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Vol. 17, No. 1

Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
First Quarter 1992
January – March

U.S. Nuclear Regulatory Commission

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Division of Freedom of Information and Publications Services
Office of Administration
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PREFACE

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

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

Secondary Report Number Index
Personal Author Index
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NRC Originating Organization Index (International Agreements)
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Contractor Index
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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. JANERP, J.S. Argonne National Laboratory. May 1981. 141 pp. 8105280299. ANL-81-3. 08632:070.

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

Contractor Report

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

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

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International Agreement Report

NUREG/IA-0001: ASSESSMENT OF TRAC-PD2 USING SUPER CANNON AND HDR EXPERIMENTAL DATA. NEUMANN, U. Kraftwerk Union. Augus. 986. 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 V16: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31, 1991.(Gray Book I) HARTFIELD,R.A. Division of Computer & Telecommunications Services (Post 890205). March 1992. 350pp. 9204060300. 61230:308.

The Nuclear Regulatory Commission's annual summary of licensed nuclear power reactor data is based primarily on the report of operating data submitted by licensees for each unit for the month of December because that report contains data for the month of December, the year to date (in this case calendar year 1991) and cumulative data, usually from the date of commercial operation. The data is not independently verified, but various computer checks are made. The report is divided into two sections. The first contains summary highlights and the second contains data on each individual unit in commercial operation. Section 1 capacity and availability factors are simple arithmetic averages. Section 2 items in the cumulative column are generally as reported by the licensee and notes as to the use of weighted averages and starting dates other than commercial operation are provided.

NUREG-0040 V15 N04: LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1991.(White Book) * Division of Reactor Inspection & Safeguards (Post 870411). January 1992. 331pp. 9202120151. 60555:002.

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from October through December 1991.

NUREG-0325 R15: U.S. NUCLEAR REGULATORY COMMISSION FUNCTIONAL ORGANIZATION CHARTS.January 31, 1992. * Ofc of Personnel (Post 870413). February 1992. 65pp. 9204100058. 61288:306.

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

NUREG-0386 D06 R01: UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission,Appeal Board And Licensing Decisions.July 1972 - March 1991. * Office of the General Counsel (Post 860701). February 1992. 728pp. 9203250316. 61088:232.

This revision of the sixth 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 of July 1, 1972, to March 31, 1991, interpreting the NRC's Rules of Practice in 10 CFR Part 2.

NUREG-0430 V11: LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1, 1990 - June 30, 1991.(Gray Book II) JOY,D.; BROWN,C. Office of Nuclear Material Safety & Safeguards. March 1992. 18pp. 9203250277. 61087:154.

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

NUREG-0540 V13 N11: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30, 1991. * Division of Freedom of Information & Publications Services (Post 890205). January 1992. 296pp. 9202110288. 60541:284.

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 of Enclosures to Principal Documents.

NUREG-0540 V13 N12: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1991. * Division of Freedom of Information & Publications Services (Post 890205). February 1992. 335pp. 9202270144. 60716:287.

See NUREG-0540,V13,N11 abstract.

NUREG-0540 V14 N01: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1992. * Division of Information Support Services (Post 890205). March 1992. 307pp. 9204060305. 61230:001.

See NUREG-0540,V13,N11 abstract.

NUREG-0750 V34 I01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-September 1991. * Division of Freedom of Information & Publications Services (Post 890205). January 1992. 51pp. 9202060417. 60494:001.

Digests and indexes for issuances of the Commission, 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 V34 N05: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1991. Pages 261-295. * Division of Freedom of Information & Publications Services (Post 890205). January 1992. 42pp. 9202060465. 60494:032.

Legal issuances of the Commission, the Atomic Safety and Licensing Board Panel, the Administrative Law Judges, and NRC Program Offices are presented.

NUREG-0750 V34 N06: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1991. Pages 297-376. * Division of Freedom of Information & Publications Services (Post 890205). February 1992. 87pp. 9203270098. 61102:135.

See NUREG-0750,V34,N05 abstract.

NUREG-0750 V35 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1992. Pages 1-46. * Division of Freedom of Information & Publications Services (Post 890205). March 1992. 52pp. 9204130029. 61315:020.

See NUREG-0750,V34,N05 abstract.

NUREG-0847 S08: SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennesse Valley Authority) TAM,P.S. Division of Reactor Projects - I/II (Post 870411). January 1992. 84pp. 9201310311. 60442:129.

Supplement No. 8 to the Safety Evaluation Report for the application filed by the Tennessee Valley Authority for license to operate Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391, located in Rhea County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation of: (1) additional information submitted by the applicant since Supplement No. 7 was issued, and (2) matters that the staff had under review when Supplement No. 7 was issued.

NUREG-0936 V10 N04: NRC REGULATORY AGENDA.Quarterly Report,October-December 1991. * Division of Freedom of Information & Publications Services (Post 890205). February 1992. 128pp. 9202240156. 60669:155.

The NRC Regulatory Agenda is a compilation of all rules on which the NRC has recently completed action, or has proposed action, 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 V10 N04: ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED.Quarterly Progress Report,October-December 1991. * Ofc of Enforcement (Post 870413). March 1992. 325pp. 9204060293. 61232:051.

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (October - December 1991) 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-1100 V08: BUDGET ESTIMATES.Fiscal Year 1993. * Division of Budget & Analysis (Post 890205). January 1992. 185pp. 9202180162. 60644:275.

This report contains the fiscal year budget justification to Congress. The budget provides estimates for salaries and expenses and for the Office of the Inspector General for fiscal year 1993.

NUREG-1135 S01: SAFETY EVALUATION REPORT RELATED TO THE CONSTRUCTION PERMIT AND OPERATING LICENSE FOR THE RESEARCH REACTOR AT THE UNIVERSITY OF TEXAS.Docket No. 50-602.(The University Of Texas) * Division of Advanced Reactors & Special Projects (901216-920516). January 1992. 80pp. 9202120148. 60548:178.

The Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRC) has prepared Supplement 1 to NUREG-1135, "Safety Evaluation Report Related to the Construction Permit and Operation License for the Research Reactor at the University of Texas" (SER) May 1985. The reactor facility is owned by The University of Texas at Austin (UT, the applicant) and is located at the University's Balcones Research Center in Austin, Texas. This supplement to the SER (SSER) describes the changes to the reactor facility design from the description in the SER. The SER and SSER together reflect the facility as built. The SSER also documents the reviews that the NRC has completed regarding the applicant's emergency plan, security plan, and technical specifications that were identified as open in the SER.

NUREG-1214 R09: HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. ALLENSPACH,F. Division of Licensee Performance & Quality Evaluation (Post 870411). February 1992. 123pp. 9203170266. 60947:010.

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-1324: PROPOSED METHOD FOR REGULATING MAJOR MATERIALS LICENSEES. HAUGHNEY,C.J.; BROWN,W.B.; ROTH,J.; et al. Office of Nuclear Material Safety & Safeguards. February 1992. 67pp. 9202240161. 60669:283.

The Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, appointed a Materials Regulatory Review Task Force to conduct a broad-based review of the Commission's current licensing and oversight programs for fuel cycle and large materials plants. The task force, as requested, defined the components and subcomponents of an ideal regulatory evaluation system for these types of licensed plants and compared them to the components and subcomponents of the existing regulatory evaluation system. This report discusses findings from this comparison and proposed recommendations on the basis of these findings.

NUREG-1449 DRFT FC: SHUTDOWN AND LOW-POWER OPERATION AT NUCLEAR POWER PLANTS IN THE UNITED STATES.Draft Report For Comment. * Division of Systems Technology (Post 890827). February 1992. 225pp. 9202260237. 60707:143.

The report contains the results of the NRC staff's evaluation of shutdown and low-power operations at commercial nuclear power plants in the United States. The report describes studies conducted by the staff in the following areas: operating experience related to shutdown and low-power operations, probabilistic risk assessment of shutdown and low- power conditions, and utility programs for planning and conducting activities during periods the plant is shut down. The report also documents evaluations of a number of technical issues regarding shutdown and low-power operations performed by the staff, including the principal findings and conclusions. Potential new regulatory requirements are discussed, as are potential changes in NRC programs. This report is currently a draft report issued for comment. It will be issued as a final report after the staff considers public comments and completes its regulatory analysis of potential new requirements in mid-1992.

NUREG/CP-0121: AGING RESEARCH INFORMATION CONFERENCE - ABSTRACTS OF PAPERS. BERANEK,A. Division of Engineering (Post 870413). March 1992. 87pp. 9203250301. 61086:162.

This report contains abstracts of papers to be presented at the Aging Research Information Conference to be held at the Holiday Inn Crowne Plaza in Rockville, Maryland, on March 24-27, 1992. This conference is held to disseminate research results in the area of nuclear power plant aging from programs sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission. The conference will also provide an opportunity for engineers and scientists from around the world to exchange technical information and discuss future international cooperation. The abstracts appear in the order in which they will be presented at the conference, and they are grouped by technical session. The full papers and the agenda for the conference will be published as separate documents.

NUREG/CR-2000 V10N12: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of December 1991. * Oak Ridge National Laboratory. January 1992. 69pp. 9202120139. ORNL/NSIC-200. 60548:107.

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 Report." 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 V11 N1: LICENSEE EVENT REPORT (LER) COMPILATION.For Month Of January 1992. * Oak Ridge National Laboratory. February 1992. 62pp. 9203130098. ORNL/NSIC-200. 60906:070.

See NUREG/CR-2000,V10,N12 abstract.

NUREG/CR-2850 V10: POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1988. BAKER,D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1992. 184pp. 9202120150. PNL-4221. 60548:249.

Population radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1988. Fifty-year dose commitments from a one-year exposure were calculated from both liquid and atmospheric releases for four population groups (infant, child, teenager and adult) residing between 2 and 80 km from each of 71 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 16 person-rem to a low of 0.001 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 75 person- rem for the 150 million people considered at risk.

NUREG/CR-4219 V08 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1990 - March 1991. PENNELL,W.E. Oak Ridge National Laboratory. February 1992. 82pp. 9202210347. ORNL/TM-9593. 60659:005.

The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission (NRC) by Oak Ridge National Laboratory (ORNL). The program focus is on the development and validation of technology for the assessment of fracture-prevention margins in commercial nuclear reactor pressure vessels. Reorganization of the original HSST Program into separate programs with emphasis on fracture- mechanics technology (HSST) and materials-irradiation effects (HSSI) was previously completed. The revised HSST Program is organized in 10 tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterization and properties, (4) special technical assistance, (5) crack-arrest technology, (6) cleavage-crack initiation, (7) cladding evaluations, (8) pressurized-thermal-shock technology, (9) analysis methods validation, and (10) fracture evaluation tests. The program tasks have been structured to place emphasis on the resolution fracture issues with near-term licensing significance. Resources to execute the research tasks are drawn from ORNL with subcontract support from universities and other research laboratories. Close contact is maintained with related research programs both in the United States and abroad.

NUREG/CR-4551 V2R1P3: EVALUATION OF SEVERE ACCIDENT RISKS: QUANTIFICATION OF MAJOR INPUT PARAMETERS.Experts' Determination Of Structural Response Issues. BREEDING,R.J.; HARPER,F.T.; BROWN,T.D.; et al. Sandia National Laboratories. March 1992. 282pp. 9204060038. SAND86-1309. 61233:016.

In support of the Nuclear Regulatory Commission's (NRC's) assessment of the risk from severe accidents at commercial nuclear power plants in the U.S. reported in NUREG-1150, the Severe Accident Risk Reduction Program (SAARP) has completed a revised calculation of the risk to the general public from severe accidents at five nuclear power plants: Surry, Sequoyah, Zion, Peach Bottom, and Grand Gulf. The emphasis in this risk analysis was not on determining a "so-called" point estimate of risk. Rather, it was to determine the distribution of risk, and to discover the uncertainties that account for the breadth of this distribution. Off-site risk initiation by events, both internal to the power station and external to the power station were assessed. Much of the important input to the logic models was generated by expert panels. This document presents the distributions and the rationale supporting the distributions for the questions posed to the Structural Response Panel.

NUREG/CR-4554 V01 R1: SCANS (SHIPPING CASK ANALYSIS SYSTEM) A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR SHIPPING CASK DESIGN REVIEW.User's Manual To Version 2a (Including Program Reference). GERHARD,M.A.; TRUMMER,D.J.; JOHNSON,G.L.; et al. Lawrence Livermore National Laboratory. February 1992. 207pp. 9203250286. UCID-20674. 61088:025.

SCANS (Shipping Cask ANalysis System) is a microcomputer-based system of computer programs and databases developed at the Lawrence Livermore National Laboratory (LLNL) for evaluating safety analysis reports on spent fuel shipping casks. SCANS is an easy-to-use system that calculates the global response to impact loads, pressure loads and thermal conditions, providing reviewers with an independent check on analyses submitted by licensees. SCANS is based on microcomputers compatible with the IBM-PC family of computers. The system is composed of a series of menus, input programs, cask analysis programs, and output display programs. All data is entered through fill-in-the-blank input screens that contain descriptive data requests. Analysis options are based on regulatory cases described in the Code of Federal Regulations (1983) and Regulatory Guides published by the U.S. Nuclear Regulatory Commission in 1977 and 1978.

NUREG/CR-4554 V03 R1: SCANS (SHIPPING CASK ANALYSIS SYSTEM) A MICROCOMPUTER BASED ANALYSIS SYSTEM FOR SHIPPING CASK DESIGN REVIEW.Theory Manual (Lead Slump In Impact Analysis And Verification Of Impact Analysis). CHUN,R.C.; LO,T.; MOK,G.C.; et al. Lawrence Livermore National Laboratory. February 1992. 103pp. 9203250288. UCID-20674. 61087:174.

A computer system called SCANS (Shipping Cask ANalysis System) has been developed for the staff of the U.S. Nuclear Regulatory Commission to perform confirmatory licensing review analyses. SCANS can handle problems associated with impact, heat transfer, thermal stress, and pressure. A new methodology was developed to allow SCANS to analyze the lead slump behavior of lead-shielded casks during a postulated impact with an unyielding surface. The methodology is an expansion of the existing lumped-parameter impact analysis method. In the new methodology, it is assumed that the lead and the steel cylinders are not bonded as opposed to the existing bonded-lead assumption. The lead shield is allowed to slide freely relative to the steel cylinders and interact with the steel cylinders only in the radial direction of the shipping cask. The lead slump methodology described in this revision (Rev. 1) of the report is an improved version of the method documented in the original report. The main improvement is in the modeling of the lead behavior. To minimize mathematical difficulty and development cost, the lead was formerly treated as an elastic material with an effective modulus which was tuned to account for the effect of plastic deformation occurring in a cask drop. Although this method gave satisfactory results for 30-ft accident drops, it produced overconservative predictions for 1- to 4-ft normal drops. Thus, the present revision of the method was undertaken to improve the range of applicability of the method. In the improved method described in this report, the lead is treated as an elastic-plastic material and the actual elastic-plastic properties of lead are used instead.

NUREG/CR-4627 R02: GENERIC COST ESTIMATES.Abstracts From Generic Studies For Use In Preparing Regulatory Impact Analyses. SCIACCA,F. Science & Engineering Associates, Inc. February 1992. 41pp. 9203250297. SEA87-25308-A:2. 61087:338.

The Nuclear Regulatory Commission has sponsored a number of generic cost estimating studies. These studies were prepared to aid NRC analysts in preparing Regulatory Impact Analyses (RIA's). These generic studies provide cost estimates that would have wide application to a large number of Regulatory Analyses being performed throughout the NRC and deal primarily with repair and modification activities that may be imposed on nuclear plants as a result of regulatory actions. Abstracts of each of the generic cost estimating studies have been prepared and assembled in this catalog. These abstracts present the results of the more detailed studies in a compact, easily understood and readily useable format. Individual abstracts have been developed to treat the main-line topics of the generic studies. In addition, abstracts have been prepared covering important sub-topics or "stand-alones" which are of broad interest in RIA preparation. This abstract catalog updates and revises information presented in NUREG/CR-4627, Rev.1. The catalog will be expanded and modified as additional generic cost studies are completed and as abstracts are modified to reflect updated conditions.

NUREG/CR-4667 V13: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April-September 1991. KASSNER,T.F.; RUTHER,W.E.; CHUNG,H.M.; et al. Argonne National Laboratory. March 1992. 48pp. 9203270025. ANL-92/6. 61101:161.

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking in light water reactors during the six months from April 1991 through September 1991. Topics that have been investigated during this period include: (l) fatigue and stress corrosion cracking (SCC) of low-alloy steel used in piping and in steam generator and reactor pressure vessels; (2) role of chromate and sulfate in simulated boiling water reactor (BWR) water on SCC of sensitized Type 304 SS; and (3) radiation-induced segregation (RIS) and irradiation-assisted SCC of Type 304 SS after accumulation of relatively high fluence. Fatigue data were obtained on medium-S- content A533-Gr B and A106-Gr B steels in high-purity (HP) deoxygenated water, in simulated pressurized water reactor (PWR) water, and in air. Crack-growth-rates (CGRs) of composite specimens of A533-Gr B/Inconel-182/Inconel-600 (plated with nickel) and homogeneous specimens of A533-Gr B were determined under small-amplitude cyclic loading in HP water with ≈300 ppb dissolved oxygen. CGR tests on sensitized Type 304 SS indicate that low chromate concentrations in BWR water (25-35 ppb) may actually have a beneficial effect on SCC if the sulfate concentration is below a critical level. Microchemical and microstructural changes in HP and commercial-purity Type 304 SS specimens from control-blade absorber tubes used in two operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy, and slow-strain-rate-tensile tests were conducted on tubular specimens in air and in simulated BWR water at 289 degrees C.

NUREG/CR-5229 V04: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1991.Annual Report. MCCONNELL,J.W.; ROGERS,R.D.; JASTROW,J.D.; et al. EG&G Idaho, Inc. January 1992. 62pp. 9202110294. EGG-2577. 60540:247.

The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission, is (a) studying the degradation effects in EPICOR-II organic ion- exchange resins caused by radiation, (b) examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Forms to meet the requirements of 10 CFR 61 using solidified EPICOR-II resins, (c) obtaining performance information on solidified EPICOR-II ion-exchange resins in a disposal environment, and (d) determining the condition of EPICOR-II liners. Results of the sixth year of data acquisition from the field testing are presented and discussed. During the continuing field testing, both Portland Type I-II cement and Dow vinyl ester-styrene waste forms are being tested in lysimeter arrays located at Argonne National Laboratory (ANL-E) in Illinois and Oak Ridge National Laboratory (ORNL). The study is designed to provide continuous data on nuclide release and movement, as well as environmental conditions, over a 20-year period.

NUREG/CR-5303 V01: SYSTEM ANALYSIS AND RISK ASSESSMENT SYSTEM (SARA) VERSION 4.0.Reference Manual. RUSSELL,K.D.; SATTISON,M.B.; SKINNER,N.L.; et al. EG&G Idaho, Inc. February 1992. 248pp. 9203170293. EGG-2628. 60947:145.

This document is the reference manual for the System Analysis and Risk Assessment (SARA) System Version 4.0, a microcomputer-based system used to analyze the safety issues of a family [i.e., a power plant, a manufacturing facility, any facility on which a probabilistic risk assessment (PRA) might be performed]. The SARA data base contains PRA data for the dominant accident sequences of a family and descriptive information about the family including event trees, fault trees, and system model diagrams. The number of facility data bases that can be accessed is limited only by the amount of disk storage available. To simulate changes to family systems, SARA users change the failure rates of initiating and basic events and/or modify the structure of the cut sets that make up the event trees, fault trees, and systems. The user then evaluates the effects of these changes through the recalculation of the resultant accident sequence probabilities and importance measures. The results are displayed in tables and graphs.

NUREG/CR-5303 V02: SYSTEM ANALYSIS AND RISK ASSESSMENT SYSTEM (SARA) VERSION 4.0.Tutorial. SATTISON,M.B.; RUSSELL,K.D.; SKINNER,N.L. EG&G Idaho, Inc. January 1992. 101pp. 9202110301. EGG-2628. 60540:146.

This document is the tutorial for the System Analysis and Risk Assessment System (SARA) Version 4.0, a microcomputer-based system used to analyze the safety issues of a family [i.e., a power plant, a manufacturing facility, any facility on which a probabilistic risk assessment (PRA) might be performed]. A series of lessons are provided that walk the user through some basic steps common to most analyses performed with SARA. The example problems presented in the lessons build on one another, and in combination, lead the user through all aspects of SARA sensitivity analysis.

NUREG/CR-5444: INSTRUMENT AVAILABILITY DURING SEVERE ACCIDENTS FOR A BOILING WATER REACTOR WITH A MARK I CONTAINMENT. ARCIERI,W.C.; HANSON,D.J. EG&G Idaho, Inc. February 1992. 126pp. 9203130127. EGG-2661. 60906:297.

In support of the U.S. Nuclear Regulatory Commission Accident Management Research Program, the availability of instruments to supply accident management information during a broad range of severe accidents is evaluated for a Boiling Water Reactor with a Mark I containment. Results from this evaluation include: (a) the identification of plant conditions that would impact instrument performance and information needs during severe accidents, (b) the definition of envelopes of parameters that would be important in assessing the performance of plant instrumentation for a broad range of severe accident sequences, and (c) assessment of the availability of plant instrumentation during severe accidents. A similar evaluation for a pressurized water reactor with a large, dry containment design is presented in NUREG/CR-5691.

NUREG/CR-5535 V05: RELAP5/MOD3 CODE MANUAL.User's Guidelines. FLETCHER,C.D.; SCHULTZ,R.R. EG&G Idaho, Inc. January 1992. 375pp. 9203270007. EGG-2596. 61100:146.

The RELAP5 code has been developed for best-estimate transient simulation of light water reactor coolant systems during a severe accident. The code models the coupled behavior of the reactor coolant system and the core during a severe accident transient, as well as large and small break loss-of-coolant accidents and operational transients, such as anticipated transients without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modeling approach is used that permits as much of a particular system to be modeled as necessary. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater conditioning systems. RELAP5/MOD3 code documentation is divided into five volumes: Volume 1 describes modeling theory and associated numerical schemes; Volume 2 contains detailed instructions for code application and input data preparation; Volume 3 provides the results of developmental assessment cases that demonstrate and verify the models used in the code; and Volume 4 presents a detailed discussion of RELAP5 models and correlations. Volumes 1-4 are in varying stages of development. This document, Volume 5, contains guidelines that have evolved over the past several years through use of the RELAP5 code.

NUREG/CR-5631 R01: CONTRIBUTION OF MATERNAL RADIONUCLIDE BURDENS TO PRENATAL RADIATION DOSES.Interim Recommendations. SIKOV,M.R.; TRAUB,R.J.; HUI,T.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1992. 179pp. 9204100139. PNL-7445. 61289:230.

This report describes approaches for calculating and expressing radiation doses to the embryo/fetus from internal radionuclides. Information was obtained for selected, occupationally significant radionuclides to provide metabolic and dosimetric characteristics. Fractional placental transfer and ratios of concentration in the embryo/fetus to that in the woman were calculated. This information was integrated with data from biokinetic transfer models to estimate the levels of radioactivity in the embryo/fetus as a function of stage of pregnancy and time after entry. The MIRD methodologies were extended to describe details for calculating radiation doses to the embryo/fetus. To accommodate the stage dependence of geometric relationships and biological behaviors, calculations were performed for a representative situation of an introduction of 1 μCi into a woman's transfer compartment (blood) at successive months of pregnancy. Detailed tables of the initial and retained fractions of activity in the embryo/fetus, and the corresponding radiation dose rates and doses are presented. These approaches yield radiation absorbed doses, and multiplication by quality factor (Q) converts them to dose equivalent. This is the most common quantity for stating prenatal dose limits and is appropriate for the unique effects of prenatal exposure. Our knowledge is currently inadequate to warrant the use of effective dose equivalent or committed dose equivalent.

NUREG/CR-5643: INSIGHTS GAINED FROM AGING RESEARCH. BLAHNIK,D.E.; CASADA,D.A.; EDSON,J.L.; et al. Brookhaven National Laboratory. March 1992. 118pp. 9204130081. BNL-NUREG-52323. 61314:047.

The USNRC Office of Nuclear Regulatory Research has implemented hardware-oriented engineering research programs to identify and resolve technical issues related to the aging of systems, structure, and components (SSCs) in operating nuclear power plants. This report provides a summary of those research results which have been compiled and published in NUREGs and related technical reports. The systems, components and structures that have been studied are organized by alphabetical order. The research results summary on the SSCs is followed by an assessment guide to emphasize inspection techniques which may be useful for detecting aging degradation in nuclear power plants. This report will be updated periodically to reflect new research results on these or other SSCs.

NUREG/CR-5650: EXTRAPOLATION OF THE J-R CURVE FOR PREDICTING REACTOR VESSEL INTEGRITY. LANDES,J.D. Tennessee, Univ. of, Knoxville, TN. * Oak Ridge National Laboratory. January 1992. 101pp. 9202060478. ORNLSUB89997321. 60490:261.

The work in this report was conducted in support of the issues studied by the U.S. Nuclear Regulatory Commission J(D)/J(M) Workers Group during the period 1987-1989. The major issues studied were the J-R curve extrapolation techniques for using small-specimen test results to predict ductile instability in larger structures where the extent of crack extension from the small-specimen test was not sufficient. An additional issue was raised during the course of this work by the testing of a low-upper-shelf A 302 steel. The results from these tests were not typical of ductile fracture in many steels and suggested that small-specimen J-R curves may not predict the behavior of large structures in some cases. The cause of this behavior was studied as well as the consequences of using the J-R curve results from small specimens of this kind of material. Finally, a discussion and recommendations are given relating to the use of extrapolated J-R curves.

NUREG/CR-5674: EVALUATION OF BEHAVIOR AND THE RADIAL SHEAR STRENGTH OF A REINFORCED CONCRETE CONTAINMENT STRUCTURE. WALTHER,H.P. Illinois, Univ. of, Urbana, IL. January 1992. 181pp. 9201310319. SAND91-7058. 60442:213.

This study is on the behavior and strength of the 1/6-scale reinforced concrete containment model tested at Sandia National Laboratories. The containment model was pressurized to more than three times its design pressure until a tear in the liner terminated the test. Deformation data from the test was used to interpret behavior and to estimate the internal forces at the wall-basemat connection. A possible mode of structural failure of containments subjected to high pressures is by radial shear failure at the wall-basemat connection. Although the containment model showed no sign that such a failure was imminent when the test was stopped, if it had been possible to increase the internal pressure, an abrupt shear failure was possible. A method based on the compressive force due to flexure at the wall-base was developed to evaluate the radial shear strength of the 1/6-scale containment. Using the developed methodology, an estimate is made of the pressure that would initiate a shear failure at the wall-basemat junction of the model. This estimate is based on a projection of the observed strength of similar 1/12-scale wall-basemat connections, which have failed in shear.

NUREG/CR-5708: POTENTIODYNAMIC POLARIZATION STUDIES ON CANDIDATE CONTAINER ALLOYS FOR THE TUFF REPOSITORY. THOMPSON,N.G.; BEAVERS,J.A.; DURR,C.L. Cortest Columbus Technologies, Inc. (formerly Cortest Columbus, Inc.). January 1992. 266pp. 9202210366. 60703:321.

Cortest Columbus Technologies, Inc. (CC Technologies) is investigating the long-term performance of container materials used for high-level radioactive waste packages. This information is being developed for the Nuclear Regulatory Commission to aid in their assessment of the Department of Energy's application to construct a geologic repository for disposal of high-level radioactive waste. This report summarizes the results of cyclic-potentiodynamic-polarization (CPP) studies performed on candidate container materials for the Tuff Repository. The CPP technique was used to provide an understanding of how specific variables such as environmental composition, temperature, alloy composition, and welding affect both the general- and localized-corrosion behavior of two copper-base and two Fe-Cr-Ni alloys in simulated repository environments. A statistically-designed test solution matrix was formulated, based on an extensive search of the literature, to evaluate the possible range of environmental species that may occur in the repository over the life of the canister. Forty-two CPP curves were performed with each alloy and the results indicated that several different types of corrosion were possible. The copper-base alloys exhibited unusual CPP behavior in that hysteresis was not always associated with pitting. The effects of temperature on the corrosion behavior were evaluated in two types of tests; isothermal tests at temperatures from 50 degrees C to 90 degrees C and heat-transfer tests where the solution was maintained at 50 degrees C and the sperimen was internally heated to 90 degrees C. In the isothermal test, CPP curves were obtained with each alloy in simulated environments at 50 degrees C, 75 degrees C, and 90 degrees C. The results of these CPP experiments indicated that no systematic trends were evident for the environments tested. In the heat-transfer test, CPP tests were performed with a specimen internally heated to 90 degrees C while maintaining the test solution at 50 degrees C. The results of these experiments indicated that in simulated J-13 well water, heat transfer appeared to have an effect on the corrosion behavior of each of the four alloys. Heat transfer did not appear to have a major effect in more aggressive simulated environments. Lastly, the effects of welding on the corrosion behavior of the alloys in simulated environments were examined. Rod material was welded into a V-shaped grove in plate material. The weld was machined and evaluated by the CPP technique. These studies showed that welding had relatively little effect on the CPP behavior of the Fe-Cr-Ni alloys in the environments that were selected. Welding was found to be detrimental to the performance of the copper-base alloys in both simulated groundwater and in a solution shown to promote pitting of the wrought copper-base alloys.

NUREG/CR-5709: PITTING,GALVANIC, AND LONG-TERM CORROSION STUDIES ON CANDIDATE CONTAINER ALLOYS FOR THE TUFF REPOSITORY. BEAVERS,J.A.; THOMPSON,N.G.; DURR,C.L. Cortest Columbus Technologies, Inc. (formerly Cortest Columbus, Inc.). January 1992. 219pp. 9202240328. 60683:001.

Cortest Columbus Technologies, Inc. (CC Technologies) investigated the long-term performance of container materials for high-level radioactive waste packages for the Tuff Repository. This report summarizes the results of Task 4 (Pitting Studies), Task 6 (Other Failure Modes) and Task 7 (Long-Term Exposures) of the program. Fe-Cr-Ni alloys (Alloy 304L and Alloy 825) and copper-base alloys (CDA 102 and CDA 715) were evaluated in a simulated J-13 well water and in solutions selected from Task 2 of the program. Pit-initiation studies of the copper-base alloys confirmed that standard interpretations of CPP tests are not always appropriate in the presence of thick oxide layers. Hysteresis in CPP tests may not always be indicative of pitting. Pit- propagation studies with Alloy CDA 102 showed that, if pits initiate, their propagation may be limited by the concentration of oxidizing species such as hydrogen peroxide (H(2)O(2)). Thermogalvanic effects on corrosion were found to be, in general, minor in comparison to the deleterious effect of increasing temperature on corrosion rate. In borehole liner-container interaction studies, performed with Alloy 304L - C1010 and Alloy 825 - Alloy 304L galvanic couples, the active member of the couple consistently experienced accelerated corrosion. Long-term, boil-down studies showed negligible general corrosion rates for Alloys 825, 304L, and CDA 715 following eighty weeks of exposure in concentrated simulated J-13 well water at 90 degrees C. Alloy CDA 102 experienced a general corrosion rate of 0.45 μm/yr in this environment. No SCC of U-bend specimens of any of the four alloys occurred. Alloy 304L and Alloy 825 exhibited no evidence of localized corrosion but some localized corrosion was evident on specimens of Alloy CDA 102 and Alloy CDA 715.

NUREG/CR-5725: PROGRESS REPORT ON HOT PARTICLE STUDIES. BAUM,J.W.; KAURIN,D.G.; WALIGORSKI,M.; et al. Brookhaven National Laboratory. February 1992. 49pp. 9203130114. BNL-NUREG-52287. 60906:137.

NCRP Report 106 on the effects of hot particles on the skin of pigs, monkeys, and humans was critically reviewed and reassessed. The analysis of the data of Forbes and Mikhail on the effects from activated UC(2) particles, ranging in diameter from 144 Im to 328 Im, led to the formulation of a new model to predict both the threshold for acute ulceration and for ulcer diameter. In this model, a point dose of 27 Gy at a depth of 1.33 mm in tissue will cause an ulcer with a diameter determined by the radius to which this dose extends. Application of the model to the Forbes and Mikhail data obtained with mixed fission product beta particles yielded a "threshold" (5% probability) of 5 x 10(9) beta particles from a point source of high energy (2.25 MeV maximum) beta particles on skin. The above model was used to predict that approximately 1.2 x 10(10) beta particles from Sr-Y-90 would produce similar effects, since few Sr-90 beta particles reach 1.33 mm depth. These emissions correspond to doses at 70-Im depth in tissue of approximately 5.3 to 5.5 Gy averaged over 1 cm(2), respectively.

NUREG/CR-5747 DRF FC: ESTIMATE OF RADIONUCLIDE RELEASE CHARACTERISTICS INTO CONTAINMENT UNDER SEVERE ACCIDENT CONDITIONS.Draft Report For Comment. NOURBAKHSH,H.P. Brookhaven National Laboratory. January 1992. 122pp. 9202240339. BNL-NUREG-52289. 60682:213.

A detailed review of the available light water reactor source term information is presented as a technical basis for development of updated source terms into the containment under severe accident conditions. Simplified estimates of radionuclide release and transport characteristics are specified for each unique combination of the reactor coolant and containment system combinations. A quantitative uncertainty analysis in the release to the containment using NUREG-1150 methodology is also presented.

NUREG/CR-5762: COMPREHENSIVE AGING ASSESSMENT OF CIRCUIT BREAKERS AND RELAYS. GLEASON,J.F. Wyle Laboratories. March 1992. 284pp. 9203270017. WYLE 60101. 61101:211.

As part of the NRC Nuclear Plant Aging Research (NPAR) Program, a comprehensive aging assessment was made of relays and circuit breakers. Relays and circuit breakers are important nuclear power plant equipment which are susceptible to degradation with time. This is a Phase II NPAR report which follows the NPAR strategy. Tests on naturally aged and degraded relays and circuit breakers were performed, in-situ measurements made and current and improved methods for inspection, surveillance and monitoring evaluated. Significant results described in this report were the identification of inspection, surveillance and monitoring methods which provide a higher level of assurance that aging will be detected and mitigated. The potential exists that implementation of the improved methods in nuclear plants would minimize the impact of aging and result in more cost effective maintenance on relays and circuit breakers.

NUREG/CR-5769: NATURAL CIRCULATION COOLING IN U.S. PRESSURIZED WATER REACTORS. MCHUGH,P.R.; HENTZEN,R.D. EG&G Idaho, Inc. January 1992. 165pp. 9201310306. EGG-2653. 60441:324.

This document is a synthesis of data and analysis concerning natural circulation cooling in U.S. Pressurized Water Reactors during off-normal operation and accident transients. Its objective is the integration of important research findings concerning PWR natural circulation phenomena into a single reference document. Sources of information include the Nuclear Regulatory Commission, reactor vendors, utility sponsored research groups, utilities, national laboratories, research reports, meeting papers, archival literature, and foreign sources. Three modes of natural circulation are discussed: single-phase, two-phase, and reflux/boiling condensation. General characteristics, analytical expressions, noncondensible gas effects, secondary effects, and nonuniform flow are described with regard to each of the natural circulation modes. Plant operational data, tests in scaled experimental facilities, and analysis with thermal hydraulic system codes have demonstrated the effectiveness of single-phase natural circulation as a cooling mechanism. Evidence suggests that two-phase natural circulation and reflux/boiling condensation can also be effective methods of alternate core cooling. Experimental test facility data and analysis are the primary components of the two-phase and reflux/boiling condensation natural circulation knowledge base.

NUREG/CR-5774: ELASTIC-PLASTIC CHARACTERIZATION OF A CAST STAINLESS STEEL PIPE ELBOW MATERIAL. JOYCE,J.A. U.S. Naval Academy, Annapolis, MD. HACKETT,E.M.; ROE,C. David Taylor Research Center. January 1992. 212pp. 9202240346. 60682:001.

Tests conducted in Japan as part of the High Level Vibration Test (HLVT) program for reactor piping systems revealed fatigue crack growth in a cast stainless steel pipe elbow. The material tested was equivalent to ASME SA-351CF8M. The David Taylor Research Center (DTRC) was tasked to develop the appropriate material property data to characterize cyclic deformation, cyclic elastic-plastic crack growth and ductile tearing resistance in the pipe elbow material. It was found that the cast stainless steel was very resistant to ductile crack extension. J-R curves essentially followed a blunting behavior to very high J levels. Low cycle fatigue crack growth rate data obtained on this material using a cyclic J integral approach was consistent with the high cycle fatigue crack growth rate and with a standard textbook correlation equation typical for this type of material. Evaluation of crack closure effects was essential to accurately determine the crack driving force for cyclic elastic-plastic crack growth in this material. SEM examination of several of the cyclic J test fracture surfaces indicated that fatigue was the primary mode of fracture with ductile crack extension intervening only during the last few cycles of loading.

NUREG/CR-5775: QUANTITATIVE EVALUATION OF SURVEILLANCE TEST INTERVALS INCLUDING TEST-CAUSED RISKS. KIM,I.S.; MARTORELL,S.; VESELY,W.E.; et al. Brookhaven National Laboratory. February 1992. 77pp. 9203250305. BNL-NUREG-52296. 61086:251.

Concerns have been raised regarding the adverse safety impact of surveillance testing and generally overburdensome surveillance requirements. To evaluate these concerns, the risk-effectiveness of surveillance tests has been studied with explicit consideration of the negative risk impact, in conjunction with the positive risk impact. This report defines the negative effects of surveillance testing from a risk perspective, and then presents the methodology by which the negative risk impact can be quantified, focusing on two important kinds of negative risk impact of surveillance testing: (1) risk impact of test-caused trips; and (2) risk impact of test-caused equipment wear. Using the methodology presented, these negative risk impacts are evaluated for a selected set of surveillance tests for demonstration examples. The results of the risk-effectiveness evaluation are provided along with the insights from the sensitivity analyses.

NUREG/CR-5787 DRF FC: TIMING ANALYSIS OF PWR FUEL PIN FAILURES.Draft Report For Comment. JONES,K.R.; WADE,N.L.; KATSMA,K.R.; et al. EG&G Idaho, Inc. March 1992. 710pp. 9204060048. EGG-2657. 61226:167.

Research has been conducted to develop and demonstrate a methodology for calculating the time interval between receipt of containment isolation signals and the first fuel pin failure for loss- of-coolant accidents (LOCAs). Demonstration calculations were performed for a Babcock and Wilcox design (Oconee) and a Westinghouse 4-loop design (Seabrook). Sensitivity studies assessed the impact of fuel pin burnup, axial peaking factor, break size, emergency core cooling system availability, and main coolant pump trip on these times. The analysis used SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1 to calculate reactor system transient thermal-hydraulic conditions and FRAPCON-2 and FRAP-T6 to calculate steady-state and transient fuel behavior. This analysis also provides a comparison of SCDAP/RELAP5/MOD3 and TRAC-PF1/MOD1 results for large-break LOCA analysis. Using SCDAP/RELAP5/MOD3 thermal-hydraulic data, the shortest time intervals calculated between containment isolation and fuel pin failure are 10.4 seconds and 19.1 seconds for the B&W and W plants, respectively. Using data generated by TRAC-PF1/MOD1, the shortest interval for the W reactor is 29.1 seconds. These intervals are for a double-ended, offset-shear, cold leg break, using maximum peaking factor applied to fuel with maximum burnup.

NUREG/CR-5788: A COMPARISON OF WEIBULL AND B(IC) ANALYSIS OF TRANSITION RANGE FRACTURE TOUGHNESS DATA. MCCABE,D.E. Oak Ridge National Laboratory. January 1992. 40pp. 9202240164. ORNL/TM-11959. 60669:112.

Characteristics of extremal statistics that are used to predict size effects on cleavage fracture toughness in the transition range were explored. A 533 grade B steel base and weld metals were tested using compact specimens ranging in size from 1/2TC(T) to 8TC(T) and with sufficient replication in some cases to provide good fits to Weibull distributions. The classical specimen size effect on data scatter and median K_{Jc} toughness at a given test temperature was observed in the low- to mid-transition range. These effects were well predicted with extremal statistics. However, the same model is not applicable on the lower shelf and it also becomes extremely weak and unreliable in the mid- to high-transition range. The Irwin β_c-β_{Ic} relationship was also explored as a model and was found to predict similar size effects. The predictive characteristics of the latter seemed better suited to deal with the diminution of size effects in the near- to low-shelf toughness range. In the rising toughness part of the transition, the predictive characteristics were about the same as the statistical model up to where β_c (β_{Jc} in this study) of the baseline (small specimen) data were π or less. This work could be used in the establishment of a framework for transition temperature test criteria. Upper- and lower-bound β_{Jc} criteria could be used to define optimum conditions for the application of either of the aforementioned models. For surveillance programs, sensible rules should be specified as to specimen size requirements and numbers of specimens to be tested in order to apply these analytical models. Another need would be the definition of a procedure for the Weibull distribution fitting. The present report suggests items to be considered for requirements in application of these predictive techniques.

NUREG/CR-5799: REVIEW OF REACTOR PRESSURE VESSEL EVALUATION REPORT FOR YANKEE ROWE NUCLEAR POWER STATION (YAEC NO. 1735). CHEVERTON,R.D.; DICKSON,T.L.; MERKLE,J.G.; et al. Oak Ridge National Laboratory. March 1992. 149pp. 9204130054. ORNL/TM-11982. 61315:073.

The Yankee Atomic Electric Company (YAEC) has performed an Integrated Pressurized Thermal Shock (IPTS)-type evaluation of the Yankee Rowe reactor pressure vessel in accordance with the PTS Rule (10 CFR 50.61) and U.S. Regulatory Guide 1.154. The Oak Ridge National Laboratory (ORNL) reviewed the YAEC document and performed an independent probabilistic fracture-mechanics analysis. The review included a comparison of the Pacific Northwest Laboratory and the ORNL probabilistic fracture-mechanics codes (VISA-II and OCA-P, respectively). The review identified minor errors and one significant difference in philosophy. Also, the two codes have a few dissimilar peripheral features. Aside from these differences, VISA-II and OCA-P are very similar and with errors corrected and when adjusted for the difference in the treatment of fracture toughness distribution through the wall, yield essentially the same value of the conditional probability of failure. The ORNL independent evaluation indicated RT(NDT) values considerably greater than those corresponding to the PTS-Rule screening criteria and a frequency of failure substantially greater than that corresponding to the "primary acceptance criterion" in U.S. Regulatory Guide 1.154. Time constraints, however, prevented as rigorous a treatment as the situation deserves. Thus, these results are very preliminary.

NUREG/CR-5802: IDENTIFICATION AND ASSESSMENT OF CONTAINMENT AND RELEASE MANAGEMENT STRATEGIES FOR A BWR MARK III CONTAINMENT. LIN,C.C.; LEHNER,J.R. Brookhaven National Laboratory. VANDENKIEBOOM Michigan, Univ. of, Ann Arbor, MI. February 1992. 109pp. 9203130120. BNL-NUREG-52305. 60906:188.

This report identifies and assesses accident management strategies which could be important for preventing containment failure and/or mitigating the release of fission products during a severe accident in a BWR plant with a Mark III type of containment. Based on information available from probabilistic risk assessments and other existing severe accident research, and using simplified containment and release event trees, the report identifies the challenges a Mark III containment could face during the course of a severe accident, the mechanisms behind these challenges, and the strategies that could be used to mitigate the challenges. The strategies are linked to the general safety objectives which apply for containment and release management by means of a safety objective tree. The strategies were assessed by applying them to certain severe accident sequence categories deemed important for a Mark III containment because of one or more of the following characteristics: high probability of core damage, high consequences, lead to a number of challenges, and involve the failure of multiple systems.

NUREG/CR-5813 V01: INTEGRATED RELIABILITY AND RISK ANALYSIS SYSTEM (IRRAS) VERSION 4.0 Reference Manual. RUSSELL,K.D.; MCKAY,M.K.; SATTISON,M.B.; et al. EG&G Idaho, Inc. January 1992. 334p. 9202110290. EGG-2664. 60540:310.

The Integrated Reliability and Risk Analysis System (IRRAS) is a state-of-the-art, microcomputer-based probabilistic risk assessment (PRA) model development and analysis tool to address key nuclear plant safety issues. IRRAS is an integrated software tool that gives the user the ability to create and analyze fault trees and accident sequences using a microcomputer. This program provides functions that range from graphical fault tree construction to cut set generation and quantification. Version 1.0 of the IRRAS program was released in February of 1987. Since that time, many user comments and enhancements have been incorporated into the program providing a much more powerful and user-friendly system. This version has been designated IRRAS 4.0 and is the subject of this Reference Manual. Version 4.0 of IRRAS provides the same capabilities as Version 1.0 and adds a relational data base facility for managing the data, improved functionality, and improved algorithm performance.

NUREG/CR-5815: EVALUATIONS OF 1990 PRISM DESIGN REVISIONS. VAN TUYLE,G.J.; SLOVIK,G.C.; CHAN,B.C.; et al. Brookhaven National Laboratory. March 1992. 215pp. 9204130073. BNL-NUREG-52311. 61514:165.

Analyses of the 1990 version of the PRISM Advanced Liquid Metal Reactor (ALMR) design are presented and discussed. Most of the calculations were performed using BNL computer codes, particularly SSC and MINET. In many cases, independent BNL calculations were compared against analyses presented by General Electric when they submitted the PRISM design revisions for evaluation by the Nuclear Regulatory Commission (NRC). The current PRISM design utilizes the metallic fuel developed by Argonne National Laboratory (ANL) which facilitates the passive/"inherent" shutdown mechanism that acts to shut down reactor power production whenever the system overheats. There are a few vulnerabilities in the passive shutdown, with the most worrisome being the positive feedback from sodium density decreases or sodium voiding. Various postulated unscrammed events were examined by GE and/or BNL, and much of the analysis discussed in this report is focused on this category of events. For the most part, the BNL evaluations confirm the information submitted by General Electric. The principal areas of concern are related to the performance of the ternary metal fuel, and may be resolved as ANL continues with its fuel development and testing program.

NUREG/CR-5817: REPORT ON RESEARCH ACTIVITIES FOR CALENDAR YEAR 1990. ABABOU,R.; CHOWDHURY,A.H.; CRAGNOLINO,G.; et al. Center for Nuclear Waste Regulatory Analyses. December 1991. 325pp. 9202260218. CNWRA 90-01A. 60706:123.

This is an annual status report on the results of research conducted on behalf of the NRC by the Center for Nuclear Waste Regulatory Analyses in support of activities under the Nuclear Waste Policy Act, as Amended. Eight specific projects are underway. The Geochemistry project is using laboratory methods and computer calculations to assess key geochemical constraints and to evaluate sorptive properties of zeolite present at the proposed repository site. The Thermohydrology project has as its focus improved understanding of heat and fluid flow in unsaturated media. Laboratory, field, and calculational studies are combined in the Seismic Rock Mechanics project to examine the effects of repeated seismic loadings on the rock- mechanical and hydrological responses of rock masses. The Integrated Waste Package Experiments have been initiated to evaluate degradation modes of candidate waste container alloys. Three-dimensional computer analysis techniques are being used to investigate spatial variability of flow and transport in variably saturated fractured porous media in the Stochastic Flow and Transport project. The recently initiated Geochemical Analogs project seeks to investigate the role of such analogs in the licensing process, and is currently focused on locating and evaluating a potential site for investigation. The Performance Assessment project is directed toward developing and evaluating methodologies for evaluation of the long-term performance of the proposed repository.

NUREG/CR-5841: VERIFICATION OF NONLINEAR PIPING RESPONSE CALCULATION WITH DATA FROM SEISMIC TESTING OF AN IN-PLANT PIPING SYSTEM. SRINIVASAN,M.G.; JOJTAHED,M.; KOT,C.A. Argonne National Laboratory. March 1992. 215pp. 9204100152. ANL-92/4. 61289:015.

The nonlinear piping analysis code NONPIPE was evaluated by modeling and performing posttest calculations for a high level seismic test performed on an in-plant piping system at the HDR test facility in Germany. The piping used only rigid struts for dynamic supports and experienced significant nonlinear response and plastic deformation in the test. The calculated responses in general had the same time trends as the measurements but were quite variable in estimating peak values. Peak strains were overestimated, the shear strains to a greater degree than bending strains. Peak strut forces were underestimated as a group. On the average the displacements were estimated close to the measurements and the accelerations were overestimated. Spatial consistency in displacement and strut force estimates was better than for accelerations. Qualitatively the analytical predictions are similar to experimental observations and they provide quantitative estimates that are useful for practical purposes. The modeling aspects that contribute to large discrepancies are identified. Both the experiments and calculations clearly indicate that piping is all but immune to gross failure even when subjected to extreme seismic loading.

NUREG/CR-5842: GASTROINTESTINAL ABSORPTION OF PLUTONIUM,URANIUM, AND NEPTUNIUM IN FED AND FASTED ADULT BABOONS:APPLICATION TO HUMANS. BHATTACHARYYA; LARSEN,R.P.; OLDHAM,R.D.; et al. Argonne National Laboratory. March 1992. 59pp. 9203250282. ANL-92/8. 61087:277.

Gastrointestinal (GI) absorption values of plutonium, uranium, and neptunium were determined in fed and fasted adult baboons. A dual isotope method of determining GI absorption, which does not require animal sacrifice, was validated and shown to compare well with the sacrifice method (summation of oral isotope in urine with that in tissues at sacrifice). For plutonium, GI absorption values in baboons were almost identical to those in mice (ca. 0.2% in fasted animals, 0.01% in fed animals), and the values for fed animals agreed with estimates for humans. For uranium, GI absorption values in fed (0.5%) and fasted (4%) baboons agreed well with those in fed and fasted humans and were 6-7 times higher than those in mice. For neptunium, GI absorption values in fed baboons were lower for small amounts of (239)Np (0.03%) than for much larger amounts of (237)Np (0.3%). Neptunium GI absorption values in fasted baboons (1.5%) were independent of amounts ingested and were considerably higher than those in fed animals. The GI absorption of (239)Np in fed animals was essentially the same in mice, baboons, and humans. In fasted animals, mice absorbed 4 times less (239)Np than did baboons (values for humans are not available). For one baboon that was not given its morning meal, both plutonium and neptunium GI absorption values at 0900 hours, 2 h after the usual mealtime (14-h overnight fast, baboon "without breakfast"), were the same as those in baboons fasted for 24 h. In contrast, for baboons that received a morning meal, plutonium and neptunium absorptions did not rise to the value found in 24-h-fasted baboons even 8 h after the meal. The authors conclude that GI absorption values for plutonium, uranium, and neptunium in adult baboons are good estimates of the values in humans (and better than those in mice) and that the values for the fasted condition need to be taken into account when standards are set for oral exposures in environmental and workplace settings. A rational way of doing this for plutonium is discussed.

NUREG/CR-5856: IDENTIFICATION AND EVALUATION OF PWR IN-VESSEL SEVERE ACCIDENT MANAGEMENT STRATEGIES. DUKELOW,J.S. Battelle Memorial Institute, Pacific Northwest Laboratory. HARRISON,D.G. Jason Associates. MORGENSTERN,M. Battelle Human Affairs Research Centers. March 1992. 103pp. 9203270010. PNL-8022. 61115:111.

This report documents work performed for the NRC/RES Accident Management Guidance Program to evaluate possible strategies for mitigating the consequences of PWR severe accidents. The selection and evaluation of strategies was limited to the in-vessel phase of the severe accident, i.e., after the initiation of core degradation and prior to reactor pressure vessel failure. A parallel project at Brookhaven National Laboratory has been considering strategies applicable to the ex-vessel phase of PWR severe accidents.

NUREG/GR-0005 V01: RISK-BASED INSPECTION - DEVELOPMENT OF GUIDELINES:General Document. * American Society of Mechanical Engineers. February 1992. 167pp. 9203250310. CRTD-VOL.20-1. 61086:327.

Inservice inspection can play a significant role in minimizing equipment and structural failures. For many industrial applications, requirements for inservice inspection are based upon prior experience or engineering judgment, or are nonexistent. Most requirements or guidelines for those inspections are based on engineers' qualitative judgment, and only implicitly take into account the probability of failure of a component under its operation and loading conditions, and the consequence of such failure, if it occurs. This document recommends appropriate methods for establishing a risk-based inspection program for any facility or structural system. The process involves four major steps: defining the system; performing a qualitative risk assessment; using this to do a quantitative risk analysis; and developing an inspection program for components and structural elements using probabilistic engineering methods. A companion document will detail specific risk-based techniques for the inspection of components of LWR nuclear power plants, applying methodology set out in Volume 1.

NUREG/IA-0040: BOIL-OFF EXPERIMENTS WITH THE EIR-NEPTUN FACILITY: ANALYSIS AND CODE ASSESSMENT OVERVIEW REPORT. AKSAN,S.N.; STIERLI,F.; ANALYTIS,G.T. Paul Scherrer Institute. March 1992. 50pp. 9204100063. EIR-BERICHT 629. 61288:255.

A series of experiments was performed in the NEPTUN test facility consisting of ten boil-off (core uncovery) and one adiabatic heat-up tests. In these tests rod power, system pressure, and initial coolant subcooling were varied. The repeatability of the experiments was also demonstrated. Some of the boil-off data obtained from the NEPTUN test facility are used for the assessment of the thermal-hydraulic transient computer codes. These calculations were performed extensively using the frozen version of TRAC-BD1/MOD1 (version 22). A limited number of assessment calculations were also done with RELAP5/MOD2 (version 36.02). In this report the main results and conclusions of these calculations are presented with the identification of problem areas in relation to the models relevant to boil-off phenomena.

NUREG/IA-0041: ASSESSMENT OF TRAC-PF1/MOD1 AGAINST AN INADVERTENT STEAM LINE ISOLATION VALVE CLOSURE IN THE RINGHALS 2 POWER PLANT. PELAYO,F. Consejo de Seguridad Nuclear (Spain). SJOBERG,A. Studsvik Energiteknik AB. March 1992. pp. 9204100075. ICSP-R2MSIV-T. 61288:197.

A TRAC-PF1/MOD1 simulation has been conducted to assess the capability of the code to predict a steam line isolation valve closure transient. Extensive use of results from Ringhals 2 data acquisition system was made to drive the initial conditions and some of the necessary boundary conditions. The results of the simulation revealed the importance of proper modeling of steam generator internals as well as the modeling of pressurizer walls and spray nozzles in order to reasonably predict the condensation phenomena.

NUREG/IA-0047: ASSESSMENT OF RELAP5/MOD2 CYCLE 36.04 AGAINST THE LOVIISA-2 STUCK-OPEN TURBINE BYPASS VALVE TRANSIENT ON SEPTEMBER 1, 1981. YRJOLA,V. Technical Research Centre of Finland (VTT). March 1992. 118pp. 9204060315. 61229:076.

RELAP5/MOD2 simulations have been conducted for an overcooling type transient that occurred at the LOVIISA Unit 2. The code assessment work in the report was based on the available plant data that were saved through the normal plant instrumentation into memory of the plant computer. The RELAP5 results matched well the main measured parameters, in particular, if the general trends were examined. The biggest quantitative differences were found between calculated and measured values of the primary pressure and pressurizer water level. The importance of the modelling of the pressurized vessel wall was demonstrated when condensation on the wall alone was able to stop and turn down the pressure increase.

NUREG/IA-0051: ASSESSMENT STUDY OF RELAP5/MOD2 CYCLE 36.05 BASED ON THE DOEL 4 REACTOR TRIP OF NOVEMBER 22, 1985. DE VLAMINCK,M.; DESCHUTTER,P.; VANHOENACKER,L. TRACTEBEL. March 1992. 63pp. 9204100097. 61286:133.

As part of the first cycle testing program for the Belgium DOEL 4 plant, a turbine trip on high steam generator level followed by a reactor trip was performed on November 22, 1985. Nine assessment runs were made using RELAP5/MOD2 Cycle 36.05 with the output compared to the data acquired during the test.

NUREG/IA-0052: AN ANALYSIS OF SEMISCALE MOD-2C S-FS-1 STEAM LINE BREAK TEST USING RELAP5/MOD2. ROGERS,J.M. United Kingdom Atomic Energy Authority. March 1992. 48pp. 9204060323. AEEW-R2476. 61232:001.

RELAP5/MOD2 is being used by CEGB to assist in an independent assessment of the sizewell B POSR. As part of the process of validation of the code for that assessment performance of the codo was verified by applying the results of Semiscale, MB-2 and LOBI. This report presents the results of the code assessment of RELAP5/MOD2 using semiscale S-FS-1 steam line break test.

NUREG/IA-0053: AN ASSESSMENT OF TRAC-PF1/MOD1 USING STRATCHCLYDE 1/10 SCALE MODEL REFILL TESTS: 2ND REPORT. DEMPSTER,W.M.; BRADFORD,A.A.; CALLANDER,T.M.; et al. Strathclyde, Univ. of, United Kingdom. March 1992. 69pp. 9204060332. 61226:062.

TRAC-PFI/MODI predictions of LOCA Refill Experiments carried out on a 1/10 scale model are compared against experimental measurements and video observations. Sensitivity studies have been carried out to determine the effect of Hydraulic Diameter and nodalisation. A simplified analysis of total penetration conditions reveals that the liquid head transfer coefficient during condensation is substantially greater than suggested by the reduction of the experimental measurements.

NUREG/IA-0055: AN ASSESSMENT OF TRAC-PF1/MOD1 USING STRATHCLYDE 1/10 SCALE MODEL REFILL TESTS. DEMPSTER,W.M.; BRADFORD,A.M.; CALLANDER,T.M.; et al. Strathclyde, Univ. of, United Kingdom. March 1992. 58pp. 9204060348. 61226:001.

TRAC-PFI/MODI predictions of LOCA Refill Experiments carried out on a 1/10 scale model PWR vessel are presented. The predictions show that TRAC underpredicts bypass for the test cases considered. Comparison results are presented and discussed. Simple sensitivity analysis of the interfacial drag models used is presented in an effort to explain the performance of the code.

NUREG/IA-0056: ASSESSMENT OF THE SUB-COOLED BOILING MODEL USED IN RELAP5/MOD2 (CYCLE 36.05, VERSION E03) AGAINST EXPERIMENTAL DATA. BRAIN,C.R. Central Electricity Generating Board. March 1992. 34pp. 9204060253. GD/PE-N/729. 61226:132.

In order to test the ability of RELAP5/MOD2 to describe sub-cooled nucleate boiling under conditions similar to those anticipated during intact circuit fault scenarios in pressurized water reactors the code has been assessed against results of high pressure sub-cooled boiling experiments reported in literature. It is concluded that RELAP5/MOD2 can be applied with reasonable confidence to the prediction of sub- cooled boiling void fraction for conditions expected during PWR intact circuit faults.

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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.

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NUREG/CR-4219 V08 N1: HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1990 - March 1991.

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NRC Originating Organization Index (Staff Reports)

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

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NRC 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.

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.

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.

50-602	Univ. of Texas Research Reactor	NUREG-1135 S01
50-390	Watts Bar Nuclear Plant, Unit 1, Tennessee Valley Authority	NUREG-0847 S08
50-391	Watts Bar Nuclear Plant, Unit 2, Tennessee Valley Authority	NUREG-0847 S08
50-29	Yankee-Rowe Nuclear Power Station, Yankee Atomic Electric Co.	NUREG/CR-5799

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

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(See instructions on the reverse)

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