NUREG-0304
Vol. 21, No. 4

# Regulatory and Technical Reports (Abstract Index Journal)

Annual Compilation for 1996

**U.S. Nuclear Regulatory Commission**

**Office of Information Resources Management**

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Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

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The following documents in the NUREG series are available for purchase from the Government Printing Office: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grantee reports, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852-2738, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018-3308.

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**Publications Branch**
**Office of Information Resources Management**
**U.S. Nuclear Regulatory Commission**
**Washington, DC 20555-0001**

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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:

Publications Branch
Office of Information Resources Management
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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555–0001

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. 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), (9) the microfiche address (for NRC internal use).

## Grant Report

NUREG/GR-0013: APPLICATIONS OF A NEW MAGNETIC MONITORING TECHNIQUE TO IN SITU EVALUATION OF FATIQUE DAMAGE IN FERROUS COMPONENTS. JILES, D.C.; BINER, S.B.; GOVINDARAJU, M.; et al. Iowa State Univ., Ames, IA. June 1994. 41 pp. 9407250286. 80328:195.

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), (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. Kraftweek Union. August 1986. 223 pp. 8608270424. 37659:138.

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

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

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

## Availability of NRC Publications

Copies of NRC staff and contractor reports may be purchased either from the Government Printing Office (GPO) or from the National Technical Information Service, Springfield, Virginia 22161. To purchase documents from the GPO, send a check or money order, payable to the Superintendent of Documents, to the following address:

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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 or the work being reported.

In addition to the NUREG and NUREG/CR codes, NUREG/CP is used for NRC-sponsored conference proceedings NUREG/GR is used for NRC grant reports, and NUREG/IA is used for international agreement reports.

All these report codes are controlled and assigned by the staff of the Publications Branch of the NRC Office of Information Resources Management.

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# 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 V20:** LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31, 1995.(Gray Book I) HARTFIELD,R.A. Office of Information Resources Management (Post 890205). June 1996. 342pp. 9606250180. 88702:001.

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 1995) 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 :actors 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 V19 N04:** LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,October-December 1995.(White Book) * Office of Nuclear Reactor Regulation (Post 941001). February 1996. 73pp. 9603190117. 87510:219.

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from October through December 1995.

**NUREG-0040 V20 N01:** LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,January-March 1996.(White Book) * Office of Nuclear Reactor Regulation (Post 941001). May 1996. 91pp. 9606250166. 88701:237.

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from January through March 1996.

**NUREG-0040 V20 N02:** LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT. Quarterly Report,April-June 1996.(White Book) * Office of Nuclear Reactor Regulation (Post 941001). August 1996. 224pp. 9609100272. 89604:009.

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from April through June 1996.

**NUREG-0090 V18 N02:** REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.April-June 1995. * Office for Analysis & Evaluation of Operational Data, Director. October 1995. 29pp. 9602120369. 87081:158.

Section 208 of the Energy Reorganization Act of 1974 identifies abnormal occurrence (AO) as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety requires a quarterly report of such occurrences to be made to Congress. This report provides a description of those incidents and events that have been determined to be AOs during the period of April 1 through June 30, 1995. This report addresses five AOs at NRC-licensed facilities. One involved a reactor coolant system blowdown at a pressurized water reactor (PWR) nuclear power plant, one involved a previously unidentified path for the potential release of radioactivity at a PWR nuclear power plant, two involved medical brachytherapy misadministrations, and one involved a medical therapeutic radiopharmaceutical misadmininstration. Four AOs submitted by the Agreement States are included. One involved a medical teletherapy misadministration, two involved medical brachytherapy misadministrations, and one involved the overexposure of personnel at a medical center. The report also contains an update of one AO previously reported by an NRC licensee and two AOs previously reported by the Agreement States. No "Other Events of Interest" items are being reported.

**NUREG-0090 V18 N03:** REPORT TO CONGRESS ON ABNORMAL OCCURRENCES.July-September 1995. * Office for Analysis & Evaluation of Operational Data, Director. February 1996. 23pp. 9602280295. 87272:268.

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence (AO) as an unscheduled incident or event that the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such occurrences to be made to Congress. This report provides a description of those incidents and events that have been determined to be AOs during the period of July 1 through September 30, 1995. This report addresses three AOs at NRC-licensed facilities. Two involved medical brachytherapy misadministrations and one involved ingestion of radioactive material by research workers. One AO submitted by the Agreement States is included. It involved importation into the United States of a package having excessive radiation. No updates of previously reported AOs are included in this report. No "Other Events of Interest" items are being reported.

**NUREG-0304 V20 N03:** REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Third Quarter 1995,July-September. * Division of Freedom of Information & Publications Services (Post 940714). January 1996. 48pp. 9602220246. 87210:306.

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

**NUREG-0304 V20 N04:** REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Annual Compilation For 1995 * Division of Freedom of Information & Publications Services (Post 940714). April 1996. 120pp. 9605220248. 88319:273.

See NUREG-0304,V20,N03 abstract.

**NUREG-0304 V21 N01:** REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For First Quarter 1996,January-March. * Division of Freedom of Information & Publications Services (Post 940714). June 1996. 43pp. 9607090048. 88955:305.

See NUREG-0304,V20,N03 abstract.

**NUREG-0304 V21 N02:** REGULATORY AND TECHNICAL REPORTS (ABSTRACT INDEX JOURNAL). Compilation For Second Quarter 1996,April-June. * Division of Freedom of Information & Publications Services (Post 940714). August 1996. 42pp. 9609100247. 89600:307.

See NUREG-0304,V20,N03 abstract.

**NUREG-0325 R19:** U.S. NUCLEAR REGULATORY COMMISSION ORGANIZATION CHARTS AND FUNCTIONAL STATEMENTS.January 31, 1996. * Ofc of Personnel (Post 870413). January 1996. 67pp. 9603190108. 87510:293.

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

**NUREG-0325 R20:** U.S. NUCLEAR REGULATORY COMMISSION ORGANIZATION CHARTS AND FUNCTIONAL STATEMENTS.July 1, 1996. * Ofc of Personnel (Post 870413). July 1996. 68pp. 9608050055. 89244:273.

See NUREG-0325,R19 abstract.

**NUREG-0325 R21:** U.S. NUCLEAR REGULATORY COMMISSION ORGANIZATION CHARTS AND FUNCTIONAL STATEMENTS.August 19, 1996. * Ofc of Personnel (Post 870413). August 1996. 68pp. 9609100243. 89609:235.

See NUREG-0325,R19 abstract.

**NUREG-0383 V01 R19:** DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Packages. * Office of Nuclear Material Safety & Safeguards. October 1996. 644pp. 9611180287. 90808:001.

The purpose of this directory is to make available a convenient source of information on packagings approved by the U.S. Nuclear Regulatory Commission. To assist in identifying packaging, an index by Model Number and corresponding Certificate of Compliance Number is included at the front of Volumes 1 and 2. An alphabetical listing by user name is included in the back of Volume 3 of approved Quality Assurance programs. The reports include a listing of all users of each package design and approved Quality Assurance programs prior to the publication date.

**NUREG-0383 V02 R19:** DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Certificates Of Compliance. * Office of Nuclear Material Safety & Safeguards. October 1996. 599pp. 9611180288. 90806:001.

See NUREG-0383,V01,R19 abstract.

**NUREG-0383 V03 R16:** DIRECTORY OF CERTIFICATES OF COMPLIANCE FOR RADIOACTIVE MATERIALS PACKAGES.Report Of NRC Approved Quality Assurance Programs For Radioactive Materials Packages. * Office of Nuclear Material Safety & Safeguards. October 1996. 180pp. 9611180289. 90819:001.

See NUREG-0383,V01,R19 abstract.

**NUREG-0386 D07:** UNITED STATES NUCLEAR REGULATORY COMMISSION STAFF PRACTICE AND PROCEDURE DIGEST.Commission, Appeal Board And Licensing Board Decisions.July 1972 - June 1995. * Office of the General Counsel (Post 860701). April 1996. 1056pp. 9605220382. 88311:001.

This 7th 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 June 30, 1995, interpreting the NRC's Rules.

**NUREG-0390 V10:** TOPICAL REPORT REVIEW STATUS.(Blue Book) * Office of Nuclear Reactor Regulation (Post 941001). March 1996. 36pp. 9604150318. 87865:283.

This report provides industry with procedures for submitting topical reports, guidance on how the U.S. Nuclear Regulatory Commission (NRC) processes and responds to topical report submittals, and an accounting, with review schedules, of all topical reports currently accepted for review by the NRC. This report is published annually.

**NUREG-0430 V15:** LICENSED FUEL FACILITY STATUS REPORT.Inventory Difference Data.July 1, 1994 - June 30, 1995.(Gray Book II). JOY,D.R. Office of Nuclear Material Safety & Safeguards. May 1996. 18pp. 9607090152. 88956:319.

NRC is committed to the periodic publication of licensed fuel facility 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 special nuclear material.

**NUREG-0525 V02 R04:** SAFEGUARDS SUMMARY EVENT LIST (SSEL).January 1, 1990 Through December 31, 1995. FADDEN,M.A.; YARDUMIAN,J. Operations Branch. July 1996. 109pp. 9608060289. 89262:238.

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, and Miscellaneous. Because of the public interest, the Miscellaneous category also includes events reported involving source material, byproduct material, and natural uranium, which are exempt from safeguards requirements. Information in the event descriptions was obtained from official NRC sources.

**NUREG-0540 V17 N11:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. November 1-30, 1995. * Division of Freedom of Information & Publications Services (Post 940714). January 1996. 286pp. 9602220190. 87212:001.

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 V17 N12:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. December 1-31, 1995. * Division of Freedom of Information & Publications Services (Post 940714). February 1996. 298pp. 9603260311. 87641:001.

See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N01:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. January 1-31, 1996. * Division of Freedom of Information & Publications Services (Post 940714). March 1996. 276pp. 9604230386. 87993:001.

See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N02:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.February 1-29, 1996. * Division of Freedom of Information & Publications Services (Post 940714). April 1996. 264pp. 9605220337. 88310:001.
See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N03:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.March 1-31, 1996. * Division of Freedom of Information & Publications Services (Post 940714). May 1996. 319pp. 9606070098. 88487:001.
See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N04:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.April 1-30, 1996. * Division of Freedom of Information & Publications Services (Post 940714). June 1996. 425pp. 9606250175. 88700:001.
See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N05:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. May 1-31, 1996. * Division of Freedom of Information & Publications Services (Post 940714). July 1996. 367pp. 9608050217. 89245:233.
See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N06:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.June 1-30, 1996. * Division of Freedom of Information & Publications Services (Post 940714). August 1996. 300pp. 9609030364. 89546:001.
See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N07:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.July 1-31, 1996. * Division of Freedom of Information & Publications Services (Post 940714). September 1996. 356pp. 9612040128. 91005:001.
See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N08:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.August 1-31, 1996. * Division of Freedom of Information & Publications Services (Post 940714). October 1996. 318pp. 9611180279. 90810:001.
See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N09:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.September 1-30, 1996. * Division of Freedom of Information & Publications Services (Post 940714). November 1996. 324pp. 9612110223. 91063:001.
See NUREG-0540,V17,N11 abstract.

**NUREG-0540 V18 N10:** TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE.October 1-31, 1996. * Division of Freedom of Information & Publications Services (Post 940714). December 1996. 350pp. 9701130175. 91398:001.
See NUREG-0540,V17,N11 abstract.

**NUREG-0654 R1 S2 DFC:** CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF NUCLEAR POWER PLANTS.Criteria For Emergency Planning In An Early Site Permit Application.Draft Report For Comment. KANTOR,F.; FOX,E.F. Office of Nuclear Reactor Regulation (Post 941001). WINGERT,V.L.; et al. Federal Emergency Management Agency. April 1996. 32pp. 9605220244. FEMA-REP-1. 88314:302.

The Nuclear Regulatory Commission (NRC) and the Federal Emergency Management Agency (FEMA) have added Supplement 2 NUREG-0654/FEMA-REP-1, Revision 1, to provide guidance for the development, review, and approval of radiological emergency information and plans submitted with an early site permit application under Subpart A of 10 CFR Part 52.

**NUREG-0654 R1 S3 DFC:** CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF NUCLEAR POWER PLANTS.Criteria For Protective Action Recommendations For Severe Accidents.Draft Report For.... CONGEL,F.; KANTOR,F.; MCKENNA,T.; et al. Office of Nuclear Reactor Regulation (Post 941001). July 1996. 19pp. 9608230202. FEMA-REP-1. 89452:321.

The Nuclear Regulatory Commission (NRC) and the Federal Emergency Management Agency (FEMA) have added Supplement 3 to NUREG-0654/FEMA-REP-1, Revision 1, which provides guidance for development of protective action recommendations for the public for severe reactor accidents involving actual or projected core damage with the potential for loss of containment. Studies of severe reactor accidents and their consequences since the issuance of NUREG-0654/FEMA-REP-1, Revision 1, have led the NRC staff to conclude that the preferred initial protective action for a severe (core damage) accident is to evacuate promptly rather than to shelter the population near the plant, barring any constraints to evacuation. The guidance in this document is intended to update and simplify the decisionmaking process for protective actions for severe reactor accidents given in Appendix 1 to NUREG-0654/FEMA-REP-1, Revision 1.

**NUREG-0700 R01 V01:** HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINES.Process And Guidelines.Final Report. * Division of Systems Technology (Post 941217). June 1996. 489pp. 9611260297. 90919:118.

NUREG-0700, Rev. 1, provides human factors engineering (HFE) guidance to the U.S. Nuclear Regulatory Commission staff for its: (1) review of the human system interface (HSI) design submittals prepared by licensees or applicants for a license or design certification of commercial nuclear power plants, and (2) performance of HSI reviews that could be undertaken as part of an inspection or other type of regulatory review involving HSI design or incidents involving human performance. The guidance consists of a review process and HFE guidelines. The document describes those aspects of the HSI design review process that are important to the identification and resolution of human engineering discrepancies that could adversely affect plant safety. Guidance is provided that could be used by the staff to review an applicants HSI design review process or to guide the development of an HSI design review plan, e.g., as part of an inspection activity. The document also provides detailed HFE guidelines for the assessment of HSI design implementations. NUREG-0700, Revision 1, consists of three stand-alone volumes. Volume 1, Human System Interface Design Review Guideline: Process and Guidelines, is the principal technical document and provides a detailed discussion of both the review procedures and HFE guidelines. Volume 2, Human System Interface Design Review Guideline: Reviewer's Checklist, provides the HFE guidelines in a checklist format. Volume 3, Human System Interface Design Review Guideline: Review Software and User's Guide, contains an interactive software application to support design reviews.

**NUREG-0700 R01 V02:** HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINES.Reviewer's Checklist.Final Report. * Division of Systems Technology (Post 941217). June 1996. 500pp. 9611260290. 90918:001.
See NUREG-0700,R01,V01 abstract.

**NUREG-0700 R01 V03:** HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINES.Review Software And User's Guide.Final Report. * Division of Systems Technology (Post 941217). June 1996. 42pp. 9611260324. 90920:305.
See NUREG-0700,R01,V01 abstract.

**NUREG-0713 V16:** OCCUPATIONAL RADIATION EXPOSURE AT COMMERICAL NUCLEAR POWER REACTORS AND OTHER FACILITIES,1994.Twenty-Seventh Annual Report. THOMAS,M.L. Division of Regulatory Applications (Post 941217). HAGEMEYER,D. Science Applications International Corp. (formerly Science Applications, Inc.). January 1996. 302pp. 9602220240. 87211:001.

This report summarizes the occupational radiation exposure information that has been reported to the NRC's Radiation Exposure Information Reporting System (REIRS) by nuclear power facilities and certain other categories of NRC licensees during the years 1969 through 1994. The bulk of the data presented in the report was obtained from annual radiation exposure reports submitted in accordance with the requirements of 10 CFR 20.220 and the technical specifications of nuclear power plants. The 1994 annual reports submitted by about 303 licensees indicated that approximately 152,028 individuals were monitored, 141,901 of whom were monitored by nuclear power facilities. They incurred an average individual dose of 0.1 rem (cSv) and an average measurable dose of about 0.31 rem (cSv). Analyses of transient worker data indicate that 18,178 individuals completed work assignments at two or more licensees during the monitoring year. The dose distributions are adjusted each year to account for the duplicate reporting of transient workers by multiple licensees. In 1994, the average measurable dose calculated from reported data was 0.28 cSv (rem). The corrected dose distribution resulted in an average measurable dose of 0.31 cSv (rem).

**NUREG-0725 R11:** PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL. * Office of Nuclear Material Safety & Safeguards. July 1996. 32pp. 9608210216. 89420:305.

This circular has been prepared to provide information on the shipment of irradiated reactor fuel (spent fuel) subject to regulation by the Nuclear Regulatory Commission (NRC), and to meet the requirements of Public Law 96-295. The report provides a brief description of NRC authority for certain aspects of transporting spent fuel. It provides descriptive statistics on spent fuel shipments regulated by the NRC from 1979 to 1995. It also lists detailed highway and railway segments used within each state from January 1, 1993, through December 31, 1995.

**NUREG-0750 V41:** NUCLEAR REGULATORY COMMISSION ISSUANCES.Opinions And Decisions Of The Nuclear Regulatory Commission With Selected Orders.January-June 1995. * Division of Freedom of Information & Publications Services (Post 940714). February 1996. 547pp. 9604020257. 87709:001.

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

**NUREG-0750 V42 I02:** INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.July-December 1995. * Division of Freedom of Information & Publications Services (Post 940714). March 1996. 40pp. 9604020329. 87710:255.

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

**NUREG-0750 V42 N05:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1995. Pages 181-219. * Division of Freedom of Information & Publications Services (Post 940714). January 1996. 45pp. 9602090022. 87066:244.
See NUREG-0750,V41 abstract.

**NUREG-0750 V42 N06:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1995. Pages 221-258. * Division of Freedom of Information & Publications Services (Post 940714). February 1996. 44pp. 9602280302. 87272:185.
See NUREG-0750,V41 abstract.

**NUREG-0750 V43 I01:** INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-March 1996. * Division of Freedom of Information & Publications Services (Post 940714). June 1996. 25pp. 9607090246. 88956:33..
See NUREG-0750,V42,I02 abstract.

**NUREG-0750 V43 I02:** INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-June 1996. * Division of Freedom of Information & Publications Services (Post 940714). September 1996. 49pp. 9612040124. 91007:012.
See NUREG-0750,V42,I02 abstract.

**NUREG-0750 V43 N01:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1996. Pages 1-11. * Division of Freedom of Information & Publications Services (Post 940714). March 1996. 17pp. 9603260308. 87624:289.
See NUREG-0750,V41 abstract.

**NUREG-0750 V43 N02:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR FEBRUARY 1996. Pages 13-49. * Division of Freedom of Information & Publications Services (Post 940714). April 1996. 44pp. 9604230336. 87976:198.
See NUREG-0750,V41 abstract.

**NUREG-0750 V43 N03:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1996. Pages 51-121. * Division of Freedom of Information & Publications Services (Post 940714). May 1996. 79pp. 9605220426. 88310:269.
See NUREG-0750,V41 abstract.

**NUREG-0750 V43 N04:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR APRIL 1996.Page 123-210. * Division of Freedom of Information & Publications Services (Post 940714). June 1996. 93pp. 9607120203. 89008:158.
See NUREG-0750,V41 abstract.

**NUREG-0750 V43 N05:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MAY 1996. * Division of Freedom of Information & Publications Services (Post 940714). July 1996. 30pp. 9608060271. 89258:145.
See NUREG-0750,V41 abstract.

**NUREG-0750 V43 N06:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JUNE 1996. * Division of Freedom of Information & Publications Services (Post 940714). August 1996. 129pp. 9609200257. 89729:124.
See NUREG-0750,V41 abstract.

**NUREG-0750 V44 N01:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JULY 1996.Pages 1-57. * Division of Freedom of Information & Publications Services (Post 940714). September 1996. 63pp. 9612040113. 91004:236.
See NUREG-0750,V41 abstract.

**NUREG-0750 V44 N02:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR AUGUST 1996. Pages 59-78. * Division of Freedom of Information & Publications Services (Post 940714). October 1996. 27pp. 9611190210. 90824:321.
See NUREG-0750,V41 abstract.

**NUREG-0750 V44 N03:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR SEPTEMBER 1996. Pages 79-106. * Division of Freedom of Information & Publications Services (Post 940714). November 1996. 34pp. 9612110149. 91064:323.
See NUREG-0750,V41 abstract.

**NUREG-0750 V44 N04:** NUCLEAR REGULATORY COMMISSION ISSUANCES FOR OCTOBER 1996. Pages 107-228. * Division of Freedom of Information & Publications Services (Post 940714). December 1996. 129pp. 9701150176. 91421:175.
See NUREG-0750,V41 abstract.

**NUREG-0800 DRFT FC:** STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS.LWR Edition.Draft Report For Comment. * Office of Nuclear Reactor Regulation (Post 941001). June 1996. 7,046pp. 9606260017. 88730:001.

The Standard Review Plan (SRP) provides guidance to staff reviewers in the Office of Nuclear Reactor Regulation in performing safety reviews of applications to construct or operate nuclear power plants. The principal purpose of the SRP is to assure the quality and uniformity of staff safety reviews. In 1991, the Standard Review Plan Update and Development Program (SRP-UDP) was established to update NUREG-0800 for use in reviewing future reactor design applications. An "Implementing Procedures Document (IPD)", NUREG-1447, was issued May 1992 to describe the SRP-UDP and establish the procedures for updating the SRP. The principal objectives of the SRP-UDP were to update the SRP to reflect the substantial changes in regulation and regulatory guidance that occurred since the 1981 revision of the SRP and to reflect the experience of the safety reviews conducted of design certification applications for evolutionary nuclear plants. This document provides the results of the update program.

**NUREG-0837 V15 N04:** NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. October-December 1995. STRUCKMEYER,R. Region 1 (Post 820201). March 1996. 326pp. 9603260299. 87623:001.

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 fourth quarter of 1995.

**NUREG-0837 V16 N01:** NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. January-March 1996. STRUCKMEYER,R. Region 1 (Post 820201). May 1996. 227pp. 9606070096. 88486:026.

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 first quarter of 1996.

**NUREG-0837 V16 N02:** NRC TLD DIRECT RADIATION MONITORING NETWORK.Progress Report. April-June 1996. STRUCKMEYER,R. Region 1 (Post 820201). August 1996. 248pp. 9609200268. 89727:134.

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 1996.

**NUREG-0847 S20:** SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF WATTS BAR NUCLEAR PLANT,UNITS 1 AND 2.Docket Nos. 50-390 And 50-391.(Tennessee Valley Authority) TAM,P.S. Office of Nuclear Reactor Regulation (Post 941001). February 1996. 36pp. 9602280321. 87272:231.

Supplement No. 20 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 with (1) additional information submitted by the applicant since Supplement No. 19 was issued, and (2) matters that the staff had under review when Supplement No. 19 was issued.

**NUREG-0910 R02 S02:** NRC COMPREHENSIVE RECORDS DISPOSITION SCHEDULE. * Information & Records Management Branch (Post 890827). February 1996. 195pp. 9605220230. 88332:076.

The approved records disposition schedules specify the appropriate duration of retention and the final disposition for records created or maintained by the NRC. NUREG-0910, Revision 2, Supplement 2 makes editorial and administrative changes to the National Archives and Records Administration's General Record Schedule (GRS) and forwards an entire updated set of the General Records Schedule including GRS Subject and Forms Indexes.

**NUREG-0933 S20:** A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R. Division of Engineering Technology (Post 941217). July 1996. 232pp. 9608050226. 89245:001.

The report presents the safety priority ranking 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 resolution 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-0933 S21:** A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT,R. Division of Engineering Technology (Post 941217). December 1996. 174pp. 9701130173. 91400:037.

See NUREG-0933,S20 abstract.

**NUREG-0936 V14 N02:** NRC REGULATORY AGENDA.Semiannual Report.July-December 1995. * Division of Freedom of Information & Publications Services (Post 940714). February 1996. 57pp. 9602280332. 87272:290.

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 semiannually.

**NUREG-0936 V15 N01:** NRC REGULATORY AGENDA.Semiannual Report.January-June 1996. * Division of Freedom of Information & Publications Services (Post 940714). August 1996. 58pp. 9609100239. 89601:267.

See NUREG-0936,V14,N02 abstract.

**NUREG-0940 V14N3&4P1:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED INDIVIDUAL ACTIONS.Semiannual Progress Report,July-December 1995. * Ofc of Enforcement (Post 870413). February 1996. 363pp. 9603050127. 87335:001.

This compilation summarizes significant enforcement actions that have been resolved during the period (July - December 1995) and includes copies of Orders and Notices of Violation sent by the Nuclear Regulatory Commission to individuals 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. The Commission believes this information may be useful to licensees in making employment decisions.

**NUREG-0940 V14N3&4P2:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES.Semiannual Progress Report,July-December 1995. * Ofc of Enforcement (Post 870413). February 1996. 241pp. 9603040093. 87314:095.

This compilation summarizes significant enforcement actions that have been resolved during the period (July - December 1995) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to reactor 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-0940 V14N3&4P3:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES.Semiannual Progress Report,July-December 1995. * Ofc of Enforcement (Post 870413). February 1996. 250pp. 9603260287. 87620:001.

This compilation summarizes significant enforcement actions that have been resolved during the period (July - December 1995) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to material 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-0940 V15 N1 P1:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED INDIVIDUAL ACTIONS.Semiannual Progress Report,January-June 1996. * Ofc of Enforcement (Post 870413). August 1996. 375pp. 9609100232. 89603:001.

This compilation summarizes significant enforcement actions that have been resolved during the period (January - June 1996) and includes copies of Orders and Notices of Violation sent by the Nuclear Regulatory Commission to individuals 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. The Commission believes this information may be useful to licensees in making employment decisions.

**NUREG-0940 V15 N1 P2:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED REACTOR LICENSEES.Semiannual Progress Report,January-June 1996. * Ofc of Enforcement (Post 870413). August 1996. 266pp. 9609100251. 89601:001.

This compilation summarizes significant enforcement actions that have been resolved during the period (January - June 1996) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to reactor 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-0940 V15 N1 P3:** ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED MATERIAL LICENSEES.Semiannual Progress Report,January-June 1996. * Ofc of Enforcement (Post 870413). August 1996. 350pp. 9609100230. 89602:001.

This compilation summarizes significant enforcement actions that have been resolved during the period (January - June 1996) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to material 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 V12:** BUDGET ESTIMATES.Fiscal Year 1997. * Division of Budget & Analysis (Post 890205). March 1996. 168pp. 9604020325. 87714:001.

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 1997.

**NUREG-1125 V17:** A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.1995 Annual. * ACRS - Advisory Committee on Reactor Safeguards. April 1996. 118pp. 9605130141. 88226:142.

This compilation contains 44 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1995. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports by project name and by chronological order within project name. Part 2 categorizes the reports by the most appropriate generic subject area and by chronological order within subject area.

**NUREG-1145 V12:** U.S. NUCLEAR REGULATORY COMMISSION 1995 ANNUAL REPORT. * Office of Administration, Director (Post 940714). August 1996. 337pp. 9611210208. 90885:001.

This report covers the major activities, events, decisions, and planning that took place during Fiscal Year 1995 within the U.S. Nuclear Regulatory Commission (NRC) or involving the NRC.

**NUREG-1214 R14:** HISTORICAL DATA SUMMARY OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE. GAMBERONI,D. Office of Nuclear Reactor Regulation (Post 941001). October 1996. 50pp. 9612030264. 90980:109.

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 of 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. Section 2 presents a chronological listing of all SALP report ratings for each operating facility since August 1, 1988.

**NUREG-1272 V09 N01:** OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1994-FY 95 Annual Report - Reactors. * Office for Analysis & Evaluation of Operational Data, Director. July 1996. 306pp. 9609100257. 89600:001.

This annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) describes activities conducted during CY 1994 and FY 1995. The report is published in three parts. NUREG-1272, Vol. 9, 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. NUREG-1272, Vol. 9, No. 2, covers nuclear materials and presents a review of the events and concerns during 1993 associated with the use of licensed material in non-reactor applications, such as personnel overexposures and medical misadministrations. Both reports also contain a discussion of the Incident Investigation Team program and summarize both the Incident Investigation Team and Augmented Inspection Team reports. Each volume contains a list of the AEOD reports issued from 1980 through 1995. NUREG-1272, Vol. 9, No. 3, covers technical training and presents the activities of the Technical Training Center in support of the NRC's mission.

**NUREG-1272 V09 N02:** OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1994-FY 95 Annual Report - Nuclear Materials. * Office for Analysis & Evaluation of Operational Data, Director. September 1996. 184pp. 9612120065. 91072:144.

See NUREG-1272,V09,N01 abstract.

**NUREG-1272 V09 N03:** OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA. 1994-FY Annual Report - Technical Training. * Office for Analysis & Evaluation of Operational Data, Director. September 1996. 43pp. 9612030287. 90980:220.

See NUREG-1272,V09,N01 abstract.

**NUREG-1307 R06:** REPORT ON WASTE BURIAL CHARGES.Escalation Of Decommissioning Waste Disposal Costs At Low-Level Waste Burial Facilities. * Division of Regulatory Applications (Post 941217). September 1996. 66pp. 9609200324. 89726:236.

One of the requirements placed upon nuclear power reactor licensees by the U.S. Nuclear Regulatory Commission (NRC) is for the licensees to periodically adjust the estimate of the cost of decommissioning their plants, in dollars of the current year, as part of the process to provide reasonable assurance that adequate funds for decommissioning will be available when needed. This report, which is scheduled to be revised periodically, contains the development of a formula for escalating decommissioning cost estimates that is acceptable to the NRC, and contains values for the escalation of radioactive waste burial costs, by site and by year. The licensees may use the formula, the coefficients, and the burial escalation from this report in their escalation analyses, or they may use an escalation rate at least equal to the escalation approach presented herein.

**NUREG-1350 V08**: NUCLEAR REGULATORY COMMISSION INFORMATION DIGEST.1996 Edition. GARVER,M. Division of Budget & Analysis (Post 890205). July 1996. 138pp. 9609030260. 89545:001.

The Nuclear Regulatory Commission Information Digest (digest) provides a summary of information about the U.S. Nuclear Regulatory Commission (NRC), NRC's regulatory responsibilities, NRC licensed activities, and general information on domestic and worldwide nuclear energy. The digest, published annually, is a compilation of nuclear- and NRC-related data and is designed to provide a quick reference to major facts about the agency and the industry it regulates. In general, the data cover 1975 through 1995, with exceptions noted. Information on generating capacity and average capacity factor for operating U.S. commercial nuclear power reactors is obtained from monthly operating reports that are submitted directly to the NRC by the licensee. This information is reviewed by the NRC for consistency only and no independent validation and/or verification is performed.

**NUREG-1415 V08 N02**: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress,October 1, 1995 - March 31, 1996. BARCHI,T.; WATKINS,B.; GRODIN,M.; et al. Office of the Inspector General (Post 890417). June 1996. 43pp. 9606180278. 88634:231.

The Inspector General Act of 1978, as amended, requires that Inspectors General submit a "Semiannual Report to Congress" summarizing program activities. The Inspector General's report is submitted to the Chairman of the NRC not later than April 30, and October 31 for each reporting period. The Chairman comments on the report and prepares the NRC's Semiannual Report to Congress as required by the Act. The Chairman then submits the agency's report and the OIG's report no later than November 30 and May 31, respectively.

**NUREG-1415 V09 N01**: OFFICE OF THE INSPECTOR GENERAL.Semiannual Report To Congress,April 1, 1996 - September 30, 1996. BARCHI,T.; WATKINS,B.; GRODIN,M.; et al. Office of the Inspector General (Post 890417). November 1996. 39pp. 9612120045. 91074:212.

See NUREG-1415,V08,N02 abstract.

**NUREG-1423 V06**: A COMPILATION OF REPORTS OF THE ADVISORY COMMITTEE ON NUCLEAR WASTE.July 1995 - June 1996. * Advisory Committee on Nuclear Waste. August 1996. 54pp. 9609030264. 89547:283.

This compilation contains 8 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the eighth year of its operation. The reports were submitted to the Chairman and Commissioners of the U.S. Nuclear Regulatory Commission. All reports prepared by the Committee have been made available to the public through the NRC Public Document Room, the U.S. Library of Congress, and the internet at http://www.nrc.gov/ACRSACNW.

**NUREG-1437 V01**: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Main Report. * Division of Regulatory Applications (Post 941217). May 1996. 638pp. 9606180460. 88629:001.

The generic environmental impact statement (geis) examines the possible environmental impacts that could occur as a result of renewing licenses of individual nuclear power plants under 10 cfr part 54. The geis, to the extent possible, establishes the bounds and significance of these potential impacts. The analyses in the geis encompass all operating light-water power reactors. For each type of environmental impact the geis attempts to establish generic findings covering as many plants as possible. This geis has three principal objectives: (1) to provide an understanding of the types and severity of environmental impacts that may occur as a result of license renewal of nuclear power plants under 10 cfr part 54, (2) to identify and assess those impacts that are expected to be generic to license renewal, and (3) to support a rulemaking (10 cfr part 51) to define the number and scope of issues that need to be addressed by the applicants in plant-by-plant license renewal proceedings. To accomplish these objectives, the geis makes maximum use of environmental and safety documentation from original licensing proceedings and information from state and federal regulatory agencies, the nuclear utility industry, the open literature, and professional contacts.

**NUREG-1437 V02**: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Appendices. * Division of Regulatory Applications (Post 941217). May 1996. 553pp. 9606180469. 88633:001.

See NUREG-1437,V01 abstract.

**NUREG-1440**: REGULATORY ANALYSIS FOR AMENDMENTS TO REGULATIONS FOR THE ENVIRONMENTAL REVIEW FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES.Final Report. * Division of Regulatory Applications (Post 941217). May 1996. 35pp. 9606180288. 88630:327.

This regulatory analysis provides the supporting information for a rule that amends the nuclear regulatory commission's requirements for environmental review of applications for renewal of nuclear power plant operating licenses. After considering various options, the staff identified and analyzed two major alternatives. Alternative a is to not amend the regulations and to perform environmental reviews under the existing regulations. Alternative b is to assess, on a generic basis, the environmental impacts of renewing the operating license of individual nuclear power plants, and define the issues that will need to be further analyzed on a case-by-case basis. The findings of this assessment are codified in 10 cfr part 51. The staff has selected alternative b as the preferred alternative.

**NUREG-1509**: RADIATION EFFECTS ON REACTOR PRESSURE VESSEL SUPPORTS. JOHNSON,R.E. Division of Engineering Technology (Post 941217). LIPINSKI,R.E. Idaho National Engineering Laboratory. May 1996. 200pp. 9605220250. 88321:001.

The NRC Generic Safety Issue No. 15, (GSI-15), "Radiation Effects on Reactor Pressure Vessel Supports," was established to evaluate the concern that low-temperature, low-flux-level neutron irradiation might embrittle reactor pressure vessel supports to a significant degree and compromise plant safety. Evaluation of the surveillance samples from the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory (ORNL) led to the conclusion that the embrittlement rates of some materials used for pressurized water reactor pressure vessel (RPV) supports could be higher than expected. This disclosure raised a concern that a brittle fracture of the RPV supports could occur during the anticipated life-span of the plant. A later study by the ORNL demonstrated that gamma radiation contributed a significant amount of the embrittlement in the HFIR surveillance specimens. However, the shielding provided by the thick steel shell of the RPV ensures that degradation of RPV supports from gamma irradiation is improbable or minimal. This report (1) describes the technical findings resulting from the work done in accord with the GSI-15 Task Action Plan and (2) was used, in part, as the basis for technical resolution of the issue.

**NUREG-1511 S01**: REACTOR PRESSURE VESSEL STATUS REPORT. ELLIOT,B.J.; HACKETT,E.M.; LEE,A.D.; et al. Office of Nuclear Reactor Regulation (Post 941001). October 1996. 42pp. 9611180284. 90820:259.

This report describes the issues raised as a result of the staff's review of Generic Letter (GL) 92-01, Revision 1, responses and plant-specific reactor pressure vessel (RPV) assessments and the actions taken or work in progress to address these issues. In addition, the report describes actions taken by the staff and the nuclear industry to develop a thermal annealing process for possible use at U.S. commercial nuclear plants to mitigate the effects of neutron radiation on the fracture toughness of RPV materials. The Nuclear Regulatory Commission (NRC) issued GL 92-01, Revision 1, Supplement 1, to obtain information needed to assess compliance with regulatory requirements and licensee commitments regarding RPV integrity. GL 92-01, Revision 1, Supplement 1, was issued as a result of generic issues raised in the NRC staff's review of licensee responses to GL 92-01, Revision 1, and plant-specific RPV evaluations. In particular, an integrated review of all data submitted in response to GL 92-01, Revision 1, indicated that licensees may not have considered all relevant data in their RPV assessments.

**NUREG-1518:** DIFFERING PROFESSIONAL VIEWS OR OPINIONS.1994 Special Review Panel. * Ofc of Personnel (Post 870413). September 1996. 79pp. 9612040132. 91007:134.

In July 1994, the Executive Director for Operations of the U.S. Nuclear Regulatory Commission (NRC) appointed a Special Review Panel to assess the Differing Professional View or Opinion (DPV/DPO)process, including "...its effectiveness, how well it is understood by employees, and the organizational climate for having such views aired and properly decided." An additional area within this review was to address "...the effectiveness of the DPO procedures as they pertain to public access and confidentiality." Further, the Panel was charged with the review of the submittals completed since the last review to identify employees who made significant contributions to the agency or to public health and safety but had not been adequately recognized for this contribution. The report presents the Special Review Panel's evaluation of the NRC's current process for dealing with Differing Professional Views or Opinions. Provided in this report are the results of an employee opinion survey on the process; highlights and suggestions from interviews with individuals who had submitted a Differing Professional View or Opinion, as well as with agency managers directly involved with the Differing Professional Views or Opinions process; and the Special Review Panel's recommendations for improving the DPV/DPO process.

**NUREG-1524:** A REASSESSMENT OF THE POTENTIAL FOR AN ALPHA-MODE CONTAINMENT FAILURE AND A REVIEW OF THE CURRENT UNDERSTANDING OF BROADER FUEL-COOLANT INTERACTION ISSUES. Report Of The Second Steam Explosion Review Group Workshop. BASU,S. Division of Systems Technology (Post 941217). GINSBERG,T. Brookhaven National Laboratory. August 1996. 200pp. 9608230217. 89450:001.

This report summarizes the review and evaluation by experts of the current understanding of the molten fuel-coolant interaction (FCI) issues covering the complete spectrum of interactions, i.e., from mild quenching to very energetic interactions including those that could lead to the alpha-mode containment failure. The experts' review and evaluation took place in the form of a Second Steam Explosion Review Group (SERG-2) Workshop, held in Annapolis, Maryland, on June 15 and 16, 1995. The first such workshop (SERG-1) took place in 1985. Extensive discussions took place at the SERG-2 workshop on the alpha-mode failure issue, based on the experts' responses to the questions raised, and consensus opinions on the status of resolution of the issue emerged from the discussions. Of the eleven experts polled, all but two concluded that the alpha-mode failure issue was resolved from a risk perspective, meaning that this mode of failure is of very low probability, that it is of little or no significance to the overall risk from a nuclear power plant, and that any further reduction in residual uncertainties is not likely to change the probability in an appreciable manner. To a lesser degree, discussions also took place on the broader FCI issues such as mild quenching of core melt during non-explosive FCI, and shock loading of lower head and ex-vessel support structures arising from explosive localized FCIs. These latter issues are relevant with regard to determining the efficacy of certain accident management strategies for operating reactors as well as for advanced light water reactors. The experts reviewed the status of understanding of the FCI phenomena in the context of these broader issues, identified residual uncertainties in the understanding, and recommended further research (both experimental and analytical) to reduce the uncertainties.

**NUREG-1529 V01:** PUBLIC COMMENTS ON THE PROPOSED 10 CFR PART 51 RULE FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES AND SUPPORTING DOCUMENTS: REVIEW OF CONCERNS AND NRC STAFF RESPONSE.Executive Summary. * Division of Regulatory Applications (Post 941217). May 1996. 40pp. 9606180325. 88634:190.

This report documents the nuclear regulatory commission (nrc) staff review of public comments provided in response to the nrc's proposed amendments to 10 code of federal regulations (cfr) part 51, which establish new requirements for the environmental review of applications for the renewal of operating licenses of nuclear power plants. The public comments include those submitted in writing, as well as those provided at public meetings that were held with other federal agencies, state agencies, nuclear industry representatives, public interest groups, and the general public. This report also contains the nrc staff response to the various concerns raised, and highlights the changes made to the final rule and the supporting documents in response to these concerns.

**NUREG-1529 V02:** PUBLIC COMMENTS ON THE PROPOSED 10 CFR PART 51 RULE FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES AND SUPPORTING DOCUMENTS: REVIEW OF CONCERNS AND NRC STAFF RESPONSE.Appendices. * Division of Regulatory Applications (Post 941217). May 1996. 671pp. 9606180333. 88627:001.

See NUREG-1529,V01 abstract.

**NUREG-1531:** DRAFT ENVIRONMENTAL IMPACT STATEMENT RELATED TO RECLAMATION OF THE URANIUM MILL TAILINGS AT THE ATLAS SITE,MOAB,UTAH. Source Material License No. SUA-917,Docket No. 40-3453. (Atlas Corporation) BLASING,T.S.; EASTERLY,C.E. Oak Ridge National Laboratory. FLIEGEL,M.; et al. Division of Waste Management (NMSS 940403). January 1996. 300pp. 9602050062. 86981:001.

This Draft Environmental Impact Statement (DEIS) has been prepared by the Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards, to address potential environmental impacts associated with a request by Atlas Corporation to amend its existing NRC License No. SUA-917 to reclaim an existing uranium mill tailings pile near Moab, Utah. The proposed reclamation would allow Atlas to (1) reclaim the tailings pile for permanent disposal and long-term custodial care by a government agency in its current location on the Moab site, (2) prepare the 162-ha (400 acre) Moab site for site closure, and (3) relinquish responsibility of the site after having its NRC license terminated. The DEIS describes and evaluates (1) the purpose of and need for the proposed action, (2) alternatives considered, (3) potentially affected environmental resources, (4) environmental consequences of the proposed action, and (5) costs and benefits associated with reclamation alternatives. Public and agency comments on this DEIS will be considered in the Final Environmental Impact Statement.

**NUREG-1532:** DRAFT TECHNICAL EVALUATION REPORT FOR THE PROPOSED REVISED RECLAMATION PLAN FOR THE ATLAS CORPORATION MOAB MILL.Source Material License No. SUA-917,Docket No. 40-3453.(Atlas Corporation) BRUMMETT,E.; FLIEGEL,M.; IBRAHIM,A.; et al. Division of Waste Management (NMSS 940403). January 1996. 128pp. 9602070045. 87015:139.

This Draft Technical Evaluation Report (DTER) summarizes the U.S. Nuclear Regulatory Commission staff's review of Atlas

Corporation's proposed reclamation plan for its uranium mill tailings pile near Moab, Utah. The proposed reclamation would allow Atlas to (1) reclaim the tailings pile for permanent disposal and long-term custodial care by a government agency in its current location on the Moab site, (2) prepare the site for closure, and (3) relinquish responsibility of the site after having its NRC license terminated. The NRC staff review has identified open issues in geology, seismology, geotechnical engineering, erosion protection, water resources protection and radon attenuation. The NRC will not approve the proposed reclamation plan until Atlas adequately resolves these open issues.

**NUREG-1536 DRFT FC:** STANDARD REVIEW PLAN FOR DRY SPENT FUEL STORAGE SYSTEMS. Draft Report For Comment. * Office of Nuclear Material Safety & Safeguards. February 1996. 172pp. 9603190122. 87516:147.

The Standard Review Plan (SRP) for Dry Cask Storage Systems provides guidance to the Nuclear Regulatory Commission staff in the Spent Fuel Project Office for performing safety reviews of dry cask storage systems. The SRP is intended to ensure the quality and uniformity of the staff reviews and present a basis for the review scope and requirements. Part 72, Supart B generally specifies the information needed in a license application for the independent storage of spent nuclear fuel and high level radioactive waste. Regulatory Guide 3.61 "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask" contains an outline of the specific information required by the staff. The SRP is divided into 14 sections which reflect the standard application format. Regulatory requirements, staff position, industry codes and standards, acceptance criteria, and other information are discussed. Comments on this draft, will be considered and incorporated into the SRP as appropriate. The SRP is scheduled for publication as an NRC NUREG document late in 1996. Comments, errors or omissions, and suggestions for improvement should be sent to the Director, Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

**NUREG-1537 PT01:** GUIDELINES FOR PREPARING AND REVIEWING APPLICATIONS FOR THE LICENSING OF NON-POWER REACTORS.Part 1: Format And Content. * Office of Nuclear Reactor Regulation (Post 941001). February 1996. 523pp. 9604020335. 87711:001.

NUREG-1537, Part 1 gives guidance to non-power reactor licensees and applicants on the format and content of applications to the Nuclear Regulatory Commission for licensing actions. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination.

**NUREG-1537 PT02:** GUIDELINES FOR PREPARING AND REVIEWING APPLICATIONS FOR THE LICENSING OF NON-POWER REACTORS.Part 2: Standard Review Plan And Acceptance Criteria. * Office of Nuclear Reactor Regulation (Post 941001). February 1996. 300pp. 9604020339. 87713:001.

NUREG-1537, Part 2 gives guidance on the conduct of licensing action reviews to NRC staff who review non-power reactor licensing applications. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination.

**NUREG-1539:** METHODOLOGY AND FINDINGS OF THE NRC'S MATERIALS LICENSING PROCESS REDESIGN. RATHBUN,P.A.; BROWN,K.D.; MADERA,J.R.; et al. Division of Industrial & Medical Nuclear Safety (Post 870729). April 1996. 82pp. 9604160379. 87884:179.

This report describes the work and vision of the team chartered to redesign the process for licensing users of nuclear materials. The Business Process Redesign team was chartered to improve the speed of the existing licensing process while maintaining or improving public safety and to achieve required resource levels. The report describes the team's methods for acquiring and analyzing information about the existing materials licensing process and the steps necessary to radically change this process to the envisioned future process.

**NUREG-1540:** BWR STEEL CONTAINMENT CORROSION. TAN,C.P.; BAGCHI,G. Office of Nuclear Reactor Regulation (Post 941001). April 1996. 217pp. 9604230397. 87994:001.

The report describes regulatory actions taken after corrosion was discovered in the drywell at the Oyster Creek Plant and in the torus at the Nine Mile Point 1 Plant. The report describes the causes of corrosion, requirements for monitoring corrosion, and measures to mitigate the corrosive environment for the two plants. The report describes the issuances of generic letters and information notices either to collect information to determine whether the problem is generic or to alert the licensees of similar plants about the existence of such a problem. Implementation of measures to enhance the containment performance under severe accident conditions is discussed. A study by Brookhaven National Laboratory (BNL) of the performance of a degraded containment under severe accident conditions is summarized. The details of the BNL study are in the appendix to the report.

**NUREG-1541 DRFT FC:** PROCESS AND DESIGN FOR CONSOLIDATING AND UPDATING MATERIALS LICENSING GUIDANCE.Draft Report For Comment. WHITTEN,J.E.; VACCA,P.C.; BROWN,K.D.; et al. Division of Industrial & Medical Nuclear Safety (Post 870729). April 1996. 39pp. 9605130135. 88220:239.

This report describes the concept and approach for developing the Materials Electronic Library (MEL). The Business Process Redesign team for the licensing of materials conceived, as an integral part of its vision for the redesign of this licensing process, the idea for MEL. To establish MEL, the NRC will consolidate and update numerous regulations and policy and guidance documents supporting the materials licensing process into a single, comprehensive electronic repository for use by the NRC, Agreement and non-Agreement States, licensees, applicants, and the public.

**NUREG-1542 V01:** ACCOUNTABILITY REPORT FISCAL YEAR 1995. CONNELLY,S.R. Office of the Controller (Post 890205). April 1996. 85pp. 9607120159. 88982:220.

The U.S. Nuclear Regulatory Commission (NRC) is one of six Federal agencies participating in a pilot project to streamline financial management reporting. The goal of this pilot is to consolidate performance-related reporting into a single accountability report in accordance with the Government Management Reform Act (GMRA) of 1994. The NRC's first accountability report consolidates the information previously reported in the NRC's annual financial statement required by the Chief Financial Officers Act of 1990, as amended; the Chairman's annual report to the President and the Congress, required by the Federal Managers' Financial Integrity Act of 1982; and the Chairman's semiannual report to the Congress on management decisions and final actions on Office of Inspector General (OIG) audit recommendations, required by the Inspector General Act of 1978, as amended. This report also includes performance measures, as required by the Chief Financial Officers Act of 1990.

**NUREG-1543:** DRAFT ENVIRONMENTAL IMPACT STATEMENT DECOMMISSIONING OF THE SHIELDALLOY METALLURGICAL CORPORATION,CAMBRIDGE,OHIO, FACILITY.Docket No. 40-8948,License No. SMB-1507. WADE,M.C.; BLASING,T.J.; CURTIS,A.H.; et al. Oak Ridge National Laboratory. July 1996. 249pp. 9608060185. 89260:063.

Shieldalloy Metallurgical Corporation holds a license from the U.S. Nuclear Regulatory Commission (NRC) for the possession of source material at its Cambridge, Ohio, facility. The source material is in the form of slag and is located in two piles that contain a total of 546,000 metric tons (606,000 tons) of material. The piles also contain chemical contaminants that may require remediation. Shieldalloy proposed to stabilize, cap, and grade the slag piles as part of decommissioning the site and terminating the NRC license. The DEIS evaluates radiological and nonradiological impacts associated with the proposed action and five alternative actions, including no action. Impacts are assessed for land use, socioeconomics and cultural resources, geology, air quality, water quality and wetlands, human health, and biological resources. The staff concludes that the environmental impacts of the on-site and the off-site disposal alternatives are not significant if mitigation as described is carried out and that there is no obviously superior alternative. A cost benefit analysis shows that the proposed action is less costly than all other alternatives except no action. The no-action alternative has no economic benefits. The on-site disposal alternatives have identical economic benefits, and the off-site disposal alternative has the greatest associated economic benefits to local residents.

**NUREG-1544:** STATUS REPORT: INTERGRANULAR STRESS CORROSION CRACKING OF BWR CORE SHROUDS AND OTHER INTERNAL COMPONENTS. * Office of Nuclear Reactor Regulation (Post 941001). March 1996. 115pp. 9604150308. 87877:165.

On July 25, 1994, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 94-03 to obtain information needed to assess compliance with regulatory requirements regarding the structural integrity of core shrouds in domestic boiling water reactors (BWRs). This report begins with a brief description of the safety significance of intergranular stress corrosion cracking (IGSCC) as it relates to the design and function of BWR core shrouds and other internal components. It then presents a brief history of shroud cracking events both in the U.S. and abroad, followed by an indepth summary of the industry actions to address the issue of IGSCC in BWR core shrouds and other internal components. This report summarizes the staff's basis for issuing GL 94-03, as well as the staff's assessment of plant-specific responses to GL 94-03. The staff is continually evaluating the licensee inspection programs and the results from examinations of BWR core shrouds and other internal components. This report is representative of submittals to and evaluations by the staff as of September 30, 1995. An update of this report will be issued at a later date.

**NUREG-1547:** METHODOLOGY FOR DEVELOPING AND IMPLEMENTING ALTERNATIVE TEMPERATURE-TIME CURVE FOR TESTING THE FIRE RESISTANCE OF BARRIERS FOR NUCLEAR POWER PLANT APPLICATIONS. COOPER,L.Y.; STECKLER,K.D. National Institute of Standards & Technology (formerly National Bureau of Standa. * Office of Nuclear Reactor Regulation (Post 941001). August 1996. 125pp. 9609200310. 89729:001.

Advances in fire science over the past 40 years have offered the potential for developing technically sound alternative temperature-time curves for use in evaluating fire barriers for areas where fire exposures can be expected to be significantly different than the ASTM E-119, standard temperature-time exposure. This report summarizes the development of the ASTM E-119, standard temperature-time curve, and the efforts by the federal government and the petrochemical industry to develop alternative fire endurance curves for specific applications. The report also provides a framework for the development of alternative curves for application at nuclear power plants. The staff has concluded that in view of the effort necessary for the development of nuclear power plant specific temperature-time curves, such curves are not a viable approach for resolving the issues concerning Thermo-Lag fire barriers. However, the approach may be useful to licensees in the development of performance-based fire protection methods in the future.

**NUREG-1550:** STANDARD REVIEW PLAN FOR APPLICATIONS FOR SEALED SOURCE AND DEVICE EVALUATIONS AND REGISTRATIONS. * Division of Industrial & Medical Nuclear Safety (Post 870729). November 1996. 96pp. 9612110179. 91064:185.

The purpose of this document is to provide the reviewer of a request for a sealed source or device safety evaluation with the information and materials necessary to make a determination that the product is acceptable for licensing purposes. It provides the reviewer with a listing of the applicable regulations and industry standards, policies affecting evaluation and registration, certain administrative procedures to be followed, and information on how to perform the evaluation and write the registration certificate. Standard review plans are prepared for the guidance of the Office of Nuclear Material Safety and Safeguards staff responsible for the review of a sealed source or device application. This document is made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Washington, DC 20555-0001.

**NUREG-1551:** FINAL REPORT OF THE NRC-AGREEMENT STATE WORKING GROUP TO EVALUATE CONTROL AND ACCOUNTABILITY OF LICENSED DEVICES. * Division of Industrial & Medical Nuclear Safety (Post 870729). October 1996. 78pp. 9611260270. 90923:132.

The U.S. Nuclear Regulatory Commission staff acknowledged that licensees were having problems maintaining control over and accountability for devices containing radioactive material. In June 1995, the Commission approved the staff's suggestion to form a joint NRC-Agreement State Working Group (WG) to evaluate the problem and propose solutions. The staff indicated that the formation of the WG was necessary to address the concerns from a national perspective, allow for a broad level of Agreement State (AS) input, and to reflect their experience. AS participation in the process is essential since some AS already have implemented effective programs for oversight of device users. This report includes the five recommendations proposed by the WG to increase regulatory oversight, increase control and accountability of devices, ensure proper disposal, and ensure disposal of orphaned devices. Specifically, the WG recommends that: 1) NRC and AS increase regulatory oversight for users of certain devices; 2) NRC and AS impose penalties on persons losing devices; 3) NRC and AS ensure proper disposal of orphaned devices; 4) NRC encourage States to implement similar oversight programs for users of Naturally-Occurring or Accelerator-Produced Material (NARM); and, 5) NRC encourage non-licensed stakeholders to take appropriate actions, such as instituting programs for material identification.

**NUREG-1552:** FIRE BARRIER PENETRATION SEALS IN NUCLEAR POWER PLANTS. BAJWA,C.S.; WEST,K.S. Office of Nuclear Reactor Regulation (Post 941001). July 1996. 55pp. 9608230207. 89455:045.

Nuclear power plants are divided into separate fire areas by fire-rated structural barriers. Fire-rated penetration seals are installed to seal certain openings in these barriers. The seals maintain the fire-resistive integrity of the barriers and provide reasonable assurance that a fire will be confined to the area in which it started. The U.S. Nuclear Regulatory Commission conducted a comprehensive technical assessment of penetration seals to address reports of potential problems, to determine if there were any problems of safety significance, and to determine if NRC requirements, review guidance, and inspection procedures are adequate. The staff did not find plant-specific problems of safety significance or concerns with generic implications. The staff concluded that the general condition of penetration seal programs in industry is satisfactory. The staff also concluded that actions it had taken in 1988 and 1994 to address potential penetration seal problems increased industry awareness of such problems and resulted in more thorough surveillances, maintenance, and corrective actions. These previous staff actions, together with continued licensee upkeep of existing penetration seal programs and continued NRC inspections, are adequate to maintain public health and safety.

**NUREG-1556 V01 DR FC:** CONSOLIDATED GUIDANCE ABOUT MATERIALS LICENSES.Program-Specific Guidance About Portable Gauge Licenses.Draft Report For Comment. VACCA,P.C.; WHITTEN,J.E.; ARREDONDO,S.A.; et al. Division of Industrial & Medical Nuclear Safety (Post 870729). September 1996. 99pp. 9612030289. 91001:001.

As part of its redesign of the materials licensing process, NRC is consolidating and updating numerous guidance documents into a single comprehensive repository as described in NUREG-1539 and draft NUREG-1541. Draft NUREG-1556, Vol. 1, is the first program-specific guidance developed for the new process and may serve as a template for subsequent program-specific guidance. This document is ultimately intended for use by applicants, licensees, and NRC staff and will also be available to Agreement States. This document combines the guidance previously found in draft Regulatory Guide DG-0008, "Applications for the Use of Sealed Sources in Portable Gauging Devices," and in NMSS Policy and Guidance Directive 2-07, "Standard Review Plan for Applications for the Use of Sealed Sources in Portable Gauging Devices." This draft NUREG takes a graded, more performance-based approach to licensing portable gauges, reducing the information (amount and level of detail) needed in support of an application to use these devices. Note that this document is strictly for public comment and NOT for use in preparation or review of portable gauge licenses until it is published in final form.

**NUREG-1557:** SUMMARY OF TECHNICAL INFORMATION AND AGREEMENTS FROM NUCLEAR MANAGEMENT AND RESOURCES COUNCIL INDUSTRY REPORTS ADDRESSING LICENSE RENEWAL. REGAN,C.; LEE,S. Office of Nuclear Reactor Regulation (Post 941001). CHOPRA,O.K.; et al. Argonne National Laboratory. October 1996. 188pp. 9611180290. 90805:141.

In about 1990, the Nuclear Management and Resources Council (NUMARC) submitted for NRC review ten industry reports (IRs) addressing aging issues associated with specific structures and components of nuclear power plants and one IR addressing the screening methodology for integrated plant assessment. The NRC staff had been reviewing the ten NUMARC IRs; their comments on each IR and NUMARC responses to the comments have been compiled as public documents. This report provides a brief summary of the technical information and NUMARC/NRC agreements from the ten IRs, except for the Cable License Renewal IR. The technical information and agreements documented herein represent the status of the NRC staffs review when the NRC staff and industry resources were redirected to address rule implementation issues. The NRC staff plans to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.

**NUREG-1560 V1 P1 DFC:** INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE.Summary Report.Draft Report For Comment. * Division of Systems Technology (Post 941217). October 1996. 276pp. 9611190256. 90817:034.

This report provides perspectives gained by reviewing 75 Individual Plant Examination (IPE) submittals pertaining to 108 nuclear power plant units. IPEs are probabilistic analysis that estimate the core damage frequency (CDF) and containment performance for accidents initiated by internal events (including internal flooding, but excluding internal fire). The IPE submittals were reviewed to gain perspectives in three major areas: (1) improvements made to individual plants as a result of their IPEs and the collective result of the IPE program, (2) plant-specific design and operational features and modeling assumptions that significantly affect the estimates of CDF and containment performance, and (3) the quality of the IPEs with respect to their potential role in risk-informed regulation. These perspectives were gained by assessing the core damage and containment performance results, including overall CDF, accident sequences, dominant contributions to component failure and human error, and containment failure modes. These results were assessed in relation to the design and operational characteristics of the various reactor and containment types, and by comparing the IPEs to attributes of a quality probabilistic risk assessment. Methods data, boundary conditions, and assumptions used in the IPEs were considered in understanding the differences and similarities observed among the various types of plants.

**NUREG-1560 V2P2-5DFC:** INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE.Parts 2-5.Draft Report For Comment. * Division of Systems Technology (Post 941217). November 1996. 554pp. 9612160337. 91122:001.

See NUREG-1560,V01,P01,DFC abstract.

**NUREG-1561:** STANDARD REVIEW PLAN MAINTENANCE PROGRAM IMPLEMENTING PROCEDURES DOCUMENT. * Office of Nuclear Reactor Regulation (Post 941001). November 1996. 68pp. 9612030270. 90980:041.

The Implementing Procedures Document (IPD) was developed by the Inspection Program Branch, Office of Nuclear Reactor Regulation, with assistance from Pacific Northwest Laboratory, for the Standard Review Plan Maintenance Program (SRP-MP). The SRP-MP was established to maintain the Standard Review Plan (SRP) on an on-going basis. The IPD provides guidance, including an overall approach and procedures, for SRP-MP tasks. The objective of the IPD is to ensure that revisions to the SRP reflect current NRC requirements and guidance, and that a consistent methodology is used to develop and revise SRP sections.

**NUREG-1563:** BRANCH TECHNICAL POSITION ON THE USE OF EXPERT ELICITATION IN THE HIGH-LEVEL RADIOACTIVE WASTE PROGRAM. KOTRA,J.P.; LEE,M.P.; EISENBERG,N.A.; et al. Division of Waste Management (NMSS 940403). November 1996. 71pp. 9612110187. 91062:268.

The U.S. Nuclear Regulatory Commission staff expects that subjective judgments of individual experts and, in some cases, groups of experts, will be used by the U.S. Department of Energy (DOE) to interpret data obtained during site characterization and to address the many technical issues and inherent uncertainties associated with predicting the performance of a repository system for thousands of years. NRC has traditionally accepted, for review, expert judgment to evaluate and interpret the factual bases of license applications and is expected to give appropriate consideration to the judgments of DOE's experts regarding the geologic repository. Such consideration, however, envisions DOE using expert judgments to complement and supplement other sources of scientific and technical information, such as data collection, analyses, and experimentation. In this document, the NRC staff has set forth technical positions that: (1) provide general guidelines on those circumstances that may warrant the use of a formal process for obtaining the judgments of more than one expert; and (2) describe acceptable procedures for conducting expert elicitation when formally elicited judgments are used to support a demonstration of compliance with NRC's geologic disposal regulation, currently set forth in 10 CFR Part 60.

**NUREG-1564:** LONG-TERM KINETIC EFFECTS AND COLLOID FORMATIONS IN DISSOLUTION OF LWR SPENT FUEL. AHN,T.M. Division of Waste Management (NMSS 940403). November 1996. 60pp. 9612110158. 91068:147.

This report evaluates continuous dissolution and colloid formation during spent-fuel performance under repository conditions in high-level waste disposal. Various observations suggest that reprecipitated layers formed on spent-fuel surfaces may not be protective. This situation may lead to continuous dissolution of highly soluble radionuclides such as C-14, Cl-36, Tc-99, I-129, and Cs-135. However, the diffusion limits of various species involved may retard dissolution significantly. For low-solubility actinindes such as Pu-(239+240) or Am-(241+243), various processes regarding colloid formation have been analyzed. The processes analyzed are condensation, dispersion, and sorption. Colloid formation may lead to significant releases of low-solubility actinides. However, because there are only limited data available on matrix dissolution, colloid formation, and solubility limits, many uncertainties still exist. These uncertainties must be addressed before the significance of radionuclide releases can be determined.

**NUREG-1565:** DRY OXIDATION AND FRACTURE OF LWR SPENT FUEL. AHN,T.M. Division of Waste Management (NMSS 940403). November 1996. 40pp. 9612110153. 91064:281.

This report evaluates the characteristics of oxidation and fracture of light-water reactor (LWR) spent fuel in dry air. It also discusses their effects on radionuclide releases in the anticipated high-level waste repository environment. A sphere model may describe diffusion-limited formation of lower oxides, such as U(4)O(9), in the oxidation of the SF matrix. Detrimental higher oxides, such as U(3)O(8), may not form at temperatures below a threshold temperature. The nucleation process suggests that a threshold temperature exists. The calculated results regarding fracture properties of the SF matrix agree with experimental observations. Oxidation and fracture of Zircaloy may not be significant under anticipated conditions. Under saturated or unsaturated aqueous conditions, oxidation of the SF matrix is believed to increase the releases of Pu-(239+240), Am-(241+243), C-14, Tc-99, I-129, and Cs-135. Under dry conditions, I-129 releases are likely to be small, unlike C-14, in lower oxides; Cl-36, Tc-99, I-129, and Cs-135 may be released fast in higher oxides.

**NUREG-1567 DRFT FC:** STANDARD REVIEW PLAN FOR SPENT FUEL DRY STORAGE FACILITIES. Draft Report For Comment. * Office of Nuclear Material Safety & Safeguards. October 1996. 655pp. 9611190199. 90872:001.

The Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities provides guidance to the staff of the U.S. Nuclear Regulatory Commission for performing safety reviews of Spent Fuel Dry Storage Facilities. The SRP is intended to ensure the quality and uniformity of the staff reviews by establishing the review scope and requirements. Part 72, Subpart B generally specifies the information needed in a license application for the independent storage of spent nuclear fuel and high level radioactive waste. Regulatory requirements, staff position, industry codes and standards, acceptance criteria, and other information are discussed. Comments on this draft, will be considered and incorporated into the SRP as appropriate. The SRP is scheduled for publication as an NRC NUREG document in 1997. A separate Standard Review Plan for Dry Cask Storage Systems (DCSRP) was issued for public comment in February 1996. The DCSRP is scheduled to be published as an NRC NUREG document in January 1997. To ensure consistency between the two standard review plans (SRPs), comments on sections common to both SRPs will be incorporated, as appropriate, in both NUREG. Comments, errors or ommissions, and suggestions for improvement should be sent to the Chief, Rules Review and Directives, Division of Freedom of Information and Publication Services, Mail Stop T-6-D-59, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

**NUREG-1568:** LICENSE RENEWAL DEMONSTRATION PROGRAM: NRC OBSERVATIONS AND LESSONS LEARNED. PRATO,R.J.; KUO,P.T.; NEWBERRY,S.F. Office of Nuclear Reactor Regulation (Post 941001). December 1996. 25pp. 9701130209. 91400:012.

This report summarizes the Nuclear Regulatory Commission staff's observations and lessons learned from the five License Renewal Demonstration Program (LRDP) site visits performed by the staff from March 25, 1996, through August 16, 1996. The LRDP was a Nuclear Energy Institute (NEI) program intended to assess the effectiveness of the guidance provided by NEI 95-10, Revision 0, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," to implement the requirements of Title 10 of the Code of Federal Regulations, Part 54 (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." In general, NEI 95-10, appeared to contain most of the guidance needed for scoping, screening, identifying aging effects, developing aging management programs, and evaluating time-limited aging analysis. However, as expected, the LRDP site-visit reviews identified the need for some improvements to assist applicants in developing license renewal applications and supporting documentation. The improvements and additions to NEI 95-10 that are needed for developing an license renewal program consistent with the intent of the rule, will be included in NEI 95-10 or the applicable Regulatory Guide.

**NUREG-1575 DRFT FC:** MULTI-AGENCY RADIATION SURVEY AND SITE INVESTIGATION MANUAL (MARSSIM).Draft For Public Comment. * Division of Regulatory Applications (Post 941217). December 1996. 601pp. 9701130202. 91396:001.

The MARSSIM provides information on planning, conducting, evaluating, and documenting environmental radiological surveys for demonstrating compliance with dose-based regulations. The MARSSIM, when finalized, will be a multi-agency consensus document. MARSSIM was developed collaboratively over the past three years by four Federal agencies having authority for control of radioactive materials; EPA, DOD, DOE, and NRC (60 FR 12555). MARSSIM's objective is to describe standardized and consistent approaches for surveys, which provide a high degree of assurance that established dose-based release criteria, limits, guidelines, and conditions of the regulatory agencies are satisfied at all stages of the process, while at the same time encouraging an effective use of resources. The techniques, methodologies, and philosophies that form the bases of this manual were developed to be consistent with current Federal limits, guidelines, and procedures. The draft manual was prepared by a multi-agency technical working group composed of representatives from DOD, DOE, EPA, and NRC. Contractors to the NRC, EPA, and DOE, and members of the public have been present during the open meetings of the MARSSIM work group.

**NUREG/CP-0149 V01:** PROCEEDINGS OF THE TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING.Plenary Session, High Burnup Fuel Behavior, Thermal Hydraulic Research. MONTELEONE,S. Brookhaven National Laboratory. March 1996. 278pp. 9604150352. 87868:001.

This three-volume report contains papers presented at the Twenty-Third Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 23-25, 1995. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Italy, Japan, Norway, Russia, Sweden, and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

**NUREG/CP-0149 V02:** PROCEEDINGS OF THE TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING.Human Factors Research, Advanced I&C Hardware & Software, Severe Accident Research, Probabilistic Risk Assessment Topics, Individual Plant Examination. MONTELEONE,S. Brookhaven National Laboratory. March 1996. 542pp. 9604150357. 87866:001.

See NUREG/CP-0149,V01 abstract.

**NUREG/CP-0149 V03:** PROCEEDINGS OF THE TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING.Structural & Seismic Engineering, Primary Systems Integrity, Equipment Operability And Aging, ECCS Strainer Blockage Research & Regulatory Issues. MONTELEONE,S. Brookhaven National Laboratory. March 1996. 246pp. 9604150363. 87865:037.

See NUREG/CP-0149,V01 abstract.

**NUREG/CP-0150:** WORKSHOP ON ROCK MECHANICS ISSUES IN REPOSITORY DESIGN AND PERFORMANCE ASSESSMENT.Held At Holiday Inn Crowne Plaza, Rockville,Maryland,September 19-20, 1994. * Center for Nuclear Waste Regulatory Analyses. April 1996. 266pp. 9606030259. 88422:001.

The Center for Nuclear Waste Regulatory Analyses organized and hosted a workshop on "Rock Mechanics Issues in Repository Design and Performance Assessment" on behalf its sponsor the U.S. Nuclear Regulatory Commission (NRC). This workshop was held on September 19-20, 1994 at the Holiday Inn Crowne Plaza, Rockville, Maryland. The objectives of the workshop were to stimulate exchange of technical information among parties actively investigating rock mechanics issues relevant to the proposed high-level waste repository at Yucca Mountain and identify/confirm rock mechanics issues important to repository design and performance assessment. The workshop contained three technical sessions and two panel discussions. The participants included technical and research staffs representing the NRC and the Department of Energy and their contractors, as well as researchers from the academic, commercial, and international technical communities. These proceedings include most of the technical papers presented in the technical sessions and the transcripts for the two panel discussions.

**NUREG/CP-0151:** PROCEEDINGS OF THE IAEA SPECIALISTS' MEETING ON CRACKING IN LWR RPV HEAD PENETRATIONS.Held At ASTM Headquarters, Philadelphia, Pennsylvania, May 2-3, 1995. PUGH,C.E.; RANNEY,S.J. Oak Ridge National Laboratory. July 1996. 304pp. 9608210221. ORNL/TM-13187. 89420:001.

This report contains 17 papers that were presented in four sessions at the IAEA Specialists' Meeting on Cracking in LWR RPV Head Penetrations held at ASTM Headquarters in Philadelphia on May 2-3, 1995. The papers are compiled here in the order they were presented in the sessions, and they relate to operational observations, inspection techniques, analytical modeling, and regulatory control. The goal of the meeting was to allow international experts to review experience in the field of ensuring adequate performance of reactor pressure vessel (RPV) heads and penetrations. The emphasis was aimed at better understanding of behavior of reactor component materials, to provide guidance and recommendations assuring reliability, adequate performance, and directions for further investigations. The international nature of the meeting is illustrated by the fact that papers were presented by researchers from 10 countries. There were technical experts present from other countries who participated in discussions of the results presented. The IAEA issued a Working Material version of the meeting papers (IAEA IWG-LMNPP-95/1), and this present document incorporates the final version of the papers as received from the authors.

**NUREG/CP-0152:** PROCEEDINGS OF THE FOURTH NRC/ ASME SYMPOSIUM ON VALVE AND PUMP TESTING.Held At The Hyatt Regency Hotel,Washington,DC, July 15-18, 1996. * Office of Nuclear Reactor Regulation (Post 941001). * American Society of Mechanical Engineers. July 1996. 700pp. 9608140271. 89352:001.

The 1996 Symposium on Valve and Pump Testing, jointly sponsored by the Board on Nuclear Codes and Standards of the American Society of Mechanical Engineers and by the Nuclear Regulatory Commission, provides a forum for the discussion of current programs and methods for inservice testing and motor-operated valve testing at nuclear power plants. The symposium also provides an opportunity to discuss the need to improve that testing in order to help ensure the reliable performance of pumps and valves. The participation of industry representatives, regulators, and consultants results in the discussion of a broad spectrum of ideas and perspectives regarding the improvement of inservice testing of pumps and valves at nuclear power plants.

**NUREG/CR-2800 S05:** GUIDELINES FOR NUCLEAR POWER PLANT SAFETY ISSUE PRIORITIZATION INFORMATION DEVELOPMENT. DALING,P.M.; LAVENDER,J.C. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1996. 237pp. 9608050237. PNL-4297. 89243:001.

This is the sixth in a series of reports to document the use of a methodology developed by Pacific Northwest Laboratories to calculate, for prioritization purposes, the risk, dose, and cost impacts of implementing resolutions to reactor safety issues (NUREG/CR-2800, Andrews, et al., 1983). This report contains the results of issue-specific analyses for 34 generic issues. The results are referenced, as one consideration in setting priorities for reactor safety issues, in NUREG-0933, A Prioritization of Generic Safety Issues.

**NUREG/CR-2850 S01:** DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES.Methodology And Data Base. BAKER,D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1996. 250pp. 9607090240. 88957:001.

This manual describes a dose assessment system used to estimate the population or collective dose commitments received via both airborne and waterborne pathways by person living within a 2- to 80-kilometer region of a commercial operating power reactor for a specific year of effluent releases. Computer programs, data files, and utility routines are included which can be used in conjunction with an IBM or compatible personal computer to produce the required dose commitments and their statistical distributions. In addition, maximum individual airborne and waterborne dose commitments are estimated and compared to 10 CFR Part 50, Appendix I, design objectives.

**NUREG/CR-2850 V14:** DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1992. AABERG,R.L.; BAKER,D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1996. 196pp. 9604170363. PNL-4221. 87903:120.

Population and individual radiation dose commitments have been estimated from reported radionuclide releases from commercial power reactors operating during 1992. Fifty-year dose commitments for a one-year exposure from both liquid and atmospheric releases were calculated for four population groups (infant, child, teenager, and adult) residing between 2 and 80 km from each of 72 reactor sites. This report tabulates the results of these calculations, showing the dose commitments for both water and airborne pathways for each age group and organ. Also included for each of the sites is an estimate of individual doses that are compared with 10 CFR Part 50, Appendix I design objectives. The total collective dose commitments (from both liquid and airborne pathways) for each site ranged from a high of 3.7 person-rem to a low of 0.0015 person-rem for the sites with plants in operation and producing power during the year. The arithmetic mean was 0.66 person-rem. The total population dose for all sites was estimated at 47 person-rem for the 130 million people considered at risk. The individual dose commitments estimated for all sites were below the Appendix I design objectives.

**NUREG/CR-4219 V11 N2:** HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For April-September 1994. PENNELL,W.E. Oak Ridge National Laboratory. April 1996. 117pp. 9605130075. ORNL/TM-9593. 88220:126.

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. The HSST Program is organized in seven tasks: (1) program management, (2) constraint effects analytical development and validation, (3) evaluation of cladding effects, (4) ductile-to-cleavage fracture-mode conversion, (5) fracture analysis methods development and application, (6) material property data and test methods, and (7) integration of results. The program tasks have been structured to place emphasis on resolution fracture issues with near-term licensing significance. Resources to execute the research tasks are drawn from ORNL with sub-contract support from universities and other research laboratories. Close contact is maintained with the sister Heavy-Section Steel Irradiation (HSSI) Program at ORNL and with related research programs both in the United States and abroad. This report provides an overview of principal developments in each of the seven program tasks from April 1994 to September 1994.

**NUREG/CR-4219 V12 N1:** HEAVY-SECTION STEEL TECHNOLOGY PROGRAM.Semiannual Progress Report For October 1994 - March 1995. PENNELL,W.E. Oak Ridge National Laboratory. July 1996. 146pp. 9608070004. ORNL/TM-9593. 89285:040.

The Heavy-Section Steel Technology (HSST) Program is conducted for the U.S. 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. The HSST Program is organized in seven tasks: (1) program management, (2) constraint effects analytical development and validation, (3) evaluation of cladding effects, (4) ductile to cleavage fracture mode conversion, (5) fracture analysis methods development and applications, (6) material property data and test methods, and (7) integration of results into a state-of-the-art methodology. 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 the sister Heavy-Section Steel Irradiation Program at ORNL with related research programs both in the United States and abroad. This report provides an overview of principal developments in each of the seven program tasks from October 1994 - March 1995.

**NUREG/CR-4667 V20:** ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1994 - March 1995. CHUNG,H.M.; CHOPRA,O.K.; GAVENDA,D.J.; et al. Argonne National Laboratory. January 1996. 72pp. 9602050045. ANL-95/41. 86980:234.

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRs) from October 1994 to March 1995. Topics that have been investigated include (a) fatigue of carbon and low-alloy steel used in reactor piping and pressure vessels, (b) EAC of Alloy 600 and 690, and (c) irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS. Fatigue tests were conducted on ferritic steels in water with several dissolved-oxygen (DO) concentrations to determine whether a slow strain rate applied during different portions of a tensile-loading cycle are equally effective in decreasing fatigue life. Tensile properties and microstructures of several heats of Alloy 600 and 690 were characterized for correlation with EAC of the alloys in simulated LWR environments. Effects of DO and electrochemical potential on susceptibility to intergranular cracking of high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath irradiated in boiling water reactors were determined in slow-strain-rate-tensile tests at 289 degrees C. Microchemical changes in the specimens were studied by Auger electron spectroscopy and scanning electron microscopy to determine whether trace impurity elements may contribute to IASCC of these materials.

**NUREG/CR-4667 V21:** ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April 1995 - December 1995. CHOPRA,O.K.; CHUNG,H.M.; GRUBER,E.E.; et al. Argonne National Laboratory. July 1996. 87pp. 9608210264. ANL-96/1. 89422:253.

This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRs) from April 1995 to December 1995. Topics that have been investigated include (a) fatigue of carbon and low-alloy steel used in reactor piping and pressure vessels, (b) EAC of Alloys 600 and 690, and (c) irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS. Fatigue tests were conducted on ferritic steels in water that contained various concentrations of dissolved oxygen (DO) to determine whether a slow strain rate applied during different portions of a tensile-loading cycle are equally effective in decreasing fatigue life. Crack-growth-rate tests were conducted on compact-tension specimens from several heats of Alloys 600 and 690 in simulated LWR environments. Effects of fluoride-ion contamination on susceptibility to intergranular cracking of high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes irradiated in boiling water reactors were determined in slow-strain-rate-tensile tests at 288 degrees C. Microchemical changes in the specimens were studied by Auger electron spectroscopy and scanning electron microscopy to determine whether trace impurity elements may contribute to IASCC of these materials.

**NUREG/CR-4918 V09:** CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report On Field Experiments At A Humid Region Site,Beltsville,Maryland. SCHULZ,R.K. California, Univ. of, Los Angeles, CA. RIDKY,R.W. Maryland, Univ. of, College Park, MD. O'DONNELL,E. Division of Regulatory Applications (Post 941217). August 1996. 30pp. 9609030360. 89546:303.

The project objective is to assess means for controlling waste infiltration through waste disposal unit covers in humid regions. Experimental work is being performed in large scale lysimeters (70'x45'xlO') at Beltsville, MD,and results of the assessment are applicable to disposal of LLW, uranium mill tailings, hazardous waste, and sanitary landfills. Three concepts are under investigation: (1) resistive layer barrier, (2) conductive layer barrier, and (3) bioengineering water management. The resistive layer barrier consists of compacted earth (clay). The conductive layer barrier is a special case of the capillary barrier and it requires a flow layer (e.g. fine sandy loam) over a capillary break. As long as unsaturated conditions are maintained water is conducted by the flow layer to below the waste. This barrier is most efficient at low flow rates and is thus best placed below a resistive layer barrier. Such a combination of the resistive layer over the conductive layer barrier promises to be highly effective provided there is no appreciable subsidence. Bioengineering water management is a surface cover that is designed to accommodate subsidence. It consists of impermeable panels which enhance run-off and limit infiltration. Vegetation is planted in narrow openings between panels to transpire water from below the panels. This system has successfully dewatered two lysimeters thus demonstrating that this procedure could be used for remedial action "drying out" existing water-logged disposal sites at low cost.

**NUREG/CR-5068:** PIPING INSPECTION ROUND ROBIN. HEASLER,P.G.; DOCTOR,S.R. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1996. 196pp. 9604230383. PNNL-10475. 87992:143.

The piping inspection round robin was conducted in 1981 at the Pacific Northwest National Laboratory (PNNL) to quantify the capability of ultrasonics for inservice inspection and to address some aspects of reliability for this type of.nondestructive evaluation (NDE). The research was sponsored by the U.S. Nuclear Regulatory Commission, Office of Research, under a program entitled, "Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors." The round robin measured the crack detection capabilities of seven field inspection teams who employed procedures that met or exceeded the 1977 edition through the 1978 addenda of the American Society of Mechanical Engineers (ASME) Section XI Code requirements. Three different types of material were employed in the study (cast stainless steel, clad ferritic, and wrought stainless steel), and two different types of flaws were implanted into the specimens (intergranular stress corrosion cracks (IGSCCs) and thermal fatigue cracks (TFCs)). When considering near-side inspection, far-side inspection, and false call rate, the overall performance was found to be best in clad ferritic, less effective in wrought stainless steel and the worst in cast stainless steel. Depth sizing performance showed little correlation with the true crack depths.

**NUREG/CR-5229 V08:** FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1995.Annual Report. MCCONNELL,J.W.; ROGERS,R.D. Idaho National Engineering Laboratory. JASTROW,J.D.; et al. Argonne National Laboratory. June 1996. 82pp. 9607090111. INEL-94/0278. 88958:118.

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 organic ion-exchange resins caused by radiation, (b) examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Form to meet the requirements of 10 CFR 61 using solidified ion-exchange resins, (c) obtaining performance information on solidified ion-exchange resins in a disposal environment, and (d) determining the condition of liners used to dispose the ion-exchange resins. Compressive test results of 12-year-old cement and vynyl ester-styrene solidified waste forms are presented, which show effects of aging and self-irradiation. Results of the tenth 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-East in Illinois and at Oak Ridge National Laboratory. 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-5442:** RELIABILITY-BASED CONDITION ASSESSMENT OF STEEL CONTAINMENTS AND LINERS. ELLINGWOOD,B.R.; BHATTACHARYA,B.; ZHENG,R-H.; et al. Johns Hopkins Univ., Baltimore, MD. November 1996. 109pp. 9612160327. ORNL/TM-13244. 91099:113.

The evaluation of steel containments and liners for continued service must provide assurance that they are able to withstand future extreme loads during the service period with a level of reliability that is sufficient for public safety. This research demonstrates the feasibility of using reliability analysis as a tool for performing condition assessments and service life predictions of steel containment and liners. Mathematical models that describe time-dependent charges in steel due to aggressive environmental factors are identified, and statistical data supporting the use of these models in time-dependent reliability analysis are summarized. The analysis of steel containment fragility is described, and simple illustrations of the impact on reliability of structural degradation are provided. The role of nondestructive evaluation in time-dependent reliability analysis, both in terms of defect detection and sizing,is examined. A Markov model provides a tool for accounting for time-dependent changes in damage condition of a structural component or system.

**NUREG/CR-5535 V07 R1:** RELAP5/MOD3 CODE MANUAL.Summaries And Reviews Of Independent Code Assessment Reports. MOORE,R.L.; SLOAN,S.M.; SCHULTZ,R.R.; et al. Idaho National Engineering Laboratory. October 1996. 139pp. 9611190220. INEL-95/0174. 90823:110.

Summaries of RELAP5/MOD3 code assessments, a listing of the assessment matrix, and a chronology of the various versions of the code are given. Results from these code assessments have been used to formulate a compilation of some of the strengths and weaknesses of the code. These results are documented in the report. Volume 7 was designed to be updated periodically and to include the results of the latest code assessments as they become available. Consequently, users of Volume 7 should ensure that they have the latest revision available.

**NUREG/CR-5591 V06 N2:** HEAVY-SECTION STEEL IRRADIATION PROGRAM.Semiannual Progress Report For April Through September 1995. CORWIN,W.R. Oak Ridge National Laboratory. August 1996. 73pp. 9612040119. ORNL/TM-11568. 91007:061.

The goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of effects of neutron irradiation on material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 14 tasks: (1) program management, (2) fracture toughness (K(Ic)) curve shift in high-copper welds, (3) crack-arrest toughness (K(Ia)) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K(Ic) and K(Ia) curve shifts in low upper-shelf welds, (6) annealing effects in low upper-shelf welds, (7) irradiation effects in a commercial low upper-shelf weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) JPDR steel examination, (13) technical assistance for JCCCNRS Working Groups 3 and 12, and (14) additional requirements for materials. This report provides an overview of the activities within each of these tasks from April Through September 1995.

**NUREG/CR-5595 R01:** FORECAST: REGULATORY EFFECTS COST ANALYSIS SOFTWARE MANUAL. Version 4.1. LOPEZ,B.; SCIACCA,F.W. Science & Engineering Associates, Inc. * S. Cohen & Associates, Inc. July 1996. 143pp. 9608050060. SEA95-2755010A1. 89250:001.

The FORECAST program was developed to facilitate the preparation of the value-impact portion of NRC regulatory analyses. This PC program integrates the major cost and benefit considerations that may result from a proposed regulatory change. FORECAST automates much of the calculations typically needed in a regulatory analysis and thus reduces the time and labor required to perform these analyses. More importantly, its integrated and consistent treatment of the different value impact considerations should help assure comprehensiveness, uniformity, and accuracy in the preparation of NRC regulatory analyses. The current FORECAST version 4.1 has been upgraded from the previous version and now includes an uncertainty package and an automatic cost escalation package. In addition, it now explicitly addresses public health impacts, occupational health impacts, onsite property damage, and government costs.

**NUREG/CR-5631 R02:** CONTRIBUTION OF MATERNAL RADIONUCLIDE BURDENS TO PRENATAL RADIATION DOSES. SIKOV,M.R.; HUI,T.E.; TRAUB,R.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1996. 389pp. 9606070092. PNL-7445. 88485:001.

This report describes approaches to calculating and expressing radiation doses to the embryo/fetus from internal radionuclides. Information for occupationally and medically significant radioelements was used to derive biokinetic transfer models and integrated with metabolic patterns. Placental transfer and radioactivty levels in the embryo/fetus were calculated as a function of stage of pregnancy and time after administration and are given as tables of deposition and retention in the embryo/fetus. Methodologies described by MIRD were extended to calculate radiation absorbed doses to the embryo/fetus using a scenario that assumed injection of a bolus into the woman's blood. Calculations were performed for administration at successive months of pregnancy to accommodate stage dependence of geometric relationships and biological behaviors of radionuclides. The gestational-stage-dependent dosimetric dose factors are based on radiation absorbed doses. Multiplication by appropriate quality factors convert these to dose equivalent, the most common quantity for stating prenatal dose limits in the United States. The dose factor tabulations are supplemented with tables of correlations and surrogate dose factors.

**NUREG/CR-5753:** AGING OF SAFETY CLASS 1E TRANSFORMERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS. ROBERTS,E.W.; EDSON,J.L.; UDY,A.C. Idaho National Engineering Laboratory. February 1996. 64pp. 9603010296. INEL-95/0573. 87307:067.

This report discusses aging effects on safety-related power transformers in nuclear power plants. It also evaluates maintenance, testing, and monitoring practices, with respect to their effectiveness in detecting and mitigating the effects of aging. The study follows the U.S. Nuclear Regulatory Commission's (NRC) Nuclear Plant-Aging Research approach. It investigates the materials used in transformer construction, identifies stressors and aging mechanisms, presents operating and testing experience with aging effects, analyzes transformer failure events reported in various databases, and evaluates maintenance practices. Databases that were analyzed included the NRC's Licensee Event Report (LER) system and the Institute for Nuclear Power Operations' Nuclear Plant Reliability Data System (NPRDS).

**NUREG/CR-5758 V06:** FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1995. SILBERNAGEL,M.; BRICHOUX,J.; DURBIN,N. Battelle Seattle Research Center. July 1996. 96pp. 9608060285. PNL-11202. 89258:230.

This report summarizes the data from the semiannual reports of fitness-for-duty program submitted to the NRC by utilities for two reporting periods: January 1 through June 30, 1995, and July 1 through December 31, 1995. During 1995, licensees reported that they conducted 150,121 tests for the presence of illegal drugs and alcohol. Of these tests, 1,476 (.98%) were confirmed positive. Positive test results varied by category of test and category of worker. The majority of positive test results (1,122) were obtained through pre-access testing. Of tests conducted on workers having access to the protected area, 180 were positive from random testing and 139 were positive from for-cause testing. Follow-up testing of workers who had previously tested positive resulted in 35 positive tests. For-cause testing resulted in the highest percentage of positive tests; about 18% of for-cause tests were positive. In comparison, 1.41% of pre-access tests and .27% of random tests were positive. Positive test rates also varied by category of worker. When all types of tests are combined (pre-access, random, for-cause and follow-up testing), short-term contractor personnel had the highest positive test rate at 1.44%. Licensee employees and long-term contractors had lower combined positive test rates (.34% and .40%, respectively). Of the substances tested, marijuana was responsible for the highest percentage of positive test results (53.08%), followed by cocaine (24.24%) and alcohol (17.17%).

**NUREG/CR-5973 R03:** CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS. NICKOLAUS,J.R.; BOHLANDER,K.L. Battelle Memorial Institute, Pacific Northwest Laboratory. August 1996. 555pp. 9609030269. PNL-8462. 89548:001.

As part of the U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan Update and Development Program (SRP-UDP), Pacific Northwest National Laboratory developed a listing of industry consensus codes and standards and other government and industry guidance referred to in regulatory documents. The SRP-UDP has been completed and the SRP-Maintenance Program (SRP-MP) is now maintaining this listing. Besides updating previous information, Revision 3 adds approximately 80 citations. This listing identifies the version of the code or standard cited in the regulatory document, the regulatory document, and the current version of the code or standard. It also provides a summary characterization of the nature of the citation. This listing was developed from electronic searches of the Code of Federal Regulations and the NRC's Bulletins, Information Notices, Circulars, Enforcement Manual, Generic Letters, Inspection Manual, Policy Statements, Regulatory Guides, Standard Technical Specifications and the Standard Review Plan (NUREG-0800).

**NUREG/CR-5985 S01:** REVIEW OF P-SCAN COMPUTER-BASED ULTRASONIC INSERVICE INSPECTION SYSTEM. HARRIS,R.V.; ANGEL,L.J. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1995. 129pp. 9604090369. PNL-8919. 87807:001.

This supplement reviews the P-scan system, a computer-based ultrasonic system used for inservice inspection of piping and other components in nuclear power plants. The supplement was prepared using the methodology described in detail in Appendix A of NUREG/CR-5985, and is based on one month of using the system in a laboratory. This supplement describes and characterizes: computer system, ultrasonic components, and mechanical components; scanning, detection, digitizing, imaging, data interpretation, operator interaction, data handling, and record-keeping. It includes a general description, a review checklist, and detailed results of all tests performed.

**NUREG/CR-6074 V02:** SEALED SOURCE AND DEVICE DESIGN SAFETY TESTING.Technical Report On The Findings Of Task 4 Investigation of Failed Nitinol Brachytherapy Wire. BENAC,D.J.; BURGHARD,H.C. Southwest Research Institute. March 1996. 268pp. 9605130090. 04-4448-010. 88228:047.

This report covers an investigation of the nature and cause of failure in Nitinol brachytherapy sourcewires. The investigation was initiated after two clinical incidents in which sourcewires failed during or immediately after a treatment. The investigation determined that the two clinical Nitinol sourcewires failed in a brittle manner, which is atypical for Nitinol. There were no material anomalies or subcritical flaws to explain the brittle failures. The bend tests also demonstrated that neither moist environment, radiation, nor low-temperature structure transformation was a likely root cause of the failures. However, degradation of the PTFE was consistently evident, and those sourcewires shipped or stored with PTFE sleeves consistently failed in laboratory bend tests. On the basis of the results of this study, it was concluded that the root cause of the in-service failures of the sourcewires was environmentally induced embrittlement due to the breakdown of the PTFE protective sleeves in the presence of the high-radiation field and subsequent reaction or interaction of the breakdown products with the Nitinol alloy.

**NUREG/CR-6096:** APACHE LEAP TUFF INTRAVAL EXPERIMENTS.Results And Lessons Learned. RASMUSSEN,T.C. Georgia, Univ. of, Athens, GA. RHODES,S.C.; GUZMAN,A.G.; et al. Arizona, Univ. of, Tucson, AZ. March 1996. 110pp. 9604020277. 87714:170.

Data from laboratory and field experiments in unsaturated fractured rock are summarized and interpreted for the purpose of evaluating conceptual and numerical models of fluid, heat and solute transport. The experiments were conducted at four scales, in small cores (2.5-cm long by 6-cm across), a large core (12-cm long by 10-cm across), a small block containing a single fracture (20 x 21 x 93 cm), and at field scales in boreholes (30-m long by 10-cm across) at three scales (1/2-, 1- and 3-meters). The smallest scale in the laboratory provided isothermal hydraulic and thermal properties of unfractured rock. Nonisothermal heat, fluid and solute transport experiments were conducted using the large core. Isothermal gas and liquid flow experiments were conducted in the fractured block. Field-scale experiments using air were used to obtain in situ permeability estimates as a function of the measurement scale. Interpretation of experimental results provides guidance for resolving uncertainties related to radionuclide migration from high level waste repositories in unsaturated fractured rock.

**NUREG/CR-6124:** CHARACTERIZATION OF RADIONUCLIDE-CHELATING AGENT COMPLEXES FOUND IN LOW-LEVEL RADIOACTIVE DECONTAMINATION WASTE. Literature Review. SERNE,R.J.; FELMY,A.R.; CANTRELL,K.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1996. 168pp. 9604160360. PNL-8856. 87884:011.

The U.S. Nuclear Regulatory Commission is responsible for regulating the safe land disposal of low-level radioactive wastes that may contain organic chelating agents. Such agents include EDTA, DTPA, picolinic acid, and citric acid, which can form radionuclide-chelate complexes that my enhance the migration of radionuclides from disposal sites. Data from the available literature indicate that chelates, can leach from solidified decontamination wastes in moderate concentration and can potentially complex certain radionuclides in the leachates. The effects of the formation of such radionuclide-chelate complexes on the migration of radionuclides in groundwater systems is still difficult to quantitatively predict. However, in general it appears that both EDTA and DTPA have the potential to mobilize radionuclides from waste disposal sites because such chelates can leach in moderate concentration, form strong radionuclide-chelate complexes, and can be recalcitrant to biodegradation. It also appears that oxalic acid and citric acid will not greatly enhance the mobility of radionuclides from waste disposal sites because these chelates do not appear to leach in high concentration tend to form relatively weak radionuclide-chelate complexes, and can be readily biodegraded. In the case of picolinic acid, insufficient data are available to make definitive predictions.

**NUREG/CR-6163:** COMPUTER PROGRAMS FOR THE ACQUISITION AND ANALYSIS OF EDDY CURRENT ARRAY PROBE DATA. PATE,J.R.; DODD,C.V. Oak Ridge National Laboratory. July 1996. 186pp. 9608210249. ORNL/TM-13212. 89424:001.

The objective of the Improved Eddy-Current ISI for Steam Generators Tubing program is to upgrade and validate eddy-current inspections, including probes, instrumentation, and data processing techniques for inservice inspection of new, used, and repaired steam generator tubes; to improve defect detection, classification and characterization as affected by diameter and thickness variations, denting, probe wobble, tube sheet, tube supports, copper and sludge deposits, even when defect types and other variables occur in combination; to transfer this advanced technology to NRC's mobile NDE laboratory and staff. This report documents computer programs that were developed for acquisition of eddy-current data from specially-designed 16-coil array probes. Complete code as well as instructions for use are provided.

**NUREG/CR-6174 V01:** REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE BOILING WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Assurance Requirements Of The Decommissioning Rule And.... SMITH,R.I.; BIERSCHBACH,M.C; KONZEK,G.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1996. 120pp. 9608210240. PNL-9975. 89428:001.

With the issuance of the Decommissioning Rule in 1988, nuclear power plant licensees are required to submit to the U.S. Nuclear Regulatory Commission (NRC) decommissioning cost estimates for review. This reevaluation study provides some of the needed bases documentation to the NRC staff that will assist them in assessing the adequacy of the licensee submittals. This report presents the results of a review and reevaluation of the PNL 1980 decommissioning study of the WNP-2 nuclear plant for the DECON, SAFSTOR, and ENTOMB decommissioning alternatives. These alternatives now include an initial 5-7 year period during which the spent fuel is stored in the spent fuel pool, prior to beginning major disassembly or extended safe storage of the plant. This report also includes NRC consideration that decommissioning activities leading to termination of the nuclear license be completed within 60 years of final reactor shutdown, consideration of packaging and disposal requirements for Greater-Than-Class C low-level waste, and reflects all costs in 1993 dollars. Sensitivity of the total license termination cost to the disposal at different low-level radioactive waste disposal sites, and to different depths of contaminated concrete surface removed, is also examined.

**NUREG/CR-6174 V02:** REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE BOILING WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Assurance Requirements Of The Decommissioning Rule And.... SMITH,R.I.; BIERSCHBACH,M.C; KONZEK,G.J.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. July 1996. 252pp. 9608210245. PNL-9975. 89422:001.

With the issuance of the Decommissioning Rule in 1988, nuclear power plant licensees are required to submit to the U.S. Nuclear Regulatory Commission (NRC) decommissioning cost estimates for review. This reevaluation study provides some of the needed bases documentation to the NRC staff that will assist them in assessing the adequacy of the licensee submittals. This report presents the results of a review and reevaluation of the PNL 1980 decommissioning study of the WNP-2 nuclear plant for the DECON, SAFSTOR, and ENTOMB decommissioning alternatives. These alternatives now include an initial 5-7 year period during which the spent fuel is stored in the spent fuel pool, prior to beginning major disassembly or extended safe storage of the plant. This report also includes NRC consideration that decommissioning activities leading to termination of the nuclear license be completed within 60 years of final reactor shutdown, consideration of packaging and disposal requirements for Greater-Than-Class C low-level waste, and reflects all costs in 1993 dollars. Sensitivity of the total license termination cost to the disposal at different low-level radioactive waste disposal sites, and to different depths of contaminated concrete surface removed, is also examined.

**NUREG/CR-6189:** A SIMPLIFIED MODEL OF AEROSOL REMOVAL BY NATURAL PROCESSES IN REACTOR CONTAINMENTS. POWERS,D.A.; WASHINGTON,K.E. Sandia National Laboratories. BURSON,S.B.; et al. Division of Systems Technology (Post 941217). July 1996. 263pp. 9608060182. SAND94-0407. 89278:001.

Simplified formulae are developed for estimating the aerosol decontamination that can be achieved by natural processes in the containments of pressurized water reactors and in the drywells of boiling water reactors under severe accident conditions. These simplified formulae were derived by correlation of results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior under accident conditions. Monte Carlo uncertainty analyses of decontamination by natural aerosol processes are reported for 1000, 2000, 3000, and 4000 MW(th) pressurized water reactors and for 1500, 2500, and 3500 MW(th) boiling water reactors. Uncertainty distributions for the decontamination factors and decontamination coefficients as functions of time were developed in the Monte Carlo analyses by considering uncertainties in aerosol processes, material properties, reactor geometry, and severe accident progression. Phenomenological uncertainties examined in this work included uncertainties in aerosol coagulation by gravitational collision, Brownian diffusion, turbulent diffusion, and turbulent inertia. Uncertainties in aerosol deposition by gravitational settling, thermophoresis, diffusiophoresis, and turbulent diffusion were examined. Electrostatic charging of aerosol particles in severe accidents is discussed. Median (50 percentile), 90, and 10 percentile values of the uncertainty distributions for effective decontamination coefficients were correlated with time and reactor thermal power. These correlations constitute a simplified model that can be used to estimate the decontamination by natural aerosol processes at three levels of conservatism. Example applications of the simplified model are described.

**NUREG/CR-6202:** LONG-TERM AGING AND LOSS-OF-COOLANT ACCIDENT (LOCA) TESTING OF ELECTRICAL CABLES.U.S./French Cooperative Research Program. NELSON,C.F. Sandia National Laboratories. GAUTHIER,G.; CARLIN,F.; et al. France. October 1996. 146pp. 9611190225. IPSN 94-03. 90821:177.

Experiments were performed to assess the aging degradation and loss-of-coolant accident (LOCA) behavior of electrical cables subjected to long-term aging exposures. Four different cable types were tested in both the U.S. and France. 1. U.S. 2 conductor with ethylene propylene rubber (EPR) insulation and a Hypalon jacket. 2. U.S. 3 conductor with cross-linked polyethylene (XLPE) insulation and a Hypalon jacket. 3. French 3 conductor with EPR insulation and Hypalon jacket. 4. French coaxial with polyethylene (PE) insulation and a PE jacket. The data represent up to 5 years of simultaneous aging where the cables were exposed to identical aging radiation doses at either 40 degrees C or 70 degrees C; however, the dose rate used for the aging irradiation was varied over a wide range (2-100 Gy/hr). Aging was followed by exposure to simulated French LOCA conditions. Several mechanical, electrical, and physical-chemical condition monitoring techniques were used to investigate the degradation behavior of the cables. All the cables, except for the French PE cable, performed acceptably during the aging and LOCA simulations. In general, cable degradation was highest for the lowest dose rate, and the amount of degradation decreased as the dose rate was increased.

**NUREG/CR-6210:** COMPUTER CODES FOR EVALUATION OF CONTROL ROOM HABITABILITY (HABIT). STAGE,S.A. Battelle Memorial Institute, Pacific Northwest Laboratory. June 1996. 173pp. 9606250169. PNL-10496. 88701:066.

This report describes the computer codes for evaluation of control room habitability (HABIT). HABIT is a package of computer codes designed to be used for the evaluation of control room habitability in the event of an accidental release of toxic chemicals or radioactive materials. Given information about the design of a nuclear power plant, a scenario for the release of toxic chemicals or radionuclides, and information about the air flows and protection systems of the control room; HABIT can be used to estimate the chemical exposure or radiological dose of control room personnel. HABIT is an integrated package of several programs that previously needed to be run separately and required considerable user intervention. These are EXTRAN, CHEM, TACT5, FPFP—2, and CONHAB. New input routines have been written for these routines using data input windows. These are designed for easy use entering, reviewing and revising the data. The programs can now be run in sequence as an integrated package. Improvements have been made in the computational methods used by some of the routines. The programs produce files containing ASCII tables of values and output files that can readily be imported into a commercial spreadsheet to be graphed or for further computations.

**NUREG/CR-6227:** PERFORMANCE DEMONSTRATION TESTS FOR EDDY CURRENT INSPECTION OF STEAM GENERATOR TUBING. KURTZ,R.J.; HEASLER,P.G.; ANDERSON,C.M. Battelle Memorial Institute, Pacific Northwest Laboratory. May 1996. 54pp. 9606180253. PNNL-9433. 88630:275.

This report describes the methodology and results for development of performance demonstration tests for eddy current (ET) inspection of steam generator tubes. Statistical test design principle were used to develop the performance demonstration tests. Thresholds on ET system inspection performance were selected to ensure that field inspection systems have a high probability of detecting and correctly sizing tube degradation. The technical basis for the ET system performance thresholds is presented in detail. Statistical test design calculations for probability of detection and flaw sizing tests are described. A recommended performance demonstration test based on the design calculations is presented. A computer program for grading the probability of detection portion of the performance demonstration test is given.

**NUREG/CR-6230:** RADIOANALYTICAL TECHNOLOGY FOR 10 CFR PART 61 AND OTHER SELECTED RADIONUCLIDES.Literature Review. THOMAS,C.W.; THOMAS,V.W.; ROBERTSON,D.E. Battelle Memorial Institute, Pacific Northwest Laboratory. March 1996. 42pp. 9604150341. PNL-9444. 87865:319.

A comprehensive literature review and assessment was conducted to identify and evaluate radioanalytical technology and procedures used for measuring 10CFR61 radionuclides and other long-lived isotopes. This review evaluated radiochemical procedures currently in use at a number of laboratories in the US, as well as identifying new advanced methods and techniques which could be adapted for routine radiochemical analyses of low-level radioactive waste. The 10CFR61 radionuclides include (14)C, (60)Co, (59,63) Ni, (90)Sr, (94)Nb, (99)Tc, (129)I, (137)Cs and TRU isotopes with half-lives greater than five years. The other low-level radionuclides of interest include (7,10)Be, (26)Al, (36)Cl, (93)Mo, (109,113m)Cd, and (121m,126)Sn, which may be present in various types of waste streams from nuclear power stations.

**NUREG/CR-6246:** EFFECTS OF AGING AND SERVICE WEAR ON MAIN STEAM ISOLATION VALVES AND VALVE OPERATORS. CLARK,R.L. Oak Ridge National Laboratory. March 1996. 108pp. 9603260298. ORNL-6814. 87640:001.

In recent years main steam isolation valve (MSIV) operating problems have resulted in significant operational transients (e.g., spurious reactor trips, steam generator dry out, excessive valve seat leakage), increased cost, and decreased plant availability. A key ingredient to an engineering-oriented reliability improvement effort is a thorough understanding of relevant historical experience. A detailed review of historical failure data available through the Institute of Nuclear Power Operation's Nuclear Plant Reliability Data System has been conducted for several types of MSIVs and valve operators for both boiling-water reactors and pressurized-water reactors. The focus of this review is on MSIV Failures modes, actuator failure modes, consequences of failure on plant operations method of failure detection, and major stressors affecting both valves and valve operators.

**NUREG/CR-6253:** PIUS CORE PERFORMANCE ANALYSIS. CAREW,J.F.; ARONSON,A.; COKINOS,D.M.; et al. Brookhaven National Laboratory. May 1996. 71pp. 9606070090. BNL-NUREG-52425. 88490:283.

A detailed evaluation of the fuel burnup dependent power distribution and the scram reactivity for the PIUS reactor design has been performed. The analyses were carried out using the CPM lattice physics and NODE-P2 core neutronics/thermal-hydraulics codes and are based on the information provided in the

PIUS Preliminary Safety Information Document. Cycle depletion calculations were performed for a set of nine representative initial core loadings and the three-dimensional core power distributions were determined. These calculations indicate that the PIUS radial F($\Delta$h) and total F(Q) power peaking is significantly stronger than indicated by the PIUS reference design values. The scram reactivity resulting from the injection of highly borated pool water was calculated for a series of time-dependent linear ramp and square-wave pool flows. The three-dimensional distribution of the pool water throughout the core was modeled and the spatial reactivity effects of the distributed boron were determined. For pool flows that increase as a linear ramp, the spatial reactivity effects of the distributed boron were very small. In this case, a constant core-average boron reactivity coefficient can be used to model the PIUS scram reactivity.

**NUREG/CR-6256 V03:** FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM LYSIMETER TEST RESULTS FOR FISCAL YEARS 1994 AND 1995. MCCONNELL,J.W.; ROGERS,R.D. Idaho National Engineering Laboratory. JASTROW,J.D.; et al. Argonne National Laboratory. June 1996. 121pp. 9607090148. INEL-95/0073. 88956:112.

The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission NRC), 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 Form 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 final 2 (10 total) years of data acquisition from operation of 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-East in Illinois and at Oak Ridge National Laboratory. The experimental equipment is described and results of waste form characterization using tests recommended by the NRC's "Technical Position on Waste Form" are presented. The study is designed to provide continuous data on nuclide release and movement, as well as environmental conditions, over a 20-year period. At the end of the tenth year, the experiment was closed down. Examination of soil and waste forms is planned to be conducted next and will be reported later.

**NUREG/CR-6270:** ESTIMATING BOILING WATER REACTOR DECOMMISSIONING COSTS.A User's Manual For The BWR Cost Estimating Computer Program (CECP) Software.Final Report. BIERSCHBACH,M.C Battelle Memorial Institute, Pacific Northwest Laboratory. June 1996. 216pp. 9607150168. PNL-10086. 89003:058.

Nuclear power plant licensees are required to submit to the U.S. Nuclear Regulatory Commission (NRC) for review decommissioning cost estimates. This user's manual and the accompanying Cost Estimating Computer Program (CECP) software provide a cost- calculating methodology to the NRC staff that will assist them in assessing the adequacy of the licensee submittals. The CECP, designed to be used on a personal computer, provides estimates for the cost of decommissioning BWR power stations to the point of license termination. Such cost estimates include component, piping, and equipment removal costs; packaging costs; decontamination costs; transportation costs; burial costs; and manpower costs. In addition to costs, the CECP also calculates burial volumes, person-hours, crew-hours, and exposure person-hours associated with decommissioning.

**NUREG/CR-6279:** APPLICATION OF FRACTURE TOUGHNESS SCALING MODELS TO THE DUCTILE-TO-BRITTLE TRANSITION. LINK,R.E. Navy, Dept. of. JOYCE,J.A. U.S. Naval Academy, Annapolis, MD. January 1996. 42pp. 9602220350. 87234:102.

An experimental investigation of fracture toughness in the ductile-brittle transition range was conducted. A large number of ASTM A533, Grade B steel, bend and tension specimens with varying crack lengths were tested throughout the transition region. Cleavage fracture toughness scaling models were utilized to correct the data for the loss of constraint in short crack specimens and tension geometries. The toughness scaling models were effective in reducing the scatter in the data, but tended to over-correct the results for the short crack bend specimens. A proposed ASTM Test Practice for Fracture Toughness in the Transition Range, which employs a master curve concept, was applied to the results. The proposed master curve over predicted the fracture toughness in the mid-transition and a modified master curve was developed that more accurately modeled the transition behavior of the material. Finally, the modified master curve and the fracture toughness scaling models were combined to predict the as-measured fracture toughness of the short crack bend and the tension specimens. It was shown that when the scaling models over correct the data for loss of constraint, they can also lead to non- conservative estimates of the increase in toughness for low constraint geometries.

**NUREG/CR-6280:** TECHNOLOGY,SAFETY,AND COSTS OF DECOMMISSIONING A REFERENCE LARGE IRRADIATOR AND REFERENCE SEALED SOURCES. HAFFNER,D.R.; VILLEGAS,A.J. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1996. 109pp. 9602290251. 87295:001.

This report contains the results of a study sponsored by the U.S. Nuclear Regulatory Commission (NRC) to examine the decommissioning of large radioactive irradiators and their respective facilities, and a broad spectrum of sealed radioactive sources and their respective devices. Conceptual decommissioning activities are identified, and the technology, safety, and costs (in early 1993 dollars) associated with decommissioning the reference large irradiator and sealed source facilities are evaluated. The study provides bases and background data for possible future NRC rulemaking regarding decommissioning, for evaluation of the reasonableness of planned decommissioning actions, and for determining if adequate funds are reserved by the licensees for decommissioning of their large irradiator or sealed source facilities. Another purpose of this study is to provide background and information to assist licensees in planning and carrying out the decommissioning of their sealed radioactive sources and respective facilities.

**NUREG/CR-6309:** SCIENTIFIC DESIGN OF PURDUE UNIVERSITY MULTI-DIMENSIONAL INTEGRAL TEST ASSEMBLY (PUMA) FOR GE SBWR. ISHII,M.; REVANKAR,S.T.; DOWLATI,R.; et al. Purdue Univ., West Lafayette, IN. April 1996. 279pp. 9605130147. PU-NE 94/1. 88238:001.

The scientific design of the scaled facility (PUMA) has been carried out under the project "Confirmatory Integral System Testing for GE SBWR Design". The design was based on the three level scaling method developed for this task. The first level of scaling is based on the integral response function. The second level scaling is for the boundary flow of mass and energy between components. The third level of scaling is focused on the key local phenomena and constitutive relations. The facility has 1/4 height and 1/100 area ratio scaling. This corresponds to a volume scale of 1/400 and power scaling of 1/200. The time will run twice as fast in the model as predicted by the present scaling method. The PUMA is scaled for full pressure and is intended to operate at and below 150 psia following scram. The facility models all the major components of SBWR.

**NUREG/CR-6314:** QUALITY ASSURANCE INSPECTIONS FOR SHIPPING AND STORAGE CONTAINERS. STROMBERG,H.M.; ROBERTS,G.D.; BRYCE,J.H. Idaho National Engineering Laboratory. April 1996. 77pp. 9605310158. INEL-95/0061. 88395:119.

This document is a guide for conducting quality assurance inspections of transportation packaging and dry spent fuel storage system suppliers. This document is used during an inspection to determine regulatory compliance with Title 10 of the Code of Federal Regulations, Part 71, Subpart H; Title 10 of the Code of Federal Regulations, Part 72, Subpart G; and Title 10 of the Code of Federal Regulations, Part 21, and quality assurance program commitments. The guidance provides a framework for transportation packaging and dry spent fuel storage system inspections. Inspectors are provided with the flexibility to adapt the methods and concepts to meet the inspection requirements for the particular facility. This guide was developed to provide a structured and consistent approach for inspections. The method separates each performance element into several areas for inspection and identifies guidelines, based on regulatory requirements, to qualitatively evaluate each area. This document was also developed to serve as a field manual to facilitate the quality assurance inspection activities.

**NUREG/CR-6317:** NUMERICAL INVESTIGATION OF 3-D CONSTRAINT EFFECTS ON BRITTLE FRACTURE IN SE(B) AND C(T) SPECIMENS. NEVALAINEN,M. Technical Research Centre of Finland (VTT). DODDS,R.H. Illinois, Univ. of, Urbana, IL. July 1996. 56pp. 9608230213. 89455:100.

This investigation employs 3-D nonlinear finite element analyses to conduct an extensive parametric evaluation of crack front stress triaxiality for deep notch SE(B) and C(T) specimens and shallow notch SE(B) specimens, with and without side grooves. Crack front conditions are characterized in terms of J-Q trajectories and the constraint scaling model for cleavage fracture toughness proposed previously by Dodds and Anderson. The 3-D computational results imply that a significantly less strict size/deformation limits indicated by previous plane-strain computations, relative to the limits indicated by previous plane strain computations, is needed to maintain small-scale yielding conditions at fracture by a stress-controlled cleavage mechanism in deep notch SE(B) and C(T) specimens. Additional new results made available from the 3-D analyses also include revised $\eta$-plastic factors for use in experimental studies to convert measured work quantities to thickness average and maximum (local) J-values over the crack front.

**NUREG/CR-6332:** MICROSTRUCTURAL CHARACTERIZATION OF SELECTED AEA/UCSB MODEL FECUMN ALLOYS. RICE,P.M.; STOLLER,R.E. Oak Ridge National Laboratory. June 1996. 63pp. 9607090195. ORNL/TM-12980. 88958:266.

A set of 22 model ferritic alloys was purchased as part of a collaborative research program by the AEA Harwell Laboratory and the University of California at Santa Barbara. Nine of these alloys were selected by the Oak Ridge National Laboratory for use in a series of ion irradiation experiments investigating dispersed barrier hardening. These nine alloys contain varying amounts of copper, manganese, titanium, carbon, and nitrogen. The alloys have been characterized by transmission electron microscopy in the as-received condition to provide a baseline for comparison with the irradiated specimens. A description of the microstructural observations is provided for future reference. This summary focuses on the type and size distributions of the precipitates present; grain size and dislocation measurements are also included.

**NUREG/CR-6336:** AGING ASSESSMENT OF LARGE ELECTRIC MOTORS IN NUCLEAR POWER PLANTS. VILLARAN,M.; SUBUDHI,M. Brookhaven National Laboratory. March 1996. 136pp. 9603260290. BNL-NUREG-52460. 87640:106.

Large electric motors serve as the prime movers to drive high capacity pumps, fans, compressors, and generators in a variety of nuclear plant systems. This study examined the stressors that cause degradation and aging in large electric motors operating in various plant locations and environments. The operating history of these machines in nuclear plant service was studied by review and analysis of failure reports in the NPRDS and LER databases. This was supplemented by a review of motor designs, and their nuclear and balance of plant applications, in order to characterize the failure mechanisms that cause degradation, aging, and failure in large electric motors. A generic failure modes and effects analysis for large squirrel cage induction motors was performed to identify the degradation and aging mechanisms affecting various components of these large motors, the failure modes that result, and their effects upon the function of the motor. The effects of large motor failures upon the systems in which they are operating, and on the plant as a whole, were analyzed from failure reports in the databases. The effectiveness of the industry's large motor maintenance programs was assessed based upon the failure reports in the databases and reviews of plant maintenance procedures and programs.

**NUREG/CR-6337:** SUMMARY OF RESULTS FROM THE IPIRG-2 ROUND-ROBIN ANALYSES. RAHMAN,S.; OLSON,R.; ROSENFIELD,A.; et al. Battelle Memorial Institute, Columbus Laboratories. February 1996. 182pp. 9602290264. BMI-2186. 87295:113.

This report presents a summary of the results from three one-day international round-robin workshops which were organized by Battelle in conjunction with the Second International Piping Integrity Research Group (IPIRG-2) Program. The objective of these workshops was to develop a consensus in handling difficult analytical problems in leak-before-break and pipe flaw evaluations. The workshops, which were held August 5, 1993, March 4, 1994, and October 21, 1994 at Columbus, Ohio, involved various technical presentations on the related research efforts by the IPIRG-2 member organizations and solutions to several round-robin problems. Following review by the IPIRG-2 members, four sets of round-robin problems were developed. They involved: (1) evaluations of fracture properties and pipe loads, (2) crack-opening and leak-rate evaluations, (3) dynamic analysis of cracked pipes, and (4) fracture evaluations of elbows. A total of 18 organizations from the United States, Japan, Korea, and Europe solved these round-robin problems. The analysis techniques employed by the participants included both finite element and engineering methods. Based on the results from these analyses, several important observations were made concerning the predictive capability of the current fracture-mechanics and thermal-hydraulics models for their applications in nuclear piping and piping welds.

**NUREG/CR-6338:** RESOLUTION OF THE DIRECT CONTAINMENT HEATING ISSUE FOR ALL WESTINGHOUSE PLANTS WITH LARGE DRY CONTAINMENTS OR SUBATMOSPHERIC CONTAINMENTS. PILCH,M.M.; ALLEN,M.D.; KLAMERUS,E.W. Sandia National Laboratories. February 1996. 241pp. 9603260307. SAND95-2381. 87624:001.

This report uses the scenarios described in NUREG/CR-6075 and NUREG/CR-6075, Supplement 1, to address the direct containment heating (DCH) issue for all Westinghouse plants with large dry or subatmospheric containments. DCH is considered resolved if the conditional containment failure probability (CCFP) is less than 0.1. Loads versus strength evaluations of the CCFP were performed for each plant using plant-specific information. The DCH issue is considered resolved for a plant if a screening phase results in a CCFP less than 0.01, which is more stringent than the overall success criterion. If the screening phase CCFP for a plant is greater than 0.01, then refined containment loads evaluations must be performed and/or the probability of high pressure at vessel breach must be analyzed. These analyses could be used separately or could be integrated together to re-calculate the CCFP for an individual plant to reduce the CCFP to meet the overall success criterion of less than 0.1. The CCFPs for all of the Westinghouse plants with dry containments were less than 0.01 at the screening phase, and thus, the DCH issue is resolved for these plants based on containment loads alone. No additional analyses are required.

**NUREG/CR-6335:** AGING ASSESSMENT OF WESTINGHOUSE PWR AND GENERAL ELECTRIC BWR CONTAINMENT ISOLATION FUNCTIONS. LEE,B.S.; TRAVIS,R.; GROVE,E.; et al. Brookhaven National Laboratory. March 1996. 154pp. 9604020321. BNL-NUREG-52462. 87715:001.

A study was performed to assess the effects of aging on the Containment Isolation (CI) functions of Westinghouse Pressurized Water Reactors and General Electric Boiling Water Reactors. This study is part of the Nuclear Plant Aging Research (NPAR) program, sponsored by the U.S. Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging progress and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age. The failure data from national databases, Nuclear Plant Reliability Data System (NPRDS) and Licensee Event Reports (LERs), as well as plant specific data were reviewed and analyzed to understand the effects of aging on the CI functions. This study provided information on the effects of aging on component failure frequency, failure modes, and failure causes. Current inspection, surveillance, and monitoring practices were also reviewed.

**NUREG/CR-6340:** AGING ASSESSMENT OF SURGE PROTECTIVE DEVICES IN NUCLEAR POWER PLANTS. DAVIS,J.F.; SUBUDHI,M. Brookhaven National Laboratory. CARROLL,D.P. Florida, Univ. of, Gainesville, FL. January 1996. 179pp. 9603040082. BNL-NUREG-52463. 87317:001.

A study has been performed that assesses the effects of aging on the performance and availability of surge protective devices (SPDs), commonly called surge arresters and surge suppressors, used in electrical power and control systems in U.S. commercial nuclear power plants. This study is one of a number of studies performed under the Nuclear Plant Aging Research (NPAR) program. One of the many purposes of that program is to provide a technical basis for the identification and evaluation of degradation caused by age. Although surge protective devices have not been classified as safety-related, they are risk important because they can minimize the initiating event frequencies associated with loss of offsite power (LOOP) and reactor trip. Conversely, their failure due to age, or other causes, might increase the initiating event frequencies. Because of their importance, especially those in (lightning) high flash density regions to the U.S., the proper application and coordination of high voltage and low voltage SPDs are important in ensuring that overvoltage transients will not increase plant risk.

**NUREG/CR-6341:** MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE.Final Report. ROGERS,R.D.; HAMILTON,M.A.; VEEH,R.H.; et al. Idaho National Engineering Laboratory. June 1996. 116pp. 9607090197. INEL-95/0215. 88958:001.

The Nuclear Regulatory Commission stipulates in 10 CFR 61 that disposed low-level radioactive waste (LLW) be stabilized. To provide guidance to disposal vendors and nuclear station waste generators for implementing those requirements, the NRC developed the Technical Position on Waste Form, Revision 1. That document details a specified set of recommended testing procedures and criteria, including several tests for determining the biodegradation properties of waste forms. Information has been presented by a number of researchers, which indicated that those tests may be inappropriate for examining microbial degradation of cement-solidified LLW. Cement has been widely used to solidify LLW; however, the resulting waste forms are sometimes susceptible to failure due to the actions of waste constituents, stress, and environment. The purpose of this research program was to develop modified microbial degradation test procedures that would be more appropriate than the existing procedures for evaluation of the effects of microbiologically influenced chemical attack on cement-solidified LLW. The procedures that have been developed in this work are presented and discussed. Groups of microorganisms indigenous to LLW disposal sites were employed that can metabolically convert organic and inorganic substrates into organic and mineral acids. Such acids aggressively react with cement and can ultimately lead to structural failure. Results on the application of mechanisms inherent in microbially influenced degradation of cement-based material are the focus of this final report. Data-validated evidence of the potential for microbially influenced deterioration of cement-solidified LLW and subsequent release of radionuclides developed during this study are presented.

**NUREG/CR-6344:** REAL-TIME 3-D SAFT-UT SYSTEM EVALUATION AND VALIDATION. DOCTOR,S.R.; SCHUSTER,G.J.; REID,L.D.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. September 1996. 277pp. 9612240090. PNNL-10571. 91198:001.

SAFT-UT technology is shown to provide significant enhancements to the inspection of materials used in U.S. nuclear power plants. This report provides guidelines for the implementation of SAFT-UT technology and shows the results from its application. An overview of the development of SAFT-UT is provided so that the reader may become familiar with the technology. Then the basic fundamentals are presented with an extensive list of references. A comprehensive operating procedure, which is used in conjunction with the SAFT-UT field system developed by Pacific Northwest National Laboratory (PNL), provides the recipe for both SAFT data acquisition and analysis. The specification for the hardware implementation is provided for the SAFT-UT system along with a description of the subsequent developments and improvements. One development of technical interest is the SAFT real-time processor. Performance of the real-time processor is impressive and comparison is made of this dedicated parallel processor to a conventional computer and to the newer high-speed computer architectures designed for image processing. Descriptions of other improvements, including a robotic scanner, are provided.

**NUREG/CR-6345:** RADIATION DOSE ESTIMATES FOR RADIOPHARMACEUTICALS. STABIN,M.G.; STUBBS,J.B.; TOOHEY,R.E. Oak Ridge Associated Universities. April 1996. 92pp. 9605290091. 88375:133.

Tables of radiation dose estimates based on the Cristy-Eckerman adult male phantom are provided for a number of radiopharmaceuticals commonly used in nuclear medicine. Radiation dose estimates are listed for all major source organs, and several other organs of interest. The dose estimates were calculated using the MIRD Technique as implemented in the MIRDOSE3 computer code, developed by the Oak Ridge Institute for Science and Education, Radiation Internal Dose Information Center. In this code, residence times for source organs are used with decay data from the MIRD Radionuclide Data and Decay Schemes to produce estimates of radiation dose to organs of standardized phantoms representing individuals of different ages.

**NUREG/CR-6346:** HYDROLOGIC EVALUATION METHODOLOGY FOR ESTIMATING WATER MOVEMENT THROUGH THE UNSATURATED ZONE AT COMMERCIAL LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITES. MEYER,P.D.; ROCKHOLD,M.L.; NICHOLS,W.E.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. January 1996. 134pp. 96030103?8. PNL-10843. 87311:068.

This report identifies key technical issues related to hydrologic assessment of water flow in the unsaturated zone at low-level radioactive waste (LLW) disposal facilities. In addition, a methodology for incorporating these issues in the performance assessment of proposed LLW disposal facilities is identified and evaluated. The issues discussed fall into four areas: estimating the water balance at a site (i.e., infiltration, runoff, water storage, evapotranspiration, and recharge); analyzing the hydrologic performance of engineered components of a facility; evaluating the application of models to the prediction of facility performance; and estimating the uncertainty in predicted facility performance. To illustrate the application of the methodology, two examples are presented. The first example is of a below ground vault located in a humid environment. The second example looks at a shallow land burial facility located in an arid environment. The examples utilize actual site-specific data and realistic facility designs. The two examples illustrate the issues unique to humid and arid sites as well as the issues common to all LLW sites. Strategies for addressing the analytical difficulties arising in any complex hydrologic evaluation of the unsaturated zone are demonstrated.

**NUREG/CR-6350:** A TECHNIQUE FOR HUMAN ERROR ANALYSIS (ATHEANA).Technical Basis And Methodology Description. COOPER,S.E. Science Applications International Corp. (formerly Science Applications, Inc.). RAMEY-SMITH,A. Division of Systems Technology (Post 941217). WREATHALL,J.; et al. Affiliation Not Assigned. May 1996. 111pp. 9607090121. BNL-NUREG-52467. 88956:001.

Probabilistic risk assessment (PRA) has become an important tool in the nuclear power industry, both for the Nuclear Regulatory Commission (NRC) and the operating utilities. Human reliability analysis (HRA) is a critical element of PRA; however, limitations in the analysis of human actions in PRAs have long been recognized as a constraint when using PRA. A multidisciplinary HRA framework has been developed with the objective of providing a structured approach for analyzing operating experience and understanding nuclear plant safety, human error, and the underlying factors that affect them. The concepts of the framework have matured into a rudimentary working HRA method. A trial application of the method has demonstrated that it is possible to identify potentially significant human failure events from actual operating experience which are not generally included in current PRAs, as well as to identify associated performance shaping factors and plant conditions that have an observable impact on the frequency of core damage. A general process was developed, albeit in preliminary form, that addresses the iterative steps of defining human failure events and estimating their probabilities using search schemes. Additionally, a knowledge-base was developed which describes the links between performance shaping factors and resulting unsafe actions.

**NUREG/CR-6353:** COMMENTS RECEIVED ON PROPOSED RULE ON RADIOLOGICAL CRITERIA FOR DECOMMISSIONING AND RELATED DOCUMENTS. PAGE,G.; CAPLIN,J.; SMITH,D.; et al. Advanced Systems Technology, Inc. March 1996. 166pp. 9604020352. 87712:160.

The Nuclear Regulatory Commission (NRC) is conducting an enhanced participatory rulemaking to establish radiological criteria for the decommissioning of NRC licensed facilities. As a part of this action, the Commission published in the Federal Register (59 FR 43200), on August 22, 1994, a proposed rule on radiological criteria for decommissioning, soliciting comments both on the rule as proposed and on certain specific items as identified in its supplementary statement of considerations. A draft Generic Environmental Impact Statement (GEIS) in support of the rule, also published in August 1994 as NUREG-1496, along with its Appendix A (NUREG-1501), were also made available for comment. A staff working draft on regulatory guidance (NUREG-1500) was also made available. This report summarized the 1,309 comments on the proposed rule and supplementary items and the 311 comments on the GEIS as excerpted from 101 docketed letters received as solicited in the Federal Register notice. Comments from two NRC/Agreement-States meetings are also summarized.

**NUREG/CR-6357:** EVALUATION AND FIELD VALIDATION OF EDDY-CURRENT ARRAY PROBES FOR STEAM GENERATOR TUBE INSPECTION. DODD,C.V.; PATE,J.R. Oak Ridge National Laboratory. July 1996. 86pp. 9608210247. ORNL/TM-13213. 89424:189.

The objective of the Improved Eddy-Current ISI for Steam Generator Tubing program is to upgrade and validate eddy-current inspections, including probes, instrumentation, and data processing techniques for inservice inspection of new, used, and repaired steam generator tubes; to improve defect detection, classification and characterization as affected by diameter and thickness variations, denting, probe wobble, tube sheet, tube supports, copper and sludge deposits, even when defect types and other variables occur in combination; to transfer this advanced technology to NRC's mobile NDE laboratory and staff. This report describes the design of specialized high-speed 16-coil eddy-current array probes. Both pancake and rejection coils are considered. Test results from inspections using the probes in working steam generators are given. Computer programs developed for probe calculations are also supplied.

**NUREG/CR-6360:** SUMMARY OF AIR PERMEABILITY DATA FROM SINGLE-HOLE INJECTION TESTS IN UNSATURATED FRACTURED TUFFS AT THE APACHE LEAP RESEARCH SITE: RESULTS OF STEADY-STATE TEST INTERPRETATION. GUZMAN,A.G.; GEDDIS,A.M.; HENRICH,M.J.; et al. Arizona, Univ. of, Tucson, AZ. March 1996. 175pp. 9604230343. 87997:011.

This document summarizes air permeability estimates obtained from single hole pneumatic injection tests in unsaturated fractured tuffs at the Covered Borehole Site within the larger Apache Leap Research Site. Only permeability estimates obtained from a steady state interpretation of relatively stable pressure and flow rate data are included. Tests were conducted in five boreholes inclined at 45 degrees to the horizontal, and one vertical borehole. Five of the boreholes are 30 m long, one has length of 45 m. Over 180 borehole segments were tested between packers set 1 m apart. Additional tests were conducted in segments of lengths 0.5, 2.0 and 3.0 m in one borehole, and 2.0 m in another borehole, bringing the total number of tests to over 270. Tests were conducted by maintaining a constant injection rate until air pressure became relatively stable and remained so for some time. The injection rate was then incremented by a constant value and the procedure repeated. Three or more such incremental steps were conducted in each borehole segment while recording the air injection rate, pressure, temperature, and relative humidity. A description of field operating procedures used to insure compliance with QA/QC requirements is included.

**NUREG/CR-6365:** STEAM GENERATOR TUBE FAILURES. MACDONALD,P.E.; SHAH,V.N.; WARD,L.W. Idaho National Engineering Laboratory. April 1996. 303pp. 9607100298. INEL-95/0383. 88955:001.

A review and summary of the available information on steam generator tubing failures and the impact of these failures on plant safety is presented. The following topics are covered: pressurized water reactor (PWR), Canadian deuterium uranium (CANDU) reactor, and Russian water moderated, watercooled energy reactor (VVER) steam generator degradation, PWR steam generator tube ruptures, the thermal-hydraulic response of a PWR plant with a faulted steam generator, the risk significance of steam generator tube rupture accidents, tubing inspection requirements and fitness-for-service criteria in various countries, and defect detection reliability. Steam generator tube damage is caused by many diverse degradation mechanisms, some of which are difficult to detect and predict. The frequency of steam generator tube ruptures can be significantly reduced through appropriate and timely inspections and repairs or removal from service. However, a continuing issue has been exactly what constitutes an appropriate and timely inspection and which degraded tubes are still fit for service. There have been many different approaches to this problem throughout the world. Although steam generator tube ruptures are small contributors to the total core damage frequency calculated in probabilistic risk assessments, they are risk significant because the radionuclides are likely to bypass the reactor containment building.

**NUREG/CR-6366:** POLYRES: A POLYGON-BASED RICHARDS EQUATION SOLVER. HILLS,R.G. New Mexico State Univ., Las Cruces, NM. MEYER,P.D.; ROCKHOLD,M.L. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1995. 68pp. 9605220418. PNL-10709. 88314:235.

This document describes the theory, implementation, and use of a software package designed to solve the transient, two-dimensional, Richards equation for water flow in unsaturated-saturated soils. This package was specifically designed to model complex geometries with minimal input from the user. The spatial variation of the hydraulic properties can be defined across individual polygon-shaped subdomains, called objects. These objects combine to form a polygon-shaped model domain. Each object can have its own distribution of hydraulic parameters. The resulting model domain and polygon-shaped internal objects are mapped onto a rectangular, finite-volume, computational grid by a preprocessor. This allows the user to specify model geometry independently of the underlying grid and greatly simplifies user input for complex geometries. In addition, this approach significantly reduces the computational requirements since complex geometries are actually modeled on a rectangular grid. This results in well-structured, finite difference-like systems of equations that require minimal storage and are very efficient to solve. The documentation for this software package includes a user's manual, a detailed description of the underlying theory, and a detailed discussion of program flow. Several example problems are presented that show the use and features of the software package. The water flow predictions for several of these example problems are compared to those of another algorithm (Kirkland et al., 1992) to test for prediction equivalency. The computer code described in this document is available from the Energy Science and Technology Software Center, P.O. Box 1020, Oak Ridge, TN 37831-1020.

**NUREG/CR-6367:** EXPERIMENTAL STUDY OF HEAD LOSS AND FILTRATION FOR LOCA DEBRIS. RAO,D.V.; SOUTO,F.J. Science & Engineering Associates, Inc. February 1996. 153pp. 9603190128. SEA95-554-06A:8. 87520:130.

A series of controlled experiments were conducted to obtain head loss and filtration characteristics of debris beds formed of NUKON(TM) fibrous fragments, and obtain data to validate the semi-theoretical head loss model developed in NUREG/CR-6224. A thermally insulated closed-loop test set-up was used to conduct experiments using beds formed of fibers only and fibers intermixed with particulate debris. A total of three particulate mixes were used to simulate the particulate debris. The head loss data were obtained for theoretical fiber bed thicknesses of 0.125 inch to 4.0 inches; approach velocities of 0.15 to 1.5 ft/s; temperatures of 75 degrees F and 125 degrees F; and sludge-to-fiber nominal concentration ratios of 0 to 60. Concentration measurements obtained during the first flushing cycle were used to estimate the filtration efficiencies of the debris beds. For test conditions where the beds are fairly uniform, the head loss data were predictable within an acceptable accuracy range by the semi-theoretical model. The model was equally applicable for both pure fiber beds and the mixed beds. Typically the model over-predicted the head losses for very thin beds and for thin beds at high sludge-to-fiber mass ratios. This is attributable to the non-uniformity of such debris beds. In this range the correlation can be interpreted to provide upper bound estimates of head loss.

**NUREG/CR-6374:** WHOLE-BODY EFFECTIVE HALF-LIVES FOR RADIOLABELED ANTIBODIES AND RELATED ISSUES. KAURIN,D.G.; CARSTEN,A.L.; et al. Brookhaven National Laboratory. BARBER,D.E. Minnesota, Univ. of, Minneapolis, MN. August 1996. 176pp. 9609100277. BNL-NUREG-52476. 89609:061.

Radiolabeled antibodies (RABs) are being developed and used in medical imaging and therapy in rapidly increasing numbers. There are concerns about the radiation exposure of caregivers and the general public from treated patents. The magnitude of this hazard is closely related to the RABs' whole-body effective half-life (T(e)). Data on whole-body effective half-lives were calculated from external dose rates obtained from attending physicians and radiation safety officers at participating institutions. Calculations of T(e)s were made using exponential regression analyses of data from patents receiving single and multiple administrations. These data were analyzed on the basis of age, sex, isotope label, radiation energy, antibody type, disease treated, method of administration, and number of administrations. The effective half-life in the blood did not correlate with the T(e). The values of T(e) varied by a factor of two for patients taking the same RAB. A single exponential clearance rate, compared with a bi-exponential clearance rate, provides an adequate fit for 95% of the data sets tested.

**NUREG/CR-6375:** STRAIN RATE AND INERTIAL EFFECTS ON IMPACT LOADED SINGLE-EDGE NOTCH BEND SPECIMENS. VARGAS,P.M.; DODDS,R.H. Illinois, Univ. of, Urbana, IL. June 1996. 27pp. 9608060180. 89258:203.

When the severity of impact loads is sufficient to produce large inelastic deformations, the assessment of crack-tip conditions must include the effects of plasticity, strain rate and inertia. This work examines the interaction of impact loading, inelastic material deformation and rate sensitivity with the goal of improving the interpretation of ductile fracture toughness values measured under dynamic loading. Three-dimensional, nonlinear dynamic analyses are performed for SE(B) fracture specimens ($a/W = 0.5, 0.15, 0.0725$) subjected to impact loading. Loading rates obtained in conventional drop tower tests (impact load-line velocities of $\approx 6$ m/sec) are applied in the analyses. Strains at key locations on the specimens and the support reactions (applied load) are extracted from the analyses to assess the accuracy of static formulas commonly used to estimate applied J values. Inertial effects on the applied J are quantified by examining the acceleration component of J evaluated through a domain integral procedure.

**NUREG/CR-6383:** CORROSION FATIGUE OF ALLOYS 600 AND 690 IN SIMULATED LWR ENVIRONMENTS. RUTHER,W.E.; SOPPET,W.K.; KASSNER,T.F. Argonne National Laboratory. April 1996. 56pp. 9604230394. ANL-95/37. 87993:276.

Crack growth data were obtained on fracture-mechanics specimens of Alloy 600 and 690 to investigate environmentally assisted cracking (EAC) in simulated boiling water reactor and pressurized water reactor environments at 289 and 320 degrees C. Preliminary information was obtained on the effect of temperature, load ratio, stress intensity K, and dissolved-oxygen and -hydrogen concentrations of the water on EAC. Specimens of Type 316NG and sensitized Type 304 stainless steel (SS) were included in several of the experiments to assess the behavior of these materials and Alloy 600 under the same water chemistry and loading conditions. The experimental data are compared with predictions from an Argonne National Laboratory (ANL) model for crack growth rates (CGRs) of SSs in water and the ASME Code Section XI correlation for CGRs in air at the K(max) and load-ratio values in the various tests. The data for all of the materials were bounded by ANL model predictions and the ASME Section XI "air line."

**NUREG/CR-6384 V01:** LITERATURE REVIEW OF ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRIC CABLES.Summary Of Past Work. SUBUDHI,M. Brookhaven National Laboratory. April 1996. 300pp. 9605130112. BNL-NUREG-52480. 88219:001.

This report summarizes the findings from a review of published documents dealing with research on the environmental qualification of safety-related electric cables used in nuclear power plants. Simulations of accelerated aging and accident conditions are important considerations in qualifying the cables. Significant research in these two areas has been performed in the United States and abroad. The results from studies in France, Germany, and Japan are described in this report. In recent years, the development of methods to monitor the condition of cables has received special attention. Tests involving chemical and physical examination of cable's insulation and jacket materials, and electrical measurements of the insulation properties of cables are discussed. Although there have been significant advances in many areas, there is no single method which can provide the necessary information about the condition of a cable currently in service. However, it is possible that further research may identify a combination of several methods that can adequately characterize the cable's condition.

**NUREG/CR-6384 V02:** LITERATURE REVIEW OF ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRIC CABLES.Literature Analysis And Appendices. LOFARO,R.; BOWERMAN,B.; CARBONARO,J.; et al. Brookhaven National Laboratory. April 1996. 146pp. 9605130115. BNL-NUREG-52480. 88265:001.

In support of the U.S. NRC Environmental Qualification (EQ) Research Program, a literature review was performed to identify past relevant work that could be used to help fully or partially resolve issues of interest related to the qualification of low-voltage electric cable. A summary of the literature reviewed is documented in Volume I of this report. In this, Volume 2 of the report, dossiers are presented which document the issues selected for investigation in this program, along with recommendations for future work to resolve the issues, when necessary. The dossiers are based on an analysis of the literature reviewed, as well as expert opinions. This analysis includes a critical review of the information available from past and ongoing work in thirteen specific areas related to EQ. The analysis for each area focuses on one or more questions which must be answered to consider a particular issue resolved. Results of the analysis are presented, along with recommendations for future work. The analysis is documented in the form of a dossier for each of the areas analyzed.

**NUREG/CR-6388:** SEISMIC RESPONSE OF ROCK JOINTS AND JOINTED ROCK MASS. GHOSH,A.; HSIUNG,S.M.; CHOWDHURY,A.H. Center for Nuclear Waste Regulatory Analyses. June 1996. 120pp. 9608060191. CNWRA 95-013. 89266:191.

Two key technical uncertainties (KTU) that can potentially pose a high risk of noncompliance with the performance objectives of 10 CFR Part 60 are the prediction of (i) thermal-mechanical effects on stability of emplacement drifts and the engineered barrier system (EBS), and (ii) thermal-mechanical-hydrological effects on the host rock surrounding the EBS. This final report summarizes the research activities concerned with the repetitive seismic load aspect of both these KTUs. This research project has the dual focus of (i) understanding the key parameters affecting repository performance under repeated seismic loading, and (ii) evaluating current capabilities for calculating such effects. Laboratory experiments on Apache Leap tuff joints using cyclic pseudostatic and dynamic loads indicate that the shear strength in the reverse direction of shearing is less than that in the forward direction. But the reverse shear strength predicted by UDEC (versions 1.82 and 1.83) is found to be inconsistent with these findings. Thus, a new joint model is desirable. Field experiments at Lucky Friday Mine, scale-model experiments conducted at CNWRA, and information in the literature show that for excavations subjected to repetitive seismic motions, accumulation of shear displacement along the joints is the primary mode of deformation for the rock mass. However, the currently available seismic design procedure for underground excavations is based on the probable peak particle motion concept and does not explicitly take into account of either the time history of individual events or the effects of repetitive seismic events. An adequate seismic design methodology is desirable. The results of this research project will be used to: conduct prelicensing reviews, provide guidance to DOE, develop CDMs, support IPA, and provide basis for developing seismic design methodology.

**NUREG/CR-6392:** THE EFFECTS OF AGING ON COMPRESSIVE STRENGTH OF LOW-LEVEL RADIOACTIVE WASTE FORMS. MCCONNELL,J.W.; NEILSON,R.M. Idaho National Engineering Laboratory. June 1996. 47pp. 9607090141. INEL-95/0506. 88956:233.

The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission (NRC), is (a) studying the degradation effects in organic ion-exchange resins caused by radiation, (b) examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Form to meet the requirements of 10 CFR 61 using solidified ion-exchange resins, (c) obtaining performance information on solidified ion-exchange resins in a disposal environment, and (d) determining the condition of liners used to dispose ion-exchange resins. Compressive tests were performed periodically over a 12-year period as part of the Technical Position testing. Results of that compressive testing are presented and discussed. During the study, both portland type I-II cement and Dow vinyl ester-styrene waste form samples were tested. This testing was designed to examine the effects of aging caused by self-irradiation on the compressive strength of the waste forms. Also presented is a brief summary of the results of waste form characterization, which had been conducted in 1986, using tests recommended in the Technical Position on Waste Form. The aging test results are compared to the results of those earlier tests.

**NUREG/CR-6396:** EXAMPLES,CLARIFICATIONS, AND GUIDANCE ON PREPARING REQUESTS FOR RELIEF FROM PUMP AND VALVE INSERVICE TESTING REQUIREMENTS. RANSOM,C.B.; HARTLEY,R.S. Idaho National Engineering Laboratory. February 1996. 186pp. 9603010300. INEL-95/0512. 87307:131.

In this report, the Idaho National Engineering Laboratory reviewers discuss issues related to requests for relief from the American Society of Mechanical Engineers code requirements for inservice testing (IST) of safety-related pumps and valves at commercial nuclear power plants. This report compiles information and examples that may be useful to licensees in developing relief requests submitted to U.S. Nuclear Regulatory Commission (NRC) for their consideration and provides insights and recommendations on related IST issues. The report also gives specific guidance on relief requests acceptable and not acceptable to the NRC and advises licensees in the use of this information for application at their facilities.

**NUREG/CR-6401:** FAULTING IN THE YUCCA MOUNTAIN REGION.Critical Review And Analyses of Tectonic Data From The Central Basin And Range. FERRILL,D.A.; STIREWALT,G.L.; HENDERSON,D.B.; et al. Center for Nuclear Waste Regulatory Analyses. March 1996. 108pp. 9606030250. CNWRA 95-017. 88421:231.

Yucca Mountain, Nevada, has been proposed as the potential site for a high-level waste (HLW) repository. The tectonic setting of Yucca Mountain presents several potential hazards for a proposed repository, such as potential for earthquake seismicity, fault disruption, basaltic volcanism, magma channeling along pre-existing faults, and faults and fractures that may serve as barriers or conduits for ground water flow. Characterization of geologic structures and tectonic processes will be necessary to assess compliance with regulatory requirements for the proposed HLW repository. In this report, we specifically investigate fault slip, seismicity, contemporary stain, and fault-slip potential in the Yucca Mountain region with regard to Key Technical Uncertainties outlined in the License Application Review Plan (Sections 3.2.1.5 through 3.2.1.9 and 3.2.2.8). These investigations center on (i) alternative methods of determining the slip history of the Bare Mountain Fault, (ii) cluster analysis of historic earthquakes, (iii) crustal strain determinations from Global Positioning System measurements, and (iv) three-dimensional slip-tendency analysis. The goal of this work is to assess uncertainties associated with neotectonic data sets critical to the Nuclear Regulatory Commission and the Center for Nuclear Waste Regulatory Analyses' ability to provide prelicensing guidance and perform license application review with respect to the proposed HLW repository at Yucca Mountain.

**NUREG/CR-6406:** ENVIRONMENTAL TESTING OF AN EXPERIMENTAL DIGITAL SAFETY CHANNEL. KORSAH,K.; WILSON,T.L.; et al. Oak Ridge National Laboratory. TANAKA,T.J. Sandia National Laboratories. September 1996. 146pp. 9612030282. ORNL/TM-13122. 91000:001.

This document presents the results of environmental stress tests performed on an experimental digital safety channel (EDSC) assembled at the Oak Ridge National Laboratory (ORNL) as part of the NRC-sponsored "Qualification of Advanced Instrumentation and Controls (I&C) System" program. The objective of this study is to investigate failure modes and vulnerabilities of microprocessor-based technologies when subjected to environmental stressors. The study contributes to the technical basis for environmental qualification of safety-related digital I&C systems for nuclear power plants. The EDSC employs technologies and digital subsystems representative of those proposed for use in advanced light-water reactors (ALWRs) or for retrofits in existing plants. It was subjected to selected stressors that are a potential risk to digital equipment in a mild environment. The selected stressors; were electromagnetic and radio-frequency interference (EMI/RFI), temperature, humidity, and smoke exposure. The stressors were applied over ranges that were considerably higher than what the channel is likely to experience in a normal nuclear power plant environment. Stressor-induced errors were logged so that failure modes that are characteristic of the technologies employed could be identified.

**NUREG/CR-6407:** CLASSIFICATION OF TRANSPORTATION PACKAGING AND DRY SPENT FUEL STORAGE SYSTEM COMPONENTS ACCORDING TO IMPORTANCE TO SAFETY. MCCONNELL,J.W.; AYERS,A.L.; TYACKE,M.J. Idaho National Engineering Laboratory. February 1996. 59pp. 9603190098. INEL-95/0551. 87522:099.

This report provides a graded approach for classification of components used in transportation packaging and dry spent fuel storage systems. This approach provides a method for identifying the classification of components according to importance to safety within transportation packagings and dry spent fuel storage systems. Record retention requirements are discussed to identify the documentation necessary to validate that the individual components were fabricated in accordance with their assigned classification. A review of the existing regulations pertaining to transportation packagings and dry storage systems was performed to identify current requirements. The general types of transportation packagings and dry storage systems were identified. Discussions were held with suppliers and fabricators of packagings and storage systems to determine current practices. The methodology used in this report is based on Regulatory Guide 7.10, Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material. This report also includes a list of generic components for each of the general types of transportation packagings and spent fuel storage systems. The safety importance of each component is discussed, and a classification category is assigned.

**NUREG/CR-6411:** A GEOSTATISTICAL METHODOLOGY TO ASSESS THE ACCURACY OF UNSATURATED FLOW MODELS. SMOOT,J.L.; WILLIAMS,R.E. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1996. 70pp. 9607090193. PNL-10866. 88958:200.

The Pacific Northwest National Laboratory (PNNL) has developed a Hydrologic Evaluation Methodology (HEM) to assist the U.S. Nuclear Regulatory Commission in evaluating the potential that infiltrating meteoric water will produce leachate at commercial low-level radioactive waste disposal sites. Two key issues are raised in the HEM: 1) evaluation of mathematical models that predict facility performance, and 2) estimation of the uncertainty associated with these mathematical model predictions. The technical objective of this research is to adapt geostatistical tools commonly used for model parameter estimation to the problem of estimating the spatial distribution of the dependent variable to be calculated by the model. To fulfill this objective, a database describing the spatiotemporal movement of water injected into unsaturated sediments at the Hanford Site in Washington State was used to develop a new method for evaluating mathematical model predictions. Measured water content data were interpolated geostatisically to a 16 x 16 x 36 grid at several time intervals. Then a mathematical model was used to predict water content at the same grid locations at the selected times. Node-by-node comparison of the mathematical model predictions with the geostatistically interpolated values was conducted. The method facilitates a complete accounting and categorization of model error at every node. The comparison suggests that model results generally are within measurement error. The worst model error occurs in [illegible] lenses and is in excess of measurement error.

**NUREG/CR-6413:** ANALYSIS OF THE IRRADIATION DATA FOR A302B AND A533B CORRELATION MONITOR MATERIALS. WANG,J.A. Oak Ridge National Laboratory. April 1996. 318pp. 9605220308. ORNL/TM-13133. 88320:029.

The results of Charpy V-notch impact tests for A302B and A533B-1 Correlation Monitor Materials (CMM) listed in the surveillance power reactor data base (PR-EDB) and material test reactor data base (TR-EDB) are analyzed. The shift of the transition temperature at 30 ft-lb (T(30)) is considered as the primary measure of radiation embrittlement in this report. The hyperbolic tangent fitting model and uncertainty of the fitting parameters for Charpy impact tests are presented in this report. For the surveillance CMM data, the transition temperature shifts at 30 ft-lb ($\Delta$T(30)) generally follow the predictions provided by Revision 2 of Regulatory Guide 1.99 (R.G. 1.99). Difference in capsule temperatures is a likely explanation for large deviations from R.G. 1.99 predictions. Deviations from the R.G. 1.99 predictions are correlated to similar deviations for the accompanying materials in the same capsules, but large random fluctuations prevent precise quantitative determination. Significant scatter is noted in the surveillance data, some of which may be attributed to variations from one specimen set to another, or inherent in Charpy V-notch testing. In general, the embrittlement behavior of both the A302B and A533B-1 plate materials is similar. There is evidence for a fluence-rate effect in the CMM data irradiated in test reactors; thus its implication on power reactor surveillance programs deserves special attention.

**NUREG/CR-6415:** APPLICATIONS OF RELIABILITY DEGRADATION ANALYSIS. VESELY,W.E. Science Applications International Corp. (formerly Science Applications, Inc.). SAMANTA,P.K. Brookhaven National Laboratory. February 1996. 47pp. 9603260303. BNL-NUREG-52488. 87624:242.

Reliability degradation analysis is the analysis of the occurrences of degradations and the times of maintenance to determine their reliability and risk implications. A program is presented for applying reliability degradation analyses to maintenance data collected at nuclear power plants. As a specific part of the program, time trending of maintenance data is illustrated. Maintenance data on residual heat removal (RHR) pumps and service water (SW) pumps at selected boiling water reactor (BWR) plants are evaluated to show how trends in maintenance data, which generally do not involve failures, can be used to understand effectiveness of maintenance. These trends also are translated to specific impacts on pump unavailability and on core-damage frequency (assuming that the trends in failure rate are the same as those observed for degradation rate). The second application shows the use of reliability degradation analysis to quantitatively evaluate the effect of maintenance, i.e., the quantitative change in component unavailability when no maintenance is performed. Assessment of these impacts are important since they measure the reliability and risk impacts of maintenance and can be fed back to the maintenance program to improve its effectiveness.

**NUREG/CR-6419:** SOLUBILITY TESTING OF ACTINIDES ON BREATHING-ZONE AND AREA AIR SAMPLES. METZGER,R.L.; JESSOP,B.H.; MCDOWELL,B.L. Radiation Safety Engineering, Inc. February 1996. 55pp. 9603180378. 87509:248.

A solubility testing method for several common actinides has been developed with sufficient sensitivity to allow profiles to be determined from routine breathing zone and area air samples in the workplace. Air samples are covered with a clean filter to form a filter-sample-filter sandwich which is immersed in an extracellular lung serum simulant solution. The sample is moved to a fresh beaker of the lung fluid simulant each day for one week, and then weekly until the end of the 28-day test period. The soak solutions are wet ashed with nitric acid and hydrogen peroxide to destroy the organic components of the lung simulant prior to the extraction of the nuclides of interest directly into an extractive scintillator for subsequent counting on a photon-electron rejecting alpha liquid scintillation (PEARLS) spectrometer. Solvent extraction methods utilizing the extractive scintillators have been developed for the isotopes of uranium, plutonium, and curium. The procedures normally produce an isotopic recovery greater than 95% and have been used to develop solubility profiles for air samples with 40 pCi or less of U3O8. Profiles developed for U3O8 samples show good agreement with in vitro and in vivo tests performed by other investigators on samples from the same uranium mills.

**NUREG/CR-6421:** A PROPOSED ACCEPTANCE PROCESS FOR COMMERCIAL OFF-THE-SHELF (COTS) SOFTWARE IN REACTOR APPLICATIONS. PRECKSHOT,G.G.; SCOTT,J.A. Lawrence Livermore National Laboratory. March 1996. 107pp. 9604150345. UCRL-ID-122526. 87867:179.

This paper proposes a process for acceptance of commercial off-the-shelf- (COTS) software products for use in reactor systems important to safety. An initial set of four criteria establishes COTS software product identification and its safety category. Based on safety category, three sets of additional criteria, graded in rigor, are applied to approve (or disapprove) the product. These criteria fall roughly into three areas: product assurance, verification of safety function and safety impact, and examination of usage experience of the COTS product in circumstances similar to the proposed application. A report addressing the testing of existing software is included as an appendix.

**NUREG/CR-6422:** POWER EXCURSION ANALYSIS FOR HIGH BURNUP CORES. DIAMOND,D.J.; NEYMOTIN,L.; KOHUT,P. Brookhaven National Laboratory. February 1996. 78pp. 9603260310. BNL-NUREG-52491. 87640:242.

A study has been undertaken to determine the fuel enthalpy during a rod drop accident (RDA) and during two thermal-hydraulic transients in a boiling water reactor (BWR). The objective was to understand the consequences to high burnup fuel and the sources of uncertainty in the RDA calculations. The analysis was done with RAMONA-4B, a computer code that models the neutron kinetics throughout the core along with the thermal-hydraulics in the core, vessel, and steamline. The results showed that the calculated maximum fuel enthalpy in high burnup fuel will be affected by core design, initial conditions, and modeling assumptions. The important parameters in each of these categories are discussed in the report. An additional objective of this study was to identify BWR and pressurized water reactor (PWR) transients in which there is significant energy deposition. This determined which BWR transients were to be calculated as part of this study and which might be calculated if analysis were to be done for PWRs.

**NUREG/CR-6424:** REPORT ON AGING OF NUCLEAR POWER PLANT REINFORCED CONCRETE STRUCTURES. NAUS,D.J.; OLAND,C.B. Oak Ridge National Laboratory. ELLINGWOOD,B.R. Johns Hopkins Univ., Baltimore, MD. March 1996. 230pp. 9604230376. ORNL/TM-13148. 87974:001.

The Structural Aging Program provides the U.S. Nuclear Regulatory Commission with potential structural safety issues and acceptance criteria for continued service assessments of safety-related nuclear power plant concrete structures. The program was organized under four task areas - Program Management, Materials Property Data Base, Structural Component Assessment/Repair Technology, and Quantitative Methodology for Continued Service Determinations. Under these tasks, over 90 papers and reports were prepared addressing pertinent aspects associated with aging management of nuclear power plant reinforced concrete structures. Contained in this report is a summary of program results in the form of information related to longevity of nuclear power plant reinforced concrete structures, a data base presenting data and information on the time variation of concrete materials under the influence of environmental stressors and aging factors, in-service inspection and condition assessments techniques, repair materials and methods, evaluation of nuclear power plant reinforced concrete structures and a reliability-based methodology for current and future condition assessments. Recommendations for future activities are also provided.

**NUREG/CR-6425:** IMPACT OF STRUCTURAL AGING ON SEISMIC RISK ASSESSMENT OF REINFORCED CONCRETE STRUCTURES IN NUCLEAR POWER PLANTS. ELLINGWOOD,B.R.; SONG,J. Johns Hopkins Univ., Baltimore, MD. * Oak Ridge National Laboratory. March 1996. 81pp. 9604230399. ORNL/TM-13149. 87971:225.

The Structural Aging Program is addressing the potential for degradation of concrete structural components and systems in nuclear power plants over time due to aging and aggressive environmental stressors. Structures are passive under normal operating conditions but play a key role in mitigating design-basis events, particularly those arising from external challenges such as earthquakes, extreme winds, fire, and floods. Structures are plant-specific and unique, often are difficult to inspect, and are virtually impossible to replace. The importance of structural failures in accident mitigation is amplified because such failures may lead to common-cause failures of other components. Structural condition assessment and service life prediction must focus on a few critical components and systems within the plant. Components and systems that are dominant contributors to risk and that require particular attention can be identified through the mathematical formalism of a probabilistic risk assessment, or PRA. To illustrate, the role of structural degradation due to aging on plant risk is examined through the framework of a Level 1 seismic PRA of a nuclear power plant. Plausible mechanisms of structural degradation are found to increase the core damage probability by approximately a factor of two.

**NUREG/CR-6428:** EFFECTS OF THERMAL AGING ON FRACTURE TOUGHNESS AND CHARPY-IMPACT STRENGTH OF STAINLESS STEEL PIPE WELDS. GAVENDA,D.J.; MICHAUD,W.F.; GALVIN,T.M.; et al. Argonne National Laboratory. May 1996. 86pp. 9605310163. ANL-95/47. 88394:253.

The degradation of fracture toughness tensile, and Charpy-impact properties of Type 308 stainless steel (SS) pipe welds due to thermal aging has been characterized at room temperature and 290 degrees C. Thermal aging of SS welds results in moderate decreases in Charpy- impact strength and fracture toughness. For the various welds in this study, upper-shelf energy decreased by 50-80 J/cm(2). The decrease in fracture toughness J-R curve or J(IC) is relatively small. Thermal aging had little or no effect on the tensile strength of the welds. Fracture properties of SS welds are controlled by the distribution and morphology of second-phase particles. Failure occurs by the formation and growth of microvoids near hard inclusions. Such processes are relatively insensitive to thermal aging. The ferrite phase has little or no effect on the fracture properties of the welds. Differences in fracture resistance of the welds arise from differences in the density and size of inclusions. The mechanical-property data from the present study are consistent with results from other investigations. The existing data have been used to establish minimum expected fracture properties for SS welds.

**NUREG/CR-6429:** THE ROLE OF ORGANIC COMPLEXANTS AND MICROPARTICULATES IN THE FACILITATED TRANSPORT OF RADIONUCLIDES. SCHILK,A.J.; ROBERTSON,D.E.; ABEL,K.H.; et al. Battelle Memorial Institute, Pacific Northwest Laboratory. December 1996. 109pp. 9701130193. PNL-10897. 91397:238.

This progress report describes the results of ongoing radiological and geochemical investigations of the mechanisms of radionuclide transport in groundwater at two low-level waste (LLW) disposal sites within the waste management area of the Chalk River Laboratories (CRL), Ontario, Canada. These sites,the Chemical Pit liquid disposal facility and the Waste Management Area C solid LLW disposal site, have provided valuable 30- to 40-year old field locations for characterizing the migration of radionuclides and evaluating a number of recent site performance objectives for LLW disposal facilities. These studies have focused on identifying the physico-chemical species of mobile radionuclides in groundwater at these field locations and characterizing their behavior in the sub-surface environment. Field and laboratory studies have shown that the mobile radionuclide species, including (55)Fe, (60)Co, (106)Ru, (125)Sb, and (239,24O)Pu, are generally anionic in nature, being sequestered by naturally occurring and/or man-made complexing materials, including fulvic and humic substances. Ion chromatographic and ultrafiltration separations of contaminated groundwater have identified a number of individual mobile chemical species of (6O)Co and (106)Ru, as well as organo-metallic macromolecules, indicating the presence of a complex mixture of these sequestered radionuclides being transported in the groundwater.

**NUREG/CR-6430:** SOFTWARE SAFETY HAZARD ANALYSIS. LAWRENCE,J.D. Lawrence Livermore National Laboratory. February 1996. 91pp. 9602290270. UCRL-ID-122514. 87296:001.

Techniques for analyzing the safety and reliability of analog-based electronic protection systems that serve to mitigate hazards in process control systems have been developed over many years, and are reasonably well understood. An example is the protection system in a nuclear power plant. The extension of these techniques to systems which include digital computers is not well developed, and there is little consensus among software engineering experts and safety experts on how to analyze such systems. One possible technique is to extend hazard analysis to include digital computer-based systems. Software is frequently overlooked during system hazard analyses, but this is unacceptable when the software is in control of a potentially hazardous operation. In such cases, hazard analysis should be extended to fully cover the software. A method for performing software hazard analysis is proposed in this paper.

**NUREG/CR-6432:** ESTIMATED NET VALUE AND UNCERTAINTY FOR AUTOMATING ECCS SWITCHOVER AT PWRS. WALSH,B.; BRIDEAU,J.; COMES,L.; et al. Science & Engineering Associates, Inc. February 1996. 125pp. 9605130079. SEASF-DR-94-001. 88220:001.

A central question for resolution of GSI-24 is whether or not PWRs that currently rely on a manual system for ECCS switchover to recirculation should be required to install an automatic system. Risk estimates are obtained by reevaluating the contributions to core damage frequencies (CDFs) associated with failures of manual and semiautomatic switchover at a representative PWR. This study considers each separate instruction of the corresponding emergency operating procedures (EOPs), the mechanism for each control, and the relationship of each control to its neighbors. Important contributions to CDF include human errors that result in completely coupled failure of both trains and failure to enter the required EOP. This detailed study finds that changeover to a semiautomatic system is not justified on the basis of cost-benefit analysis: going from a manual to a semiautomatic system reduces the CDF by 1.7x1O-5 per reactor year, but the probability that the net cost associated with the modification being less than $1,000 per person-rem is about 20% without license renewal. Scoping analyses, using optimistic assumptions, were performed for a changeover to a semiautomatic system with automatic actuation and to a fully automatic system; in these cases the probability of having a net cost being less than $1,000/person-rem is about 50% without license renewal and over 95% with license renewal.

**NUREG/CR-6435:** AN ANALYSIS OF THE IMPACTS OF ECONOMIC INCENTIVE PROGRAMS ON COMMERCIAL NUCLEAR POWER PLANT OPERATIONS AND MAINTENANCE COSTS. KAVANAUGH,D.C.; MONROE,W.K. Battelle Memorial Institute, Pacific Northwest Laboratory. WOOD,R.S. Office of Nuclear Reactor Regulation (Post 941001). February 1996. 47pp. 9603190137. PNL-10934. 87510:172.

This study presents an analysis of the determinants of nonfuel Operations and Maintenance expenses (O&M) for the period 1986 to 1990. Since the determinants of O&M outlays are likely to be many and varied, the potential linkages, both direct and indirect, to plant safety can be substantial. This study develops a framework to analyze the elements that affect these expenditures to provide a basis for understanding these linkages.

**NUREG/CR-6436:** SURVEY OF AMBIENT ELECTROMAGNETIC AND RADIO-FREQUENCY INTERFERENCE LEVELS IN NUCLEAR POWER PLANTS. KERCEL,S.W.; MOORE,M.R.; BLAKEMAN,E.D.; et al. Oak Ridge National Laboratory. November 1996. 111pp. 9701060156. ORNL/TM-13171. 91280:149.

This document reports the results of a survey of ambient electromagnetic conditions in representative nuclear power plants. The U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research engaged the Oak Ridge National Laboratory (ORNL) to perform these measurements to characterize the electromagnetic interference (EMI) and radio-frequency interference (RFI) levels that can be expected in nuclear power plant environments. This survey is the first of its kind, being based on long-term unattended observations. The data presented in this report were measured at eight different nuclear units and required 14 months to collect. A representative sampling of power plant conditions (reactor type, operating mode, site location) monitored over extended observation periods (up to 5 weeks) were selected to more completely determine the characteristic electromagnetic environment for nuclear power plants. Radiated electric fields were measured over the frequency range of 5 MHz to 8 GHz. Radiated magnetic fields and conducted EMI events were measured over the frequency range of 305 Hz to 5 MHz. Highest strength observations of the electromagnetic ambient environment across all measurement conditions at each site provide frequency dependent profiles for EMI/RFI levels in nuclear power plants.

**NUREG/CR-6438:** THE EFFECT OF CYCLIC AND DYNAMIC LOADS ON CARBON STEEL PIPE. RUDLAND,D.L.; SCOTT,P.M.; WILKOWSKI,G.M. Battelle Memorial Institute, Columbus Laboratories. February 1996. 102pp. 9603190145. BMI-2188. 87511:187.

This report presents the results of four 152-mm (6-inch) diameter, unpressurized, circumferential through-wall-cracked, dynamic pipe experiments fabricated from STS410 carbon steel pipe manufactured in Japan. For three of these experiments, the through-wall crack was in the base metal. The displacement histories applied to these experiments were a quasi-static monotonic, dynamic monotonic, and dynamic, cyclic (R = -1) history. The through-wall crack for the third experiment was in a tungsten-inert-gas weld, fabricated in Japan, joining two lengths of STS410 pipe. The displacement history for this experiment was the same history applied to the dynamic, cyclic base metal experiment. The test temperature for each experiment was 300 C (572 F). The objective of these experiments was to compare a Japanese carbon steel pipe material with United States pipe material, to ascertain whether this Japanese steel was as sensitive to dynamic and cyclic effects as United States carbon steel pipe. In support of these pipe experiments, quasi-static and dynamic, tensile and fracture toughness tests were conducted. An analysis effort was performed that involved comparing experimental crack initiation and maximum moments with predictions based on available fracture prediction models, and calculating J-R curves for the pipe experiments using the **-factor method.

**NUREG/CR-6439:** DESIGN OF THE IPIRG-2 SIMULATED SEISMIC FORCING FUNCTION. OLSON,R.; SCOTT,P.M.; WILKOWSKI,G.M. Battelle Memorial Institute, Columbus Laboratories. February 1996. 86pp. 9602290278. BMI-2186. 87296:092.

A series of pipe system experiments was conducted in IPIRG-2 that used a realistic seismic forcing function. Because the seismic forcing function was more complex than the single-frequency increasing-amplitude sinusoidal forcing function used in the IPIRG-1 pipe system experiments, considerable effort went into designing the function. This report documents the design process for the seismic forcing function used in the IPIRG-2 pipe system experiments.

**NUREG/CR-6440:** THE EFFECTS OF CYCLIC AND DYNAMIC LOADING ON THE FRACTURE RESISTANCE OF NUCLEAR PIPING STEELS.Technical Report:October 1992 - April 1996. RUDLAND,D.L.; BRUST,F.W.; WILKOWSKI,G.M. Battelle Memorial Institute, Columbus Laboratories. December 1996. 200pp. 9701160171. BMI-2190. 91456:002.

This report presents the results of the material property evaluation efforts performed within Task 3 of the IPIRG-2 Program. Several related investigations were conducted. (1) Quasi-static, cyclic-load compact tension specimen experiments were conducted using parameters similar to those used in IPIRG-1 experiments on 6-inch nominal diameter through-wall-cracked pipes. These experiments were conducted on a TP304 base metal, an A106 Grade B base metal, and their respective submerged-arc welds. The results showed that when using a constant cyclic displacement increment, the compact tension experiments could predict the through-wall-cracked pipe crack initiation toughness, but a different control procedure is needed to reproduce the pipe cyclic crack growth in the compact tension tests. (2) Analyses conducted showed that for 6-inch diameter pipe, the quasi-static, monotonic J-R curve can be used in making cyclic pipe moment predictions; however, sensitivity analyses suggest that the maximum moments decrease slightly from cyclic toughness degradation as the pipe diameter increases. (3) Dynamic stress-strain and compact tension tests were conducted to expand on the existing dynamic database. Results from dynamic moment predictions suggest that the dynamic compact tension J-R and the quasi-static stress-strain curves are the appropriate material properties to use in making dynamic pipe moment predictions.

**NUREG/CR-6442:** EVIDENCE OF AGING EFFECTS ON CERTAIN SAFETY-RELATED COMPONENTS. MAGLEBY,H.L.; ATWOOD,C.L.; MACDONALD,P.E.; et al. Idaho National Engineering Laboratory. January 1996. 74pp. 9603180353. INEL-95/0654. 87511:281.

In response to interest shown by the Nuclear Energy Agency (NEA), Principal Working Group I (PWG-1) of the Committee on the Safety of Nuclear Installations (CSNI) conducted a generic study on the effects of aging of active components in nuclear power plants. (This focus on active components is consistent with PWG-1's mandate; passive components are primarily within the mandate of PWG-3.) Representatives from France, Sweden, Finland, Japan, the United States, and the United Kingdom participated in the study by submitting reports documenting aging studies performed in their countries. This report consists of summaries of those reports, along with a comparison of the various statistical analysis methods used in the studies. The studies indicate that with some exceptions, active components generally do not present a significant aging problem in nuclear power plants. Design criteria and effective preventative maintenance programs, including timely replacement of components, are effective in mitigating potential aging problems. However, aging studies (such as qualitative and statistical analyses of failure modes and maintenance data) are an important part of efforts to identify and solve potential aging problems. Solving these problems typically includes such strategies as replacing suspect components with improved components, and implementing improved maintenance programs.

**NUREG/CR-6443:** DETERMINISTIC AND PROBABILISTIC EVALUATIONS FOR UNCERTAINTY IN PIPE FRACTURE PARAMETERS IN LEAK-BEFORE-BREAK AND IN-SERVICE FLAW EVALUATIONS. GHADIALI,N.; RAHMAN,S.; CHOI,Y.H. Battelle Memorial Institute, Columbus Laboratories. June 1996. 134pp. 9607120198. BMI-2191. 88980:141.

This report presents new results from deterministic and probabilistic analyses to evaluate the significance of a number of technical aspects that may affect LBB or in-service flaw evaluations. In most cases there are both deterministic and probabilistic results. The deterministic analyses were conducted independently of the probabilistic analyses, which offered the opportunity to validate conclusions from each of these independent studies. The technical aspects evaluated relative to LBB uncertainties were: (1) evaluation of different crack morphology default values, (2) evaluation of COD dependent and independent crack morphology models for tight crack leak-rate analyses, (3) changes of normal operating and N+SSE stress levels on conditional failure probability, (4) dynamic and cyclic loads history effects on load-carrying capacity of through-wall-cracked pipe, (5) evaluation of the effect of off-centered cracks, (6) evaluation of the effect of restraint of pressure induced bending, and (7) evaluation of the effect of residual stresses on leak-rate analyses. Uncertainty analyses conducted relative to in-service flaw evaluations were: (1) dynamic and cyclic load history effects on load-carrying capacity of surface-cracked pipe, and (2) effect of uncertainty in UT flaw sizing. The relative ranking of importance is given for the significance of each technical aspect investigated.

**NUREG/CR-6444:** FRACTURE BEHAVIOR OF CIRCUMFERENTIALLY SURFACE-CRACKED ELBOWS. KILINSKI,T.; MOHAN,R.; RUDLAND,D.L.; et al. Battelle Memorial Institute, Columbus Laboratories. December 1996. 183pp. 9701160168. BMI-2192. 91457:001.

This report presents the results from Task 2 of the Second International Piping Integrity Research Group (IPIRG-2) program. The focus of the Task 2 work was directed towards furthering the understanding of the fracture behavior of long-radius elbows. This was accomplished through a combined analytical and experimental program. J-estimation schemes were developed for both axial and circumferential surface cracks in elbows. Large-scale, quasi-static and dynamic, pipe-system, elbow fracture experiments under combined pressure and bending loads were performed on elbows containing an internal surface crack at the extrados. In conjunction with the elbow experiments, material property data were developed for the A106-90 carbon steel and WP304L stainless steel elbow materials investigated. A comparison of the experimental data with the maximum stress predictions using existing straight pipe fracture prediction analysis methods, and elbow fracture prediction methods developed in this program was performed. This analysis was directed at addressing the concerns regarding the validity of using analysis predictions developed for straight pipe to predict the fracture stresses of cracked elbows. Finally, a simplified fitting flaw acceptance criteria incorporating ASME B(2) stress indices and straight pipe, circumferential-crack analysis was developed.

**NUREG/CR-6445:** DEVELOPMENT OF A J-ESTIMATION SCHEME FOR INTERNAL CIRCUMFERENTIAL AND AXIAL SURFACE CRACKS IN ELBOWS. MOHAN,R.; BRUST,F.W.; GHADIALI,N.; et al. Battelle Memorial Institute, Columbus Laboratories. June 1996. 68pp. 9607090154. BMI-2193. 88957:252.

This report summarizes efforts to develop elastic and elastic-plastic fracture mechanics analyses for internal surface cracks in elbows. The analyses involved development of a GE/EPRI type J-estimation scheme from a matrix of finite element analyses. The following parameters were covered; 90-degree long-radius elbows, various R(m)/t ratios, and combined pressure and in-plane bending. For the combined pressure and bending analyses, the hoop stress was fixed to correspond to an average S(m) value for typical U.S. nuclear piping materials. Further factors included in the analyses were various strain-hardening exponents and a/t values; however, the circumferential cracked elbow was limited to one crack length, and the axial cracked elbow was limited to one crack length on the elbow flank. These analyses were implemented into a computer code called IP2ELBOW. The results from the computer code calculations showed that the moment values at crack initiation were 1.5 to 2 times lower for the axially cracked elbow than for the circumferentially cracked elbow. The moment values for pressurized straight pipe were somewhat lower than for circumferentially cracked elbows for R(m)/t of 10, but were 1.5 greater for R(m)/t of 20.

**NUREG/CR-6448 V01:** EVALUATION OF NATIONAL SEISMOGRAPH NETWORK DETECTION CAPABILITIES.Annual Report,July 1994 - July 1995. MCLAUGHLIN,K.L.; BENNETT,T.J. S-Cubed. March 1996. 58pp. 9604150331. SSS-TR-95-15216. 87867:286.

This first annual report presents detection thresholds, detection probabilities, and location error ellipse projections for the United States National Seismic Network (USNSN) with and without cooperative stations in the eastern United States. Network simulation methods are used with spectral noise levels at stations to simulate the processes of excitation, propagation, detection, and processing of seismic phases. The USNSN alone should be capable of detecting 4 or more P waves for shallow crustal earthquakes in nearly all of the eastern and central United States at the magnitude 3.8 level. When cooperative stations are added, the network should be capable of detecting 4 or more P waves from events 0.2 to 0.3 magnitude units lower. The planned expansion of the USNSN and cooperative stations should improve detection levels by an additional 0.2 to 0.3 magnitudes units in any areas. Location uncertainties for the USNSN can be significantly improved by addition of real-time cooperative stations. Median error ellipses for magnitude 4.5 earthquakes depend strongly on location, but uncertainties should be less than 100 km(2) in the central United States and degrade to 200 km(2) or more off-shore and south and north of the international boundaries. Close cooperation with the Canadian National Network should substantially improve detection thresholds and location uncertainties along the Canadian border.

**NUREG/CR-6450:** CHARACTERIZATION OF CONTAMINATION THROUGH THE USE OF POSITION SENSITIVE DETECTORS AND DIGITAL IMAGE PROCESSING. SHONKA,J.J.; DEBORD,D.M.; BENNETT,T.E.; et al. Shonka Research Associates, Inc. June 1996. 46pp. 9609200298. 89726:188.

This report describes development of a significant new method for monitoring radioactive surface contamination. A floor monitor prototype has been designed which uses position sensitive proportional counter based radiation detectors. The system includes a novel operator interface consisting of an enhanced reality display providing the operator with 3 dimensional contours of contamination and background subtracted stereo "clicks". The process software saves electronic files of survey data at very high rates along with time stamped video recording and provides completely documented surveys in a visualization oriented data management system. The data management system allows simple re-assembly of strips of data that are taken with a linear PSPC and allows visualization and treatment of the data using algorithms developed for processing images from earth resource satellites. This report includes a brief history of the development path for the floor monitor, a discussion of position sensitive proportional counter technology, and details concerning the process software, post processor and hardware. The last chapter discusses the field tests that were conducted at five sites and an application of the data management system for data not associated with detector systems.

**NUREG/CR-6455:** DATA ANALYSIS FOR STEAM GENERATOR TUBING SAMPLES. DODD,C.V. Oak Ridge National Laboratory. July 1996. 64pp. 9608210230. ORNL/TM-13206. 89423:282.

The objective of the Improved Eddy-Current ISI for Steam Generators program is to upgrade and validate eddy-current inspections, including probes, instrumentation, and data processing techniques for inservice inspection of new, used, and repaired steam generator tubes; to improve defect detection, classification and characterization as affected by diameter and thickness variations, denting, probe wobble, tube sheet, tube supports, copper and sludge deposits, even when defect types and other variables occur in combination; to transfer this advanced technology to NRC's mobile NDE laboratory and staff. This report provides a description of the application of advanced eddy-current neural network analysis methods for the detection and evaluation of common steam generator tubing flaws including axial and circumferential outer-diameter stress-corrosion cracking and intergranular attack. The report describes the training of the neural networks on tubing samples with known defects and the subsequent evaluation results for unknown samples. Evaluations were done in the presence of artifacts. Computer programs are given in the appendix.

**NUREG/CR-6457:** RAPID ESTIMATE OF SOLID VOLUME IN LARGE TUFF CORES USING A GAS PYCNOMETER. GEDDIS,A.M.; GUZMAN,A.G.; BASSETT,R.L. Arizona, Univ. of, Tucson, AZ. September 1996. 81pp. 9612030279. 90992:158.

A thermally insulated, rigid-volume gas pycnometer system has been developed. The pycnometer chambers have been machined from solid PVC cylinders. Two chambers confine dry high-purity helium at different pressures. A thick-walled design ensures minimal heat exchange with the surrounding environment and a constant volume system, while expansion takes place between the chambers. The internal energy of the gas is assumed constant over the expansion. The ideal gas law is used to estimate the volume of solid material sealed in one of the chambers. Temperature is monitored continuously and incorporated into the calculation of solid volume. Temperature variation between measurements is less than 0.1 degrees C. The data are used to compute grain density for oven-dried Apache Leap tuff core samples. The measured volume of solid and the sample bulk volume are used to estimate porosity and bulk density. Intrinsic permeability was estimated from the porosity and measured pore surface area and is compared to in-situ measurements by the air permeability method. The gas pycnometer accommodates large core samples (0.25 m length x 0.11 1 m diameter) and can measure solid volume greater than 220 cm (3) with less than 1% error.

**NUREG/CR-6458:** MOISTURE CHARACTERISTIC CURVES FOR APACHE LEAP TUFF: TEMPERATURE EFFECTS AND HYSTERESIS. RHODES,C.R.; WILSON,L.G.; RASMUSSEN,T.C.; et al. Arizona, Univ. of, Tucson, AZ. September 1996. 72pp. 9612030275. 90992:239.

Laboratory methods were use to define matrix hydraulic properties for low-permeability Apache Leap Tuff core segments. Moisture content/matric potential relationships, including hysteresis, and measured hydraulic conductivity data were determined at a constant laboratory temperature of 20 C. To investigate the effects of temperature on those relationships, additional retention data were obtained at 5 C and 45 C. Measured retention data at all temperatures were applied to the van Genuchten model RETC, which performs curve-fitting and calculation of the flow parameter hydraulic conductivity. Although data at 5 C proved to be inconclusive, increasing the temperature from 20 to 45 C produced a shift of the moisture characteristic curve toward a higher potential for a given water saturation. Model calculated hydraulic conductivity also increased as temperature increased, with respect to water saturation. The temperature-dependent change in the viscosity of water proved inadequate to explain the increases of hydraulic conductivity with temperature.

**NUREG/CR-6460:** CSNI PROJECT FOR FRACTURE ANALYSES OF LARGE-SCALE INTERNATIONAL REFERENCE EXPERIMENTS (FALSIRE II). BASS,B.R.; PUGH,C.E.; KEENEY,J.; et al. Oak Ridge National Laboratory. November 1996. 176pp. 9612030256. ORNL/TM-13207. 90999:001.

A summary of Phase II of the Project for Fracture Analysis of Large-Scale International Reference Experiments (FALSIRE) is presented. Project FALSIRE was created by the Fracture Assessment Group (FAG) of the Organization for Economic Cooperation and Development/Nuclear Energy Agency's (OECD/NEA's) Committee on the Safety of Nuclear Installations (CSNI) Principal Working Group No. 3 (PWG-3). The CSNI/FAG was formed to evaluate fracture prediction capabilities currently used in safety assessments of nuclear components. Members were from laboratories and research organizations in Western Europe, Japan, and the United States. The CSNI/FAG initiated an international project (FALSIRE 1) in 1988 to assess various fracture methodologies through interpretive analyses of six large-scale fracture experiments. These experiments were conducted by research organizations in Europe, Japan, and the United States. Following the successful completion of FALSIRE I in 1992, several participating organizations indicated a desire to proceed with further evaluation of fracture analysis methods in a Phase II program. FALSIRE II included seven reference cleavage fracture experiments that focused primarily on behavior of relatively shallow cracks in the transition temperature region.

**NUREG/CR-6462:** FIELD TESTING PLAN FOR UNSATURATED ZONE MONITORING AND FIELD STUDIES. YOUNG,M.H.; WIERENGA,P.J.; WARRICK,A.W.; et al. Arizona, Univ. of, Tucson, AZ. October 1996. 63pp. 9611190243. 90824:258.

The U.S. Nuclear Regulatory Commission has requested development of these field testing plans for evaluating subsurface monitoring systems for low-level radioactive waste disposal sites (LLW) and for monitoring at decommissioned facilities designated under the "Site Decommissioning Management Plan" (SDMP). The tests are conducted on a 50 m by 50 m plot on the University of Arizona's Maricopa Agricultural Center. Within the 50 m by 50 m plot one finds: 1) an instrumented buried trench, 2) monitoring islands similar to those proposed for the Ward Valley, California LLW Facility, 3) deep borehole monitoring sites, 4) gaseous transport monitoring, and 5) locations for testing non-invasive geophysical measurement techniques. The various subplot areas are instrumented with commercially available instruments such as neutron probes, time domain reflectometry probes, tensiometers, psychrometers, heat dissipation sensors, thermocouples, solution samplers, and cross-hole geophysics electrodes. Measurement depths vary from ground surface to 15 m. The data from the controlled flow and transport experiments will be used to develop an integrated approach to long-term monitoring of the unsaturated zone at waste disposal sites. The data will be used to test field-scale flow and transport models. This report describes the design of the experiment and the methodology proposed for evaluating the data.

**NUREG/CR-6463:** REVIEW GUIDELINES ON SOFTWARE LANGUAGES FOR USE IN NUCLEAR POWER PLANT SAFETY SYSTEMS.Final Report. HECHT,H.; HECHT,M.; GRAFF,S.; et al. SoHaR, Inc. June 1996. 400pp. 9608060265. 89259:001.

Guidelines for the programming and auditing of software written in high level languages for safety systems are presented. The guidelines are derived from a framework of issues significant to software safety which was gathered from relevant standards and research literature. Language-specific adaptations of these guidelines are provided for the following high level languages: Ada, C/C++, Programmable Logic Controller (PLC) Ladder Logic, International Electrotechnical Commission (IEC) Standard 1131-3 Sequential Function Charts, Pascal, and PL/M. Appendices to the report include a tabular summary of the guidelines and additional information on selected languages.

**NUREG/CR-6465:** DEVELOPMENT OF TOOLS FOR SAFETY ANALYSIS OF CONTROL SOFTWARE IN ADVANCED REACTORS. GUARRO,S.; YAU,M.; MOTAMED,M. ASCA, Inc. April 1996. 144pp. 9605130069. 88226:001.

Software based control systems have gained a pervasive presence in a wide variety of applications, including nuclear power plant control and protection systems which are within the oversight and licensing responsibility of the U.S. Nuclear Regulatory Commission. While the cost effectiveness and flexibility of software based plant process control is widely recognized, it is very difficult to achieve high levels of dependability and safety assurance for the functions performed by process control software, due to the very flexibility and potential complexity of the software itself. The development of tools to model, analyze and test software design and implementations in the context of the system that the software is designed to control can greatly assist the task of providing higher levels of assurance than those obtainable by software testing alone. This report presents and discusses the development of the Dynamic flowgraph methodology (DFM) and its application in the dependability and assurance analysis of software-based control systems. The features of the methodology and full-scale examples of application to both generic process and nuclear power plant control systems are presented and discussed in detail. The features of a workstation software tool developed to assist users in the application of DFM are also described.

**NUREG/CR-6466:** GROUND MOTION INPUT IN SEISMIC EVALUATION STUDIES.Impacts Of Artificial Time History Input On In-Structure Demand Spectra. SEWELL,R.T.; WU,S.C. Risk Engineering, Inc. July 1996. 315pp. 9608210251. 89421:001.

This report documents research pertaining to conservatism and variability in seismic risk estimates. Specifically, it examines whether or not artificial motions produce unrealistic evaluation demands, i.e., demands significantly inconsistent with those expected from real earthquakes motions. To study these issues, two types of artificial motions are considered; (a) motions with smooth response spectra, and (b) motions with realistic variations in spectral amplitude across vibration frequency. For both types of artificial motion, time histories are generated to match target spectral shapes. For comparison, empirical motions representative of those that might result from strong earthquakes in the Eastern U.S. are also considered. The study findings suggest that artificial motions resulting from typical simulation approaches (aimed at matching a given target spectrum) are generally adequate and appropriate in representing the peak-response demands that may be induced in linear structures and equipment responding to real earthquake motions. Also, given similar input Fourier energies at high-frequencies, levels of input Fourier energy at low frequencies observed for artificial motions are substantially similar to those levels noted in real earthquake motions. In addition, the study reveals specific problems resulting from the application of Western U.S. type motions for seismic evaluation of Eastern U.S. nuclear power plants.

**NUREG/CR-6467:** IMPACT OF GROUND MOTION CHARACTERIZATION ON CONSERVATISM AND VARIABILITY IN SEISMIC RISK ESTIMATES. SEWELL,R.T.; TORO,G.R.; MCGUIRE,R.K. Risk Engineering, Inc. July 1996. 151pp. 9608210260. 89423:001.

This study evaluates the impact of alternative methods in treatment and characterization of earthquake ground motions on estimates of seismic risk and its uncertainty. The objective is to formulate specific procedures and characterizations that may lead to less biased and more precise estimates of risk. This report focuses on sources of conservatism and uncertainty in risk that may be introduced by simplifications that are made at the interface of seismic hazard and fragility assessments, particularly the use of a fixed spectral shape for all magnitudes and the anchoring of this shape to PGA. Results indicate significant conservatism in the use of standard review spectra at eastern U.S. Nuclear plant sites and a strong dependence of seismic fragility on earthquake magnitude when PGA is used as the ground-motion characterization. This study concludes that a single, composite-magnitude spectrum of the appropriate shape can generally be used to characterize ground motion for fragility assessment without introducing significant bias or uncertainty in seismic risk estimates. Results also show that the inelastic or elastic spectral acceleration are superior to PGA as spectral anchors, but they bring only a modest benefit in uncertainty reduction because uncertainty in the risk is dominated by the large uncertainty in the hazard.

**NUREG/CR-6468:** GROUND MOTION INPUT IN SEISMIC EVALUATION STUDIES.Impacts On Risk Assessment Of Uniform Hazard Spectra. WU,S.C.; SEWELL,R.T. Risk Engineering, Inc. July 1996. 130pp. 9608210262. 89423:152.

This report documents research on the subject of conservatism and variability in seismic risk estimates. Particularly, it examines the effects of the uniform hazard spectrum (UHS) for deriving probabilistic estimates of risk and in-structure demand levels, as compared to the more-exact use of realistic time history inputs (of given probability) that depend explicitly on magnitude and distance. The approach differs significantly from the conventional procedure in its exhaustive treatment of the ground-motion threat, and in its more detailed assessment of component responses to that threat. It is found that the approximate uniform hazard in-structure spectrum (UH-ISS) obtained based on UHS appear to be very close to the more-exact results directly computed from scenario earthquakes. The conclusion does not depend on site configurations and structural characteristics. In addition, UH-ISS has composite shapes and may not correspond to the characteristics possessed in a single earthquake. The shape is largely affected by the structural property in most cases and can be derived approximately from the corresponding UHS. Motions with smooth spectra, however, will not have the same damage potential as those of more realistic motions with jagged spectral shapes. As a result, UHS-based analysis may underestimate the real demands in non-linear structural analyses.

**NUREG/CR-6470:** FITNESS FOR DUTY IN THE NUCLEAR INDUSTRY: UPDATE OF THE TECHNICAL ISSUES 1996. DURBIN,N.; GRANT,T.; BITTNER,A.; et al. Battelle Seattle Research Center. May 1996. 272pp. 9605220322. BSRC-700/96/004. 88319:001.

This report provides an update of information on the technical issues surrounding the creation, implementation, and maintenance of fitness-for-duty (FFD) policies and programs. It has been prepared as a resource for Nuclear Regulatory Commission (NRC) and nuclear power plant personnel who deal with FFD programs. It contains a general overview and update on the technical issues that the NRC considered prior to the publication of its original FFD rule and the revisions to that rule (presented in earlier NUREG/CRs). It also includes chapters that address issues about which there is growing concern and/or about which there have been substantial changes since NUREG/CR-5784 was published. Although this report is intended to support the NRC's rulemaking on fitness for duty, the conclusions of the authors are their own and do not necessarily represent the opinions of the NRC.

**NUREG/CR-6473:** GLOBAL POSITIONING SYSTEM REOBSERVATIONS OVER THE EASTERN UNITED STATES STRAIN MONITORING NETWORK. STRANGE,W.E. Commerce, Dept. of, National Oceanic & Atmospheric Administration. June 1996. 33pp. 9607090248. 88957:315.

In the period March-May, 1990, a 45 station geodetic network, originally established in November-December, 1987, was reobserved using global positioning system (GPS) technology. This network, known as the Eastern U.S. Strain Network, was established for the purpose of determining strain and deformation in the central and eastern United States. This 1990 reobservation was the first of a series of reobservations scheduled to take place over a decade in order to place meaningful constraints on the small differential movements involved.

**NUREG/CR-6476:** CIRCUIT BRIDGING OF COMPONENTS BY SMOKE. TANAKA,T.J.; NOWLEN,S.P.; ANDERSON,D.J. Sandia National Laboratories. October 1996. 205pp. 9612120081. SAND96-2633. 91074:007.

Smoke can adversely affect digital electronics; in the short term, it can lead to circuit bridging and in the long term to corrosion of metal parts. This report is a summary of the work to date and component-level tests by Sandia National Laboratories for the Nuclear Regulatory Commission to determine the impact of smoke on digital instrumentation and control equipment. The component tests focused on short-term effects such as circuit bridging in typical components and the factors that can influence how much the smoke will affect them. These factors include the component technology and packaging, physical board protection, and environmental conditions such as the amount of smoke, temperature of burn, and humidity level. The likelihood of circuit bridging was tested by measuring leakage currents and converting those currents to resistance in ohms. Hermetically sealed ceramic packages were more resistant to smoke than plastic packages. Coating the boards with an acrylic spray provided some protection against circuit bridging. The smoke generation factors that affect the resistance the most are humidity, fuel level, and burn temperature. The use of CO(2) as a fire suppressant, the presence of galvanic metal, and the presence of PVC did not significantly affect the outcome of these results.

**NUREG/CR-6483:** GUIDE TO VERIFICATION AND VALIDATION OF THE SCALE-4 CRITICALITY SAFETY SOFTWARE. EMMETT,M.B.; JORDAN,W.C. Oak Ridge National Laboratory. December 1996. 121pp. 9701130200. ORNL/TM-12834. 91393:171.

Whenever a decision is made to newly install the SCALE nuclear criticality safety software on a computer system, the user should run a set of verification and validation (V&V) test cases to demonstrate that the software is properly installed and functioning correctly. This report is intended to serve as a guide for this V&V in that it specifies test cases to run and gives expected results. The report describes the V&V that has been performed for the nuclear criticality safety software in a version of SCALE-4. The verification problems specified by the code developers have been run, and the results compare favorably with those in the SCALE 4.2 baseline. The results reported in this document are from the SCALE 4.2P version which was run on an IBM RS/6000 Workstation. These results verify that the SCALE-4 nuclear criticality safety software has been correctly installed and is functioning properly. A validation has been performed for KENO V.a utilizing the CSAS25 criticality sequence and the SCALE 27-group cross-section library for (233)U, (235)U, and (239)Pu fissile systems in a broad range of geometries and fissile fuel forms. The experimental models used for the validation were taken from three previous validations of KENO V.a. A statistical analysis of the calculated results was used to determine the average calculational bias and a subcritical k(eff) criteria for each class of systems validated. Included in the statistical analysis is a means of estimating the margin of subcriticality in k(eff). This validation demonstrates that KENO V.a and the 27-group library may be used for nuclear criticality safety computations provided the system being analyzed falls within the range of the experiments used in the validation.

**NUREG/CR-6484:** GUIDE TO VERIFICATION AND VALIDATION OF THE SCALE-4 RADIATION SHIELDING SOFTWARE. BROADHEAD,B.L.; EMMETT,M.B.; TANG,J.S. Oak Ridge National Laboratory. December 1996. 55pp. 9701130196. ORNL/TM-13277. 91400:234.

Whenever a decision is made to newly install the SCALE radiation shielding software on a computer system, the user should run a set of verification and validation (V&V) test cases to demonstrate that the software is properly installed and functioning correctly. This report is intended to serve as a guide for this V&V in that it specifies test cases to run and gives expected results. The report describes the V&V that has been performed for the radiation shielding software in a version of SCALE-4. This report provides documentation of sample problems which are recommended for use in the V&V of the SCALE-4 system for all releases. The results reported in this document are from the SCALE-4.2P version which was run on an IBM RS/6000 work station. These results verify that the SCALE-4 radiation shielding software has been correctly installed and is functioning properly. A set of problems for use by other shielding codes (e.g., MCNP, TWOTRAN, MORSE) performing similar V&V are discussed. A validation has been performed for XSDRNPM and MORSE-SGC utilizing SAS1 and SAS4 shielding sequences and the SCALE 27-18 group (27N-18COUPLE) cross-section library for typical nuclear reactor spent fuel sources and a variety of transport package geometries. The experimental models used for the validation were taken from two previous applications of the SAS1 and SAS4 methods.

**NUREG/CR-6487:** CONTAINMENT ANALYSIS FOR TYPE B PACKAGES USED TO TRANSPORT VARIOUS CONTENTS. ANDERSON,B.L.; CARLSON,R.W.; FISCHER,L.E. Lawrence Livermore National Laboratory. November 1996. 65pp. 9612030267. UCRL-ID-124822. 90980:159.

This report presents sample calculations and examples of leakage rates for various contents in Type B packages. Samples of acceptance standard leakage rates are developed for specific contents types at normal transport conditions and at hypothetical accident conditions. The leakage rates are expressed as allowable standard leakage rates. The types of contents considered include: (1) powders, (2) liquids, (3) irradiated fuel rods,. (4) gases, and (5) solids.

**NUREG/CR-6489:** SUMMARY OF FAILURE ANALYSIS ACTIVITIES AT BROOKHAVEN NATIONAL LABORATORY. COWGILL,M.G.; CZAJKOWSKI,C.J.; FRANZ,E.M. Brookhaven National Laboratory. October 1996. 119pp. 9611190248. BNL-NUREG-52508. 90824:139.

Brookhaven National Laboratory has for many years conducted examinations related to the failures of nuclear materials and components. These examinations include the confirmation of root cause analyses, the determination of the causes of failure, identification of the species that accelerate corrosion, and comparison of the results of nondestructive examinations with those obtained by destructive examination. The results of those examinations, which had previously appeared in various formats (formal and informal reports, journal articles, etc,), have been collected together and summarized in the present report. The report is divided into sections according to the general subject matter (for example, corrosion, fatigue, etc.). Each section presents summaries of the information contained in specific reports and publications, all of which are fully identified as to title, authors, report number, or journal reference, date of publication, and FIN number under which the work was performed.

**NUREG/CR-6490 V01:** NUCLEAR POWER PLANT GENERIC AGING LESSONS LEARNED (GALL). Main Report And Appendix A. KASZA,K.E.; DIERCKS,D.R.; HOLLAND,J.W.; et al. Argonne National Laboratory. December 1996. 443pp. 9701130206. ANL-96/13. 91394:001.

Argonne National Laboratory and Idaho National Engineering Laboratory in support of the License Renewal Project Directorate of the U.S. Nuclear Regulatory Commission (NRC) performed a comprehensive review of literature pertaining to nuclear power plant aging effects. This generic aging lessons learned (GALL) effort was a systematic review of plant aging information in order to assess materials and component aging issues related to continued operation and license renewal of operating reactors. Literature on mechanical, structural, thermal-hydraulic, and electrical components and systems reviewed consisted of 163 Nuclear Plant Aging Research Reports, 31 NRC Generic Letters, 265 Information Notices, 82 Licensee Event reports, 5 Bulletins, and 10 Nuclear Management and Resources Council Industry Reports. The results of these reviews were systematized using a standardized GALL tabular format and standardized definitions of aging related degradation mechanisms and effects. A computerized data base has also been developed for all review tables and can be used to search for information on structures, components, and relevant aging effects. A survey of the GALL tables reveals that all significant component and structure aging issues are currently being addressed by the regulatory process. However, aging of what are termed passive components and structures has been highlighted for continued scrutiny.

**NUREG/CR-6490 V02:** NUCLEAR POWER PLANT GENERIC AGING LESSONS LEARNED (GALL). Appendix B. KASZA,K.E.; DIERCKS,D.R.; HOLLAND,J.W.; et al. Argonne National Laboratory. December 1996. 278pp. 9701130171. ANL-96/13. 91395:080.

See NUREG/CR-6490,V01 abstract.

**NUREG/CR-6491:** RECOMMENDATIONS FOR PROTECTING AGAINST FAILURE BY BRITTLE FRACTURE.Category II And III Ferritic Steel Shipping Containers With Wall Thickness Greater Than Four Inches. SCHWARTZ,M.W.; FISCHER,L.E. Lawrence Livermore National Laboratory. August 1996. 26pp. 9609030358. UCRL-ID-124583. 89546:334.

This report provides criteria for selecting ferritic steels that would prevent brittle fracture in Category II and III shipping containers with wall thickness greater than four inches. These methods are extensions of those previously used for Category II and III containers with wall thickness less than four inches and Category I containers with wall thickness greater than four inches.

**NUREG/CR-6492:** BLT-MS (BREACH,LEACH, AND TRANSPORT-MULTIPLE SPECIES) DATA INPUT GUIDE.A Computer Model For Simulating Release Of Contaminant From Subsurface Low-Level Waste Disposal Facility. SULLIVAN,T.M.; KINSEY,R.R.; et al. Brookhaven National Laboratory. MACKINNON,R.J. Ecodynamics Research Associates, Inc.,. November 1996. 184pp. 9612110212. BNL-NUREG-52509. 91064:001.

The BLT-MS computer code has been developed, implemented, and tested. BLT-MS is a two-dimensional finite element computer code capable of simulating the time evolution of concentration resulting from the time-dependent release and transport of aqueous phase species in a subsurface soil system. BLT-MS contains models to simulate the processes (water flow, container degradation, waste form performance, transport, and radioactive production and decay) most relevant to estimating the release and transport of contaminants from a subsurface disposal system. Water flow is simulated through tabular input or auxiliary files. Container degradation considers localized failure due to pitting corrosion and general failure due to uniform surface degradation processes. Waste form performance considers release to be limited by one of our mechanism: rinse with partitioning, diffusion, uniform surface degradation, or solubility. Radioactive production and decay in the waste form are simulated. Transport considers the processes of advection, dispersion, diffusion, radioactive production and decay, reversible linear sorption, and sources (waste forms releases). To improve the usefulness of BLT-MS a pre-processor, BLTMSIN, which assists in the creation of input files, and a post-processor, BLTPLOT, which provides a visual display of the data have been developed. This document reviews the models implemented in BLT-MS and serves as a guide to creating input files for BLT-MS.

**NUREG/CR-6494:** CONTINUOUS ANALYSIS FOR VANADIUM IN LOMI DECONTAMINATION. KOTTLE,S.; STOWE,R.A.; BISHOP,J.V. Omni Tech International, Ltd. December 1996. 38pp. 9612240085. 91197:244.

This report covers research into methods for real-time, on-line analysis of vanadium ion concentration during LOMI chemical decontamination of piping in nuclear power plants. An on-line colorimeter was developed which performed very well in laboratory simulations, but was subject to excessive interference by colored materials in the field. A titrimetric method was investigated using ferric ion with potentiometric electrodes to follow the course of the reaction. By proper choice of reagents it was specific to vanadous ion and free from interference. An automatic instrument was developed, tested under simulated LOMI conditions, and successfully evaluated in the field during a LOMI chemical decontamination.

**NUREG/CR-6495:** CASE STUDY OF LIQUEFACTION INDUCED BY THE 1944 MASSENA, NEW YORK - CORNWALL, ONTARIO EARTHQUAKE. TUTTLE,M.P. Lamont-Doherty Earth Observatory. September 1996. 33pp. 9612040122. 91006:317.

Despite surveying with ground-penetration radar and trenching at four different locations, no earthquake-induced sand dikes were observed at the site where three sand fissures were documented by Berkey (1945) following the 1944 Massena, New York-Cornwall, Ontario, earthquake. The site was found to be very disturbed by human activity, especially road building related to the construction of the St. Lawrence-Franklin D. Roosevelt Power Project. Sand diapirs and dewatering structures were observed in two of the trenches. These features may be related to incipient liquefaction but their origin is equivocal. The epicentral location of the 1944 earthquake probably occurred within 10 km of the site; therefore, the epicentral distance of this site of liquefaction is within the expected range even for western earthquakes of similar magnitude. A nearshore sandy facies of a glaciolacustrine deposit is thought to be the material that liquefied during the 1944 event A geotechnical investigation that would have assessed the liquefaction potential of subsurface material at the site was canceled due to downsizing at Ontario Hydro. Until a geotechnical investigation is conducted, additional trenching seems unwarranted. Poor cutbank exposures limited search for liquefaction features in the Massena area.

**NUREG/CR-6500:** OWNERS OF NUCLEAR POWER PLANTS. HUDSON,C.R.; WHITE,V.S. Oak Ridge National Laboratory. November 1996. 61pp. 9611260261. ORNL/TM-13297. 90920:241.

Commercial nuclear power plants in this country can be owned by a number of separate entities, each with varying ownership portions. Each of these owners may, in turn, have a parent/subsidiary relationship to other companies. In addition, the operator of the plant may be a different entity as well. This report provides a compilation on the owners/operators for all commercial power reactors in the United States. While the utility industry is currently experiencing changes in organizational structure which may affect nuclear plant ownership, the data in this report is current as of July 1996. The report is divided into the following sections representing different aspects of nuclear plant ownership: Nuclear power plant percentage ownership ordered by plant name; Nuclear power plant percentage ownership ordered by utility name; Utility/company relationships ordered by parent/holding company; Utility/company relationships ordered by subsidiary; Nuclear power plants listed by operator; Nuclear power plant operators listed by plant name; Nuclear power plants listed by state.

**NUREG/GR-0015:** BULK TEMPERATURE MEASUREMENT IN THERMALLY STRIPED PIPE FLOWS. LEMURE,N.; OLVERA,J.R.; RUGGLES,A.E. Tennessee, Univ. of, Knoxville, TN. December 1995. 130pp. 9608050231. 89250:144.

The bulk temperature measurement of pipe flows with thermal striping is explored. An experiment is conducted to examine the feasibility of using temperature measurements on the external surface of the pipe to estimate the bulk temperature of the flow. Simple mixing models are used to characterize the development of the temperature profile in the flow. Simple averaging techniques and Backward Propagating Neural Net are used to predict bulk temperature from the external temperature measurements. Accurate bulk temperatures can be predicted. However, some temperature distributions in the flow effectively mask the bulk temperature from the wall and cause significant error in the bulk temperature predicted using this technique.

**NUREG/IA-0129 P01:** AN ASSESSMENT OF THE CORCON-MOD3 CODE.Part I: Thermal-Hydraulic Calculations. STRIZHOV,V.; KANUKOVA,V.; VINOGRADOVA,T.; et al. Russia. September 1996. 201pp. 9612040179. 91006:001.

This report deals with the subject of CORCON-Mod3 code validation (thermal-hydraulics modeling capability only) based on MCCI experiments conducted under different programs in the past decade. Thermal-hydraulic calculations (i.e., concrete ablation, melt temperature, melt energy, concrete temperature, and condensable and non-condensible gas generation) were performed with the code, and compared with the data from 15 experiments, coducted at different scales using both simulant (metallic and oxidic) and prototypic melt materials, using different concrete types, and with and without an overlying water pool. Sensitivity studies were performed in a few cases involving, for example, heat transfer from melt to concrete, condensed phase chemistry, etc. Further, special analysis was performed using the ACE L8 experimental data to illustrate the differences between the experimental and the reactor conditions, and to demonstrate that with proper corrections made to the code, the calculated results were in better agreement with the experimental data. Generally, in the case of dry cavity and metallic melts, CORCON-Mod3 thermal-hydraulic calculations were in good agreement with the test data. For oxidic melts in a dry cavity, uncertainties in heat transfer models played an important role for two melt configurations - a stratified geometry with segregated metal and oxide layers, and a heterogeneous mixture. Some discrepancies in the gas release data were noted in a few cases. These discrepancies were attributed, in part, to condensed phase chemical reactions modeling and, in part, to experimental uncertainties. In the case of wet cavity, good agreement was found between the experimental data and code calculations except, again, for the gas release data. With proper corrections made to the code to account for correct condensed phase chemistry and with corrections made to the input data to account for experimental uncertainties, better agreement between code calculations and experimental data was noted.

**NUREG/IA-0130:** ASSESSMENT OF RELAP5/MOD3.1 WITH THE LSTF SB-SG-06 EXPERIMENT SIMULATING A STEAM GENERATOR TUBE RUPTURE TRANSIENT. SEUL,K.W.; BANG,Y.S.; LEE,S.; et al. Korea Institute of Nuclear Safety. September 1996. 112pp. 9612040171. CAMP002. 91006:206.

The objective of the present work is to identify the predictability of RELAP5/MOD3.1 regarding thermal-hydraulic behavior during a steam generator tube rupture (SGTR). To evaluate the computed results, LSTF SB-SG-06 test data simulating the SGTR that occurred at the Mihama Unit 2 in 1991 are used. Also, some sensitivity studies of the code change in RELAP5, the break simulation model, and the break valve discharge coefficient are performed. The calculation results indicate that the RELAP5/MOD3.1 code predicted well the sequence of events and the major phenomena during the transient, such as the asymmetric loop behavior, reactor coolant system (RCS) cool-down and heat transfer by natural circulation, the primary and secondary system depressurization by the pressurizer auxiliary spray and the steam dump using the intact loop steam generator (SG) relief valve, and so on. However, there are some differences from the experimental data in the number of the relief valve cycling in the affected SG, and the flow regime of the hot leg with the pressurizer, and the break flow rates. Finally, the calculation also indicates that the coolant in the core could remain in a subcooled state as a result of the heat transfer caused by the natural circulation flow even if the reactor coolant pumps (RCPs) turned off and that the affected SG could be properly isolated to minimize the radiological release after the SGTR.

**NUREG/IA-0131:** ASSESSMENT OF RELAP5/MOD3 USING BETHSY 6.2TC 6-INCH COLD LEG SIDE BREAK COMPARATIVE TEST. CHUNG,Y-J.; JEONG,J-J.; KIM,D-S.; et al. Korea Atomic Energy Research Institute. October 1996. 138pp. 9611190262. CAMP003. 90824:001.

This report presents the results of the RELAP5/MOD3 Version 7j assessment on BETHSY 6.2 TC test corresponding to a six inch cold leg break LOCA of the Pressurizer Water Reactor (PWR). The primary objective of the test was to provide reference data of two facilities of different scales (BETHSY and LSTF facilities). The present calculation aims at analysis of RELAP5/MOD3 capability on the small break LOCA simulation. The results of calculation have shown that the RELAP5/MOD3 reasonably predicts occurrences as well as trends of the major phenomena such as primary pressure, timing of loop seal clearing, liquid hold up, etc. However, some differences also have been found in the predictions of loop seal clearing, collapsed core water level after loop seal clearing, and accumulator injection behaviors. For understanding of discrepancies in the same predictions, several sensitivity calculations have been performed as well. These include the changes of two-phase discharge coefficients at the break junction and some corrections of the interphase drag term. As a result, change of a single parameter has not improved the overall predictions and it has been found that the interphase drag model still has large uncertainties.

**NUREG/IA-0132:** IMPROVEMENTS TO THE RELAP5/MOD3 REFLOOD MODEL AND UNCERTAINTY QUANTIFICATION OF REFLOOD PAK CLAD TEMPERATURE. CHUNG,B.D.; LEE,Y.J.; PARK,C.E.; et al. Korea Atomic Energy Research Institute. October 1996. 106pp. 9611190273. CAMP001. 90823:001.

Assessment of the original RELAP5/MOD 3.1 code against the FLECHT SEASET series of experiment has identified some weaknesses of the reflood model, such as the lack of quenching temperature model, the shortcoming of Chen transition boiling model, and the incorrect prediction of droplet size and interfacial heat transfer. Also high pressure spikes during the reflood calculation resulted in the high steam flow oscillation and liquid carryover. An effort has been made to improve the code with respect to the above weakness and the necessary model for wall heat transfer package and numerical scheme had been modified. Some important FLECHT-SEASET experiment were assessed using the improved version and standard version. The result from the improved version of RELAP5/MOD3.1 shows the weaknesses of RELAP5/MOD3.1 was much improved when compared to the standard MOD3.1 code. The prediction of void profile and cladding temperature agreed better with test data especially for the gravity feed test. The scatter diagram of peak cladding temperatures (PCTs) is made from the comparison of all the calculated PCTs and the corresponding experimental values. The deviation between experimental and calculated PCTs were calculated for 2793 data points. The deviations are shown to be normally distributed, and used to quantify statistically the PCT uncertainty of the code. The upper limit of PCT uncertainty at 95% confidence level is evaluated to be about 99K.

**NUREG/IA-0133:** DEVELOPMENT,IMPLEMENTATION, AND ASSESSMENT OF SPECIFIC CLOSURE LAWS FOR INVERTED-ANNULAR FILM-BOILING IN A TWO-FLUID MODEL. DE CACHARD,F. Paul Scherrer Institute. October 1996. 103pp. 9611190277. CAMP004. 90823:249.

Inverted-Annular Film-Boiling (IAFB) is one of the post-burnout heat transfer modes taking place during the reflooding phase of the loss-of-coolant accident, when the liquids at the quench front is subcooled. Under IAFB conditions, a continuous liquid core is separated from the wall by a superheated vapour film. A key issue in IAFB modeling is to predict how the heat flux reaching the vapour-liquid interface is split into a liquid heating term and a vaporization term. In the model proposed, convective liquid heating is related to the liquid velocity relative to the interface and not the absolute liquid velocity, as in previous models.

**NUREG/IA-0134:** ASSESSMENT OF RELAP5/MOD3.1 FOR GRAVITY-DRIVEN INJECTION EXPERIMENT IN THE CORE MAKEUP TANK OF THE CARR PASSIVE REACTOR (CP-1300). LEE,S.; NO,H.C. Korea Advanced Institute of Science and Technology. BANG,Y.S.; et al. Korea Institute of Nuclear Safety. October 1996. 150pp. 9611180277. CAMP005. 90820:001.

The objective of the present work is to improve the analysis capability of RELAP5/MOD3.1 on the direct contact condensation in the core makeup tank (CMT) of passive high-pressure injection system (PHIS) in the CARR Passive Reactor (CP-1300). The gravity-driven injection experiment is conducted by using a small scale test facility to identify the parameters having significant effects on the gravity-driven injection and the major condensation modes. The condensation modes are divided into three modes: sonic jet, subsonic jet and steam cavity. RELAP5/MOD3.1 is chosen to evaluate the code predictability on the direct contact condensation in the CMT. It is found that the predictions of MOD3.1 are in better agreement with the experimental data than those of MOD3.0. From the nodalization study of the test section, the 1-node model shows better agreement with the experimental data than the multi-node models. RELAP5/MOD3.1 identifies the flow regime of the test section as vertical stratification. However, the flow regime observed in the experiment is the subsonic jet with the bubble having the vertical cone shape. To accurately predict the direct contact condensation in the CMT with RELAP5/MOD3.1, it is essential that a new set of the interfacial heat transfer coefficients and a new flow regime map for direct contact in the CMT be developed.

**NUREG/IA-0135:** POST-TEST ANALYSIS OF PIPER-ONE PO-IC-2 EXPERIMENT BY RELAP5/MOD3 CODES. BOVALINI,R.; AURIA,F.D.; GALASSI,G.M.; et al. Universita' Degli Studi Di Pisa, Pisa, Italy. November 1996. 200pp. 9612110160. CAMP006. 91065:001.

RELAP5/MOD3.1 was applied to the PO-IC-2 experiment performed in PIPER-ONE facility, which has been modified to reproduce typical isolation condenser thermal-hydraulic conditions. RELAP5 is a well known code widely used at the University of Pisa during the past seven years. RELAP5/MOD3.1 was the latest version of the code made available by the Idaho National Engineering Laboratory at the time of the reported study. PIPER-ONE is an experimental facility simulating a General Electric BWR-6 with volume and height scaling ratios of 1/2200 and 1./1, respectively. In the frame of the present activity a once-through heat exchanger immersed in a pool of ambient temperature water, installed approximately 10 m above the core, was utilized to reproduce qualitatively the phenomenologies expected for the Isolation Condenser in the simplified BWR (SBWR). The PO-IC-2 experiment is the flood up of the PO-SD-8 and has been designed to solve some of the problems encountered in the analysis of the PO-SD-8 experiment. A very wide analysis is presented hereafter including the use of different code versions.

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# 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.

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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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NUREG/CR-6309: SCIENTIFIC DESIGN OF PURDUE UNIVERSITY MULTI-DIMENSIONAL INTEGRAL TEST ASSEMBLY (PUMA) FOR GE SBWR.

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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.

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# NRC Contract Sponsor Index (Contractor Reports)

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

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NUREG/CR-6442: EVIDENCE OF AGING EFFECTS ON CERTAIN SAFETY-RELATED COMPONENTS.

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NUREG/CR-6484: GUIDE TO VERIFICATION AND VALIDATION OF THE SCALE-4 RADIATION SHIELDING SOFTWARE.
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NUREG/CR-5985 S01: REVIEW OF P-SCAN COMPUTER-BASED ULTRASONIC INSERVICE INSPECTION SYSTEM.
NUREG/CR-6163: COMPUTER PROGRAMS FOR THE ACQUISITION AND ANALYSIS OF EDDY CURRENT ARRAY PROBE DATA.
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NUREG/CR-6309: SCIENTIFIC DESIGN OF PURDUE UNIVERSITY MULTI-DIMENSIONAL INTEGRAL TEST ASSEMBLY (PUMA) FOR GE SBWR.
NUREG/CR-6338: RESOLUTION OF THE DIRECT CONTAINMENT HEATING ISSUE FOR ALL WESTINGHOUSE PLANTS WITH LARGE DRY CONTAINMENTS OR SUBATMOSPHERIC CONTAINMENTS.
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NUREG/CR-2850 S01: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES.Methodology And Data Base.
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NUREG/CR-6490 V01: NUCLEAR POWER PLANT GENERIC AGING LESSONS LEARNED (GALL). Main Report And Appendix A.
NUREG/CR-6490 V02: NUCLEAR POWER PLANT GENERIC AGING LESSONS LEARNED (GALL). Appendix B.
NUREG/CR-6500: OWNERS OF NUCLEAR POWER PLANTS.

# 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.

**ADVANCED SYSTEMS TECHNOLOGY, INC.**
- NUREG/CR-6353: COMMENTS RECEIVED ON PROPOSED RULE ON RADIOLOGICAL CRITERIA FOR DECOMMISSIONING AND RELATED DOCUMENTS.

**AMERICAN SOCIETY OF MECHANICAL ENGINEERS**
- NUREG/CP-0152: PROCEEDINGS OF THE FOURTH NRC/ASME SYMPOSIUM ON VALVE AND PUMP TESTING.Held At The Hyatt Regency Hotel,Washington,DC, July 15-18, 1996.

**ARGONNE NATIONAL LABORATORY**
- NUREG-1557: SUMMARY OF TECHNICAL INFORMATION AND AGREEMENTS FROM NUCLEAR MANAGEMENT AND RESOURCES COUNCIL INDUSTRY REPORTS ADDRESSING LICENSE RENEWAL.
- NUREG/CR-4667 V20: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,October 1994 - March 1995.
- NUREG/CR-4667 V21: ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. Semiannual Report,April 1995 - December 1995.
- NUREG/CR-5229 V08: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM FOR FISCAL YEAR 1995.Annual Report.
- NUREG/CR-6256 V03: FIELD LYSIMETER INVESTIGATIONS: LOW-LEVEL WASTE DATA BASE DEVELOPMENT PROGRAM LYSIMETER TEST RESULTS FOR FISCAL YEARS 1994 AND 1995.
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- NUREG/CR-6490 V01: NUCLEAR POWER PLANT GENERIC AGING LESSONS LEARNED (GALL). Main Report And Appendix A.
- NUREG/CR-6490 V02: NUCLEAR POWER PLANT GENERIC AGING LESSONS LEARNED (GALL). Appendix B.

**ARIZONA, UNIV. OF, TUCSON, AZ**
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- NUREG/CR-6360: SUMMARY OF AIR PERMEABILITY DATA FROM SINGLE-HOLE INJECTION TESTS IN UNSATURATED FRACTURED TUFFS AT THE APACHE LEAP RESEARCH SITE: RESULTS OF STEADY-STATE TEST INTERPRETATION.
- NUREG/CR-6457: RAPID ESTIMATE OF SOLID VOLUME IN LARGE TUFF CORES USING A GAS PYCNOMETER.
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- NUREG/CR-6438: THE EFFECT OF CYCLIC AND DYNAMIC LOADS ON CARBON STEEL PIPE.
- NUREG/CR-6439: DESIGN OF THE IPIRG-2 SIMULATED SEISMIC FORCING FUNCTION.
- NUREG/CR-6440: THE EFFECTS OF CYCLIC AND DYNAMIC LOADING ON THE FRACTURE RESISTANCE OF NUCLEAR PIPING STEELS.Technical Report:October 1992 - April 1996.
- NUREG/CR-6443: DETERMINISTIC AND PROBABILISTIC EVALUATIONS FOR UNCERTAINTY IN PIPE FRACTURE PARAMETERS IN LEAK-BEFORE-BREAK AND IN-SERVICE FLAW EVALUATIONS.
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- NUREG/CR-5631 R02: CONTRIBUTION OF MATERNAL RADIONUCLIDE BURDENS TO PRENATAL RADIATION DOSES.
- NUREG/CR-5758 V06: FITNESS FOR DUTY IN THE NUCLEAR POWER INDUSTRY.Annual Summary Of Program Performance Reports CY 1995.
- NUREG/CR-5973 R03: CODES AND STANDARDS AND OTHER GUIDANCE CITED IN REGULATORY DOCUMENTS.
- NUREG/CR-5985 S01: REVIEW OF P-SCAN COMPUTER-BASED ULTRASONIC INSERVICE INSPECTION SYSTEM.
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- NUREG/CR-6174 V02: REVISED ANALYSES OF DECOMMISSIONING FOR THE REFERENCE BOILING WATER REACTOR POWER STATION.Effects Of Current Regulatory And Other Considerations On The Financial Assurance Requirements Of The Decommissioning Rule And....
- NUREG/CR-6210: COMPUTER CODES FOR EVALUATION OF CONTROL ROOM HABITABILITY (HABIT).
- NUREG/CR-6227: PERFORMANCE DEMONSTRATION TESTS FOR EDDY CURRENT INSPECTION OF STEAM GENERATOR TUBING.
- NUREG/CR-6230: RADIOANALYTICAL TECHNOLOGY FOR 10 CFR PART 61 AND OTHER SELECTED RADIONUCLIDES.Literature Review.
- NUREG/CR-6270: ESTIMATING BOILING WATER REACTOR DECOMMISSIONING COSTS.A User's Manual For The BWR Cost Estimating Computer Program (CECP) Software.Final Report.
- NUREG/CR-6280: TECHNOLOGY,SAFETY,AND COSTS OF DECOMMISSIONING A REFERENCE LARGE IRRADIATOR AND REFERENCE SEALED SOURCES.
- NUREG/CR-6344: REAL-TIME 3-D SAFT-UT SYSTEM EVALUATION AND VALIDATION.
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## 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.

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# 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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