Regulatory and Technical Reports
(Abstract Index Journal)

Compilation for
First Quarter 1997
January – March

U.S. Nuclear Regulatory Commission
Office of Information Resources Management

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Regulatory and Technical Reports
(Abstract Index Journal)

Compilation for
First Quarter 1997
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M. A. Sheehan, Project Manager

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Office of Information and Resources Management
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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/CRR-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


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


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


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


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


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

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

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

Availability of NRC Publications

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NRC Report Codes

The NUREG designation, NUREG-XXXX, indicates that the document is a formal NRC staff-generated report. Contractor-prepared formal NRC reports carry the report code NUREG/CR-XXXX. This type of identification replaces contractor-established codes such as ORNL/NUREG/TM-XXX and TREE-NUREG-XXX, 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.
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.


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 July through September 1996.


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 - December 1996.


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.


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.


See NUREG-0540,V18,N11 abstract.


See NUREG-0540,V19,N11 abstract.


This report summarizes the occupational exposure data that are maintained in the U.S. Nuclear Regulatory Commission's Radiation Exposure Information and Reporting System (REIRS). The bulk of the information contained in this report was compiled from the 1995 annual reports submitted by the classes of NRC licensees subject to the reporting requirements of 10 CFR 20.2205. Annual reports for 1995 were received from a total of 294 NRC licensees, of which 109 were operators of nuclear power reactors in commercial operation. Compilations of the reports submitted by the 294 licensees indicated that 142,516 individuals were monitored, 76,822 of whom received a measurable dose. The collective dose incurred by these individuals was 24,536 person-cSv (person-rem) which represents a 1% increase from the 1994 value. The number of workers receiving a measurable dose also increased, resulting in the average measurable dose of 0.32 cSv (rem) for 1995. The average measurable dose is defined to be the total collective dose divided by the number of workers receiving a measurable dose. The figures have been adjusted to account for transient reactor workers. In 1995, the annual collective dose per reactor for light water reactor licensees was 193 person-cSv (person-rem). This is the same value that was reported for 1994. The annual collective dose per reactor for boiling water reactors was 256 person-cSv (person-rem) and, for pressurized water reactors it was 170 person-cSv (person-rem). Analyses of transient worker data indicated that 17,153 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 1995, the average measurable dose calculated from reported data was 0.26 cSv (rem). The corrected dose distribution resulted in an average measurable dose of 0.32 cSv (rem).


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.


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


NUREG-0936 V15 NO2: NRC REGULATORY AGENDA.Semiannual Report.July-December 1996. * Rules & Directives Review Branch (Post 920323). March 1997. 58pp. 9704080379. 92889:001. 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 semianually.

NUREG-1021 INT R08: OPERATOR LICENSING EXAMINATION STANDARDS FOR POWER REACTORS. * Office of Nuclear Reactor Regulation (Post 941001). January 1997. 460pp. 9703050343. 91963:001. NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," establishes the policies, procedures, and practices for examining licensees and applicants for reactor operator and senior reactor operator licenses at power reactor facilities pursuant to Title 10, Part 55, of the Code of Federal Regulations (10 CFR Part 55). The examination standards are intended to assist NRC examiners and facility licensees to better understand the processes associated with initial and re-qualification examinations. The standards also ensure the equitable and consistent administration of examinations for all applicants. The standards are for guidance purposes and are not a substitute for the operator licensing regulations (i.e., 10 CFR Part 55), and they are subject to revision or other changes in internal operator licensing policy. This interim revision permits facility licensees to prepare their initial operator licensing examinations on a voluntary basis pending an amendment to 10 CFR Part 55 that will require facility preparation. The NRC intends to solicit comments on this revision during the rulemaking process and to issue a final Revision 8 in conjunction with the final rule.


NUREG-1275 V12: OPERATING EXPERIENCE FEEDBACK REPORT.Assessment Of Spent Fuel Cooling. IBARRA,J.G.; JONES,W.R.; LANIK,G.F.; et al. Division of Safety Programs (Post 870413). February 1997. 48pp. 9703170237. 92131:199. This report is an assessment of the likelihood and consequences of loss of spent fuel pool cooling in the nuclear power industry. A generic pressurized water reactor spent fuel pool configuration is developed and a generic boiling water reactor spent fuel pool configuration is developed. Over twelve years of operational data is reviewed and assessed. Six site visits were conducted to gather specific information on spent fuel pool physical configuration, licensee practices, and licensee procedures. The regulations on spent fuel pools were reviewed. Independent engineering assessments on the spent fuel pool system were performed on the electrical system, instrumentation, heat loads, and radiation. An assessment on the risk of loss of spent fuel cooling was performed. The overall conclusions are that the typical plant may need improvements in spent fuel pool instrumentation, operator procedures and training, and configuration control.

NUREG-1492: REGULATORY ANALYSIS ON CRITERIA FOR THE RELEASE OF PATIENTS ADMINISTERED RADIOACTIVE MATERIAL.Final Report. SCHNEIDER,S.; MCGUIRE,S.A. Division of Regulatory Applications (Post 941217). February 1997. 79pp. 9703200259. 92191:166. This regulatory analysis was developed to respond to three petitions for rulemaking to amend 10 CFR Parts 20 and 35 regarding release of patients administered radioactive material. The petitions requested revision of these regulations to remove the ambiguity that existed between the 1-mSv (0.1-rem) total effective dose equivalent (TEDE) public dose limit in Part 20, adopted in 1991, and the activity-based release limit in 10 CFR 35.75 that in some instances, would permit release of individuals in excess of the current public dose limit. Three alternatives for resolution of the petitions were evaluated. Under Alternative 1, NRC would amend its patient release criteria in 10 CFR 35.75 to match the annual public dose limit in Part 20 of 1 mSv (0.1 rem) TEDE. Alternative 2 would maintain the status quo of using the activity-based release criteria currently found in 10 CFR 35.75. Under Alternative 3, the NRC would revise the release criteria in 10 CFR 35.75 to specify a dose limit of 5 mSv (0.5 rem) TEDE. The evaluation demonstrates that adoption of Alternative 1 would be considerably more expensive to the public compared to Alternative 2 (the status quo), primarily due to increased health care costs associated with more patients remaining in the hospital under the current criteria. Alternative 2 would result in a higher net value to the public compared to Alternative 2 (the status quo), primarily due to lower health care costs and the increased psychological benefits to patients and their families by permitting earlier release from the hospital. Based on this analysis, the decision was made that adoption of the 5-mSv (0.5-rem) TEDE limit is consistent with the provisions in 10 CFR 20.1301(c), and the recommendations of the International Commission on Radiological Protection that an individual be allowed to receive annual doses up to 5 mSv (0.5 rem) TEDE under certain circumstances. Further, it no longer restricts patient release to a specific activity, and therefore, permits release of patients with activities that are greater than currently allowed. The primary benefit is in reduced hospital stays that provide emotional benefits to patients and their families, and result in lower health care costs.

This Final Environmental Impact Statement (FEIS) addresses issues concerning spent fuel storage facilities (ISFSIs) and proposes plans for Federal and Indian lands to Hydro Resources, Inc. (HR). This action would authorize the company to conduct in situ leach uranium mining in McKinley County, New Mexico. Such mining would involve drilling wells to the ore bodies, then recirculating ground water fortified with dissolved oxygen and sodium bicarbonate to mobilize uranium minerals found in the rock. Uranium would then be removed from the aqueous mining solutions using ion exchange technology in processing plants located in three separate project areas. A central plant would provide drying and packaging equipment for yellow-cake production for the entire project.

The FEIS was prepared by a joint interagency review group, including the U.S. Nuclear Regulatory Commission (NRC), the U.S. Bureau of Land Management (BLM) and the U.S. Bureau of Indian Affairs (BIA). This FEIS describes the staff's analyses concerning the evaluation of: (1) the purpose and need for the proposed action; (2) alternatives to the proposed action; (3) the environmental resources that could be affected by the proposed action and alternatives; (4) the potential environmental consequences of the proposed action and alternatives; and (5) the economic costs and benefits associated with the proposed action. The evaluation is based on a comprehensive review of NRC's license application, environmental reports, related submissions, independent information sources, and written and oral communications received on the Draft Environmental Impact Statement. On the basis of its independent review, the staff concludes that the potential significant impacts of the proposed project can be mitigated, and that NRC should be issued a combined source and 116(2) byproduct material license from NRC, and miners operating leases from BLM and BIA.


This final Technical Evaluation Report (TER) 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 to implement 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 concludes that, subject to license conditions identified in the TER, the proposed reclamation plan meets the requirements identified in NRC regulations, which appear primarily in 10 CFR Part 40.


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 operating for Federal and Indian lands to Hydro Resources, Inc. (HR). This action would authorize the company to conduct in situ leach uranium mining in McKinley County, New Mexico. Such mining would involve drilling wells to the ore bodies, then recirculating ground water fortified with dissolved oxygen and sodium bicarbonate to mobilize uranium minerals found in the rock. Uranium would then be removed from the aqueous mining solutions using ion exchange technology in processing plants located in three separate project areas. A central plant would provide drying and packaging equipment for yellow-cake production for the entire project.

This final Technical Evaluation Report (TER) 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 to implement 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 concludes that, subject to license conditions identified in the TER, the proposed reclamation plan meets the requirements identified in NRC regulations, which appear primarily in 10 CFR Part 40.


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 operating for Federal and Indian lands to Hydro Resources, Inc. (HR). This action would authorize the company to conduct in situ leach uranium mining in McKinley County, New Mexico. Such mining would involve drilling wells to the ore bodies, then recirculating ground water fortified with dissolved oxygen and sodium bicarbonate to mobilize uranium minerals found in the rock. Uranium would then be removed from the aqueous mining solutions using ion exchange technology in processing plants located in three separate project areas. A central plant would provide drying and packaging equipment for yellow-cake production for the entire project.

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This document provides guidance and criteria for U.S. Nuclear Regulatory Commission (NRC) personnel to use in evaluating corrective action plans for nuclear power plant communications. The document begins by describing the purpose, scope, and applicability of the evaluation criteria. Next, it presents background information concerning the communications process, root causes of communication errors, and development and implementation of corrective actions. The document then defines specific criteria for evaluating the effectiveness of the corrective action plan, interview protocols, and an observation protocol related to communication processes. This document is intended only as guidance. It is not intended to have the effect of a regulation, and it does not establish any binding requirements or interpretations of NRC regulations.


Exemptions from the requirements for an NRC license to persons who receive, possess, use, transfer, own, or acquire byproduct material in exempt distribution products are provided in 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material." Exempt distribution products include silicon chips, electron tubes, resins, check sources, gun sights, and smoke detectors and are generally distributed by persons who have a specific license from the Commission authorizing such distribution to persons exempt from the requirements for an NRC license. This document provides assistance to applicants and licensees in preparing license applications and describes the methods acceptable to NRC license reviewers in implementing the regulations and the techniques used by the reviewers in evaluating the applications to determine if the proposed exempt distribution activity is acceptable for licensing purposes. The guidance contained herein does not represent new or proposed regulatory requirements, and licensees will not be inspected against any portion of it. In accordance with NRC usage, the word "should" is used when describing the interpreting NRC regulations. Additionally, regulatory compliance with all applicable regulations is not assured by licensees who adopt any portion of, or apply the principles described in, this guidance.


In this information handbook, the staff of the U.S. Nuclear Regulatory Commission describes (1) background information regarding the licensing history of independent spent fuel storage installations (ISFSIs), (2) a discussion of the licensing process, (3) a description of all currently approved or certified models of dry cask storage systems (DCSSs), and (4) a description of sites currently storing spent fuel in an ISFSI. Storage of spent fuel in ISFSIs must be in accordance with the provisions of 10 CFR, Part 72. The information in this handbook is current as of December 1996. The accuracy of any information herein is not guaranteed. For verification or for more details, the reader should refer to the respective docket files for each DCSS and ISFSI site. The information in this handbook is current as of September 1, 1996.
4 Main Citations and Abstracts


The Nuclear Regulatory Commission is issuing this draft Standard Review Plan to describe the procedure used to implement the antitrust review at enforcement prescribed in Sections 105 and 186 of the Atomic Energy Act of 1954, as amended. This draft SRP reflects current regulations and policy, and will be updated to reflect changes in NRC regulations.


The Nuclear Regulatory Commission is issuing this draft Standard Review (SRP) to describe the process it uses to review the financial qualifications and methods of providing decommissioning funding assurance required of power reactor licensees. This draft SRP reflects current regulations and policy, and will be updated to reflect changes in NRC regulations.


An International Workshop on Steam Generator Tube Integrity in Nuclear Power Plants, sponsored by the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI) of the OECD-NEA, was held at Oak Brook (suburban Chicago), Illinois, on October 30-­November 2, 1995. The USNRC Office of Nuclear Regulatory Research served as host. The objective of the workshop was to provide a working forum for the exchange of information by contributing experts on current issues related to PWR steam generator tube integrity. One hundred persons from 15 countries attended the workshop, including 36 from regulatory and nuclear policy agencies, 28 from research and development laboratories, 18 from nuclear vendors and consulting firms, and 18 from electrical utilities. The workshop opened with a plenary session; the first part of the session covered international steam generator tube integrity issues, featuring speakers from regulatory bodies in Belgium, France, Japan, Spain, and the United States. In Part 2 of the plenary session, comprehensive technical overviews on steam generator tube degradation, inspection, and integrity were presented by authorities in these fields from the United States, France, and Belgium. Parallel working sessions on the second and third days of the workshop then developed findings and recommendations in the areas of (1) tube degradation, (2) tube inspection, (3) tube integrity, (4) preventative and corrective measures, and (5) operational aspects and risk analysis. On the final day of the workshop, the working-session facilitators presented summaries of their sessions to the workshop attendees.


This three-volume report contains papers presented at the Twenty-Fourth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 21-23, 1996. 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 Finland, France, Japan, Norway, Russia and the United Kingdom. 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-0157 V02: PROCEEDINGS OF THE TWENTY­FOURTH WATER REACTOR SAFETY INFORMATION MEETING. Reactor Pressure Vessel Embrittlement And Thermal Annealing. Reactor Vessel Lower Head Integrity And Evaluation And Projection of Steam Generator tube... MONTELEONE, S. Brookhaven National Laboratory. February 1997. 444pp. 9703120276. 92075.001.

See NUREG/CP-0157 V01 abstract.

NUREG/CP-0157 V03: PROCEEDINGS OF THE TWENTY­FOURTH WATER REACTOR SAFETY INFORMATION MEETING. PRA And HRA, And Probabilistic Seismic Hazard As­sessment And Seismic Site Criteria. MONTELEONE, S. Brook­haven National Laboratory. February 1997. 160pp. 9703120285. 92076.081.

See NUREG/CP-0157 V01 abstract.


The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission (NRC) by Oak Ridge National Laboratory (ORNL). The program's focus is on the development and validation of technology for the assessment of fracture-prevention margins in commercial nuclear reactor 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) duct­tile-to-cleavage fracture-mode conversion, (5) fracture analysis methodologies 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 of fracture mechanics 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 1995 through September 1995.


This is the sixth volume in a series of reports that provide information on dose reduction research and health physics technology for nuclear power plants. The information is taken from two of several databases maintained by Brookhaven National Laboratory's ALARA Center for the U.S. Nuclear Regulatory Commission. The research section of the report covers dose reduction projects that are in the experimental or development phase. It includes topics such as need for cost-effective measures to control radiation fields, the highly effective full-system decontamination, progress in addressing the increase in radiation fields upon switching from normal water chemistry to hydrogen water chemistry in BWRs, addition of depleted zinc to reduce radiation fields, and cobalt free wear-resistant alloys. The section on health physics technology discusses dose reduction efforts that are in place or in the process of being implemented at nuclear power plants. A total of 67 new or updated projects are described. The appendix provides a complete listing of all the material in this area, including that from previous reports. The material is available through fax machine from our ACEFAX on-line service. The procedure for accessing ACEFAX is also described.

As part of the Nondestructive Evaluation Reliability Program sponsored by the U.S. Nuclear Regulatory Commission, the Pacific Northwest National Laboratory has developed risk-informed approaches for inspecting plant-specific inspection plans of nuclear power plants. This method uses probabilistic risk assessment (PRA) results to identify and prioritize the most risk-important components for inspection. The Surry Nuclear Power Station Unit 1 was selected for pilot application of this methodology. This report, which incorporates more recent plant-specific information and improved risk-informed methodology and tools, is Revision 1 of the earlier report (NUREG/CR-6181). The methodology discussed in the original report is no longer current and a preferred methodology is presented in this Revision. This report, NUREG/CR-6181, Rev. 1, therefore supersedes the earlier NUREG/CR-6181 published in August 1994. The specific systems addressed in this report are the auxiliary feedwater, the lower plenum, and the reactor coolant systems. The results provide a risk-informed ranking of components within these systems.


Brookhaven National Laboratory has performed a series of probabilistic consequence assessment calculations for nuclear reactor siting. This study takes into account recent insights into severe accident source terms and examines consequences in a risk-based format consistent with the quantitative health objectives (QHOs) of the NRC's Safety Goal Policy. Simplified severe accident source terms developed in this study are based on the risk insights of NUREG-1150 and compared to those used in earlier studies, particularly the Sandia Siting Study. The results of the present study indicate that both the quantity of radioactivity released in a severe accident as well as the likelihood of a release are lower than those predicted in earlier studies. The accident risks using the simplified source terms are examined at a series of generic plant sites that vary in population distribution, meteorological characteristics, and exclusion boundary distances. Sensitivity calculations are performed to evaluate the effects of emergency protective action assumptions on the risk of prompt fatality and latent cancers fatality, and population relocation. The study finds that based on the new source terms, the prompt and latent fatality risks at all generic sites meet the QHOs of the NRC's Safety Goal Policy by margins ranging from one to more than three orders of magnitude.


The BLOCKAGE 2.5 code described in this User's Manual was developed by the United States Nuclear Regulatory Commission (NRC) as a tool to evaluate licensee compliance with NRC Bulletin 98-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boring Water Reactors". As such, BLOCKAGE 2.5 provides a generalized framework into which a user can input plant-specific and insulation-specific data for performing analyses in accordance with Regulatory Guide 1.82, Rev. 2. This user's manual describes the capabilities of BLOCKAGE 2.5 along with a description of the graphics user's interface provided for data entry. Each input/output dialog is described in detail along with special considerations related to developing and executing BLOCKAGE. Also, several sample problems are provided such that user can easily modify them to suit a particular plant of interest. The models used in BLOCKAGE 2.5 and their validation are presented in the accompanying NUREG/CR-6371. The BLOCKAGE 2.5 models were designed to be parametric in nature, allowing the user flexibility to examine the impact of several modeling assumptions and to conduct sensitivity analyses. As a result, BLOCKAGE 2.5 results are known to be very sensitive to the user provided input. It is therefore strongly recommended that users become thoroughly familiar with BLOCKAGE models and their limitations as described in NUREG/CR-6224.


The BLOCKAGE 2.5 code was developed by the United States Nuclear Regulatory Commission (NRC) as a tool to evaluate licensee compliance regarding the design of suction strainers for emergency core cooling system (ECCS) pumps in boiling water reactors (BWR) as required by NRC Bulletin 98-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boring Water Reactors." Science and Engineering Associates, Inc. (SEA) and Software Edge, Inc. (SE) developed this PC-based code. The Instruction Manual on CD-ROM and this code to evaluate the potential of debris to sufficiently block a pump suction strainer such that a pump could lose NPSH margin was documented in a User's Manual [NRC, NUREG/CR-6370]. The Reference Manual contains additional information that supports the use of BLOCKAGE 2.5. It contains descriptions of the analytical models contained in the code, programmers illustrating the structure of the code, and summaries of coding verification and model validation exercises that were performed to ensure that the analytical model is correctly coded and applicable to the evaluation of BWR pump suction strainers. The BLOCKAGE code was developed by SEA and programmed in FORTRAN as a code that can be executed from the DOS level on a PC. A graphical users interface (GUI) was then developed by SEA to make BLOCKAGE easier to use and to provide graphical output capability. The GUI was programmed in the C language. The user has the option of executing BLOCKAGE 2.5 with the GUI or from the DOS level and the Users Manual provides instructions for both methods of execution.


To examine the potential for using subsize Charpy specimens to evaluate the material properties of vessel materials for life extension, a study was conducted on the behavior of subsize impact specimens of five different geometries. Effects of notch depth, angle, and radius, as well as overall specimen dimensions were determined. Correlations of the transition temperature determined by the different subsize specimens as compared to full-size specimens were evaluated. A new procedure for transforming data from subsize specimens was developed.


This report presents the results from Task 1 of the Second International Pipeline Integrity Research Group (IPIRG-2) program. The rationale for and objective of Task 1 was to build on the results of the first IPIRG-1 program by evaluating: (1) the fracture behavior of circumferentially cracked pipe subjected to more complex load histories, such as simulated seismic load histories; (2) cracks at geometric discontinuities, such as elbow...

The U.S. Nuclear Regulatory Commission reviews the human factors engineering (HFE) aspects of advanced nuclear power plant designs. In order to support the advanced reactor design certification review process, the HFE Program Certification Review Model was developed. The model describes the HFE program elements that are necessary and sufficient to develop an acceptable detailed design and provides the review criteria for their evaluation. One of the review elements is verification and validation. The purpose of this document is to discuss the detailed methodological considerations necessary for a review of an HFE integrated system validation. A conceptual approach, or paradigm, to integrated system validation is presented which identifies important validation principles and their relationships. The validation paradigm was used to identify the methodological aspects of the validation process that are needed to meet the general paradigm requirements. The methodology must support a logical and defensible inference to be made from validation tests to predicted integrated system performance under actual operating conditions. The validation paradigm is based upon four general forms of validity: system representation, performance representation, test design and statistical conclusion validity. Validating an integrated system is based on establishing that these four types of validity are satisfied. Such assessments are made by reviewing the methodology used to conduct validation tests. Methodological factors relevant to each of the aspects of validity are discussed.


For many years, protecting the fetus has been a concern of the National Council on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP). Early recommendations focused on the possibility of a wide variety of detrimental developmental effects while later recommendations focused on the potential for severe mental retardation and/or reduction in the intelligence quotient (I.Q.). The latest recommendations also note that the risk of cancer or the fetus is probably two to three times greater for a three-month gestation than in the adult. For all these reasons, the NCRP and the ICRP have provided guidance to physicians on taking all reasonable steps to ascertain whether any woman requiring a radiological or nuclear medicine procedure is pregnant or nursing a child. The NCRP and the ICRP also advise the clinician to postpone such procedures until after delivery or cessation of nursing, if possible.


The NRC Human Factors Engineering Program Review Model (HFE PRM, NUREG-0711) was developed to support a design process review for advanced reactor design certification under 10CFR52. The HFE PRM defines ten fundamental elements of a human factors engineering program. An Operating Experience Review (OER) is one of these elements. The main purpose of an OER is to identify potential safety issues from operating plant experience and ensure that they are addressed in a new design. Broad-based experience reviews have typically been performed in the past by reactor designers. For the HFE PRM, the intent is to have a more focussed OER that concentrates on HFE issues or experience that would be relevant to the human-system interface (HSI) design process for new advanced reactors. This document provides a detailed list of HFE-relevant operating experience pertinent to the HSI design process for advanced nuclear power plants. This document is intended to be used by NRC reviewers as part of the HFE PRM review process in determining the completeness of an OER performed by an applicant for advanced reactor design certification.


To satisfy the need for verification of the computer programs and modeling techniques that will be used to perform the final piping analyses for the Westinghouse AP600 Standardized Plant, three benchmark problems were developed. The problems are representative piping systems subjected to representative dynamic loads with solutions developed using the methods being proposed for analysis for the AP600 standard design. It will be required that the combined licensees demonstrate that their solutions to these problems are in agreement with the benchmark problem set.


The objective of this work was to develop ductile fracture toughness data in the form of J-R curves for modified A 302 grade B plate materials typical of those used in fabricating reactor pressure vessels. A previous experimental study at Materials Engineering Associates, Lanham, Maryland, on one particular heat of A 302 grade B plate showed decreasing J-R curves with increased specimen thickness. This characteristic has not been observed in numerous tests made on the more recent production materials of A 533 grade B and A 508 class 2 pressure vessel steels. It was unknown if the departure from norm for the MEA material was a generic characteristic for all heats of A 302 grade B steels or just unique to that one particular plate.


The objective of this work was to develop ductile fracture toughness data in the form of J-R curves for modified A 302 grade B plate materials typical of those used in fabricating reactor pressure vessels. A previous experimental study at Materials Engineering Associates (MEA) on one particular heat of A 302 grade B plate showed decreasing J-R curves with increased specimen thickness. This characteristic has not been observed in numerous tests made on the more recent production materials of A 533 grade B and A 508 class 2 pressure vessel steels. It was unknown if the departure from norm for the MEA material was a generic characteristic for all heats of A 302 grade B steels or just unique to that one particular plate. Seven heats of modified A 302 grade B steel and one heat of vintage A 533...
grade B steel were provided to this project by the General Electric Company of San Jose, California. All plates were tested for chemical content, tensile properties, Charpy transition temperature curves, drop-weight nil-ductility transition (NDT) temperature, and J-R curves. Tensile tests were made in the three principal orientations and at four temperatures, ranging from room temperature to 550 degrees F (288 degrees C). Charpy V-notch transition temperature curves were obtained in longitudinal, transverse, and short transverse orientations. J-R curves were made using four specimen sizes (1/2T, IT, 2T, and 4T). The fracture mechanics-based evaluation method covered three test orientations and three test temperatures [180, 400, and 550 degrees F (82, 204, and 288 degrees C)]. However, the coverage of these variables was contingent upon the amount of material provided. Drop-weight NDT temperature was determined for the T-L orientation only. None of the seven heats of modified A 302 grade B showed size effects of any consequence on the J-R curve behavior. Crack orientation effects were present, but none were severe enough to be reported as atypical. A test temperature increase from 180 to 550 degrees F (82 to 288 degrees C) produced the usual loss in J-R curve fracture toughness. Gener- ic J-R curves and mathematical curve fits to the same were generated to represent each heat of material. Volume 1 deals with evaluation of data and discussion of technical findings. This volume (Volume 2) is a compilation of all data developed.


In SECY-80-018, the NRC proposed a safety goal of a condi- tional containment failure probability (CCFP) of 0.1 and the alternative acceptance criteria allowed for steel containments, which specifies that the stresses should not exceed ASME Level C allowable for severe accident pressures and tempera- tures. In this work, the need for an equivalent criterion for con- crete containments was studied. Six surrogate containments were designed and analyzed in order to compare the margins between design pressure, pressure resulting in exceedance of Level C (or yield) stress limits, and ultimate pressure. For compar- ability, each containment has an identical internal volume and design pressure. Results from the analysis showed margins to yield are comparable and display a similar margin for both steel and concrete containments. In addition, the margin to fail- ure, although slightly higher in the steel containments, were also comparable. Results of a CCFP for containment design was determined based on general membrane behavior and imposing an upper bound severe accident curve developed in the DCH studies. The resulting CCFPs were less than 0.02 (or 2%) for all the surrogate containments studied, showing that these contain- ment designs all achieved the NRC safety goal.


In the IPIRG-1 program, the J-R curve calculated for a 16-inch nominal diameter, Schedule 100 TP304 stainless steel (DP2-A8) surface-cracked pipe experiment (Experiment 1.3-3) was considerably lower than the quasi-static, monotonic J-R curve calculated from a C(T) specimen (A8-12a). The results from several related investigations conducted to determine the cause of the observed toughness difference are: (1) Chemical analyses on sections of Pipe DP2-A8 from several surface-cracked pipe and material property specimen fracture surfaces indicate that there are two distinct heats of material within Pipe DP2-A8 that differ in chemical composition. (2) SEN(T) specimen experimental results indicate that the toughness of a sur- face-cracked specimen is highly dependent on the depth of the initial crack. In addition, the J-R curves from the SEN(T) speci- mens closely match the J-R curve from the surface-cracked pipe experiment. (3) C(T) experimental results suggest that there is a large difference in the quasi-static, monotonic tough- ness between the two heats of DP2-A8, as well as a toughness degradation in the lower toughness heat of material (DP2-A8II) when loaded with a dynamic, cyclic (R = -0.3) loading history.


The IPIRG-2 program was an international group program managed by the U.S. NRC and funded by organizations from 15 nations. The emphasis of the IPIRG-2 program was the development of data to verify fracture analyses for cracked pipes and fittings subjected to dynamic/cyclic load histories typical of seismic events. The scope included: (1) the study of more complex dynamic/cyclic load histories, i.e., multi-frequency, variable am- puls, simulated seismic excitations, than those considered in the IPIRG-1 program, (2) crack sizes more typical of those con- sidered in Leak-Before-Break (LBB) and in-service flaw evaluations, (3) through-wall-cracked pipe experiments which can be used to validate LBB-type fracture analyses, (4) cracks in and around pipe fittings, such as elbows, and (5) laboratory specimen and separate effect pipe experiments to provide better in- sight into the effects of dynamic and cyclic load histories. Also undertaken were an uncertainty analysis to identify the issues most important for LBB or in-service flaw evaluations, updating computer codes and databases, the development and conduct of a series of round-robin analyses, and analyst's group meet- ings to provide a forum for nuclear piping experts from around the world to exchange information on the subject of pipe frac- ture technology.


Review of industry efforts to manage thermal fatigue, flow-ac- celerated corrosion, and steam generator water damage to Pressurized Water Reactor (PWR) feedwater nozzles, piping, and feedings is presented in this report. The review includes an evaluation of design modifications, operating procedure changes, augmented inspection and monitoring pro- grams, and mitigation, repair and replacement activities. Four specific actions were taken to perform the evaluation (a) review of field experience to identify trends of operating events; (b) review of the related technical literature; (c) visits to three PWR plants and a PWR vendor; and (d) solicitation of information from foreign utilities. Our assessment of field experience indi- cates the USNRC licensees have apparently taken sufficient action to minimize the feedwater nozzle cracking caused by thermal fatigue, wall thinning of J-tubes and feedwater piping, and steam generator water hammer in both top-feed and pre- heat steam generators. A major finding of this review is that the analysis, inspection, monitoring, mitigation, and replacement techniques have been developed for managing thermal fatigue and flow-accelerated corrosion damage to feedwater nozzles, piping, and feedings. Adequate training and appropriate applica- tions of these techniques would ensure effective manage- ment of this damage. Several PWR plant operators have been proactive in managing this damage.

NUREG/CR-6469: EXPERIMENTS TO INVESTIGATE DIRECT
8 Main Citations and Abstracts


The Surtsey Test Facility at Sandia National Laboratories (SNL) is used to perform scaled experiments for the Nuclear Regulatory Commission (NRC) that simulate High Pressure Melt Ejection (HPME) accidents in a nuclear power plant (NPP). These experiments are designed to investigate the effects of direct containment heating (DCH) phenomena on the containment load. In previous experiments, high-temperature, chemically reactive (thermitic) metal was ejected by high-pressure steam into a scale model of either the Zion or Surry NPP. The results from the Zion and Surry experiments were extrapolated to other Westinghouse plants. This report describes tests performed with Combustion Engineering plant geometries (in particular, Calvert Cliffs-like) and the impact of codispersed water as part of the overall DCH issue resolution. Integral effects tests were performed with a 1/10th scale model of the Calvert Cliffs NPP inside the Surtsey test vessel. The objective was to investigate the effects of codispersion of water, steam, and molten core simulant materials on DCH loads under prototypic accident conditions and plant configurations. The results indicated that large amounts of codispersed water reduced the DCH load by a small amount. Large amounts of debris were dispersed from the cavity to the upper dome (via the annular gap).


Phenomena Identification and Ranking Tables (PIRT) have been developed for a start-up transient for SBWR. The information used for PIRT came from RAMONA-4B and TRACG analyses of the transient and from related small scale tests. The transient was divided into four distinct phases, namely, Sub-Cooled Core Heat-up, Subcooled Chirrney, Saturated Chirrney, and Power Ascension. The assessment criterion selected was Minimum Critical Power Ratio. The SBWR system was divided into ten components. A total of 35 distinct phenomena among the core components were identified. The Phase I has 26 ranked phenomena with 17 low, 6 medium, and 5 high ranking. Phase II has 39 ranked phenomena with 18 low, 13 medium, and 8 high ranking. The Phase III has 47 ranked phenomena with 22 low, 10 medium and 15 high ranking. The Phase IV has 48 ranked phenomena with 16 low, 12 medium and 18 high ranking.


Modular construction techniques have been successfully used in a number of industries, both domestically and internationally. Recently, the use of structural modules has been proposed for advanced nuclear power plants. The objective in utilizing modular construction is to reduce the construction schedule, reduce construction costs, and improve the quality of construction. This report documents the results of a program which evaluated the proposed use of modular construction for safety-related structures in advanced nuclear power plant designs. The program included a review of current design techniques, development of licensing review criteria for modular construction, and initial validation of currently available analytical techniques applied to concrete-filled steel structural modules. The program was conducted in three phases. The objective of the first phase was to identify the technical issues and the need for further study in order to support NRC licensing review activities. The two key findings were the need for supplementary review criteria to augment the Standard Review Plan and the need for verified design/analysis methodology for unique types of modules, such as the concrete-filled steel module. In the second phase of this program, Modular Construction Review Criteria were developed to provide guidance for licensing reviews. In the third phase, an analysis effort was conducted to determine if currently available finite element analysis techniques can be used to predict the response of concrete-filled steel modules.


Exposure of the hands of medical personnel administering radiolabeled antibodies (RABS) was evaluated on the basis of (a) observing and photo-documenting administration techniques, and (b) experimental data on doses to thermoluminescent dosimeters (TLDs) on fingers of phantom hands holding syringes, and on syringes, with radionuclides in the syringes in each case. Dose rate coefficients to the skin, if in contact with the syringe wall, were 89, 1.9, 3.8, and 0.41 uSv s(-1) averaged over 1 CM(2) at 7 mg CM(-2) per 37 MBq (1 mCi) for Y-90, Tc-99m, I-131, and Lu-177, respectively. When using Y-90 the importance of avoiding direct contact with syringes containing RABS and of using a beta-particle shield on the syringe was indicated. In using a syringe for injection, doses can be kept to a small fraction of the geometry studied by (a) wearing a finger dosimeter on the middle finger, toward the outside of the hand, on the hand operating the plunger, and (b) wearing finger dosimeters on the inner (palm) side of the finger on the hand that supports the syringe for energetic beta-particle emitters, such as Y-90 and Re-188.


This annual status report for fiscal year 1996 documents technical work performed on ten key technical issues (KTIs) that are most important to performance of the proposed geologic repository at Yucca Mountain. This report was prepared jointly by the staff of the Nuclear Regulatory Commission (NRC) Division of Waste Management and the Center for Nuclear Waste Regulatory Analyses. The programmatic aspects of restructuring the NRC repository program in terms of KTIs is discussed and a brief summary of work accomplished is provided in Chapter 1. The other ten chapters provide a comprehensive summary of the work in each KTI. Discussions on probability of future volcanic activity and its consequences, impacts of structural deformation and seismicity, the nature of the near-field environment and its effects on container life and source term, flow and transport including effects of thermal loading, aspects of repository design, estimates of system performance, and activities related to the U.S. Environmental Protection Agency standard are provided.


This Environmental Assessment was prepared to evaluate environmental issues associated with the renewal of NRC Licensees Nos. SMB-179 and SUB-1452 for facilities operated by Nuclear Metals, Inc. (NMI) in Concord Massachusetts. License renewal was needed to permit the continuation of NMI operations involving depleted and natural uranium.

This report documents the results of a program which evaluated the proposed use of modular construction for safety-related structures in advanced nuclear power plant designs. The program included a review of current design techniques, development of licensing review criteria for modular construction, and initial validation of currently available analytical techniques applied to concrete-filled steel structural modules. The program was conducted in three phases. The objective of the first phase was to identify the technical issues and the need for further study in order to support NRC licensing review activities. The two key findings were the need for supplementary review criteria to augment the Standard Review Plan and the need for verified design/analysis methodology for unique types of modules, such as the concrete-filled steel module. In the second phase of this program, Modular Construction Review Criteria were developed to provide guidance for licensing reviews. In the third phase, an analysis effort was conducted to determine if currently available finite element analysis techniques can be used to predict the response of concrete-filled steel modules.
Although the average strain rate in intraplate settings is 2-3 orders of magnitude lower than at plate boundaries, there are pockets of high strain rates within intraplate regions. The results of a Global Positioning System survey near the location of current seismicity (and the inferred location of the destructive 1886 Charleston, South Carolina earthquake) suggest that there is anomalous strain build-up occurring there. By reoccupying 1930 triangulation and 1980 GPS sites with six Trimble SST dual frequency receivers, a strain rate of $0.4 \times 10^{-7}$ yr$^{-1}$ was observed. At the 95% confidence level, this value is not significant; however, at a lower level of confidence ($\sim 85\%$) it is about two orders of magnitude greater than the background of $10^{-9}$ to $10^{-10}$ yr$^{-1}$. The direction of contraction inferred from the GPS survey 66 degrees $\pm$ 11 degrees is in excellent agreement with the direction of the maximum horizontal stress (N 60 degrees E) in the area, suggesting that the observed strain rate is also real.
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NUREG/CR-0444: PIPING BENCHMARK PROBLEMS FOR THE WESTINGHOUSE APE60 STANDARDIZED PLANT.

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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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- DIVISION OF BUDGET & ANALYSIS (POST 890205) NUREG-1100 V16: BUDGET ESTIMATES. Fiscal Year 1996.

**EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA**
- DIVISION OF SAFETY PROGRAMS (POST 870418) NUREG-1275 V12: OPERATING EXPERIENCE FEEDBACK REPORT. Assessment of Spent Fuel Cooling.

**EDO - OFFICE OF INFORMATION RESOURCES MANAGEMENT & ARM (POST 861100)**
- NUREG-0750 V44 N01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES. July-September 1996.
- NUREG-0750 V44 N02: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR NOVEMBER 1996, Pages 229-314.
- NUREG-0750 V44 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR DECEMBER 1996, Pages 315-432.
- NUREG-0750 V44 N01: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR JANUARY 1997, Pages 1-47.

**EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS**
- NUREG-1571: INFORMATION HANDBOOK ON INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS.
- DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870209)
- NUREG-1580 DRFT FC: STANDARD REVIEW PLAN FOR APPLICATIONS FOR LICENSES TO DISTRIBUTE BYPRODUCT MATERIAL TO PERSONS EXEMPT FROM THE REQUIREMENTS FOR AN NRC LICENSE. 10 CFR Parts 30.14, 30.15, 30.16, 30.18, 30.19 & 30.20.
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- DIVISION OF SAFETY PROGRAMS (POST 870413)
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- DIVISION OF ENGINEERING TECHNOLOGY (POST 941017)
  - NUREG/CR-6181 R01: A PILOT APPLICATION OF RISK-INFORMED METHODS TO ESTABLISH INSERVICE INSPECTION PRIORITIES FOR NUCLEAR COMPONENTS AT SURRY UNIT 1 NUCLEAR POWER STATION.
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  - NUREG/CR-6446: FRACTURE TOUGHNESS EVALUATIONS OF TP304 STAINLESS STEEL PIPES.
  - NUREG/CR-6456: ASSESSMENT OF MODULAR CONSTRUCTION FOR SAFETY-RELATED STRUCTURES AT ADVANCED NUCLEAR POWER PLANTS.
  - NUREG/CR-6529: VALIDATION OF TECTONIC MODELS FOR AN INTRAPLATE SEISMIC ZONE, CHARLESTON, SOUTH CAROLINA WITH GPS GEODETIC DATA.
- DIVISION OF REGULATORY APPLICATIONS (POST 941217)
- DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)
  - NUREG/CR-6393: INTEGRATED SYSTEM VALIDATION METHODOLOGY AND REVIEW CRITERIA.
  - NUREG/CR-6400: HUMAN FACTORS ENGINEERING (HFE) INSIGHTS FOR ADVANCED REACTORS BASED UPON OPERATING EXPERIENCE.
  - NUREG/CR-6414: PIPING BENCHMARK PROBLEMS FOR THE WESRINGHOUSE AP600 STANDARDIZED PLANT.
**Contractor Index**

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

ARGONNE NATIONAL LABORATORY  
NUREG/CP-0154: PROCEEDINGS OF THE CNRA/CSNI WORKSHOP ON STEAM GENERATOR TUBE INTEGRITY IN NUCLEAR POWER PLANTS.

BATTelle MEMORIAL INSTITUTE, COLUMBUS LABORATORIES  
NUREG/CR-6446: FRACTURE TOUGHNESS EVALUATIONS OF TP304 STAINLESS STEEL PIPES.  

BATTelle MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORY  
NUREG/CR-6181 R01: A PILOT APPLICATION OF RISK-INFORMED METHODS TO ESTABLISH INSERVICE INSPECTION PRIORITIES FOR NUCLEAR COMPONENTS AT SURRY UNIT 1 NUCLEAR POWER STATION.

BROOKHAVEN NATIONAL LABORATORY  
NUREG/CP-0157 V01: PROCEEDINGS OF THE TWENTY-FOURTH WATER REACTOR SAFETY INFORMATION MEETING.Plenary Session, High Burnup Fuel, Containment And Structural Aging.

SCIENCE APPLICATIONS, INC.  
NUREG/CR-6414: PIPING BENCHMARK PROBLEMS FOR THE WESTINGHOUSE AP600 STANDARDIZED PLANT.

SCIENCE APPLICATIONS, INC.  
NUREG/CR-6474: PRELIMINARY PHENOMENA IDENTIFICATION AND RANKING TABLES (PIRT) FOR SBWR STARTUP STABILITY.

BOKA RIDGE NATIONAL LABORATORY  

BROOKHAVEN NATIONAL LABORATORY  
NUREG/CP-0157 V02: PROCEEDINGS OF THE TWENTY-FOURTH WATER REACTOR SAFETY INFORMATION MEETING.Reactor Pressure Vessel Embrittlement And Thermal Annealing,Reactor Vessel Lower Head Integrity And Evaluation And Projection Of Steam Generator tube...

SCIENCE APPLICATIONS, INC.  
NUREG/CP-0157 V03: PROCEEDINGS OF THE TWENTY-FOURTH WATER REACTOR SAFETY INFORMATION MEETING.Pla and HRA, And Probabilistic Seismic Hazard Assessment And Seismic Siting Criteria.

BROOKHAVEN NATIONAL LABORATORY  
NUREG/CR-4409 V06: DATA BASE ON DOSE REDUCTION PROJECTS FOR NUCLEAR POWER PLANTS.

SANDIA NATIONAL LABORATORIES  
NUREG/CR-6433: CONTAINMENT PERFORMANCE OF PROTOTYPE REACTOR CONTAINMENTS SUBMITTED TO SEVERE ACCIDENT CONDITIONS.

SANDIA NATIONAL LABORATORIES  
NUREG/CR-6456: REVIEW OF INDUSTRY EFFORTS TO MANAGE PRESSURIZED WATER REACTOR FEEDWATER NOZZLE, PIPING, AND FEEDING CRACKING AND WALL THINNING.

SANDIA NATIONAL LABORATORIES  
NUREG/CR-6439: EXPERIMENTS TO INVESTIGATE DIRECT CONTAINMENT HEATING PHENOMENA WITH SCALED MODELS OF THE CALVERT CLIFFS NUCLEAR POWER PLANT.

SCIENCE & ENGINEERING ASSOCIATES, INC.  
NUREG/CR-6370: BLOCKAGE 2.5 USER'S MANUAL.

SCIENCE APPLICATIONS INTERNATIONAL CORP. (FORMERLY SCIENCE APPLICATIONS)  

SOFMlARE EDGE, INC.  
NUREG/CR-6372: BLOCKAGE 2.5 USER'S MANUAL.

SOUTH CAROLINA, UNIV. OF, COLUMBIA, SC  
NUREG/CR-6526: VALIDATION OFTECTONIC MODELS FOR AN INTRAPLATE SEISMIC ZONE,CHARLESTON,SOUTH CAROLINA WITH GPS GEODETIC DATA.
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.

There were no NUREG/IA reports published this quarter.
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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## Regulatory and Technical Reports (Abstract Index Journal)

Compilation for First Quarter 1997
January – March

### 5. Author(s)

M. A. Sheehan, Project Manager

### 11. Abstract (200 words or less)

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.

### 12. Keywords/Descriptors (List words or phrases that will assist researchers in locating the report.)

- compilation
- abstract index