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Compilation of Reports From Research Supported by the Electrical, Materials, and Mechanical Engineering Branch, Division of Engineering

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Electrical, Materials, and Mechanical Engineering Branch
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

Since 1965, the Materials Engineering Branch, Division of Engineering, of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, and its predecessors dating back to the Atomic Energy Commission (AEC), has sponsored research programs concerning the integrity of the primary system pressure boundary of light water reactors. The components of concern in these research programs have included the reactor pressure vessel (RPV), steam generators, and the piping. These research programs have covered a broad range of topics, including fracture mechanics analysis and experimental work for RPV and piping applications, inspection method development and qualification, and evaluation of irradiation effects to RPV steels.

This report provides as complete a listing as practical of formal technical reports submitted to the NRC by the investigators working on these research programs. This listing includes topical, final and progress reports, and is segmented by topic area. In many cases a report will cover several topics (such as in the case of progress reports of multi-faceted programs), but is listed under only one topic. Therefore, in searching for reports on a specific topic, other related topic areas should be checked also. The separate volumes of this report cover the following periods:

Volume 1: 1965 - 1990
Volume 2: 1991 - 1993
Volume 3: 1994 - 1998

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Introduction

Since 1965, the Materials Engineering Branch, Division of Engineering, of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, and its predecessors dating back to the Atomic Energy Commission (AEC), has sponsored research programs concerning the integrity of the primary system pressure boundary of light water reactors. The components of concern in these research programs have included the reactor pressure vessel (RPV), steam generators, and the piping. These research programs have covered a broad range of topics, including fracture mechanics analysis and experimental work for RPV and piping applications, inspection method development and qualification, and evaluation of irradiation effects to RPV steels.1

The branch sponsoring these research programs has had various names and affiliations over the years, including the following:

1965-1973	Reactor Vessels Branch, Division of Reactor Development and Technology, U.S. Atomic Energy Commission
1973-1975	Metallurgy and Materials Research Branch, Divisionl of Reactor Safety Research, U.S. Atomic Energy Commission
1975-1981	Metallurgy and Materials Research Branch, Division of Reactor Safety Research, U.S. Nuclear Regulatory Commission
1981-1986	Materials Engineering Branch, Division of Engineering Technology, U.S. Nuclear Regulatory Commission
1986-1993	Materials Engineering Branch, Division of Engineering, U.S. Nuclear Regulatory Commission
1995-1998	Electrical, Materials, and Mechanical Engineering Branch, Division of Engineering, U.S. Nuclear Regulatory Commission

This report provides as complete a listing as practical of formal technical reports submitted to the NRC by the investigators working on these research programs. This listing includes topical, final and progress reports, and is segmented by topic area. In many cases a report will cover several topics (such as in the case of progress reports of multi-faceted programs), but is listed under only one topic. Therefore, in searching for reports on a specific topic, other related topic areas should be checked also.

Compilation of Reports - 1994-1998

The separate volumes of this report cover the following periods:

- Volume 1: 1965 - 1990
- Volume 2: 1991 - 1993
- Volume 3: 1994 - 1998

Advanced Reactors

Title: Review of the proposed materials of construction for the SBWR and AP600 advanced reactors
Author(s)/Editor(s): Diercks, D.R. ; Shack, W.J. ; Chung, H.M. ; Kassner, T.F. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1994
Report Number(s): NUREG/CR-6223; ANL--94/13
Order Number: TI94013716
Abstract: Two advanced light water reactor (LWR) concepts, namely the General Electric Simplified Boiling Water Reactor (SBWR) and the Westinghouse Advanced Passive 600 Mwe Reactor (AP600), were reviewed in detail by Argonne National Laboratory. The objectives of these reviews were to (a) evaluate proposed advanced-reactor designs and the materials of construction for the safety systems, (b) identify all aging and environmentally related degradation mechanisms for the materials of construction, and (c) evaluate from the safety viewpoint the suitability of the proposed materials for the design application. Safety-related systems selected for review for these two LWRs included (a) reactor pressure vessel, (b) control rod drive system and reactor internals, (c) coolant pressure boundary, (d) engineered safety systems, (e) steam generators (AP600 only), (f) turbines, and (g) fuel storage and handling system. In addition, the use of cobalt-based alloys in these plants was reviewed. The selected materials for both reactors were generally sound, and no major selection errors were found. It was apparent that considerable thought had been given to the materials selection process, making use of lessons learned from previous LWR experience. The review resulted in the suggestion of alternate an possibly better materials choices in a number of cases, and several potential problem areas have been cited.

Annealing

Title: Marble Hill Annealing Demonstration Evaluation
Author(s)/Editor(s): C.B. Oland, B.R. Bass, J.W. Bryson, L.J. Ott, J.A. Crabtree (Oak Ridge National Laboratory)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: February 1998
Report Number(s): NUREG/CR-6552
Abstract: During thc summer of 1996, an unirradiated reactor pressurc vesscl at the abandoned Marble Hill nuclear power plant was annealed to demonstrate that existing technology can he used to thermally anneal reactor pressure vessels at commercial pressurized water reactor nuclear power plants in the United States. Instrumentation installed on the reactor pressure vessel and interfacing plant components provided evidence that the

demonstration was successful. An independent evaluation of engineering issues associated with the annealing demonstration was conducted at the Oak Ridge National Laboratory for the Nuclear Regulatory Commission. Temperature, strain, and displacement data acquired during the annealing demonstration were used to verify thermal and structural analysis results. Based on findings and observations from the annealing demonstration and results of thermal and structural analysis, an instrumentation system was developed for use in assessing thermal annealing at other nuclear power plants similar to Marble Hill. The objective of the instrumentation system is to provide sufficient data for determining if the observed time and temperature profile satisfies or exceeds the required thermal annealing conditions, and for verifying thermal and structural analysis results. Development of the instrumentation system involved consideration of technical requirements as well as issues related to minimizing occupational exposure to radiation in accordance with the "as low as is reasonably achievable" principle.

Correlations

Title: Models for embrittlement recovery due to annealing of reactor pressure vessel steels
Author(s)/Editor(s): Eason, E.D. ; Wright, J.E. ; Nelson, E.E. (Modeling and Computing Services, Boulder, CO (United States)); Odette, G.R. ; Mader, E.V. (California Univ., Santa Barbara, CA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: May 1995
Report Number(s): NUREG/CR-6327; MCS--950302
Order Number: TI95011539
Abstract: The reactor pressure vessel (RPV) surrounding the core of a commercial nuclear power plant is subject to embrittlement due to exposure to high energy neutrons. The effects of irradiation embrittlement can be reduced by thermal annealing at temperatures higher than the normal operating conditions. However, a means of quantitatively assessing the effectiveness of annealing for embrittlement recovery is needed. The objective of this work was to analyze the pertinent data on this issue 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. Data were gathered from the Test Reactor Embrittlement Data Base and from various annealing reports. 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 chemistries, annealing time and temperature, irradiation time and temperature, fluence, and flux. To identify important variables and functional forms for predicting embrittlement recovery, advanced statistical

techniques, including pattern recognition and transformation analysis, were applied together with current understanding of the mechanisms governing embrittlement and recovery. Models were calibrated using multivariable surface-fitting techniques. Several iterations of model calibration, evaluation with respect to mechanistic and statistical considerations, and comparison with the trends in hardness data produced correlation models for estimating Charpy upper shelf energy and transition temperature after irradiation and annealing. This work provides a clear demonstration that (1) microhardness recovery is generally a very good surrogate for shift recovery, and (2) there is a high level of consistency between the observed annealing trends and fundamental models of embrittlement and recovery processes.

Degradation of Mechanical Components

Title: Insights for aging management of light water reactor components: Metal containments
Author(s)/Editor(s): Shah, V.N. ; Sinha, U.P. (EG and G Idaho, Inc., Idaho Falls, ID (United States)); Smith, S.K. (Ogden Environmental and Energy Services, Southfield, MI (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1994
Report Number(s): NUREG/CR-5314-Vol.5; EGG--2562-Vol.5
Order Number: TI94009041
Abstract: This report evaluates the available technical information and field experience related to management of aging damage to light water reactor metal containments. A generic aging management approach is suggested for the effective and comprehensive aging management of metal containments to ensure their safe operation. The major concern is corrosion of the embedded portion of the containment vessel and detection of this damage. The electromagnetic acoustic transducer and half-cell potential measurement are potential techniques to detect corrosion damage in the embedded portion of the containment vessel. Other corrosion-related concerns include inspection of corrosion damage on the inaccessible side of BWR Mark I and Mark II containment vessels and corrosion of the BWR Mark I torus and emergency core cooling system piping that penetrates the torus, and transgranular stress corrosion cracking of the penetration bellows. Fatigue-related concerns include reduction in the fatigue life (a) of a vessel caused by roughness of the corroded vessel surface and (b) of bellows because of any physical damage. Maintenance of surface coatings and sealant at the metal-concrete interface is the best protection against corrosion of the vessel.

Title: The effects of age on nuclear

power plant containment cooling systems
Author(s)/Editor(s): Lofaro, R. ; Subudhi, M. ; Travis, R. ; DiBiasio, A. ; Azarm, A. (Brookhaven National Lab., Upton, NY (United States)); Davis, J. (Science Applications International Corp., New York, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1994
Report Number(s): NUREG/CR-5939; BNL-NUREG--52345
Order Number: TI94011190
Abstract: A study was performed to assess the effects of aging on the performance and availability of containment cooling systems in US commercial nuclear power plants. This study is part of the Nuclear Plant Aging Research (NPAR) program sponsored by the US Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging process and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age. The effects of age were characterized for the containment cooling system by reviewing and analyzing failure data from national databases, as well as plant-specific data. The predominant failure causes and aging mechanisms were identified, along with the components that failed most frequently. Current inspection, surveillance, and monitoring practices were also examined. A containment cooling system unavailability analysis was performed to examine the potential effects of aging by increasing failure rates for selected components. A commonly found containment spray system design and a commonly found fan cooler system design were modeled. Parametric failure rates for those components in each system that could be subject to aging were accounted for in the model to simulate the time- dependent effects of aging degradation, assuming no provisions are made to properly manage it. System unavailability as a function of increasing component failure rates was then calculated.

Title: Development and application of degradation modeling to define maintenance practices
Author(s)/Editor(s): Stock, D. ; Samanta, P. (Brookhaven National Lab., Upton, NY (United States)); Vesely, W. (Science Applications International Corp., Dublin, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1994
Report Number(s): NUREG/CR-5967; BNL-NUREG--52353
Order Number: TI94013791
Abstract: This report presents the development and application of component degradation modeling to analyze degradation effects on reliability and to identify aspects of maintenance practices that mitigate degradation and aging effects. Using continuous time Markov approaches, a component degradation model is

discussed that includes information about degradation and maintenance. The component model commonly used in probabilistic risk assessments is a simple case of this general model. The parameters used in the general model have engineering interpretations and can be estimated using data and engineering experience. The generation of equations for specific models, the solution of these equations, and a methodology for estimating the needed parameters are all discussed. Applications in this report show how these models can be used to quantitatively assess the benefits that are expected from maintaining a component, the effects of different maintenance efficiencies, the merits of different maintenance policies, and the interaction of surveillance test intervals with maintenance practices.

Title: The effects of aging on BWR core isolation cooling systems
Author(s)/Editor(s): Lee, B.S. (Brookhaven National Lab., Upton, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Oct 1994
Report Number(s): NUREG/CR-6087; BNL-NUREG--52390
Order Number: TI95002269
Abstract: A study was performed to assess the effects of aging on the Reactor Core Isolation Cooling (RCIC) system in commercial Boiling Water Reactors (BWRs). This study is part of the Nuclear Plant Aging Research (NPAR) program sponsored by the US Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging process and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age. The failure data from national databases, as well as plant specific data were reviewed and analyzed to understand the effects of aging on the RCIC system. This analysis identified important components that should receive the highest priority in terms of aging management. The aging characterization provided information on the effects of aging on component failure frequency, failure modes, and failures causes. Current inspection, surveillance, and monitoring practices were also reviewed.

Title: The effects of age on nuclear power plant containment cooling systems
Author(s)/Editor(s): Lofaro, R. ; Subudhi, M. ; Travis, R ; DiBiasio, A. ; Azarm, A. (Brookhaven National Lab., Upton, NY (United States)); Davis, J. (Science Applications International Corp., New York, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1994
Report Number(s): NUREG/CR-5939; BNL-NUREG--52345
Order Number: TI94011190
Abstract: A study was performed to assess the effects of aging on the

performance and availability of containment cooling systems in US commercial nuclear power plants. This study is part of the Nuclear Plant Aging Research (NPAR) program sponsored by the US Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging process and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age. The effects of age were characterized for the containment cooling system by reviewing and analyzing failure data from national databases, as well as plant-specific data. The predominant failure causes and aging mechanisms were identified, along with the components that failed most frequently. Current inspection, surveillance, and monitoring practices were also examined. A containment cooling system unavailability analysis was performed to examine the potential effects of aging by increasing failure rates for selected components. A commonly found containment spray system design and a commonly found fan cooler system design were modeled. Parametric failure rates for those components in each system that could be subject to aging were accounted for in the model to simulate the time- dependent effects of aging degradation, assuming no provisions are made to properly manage it. System unavailability as a function of increasing component failure rates was then calculated.

Title: Valve actuator motor degradation
Author(s)/Editor(s): Kueck, J.D. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Dec 1994
Report Number(s): NUREG/CR-6205; ORNL--6796
Order Number: TI95004904
Abstract: Valve actuator motor degradation and failure has been a significant, but little studied, problem in the nuclear industry. This study provides a discussion of the primary failure mode --thermal degradation-- and reviews the basis for the solution to thermal degradation --thermal protection. The study also provides reviews of various industry data bases, discusses effects of other failure modes such as corrosion, and provides a review of other considerations the user should entertain when assessing thermal protection.

Title: Aging and service wear of spring-loaded pressure relief valves used in safety-related systems at nuclear power plants
Author(s)/Editor(s): Staunton, R.H. ; Cox, D.F. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1995

Report Number(s): NUREG/CR-6192; ORNL--6791
Order Number: TI95009655
Abstract: Spring-loaded pressure relief valves (PRVS) are used in some safety- related applications at nuclear power plants. In general, they are used in systems where, during accidents, pressures may rise to levels where pressure safety relief is required for protection of personnel, system piping, and components. This report documents a study of PRV aging and considers the severity and causes of service wear and how it is discovered and corrected in various systems, valve sizes, etc. Provided in this report are results of the examination of the recorded failures and identification of trends and relationships/correlations in the failures when all failure-related parameters are considered. Components that comprise a typical PRV, how those components fail, when they fail, and the current testing frequencies and methods are also presented in detail.

Title: Aging study of boiling water reactor high pressure injection systems
Author(s)/Editor(s): Conley, D.A. ; Edson, J.L. ; Fineman, C.F. (Lockheed Idaho Technologies Co., Idaho Falls, ID (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1995
Report Number(s): NUREG/CR-5462; INEL--94/0090
Order Number: TI95009514
Abstract: The purpose of high pressure injection systems is to maintain an adequate coolant level in reactor pressure vessels, so that the fuel cladding temperature does not exceed 1,200[degrees]C (2,200[degrees]F), and to permit plant shutdown during a variety of design basis loss-of-coolant accidents. This report presents the results of a study on aging performed for high pressure injection systems of boiling water reactor plants in the United States. The purpose of the study was to identify and evaluate the effects of aging and the effectiveness of testing and maintenance in detecting and mitigating aging degradation. Guidelines from the United States Nuclear Regulatory Commission's Nuclear Plant Aging Research Program were used in performing the aging study. Review and analysis of the failures reported in databases such as Nuclear Power Experience, Licensee Event Reports, and the Nuclear Plant Reliability Data System, along with plant-specific maintenance records databases, are included in this report to provide the information required to identify aging stressors, failure modes, and failure causes. Several probabilistic risk assessments were reviewed to identify risk-significant components in high pressure injection systems. Testing, maintenance, specific safety issues, and codes and standards are also discussed.

Title: Effect of aging on the PWR Chemical and Volume Control System
Author(s)/Editor(s): Grove, E.J. ;

Travis, R.J. ; Aggarwal, S.K. (Brookhaven National Lab., Upton, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1995
Report Number(s): NUREG/CR-5954; BNL-NUREG--52410
Order Number: TI95014434
Abstract: The PWR Chemical and Volume Control System (CVCS) is designed to provide both safety and non-safety related functions. During normal plant operation it is used to control reactor coolant chemistry, and letdown and charging flow. In many plants, the charging pumps also provide high pressure injection, emergency boration, and RCP seal injection in emergency situations. This study examines the design, materials, maintenance, operation and actual degradation experiences of the system and main sub-components to assess the potential for age degradation. A detailed review of the Nuclear Plant Reliability Data System (NPRDS) and Licensee Event Report (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 design (W, CE, and B and W) 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 increase in response to repeated failure events. A parametric study was performed to assess the effect of system aging on Core Damage Frequency (CDF). This study showed that as motor-operated valve (MOV) operating failures increased, the contribution of the High Pressure Injection to CDF also increased.

Title: Aging of turbine drives for safety-related pumps in nuclear power plants
Author(s)/Editor(s): Cox, D.F. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1995
Report Number(s): NUREG/CR-5857; ORNL--6713
Order Number: TI95014753
Abstract: This study was performed to examine the relationship between time-dependent degradation and current industry practices in the 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 in the Reactor Core Isolation Cooling and High-Pressure

Coolant Injection systems for boiling-water reactor plants. This research has been conducted by examination of failure data in the Nuclear Plant Reliability Data System, review of Licensee Event Reports, discussion of problems with operating plant personnel, and personal observation. The reported failure data were reviewed to determine the cause of the event and the method of discovery. Based on the research results, attempts have been made to determine the predictability of failures and possible preventive measures that may be implemented. Findings in a recent study of AFW systems indicate that the turbine drive is the single largest contributor to AFW system degradation. However, examination of the data shows 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 overspeed trip mechanism; these apparently stem from three major causes which are discussed in the text. Recent improvements in maintenance practices and procedures, combined with a stabilization of the design, have led to improved performance resulting in a reliable safety-related component. However, these improvements have not been universally implemented.

Title: Robust, accurate, and non-contacting vibration measurement systems: Supplemental appendices presenting comparison measurements of the robust laser interferometer and typical accelerometer systems. Volume 2
Author(s)/Editor(s): Goodenow, T.C. ; Shipman, R.L. ; Holland, H.M. (Epoch Engineering, Inc., Gaithersburg, MD (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1995
Report Number(s): NUREG/CR-6313-Vol.2
Order Number: TI95015065
Abstract: Epoch Engineering, Incorporated (EEI) has completed a series of vibration measurements comparing their newly-developed Robust Laser Interferometer (RLI) with accelerometer-based instrumentation systems. EEI has successfully demonstrated, on several pieces of commonplace machinery, that non-contact, line-of-sight measurements are practical and yield results equal to or, in some cases, better than customary field implementations of accelerometers. The demonstration included analysis and comparison of such phenomena as nonlinearity, transverse sensitivity, harmonics, and signal-to-noise ratio. Fast Fourier Transformations were performed on the accelerometer and the laser system outputs to provide a comparison basis. The RLI was demonstrated, within the limits o the task, to be a viable, line-of-sight, non-contact alternative to accelerometer systems. Several different kinds of machinery were instrumented and compared, including a small pump, a gear-driven cement mixer, a rotor kit, and two small fans. Known machinery vibration sources were verified and RLI system output file

formats were verified to be compatible with commercial computer programs used for vibration monitoring and trend analysis. The RLI was also observed to be less subject to electromagnetic interference (EMI) and more capable at very low frequencies. This document, Volume 2, provides the appendices to this report.

Title: Robust, accurate, and non-contacting vibration measurement systems: Summary of comparison measurements of the robust laser interferometer and typical accelerometer systems. Volume 1
Author(s)/Editor(s): Goodenow, T.C. ; Shipman, R.L. ; Holland, H.M. (Epoch Engineering, Inc., Gaithersburg, MD (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1995
Report Number(s): NUREG/CR-6313-Vol.1
Order Number: TI95015064
Abstract: Epoch Engineering, Incorporated (EEI) has completed a series of vibration measurements comparing their newly-developed Robust Laser Interferometer (RLI) with accelerometer-based instrumentation systems. EEI has successfully demonstrated, on several pieces of commonplace machinery, that non-contact, line-of-sight measurements are practical and yield results equal to or, in some cases, better than customary field implementations of accelerometers. The demonstration included analysis and comparison of such phenomena as nonlinearity, transverse sensitivity, harmonics, and signal-to-noise ratio. Fast Fourier Transformations were performed on the accelerometer and the laser system outputs to provide a comparison basis. The RLI was demonstrated, within the limits of the task, to be a viable, line-of-sight, non-contact alternative to accelerometer systems. Several different kinds of machinery were instrumented and compared, including a small pump, a gear-driven cement mixer, a rotor kit, and two small fans. Known machinery vibration sources were verified and RLI system output file formats were verified to be compatible with commercial computer programs used for vibration monitoring and trend analysis. The RLI was also observed to be less subject to electromagnetic interference (EMI) and more capable at very low frequencies.

Title: Detection of pump degradation
Author(s)/Editor(s): Greene, R.H. ; Casada, D.A. ; Ayers, C.W. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Aug 1995
Report Number(s): NUREG/CR-6089
Order Number: TI95017245
Abstract: This Phase II Nuclear Plant Aging Research study examines the methods of detecting pump degradation that are currently employed in domestic and overseas nuclear facilities. This report evaluates the criteria mandated by required pump testing at U.S. nuclear power plants and compares them

to those features characteristic of state-of-the-art diagnostic programs and practices currently implemented by other major industries. Since the working condition of the pump driver is crucial to pump operability, a brief review of new applications of motor diagnostics is provided that highlights recent developments in this technology. The routine collection and analysis of spectral data is superior to all other technologies in its ability to accurately detect numerous types and causes of pump degradation. Existing ASME Code testing criteria do not require the evaluation of pump vibration spectra but instead overall vibration amplitude. The mechanical information discernible from vibration amplitude analysis is limited, and several cases of pump failure were not detected in their early stages by vibration monitoring. Since spectral analysis can provide a wealth of pertinent information concerning the mechanical condition of rotating machinery, its incorporation into ASME testing criteria could merit a relaxation in the monthly-to-quarterly testing schedules that seek to verify and assure pump operability. Pump drivers are not included in the current battery of testing. Operational problems thought to be caused by pump degradation were found to be the result of motor degradation. Recent advances in nonintrusive monitoring techniques have made motor diagnostics a viable technology for assessing motor operability. Motor current/power analysis can detect rotor bar degradation and ascertain ranges of hydraulically unstable operation for a particular pump and motor set. The concept of using motor current or power fluctuations as an indicator of pump hydraulic load stability is presented.

Title: Fire modeling of the Heiss Dampf Reaktor containment
Author(s)/Editor(s): Nicolette, V.F. (Sandia National Labs., Albuquerque, NM (United States)); Yang, K.T. (Notre Dame Univ., IN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Sep 1995
Report Number(s): NUREG/CR-6017; SAND--93-0528
Order Number: TI96001168
Abstract: This report summarizes Sandia National Laboratories' participation in the fire modeling activities for the German Heiss Dampf Reaktor (HDR) containment building, under the sponsorship of the United States Nuclear Regulatory Commission. The purpose of this report is twofold: (1) to summarize Sandia's participation in the HDR fire modeling efforts and (2) to summarize the results of the international fire modeling community involved in modeling the HDR fire tests. Additional comments, on the state of fire modeling and trends in the international fire modeling community are also included. It is noted that, although the trend internationally in fire modeling is toward the development of the more complex fire field models, each type of fire model has something to contribute

to the understanding of fires in nuclear power plants.

Title: Gate valve and motor-operator research findings
Author(s)/Editor(s): Steele, R. Jr. ; DeWall, K.G. ; Watkins, J.C. ; Russell, M.J. ; Bramwell, D.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Sep 1995
Report Number(s): NUREG/CR-6100; INEL--94/0156
Order Number: TI96000305
Abstract: This report provides an update on the valve research being sponsored by the US Nuclear Regulatory Commission (NRC) and conducted at the Idaho National Engineering Laboratory (INEL). The research addresses the need to provide assurance that motor-operated valves can perform their intended safety function, usually to open or close against specified (design basis) flow and pressure loads. This report describes several important developments: Two methods for estimating or bounding the design basis stem factor (in rising-stem valves), using data from tests less severe than design basis tests; a new correlation for evaluating the opening responses of gate valves and for predicting opening requirements; an extrapolation method that uses the results of a best effort flow test to estimate the design basis closing requirements of a gate valve that exhibits atypical responses (peak force occurs before flow isolation); and the extension of the original INEL closing correlation to include low-flow and low-pressure loads. The report also includes a general approach, presented in step-by-step format, for determining operating margins for rising-stem valves (gate valves and globe valves) as well as quarter-turn valves (ball valves and butterfly valves).

Title: A summary of the Fire Testing Program at the German HDR Test Facility
Author(s)/Editor(s): Nowlen, S.P. (Sandia National Labs., Albuquerque, NM (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Nov 1995
Report Number(s): NUREG/CR-6173; SAND--94-1795
Order Number: TI96002834
Abstract: This report provides an overview of the fire safety experiments performed under the sponsorship of the German government in the containment building of the decommissioned pilot nuclear power plant known as HDR. This structure is a highly complex, multi-compartment, multi-level building which has been used as the test bed for a wide range of nuclear power plant operation safety experiments. These experiments have included numerous fire tests. Test fire fuel sources have included gas burners, wood cribs, oil pools, nozzle release oil fires, and cable in cable trays. A wide range of ventilation conditions including full natural ventilation, full forced ventilation, and combined natural and

forced ventilation have been evaluated. During most of the tests, the fire products mixed freely with the full containment volume. Macro- scale building circulation patterns which were very sensitive to such factors as ventilation configuration were observed and characterized. Testing also included the evaluation of selective area pressurization schemes as a means of smoke control for emergency access and evacuation stairwells.

Title: Gate valve and motor-operator research findings
Author(s)/Editor(s): Steele, R. Jr. ; DeWall, K.G. ; Watkins, J.C. ; Russell, M.J. ; Bramwell, D.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Sep 1995
Report Number(s): NUREG/CR-6100; INEL--94/0156
Order Number: TI96000305
Abstract: This report provides an update on the valve research being sponsored by the US Nuclear Regulatory Commission (NRC) and conducted at the Idaho National Engineering Laboratory (INEL). The research addresses the need to provide assurance that motor-operated valves can perform their intended safety function, usually to open or close against specified (design basis) flow and pressure loads. This report describes several important developments: Two methods for estimating or bounding the design basis stem factor (in rising-stem valves), using data from tests less severe than design basis tests; a new correlation for evaluating the opening responses of gate valves and for predicting opening requirements; an extrapolation method that uses the results of a best effort flow test to estimate the design basis closing requirements of a gate valve that exhibits atypical responses (peak force occurs before flow isolation); and the extension of the original INEL closing correlation to include low-flow and low-pressure loads. The report also includes a general approach, presented in step-by-step format, for determining operating margins for rising-stem valves (gate valves and globe valves) as well as quarter-turn valves (ball valves and butterfly valves).

Title: Aging assessment of surge protective devices in nuclear power plants
Author(s)/Editor(s): Davis, J.F. ; Subudhi, M. (Brookhaven National Lab., Upton, NY (United States)); Carroll, D.P. (Florida Univ., Gainesville, FL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1996
Report Number(s): NUREG/CR-6340; BNL-NUREG--52463
Order Number: TI96006194
Abstract: An assessment was performed to determine the effects of aging on the performance and availability of surge protective devices (SPDs), used in electrical power and control systems in nuclear power plants. Although SPDs

have not been classified as safety-related, they are risk-important because they can minimize the initiating event frequencies associated with loss of offsite power and reactor trips. Conversely, their failure due to age might cause some of those initiating events, e.g., through short circuit failure modes, or by allowing deterioration of the safety-related component(s) they are protecting from overvoltages, perhaps preventing a reactor trip, from an open circuit failure mode. From the data evaluated during 1980--1994, it was found that failures of surge arresters and suppressers by short circuits were neither a significant risk nor safety concern, and there were no failures of surge suppressers preventing a reactor trip. Simulations, using the ElectroMagnetic Transients Program (EMTP) were performed to determine the adequacy of high voltage surge arresters.

Title: Estimated net value and uncertainty for automating ECCS switchover at PWRs
Author(s)/Editor(s): Walsh, B. ; Brideau, J. ; Comes, L. ; Darby, J. ; Guttmann, H. ; Sciacca, F. ; Souto, F. ; Thomas, W. ; Zigler, G. (Science and Engineering Associates, Inc., Albuquerque, NM (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1996
Report Number(s): NUREG/CR-6432; SEASF-DR--94-001
Order Number: TI96009348
Abstract: Question for resolution of Generic Safety Issue No. 24 is whether or not PWRs that currently rely on a manual system for ECCS switchover to recirculation should be required to install an automatic system. Risk estimates are obtained by reevaluating the contributions to core damage frequencies (CDFs) associated with failures of manual and semiautomatic switchover at a representative PWR. This study considers each separate instruction of the corresponding emergency operating procedures (EOPs), the mechanism for each control, and the relation of each control to its neighbors. Important contributions to CDF include human errors that result in completely coupled failure of both trains and failure to enter the required EOP. It is found that changeover to a semiautomatic system is not justified on the basis of cost-benefit analysis: going from a manual to a semiautomatic system reduces the CDF by 1.7 [times] 10[sup [minus]5] per reactor- year, but the probability that the net cost of the modification being less than $1, 000 per person-rem is about 20% without license renewal. Scoping analyses, using optimist assumptions, were performed for a changeover to a semiautomatic system with automatic actuation and to a fully automatic system; in these cases the probability of a net cost being less than $1,000/person-rem is about 50% without license renewal and over 95% with license renewal.

Title: Applications of reliability degradation analysis
Author(s)/Editor(s): Vesely, W.E. (Science Applications International Corp., Dublin, OH (United States)); Samanta, P.K. (Brookhaven National Lab., Upton, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washingt‹n, DC (United States)
Publication Date: Feb 1996
Report Number(s): NUREG/CR-6415; BNL-NUREG--52488
Order Number: TI96006221
Abstract: Reliability degradation analysis is the analysis of the occurrences of degradations and the times of maintenance to determine their reliability and risk implications. A program is presented for applying reliability degradation analyses to maintenance data collected at nuclear power plants. As a specific part of the program, time trending of maintenance data is illustrated. Maintenance data on residual heat removal (RHR) pumps and service water (SW) pumps at selected boiling water reactor (BWR) plants are evaluated to show how trends in maintenance data, which generally do not involve failures, can be used to understand effectiveness of maintenance. These trends also are translated to specific impacts on pump unavailability and on core-damage frequency (assuming that the trends in failure rate are the same as those observed for degradation rate). The second application shows the use of reliability degradation analysis to quantitatively evaluate the effect of maintenance, i.e., the quantitative change in component unavailability when no maintenance is performed. Assessment of these impacts are important since they measure the reliability and risk impacts of maintenance and can be fed back to the maintenance program to improve its effectiveness.

Title: Aging of safety class 1E transformers in safety systems of nuclear power plants
Author(s)/Editor(s): Roberts, E.W. ; Edson, J.L. ; Udy, A.C. (Lockheed Idaho Technologies Co., Idaho Falls, ID (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1996
Report Number(s): NUREG/CR-5753; INEL--95/0573
Order Number: TI96006598
Abstract: This report discusses aging effects on safety-related power transformers in nuclear power plants. It also evaluates maintenance, testing, and monitoring practices with respect to their effectiveness in detecting and mitigating the effects of aging. The study follows the US Nuclear Regulatory Commission's (NRC's) Nuclear Plant-Aging Research approach. It investigates the materials used in transformer construction, identifies stressors and aging mechanisms, presents operating and testing experience with aging effects, analyzes transformer failure events reported in various databases, and evaluates maintenance practices. Databases

maintained by the nuclear industry were analyzed to evaluate the effects of aging on the operation of nuclear power plants.

Title: Aging assessment of Westinghouse PWR and General Electric BWR containment isolation functions
Author(s)/Editor(s): Lee, B.S. ; Travis, R. ; Grove, E. ; DiBiasio, A.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1996
Report Number(s): NUREG/CR-6339; BNL-NUREG--52462
Order Number: TI96008079
Abstract: A study was performed to assess the effects of aging on the Containment Isolation (CI) functions of Westinghouse Pressurized Water Reactors and General Electric Boiling Water Reactors. This study is part of the Nuclear Plant Aging Research (NPAR) program, sponsored by the U.S. Nuclear Regulatory Commission. The objectives of this program are to provide an understanding of the aging process and how it affects plant safety so that it can be properly managed. This is one of a number of studies performed under the NPAR program which provide a technical basis for the identification and evaluation of degradation caused by age. Failure data from two national databases, Nuclear Plant Reliability Data System (NPRDS) and Licensee Event Reports (LERs), as well as plant specific data were reviewed and analyzed to understand the effects of aging on the CI functions. This study provided information on the effects of aging on component failure frequency, failure modes, and failure causes. Current inspection, surveillance, and monitoring practices were also reviewed.

Title: Effects of aging and service wear on main steam isolation valves and valve operators
Author(s)/Editor(s): Clark, R.L. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1996
Report Number(s): NUREG/CR-6246; ORNL--6814
Order Number: TI96008272
Abstract: In recent years main steam isolation valve (MSIV operating problems have resulted in significant operational transients (e.g., spurious reactor trips, steam generator dry out, excessive valve seat leakage), increased cost, and decreased plant availability. A key ingredient to an engineering-oriented reliability improvement effort is a thorough understanding of relevant historical experience. A detailed review of historical failure data available through the Institute of Nuclear Power Operation's Nuclear Plant Reliability Data System has been conducted for several types of MSIVs and valve operators for both boiling- water reactors and pressurized-water reactors. The focus of this review is on MSIV failures modes, actuator

failure modes, consequences of failure on plant operations, method of failure detection, and major stressors affecting both valves and valve operators.

Title: Aging assessment of large electric motors in nuclear power plants

Author(s)/Editor(s): Villaran, M. ; Subudhi, M. (Brookhaven National Lab., Upton, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1996
Report Number(s): NUREG/CR-6336; BNL-NUREG--52460
Order Number: TI96008243
Abstract: Large electric motors serve as the prime movers to drive high capacity pumps, fans, compressors, and generators in a variety of nuclear plant systems. This study examined the stressors that cause degradation and aging in large electric motors operating in various plant locations and environments. The operating history of these machines in nuclear plant service was studied by review and analysis of failure reports in the NPRDS and LER databases. This was supplemented by a review of motor designs, and their nuclear and balance of plant applications, in order to characterize the failure mechanisms that cause degradation, aging, and failure in large electric motors. A generic failure modes and effects analysis for large squirrel cage induction motors was performed to identify the degradation and aging mechanisms affecting various components of these large motors, the failure modes that result, and their effects upon the function of the motor. The effects of large motor failures upon the systems in which they are operating, and on the plant as a whole, were analyzed from failure reports in the databases. The effectiveness of the industry's large motor maintenance programs was assessed based upon the failure reports in the databases and reviews of plant maintenance procedures and programs.The Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research wrote this draft report at the request of NRC's Office of Nuclear Reactor Regulation. This report is to serve as a reference that the NRC staff and the nuclear industry and its suppliers can use when writing and applying sampling programs for commercial grade dedication.The RES staff reviewed the history, practices, and guidelines for commercial grade dedication in the nuclear industry to understand the particular needs for a new sampling reference. Additionally, it analyzed various material standards such as those in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section II, and those of the American Society for Testing and Materials, as well as standard industrial steel-making practices. As a result of this review and analysis, the staff identified important principles that must be applied to ensure the integrity of the dedication process for simple, metallic commercial grade items. This report

is suitable for field application.

Title: Motor-Operated Valve (MOV) Actuator Motor and Gearbox Testing
Author(s)/Editor(s): K.Dewall, J.C. Watkins, D. Bramwell (Idaho National Engineering Laboratory)
Sponsoring Organization: NRC, Washington DC (United States)
Publication Date: July 1997
Report Number: NUREG/CR-6478; INEL-96/0219
Abstract: This report documents the results of valve research sponsored by the U.S. Nuclear Regulatory Commission (NRC) and conducted at the Idaho National Engineering and Environmental Laboratory (INEEL). The research provides technical bases to the NRC in support of their effort regarding motor-operated valves (MOVs) in nuclear power plants. Specifically, the research measured the capabilities of typical valve actuators during operation at simulated design basis loads and operating conditions. Using a test stand that simulates the stem load profiles a valve actuator would experience when closing a valve against flow and pressure, we tested five typical electric motors (four ac motors and one dc motor) and three gearboxes at conditions a motor might experience in a power plant, including such off-normal conditions as operation at high temperature and reduced voltage. We also monitored the efficiency of the actuator gearbox. The testing produced the following results:

· All five motors operated at or above their rated starting torque during tests at normal voltages and temperatures.

· For all five motors (dc as well as ac), actual motor torque losses due to voltage degradation were greater than the losses calculated by methods typically used for predicting motor torque at degraded voltage conditions.

· Starting torques in motor tests compared well with stall torques in dynamometer-type tests.

· For three of the ac motors, actual motor torque l ses due to elevated operating temperatures were equal to or lower than losses calculated by the typical predictive method; for the dc motor, the actual losses were significantly greater than the predictions.

· For a`l three actuator gearboxes, the tual running efficiencies determined from testing were lower than the running efficiencies published by the manufacturer. In most instances, the actual pullout efficiencies were lower than the published pullout efficiencies.

· Operation of the gearbox at elevemperature did not affect the operating efficiency.

Title: Component unavailability versus inservice test (IST) interval: Evaluations of component aging effects with applications to check valves
Author(s)/Editor(s): Vesely, W.E. (Vesely, (W.E.), Dublin, OH (United States)); Poole, A.B. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear

Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1997
Report Number(s): NUREG/CR-6508; ORNL--6909
Order Number: TI97007394
Abstract: Methods are presented for calculating component unavailabilities when inservice test (IST) intervals are changed and when component aging is explicitly included. The methods extend usual approaches for calculating unavailability and risk effects of changing IST intervals which utilize Probabilistic Risk Assessment (PRA) methods that do not explicitly include component aging. Different IST characteristics are handled including ISTs which are followed by corrective maintenances which completely renew or partially renew the component. ISTs which are not followed by maintenance activities needed to renew the component are also handled. Any downtime associated with IST, including the test downtime and the following maintenance downtime, is included in the unavailability evaluations. A range of component aging behaviors is studied including both linear and nonlinear aging behaviors. Based upon evaluations completed to date, pooled failure data on check valves show relatively small aging (e.g., less than 7% per year). However, data from some plant systems could be evidence for larger aging rates occurring in time periods less than 5 years. The methods are utilized in this report to carry out a range of sensitivity evaluations to evaluate aging effects for different possible applications. Based on the sensitivity evaluations, summary tables are constructed showing how optimal IST interval ranges for check valves can vary relative to different aging behaviors which might exist. The evaluations are also used to identify IST intervals for check valves which are robust to component aging effects. General insights on aging effects are also extracted. These sensitivity studies and extracted results provide useful information which can be supplemented or be updated with plant specific information. The models and results can also be input to PRAs to determine associated risk implications.

Title: Results of Pressure Locking and Thermal Binding Tests of Gate Vavles
Author(s)/Editor(s): K.G.DeWall, J.C. Watkins, M.G. McKellar, D. Bramwell (Idaho National Engeering and Environmental Laboratory)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: May 1998
Report Number(s): NUREG/CR-6611; INEEL/EXT-98/00161
Abstract: The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, is funding the Idaho National Engineering and Environmental Laboratory (INEEL) in performing research investigating the performance of gate valves subjected to pressure locking and thermal binding conditions. Pressure locking and thermal binding are phenomena that make a closed gate valve difficult to open. Pressure locking can occur when operating sequences or temperature

changes cause the pressure of the fluid in the bonnet (and, in most gate valves, between the discs) to be higher than the pressure on the upstream and downstream sides of the disc assembly. Thermal binding can occur when thermal expansion/contraction effects cause the disc to be squeezed between the valve body seats. If the loads associated with pressure locking or thermal binding are very high, the actuator might not have the capacity to open the valve. We tested a flexible-wedge gate valve and a double-disc gate valve under pressure locking and thermal binding conditions. The results show that these valves are susceptible to pressure locking; however, they are not significantly affected by thermal binding. For the flexible-wedge gate valve, pressure locking loads (in terms of stem thrust) were higher than corresponding hydrostatic opening loads by a factor of 1.1 to 1.5. For the parallel disc gate valve, pressure locking loads were higher by a factor of 2.05 to 2.4. The results also show that seat leakage affects the bonnet pressurization rate when the valve is subjected to thermally induced pressure locking conditions.

Dosimetry

Title: Transport calculations of radiation exposure to vessel support structures in the Trojan Reactor
Author(s)/Editor(s): Asgari, M. ; Williams, M.L. (Louisiana State Univ., Baton Rouge, LA (United States). Nuclear Science Center); Kam, F.B.K. (Oak Ridge National Lab., TN (United States)); McGarry, E.D. (National Inst. of Standards and Technology, Gaithersburg, MD (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1994
Report Number(s): NUREG/CR-6206; ORNL/TM--12693
Order Number: TI94015398
Abstract: Comparison of transport calculations of the dosimeter activities with the experimental measurements shows that the values obtained with ENDF/B-VI cross-section data overestimate the measured results for high-energy-threshold reactions in the cavity by up to 41%, and thermal reactions by up to a factor of 3.0. The transport calculations performed with the original SAILOR cross-section library (based on ENDF/B-VI data) overestimate measured threshold reactions by only 15% and the thermal reactions by about a factor of 2.50. These results are inconsistent with those obtained in earlier studies that compared transport calculations done with SAILOR vs ENDF/B-VI, which indicate that SAILOR tends to underestimate cavity dosimeter activities for threshold reactions, while the ENDF/B-VI values usually agree better with experimental results. One factor that probably contributes to the rather large discrepancy between the computed and measured activities is the core power distribution used in the transport calculations. Because of unavailability of plant-specific data,

a generic power distribution provided by Westinghouse was used. Since the calculated cavity flux levels appear to be over-estimated, the results estimated for the exposure to the support structure should be conservative.

Title: Production and testing of the VITAMIN-B6 fine group and the BUGLE-93 broad- group neutron/photon cross-section libraries derived from ENDF/B-VI nuclear data
Author(s)/Editor(s): Ingersoll, D.T. ; White, J.E. ; Wright, R.Q. ; Hunter, H.T. ; Slater, C.O. (Nuclear Regulatory Commission, Washington, DC (United States)); Greene, N.M. ; Roussin, R.W. (Oak Ridge National Lab., TN (United States)); MacFarlane, R.E. (Los Alamos National Lab., NM (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1995
Report Number(s): NUREG/CR-6214; ORNL--6795
Order Number: TI95005715
Abstract: A new multigroup cross-section library based on ENDF/B-VI data has been produced and tested for light water reactor shielding and reactor pressure vessel dosimetry applications. The broad-group library, which is designated BUGLE-93, is intended to replace the aging BUGLE-80 and SAILOR libraries. The processing methodology is consistent with ANSI/ANS 6.1.2, since the ENDF data were first processed into a fine-group, pseudo-problem-independent format and then collapsed into the final broad-group format. The fine-group library, which is designated VITAMIN-B6, contains 120 nuclides. The BUGLE-93 47-neutron-group/20-gamma-ray- group library contains the same 120 nuclides processed as infinitely dilute and collapsed using a weighing spectrum typical of a concrete shield. Additionally, BUGLE-93 contains 105 nuclides processed with resonance self-shielding and weighted using spectra specific to BWR and PWR material compositions and reactor models. Several dosimetry response functions and kerma factors for all 120 nuclides are also included with the library. An extensive integral data testing effort was performed to qualify the new library. In general, results using the new data show significant improvements relative to earlier ENDF data.

Title: Pool critical assembly pressure vessel facility benchmark
Author(s)/Editor(s): Remec, I. ; Kam, F.B.K. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1997
Report Number(s): NUREG/CR-6454; ORNL/TM--13205
Order Number: TI97008288
Abstract: This pool critical assembly (PCA) pressure vessel wall facility benchmark (PCA benchmark) is described and analyzed in this report. Analysis

of the PCA benchmark can be used for partial fulfillment of the requirements for the qualification of the methodology for pressure vessel neutron fluence calculations, as required by the US Nuclear Regulatory Commission regulatory guide DG-1053. Section 1 of this report describes the PCA benchmark and provides all data necessary for the benchmark analysis. The measured quantities, to be compared with the calculated values, are the equivalent fission fluxes. In Section 2 the analysis of the PCA benchmark is described. Calculations with the computer code DORT, based on the discrete-ordinates method, were performed for three ENDF/B-VI-based multigroup libraries: BUGLE-93, SAILOR-95, and BUGLE-96. An excellent agreement of the calculated (C) and measures (M) equivalent fission fluxes was obtained. The arithmetic average C/M for all the dosimeters (total of 31) was 0.93 [+-] 0.03 and 0.92 [+-] 0.03 for the SAILOR-95 and BUGLE-96 libraries, respectively. The average C/M ratio, obtained with the BUGLE-93 library, for the 28 measurements was 0.93 [+-] 0.03 (the neptunium measurements in the water and air regions were overpredicted and excluded from the average). No systematic decrease in the C/M ratios with increasing distance from the core was observed for any of the libraries used.

Title: H.B. Robinson-2 Pressure Vessel Benchmark
Author(s)/Editor(s): I. Remec, F.B.K. Kam (Oak Ridge National Laboratory)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: February 1998
Report Number(s): NUREG/CR-6453; ORNL/TM-13204
Abstract: The H. B. Robinson Unit 2 Pressure Vessel Benchmark (HBR-2 benchmark) is described and analyzed in this report. Analysis of the HBR-2 benchmark can be used as partial fulfillment of the requirements for the qualification of the methodology for calculating neutron fluence in pressure vessels, as required by the U.S. Nuclear Regulatory Commission Regulatory Guide DG-1053, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*.

Section 1 of this report describes the BR-2 benchmark and provides all the dimensions, material compositions, and neutron source data necessary for the analysis. The measured quantities, to be compared with the calculated values, are the specific activities at the end of fuel cycle 9. The characteristic feature ofthe HBR-2 benchmark is that it provides measurements on both sides of the pressure vessel: in the surveillance capsule attached to the thermal shield and in the reactor cavity.

In Section 2, the analysis ofthe HBR-2 benchmark is described. Calculations with the computer code DORT, based on the discrete-ordinates method, were performed with three multigroup libraries based on ENDF/B-VI: BUGLE-93, SAILOR-95 and BUGLE-96. The average ratio of the calculated-tomeasured specific activities (C/M) for the six dosimeters in the surveillance capsule was 0.90 ± 0.04 for all three libraries. The average C/Ms for the cavity dosimeters (without neptunium dosirneter) were 0.89 ± 0.10, 0.91 ± 0.10, and 0.90 ± 0.09 for the BUGLE-93, SAILOR-95 and BUGLE-96 libraries, respectively.

It is expected that the agreement ofthe calculations with the measurements, similar to the agreement obtained in this research, should typically be observed when the discrete-ordinates method and ENDFIB-VI libraries are used for the HBR-2 benchmark analysis.

Electrical Systems

Title: The effects of solar-geomagnetically induced currents on electrical systems in nuclear power stations
Author(s)/Editor(s): Subudhi, M. (Brookhaven National Lab., Upton, NY (United States)); Carroll, D.P. (Florida Univ., Gainesville, FL (United States)); Kasturi, S. (MOS, Inc., Melville, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1994
Report Number(s): NUREG/CR-5990; BNL-NUREG--52359
Order Number: TI94005979
Abstract: This report presents the results of a study to evaluate the potential effects of geomagnetically induced currents (GICs) caused by the solar disturbances on the in-plant electrical distribution system and equipment in nuclear power stations. The plant-specific electrical distribution system for a typical nuclear plant is modeled using the ElectroMagnetic Transient Program (EMTP). The computer model simulates online equipment and loads from the station transformer in the switchyard of the power station to the safety-buses at 120 volts to which all electronic devices are connected for plant monitoring. The analytical model of the plant's electrical distribution system is studied to identify the transient effects caused by the half-cycle saturation of the station transformers due to GIC. This study provides results of the voltage harmonics levels that have been noted at various electrical buses inside the plant. The emergency circuits appear to be more susceptible to high harmonics due to the normally light load conditions. In addition to steady-state analysis, this model was further analyzed simulating various plant transient conditions (e.g., loss of load or large motor start-up) occurring during GIC events. Detail models of the plant's protective relaying system employed in bus transfer application were included in this model to study the effects of the harmonic distortion of the voltage input. Potential harmonic effects on the uniterruptable power system (UPS) are qualitatively discussed as well.

Title: Selected fault testing of electronic isolation devices used in nuclear power plant operation
Author(s)/Editor(s): Villaran, M. ; Hillman, K. ; Taylor, J. ; Lara, J. ; Wilhelm, W. (Brookhaven National Lab., Upton, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: May 1994
Report Number(s): NUREG/CR-6086; BNL-NUREG--52385
Order Number: TI94012160
Abstract: Electronic isolation devices are used in nuclear power plants to provide electrical separation between safety and non-safety circuits and systems. Major fault testing in an earlier program indicated that some energy may pass through an isolation device when a fault at the maximum credible potential is applied in the transverse mode to its output terminals. During subsequent field qualification testing of isolators, concerns were raised that the worst case fault, that is, the maximum credible fault (MCF), may not occur with a fault at the maximum credible potential, but rather at some lower potential. The present test program investigates whether problems can arise when fault levels up to the MCF potential are applied to the output terminals of an isolator. The fault energy passed through an isolated device during a fault was measured to determine whether the levels are great enough to potentially damage or degrade performance of equipment on the input (Class 1E) side of the isolator.

Title: Summary of work completed under the Environmental and Dynamic Equipment Qualification research program (EDQP)
Author(s)/Editor(s): Steele, R. Jr. ; Bramwell, D.L. ; Watkins, J.C. ; DeWall, K.G. (EG and G Idaho, Inc., Idaho Falls, ID (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1994
Report Number(s): NUREG/CR-5935; EGG--2686
Order Number: TI94007115
Abstract: This report documents the results of the main projects undertaken under the Environmental and Dynamic Equipment Qualification Research Program (EDQP) sponsored by the U.S. Nuclear Regulatory Commission (NRC) under FIN A6322. Lasting from fiscal year 1983 to 1987, the program dealt with environmental and dynamic (including seismic) equipment qualification issues for mechanical and electromechanical components and systems used in nuclear power plants. The research results have since been used by both the NRC and industry. The program included seven major research projects that addressed the following issues: (a) containment purge and vent valves performing under design basis loss of coolant accident loads, (b) containment piping penetrations and isolation valves performing under seismic loadings and design basis and severe accident containment wall displacements, (c) shaft seals for primary coolant pumps performing under station blackout conditions, (d)

electrical cabinet internals responding to in-structure generated motion (rattling), and (e) in situ piping and valves responding to seismic loadings. Another project investigating whether certain containment isolation valves will close under design basis conditions was also started under this program. This report includes eight main section, each of which provides a brief description of one of the projects, a summary of the findings, and an overview of the application of the results. A bibliography lists the journal articles, papers, and reports that document the research.

Title: Workshop on environmental qualification of electric equipment
Author(s)/Editor(s): Lofaro, R. ; Gunther, W. ; Villaran, M. ; Lee, B.S. ; Taylor, J. (comps.) (Brookhaven National Lab., Upton, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: May 1994
Report Number(s): NUREG/CP-0135; BNL-NUREG--52409; CONF-9311207--
Order Number: TI94012761
Abstract: Questions concerning the Environmental Qualification (EQ) of electrical equipment used in commercial nuclear power plants have recently become the subject of significant interest to the US Nuclear Regulatory Commission (NRC). Initial questions centered on whether compliance with the EQ requirements for older plants were adequate to support plant operation beyond 40 years. After subsequent investigation, the NRC Staff concluded that questions related to the differences in EQ requirements between older and newer plants constitute a potential generic issue which should be evaluated for backfit, independent of license renewal activities. EQ testing of electric cables was performed by Sandia National Laboratories (SNL) under contract to the NRC in support of license renewal activities. Results showed that some of the environmentally qualified cables either failed or exhibited marginal insulation resistance after a simulated plant life of 20 years during accident simulation. This indicated that the EQ process for some electric cables may be non-conservative. These results raised questions regarding the EQ process including the bases for conclusions about the qualified life of components based upon artificial aging prior to testing.

Title: Literature review of environmental qualification of safety-related electric cables: Summary of past work. Volume 1
Author(s)/Editor(s): Subudhi, M. (Brookhaven National Lab., Upton, NY (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1996
Report Number(s): NUREG/CR-6384-Vol.1; BNL-NUREG--52480-Vol.1
Order Number: TI96009367
Abstract: 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 US 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.

Title: Literature review of environmental qualification of safety-related electric cables: Literature analysis and appendices. Volume 2
Author(s)/Editor(s): Lofaro, R. ; Bowerman, B. ; Carbonaro, J. (Brookhaven National Lab., Upton, NY (United States)) (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1996
Report Number(s): NUREG/CR-6384-Vol.2; BNL-NUREG--52480-Vol.2
Order Number: TI96009368
Abstract: In support of the US 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 1 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.

Title: Long-term aging and loss-of-coolant accident (LOCA) testing of electrical cables
Author(s)/Editor(s): Nelson, C.F. ;

Gauthier, G. ; Carlin, F. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Oct 1996
Report Number(s): NUREG/CR-6202; IPSN--94-03; SAND--94-0485
Order Number: TI97000454
Abstract: Experiments were performed to assess the aging degradation and loss-of- coolant accident (LOCA) behavior of electrical cables subjected to long-term aging exposures. Four different cable types were tested in both the U.S. and France: (1) U.S. 2 conductor with ethylene propylene rubber (EPR) insulation and a Hypalon jacket. (2) U.S. 3 conductor with cross-linked polyethylene (XLPE) insulation and a Hypalon jacket. (3) French 3 conductor with EPR insulation and a Hypalon jacket. (4) French coaxial with polyethylene (PE) insulation and a PE jacket. The data represent up to 5 years of simultaneous aging where the cables were exposed to identical aging radiation doses at either 40[degrees]C or 70[degrees]C; however, the dose rate used for the aging irradiation was varied over a wide range (2-100 Gy/hr). Aging was followed by exposure to simulated French LOCA conditions. Several mechanical, electrical, and physical-chemical condition monitoring techniques were used to investigate the degradation behavior of the cables. All the cables, except for the French PE cable, performed acceptably during the aging and LOCA simulations. In general, cable degradation at a given dose was highest for the lowest dose rate, and the amount of degradation decreased as the dose rate was increased.

EAC and Fatigue

Title: Environmentally assisted cracking in Light Water Reactors: Semiannual report, April 1993--September 1993
Author(s)/Editor(s): Chopra, O.K. ; Chung, H.M. ; Karlsen, T. ; Kassner, T.F. ; Michaud, W.F. ; Ruther, W.E. ; Sanecki, J.E. ; Shack, W.J. ; Soppet, W.K. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1994
Report Number(s): NUREG/CR-4667-Vol.17; ANL--94/16-Vol.17

Order Number: TI94014862
Abstract: This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRS) during the six months from April 1993 to September 1993. EAC and fatigue of piping, pressure vessels, and core components in LWRs are important concerns as extended reactor lifetimes are envisaged. Topics that have been investigated include (a) fatigue of low-alloy steel used in piping, steam generators, and reactor pressure vessels; (b) EAC of cast stainless steels (SSs); and (c) radiation-induced

segregation and irradiation-assisted stress corrosion cracking of Type 304 SS after accumulation of relatively high fluence. Fatigue tests were conducted on medium-sulfur-content A106-Gr B piping and A533-Gr B pressure vessel steels in simulated PWR water and in air. Additional crack growth data were obtained on fracture-mechanics specimens of cast austenitic SSs in the as- received and thermally aged conditions in simulated boiling-water reactor (BWR) water at 289[degree]C. The data were compared with predictions based on crack growth correlations for wrought austenitic SS in oxygenated water developed at ANL and rates in air from Section 11 of the ASME Code. Microchemical and microstructural changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy.

Title: Environmentally assisted cracking in light water reactors. Semiannual report, October 1993--March 1994. Volume 18
Author(s)/Editor(s): Chung, H.M. ; Chopra, O.K. ; Erck, R.A. ; Kassner, T.F. ; Michaud, W.F. ; Ruther, W.E. ; Sanecki, J.E. ; Shack, W.J. ; Soppet, W.K. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1995
Report Number(s): NUREG/CR-4667-Vol.18; ANL--95/2-Vol.18
Order Number: TI95009017
Abstract: This report summarizes work performed by Argonne National Laboratory (ANL) on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRs) during the six months from October 1993 to March 1994. EAC and fatigue of piping, pressure vessels, and core components in LWRs are important concerns in operating plants and as extended reactor lifetimes are envisaged. Topics that have been investigated include (a) fatigue of low-alloy steel used in piping, steam generators, and reactor pressure vessels, (b) EAC of wrought and cast austenitic stainless steels (SSs), and (c) radiation-induced segregation and irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS after accumulation of relatively high fluence. Fatigue tests have been conducted on A302-Gr B low-alloy steel to verify whether the current predictions of modest decreases of fatigue life in simulated pressurized water reactor water are valid for high-sulfur heats that show environmentally enhanced fatigue crack growth rates. Additional crack growth data were obtained on fracture-mechanics specimens of austenitic SSs to investigate threshold stress intensity factors for EAC in high-purity oxygenated water at 289[degrees]C. The data were compared with predictions based on crack growth correlations for wrought austenitic SS in oxygenated water developed at ANL

and rates in air from Section XI of the ASME Code. Microchemical and microstructural changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating boiling water reactors were studied by Auger electron spectroscopy and scanning electron microscopy to determine whether trace impurity elements, which are not specified in the ASTM specifications, may contribute to IASCC of solution-annealed materials.

Title: Environmentally Assisted Cracking in Light Water Reactors Semiannual Report April 1 994-September 1994
Author(s)/Editor(s): O. K. Chopra, H. M. Chung, D. J. Gavenda, E. E. Gruber, A. G. Hins, T. H. Hughes, T. F. Kassner, W. E. Ruther, W. J. Shack, and W. K. Soppet (Argonne National Laboratory)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: September 1995
Report Number(s): NUREG/CR-4667/ANL-95/2 Vol 19
Abstract: This report summarizes work performed by Argonne National Laboratory (ANL) on fatigue and environmentally assisted cracking (EAC) in light water reactors from April to September 1994. Topics that have been investigated include (a) fatigue of carbon and low-alloy steel used in piping and reactor pressure vessels, (b) EAC of austenitic stainless steels (SSs) and Alloy 600, and (c) irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS. Fatigue tests have been conducted on A106-Gr B and A533-Gr B steels in oxygenated water to determine whether a slow strain rate applied during different portions of a tensile-loading cycle are equally effective in decreasing fatigue life. Crack growth data were obtained on fracture-mechanics specimens of SSs and Alloy 600 to investigate EAC in simulated boiling water reactor (BOOR) and pressurized water reactor environments at 289°C. The data were compared with predictions from crack growth correlations developed at ANL for SSs in water and from rates in air from Section Xl of the ASME Code. Microchemical changes in high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath from operating BWRs were studied by Auger electron spectroscopy and scanning electron microscopy to determine whether trace impurity elements may contribute to IASCC of these materials.

Title: Environmentally assisted cracking in Light Water Reactors: Semiannual report, October 1994--March 1995. Volume 20
Author(s)/Editor(s): Chung, H.M. ; Chopra, O.K. ; Gavenda, D.J. ; Hins, A.G. ; Kassner, T.F. ; Ruther, W.E. ; Shack, W.J. ; Soppet, W.K. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)

Publication Date: Jan 1996
Report Number(s):
NUREG/CR-4667-Vol.20; ANL--95/41-Vol.20

Order Number: TI96005922
Abstract: This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRS) from October 1994 to March 1995. Topics that have been investigated include (a) fatigue of carbon and low-alloy steel used in reactor piping and pressure vessels, (b) EAC of Alloy 600 and 690, and (c) irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS. Fatigue tests were conducted on ferritic steels in water with several dissolvedoxygen (DO) concentrations to determine whether a slow strain rate applied during different portions of a tensile-loading cycle are equally effective in decreasing fatigue life. Tensile properties and microstructures of several heats of Alloy 600 and 690 were characterized for correlation with EAC of the alloys in simulated LWR environments. Effects of DO and electrochemical potential on susceptibility to intergranular cracking of high- and commercial-purity Type 304 SS specimens from control-blade absorber tubes and a control-blade sheath irradiated in boiling water reactors were determined in slow-strain-rate-tensile tests at 289[degrees]C. Microchemical changes in the specimens were studied by Auger electron spectroscopy and scanning electron microscopy to determine whether trace impurity elements may contribute to IASCC of these materials.

Title: Corrosion fatigue of alloys 600 and 690 in simulated LWR environments
Author(s)/Editor(s): Ruther, W.E. ; Soppett, W.K. ; Kassner, T.F. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1996
Report Number(s): NUREG/CR-6383; ANL--95/37
Order Number: TI96008966
Abstract: Crack growth data were obtained on fracture-mechanics specimens of Alloys 600 and 690 to investigate environmentally assisted cracking (EAC) in simulated boiling water reactor and pressurized water reactor environments at 289 and 320 C. Preliminary information was obtained on the effect of temperature, load ratio, stress intensity (K), and the dissolved-oxygen and -hydrogen concentrations of the water on EAC. Specimens of Type 316NG and sensitized Type 304 stainless steel (SS) were included in several of the experiments to assess the behavior of these materials and Alloy 600 under the same water chemistry and loading conditions. The experimental data are compared with predictions from an Argonne National Laboratory (ANL) model for crack growth rates (CGRs) of SSs in water and the ASME Code Section 11 correlation for CGRs in air at the K[sub 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 11 air line.''

Title: Environmentally assisted cracking in light water reactors
Author(s)/Editor(s): Chopra, O.K. ; Chung, H.M. ; Gruber, E.E. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1996
Report Number(s):
NUREG/CR-4667-Vol.21; ANL--96/1-Vol.21
Order Number: TI96013829
Abstract: This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors (LWRs) from April 1995 to December 1995. Topics that have been investigated include fatigue of carbon and low-alloy steel used in reactor piping and pressure vessels, EAC of Alloy 600 and 690, and irradiation-assisted stress corrosion cracking (IASCC) of Type 304 SS. Fatigue tests were conducted on ferritic steels in water that contained various concentrations of dissolved oxygen (DO) to determine whether a slow strain rate applied during different portions of a tensile-loading cycle are equally effective in decreasing fatigue life. Crack- growth-rate tests were conducted on compact-tension specimens from several heats of Alloys 600 and 690 in simulated LWR environments. Effects of fluoride-ion contamination on susceptibility to intergranular cracking of high- and commercial-purity Type 304 SS specimens from control-tensile tests at 288 degrees Centigrade. Microchemical changes in the specimens were studied by Auger electron spectroscopy and scanning electron microscopy to determine whether trace impurity elements may contribute to IASCC of these materials.

Title: Environmentally assisted cracking in light water reactors. Semiannual progress report, January 1996--June 1996
Author(s)/Editor(s): Chopra, O.K. ; Chung, H.M. ; Gruber, E.E. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: May 1997
Report Number(s):
NUREG/CR-4667-Vol.22; ANL--97/9-Vol.22
Order Number: TI97006378
Abstract: This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors from January 1996 to June 1996. Topics that have been investigated include (a) fatigue of carbon, low-alloy, and austenitic stainless steels (SSs) used in reactor piping and pressure vessels, (b) irradiation-assisted stress corrosion cracking of Type 304 SS, and (c) EAC of Alloys 600 and 690. Fatigue tests were conducted on ferritic and austenitic SSs in water that contained various concentrations of dissolved oxygen (DO) to determine whether a slow strain rate applied during various portions of a tensile-loading cycle are equally effective in decreasing fatigue life. Slow-strain-rate-tensile tests were

conducted in simulated boiling water reactor (BWR) water at 288[degrees]C on SS specimens irradiated to a low fluence in the Halden reactor and the results were compared with similar data from a control-blade sheath and neutron-absorber tubes irradiated in BWRs to the same fluence level. Crack-growth-rate tests were conducted on compact-tension specimens from several heats of Alloys 600 and 690 in air and high-purity, low-DO water. 83 refs., 60 figs., 14 tabs.

Title: Effects of fluoride and other halogen ions on the external stress corrosion cracking of Type 304 austenitic stainless steel
Author(s)/Editor(s): Whorlow, K.M. ; Hutto, F.B. Jr. (Tutco Scientific Corp., Grand Junction, CO (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1997
Report Number(s): NUREG/CR-6539
Order Number: TI97007395
Abstract: The drip procedure from the Standard Test Method for Evaluating the Influence of Thermal Insulation on External Stress Corrosion Cracking Tendency of Austenitic Stainless Steel (ASTM C 692-95a) was used to research the effect of halogens and inhibitors on the External Stress Corrosion Cracking (ESCC) of Type 304 stainless steel as it applies to Nuclear Regulatory Commission Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel. The solutions used in this research were prepared using pure chemical reagents to simulate the halogens and inhibitors found in insulation extraction solutions. The results indicated that sodium silicate compounds that were higher in sodium were more effective for preventing chloride-induced ESCC in Type 304 austenitic stainless steel. Potassium silicate (all-silicate inhibitor) was not as effective as sodium silicate. Limited testing with sodium hydroxide (all-sodium inhibitor) indicated that it may be effective as an inhibitor. Fluoride, bromide, and iodide caused minimal ESCC which could be effectively inhibited by sodium silicate. The addition of fluoride to the chloride/sodium silicate systems at the threshold of ESCC appeared to have no synergistic effect on ESCC. The mass ratio of sodium + silicate (mg/kg) to chloride (mg/kg) at the lower end of the NRC RG 1.36 Acceptability Curve was not sufficient to prevent ESCC using the methods of this research.

Title: Environmentally assisted cracking in light water reactors. Semiannual report July 1996--December 1996
Author(s)/Editor(s): Chopra, O.K. ; Chung, H.M. ; Gavenda, D.J. (Argonne National Lab., IL (United States)) (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Oct 1997
Report Number(s): NUREG/CR-4667-Vol.23; ANL--97/10

Order Number: TI98000789
Abstract: This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors from July 1996 to December 1996. Topics that have been investigated include (a) fatigue of carbon, low-alloy, and austenitic stainless steels (SSs) used in reactor piping and pressure vessels, (b) irradiation-assisted stress corrosion cracking of Type 304 SS, (c) EAC of Alloy 600, and (d) characterization of residual stresses in welds of boiling water reactor (BWR) core shrouds by numerical models. Fatigue tests were conducted on ferritic and austenitic SSs in water that contained various concentrations of dissolved oxygen to determine whether a slow strain rate applied during various portions of a tensile-loading cycle are equally effective in decreasing fatigue life. Slow-strain-rate-tensile tests were conducted in simulated BWR water at 288 C on SS specimens irradiated to a low fluence in the Halden reactor and the results were compared with similar data from a control-blade sheath and neutron-absorber tubes irradiated in BWRs to the same fluence level. Crack-growth-rate tests were conducted on compact- tension specimens from a low-carbon content heat of Alloy 600 in high-purity oxygenated water at 289 C. Residual stresses and stress intensity factors were calculated for BWR core shroud welds.

Title: Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels.
Author(s)/Editor(s): O. K. Chopra and W. J. Shack (Argonne National Laboratory)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington DC (United States)
Publication Date: March 1998
Report Number(s): NUREG/CR-6583; ANL-97/18
Abstract: The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures 1-9.1 through 1-9.6 of Appendix I to Section 111 of the Code specify fatigue design curves for structural materials. While effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate in coolant environments. This report summarizes work performed by Argonne National Laboratory on fatigue of carbon and lowalloy steels in light water reactor (LWR) environments. The existing fatigue S-N data have been evaluated to establish the effects of various material and loading variables such as steel type, dissolved oxygen level, strain range, strain rate, temperature, orientation, and sulfur content on the fatigue life of these steels. Statistical models have been developed for estimating the fatigue S-N curves as a function of material, loading, and environmental variables. The results have i een used to estimate the probability of fatigue cracking of reactor components. The different methods for incorporating the effects

of LWR coolant environments on the ASME Code fatigue design curves are presented.

Title: Environmentally Assisted Cracking in Light Water Reactors Semiannual Report January 1997 - June 1997
Author(s)/Editor(s): Chopra, O.K. ; Chung, H.M. ; Gavenda, D.J. (Argonne National Lab., IL (United States)) (and others)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: April 1998
Report Number(s): NUREG/CR-4667, Vol 24
Abstract: This report summarizes work performed by Argonne National Laboratory on fatigue and environmentally assisted cracking (EAC) in light water reactors from January 1997 to June 1997. Topics that have been investigated include (a) fatigue of carbon, low-alloy, and austenitic stainless steels (SSs) used in reactor piping and pressure vessels, (b) irradiationassisted stress corrosion cracking of Types 304 and 304L SS, and (c) EAC of AUoys 600 and 690. Fatigue tests were conducted on ferritic and austenitic SSs in water that contained various concentrations of dissolved oxygen (DO) to determine whether a slow strain rate applied during various portions of a tensile-loading cycle is equally effective in decreasing fatigue life. Slow-strain-rate-tensile tests were conducted in simulated boiling water reactor (BWR) water at 288°C on SS specimens irradiated to a low fluence in the Halden reactor and the results were compared with similar data from a control-blade sheath and neutronabsorber tubes irradiated in BWRs to the same fluence level. Crack-growth-rate tests were conducted on compact-tension specimens from several heats of Alloys 600 and 690 in low-DO, simulated pressurized water reactor environments.

Fracture Mechanics Testing and Analysis

Title: Biaxial loading and shallow-flaw effects on crack-tip constraint and fracture toughness
Author(s)/Editor(s): Bass, B.R. ; Bryson, J.W. ; Theiss, T.J. ; Rao, M.C. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1994
Report Number(s): NUREG/CR-6132; ORNL/TM--12498
Order Number: TI94007598
Abstract: A program to develop and evaluate fracture methodologies for the assessment of crack-tip constraint effects on fracture toughness of reactor pressure vessel (RPV) steels has been initiated in the Heavy-Section Steel Technology (HSST) Program. Crack-tip constraint is an issue that significantly impacts fracture mechanics technologies employed in safety assessment procedures for commercially licensed nuclear RPVs.

The focus of studies described herein is on the evaluation of two stressed-based methodologies for quantifying crack-tip constraint (i.e., J-Q theory and a micromechanical scaling model based on critical stressed volumes) through applications to experimental and fractographic data. Data were utilized from single-edge notch bend (SENB) specimens and HSST-developed cruciform beam specimens that were tested in HSST shallow-crack and biaxial testing programs. Results from applications indicate that both the J-Q methodology and the micromechanical scaling model can be used successfully to interpret experimental data from the shallow- and deep-crack SENB specimen tests. When applied to the uniaxially and biaxially loaded cruciform specimens, the two methodologies showed some promising features, but also raised several questions concerning the interpretation of constraint conditions in the specimen based on near-tip stress fields. Fractographic data taken from the fracture surfaces of the SENB and cruciform specimens are used to assess the relevance of stress-based fracture characterizations to conditions at cleavage initiation sites. Unresolved issues identified from these analyses require resolution as part of a validation process for biaxial loading applications. This report is designated as HSST Report No. 142.

Title: Effects of tensile loading on upper shelf fracture toughness
Author(s)/Editor(s): Joyce, J.A. (Naval Academy, Annapolis, MD (United States)); Link, R.E. (Naval Surface Warfare Center, Annapolis, MD (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1994
Report Number(s): NUREG/CR-6051
Order Number: TI94009593
Abstract: Constraint has been an important consideration in fracture mechanics from the earliest work that was done to develop the 1974 version of the ASTM Standard E399. O'Dowd and Shih (1991) have proposed that the difference in crack tip stress fields can be quantified in terms of a field quantity that they have call Q. The Q quantity is a function of J, the crack shape and size, the structural geometry, mode of loading and on the level of deformation and can only be calculated from a high resolution elastic-plastic computational analysis. A similar, simpler, but more controversial approach has been suggested by Betegon and Hancock (1991), who use the non-singular term of the elastic, crack singularity solution, called the T-Stress, as a measure of elastic-plastic crack tip constraint. The objective of this work is to develop some upper shelf, elastic-plastic experimental results to attempt to investigate the applicability of the Q and T stress parameters to the correlation of upper shelf initiation toughness and J resistance curves. The first objective was to obtain upper shelf J resistance curves, J[sub Ic], and tearing

resistance results for a range of applied constraint. The J-Q and J-T stress loci were developed and compared with the expectations of the O'Dowd and Shih and the Betegon and Hancock analyses. Constraint was varied by changing the crack length and also by changing the mode of loading from bending to predominantly tensile. The principle conclusions of this work are that J[sub Ic] does not appear to be dependent on T stress or Q while the material tearing resistance is dependent on T stress and Q, with the tearing modulus increasing as constraint decreases.

Title: Numerical modeling of ductile tearing effects on cleavage fracture toughness
Author(s)/Editor(s): Dodds, R.H. Jr. ; Tang, M. (Univ. of Illinois, Urbana (United States)); Anderson, T.L. (Texas A M Univ., College Station, TX (United States))
Sponsoring Organization: NRC; DOD; Nuclear Regulatory Commission, Washington, DC (United States); Department of Defense, Washington, DC (United States)
Publication Date: May 1994
Report Number(s): NUREG/CR-6162; UILU-ENG--93-2014
Order Number: TI94015146
Abstract: Experimental studies demonstrate a significant effect of specimen size, a/W ratio and prior ductile tearing on cleavage fracture toughness values (J[sub c]) measured in the ductile-to-brittle transition region of ferritic materials. In the lower-transition region, cleavage fracture often occurs under conditions of large-scale yielding but without prior ductile crack extension. The increased toughness develops when plastic zones formed at the crack tip interact with nearby specimen surfaces which relaxes crack-tip constraint (stress triaxiality). In the mid-to-upper transition region, small amounts of ductile crack extension (often < 1-2 mm) routinely precede termination of the J-[Delta]a curve by brittle fracture. Large-scale yielding, coupled with small amounts of ductile tearing, magnifies the impact of small variations in microscale material properties on the macroscopic fracture toughness which contributes to the large amount scatter observed in measured J[sub c]-values. Previous work by the authors described a micromechanics fracture model to correct measured J[sub c]-values for the mechanistic effects of large-scale yielding. This new work extends the model to also include the influence of ductile crack extension prior to cleavage. The paper explores development of the new model, provides necessary graphs and procedures for its application and demonstrates the effects of the model on fracture data sets for two pressure vessel steels (A533B and A515).

Title: Crack-speed relations inferred from large single-edge notched specimens of a 533 B steel
Author(s)/Editor(s): Schwartz, C.W. (Maryland Univ., College Park, MD

(United States). Dept. of Civil Engineering)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1994
Report Number(s): NUREG/CR-5861; ORNL/SUB--79-7778/9
Order Number: TI94016708
Abstract: A relationship between instantaneous crack-tip velocity [dot a], dynamic stress-intensity factor K[sub I], and temperature T for A 533 B steel is estimated using dynamic crack position us time data measured in a series of very large-scale crack-arrest tests. The corresponding dynamic stress intensity us time history and the dynamic-arrest toughness for each test are obtained from generation-mode elastodynamic analyses based on cubic polynomial fits to elastodynamic analytical predictions based on the proposed [dot a]-K[sub I]-T relation are within 7% of experimentally measured arrested crack lengths and within 50% of measured arrest times. These predictions within 50% of measured arrest times. These predictions represent significant improvements over results obtained using previous preliminary estimates of the [dot a]-K[sub I]-T relation for A 533 B steel. The influence of nonlinear material behavior on the results is also evaluated.

Title: Heavy-Section Steel Technology Program Semiannual progress report, October 1992 -- March 1993. Volume 10, No. 1
Author(s)/Editor(s): Pennell, W.E. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: September 1994
Report Number(s): NUREG/CR-4219-Vol.10-No.1; ORNL/TM--9593/V10-N1
Abstract: The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission 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 12 tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterization and properties, (4) special technical assistance, (5) fracture analysis computer programs, (6) cleavage-crack initiation, (7) cladding evaluations (S) pressurized-thermal shock technology, (9) analysis methods validation, (10) fracture evaluation tests, (11) warm prestressing, and (12) biaxial loading effects. The program tasks have been structured to place emphasis on the resolution fracture issues with near-term licensing significance. Resources to execute the research tasks are drawn from ORNL with subcontract support from universities and other research laboratories. Close contact is maintained with the sister Heavy-Section Steel Irradiation Program at ORNL 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 October 1992 to March 1993.

Title: Preliminary assessment of the fracture behavior of weld material in full- thickness clad beams
Author(s)/Editor(s): Keeney, J.A. ; Bass, B.R. ; McAfee, W.J. ; Iskander, S.K. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Oct 1994
Report Number(s): NUREG/CR-6228; ORNL/TM--12735
Order Number: TI95001501
Abstract: This report describes a testing program that utilizes full-thickness clad beam specimens to quantify fracture toughness for shallow cracks in material for which metallurgical conditions are prototypic of those found in reactor pressure vessels (RPVs). The beam specimens are fabricated from a section of an RPV wall (removed from a canceled nuclear plant) that includes weld, plate, and clad material. Metallurgical factors potentially influencing fracture toughness for shallow cracks in the beam specimens include material gradients due to welding and cladding applications, as well as material inhomogeneities in welded regions due to reheating in multiple weld passes. A summary of the testing program includes a description of the specimen geometry, material properties, the testing procedure, and the experimental results form three specimens. The yield strength of the weld material was determined to be 36% higher than the yield strength of the base material. An irradiation-induced increase in yield strength of the weld material could result in a yield stress that exceeds the upper limit where code curves are valid. The high yield strength for prototypic weld material may have implications for RPV structural integrity assessments. Analyses of the test data are discussed, including comparisons of measured displacements with finite-element analysis results, applications of toughness estimation techniques, and interpretations of constraint conditions implied by stress-based constraint methodologies. Metallurgical conditions in the region of the cladding heat-affected zone are proposed as a possible explanation for the lower-bound fracture toughness measured with one of the shallow-crack clad beam specimens. Fracture toughness data from the three clad beam specimens are compared with other shallow- and deep-crack uniaxial beam and cruciform data generated previously from A 533 Grade B plate material.

Title: Cleavage behaviors in nuclear vessel steels
Author(s)/Editor(s): Irwin, G.R. ; Zhang, X.J. (Univ. of Maryland, College Park, MD (United States). Dept. of Mechanical Engineering)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC

(United States)
Publication Date: Nov 1994
Report Number(s): NUREG/CR-6262; ORNL/Sub--79-7778/11
Order Number: TI95004174
Abstract: Cleavage behaviors of nuclear vessel steels in the transition temperature range are reviewed. Viewpoints are presented to assist understanding of cleavage crack speed, cleavage initiation, cleavage arrest, and the sensitivity of fracture toughness to constraint and temperature. The importance of high local stress elevations by high strain rate is emphasized. This report is designated as HSST Report No. 149.

Title: Constraint effects on fracture initiation loads in HSST wide-plate tests
Author(s)/Editor(s): Dodds, R.H. Jr. (Illinois Univ., Urbana, IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Dec 1994
Report Number(s): NUREG/CR-6259; UILU-ENG--94-2009; ORNL/TM--12796
Order Number: TI95005273
Abstract: During the period 1984--1987, researchers of the Heavy-Section Steel Technology program at the Oak Ridge National Laboratory performed a unique series of fracture mechanics tests using exceptionally large, SE(T) specimens (a/W=0.2) fabricated from a reactor pressure vessel material, A533B Class 1 steel. This study re-examines fracture initiation loads in the wide-plate tests using two constraint assessment methodologies developed over the past five years: the J-Q toughness locus approach and the toughness scaling approach based on a local failure criterion for cleavage. Both approaches demonstrate a significant loss of constraint in the elastic-plastic fields ahead of the crack in the wide-plate specimens caused by the inherent negative T-stress of the shallow notch SE(T) configuration. Moreover, the 25mm wide machined notch required for specimen fabrication is shown to further reduce constraint by introducing a traction free surface very near the crack tip. Both of these factors combined to reduce near- tip stresses by 10% below those of the small-scale yielding, SSY (T=0), fields. This reduction places fracture results for the wide-plate specimens within the J-Q toughness locus defined by fracture toughness tests on the A533B material and within the constraint corrected J[sub c] values defined by the toughness scaling methodology.

Title: Validity limits in J-resistance curve determination: An assessment of the J[sub M] Parameter. Volume 1
Author(s)/Editor(s): Shih, C.F. ; Liu, X.H. (Brown Univ., Providence, RI (United States). Div. of Engineering)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1995
Report Number(s): NUREG/CR-6264-Vol.1;

BMI--2181-Vol.1
Order Number: TI95008150
Abstract: Significant advances in elastic-plastic fracture became possible with the introduction of Rice's path independent J-integral which has two physical meanings. First, the J-integral is equivalent to the energy release rate associated with a virtual crack advance. Secondly, J can be regarded as the strength of the stress and strain singularity near a stationary crack tip. As a result of several experimental studies, the J-integral is generally accepted as a valid parameter to characterize a material's resistance to the onset of crack growth under large-scale yielding. Driven by simplicity and the practical benefits that could be derived from a geometry and size-independent material resistance curve for large amounts of crack growth, J[sub M], a modified J parameter was introduced. Initial results using J[sub M] were encouraging but subsequent studies did not support the earlier results. The present computational study presented in Volume 1 of this report investigates several forms of this parameter, how they are derived and the validity of these parameters for small and large amounts of crack growth. It is concluded that neither J nor J[sub M] (nor any single parameter) can satisfactorily capture the full range of near-tip fracture states. A discussion on the range of validity of J[sub M] is given in Volume 2. This work is relevant for assessing structural integrity of nuclear pressure vessels and piping.

Title: Validity limits in J-resistance curve determination: A computational approach to ductile crack growth under large-scale yielding conditions. Volume 2
Author(s)/Editor(s): Shih, C.F. ; Xia, L. (Brown Univ., Providence, RI (United States). Div. of Engineering); Hutchinson, J.W. (Harvard Univ., Cambridge, MA (United States). Div. of Applied Sciences)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1995
Report Number(s): NUREG/CR-6264-Vol.2; BMI--2181-Vol.2
Order Number: TI95008149
Abstract: In this report, Volume 2, Mode I crack initiation and growth under plane strain conditions in tough metals are computed using an elastic/plastic continuum model which accounts for void growth and coalescence ahead of the crack tip. The material parameters include the stress-strain properties, along with the parameters characterizing the spacing and volume fraction of voids in material elements lying in the plane of the crack. For a given set of these parameters and a specific specimen, or component, subject to a specific loading, relationships among load, load-line displacement and crack advance can be computed with no restrictions on the extent of plastic deformation. Similarly, there is no limit on crack advance, except that it must take place on the symmetry plane ahead of the initial crack. Suitably

defined measures of crack tip loading intensity, such as those based on the J-integral, can also be computed, thereby directly generating crack growth resistance curves. In this report, the model is applied to five specimen geometries which are known to give rise to significantly different crack tip constraints and crack growth resistance behaviors. Computed results are compared with sets of experimental data for two tough steels for four of the specimen types. Details of the load, displacement and crack growth histories are accurately reproduced, even when extensive crack growth takes place under conditions of fully plastic yielding. A description of material resistance to crack initiation and subsequent growth is essential for assessing structural integrity such as nuclear pressure vessels and piping.

Title: Biaxial loading effects on fracture toughness of reactor pressure vessel steel
Author(s)/Editor(s): McAfee, W.J. ; Bass, B.R. ; Bryson, J.W. Jr. ; Pennell, W.E. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1995
Report Number(s): NUREG/CR-6273; ORNL/TM--12866
Order Number: TI95008768
Abstract: The preliminary phases of a program to develop and evaluate fracture methodologies for assessing crack-tip constraint effects on fracture toughness of reactor pressure vessel (RPV) steels have been completed by the Heavy-Section Steel Technology (HSST) Program. Objectives were to investigate effect of biaxial loading on fracture toughness, quantify this effect through existing stress-based, dual-parameter, fracture-toughness correlations, or propose and verify alternate correlations. A cruciform beam specimen with 2-D, shallow, through-thickness flaw and a special loading fixture was designed and fabricated. Tests were performed using biaxial loading ratios of 0:1 (uniaxial), 0.6:1, and 1:1 (equi-biaxial). Critical fracture-toughness values were calculated for each test. Biaxial loading of 0.6:1 resulted in a reduction in the lower bound fracture toughness of [approximately]12% as compared to that from the uniaxial tests. The biaxial loading of 1:1 yielded two subsets of toughness values; one agreed well with the uniaxial data, while one was reduced by [approximately]43% when compared to the uniaxial data. Results were evaluated using J-Q theory and Dodds-Anderson (D-A) micromechanical scaling model. The D-A model predicted no biaxial effect, while the J-Q method gave inconclusive results. When applied to the 1:1 biaxial data, these constraint methodologies failed to predict the observed reduction in fracture toughness obtained in one experiment. A strain- based constraint methodology that considers the relationship between applied biaxial load, the plastic zone width in the

crack plane, and fracture toughness was formulated and applied successfully to the data. Evaluation of this dual-parameter strain-based model led to the conclusion that it has the capability of representing fracture behavior of RPV steels in the transition region, including the effects of out-of-plane loading on fracture toughness. This report is designated as HSST Report No. 150.

Title: Stainless steel submerged arc weld fusion line toughness
Author(s)/Editor(s): Rosenfield, A.R. ; Held, P.R. ; Wilkowski, G.M. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-6251; BMI--2180
Order Number: TI95010950
Abstract: 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, Ji, was greater than the SAW toughness but much less than the base metal. Of greater importance may be that the crack growth resistance (JD-R) of the fusion line appeared to reach a steady-state value, while the SAW had a continually increasing JD-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.

Title: Heavy-Section Steel Technology Program Semiannual progress report, April-- September 1993. Volume 10, No. 2
Author(s)/Editor(s): Pennell, W.E. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: May 1995
Report Number(s): NUREG/CR-4219-Vol.10-No.2; ORNL/TM--9593/V10-N2
Order Number: TI95012816
Abstract: 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: Program management, fracture methodology and analysis,

material characterizations and properties, special technical assistance, fracture analysis computer programs, cleavage-crack initiation, cladding evaluations, pressurized-thermal-shock technology, analysis methods validation, fracture evaluation tests, warm prestressing, and 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 provide s an overview of principal developments in each of the 12 program tasks from April -- September 1993.

Title: Size and deformation limits to maintain constraint in K[sub Ic] and J[sub c] testing of bend specimens
Author(s)/Editor(s): Koppenhoefer, K.C. ; Dodds, R.H. Jr. (Illinois Univ., Urbana, IL (United States). Dept. of Civil Engineering)
Sponsoring Organization: NRC; DOD; Nuclear Regulatory Commission, Washington, DC (United States); Department of Defense, Washington, DC (United States)
Publication Date: Oct 1995
Report Number(s): NUREG/CR-6191; UILU-ENG--94-2002
Order Number: TI96002335
Abstract: The ASTM Standard Test Method for Plane-Strain Fracture Toughness of metallic Materials (E399-90) restricts test specimen dimensions to insure the measurement of highly constrained fracture toughness values (K[sub Ic]). These requirements insure small-scale yielding (SSY) conditions at fracture, and thereby the validity of linear elastic fracture mechanics. Recently, Dodds and Anderson have proposed a less restrictive size requirement for cleavage fracture toughness measured in terms of the J-integral (J[sub c]), as given by a, b, B [ge] 200 J[sub c]/[sigma][sub 0]. The size requirement proposed by Dodds and Anderson increases the applicability of fracture toughness experiments by expanding the range of conditions over which fracture toughness data meeting SSY conditions can be reliably measured. This investigation compares the proposed size requirement with that of ASTM Standard Test Method E399 and, by comparison with published experimental data for various alloys, provides validation of the new requirements.

Title: Heavy-section steel technology program: Semiannual progress report, October 1993--March 1994. Volume 11, No. 1
Author(s)/Editor(s): Pennell, W.E. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC

(United States)
Publication Date: Nov 1995
Report Number(s):
NUREG/CR-4219-Vol.11-No.1;
ORNL/TM--9593/V11-N1
Order Number: TI96003353
Abstract: The Heavy-Section Steel Technology (HSST) Program is conducted for the US Nuclear Regulatory Commission (NRC) by Oak Ridge National Laboratory (ORNL). The Program focus is on the development and validation of technology for the assessment Of fracture-prevention margins in commercial nuclear reactor pressure vessels. The HSST Program is organized in seven tasks: (1) program management (2) constraint effects analytical development and validation, (3) evaluation of cladding effects, (4) ductile to cleavage fracture mode conversion, (5) fracture analysis methods development and applications, (6) material Property data and test methods, and (7) integration of results into a state-of-the-art methodology. The program tasks have been structured to place emphasis on the resolution fracture issues with near-term licensing significance. Resources to execute the research tasks are drawn from ORNL with subcontract support from universities and other research laboratories. Close contact is maintained with the sister Heavy-Section Steel Irradiation Program at ORNL 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 October 1993--March 1994.

Title: Application of fracture toughness scaling models to the ductile-to- brittle transition
Author(s)/Editor(s): Link, R.E. (Naval Surface Warfare Center, Annapolis, MD (United States)); Joyce, J.A. (Naval Academy, Annapolis, MD (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1996
Report Number(s): NUREG/CR-6279
Order Number: TI96005815
Abstract: An experimental investigation of fracture toughness in the ductile- brittle transition range was conducted. A large number of ASTM A533, Grade B steel, bend and tension specimens with varying crack lengths were tested throughout the transition region. Cleavage fracture toughness scaling models were utilized to correct the data for the loss of constraint in short crack specimens and tension geometries. The toughness scaling models were effective in reducing the scatter in the data, but tended to over-correct the results for the short crack bend specimens. A proposed ASTM Test Practice for Fracture Toughness in the Transition Range, which employs a master curve concept, was applied to the results. The proposed master curve over predicted the fracture toughness in the mid-transition and a modified master curve was developed that more accurately modeled the transition behavior of the material. Finally, the modified master curve and the fracture toughness scaling models were combined

to predict the as- measured fracture toughness of the short crack bend and the tension specimens. It was shown that when the scaling models over correct the data for loss of constraint, they can also lead to non-conservative estimates of the increase in toughness for low constraint geometries.

Title: Heavy-Section Steel Technology Program: Semiannual progress report for April--September 1994. Volume 11, Number 2
Author(s)/Editor(s): Pennell, W.E. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1996
Report Number(s): NUREG/CR-4219-Vol.11-No.2; ORNL/TM--9593/V11 N2
Order Number: TI96010062
Abstract: 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 applications, (6) material property data and test methods, and (7) integration of results. The program tasks have been structured to place emphasis on the resolution fracture issues with near-term licensing significance. Resources to execute the research tasks are drawn from ORNL with subcontract support from universities and other research laboratories. Close contact is maintained with the sister Heavy-Section Steel Irradiation (HSSI) Program at ORNL and with related research programs both in the US and abroad. This report provides an overview of principal developments in each of the seven program tasks from April 1994 to September 1994.

Title: Numerical investigation of 3-D constraint effects on brittle fracture in SE(B) and C(T) specimens
Author(s)/Editor(s): Nevalainen, M. (Valtion Teknillinen Tutkimuskeskus, Espoo (Finland)); Dodds, R.H. Jr. (Illinois Univ., Urbana, IL (United States). Dept. of Civil Engineering)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1996
Report Number(s): NUREG/CR-6317; UILU-ENG--95-2001
Order Number: TI96013827
Abstract: This investigation employs 3-D nonlinear finite element analyses to conduct an extensive parametric evaluation of crack front stress triaxiality for deep notch SE(B) and C(T) specimens and shallow notch SE(B)

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

Title: Strain rate and inertial effects on impact loaded single-edge notch bend specimens
Author(s)/Editor(s): Vargas, P.M. ; Dodds, R.H. Jr.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1996
Report Number(s): NUREG/CR-6375; UILU-ENG--94-2018
Order Number: TI96013209
Abstract: When the severity of impact loads is sufficient to produce large inelastic deformations, the assessment of crack-tip conditions must include the effects of plasticity, strain rate and inertia. This work examines the interaction of impact loading, inelastic material deformation and rate sensitivity with the goal of improving the interpretation of ductile fracture toughness values measured under dynamic loading. Three-dimensional, nonlinear dynamic analyses are performed for SE(B) fracture specimens (a/W = 0.5, 0.15, 0.0725) subjected to impact loading. Loading rates obtained in conventional drop tower tests (impact load-line velocities of [approx]6 m/sec) are applied in the analyses. Strains at key locations on the specimens and the support reactions (applied load) are extracted from the analyses to assess the accuracy of static formulas commonly used to estimate applied J values. Inertial effects on the applied J are quantified by examining the acceleration component of J evaluated through a domain integral procedure.

Title: Heavy-section steel technology program. Semiannual progress report October 1994--March 1995
Author(s)/Editor(s): Pennell, W.E.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1996
Report Number(s): NUREG/CR-4219-Vol.12-No.1; ORNL/TM--9593
Order Number: TI96013666
Abstract: 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 applications, (6) material property data and test methods, and (7) integration of results. The program tasks have been structured to place emphasis on the resolution of 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 seven program tasks from October 1994-March 1995.

Title: CSNI Project for Fracture Analyses of Large-Scale International Reference Experiments (FALSIRE II)
Author(s)/Editor(s): Bass, B.R. ; Pugh, C.E. ; Keeney, J. (Oak Ridge National Lab., TN (United States)); Schulz, H. ; Sievers, J. (Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) mbH, Koeln (Gemany))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Nov 1996
Report Number(s): NUREG/CR-6460; ORNL/TM--13207
Order Number: TI97001357
Abstract: A summary of Phase II of the Project for FALSIRE is presented. FALSIRE was created by the Fracture Assessment Group (FAG) of the OECD/NEA's Committee on the Safety of Nuclear Installations (CNSI) Principal Working Group No. 3. FALSIRE I in 1988 assessed fracture methods through interpretive analyses of 6 large-scale fracture experiments in reactor pressure vessel (RPV) steels under pressurized- thermal-shock (PTS) loading. In FALSIRE II, experiments examined cleavage fracture in RPV steels for a wide range of materials, crack geometries, and constraint and loading conditions. The cracks were relatively shallow, in the transition temperature region. Included were cracks showing either unstable extension or two stages of extensions under transient thermal and mechanical loads. Crack initiation was also investigated in connection with clad surfaces and with biaxial load. Within FALSIRE II, comparative assessments were performed for 7 reference fracture experiments based on 45 analyses received from 22 organizations representing 12 countries. Temperature distributions in thermal shock loaded samples were approximated with high accuracy and small scatter bands. Structural response was predicted reasonably well; discrepancies could usually be traced to the assumed

material models and approximated material properties. Almost all participants elected to use the finite element method.

Title: Heavy-section steel technology program. Semiannual progress report, April-- September 1995 Vol. 12, No. 2
Author(s)/Editor(s): Pennell, W.E.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1997
Report Number(s): NUREG/CR-4219-Vol.12-No.2; ORNL/TM--9593/V12 N2
Order Number: TI97002993
Abstract: The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission by Oak Ridge National Laboratory (ORNL). The program focus is on the development and validation of technology for the assessment of fracture-prevention margins in commercial nuclear reactor pressure vessels. The HSST Program is organized in seven tasks: (1) program management, (2) constraint effects analytical development and validation, (3) evaluation of cladding effects, (4) ductile-to-cleavage fracture-mode conversion, (5) fracture analysis methods development and applications, (6) material property data and test methods, and (7) integration of results. The program tasks have been structured to place emphasis on the resolution of 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 seven program tasks from April 1995 to September 1995.

Title: Ductile fracture toughness of modified A 302 Grade B Plate materials, data analysis. Volume 1
Author(s)/Editor(s): McCabe, D.E. ; Manneschmidt, E.T. ; Swain, R.L.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1997
Report Number(s): NUREG/CR-6426-Vol.1; ORNL--6892-Vol.1
Order Number: TI97004262
Abstract: The goal of this work was to develop ductile fracture toughness data in the form of J-R curves for modified A302 grade B plate materials typical of those used in reactor pressure vessels. A previous experimental study on one heat of A302 grade B plate showed decreasing J-R curves with increased specimen thickness. This characteristic has not been observed in tests made on recent production materials of A533 grade B and A508 class 2 pressure vessel steels. It was unknown if the departure from norm for the material was a generic characteristic for all heats of A302

grade B steels or unique to that particular plate. Seven heats of modified A302 grade B steel and one heat of vintage A533 grade B steel were tested for chemical content, tensile properties, Charpy transition temperature curves, drop-weight nil-ductility transition (NDT) temperature, and J- R curves. Tensile tests were made in the three principal orientations and at four temperatures, ranging from room temperature to 550F. Charpy V-notch transition temperature curves were obtained in longitudinal, transverse, and short transverse orientations. J-R curves were made using four specimen sizes (1/2T, 1T, 2T, and 4T). The fracture mechanics-based evaluation method covered three test orientations and three test temperatures (80, 400, and 550F). However, the coverage of these variables was contingent upon the amount of material provided. Drop-weight NDT temperature was determined for the T-L orientation only. None of the heats of modified A302 grade B showed size effects of any consequence on the J-R curve behavior. Crack orientation effects were present, but none were severe enough to be reported as atypical. A test temperature increase from 180 to 550F produced the usual loss in J-R curve fracture toughness. Generic J-R curves and curve fits were generated to represent each heat of material. This volume deals with the evaluation of data and the discussion of technical findings. 8 refs., 18 figs., 8 tabs.

Title: Ductile fracture toughness of modified A 302 grade B plate materials. Volume 2
Author(s)/Editor(s): McCabe, D.E. ; Manneschmidt, E.T. ; Swain, R.L.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1997
Report Number(s): NUREG/CR-6426-Vol.2; ORNL--6892/V2
Order Number: TI97003758

Abstract: The objective of this work was to develop ductile fracture toughness data in the form of J-R curves for modified A 302 grade B plate materials typical of those used in fabricating reactor pressure vessels. A previous experimental study at Materials Engineering Associates (MEA) on one particular heat of A 302 grade B plate showed decreasing J-R curves with increased specimen thickness. This characteristic has not been observed in numerous tests made on the more recent production materials of A 533 grade B and A 508 class 2 pressure vessel steels. It was unknown if the departure from norm for the MEA material was a generic characteristic for all heats of A 302 grade B steels or just unique to that one particular plate. Seven heats of modified A 302 grade B steel and one heat of vintage A 533 grade B steel were provided to this project by the General Electric Company of San Jose, California. All plates were tested for chemical content, tensile properties, Charpy transition temperature curves, drop-weight nil-ductility transition (NDT) temperature, and J-R curves. Tensile

tests were made in the three principal orientations and at four temperatures, ranging from room temperature to 550[degrees]F (288[degrees]C). Charpy V-notch transition temperature curves were obtained in longitudinal, transverse, and short transverse orientations. J-R curves were made using four specimen sizes (1/2T, IT, 2T, and 4T). None of the seven heats of modified A 302 grade showed size effects of any consequence on the J-R curve behavior. Crack orientation effects were present, but none were severe enough to be reported as atypical. A test temperature increase from 180 to 550[degrees]F (82 to 288[degrees]C) produced the usual loss in J-R curve fracture toughness. Generic J-R curves and mathematical curve fits to the same were generated to represent each heat of material. This volume is a compilation of all data developed.

<u>Title:</u> Heavy-section steel technology program: Semiannual progress report for October 1995--March 1996. Volume 13, Number 1
<u>Author(s)/Editor(s):</u> Pennell, W.E. (Oak Ridge National Lab., TN (United States))
<u>Sponsoring Organization:</u> NRC: Nuclear Regulatory Commission, Washington, DC (United States)
<u>Publication Date:</u> Sep 1997
<u>Report Number(s):</u> NUREG/CR-4219-Vol.13-No.1; ORNL/TM--9593/V13 N1
<u>Order Number:</u> TI98000428
<u>Abstract:</u> The Heavy-Section Steel Technology (HSST) Program is conducted for the US Nuclear Regulatory Commission (NRC) by the Oak Ridge National Laboratory (ORNL). The program focus is on the development and validation of technology for the assessment of fracture-prevention margins in commercial nuclear reactor pressure vessels. The HSST Program is organized in seven tasks: (1) program management, (2) constraint effects analytical development and validation, (3) evaluation of cladding effects, (4) ductile to cleavage fracture mode conversion, (5) fracture analysis methods development and applications, (6) material property data and test methods, and (7) integration of results into a state-of-the-art methodology. The program tasks have been structured to place emphasis on the resolution fracture issues with near-term licensing significance. Resources to execute the research tasks are drawn from ORNL with subcontract support from universities and other research laboratories. Close contact is maintained with the sister Heavy-Section Steel Irradiation Program at ORNL with related research programs both in the US and abroad. This report provides an overview of principal developments in each of the seven program tasks from October 1995--March 1996.

Non Destructive Examination

<u>Title:</u> Evaluation of computer-based ultrasonic inservice inspection systems

Author(s)/Editor(s): Harris, R.V. Jr. ; Angel, L.J. ; Doctor, S.R. ; Park, W.R. ; Schuster, G.J. ; Taylor, T.T. (Pacific Northwest Lab., Richland, WA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1994
Report Number(s): NUREG/CR-5985; PNL--8919
Order Number: TI94010311
Abstract: This report presents the principles, practices, terminology, and technology of computer-based ultrasonic testing for inservice inspection (UT/ISI) of nuclear power plants, with extensive use of drawings, diagrams, and LTT images. The presentation is technical but assumes limited specific knowledge of ultrasonics or computers. The report is divided into 9 sections covering conventional LTT, computer-based LTT, and evaluation methodology. Conventional LTT topics include coordinate axes, scanning, instrument operation, RF and video signals, and A-, B-, and C-scans. Computer-based topics include sampling, digitization, signal analysis, image presentation, SAFI, ultrasonic holography, transducer arrays, and data interpretation. An evaluation methodology for computer-based LTT/ISI systems is presented, including questions, detailed procedures, and test block designs. Brief evaluations of several computer-based LTT/ISI systems are given; supplementary volumes will provide detailed evaluations of selected systems.

Title: Feasibility of developing risk-based rankings of pressure boundary systems for inservice inspection
Author(s)/Editor(s): Vo, T.V. ; Smith, B.W. ; Simonen, F.A. ; Gore, B.F.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Aug 1994
Report Number(s): NUREG/CR-6151; PNL--8912
Order Number: TI94018068
Abstract: The goals of the Evaluation and Improvement of Non-destructive Examination Reliability for the In-service Inspection of Light Water Reactors Program sponsored by the Nuclear Regulatory Commission at Pacific Northwest Laboratory (PNL) are to (1) assess current ISI techniques and requirements for all pressure boundary systems and components, (2) determine if improvements to the requirements are needed, and (3) if necessary, develop recommendations for revising the applicable ASME Codes and regulatory requirements. In evaluating approaches that could be used to provide a technical basis for improved inservice inspection plans, PNL has developed and applied a method that uses results of probabilistic risk assessment (PRA) to establish piping system ISI requirements. In the PNL program, the feasibility of generic ISI requirements is being addressed in two phases. Phase I involves identifying and prioritizing the systems most relevant to plant safety. The results of these evaluations will be later consolidated into requirements for

comprehensive inservice inspection of nuclear power plant components that will be developed in Phase II. This report presents Phase I evaluations for eight selected plants and attempts to compare these PRA-based inspection priorities with current ASME Section XI requirements for Class 1, 2 and 3 systems. These results show that there are generic insights that can be extrapolated from the selected plants to specific classes of light water reactors.

Title: Review of P-scan computer-based ultrasonic inservice inspection system. Supplement 1
Author(s)/Editor(s): Harris, R.V. Jr. ; Angel, L.J. (Pacific Northwest Lab., Richland, WA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Dec 1995
Report Number(s): NUREG/CR-5985-Suppl.1; PNL--8919-Suppl.1
Order Number: TI96008319
Abstract: This Supplement reviews the P-scan system, a computer-based ultrasonic system used for inservice inspection of piping and other components in nuclear power plants. The Supplement was prepared using the methodology described in detail in Appendix A of NUREG/CR-5985, and is based on one month of using the system in a laboratory. This Supplement describes and characterizes: computer system, ultrasonic components, and mechanical components; scanning, detection, digitizing, imaging, data interpretation, operator interaction, data handling, and record-keeping. It includes a general description, a review checklist, and detailed results of all tests performed.

Title: A pilot application of risk-based methods to establish in-service inspection priorities for nuclear components at Surry Unit 1 Nuclear Power Station
Author(s)/Editor(s): Vo, T. ; Gore, B. ; Simonen, F. ; Doctor, S. (Pacific Northwest Lab., Richland, WA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Aug 1994
Report Number(s): NUREG/CR-6181; PNL--9020
Order Number: TI94018066
Abstract: As part of the Nondestructive Evaluation Reliability Program sponsored by the US Nuclear Regulatory Commission, the Pacific Northwest Laboratory is developing a method that uses risk-based approaches to establish in-service inspection plans for nuclear power plant components. This method uses probabilistic risk assessment (PRA) results and Failure Modes and Effects Analysis (FEMA) techniques to identify and prioritize the most risk-important systems and components for inspection. The Surry Nuclear Power Station Unit 1 was selected for pilot applications of this method. The specific systems addressed in this report are the

reactor pressure vessel, the reactor coolant, the low-pressure injection, and the auxiliary feedwater. The results provide a risk-based ranking of components within these systems and relate the target risk to target failure probability values for individual components. These results will be used to guide the development of improved inspection plans for nuclear power plants. To develop inspection plans, the acceptable level of risk from structural failure for important systems and components will be apportioned as a small fraction (i.e., 5%) of the total PRA-estimated risk for core damage. This process will determine target (acceptable) risk and target failure probability values for individual components. Inspection requirements will be set at levels to assure that acceptable failure probabilistics are maintained.

Title: Real-time 3-D SAFT-UT system evaluation and validation Title Augmentation: Synthetic Aperture Focusing Technique for Ultrasonic Testing
Author(s)/Editor(s): Doctor, S.R. ; Schuster, G.J. ; Reid, L.D. ; Hall, T.E. (Pacific Northwest National Lab., Richland, WA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Sep 1996
Report Number(s): NUREG/CR-6344; PNNL--10571
Order Number: TI97001474
Abstract: SAFT-UT technology is shown to provide significant enhancements to the inspection of materials used in US nuclear power plants. This report provides guidelines for the implementation of SAFT-UT technology and shows the results from its application. An overview of the development of SAFT-UT is provided so that the reader may become familiar with the technology. Then the basic fundamentals are presented with an extensive list of references. A comprehensive operating procedure, which is used in conjunction with the SAFT-UT field system developed by Pacific Northwest Laboratory (PNL), provides the recipe for both SAFT data acquisition and analysis. The specification for the hardware implementation is provided for the SAFT-UT system along with a description of the subsequent developments and improvements. One development of technical interest is the SAFT real time processor. Performance of the real-time processor is impressive and comparison is made of this dedicated parallel processor to a conventional computer and to the newer high-speed computer architectures designed for image processing. Descriptions of other improvements, including a robotic scanner, are provided. Laboratory parametric and application studies, performed by PNL and not previously reported, are discussed followed by a section on field application work in which SAFT was used during inservice inspections of operating reactors.

Title: A pilot application of

risk-informed methods to establish inservice inspection priorities for nuclear components at Surry Unit 1 Nuclear Power Station. Revision 1
Author(s)/Editor(s): Vo, T.V. ; Phan, H.K. ; Gore, B.F. ; Simonen, F.A. ; Doctor, S.R. (Pacific Northwest National Lab., Richland, WA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1997
Report Number(s): NUREG/CR-6181-Rev.1; PNNL--9020-Rev.1
Order Number: TI97004344
Abstract: As part of the Nondestructive Evaluation Reliability Program sponsored by the US Nuclear Regulatory Commission, the Pacific Northwest National Laboratory has developed risk-informed approaches for inservice inspection plans of nuclear power plants. This method uses probabilistic risk assessment (PRA) results to identify and prioritize the most risk-important components for inspection. The Surry Nuclear Power Station Unit 1 was selected for pilot application of this methodology. This report, which incorporates more recent plant-specific information and improved risk-informed methodology and tools, is Revision 1 of the earlier report (NUREG/CR-6181). The methodology discussed in the original report is no longer current and a preferred methodology is presented in this Revision. This report, NUREG/CR-6181, Rev. 1, therefore supersedes the earlier NUREG/CR-6181 published in August 1994. The specific systems addressed in this report are the auxiliary feedwater, the low-pressure injection, and the reactor coolant systems. The results provide a risk-informed ranking of components within these systems.

Title: An Evaluation of Human Factors Research for Ultrasonic Inservice Inspection
Author(s)/Editor(s): D.J. Pond, D.T. Donohoo, R.V. Harris, Jr. (Pacific Northwest National Laboratory)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: March 1998
Report Number(s): NUREG/CR- 6605; PNNL-11797
Abstract: This work was undertaken to determine if human factors research has yielded information applicable to upgrading requirements in ASME Boiler and Pressure Vessel Code Section XI, improving methods and techniques in Section V, and/or suggesting relevant research. A preference was established for information and recommendations which have become accepted and standard practice.

Manual Ultrasonic Testing/Inservice Inspection (UT/ISI) is a complex task subject to influence by dozens of variables. This review frequently revealed equivocal findings regarding affects of environmental variables as well as repeated indications that inspection performance may bo more, and more reliably, influenced by tho workers' social environment, including managerial practices, than by other situational variables. Also of significance are each inspector's relevant knowledge, skills, and abilities, and determination of these is seen as a necessary first step in upgrading requirements, methods, ant techniques as well as in focusing research in support of such programs. While understanding tho effects and mediating mechanisms of the variables impacting inspection performance is a worthwhile pursuit for researchers, initial improvements in industrial UT/IS performance may be achieved by implementing practices already known to mitigate the effects of potentially adverse conditions.

Piping

Title: Short cracks in piping and piping wells
Author(s)/Editor(s): Wilkowski, G.M. ; Brust, F. ; Francini, R. (Battelle, Columbus, OH (United States)) (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1994
Report Number(s): NUREG/CR-4599-Vol.3-No.2; BMI--2173-Vol.3-No.2
Order Number: TI94008267
Abstract: This is the sixth semiannual report of the US Nuclear Regulatory Commission's 4-year research program Short Cracks in Piping and Piping Welds'' which began in March 1990. The objective is to verify and improve fracture analyses for circumferentially cracked nuclear piping with cracks sizes typically found during in-service flaw evaluations. Progress is the through-wall-cracked pipe efforts involved (1) verification of deformation plasticity under nonproportional loading, (2) evaluation of the effect of weld metal strength on various J-estimation schemes, and (3) development of new GE/EPRI functions. Surface-cracked pipe evaluations involved (1) material characterization of B W C- Mn-Mo submerged arc weld metal, and (2) 3D finite-element mesh refinement study. The toughness of the bimetallic weld fusion line was evaluated and showed unusual fracture behavior based on the results of the Charpy tests. The dynamic strain aging J-R tests confirmed the screening criterion developed earlier in the program. The results from this program to date necessitated several additional efforts. These were initiated and have been reported here. Presentation of the results from this program to the ASME Section XI Pipe Flaw Evaluation Working Group is also summarized here.

Title: Review of Elastic Stress and Fatigue-to-Failure Data for Branch Connections and Tees in Relation to ASME Design Criteria for Nuclear Power Piping Systems
Author(s)/Editor(s): E.C.Rodabaugh, S.E. Moore, R.C. Gwaltney
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: May 1994
Report Number(s): NUREG/CR-

5359;ORNL/TM-11152

Abstract: This is the third in a series of reports on the state-of-the art design guidance for piping system branch connections and tees provided by Section III of the ASME Boiler and Pressure Vessel Code. The other reports covered pnmary or limit-loads and nozzle flexibility. The principal objective of this report, as with the others, was to identify and collect the pertinent literature on the subject and to identify needed improvements in the design methods and criteria of the Code based on the evaluation of the available information. This report does not propose changes in the design pracedure of the Code. This report discusses the evaluation of stresses in branch connections and tees, correlation of these stresses with fatigue failures, and the Code rules for protection against fatigue failure in design applications. Because of the extensive amount of available information, the report was divided into two parts. Part I discusses cyclic internal pressure loading and Part II discusses moment loadings for the branch and run. The cyclic pressure loading fatigue parameters are mostly based on leakage, whereas, if the parameters were based on crack initiation, different and possibly higher valves would be developed. The fatigue evaluation procedure, which attempts to relate fatigue strength of piping components to strain controlled, polished bar, and fatigue data appears to be inaccurate on the conservative side for high amplitude cycles and on the unconservative side for low amplitude cycles. The report proposes addidonal analytical and experimental work.

Title: Evaluation and refinement of leak-rate estimation models

Author(s)/Editor(s): Paul, D.D. ; Ahmad, J. ; Scott, P.M. ; Flanigan, L.F. ; Wilkowski, G.M. (Battelle, Columbus, OH (United States))

Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1994
Report Number(s): NUREG/CR-5128-Rev.1; BMI--2164-Rev.1
Order Number: TI94015006

Abstract: Leak-rate estimation models are important elements in developing a leak- beforebreak methodology in piping integrity and safety analyses. Existing thermalhydraulic and crack-opening-area models used in current leak-rate estimations have been incorporated into a single computer code for leak-rate estimation. The code is called SQUIRT, which stands for Seepage Quantification of Upsets In Reactor Tubes. The SQUIRT program has been validated by comparing its thermalhydraulic predictions with the limited experimental data that have been published on two-phase flow through slits and cracks, and by comparing its crack-opening-area predictions with data from the Degraded Piping Program. In addition, leak-rate experiments were conducted to obtain validation data for a circumferential fatigue crack in a carbon steel pipe girth weld.

Title: Validation of analysis methods for assessing flawed piping subjected to dynamic loading
Author(s)/Editor(s): Olson, R.J. ; Wolterman, R.L. ; Wilkowski, G.M. (Battelle, Columbus, OH (United States)); Kot, C.A. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Aug 1994
Report Number(s): NUREG/CR-6234; ANL--94/22; BMI--2178
Order Number: TI94017485
Abstract: Argonne National Laboratory and Battelle have jointly conducted a research program for the USNRC to evaluate the ability of current engineering analysis methods and one state-of-the-art analysis method to predict the behavior of circumferentially surface-cracked pipe system water-hammer experiment. The experimental data used in the evaluation were from the HDR Test Group E31 series conducted by the Kernforschungszentrum Karlsruhe (KfK) in Germany. The incentive for this evaluation was that simplified engineering methods, as well as newer state-of-the-art'' fracture analysis methods, have been typically validated only with static experimental data. Hence, these dynamic experiments were of high interest. High-rate dynamic loading can be classified as either repeating, e.g., seismic, or nonrepeating, e.g., water hammer. Development of experimental data and validation of cracked pipe analyses under seismic loading (repeating dynamic loads) are being pursued separately within the NRC's International Piping Integrity Research Group (IPIRG) program. This report describes developmental and validation efforts to predict crack stability under water hammer loading, as well as comparisons using currently used analysis procedures. Current fracture analysis methods use the elastic stress analysis loads decoupled from the fracture mechanics analysis, while state-of-the-art methods employ nonlinear cracked-pipe time-history finite element analyses. The results showed that the current decoupled methods were conservative in their predictions, whereas the cracked pipe finite element analyses were more accurate, yet slightly conservative. The nonlinear time-history cracked-pipe finite element analyses conducted in this program were also attractive in that they were done on a small Apollo DN5500 workstation, whereas other cracked-pipe dynamic analyses conducted in Europe on the same experiments required the use of a CRAY2 supercomputer, and were less accurate.

Title: Stability of cracked pipe under inertial stresses
Author(s)/Editor(s): Scott, P. ; Wilson, M. ; Olson, R. ; Marschall, C. ; Schmidt, R. ; Wilkowski, G. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)

Publication Date: Aug 1994
Report Number(s): NUREG/CR-6233-Vol.1; BMI--2177-Vol.1
Order Number: TI94018624
Abstract: This report presents the results of the pipe fracture experiments, analyses, and material characterization efforts performed within Subtask 1.1 of the IPIRG Program. The objective of Subtask 1.1 was to experimentally verify the analysis methodologies for circumferentially cracked pipe subjected primarily to inertial stresses. Eight cracked-pipe experiments were conducted on 6-inch nominal diameter TP304 and A106B pipe. The experimental procedure was developed using nonlinear time-history finite element analyses which included the nonlinear behavior due to the crack. The model did an excellent job of predicting the displacements, forces, and times to maximum moment. The comparison of the experimental loads to the predicted loads by the Net-Section-Collapse (NSC), Dimensionless Plastic-Zone Parameter, J-estimation schemes, R6, and ASME Section XI in-service flaw assessment criteria tended to underpredict the measured bending moments except for the NSC analysis of the A106B pipe. The effects of flaw geometry and loading history on toughness were evaluated by calculating the toughness from the pipe tests and comparing these results to C(1) values. These effects were found to be variable. The surface-crack geometry tended to increase the toughness (relative to CM results), whereas a negative load-ratio significantly decreased the TP304 stainless steel surface-cracked pipe apparent toughness. The inertial experiments tended to achieve complete failure within a few cycles after reaching maximum load in these relatively small diameter pipe experiments. Hence, a load-controlled fracture mechanics analysis may be more appropriate than a displacement-controlled analysis for these tests.

Title: Effect of dynamic strain aging on the strength and toughness of nuclear ferritic piping at LWR temperatures
Author(s)/Editor(s): Marschall, C.W. ; Mohan, R. ; Krishnaswamy, P. ; Wilkowski, G.M. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Oct 1994
Report Number(s): NUREG/CR-6226; BMI--2176
Order Number: TI95002451
Abstract: This topical report is on the phenomenon of dynamic strain aging (DSA) in ferritic nuclear piping steels and its effect on fracture at LWR temperatures. The report was a deliverable from the US NRC's program entitled Short Cracks in Piping and Piping Welds''. The objective of this work was to predict the occurrence of and evaluate the effects of ductile crack instabilities, which occur frequently in ferritic steel pipe fracture tests at 288 C (550 F), and

are believed to be due to dynamic stain aging. Numerous laboratory tests and one numerical simulation of a C(T) test with crack instabilities were undertaken.

Title: Refinement and evaluation of crack-opening-area analyses for circumferential through-wall cracks in pipes
Author(s)/Editor(s): Rahman, S. ; Brust, F. ; Ghadiali, N. ; Krishnaswamy, P. ; Wilkowski, G. (Battelle, Columbus, OH (United States)); Choi, Y.H. (Battelle, Columbus, OH (United States) Korea Inst. of Nuclear Safety, Taejeon (Korea, Republic of)); Moberg, F. ; Brickstad, B. (Battelle, Columbus, OH (United States) Swedish Plant Inspection Ltd., Stockholm (Sweden))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-6300; BMI--2184
Order Number: TI95010503
Abstract: 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 effects from pipe rupture. The application of the LBB methodology frequently requires calculation of leak rates. These leak rates depend on the crack-opening area of a 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 11. 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. 140 refs., 105 figs., 41 tabs.

Title: Short cracks in piping and piping welds. Seventh program report, March 1993- December 1994. Volume 4, Number 1
Author(s)/Editor(s): Wilkowski, G.M. ; Ghadiali, N. ; Rudland, D. ; Krishnaswamy, P. ; Rahman, S. ; Scott, P. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-4599-Vol.4-No.1;

BMI--2173-Vol.4-No.1
Order Number: TI95010955
Abstract: This is the seventh progress report of the U.S. Nuclear Regulatory Commission's research program entitled [open quotes]Short Cracks in Piping and Piping Welds[close quotes]. The program objective is to verify and improve fracture analyses for circumferentially 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: (1) Improvements to the two computer programs NRCPIPE and NRCPIPES to assess the failure behavior of circumferential through-wall and surface-cracked pipe, respectively; (2) Pipe material property database PIFRAC; (3) Circumferentially cracked pipe database CIRCUMCK.WKI; (4) An assessment of the proposed ASME Section III design stress rule changes on pipe flaw tolerance; and (5) 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.

Title: Fracture evaluations of fusion line cracks in nuclear pipe bimetallic welds
Author(s)/Editor(s): Scott, P. ; Francini, R. ; Rahman, S. ; Rosenfield, A. ; Wilkowski, G. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-6297; BMI--2182
Order Number: TI95010502
Abstract: In both BWRs 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.

Title: Effects of toughness anisotropy and combined tension, torsion, and bending loads on fracture behavior of ferritic nuclear pipe
Author(s)/Editor(s): Mohan, R. ; Marschall, C. ; Krishnaswamy, P. ; Brust, F. ; Ghadiali, N. ; Wilkowski, G. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-6299
Order Number: TI95012202
Abstract: This topical report summarizes the work on angled crack growth and combined loading effects performed within the Nuclear Regulatory Commission's research program entitled [open quotes]Short Cracks in Piping and Piping Welds[close quotes]. 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 mariner.

Title: Probabilistic pipe fracture evaluations for leak-rate-detection applications
Author(s)/Editor(s): Rahman, S. ; Ghadiali, N. ; Paul, D. ; Wilkowski, G. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-6004; BMI--2174
Order Number: TI95011013
Abstract: Regulatory Guide 1.45, [open quotes]Reactor Coolant Pressure Boundary Leakage Detection Systems,[close quotes] 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, [open quotes]Leak Before Break Evaluation Procedures[close quotes] 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.

Title: Assessment of short through-wall circumferential cracks in pipes. Experiments and analysis: March 1990--December 1994
Author(s)/Editor(s): Brust, F.W. ; Scott, P. ; Rahman, S. (Battelle, Columbus, OH (United States)) (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-6235; BMI--2179
Order Number: TI95010506
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 focuses 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: (1) material property determinations, (2) pipe fracture experiments, and (3) 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 PIFRAC 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 11 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. 69 refs., 124 figs, 49 tabs.

Title: Fracture behavior of short circumferentially surface-cracked pipe
Author(s)/Editor(s): Krishnaswamy, P. ; Scott, P. ; Mohan, R. (Battelle, Columbus OH (United States)) (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Nov 1995
Report Number(s): NUREG/CR-6298; BMI--2183
Order Number: TI96004052
Abstract: This topical report summarizes the work performed for the Nuclear Regulatory Comniission's (NRC) research program entitled Short Cracks in Piping and Piping Welds'' that specifically focuses on pipes with short, circumferential surface cracks. The following details are provided in this report: (i) material property deteminations, (ii) pipe fracture experiments, (iii) development, modification and validation of fracture analysis methods, and (iv) impact of this work on the ASME Section XI Flaw Evaluation Procedures. The material properties developed and used in the analysis of the experiments are included in this report and have been implemented into the NRC's PIFRAC database. Six full-scale pipe experiments were conducted during this program. The analyses methods reported here fall into three categories (i) limit-load approaches, (ii) design criteria, and (iii) elastic-plastic fracture methods. These methods were evaluated by comparing the analytical predictions with experimental data. The results, using 44 pipe experiments from this and other programs, showed that the SC.TNP1 and DPZP analyses were the most accurate in predicting maximum load. New Z-factors were developed using these methods. These are being considered for updating the ASME Section XI criteria.

Title: The effect of cyclic and dynamic loads on carbon steel pipe
Author(s)/Editor(s): Rudland, D.L. ; Scott, P.M. ; Wilkowski, G.M. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1996
Report Number(s): NUREG/CR-6438; BMI--2188
Order Number: TI96007099
Abstract: This report presents the results of four 152-mm (6-inch) diameter, unpressurized, circumferential through-wall-cracked, dynamic pipe experiments fabricated from STS410 carbon steel pipe manufactured in Japan. For three of these experiments, the through-wall crack was in the base metal. The displacement histories applied to these experiments were a quasi-static monotonic, dynamic monotonic, and dynamic, cyclic (R = [minus]1) history. The through-wall crack for the third

experiment was in a tungsten-inert-gas weld, fabricated in Japan, joining two lengths of STS410 pipe. The displacement history for this experiment was the same history applied to the dynamic, cyclic base metal experiment. The test temperature for each experiment was 300 C (572 F). The objective of these experiments was to compare a Japanese carbon steel pipe material with US pipe material, to ascertain whether this Japanese steel was as sensitive to dynamic and cyclic effects as US carbon steel pipe. In support of these pipe experiments, quasi-static and dynamic, tensile and fracture toughness tests were conducted. An analysis effort was performed that involved comparing experimental crack initiation and maximum moments with predictions based on available fracture prediction models, and calculating J-R curves for the pipe experiments using the [eta]-factor method.

Title: Summary of results from the IPIRG-2 round-robin analyses
Author(s)/Editor(s): Rahman, S. ; Olson, R. ; Rosenfield, A. ; Wilkowski, G. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1996
Report Number(s): NUREG/CR-6337; BMI--2186
Order Number: TI96006050
Abstract: This report presents a summary of the results from three one-day international round-robin workshops which were organized by Battelle in conjunction with the Second International Piping Integrity Research Group (IPIRG- 2) Program. The objective of these workshops was to develop a consensus in handling difficult analytical problems in leak-before-break and pipe flaw evaluations. The workshops, which were held August 5, 1993, March 4, 1994, and October 21, 1994 at Columbus, Ohio, involved various technical presentations on the related research efforts by the IPIRG-2 member organizations and solutions to several round-robin problems. Following review by the IPIRG-2 members, four sets of round-robin problems were developed. They involved: (1) evaluations of fracture properties and pipe loads, (2) crack-opening and leak-rate evaluations, (3) dynamic analysis of cracked pipes, and (4) evaluations of elbows. A total of 18 organizations from the United States, Japan, Korea, and Europe solved these round-robin problems. The analysis techniques employed by the participants included both finite element and engineering methods. Based on the results from these analyses, several important observations were made concerning the predictive capability of the current fracture-mechanics and thermal-hydraulics models for their applications in nuclear piping and piping welds.

Title: Design of the IPIRG-2 simulated seismic forcing function

Author(s)/Editor(s): Olson, R. ; Scott, P. ; Wilkowski, G. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1996
Report Number(s): NUREG/CR-6439; BMI--2189
Order Number: TI96006110
Abstract: A series of pipe system experiments was conducted in IPIRG-2 that used a realistic seismic forcing function. Because the seismic forcing function was more complex than the single-frequency increasing-amplitude sinusoidal forcing function used in the IPIRG-1 pipe system experiments, considerable effort went into designing the function. This report documents the design process for the seismic forcing function used in the IPIRG-2 pipe system experiments.

Title: Development of a J-estimation scheme for internal circumferential and axial surface cracks in elbows
Author(s)/Editor(s): Mohan, R. ; Brust, F.W. ; Ghadiali, N. ; Wilkowski, G.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1996
Report Number(s): NUREG/CR-6445; BMI--2193
Order Number: TI96012174
Abstract: This report summarizes efforts to develop elastic and elastic-plastic fracture mechanics analyses for internal surface cracks in elbows. The analyses involved development of a GE/EPRI type J-estimation scheme which requires an elastic and fully plastic contribution to crack-driving force in terms of the J- integral parameter. The elastic analyses require the development of F-function values to relate the J[sub e] term to applied loads. Similarly, the fully plastic analyses require the development of h-functions to relate the J[sub p] term to the applied loads. The F- and h-functions were determined from a matrix of finite element analyses. To minimize the cost of the analyses, three- dimensional ABAQUS finite element analyses were compared to a simpler finite element technique called the line-spring method. The line-spring method provides a significant computational savings over the full three-dimensional analysis. The comparison showed excellent agreement between the line-spring and three- dimensional analysis. This experience was consistent with comparisons with circumferential surface-crack analyses in straight pipes during the NRC's Short Cracks in Piping and Piping Welds program.

Title: Deterministic and probabilistic evaluations for uncertainty in pipe fracture parameters in leak-before-break and in-service flaw evaluations
Author(s)/Editor(s): Ghadiali, N. ; Wilkowski, G. (Battelle, Columbus, OH (United States)); Rahman, S. (Univ. of Iowa, Iowa City, IA (United States)); Choi, Y.H. (Korea Inst. of Nuclear

Safety (KINS), Daeduk-danji Taejon (Korea, Republic of))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1996
Report Number(s): NUREG/CR-6443; BMI--2191
Order Number: TI96012372
Abstract: 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. The following summarizes the objectives and results from both the deterministic and probabilistic studies. The reasons for including each technical aspect being evaluated are given first. Then a table is given that summarizes the relative significance of each technical aspect. In most cases there are both deterministic and probabilistic results. The deterministic analyses were conducted independently of the probabilistic analysis, which offered the opportunity to validate conclusions from each of these studies.

Title: Fracture behavior of circumferentially surface-cracked elbows. Technical report, October 1993--March 1996
Author(s)/Editor(s): Kilinski, T. ; Mohan, R. ; Rudland, D. ; Fleming, M. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Dec 1996
Report Number(s): NUREG/CR-6444; BMI--2192
Order Number: TI97002497
Abstract: This report presents the results from Task 2 of the Second International Piping Integrity Research Group (IPIRG-2) program. The focus of the Task 2 work was directed towards furthering the understanding of the fracture behavior of long-radius elbows. This was accomplished through a combined analytical and experimental program. J-estimation schemes were developed for both axial and circumferential surface cracks in elbows. Large-scale, quasi-static and dynamic, pipe-system, elbow fracture experiments under combined pressure and bending loads were performed on elbows containing an internal surface crack at the extrados. In conjunction with the elbow experiments, material property data were developed for the A106-90 carbon steel and WP304L stainless steel elbow materials investigated. A comparison of the experimental data with the maximum stress predictions using existing straight pipe fracture prediction analysis methods, and elbow fracture prediction methods developed in this program was performed. This analysis was directed at addressing the concerns regarding the validity of using analysis predictions developed for straight pipe to predict the fracture stresses of cracked elbows. Finally, a simplified fitting flaw acceptance criteria incorporating ASME B2 stress indices and straight pipe, circumferential- crack analysis was developed.

Title: The effects of cyclic and dynamic loading on the fracture resistance of nuclear piping steels. Technical report, October 1992--April 1996
Author(s)/Editor(s): Rudland, D.L. ; Brust, F. ; Wilkowski, G.M.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Dec 1996
Report Number(s): NUREG/CR-6440; BMI--2190
Order Number: TI97002498
Abstract: This report presents the results of the material property evaluation efforts performed within Task 3 of the IPIRG-2 Program. Several related investigations were conducted. (1) Quasi-static, cyclic-load compact tension specimen experiments were conducted using parameters similar to those used in IPIRG-1 experiments on 6-inch nominal diameter through-wall-cracked pipes. These experiments were conducted on a TP304 base metal, an A106 Grade B base metal, and their respective submerged-arc welds. The results showed that when using a constant cyclic displacement increment, the compact tension experiments could predict the through-wall-cracked pipe crack initiation toughness, but a different control procedure is needed to reproduce the pipe cyclic crack growth in the compact tension tests. (2) Analyses conducted showed that for 6-inch diameter pipe, the quasi-static, monotonic J-R curve can be used in making cyclic pipe moment predictions; however, sensitivity analyses suggest that the maximum moments decrease slightly from cyclic toughness degradation as the pipe diameter increases. (3) Dynamic stress-strain and compact tension tests were conducted to expand on the existing dynamic database. Results from dynamic moment predictions suggest that the dynamic compact tension J-R and the quasi-static stress-strain curves are the appropriate material properties to use in making dynamic pipe moment predictions.

Title: IPIRG-2 task 1 - pipe system experiments with circumferential cracks in straight-pipe locations. Final report, September 1991--November 1995
Author(s)/Editor(s): Scott, P. ; Olson, R. ; Marschall, C. ; Rudland, D. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1997
Report Number(s): NUREG/CR-6389; BMI--2187
Order Number: TI97003757
Abstract: This report presents the results from Task 1 of the Second International Piping Integrity Research Group (IPIRG-2) program. The IPIRG-2 program is an international group program managed by the US Nuclear Regulatory Commission (US NRC) and funded by a consortium of organizations from 15 nations including: Bulgaria, Canada, Czech Republic, France, Hungary, Italy, Japan, Republic of Korea, Lithuania, Republic of China, Slovak Republic, Sweden, Switzerland,

the United Kingdom, and the United States. The objective of the program was to build on the results of the IPIRG-1 and other related programs by extending the state- of-the-art in pipe fracture technology through the development of data needed to verify engineering methods for assessing the integrity of nuclear power plant piping systems that contain defects. The IPIRG-2 program included five main tasks: Task 1 - Pipe System Experiments with Flaws in Straight Pipe and Welds Task 2 - Fracture of Flawed Fittings Task 3 - Cyclic and Dynamic Load Effects on Fracture Toughness Task 4 - Resolution of Issues From IPIRG-1 and Related Programs Task 5 - Information Exchange Seminars and Workshops, and Program Management. The scope of this report is to present the results from the experiments and analyses associated with Task 1 (Pipe System Experiments with Flaws in Straight Pipe and Welds). The rationale and objectives of this task are discussed after a brief review of experimental data which existed after the IPIRG- 1 program.

Title: Fracture toughness evaluations of TP304 stainless steel pipes
Author(s)/Editor(s): Rudland, D.L. ; Brust, F.W. ; Wilkowski, G.M. (Battelle, Columbus, OH (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1997
Report Number(s): NUREG/CR-6446; BMI--2194
Order Number: TI97004340
Abstract: In the IPIRG-1 program, the J-R curve calculated for a 16-inch nominal diameter, Schedule 100 TP304 stainless steel (DP2-A8) surface-cracked pipe experiment (Experiment 1.3-3) was considerably lower than the quasi-static, monotonic J-R curve calculated from a C(T) specimen (A8-12a). The results from several related investigations conducted to determine the cause of the observed toughness difference are: (1) chemical analyses on sections of Pipe DP2-A8 from several surface-cracked pipe and material property specimen fracture surfaces indicate that there are two distinct heats of material within Pipe DP2-A8 that differ in chemical composition; (2) SEN(T) specimen experimental results indicate that the toughness of a surface-cracked specimen is highly dependent on the depth of the initial crack, in addition, the J-R curves from the SEN(T) specimens closely match the J-R curve from the surface-cracked pipe experiment; (3) C(T) experimental results suggest that there is a large difference in the quasi- static, monotonic toughness between the two heats of DP2-A8, as well as a toughness degradation in the lower toughness heat of material (DP2-A8II) when loaded with a dynamic, cyclic (R = [minus]0.3) loading history.

Title: The Second International Piping Integrity Research Group (IPIRG-2) program. Final report, October 1991--April 1996

Author(s)/Editor(s): Hopper, A. ; Wilowski, G. ; Scott, P. ; Olson, R. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1997
Report Number(s): NUREG/CR-6452; BMI--2195
Order Number: TI97004743
Abstract: The IPIRG-2 program was an international group program managed by the US NRC and funded by organizations from 15 nations. The emphasis of the IPIRG-2 program was the development of data to verify fracture analyses for cracked pipes and fittings subjected to dynamic/cyclic load histories typical of seismic events. The scope included: (1) the study of more complex dynamic/cyclic load histories, i.e., multi-frequency, variable amplitude, simulated seismic excitations, than those considered in the IPIRG-1 program, (2) crack sizes more typical of those considered in Leak-Before-Break (LBB) and in-service flaw evaluations, (3) through-wall-cracked pipe experiments which can be used to validate LBB-type fracture analyses, (4) cracks in and around pipe fittings, such as elbows, and (5) laboratory specimen and separate effect pipe experiments to provide better insight into the effects of dynamic and cyclic load histories. Also undertaken were an uncertainty analysis to identify the issues most important for LBB or in- service flaw evaluations, updating computer codes and databases, the development and conduct of a series of round-robin analyses, and analyst's group meetings to provide a forum for nuclear piping experts from around the world to exchange information on the subject of pipe fracture technology. 17 refs., 104 figs., 41 tabs.

Title: Proceedings of the Seminar on Leak Before Break in Reactor Piping and Vessels
Author(s)/Editor(s): C. Faidy, (Electricit'e de France), Ph. Gilles (Framatome)
Sponsoring Organization: Electricit'e de France, Framatome, Commissariat a l'Energie Atomique, European Community, DGXI-WGCS, Nuclear Electric, International Atomic Energy Agency, OECD-Nuclear Energy Agency, USNuclear Regulatory Commission, French Nuclear Energy Society
Publication Date: April 1997
Report Number(s): NUREG/CP-0155
Abstract: The sixth in a series of international Leak-Before-Break (LBB) Seminars was held at Hotel Sofitel in Lyon, France on October 9 through 11, 1995. The seminar updated international policies and supporting research on LBB. The more than 210 attendees that joined the meeting included representatives from regulatory agencies, electric utility representatives, fabricators of nuclear power plants, research organizations, and academic institutions.
The objective of the seminar was to present the current state of the art in LBB methodology development, validation, and application in an international forum. With particular

emphasis on industrial applications and regulatory policies, the seminar provided an opportunity to compare approaches, experiences, and codifications developed by different countries.
The seminar was organized into four topic areas:
- Status of LBB Applications
- Technical Issues in LBB Methodology
- Complementary Requirements (Leak Detection and Inspection)

- LBB Assessment and Margins.

In addition to the formal sessions where papers were presented by participants from France, Germany, Japan, Korea, Belgium, the United Kingdom, the Czech Republic, Finland, Russia, Sweden, Canada, the Netherlands, and the United States, informal LBB poster sessions were available outside the presentation hall. A keynote address (see Appendix B) by Mr. J. Branchu, Head of the Primary Nuclear Components Division of Framatome, was delivered at the LBB 95 Banquet and summarized the goals and objectives of the seminar. As a result of this seminar, an improved understanding of LBB gained through sharing of different viewpoints from different countries, permits consideration of:
- Simplified pipe support design and possible elimination of loss-of-coolant- accident (LOCA) mechanical consequences for specific cases
- Defense-in-Depth type of applications without support modifications
- Support of safety cases for plants designed without the LOCA hypothesis.

In support of these activities, better estimates of the limits to the LBB approach should follow, as well as an improvement in codifying methodologies.

Title: International Piping Integrity Research Group (IPIRG) Program. Final report
Author(s)/Editor(s): Wilkowski, G. ; Schmidt, R. ; Scott, P. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1997
Report Number(s): NUREG/CR-6233-Vol.4
Order Number: TI97006968
Abstract: This is the final report of the International Piping Integrity Research Group (IPIRG) Program. The IPIRG Program was an international group program managed by the U.S. Nuclear Regulatory Commission and funded by a consortium of organizations from nine nations: Canada, France, Italy, Japan, Sweden, Switzerland, Taiwan, the United Kingdom, and the United States. The program objective was to develop data needed to verify engineering methods for assessing the integrity of circumferentially-cracked nuclear power plant piping. The primary focus was an experimental task that investigated the behavior of circumferentially flawed piping systems subjected to high-rate loadings typical of seismic events. To accomplish these objectives a pipe system fabricated as an expansion loop with over 30 meters of 16-inch diameter pipe and five long radius elbows was constructed. Five dynamic, cyclic, flawed piping

experiments were conducted using this facility. This report: (1) provides background information on leak-before-break and flaw evaluation procedures for piping, (2) summarizes technical results of the program, (3) gives a relatively detailed assessment of the results from the pipe fracture experiments and complementary analyses, and (4) summarizes advances in the state-of-the-art of pipe fracture technology resulting from the IPIRG program.

Title: Stability of cracked pipe under seismic/dynamic displacement-controlled stresses. Subtask 1.2 final report
Author(s)/Editor(s): Kramer, G. ; Veith, P. ; Marschall, C. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1997
Report Number(s): NUREG/CR-6233-Vol.2; BMI--2177
Order Number: TI97006961
Abstract: Results of displacement-controlled pipe fracture experiments, analyses, and material characterization efforts performed within the International Piping Integrity Research Group, IPIRG, Program Subtask 1.2 are discussed. Effects of dynamic versus quasi-static and monotonic versus cyclic loading were evaluated for ductile tearing of two materials, A106 Grade B ferritic steel and TP304 austenitic steel. Twelve through-wall-cracked pipe experiments were conducted on 6-inch diameter Schedule 120 pipe at 288 C (550 F). The results indicated dynamic loading at seismic strain rates marginally increased the load-carrying capacity of austenitic steel. The ferritic steel tested was sensitive to dynamic strain-aging, and consequently, its load-carrying capacity decreased at dynamic strain rates. Two parameters were found to affect the apparent ductile crack growth resistance during cyclic loading, load ratio (R) and incremental plastic displacement that occurs in a cycle. Cyclic (R = 0) loading had minimal effect on ductile tearing for both materials. However, fully reversed loading decreased the load-carrying capacity and toughness for both materials. The incremental plastic displacement can be as important as the load ratio; however, it is harder to quantify from design stress reports. Large plastic displacements will minimize the effect of negative load ratios.

Title: Crack stability in a representative piping system under combined inertial and seismic/dynamic displacement-controlled stresses. Subtask 1.3 final report
Author(s)/Editor(s): Scott, P. ; Olson, R. ; Wilkowski, O.G. ; Marschall, C. ; Schmidt, R.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1997
Report Number(s): NUREG/CR-6233-Vol.3; BMI--2177
Order Number: TI97006967
Abstract: This report presents the

results from Subtask 1.3 of the International Piping Integrity Research Group (IPIRG) program. The objective of Subtask 1.3 is to develop data to assess analysis methodologies for characterizing the fracture behavior of circumferentially cracked pipe in a representative piping system under combined inertial and displacement-controlled stresses. A unique experimental facility was designed and constructed. The piping system evaluated is an expansion loop with over 30 meters of 16-inch diameter Schedule 100 pipe. The experimental facility is equipped with special hardware to ensure system boundary conditions could be appropriately modeled. The test matrix involved one uncracked and five cracked dynamic pipe-system experiments. The uncracked experiment was conducted to evaluate piping system damping and natural frequency characteristics. The cracked-pipe experiments evaluated the fracture behavior, pipe system response, and stability characteristics of five different materials. All cracked-pipe experiments were conducted at PWR conditions. Material characterization efforts provided tensile and fracture toughness properties of the different pipe materials at various strain rates and temperatures. Results from all pipe-system experiments and material characterization efforts are presented. Results of fracture mechanics analyses, dynamic finite element stress analyses, and stability analyses are presented and compared with experimental results.

<u>Title:</u> State-of-the-Art Report on Piping Fracture Mechanics
<u>Author(s)/Editor(s):</u> G.M. Wilkowski, R.J. Olsol, P.M. Scott (Battelle)
<u>Sponsoring Organization:</u> NRC; Washington DC (United States)
<u>Publication Date:</u> January 1998
<u>Report Number(s):</u> NUREG/CR-6540; BMI-2196
<u>Abstract:</u> This report is an in-depth summary of the state-of-the-art in nuclear piping fracture mechanics. It represents the culmination of 20 years of work done primarily in the U.S., but also attempts to include important aspects from other international efforts. Although the focus of this work was for the nuclear industry, the technology is also applicable in many cases to fossil plants, petrochemical/refinery plants, and the oil and gas industry. In compiling this detailed summary report, all of the equations and details of the analysis procedure or experimental results are not necessarily included. Rather, the report describes the important aspects and limitations, tells the reader where he can go for further information, and more importantly, describes the accuracy of the models. Nevertheless, the report still contains over 150 equations and over 400 references. The main sections of this report describe: (1) the evolution of piping fracture mechanics history relative to the developments of the nuclear industry, (2) technical developments in stress analyses, material property aspects, and fracture mechanics analyses, (3) unresolved issues and technically evolving areas, and (4) a summary of

conclusions of major developments to date.

Pressure Vessel Steels

Title: Peer review of the Three Mile Island Unit 2 Vessel Investigation Project metallurgical examinations
Author(s)/Editor(s): Bohl, R.W. ; Gaydos, R.G. ; Vander Voort, G.F. ; Diercks, D.R. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1994
Report Number(s): NUREG/CR-6183; ANL--94/3
Order Number: TI94016723
Abstract: Fifteen samples recovered from the lower head of the Three Mile Island (TMI) Unit 2 nuclear reactor pressure vessel were subjected to detailed metallurgical examinations by the Idaho National Engineering Laboratory (INEL), with supporting work carried out by Argonne National Laboratory (ANL) and several of the European participants. These examinations determined that a portion of the lower head, a so-called elliptical ``hot spot'' measuring [approx]0.8 [times] 1 m, reached temperatures as high as 1100[degrees]C during the accident and cooled from these temperatures at [approx]10--100[degrees]C/min. The remainder of the lower head was found to have remained below the ferrite-toaustenite transformation temperature of 727[degrees]C during the accident. Because of the significance of these results and their importance to the overall analysis of the TMI accident, a panel of three outside peer reviewers, Dr. Robert W. Bohl, Mr. Richard G. Gaydos, and Mr. George F. Vander Voort, was formed to conduct an independent review of the metallurgical analyses. After a thorough review of the previous analyses and examination of photo-micrographs and actual lower head specimens, the panel determined that the conclusions resulting from the INEL study were fundamentally correct. In particular, the panel reaffirmed that four lower head samples attained temperatures as high as 1100[degrees]C, and perhaps as high as 1150--1200[degrees]C in one case, during the accident. They concluded that these samples subsequently cooled at a rate of [approx]50--125[degrees]C/min in the temperature range of 600--400[degrees]C, in good agreement with the original analysis. The reviewers also agreed that the remainder of the lower head samples had not exceeded the ferrite-to-austenite transformation temperature during the accident and suggested several refinements and alternative procedures that could have been employed in the original analysis.

Title: Review of the proposed materials of construction for the SBWR and AP600 advanced reactors
Author(s)/Editor(s): Diercks, D.R. ; Shack, W.J. ; Chung, H.M. ; Kassner,

T.F. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1994
Report Number(s): NUREG/CR-6223; ANL--94/13
Order Number: TI94013716
Abstract: Two advanced light water reactor (LWR) concepts, namely the General Electric Simplified Boiling Water Reactor (SBWR) and the Westinghouse Advanced Passive 600 MWe Reactor (AP600), were reviewed in detail by Argonne National Laboratory. The objectives of these reviews were to (a) evaluate proposed advanced-reactor designs and the materials of construction for the safety systems, (b) identify all aging and environmentally related degradation mechanisms for the materials of construction, and (c) evaluate from the safety viewpoint the suitability of the proposed materials for the design application. Safety-related systems selected for review for these two LWRs included (a) reactor pressure vessel, (b) control rod drive system and reactor internals, (c) coolant pressure boundary, (d) engineered safety systems, (e) steam generators (AP600 only), (f) turbines, and (g) fuel storage and handling system. In addition, the use of cobalt-based alloys in these plants was reviewed. The selected materials for both reactors were generally sound, and no major selection errors were found. It was apparent that considerable thought had been given to the materials selection process, making use of lessons learned from previous LWR experience. The review resulted in the suggestion of alternate an possibly better materials choices in a number of cases, and several potential problem areas have been cited.

Title: Unirradiated material properties of Midland weld WF-70
Author(s)/Editor(s): McCabe, D.E. ; Nanstad, R.K. ; Iskander, S.K. ; Swain, R.L. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Oct 1994
Report Number(s): NUREG/CR-6249; ORNL/TM--12777
Order Number: TI95003010
Abstract: Weld metal, designated WF-70, taken from the nozzle course and beltline welds of the Midland Reactor, Unit 1, has been given a preliminary evaluation using the conventional Charpy V-notch (CVN), drop-weight (DWT), and chemical analyses. These tests indicated essentially identical fracture toughness at both locations, but there was a significant deference in copper content, nominally 0.25% versus 0.40%. Because the objective of this study was to evaluate the before-and-after irradiation properties, these are regarded as different materials. This report summarizes material characterization results and presents the results of fracture mechanics tests on the unirradiated material to establish

baseline transition temperature and J-R curves. Tensile properties were also determined. Five experimental objectives to be accomplished from the testing of irradiated materials were identified. One of the more important objectives is to improve the precision of transition temperature shift and to identify any curve shape changes after irradiation, concentrating on utilizing data from small surveillance capsule size specimens.

Title: CASKS (Computer Analysis of Storage casKS): A microcomputer based analysis system for storage cask design review. User's manual to Version 1b (including program reference)
Author(s)/Editor(s): Chen, T.F. ; Gerhard, M.A. ; Trummer, D.J. ; Johnson, G.L. ; Mok, G.C. (Lawrence Livermore National Lab., CA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1995
Report Number(s): NUREG/CR-6242; UCRL-ID--117418
Order Number: TI95007829
Abstract: CASKS (Computer Analysis of Storage casKS) is a microcomputer-based system of computer programs and databases developed at the Lawrence Livermore National Laboratory (LLNL) for evaluating safety analysis reports on spent-fuel storage casks. The bulk of the complete program and this user's manual are based upon the SCANS (Shipping Cask ANalysis System) program previously developed at LLNL. A number of enhancements and improvements were added to the original SCANS program to meet requirements unique to storage casks. CASKS is an easy-to-use system that calculates global response of storage casks to impact loads, pressure loads and thermal conditions. This provides reviewers with a tool for an independent check on analyses submitted by licensees. CASKS is based on microcomputers compatible with the IBM-PC family of computers. The system is composed of a series of menus, input programs, cask analysis programs, and output display programs. All data is entered through fill-in-the-blank input screens that contain descriptive data requests.

Title: Microstructural characterization of selected AEA/UCSB model FeCuMn alloys
Author(s)/Editor(s): Rice, P.M. ; Stoller, R.E.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1996
Report Number(s): NUREG/CR-6332; ORNL/TM--12980
Order Number: TI96012332
Abstract: A set of 22 model ferritic alloys was purchased as part of a collaborative research program by the AEA Harwell Laboratory and the University of California at Santa Barbara. Nine of these alloys were selected by the Oak Ridge National Laboratory for use in a series of ion irradiation experiments investigating dispersed barrier hardening. These

nine alloys contain varying amounts of copper, manganese, titanium, carbon, and nitrogen. The alloys have been characterized by transmission electron microscopy in the as-received condition to provide a baseline for comparison with the irradiated specimens. A description of the microstructural observations is provided for future reference. This summary focuses on the type and size distributions of the precipitates present; grain size and dislocation measurements are also included.

Title: Proceedings of the IAEA Specialists' Meeting on Cracking in LWR RPV Head Penetrations
Author(s)/Editor(s): C.E. Pugh, S.J. Raney
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington DC (United States)
Publication Date: July 1996
Report Number(s): NUREG/CP-0151; ORNL/TM-13187
Abstract: This report contains 17 papers that were presented in four sessions at the IAEA Specialists' meeting on *Cracking in LWR RPV Head Penetrations* held at ASTM Headquarters in Philadelphia on May 2-3, 1995. The papers are compiled here in the order that presentations were made in the sessions, and they relate to operational observations, inspection techniques, analytical modeling, and regulatory control. The goal of the meeting was to allow international experts to review experience in the field of ensuring adequate performance of reactor pressure vessel (RPV) heads and penetrations. The emphasis was to allow a better understanding of RPV material behavior, to provide guidance supporting reliability and adequate performance, and to assist in defining directions for further invesdgations. The intemational nature of the meeting is illustrated by the fact that papers were presented by researchers from 10 countries. There were technical experts present from other countries who participated in discussions of the results presented. The IAEA issued a Worlcing Material version of the meeting papers (IAEA IWG-LMNPP-95/1), and this present document incorporates the final version of the papers as received from the authors. The final chapter includes conclusions and recommendations.

Title: An Improved Correlation Procedure for Subsize and Full-Size Charpy Impact Specimen Data
Author(s)/Editor(s): M.A. Sokolov, D.J. Alexander (Oak Ridge National Laboratory)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: March 1997
Report Number(s): NUREG/CR-6379; ORNL-6888
Abstract: The possibility of using subsize specimens to monitor the properties of reactor pressure vessel steels is receiving increasing attention for light-water reactor plant life extension. This potential results from the possibility of cutting samples of small volume from the internal

surface of the pressure vessel for determination of the actual properties of the operating pressure vessel. In addition, plant life extension will require supplemental data that cannot be provided by existing surveillance programs. Testing of subsize specimens manufactured from broken halves of previously tested surveillance Charpy specimens offers an attractive means of extending existing surveillance programs. Using subsize Charpy V-notch-type specimens requires the establishment of a specimen geometry that is adequate to obtain a ductile-to-brittle transition curve similar to that obtained from full size specimens, and the development of correlations for transition temperature and upper-shelf energy (USE) level between sub size and full-size specimens. Five different geometries of subsize specimens were selected for testing and evaluation. The specimens were made from several types of pressure vessel steels with a wide range of yield strengths, transition temperatures, and USES. The effects of specimen dimensions, including notch depth, angle, and radius, have been studied. The correlations of transition temperatures determined from different types of subsize specimens and the full-size specimens are presented. A new procedure for transforming data from subsize specimens is developed. The transformed data are in good agreement with data from full-size specimens for materials that have USE levels less than 200 J.

Title: The Characterization of Vicker's Microhardness Indentations and Pile-Up Profiles as a Strain - Hardening Microprobe
Author(s)/Editor(s): C. Santos Jr., G.R. Odette, G.E.Lucas, B. Schroeter, D. Klinginsmith, T. Yamamoto
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: April 1998
Report Number(s): NUREG-1629
Abstract: Microhardness measurements have long been used to examine strength properties and changes in strength properties in metals, for example, as induced by irradiation. Microhardness affords a relatively simple test that can be applied to very small volumes of material. Microhardness is nominally related to the flow stress of the material at a fixed level of plastic strain. Further, the geometry of the pile-up of material around the indentation is related to the strain-hardening behavior of the material; steeper pile-ups correspond to smaller strain hardening rates. In this study the relationship between pile-up profiles and strain hardening is examined using both experimental and analytical methods. Vicker's microhardness tests have been performed on a variety of metal alloys including low alloy, high Cr and austenitic stainless steels. The pile-up topology around the indentations has been quantified using confocal microscopy techniques. In addition, the indentation and pile-up geometry has been simulated using finite element method techniques. These results have been used to develop improved

quantification of the relationship between pile-up geometry and the strain hardening constitutive behavior of the test material.

Radiation Embrittlement

Title: PR-EDB: Power Reactor Embrittlement Data Base, Version 2
Author(s)/Editor(s): Stallmann, F.W. ; Wang, J.A. ; Kam, F.B.K. ; Taylor, B.J. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1994
Report Number(s): NUREG/CR-4816-Rev.2; ORNL/TM--10328/R2
Order Number: TI94006457
Abstract: Investigations of regulatory issues such as vessel integrity over plant life, vessel failure, and sufficiency of current codes Standard Review Plans (SRP's) and Guides for license renewal can be greatly expedited by the use of a well-designed computerized data base. Also, such a data base is essential for the validation of embrittlement prediction models by researchers. The Power Reactor Embrittlement Data Base (PR-EDB) is such a comprehensive collection of data for US commercial nuclear reactors. The current version of the PR-EDB contains the Charpy test data that were irradiated in 252 capsules of 96 reactors and consists of 207 data points for heat-affected-zone (HAZ) materials (98 different HAZ), 227 data points for weld materials (105 different welds), 524 data points for base materials (136 different base materials), including 297 plate data points (85 different plates), 119 forging data points (31) different forging), and 108 correlation monitor materials data points (3 different plates). The data files are given in dBASE format and can be accessed with any computer using the DOS operating system. User-friendly'' utility programs are used to retrieve and select specific data, manipulate data, display data to the screen or printer, and to fit and plot Charpy impact data. The results of several studies investigated are presented in Appendix D.

Title: TR-EDB: Test Reactor Embrittlement Data Base, Version 1
Author(s)/Editor(s): Stallmann, F.W. ; Wang, J.A. ; Kam, F.B.K. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jan 1994
Report Number(s): NUREG/CR-6076; ORNL/TM--12415
Order Number: TI94007600
Abstract: The Test Reactor Embrittlement Data Base (TR-EDB) is a collection of results from irradiation in materials test reactors. It complements the Power Reactor Embrittlement Data Base (PR-EDB), whose data are restricted to the results from the analysis of surveillance capsules in commercial power reactors. The

rationale behind their restriction was the assumption that the results of test reactor experiments may not be applicable to power reactors and could, therefore, be challenged if such data were included. For this very reason the embrittlement predictions in the Reg. Guide 1.99, Rev. 2, were based exclusively on power reactor data. However, test reactor experiments are able to cover a much wider range of materials and irradiation conditions that are needed to explore more fully a variety of models for the prediction of irradiation embrittlement. These data are also needed for the study of effects of annealing for life extension of reactor pressure vessels that are difficult to obtain from surveillance capsule results.

Title: Crack-arrest tests on two irradiated high-copper welds
Author(s)/Editor(s): Iskander, S.K. ; Corwin, W.R. ; Nanstad, R.K. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1994
Report Number(s): NUREG/CR-6139; ORNL/TM--12513
Order Number: TI94007831
Abstract: The objective of the Heavy-Section Steel Irradiation Program Sixth Irradiation Series is to determine the effect of neutron irradiation on the shift and shape of the lower-bound curve to crack-arrest toughness data. Two submerged- arc welds with copper contents of 0.23 and 0.31 wt % were commercially fabricated in 220-mm-thick plate. Crack-arrest specimens fabricated from these welds were irradiated at a nominal temperature of 288[degrees]C to an average fluence of 1.9 [times] 10[sup 19] neutrons/cm[sup 2] (>1 MeV). This is the second report giving the results of the tests on irradiated duplex-type crack-arrest specimens. A previous report gave results of tests on irradiated weld-embrittled-type specimens. Charpy V-notch (CVN) specimens irradiated in the same capsules as the crack-arrest specimens were also tested, and a 41-J transition temperature shift was determined from these specimens. [open quotes]Mean[close quote] curves of the same form as the American Society of Mechanical Engineers (ASME) K[sub Ia] curve were fit to the data with only the [open quotes]reference temperature[close quotes] as a parameter. The shift between the mean curves agrees well with the 41-J transition temperature shift obtained from the CVN specimen tests. Moreover, the four data points resulting from tests on the duplex crack-arrest specimens of the present study did not make a significant change to mean curve fits to either the previously obtained data or all the data combined.

Title: Heavy-Section Steel Irradiation Program. Volume 2, No. 1: Semiannual progress report, October 1990--March 1991
Author(s)/Editor(s): Corwin, W.R. (Oak Ridge National Lab., TN (United

States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1994
Report Number(s): NUREG/CR-5591-Vol.2-No.1; ORNL/TM--11568-Vol.2-No.1
Order Number: TI94016626
Abstract: Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. The RPV is the only key safety-related component of the plant for which a duplicate or redundant backup system does not exist. It is therefore imperative to understand and be able to predict the capabilities and limitations of the integrity inherent in the RPV. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established with its primary goal to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, and in particular the fracture toughness properties, of typical pressure-vessel steels as they relate to light-water reactor pressure-vessel integrity. The HSSI Program is arranged into nine tasks: (1) program management, (2) K[sub ic] curve shift in high-copper welds, (3) K[sub ia] curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K[sub ic] and K[sub ia] curve shifts in low upper-shelf (LUS) weld, (6) irradiation effects in a commercial LUS weld, (7) microstructural analysis of irradiation, (8) in- service aged material evaluations, and (9) correlation monitor materials. During this period, additional analyses on the effects of precleavage stable ductile tearing on the toughness of high-copper welds 72W and 73W demonstrated that the size effects observed in the transition region are not due to substantial differences in ductile tearing behavior. Possible modifications to irradiated duplex crack-arrest specimens were examined to increase the likelihood of their successful testing. Characterization of a second batch of 72W and 73W welds was begun and results of the Charpy V-notch testing is provided. A review of literature on the annealing response of reactor pressure vessel steels was initiated.

Title: Heavy-Section Steel Irradiation Program
Author(s)/Editor(s): Corwin, W.R. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Oct 1994
Report Number(s): NUREG/CR-5591-Vol.2-No.2; ORNL/TM--11568-Vol.2-No.2
Order Number: TI95001892
Abstract: Goal is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, and in particular the fracture toughness properties, of typical pressure vessel stools 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 into 10 tasks: (1) program management, (2) K[sub Ic] curve shift in high-copper welds, (3) K[sub Ia] curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K[sub Ic] and K[sub Ia] curve shifts in low upper-shelf welds, (6) irradiation effects in a commercial low upper-sheer weld, (7) microstructural analysis of irradiation effects, (8) in-service aged material evaluations, (9) correlation monitor materials, and (10) special technical assistance. This report provides an overview of the activities within each of these tasks from April to September 1991.

Title: A Comparison of the Relative Importance of Copper Precipitates and Point Defect Clusters in Reactor Pressure Vessel Embrittlement
Author(s)/Editor(s): R.E. Stoller (Oak Ridge National Laboratory)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: December 1994
Report Number(s): NUREG/CR-6231; ORNL/TM-6811
Abstract: The embrittlement of irradiated reactor pressurevessel (RPV, steels is believed to arise primarily from the hardening of the material due to the formation of extended defects that act to impede dislocation motion. Radiation-induced point defect clusters (PDC) and radiation-enhanced, copper-rich precipitates (CRP) provide two plausible sources of this matrix hardening. These PDC can be of either interstitial or vacancy type and could exist in either two- or three-dimensional morphologies, e.g., small loops, voids, or stacking fault tetrahedra. The formation and evolution of PDC are primarily determined by the displacement damage rate and irradiation temperature. There is experimental evidence that the the distributions of these clusters are also influenced by impurities such as copper. A theoretical model has been developed to investigate the relative importance ofPDC and CRP in RPV embrittlement. The model includes a detailed description of the interstial cluster population; vacancy clustering and copper precipitation are treated in a more approximate fashion. The model has been used to examine a broad range of irradiation and material parameters. The results indicate that there are temperature and displacement rate regimes wherein either CRP or PDC can dominate the material's response to irradiation. Both interstitial and vacancy-type defects contribute to the PDC component, wth their relative importance determined by the specific irradiation conditions. The varying dependencies of the CRP and PDC on temperature and displacement rate indicate that simple data extrapolations could lead to poor predictions of RPV embrittlement.

Title: Heavy-Section Sdteel Irradiation Program: Volume 3, Progress report, October 1991--September 1992
Author(s)/Editor(s): Corwin, W.R. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1995
Report Number(s): NUREG/CR-5591-Vol.3; ORNL/TM--11568-Vol.3
Order Number: TI95007768
Abstract: The primary goal of the Heavy-Section Steel Irradiatic. Program is to provide a thorough, quai.citative assessment of the effects of neutron irradiation on the 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 postirradiation annealing are being examined on a wide range of fracture pronerties. The HSSI Program is arranged into 10 tasks: (1) program management, (2) K[sub Ic] curve shift in high-copper welds, (3) K[sub Ia] curve shift in high- copper welds, (4) irradiation effects on cladding, (5) K[sub Ic] and K[sub Ia] curve shifts in low upper-shelf welds, (6) irradiation ffects in a commercial low upper-shelf weld, (7) microstructural analysis of irradiation effects, (8) in- service aged material evaluations, (9) correlation monitor materials, and (10) special technical assistance. This report provides an overview of the activities within each of these tasks from October 1991 to September 1992.

Title: Heavy-section steel irradiation program. Volume 4, No. 2. Semiannual progress report, April 1993--September 1993
Author(s)/Editor(s): C. win, W.R. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Mar 1995
Report Number(s): NUREG/CR-5591-Vol.4-No.2; ORNL/TM--11568/V4-N2
Order Number: TI95009673
Abstract: Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents which have the potential for major contamination release. The RPV is the only key safety-related component of the plant for which a duplicate or redundant backup system does not exist. In particular, it is vital to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance which occurs during service, since without that radiation damage, it 's virtually impossible to postulate a realistic scenario that would result in RPV failure. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established to provide

a quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 14 tasks: (1) program management, (2) fracture toughness (K[sub 1c]) curve shift in high-copper welds, (3) crack-arrest toughness (K[sub 1a]) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K[sub 1c] and K[sub 1a] curve shifts in low upper-shelf (LUS) welds, (6) annealing effects in LUS welds, (7) irradiation effects in a commercial LUS weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) Japan Power Development Reactor steel examination, (13) technical assistance for Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, and (14) additional requirements for materials.

Title: Heavy-section steel irradiation program. Semiannual progress report, September 1993--March 1994
Author(s)/Editor(s): Corwin, W.R. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-5591-Vol.5-No.1; ORNL/TM--11568/V5-N1
Order Number: TI95010954
Abstract: Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. The RPV is the only component in the primary pressure boundary for which, if it should rupture, the engineering safety systems cannot assure protection from core damage. It is therefore imperative to understand and be able to predict the capabilities and limitations of the integrity inherent in the RPV. In particular, ft is vital to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance that occurs during service. The Heavy-Section Steel (HSS) Irradiation Program has been established; its primary goal is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, and in particular the fracture toughness properties of typical pressure-vessel steels, as they relate to light-water RPV integrity. The program includes the direct continuation of irradiation studies previously conducted within the HSS Technology Program augmented by enhanced examinations of the accompanying microstructural changes.

During this period, the report on the duplex-type crack-arrest specimen tests from Phase 11 of the K[sub 1a] program was issued, and final preparations for testing the large, irradiated crack-arrest specimens from the Italian Committee for Research and Development of Nuclear Energy and Alternative Energies were completed. Tests on undersize Charpy V-notch (CVN) energy specimens in the irradiated and annealed weld 73W were completed. The results are described in detail in a draft NUREG report. In addition, the ORNL investigation of the embrittlement of the High Flux Isotope RPV indicated that an unusually large ratio of the high-energy gamma- ray flux to fast-neutron flux is most likely responsible for the apparently accelerated embrittlement.

Title: Heavy-Section Steel Irradiation Program. Volume 5, No. 2, Progress report, April 1994--September 1994.
Author(s)/Editor(s): Corwin, W.R. (Oak Ridge National Lab., TN (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1995
Report Number(s):
NUREG/CR-5591-Vol.5-No.2;
ORNL/TM--11568-Vol.5-No.2
Order Number: TI95016159
Abstract: The Heavy-Section Steel Irradiation (HSSI) Program has been established with its primary goal to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and the fracture toughness properties of typical pressure-vessel steels as they relate to light- water RPV integrity. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation 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 curve shift in high-copper weldments (Series 5 and 6), (3) K[sub 1c] and K[sub 1a] curve shifts in low upper-shelf (LUS) welds (Series 8), (4) irradiation effects in a commercial LUS weld (Series 10), (5) irradiation effects on weld heat-affected zone and plate materials (Series 11), (6) annealing effects in LUS welds (Series 9), (7) microstructural and microfracture analysis of irradiation effects, (8) in-service irradiated and aged material evaluations, (9) Japan Power Development Reactor (JPDR) steel examination, (10) fracture toughness curve shift method, (11) special technical assistance, (12) technical assistance for Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, (13) correlation monitor materials, and (14) test reactor coordination. Progress on each task is reported.

Title: Heavy-section steel irradiation program. Progress report, October 1994-- March 1995
Author(s)/Editor(s): Corwin, W.R. (Oak

Ridge National Lab., TN (United States))
Publication Date: Oct 1995
Report Number(s): NUREG/CR-5591-Vol.6-No.1
Order Number: DE96002233
Abstract: This task was established to supply and coordinate irradiation services needed by NRC contractors other than ORNL. These services include the design and assembly of irradiation capsules as well as arranging for their exposure, disassembly, and return of specimens. During this period, the final design of the facility and specimen baskets was determined through an iterative process involving the designers and thermal analysts. The resulting design should permit the irradiation of all test specimens to within 5[degrees]C of their desired temperature. Detailing of all parts is ongoing and should be completed during the next reporting period. Procurement of the facility will also be initiated during the next review period.

Title: Radiation Effects of Reactor Pressure Vessel Supports
Author(s)/Editor(s): R.E. Johnson, and R.E. Lipinski
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: May 1996
Report Number(s): NUREG-1509
Abstract: The purpose of this report is to present the findings from the work done in accordance with the Task Action Plan developed to resolve the Nuclear Regulatory Commission (NRC) Generic Safety Issue No. 15, (GSI-15), "Radiation Effects On Reactor Pressure Vessel Supports." GSI-15 was established to evaluate the potential for low-temperature, low-flux level neutron irradiation to embrittle reactor pressure vessel (RPV) supports to the point of compromising plant safety. An evaluation of surveillance samples from the high flux isotope reactor (HFIR) at the Oak Ridge National Laboratory (ORNL) had suggested that some materials used for RPV supports in pressurized-water reactors could exhibit higher than expected embrittlement rates. However, further tests designed to evaluate the applicability of the HFIR data to reactor RPV supports under operating conditions led to the conclusion that RPV supports could be evaluated using traditional methods. It was found that the unique HFIR radiation environment allowed the gamma radiation to contribute significantly to the embrittlement. The shielding provided by the thick steel RPV shell ensures that degradation of RPV supports from gamma irradiation is improbable or minimal.

The findings reported herein were used, in part, as the basis for technical resolution of the issue.

Title: Analysis of the irradiation data for A302B and A533B correlation monitor materials
Author(s)/Editor(s): Wang, J.A.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC

(United States)
Publication Date: Apr 1996
Report Number(s): NUREG/CR-6413; ORNL/TM--13133
Order Number: TI96010415
Abstract: 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[sub 30]) is considered as the primary measure of radiation embrittlement in this report. The hyperbolic tangent fitting model and uncertainty of the fitting parameters for Charpy impact tests are presented in this report. For the surveillance CMM data, the transition temperature shifts at 30 ft-lb ([Delta]T[sub 30]) generally follow the predictions provided by Revision 2 of Regulatory Guide 1.99 (R.G. 1.99). Difference in capsule temperatures is a likely explanation for large deviations from R.G. 1.99 predictions. Deviations from the R.G. 1.99 predictions are correlated to similar deviations for the accompanying materials in the same capsules, but large random fluctuations prevent precise quantitative determination. Significant scatter is noted in the surveillance data, some of which may be attributed to variations from one specimen set to another, or inherent in Charpy V-notch testing. The major contributions to the uncertainty of the R.G. 1.99 prediction model, and the overall data scatter are from mechanical test results, chemical analysis, irradiation environments, fluence evaluation, and inhomogeneous material properties. Thus in order to improve the prediction model, control of the above-mentioned error sources needs to be improved. 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.

Title: Heavy Section Steel Irradiation Program Progress Report for April - September 1995
Author(s)/Editor(s): W.R. Corwin
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: August 1996
Report Number(s): NUREG/CR-5591 Vol 6 No 2/ORNL/TM-11568/V6&N2
Abstract: Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents which have the potential for major contamination release. The RPV is the only key safety-related component of the plant for which a duplicate or redundant backup system does not exist. It is therefore imperative to understand and be able to predict the capabilities and limitations of the integrity inherent in the RPV. In particular, it is vital to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance which occurs

during Service, since without that radiation damage, H is virtually impossible to postulate a realistic scenario that would result in RPV failure.

For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established with its primary goal to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels as they relate to light-water RPV integrity. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation 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 curve shift in high-copper weldments (Series 5 and 6), (3) KIC and Kl, curve shifts in low upper-shelf (LUS) welds (Series 8), (4) irradiation effects in a commercial LUS weld (Series 10), (5) irradiation effects on weld heat-affected zone (HAZ) and plate materials (Series 11), (6) annealing effects in LUS welds (Series 9), (7) microstructural and microfracture analysis of irradiation effects, (8) in-service irradiated and aged material evaluations, (9) Japan Power Development Reactor (JPDR) steel examination, (10) fracture toughness curve shift method, (11) special technical assistance, (12) technical assistance for Joint Coordination Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, (13) correlation monitor materials, and (14) test reactor coordination.

During this period, results of testing the Italian crack-arrest specimens were analyzed and a draft NUREG report prepared. A test plan was developed for irradiation of HSSI weld 73W to a high fluence [5 x 10'9 neutrons/cm2 (> 1 MeV)] to determine whether the KJC curve shape change observed in the Fifth Series is exacerbated. The fabrication of the third of the three trial LUS scoping welds to identify possible materials for studies on KIC shifts in LUS materials was completed. Data from fracture mechanics testing of specimens of the irradiated LUS Midland Weld WF-70 from both scoping capsules and the first large capsule [exposed to 0.5 and 1.0 x 1019 neutrons/cm2 (~1 MeV), respectively] was completed and the results reported. Precracked Charpy specimens of beltline weld were also tested in the unirradiated and irradiated condHions and showed an irradiation-induced fracture toughness shift very close to that indicated by the compact specimens. The second large capsule that was shipped to Oak Ridge National Laboratory (ORNL) has been disassembled, and the specimens are awaiting testing in the hot cell. Arrangements were made with Yankee Atomic Electric Company for the procurement of two A 302 grade B plates, identified for examination of the effects of neutron irradiation on the fracture toughness of the HAZ of welds of plate materials typical of those used in fabricating older RPVs. Microstructural characterizations of long-term (~ 100,000 h) thermally aged, neutron-irradiated, and annealed surveillance materials were completed. An atom-probe field-ion microscopy characterization of a simple thermally aged model alloy was performed to investigate its suitability as a model for commercial RPV steels. The validity of the low-load "nanoindentation" technique to monitor strength changes was established. A comparison of model predictions and available data seemed to confirm that the lead factors commonly employed in commercial surveillance programs should have negligible impact on the validity of the data obtained. Modification of the computer numerically controlled machining center for irradiated materials continued with the completion of the drawings, new cables and table, machine enclosure, fittings, and a floor tub for installation inside the hot cell. Tensile and Charpy V-notch impact tests of type 308 stainless steel weld metals aged at 343°C for up to 50,000 h showed that aging had little effect on the tensile properties, but did result in embrittlement as shown by the impact testing. The baseline testing for the cross comparison of the effects of the different tups used in U.S. and Japanese Charpy impact machines was performed to provide a basis for understanding any differences that might later arise from jointly testing the JPDR materials. Available fracture toughness databases were analyzed for plates, forgings, and welds, and H was shown that, on average, the fracture toughness shifts generally exceeded the Charpy 41-J shifts by about 20, 50, and 8% for plates, forgings, and welds, respectively. Overall, the fracture toughness shifts exceed the CVN shifts by about 14%, similar to the results reported previously from analysis of HSSI Program data. Evaluation of the precracked cylindrical tensile specimen continued with a report produced for ORNL by SRI International and AEA Technology, Harwell, United Kingdom, regarding their test results. A detailed plan was developed for removal of material from the pressurized-water RPV, the Pressure Vessel Research User's Facility located at the Oak Ridge Gaseous Diffusion Plant (K-25 site). The remaining correlation monitor materials were moved from the storage area at the Y-12 Plant and placed into the HSSI storage facility at the ORNL site. Two blocks of Heavy-Section Steel Technology (HSST) plate 03 were sent to the Hanjung America Corp. for use as correlation monitor materials in Units 3 and 4 of the Ulchin Nuclear Power Plant in Korea. Most of the engineering drawings for the irradiation facility and specimen baskets for University of California, Santa Barbara, irradiations were completed, and procurement and fabrication of selected portions of the facility were initiated.

Title: Heavy Section Steel Irradiation Program Semiannual Progress Report for October 1995-March 1996
Author(s)/Editor(s): W.R. Corwin

Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: April 1997
Report Number(s): NUREG/CR-5591 Vol7 No 1/ORNL/TM-11568V7&N1

Abstract: Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents which have the potential for major contamination release. The RPV is the only key safety-related component of the plant for which a duplicate or redundant backup system does not exist. It is therefore imperative to understand and be able to predict the capabilities and limitations of the integrity inherent in the RPV. In particular, it is vital to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance which occurs during service, since without that radiation damage, it is virtually impossible to postulate a realistic scenario that would result in RPV failure.

For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established with its primary goal to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels as they relate to light-water RPV integrity. Effects of specimen size; material chemistry; product form and microstructure, irradiation fluence, flux temperature, and spectrum; and postirradiation 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 curve shift in high-copper weldments (Series 5 and 6), (3) KIc and K., curve shifts in low upper-shelf (LUS) welds (Series 8), (4) irradiation effects in a commercial LUS weld (Series 10), (5) irradiation effects on weld heat-affected zone (HAZ) and plate materials (Series 11), (6) annealing effects in LUS welds (Series 9), (7) microstructural and microfracture analysis of irradiation effects, (8) in-service irradiated and aged material evaluations, (9) Japan Power Development Reactor (JPDR) steel examination, (10) fracture toughness curve shift method, (11) special technical assistance, (12) technical assistance for Joint Coordinabng Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, (13) correlation monitor materials, and (14) test reactor coordination.

During this period, a draft NUREG report was prepared describing the results of testing the irradiated Italian crack arrest specimens. The fabrication of the third of the three trial LUS scoping welds was completed. Charpy V-notch (CVN) specimens from the trial weld which was fabricated with HSSI weld 73W weld wire and Linde 80 flux to identify possible materials for studies on K,e shifts in LUS materials have been tested showing a relatively small reduction in upper-shelf energy from 136 J for the original weld (fabricated with Linde 124 flux) to 121

J for the biaxial weld made with Linde 80 flux. Data from fracture mechanics testing of specimens of the irradiated LUS Midland Weld WF-70 from both scoping capsules and both large capsules [exposed to 0.5 and 1.0 x 10~9 neutrons/cm2 (> 1 MeV), respectively were analyzed. An A 302 grade B plate was procured from Yankee Atomic Electric Company for examination of the effects of neutron irradiation on the fracture toughness of the HAZ of welds of plate materials typical of those used in fabricating older RPVs. Detailed planning was performed and work begun to examine grain boundary segregation of phosphorous and resultant intergranular fracture of steel heat treated to give large, prior austenite grains such as would be found in the HAZ. The annealing and testing of specimens irradiated within capsule 10.06 was completed and planning of the specimen complement for the first reirradiation capsule begun. Two irradiation, annealing, and reirradiation facilities; data acquisition and control instrumentation; and the associated reusable temperature verification capsule have been fabricated and assembled. The analysis of solute effects in ion-irradiated model alloys was largely completed, indicating a strong effect of copper and a strong copper-manganese interaction. The effect of the interstitial solutes nitrogen and carbon was more modest. A detailed comparison of neutron flux and spectral effects on tensile properties at 50 to 60°C was completed. The results of molecular dynamics cascade simulations were used to develop effective defect production cross sections for relevant reactor neutron spectra. Modification of the computer numerically controlled (CNC) machining center continued with all drawings completed and new cables and table, machine enclosure, fittings, and a floor tub for installation inside the hot cell procured. Tensile and CVN impact tests of type 308 stainless steel weld metals aged at 343°C for up to 50,000 h showed that aging had little effect on the tensile properties but did result in embrittlement as shown by the impact testing. Planning was initiated at Oak Ridge National Laboratory (ORNL) for the machining of the JPDR vessel trepans, and recent studies conducted in Japan have shown that the through-wall attenuation is somewhat greater than would be predicted by the attenuation formula in *Regulatory Guide 1.99*. As part of the evaluation of the database of Charpy impact and fracture toughness data for RPV steels, instrumented CVN and dynamic precracked CVN tests were analyzed for potential use in estimating various toughness parameters. The end of unstable crack propagation indicated by the load-displacement record was compared to the drop-weight nil-ductility-transition temperature (NDT) and crack-arrest toughness tests showing results that are encouraging with regard to the potential use of the instrumented CVN test record to provide a reasonable estimate of the NDT temperature and the crack-arrest toughness for RPV steels. Preparations continued for the characterization of the beltline weld from a pressurized-water reactor pressure vessel, the Pressure Vessel Research User's Facility (PVRUF), located at the Oak Ridge K-25 plant site. Material will be removed for experimental projects within the HSSI and Heavy-Section Steel Technology (HSST) programs at ORNL, as well as the Pacific Northwest National Laboratory Nondestructive Evaluation Program. The CVN and round tensile specimens of two Russian weld metals irradiated in HSSI capsule 10.06 were returned to ORNL where the capsule was disassembled and preparations made for specimens to be tested in both the irradiated and thermally annealed conditions. The remainder of the engineering drawings for the irradiation facility and specimen baskets for University of California, Santa Barbara, irradiations were completed, and procurement and fabrication of selected portions of the facility were continued.

Title: Results of charpy V-notch impact testing of structural steel specimens irradiated at [approximately]30[degrees]C to 1 x 10[sup 16] neutrons/cm[sup 2] in a commercial reactor cavity
Author(s)/Editor(s): Iskander, S.K. ; Stoller, R.E.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1997
Report Number(s): NUREG/CR-6399; ORNL--6886
Order Number: TI97005912
Abstract: A capsule containing Charpy V-notch (CVN) and mini-tensile specimens was irradiated at [approximately] 30[degrees]C ([approximately] 85[degrees]F) in the cavity of a commercial nuclear power plant to a fluence of 1 x 10[sup 16] neutrons/cm[sup 2] (> 1MeV). The capsule included six CVN impact specimens of archival High Flux Isotope Reactor A212 grade B ferritic steel and five CVN impact specimens of a well-studied A36 structural steel. This irradiation was part of the ongoing study of neutron-induced damage effects at the low temperature and flux experienced by reactor supports. The plant operators shut down the plant before the planned exposure was reached. The exposure of these specimens produced no significant irradiation-induced embrittlement. Of interest were the data on unirradiated specimens in the L-T orientation machined from a single plate of A36 structural steel, which is the same specification for the structural steel used in some reactor supports. The average CVN energy of five unirradiated specimens obtained from one region of the plate and tested at room temperature was [approximately] 99 J, while the energy of 11 unirradiated specimens from other locations of the same plate was 45 J, a difference of [approximately] 220%. The CVN impact energies for all 18 specimens ranged from a low of 32 J to a high of 111 J. Moreover, it appears that the University of Kansas CVN impact energy data of the unirradiated specimens at

the 100-J level are shifted toward higher temperatures by about 20 K. The results were an example of the extent of scatter possible in CVN impact testing. Generic values for the CVN impact energy of A36 should be used with caution in critical applications.

Title: Heavy-section steel irradiation program. Progress report, April 1996--September 1996
Author(s)/Editor(s): Corwin, W.R.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Sep 1997
Report Number(s):
NUREG/CR-5591-Vol.7-No.2;
ORNL/TM--11568-Vol.7-No.2
Order Number: TI98000039
Abstract: The Heavy-Section Steel Irradiation Program was established to quantitatively assess the effects of neutron irradiation on the material behavior of typical reactor pressure vessel (RPV) steels. During this period, fracture mechanics testing of specimens of the irradiated low upper shelf (LUS) weld were completed and analyses performed. Heat treatment of five RPV plate materials was initiated to examine phosphorus segregation effects on the fracture toughness of the heat affected zone of welds. Initial results show that all five materials exhibited very large prior austenite grain sizes as a consequence of the initial heat treatment. Irradiated and annealed specimens of LUS weld material were tested and analyzed. Four sets of Charpy V-notch (CVN) specimens were aged at various temperatures and tested to examine the reason for overrecovery of upper shelf energy that has been observed. Molecular dynamics cascade simulations were extended to 40 keV and have provided information representative of most of the fast neutron spectrum. Investigations of the correlation between microstructural changes and hardness changes in irradiated model alloys was also completed. Preliminary planning for test specimen machining for the Japan Power Development Reactor was completed. A database of Charpy impact and fracture toughness data for RPV materials that have been tested in the unirradiated and irradiated conditions is being assembled and analyzed. Weld metal appears to have similar CVN and fracture toughness transition temperature shifts, whereas the fracture toughness shifts are greater than CVN shifts for base metals. Draft subcontractor reports on precracked cylindrical tensile specimens were completed, reviewed, and are being revised. Testing on precracked CVN specimens, both quasi-static and dynamic, was evaluated. Additionally, testing of compact specimens was initiated as an experimental comparison of constraint limitations. 16 figs., 2 tabs.

Title: Results of Crack-Arrest Tests on Irradiated A508 Class 3 Steel
Author(s)/Editor(s): S.K. Iskander, P.P. Milella, A. Pini
Sponsoring Organization: NRC; Washington DC (United States)

Publication Date: February 1998
Report Number(s): NUREG/CR-6447
Abstract: Ten crack-arrest toughness values for irradiated specimens of A 508 class 3 forging steel have been obtained. The tests were performed according to the American Society for Testing and Materials (ASTM) Standard Test Method for Determining Plane-Strain Crack-Arrest Fracture Toughness, KIa, of Ferritic Steels, E 1221-88. None of these values are strictly "valid, in all five ASTM E 1221-88 validity criteria. However, they are useful when compared to unirradiated crack-arrest specimen toughness values since they show the small (averaging approximately 10°C) shifts in the mean and lower-bound crack-arrest toughness curves. This confirms that a low copper content in ASTM A 508 class 3 forging material can be expected to result in small shifts of the transition toughness curve. The shifts due to neutron irradiation of the lower bound and mean toughness curves are approximately the same as the Charpy V-notch (CVN) 41-J temperature shift. The nine crack-arrest specimens were irradiated at temperatures varying from 243 to 280°C, and to a fluence varying from 1.7 to 2.7 x 1019 neutrons/cm2 (> 1 MeV). The test results were "normalized" to reference values that correspond to those of CVN specimens irradiated at 284°C to a fluence of 3.2 x 1019 neutrons/cm2 (> 1 MeV) in the same capsule as the crack-arrest specimens. This adjustment resulted in a shift to lower temperatures of all the data, and in particular moved two data points that appeared to lie close to or lower than the American Society of Mechanical Engineers K.. curve to positions that seemed more reasonable with respect to the remaining data. A special fixture was designed, fabricated, and successfully used in the testing. For reasons explained in the text, special blocks to receive the Oak Ridge National Laboratory clip gage were designed, and greater-than-standard crack-mouth opening displacements measured were accounted for.

Title: Heavy-Section Steel Irradiation Program Semiannual Progress Report for October 1996 - March 1997
Author(s)/Editor(s): T.M. Rosseel
Sponsoring Organization: NRC: Washington DC (United States)
Publication Date: February 1998
Report Number(s): NUREG/CR-5591 Vol8 No 1/ORNL/TM-11568Vol8 No 1
Abstract: Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. Because the RPV is the only key safety-related component of the plant for which a redundant backup system does not exist, it is imperative to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance that occurs during service. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established. Its primary goal is to provide a thorough,

quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels as they relate to light-water RPV integrity. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into seven tasks: (1) program management, (2) irradiation effects in engineering materials, (3) annealing, (4) microstructural analysis of radiation effects, (5) in-service irradiated and aged material evaluations, (6) fracture toughness curve shift method, (7) special technical assistance, and (8) foreign research interactions. The work is performed by the Oak Ridge National Laboratory.

Steam Generator Tube Integrity

Title: Evaluation of sampling plans for in-service inspection of steam generator tubes
Author(s)/Editor(s): Kurtz, R.J. ; Heasler, P.G. ; Baird, D.B. (Pacific Northwest Lab., Richland, WA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1994
Report Number(s): NUREG/CR-5161-Vol.2; PNL--6462-Vol.2
Order Number: TI94007939
Abstract: This report summarizes the results of three previous studies to evaluate and compare the effectiveness of sampling plans for steam generator tube inspections. An analytical evaluation and Monte Carlo simulation techniques were the methods used to evaluate sampling plan performance. To test the performance of candidate sampling plans under a variety of conditions, ranges of inspection system reliability were considered along with different distributions of tube degradation. Results from the eddy current reliability studies performed with the retired-from-service Surry 2A steam generator were utilized to guide the selection of appropriate probability of detection and flaw sizing models for use in the analysis. Different distributions of tube degradation were selected to span the range of conditions that might exist in operating steam generators. The principal means of evaluating sampling performance was to determine the effectiveness of the sampling plan for detecting and plugging defective tubes. A summary of key results from the eddy current reliability studies is presented. The analytical and Monte Carlo simulation analyses are discussed along with a synopsis of key results and conclusions.

Title: Applications of a new magnetic monitoring technique to in situ evaluation of fatigue damage in ferrous

components
Author(s)/Editor(s): Jiles, D.C. ; Biner, S.B. ; Govindaraju, M.R. ; Chen, Z.J. (Iowa State Univ. of Science and Technology, Ames, IA (United States). Center for Nondestructive Evaluation)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jun 1994
Report Number(s): NUREG/GR-0013
Order Number: TI94015174
Abstract: This project consisted of research into the use of magnetic inspection methods for the estimation of fatigue life of nuclear pressure vessel steel. Estimating the mechanical and magnetic properties of ferromagnetic materials are closely interrelated, therefore, measurements of magnetic properties could be used to monitor the evolution of fatigue damage in specimens subjected to cyclic loading. Results have shown that is possible to monitor the fatigue damage nondestructively by magnetic techniques. For example, in load-controlled high- cycle fatigue tests, it has been found that the plastic strain and coercivity accumulate logarithmically during the fatigue process. Thus a quantitative relationship between coercivity and the number of fatigue cycles could be established based on two empirical coefficients, which can be determined from the test conditions and material properties. Also it was found that prediction of the onset of fatigue failure in steels was possible under certain conditions. In strain-controlled low cycle fatigue, critical changes in Barkhausen emissions, coercivity and hysteresis loss occurred in the last ten to twenty percent of fatigue life.

Title: Piping inspection round robin
Author(s)/Editor(s): Heasler, P.G. ; Doctor, S.R. (Pacific Northwest National Lab., Richland, WA (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1996
Report Number(s): NUREG/CR-5068; PNNL--10475
Order Number: TI96009923
Abstract: The piping inspection round robin was conducted in 1981 at the Pacific Northwest National Laboratory (PNNL) to quantify the capability of ultrasonics for inservice inspection and to address some aspects of reliability for this type of nondestructive evaluation (NDE). The 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 11 Code requirements. Three different types of materials 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.

Title: Performance demonstration tests for eddy current inspection of steam generator tubing
Author(s)/Editor(s): Kurtz, R.J. ; Heasler, P.G. ; Anderson, C.M.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: May 1996
Report Number(s): NUREG/CR-6227; PNNL--9433
Order Number: TI96011419
Abstract: This report describes the methodology and results for development of performance demonstration tests for eddy current (ET) inspection of steam generator tubes. Statistical test design principles were used to develop the performance demonstration tests. Thresholds on ET system inspection performance were selected to ensure that field inspection systems would have a high probability of detecting and and correctly sizing tube degradation. The technical basis for the ET system performance thresholds is presented in detail. Statistical test design calculations for probability of detection and flaw sizing tests are described. A recommended performance demonstration test based on the design calculations is presented. A computer program for grading the probability of detection portion of the performance demonstration test is given.

Title: Data analysis for steam generator tubing samples
Author(s)/Editor(s): Dodd, C.V.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1996
Report Number(s): NUREG/CR-6455
Order Number: TI96013809
Abstract: The objective of the Improved Eddy-Current ISI for Steam Generators program is to upgrade and validate eddy-current inspections, including probes, instrumentation, and data processing techniques for inservice inspection of new, used, and repaired steam generator tubes; to improve defect detection, classification and characterization as affected by diameter and thickness variations, denting, probe wobble, tube sheet, tube supports, copper and sludge deposits, even when defect types and other variables occur in combination; to transfer this advanced technology to NRC's mobile NDE laboratory and staff. This report provides a description of the application of advanced eddy-current neural network analysis methods for the detection and evaluation of common steam generator tubing flaws including axial and circumferential outer-diameter stress-corrosion cracking and intergranular attack. The report describes the training of the neural networks on tubing samples with known defects and the subsequent evaluation results for

unknown samples. Evaluations were done in the presence of artifacts. Computer programs are given in the appendix.

Title: Computer programs for the acquisition and analysis of eddy-current array probe data
Author(s)/Editor(s): Pate, J.R. ; Dodd, C.V.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1996
Report Number(s): NUREG/CR-6163; ORNL/TM--13212
Order Number: TI96013016
Abstract: Objective of the Improved Eddy-Curent ISI (in-service inspection) for Steam Generators Tubing program is to upgrade and validate eddy-current inspections, including probes, instrumentation, and data processing techniques for ISI of new, used, and repaired steam generator tubes; to improve defect detection, classification and characterization as affected by diameter and thickness variations, denting, probe wobble, tube sheet, tube supports, copper and sludge deposits, even when defect types and other variables occur in combination; to transfer this advanced technology to NRC's mobile NDE laboratory and staff. This report documents computer programs that were developed for acquisition of eddy-current data from specially designed 16-coil array probes. Complete code as well as instructions for use are provided.

Title: Evaluation and field validation of Eddy-Current array probes for steam generator tube inspection
Author(s)/Editor(s): Dodd, C.V. ; Pate, J.R.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Jul 1996
Report Number(s): NUREG/CR-6357
Order Number: TI96013374
Abstract: The objective of the Improved Eddy-Current ISI for Steam Generator Tubing program is to upgrade and validate eddy-current inspections, including probes, instrumentation, and data processing techniques for inservice inspection of new, used, and repaired steam generator tubes; to improve defect detection, classification, and characterization as affected by diameter and thickness variations, denting, probe wobble, tube sheet, tube supports, copper and sludge deposits, even when defect types and other variables occur in combination; to transfer this advanced technology to NRC's mobile NDE laboratory and staff. This report describes the design of specialized high-speed 16-coil eddy-current array probes. Both pancake and reflection coils are considered. Test results from inspections using the probes in working steam generators are given. Computer programs developed for probe calculations are also supplied.

Title: Proceedings of the CNRA/CSNI workshop on steam generator tube integrity in nuclear power plants

Author(s)/Editor(s): Diercks, D.R. (Argonne National Lab., IL (United States))
Publication Date: Feb 1997
Report Number(s): NUREG/CP-0154; ANL--96/14; NEA/CNRA/R--(96)1; CONF-9510423--
Order Number: TI97004312
Abstract: Steam generator (SG) tubes in pressurized water reactor plants are exposed to various types of degradation processes, among which stress corrosion cracking in particular has been observed. To be able to evaluate the safety importance of such cracking of SG-tubes one has to have a good and empirically founded knowledge about the scope and the size of the cracks as well as the rate of their continuous growth. The basis of experience is to a large extent constituted of the annually performed SG-inspections and crack sizing procedures. On the basis of this experience one can estimate the distribution of existing crack lengths, and modify this distribution with regard to maintenance (plugging) and the predicted rate of crack propagation. Finally, one can calculate the rupture probability of SG-tubes as a function of a given critical crack length. On account of the Swedish Nuclear Power Inspectorate an introductory study has been performed in order to get a survey of what has been done elsewhere in this field. The study resulted in a proposal of a computerizable model to be able to estimate the distribution of true cracks, to modify this distribution due to the crack growth and to compute the probability of tube rupture. The model has now been implemented in a compute code, called STAC (STatistical Analysis of Cracks). This paper is aimed to give a brief outline of the model to facilitate the understanding of the possibilities and limitations associated with the model.

Title: Steam generator tube integrity program. Semiannual report, August 1995-- March 1996
Author(s)/Editor(s): Diercks, D.R. ; Bakhtiari, S. ; Chopra, O.K. (and others)
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1997
Report Number(s): NUREG/CR-6511-Vol.1; ANL--96/17
Order Number: TI97005975
Abstract: This report summarizes work performed by Argonne National Laboratory on the Steam Generator Tube Integrity Program from the inception of that program in August 1995 through March 1996. The program is divided into five tasks, namely (1) Assessment of Inspection Reliability, (2) Research on ISI (in-service- inspection) Technology, (3) Research on Degradation Modes and Integrity, (4) Development of Methodology and Technical Requirements for Current and Emerging Regulatory Issues, and (5) Program Management. Under Task 1, progress is reported on the preparation of and evaluation of nondestructive evaluation (NDE) techniques for inspecting a mock-up steam generator for round-robin testing, the development of better ways

to correlate burst pressure and leak rate with eddy current (EC) signals, the inspection of sleeved tubes, workshop and training activities, and the evaluation of emerging NDE technology. Under Task 2, results are reported on closed-form solutions and finite element electromagnetic modeling of EC probe response for various probe designs and flaw characteristics. Under Task 3, facilities are being designed and built for the production of cracked tubes under aggressive and near-prototypical conditions and for the testing of flawed and unflawed tubes under normal operating, accident, and severe accident conditions. In addition, crack behavior and stability are being modeled to provide guidance on test facility design, to develop an improved understanding of the expected rupture behavior of tubes with circumferential cracks, and to predict the behavior of flawed and unflawed tubes under severe accident conditions. Task 4 is concerned with the cracking and failure of tubes that have been repaired by sleeving, and with a review of literature on this subject.

Title: Steam Generator Tube Integrity Program:
Annual Report August 1995-September 1996
Author(s)/Editor(s): . R. Diercks, S. Bakhtiari, K. E. Kasza, D. S. Kupperman, S. Ma.lumdar, J. Y. Park, and W. J. Shack (Argonne National Laboratory (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington DC (United States)
Publication Date: February 1998
Report Number(s): NUREG/CR-6511 Vol2, ANL-97/3
Abstract: This report summarizes work performed by Argonne National Laboratory on the Steam Generator Tube Integrity Program from the inception of the program in August 1995 through September 1996. The program is divided into five tasks: (1) Assessment of Inspection Reliability, (2) Research on ISI (in-service-inspection) Technology, (3) Research on Degradation Modes and Integrity, (4) Tube Removals from Steam Generators, and (5) Program Management. Under Task 1, progress is reported on the preparation of facilities and evaluation of nondestructive evaluation techniques for inspecting a mock-up steam generator for round-robin testing, the development of better correlations between failure pressure and leak rate with eddy current (EC) signals, the inspection of sleeved tubes, workshop and training activities, and the evaluation of emerging NDE technology. Under Task 2, results are reported on closed-form solutions and finite-element electromagnetic modeling of EC probe responses for various probe designs and flaw characteristics. Under Task 3, facilities are being designed and built for the production of cracked tubes under aggressive and near-prototypical conditions and for the testing of flawed and unflawed tubes under normal operating, accident, and severe accident conditions. In addition, crack behavior and stability are being modeled to provide guidance for test

facility design, develop an improved understanding of the expected failure behavior of tubes with circumferential cracks, and predict the behavior of flawed and unflawed tubes under severe accident conditions. Task 4 is concerned with the acquisition of tubes and tube sections from retired steam generators for use in the other research tasks. Progress on the acquisition of tubes from the Salem and McGuire 1 nuclear plants is reported.

Title: Failure Behavior of Internally Pressurized Flawed and Unflawed Steam Generator Tubing at High Temperatures - Experiments and Comparison with Model Predictions
Author(s)/Editor(s): S. Majumdar, W.J. Shack, D.R. Diercks, K. Mruk, J. Franklin, and L. Knoblich
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: March 1998
Report Number(s): NUREG/CR-6575
Abstract: This report summarizes experimental work performed at Argonne National Laboratory on the failure of internally pressurized steam generator tubing at high temperatures (~700°C). A model was developed for predicting failure of flawed and unflawed steam generator tubes under internal pressure and temperature histories postulated to occur during severe accidents. The model was validated by failure tests on specimens with part-through-wall axial and circumferential flaws of various lengths and depths, conducted under various constant and ramped internal pressure and temperature conditions. The failure temperatures predicted by the model for two temperature and pressure histories, calculated for severe accidents initiated by a station blackout, agree very well with tests performed on both flawed and unflawed specimens.

Title: The Role of Time-Dependent Deformation in Intergranular Crack Initiation of Alloy 600 Steam Generator Tubing Material
Author(s)/Editor(s): G.S. Was, K. Lian
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: March 1998
Report Number( ): NUREG/GR-0016
Abstract: Intergranular stress corrosion cracking (IGSCC) of two commercial alloy 600 conditions (600LT, 600HT) and controlled purity Ni-18Cr-9Pe alloys (CDMA, CDTT) were investigated using constant extension rate tensile (CERT) tests in primary water (0.01M LiOH + 0.01M H3BO3) with 1 bar hydrogen overpressure at 360°C and 320°C. Heat treatments produced two types of microstructures in both commercial and controlled-purity alloys: one dominated by grain boundary carbides (600HT and CDTT) and one dominated by intragranular carbides (600LT and CDMA). CERT tests were conducted over a range of strain rates and at two temperatures with interruptions at specific strains to determine the crack depth distributions. Results show that in all samples, IGSCC was the dominant failure mode. For both the commercial alloy and

the controlled-purity alloys, the microstructure with grain boundary carbides showed delayed crack initiation and shallower crack depths than did the intragranular carbide microstructure under all experimental conditions. This data indicates that a grain boundary carbide microstructure is more resistant to IGSCC than an intragranular carbide microstructure. Observations support both the f~lm rupture/slip dissolution mechanism and enhanced localized plasticity. The advantage of these results over previous studies is that the different carbide distributions were obtained in the same commercial alloy using different heat treatments, and in the other case, in nearly identical controlled-purity alloys. Therefore, observations of the effects of carbide distribution on IGSCC can more confidently be attributed to the carbide distribution *alone* rather than other potentially significant differences in microstructure or composition. Crack growth rates increased with increasing strain rate according to a power law relation with a strain rate exponent between 0.4 and 0.64. However, crack growth rate measured in m/unit strain decreased with increasing strain rate indicating an effect of either the environment or creep. The temperature dependence of the crack growth rate was consistent with the literature.

Thermal Aging

Title: Tensile-property characterization of thermally aged cast stainless steels
Author(s)/Editor(s): Michaud, W.F. ; Toben, P.T. ; Soppet, W.K. ; Chopra, O.K. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Feb 1994
Report Number(s): NUREG/CR-6142; ANL--93/35
Order Number: TI94007118
Abstract: The effect of thermal aging on tensile properties of cast stainless steels during service in light water reactors has been evaluated. Tensile data for several experimental and commercial heats of cast stainless steels are presented. Thermal aging increases the tensile strength of these steels. The high-C Mo-bearing CF-8M steels are more susceptible to thermal aging than the Mo- free CF-3 or CF-8 steels. A procedure and correlations are presented for predicting the change in tensile flow and yield stresses and engineering stress- vs.-strain curve of cast stainless steel as a function of time and temperature of service. The tensile properties of aged cast stainless steel are estimated from known material information, i.e., chemical composition and the initial tensile strength of the steel. The correlations described in this report may be used for assessing thermal embrittlement of cast stainless steel components.

Title: Estimation of fracture

toughness of cast stainless steels during thermal aging in LWR systems-revision 1
Author(s)/Editor(s): Chopra, O.K.
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Aug 1994
Report Number(s): NUREG/CR-4513-Rev.1; ANL--93/22-Rev.1
Order Number: TI94018628
Abstract: This report presents a revision of the procedure and correlations presented earlier in NUREG/CR-4513, ANL-90/42 (June 1991) for predicting the change in mechanical properties of cast stainless steel components due to thermal aging during service in light water reactors at 280-330[degrees]C (535-625[degrees]F). The correlations presented in this report are based on an expanded data base and have been optimized with mechanical-property data on cast stainless steels aged up to [approx]58,000 h at 290-350[degrees]C (554- 633[degrees]F). The fracture toughness J-R curve, tensile stress, and Charpy- impact energy of aged cast stainless steels are estimated from known material information. Mechanical properties of a specific cast stainless steel are estimated from the extent and kinetics of thermal embrittlement. Embrittlement of cast stainless steels is characterized in terms of room-temperature Charpy-impact energy. Charpy-impact energy as a function of time and temperature of reactor service is estimated from the kinetics of thermal embrittlement, which are also determined from the chemical composition. The initial impact energy of the unaged steel is required for these estimations. Initial tensile flow stress is needed for estimating the flow stress of the aged material. The fracture toughness J-R curve for the material is then obtained by correlating room-temperature Charpy-impact energy with fracture toughness parameters. The values of J[sub IC] are determined from the estimated J-R curve and flow stress. A common [open quotes]predicted lower-bound[close quotes] J-R curve for cast stainless steels of unknown chemical composition is also defined for a given grade of steel, range of ferrite content, and temperature. Examples of estimating mechanical properties of cast stainless steel components during reactor service are presented.

Title: Mechanical properties of thermally aged cast stainless steels from Shippingport reactor components
Author(s)/Editor(s): Chopra, O.K. ; Shack, W.J. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: Apr 1995
Report Number(s): NUREG/CR-6275; ANL--94/37
Order Number: TI95012084
Abstract: 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 shutoff valves, and two pump volutes. The actual time-at-temperature for the materials was [approximately]13 y at [approximately]281 C (538 F) for the hot-leg components and [approximately]264 C (507 F) for the cold- leg components. Baseline mechanical properties for as-cast material were determined from tests on either recovery-annealed material, i.e., annealed for 1 h at 550 C and then water quenched, 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[sub IC] of the materials. The kinetics of thermal embrittlement and degree of embrittlement at saturation, i.e., the minimum impact energy achieved after long-term aging, 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 crossover-leg elbows (CF-8M steel) after service of [approximately] 15 y and the KRB reactor pump cover plate (CF-8) after [approximately] 8 y of service.

Title: Effects of thermal aging on fracture toughness and Charpy-impact strength of stainless steel pipe welds
Author(s)/Editor(s): Gavenda, D.J. ; Michaud, W.F. ; Galvin, T.M. ; Burke, W.F. ; Chopra, O.K. (Argonne National Lab., IL (United States))
Sponsoring Organization: NRC; Nuclear Regulatory Commission, Washington, DC (United States)
Publication Date: May 1996
Report Number(s): NUREG/CR-6428; ANL--95/47
Order Number: TI96011018
Abstract: Degradation of fracture toughness, tensile, and Charpy-impact properties of Type 304 and 304/308 SS pipe welds due to thermal aging was studied at room temperature and 290 C. Thermal aging of SS welds results in moderate decreases in charpy-impact strength and fracture toughness. Upper-shelf energy decreased by 50-80 J/cm[sup 2]. Decrease in fracture toughness J-R curve or J[sub IC] is relatively small. Thermal aging had no or little effect on tensile strength of the welds. Fracture properties of SS welds are controlled by the distribution and morphology of second-phase particles. Failure occurs by 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 fracture properties of the welds. Differences in fracture

resistance of the welds arise from differences in the density and size of inclusions. 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.

Title: Effects of Thermal Aging and Neutron Irradiation on the Mechanical Properties of Three-Wire Stainless Steel Weld Overlay Cladding
Author(s)/Editor(s): F.M. Haggag, R.K. Nanstad
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: May 1997
Report Number(s): NUREG/CR-6363/ORNL/TM-13047
Abstract: Thermal aging of three-wire series-arc stainless steel weld overlay cladding at 288°C for 1605 h resulted in an appreciable decrease (16%) in the Charny V-notch (CVN) upper-shelf energy (USE), but the effect on the 41-transition temperature shift was very small (3°C). The combined effect of aging and neutron irradiation at 288°C to a fluence of 5 x 10~9 neutrons/cm2 (> 1 MeV) was a 22% reduction in the USE and a 29°C shift in the 41-transition temperature. The effect of thermal aging on tensile properties was very small. However, the combined effect of irradiation and aging was an increase in the yield strength (6 to 34% at test temperatures from 288 to -125°C) but no apparent change in ultimate tensile strength or total elongation. Neutron irradiation reduced the initiation fracture toughness (J..) much more man did thermal aging alone. Irradiation slightly decreased the tearing modulus, but no reduction was caused by thermal aging alone. Other results from tensile, CVN, and fracture toughness specimens showed that the effects of thermal aging at 288 or 343°C for 20,000 h each were very small and similar to those at 288°C for 1605 h. The effects of long-term thermal exposure time (50,000 h and greater) at 288°C will be investigated as the specimens become available in 1996 and beyond.

Title: Influence of Long-Term Thermal Aging on the Microstructural Evolution of Nuclear Reactor Pressure Vessel Materials
Author(s)/Editor(s): P. Pareige, K.F. Russell, R.E. Stoller, M.K. Miller (Oak Ridge National Laboratory)
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: March 1998
Report Number(s): NUREG/CR-6537;ORNL/TM-13406
Abstract: Atom probe field ion microscopy (APFIM) investigations of the microstructure of unaged (as-fabricated) and long-term thermally aged (~100,000 h at 280°C) surveillance materials from commercial reactor pressure vessel steels were performed. This combination of materials and conditions permitted the investigation of potential thermal-aging effects. This microstructural study focused on the quantification of the compositions of the matrix and carbides. The APFIM

results indicate that there was no significant microstructural evolution after a long-term thermal exposure in weld, plate, or forging materials. The matrix depletion of copper that was observed in weld materials was consistent with the copper concentration in the matrix after the stress-relief heat treatment. The compositions of cementite carbides aged for 100,000 h were compared with the Thermocalc™ prediction. The APFIM comparisons of materials under these conditions are consistent with the measured change in mechanical properties such as the Charpy transition temperature.

Underwater Welding

Title: Underwater Welding of Highly Irradiated Boiling Water Reactor In-Vessel Components
Author(s)/Editor(s): L. Lund
Sponsoring Organization: NRC; Washington DC (United States)
Publication Date: November 1997
Report Number(s): NUREG-1616
Abstract: In February 1997, the U. S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research (RES), initiated a literature review to assess the state of underwater welding technology. In particular, the objective of this literature review was to evaluate the viability of underwater welding in-vessel components of boiling water reactor (BWR) in-vessel components, especially those components fabricated from stainless steels that are subjected to high neutron fluences. This assessment was requested because of the recent increased level of activity in the commercial nuclear industry to address generic issues concerning the reactor vessel and internals, especially those issues related to repair options. This literature review revealed a preponderance of general information about underwater welding technology, as a result of the active research in this field sponsored by the U. S. Navy and offshore oil and gas industry concerns. However, the literature search yielded only a limited amount of information about underwater welding of components in low-fluence areas of BWR in-vessel environments, and no information at all concerning underwater welding experiences in high-fluence environments.

Research reported by the staff of the U. S. Department of Energy (DOE) Savannah River Site and researchers from the DOE fusion reactor program proved more fruitful. This research documented relevant experience concerning welding of stainless steel materials in air environments exposed to high neutron fluences. It also addressed problems with welding highly irradiated materials, and primarily attributed those problems to helium-induced cracking in the material. (Helium is produced from the neutron irradiation of boron, an impurity, and nickel.) The researchers found that the amount of helium-induced cracking could be controlled, or even eliminated, by reducing the heat input

into the weld and applying a compressive stress perpendicular to the weld path.

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11. ABSTRACT *(200 words or less)*

Since 1965, the Materials Engineering Branch, Division of Engineering, of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, and its predecessors dating back to the Atomic Energy Commission (AEC), have sponsored research programs concerning the integrity of the primary system pressure boundary of light-water reactors. The components of concern in these research programs have included the reactor pressure vessel (RPV), steam generators, and the piping. These research programs have covered a broad range of topics, including fracture mechanics analysis and experimental work for RPV and piping applications, inspection method development and qualification, and evaluation of irradiation effects on RPV steels.

This report provides as complete a listing as practical of formal technical reports submitted to the NRC by the investigators working on these research programs. This listing includes topical, final, and progress reports and is divided by topic area. In many cases, a report will cover several topics (such as in the case of progress reports of multi-faceted programs) but is listed under only one topic. Therefore, in searching for reports on a specific topic, other related topic areas should be checked also. The separate volumes of this report cover the following periods:

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