Regulatory and Technical Reports (Abstract Index Journal)

Compilation for
Second Quarter 1995
April – June

U.S. Nuclear Regulatory Commission

Office of Administration
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A year's subscription of this report consists of four quarterly issues.
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PREFACE

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

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

- Secondary Report Number Index
- Personal Author Index
- 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).
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 Technical Publications Section of the NRC Division of Freedom of Information and Publications Services.
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.


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 1994) and cumulative data, usually from the date of commercial operation. The data is not independently verified, but various computer checks are made. The report is divided into two sections. The first contains summary highlights and the second contains data on each individual unit in commercial operation. Section 1 capacity and availability factors are simple arithmetic averages. Section 2 items in the cumulative column are generally as reported by the licensee and notes as to the use of weighted averages and starting dates other than commercial operation are provided.


This periodic 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 1995.


The Final Environmental Statement (FES) issued in 1978 represents the Nuclear Regulatory Commission's (NRC's) previous environmental review related to the operation of Watts Bar Nuclear (WBN) Plant. The purpose of this NRC review is to discuss the effects of observed changes in environment and to evaluate the changes in environmental impacts that have occurred as a result of changes in the WBN Plant design and proposed methods of operation since the last environmental review. A full scope of environmental topics has been evaluated, including regional demography, land and water use, meteorology, terrestrial and aquatic ecology, radiological and nonradiological impacts on humans and the environment, socioeconomic impacts, and environmental justice. The staff concluded that there are no significant changes in the environmental impacts since the NRC 1978 FES-CI from changes in plant design, proposed methods of operation, or changes in the environment. The applicant's preoperational and operational monitoring programs were reviewed and found to be appropriate for establishing baseline conditions and ongoing assessments of environmental impacts. The staff also conducted an analysis of plant operation with severe accident mitigation design alternatives (SAMDAs) and concluded that none of the SAMDAs, beyond the three procedural changes that the applicant committed to implement, would be cost-beneficial for further mitigating environmental impacts.


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 1994. It also lists detailed highway and railway segments used within each state from January 1, 1993, through December 31, 1994.


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.

Main Citations and Abstracts

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


See NUREG-0750,V41,N02 abstract.


See NUREG-0750,V41,N02 abstract.


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 1995.


Supplement No. 15 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. 14 was issued, and (2) matters that the staff had under review when Supplement No. 14 was issued.


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.


This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January-March 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.


This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January-March 1995) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to medical 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.


This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January-March 1995) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to Material Licensees (non-Medical) 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.


This compilation contains 30 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1994. 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 categorized by the most appropriate generic subject area and by chronologi- cal order within subject area.


The report presents the results of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEDO) review of operating experience of nine turbine-generator overspeed and overspeed protection systems. AEDO's study provides insight into the shortcomings in the design, operation, maintenance, testing, and human factors associated with turbine overspeed protection systems. It includes an indepth examination of the turbine overspeed event that occurred on November 9, 1991, at the Salam Unit 2 Nuclear Power Plant. It also provides information concerning actions taken by other utilities and the turbine manufacturers as a result of the Salem overspeed event. AEDO's study reviewed operating procedures and plant practices. It noted differences between turbine manufacturer designs and recommendations for operations, maintenance, and testing, and also identified significant variations in the manner that individual plants maintain and test their turbine overspeed protection systems.

In 1987 the NRC revised the material control and accounting requirements for NRC licensees authorized to possess and use a formula quantity (i.e., 5 formula kilograms or more) of strategic special nuclear material. Those revisions issued as 10CFR 47.51-59 require timely monitoring of in-process inventory and discrete items to detect anomalies potentially indicative of material losses. Timely detection and enhanced loss localization capabilities are beneficial to alarm resolution and also for material recovery in the event of an actual loss. NUREG-1280 was issued in 1987 to present criteria that could be used by applicants, licensees, and NRC license reviewers in the initial preparation and subsequent review of fundamental material control (FMNC) plans submitted in response to the Reform Amendment. This document is also intended for both licensees and license reviewers with respect to FNMC plan revisions. General performance objectives, system capabilities, process monitoring, item monitoring, alarm resolution, quality assurance, and accounting are addressed. This revision to NUREG-1280 is an expansion of the initial edition, which clarifies and expands upon several topics and addresses issues identified under Reform Amendment implementation experience.


This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for Babcock & Wilcox Plants. This revision to NUREG-1280 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.9 of the improved STS.


This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS), Revision 1 for Westinghouse Plants. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.9 of the improved STS.
result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.9 of the improved STS.


NUREG-1433 V01 R01: STANDARD TECHNICAL SPECIFICATIONS GENERAL ELECTRIC PLANTS BWR/4 Specifications. * Office of Nuclear Reactor Regulation (Post 941001). April 1995. 388pp. 9506290155. 84461:001. This report documents the results of the combined effort of the NRC and the industry to produce improved Standard Technical Specifications (STS). Revision 1 for General Electric BWR/4 Plants. The changes reflected in Revision 1 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. This report is the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, NSSS vendors, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993. The improved STS will be used as the basis for individual nuclear power plant licensees to develop improved plant-specific technical specifications. This report contains three volumes. Volume 1 contains the Specifications for all chapters and sections of the improved STS. Volume 2 contains the Bases for Chapters 2.0 and 3.0, and Sections 3.1 - 3.3 of the improved STS. Volume 3 contains the Bases for Sections 3.4 - 3.10 of the improved STS.


NUREG-1470 V04: FINANCIAL STATEMENT FOR FISCAL YEAR 1994. * Office of the Controller (Post 890205). March 1995. 84pp. 9505180277. 83967:146. The Chief Financial Officer’s Act of 1990 requires the NRC Chief Financial Officer to prepare and submit an annual financial statement to the Director of the Office of Management and Budget (OMB). The OMB has replaced the requirement for the CFO’s Annual Report with the annual financial statement. The annual financial statement was previously included in the Chief Financial Officer’s Annual Report. This report is the fourth annual report for the NRC and includes an overview of the NRC, the audited principal financial statements and audit reports for fiscal year 1994, and supplemental financial and management information.

NUREG-1478 R01: NON-POWER REACTOR OPERATOR LICENSING EXAMINER STANDARDS. * Office of Nuclear Reactor Regulation (Post 941001). June 1995. 172pp. 9507060351. 84533:100. The Non-Power Reactor Operator Licensing Examiner Standards provide policy and guidance to NRC examiners and establish the procedures and practices for examining licensees and applicants for NRC operator licenses pursuant to Part 55 of Title 10 of the Code of Federal Regulations (10 CFR Part 55). They are intended to assist NRC examiners and facility licensees to better understand the examination process and to ensure the equitable and consistent administration of examinations to all applicants. These standards are not a substitute for the operator licensing regulations and are subject to revision or other internal operator examination licensing policy changes. As appropriate, these standards will be revised periodically to accommodate comments and reflect new information or experience.

NUREG-1482: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR POWER PLANTS. CAMPBELL P.L. Office of Nuclear Reactor Regulation, Director (Post 870411). April 1995. 328pp. 9505030476. 83746:022. In this report, the staff gives licensees guidelines for developing and implementing programs for the inservice testing of pumps and valves at commercial nuclear power plants. The report includes U.S. Nuclear Regulatory Commission guidance...
and recommendations on in-service testing issues. The staff discusses the regulations, the components to be included in an in-service testing program, and the preparation and content of cold shutdown and refueling outage justifications and requests for relief from the American Society of Mechanical Engineers Code requirements. The staff also gives specific guidance on relief acceptable to the NRC and advises licenses in the use of this information for application at their facilities. The staff discusses the revised standard technical specifications for the in-service testing program requirements and gives guidance on the process a licensee may follow upon finding an instance of non-compliance with the Code.


On May 13, 1994, the Nuclear Regulatory Commission's (NRC's) Executive Director for Operations established a Review Team to Assess the NRC Enforcement Program. The team evaluated the current system, and solicited comments from various NRC Offices, other Federal agencies, members of industry, and the public. The team's assessment report presents the team's assessment of the NRC enforcement program. The report summarizes current processes and suggests certain changes. It proposes: (1) increased clarity, focus, and simplicity in the enforcement program; (2) retention of four severity levels of violations, with a clear focus on safety; (3) holding enforcement conferences only when needed, clarifying their status as predecisional, and making open conferences the norm; (4) a streamlined civil penalty assessment process, with fewer decision points and limited outcomes, and the use of discretion where appropriate; and (5) implementation changes to increase efficiency. Recommendations are given in each area.


This report summarizes the lessons learned from the nine pilot site visits that were performed to review early implementation of the maintenance rule using the draft NRC Maintenance Inspection Procedure. Licensees followed NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." In general, the licensees were thorough in determining which structures, systems, and components (SSCs) were within the scope of the maintenance rule at each site. The use of an expert panel was an appropriate and practical method of determining which SSCs are risk significant. When setting goals, all licensees considered safety but many licensees did not consider operating experience throughout the industry. Although required to do so, licensees were not monitoring at the system or train level the performance or condition of some systems used in standby service but not significant to risk. Most licensees had not established adequate monitoring of structures under the rule. Licensees established reasonable plans for doing periodic evaluations, balancing unavailability and reliability, and assessing the effect of taking equipment out of service for maintenance. However, these plans were not evaluated because they had not been fully implemented at the time of the site visits.


As part of a comprehensive simulator upgrade program, the simulator computer systems associated with the Nuclear Regulatory Commission's (NRC) nuclear power plant simulators were replaced. Because the original instructor stations for two of the simulators were dependent on the original computer equipment, it was necessary to develop and implement new instructor stations. This report describes the Macintosh-based Instructor Station developed by NRC engineers for the General Electric (GE) and Babcock and Wilcox (B&W) simulators.


This three-volume report contains papers presented at the Twenty-Second Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, during the week of October 24-26, 1994. 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, Italy, Japan, Russia and 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.


See NUREG/CP-0140,V01 abstract.


See NUREG/CP-0140,V01 abstract.


SCALE-a modular code system for Standardized Computer Analyses Licensing Evaluation-has been developed by Oak Ridge National Laboratory at the request of the U.S. Nuclear Regulatory Commission. The SCALE system utilizes well-established computer codes and methods within standard analysis sequences that (1) allow an input format designed for the occasional user and/or novice, (2) automate the data processing and coupling between modules, and (3) provide accurate and reliable results. System development has been directed at problem-dependent cross-section processing and analysis of critical safety, shielding, heat transfer, and depletion/deady problems. Since the initial release of SCALE in 1980, the code system has been heavily used for evaluation of nuclear fuel facility and package designs. This revision documents Version 4.2 of the system.


See NUREG/CR-0200,V01,R4 abstract.


See NUREG/CR-0200,V01,R4 abstract.

NUREG/CR-2850 V13: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1991. BAKER,D.A. Battelle Memorial Institute, Pacific Northwest Laboratory. April 1995. 192pp. 9505090280. PNL-4221. ACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES commitments estimated for all sites below the Appendix I the 130 million people considered at risk. The individual dose from a high of 22 person-rem to a low of 0.002 person-rem for the sites with plants in operation and producing power during the year. The arithmetic mean was 1.2 person-rem. The total population dose for all sites was estimated at 88 person-rem for the 130 million people considered at risk. The individual dose commitments estimated for all sites below the Appendix I design objectives.


The ALARA Center at Brookhaven National Laboratory publishes a series of bibliographies of selected readings in radiation protection and ALARA in a continuing effort to collect and disseminate information on radiation dose reduction at nuclear power plants. This is volume 8 of the series. The abstracts in this bibliography were selected from proceedings of technical meetings and conference journals, research reports, and searches of the Energy Science and Technology database of the U.S. Department of Energy. The subject material of these abstracts relates to the many aspects of radiation protection and dose reduction, and ranges from use of robotics, to operational health physics, to water chemistry. Material on the design, planning, and management of nuclear power stations is included, as well as information on decommissioning and safe storage efforts. Volume 8 contains 232 abstracts, an author index, and a subject index. The author index is specific for this volume, the subject index is cumulative and lists all abstract numbers from volumes 1 to 8. The numbers in boldface indicate the abstracts in this volume; the numbers not in boldface represent abstracts in previous volumes.


The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission by Oak Ridge National Laboratory (ORNL). The program focuses 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 12 tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterizations and properties, (4) special technical assistance, (5) fracture analysis computer programs, (6) cleavage-crack initiation, (7) cladding evaluations, (8) pressurized-thermal-shock technology, (9) analysis methods validation, (10) fracture evaluation tests, (11) warm prestressing, and (12) biaxial loading effects on fracture toughness. The program tasks have been structured to emphasize 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 and with related research programs both in the United States and abroad. This report provides an overview of principal developments in each of the 12 program tasks from April - September 1993.


This is the seventh progress report of the U.S. Nuclear Regulatory Commission's research program entitled "Short Cracks in Piping and Piping Welds." The program objective is to verify and improve fracture analyses for circularly cracked large- diameter nuclear piping with crack sizes typically used in leak- before-break (LBB) analyses and in-service flaw evaluations. All work in the eight technical tasks have been completed. Ten topical reports are scheduled to be published. Progress only during the reporting period, March 1993 - December 1994, not covered in the topical reports is presented in this report. Details about the following efforts are covered in this report: Improvements to the two computer programs NRCPIPE and NRCPIFES to assess the failure behavior of circumferential through-wall and surface-cracked pipe, respectively; Pipe material property database PIFRAC; Circumferentially cracked pipe database CIRCUMCK; An assessment of the proposed ASME Section III design stress rule changes on pipe flaw tolerance; and A pipe fracture experiment on a section of pipe removed from service degraded by microbiologically induced corrosion (MIC) which contained a girth weld crack. Progress in the other tasks is not repeated here as it has been covered in great detail in the topical reports.


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'x10') 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 provides the best protection 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.


The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission, is (a) studying the degradation effects in EPICOR-II organic ion-exchange resins caused by radiation, (b) examining the adequacy of test procedures recommended in the Branch Technical Position on Waste 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. Compressive test results of 11-year-old cement and vinyl ester-styrene solidified waste forms are presented, which show effects of aging and self-irradiation. Results of the ninth 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.


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 (Kt(0)) curve shift in high-copper welds, (3) crack-arrest toughness (Kt(a)) curve shift in high-copper welds, (4) irradiation effects on crack-arrest, (5) Kt(0) and Kt(a) 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) JFDR steel examination, (13) technical assistance for JCCNRS 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 September 1993 through March 1994.


AUTOCASK (Automatic Generation of 3-D Cask Models) is a microcomputer-based system of computer programs and databases developed at the Lawrence Livermore National Laboratory for the structural analysis of shipping casks for radioactive material. Model specification is performed on the microcomputer, and the analyses are performed on the engineering workstation or mainframe computer. AUTOCASK is based on 80386/60486 compatible microcomputers. The system is composed of a series of menus, input programs, display programs, a mesh generation program, and archive programs. All data is entered through fill-in-the-blank input screens that contain descriptive data requests.


This study examines the relationship between aging, and current industry practices in areas of maintenance, surveillance, and operation of steam turbine drives for safety related pumps. These pumps are located in the Auxiliary Feedwater (AFW) system for pressurized water reactor plants, and the Reactor Core Isolation Cooling and High Pressure Coolant Injection systems for Boiling Water Reactor facilities. Findings in a recent study on the AFW (NUREG/CR-5404) indicate that the turbine drive is the single largest contributor to AFW system degradation. However, examination of the data show that the turbine itself is a reliable piece of equipment with a good service record. Most of the problems documented are the result of problems with the turbine controls and the mechanical over-sided trip mechanism, which apparently stem from three major causes.


The effect of aging on the PWR Chemical and Volume Control System (CVCS) has been evaluated. A detailed review of the NPRDS and LER databases for the 1988-1991 time period, together with a review of industry and NRC experience and research, indicate that age-related degradations and failures have occurred. These failures had significant effects on plant operation, including reactivity excursions, and pressurizer level transients. The majority of these component failures resulted in leakage of reactor coolant outside the containment. A representative plant of each PWR NSSS design (W, CE, and B&S) was visited to obtain specific information on system inspection, surveillance, monitoring, and inspection practices. The results of these visits indicate that adequate system maintenance and inspection is being performed. In some instances, the frequencies of inspection were increased in response to repeated failure events. A parametric study was performed to assess the effects of system aging on Core Damage Frequency (CDF). This study showed that as MOV operating failures increased the contribution of the High Pressure Injection to CDF also increased.


The U.S. Nuclear Regulatory Commission (NRC) periodically surveys utilities that operate nuclear plants and state regulatory commissions that regulate utility owners of nuclear power plants. The NRC is interested in identifying states that have established economic or performance incentive programs applicable to nuclear power plants, including states with new programs, how the programs are being implemented, and in determining the financial impact of the programs on the utilities. The NRC interest stems from the fact that such programs have the potential to adversely affect the safety of nuclear power plants. The information in this report was obtained from interviews conducted with each state regulatory agency that administers an incentive program and each utility that owns at least 10% of an affected nuclear power plant. The agreements, orders, and settlements, that form the basis for each incentive program were reviewed, and the interviews and supporting documentation were viewed as required. The interviews and supporting documentation were viewed as required. The interviews and supporting documentation were viewed as required. The interviews and supporting documentation were viewed as required. The interviews and supporting documentation were viewed as required. The interviews and supporting documentation were viewed as required. The interviews and supporting documentation were viewed as required.

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," was published by the U.S. Nuclear Regulatory Commission (NRC) in May 1973, and provides guidance on leak detection methods and system requirements for Light Water Reactors. Additionally, leak detection limits are specified in plant Technical Specifications and are different for Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs). These leak detection limits are also used in leak-before-break evaluations performed in accordance with Draft Standard Review Plan, Section 3.6.3, "Leak Before Break Evaluation Procedures" where a margin of 10 on the leak detection limit is used in determining the crack size considered in subsequent fracture analyses. This study was requested by the NRC to: (1) evaluate the conditional failure probability for BWR and PWR piping for pipes that were leaking at the allowable leak detection limit, and (2) evaluate the margin of 10 to determine if it was unnecessarily large. A probabilistic approach was undertaken to conduct fracture evaluations of circumferentially cracked pipes for leak-rate-detection applications. Sixteen nuclear piping systems in BWR and PWR plants were analyzed to evaluate conditional failure probability and effects of crack- morphology-variability on the current margins used in leak rate detection for leak-before-break.


Air-operated valves (AOVs) are used in a variety of safety-related applications at nuclear power plants. They are often used where rapid stroke times are required or precise control of the valve actuator is required. They can be designed to operate automatically upon loss of power, which is often desirable when selecting components for response to design basis conditions. The purpose of this report is to examine the reported failures of AOVs and determine whether there are identifiable trends in the failures related to predictable causes. This report examines the specific components that comprise a typical AOV, how those components fail, when they fail, and how such failures are discovered. It also examines whether current testing frequencies and methods are effective in predicting such failures.


This study of simulated 8-hour and 12-hour work shifts compares alertness, speed, and accuracy of subjects assigned to the two shift times. Twelve male subjects, ages 20 to 45, were randomly assigned to either an 8-hour or 12-hour shift protocol, working in teams of two in a process control simulator. In addition to the simulator monitoring task, subjects completed a 15-minute performance assessment battery and a vigor and affect scale hourly. Throughout the simulated work shift, alertness was monitored by recording electroencephalogram (EEG), electrocorticogram (EOG) and electromyogram (EMG) data. At 2-hour intervals, subjects’ sleepiness-alertness level was assessed via standardized nap tests. During off-shift hours, subjects lived in attached, self-contained apartments. Subjects were restricted to bed for an 8-hour sleep period in total darkness one hour after work shifts; EEG, EOG, and EMG recordings were continued during sleep. During days off between shifts of blocks of shifts, subjects left the facility, but kept detailed logs of daytime sleep. Exceptions measured showed better alertness, mood, and off-duty sleep on evening shifts than on night shifts.


This report uses the methodology and scenarios described in NUREG/CR-6075 and its supplement to address the direct containment heating (DCH) issue for the Surry nuclear power plant (NPP). Consistency of the initial distributions has been ensured by using insights from system-level codes, specifically SCDAHP, RELAP5 and CONTAIN. The most useful insights are that the reactor coolant system (RCS) pressure is low at vessel breach, and the RCS pressure will be reduced based on containment loads alone. However, the likelihood of high RCS pressures at vessel breach was evaluated for Surry for a limited number of sequences. The probability of RCS pressures greater than 1.38 MPa for all station blackout scenarios without power recovery or operator intervention was found to be low (0.077). This probability could have been factored into the containment failure probability for Surry if there had been intersections of the load and strength distributions.


This report summarizes information required to estimate, at least qualitatively, the potential impacts of reducing occupational dose limits below those given in 10 CFR 20 (Revised). The following overall conclusions were reached: (1) Although 10 mSv yr(-1) is a reasonable limit for many licensees, such a limit could be extraordinarily difficult to achieve and potentially destructive to the continued operation of some licensees, such as nuclear power, fuel fabrication, and medicine, (2) Twenty mSv yr(-1) as a limit is possible for some of these groups, but for others it would prove difficult, (3) Fifty mSv yr(-1) and age in 10s yr(-1) as a limit is possible for some of these groups, but for others it would prove difficult, (4) Twenty mSv yr(-1) as a limit is possible for some of these groups, but for others it would prove difficult, and (5) Fifty mSv yr(-1) and age in 10s yr(-1) as a limit is possible for some of these groups, but for others it would prove difficult. A numerical model of multiphase air-water flow and contaminant transport in the unsaturated zone is presented. The multiphase flow equations are solved using the two-pressure, mixed form of the equations with a modified Picard linearization of the equations and a finite element spatial approximation. A volatile contaminant is assumed to be transported in either phase, or in both phases simultaneously. The contaminant partitions between phases with an equilibrium distribution given by Henry's Law or via kinetic mass transfer. The transport equations are...
solved using a Galerkin finite element method with reduced integration to lump the resultant matrices. The numerical model is applied to published experimental studies to examine the behavior of the air phase and associated contaminant movement under water infiltration. The model is also used to evaluate a hypothetical design for a low-level radioactive waste disposal facility. The model has been developed in both one and two dimensions; documentation and computer codes are available for the one-dimensional flow and transport model.


The Systems Analysis Programs for the Hands-on Integrated Reliability Evaluations (SAPHIRE) suite of programs can be used to organize and standardize in an electronic format information from probabilistic risk assessments or individual plant examinations. The Models and Results Database (MAR-D) program of the SAPHIRE suite serves as the repository for probabilistic risk assessment and individual plant examination data and information. This report demonstrates by examples the common electronic and manual methods used to load these types of data. It is not a stand-alone document but references documents that contribute information relative to the data loading process. This document provides a more detailed discussion and instructions for using SAPHIRE 5.0 only when enough information on a specific topic is not provided by another available source.


MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code Package. A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework. These include: thermal-hydraulic response in the reactor coolant system, reactor core, containment, and confinement buildings; core heatup, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport; and the impact of engineered safety features on thermal-hydraulic and radionuclide behavior. Current uses of MELCOR include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications.


See NUREG/CR-6119, V01 abstract.


Remote Afterloading Brachytherapy (RAB) is a medical process used in the treatment of cancer. RAB uses a computer-controlled device to remotely insert and remove radioactive sources close to a target (or tumor) in the body. Some RAB problems affecting the radiation dose to the patient have been reported and attributed to human error. To determine the root cause of human error in the RAB system, a human factors team visited 23 RAB treatment sites in the U.S. The team observed RAB treatment planning and delivery, interviewed RAB personnel, and performed walk-throughs, during which staff demonstrated the procedures and practices used in performing RAB tasks. Factors leading to human error in the RAB system were identified. The impact of those factors on the performance of RAB was then evaluated and prioritized in terms of safety significance. Finally, the project identified and evaluated alternative approaches for resolving the safety significant problems related to human error.


A human factors project on the use of nuclear by-product material to treat cancer using remotely operated afterloaders was undertaken by the Nuclear Regulatory Commission. The purpose of the project was to identify factors that contribute to human error in the system for remote afterloading brachytherapy (RAB). This report documents the findings from the first phase of the project, which involved an extensive function and task analysis of RAB. The analysis identified the functions and tasks in RAB, made preliminary estimates of the likelihood of human error in each task, and determined the skills needed to perform each RAB task. The findings of the function and task analysis served as the foundation for the remainder of the project, which evaluated four major aspects of the RAB system linked to human error: human-system interfaces; procedures and practices; training and qualifications of RAB staff; and organizational practices and policies. At its completion, the project identified and prioritized areas for recommended NRC and industry attention based on all of the evaluations and analyses.


This document contains the accident progression analyses of internally initiated events for Surry Unit 1 as it operates in mid-loop operation during drained maintenance or a refueling outage. The report documents the methodology used during the analysis, describes the results from the application of the methodology, and compares the results with the results from a full power analysis performed on Surry as a part of the Nureg-1150 study.


See NUREG/CR-6144, V06, P1 abstract.


This topical report summarizes the work performed for the Nuclear Regulatory Commission's (NRC) research program entitled "Short Cracks in Piping and Piping Welds" that specifically focused on pipes with short through-wall cracks. Previous NRC efforts, conducted under the Degraded Piping Program, focused on understanding the fracture behavior of larger cracks in piping.
and fundamental fracture mechanics developments necessary for this technology. This report gives details on: (i) material property determinations (ii) pipe fracture experiments, and (iii) development, modification, and validation of fracture analysis methods. The material property data required to analyze the experimental results are included. These data were also implemented into the NRC's PIFAC database. Three pipe experiments with short through-wall cracks were conducted on large diameter pipe. Also, experiments were conducted on a large-diameter uncracked pipe and a pipe with a moderate-size through-wall crack. The analysis results reported here focus on simple predictive methods based on the J-Tearing theory as well as limit-load and ASME Section XI analyses. Some of these methods were improved for short-crack-length predictions. The accuracy of the various methods was determined by comparisons with experimental results from this and other programs.


This effort evaluated the fracture toughness of austenitic steel submerged-arc weld (SAW) fusion lines. The incentive was to explain why cracks grow into the fusion line in many pipe tests conducted with cracks initially centered in SAWs. The concern was that the fusion line may have a lower toughness than the SAW. It was found that the fusion line, J(Tc) was greater than the SAW toughness but much less than the base metal. Of greater importance may be that the crack growth resistance (J(D)-R) of the fusion line appears to reach a steady-state value, while the SAW had a continually increasing J(D)-R curve. This explains why the cracks eventually turn to the fusion line in the pipe experiments. A method of incorporating these results would be to use the weld metal J-R curve up to the fusion-line steady-state J value. These results may be more important to LBB analyses than the ASME flaw evaluation procedures, since there is more crack growth with through-wall cracks in LBB analyses than for surface cracks in pipe flaw evaluations.


Field LIQMET Investigation of 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 first 4 years 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 Anarbonate 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.


Thermal embrittlement of static-cast CF-8 stainless steel components from the decommissioned Shippingport reactor has been characterized. Cast stainless steel materials were obtained from four cold-leg check valves, three hot-leg main shut-off valves, and two pump volutes. The actual time-at-temperature for the materials was 139 days at 281 degrees C for the hot-leg components and 264 degrees C for the cold-leg components. Baseline mechanical properties for as-cast material were determined from tests on either recovery-annaled material or material from the cooler region of the component. The Shippingport materials show modest decreases in fracture toughness and Charpy-impact properties and a small increase in tensile strength because of relatively low service temperatures and ferrite content of the steel. The procedure and correlations developed at Argonne National Laboratory for estimating mechanical properties of cast stainless steels predict accurate or slightly lower values for Charpy-impact energy, tensile flow stress, fracture toughness J-R curve, and J(D) of the materials. The kinetics of thermal embrittlement and degree of embrittlement at saturation were established from materials that were aged further in the laboratory. The results were consistent with the estimates. The correlations successfully predicted the mechanical properties of the Ringhals 2 reactor hot- and cross-over-leg elbows (CF-8M) after service of 15 y and the KRB reactor pump cover plate (CF-8) after 9 y of service.


The results of a field investigation on the groundwater-hydrologic effect of mining-induced earthquakes are presented. The investigation was conducted at the Lucky Friday Mine Idaho, the groundwater pressure in three fracture zones was monitored over a 24-mo period. The magnitude, source location, and associated ground motions of mining-induced seismic events were also monitored. Several seismic events of magnitude 1.0 or larger were recorded, many of which caused a change in the groundwater pressure. The magnitude of groundwater-pressure change varied with the seismic-event magnitude and source distance. The data was examined using regression analysis. The statistical models obtained predicted the effects of small-magnitude seismic events more satisfactorily than those of larger ones. The observed change in groundwater pressure due to seismic events of magnitude 3.0 or more were larger than those predicted using the statistical model. Based on these results, it suggests that the effect of earthquakes on groundwater flow may be better understood through mechanistic modeling. The mechanical processes and material behavior that would need to be incorporated in such a model are examined. They include a description of the effect of stress change on the permeability and water storage capacity of a fractured rock mass; transient fluid flow; and the generation and transmission of seismic waves through the rock mass.


This report presents an overview of effectiveness of management control of safety. It reviews several modern management control theories as well as the general functions of management and relates them to safety issues at the corporate and at the process safety management (PSM) program level. Following these discussions, a formal and structured technique for assessing management of the safety function is suggested. Seven modern management control theories are summarized, including business process reengineering, the learning organization, capability maturity, total quality management, quality assurance and control, reliability centered maintenance, and industrial process
safety. Each of these theories is examined for its principal characteristics and implications for safety management. The five general management functions of planning, organizing, directing, monitoring, and integrating, which together provide control over all company operations, are discussed. Under the broad categories of Safety Culture, Leadership and Commitment, and Operating Excellence, key corporate safety elements and their subelements are examined. The three categories under which PSM program-level safety issues are described are Technology, Personnel, and Faculties.


In both SWRs and PWRs there are many locations where carbon steel pipe or components are joined to stainless steel pipe or components with a bimetallic weld. The objective of the research described in this report was to assess the accuracy of current fracture analyses for the case of a crack along a carbon steel to austenitic weld fusion line. To achieve the program objective, material property data and data from a large-diameter pipe fracture experiment were developed to assess current analytical methods. The bimetallic welds evaluated in this program were bimetallic welds obtained from a cancelled Combustion Engineering plant. The welds joined sections of the carbon steel cold-leg piping system to stainless steel safe ends that were to be welded to stainless steel pump housings. The major conclusion drawn as a result of these efforts was that the fracture behavior of the bimetallic weld evaluated in this program could be evaluated with reasonable accuracy using the strength and toughness properties of the carbon steel pipe material in conjunction with conventional elastic-plastic fracture mechanics or limit-load analyses. This may not be generally true for all bimetallic welds, as discussed in this report.


This topical report summarizes the work on angled crack growth and combined loading effects performed within the Nuclear Regulatory Commission's research program, referred to as "Short Cracks in Piping and Piping Welds". The major impetus for this work stemmed from the observation that initial circumferential cracks in carbon steel pipes exhibited angular crack growth. This failure mode was little understood, and the effect of angled crack growth from an initially circumferential crack raised questions of how pipes under combined loading with torsional stresses would behave. There were three major conclusions from this work. The first was that virtually all ferritic nuclear pipes will have toughness anisotropy. The second was that the ratio of the normalized crack driving force (as a function of angle) to the normalized toughness (also as a function of the angle of crack growth) showed that there was an equal likelihood of cracks growing at any angle between 25 and 65 degrees. This agreed with the scatter of crack growth angles observed in pipe tests. Third, for combined loads with torsional stresses, an effective moment allows pure bending analyses to be used up to crack initiation. Crack opening area under combined loads could also be determined in this manner.


Leak-before-break (LBB) analyses for circumferentially cracked pipes are currently being conducted in the nuclear industry to justify elimination of pipe whip restraints and jet impingement shields which are present because of the expected dynamic force. The methodology frequently requires calculation of leak rates. The leak rates depend on the crack-opening area of the through-wall crack in the pipe. In addition to LBB analyses which assume a hypothetical flaw size, there is also interest in the integrity of actual leaking cracks corresponding to current leakage detection requirements in NRC Regulatory Guide 1.45, or for assessing temporary repair of Class 2 and 3 pipes that have leaks as are being evaluated in ASME Section XI. This study was requested by the NRC to review, evaluate, and refine current analytical models for crack-opening-area analyses of pipes with circumferential through-wall cracks. Twenty-five pipe experiments were analyzed to determine the accuracy of the predictive models. Several practical aspects of crack-opening such as: crack-face pressure, off-center cracks, restraint of pressure-induced bending, cracks in thickness transition regions, weld residual stresses, crack-morphology models, and thermal-hydraulic analysis, were also investigated.


The Nuclear Regulatory Commission (NRC) is conducting an enhanced participatory rulemaking to establish radiological criteria for the decommissioning of NRC-licensed facilities. As part of this rulemaking, on August 20, 1994 the NRC published a proposed rule for public comment. Paragraph 20.1406(b) of the proposed rule would require that the licensee convene a Site Specific Advisory Board (SSAB) if the licensee proposed release of the site for restricted use after decommissioning. To encourage comment the NRC held a workshop on the subject of SSABs on December 6, 7, and 8, 1994. This report summarizes the 567 comments categorized from the transcript of the workshop. The commenters at the workshop generally supported public participation in decommissioning cases. Many participants favored promulgating requirements in the NRC’s rules. Some industry participants favored relying on voluntary exchange between the public and the licensees. Many participants indicated that a SSAB or something functionally equivalent is needed in controversial decommissioning cases, but that some lesser undertaking can achieve meaningful public participation in other cases. No analysis or response to the comments is included in this report.


Wetting front instability is an important phenomenon affecting fluid flow and contaminant transport in unsaturated soils and rocks. It causes the development of fingers which travel faster than would a uniform front and thus bypass much of the medium. Water saturation and solute concentration in such fingers tend to be higher than in the surrounding medium. During infiltration, fingering may cause unexpectedly rapid arrival of water and solute at the water table. This notwithstanding, most models of subsurface flow and transport ignore instability and fingering. In this report, we survey the literature to assess the extent to which this may or may not be justified. Our overview covers experiments, theoretical studies, and computer simulations of instability and fingering during immiscible two-phase flow and transport, with emphasis on infiltration into soils and fractured rocks. Our description of instability and fingering (Hele-Shaw cell) includes an extension of existing theory to fractures and interfaces having arbitrary orientations in space. Our discussion of instability in porous media includes a slight
but important correction of existing theory for the case of an inclined interface. We conclude by outlining some potential directions for future research. Among these, we single out the effect of soil and rock heterogeneities on instability and preferential flow as meriting special attention in the context of nuclear waste storage in unsaturated media.


The principle of operation of fiber optic pressure sensors and the potential of these sensors for use in nuclear power plants are described in this report. Also included is a review of current research on fiber optic sensing technologies, a comparison of fiber optic pressure sensors with conventional pressure sensors, a discussion on advantages and disadvantages of fiber optic pressure sensors, a review of failure modes of these sensors, and results of a survey of fiber optic sensor manufacturers.


The fuel had experienced a burnup of ~40 Mwd/kg U. It was the upper end. The release behaviors for the most volatile elements, Kr and Cs, were in good agreement with the ORNL-Booth Model.


The basket for a spent fuel shipping cask is subjected to compressive stresses that may cause global instability of the basket assemblies or local buckling of the individual members. Adopting the common buckling design practice in which the stability capacity of the entire structure is based on the performance of the individual members of the assemblies, the typical spent fuel basket, which is composed of plates and tubular structural members, can be idealized as an assemblage of columns, beam-columns and plates. This report presents the flexural buckling formulas for five load cases that are common in the basket buckling analysis: column under axial loads, column under axial and bending loads, plates under uniaxial loads, plates under biaxial loadings, and plate under biaxial loads and lateral pressure. The acceptance criteria from the ASME Boiler and Pressure Vessel Code are used to determine the adequacy of the basket components. Special acceptance criteria are proposed to address the unique material characteristics of austenitic stainless steel, a material which is frequently used in the basket assemblies.


The irradiation embrittlement of the reactor pressure vessel in nuclear power plants can be reduced by thermal annealing at temperatures higher than the normal operating conditions. The objective of this work was to analyze the pertinent data and develop quantitative models for estimating the recovery in 30 ft-lb (41 J) Charpy transition temperature and Charpy upper shelf energy due to annealing. An analysis data base was developed, reviewed for completeness and accuracy, and documented as part of this work. Independent variables considered in the analysis included material characteristics, annealing time and temperature, irradiation time and temperature, fluence, and flux. To identify important variables and functional forms for predicting embrittlement recovery, pattern recognition, transformation analysis, and residual analysis were applied together with current understanding of the mechanisms governing embrittlement and recovery. Models were calibrated using multivariable surface-fitting techniques. The quality of fit was evaluated by considering both the Charpy annealing data for fitting and a surrogate hardness data base. Several iterations of model calibration, evaluation with respect to mechanistic and statistical considerations, and comparison with the trends in hardness data produced improved correlation models for estimating Charpy upper shelf energy and transition temperature after irradiation and annealing.

NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF NUCLEAR MATERIAL LICENSEES OF THE UNITED STATES NUCLEAR REGULATORY COMMISSION. Commissioners of the NRC directed staff to provide the Commission with first hand information from licensees that could be used to improve the overall regulatory program. A self-administered, mail-out survey questionnaire was used to collect data from a sample of licensees who had interaction with the NRC during the previous 12 months. A total of 371 respondents of the 569 who were sent
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questionnaires returned completed surveys, for a response rate of 63%. The body of the report presents the findings of the survey including a brief introduction to the approach used, followed by survey findings regarding regulations, policies and regulatory guidance; experience with licensing applications, renewals and amendments; inspections; reporting requirements; and enforcement actions. The appendices of the report include a copy of the survey as administered to licensees, a fuller description of the survey design and data collection methods, and detailed graphic material describing survey responses.


This report documents the ARCON95 computer code developed for the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research for use in control room habitability assessments. The document includes a user’s guide to the code, a description of the technical basis for the code, and a programmer’s guide to the code. The ARCON95 code uses hourly meteorological data and recently developed methods for estimating dispersion in the vicinity of buildings to calculate relative concentrations at control room air intakes that would be exceeded no more than five percent of the time. These concentrations are calculated for averaging periods ranging from one hour to 30 days in duration. Relative concentrations calculated by ARCON95 are significantly lower than concentrations calculated using the currently accepted procedure when winds are less than two meters per second. For higher wind speeds, ARCON95 calculates about the same concentrations as the current procedure.
Secondary Report Number Index

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

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Personal Author Index

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ALLEN, M.D.
NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

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posal.

Valve
NUREG-1492: GUIDELINES FOR INSERVICE TESTING AT NUCLEAR
POWER PLANTS.

Vendor Inspection
NUREG-0040 V19 N01: LICENSEE CONTRACTOR AND VENDOR IN­
SPENSION STATUS REPORT, Quarterly Report,January-March
1995.(White Book)

Violation
NUREG-1625: ASSESSMENT OF THE NRC ENFORCEMENT PRO­
GRAM.

Water Infiltration
NUREG/CR-4518 V08: CONTROL OF WATER INFILTRATION INTO
NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Ex­
periments At A Humid Region Site,Beltaville,Maryland.

Weld
NUREG/CR-4599 V04 N1: SHORT CRACKS IN PIPING AND PIPING
NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD
FUSION LINE TOUGHNESS.
NRC Originating Organization Index (Staff Reports)

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

ADVISORY COMMITTEE(S)

ACRS - ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)

OFFICE OF INFORMATION RESOURCES MANAGEMENT & ARM

DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)

NUREG-0725 R10: PUBLIC INFORMATION CIRCULAR FOR SHIPMENTS OF IRRADIATED REACTOR FUEL.

OPERATIONS BRANCH

NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY.Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.

NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function and Task Analysis.

DIVISION OF FUEL CYCLE SAFETY & SAFEGUARDS (POST 930207)

NUREG-1280 R01: STANDARD FORMAT AND CONTENT ACCEPTANCE CRITERIA FOR THE MATERIAL CONTROL AND ACCOUNTING (MC&A) REFORM AMENDMENT.

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF THE INSPECTOR GENERAL (POST 890417)


EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)

DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)

NUREG-0693 518: A PRIORITIZATION OF GENERIC SAFETY ISSUES.

WASTE MANAGEMENT BRANCH (POST 941217)

NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Experiments At A Humid Region Site, Beltsville, Maryland.

NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.


NUREG/CR-6283: CHALLENGE ANALYSIS.COMBUSTION ENGINEERING PLANTS.Specifications.

NUREG/CR-6283: HUMAN ERROR AND CRITICAL TASKS IN REMOTE AFTERLOADING BRACHYTHERAPY. Approach for Improved System Performance.

NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function and Task Analysis.

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 840282)

OFFICE OF NUCLEAR REACTOR REGULATION (POST 941001)


NUREG-1430 V01 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Specifications.

NUREG-1430 V02 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Specifications.

NUREG-1430 V03 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 2.0 - 3.3).

NUREG-1430 V03 R01: STANDARD TECHNICAL SPECIFICATIONS BABCOCK AND WILCOX PLANTS. Bases (Sections 3.4 - 3.9).

NUREG-1431 V01 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Specifications.

NUREG-1431 V02 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Specifications.

NUREG-1431 V03 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Bases (Sections 2.0 - 3.3).

NUREG-1431 V03 R01: STANDARD TECHNICAL SPECIFICATIONS WESTINGHOUSE PLANTS. Bases (Sections 3.4 - 3.9).

NUREG-1432 V01 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Specifications.

NUREG-1432 V02 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Specifications.

NUREG-1432 V03 R01: STANDARD TECHNICAL SPECIFICATIONS COMBUSTION ENGINEERING PLANTS. Specifications.
NRC Originating Organization Index (International Agreements)

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

There were no NUREG/IA reports published during this quarter.
This index lists the NRC organizations that sponsored the contractor reports listed in this compilation. It is arranged alphabetically by major NRC organization (e.g., program office) and then by subsections of these (e.g., divisions) where appropriate. The sponsor organization is followed by the NUREG/CR number and title of the report(s) prepared by that organization. If further information is needed, refer to the main citation by the NUREG/CR number.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 670729)
NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION.Control Modules.
NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION; Miscellaneous.
NUREG/CR-6832: BUCKLING ANALYSIS OF SPENT FUEL BASKET.
NUREG/CR-6350: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.
DIVISION OF FUEL CYCLE SAFETY & SAFEGUARDS (POST 930207)
NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)
OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 941217)
NUREG/CR-6285: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.
DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)
NUREG/CR-6589: AGING OF DRIVE RINGS FOR SAFETY-RELATED PUMPS IN NUCLEAR POWER PLANTS.
NUREG/CR-6584: EFFECT ON AGING ON FWR CHEMICAL AND VOLUME CONTROL SYSTEM.
NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.
NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.
NUREG/CR-6281: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.
NUREG/CR-6267: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.
NUREG/CR-6288: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.
NUREG/CR-6327: MODELS FOR EMBRYMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.
DIVISION OF REGULATORY OPERATIONS (POST 941217)
NUREG/CR-9349 V06: OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA.
NUREG/CR-4919 V06: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS.Progress Report Of Field Experiments At A Humid Region Site, Beltsville, Maryland.
NUREG/CR-6114 V02: AUXILIARY ANALYSES IN SUPPORT OF PERFORMANCE ASSESSMENT OF A HYPOTHETICAL LOW-LEVEL WASTE FACILITY. Two-Phase Flow And Contaminant Transport In Unsaturated Soils With Application To Low-Level Radioactive Waste Disposal.
NUREG/CR-6283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.
NUREG/CR-6327: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.
DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)
NUREG/CR-6046: ALERTNESS, PERFORMANCE, AND OFF-DUTY SLEEP ON 8-HOUR AND 12-HOUR NIGHT ShiftS IN A SIMULATED CONTINUOUS OPERATIONS CONTROL ROOM SETTING.
NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.
NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.
NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function And Task Analysis.
NUREG/CR-6312: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.
NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.
DIVISION OF RISK ANALYSIS & OPERATIONS (POST 840429-000720)
NUREG/CR-6258: AN OVERVIEW OF INSTANTANEOUS FINGERING DURING IMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA.
EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800428)
OFFICE OF NUCLEAR REACTOR REGULATION (POST 941001)

LICENSE RENEWAL & ENVIRONMENTAL REVIEW PROJECT DIRECTORATE (PDLR) (POST
NUREG/CR-5975 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.
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-6307: SUMMARY OF COMMENTS RECEIVED AT WORKSHOP ON USE OF A SITE SPECIFIC ADVISORY BOARD (SSAB) TO FACILITATE PUBLIC PARTICIPATION IN DECOMMISSIONING CASES.

ANALYSIS & MEASUREMENT SERVICES CORP.
NUREG/CR-6912: ASSESSMENT OF FIBER OPTIC PRESSURE SENSORS.

ARGONNE NATIONAL LABORATORY
NUREG/CR-8109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.
NUREG/CR-6275: MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEELS FROM SHIPPINGPORT REACTOR COMPONENTS.
NUREG/CR-6315: CANDU REACTORS, THEIR REGULATION IN CANADA, AND THE IDENTIFICATION OF RELEVANT NRC SAFETY ISSUES.

ARIZONA, UNIV. OF, TUCSON, AZ
NUREG/CR-6008: AN OVERVIEW OF INSTABILITY AND FINGERING DURING IMMISCIBLE FLUID FLOW IN POROUS AND FRACTURED MEDIA.

BATTelle HUMAN AFFAIRS RESEARCH CENTERS
NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.

BATTelle MEMORIAL INSTITUTE, COLUMBUS LABORATORIES
NUREG/CR-6004: PROBABILISTIC PIPE FRACTURE EVALUATIONS FOR LEAK-RATE-DETECTION APPLICATIONS.
NUREG/CR-6251: STAINLESS STEEL SUBMERGED ARC WELD FUSION LINE TOUGHNESS.
NUREG/CR-6257: FRACTURE EVALUATIONS OF FUSION LINE CRACKS IN NUCLEAR PIPE BIMETALLIC WELDS.
NUREG/CR-6299: EFFECTS OF TOUGHNESS ANISOTROPY AND COMBINED TENSION, TORSION, AND BENDING LOADS ON FRACTURE BEHAVIOR OF FERRITIC NUCLEAR PIPE.
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

BATTelle MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORY
NUREG/CR-6075 R01: INCENTIVE REGULATION OF INVESTOR-OWNED NUCLEAR POWER PLANTS BY PUBLIC UTILITY REGULATORS.
NUREG/CR-6330: RESULTS OF REGULATORY IMPACT SURVEY OF INDUSTRIAL AND MEDICAL MATERIALS LICENSEES OF THE OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS.
NUREG/CR-6331: ATMOSPHERIC RELATIVE CONCENTRATIONS IN BUILDING WAKES.

BROOKHAVEN NATIONAL LABORATORY
NUREG/CR-0140 V01: PROCEEDINGS OF THE TWENTY-SECOND WATER REACTOR SAFETY INFORMATION MEETING,Plenary Session, Advanced Instrumentation & Control Hardware & Software, Human Factors Research, IPE & PRA.

CALIFORNIA, UNIV. OF, LOS ANGELES, CA
NUREG/CR-4918 V08: CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITS, Progress Report Of Field Experiments At A Humid Region Site,Beltsville, Maryland.

CALIFORNIA, UNIV. OF, SAN DIEGO, CA
NUREG/CR-6125 V01: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY,Human Error And Critical Tasks In Remote Afterloading Brachytherapy And Approaches For Improved System Performance.

CALIFORNIA, UNIV. OF, SANTA BARBARA, CA
NUREG/CR-6327: MODELS FOR EMBRITTLEMENT RECOVERY DUE TO ANNEALING OF REACTOR PRESSURE VESSEL STEELS.

CENTER FOR NUCLEAR WASTE REGULATORY ANALYSES
NUREG/CR-8283: FIELD SITE INVESTIGATION: EFFECT OF MINE SEISMICITY ON GROUNDWATER HYDROLOGY.

CENTRAL RESEARCH INSTITUTE OF ELECTRIC POWER INDUSTRY

GEORGE WASHINGTON UNIV., WASHINGTON, DC
NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.

IDAHO NATIONAL ENGINEERING LABORATORY
NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.
INSTITUTE FOR CIRCADIUM PHYSIOLOGY
NUREG/CR-6046: ALERTNESS, PERFORMANCE, AND OFF-DUTY SLEEP ON 8-HOUR AND 12-HOUR NIGHT SHIFTS IN A SIMULATED CONTINUOUS OPERATIONS CONTROL ROOM SETTING.

KOREA INSTITUTE OF NUCLEAR SAFETY
NUREG/CR-6000: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

LAWRENCE LIVERMORE NATIONAL LABORATORY
NUREG/CR-5657: AUTOCASK (AUTOMATIC GENERATION OF 3-D CASK MODELS), A Microcomputer Based System For Shipping Cask Design Review Analysis.
NUREG/CR-6322: BUCKLING ANALYSIS OF SPENT FUEL BASKET.

MARYLAND, UNIV. OF, COLLEGE PARK, MD

MODELING & COMPUTER SERVICES
NUREG/CR-6327: MODELS FOR EMBrittlement recovery due to annealing of reactor pressure vessel steels.

OAK RIDGE NATIONAL LABORATORY
NUREG/CR-0200 V1 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Control Modules.
NUREG/CR-0200 V3 R04: SCALE: A MODULAR CODE SYSTEM FOR PERFORMING STANDARDIZED COMPUTER ANALYSES FOR LICENSING EVALUATION. Miscellaneous.

NUREG/CR-5857: AGING OF TURBINE DRIVES FOR SAFETY-RELATED PUMPS IN NUCLEAR POWER PLANTS.
NUREG/CR-6016: AGING AND SERVICE WEAR OF AIR-OPERATED VALVES USED IN SAFETY-RELATED SYSTEMS AT NUCLEAR POWER PLANTS.

PACIFIC SCIENCE & ENGINEERING GROUP, INC.
NUREG/CR-6125 V02: HUMAN FACTORS EVALUATION OF REMOTE AFTERLOADING BRACHYTHERAPY. Function and Task Analysis.

PRINCETON UNIV., PRINCETON, NJ

SANDIA NATIONAL LABORATORIES
NUREG/CR-6109: THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN SURRY.

SWEDEN, GOVT. OF
NUREG/CR-6300: REFINEMENT AND EVALUATION OF CRACK-OPENING-AREA ANALYSES FOR CIRCUMFERENTIAL THROUGH-WALL CRACKS IN PIPES.

THE KEVIRC COMPANY, INC.
NUREG/CR-6287: MANAGEMENT CONCEPTS AND SAFETY APPLICATIONS FOR NUCLEAR FUEL FACILITIES.
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 during 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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BIBLIOGRAPHIC DATA SHEET
(See instructions on the reverse)

1. REPORT NUMBER
NUREG-0304
Vol. 20, No. 2

2. TITLE AND SUBTITLE
Regulatory and Technical Reports (Abstract Index Journal)
Compilation for Second Quarter 1995
April – June

3. DATE REPORT PUBLISHED
September 1995

4. TYPE OF REPORT
Reference

5. AUTHOR(S)

6. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; If contractor, provide name and mailing address.)
Division of Freedom of Information and Publications Services
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

7. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; If contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)
Same as 8, above.

8. SUPPLEMENTARY NOTES

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. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

compilation
abstract index

13. AVAILABILITY STATEMENT
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14. SECURITY CLASSIFICATION
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15. NUMBER OF PAGES

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