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Vol. 15, No. 3

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
Third Quarter 1990
July – September

U.S. Nuclear Regulatory Commission

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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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Woodmont 537
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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. JANBERG, J.S. Argonne National Laboratory. May 1981. 141 pp. 8105280299. ANL-81-3. 08632:070.

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

Contractor Report

NUREG/CR-1556: STUDY OF ALTERNATE DECAY HEAT REMOVAL CONCEPTS FOR LIGHT WATER REACTORS-CURRENT SYSTEMS AND PROPOSED OPTIONS. BERRY, D.L.; BENNETT, P.R. Sandia Laboratories. May 1981. 100 pp. 8107010449. SAND80-0929. 08912:242.

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

International Agreement Report

NUREG/IA-0001: ASSESSMENT OF TRAC-PD2 USING SUPER CANNON AND HDR EXPERIMENTAL DATA. NEUMANN, U. Kraftwerk Union. August 1980. 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

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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 V14 N03: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of February 28,1990.(Gray Book I) HARTFIELD,R.A. Division of Computer & Telecommunications Services (Post 890205). May 1990. 531pp. 9008070321. 54860:231.

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 Information 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-0040 V14 N01: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1990. * Division of Reactor Inspection & Safeguards (Post 870411). July 1990. 213pp. 9008140499. 54924:276.

This periodical covers the results of inspections performed by NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from January 1990 through March 1990.

NUREG-0090 V13 N01: REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.January-March 1990. * Office for Analysis & Evaluation of Operational Data, Director. July 1990. 42pp. 9008160098. 54947:281.

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 January 1 through March 31, 1990. For this reporting period, there were 10 abnormal occurrences. One involved the loss of vital ac power with a subsequent reactor coolant system heat-up at the Vogtle Unit 1 power plant during shutdown. The event was investigated by an NRC Incident Investigation Team (IIT). The other nine abnormal occurrences involved nuclear material licensees and are described in detail under other NRC-issued licenses: eight of these involved medical therapy misadministrations; the other involved the receipt of an unshielded radioactive source at Amersham Corporation in Burlington, Massachusetts. The latter event was also investigated by an NRC IIT. No abnormal occurrences were reported by the Agreement States. The report also contains information that updates a previously reported abnormal occurrence.

NUREG-0325 R14: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.August 15, 1990. * Ofc of Personnel (Post 870413). September 1990. 62pp. 9010090054. 55333:027.

Functional organization charts for the U.S. Nuclear Regulatory Commission offices, divisions, and branches are presented.

NUREG-0386 D6 R07: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - March 1990. * Office of the General Counsel (Post 860701). August 1990. 65[illegible]pp. 9009120095. 55125:145.

This Revision Number 7 of the fifth edition of the NRC Practice and Procedure Digest contains a digest of a number of Commission, Atomic Safety and Licensing Appeal Board, and Atomic Safety and Licensing Board decisions issued during the period July 1, 1972 to March 31, 1990, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0525 R16: SAFEGUARDS SUMMARY EVENT LIST (SSEL). * Division of Safeguards & Transportation (Post 870413). July 1990. 369pp. 9009070011. 55070:128.

The Safeguards Summary Event List provides brief summaries of hundreds of safeguards-related events involving nuclear material or facilities regulated by the U.S. Nuclear Regulatory Commission. Events are described under the categories: bomb-related, intrusion, missing/allegedly stolen, transportation-related, tampering/vandalism, arson, firearms-related, radiological sabotage, non-radiological sabotage, alcohol and drug related, and miscellaneous. Because of public interest, the miscellaneous section also includes events reported involving source material, byproduct material, and natural uranium, which are exempt from safeguards requirements. Information in the event descriptions were obtained from official NRC reports.

NUREG-0540 V12 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 453pp. 9008070384. 54847:012.

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

NUREG-0540 V12 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.April 1-30, 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 322pp. 9008160107. 54949:016.

See NUREG-0540,V12,N03 abstract.

NUREG-0540 V12 N05: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.May 1-31, 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 414pp. 9008160102. 54947:322.
See NUREG-0540,V12,N03 abstract.

NUREG-0540 V12 N06: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. June 1-30, 1990. * Division of Freedom of Information & Publications Services (Post 890205). August 1990. 333pp. 9009040041. 55050:227.
See NUREG-0540,V12,N03 abstract.

NUREG-0540 V12 N07: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1990. * Division of Freedom of Information & Publications Services (Post 890205). September 1990. 364pp. 9009210224. 55232:198.
See NUREG-0540,V12,N03 abstract.

NUREG-0750 V31 '01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-March 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 53pp. 9009170093. 55202:102.
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 Judges, the Directors' Decisions, and the Denials of Petitions for Rulemaking are presented.

NUREG-0750 V31 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1990.Pages 371-481. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 117pp. 9008070350. 54863:075.
Legal issuances of the Commission, the Atomic Safety and Licensing Appeal Panel, the Atomic Safety and Licensing Board Panel, the Administrative Law Judges, and NRC Program Offices are presented.

NUREG-0750 V31 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1990.Pages 483-604. * Division of Freedom of Information & Publications Services (Post 890205). August 1990. 133pp. 9009040088. 55046:178.
See NUREG-0750,V31,N05 abstract.

NUREG-0750 V32 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1990.Pages 1-55. * Division of Freedom of Information & Publications Services (Post 890205). August 1990. 62pp. 9009130045. 55194:041.
See NUREG-0750,V31,N05 abstract.

NUREG-0800 17.3 R00: STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Revision 0 To SRP Section 17.3, "Quality Assurance Program Description." * Division of Licensee Performance & Quality Evaluation (Post 870411). August 1990. 20pp. 9009070001. 55071:137.
SRP Section 17.3, "Quality Assurance Program Description," is a new section in Chapter 17. It puts in place a performance-oriented quality assurance program review plan that (1) minimizes the current fragmentation and overlap of the self-assessment function responsibilities, including safety committee activities, audits, and other independent assessments, (2) simplifies the format, clarifies the intent, and consolidates the text of the present SRP Sections 17.1 and 17.2, (3) places emphasis on management, performance/verification, and self-assessment, the three components of quality assurance, and (4) permits the use of up-to-date industry consensus standards.

NUREG-0837 V10 N02: NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. April-June 1990. STRUCKMEYER,R.; MCNAMARA,N. Region 1 (Post 820201). September 1990. 226pp. 9010090064. 55319:345.
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 facilities throughout the country for the second quarter of 1990.

NUREG-0933 S11: A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R.; RIGGS,R.; MILSTEAD,W., et al. Division of Regulatory Applications (Post 870413). July 1990. 180pp. 9008070346. 54862:255.
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-0936 V09 N02: NRC REGULATORY AGENDA.Quarterly Report,April-June 1990. * Division of Freedom of Information & Publications Services (Post 890205). July 1990. 153pp. 9008140487. 54925:129.
The NRC Regulatory Agenda is a compilation of all rules on which the NRC has proposed or is considering action and all petitions for rulemaking which have been received by the Commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued each quarter.

NUREG-0940 V09 N02: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report April-June 1990. * Ofc of Enforcement (Post 870413). September 1990. 502pp. 9[illegible]0062. 55242:170.
This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April - June 1990) 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-1145 V06: U.S. NUCLEAR REGULATORY COMMISSION 1989 ANNUAL REPORT. * Office of Administration (Post 890205). July 1990. 248pp. 9009040091. 55047:155.
This report covers the major activities, events, decisions, and planning that took place during fiscal year 1989 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

NUREG-1214 R06: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F.; WHARTON,R. Division of Licensee Performance & Quality Evaluation (Post 870411). August 1990. 114pp. 9008310213. 55043:056.
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 sections: Section 1 presents the most recent SALP report ratings for facilities in operation and under construction. 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 ratings for facilities that recently have been licensed also are listed in Section 3.

NUREG-1272 V04 N01: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA 1989 ANNUAL REPORT.Power Reactors. * Office for Analysis & Evaluation of Operational Data Director. July 1990. 264pp. 9009040081. 55048:295.
The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1989. The

report is published in two separate parts. NUREG-1272, Vol. 4, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. This report also compiles the status of staff actions resulting from previous Incident Investigation Team (IIT) reports. NUREG-1272, Vol. 4, No. 2, covers nonreactors and presents a review of the events and concerns during 1989 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Each volume contains a list of the AEOD reports issued for 1980-1989.

NUREG-1272 V04 N02: OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA 1989 ANNUAL REPORT.Nonreactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1990. 68pp. 9009040059. 55046:199.

See NUREG-1272,V04,N01 abstract.

NUREG-1339: RESOLUTION OF GENERIC SAFETY ISSUE 29:BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS. JOHNSON,R.E. Division of Safety Issue Resolution (Post 880717). June 1990. 23pp. 9008140466. 54927:158.

This report describes the U.S. Nuclear Regulatory Commission's (NRC's) Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants," including the bases for establishing the issue and its historical highlights. The report also describes the activities of the Atomic Industrial Forum (AIF) relevant to this issue, including its cooperation with the Materials Properties Council (MPC) to organize a task group to help resolve the issue. The Electric Power Research Institute, supported by the AIF/MPC task group, prepared and issued a two-volume document that provides, in part, the technical basis for resolving Generic Safety Issue 29. This report presents the NRC's review and evaluation of the two-volume document and NRC's conclusion that this document, in conjunction with other information from both industry and NRC, provides the bases for resolving this issue.

NUREG-1362 DRFT FC: REGULATORY ANALYSIS FOR PROPOSED RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). July 1990. 213pp. 9008070329. 54652:042.

This regulatory analysis provides the supporting information for a proposed rule that will define the Nuclear Regulatory Commission's requirements for renewing the operating licenses of commercial nuclear power plants. A set of four specific alternatives for the safety review of license renewal applications is defined and evaluated. These are: Alternative A--current licensing basis; Alternative B--extension of Alternative A to require assessment and managing of aging; Alternative C--extension of Alternative B to require assessment of design differences against selected new-plant standards using probabilistic risk assessment; and Alternative D--extension of Alternative B to require compliance with all new-plant standards. A quantitative comparison of the four alternatives in terms of impact-to-value ratios is presented, and Alternative B is the most cost-beneficial safety review alternative.

NUREG-1363 V02: ATOMIC SAFETY AND LICENSING BOARD PANEL ANNUAL REPORT FISCAL YEAR 1989. COTTER,B.P. Atomic Safety and Licensing Board Panel. July 1990. 63pp. 9008070362. 54845:098.

In Fiscal Year 1989, the Atomic Safety and Licensing Board Panel (ASLBP) handled 40 proceedings involving the construction, operation and maintenance of commercial nuclear power reactors or other activities requiring a license from the Nuclear Regulatory Commission. This report summarizes, highlights and analyzes how the wide-ranging issues raised in these proceedings were addressed by the Judges and Licensing Boards of the ASLBP during the year.

NUREG-1377 R01: NRC RESEARCH PROGRAM ON PLANT AGING: LISTINGS AND SUMMARIES OF REPORTS ISSUED THROUGH MAY 1990. KONDIC,N.N. HILL,E.L. Division of Engineering (Post 870413). July 1990. 62pp. 9008070371. 54845:036.

The U.S. Nuclear Regulatory Commission is conducting the Nuclear Plant Aging Research (NPAR) Program. This is a comprehensive hardware-oriented engineering research program focused on understanding the aging mechanisms of components and systems in nuclear plants. The NPAR program also focuses on methods for simulating and monitoring the aging-related degradation of these components and systems. In addition, it provides recommendations for effective maintenance to manage aging and for the implementation of the research results in the regulatory process. This document contains a listing and index of reports generated in the NPAR program that were issued through May 1990 and summaries of those reports. Each summary describes the elements of the research covered in the report and outlines the significant results. For the convenience of the user, the reports are indexed by personal author, corporate author, and subject.

NUREG-1396: ENVIRONMENTAL ASSESSMENT OF THE THERMAL NEUTRON ACTIVITATION EXPLOSIVE DETECTION SYSTEM FOR CONCOURSE USE AT U.S. AIRPORTS. JONES,C.G. Division of Industrial & Medical Nuclear Safety (Post 870729). August 1990. 150pp. 9010090050. 55322:093.

This document is an environmental assessment of a system designed to detect the presence of explosives in checked airline baggage or cargo. The system is meant to be installed at the concourse or lobby ticketing areas of U.S. commercial airports and uses a sealed radioactive source of californium-252 to irradiate baggage items. The major impact of the use of this system arises from direct exposure of the public to scattered or leakage radiation from the source and to induced radioactivity in baggage items. Under normal operation and the most likely accident scenarios, the environmental impacts that would be created by the proposed licensing action would not be significant.

NUREG-1398 DRFT FC: ENVIRONMENTAL ASSESSMENT FOR PROPOSED RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). July 1990. 57pp. 9008070358. 54845:250.

The possible environmental effects of promulgating nuclear power plant license renewal standards by the proposed rule, 10 CFR Part 54, rather than applying requirements in an ad hoc manner in individual licensing actions, are assessed. The rule requires the development of information and analyses to identify aging problems of systems, structures, and components that will be of concern during the renewal term and will not be controlled by existing regulatory programs. Required actions may be replacement, refurbishment, inspection, testing or monitoring. Such actions will generally be within the range of similar actions taken for plants during the initial operating term. They would be primarily confined within the plants with potential for only minor disruption to the environment. It is unlikely that these actions would change the operating conditions of plants in ways that would change the environmental effects already being experienced. The promulgation of 10 CFR Part 54 has clear advantages relative to regulatory stability and administrative efficiency. However, it will not result in environmental effects significantly different from those arising from relicensing under existing regulations. The NRC concludes that promulgation of 10 CFR Part 54 would not significantly affect the environment and, therefore, a full environmental impact statement is not required and a Finding of No Significant Impact can be made.

NUREG-1407 DRFT FC: PROCEDURAL AND SUBMITTAL GUIDANCE FOR INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES.Draft Report For Comment. * Division of Safety Issue Resolution (Post 880717). July 1990. 41pp. 9008070367. 54845:307.

Based on a Policy Statement on Severe Accidents, the licensee of each nuclear power plant is requested to perform an individual plant examination. The plant examination systematically looks for vulnerabilities to severe accidents and cost-effective safety improvements that reduce or eliminate the important vulnerabilities. This document presents guidance for performing and reporting the results of the individual plant examination of external events (IPEEE). The guidance for reporting the results of the individual plant examination of internal events (IPE) is presented in NUREG-1335.

NUREG-1409: BACKFITTING GUIDELINES. ALLISON,D.P.; CONRAN,J.H.; TROTTIER,C.A. Office for Analysis & Evaluation of Operational Data, Director. July 1990. 102pp. 9008070378. 54846:270.

The backfitting process is the process by which the U.S. Nuclear Regulatory Commission (NRC) decides whether to issue new or revised requirements or staff positions to licensees of nuclear power reactor facilities. Requirements for proper justification of backfits and information requests are provided by two NRC rules (Title 10, Code of Federal Regulations, Sections 50.109 and 50.54(f)). NRC procedures include the charter of the comm[illegible] to Review Generic Requirements, NRC Manual Cha[illegible] 514, and individual office procedures. Three types of back[illegible] are recognized. Cost-justified substantial safety improvements require backfit analyses and findings of (1) substantial safety improvement and (2) justified costs. Compliance exceptions and adequate protection exceptions do not require findings of substantial safety improvements and costs are not considered. However, they are still backfits and require documented evaluations to support the use of the exceptions. Information requests (as opposed to backfits) require an analysis of the burden to be imposed to ensure that they are justified in view of the potential safety significance of the information requested.

NUREG-1411: RESPONSE TO PUBLIC COMMENTS RESULTING FROM THE PUBLIC WORKSHOP ON NUCLEAR POWER PLANT LICENSE RENEWAL. * Office of Nuclear Regulatory Research (Post 860720). July 1990. 75pp. 9008070323. 54845:348.

On October 13, 1989, the U.S. Nuclear Regulatory Commission (NRC) issued an Advance Notice of Proposed Rulemaking on nuclear power plant license renewal. The notice presented the NRC's preliminary regulatory philosophy and approach for developing license renewal regulations and solicited comments on a number of technical and policy issues. It also announced plans for a public workshop to discuss the issues and to receive comments and information. The workshop was held on November 13-14, 1989, in Reston, Virginia. This document reports on the NRC's response to the public comments from the workshop and written comments on the workshop topics received shortly after the workshop. (The proceedings of the workshop were reported in NUREG/CP-0108).

NUREG-1412 DRFT FC: FOUNDATION FOR THE ADEQUACY OF THE LICENSING BASES.A Supplement To The Statement Of Considerations For The Proposed Rule On Nuclear Power Plant License Renewal (10 CFR Part 54).Draft Report For Comment. * Office of Nuclear Reactor Regulation, Director (Post 870411). July 1990. 106pp. 9008070326. 54860:125.

In order to limit the Commission's license renewal decision to consideration of whether age-related degradation has been adequately addressed, the Part 54 rulemaking must make a generic finding for all nuclear power plants that the finding of reasonable assurance of adequate protection for issuance of an operating license continue to be true at the time of the renewal application and accordingly need not be made anew at the time of license renewal. This analysis describes the regulatory processes that form the basis for such a finding. This document discusses how the licensing process has evolved in major safety issue areas under existing regulatory processes that have ensured continued adequacy of the licensing bases of all operating plants. The document presents the described regulatory processes as the Commission's reasons for considering it unnecessary to re-review an operating plant's licensing basis, except for age-related degradation concerns, at the time of license renewal. This report is a supplement to the Statement of Considerations for the Nuclear Regulatory Commission's proposed rule (10 CFR Part 54) that would establish the criteria and standards governing nuclear power plant license renewal.

NUREG-1420: SPECIAL COMMITTEE REVIEW OF THE NUCLEAR REGULATORY COMMISSION'S SEVERE ACCIDENT RISKS REPORT (NUREG-1150). KOUTS,H.J.C.; APOSTOLAKIS,G.; BIRKHOFER,E.H.; et al. Office of Nuclear Regulatory Research (Post 860720). August 1990. 90pp. 9009110129. 55124:136.

In April 1989, the Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research (RES) published a draft report "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150. This report updated, extended and improved upon the information presented in the 1974 "Reactor Safety Study," WASH-1400. Because the information in NUREG-1150 will play a significant role in implementing the NRC's Severe Accident Policy, its quality and credibility are of critical importance. Accordingly, the Commission requested that RES conduct a peer review of NUREG-1150 to ensure that the methods, safety insights and conclusions presented are appropriate and adequately reflect the current state of knowledge with respect to reactor safety. To this end, RES formed a special committee in June of 1989 under the provisions of the Federal Advisory Committee Act. The Committee, composed of a group of recognized national and international experts in nuclear reactor safety, was charged with preparing a report reflecting their review of NUREG-1150 with respect to the adequacy of the methods, data, analysis and conclusions it set forth. The report which precedes reflects the results of this peer review.

NUREG-1425: WELDING AND NONDESTRUCTIVE EXAMINATION ISSUES AT SEABROOK NUCLEAR STATION.An Independent Review Team Report. SPESSARD,R.L.; COLEY,J.; CROWLEY,W.; et al. Ofc of the Executive Director for Operations. July 1990. 404pp. 9008140491. 54925:282.

In response to congressional concerns about the adequacy of the welding and nondestructive examination (NDE) programs at the Seabrook Nuclear Station, NRC senior management established an independent review team (IRT) to conduct an assessment. The IRT focused on the quality of the finished hardware and associated records, as well as on the adequacy of the overall quality assurance program as applied to the fabrication and NDE programs for pipe welds. This report documents the findings of that investigation.

NUREG/CP-0112 V01: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUBCRITICAL CRACK GROWTH.Opening Session And Technical Session I.Held At Moscow,USSR,May 14-17,1990. CULLEN,W.H. Materials Engineering Associates, Inc. * Argonne National Laboratory. August 1990. 312pp. 9009200013. ANL-90/22. 55214:324.

This report is a compilation of papers which were presented at the Third IAEA Specialists' Meeting on Subcritical Crack Growth, held in Moscow, USSR, on May 14-17, 1990. Volume 1 contains the welcoming remarks and attendance records, as well as the contributed papers for Session I, covering Corrosion Fatigue. Volume 2 contains the contributed papers for Sessions II, III, and IV, covering Stress-Corrosion Cracking, Test Methods, Models and Mechanisms, and summaries of National Programs in Argentina and the United Kingdom. Included as well are the Conclusions and Recommendations (Session V) developed by the organizing committee of the meeting, and discussed and approved by the participants.

NUREG/CP-0112 V02: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUBCRITICAL CRACK GROWTH.Technical Sessions II, III, And IV And Recommendations And Conclusions Session V.Held At Moscow,USSR,May 14-17, 1990. CULLEN,W.H. Materials Engineering Associates, Inc. * Argonne National Laboratory. August 1990. 219pp. 9009200015. ANL-90/22. 55215:276.

See NUREG/CP-0112,V01 abstract.

NUREG/CR-2000 V09 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1990. * Oak Ridge National Laboratory. July 1990. 89pp. 9008140508. ORNL/NSIC-200. 54927:069.

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

NUREG/CR-2000 V09 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1990. * Oak Ridge National Laboratory. August 1990. 101pp. 9009110143. ORNL/NSIC-200. 55194:181.

See NUREG/CR-2000,V09,N06 abstract.

NUREG/CR-2000 V09 N8: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of August 1990. * Oak Ridge National Laboratory. September 1990. 88pp. 9010090032. ORNL/NSIC-200. 55319:255.

See NUREG/CR-2000,V09,N06 abstract.

NUREG/CR-2850 V09: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1987. BAKER,D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1990. 187pp. 9009120104. PNL-4221. 55194:282.

Population radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1987. Fifty-year dose commitments from a one-year exposure were calculated from both liquid and atmospheric releases for four population groups (infant, child, teen-ager and adult) residing between 2 and 80 km from each of 70 sites. This report tabulates the results of these calculations, showing the dose commitments for both liquid and airborne pathways for each age group and organ. Also included for each of the sites is a histogram showing the fraction of the total population within 2 to 80 km around each site receiving various average dose commitments from the airborne pathways. The total dose commitments (from both liquid and airborne pathways) for each site ranged from a high of 15 person-rem to a low of 0.0016 person-rem for the sites with plants operating throughout the year with an arithmetic mean of 1.1 person-rem. The total population dose for all sites was estimated at 78 person-rem for the 150 million people considered at risk.

NUREG/CR-3444 V07: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION, WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE. SOO,P.; MILIAN,L.W. Brookhaven National Laboratory. July 1990. 74pp. 9008070400. BNL-NUREG-51699. 54844:188.

Studies were carried out to investigate if simulated decontamination reagent/resin waste combinations could give rise to gas generation and thermal excursions during dewatering events. The results of temperature measurements and visual observations are given. Some limited work was also carried out to determine if gamma irradiation of ion-exchange resins causes structural changes and losses in ion-exchange capacity. In addition, the corrosion of various container materials in simulated decontamination resin waste was studied. In particular, the effects of gamma irradiation were quantified.

NUREG/CR-3469 V05: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. KHAN,T.A.; TAN,H.; BAUM,J.W.; et al. Brookhaven National Laboratory. September 1990. 93pp. 9010090029. BNL-NUREG-51706. 55319:160.

One of the functions of the ALARA Center is to collect and disseminate information on dose reduction at nuclear power plants. This is the fifth report in the series of bibliographies of selected readings in radiation protection and ALARA that the Center publishes periodically. The abstracts in this bibliography were selected from proceedings of technical meetings, journals, research reports, searches of information data bases and reprints of published articles provided to us by the authors. The abstracts relate in one way or another to dose reduction at nuclear power plants, whether it is through good water chemistry, improvements in nuclear materials, better control of corrosion, robotics, and remote tooling or good operational health physics. The report contains 278 abstracts. Subject and author indices are provided. The subject index covers all previous volumes in this series. All information in the current volume is also available from the ALARA Center's on-line service, which is accessible by personal computer with the help of a modem. The preface of the report explains how the service may be accessed. The on-line service will be updated as new information is received.

NUREG/CR-4624 V06: RADIONUCLIDE RELEASE CALCULATIONS FOR SELECTED SEVERE ACCIDENT SCENARIOS.Supplemental Calculations. DENNING,R.S.; LEONARD,M.T.; CYBULSKIS,P.; et al. Battelle Memorial Institute, Columbus Laboratories. August 1990. 385pp. 9009200004. BMI-2139. 55216:135.

The results of source term calculations are reported. These calculations were performed in support of the NUREG-1150 study. Analyses were performed for three plants: Peach Bottom, a Mark I, boiling water reactor; Surry, a subatmospheric containment, pressurized water reactor; and Sequoyah, a subatmospheric containment, pressurized water reactor. Complete source term results are presented for the following sequences: short term station blackout with failure of the ADS system in the Peach Bottom plant; station blackout with pump seal LOCA in the Surry plant; station blackout with a pump seal LOCA in the Sequoyah plant; and a very small break with loss of ECC and spray recirculation in the Sequoyah plant. In addition, some partial analyses were performed which did not require running all of the modules of the Source Term Code Package. Thermal-hydraulic calculations were performed for the Surry and Sequoyah plants to evaluate the effects of alternative emergency operating procedures involving primary and secondary depressurization. For the Surry plant, calculations were performed of radionuclide transport through the primary system during accident-induced failure of steam generator tubes.

NUREG/CR-4674 V11: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Main Report And Appendix A. MINARICK,J.W.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. August 1990. 177pp. 9009130048. ORNL/NOAC-232. 55192:068.

Thirty operational events with conditional probabilities of core damage of 1.0 x 10(-6) or higher occurring at commercial light-water reactors during 1989 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 1984-1988 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 plant models used in the analysis process.

NUREG/CR-4674 V12: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Appendices B And C. MINARICK,J.W.; CLETCHER,J.W.; COPINGER,D.A.; et al. Oak Ridge National Laboratory. August 1990. 516pp. 9009130058. ORNL/NOAC-232. 55192:245.

See NUREG/CR-4674,V11 abstract.

NUREG/CR-4744 V03 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report,April-September 1988. CHOPRA,O.K.; CHUNG,H.M. Argonne National Laboratory. August 1990. 50pp. 9009250047. ANL-90/5. 55244:021.

This progress report summarizes work performed by Argonne National Laboratory on long-term embrittlement of cast duplex stainless steels in LWR systems during the six months from April to September 1988. Characteristics of the primary mechanism of aging embrittlement (i.e., spinodal decomposition of ferrite) and synergistic effects of alloying and impurity elements that influence the kinetics of the primary mechanism are discussed. Several secondary metallurgical processes of embrittlement, strongly dependent on the C, N, Ni, Mo, and Si content of various heats, are identified. Information on kinetics and data on impact properties are analyzed and correlated with microstructural characteristics to provide a unified method of extrapolating accelerated-aging data to reactor operating conditions. Fracture toughness data are presented for several heats of cast stainless steel aged at temperatures between 320 and 450 degrees C for times up to 10,000 h. Mechanical-property data are analyzed to develop the procedure and correlations for predicting the kinetics and extent of embrittlement of reactor components from known material parameters. The method and examples of estimating the impact strength and fracture toughness of cast components during reactor service are described. The lower-bound values of impact strength and fracture toughness for cast stainless steels at LWR operating temperatures are defined.

NUREG/CR-[illegible]94: EXPERIMENTAL RESULTS OF CORE-CONCRETE INTERACTIONS USING MOLTEN STEEL WITH ZIRCONIUM. COPUS,E.R.; BLOSE,R.E.; BROCKMANN,J.E.; et al. Sandia National Laboratories. July 1990. 386pp. 9008070412. SAND86-2638. 54849:001.

Four experiments were performed in order to evaluate the additional effects of zirconium metal oxidation on core debris interactions with limestone concrete using molten stainless steel as the core debris simulant. The QT-D, QT-E, SURC-3 and SURC-3A experiments eroded between 10 and 33 cm of limestone concrete during sustained interactions which lasted 35 to 120 minutes. Melt pool temperatures during the tests ranged from 1900 K before zirconium addition to 2100 K during the zirconium-steel-concrete phase of the tests. Large increases in erosion rate, gas production and aerosol release were also measured shortly after Zr metal was added to the melt.

NUREG/CR-4908: ULTRASONIC INSPECTION RELIABILITY FOR INTERGRANULAR STRESS CORROSION CRACKS.A Round Robin Study Of The Effects Of Personnel, Procedures, Equipment, And Crack Characteristics. HEASLER,P.G.; TAYLOR,T.T.; SPANNER,J.C.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1990. 153pp. 9008310179. PNL-6196. 55043:262.

A pipe inspection round robin entitled "Mini-Round Robin" was conducted at Pacific Northwest Laboratory from May 1985 through October 1985. The research was sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under a program entitled "Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors." The Mini-Round Robin (MRR) measured the IGSC crack detection and sizing capabilities of inservice inspection (ISI) inspectors that had passed the requirements of IEB 83-02 and the EPRI sizing training course. The MRR data base was compared with an earlier Pipe Inspection Round Robin (PIRR) that had measured effective detection prior to 1982. Comparison of the MRR and PIRR data bases indicated no difference in detection capability was measured for long and short cracks. In addition to the pipe inspection round robin, a human factors study was conducted in conjunction with the MRR. The most important result of the human factors study is that the Relative Operating Characteristics (ROC) curves provide a better methodology for describing inspector performance than only POD or single-point crack/no crack data.

NUREG/CR-5117: STEAM GENERATOR TUBE INTEGRITY PROGRAM/STEAM GENERATOR GROUP PROJECT.Final Project Summary Report. KURTZ,R.J.; CLARK,R.A.; BRADLEY,E.R.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1990. 210pp. 9008070393. PNL-6446. 54846:060.

The Steam Generator Tube Integrity Program/Steam Generator Group Project was a three-phase program conducted for the U.S. Nuclear Regulatory Commission (NRC) by Pacific Northwest Laboratory. The main goal of the program was to provide the NRC with validated information on the reliability of nondestructive examination techniques to detect and size flaws in steam generator tubing and to determine the remaining integrity of service-degraded tubing. The program was performed in three phases. The first phase involved burst and collapse tests and single-frequency eddy-current (EC) examinations of typical steam generator tubing with precision machined flaws. The goal of Phase I was to develop empirical models of remaining tube integrity as a function of flaw type and size, and to determine the capability of EC inspection methods to detect and size tube degradation. In Phase II, a smaller number of specimens with the same flaw types were investigated, but tube specimens were degraded by chemical means rather than machining methods. This approach was used to better simulate the irregular geometry of service-induced degradation. In the final phase of the program, the retired-from-service Surry 2A Steam Generator was used as a test bed to investigate the reliability of inservice EC inspection equipment, personnel, and procedures, and as a source of service-degraded tubes for further validating the empirical equations of remaining tube integrity.

NUREG/CR-5254: BIAS IN PEAK CLAD TEMPERATURE PREDICTIONS DUE TO UNCERTAINTIES IN MODELING OF ECC BYPASS AND DISSOLVED NON-CONDENSABLE GAS PHENOMENA. ROHATGI,U.S.; NEYMOTIN,L.Y.; JO,J.; et al. Brookhaven National Laboratory. September 1990. 175pp. 9010090047. BNL-NUREG-52168. 55322:253.

The U.S. Nuclear Regulatory Commission (USNRC), its contractors and consultants have developed a methodology for evaluating Code Scaling, Applicability and Uncertainty (CSAU). The CSAU method has been demonstrated by applying it to the

TRAC-PF1/MOD1, Version 14.3 code and its analysis of a Large Break Loss of Coolant Accident (LBLOCA) for a Westinghouse four-loop plant. In applying the methodology, the accident course is divided into three different phases, namely: Blowdown, Refill and Reflood. There are two distinct peak clad temperatures (PCT), one in the Blowdown Phase and one in the Reflood Phase. The Reflood Phase PCT is affected by the phenomena related to Emergency Core Cooling System (ECCS) in the downcomer and lower plenum of the reactor vessel. This report describes a general method for estimating the biases in the Reflood Phase PCT from systematic errors (biases) associated with the modelling of the ECCS and dissolved nitrogen, and the application of this method. The bias in the Reflood Phase PCT due to the uncertainty in the existing code models for ECCS related phenomena is -19 degrees K (-34 degrees F). The bias in the PCT due to the lack of modelling of dissolved N(2) in the code is estimated to be 9.9 degrees K (17.8 degrees F). The code prediction for PCT is conservative if the bias is negative, and nonconservative if the bias is positive. The bias estimated here is based on full scale data from the Upper Plenum Test facility and is unaffected by the scale distortions.

NUREG/CR-5280 V01: AGE-RELATED DEGRADATION OF WESTINGHOUSE 480-VOLT CIRCUIT BREAKERS.Aging Assessment And Recommendations For Improving Breaker Reliability. SUBUDHI,M.; SHIER,W.; MACDOUGALL,E. Brookhaven National Laboratory. July 1990. 90pp. 9008070370. BNL-NUREG-52178. 54845:160.

An aging assessment of Westinghouse DS-series low-voltage air circuit breakers was performed as part of the Nuclear Plant Aging Research (NPAR) program. The objectives of this study are to characterize age-related degradation within the breaker assembly and to identify maintenance practices to mitigate their effects. Since this study has been promulgated by the failures of the reactor trip breakers at the McGuire Nuclear Station in July 1987, results relating to the welds in the breaker pole lever welds are also discussed. The design and operation of DS-206 and DS-416 breakers were reviewed. Failure data from various national data bases were analyzed to identify the predominant failure modes, causes, and mechanisms. Additional operating experiences from one nuclear station and two industrial breaker-service companies were obtained to develop aging trends of various subcomponents. The responses of the utilities to the NRC Bulletin 88-01, which discusses the center pole lever welds, were analyzed to assess the final resolution of failures of welds in the reactor trips.

NUREG/CR-5297: CLOSEOUT OF IE BULLETIN 83-05:ASME NUCLEAR CODE PUMPS AND SPARE PARTS MANUFACTURED BY THE HAYWARD TYLER PUMP COMPANY. FOLEY,W.J.; DEAN,R.S.; HENNICK,A. PARAMETER, Inc. August 1990. 33pp. 9009180017. PARAMETER IE188. 55214:290.

Documentation is provided in this report to close IE Bulletin 83-05 regarding ASME nuclear code pumps and spare parts manufactured by the Hayward Tyler Pump Company (HTPC). The bulletin was issued (1) to alert holders of operating licenses and construction permits of nuclear power plants that HTPC failed to implement effectively their quality assurance (QA) program from 1977 to 1981 and (2) to require affected utilities to take action to resolve the potential for failure of the subject pumps and their spare parts. Evaluation of utility responses and NRC/Region inspection reports shows that reliability of the affected pumps was ensured by means of procedures and performance testing of the pumps as required by the bulletin. Based on the evaluation, in accordance with specific criteria, the bulletin is closed for 116 (98%) of the 118 facilities to which it was issued for action and which were not shut down indefinitely or permanently at the time of issuance of this report. A follow-up item is proposed for the two facilities with open bulletin status. Based on favorable results, a conclusion is presented to indicate that the bulletin concerns have been resolved. Background information is supplied in the Introduction and Appendix A.

NUREG/CR-5373: A DEMONSTRATION EXPERIMENT OF STEAM-DRIVEN, HIGH-PRESSURE MELT EJECTION.The HIPS-10S Test. ALLEN,M.D.; NICHOLS,R.T.; PILCH,M. Sandia National Laboratories. July 1990. 92pp. 9008310209. SAND89-1135. 55043:170.

A steam blowdown test was performed at the Surtsey Direct Heating Test Facility to test the steam supply system and burst diaphragm arrangement that will be used in subsequent Surtsey Direct Containment Heating (DCH) experiments. Following successful completion of the steam blowdown test, the HIPS-10S (High-Pressure Melt Streaming) experiment was conducted to demonstrate that the technology to perform steam-driven, high-pressure melt ejection (HPME) experiments had been successfully developed. In addition, the HIPS-10S experiment was used to assess techniques and instrumentation designed to create the proper timing of events in HPME experiments.

NUREG/CR-5374: SUMMARY OF INADEQUATE CORE COOLING INSTRUMENTATION FOR UNITED STATES NUCLEAR POWER PLANTS. ANDERSON,J.L.; HAGEN,E.W.; MORELOCK,T.C. Oak Ridge National Laboratory. July 1990. 197pp. 9008070334. ORNL/TM-11200. 54850:027.

This report summarizes a review of Inadequate Core Cooling Instrumentation installed in U.S. nuclear power plants in response to the requirements of NUREG-0737, Clarification of TMI Action Plan Requirements, and related orders. The review includes descriptions of generic systems developed by Westinghouse and Combustion Engineering, as well as plant specific reviews of each pressurized water reactor installation. Performance characteristics are discussed, including feedback from plant personnel concerning installation, operational experience, and operator acceptance. An evaluation of boiling water reactor systems is included.

NUREG/CR-5385: INITIAL ASSESSMENT OF THE MECHANISMS AND SIGNIFICANCE OF LOW-TEMPERATURE EMBRITTLEMENT OF CAST STAINLESS STEELS IN LWR SYSTEMS. CHOPRA,O.K.; SATHER,A. Argonne National Laboratory. August 1990. 252pp. 9009040085. ANL-89/17. 55048:043.

This report summarizes work performed by Argonne National Laboratory on long-term embrittlement of cast duplex stainless steels in LWR systems. Metallurgical characterization and mechanical property data from Charpy-impact, tensile, and J-R curve tests are presented for several experimental and commercial heats, as well as for reactor-aged CF-3, CF-8, and CF-8M cast stainless steels. The effects of material variables on the embrittlement of cast stainless steels are evaluated. Chemical composition and ferrite morphology strongly affect the extent and kinetics of embrittlement. In general, the low-carbon CF-3 stainless steels are the most resistant and the molybdenum-containing high-carbon CF-8M stainless steels are most susceptible to embrittlement. The microstructural and mechanical-property data are analyzed to establish the mechanisms of embrittlement. The procedure and correlations for predicting the impact strength and fracture toughness of cast components during reactor service are described. The lower bound values of impact strength and fracture toughness for low-temperature-aged cast stainless steel are defined.

NUREG/CR-5395 V11: MULTILOOP INTEGRAL SYSTEM TEST (MIST):FINAL REPORT.MIST Phase IV Tests. GEISSLER,G.O. Babcock & Wilcox Co. August 1990. 528pp. 9009210290. EPRI/NP-6480. 55237:169.

The Multiloop Integral System Test (MIST) is part of a multiphase program started in 1983 to address small-break loss-of-coolant accidents (SBLOCAs) specific to Babcock and Wilcox designed plants. MIST is sponsored by the U.S. Nuclear Regulatory Commission, the Babcock & Wilcox Owners Group, the Electric Power Research Institute, and Babcock and Wilcox. The unique features of the Babcock and Wilcox design, specifically the hot leg U-bends and steam generators, prevented the use of existing integral system data or existing integral facilities to address the thermal-hydraulic SBLOCA questions. MIST and two other supporting facilities were specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and the other facilities will be used to benchmark the adequacy of system codes, such as RELAP5 and TRAC, for predicting abnormal plant transients. The MIST program is reported in 11 volumes. The program is summarized in Volume 1; Volumes 2 through 8 describe groups of tests by test type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the calculations of RELAP5/MOD2 and MIST observations, and Volume 11 presents the later Phase 4 tests. This Volume 11 pertains to MIST Phase IV tests performed to investigate risk dominant transients and non-LOCA events.

NUREG/CR-5395 V11 AD: MULTILOOP INTEGRAL SYSTEM TEST (MIST):FINAL REPORT.MIST Phase IV Tests. GEISSLER,G.O. Babcock & Wilcox Co. August 1990. 273pp. 9009210229. EPRI/NP-6480. 55233:202.

The Multiloop Integral System Test (MIST) is part of a multiphase program started in 1983 to address small-break loss-of-coolant accidents (SBLOCAs) specific to Babcock and Wilcox designed plants. MIST is sponsored by the U.S. Nuclear Regulatory Commission, the Babcock & Wilcox Owners Group, the Electric Power Research Institute, and Babcock and Wilcox. The unique features of the Babcock and Wilcox design, specifically the hot leg U-bends and steam generators, prevented the use of existing integral system data or existing integral facilities to address the thermal-hydraulic SBLOCA questions. MIST and two other supporting facilities were specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and the other facilities will be used to benchmark the adequacy of system codes, such as RELAP5 and TRAC, for predicting abnormal plant transients. The MIST program is reported in 11 volumes. The program is summarized in Volume 1; Volumes 2 through 8 describe groups of tests by test type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the calculations of RELAP5/MOD2 and MIST observations, and Volume 11 presents the later Phase 4 tests. This Volume 11 addendum pertains to MIST natural circulation tests.

NUREG/CR-5448: AGING EVALUATION OF CLASS 1E BATTERIES: SEISMIC TESTING. EDSON,J.L. EG&G Idaho, Inc. (subs. of EG&G, Inc.). August 1990. 233pp. 9008310221. EGG-2576. 55042:138.

This report presents the results of a seismic testing program on naturally aged class 1E batteries obtained from a nuclear plant. The testing program is a Phase II activity resulting from a Phase I aging evaluation of class 1E batteries in safety systems of nuclear power plants, performed previously as a part of the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research Program and reported in NUREG/CR-4457. The primary purpose of the program was to evaluate the seismic ruggedness of naturally aged batteries to determine if aged batteries could have adequate electrical capacity, as determined by tests recommended by IEEE Standards, and yet have inadequate seismic ruggedness to provide needed electrical power during and after a safe shutdown earthquake (SSE) event. A secondary purpose of the program was to evaluate selected advanced surveillance methods to determine if they were likely to be more sensitive to the aging degradation that reduces seismic ruggedness. The program used twelve batteries naturally aged to about 14 years of age in a nuclear facility and tested them at four different seismic levels representative of the levels of possible earthquakes specified for nuclear plants in the United States. Seismic testing of the batteries did not cause any loss of electrical capacity.

NUREG/CR-5451: CRACK-ARREST BEHAVIOR IN SEN WIDE PLATES OF LOW-UPPER-SHELF BASE METAL TESTED UNDER NONISOTHERMAL CONDITIONS: WP-2 SERIES. NAUS,D.J.; KEENEY-WALKER; BASS,B.R.; et al. Oak Ridge National Laboratory. August 1990. 299pp. 9008170076. ORNL-6584. 55201:163.

Six wide-plate crack-arrest tests (WP-2 Series) are discussed in this report. Each test utilized either a 1 x 1 x 0.1-m or a 1 x 1 x 0.15-m thick single-edge notch specimen (a/w = 0.2), fabricated from a low-upper-shelf base material, that was subjected to a linear thermal gradient along the plane of crack propagation. The tests were conducted at the National Institute of Standards and Technology and were designed to provide fracture-toughness measurements at temperatures approaching or above the onset of the Charpy upper-shelf regime, in a rising toughness region, and with an increasing driving force. Results obtained from these tests have produced crack-arrest toughness values well above the limit recognized by the current ASME guidelines (220 MPa·$\sqrt{m}$) with arrests occurring at up to 102 degrees C above the material (DW)NDT (60 degrees C). The fracture data support: (1) use of fracture mechanics concepts to analyze cleavage run-arrest events, (2) treatment of cleavage and ductile fracture modes as separate events, and (3) fact that cleavage arrest occurs above the ASME limit.

NUREG/CR-5453 V05: BACKGROUND INFORMATION FOR THE DEVELOPMENT OF A LOW-LEVEL WASTE PERFORMANCE ASSESSMENT METHODOLOGY.Computer Code Implementation And Assessment. KOZAK,M.W.; CHU,M.S.Y.; MATTINGLY,P.A.; et al. Sandia National Laboratories. August 1990. 105pp. 9008310184. SAND89-2509. 55044:055.

This report documents the implementation and assessment of computer codes for a low-level waste performance assessment methodology. Computer codes and analytical solutions are implemented for ground-water flow and transport analyses, source-term analyses, surface-water transport analyses, air-transport analyses, food-chain analyses, and dosimetry analyses. The capability has been retained to perform either simple or more complicated analyses of the source term and ground-water transport aspects of the performance assessment. The simple approaches consist of analytical and simple numerical analyses that are appropriate for relatively simple conceptual models. For fully multi-dimensional or transient problems, more complicated numerical solutions are recommended. Details are given of the recommended analytical methods, together with sensitivity analyses that demonstrate important aspects of the solutions. The implementation processes for the more complicated computer codes and those problems that arose during implementation are discussed. Finally, a comparison is given between the simple and complicated ground-water transport analyses for a simple conceptual model.

NUREG/CR-5461: AGING OF CABLES, CONNECTIONS, AND ELECTRICAL PENETRATION ASSEMBLIES USED IN NUCLEAR POWER PLANTS. JACOBUS,M.J. Sandia National Laboratories. July 1990. 81pp. 9008080195. SAND89-2369. 54871:328.

This report examines effects of aging on cables, connections, and containment electrical penetration assemblies (EPAs). Aging is defined as the cumulative effects that occur to a component with the passage of time. If unchecked, these effects can lead to a loss of function and a potential impairment of plant safety. This study includes a review of component usage in nuclear power plants; a review of some commonly used components and their materials of construction; a review of the stressors that the components might be exposed to in both normal and accident environments; a compilation and evaluation of industry failure data; a discussion of component failure modes and causes; and a brief description of current industry testing and maintenance practices.

NUREG/CR-5515: LIGHT WATER REACTOR PRESSURE ISOLATION VALVE PERFORMANCE TESTING. NEELY,H.H.; JEANMOUGIN,N.M.; CORUGEDO,J.J. Energy Technology Engineering Center. July 1990. 111pp. 9008070338. ETEC 88-01. 54860:014.

The Light Water Reactor Valve Performance Testing Program was initiated by the NRC to evaluate leakage as an indication of valve condition, provide input to Section XI of the ASME Code, evaluate motor signature testing to measure valve operability, evaluate acoustic emission monitoring for condition and degradation and in-service inspection techniques. Six typical check and gate valves were purchased for testing at typical plant conditions (550 F at 2250 psig) for an assumed number of cycles for a 40-year plant lifetime. Tests revealed that there were variances between the test results and the present statement of the Code; however, the testing was not conclusive. The lifecycle tests showed that high tech acoustic emission can be utilized to trend small leaks, that specific motor signature measurement on gate valves can trend and indicate potential failure, and that in-service inspection techniques for check valves was shown to be both feasible and an excellent preventive maintenance indicator. Lifecycle testing performance here did not cause large valve leakage typical of some plant operation. Other testing is required to fully understand the implication of these results and the required program to fully implement them.

NUREG/CR-5519 V01: AGING OF CONTROL AND SERVICE AIR COMPRESSORS AND DRYERS USED IN NUCLEAR POWER PLANTS. MOYERS,J.C. Oak Ridge National Laboratory. July 1990. 129pp. 9008070332. ORNL-6607. 54850:224.

This report was produced under the Detection of Defects and Degradation Monitoring of Nuclear Plant Safety Equipment element of the Nuclear Plant Aging Research Program. This element includes the identification of practical and cost-effective methods for detecting, monitoring, and assessing the severity of time-dependent degradation (aging) of control and service air compressors and dryers in nuclear power plants. These methods are to provide capabilities for establishing degradation trends prior to failure and developing guidance for effective maintenance. The topics of this Phase I assessment report are failure modes and causes resulting from aging, manufacturer-recommended maintenance and surveillance practices, and measurable parameters (including functional indicators) for use in assessing operational readiness, establishing degradation trends, and detecting incipient failure. The results presented are based on information derived from operating experience records, manufacturer-supplied information, and input from plant operators. For each failure mode, failure causes are listed by sub-component, and parameters potentially useful for detecting degradation that could lead to failure are identified.

NUREG/CR-5524 V01: TMI-2 VESSEL INVESTIGATION PROJECT (VIP) METALLURGICAL PROGRAM.Project Report,January-September 1989. DIERCKS,D.R. Argonne National Laboratory. March 1990. 43pp. 9008090018. ANL-90/2. 54872:048.

This report summarizes the work performed by Argonne National Laboratory on the TMI-2 Vessel Investigation Project (VIP) Metallurgical Program during the nine months from the initiation of the program in January 1989 through September 1989. During the reporting period, archive material for the program was obtained from the lower head of the cancelled Midland nuclear reactor in Midland, MI, in the form of four plates. Chemical analyses and hardness measurements were performed on samples from the four plates, the as-received microstructure was characterized, and a tentative determination of rolling direction was made. Initial results from heat treatment experiments on the archive material indicate that those regions of the TMI-2 material where the maximum temperature exceeded 727 degrees C should be readily identifiable on the basis of microstructural observations. A series of round-robin mechanical tests and microstructural studies on the as-received archive material was developed, and specimens and specimen blanks for tensile and stress-rupture tests were distributed to the participating OECD laboratories. Two trial specimens cut from a plate of A36 plain-carbon structural steel by PCI Energy Systems using metal disintegration machining (MDM) were examined metallographically.

NUREG/CR-5528: AN ASSESSMENT OF BWR MARK II CONTAINMENT CHALLENGES,FAILURE MODES, AND POTENTIAL IMPROVEMENTS IN PERFORMANCE. KELLY,D.L.; JONES,K.R.; DALLMAN,R.J.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1990. 317pp. 9009040046. EGG-2593. 55049:270.

This report assesses challenges to BWR Mark II containment integrity that could potentially arise from severe accidents. Also assessed are some potential improvements that could prevent core damage or containment failure, or could mitigate the consequences of such failure by reducing the release of fission products to the environment. These challenges and improvements are analyzed via a limited quantitative risk/benefit analysis of a generic BWR/4 reactor with a Mark II containment. Point estimate frequencies of the dominant core damage sequences are obtained and simple containment event trees are constructed to evaluate the response of the containment to these severe accident sequences. The resulting containment release modes are then binned into source term release categories, which provide inputs to the consequence analysis. The output of the consequence analysis is used to construct an overall base case risk profile. Potential improvements and sensitivities are evaluated by modifying the event tree split fractions, thus generating a revised risk profile. Several important sensitivity cases are examined in order to evaluate the impact of phenomenological uncertainties on the final results.

NUREG/CR-5532: A PERFORMANCE ASSESSMENT METHODOLOGY FOR LOW-LEVEL WASTE FACILITIES. KOZAK,M.W.; CHU,M.S.Y.; MATTINGLY,P.A. Sandia National Laboratories. July 1990. 85pp. 9008[illegible]187. SAND90-0375. 54871:243.

A performance assessment methodology has been developed for use by the U.S. Nuclear Regulatory Commission in evaluating license applications for low-level waste disposal facilities. This report provides a summary of background reports on the development of the methodology and an overview of the models and codes selected for the methodology. The overview includes discussions of the philosophy and structure of the methodology and a sequential procedure for applying the methodology. Discussions are provided of models and associated assumptions that are appropriate for each phase of the methodology, the goals of each phase, data required to implement the models, significant sources of uncertainty associated with each phase, and the computer codes used to implement the appropriate models. In addition, a sample demonstration of the methodology is presented for a simple conceptual model.

NUREG/CR-5542: MODELS FOR ESTIMATION OF SERVICE LIFE OF CONCRETE BARRIERS IN LOW-LEVEL RADIOACTIVE WASTE DISPOSAL. WALTON,J.C.; PLANSKY,L.E.; SMITH,R.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). September 1990. 52pp. 9010090060. EGG-2597. 55319:043.

Concrete barriers will be used as intimate parts of systems for isolation of low-level radioactive wastes subsequent to disposal. This work reviews mathematical models for estimating degradation rate of concrete in typical service environments. The models considered cover sulfate attack, reinforcement corrosion, calcium hydroxide leaching, carbonation, freeze/thaw and cracking. Additionally, fluid flow, mass transport, and geochemical properties of concrete are briefly reviewed. Example calculations included illustrate the types of predictions expected of the models.

NUREG/CR-5552: AN OVERVIEW OF THE LOW UPPER SHELF TOUGHNESS SAFETY MARGIN ISSUE. MERKLE,J.G. Oak Ridge National Laboratory. August 1990. 62pp. 9009040031. ORNL/TM-11314. 55052:266.

The low upper shelf toughness issue has a long history, beginning with the choice of materials for the submerged arc welding process, but also potentially involving the use of A302-B plate. Criteria for vessels containing low upper shelf materials have usually been expressed in terms of the Charpy upper shelf impact energy. Although these criteria have had several different bases, the range of limiting values for wall thicknesses approaching nine inches has remained between 40 and 50 ft.lbs. Values for vessels with thinner walls and/or only circumferential low upper shelf welds could be less. A decision on criteria to be incorporated into the ASME Code is approaching. Choices to be made concern the method for estimating the decrease in upper shelf impact energy, flaw geometry for circumferential welds, statistical significance of toughness values, the choice between J(D) and J(M), reference pressure, safety factors, and the inclusion of instability pressure calculations by means of R curve extrapolation. This report presents a comprehensive overview of the issue, including history and recommendations for expediting its resolution.

NUREG/CR-5554: RECOMMENDATIONS FOR THE SHALLOW-CRACK FRACTURE TOUGHNESS TESTING TASK WITHIN THE HSST PROGRAM. THEISS,T.J. Oak Ridge National Laboratory. September 1990. 55pp. 9010090068. ORNL/TM-11509. 55318:340.

Recommendations for the Heavy-Section Steel Technology Program's investigation into the influence of crack depth on the fracture toughness of a steel under conditions prototypic of those in a reactor pressure vessel are included in this report. The primary goal of the shallow- crack project is to investigate the influence of crack depth on fracture toughness under conditions prototypic of a reactor vessel. A limited data base of fracture-toughness values will be assembled using a beam specimen with a depth of 100 mm (4 in.) using prototypic reactor vessel material. Results of the investigation are expected to improve the understanding of shallow-flaw behavior in pressure vessels, thereby providing more realistic information for application to the pressurized- thermal-shock issues.

NUREG/CR-5556: REVIEW OF CURRENT LITERATURE RELATED TO GENERIC SAFETY ISSUE 15. LIPINSKI,R.E.; GARNER,R.W. EG&G Idaho, Inc. (subs. of EG&G, Inc.). July 1990. 28pp. 9008080197. EGG-2598. 54881:320.

Recent evaluations of surveillance samples in the High Flux Isotope Reactor at the Oak Ridge National Laboratory led to the conclusion that the embrittlement rates of several reactor pressure vessel (RPV) steels may be greater than originally anticipated. In June 1987, the Advisory Committee on Reactor Safeguards requested that the U.S. Nuclear Regulatory Commission investigate the consequences of embrittlement of RPV supports. This report summarizes the current literature related to these studies, evaluates their contribution toward resolving Generic Safety Issue 15 concerning material embrittlement, and recommends any further action considered appropriate. This review also contains a short discussion of the uses of structural mechanics and fracture mechanics to analyze embrittlement.

NUREG/CR-5567: PWR DRY CONTAINMENT ISSUE CHARACTERIZATION. YANG,J.W. Brookhaven National Laboratory. August 1990. 204pp. 9009040028. BNL-NUREG-52234. 55046:311.

Severe accident issues have been characterized for pressurized water reactors with large dry containments. A description of PWR dry containment performance under severe accident conditions is provided. Reviews and discussions of early containment failure due to direct containment heating (DCH), in-vessel steam explosions, hydrogen burns and steam spikes, late containment failure due to gradual overpressurization and basemat melt-through, and containment bypass (interfacing systems LOCA) events are included. An assessment of potential improvements such as RCS depressurization, reactor cavity flooding, hydrogen control, containment venting and accident management strategy is presented. The review and discussion are largely based on existing information obtained from the nuclear industry and the NRC's severe accident research programs. Additional analyses related to operator actions were performed and are presented in the appendices.

NUREG/CR-5568 V01: INDUSTRY BASED PERFORMANCE INDICATORS FOR NUCLEAR POWER PLANTS.Phase 1 Report: June 1989 - February 1990. CONNELLY,E.M.; VAN HEMEL,S.B. Communications Technology Applications, Inc. HAAS,P.M. Concord Associates, Inc. July 1990. 98pp. 9008090024. CTA 900215-025. 54874:094.

This report presents the results of the first phase of a two-phase study performed with the goal of developing indirect (leading) indicators of nuclear power plant safety, using other industries as a model. It was hypothesized that other industries with similar public safety concerns could serve as analogs to the nuclear power industry. Many process industries have many more years of operating experience, and many more plants than the nuclear power industry, and thus should have accumulated much useful safety data. In Phase 1, the investigators screened a variety of potential industry analogs and chose a chemical/petrochemical manufacturing industry as the primary analog for further study. Information was gathered on safety programs and indicators in the chemical industry, as well as in the nuclear power industry. Frameworks were selected for the development of indicators which could be transferred from the chemical to the nuclear power environment, and candidate sets of direct and indirect safety indicators were developed. Estimates were made of the availability and quality of data in the chemical industry, and plans were developed for further investigating and testing these candidate indicators against safety data in both the chemical and nuclear power industries in Phase 2.

NUREG/CR-5572: AN EVALUATION OF THE EFFECTS OF LOCAL CONTROL STATION DESGIN CONFIGURATIONS ON HUMAN PERFORMANCE AND NUCLEAR POWER PLANT RISK. O'HARA,J.; RUGER,C.; HIGGINS,J.; et al. Brookhaven National Laboratory. September 1990. 71pp. 9009250057. BNL-NUREG-52236. 55243:312.

A human factors analysis was performed to assess how identified upgrades to local control stations (LCSs) in nuclear power plants affect both human performance and plant risk. Upgrades in the design of individual control panels and overall improvement of functional centralization were considered. The analysis methodology was accomplished in four stages. First, a list of LCS human engineering design deficiencies was developed using data collected from a variety of sources including visits to nuclear power plants. From these data, a set of potential upgrades were defined to correct the deficiencies. Second, the effects of the upgrades on human error probabilities (HEPs) were determined using a computer-based methodology for soliciting expert judgement. Third, the HEPs were propagated through a plant probabilistic risk assessment (PRA), and new core melt frequencies were established. A preliminary, scoping value-impact assessment was performed to evaluate the regulatory need for further review of possible action to improve the human factors engineering aspects of local control stations. The results indicated that implementation of both types of upgrades would improve human performance and lower risk, but that only the panel design improvements would be cost beneficial.

NUREG/CR-5575: QUANTITATIVE ANALYSIS OF POTENTIAL PERFORMANCE IMPROVEMENTS FOR THE DRY PWR CONTAINMENT. KELLY,D.L.; PAFFORD,D.J.; SCHROEDER,J.A.; et al. EG&G Idaho, Inc. (subs. of EG&G, Inc.). August 1990. 163pp. 9009200008. EGG-2602. 55232:035.

This report calculates the risk benefit associated with potential performance improvements for the large dry pressurized water reactor (PWR) containment. The analysis is based on the June 1989 draft NUREG-1150 results for the Zion commercial nuclear reactor. Simplified containment event trees and the large accident progression event trees from draft NUREG-1150 are used to evaluate the effects of potential improvements on the response of the Zion containment to dominant severe accident sequences. Source terms are generated parametrically using the ZISOR code and offsite consequences are calculated with the MELCOR Accident Consequence Code System (MACCS). These results give point estimates of the risk reduction associated with each containment improvement identified by Brookhaven National Laboratory in their draft Issues Characterization Report.

NUREG/CR-5583: PREDICTION OF CHECK VALVE PERFORMANCE AND DEGRADATION IN NUCLEAR POWER PLANT SYSTEMS WEAR AND IMPACT TESTS.Final Report,September 1988 - April 1990. KALSI,M.S.; HORST,C.L.; WANG,J.K.; et al. Kalsi Engineering, Inc. August 1990. 118pp. 9009070007. KEI 1656. 55070:010.

Check valve failures in nuclear power plants have led to safety concerns as well as extensive damage and loss of plant availability in recent years. Swing check valve internals may experience premature degradation if the disc is not firmly held open against its stop and significant flow disturbances are present upstream within 10 pipe diameters. The objective of the current Phase II research was to develop and experimentally verify a quantitative methodology for predicting swing check valve performance and the degradation of internals caused by hinge pin wear or disc stud impact. Phase I research had focussed on investigating the stability of the swing check valve disc at different flow velocities for a wide variety of upstream flow disturbances located within 10 pipe diameters of the check valve. Valve performance predictions based on methodology developed as a result of Phase I and II research correlate well with actual valve operating history at plants. The conservative guidelines provided by this methodology, tempered and refined by actual performance history and integrated with preventive maintenance activities, have the potential for significantly improving the overall reliability of check valves in nuclear power plants.

NUREG/CR-5588 V01: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.Theoretical Manual. XU,J.; PHILIPPACOPOULO; MILLER,C.A.; et al. Brookhaven National Laboratory. July 1990. 80pp. 9008070356. BNL-NUREG-52241. 54844:316.

During FY's 1988 and 1989, Brookhaven National Laboratory (BNL) developed the CARES system (Computer Analysis for Rapid Evaluation of Structures) for the U.S. Nuclear Regulatory Commission (NRC). CARES is a PC software system which has been designed to perform structural response computations similar to those encountered in licensing reviews of nuclear power plant structures. The documentation of the Seismic Module of CARES consists of three volumes. This report represents Volume 1 of the three volume documentation of the Seismic Module of CARES. It concentrates on the theoretical basis of the system and presents modeling assumptions and limitations as well as solution schemes and algorithms of CARES. The User's Manual is published as Volume 2 while solutions and results from a set of sample problems are published as Volume 3 of the CARES documentation.

NUREG/CR-5588 V02: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.User's Manual. XU,J.; PHILIPPACOPOULO; MILLER,C.A.; et al. Brookhaven National Laboratory. July 1990. 110pp. 9008070352. BNL-NUREG-52241. 54848:105.

During FY's 1988 and 1989, Brookhaven National Laboratory (BNL) developed the CARES system (Computer Analysis for Rapid Evaluation of Structures) for the U.S. Nuclear Regulatory Commission (NRC). CARES is a PC software system which has been designed to perform structural response computations similar to those encountered in licensing reviews of nuclear power plant structures. The documentation of the Seismic Module of CARES consists of three volumes. This report is Volume 2 of the three volume documentation of the Seismic Module of CARES and represents the User's Manual. Volume 1 concentrates on the theoretical basis of the system and presents modeling assumptions and limitations as well as solution schemes and algorithms of CARES. Solutions and results from a set of sample problems are published as Volume 3 of the CARES documentation.

NUREG/CR-5588 V03: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.Sample Problems. XU,J.; PHILIPPACOPOULO; MILLER,C.A.; et al. Brookhaven National Laboratory. July 1990. 146pp. 9008070359. BNL-NUREG-52241. 54848:215.

During FY's 1988 and 1989, Brookhaven National Laboratory (BNL) developed the CARES system (Computer Analysis for Rapid Evaluation of Structures) for the U.S. Nuclear Regulatory Commission (NRC). CARES is a PC software system which has been designed to perform structural response computations similar to those encountered in licensing reviews of nuclear power plant structures. The documentation of the Seismic Module of CARES consists of three volumes. This report represents Volume 3 of the three volume documentation of the Seismic Module of CARES. It presents three sample problems typically encountered in the Soil-Structure Interaction analyses. The theoretical bases, modeling assumptions and limitations as well as solution schemes and algorithms of the Seismic Module of CARES are given in Volume 1. The User's Manual is published as Volume 2.

NUREG/CR-5589: ASSESSMENT OF ICE-CONDENSER CONTAINMENT PERFORMANCE ISSUES. NOURBAKHSH,H.P. Brookhaven National Laboratory. July 1990. 57pp. 9008070396. BNL-NUREG-52242. 54844:259.

Vulnerabilities of an ice-condenser containment to challenges that could arise from severe accidents have been assessed. The phenomenological issues associated with containment challenges have been evaluated. A number of containment improvements which have the potential to mitigate severe accident challenges have been evaluated. This report is intended to provide a comprehensive statement of the relevant issues that can be used in the NRC staff's evaluation process and by the utilities during their individual plant examinations (IPEs).

NUREG/CR-5591 V01 N1: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1989 - March 1990. CORWIN,W.R. Oak Ridge National Laboratory. August 1990. 44pp. 9008310217. ORNL/TM-11568. 55043:011.

The primary goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior (particularly the fracture toughness properties) of typical pressure-vessel steels as they relate to light- water-reactor pressure-vessel integrity. The program includes direct continuation of irradiation studies previously conducted by the Heavy- Section Steel Technology Program augmented by enhanced examinations of the accompanying microstructural changes. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are examined on a wide range of fracture properties. Detailed statistical analyses of the fracture data on K(Ic) shift of high-copper welds were performed. Analysis of the first phase of irradiated crack-arrest testing on high-copper welds was completed. Final analysis and publication of the results of the second phase of the irradiation studies on stainless steel weld-overlay cladding were completed. Determinations were made of the variations in chemistry and unirradiated RT(NDT) of low upper-shelf weld metal from the Midland reactor. Final analyses were performed on the Charpy impact and tensile data from the Second and Third Irradiation series on low upper-shelf welds, and the report on the series was drafted. A detailed survey of existing data on microstructural models and data bases of irradiation damage was performed, and initial development of a reaction-rate-based model was completed.

NUREG/CR-5596: UNSATURATED FRACTURED ROCK CHARACTERIZATION METHODS AND DATA SETS AT THE APACHE LEAP TUFF SITE. RASMUSSEN,T.C.; EVANS,D.D.; SHEETS,P.J.; et al. Arizona, Univ. of, Tucson, AZ. August 1990. 146pp. 9009110122. 55123:350.

Performance assessment of high-level nuclear waste containment feasibility requires representative values of parameters as input, including parameter moments, distributional characteristics, and covariance structures between parameters. To meet this need, characterization methods and data sets for interstitial, hydraulic, pneumatic and thermal parameters for a slightly welded fractured tuff at the Apache Leap Tuff Site situated in central Arizona are reported in this document. The data sets include the influence of matric suction on measured parameters. Spatial variability is investigated by sampling along nine boreholes at regular distances. Laboratory parameter estimates for 105 core segments are provided, as well as field estimates centered on the intervals where the core segments were collected. Measurement uncertainty is estimated by repetitively testing control samples.

NUREG/CR-5607: FLOW AND TRANSPORT AT THE LAS CRUCES TRENCH SITE.Experiments 1 And 2. WIERENGA,P.J.; HUDSON,D.B. Arizona, Univ. of, Tucson, AZ. HILLS,R.G.; et al. New Mexico State Univ., Las Cruces, NM. August 1990. 426pp. 9009040036. 55051:200.

Two water flow and solute transport experiments were performed as part of a comprehensive field trench study near Las Cruces, New Mexico. These experiments were designed to provide data to test deterministic and stochastic models of vadose zone flow and transport. In Experiment 1, a 4 m by 9 m area was irrigated for 10 days with water containing tritium. Thereafter, water was applied without tritium for an additional 76 days. Simple one-dimensional uniform and layered soil deterministic models for infiltration adequately predicted the overall movement of the wetting front during infiltration, but poorly predicted point values for water content due to spatial variability. Use of the layered soil model, rather than the uniform soil model, did not consistently improve prediction accuracy for this particular field application. In Experiment 2, a 1.22 m by 12 m area was irrigated for 11.5 days with water containing tritium and bromide. Thereafter, water was applied without tracers for an additional 6[illegible] days. Water and bromide moved fairly uniformly during infiltration, whereas high concentrations of tritium developed on one side of the irrigated area. During redistribution, tritium moved little, whereas bromide displayed significant movement both downward and to one side. A two-dimensional deterministic model for water flow showed qualitative, but not quantitative, agreement with observations. A two-dimensional deterministic model for solute transport poorly described tritium and bromide movement during redistribution.

NUREG/CR-5622: ANALYSIS OF REACTOR TRIPS ORIGINATING IN BALANCE OF PLANT SYSTEMS. STETSON,F.T.; GALLAGHER,D.W.; LE,P.T.; et al. Science Applications International Corp. (formerly Science Applications, Inc.). September 1990. 200pp. 9010090026. SAIC-89/1148. 55323:064.

This report documents the results of an analysis of balance-of- plant (BOP) related reactor trips at commercial U.S. nuclear power plants over a 5-year period, from January 1, 1984, through December 31, 1988. The study was performed for the Plant Systems Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. The objectives of the study were: 1) to improve the level of understanding of BOP-related challenges to safety systems by identifying and categorizing such events; 2) to prepare a computerized data base of BOP-related reactor trip events and use the data base to identify trends and patterns in the population of these events; 3) to investigate the risk implications of BOP events that challenge safety systems; and 4) to provide recommendations on how to address BOP-related concerns in a regulatory context.

NUREG/CR-5640: OVERVIEW AND COMPARISON OF U.S. COMMERCIAL NUCLEAR POWER PLANTS.Nuclear Power Plant System Sourcebook. LOBNER,P.; DONAHOE,C.; CAVALLIN,C. Science Applications International Corp. (formerly Science Applications, Inc.). September 1990. 600pp. 9010090058. SAIC-89/1541. 55320:212.

This report is the introductory volume to the Nuclear Power Plant Sourcebook Series and is intended as a source of current summary and comparative information on U.S. commercial light water reactors (LWRs). The summary and comparative information is organized into the following four parts: (a) general U.S. LWRs, (b) pressurized water reactors (PWRs), (c) boiling water reactors (BWRs), and (d) bibliographies of general PWR and BWR references, plant-specific references, system- specific references, and component-specific references. This report is supplemented by a set of Sourcebooks that provides more detailed information on specific U.S. LWR plants.

NUREG/IA-0033: ASSESSMENT OF RELAP5/MOD2 CYCLE 36.04 AGAINST LOFT SMALL BREAK EXPERIMENT L3-6. ERIKSSON,J. Sweden, Govt. of. July 1990. 103pp. 9008140510. STUDSVIKNP87128. 54926:326.

The LOFT small break experiment L3-6 has been analyzed as part of Sweden's contribution to the International Thermal Hydraulic Code Assessment and Applications Program (ICAP). Three calculations, of which two were sensitivity studies, were carried out. The following quantities were varied: (1) the content of secondary side fluid and the feed water valve closure, and (2) the two-phase characteristics of the main pumps. All three predictions agreed reasonably well with most of the measured data. The sensitivity calculations resulted only in marginal improvements. The predicted and measured data are compared on plots and their differences are quantified over intervals in real time.

NUREG/IA-0034: ASSESSMENT STUDY OF RELAP5/MOD2 CYCLE 36.04 BASED ON PRESSURIZER SAFETY AND RELIEF VALVE TESTS. STUBBE,E.J.; VANHOENACKER,L. TRACTEBEL. July 1990. 86pp. 9008080178. 54871:157.

This report presents a code assessment study based on full size relief and assisted safety valve (called SEBIM) tests performed on the CUMULUS valve test rig operated by Electricite de France (EDF). The increased awareness that the pressurizer safety and relief valves are not reliable under water blowdown conditions, has led to the design, testing and installation of so called assisted safety valves of which the SEBIM (TM) valves are an example. These valves, used in tandem, are gradually replacing the safety and relief valves on pressurizers in some European PWR's. Before installation at the plant, the Belgian safety authorities requested a thorough full scale testing of these valves on a test rig (CUMULUS) equipped with sufficient diagnostics to measure the characteristics of the valve. The Belgian architect-engineering firm TRACTEBEL was called upon to specify, order and test these valves for installation at the DOEL 1 and DOEL 2 power plants. These tests do not provide sufficient data of high quality to justify an assessment study of the code RELAP-5 MOD-2 CYCLE 36 in the ICAP framework which is the subject of this report.

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.

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.

CORWIN,W.R.
NUREG/CR-5591 V01: HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For October 1989 - March 1990.

COSTANTINO,C.
NUREG/CR-5588 V01: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.Theoretical Manual.
NUREG/CR-5588 V02: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.User's Manual.
NUREG/CR-5588 V03: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.Sample Problems.

COTTER,B.P.
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CROWLEY,W.
NUREG-1425: WELDING AND NONDESTRUCTIVE EXAMINATION ISSUES AT SEABROOK NUCLEAR STATION.An Independent Review Team Report.

CULLEN,W.H.
NUREG/CP-0112 V01: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUBCRITICAL CRACK GROWTH.Opening Session And Technical Session I.Held At Moscow,USSR,May 14-17,1990.
NUREG/CP-0112 V02: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUBCRITICAL CRACK GROWTH.Technical Sessions II, III, And IV And Recommendations And Conclusions Session V.Held At Moscow,USSR,May 14-17, 1990.

CURTIS,L.A.
NUREG/CR-4624 V06: RADIONUCLIDE RELEASE CALCULATIONS FOR SELECTED SEVERE ACCIDENT SCENARIOS.Supplemental Calculations.

CYBULSKIS,P.
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DALLMAN,R.J.
NUREG/CR-5528: AN ASSESSMENT OF BWR MARK II CONTAINMENT CHALLENGES,FAILURE MODES, AND POTENTIAL IMPROVEMENTS IN PERFORMANCE.

DEAN,R.S.
NUREG/CR-5297: CLOSEOUT OF IE BULLETIN 83-05:ASME NUCLEAR CODE PUMPS AND SPARE PARTS MANUFACTURED BY THE HAYWARD TYLER PUMP COMPANY.

DEFFENBAUGH,J.
NUREG/CR-4908: ULTRASONIC INSPECTION RELIABILITY FOR INTERGRANULAR STRESS CORROSION CRACKS.A Round Robin Study Of The Effects Of Personnel, Procedures, Equipment, And Crack Characteristics.

DENNING,R.S.
NUREG/CR-4624 V06: RADIONUCLIDE RELEASE CALCULATIONS FOR SELECTED SEVERE ACCIDENT SCENARIOS.Supplemental Calculations.

DEWIT,R.
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LEONARD,M.T.
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SATHER,A.
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WAGNER,K.C.
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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.

High-Pressure Melt Streaming
NUREG/CR-5373: A DEMONSTRATION EXPERIMENT OF STEAM-DRIVEN, HIGH-PRESSURE MELT EJECTION.The HIPS-10S Test.

Human Performance
NUREG/CR-5572: AN EVALUATION OF THE EFFECTS OF LOCAL CONTROL STATION DESGIN CONFIGURATIONS ON HUMAN PERFORMANCE AND NUCLEAR POWER PLANT RISK.

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Ice-Condenser Containment
NUREG/CR-5589: ASSESSMENT OF ICE-CONDENSER CONTAINMENT PERFORMANCE ISSUES.

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NUREG/CR-5374: SUMMARY OF INADEQUATE CORE COOLING INSTRUMENTATION FOR UNITED STATES NUCLEAR POWER PLANTS.

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NUREG-1407 DRFT FC: PROCEDURAL AND SUBMITTAL GUIDANCE FOR INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES.Draft Report For Comment.

Inservice Inspection
NUREG/CR-5117: STEAM GENERATOR TUBE INTEGRITY PROGRAM/STEAM GENERATOR GROUP PROJECT.Final Project Summary Report.

Intergranular Stress Corrosion Crack
NUREG/CR-4908: ULTRASONIC INSPECTION RELIABILITY FOR INTERGRANULAR STRESS CORROSION CRACKS.A Round Robin Study Of The Effects Of Personnel, Procedures, Equipment, And Crack Characteristics.

J-R Curve
NUREG/CR-5552: AN OVERVIEW OF THE LOW UPPER SHELF TOUGHNESS SAFETY MARGIN ISSUE.

LER
NUREG/CR-2000 V09 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1990.
NUREG/CR-2000 V09 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1990.
NUREG/CR-2000 V09 N8: LICENSEE EVENT REPORT (LER) COMPILATION:For Month Of August 1990.

LOFT
NUREG/IA-0033: ASSESSMENT OF RELAP5/MOD2 CYCLE 36.04 AGAINST LOFT SMALL BREAK EXPERIMENT L3-6.

LWR
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NUREG/CR-3444 V07: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION, WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-4674 V11: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Main Report And Appendix A.
NUREG/CR-4674 V12: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Appendices B And C.
NUREG/CR-4744 V03 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report.April-September 1988.
NUREG/CR-5385: INITIAL ASSESSMENT OF THE MECHANISMS AND SIGNIFICANCE OF LOW-TEMPERATURE EMBRITTLEMENT OF CAST STAINLESS STEELS IN LWR SYSTEMS.
NUREG/CR-5451: CRACK-ARREST BEHAVIOR IN SEN WIDE PLATES OF LOW-UPPER-SHELF BASE METAL TESTED UNDER NONISOTHERMAL CONDITIONS: WP-2 SERIES.
NUREG/CR-5515: LIGHT WATER REACTOR PRESSURE ISOLATION VALVE PERFORMANCE TESTING.

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License Renewal
NUREG-1362 DRF FC ER: Errata to "Regulatory Analysis for Proposed Rule on Nuclear Power Plant License Renewal,Draft Rept for Comment," revising Section 7.2 on Page 7-3 to read as listed.
NUREG-1398 DRFT FC: ENVIRONMENTAL ASSESSMENT FOR PROPOSED RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL.Draft Report For Comment.
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NUREG/CR-3444 V07: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION, WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.
NUREG/CR-4674 V11: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Main Report And Appendix A.
NUREG/CR-4674 V12: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Appendices B And C.
NUREG/CR-4744 V03 N2: LONG-TERM EMBRITTLEMENT OF CAST DUPLEX STAINLESS STEELS IN LWR SYSTEMS.Semiannual Report.April-September 1988.
NUREG/CR-5385: INITIAL ASSESSMENT OF THE MECHANISMS AND SIGNIFICANCE OF LOW-TEMPERATURE EMBRITTLEMENT OF CAST STAINLESS STEELS IN LWR SYSTEMS.
NUREG/CR-5451: CRACK-ARREST BEHAVIOR IN SEN WIDE PLATES OF LOW-UPPER-SHELF BASE METAL TESTED UNDER NONISOTHERMAL CONDITIONS: WP-2 SERIES.
NUREG/CR-5515: LIGHT WATER REACTOR PRESSURE ISOLATION VALVE PERFORMANCE TESTING.

Loss Of Coolant
NUREG/CR-5254: BIAS IN PEAK CLAD TEMPERATURE PREDICTIONS DUE TO UNCERTAINTIES IN MODELING OF ECC BYPASS AND DISSOLVED NON-CONDENSABLE GAS PHENOMENA.

Loss-Of-Coolant Accident
NUREG/CR-5395 V11: MULTILOOP INTEGRAL SYSTEM TEST (MIST):FINAL REPORT.MIST Phase IV Tests.
NUREG/CR-5395 V11 AD: MULTILOOP INTEGRAL SYSTEM TEST (MIST):FINAL REPORT.MIST Phase IV Tests.

5

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.

6

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.

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 V09 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1990.

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DIVISION OF SAFETY PROGRAMS (POST 870413)

NUREG/CR-4674 V11: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Main Report And Appendix A.

NUREG/CR-4674 V12: PRECURSORS TO POTENTIAL SEVERE CORE DAMAGE ACCIDENTS:1989 A STATUS REPORT.Appendices B And C.

EDO - OFFICE OF INFORMATION RESOURCES MANAGEMENT & ARM (POST 861109)

DIVISION OF COMPUTER & TELECOMMUNICATIONS SERVICES (POST 890205)

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NUREG/CR-5117: STEAM GENERATOR TUBE INTEGRITY PROGRAM/STEAM GENERATOR GROUP PROJECT.Final Project Summary Report.

NUREG/CR-5280 V01: AGE-RELATED DEGRADATION OF WESTINGHOUSE 480-VOLT CIRCUIT BREAKERS.Aging Assessment And Recommendations For Improving Breaker Reliability.

NUREG/CR-5385: INITIAL ASSESSMENT OF THE MECHANISMS AND SIGNIFICANCE OF LOW-TEMPERATURE EMBRITTLEMENT OF CAST STAINLESS STEELS IN LWR SYSTEMS.

NUREG/CR-5448: AGING EVALUATION OF CLASS 1E BATTERIES: SEISMIC TESTING.

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DIVISION OF REGULATORY APPLICATIONS (POST 870413)

NUREG/CR-3469 V05: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

DIVISION OF SAFETY ISSUE RESOLUTION (POST 880717)

NUREG/CR-5528: AN ASSESSMENT OF BWR MARK II CONTAINMENT CHALLENGES,FAILURE MODES, AND POTENTIAL IMPROVEMENTS IN PERFORMANCE.

NUREG/CR-5556: REVIEW OF CURRENT LITERATURE RELATED TO GENERIC SAFETY ISSUE 15.

NUREG/CR-5567: PWR DRY CONTAINMENT ISSUE CHARACTERIZATION.

NUREG/CR-5575: QUANTITATIVE ANALYSIS OF POTENTIAL PERFORMANCE IMPROVEMENTS FOR THE DRY PWR CONTAINMENT.

NUREG/CR-5589: ASSESSMENT OF ICE-CONDENSER CONTAINMENT PERFORMANCE ISSUES.

DIVISION OF SYSTEMS RESEARCH (POST 880717)

NUREG/CR-4624 V06: RADIONUCLIDE RELEASE CALCULATIONS FOR SELECTED SEVERE ACCIDENT SCENARIOS.Supplemental Calculations.

NUREG/CR-4794: EXPERIMENTAL RESULTS OF CORE-CONCRETE INTERACTIONS USING MOLTEN STEEL WITH ZIRCONIUM.

NUREG/CR-5254: BIAS IN PEAK CLAD TEMPERATURE PREDICTIONS DUE TO UNCERTAINTIES IN MODELING OF ECC BYPASS AND DISSOLVED NON-CONDENSABLE GAS PHENOMENA.

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8

Contractor Index

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

ARGONNE NATIONAL LABORATORY

NUREG/CP-0112 V01: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUB-CRITICAL CRACK GROWTH.Opening Session And Technical Session I.Held At Moscow,USSR,May 14-17,1990.

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NUREG/CR-5385: INITIAL ASSESSMENT OF THE MECHANISMS AND SIGNIFICANCE OF LOW-TEMPERATURE EMBRITTLEMENT OF CAST STAINLESS STEELS IN LWR SYSTEMS.

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BABCOCK & WILCOX CO.

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BATTELLE MEMORIAL INSTITUTE, COLUMBUS LABORATORIES

NUREG/CR-4624 V06: RADIONUCLIDE RELEASE CALCULATIONS FOR SELECTED SEVERE ACCIDENT SCENARIOS.Supplemental Calculations.

BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORY

NUREG/CR-2850 V09: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1987.

NUREG/CR-4908: ULTRASONIC INSPECTION RELIABILITY FOR INTERGRANULAR STRESS CORROSION CRACKS.A Round Robin Study Of The Effects Of Personnel, Procedures, Equipment, And Crack Characteristics.

NUREG/CR-5117: STEAM GENERATOR TUBE INTEGRITY PROGRAM/STEAM GENERATOR GROUP PROJECT.Final Project Summary Report.

BROOKHAVEN NATIONAL LABORATORY

NUREG/CR-3444 V07: THE IMPACT OF LWR DECONTAMINATIONS ON SOLIDIFICATION, WASTE DISPOSAL AND ASSOCIATED OCCUPATIONAL EXPOSURE.

NUREG/CR-3469 V05: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.

NUREG/CR-5254: BIAS IN PEAK CLAD TEMPERATURE PREDICTIONS DUE TO UNCERTAINTIES IN MODELING OF ECC BYPASS AND DISSOLVED NON-CONDENSABLE GAS PHENOMENA.

NUREG/CR-5280 V01: AGE-RELATED DEGRADATION OF WESTINGHOUSE 480-VOLT CIRCUIT BREAKERS.Aging Assessment And Recommendations For Improving Breaker Reliability.

NUREG/CR-5567: PWR DRY CONTAINMENT ISSUE CHARACTERIZATION.

NUREG/CR-5572: AN EVALUATION OF THE EFFECTS OF LOCAL CONTROL STATION DESGIN CONFIGURATIONS ON HUMAN PERFORMANCE AND NUCLEAR POWER PLANT RISK.

NUREG/CR-5588 V01: CARES (COMPUTER ANALYSIS FOR RAPID EVALUATION OF STRUCTURES) VERSION 1.0.Seismic Module.Theoretical Manual.

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NUREG/CR-5589: ASSESSMENT OF ICE-CONDENSER CONTAINMENT PERFORMANCE ISSUES.

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NUREG/CR-5568 V01: INDUSTRY BASED PERFORMANCE INDICATORS FOR NUCLEAR POWER PLANTS.Phase 1 Report: June 1989 - February 1990.

CONCORD ASSOCIATES, INC.

NUREG/CR-5568 V01: INDUSTRY BASED PERFORMANCE INDICATORS FOR NUCLEAR POWER PLANTS.Phase 1 Report: June 1989 - February 1990.

EG&G IDAHO, INC. (SUBS. OF EG&G, INC.)

NUREG/CR-5448: AGING EVALUATION OF CLASS 1E BATTERIES: SEISMIC TESTING.

NUREG/CR-5528: AN ASSESSMENT OF BWR MARK II CONTAINMENT CHALLENGES,FAILURE MODES, AND POTENTIAL IMPROVEMENTS IN PERFORMANCE.

NUREG/CR-5542: MODELS FOR ESTIMATION OF SERVICE LIFE OF CONCRETE BARRIERS IN LOW-LEVEL RADIOACTIVE WASTE DISPOSAL.

NUREG/CR-5556: REVIEW OF CURRENT LITERATURE RELATED TO GENERIC SAFETY ISSUE 15.

NUREG/CR-5575: QUANTITATIVE ANALYSIS OF POTENTIAL PERFORMANCE IMPROVEMENTS FOR THE DRY PWR CONTAINMENT.

ENERGY TECHNOLOGY ENGINEERING CENTER

NUREG/CR-5515: LIGHT WATER REACTOR PRESSURE ISOLATION VALVE PERFORMANCE TESTING.

KALSI ENGINEERING, INC.

NUREG/CR-5583: PREDICTION OF CHECK VALVE PERFORMANCE AND DEGRADATION IN NUCLEAR POWER PLANT SYSTEMS - WEAR AND IMPACT TESTS.Final Report,September 1988 - April 1990.

MATERIALS ENGINEERING ASSOCIATES, INC.

NUREG/CP-0112 V01: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUB-CRITICAL CRACK GROWTH.Opening Session And Technical Session I.Held At Moscow,USSR,May 14-17,1990.

NUREG/CP-0112 V02: PROCEEDINGS OF THE THIRD INTERNATIONAL ATOMIC ENERGY AGENCY SPECIALISTS' MEETING ON SUB-CRITICAL CRACK GROWTH.Technical Sessions II, III, And IV And Recommendations And Conclusions Session V.Held At Moscow,USSR,May 14-17, 1990.

NEW MEXICO STATE UNIV., LAS CRUCES, NM

NUREG/CR-5607: FLOW AND TRANSPORT AT THE LAS CRUCES TRENCH SITE.Experiments 1 And 2.

OAK RIDGE NATIONAL LABORATORY

NUREG/CR-2000 V09 N6: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of June 1990.

NUREG/CR-2000 V09 N7: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of July 1990.

NUREG/CR-2000 V09 N8: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of August 1990.

9

International Organization Index

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

Licensed Facility Index

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

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