The following report is based upon the information obtained from nine countries currently operating pressurized water reactors (PWRs).

NUREG/CR-5019: NEUTRON EXPOSURE PARAMETERS FOR THE METALLURGICAL TEST SPECIMENS IN THE FIFTH HEAVY-SECTION STEEL TECHNOLOGY IRRADIATION SERIES CAPSULES. MILLER,L.F.; BALDWIN,C.A.; STALLMANN,F.W.; et al. Oak Ridge National Laboratory. March 1988. 127pp. 8804280633. ORNL/TM-10582. 45274:024.

The Heavy-Section Steel Technology (HSST) Program, supported by the U.S. Nuclear Regulatory Commission, has completed the Series 5 (HSST5) irradiation experiments. Twelve capsules which contain metallurgical test specimens have been irradiated at the Oak Ridge Research Reactor located at the Oak Ridge National Laboratory. These capsules have been disassembled, internal dosimeters have been analyzed, and exposure parameters are presented for each irradiation test specimen. This report describes the computational methodology for the least-squares adjustment of the dosimetry data with neutronics calculations, and it presents exposure parameters at each test specimen location for the fluence rate greater than 1.0 MeV, fluence rate greater than 0.1 MeV, and displacements per atom. The specific activity of each dosimeter at the end of irradiation is listed in the Appendix.

NUREG/CR-5021 V01: USER'S GUIDE FOR PRISIM ARKANSAS NUCLEAR ONE - UNIT 1.Volume 1,Program For Inspectors. CAMPBELL,D.J.; GUTHRIE,V.H.; KIRCHNER,J.R.; et al. Oak Ridge National Laboratory. March 1988. 192pp. 8805090144. ORNL/TM-10604. 45434:154.

This user's guide is a two-volume document designed to teach NRC inspectors and NRC regulators how to access probabilistic risk assessment information from the two Plant Risk

Status Information Management System (PRISIM) programs developed for Arkansas Nuclear One - Unit One (ANO-1). Volume 1 describes how the PRA information available in Version 1.0 of PRISIM is useful for planning inspections. Using PRISIM, inspectors can quickly access PRA information and use that information to update risk analysis results,reflecting a plant's status at any particular time. Volume 2 describes how the PRA information available in Version 2.0 of PRISIM is useful as an evaluation tool for regulatory activities. Using PRISIM regulators can both access PRA information and modify the information to assess the impact these changes may have on plant safety. Both volumes are stand-alone documents; each volume presents several sample computer sessions designed to lead the user through a variety of PRISIM applications used to obtain PRA-related information for monitoring and controlling plant risk.

NUREG/CR-5021 V02: USER'S GUIDE FOR PRISIM ARKANSAS NUCLEAR ONE - UNIT 1.Volume 2,Program For Regulators. CAMPBELL,D.J.; GUTHRIE,V.H.; KIRCHNER,J.R., et al. Oak Ridge National Laboratory. March 1988. 224pp. 8805090170. ORNL/TM-10604. 45433:290.

This user's guide is a two-volume document designed to teach NRC inspectors and NRC regulators how to access probabilistic risk assessment information from the two Plant Risk Status Information Management System (PRISIM) programs developed for Arkansas Nuclear One - Unit One (ANO-1). Volume 1 describes how the PRA information available in Version 1.0 of PRISIM is useful for planning inspections. Using PRISIM, inspectors can quickly access PRA information and use that information to update risk analysis results, reflecting a plant's status at any particular time. Volume 2 describes how the PRA information available in Version 2.0 of PRISIM is useful as an evaluation tool for regulatory activities. Using PRISIM, regulators can both access PRA information and modify the information to assess the impact these changes may have on plant safety. Each volume is a stand-alone document.

NUREG/CR-5038: OPTIMIZATION OF THE CONTROL OF CONTAMINATION AT NUCLEAR POWER PLANTS. KHAN,T.A.; BAUM,J.W. Brookhaven National Laboratory. May 1988. 146pp. 8806230171. BNL-NUREG-52073. 45898:084.

A methodology is described for the optimization of the actions taken to control contamination. It deals with many aspects of contamination, such as the monetary value assigned to a unit of radiation dose, the treatment of skin and extremity dose, and the inefficiencies introduced from working in a contaminated environment. The optimization method is illustrated with two case studies based on cleanup projects at nuclear power plants. Guidelines for the use of protective apparel, and for monitoring radiation and contamination at various levels of contamination are presented. The report concludes that additional research is required to quantify the effect of a contaminated environment on work efficiencies.

NUREG/CR-5042 S01: EVALUATION OF EXTERNAL HAZARDS TO NUCLEAR POWER PLANTS IN THE UNITED STATES.Seismic Hazard. PRASSINOS,P.G. Lawrence Livermore National Laboratory. April 1988. 64pp. 8805030107. UCID-21223. 45343:194.

As part of the research program supporting the implementation of the NRC Policy Statement on Severe Accidents, the Lawrence Livermore National Laboratory (LLNL) has performed a study of the risk of core damage to nuclear power plants in the United States due to seismic initiated events. The broad objective has been to gain an understanding of whether or not seismic events are among the major potential accident initiators that may pose a threat of severe reactor core damage or of large radioactive release to the environment from the reactor. The analysis was based on two figures-of-merit, one based on core damage frequency and the other based on the frequency of large radioactive releases. Using these two figures-of-merit as evaluation criteria, it has been possible to ascertain that the risk from seismic initiated accidents is an important contributor to overall risk for the U.S. nuclear power plants studies.

NUREG/CR-5044: ESTIMATION TECHNIQUES FOR COMMON CAUSE FAILURE EVENTS. KELLY,E.J.; HEMPHILL,G.M. Los Alamos National Laboratory. March 1988. 53pp. 8806100174. LA-11179-MS. 45788:205.

Common cause failure probability estimation techniques, including B factor, basic parameter, binomial failure rate, multiple Greek, and C-factor estimators, are evaluated and compared using simulation data that captures the real world problem of sparse data from different plants. The effects on the estimators' performances from underlying factors such as common cause shock rates, lethal shock rates probability of failing given a shock, independent failure rates, and system operational time are discussed. Worst case results are reported, and it is seen that for extremely small common cause failure probabilities the binomial failure rate estimators are best. However, these estimators can underestimate the true probabilities when the failures deviate from the binomial failure rate model. The B-factor technique is shown to be conservative, and in some cases to overestimate the true probability by several orders of magnitude. When there are observed failures for each failure event, the basic parameter technique is best and is easily calculated. This estimator is investigated in detail and is used to develop an estimator for the probability of K or more units failing due to a common cause. Uncertainty limits for this probability are also developed.

NUREG/CR-5045: KANSAS-NEBRASKA SEISMICITY STUDIES USING THE KANSAS-NEBRASKA MICROEARTHQUAKE NETWORK.Final Report. STEEPLES,D.W.; HILDEBRAND,G.M.; BENNETT,B.C.; et al. Kansas, Univ. of, Lawrence, KS. March 1988. 78pp. 8806020039. 45706:222.

The Kansas Geological Survey (KGS) operates a 15 station seismograph network with stations located in northeast Kansas and southeast Nebraska. The network is supported in part by funding from the United States Nuclear Regulatory Commission (NRC). This report discusses operation of the network and summarizes the results of research that allows a better understanding of the seismicity of the region and the link between the seismicity and the tectonic setting of the region. Results from a crustal seismic refraction survey along a 500 km long line are discussed in Appendix B to the report.

NUREG/CR-5055: ATMOSPHERIC DIFFUSION FOR CONTROL ROOM HABITABILITY ASSESSMENTS. RAMSDELL,J.V. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1988. 120pp. 8805260384. PNL-6391. 45632:292.

This report presents the results of an evaluation of the procedure used by the NRC staff to assess nuclear reactor control room habitability. The evaluation is based on experimental data collected in seven sets of field experiments at nuclear power plant sites. The procedure is generally conservative, but the models in the procedure show little skill in predicting the effects of different atmospheric conditions on maximum effluent concentrations in building wakes. Two alternative building-wake models have been developed using the experimental data. The first model differs significantly from current models in the manner in which wind speed enters the model. The second model is an extension of the first model that has more desirable asymptotic behavior and includes consideration of the mitigating effect of plume rise on concentrations in building wakes. A set of non-mathematical guidelines is offered for use in evaluating potential control room air intake locations.

NUREG/CR-5067: EARLY AND CONTINUING EFFECTS OF COMBINED ALPHA AND BETA IRRADIATION OF THE LUNG.Phase II Report. SCOTT,B.R.; HAHN,F.F.; SNIPES,M.B.; et al. Lovelace Biomed & Environmental Research Institute. March 1988. 41pp. 8804280444. LMF-119. 45269:001.

This report summarizes an inhalation exposure experiment that concerns early and continuing effects of combined alpha

and beta irradiation of the lung of rats. Both morbidity at 18 months and mortality within 18 months after exposure were examined for rats exposed to the beta-emitter (147)Pm, the alpha-emitter (238)Pu, or both combined. The results were used to validate hazard-function models that were developed (1) for pulmonary functional morbidity at 18 months and (2) for lethality from radiation pneumonitis and pulmonary fibrosis within 18 months. Both models were found to adequately predict the experimental observations after combined chronic alpha and beta irradiation of the lung. A relative biological effectiveness of approximately 7 was obtained for (238)Pu alpha radiation compared to (147)Pm beta radiation for both pulmonary functional morbidity and lethality from radiation pneumonitis and pulmonary fibrosis.

NUREG/CR-5072: DECAY HEAT REMOVAL USING FEED-AND-BLEED FOR U.S. PRESSURIZED WATER REACTORS. LOOMIS,G.; COZZUOL,J.M. EG&G Idaho, Inc. (subs. of EG&G, Inc.). * Idaho National Engineering Laboratory. June 1988. 48pp. 8807110521. EGG-2526. 46085:024.

As part of a United States Nuclear Regulatory Commission (USNRC) evaluation of current decay heat removal methods, the adequacy of feed-and-bleed decay heat removal has been assessed for United States pressurized water reactors (USPWRs). Use of feed-and-bleed for decay heat removal becomes necessary in a pressurized water reactor (PWR) system if there is a loss of steam generator heat sink capability. The feed-and-bleed technique involves passing hot fluid out of the primary system through a pressurizer power operated relief valve (PORV), while simultaneously feeding the primary system with subcooled high pressure injection system (HPIS) flow, charging flow, or both. This document summarizes the results of available experimental and analytical work that has been performed to investigate the thermal- hydraulic phenomena associated with the feed-and-bleed technique. These results are then synthesized into a statement of adequacy of the feed-and-bleed technique as a means of decay heat removal. It is concluded that feed-and-bleed can be a viable alternative form of decay heat removal for USPWRs, but that successful use of feed-and-bleed is contingent upon the existence and use of procedures for its implementation, as well as upon the specifics of plant design.

NUREG/CR-5076: AN APPROACH TO THE QUANTIFICATION OF SEISMIC MARGINS IN NUCLEAR POWER PLANTS.The Importance Of BWR Plant Systems And Functions To Seismic Margins. AMICO,P.J. Lawrence Livermore National Laboratory. * Applied Risk Technology Corp. May 1988. 40pp. 8806230148. UCRL-15985. 45896:060.

In NUREG/CR-4334 ("An Approach to the Quantification of Seismic Margins in Nuclear Power Plants"), the Expert Panel on Quantification of Seismic Margins presented a technique for studying the issue of quantifying seismic margins. As part of that technique, the panel included methods for simplifying the margins assessment by screening out components and systems using both systems and fragilities screening out components and systems using both systems and fragilities screening guidelines. At the time of that report, the panel was able to develop fragilities screening guidelines for all plants, however the systems screening guidelines applied only to PWRs (due to a shortage of BWR seismic PRAs upon which to base BWR systems screening guidelines). This report develops the BWR systems screening guidelines by utilizing the results of a number of BWR PRAs which have become available since the publication of NUREG/CR-4334.

NUREG/CR-5078 V01: A RELIABILITY PROGRAM FOR EMERGENCY DIESEL GENERATORS AT NUCLEAR POWER PLANTS.Program Structure. LOFGREN,E.V.; DEMOSS,G.M.; FRAGOLA,J.R.; et al. Sandia National Laboratories. April 1988. 155pp. 8804280386. SAND87-7178. 45271:304.

The purpose of the report is to develop guidance for the NRC staff to evaluate emergency diesel generator (EDG) reliability programs. Such reviews will likely result following the resolution of USI A-44 and GSI B-56. The diesel generator reliability program is a management system for achieving and maintaining a selected (or target) level of reliability. This can be achieved by: (1) understanding the factors that control the EDG reliability and (2) then applying reliability and maintenance techniques in proper proportion to achieve selected performance goals. The concepts and guidelines discussed in this report are concepts and approaches that have been successful in applications where high levels of reliability must be maintained. Both the EDG reliability program process and a set of review items for NRC use are provided. The review items represent a checklist for reviewing EDG reliability programs. They do not, in themselves, constitute a reliability program. Rather, the review items are those distinctive features of a reliability program that must be present for the program to be acceptable.

NUREG/CR-5082: SIMULATION EXPERIMENTS ON TWO-PHASE NATURAL CIRCULATION IN A FREON-113 FLOW VISUALIZATION LOOP. LEE,S.Y.; ISHII,M. Argonne National Laboratory. January 1988. 58pp. 8805200070. ANL-88-1. 45549:096.

In order to study the two-phase natural circulation and flow termination during a small break loss of coolant accident in LWR, simulation experiments have been performed using a Freon-113 flow visualization loop. The main focus of the present experiment was placed on the two-phase flow behavior in the hot-leg U-bend typical of B&W LWR systems. The loop was built based on the two-phase flow scaling criteria developed under this program to find out the effect of fluid properties, phase changes and coupling between hydrodynamic and heat transfer phenomena. Significantly different flow behaviors have been observed due to the non-equilibrium phase change phenomena such as the flashing and condensation in the Freon loop in comparison with the previous adiabatic experiment. The phenomena created much more unstable hydrodynamic conditions which lead to cyclic or oscillatory flow behaviors. Also, the void distribution and primary loop flow rate were measured in detail in addition to the important key parameters, such as the power input, loop friction and the liquid level inside the simulated steam generator.

NUREG/CR-5095 V01: THERMODYNAMIC NONEQUILIBRIUM IN POST-CRITICAL-HEAT-FLUX BOILING IN A ROD BUNDLE.Description Of Experiments And Sample Results. TUZLA,K.; UNAL,C.; BADR,O.; et al. Lehigh Univ., Bethlehem, PA. June 1988. 150pp. 8807110494. 46085:169.

This report describes the post-CHF heat transfer experiments in a 3 x 3 rod bundle. The objective was to obtain measurements of thermodynamic nonequilibrium in the post-CHF regime and to characterize its effects on two-phase heat transfer for verification of models used in thermal-hydraulic codes. A two-phase loop was constructed. The nine rod test bundle incorporated a heated shroud to simulate the operating characteristics of a large rod bundle. An original "hot patch" technique was developed to achieve steady-state post-CHF conditions in a rod bundle. Steam temperature probes, developed earlier at Lehigh for tests in single tubes, were modified for use in the rod bundle. Each test provided measurements of system pressure, coolant flow rate, wall heat flux, wall temperatures, two- phase equilibrium qualities, and vapor superheat temperatures. These primary data permitted determination of wall heat transfer coefficients, nonequilibrium vapor qualities, and quench front propagation velocities. Experiments were conducted in three different modes: a) steady-state with fixed CHF location, b) reflood with advancing CHF locations (propagating quench front), c) boil-off with retreating CHF locations. Tests were expected over the following range of conditions. Coolant mass flux, 0.1 to 26 kg/m(2)s, (inlet) quality, 40 degree C subcooled to 0.40, Pressure, 105 to 120 KPa, Heat flux, 5 to 43 KW/m(2)

NUREG/CR-5095 V02: THERMODYNAMIC NONEQUILIBRIUM IN POST-CRITICAL-HEAT-FLUX BOILING IN A ROD BUNDLE.Data For Stabilized Quench Front Tests. TUZLA,K.; UNAL,C.; BADR,O.; et al. Lehigh Univ., Bethlehem, PA. June 1988. 205pp. 8807110524. 46080:293.

See NUREG/CR-5095,V01 abstract.

NUREG/CR-5095 V03: THERMODYNAMIC NONEQUILIBRIUM IN POST-CRITICAL-HEAT-FLUX BOILING IN A ROD BUNDLE.Data For Advancing Quench Front Tests. TUZLA,K.; UNAL,C.; BADR,O.; et al. Lehigh Univ., Bethlehem, PA. June 1988. 567pp. 8807110530. 46079:081.

See NUREG/CR-5095,V01 abstract.

NUREG/CR-5095 V04: THERMODYNAMIC NONEQUILIBRIUM IN POST-CRITICAL-HEAT-FLUX BOILING IN A ROD BUNDLE.Data For Retreating Quench Front Tests. TUZLA,K.; UNAL,C.; BADR,O.; et al. Lehigh Univ., Bethlehem, PA. June 1988. 201pp. 8807110476. 46085:322.

See NUREG/CR-5095,V01 abstract.

NUREG/CR-5105: RESPONSE MARGINS INVESTIGATION OF PIPING DYNAMIC ANALYSES USING THE INDEPENDENT SUPPORT MOTION METHOD AND PVRC DAMPING. BEZLER,P.; WANG,Y.K.; REICH,M. Brookhaven National Laboratory. March 1988. 393pp. 8805260282. BNL-NUREG-52137. 45633:052.

An evaluation of Independent Support Motion (ISM) response spectrum methods of analysis coupled with the Pressure Vessel Research Committee (PVRC) recommendation for damping, to compute the dynamic component of the seismic response of piping systems, was completed. Response estimates for five piping/structural systems were developed using fourteen variants of the ISM response spectrum method, the Uniform Support Motion response spectrum method and the ISM time history analysis method, all based on the PVRC recommendations for damping. The ISM/PVRC calculational procedures were found to exhibit orderly characteristics with levels of conservatism comparable to those obtained with the ISM/uniform damping procedures. Using the ISM/PVRC response spectrum method with absolute combination between group contributions provided consistently conservative results while using the ISM/PVRC response spectrum method with square root sum of squares combination between group contributions provided estimates of response which were deemed to be acceptable.

NUREG/CR-5106: USER'S GUIDE FOR THE TACT5 COMPUTER CODE. WEST,D.B.; GILPIN,H.E. Science Applications International Corp. (formerly Science Applications, Inc.). June 1988. 205pp. 8807070447. SAIC-88/3023. 46013:011.

The TACT5 computer code, a successor to TACT III and earlier versions of TACT (an acronym for Transport of ACTivity), simulates the movement of radioactivity hypothetically released from a reactor core as it migrates through user-defined regions (nodes) of the containment, is immobilized by filters and sprays, and is released to the outside environment. The code has been modified to run on a personal computer (PC). A series of interactive BASIC pre-processor codes assists the user in compiling a nuclide input data file and a plant model data file. The plant model data file specifies a dynamic compartment model, which is represented by systems of ordinary differential equations with constant coefficients. The equations are solved explicitly by matrix transformation methods. A code run carries out the integration of these systems of equations over a succession of time intervals following reactor shutdown, with the interval boundaries corresponding to transitions of system parameter values which must be constant within each time interval. Output generated includes the level of radioactivity in each node of the containment and in the environment, and radiation doses to reference individuals at up to three different receptor points.

NUREG/CR-5113: FINDINGS OF THE PEER REVIEW PANEL ON THE DRAFT REACTOR RISK REFERENCE DOCUMENT,NUREG-1150. KASTENBERG,W.E.; APOSTOLAKIS,G.; PICKEL,J.H.; et al. Lawrence Livermore National Laboratory. 1988. 238pp. 8806230152. UCID-21346. 45900:160.

This report [illegible] the finding of the "NUREG-1150 Peer Review Panel on the methodology and results presented in the Draft Reactor Risk Reference Study (NUREG-1150) and supporting documentation. The report consists of two principal parts: the chairman's report, which summarizes the principal issues of the review, and the individual member's reports, which supply the individual findings and technical details of the fourteen reviewers.

NUREG/CR-5123: STUDIES OF THE PATTERN AND AGES OF POST-METAMORPHIC FAULTS IN THE PIEDMONT OF VIRGINIA AND NORTH CAROLINA. GLOVER,L.; COSTAIN,J.K.; CORUH,C. Virginia Polytechnic Institute & State Univ., Blacksburg, VA. April 1988. 67pp. 8805060296. 45430:226.

A geologic corridor from the Blue Ridge to the eastern Piedmont of Virginia is integrated into a tectonic model and extrapolated downward 10 to 15 km by means of seismic reflection and gravity studies. The Blue Ridge appears to be a hinge zone that faced a rift- generated Iapetus Ocean. An Eastern continent with an Eocambrian and Cambrian magmatic arc and sediments of the same age, collided with the North American continental margin in the Middle and Late Ordovician. Subsequent Devono-Mississippian and Mississippian-Permian orogenesis continued to drive thin thrust nappes onto North America. Early Mesozoic rift basins record the beginning of the Atlantic basin and, from Middle Jurassic to Present, the margin of North America was covered by Coastal Plain sediments. Several constrained hypocenters of the central Virginia seismic zone, adjacent to a reflection profile, show an apparent relation to structure. We tentatively conclude that flat and ramp faults formed during Paleozoic nappe emplacement are currently being reactivated. The reactivation may be largely aseismic on the old thrust faults, but seismicity appears to be related to high angle transcurrent faults where new rock breakage may be occurring.

NUREG/CR-5129: EXPERIMENTAL ASSESSMENT OF THE INFLUENCE OF DYNAMIC LOADING ON THE PERMEABILITY OF WET AND OF DRIED CEMENT BOREHOLE SEALS. ADISOMA,G.; DAEMEN,J.J. Arizona, Univ. of, Tucson, AZ. April 1988. 279pp. 8805090129. 45433:011.

An experimental sealing performance assessment of cement borehole plugs that have been subjected to dynamic loading is provided. This includes a study of plugs that have dried, as well as of plugs that have remained wet throughout the testing period. An introductory literature review indicates that deep underground structures in competent rock are safer than surface structures, openings at shallow depth, and openings in fractured rocks, when subjected to earthquakes and subsurface blasts. Cement plugs are installed in 2.5 cm diameter coaxial holes in 15 cm diameter granite cylinders. Water is injected under pressure on top of the plugs and is collected below the plugs. Hydraulic conductivities are calculated. Once a long-term steady-state flow trend has been established, the sample are subjected to dynamic loading on a shaking table. Shaking is performed at accelerations up to 2 g and for up to 300 seconds. Wet cement seals are less permeable than intact Charcoal granite. Sealing performance can degrade severely when cement seals are allowed to dry. Dye injection shows very limited and uniform penetration into wet plugs, but strongly preferential flow along the plug/rock interface of dried plugs. The permeability of wet and of rewetted previously dried cement seals does not change significantly after the application of dynamic loads. Sealing in an unsaturated environment may affect the drying (curing, aging) conditions of cementitious seals, as well as the structure of earthen seals. An unsaturated environment will need to be integrated realistically into sealing performance tests and analyses.

NUREG/CR-5130: BENTONITE BOREHOLE PLUG FLOW TESTING WITH FIVE WATER TYPES. GAUDETTE,M.V.; DAEMEN,J.J. Arizona, Univ. of, Tucson, AZ. April 1988. 223pp. 8805090139. 45432:010.

The hydraulic conductivity has been determined of plugs constructed with commercial precompressed bentonite pellets. Bentonite has been hydrated and tested with waters of five different chemical compositions, including one groundwater (Ogallala aquifer, Texas). The groundwater contained a significant amount of solids; waters prepared in the laboratory did not. Prepared waters used for testing included distilled water, a high (1000 ppm) and a low (45 ppm) calcium solution, and a 39 ppm sodium water. Uncompacted plugs were constructed by dropping bentonite tablets into waterfilled cylinders, or by mixing powdered bentonite with preselected water volumes in order to obtain controlled initial water contents. The hydraulic conductivity of all plugs tested with all waters would result in a classification of practically impervious, by conventional soil mechanics standards. Variations of several orders of magnitude of the hydraulic conductivity are observed. Clay plugs constructed from bentonite tablets hydrated with unfiltered Ogallala groundwater exhibited reduced swelling and lower hydraulic conductivity than similar plugs constructed from tablets and distilled water. The differences in observed conductivities may be governed by factors such as swelling characteristics and permeant colloidal matter clogging plug pore spaces. Constant pressure injection and transient pulse testing methods have been tried to determine hydraulic conductivity. Especially in constant pressure injection tests, outflow volumes may require adjustment to account for consolidation drainage. Attempts made here at such corrections have not been successful. Consolidation testing requires extremely long time periods to approach a constant limit. It appears probable that the flow tests have not been pursued for a sufficient time to assure complete determination of water chemistry effects on hydraulic conductivity.

NUREG/CR-5137: BIODEGRADATION TESTING OF TMI-2 EPICOR-II WASTE FORMS. ROGERS,R.D.; MCCONNELL,J.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). June 1988. 40pp. 8807070421. EGG-2540. 46011:344.

Waste-form testing is being conducted by the Three Mile Island Unit 2 (TMI-2) EPICOR-II Resin/Liner Investigation; Low-Level Waste Data Base Development Program of the U.S. Nuclear Regulatory Commission (NRC) in accordance with the NRC Branch Technical Position on Waste Form. Waste forms which were tested contain ion exchange resins solidified with vinyl ester-styrene and Portland Type I-II cement. This report describes the biodegradation testing of those waste forms, presents the test results, and provides recommendations for alternate test methodology.

NUREG/CR-5138: VALIDATION OF GENERIC COST ESTIMATES FOR CONSTRUCTION-RELATED ACTIVITIES AT NUCLEAR POWER PLANTS.Final Report. SIMION,G.; SCIACCA,F.; CLAIBORNE,E.; et al. Science & Engineering Associates, Inc. May 1988. 53pp. 8805200123. SEA 87-253-04A1. 45561:334.

This report represents a validation study of the cost methodologies and quantitative factors derived in "Labor Productivity Adjustment Factors" (NUREG/CR-4546) and "Generic Methodology for Estimating the Labor Cost Associated with the Removal of Hardware, Materials, and Structures From Nuclear Power Plants" (SEA Report 84- 116-05-A:1). This cost methodology was developed to support NRC analysts in determining generic estimates of removal, installation, and total labor costs for construction-related activities at nuclear generating stations. In addition to the validation discussion, this report reviews the generic cost analysis methodology employed. It also discusses each of the individual cost factors used in estimating the costs of physical modifications at nuclear power plants. The generic estimating approach presented uses the "greenfield" or new plant construction installation costs compiled in the Energy Economic Data Base (EEDB) as a baseline. These baseline costs are then adjusted to account for labor productivity, radiation fields, learning curve effects, and impacts on ancillary systems or components. For comparisons of estimated vs actual labor costs, approximately four dozen actual cost data points (as reported by 14 nuclear utilities) were obtained. Detailed background information was collected on each individual data point to give the best understanding possible so that the labor productivity factors, removal factors, etc., could judiciously be chosen. This study concludes that cost estimates that are typically within 40% of the actual values can be generated by prudently using the methodologies and cost factors investigated herein.

NUREG/CR-5142: DUCTILE TO BRITTLE TOUGHNESS TRANSITION CHARACTERIZATION OF A533B STEEL. JOYCE,J.A. David W. Taylor Naval Research & Development Center. June 1988. 37pp. 8807110516. 46099:318.

Central to the understanding of nuclear reactor containment vessel fracture safe design is the development of techniques to characterize and predict the fracture of ductile metals in the upper region of the ductile to brittle toughness transition. This project has utilized the J integral parameter to develop toughness transition information for static and dynamic loading rates throughout the toughness transition for an A533 grade B class 1 containment vessel steel. The J at cleavage (J(clv)) is evaluated when cleavage occurs, and the J(Ic) is evaluated when ductile crack extension is present. In many specimens, for both static and dynamic loading rates, both J(Ic) and J(clv) are evaluated and plotted as a function of test temperature and test rate. A principal conclusion is that if dynamic loading rates are used, very little size dependence seems to be present, though larger specimens produce lower bound results. Close agreement is also shown between dynamic initiation cleavage toughness values and the arrest toughness K(Ia) values obtained by NBS (Gaithersburg) dynamic crack arrest tests conducted on large edge cracked plates of this same steel alloy.

NUREG/CR-5147: FUNDAMENTAL ATTRIBUTES OF A PRACTICAL CONFIGURATION MANAGEMENT PROGRAM FOR NUCLEAR PLANT DESIGN CONTROL. KLEIN,S.M. ERC International, Inc. June 1988. 80pp. 8807080362. 46077:265.

This report summarizes the results of an evaluation of findings identified during a number of Safety-System Functional Inspections and Safety System Outage Modification Inspections which are related to configuration management. A computerized database of these findings was generated from a review of the design inspection reports. Based on the results of the evaluation, attributes of a configuration management program were developed which are responsive to minimizing these types of inspection findings. Incorporation of these key attributes is considered good practice in the development of a configuration management program at operating nuclear plants.

NUREG/CR-5150: STEAM GENERATOR OPERATING EXPERIENCE Update For 1984-1986. FRANK,L.; STOKLEY,J. Science Applications International Corp. (formerly Science Applications, Inc.). June 1988. 85pp. 8807110505. SAIC-87/3014. 46099:233.

This report summarizes operational events and degradation mechanisms affecting pressurized water reactor steam generator integrity, provides updated inspection results reported in 1984, 1985, and 1986, and highlights both prevalent problem areas and advances in improved equipment test practices, preventive measures, repair techniques, and replacement procedures. It describes equipment design features of the three major suppliers and discusses 68 plants in detail. Steam generator degradation mechanisms include intergranular stress corrosion cracking, primary water stress corrosion cracking, pitting, intergranular attack, and vibration wear that affects tube integrity and causes leakage. Plugging, sleeving, heat treatment, peening, chemical cleaning, and steam generator replacements are described and regulatory instruments and inspection guidelines for non-destructive evaluations and girth weld cracking are dis-

cussed. The report concludes that although degradation mechanisms are generally understood, the elimination of unscheduled plant shutdowns and costly repairs resulting from leaking tubes has not been achieved. Highlights of steam generator research and unresolved safety issues are discussed.

NUREG/CR-5151: PERFORMANCE-BASED INSPECTIONS. HAWKINS,F. NRC - No Detailed Affiliation Given. JOHNSON,J. Weirich & Associates. LINER,R.T.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). June 1988. 27pp. 8806230162. SAIC-88/3014. 45907:234.

This report describes the concept of performance-based inspections that is being taught in NRC's training course, "Inspecting for Performance." This concept has been endorsed and is being implemented by the Nuclear Regulatory Commission (NRC). NRC performance-based inspections concentrate on activities that impact plant reliability and safety. The inspections begin with performance- based observations, and then the inspectors let discrepancies or uncertainties lead to inspection of other areas, such as quality verification organization effectiveness, training adequacy, and procedural controls. This inspection approach departs from past NRC practices that emphasized documentation and program review as a means to measure operational safety. The first goal of performance-based inspections is to improve the NRC inspector's ability to accurately evaluate plant safety and reliability. This goal will be achieved by increasing the inspector's emphasis on actual observation of ongoing facility work activities and reducing the emphasis on document and program reviews. The secondary goal of performance-based level inspections is to encourage licensees to manage their facilities in a performance-based manner.

NUREG/CR-5159: PREDICTION OF CHECK VALVE PERFORMANCE AND DEGRADATION IN NUCLEAR POWER PLANT SYSTEMS. KALSI,M.S.; HORST,C.L.; WANG,J.K. Kalsi Engineering, Inc. May 1988. 78pp. 8806230210. KEI 1559. 45897:233.

Degradation and failure of swing check valves and resulting damage to plant equipment has led to a need to develop a method to predict performance and degradation of these valves in nuclear power plant systems. This Phase I investigation developed methods which can be used to predict the stability of the check valve disk when piping disturbances such as elbows, reducers, and generalized turbulence sources are present upstream of the valve within 10 pipe diameters. Major findings include the flow velocity required to achieve a full open, stable disk position, the magnitude of disk motion developed with these upstream disturbances with flow velocities below full open conditions, as well as disk natural frequency data which can be used to predict wear and fatigue damage. Reducers were found to cause little or no performance degradation. Elbow effects must be considered when located within 5 diameters of the check valve, while severe turbulence sources have significant effect at distances to 10 diameters. Clearway swing check designs were found to be particularly sensitive to manufacturing tolerances and installation variables making them likely candidates for premature failure. Reducing the disk full opening angle on these clearway designs results in significant performance improvement.

NUREG/CR-5160: GUIDELINES FOR THE USE OF THE EEDB AT THE SUB-COMPONENT AND SUBSYSTEM LEVEL.Final Report. ROBINSON,T.; SIMION,G.; SCIACCA,F.; et al. Science & Engineering Associates, Inc. May 1988. 72pp. 8806230166. SEA 87-253-06A1. 45898:232.

The Energy Economic Data Base (EEDB) provides complete construction cost estimates for boiling water reactors (BWRs) and pressurized water reactors (PWRs). The generic cost estimating methods developed for the NRC utilize the EEDB cost data as a basis for estimating the costs of physical modifications to nuclear plants. Such modifications may be mandated by new or revised NRC regulatory requirements. The costs presented in the EEDB are often given at a relatively high level of aggregation. The definition of material, labor, and equipment costs is seldom given at the same level as that encountered in plant modification projects. Additional definition is needed to adequately estimate the costs of many proposed changes. This report presents material and equipment costs and labor manhours/costs at the component, subcomponent, and subsystem level that support and correspond to the more highly aggregated data presented in the EEDB. It is intended to be a supplement to the EEDB. Specifically, unit labor and material/equipment costs are defined for the following types of material and equipment: Rotating machinery, piping related commodities (piping, valves, hangers, and insulation), instrumentation and control, lighting and service power, skid mounted equipment.

Secondary Report Number Index

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

SECONDARY REPORT NUMBER	REPORT NUMBER
ANL-88-1	NUREG/CR-5082
BMI-2120	NUREG/CR-4082 V06
BNL-NUREG-52059	NUREG/CR-4881
BNL-NUREG-52073	NUREG/CR-5038
BNL-NUREG-52119	NUREG/CR-5000
BNL-NUREG-52121	NUREG/CR-5015
BNL-NUREG-52137	NUREG/CR-5105
CSNI 97	NUREG/CP-0075
EGG-2458	NUREG/CR-4639 V01
EGG-2516	NUREG/CR-4971
EGG-2526	NUREG/CR-5072
EGG-2540	NUREG/CR-5137
IEB-80-03	NUREG/CR-4932
IEB-80-19	NUREG/CR-4933
IEB-83-08	NUREG/CR-4665
KEI 1559	NUREG/CR-5159
LA-11179-MS	NUREG/CR-5044
LBL-22860	NUREG/CR-4864 V01
LMF-119	NUREG/CR-5067
MEA-2232	NUREG/CR-5013
ORNL/NOAC-232	NUREG/CR-4674 V05
ORNL/NOAC-232	NUREG/CR-4674 V06
ORNL/NSIC-200	NUREG/CR-2000 V07 N3
ORNL/NSIC-200	NUREG/CR-2000 V07 N4
ORNL/NSIC-200	NUREG/CR-2000 V07 N5
ORNL/TM-10272	NUREG/CR-4777
ORNL/TM-10273	NUREG/CR-4778
ORNL/TM-10582	NUREG/CR-5019
ORNL/TM-10604	NUREG/CR-5021 V01
ORNL/TM-10604	NUREG/CR-5021 V02
ORNL/TM-10604/V	NUREG/CR-5021 V01
ORNL/TM-9593	NUREG/CR-4219 V04 N2
PARAMETER IE162	NUREG/CR-4665
PARAMETER IE167	NUREG/CR-4932
PARAMETER IE168	NUREG/CR-4933
PNL-5855	NUREG/CR-4605
PNL-6175	NUREG/CR-4879 V01
PNL-6341	NUREG/CR-5016
PNL-6391	NUREG/CR-5055
PNL-6511	NUREG/CP-0093
SAIC-87/3014	NUREG/CR-5150
SAIC-88/3014	NUREG/CR-5151
SAIC-88/3023	NUREG/CR-5106
SAND86-1938	NUREG/CR-4728
SAND86-2280	NUREG/CR-4763
SAND87-0323	NUREG/CR-4864 V01
SAND87-0871	NUREG/CR-4836
SAND87-7176	NUREG/CR-5078 V01
SEA 87-253-04A1	NUREG/CR-5138
SEA 87-253-06A1	NUREG/CR-5160
UCID-21223	NUREG/CR-5042 S01
UCID-21346	NUREG/CR-5113
UCRL-15985	NUREG/CR-5076

3

Personal Author Index

This index lists the personal authors of NRC staff, contractor, and international agreement reports in alphabetical order. Each name is followed by the NUREG number and the title of the report(s) prepared by the author. If further information is needed, refer to the main citation by the NUREG number.

4

Subject Index

This index was developed from keywords and word strings in titles and abstracts. During this development period, there will be some redundancy, which will be removed later when a reasonable thesaurus has been developed through experience. Suggestions for improvements are welcome.

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.

ADVISORY COMMITTEE(S)
ACRS - ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUREG-1125 V09: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS,1987 Annual.

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)
REGION 1, OFC OF THE DIRECTOR
NUREG-0837 V07 N04: NRC TLD DIRECT RADIATION MONITORING NETWORK,Progress Report, October-December 1987.
NUREG-0837 V08 N01: NRC TLD DIRECT RADIATION MONITORING NETWORK,Progress Report, January-March 1988.
OFC OF ENFORCEMENT (POST 870413)
NUREG-0940 V07 N01: ENFORCEMENT ACTIONS:SIGNIFICANT ACTIONS RESOLVED,Quarterly Progress Report,January-March 1988.
OFC OF SPECIAL PROJECTS
NUREG-1232 V02: SAFETY EVALUATION REPORT ON TENNESSEE VALLEY AUTHORITY. Sequoyah Nuclear Performance Plan.

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA
OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR
NUREG-0090 V10 N03: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES,July-September 1987.
NUREG-0090 V10 N04: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES,October-December 1987.

OFFICE OF GOVERNMENTAL & PUBLIC AFFAIRS
STATE, LOCAL & INDIAN TRIBE PROGRAMS
NUREG-1311: FUNDING THE NRC TRAINING PROGRAM FOR STATES.

OFFICE OF INFORMATION RESOURCES MANAGEMENT
DIVISION OF FREEDOM OF INFORMATION & PUBLICATION SERVICES (POST 880515)
NUREG-0304 V13 N01: REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1988,January-March.
NUREG-0540 V10 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30,1988.
NUREG-0750 V27 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. January-March 1988.
DIVISION OF PUBLICATION SERVICES (870413-880514)
NUREG-0540 V10 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-29,1988.
NUREG-0540 V10 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31,1988.
NUREG-0750 V26 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1987.
NUREG-0750 V26 I02: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-December 1987.
NUREG-0750 V27 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1988. Pages 41-255.
NUREG-0750 V27 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1988. Pages 257-334.
DIVISION OF COMPUTER & TELECOMMUNICATIONS SERVICES (POST 870413)
NUREG-0020 V12 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT,Data As Of February 29,1988.(Gray Book I)
NUREG-0020 V12 N04: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT,Data As Of March 31,1988 (Gray Book I)
NUREG-0020 V12 N05: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT,Data As Of April 30,1988.(Gray Book I)

OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
DIVISION OF SAFEGUARDS & TRANSPORTATION (POST 870413)
NUREG-0725 R06: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.
DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)
NUREG-1178: RADIOACTIVE MATERIAL IN THE WEST LAKE LANDFILL,Summary Report.
NUREG-1320: NUCLEAR FUEL CYCLE FACILITY ACCIDENT ANALYSIS HANDBOOK.
DIVISION OF HIGH LEVEL WASTE MANAGEMENT (POST 870413)
NUREG-1318: TECHNICAL POSITION ON ITEMS AND ACTIVITIES IN THE HIGH-LEVEL WASTE GEOLOGIC REPOSITORY PROGRAM SUBJECT TO QUALITY ASSURANCE REQUIREMENTS.

U.S. NUCLEAR REGULATORY COMMISSION
NRC - NO DETAILED AFFILIATION GIVEN
NUREG/CR-5151: PERFORMANCE-BASED INSPECTIONS.

OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 4/05/81)
OFFICE OF NUCLEAR REGULATORY RESEARCH, DIRECTOR (POST 860720)
NUREG-1032: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS,Technical Findings Related To Unresolved Safety Issue A-44,Final Report.
NUREG-1109: REGULATORY/BACKFIT ANALYSIS FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUE A-44,STATION BLACKOUT.
NUREG-1226: DEVELOPMENT AND UTILIZATION OF THE NRC POLICY STATEMENT ON THE REGULATION OF ADVANCED NUCLEAR POWER PLANTS.
NUREG-1266 V02: NRC SAFETY RESEARCH IN SUPPORT OF REGULATION - 1987.
NUREG-1325: DISPOSITION OF RECOMMENDATIONS OF THE NATIONAL RESEARCH COUNCIL IN THE REPORT "REVITALIZING NUCLEAR SAFETY RESERCH."
DIVISION OF ENGINEERING (POST 870413)
NUREG-1217 DRFT FC: EVALUATION OF SAFETY IMPLICATIONS OF CONTROL SYSTEMS IN LWR NUCLEAR POWER PLANTS,Technical Findings Related To Unresolved Safety Issue A-47,Draft Report For Comment.
NUREG-1218 DRFT FC: REGULATORY ANALYSIS FOR PROPOSED RESOLUTION OF USI A-47, Safety Implications Of Control Systems,Draft Rept For Comment.
NUREG-1221: SUMMARY,ANALYSIS AND RESPONSE TO PUBLIC COMMENTS ON PROPOSED AMENDMENTS TO 10 CFR PARTS 30,40,50,51,70 AND 72: DECOMMISSIONING CRITERIA FOR NUCLEAR FACILITIES.
NUREG-1233 DRFT FC: REGULATORY ANALYSIS FOR USI A-40, "SEISMIC DESIGN CRITERIA," Draft Report For Comment.
NUREG-1236: THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON SAFETY-RELATED MOTOR-OPERATED VALVES - GENERIC ISSUE II.E.6.1.
MATERIALS ENGINEERING BRANCH
NUREG-0975 V06: COMPILATION OF CONTRACT RESEARCH FOR THE MATERIALS ENGINEERING BRANCH,DIVISION OF ENGINEERING,Annual Report For FY 1987.
DIVISION OF REACTOR & PLANT SYSTEMS (870413-880716)
NUREG-1273: TECHNICAL FINDINGS AND REGULATORY ANALYSIS FOR GENERIC SAFETY ISSUE II.E.4.3, "CONTAINMENT INTEGRITY CHECK."
DIVISION OF REGULATORY APPLICATIONS (POST 870413)
NUREG-0713 V07: OCCUPATIONAL RADIATION EXPOSURE AT COMMERCIAL NUCLEAR POWER REACTORS AND OTHER FACILITIES 1985,Eighteenth Annual Report.
NUREG-0933 S07: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

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

OFFICE OF NUCLEAR REACTOR REGULATION, DIRECTOR (POST 870411)

NUREG-0683 S03 DRFT: PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT RELATED TO DECONTAMINATION AND DISPOSAL OF RADIOACTIVE WASTES RESULTING FROM MARCH 28,1979 ACCIDENT,THREE MILE ISLAND NUCLEAR STATION,UNIT 2.Docket No. 50-320.(GPU Nuclear,Incorporated)

NUREG-1032: EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS.Technical Findings Related To Unresolved Safety Issue A-44.Final Report.

NUREG-1109: REGULATORY/BACKFIT ANALYSIS FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUE A-44,STATION BLACKOUT.

DIVISION OF REACTOR PROJECTS - III,IV,V & SPECIAL PROJECTS (POST 870411

NUREG-1002 S06: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF BRAIDWOOD STATION,UNITS 1 AND 2.Docket Nos. 50-456 And 50-457.(Commonwealth Edison Company)

NUREG-1306: NRC SAFETY SIGNIFICANCE ASSESSMENT TEAM REPORT ON ALLEGATIONS RELATED TO THE SOUTH TEXAS PROJECT,UNITS 1 & 2.

STANDARDIZATION & NON-POWER REACTOR PROJECT DIRECTORATE

NUREG-1283: SAFETY EVALUATION REPORT RELATED TO THE RENEWAL OF THE OPERATING LICENSE FOR THE RESEARCH REACTOR AT PURDUE UNIVERSITY.

DIVISION OF REACTOR INSPECTION & SAFEGUARDS (POST 870411)

NUREG-0040 V12 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1988.(White Book)

DIVISION OF LICENSEE PERFORMANCE & QUALITY EVALUATION (POST 870411)

NUREG-1214 R03: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE.

NRC Originating Organization Index (International Agreements)

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

There were no NUREG/IA reports for this quarter.

NRC Contract Sponsor Index (Contractor Reports)

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

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA

OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA, DIRECTOR

NUREG/CR-2000 V07 N3: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of March 1988.

NUREG/CR-2000 V07 N4: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of April 1988.

NUREG/CR-2000 V07 N5: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of May 1988.

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DIVISION OF REACTOR ACCIDENT ANALYSIS (870413-880716)

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