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Secondary Report Number Index  
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NRC Originating Organization Index (International Agreements)  
NRC Contract Sponsor Index (Contractor Reports)  
Contractor Index  
International Organization Index  
Licensed Facility Index

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

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

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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.

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Main Citations and Abstracts

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The Nuclear Regulatory Commission's annual summary of licensed nuclear power reactor data is based primarily on the report of operating data submitted by licensees for each unit for the month of December because that report contains data for the month of December, the year to date (in this case calendar year 1995) and cumulative data, usually from the date of commercial operation. The data is not independently verified, but various computer checks are made. The report is divided into two sections. The first contains summary highlights and the second contains data on each individual unit in commercial operation. Section 1 capacity and availability 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 periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from January through March 1996.


This journal includes all formal reports in the NUREG series prepared by the NRC staff and contractors, proceedings of conferences and workshops, grants, and 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 agreement contractors, contractor, international organization, and licensed facility.


See NUREG-0304, V20, N04 abstract.


This 7th edition of the NRC Practice and Procedure Digest contains a digest of a number of Commission, Atomic Safety and Licensing Appeal Board, and Atomic Safety and Licensing Board decisions issued during the period of July 1, 1972 to June 30, 1995, interpreting the NRC's Rules.


NRC is committed to the periodic publication of licensed fuel facility inventory difference data, following agency review of the information and completion of any related NRC investigations. Information in this report includes inventory difference data for active fuel fabrication facilities possessing more than one effective kilogram of special nuclear material.


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, N01 abstract.


See NUREG-0540, V18, N01 abstract.


See NUREG-0540, V18, N01 abstract.


The Nuclear Regulatory Commission (NRC) and the Federal Emergency Management Agency (FEMA) have added Supplement 2 NUREG-0564/FEMA-REP-1. Revision 1, to provide guidance for the development, review, and approval of radiological...
two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports by project name and by chronological order within project name. Part 2 categorizes the reports by the most appropriate generic subject area and by chronological order within subject area.


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


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


See NUREG-1437 V01 abstract.


This regulatory analysis provides the supporting information for a rule that amends the nuclear regulatory commission’s requirements for environmental review of applications for renewal of nuclear power plant operating licenses. After considering various options, the staff identified and analyzed two major alternatives. Alternative a is to amend the regulations and perform environmental reviews under the existing regulations. Alternative b is to assess, on a generic basis, the environmental impacts of renewing the operating license of individual nuclear power plants, and define the issues that will need to be further analyzed on a case-by-case basis. The findings of this assessment are codified in 10 cfr part 51. The staff has selected alternative b as the preferred alternative.

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


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


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


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


The Center for Nuclear Waste Regulatory Analyses organized and hosted a workshop on "Rock Mechanics Issues in Repository Design and Performance Assessment" on behalf of the U.S. Nuclear Regulatory Commission (NRC). This workshop was held on September 19-20, 1994 at the Holiday Inn Crowne Plaza, Rockville, Maryland. The objectives of the workshop were to stimulate an exchange of technical information among parties actively investigating rock mechanics issues relevant to the proposed high-level waste repository at Yucca Mountain and identify/confirm rock mechanics issues important to repository design and performance assessment. The workshop contained three technical sessions and two panel discussions. The participants included technical and research staffs representing the NRC and the Department of Energy and their
4 Main Citations and Abstracts

contractors, as well as researchers from the academic, commercial, and international technical communities. These proceedings include most of the technical papers presented in the technical sessions and the transcripts for the two panel discussions.


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


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


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


The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission, is (a) studying the degradation effects in organic ion-exchange resins caused by radiation, (b) examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Form to meet the requirements of 10 CFR 61 using solidified ion-exchange resins, (c) obtaining performance information on solidified ion-exchange resins in a disposal environment, and (d) determining the condition of liners used to dispose the ion-exchange resins. Compressive test results of 12-year-old cement and vinyl ester-styrene solidified waste forms are presented, which show effects of aging and self-irradiation. Results of the tenth year of data acquisition from the field testing are presented and discussed. During the continuing field testing, both Portland type I-I 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.


This report describes approaches to calculating and expressing radiation doses to the embryo/fetus from internal radionuclides. Information for occupationally and medically significant radionuclides was used to derive biokinetic transfer models and integrated with metabolic patterns. Placental transfer and radioactivity levels in the embryo/fetus were calculated as a function of stage of pregnancy and time after administration and are given as tables of deposition and retention in the embryo/fetus. Methodologies described by MIRD were extended to calculate radiation absorbed doses to the embryo/fetus using a scenario that assumed injection of a bolus of 131I into the mother's blood. Calculations were performed for administration at successive months of pregnancy to accommodate stage dependence of geometric relationships and biological behaviors of radionuclides. The gestational-stage-dependent dosimetric dose factors are based on radiation absorbed doses. Multiplication by appropriate quality factors convert these to dose equivalent, the most common quantity for stating prenatal dose limits in the United States. The dose factor tabulations are supplemented with tables of correlations and surrogate dose factors.


This report covers an investigation of the nature and cause of failure in Nitinol brachytherapy sourcewires. The investigation was initiated after two clinical incidents in which sourcewires failed during or immediately after a treatment. The investigation determined that the two clinical Nitinol sourcewires failed in a brittle manner, which is atypical for Nitinol. There were no material anomalies or subcritical flaws to explain the brittle failures. The failed tests also showed evidence of the environment, radiation, nor low-temperature structure transformation was a likely root cause of the failures. However, degradation of the PTFE was consistently evident, and those sourcewires
shipped or stored with PTFE sleeves consistently failed in laboratory bend tests. On the basis of the results of this study, it was concluded that the root cause of the in-service failures of the sourcewires was environmentally induced embrittlement due to the breakdown of the PTFE protective sleeves in the presence of the high-radiation field and subsequent reaction or interaction of the breakdown products with the Nitinol alloy.


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


The Field Lysimeter Investigations: Low-Level Waste Data Base Development Program, funded by the U.S. Nuclear Regulatory Commission NRC, is (a) studying the degradation effects in EPICOR-II organic ion-exchange resins caused by radiation, (b) examining the adequacy of test procedures recommended in the Branch Technical Position on Waste Form to meet the requirements of 10 CFR 61 using solidified EPICOR-II resins, (c) obtaining performance information on solidified EPICOR-II ion-exchange resins in a disposal environment, and (d) determining the condition of EPICOR-II liners. Results of the final 2 (10 total) years of data acquisition from operation of the field testing are presented and discussed. During the continuing field testing, both portland type I-I cement and Dow vinyl ester-styrene waste forms are being tested in lysimeter arrays located at Argonne National Laboratory-East in Illinois and at Oak Ridge National Laboratory. The experimental equipment is described and results of waste form characterization using tests recommended by the NRC's "Technical Position on Waste Form" are presented. The study is designed to provide continuous data on nuclide release and movement, as well as environmental conditions, over a 20-year period. At the end of the tenth year, the experiment was closed down. Examination of soil and waste forms is planned to be conducted next and will be reported later.
The facility has 1/4 height and 1/100 area ratio scaling. This corresponds to a volume scale of 1/400 and power scaling of 1/200. The time will run twice as fast in the model as predicted by the present scaling method. The PUMA is scaled for full pressure and is intended to operate at and below 150 psia following scram. The facility models all the major components of SBWR.


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


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


The Nuclear Regulatory Commission stipulates in 10 CFR 61 that disposed low-level radioactive waste (LLW) be stabilized. To provide guidance to disposal vendors and nuclear station age system suppliers. This document is used during an inspection to determine regulatory compliance with Title 10 of the Code of Federal Regulations, Part 71, Subpart H; Title 10 of the Code of Federal Regulations, Part 21, and quality assurance program commitments. The guidance provides a framework for transportation packaging and dry spent fuel storage system inspections. Inspectors are provided with the flexibility to adapt the methods and concepts to meet the inspection requirements for the particular facility. This guide was developed to provide a structured and consistent approach for inspections. The method separates each performance element into several areas for inspection and identifies guidelines, based on regulatory requirements, to qualitatively evaluate each area. This document was also developed to serve as a field manual to facilitate the quality assurance inspection activities.


Tables of radiation dose estimates based on the Cristy-Ecker adult male phantom are provided for a number of radiopharmaceuticals commonly used in nuclear medicine. Radiation dose estimates are listed for all major source organs, and several other organs of interest. The dose estimates were calculated using the MIRD Technique as implemented in the MIRDose3 computer code, developed by the Oak Ridge Institute for Science and Education, Radiation Internal Dose Information Center. In this code, residence times for source organs are used with decay data from the MIRD Radionuclide Data and Decay Schemes to produce estimates of radiation dose to organs of standardized phantoms representing individuals of different ages.


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


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


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


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


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


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


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


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


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


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


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


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


The degradation of fracture toughness tensile, and Charpy-impact properties of Type 304 stainless steel (SS) pipe welds due to thermal aging has been characterized at room temperature and 290 degrees C. Thermal aging of SS welds results in moderate decreases in Charpy-impact strength and fracture toughness. For the various welds in this study, upper-shelf energy decreased by 50-80 J/cm(2). The decrease in fracture toughness J-R curve or J(OC) is relatively small. Thermal aging had little or no effect on the tensile strength of the welds. Fracture properties of SS welds are controlled by the distribution and morphology of second-phase particles. Failure occurs by the formation and growth of microvoids near hard inclusions. Such processes are relatively insensitive to thermal aging. The ferrite phase has little or no effect on the fracture properties of the welds. Differences in fracture resistance of the welds arise from differences in the density and size of inclusions. The mechanical-property data from the present study are consistent with results from other investigations. The existing data have been used to establish minimum expected fracture properties for SS welds.


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


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


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


In the period March-May, 1990, a 45 station geodetic network, originally established in November-December, 1987, was reobserved using global positioning system (GPS) technology. This network, known as the Eastern U.S. Strain Network, was
established for the purpose of determining strain and deformation in the central and eastern United States. This 1990 reobservation was the first of a series of reobservations scheduled to take place over a decade in order to place meaningful constraints on the small differential movements involved.
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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<td>Traub, R.J.</td>
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<td>Wang, W.</td>
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NUREG/CR-6455: DEVELOPMENT OF TOOLS FOR SAFETY ANALYSIS OF CONTROL SOFTWARE IN ADVANCED REACTORS.

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Subject Index

This index was developed from keywords and word strings in titles and abstracts. During this development period, there will be some redundancy, which will be removed later when a reasonable thesaurus has been developed through experience. Suggestions for improvements are welcome.

10 CFR Part 51
NUREG-1437 V01: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Main Report.
NUREG-1437 V02: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Appendixes.
NUREG-1529 V01: PUBLIC COMMENTS ON THE PROPOSED 10 CFR PART 51 RULE FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES AND SUPPORTING DOCUMENTS: REVIEW OF CONCERNS AND NRC STAFF RESPONSE.Executive Summary.
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Aging
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NUREG/CR-6392: THE EFFECTS OF AGING ON COMPRESSIVE STRENGTH OF LOW-LEVEL RADIOACTIVE WASTE FORMS.
NUREG/CR-6424: REPORT ON AGING OF NUCLEAR POWER PLANT REINFORCED CONCRETE STRUCTURES.
NUREG/CR-6425: IMPACT OF STRUCTURAL AGING ON SEISMIC RISK ASSESSMENT OF REINFORCED CONCRETE STRUCTURES IN NUCLEAR POWER PLANTS.

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NUREG/CR-6383: CORROSION FATIGUE OF ALLOYS 600 AND 690 IN SIMULATED LWR ENVIRONMENTS.

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NUREG-1540: BWR STEEL CONTAINMENT CORROSION.

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NUREG-1540: BWR STEEL CONTAINMENT CORROSION.

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NUREG/CR-6392: THE EFFECTS OF AGING ON COMPRESSIVE STRENGTH OF LOW-LEVEL RADIOACTIVE WASTE FORMS.

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NUREG/CR-6210: COMPUTER CODES FOR EVALUATION OF CONTROL ROOM HABITABILITY (HABIT).

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NUREG-1541 DRFT FC: PROCESS AND DESIGN FOR CONSOLIDATING AND UPDATING MATERIALS LICENSING GUIDANCE.Draft Report For Comment.

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NUREG-1540: BWR STEEL CONTAINMENT CORROSION.

Control Room Habitability
NUREG/CR-6210: COMPUTER CODES FOR EVALUATION OF CONTROL ROOM HABITABILITY (HABIT).

Control Software
NUREG/CR-6465: DEVELOPMENT OF TOOLS FOR SAFETY ANALYSIS OF CONTROL SOFTWARE IN ADVANCED REACTORS.

Core Performance
NUREG/CR-6253: PIUS CORE PERFORMANCE ANALYSIS.

Correlation Monitor Material
NUREG/CR-6413: ANALYSIS OF THE IRRADIATION DATA FOR A302B AND A533B CORRELATION MONITOR MATERIALS.

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NUREG-1540: BWR STEEL CONTAINMENT CORROSION.
NUREG-1542: IMPACT OF STRUCTURAL AGING ON SEISMIC RISK ASSESSMENT OF REINFORCED CONCRETE STRUCTURES IN NUCLEAR POWER PLANTS.

Corrosion Fatigue
NUREG/CR-6383: CORROSION FATIGUE OF ALLOYS 600 AND 690 IN SIMULATED LWR ENVIRONMENTS.

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Crack Growth
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Eddy Current
NUREG/CR-6227: PERFORMANCE DEMONSTRATION TESTS FOR EDDY CURRENT INSPECTION OF STEAM GENERATOR TUBING.

Elbow
NUREG/CR-6445: DEVELOPMENT OF A J-ESTIMATION SCHEME FOR INTERNAL CIRCUMFERENTIAL AND AXIAL SURFACE CRACKS IN ELBOWS.

Electric Cable
NUREG/6844 V01: LITERATURE REVIEW OF ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRIC CABLES.Summary Of Past Work.

Embrittlement
NUREG/CR-6413: ANALYSIS OF THE IRRADIATION DATA FOR A302B AND A533B CORRELATION MONITOR MATERIALS.

Emergency Planning
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NUREG-1437 V01: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Main Report.NUREG-1437 V02: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Appendices.

NUREG-1529 V01: PUBLIC COMMENTS ON THE PROPOSED 10 CFR PART 51 RULE FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES AND SUPPORTING DOCUMENTS: REVIEW OF CONCERNS AND NRC STAFF RESPONSE.Executive Summary.

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NUREG/CR-6401: FAULTING IN THE YUCCA MOUNTAIN REGION.Critical Review And Analysis Of Tectonic Data From The Central Basin And Range.

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Fracture Toughness
NUREG/CR-6426: EFFECTS OF THERMAL AGING ON FRACRTURE TOUGHNESS AND CHARPY-IMPACT STRENGTH OF STAINLESS STEEL PIPE WELDS.

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NUREG/CR-6360: SUMMARY OF AIR PERMEABILITY DATA FROM SINGLE-HOLE INJECTION TESTS IN UNSATURATED FRACTURED TUFFS AT THE APACHE LEAP RESEARCH SITE: RESULTS OF STEADY-STATE TEST INTERPRETATION.

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NUREG/CR-6411: A GEOSTATISTICAL METHODOLOGY TO ASSESS THE ACCURACY OF UNSATURATED FLOW MODELS.

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NUREG/CR-6473: GLOBAL POSITIONING SYSTEM REOBSERVATIONS OVER THE EASTERN UNITED STATES STRAIN MONITORING NETWORK.

Health Physic
NUREG/CR-6345: RADIATION DOSE ESTIMATES FOR RADIO-PHARMACEUTICALS.

Heat Transfer
NUREG/CR-6253: PIUS CORE PERFORMANCE ANALYSIS.

Heavy-Section Steel Technology Program

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NUREG/CR-5068: PIPING INSPECTION ROUND ROBIN.

Human Error Analysis

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NUREG/CR-6411: A GEOSTATISTICAL METHODOLOGY TO ASSESS THE ACCURACY OF UNSATURATED FLOW MODELS.

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NUREG/CR-6443: DETERMINISTIC AND PROBABILISTIC EVALUATIONS FOR UNCERTAINTY IN PIPE FRACRTURE PARMETERS IN LEAK-BEFORE-BREAK AND IN-SERVICE FLAW EVALUATIONS.

Inservice Inspection
NUREG/CR-5066: PIPING INSPECTION ROUND ROBIN.

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NUREG/CR-6413: ANALYSIS OF THE IRRADIATION DATA FOR A302B AND A533B CORRELATION MONITOR MATERIALS.
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NUREG/CR-6383: CORROSION FATIGUE OF ALLOYS 600 AND 690 IN SIMULATED LWR ENVIRONMENTS.

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Legal Issuances
NUREG-0750 V43 101: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUANCES.January-March 1996.
NUREG-0750 V43 N03: NUCLEAR REGULATORY COMMISSION ISSUANCES FOR MARCH 1996.Pages 51-121.

License Renewal
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NUREG-1437 V02: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Main Report.
NUREG-1529 V01: PUBLIC COMMENTS ON THE PROPOSED 10 CFR PART 51 RULE FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES.Final Report.
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Licensed Fuel Facility Status Report

Licensed Operating Reactor
NUREG-0020 V25: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31, 1995.(Gray Book I)

Licensing Process
NUREG-1539: METHODOLOGY AND FINDINGS OF THE NRC'S MATERIALS LICENSING PROCESS REDESIGN.

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

Low-Level Radioactive Waste
NUREG/CR-6341: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE FROM SIMULATED LWR ENVIRONMENTS.
NUREG/CR-6383: CORROSION FATIGUE OF ALLOYS 600 AND 690 IN SIMULATED LWR ENVIRONMENTS.

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NUREG-0540 V18 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE, April 1-30, 1996.

Transportation Package
NUREG/CR-6314: QUALITY ASSURANCE INSPECTIONS FOR SHIPPING AND STORAGE CONTAINERS.

Tube Failure
NUREG/CR-6365: STEAM GENERATOR TUBE FAILURES.

Tube Inspection
NUREG/CR-6227: PERFORMANCE DEMONSTRATION TESTS FOR EDDY CURRENT INSPECTION OF STEAM GENERATOR TUBING.

Unsaturated Flow
NUREG/CR-6360: SUMMARY OF AIR PERMEABILITY DATA FROM SINGLE-HOLE INJECTION TESTS IN UNSATURATED FRACTURED TUFFS AT THE APACHE LEAP RESEARCH SITE: RESULTS OF STEADY-STATE TEST INTERPRETATION.

Unsaturated Zone
NUREG/CR-6411: A GEOSTATISTICAL METHODOLOGY TO ASSESS THE ACCURACY OF UNSATURATED FLOW MODELS.

Vendor Inspection

Weld
NUREG/CR-6428: EFFECTS OF THERMAL AGING ON FRACTURE TOUGHNESS AND CHARPY-IMPACT STRENGTH OF STAINLESS STEEL PIPE WELDS.

Yucca Mountain
NUREG/CR-6401: FAULTING IN THE YUCCA MOUNTAIN REGION, Critical Review and Analyses of Tectonic Data From The Central Basin And Range.
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)
AACS - ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OFFICE OF EXECUTIVE DIRECTOR FOR OPERATIONS (EDO)
REGION 1 (POST 820201)

EDO - OFFICE OF ADMINISTRATION (PRE 870413 & POST 890205)
DIVISION OF FREEDOM OF INFORMATION & PUBLICATIONS SERVICES (POST 940714)
NUREG-0540 V18 N02: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. February 1-29, 1996.
NUREG-0540 V18 N03: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. March 1-31, 1996.
NUREG-0540 V18 N04: TITLE LIST OF DOCUMENTS MADE PUBLICLY AVAILABLE. April 1-30, 1996.
NUREG-0750 V43 N01: INDEXES TO NUCLEAR REGULATORY COMMISSION ISSUES.January-March 1996.
NUREG-0750 V43 N02: NUCLEAR REGULATORY COMMISSION ISSUES FOR FEBRUARY 1996. Pages 13-49.
NUREG-0750 V43 N03: NUCLEAR REGULATORY COMMISSION ISSUES FOR MARCH 1996. Pages 51-121.

EDO - OFFICE OF THE CONTROLLER (PRE 820201 & POST 890205)
OFFICE OF THE CONTROLLER (POST 890205)

EDO - OFFICE OF INFORMATION RESOURCES MANAGEMENT & ARM (POST 861109)
OFFICE OF INFORMATION RESOURCES MANAGEMENT (POST 960205)
NUREG-0020 V20: LICENSED OPERATING REACTORS STATUS SUMMARY REPORT.Data As Of December 31, 1995.(Gray Book I)

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS (POST 800205)
OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)
NUREG-1530: METHODOLOGY AND FINDINGS OF THE NRC'S MATERIALS LICENSING PROCESS REDESIGN.
NUREG-1541 DRFT FC: PROCESS AND DESIGN FOR CONSOLIDATING AND UPDATING MATERIALS LICENSING GUIDANCE.Draft Report For Comment.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF THE GENERAL COUNSEL (POST 860701)

OFFICE OF THE INSPECTOR GENERAL (POST 890417)

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820205)
DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)
NUREG-1509: RADIATION EFFECTS ON REACTOR PRESSURE VESSEL SUPPORTS.
DIVISION OF REGULATORY APPLICATIONS (POST 941217)
NUREG-1437 V01: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Main Report.
NUREG-1437 V02: GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS.Appendices.
NUREG-1440: REGULATORY ANALYSIS FOR AMENDMENTS TO REGULATIONS FOR THE ENVIRONMENTAL REVIEW FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES.Final Report.
NUREG-1529 V01: PUBLIC COMMENTS ON THE PROPOSED 10 CFR PART 51 RULE FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES AND SUPPORTING DOCUMENTS: REVIEW OF CONCERNS AND NRC STAFF RESPONSE.Executive Summary.
NUREG-1529 V02: PUBLIC COMMENTS ON THE PROPOSED 10 CFR PART 51 RULE FOR RENEWAL OF NUCLEAR POWER PLANT OPERATING LICENSES AND SUPPORTING DOCUMENTS: REVIEW OF CONCERNS AND NRC STAFF RESPONSE.Appendices.
DIVISION OF SYSTEMS TECHNOLOGY (POST 941217)
NUREG/CR-0309: SCIENTIFIC DESIGN OF PURDUE UNIVERSITY MULTI-DIMENSIONAL INTEGRAL TEST ASSEMBLY (PUMA) FOR GE SBWR.

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800205)
OFFICE OF NUCLEAR REACTOR REGULATION (POST 941001)
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.
NRC Contract Sponsor Index (Contractor Reports)

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

EDO - OFFICE FOR ANALYSIS & EVALUATION OF OPERATIONAL DATA
DIVISION OF SAFETY PROGRAMS (POST 870413)
NUREG/CR-6366: STEAM GENERATOR TUBE FAILURES.

EDO - OFFICE OF NUCLEAR MATERIAL SAFETY & SAFEGUARDS
DIVISION OF SAFETY PROGRAMS (POST 870413)
NUREG/CR-6314: QUALITY ASSURANCE INSPECTIONS FOR SHIP-PING AND STORAGE CONTAINERS.

DIVISION OF INDUSTRIAL & MEDICAL NUCLEAR SAFETY (POST 870729)
NUREG/CR-6345: RADIATION DOSE ESTIMATES FOR RADIOPHARMACEUTICALS.

EDO - OFFICE OF NUCLEAR REGULATORY RESEARCH (POST 820405)
DIVISION OF ENGINEERING TECHNOLOGY (POST 941217)
NUREG/CR-5066: PIPE INSPECTION ROUND ROBIN.
NUREG/CR-6227: PERFORMANCE DEMONSTRATION TESTS FOR EDDY CURRENT INSPECTION OF STEAM GENERATOR TUBING.
NUREG/CR-6332: MICROSTRUCTURAL CHARACTERIZATION OF SELECTED AEA/UCSB MODEL FECUMN ALLOYS.
NUREG/CR-6360: SUMMARY OF AIR PERMEABILITY DATA FROM SINGLE-HOLE INJECTION TESTS IN UNSATURATED FRACTURED TUFFS AT THE APACHE LEAP RESEARCH SITE: RESULTS OF STEADY-STATE TEST INTERPRETATION.
NUREG/CR-6392: THE EFFECTS OF AGING ON COMPRESSIVE STRENGTH OF LOW-LEVEL RADIOACTIVE WASTE FORMS.

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 800428)
NUREG/CR-2850 S01: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES.Methodology And Data Base.
NUREG/CR-6465: DEVELOPMENT OF TOOLS FOR SAFETY ANALYSIS OF CONTROL SOFTWARE IN ADVANCED REACTORS.

EDO - OFFICE OF NUCLEAR REACTOR REGULATION (POST 841001)
NUREG/CR-5591 R02: CONTRIBUTION OF MATERNAL RADIONUCLIDE BURDENS TO PRENATAL RADIATION DOSES.
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/CR-6393: CORROSION FATIGUE OF ALLOYS 600 AND 690 IN SIMULATED LWR ENVIRONMENTS.
NUREG/CR-6428: EFFECTS OF THERMAL AGING ON FRACTURE TOUGHNESS AND CHARYP-IMPACT STRENGTH OF STAINLESS STEEL PIPE WELDS.

ARIZONA, UNIV. OF, TUCSON, AZ
NUREG/CR-6360: SUMMARY OF AIR PERMEABILITY DATA FROM SINGLE-HOLE INFILTRATION TESTS IN UNSATURATED FRACTURED TUFFS AT THE APACHE LEAP RESEARCH SITE: RESULTS OF STEADY-STATE TEST INTERPRETATION.

ASCIA, INC
NUREG/CR-6645: DEVELOPMENT OF TOOLS FOR SAFETY ANALYSIS OF CONTROL SOFTWARE IN ADVANCED REACTORS.

BATTELLE MEMORIAL INSTITUTE, COLUMBUS LABORATORIES
NUREG/CR-6443: DETERMINISTIC AND PROBABILISTIC EVALUATIONS FOR UNCERTAINTY IN PIPE FRACTURE PARAMETERS IN LEAK-BEFORE-BREAK AND IN-SERVICE FLAW EVALUATIONS.
NUREG/CR-6445: DEVELOPMENT OF A J-ESTIMATION SCHEME FOR INTERNAL CIRCUMFERENTIAL AND AXIAL SURFACE CRACKS IN ELBOWS.

BATTELLE MEMORIAL INSTITUTE, PACIFIC NORTHWEST LABORATORY
NUREG/CR-2850 S01: DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES.Methodology And Data Base.
NUREG/CR-5068: PIPING INSPECTION ROUND ROBIN.
NUREG/CR-5631 R02: CONTRIBUTION OF MATERNAL RADIONUCLIDE BURDENS TO PRENATAL RADIATION DOSES.
NUREG/CR-6257: COMPUTER CODES FOR EVALUATION OF CONSTRUCTION schw-INSPECTION OF STEAM GENERATOR TUBING.
NUREG/CR-6365: STEAM GENERATOR TUBE FAILURES.
NUREG/CR-6365: STEAM GENERATOR TUBE FAILURES.
NUREG/CR-6392: THE EFFECTS OF AGING ON COMPRESSIVE STRENGTH OF LOW-LEVEL RADIOACTIVE WASTE FORMS.

BATTELLE SEATTLE RESEARCH CENTER

BROOKHAVEN NATIONAL LABORATORY
NUREG/CR-6259: PIUS CORE PERFORMANCE ANALYSIS.
NUREG/CR-6341: MICROBIAL DEGRADATION OF LOW-LEVEL RADIOACTIVE WASTE FORMS. REINFORCED CONCRETE STRUCTURES.
NUREG/CR-6384 V02: LITERATURE REVIEW OF ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRIC CABLES.Literature Analysis And Appendices.

CENTER FOR NUCLEAR WASTE REGULATORY ANALYSES
NUREG/CP-0150: WORKSHOP ON ROCK MECHANICS ISSUES IN REPOSITORY DESIGN AND PERFORMANCE ASSESSMENT.Held At Holiday Inn Crowne Plaza, Rockville, Maryland, September 19-20, 1994.
NUREG/CP-6401: FAULTING IN THE YUCCA MOUNTAIN REGION.Critical Review And Analyses of Tectonic Data From The Central Basin And Range.

COMMERCIAL, DEPT. OF, NATIONAL OCEANIC & ATMOSPHERIC ADMINISTRATION
NUREG/CR-6473: GLOBAL POSITIONING SYSTEM REOBSERVATIONS OVER THE EASTERN UNITED STATES STRAIN MONITORING NETWORK.

FEDERAL EMERGENCY MANAGEMENT AGENCY
NUREG/CP-5654 R1 S2 DFC: CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF NUCLEAR POWER PLANTS.Criteria For Emergency Planning In An Early Site Permit Application.Draft Report For Comment.

HALLIBURTON NUS ENVIRONMENTAL CORP.

IDAHO NATIONAL ENGINEERING LABORATORY

JOHNS HOPKINS UNIV., BALTIMORE, MD
NUREG/CR-6424: REPORT ON AGING OF NUCLEAR POWER PLANT REINFORCED CONCRETE STRUCTURES IN NUCLEAR POWER PLANTS.

OAK RIDGE ASSOCIATED UNIVERSITIES
NUREG/CP-6345: RADIATION DOSE ESTIMATES FOR RADIOPHARMACEUTICALS.

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

SCIENCE APPLICATIONS INTERNATIONAL CORP. (FORMERLY SCIENCE APPLICATIONS),

SOUTHWEST RESEARCH INSTITUTE

WASHINGTON, UNIV. OF, SEATTLE, WA

WESTINGHOUSE HANFORD CO.
NUREG/CR-5631 R02: CONTRIBUTION OF MATERNAL RADIONUCLIDE BURDENS TO PRENATAL RADIATION DOSES.
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.

52-004  GE Simplified BWR (SBWR) Design, General Electric Co.  NUREG/CR-6309
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