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Heavy-Section Steel Irradiation Program

Semiannual Progress Report for October 1995 – March 1996

Prepared by
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Prepared for
U.S. Nuclear Regulatory Commission

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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) K_{Ic} and K_{Ia} 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 Coordinating 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_{Ic} 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 trial 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^{19} neutrons/cm^2 (> 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.

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Acknowledgments

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Preface

The primary goal of the Heavy-Section Steel Irradiation (HSSI) Program 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 reactor pressure vessel (RPV) integrity. The program includes studies of the effects of irradiation on the degradation of mechanical and fracture properties of vessel materials augmented by enhanced examinations and modeling of the accompanying microstructural changes. 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. Results from the HSSI studies will be incorporated into codes and standards directly applicable to resolving major regulatory issues which involve RPV irradiation embrittlement such as pressurized-thermal shock, operating pressure-temperature limits, low-temperature overpressurization, and the specialized problems associated with low upper-shelf welds.

This HSSI Program progress report covers work performed from October 1995 to March 1996. The work performed by Oak Ridge National Laboratory (ORNL) is managed by the Metals and Ceramics (M&C) Division of ORNL. Major tasks at ORNL are carried out by the M&C, Computing Physics and Engineering, and Engineering Technology Divisions.

Previous HSSI Progress Reports in this series are:

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Some of the series of irradiation studies conducted within the HSSI Program were begun under the Heavy-Section Steel Technology (HSST) Program prior to the separation of the two programs in 1989. Previous HSST Program progress reports contain much information on the irradiation assessments being continued by the HSSI Program as well as earlier related studies. The HSST Program progress reports issued before formation of the HSSI Program are also tabulated here as a convenience to the reader.

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ORNL-4315
ORNL-4377
ORNL-4463
ORNL-4512
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Summary

1. Program Management

The Heavy-Section Steel Irradiation (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_{Ic} and K_{Ia} 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 Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, (13) correlation monitor materials, and (14) test reactor coordination. Report chapters correspond to the tasks. The work is performed by the Oak Ridge National Laboratory (ORNL). During the report period, six program briefings, reviews, or presentations were made by the HSSI staff during program reviews and visits with Nuclear Regulatory Commission (NRC) staff or others. Four technical papers, one letter report, and two foreign trip reports were published. In addition, 14 technical presentations were made.

2. Fracture Toughness Shifts in High-Copper Weldments (Series 5 and 6)

The objective of this task is to develop data addressing the current method of shifting the American Society of Mechanical Engineers (ASME) fracture toughness (K_{Ic}, K_{Ia}, and K_{IR}) curves to account for irradiation embrittlement in high-copper welds. The specific activities to be performed in this task are (1) the continuation of Phase 2 of the Fifth Irradiation Series and (2) the completion of the Sixth Irradiation Series, including the testing of the nine irradiated Italian crack-arrest specimens. The results of testing the Italian crack-arrest specimens have been analyzed, a draft NUREG report has been reviewed, and the report is being finalized prior to submission to the NRC for printing. As part of this task, irradiation of HSSI weld 73W to a high fluence [5×10^{19} n/cm^2 (> 1 MeV)] will be performed to determine whether the K_{Jc} curve shape change observed in the Fifth Series is exacerbated. A test plan has been developed for this task, but all work associated with Phase 2 has been placed on hold pending a review of funding. The design and fabrication of the temperature and dosimetry verification capsules are being performed under this task, but, for purposes of continuity, their progress will be reported under Task 6, where the design of the new irradiation facilities and capsules is performed.

3. Fracture Toughness Curve Shifts in Low Upper-Shelf Welds (Series 8)

This task examines the fracture toughness curve shifts and changes in shape for irradiated welds with low Charpy V-notch (CVN) upper-shelf energy (USE). This task was specifically designed to address questions raised by the Advisory Committee for Reactor Safeguards concerning the shape of the K_{Ic} curve for irradiated welds with a low USE. In particular, it will clarify whether the high concentration of inclusions in low-USE welds results in a transition relationship and behavior significantly different from high-USE welds. The information developed under this task will augment information obtained from other HSSI tasks performed on two high-USE weldments under the Fifth and Sixth Irradiation Series and on a commercial low USE under the Tenth Irradiation Series. The results will provide an expanded basis for accounting for irradiation-induced embrittlement in reactor pressure vessel materials.

To provide material for this and for the annealing task (HSSI Series 9), three trial low-USE welds were ordered from ABB-Combustion Engineering (ABB-CE), Chattanooga, Tennessee, who had also fabricated the welds for the Fifth and Sixth Series. Chemical analyses and mechanical properties for the welds have been reported in previous semiannual reports. During this reporting period, CVN specimens from the trial weld that was fabricated with HSSI weld 73W weld wire and Linde 80 flux have been tested and compared with the CVN data from HSSI weld 73W which was fabricated with Linde 124 flux. The results show a relatively small reduction in USE from 136 J for the original weld to 121 J for the trial weld made with Linde 80 flux. Two additional trial welds will be tested during the next reporting period.

4. Irradiation Effects in a Commercial Low Upper-Shelf Weld (Series 10)

The purpose of the Tenth Irradiation Series was to evaluate the before-and-after irradiation fracture toughness properties of commercially produced WF-70 weld metal. The material has been obtained from Unit 1 of the Midland Reactor of Consumers Power, Midland, Michigan. This vessel became available for test sampling and evaluation when Consumers Power aborted plans to operate the facility. Weld metal WF-70 was used in the girth welds examined; this designation indicates that a specific lot of Linde 80 weld flux was used that produces low CVN upper-shelf toughness. Low upper-shelf welds and weld metal WF-70, in particular, have been a source of concern for several currently operating nuclear power production facilities. The beltline weld of the Midland vessel was sampled completely around the girth, and the Tenth Irradiation Series received seven segments approximately 1 m long (40 in.). The nozzle course weld was similarly sampled, but this project received only two of the available segments. These two segments were spaced about 180° apart.

Tests to establish the transition temperature of the irradiated weld metal have been ongoing during this reporting period. Irradiation exposures have involved two scoping capsules, 10.01 and 10.02, that have received a neutron fluence of 0.5 x 10^{19} n/cm^2 (> 1 MeV), and two large capsules, 10.05 and 10.06, that have received a neutron fluence of 1 x 10^{19} n/cm^2 (> 1 MeV). All were irradiated at 288°C (550°F). The scope of test methods applied covers tensile properties, chemistry surveys, drop-weight (DWT) nil-ductility transition (NDT), CVN transition curves, and fracture mechanics testing for K_{Jc} transition temperature curves. Because a difference in the copper content was found between the nozzle course and beltline WF-70 welds, the two were analyzed separately for postirradiation properties. The best data have been developed on the beltline weld metal, since there was far more of that material available. The test program was designed to satisfy the need for two approaches of establishing the transition temperature. One approach is the *ASME Boiler and Pressure Vessel Code*, Sects. III and XI; the other is the master curve method. The CVN, DWT, and fracture toughness results were used to compare the two methods, and the comparisons are reported here.

5. Thermal Embrittlement Potential and Irradiation Effects in Reactor Pressure Vessel Weld Heat-Affected Zones (Series 11)

The original purpose of this task was to examine the effects of neutron irradiation on the fracture toughness (ductile and brittle) of the HAZ of welds of A 302 grade B (A302B) plate materials typical of those used in fabricating older reactor pressure vessels (RPVs). The initial plate material of emphasis will be A 302 grade B steel, not the A302B modified with nickel additions. A plate of A302B has been received from Yankee Atomic Electric Company for use in this irradiation task. Recently, however, the objectives of this task have been augmented to evaluate the potential for temper embrittlement in RPV steels. The initial task will be to duplicate an experiment reported by AEA Technology, Harwell, United Kingdom, in which they reported significant grain boundary segregation of phosphorous and resultant intergranular fracture of steel heat treated to give large, prior austenite grains. The task will use five commercial RPV steels, A302B, A 533 grade B class 1, two heats of A 302B (modified), and A 508 class 2. A second task will evaluate the potential for local brittle zone (LBZ) development in multipass submerged-arc welds made in RPV joints. The third task will be to select two materials of highest interest resulting from the two preceding tasks and produce a simulated commercial submerged-arc weld to (1) determine if there are LBZs, (2) if so, demonstrate the significance of these zones to structural integrity performance of RPVs, and (3) determine if irradiation and thermal annealing will promote temper embrittlement of LBZs.

6. Annealing Effects in Low Upper-Shelf Welds (Series 9)

The purpose of the Ninth Irradiation Series is to evaluate the correlation between fracture toughness and CVN impact energy during irradiation, annealing, and reirradiation (IAR). The annealing and testing of specimens irradiated within capsule 10.06 has been completed. Planning of the specimen complement for the first reirradiation capsule is under way. It will contain specimens that were previously irradiated and annealed. Two IAR facilities have been fabricated, assembled, leak-checked, and cleaned. They are ready for the final electrical connections and the heatup tests to be performed by the Instrumentation and Controls Division of ORNL. The reusable temperature verification capsule has been fabricated, assembled, leak-tested, and cleaned. It is also awaiting delivery for electrical lead terminations and testing. It is currently planned for the IAR facilities to undergo a 1-week annealing test at ORNL before they are delivered to the University of Michigan Ford Nuclear Reactor. The dosimeters and the dosimetry capsule have all been completed and are awaiting installation during the first in-reactor tests currently scheduled to be performed in late September 1996. The design of a reusable capsule capable of reirradiating previously irradiated and annealed CVN and compact tension specimens is also progressing. The data acquisition and control instrumentation for the first two IAR facilities is complete and is awaiting completion of the IAR facilities and temperature test capsule for checkout and control algorithm development.

7. Microstructural and Microfracture Analysis of Irradiation Effects

The overall long-term goal of this task is to develop a physically based model which can be used to predict irradiation-induced embrittlement in reactor vessel steels over the full range of their service conditions. The model should be tethered soundly on the microstructural level by results from advanced microstructural analysis techniques and constrained at the macroscopic level to produce predictions consistent with the large array of macroscopic embrittlement measurements that are available.

The work involving the analysis of solute effects in ion-irradiated model alloys was largely completed during this period. The bulk of this report is focused on these results, in which a strong effect of copper and a strong copper-manganese interaction were observed. The effect of the interstitial solutes nitrogen and carbon was more modest. The results will be described in detail in a topical report to be prepared during the next reporting period. Additional highlights included the completion of a detailed comparison of neutron flux and spectral effects on tensile properties at 50 to 60°C and the use of the results of molecular dynamics cascade simulations to develop effective defect production cross sections for relevant reactor neutron spectra.

8. In-Service Irradiated and Aged Material Evaluations

The objective of this task is to provide a direct assessment of actual material properties in irradiated components of nuclear reactors, including the effects of irradiation and aging. Four activities are currently in progress: (1) establishing a machining capability for contaminated or activated materials by completing procurement and installation of a computer-based milling machine in a hot cell, (2) machining and testing specimens from cladding materials removed from the Gundremmingen reactor to establish their fracture properties, (3) preparing an interpretive report on the effects of neutron irradiation on cladding, and (4) continuing the evaluation of long-term aging at low temperatures of austenitic structural stainless steel weld metal.

Modification of the computer numerically controlled (CNC) machining center is in progress. Several items have been completed, including drawings, new cables and table, machine enclosure, fittings, and a floor tub for installation inside the hot cell. Additionally, a new saw was purchased for slicing specimens into suitable sizes for machining on the CNC machine. 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. The embrittlement continues to increase with increasing aging time. Aging of three-wire cladding at 288°C for 50,000 h and greater is continuing to better quantify the effects of long-term thermal aging on cladding.

9. Evaluation of Steel from the JPDR Pressure Vessel

There is a need to validate the results of irradiation effects research by the examination of material taken directly from the wall of a pressure vessel which has been irradiated during normal service. This task has been included with the HSSI Program to provide just such an evaluation on material from the wall of the pressure vessel from the JPDR. During this reporting period, planning was initiated at ORNL for the machining of the JPDR vessel trepans in conjunction with the completion of the installation of the CNC to be used for hot-cell fabrication of material irradiated in service (see Task 8). Recent studies conducted at the Japan Atomic Energy Research Institute have shown that the through-wall attenuation is somewhat greater than would be predicted by the attenuation formula in *Regulatory Guide 1.99*.

10. Fracture Toughness Curve Shift Method

The purpose of this task is to examine the technical basis for the currently accepted methods for shifting fracture toughness curves to account for irradiation damage and to work through national codes and standards bodies to revise those methods, if a change is warranted. Specific activities under this task include: (1) collection and statistical analysis of pertinent fracture toughness data to assess the shift and potential change in shape of the fracture toughness curves due to neutron irradiation, thermal aging, or both; (2) evaluation of methods for indexing fracture toughness curves to values that can be deduced from material surveillance programs required under the *Code of Federal Regulations*, Appendix H; (3) participation in the pertinent ASME Section XI, American Society for Testing and Materials (ASTM) E-8, and ASTM E-10 committees; (4) interaction with other researchers in the national and international technical community addressing similar problems; and (5) frequent interactions and detailed technical meetings with the NRC staff.

During this reporting period, a database of Charpy impact and fracture toughness data for RPV steels which have been tested in the unirradiated and irradiated conditions is being assembled. Once the database is assembled, the raw data will be analyzed in a consistent manner, and the radiation-induced transition temperature shifts for Charpy and fracture toughness data will be compared. As part of this evaluation, instrumented CVN and dynamic precracked CVN tests are being analyzed for potential use in estimation of various toughness parameters from testing of small specimens in impact. The end of unstable crack propagation indicated by the load-displacement record has been analyzed and a preliminary comparison made with DWT-NDT temperature and crack-arrest toughness test results. These comparisons have been made for a number of steels tested within the HSSI Program, and the preliminary results 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.

11. Special Technical Assistance

This task has been included with the HSSI Program to provide a vehicle by which to conduct and monitor short-term, high-priority subtasks and provide technical expertise and assistance in the review of national codes and standards that may be referenced in NRC regulations or guides related to nuclear reactor components. This task currently addresses two major areas: (1) providing technical expertise and assistance in the review of national codes and standards and (2) experimental evaluations of test specimens and practices and material properties. The following activities occurred during this reporting period.

Evaluation of the precracked cylindrical tensile specimen is continuing. The report by SRI International has been prepared as a NUREG/ORNL-Sub report and will be published during the next reporting period. A preliminary report prepared by AEA Technology, Harwell, United Kingdom, regarding their test results, has been reviewed by ORNL and comments will be discussed with the authors prior to final publication. 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 (PNNL) Nondestructive Evaluation Program. Various options are under consideration for removal of material from the PVRUF, and coordination between the HSSI, HSST, and PNNL staffs continues.

12. Technical Assistance for JCCCNRS Working Groups 3 and 12

The purpose of this task is to provide technical support for the efforts of the U.S.-Russian JCCCNRS Working Group 3 on radiation embrittlement and Working Group 12 on aging. Specific activities under this task are: (1) supply of materials and preparation of test specimens for collaborative IAR studies to be conducted in Russia, (2) irradiation of Russian specimens within the United States, and (3) preparation for and participation in Working Groups 3 and 12 meetings.

The CVN and round tensile specimens of two Russian weld metals irradiated in HSSI capsule 10.06 were returned to ORNL. The capsule has been disassembled, the specimens identified, and the specimens of Russian steels have been transferred from the disassembly hot cells to the testing hot cells. Some of the specimens will be tested in the irradiated condition while the remainder will be thermally annealed and tested. Testing is anticipated to be completed prior to the end of August 1996, depending on funding and scheduling of the next JCCCNRS Working Group 3 meeting. The original schedule for the Working Group meetings was changed, and the meeting is now tentatively scheduled for September 1996 in Moscow, Russia.

13. Correlation Monitor Materials

This is a task that has been established with the explicit purpose of ensuring the continued availability of the pedigreed and extremely well-characterized material now required for inclusion in all additional and future surveillance capsules in commercial light-water reactors. Having recognized that the only remaining materials qualified for use as a correlation monitor in reactor surveillance capsules are the pieces remaining from the early HSST plates 01, 02, and 03, this task will provide for cataloging, archiving, and distributing the material on behalf of the NRC. During this reporting period, the transfer of the residual correlation monitor material found at Y-12 to ORNL was completed, and archival storage of the correlation monitor material was maintained.

14. Test Reactor Irradiation Coordination

The purpose of this task is to provide the support required 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. Currently, the University of California, Santa Barbara, is the only other NRC contractor for whom irradiations are to be conducted. During this reporting period, the engineering drawings for the remainder of the facility were completed. Procurement and fabrication of selected portions of the facility were continued.

Heavy-Section Steel Irradiation Program Semiannual Progress Report for October 1995 through March 1996*, †

W. R. Corwin

1. Program Management

The Heavy-Section Steel Irradiation (HSSI) Program, a major safety program sponsored by the Nuclear Regulatory Commission (NRC) at Oak Ridge National Laboratory (ORNL), is an engineering research activity devoted to providing a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, particularly the fracture toughness properties, of typical pressure-vessel steels as they relate to light-water reactor pressure vessel (RPV) integrity. The program centers on experimental assessments of irradiation-induced embrittlement augmented by detailed examinations and modeling of the accompanying microstructural changes. 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. Fracture toughness (K_{Ic} and J_{Ic}), crack-arrest toughness (K_{Ia}), ductile tearing resistance (dJ/da), Charpy V-notch (CVN) impact energy, drop-weight (DWT) nil-ductility transition (NDT), and tensile properties are included. Models based on observations of radiation-induced microstructural changes using the atom-probe field-ion microscope (APFIM) and the high-resolution transmission electron microscope (TEM) are being developed to provide a firm basis for extrapolating the measured changes in fracture properties to wide ranges of irradiation conditions. The principal materials examined within the HSSI Program are high-copper welds because their postirradiation properties frequently limit the continued safe operation of commercial RPVs. In addition, a limited effort will focus on stainless steel weld-overlay cladding typical of that used on the inner surfaces of RPVs because its postirradiation fracture properties have the potential for strongly affecting the extension of small surface flaws during overcooling transients.

Results from the HSSI studies will be integrated to aid in resolving major regulatory issues facing the NRC. Those issues involve RPV irradiation embrittlement such as pressurized-thermal shock, operating pressure-temperature limits, low-temperature overpressurization, and the specialized problems associated with low upper-shelf (LUS) welds. Together, the results of these studies also provide guidance and bases for evaluating the overall aging behavior of light-water RPVs.

The program is coordinated with those of other government agencies and the manufacturing and utility sectors of the nuclear power industry in the United States and abroad. The overall objective is the quantification of irradiation effects for safety assessments of regulatory agencies, professional code-writing bodies, and the nuclear power industry.

The program is broken down into 1 task responsible for overall program management and 13 technical tasks: (1) program management, (2) fracture toughness curve shift in high-copper weldments (Series 5 and 6), (3) K_{Ic} and K_{Ia} curve shifts in 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

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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.

During this period, six program briefings, reviews, or presentations were made by the HSSI staff during program reviews and visits with NRC staff or others. Four technical papers,[1-4] one letter report,[5] and two foreign trip reports[6,7] were published. In addition, 14 technical presentations were made.[8-21]

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3. S. K. Iskander, M. A. Sokolov, and R. K. Nanstad, "A Perspective on Thermal Annealing of Reactor Pressure Vessel Materials from the Viewpoint of Experimental Results," pp. 479-491 in *ICONE-4, Proceedings of the ASME-JSME 4th International Conference on Nuclear Engineering*, Vol. 1, Pt. A, ASME International, New York, 1996.†

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10. D. E. McCabe, "Development of a Standard Practice for Transition Region Fracture Toughness Testing," presented at the Westinghouse/NRC/ORNL Information Exchange Meeting, Oak Ridge, Tennessee, Oct. 10-11, 1995.

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*Available for purchase from National Technical Information Service, Springfield, VA 22161.
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20. R. E. Stoller, "The Impact of Point Defect Cluster Mobility in Radiation Damage Models," presented at the Symposium on Radiation Materials Science in Technology, 1995 Annual Meeting of TMS, Anaheim, California, Feb. 4-8, 1995.

21. P. M. Rice and R. E. Stoller, "The Effect of Solutes on Defect Distributions in Ion Irradiated Model LWR Pressure Vessel Steels," presented at the Symposium on Radiation Materials Science in Technology, 1995 Annual Meeting of TMS, Anaheim, California, Feb. 4-8, 1995.

2. Fracture Toughness Shifts in High-Copper Weldments (Series 5 and 6)

S. K. Iskander

The objective of this task is to develop data addressing the current method of shifting the American Society of Mechanical Engineers (ASME) fracture toughness (K_{Ic}, K_{Ia}, and K_{IR}) curves to account for irradiation embrittlement in high-copper welds. The specific activities to be performed in this task are (1) the continuation of Phase 2 of the Fifth Irradiation Series and (2) the completion of the Sixth Irradiation Series, including testing nine irradiated Italian crack-arrest specimens. The continuation of Phase 2 of the Fifth Irradiation Series includes irradiation of HSSI weld 73W to a high fluence [5×10^{19} n/cm^2 (> 1 MeV)] to determine whether the K_{Jc} curve shape change observed in the Fifth Series is exacerbated; all work associated with Phase 2 has been placed on hold pending a review of funding available for this task.

The NUREG report, *Results of Crack-Arrest Tests on Irradiated A 508 Class 3 Steel*, by S. K. Iskander, P. P. Milella, and A. Pini [NUREG/CR-7447 (ORNL-6894)], which contains detailed results of testing and analysis of the Italian crack-arrest specimens, has been completed. The report is being finalized prior to submission to the NRC for printing.

The design and fabrication of the temperature and dosimetry verification capsules are performed under this task, but, for purposes of continuity, that progress will be reported under Task 6, where the design of the new irradiation facilities and capsules is performed.

An abstract by S. K. Iskander, P. P. Milella, A. Pini, and E. T. Manneschmidt, "Crack-Arrest Testing of Irradiated Nuclear Reactor Pressure Vessel Steels at Oak Ridge National Laboratory," has been submitted for an American Society for Testing and Materials (ASTM) Symposium on User's Experience in Crack-Arrest Testing, New Orleans, November 18, 1996.

3. Fracture Toughness Curve Shifts in Low Upper-Shelf Welds (Series 8)

S. K. Iskander, R. K. Nanstad, D. D. Randolph,* and J. J. Henry, Jr.

This task examines the fracture toughness curve shifts and changes in shape for irradiated welds with low CVN upper-shelf energy (USE). This task was specifically designed to address questions raised by the Advisory Committee for Reactor Safeguards concerning the shape of the K_{Ic} curve for irradiated welds with a low USE. In particular, it will clarify whether the high concentration of inclusions in low-USE welds results in a transition relationship and behavior significantly different from high-USE welds. The information developed under this task will augment information obtained from other HSSI tasks performed on two high-USE weldments under the Fifth and Sixth Irradiation Series and on a commercial low USE under the Tenth Irradiation Series. The results will provide an expanded basis for accounting for irradiation-induced embrittlement in RPV materials.

To provide material for this task and for the annealing task (HSSI Series 9), three trial low-USE welds have been fabricated by ABB-Combustion Engineering (ABB-CE), Chattanooga, Tennessee; ABB-CE also fabricated the welds for the Fifth and Sixth Irradiation Series. The semiannual report for October 1994 through March 1995 (ref. 1) gave the chemical and mechanical properties obtained by ABB-CE for welds 1 and 3. The chemical analysis for the third weld, the current status, and test plans for all three welds were given in the semiannual report for April through September 1995 (ref. 2).

CVN tests on 30 specimens have been performed on the weld fabricated using HSSI weld wire 73W and Linde 80 flux. The test results show that there is no significant difference between specimens machined from 13 mm below the top of the weld and those through the depth of the weld. The USE for the welds made with Linde 80 flux is only 10% less than the HSSI welds 72W and 73W made using the same weld wire and Linde 124 flux. This indicates that in this case the Linde 80 flux alone does not cause significant decreases in the USE. There was no significant difference in the 41-J transition temperatures of the Linde 80 and Linde 124 flux welds.

Linde 80 flux was used for all three test welds fabricated by ABB-CE, which have been designated welds 1, 2, and 3. Weld 1 was made with the 73W weld wire which had copper added to the melt to reduce the variations that are associated with copper-coated weld wires. The other two welds were fabricated with a commercially available copper-coated weld wire, L-TEC heat 44112. Welds 2 and 3 had a target copper level of 0.31 and 0.45% by weight, respectively. The copper level of 0.31% for weld 2 could not be attained using the copper-coated weld wire, and a new test weld had to be fabricated, designated weld 2A. For weld 2A, the copper coating was stripped from the weld wire, which contains 0.07% Cu, and, to attain the target copper level, supplemental copper was added to the weld puddle using an ABB-CE proprietary process. Weld 3 was fabricated with the same heat of the L-TEC 44 copper-coated weld wire as used for weld 2, but with supplemental copper added to the weld puddle, which resulted in a weldment containing an average of 0.424% Cu. The semiannual report for October 1993 through March 1994 (ref. 3) discusses the reasons for the choices of copper content and welding wire.

The CVN impact properties for HSSI weld 73W are based on 83 specimens; thus, to get a better statistical basis for the three new test welds and, in particular the values of USE, more CVN impact testing must be performed. Also, the initiation fracture toughness in the transition region, K_{Jc}, for the three welds will be determined. Both welds 1 and 3 have been postweld heat treated (PWHT) at 607°C (1125°C) for 40 h. Except for the 8-h hydrogen bake at 204.4°C (400°F) to prevent cracking, weld 2A was not given the PWHT of welds 1 and 3 to study the effect of PWHTs on copper precipitation.

Sufficient material for the testing of weld 2A has recently been given the same PWHT as that of welds 1 and 3, and atom probe studies are planned before and after PWHT. The room-temperature tensile properties of weld 2A will also be measured. Other phases of this study could include more tests after aging the material at candidate annealing temperatures of 343 and 454°C (650 and 850°F).

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*ABB-Combustion Engineering, Chattanooga, Tennessee.

Thirty CVN specimens were machined through the thickness of weld 1 (0.31% Cu); those fabricated with HSSI weld wire 73W and Linde 80 flux have been tested. ABB-CE previously tested 12 specimens machined from 13 mm below the top layer of this weld. The test results are given in Tables 3.1 and 3.2 for specimens from the top layer and through the depth, respectively, and are shown in Figure 3.1. It may be seen that the difference between the Charpy energy for the specimens from either location in the weld is insignificant; however, as yet unexplained differences in lateral expansion values have been observed.

Table 3.1. ABB-Combustion Engineering test results of Weld Test Plate No. 001 fabricated with HSSI weld wire 73W and Linde 80 flux and stress-relieved 607°C (1125°F) for 40 h (specimens were machined 13 mm below top of weld)

Test temperature		Energy		Lateral expansion		Fracture appearance (% shear)
(°C)	(°F)	(J)	(ft-lb)	(mm)	(mils)	
–40	–40	19	14	0.178	7	0
–40	–40	24	18	0.279	11	0
–12	10	34	25	0.381	15	10
–12	10	52	38	0.610	24	15
4	40	79	58	1.067	42	40
4	40	91	67	1.321	52	60
24	75	83	61	1.118	44	60
24	75	98	72	1.270	50	70
49	120	106	78	1.524	60	90
49	120	111	82	1.575	62	100
71	160	113	83	1.600	63	100
71	160	115	85	1.676	66	100

The results from both locations have been combined into a single set of 42 specimens and are compared in Figure 3.2 to the results of Charpy testing 83 specimens from the Fifth Irradiation Series.[4] The upper shelf of the weld fabricated using Linde 80 flux is about 10% less than that of the Fifth Series, which was made with the same weld wire, but with Linde 124 flux. The ~10% difference in USE obtained using 12 specimens and that obtained using 30 specimens is probably due to the small number of specimens from the 12-specimen set tested at upper-shelf temperatures. Moreover, the difference in transition temperatures is within the range of data scatter. The latter observation is supported by the small differences between the mid-transition temperatures for the various welds. Various parameters from the test results, as well as from fitting of the experimental data with a hyperbolic tangent equation, are indicated in Table 3.3.

It is useful to compare the USE of the new weldment to the range of USEs of several weldments studied in the various HSSI Irradiation Series.[5–7] The USEs of these weldments, fabricated with Linde 80 flux, have been compiled in Table 3.4, and the distribution is shown in Figure 3.3. For purposes of Figure 3.3, the distribution of USEs has been accumulated in "bins," which encompass a range of 5 J, e.g., > 80 to 85 J. The frequency of occurrence, and the cumulative frequency, for each USE bin is shown in Table 3.5. It may be noted from Table 3.4 that the range of USE of the Midland welds almost spans the entire range of USEs of all the welds tested from the four series, except for two welds from the Second and Third Series, which have the lowest and highest USEs, respectively. The USE of the new test weld 2A [122 J (90 ft-lb)] is higher than the maximum value of any of the welds in Table 3.4 [111 J (82 ft-lb)]. The USEs in Table 3.4 range from 74.6 to 111.2 J (55 to 82 ft-lb).

When all testing is completed, an evaluation of all three welds will be performed so that one or possibly two welds can be chosen for fabrication in sufficient quantities for the Eighth and Ninth Irradiation Series. To estimate the linear feet of weld needed for the Eighth and Ninth Series, a detailed test plan must first be developed, particularly

Table 3.2. ORNL test results on Weld Test Plate No. 001 fabricated with HSSI weld wire 73W and Linde 80 flux and stress-relieved 607°C (1125°F) for 40 h (specimens were machined through the thickness of the weld)

Specimen	Test temperature		Energy		Lateral expansion		Fracture appearance (% shear)
	(°C)	(°F)	(J)	(ft-lb)	(mm)	(mils)	
7317	−84	−120	7	5.1	0.076	3	0
7306	−57	−70	13	9.4	0.203	8	5
7303	−40	−40	34	25.4	0.660	26	15
7302	−34	−30	16	11.7	0.406	16	10
7327	−34	−30	47	34.5	0.737	29	15
7309	−29	−20	51	37.4	0.940	37	30
7325	−23	−10	55	40.8	0.864	34	20
7330	−12	10	70	51.5	1.143	45	40
7315	−12	10	76	56.3	1.194	47	40
7310	4	40	74	54.9	1.219	48	40
7329	10	50	80	59.0	1.295	51	50
7311	10	50	82	60.2	1.372	54	80
7323	16	60	86	63.5	1.448	57	80
7304	24	75	98	72.5	1.575	62	70
7324	24	75	124	91.2	1.829	72	70
7318	49	120	104	76.7	1.727	68	85
7307	49	120	114	84.4	1.803	71	85
7321	71	160	118	87.3	2.032	80	100
7316	71	160	130	96.2	2.032	80	100
7326	93	200	112	82.5	1.956	77	100
7313	93	200	124	91.8	1.956	77	100
7314	116	240	117	86.6	1.880	74	100
7301	116	240	124	91.4	1.880	74	100
7320	138	280	120	88.2	1.956	77	100
7328	160	320	119	88.0	2.032	80	100
7308	160	320	122	90.2	1.956	77	100
7319	204	400	122	90.0	2.032	80	100
7312	204	400	125	92.0	1.981	78	100
7322	260	500	122	89.8	2.134	84	100
7305	260	500	122	90.2	1.956	77	100

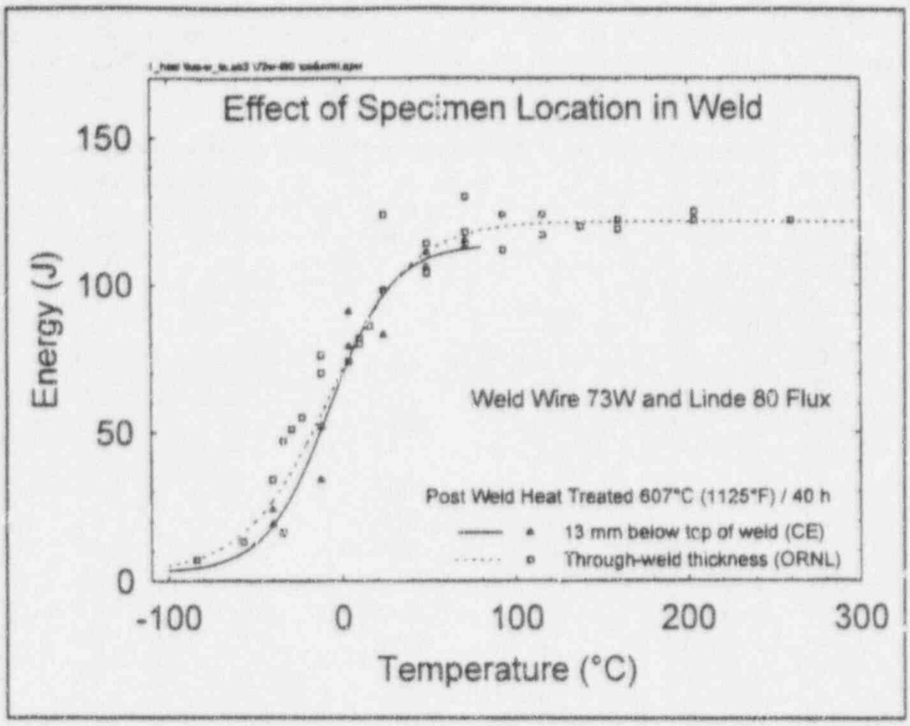

Figure 3.1 Effect of Charpy specimen location on energy.

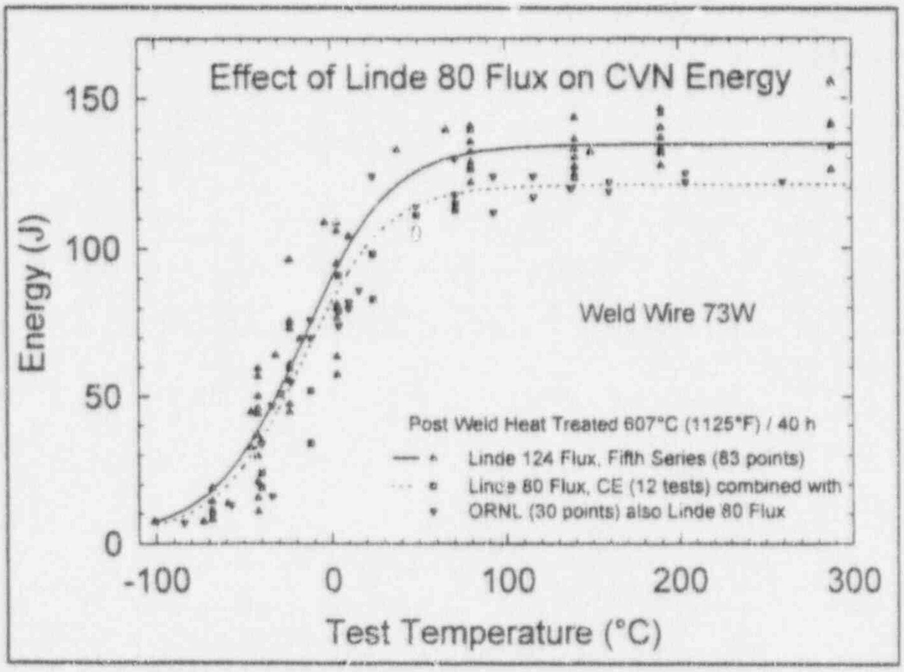

Figure 3.2. Effect of weld flux on Charpy energy.

Table 3.3. Various parameters associated with results obtained by testing welds fabricated using 73W weld wire, Linde 80 flux, Linde 124 flux, and stress-relieved for 40 h at 607°C (1125°F) [specimens fabricated with the Linde 80 flux were tested from two locations on a single weld]

Material	Tanh fit parameters[a]			Transition temperature (°C) at Charpy impact energy levels of:			Drop-weight NDT (°C)	Energy at NDT (J)	Number of specimens
	Upper-shelf energy (J)	Mid-transition temperature (°C)	Transition zone width (°C)	27 J	41 J	68 J			
13 mm below top of weld, Linde 80 flux (ABB-CE results)	113.7	−7.16	73.1	−30	−19	−1	−29	29	12
Through thickness of weld, Linde 80 flux (ORNL results)	121.7	−13.38	96.3	−46	−32	−9	−29[b]	44	30
Combined ABB-CE and ORNL results, Linde 80 flux	121.1	−9.90	95.1	−42	−27	−5	−29[b]	39	42
Fifth Series Charpy tests, Linde 124 flux	134.6	−17.0	101.1	−54	−40	−18	−34	47	83

[a]Obtained by fitting the equation: Energy = (USE + 2.7)/2 + [(USE − 2.7)/2] tanh [(T − MTT)/(TZW/2)], where USE = upper-shelf energy, 2.7 = lower-shelf energy, MTT = mid-transition temperature, and TZW = transition zone width. The 2.7 J is the lower-shelf energy and was determined experimentally from five tests conducted at liquid nitrogen temperature, −196°C, on a submerged-arc weld from the Midland reactor pressure vessel.

[b]Assumed to be the same as the value obtained by ABB-CE.

Table 3.4. Upper-shelf energy of Linde 80 welds tested in the HSSI Second, Third, Fourth, and Tenth Irradiation Series

Material	Upper-shelf energy	
	(J)	(ft-lb)
Tenth Irradiation Series (Midland weld)		
Beltline, Sect. 1-9, 1/4(t)	76.7	56.6
Beltline, Sect. 1-9, 1/2(t)	83.0	61.2
Beltline, Sect. 1-9, 5/8(t)	88.3	65.1
Beltline, Sect. 1-9, 3/4(t)	80.9	59.7
Beltline, Sect. 1-9, 7/8(t)	78.1	57.6
Beltline, Sect. 1-11, 1/4(t)	90.6	66.8
Beltline, Sect. 1-11, 1/2(t)	91.4	67.4
Beltline, Sect. 1-11, 5/8(t)	90.2	66.5
Beltline, Sect. 1-11, 3/4(t)	84.5	62.3
Beltline, Sect. 1-11, 7/8(t)	78.5	57.9
Beltline, Sect. 1-13, 1/4(t)	101.1	74.6
Beltline, Sect. 1-13, 1/2(t) D	104.4	77.0
Beltline, Sect. 1-13, 1/2(t) E	97.5	71.9
Beltline, Sect. 1-13, 5/8(t)	108.2	79.8
Beltline, Sect. 1-13, 3/4(t)	90.2	66.5
Beltline, Sect. 1-15, 1/4(t)	82.4	60.8
Beltline, Sect. 1-15, 1/2(t)	87.6	64.6
Beltline, Sect. 1-15, 5/8(t)	84.9	62.6
Beltline, Sect. 1-15, 3/4(t)	88.9	65.6
Beltline, Sect. 1-15, 7/8(t)	82.8	61.1
Nozzle, Sect. 1-31, 1/2(t)	85.6	63.1
Nozzle, Sect. 1-31, 3/4(t)	88.8	65.5
Nozzle, Sect. 1-31, 7/8(t)	89.8	66.2
Nozzle, Sect. 1-34, 1/2(t)	88.0	64.9
Nozzle, Sect. 1-34, 3/4(t)	85.4	63.0
Nozzle, Sect. 1-34, 7/8(t)	88.9	65.6
Second and Third Irradiation Series		
61W	85.4	63.0
62W	93.6	69.0
63W	89.5	66.0
64W	101.7	75.0
65W	111.2	82.0
66W	74.6	55.0
67W	103.0	76.0
Fourth Irradiation Series		
71W	104.5	77.1

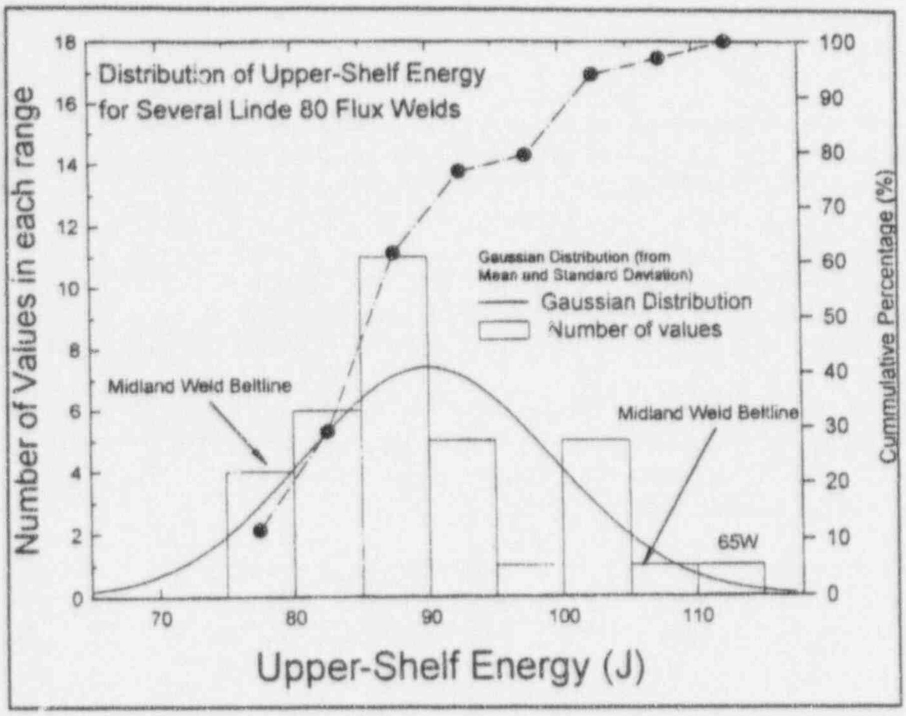

Figure 3.3. Distribution of upper-shelf energy for the HSSI Second, Third, Fourth, and Tenth Irradiation Series.

Table 3.5. Distribution of upper-shelf energy of Linde 80 welds tested in the HSSI Second, Third, Fourth, and Tenth Irradiation Series

Bin (J)	Frequency	Cumulative (%)
80	4	11.76
85	6	29.41
90	11	61.76
95	5	76.47
100	1	79.41
105	5	94.12
110	1	97.06
115	1	100
Total	34	100

with regard for the thickness of the compact specimens. A sufficiently thick compact specimen is required so that the fracture toughness at high levels can be determined, which is necessary to determine whether the shape of the fracture toughness curve in the transition region changes due to irradiation. The development of the new test practice for fracture toughness in the transition region is very promising, since it appears to give a more accurate transition temperature shift at the 100-MPa√m level using 25-mm-thick compact tension [1TC (T)] specimens, perhaps even with smaller specimens. However, to determine the possible changes in slope, a specimen with a capacity of 150 to 200 MPa√m is needed, which may require a larger specimen. The results from the ongoing Tenth Irradiation Series, which investigates a commercial low-USE weld, will also be taken into account in planning the Eighth and Ninth Series.

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*Available for purchase from National Technical Information Service, Springfield, VA 22161.

4. Irradiation Effects in a Commerical Low Upper-Shelf Weld (Series 10)

D. E. McCabe, S. K. Iskander, and D. W. Heatherly

The purpose of the Tenth Irradiation Series was to evaluate the before-and-after irradiation fracture toughness properties of commercially produced WF-70 weld metal. The material has been obtained from Unit 1 of the Midland Reactor of Consumers Power, Midland, Michigan. This vessel became available for test sampling and evaluation when Consumers Power aborted plans to operate the facility. Weld metal WF-70 was used in all girth welds; this designation indicates that a specific lot of Linde 80 weld flux was used that produces low CVN upper-shelf toughness. LUS welds and weld metal WF-70, in particular, have been a source of concern for several currently operating nuclear power production facilities. The beltline weld of the Midland vessel was sampled completely around the girth, and the Tenth Irradiation Series received seven segments approximately 1 m long (40 in.). The nozzle course weld was similarly sampled, but this project received only two of the available segments. These two were spaced about 180° apart.

Tests to establish the transition temperature of the irradiated weld metal have been ongoing during this reporting period. Irradiation exposures have involved two scoping capsules, 10.01 and 10.02, that have received a neutron fluence of 0.5×10^{19} n/cm^2 (> 1 MeV) and two large capsules, 10.05 and 10.06, that have received a neutron fluence of 1×10^{19} n/cm^2 (> 1 MeV). All were irradiated at 288°C (550°F). The scope of test methods applied covers tensile properties, chemistry surveys, DWT-NDT, CVN transition curves, and fracture mechanics testing for K_{Jc} transition temperature curves. Because a difference in the copper content was found between the nozzle course and beltline WF-70 welds, the two were analyzed separately for postirradiation properties. The best data have been developed on the beltline weld metal, since there was far more of that material available.

4.1 Transition Temperature Results

The test program was designed to satisfy the need for two approaches of establishing the transition temperature. One approach is the *ASME Boiler and Pressure Vessel Code*,[1] Sects. III and XI; the other is the master curve method. The data required to establish the ASME lower-bound K_{Ic} curve are CVN transition curves and DWT-NDT temperatures. Both were performed on unirradiated weld metals. There were 19 CVN curves on the beltline weld and 6 on the nozzle course weld. For irradiated conditions, only one CVN curve per irradiated condition was obtained.

The data needed for the master curve definition are elastic-plastic stress intensity factors, K_{Jc}. Such values were determined using compact specimens of various sizes and precracked CVN specimens. All were used to identify a reference temperature, T_o (see Tables 4.1 and 4.2). The master curve reference temperature is where 1T size specimens will give an average K_{Jc} equal to 100 MPa$\sqrt{m}$. These values are summarized in Table 4.3 for the WF-70 beltline weld and in Table 4.4 for the nozzle course weld.

In the case of the ASME Code, the reference temperature is RT_{NDT}, and for low-USE materials, it is established from the CVN transition curve. The master curve concept implies that there is a universal transition curve shape; this curve is positioned using the reference temperature, T_o. The advantage in the latter case is that fracture mechanics data are used to position a fracture mechanics-based transition curve. Only postulated correlations to fracture mechanics data are used in the ASME Code method. The result is a higher degree of certainty in the master curve method. A comparison example based on the Midland WF-70 beltline data is presented in the next section.

Table 4.1. Reference temperature, T_o, determinations on Midland WF-70 beltline weld metal

Test temperature (°C)	Specimen size (T)	$K_{Jc(med)}$ (MPa√m)	T_o (°C)
Unirradiated			
–25	1	170.6	–62
–25	2	153.7	–55
–50	1/2	112.3	–58
–50	1	97.4	–47
–50	2	113.8	–60
–60	0.4	104.2	–59
–60	0.4	102.6	–62
–70	0.4	72.1	–43
–75	1	69.2	–44
–100	1	53.5	–42
Irradiated at 1 × 10^{19} n/cm^2			
90	1	211.8	40.7
75	1	220.1	23.2
50	1/2	130.2	32.2
35	1	97.0	38.3
26	0.4	94.0	31.5
20	1/2	85.1	33.5
0	0.4	62.4	41.0

Table 4.2. Reference temperature, T_o, determinations on Midland WF-70 nozzle course weld metal

Test temperature (°C)	Specimen size (T)	$K_{Jc(med)}$ (MPa√m)	T_o (°C)
Unirradiated			
0	1	213.8	–51
–25	1	106.2	–29.5
–50	1	68.8	–19
–50	1/2	101.3	–27
–100	1	50.3	–35
			Total–36.1
Irradiated at 1 × 10^{19} n/cm^2			
45	1	77.9	65
75	1	113.8	65.5
65	1/2	92.3	71.1
25	0.4	78.3	48.2
			Total 59.1

Table 4.3. WF-70 Beltline weld fracture toughness

	41 J	NDT	RT_{NDT}	T_o
Unirradiated				
Range (low/high), °C	–25/14		–20/37	–63/–43
Number of determinations	19	4	19	9
Average, °C	–8	–55	0	–54

	T_{41J}	T_o	ΔRT_{NDT}	ΔT_o
Irradiated, 0.5 × 10^{19} n/cm²				
Number of determinations	1	1	1	1
Average, °C	36	5	44	59
Irradiated, 1.0 × 10^{19} n/cm²				
Range (low/high), °C		23/41		
Number of determinations	1	7	1	7
Average, °C	94	33	102	87

Table 4.4. WF-70 Nozzle course weld fracture toughness

	41 J	NDT	RT_{NDT}	T_o
Unirradiated				
Range (low/high), °C	−11/5		−8/18	−51/−19
Number of determinations	6	2	6	5
Average, °C	−9	−50	11	−36

	T_{41J}	T_o	ΔRT_{NDT}	ΔT_o
Irradiated, 0.5 × 10¹⁸ n/cm²				
Range (low/high), °C	62		71	
Number of determinations	1		1	
Average, °C	62		71	
Irradiated, 1.0 × 10¹⁹ n/cm²				
Range (low/high), °C	89	48/71	98	
Number of determinations	1	4	1	4
Average, °C	89	59	98	95

4.2 ASME Analysis Versus Master Curve

The end of life (EOL) for axial weld metal is specified in "Title 10, Part 50," of the *Code of Federal Regulations* (10CFR50).[2] The irradiated reference temperature, represented by RT_{PTS}, is not to exceed 132°C (270°F), otherwise a plant cannot continue to operate without demonstrated justifications. The significance of this rule to the Midland beltline weld metal can be evaluated using the available data at 1×10^{19} n/cm^2. There are two evaluation schemes: (1) determine a *Regulatory Guide 1.99* (ref. 3) chemistry factor experimentally using a known ΔRT_{NDT} from CVN tests to project to an EOL fluence, and (2) establish reference toughness curves, i.e., K_{Ic} or master curve K_{Jc} at 1×10^{19} n/cm^2, to predict the reserve of fracture toughness at 132°C (270°F). This information would be input into a pressurized thermal shock analysis of a reactor vessel.

4.2.1 End of Life by Fluence, ASME versus Master Curve

The experimental determination of the chemistry factor is obtained by the following equation:

$$\Delta RT_{NDT} = (CF) \cdot f^{(x)}$$

$$\text{where } (x) = 0.28 - 0.1 \log_{10}(f) \; . \tag{1}$$

The ASTM standard E 185-73 is referenced for the determination of ΔRT_{NDT}, and for beltline material it is 102°C. Note that $f^{(x)} = 1$ at $f = 1 \times 10^{19}$ n/cm² so that ΔRT_{NDT} equals the chemistry factor, namely 102°C (184°F). The median chemistry factor is used to solve for the fluence to EOL at 132°C (270°F).

Fluence to end of life by American Society of Mechanical Engineers

Material	Case	ΔRT_{NDT} to EOL	$f^{(x)}$	EOL fluence ($\times 10^{19}$ n/cm²)
Beltline WF-70	Worst	171°F	0.93	0.8
	Median	238°F	1.29	3
	Best	266°F	1.44	7

The same problem worked in terms of the master curve gives a lower chemistry factor, 87°C (157°F), and lower initial reference temperature, T_o = –54°C (–65°F); see Table 4.3.

Fluence to end of life by master curve

Material	Case	ΔRT_{NDT} to EOL	$f^{(x)}$	EOL fluence ($\times 10^{19}$ n/cm²)
Beltline WF-70	Worst	315°F	2.00	> 10
	Median	335°F	2.13	> 10
	Best	351°F	2.23	> 10

Equation (1) does not cover cases of fluence greater than 10^{20} n/cm².

4.2.2 Fracture Toughness at 132°C (270°F) after 1×10^{19} n/cm² Irradiation

The value of K_{Ic} is set by the following equations:

$$K_{Ic} = 33.5 + 21.04 \exp\left[0.02\left(T_{270} - RT_{PTS}\right)\right] \quad \mathrm{ksi}\sqrt{\mathrm{in.}} ,$$

$$\text{where } RT_{PTS} = RT_{NDT} + \Delta RT_{NDT} \; (°\mathrm{F}) \; (\mathrm{ASME}) . \tag{2}$$

$$K_{Jc(0.02)} = 22.09 + 27.3 \exp\left[0.01055\left(T_{270} - T_o\right)\right] \quad \mathrm{ksi}\sqrt{\mathrm{in.}} , \tag{3}$$

$$\text{or } RT_{PTS} = T_o \; (°\mathrm{F}) \; (\text{Master Curve}) \quad \mathrm{ksi}\sqrt{\mathrm{in.}} .$$

Predicted fracture toughness at 132°C (270°F) after 1.0×10^{19} n/cm²

Method	Case	RT_{NDT} (°C)	RT_{PTS} (°F)	K_J at 132°C (ksi√in.)
By ASME	Worst	37	282	50
	Median	0	216	96
	Best	–20	179	162
By master curve	Worst		106	176
	Median		91	182
	Best		73	239
K_{Ic} at 132°C by Eq. (2)				54.5

The example problem worked on the Midland reactor beltline weld by the ASME method shows a scatter in fluence to EOL by a factor of almost ten. By chance alone, the screening criteria (worst case) could be exceeded at $f = 1.0 \times 10^{19}$ neutrons/cm².

With the master curve there is a direct determination of the as-irradiated fracture toughness and hence reduced data scatter. Even the worst-case scenario by the master curve predicts that the Midland reactor, based on the beltline weld metal, would not have to deal with the screening criterion (54.5 ksi√in. minimum) of 10CFR50.

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5. Thermal Embrittlement Potential and Irradiation Effects in Reactor Pressure Vessel Weld Heat-Affected Zones (Series 11)

R. K. Nanstad and D. E. McCabe

The task title for the Eleventh Irradiation Series has been changed to reflect some additional new objectives. Emphasis is now being given to determine if there is a potential problem with thermal embrittlement in RPV steels. An annealing experiment on laboratory heats made with steels having typical pressure-vessel chemical compositions was conducted by AEA Technology, Harwell, United Kingdom. They showed quite clearly that there can be grain boundary embrittlement in RPV steels given large, prior austenite grain size and high phosphorus on the order of 0.017 wt %. The work of this task will be to reexamine the AEA Technology heat treatments using five commercially made RPV steels, representing A 302 grade B, A 533 grade B, two modified A 302 grade B, and A 508 class 2 heats. The phosphorus content covers the range typical of commercial RPV production heats. The AEA Technology heat treatment exaggerates the problem by creating a microstructure that optimizes embrittlement sensitivity. This plan has the experimental objective of screening materials for sensitivity to embrittlement.

A second task will evaluate the potential for local brittle zone (LBZ) development in multipass submerged-arc welds made in RPV joints. LBZs are thin layers of coarse-grain base metal adjacent to the fusion line. The plan is to simulate the thermal cycle of the heat-affected LBZ using an electrical resistance device used for such purposes by welding engineers, i.e., the Gleeble.

Finally, the third task will be to select two materials of highest interest resulting from the two preceding tasks and produce a simulated commercial submerged-arc weld. The purpose will be to (1) determine if there are LBZs, and (2) if so, demonstrate the significance of these zones to the structural integrity performance of RPVs. An additional objective will be to determine if irradiation will promote temper embrittlement of LBZs.

6. Annealing Effects in Low Upper-Shelf Welds (Series 9)

S. K. Iskander

The purpose of the Ninth Irradiation Series is to evaluate the correlation between fracture toughness and CVN impact energy during irradiation, annealing, and reirradiation (IAR). The annealing and testing of specimens from capsule 10.6 has been completed. Planning of the specimen complement for the first reirradiation capsule is under way. It will contain specimens that were previously irradiated and annealed. Two IAR facilities have been fabricated, assembled, leak-checked, and cleaned. They are ready for the final electrical connections and the heat-up tests to be performed by the Instrumentation and Controls (I&C) Division of ORNL. The reusable temperature verification capsule has been fabricated, assembled, leak-tested, and cleaned. It is also awaiting delivery to the I&C shop for electrical lead terminations and testing. It is currently planned for the IAR facilities to undergo a 1-week annealing test at ORNL before they are delivered to the University of Michigan Ford Nuclear Reactor (FNR). The dosimeters and the dosimetry capsule have all been completed and are awaiting installation during the first in-reactor tests scheduled to be performed in late August 1996. The design of a reusable capsule capable of reirradiating previously irradiated and annealed CVN and compact tension specimens is also progressing. The data acquisition and control instrumentation for the first two IAR facilities is complete and is awaiting completion of the IAR facilities and temperature test capsule for checkout and control algorithm development.

6.1 Testing of Annealed and Irradiated Specimens (M. A. Sokolov, S. K. Iskander, R. L. Swain, E. T. Manneschmidt, and J. J. Henry, Jr.)

The annealing and testing of specimens from capsule 10.6 has been completed. Six 1T C(T) specimens of the Midland beltline weld were tested in the transition region after annealing at 343°C for 168 h, and six 1T C(T) specimens of the Midland nozzle course weld were tested after annealing at 454°C, also for 168 h. Based on previous Charpy results, test temperatures were selected to result in median initiation fracture toughness values (K_{Jc}) of about 100 MPa$\sqrt{m}$. Fracture surfaces of specimens have been photographed, and data analysis will be performed shortly. Five precracked CVN specimens of the HSSI weld 64W were also tested. Two tensile specimens each of the Midland beltline and nozzle course welds were annealed at 343 and 454°C for 168 h, respectively. Only these eight tensile specimens have not yet been tested.

Planning of the specimen complement for the IAR capsule is under way. It will contain specimens that were previously irradiated and annealed. The specimen complement is needed to aid in the design of the IAR capsule. The target fluence planned for the first capsule is 0.5×10^{19} n/cm^2 (> 1 MeV).

Reviewers' comments on two papers presented at the ASTM Seventeenth International Symposium on Effects of Radiation on Materials, held in Sun Valley, Idaho, June 20-23, 1994, have been received, incorporated, prepared in the form of camera-ready manuscripts, and transmitted to ASTM for publication in STP 1270: "Effects of Annealing Time on the Recovery of Charpy V-Notch Properties of Irradiated High-Copper Weld Metal," by S. K. Iskander, M. A. Sokolov, and R. K. Nanstad, and "The Effect of Thermal Annealing on Fracture Toughness of Low Upper-Shelf Welds," by M. A. Sokolov, R. K. Nanstad, and S. K. Iskander.

A paper, "A Perspective on Thermal Annealing of Reactor Pressure Vessel Materials from the Viewpoint of Experimental Results," by S. K. Iskander, M. A. Sokolov, and R. K. Nanstad, was presented at the 1996 ASME/JSME Fourth International Conference on Nuclear Energy (ICONE-4), held in New Orleans, March 10–14, 1996, and published in the proceedings.

6.2 Design, Fabrication, and Installation of New Irradiation Facilities (D. W. Heatherly, C. A. Baldwin, D. W. Sparks , and G. E. Giles, Jr.)

The two IAR facilities and the heater connectors to the instrumentation have been assembled, leak-tested, and cleaned. During testing, it was discovered that one of the facilities had a leak in the flexible steel lead tube and had to be replaced; both HSSI IAR facilities are now leak-tight. The two facilities will be functionally tested at ORNL with the temperature verification capsule before they are shipped to the FNR. Functional testing consists of an "underwater" test at 454°C for 1 week to simulate annealing conditions.

The temperature verification capsule has been fabricated; functional tests of the capsule in the facility are to be conducted as mentioned above. In preparation for the functional fit tests of the temperature capsule in the facility, a potential problem was uncovered. Such a problem could have caused warping, leading to a reduction of the working clearances; the necessary modifications to the facilities have been made and, thus, the problem has been avoided. The temperature verification capsule will contain 24 thermocouples for controlling and recording specimen temperatures. A dosimeter capsule has also been fabricated and is ready for use in the dosimetry verification at the FNR. All of the dosimeters for the first two experiments have been fabricated and are generally inserted on-site at the FNR.

Both the temperature and dosimetry verification capsules are designed to be reusable and will be used in all positions on the east face of the reactor, as well as on the south face when the latter facilities are completed. If one of the two facility positions becomes vacant, to preserve the proper irradiation geometry, the temperature verification capsule could be used as a dummy capsule. A description of the dosimetry capsule has been given in the last semiannual progress report.[1]

A purchase order has also been awarded for the fabrication of two sets of IAR capsule parts. These parts will be used to assemble two capsules with unirradiated specimens to be irradiated and possibly annealed and reirradiated in situ.

The basic design of the capsule for previously irradiated and annealed specimens has been completed, and preparation of engineering drawings is progressing. The capsule has been designed so that the specimens can be exchanged in the hot cell of the FNR. The capsule hardware and thermocouples can be reused for several reirradiations.

6.3. Data Acquisition and Control System (M. T. Hurst)

The data acquisition and control system (DACS) instrumentation and the control algorithm development for the first two IAR facilities are complete and awaiting checkout. A brief description of the DACS has been included in a previous semiannual progress report.[2] The burn-in of the circuit boards and testing of the control algorithms of the DACS are ongoing. Equipment for a special junction box, termed the "electrical experiment tie-in," is being assembled. The tie-in is the location where the sheathed electric heater and thermocouple cables from the capsules are connected and serves as an intermediate junction between the sheathed cables and the control cabinet cables. It is kept at a negative pressure in the remote event that radioactive gases escape from the capsule up through the cable sheath. Design drawings are being prepared for fabrication of the electrical and thermocouple connections. A conceptual pressure loop for the IAR facility is nearing completion and has been submitted for final approval.

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1. S. K. Iskander, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., "Annealing Effects in Low Upper-Shelf Welds," in *Heavy-Section Steel Irradiation Program Semiannual Progress Report April-September 1995*, USNRC Report NUREG/CR-5591, Vol. 6, No. 2 (ORNL/TM-11568/V6&N2), 1996.*

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*Available for purchase from National Technical Information Service, Springfield, VA 22161

7. Microstructural and Microfracture Analysis of Irradiation Effects

R. E. Stoller, P. M. Rice, and K. Farrell

7.1 Mechanical Properties and Microstructural Examinations

A series of nine model light-water reactor (LWR) pressure vessel steels were ion irradiated at ~ 300°C using 2.5 MeV He ions, to a dose of 1.4×10^{17} ions/cm^2, which corresponds to ~ 0.1 displacements per atom (dpa) at a depth of 2 µm and ~ 3.5 dpa at the peak damage region which occurs at about 4 µm deep. The resultant changes in hardness as a function of depth were measured using an ultralow-load hardness tester. TEM was used to investigate the defect distributions. The alloy compositions are given in Table 7.1, and the TEM characterization of these alloys in the as-received condition is given in ref. 1. A strong effect of the solutes Cu and Mn and a weaker effect of N, C, and Ti was observed on the change of hardness and the defect distribution due to the ion irradiation. Similar results were reported previously when these same alloys were irradiated with Fe ions.[2]

Table 7.1. List of model alloys and their compositions

Alloy Number	N (appm)	Composition (wt %)			
		Cu	Mn	C	Ti
VM 348	5	—	—	—	—
VM 349	80	—	—	—	—
VM 350	120	—	—	—	—
VM 360	10	0.89	1.03	< 0.003	—
VM 373	100	< 0.01	0.01	< 0.003	0.3
VM 387	10	0.51	0.05	0.17	0.003
VM 390	20	0.51	0.06	< 0.005	0.002
VM 397	20	0.91	< 0.01	< 0.01	< 0.01
VM 399	120	0.51	0.01	< 0.01	< 0.01

The specimens were irradiated as TEM discs in the ORNL Triple Ion Facility using 2.5 MeV He ions to a dose of 1.4×10^{21} ion/m^2 at a temperature of ~ 300°C. The accelerator is capable of irradiating nine TEM discs at a time. Each target included three discs of three different alloys. The first target held three discs of each of the alloys VM348, VM390, and VM399. The second target held three discs each of alloys VM349, VM360, and VM373. The third target held three discs each of the alloys VM350, VM397, and VM387. The targets where first heated to approximately 150°C and then the ion beam, at a current of ~ 0.6 mA/cm^2, was used to heat the specimens to ~ 300°C. The temperature of the specimens was measured with a thermocouple spot welded to the front of one of the target specimens. Each target was irradiated for a total of ~ 11 h over 2 d.

The TRIM code was used to calculate the doses obtained for the specified irradiation conditions. At a depth of 2 μm below the irradiated surface the dose is ~ 0.1 dpa; a peak dose of ~ 3.5 dpa occurs just beyond 4 μm in depth. At the 2 μm depth the level of implanted helium is ~ 10 appm, with the peak occurring at 60,000 appm (6 at. %) just beyond 4 μm. Thus, both dose and helium content are varying with depth. The dose variations discussed below correspond to measurements made at different depths.

Hardness Measurements

The technique developed to prepare cross-section specimens from ion irradiated discs has been described in detail in a previous paper and so will not be discussed here.[2] The change in hardness caused by the irradiation was measured as a function of depth (i.e., distance below the irradiated surface) by using a low load hardness measuring device called the Nanoindenter.[2] The method used for these experiments has also been described in ref. 2. Briefly, a row of approximately 100 indents is made at a very shallow angle to the original irradiated surface, starting a few microns beyond the irradiated region and ending in the electroplated layer of Fe at the specimen surface. The hardness values reported here were obtained from indents at a contact depth of 50 nm to ensure that a constant and reasonable volume of material was sampled. The hardness measured from indents beyond the irradiated region (> 5 μm) was averaged to obtain a baseline, and the change in hardness was measured with respect to this value.

The most significant result of the hardness measurements can be seen by separating the alloys into two categories, Cu-free alloys and Cu-containing alloys. A plot of the change in hardness as a function of dose for the Cu-free alloys is displayed in Figure 7.1. The indent contact depths were ~ 50 nm, and each point on the graph represents the average of four or five measurements made within 0.25 μm of the specific depths 1, 2, 3, and 3.5 μm.

Nitrogen was observed to have only a weak effect on the hardening, as can be seen by comparing alloys VM349 and VM350 with VM348. One of the possible reasons for the weakness of the observed effect is the fact that a significant portion of the nitrogen was observed to have precipitated out in all the high-N alloys in the as-received condition with the exception of the titanium-containing alloy, VM373, as reported earlier.[1] An approximate average fit to the data for the three Fe-N alloys is shown and compared to the Fe-Ti-N alloy. All four alloys indicate that the initial hardening rate is relatively low but increases almost linearly up to the highest dose. This is in strong contrast to the copper-containing alloys in Figure 7.2. The Cu-containing alloys evidence a rapid hardening at low dose with an apparent saturation at higher doses.

Comparing alloys VM390 and VM397 shows that an increase in copper increases the change in hardness at all doses. However, of potentially greater interest is the fact that the two low-nitrogen, Cu-containing alloys VM390 and VM397 seemed to be unaffected by the high He concentrations. The change in hardness for both alloys remained constant even out to the peak dose/end of range ([He] = 60,000 appm), and neither alloy showed evidence of helium bubble formation while the high helium content led to extensive bubble formation in all the other alloys. This absence of bubble formation indicates a strong copper-vacancy binding which is, no doubt, responsible for part of the effect of copper in irradiated steels.

Comparing alloy VM360 with VM397 shows that the addition of 1 wt % Mn to the Fe matrix, in conjunction with the 0.9% Cu, doubles the change in hardness due to the irradiation at all doses.

TEM Imaging

The cross-sectional specimens were ground to approximately 100-μm thickness and then electrochemically polished to perforation with a twin jet polisher using a perchloric acid-based electrolyte cooled to -60°C. All TEM imaging was done with a 300-kV Philips CM30 microscope, using g = {330} imaging conditions so that the background intensity was weak and the dislocation loops appeared as sharp black spots.

The defect microstructure in each of the irradiated alloys was investigated in cross section using conventional TEM imaging techniques. Figure 7.3 shows a survey of the cross-sectional, low-magnification micrographs of the irradiated region for each of the alloys. The alloy numbers and their compositions are listed at the base of

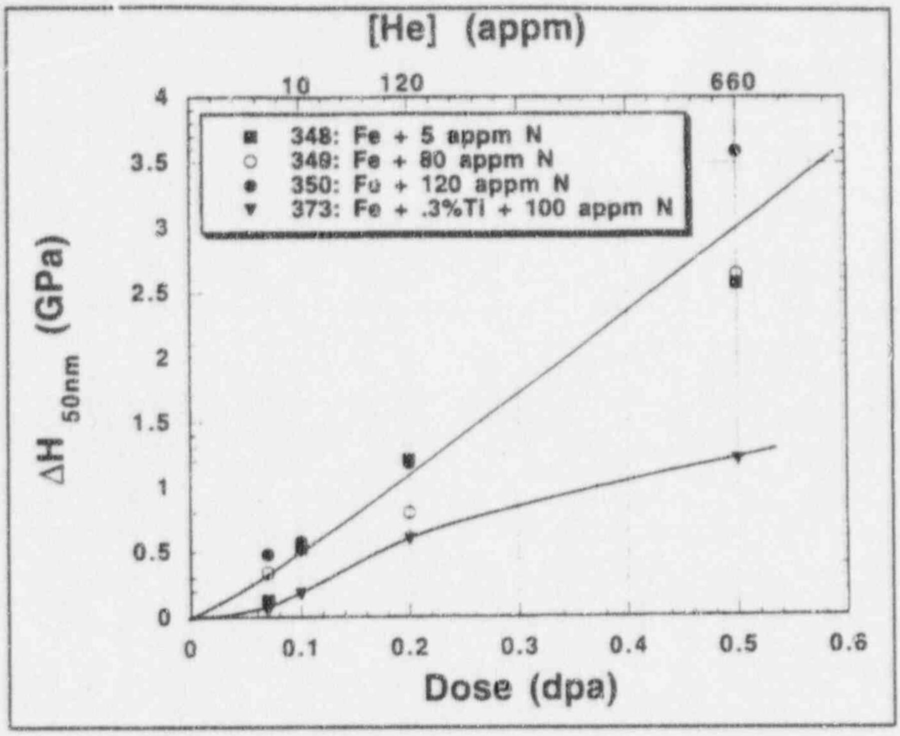

Figure 7.1. Dose dependence of the change in hardness for the copper-free alloys. The upper line represents an approximate average for the three nitrogen levels' fits of the data. Note that they all show a low initial hardening rate.

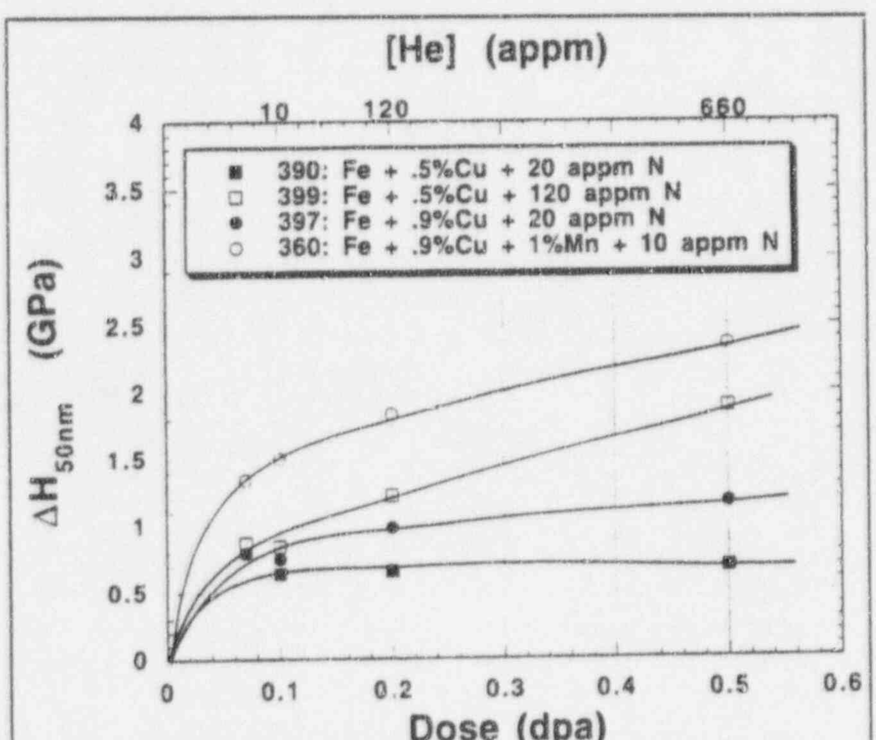

Figure 7.2. Dose dependence of the change in hardness for the copper-containing alloys. Note that they all show evidence of a high initial hardening rate.

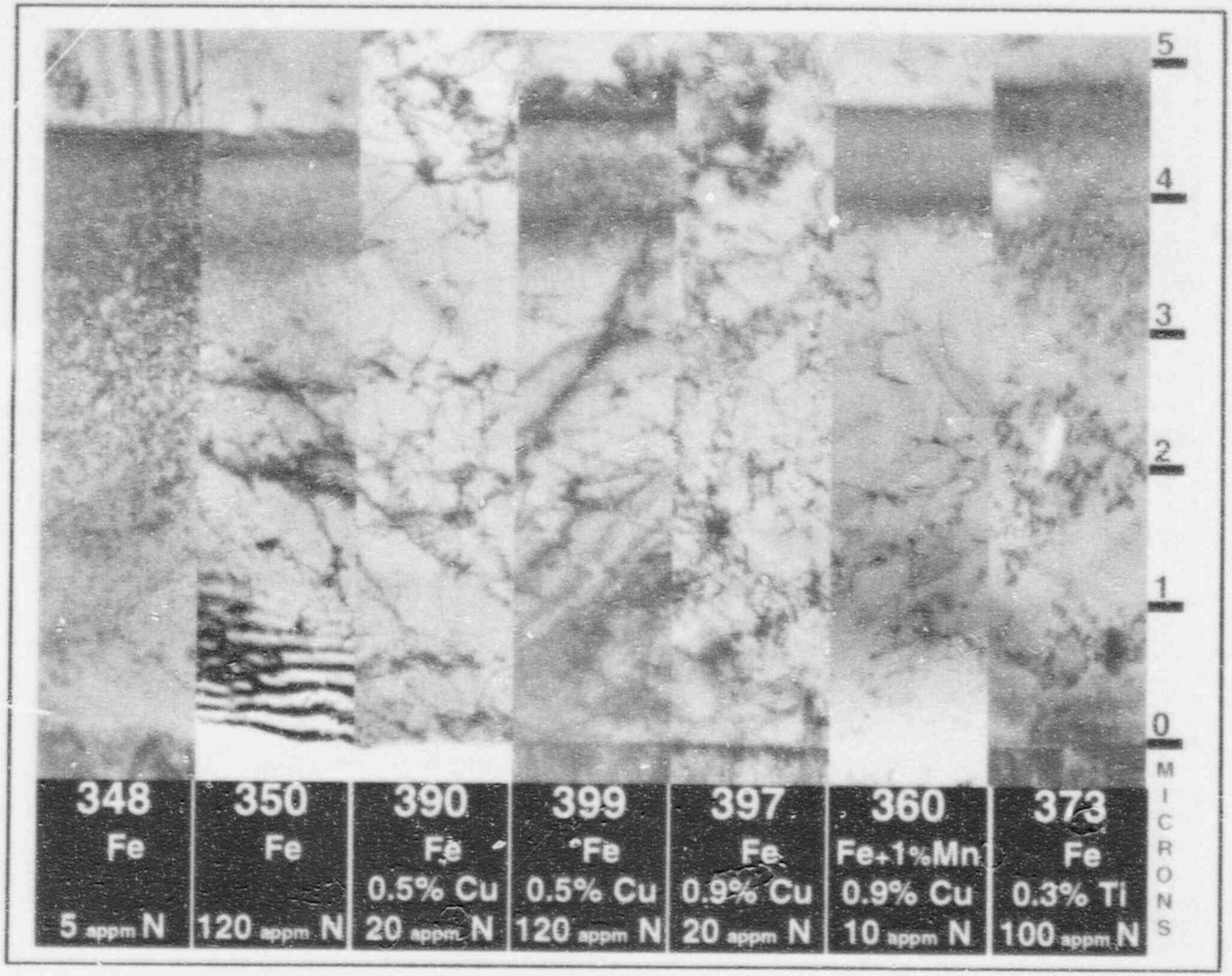

Figure 7.3. A survey of the low-magnification cross-sectional TEM micrographs showing the entire irradiated region of the major alloy. The scale is on the left and the alloy contents are listed below each image. Note that alloys VM390 and VM397 show no obvious irradiation damage even at the end of range of the He ions.

each micrograph, and the depth scale is shown on the right-hand side of the figure. Most alloys showed an extremely high defect density at the end of range of the He ions, at and beyond 4 μm. The two low-N, Cu-containing alloys, VM390 and VM397, show no such peak damage region.

Comparing alloys VM348 and VM350 indicated that the increase in nitrogen content seems to have slightly decreased the average diameter of the dislocation loops from ~ 10 nm (VM348) to ~ 8 nm (VM350). Comparing alloys VM390 and VM397 with VM348 shows that the addition of Cu (while maintaining low N) seems to have resulted in a complete lack of visible defects being formed (i.e., dislocation loops, if present, must be less than 2 nm in diameter). A comparison of alloy VM399 with VM390 shows that when sufficient nitrogen is added to the FeCu alloy, irradiation causes dislocation loops large enough to be visible (d > 2 nm) to form.

A comparison of alloy VM360 with VM397 shows that the addition of Mn to the matrix also results in visible dislocation loops. Although they are just barely visible (d ~ 2 nm), there is definitely a higher density of them than in any of the other alloys. This high density of very small dislocation loops correlates well with alloy VM360 showing the greatest change in hardness in the low-dose, relatively low-He region.

As mentioned above, it is of significant interest that alloys VM390 and VM397 showed no bubble formation even at the end of range where the helium concentration was ~ 60,000 appm. This ability of the copper to suppress He bubble formation appears to be negated by either the addition of a substitutional solute like Mn (see alloy VM360) or an interstitial solute like N (see alloy VM399). This surprisingly strong copper-vacancy interaction is consistent with a postulated high-copper-vacancy binding energy but may be much greater in a clean Fe-Cu binary alloy.[3-5] A postirradiation anneal of a VM397 specimen at 600°C for 24 h resulted in extensive bubble/void formation at the end of range depth.

Although the formation of visible defects was suppressed in alloys VM390 and VM397, significant hardening was measured. This implies that the defects responsible for the hardening have a diameter of less than 2 nm, the approximate limit of TEM visibility in these magnetic materials. A preliminary calculation of the hardening that could be attributed to the visible defects in the other alloys indicates that sub-visible defects must be responsible for some of the hardening in these alloys as well. Thus, the TEM technique has proved to be limited in characterizing all the necessary defects in these alloys that are representative of RPV steels.

7.2 Modeling and Data Analysis

The results of the molecular dynamics (MD) cascade simulations have been used to obtain effective cross sections for both total point defect production and point defect cluster production. The new cross sections were calculated in collaboration with L. R. Greenwood of the Pacific Northwest National Laboratory (PNNL) using the SPECTER code and cover neutron energies from below 1×10^{-9} MeV to 20 MeV. The energy dependence of the cross sections generally follows that of the NRT dpa cross section, but significant differences are observed between these results and secondary defect production models proposed previously by other researchers.

These MD-based damage production cross sections were used to compute spectrally averaged cross sections for several pressurized-water reactor (PWR) and boiling-water reactor (BWR) pressure-vessel locations to investigate possible defect production differences due to neutron spectral differences. The radial locations for each reactor type were: (1) in the water adjacent to the inside of the RPV, (2) the 1/4-T location in the RPV, (3) the 3/4-T location, and (4) in the cavity adjacent to the outside of the RPV. The BWR and PWR neutron spectra were obtained from J. V. Pace of ORNL. An initial review of the results indicates that differences between the two reactor types are modest.

7.3 Neutron Flux and Spectrum Effects

Analysis of tensile data obtained by testing SS3 tensile specimens that were irradiated at 50 to 60°C in the High Flux Beam Reactor (HFBR) at the Brookhaven National Laboratory, the High Flux Isotope Reactor at ORNL, and FNR at the University of Michigan has been completed. These results are described in detail in ORNL/NRC/LTR-96/3. The experimental conditions permitted an examination of the effect of fast flux over the range of 1.8×10^{12} n/m^2 to 4×10^{18} n/m^2 and the effect of thermal neutrons for thermal-to-fast flux ratios from essentially 0 to almost 400.

Typical results are shown in Figure 7.4 for the HSST-02 correlation monitor material. When the radiation-induced yield strength change is evaluated on the basis of fast fluence as shown in Figure 7.4(*a*), the effect of the high thermal-to-fast flux ratio in the HFBR V10 site can be seen in the enhanced strengthening at low fluences. In particular, the data from the cadmium-shielded V10 capsules (filled circles) should be compared with unshielded data (filled circles). However, when the data are plotted against dpa in Figure 7.4(*b*), the correlation is significantly improved. The remaining difference between the data from the two types of HFBR V10 capsules is smaller than the overall data scatter in the complete data set. This difference may be due to uncertainties in neutron dosimetry, or there may be a slight residual enhancement that is not accounted for through the use of dpa as a correlation parameter. Alternately, since there is no remaining systematic variation within the total data set, these differences may simply represent material property data scatter.

Similar results were seen for several other ferritic steels. Overall, these results demonstrate that there is no significant effect of displacement rate (fast flux) over a broad range at these temperatures and that neutron energy spectrum effects are reasonably well correlated on the basis of dpa.

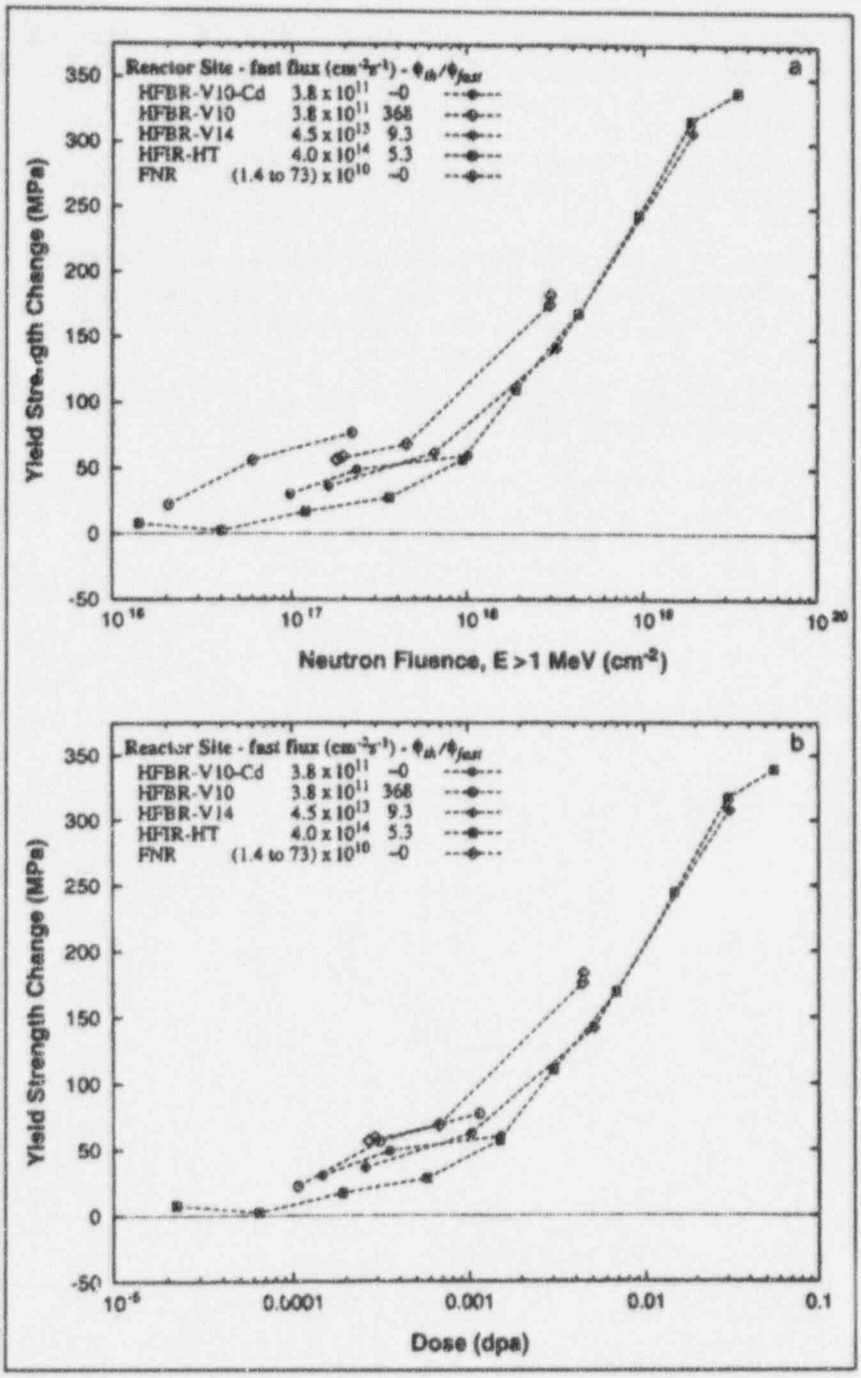

Figure 7.4. Yield strength change in A533B (HSST-02 correlation monitor) irradiated with a range of fast fluxes and thermal-to-fast flux ratios: *(a)* dependence on fast fluence and *(b)* dependence on dpa.

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*Available for purchase from National Technical Information Service, Springfield, VA 22161.
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8. In-Service Irradiated and Aged Material Evaluations

F. M. Haggag, R. K. Nanstad, and D. J. Alexander

The objective of this task is to provide a direct assessment of actual material properties in irradiated components of nuclear reactors, including the effects of irradiation and aging. Four activities are currently in progress: (1) establishing a machining capability for contaminated or activated materials by completing procurement and installation of a computer-based milling machine in a hot cell; (2) machining and testing specimens from cladding materials removed from the Gundremmingen reactor to establish their fracture properties; (3) preparing an interpretive report on the effects of neutron irradiation on cladding; and (4) continuing the evaluation of long-term aging of austenitic structural stainless steel weld metal by metallurgically examining and testing specimens aged at 288 and 343°C and reporting the results, as well as continuing the aging of the stainless steel cladding to and beyond a total time of 50,000 h.

8.1 In-Service Aging

Type 308 stainless steel weldments were aged at 343°C for up to 50,000 h. Tensile and CVN specimens were fabricated from the weld metal. Aging had little effect on the tensile properties but did result in embrittlement as shown by the impact testing. The degree of embrittlement increased with increasing ferrite content; the ductile-to-brittle temperature increased, and the USE decreased. The embrittlement continues to increase with increasing aging time. A draft report summarizing the results of these tests has been prepared and is in the internal review process.

8.2 Remotely Operated Machining Center

Modification of the computer numerically controlled (CNC) machining center (model VMC-100) is in progress. Several items have been completed, including drawings, new cables and table, machine enclosure, fittings, and a floor tub for installation inside the hot cell. Also, the automated ball indentation machine was moved from cell 6 to cell 2 to provide a place for the CNC machine. Hot cell decontamination and new crane installation are in progress. Furthermore, a new saw was purchased for slicing specimens into suitable sizes for machining on the CNC machine. This work has been slowed pending budget review.

8.3 Thermal Aging of Stainless Steel Weld-Overlay Cladding

Tensile, CVN, and fracture toughness testing of three-wire stainless steel cladding, thermally aged for 20,000 h at 288°C and at 343°C, was completed. The test results show that the effects of thermal aging at both temperatures were relatively small and similar to those reported earlier for 1605 h aging at 288°C. Hence, aging of additional three-wire cladding at 288°C for 50,000 h (completion expected in July 1996) and possibly greater is continuing to better quantify the effects of long-term thermal aging.

A NUREG report, *Effects of Thermal Aging and Neutron Irradiation on the Mechanical Properties of Three-Wire Stainless Steel Weld Overlay Cladding* [NUREG/CR-6363 (ORNL/TM-13047)], by F. M. Haggag and R. K. Nanstad, was completed and has been prepared for publication.

9. Evaluation of Steel from the JPDR Pressure Vessel

W. R. Corwin and M. A. Sokolov

There is a need to validate the results of irradiation effects research by the examination of material taken directly from the wall of a pressure vessel which has been irradiated during normal service. This task has been included within the HSSI Program to provide just such an evaluation on material from the wall of the pressure vessel from the JPDR.

The JPDR was a small BWR that began operation in 1963. It operated until 1976, accumulating ≈17,000 h of operation, of which a little over 14,000 h were with the original 45-MWTh core, and the remaining fraction, late in life, with an upgraded 90-MWTh core. The pressure vessel of the JPDR, fabricated from A 302, grade B, modified steel with an internal weld-overlay cladding of 304 stainless steel, is approximately 2 m ID and 73 mm thick. It was fabricated from two shell halves joined by longitudinal seam welds located 180° from each other. The rolling direction of the shell plates is parallel to the axis of the vessel. It operated at 273°C and reached a maximum fluence of about 2.3×10^{18} n/cm^2 (> 1 MeV). The impurity contents in the base metal are 0.10 to 0.11% Cu and 0.010 to 0.017% P with a nickel content of 0.63 to 0.65%. Impurity contents of the weld metal are 0.11 to 0.14% Cu and 0.025 to 0.039% P with a nickel content of 0.59%.

The current status of the JPDR pressure vessel is that it has been cut into pieces, roughly 800 x 800 mm x the original local wall thickness. Full-thickness trepans have been cut from one of the sections originally located at the core beltline and from one of the sections near the upper flange, well away from the beltline. Eight beltline trepans were removed containing the longitudinal fabrication weld as were eight beltline trepans located completely within the base metal. Nine remotely located trepans were taken containing the longitudinal fabrication weld as were 14 containing only base metal. Japan Atomic Energy Research Institute (JAERI) has shipped the irradiated material from the wall of the JPDR that will be examined at ORNL, where it was subsequently received and moved into the hot cells, where it is to be machined and examined. The material received at ORNL consists of 16 full-thickness trepans, each approximately 87 mm in diameter. The trepans contain four types of material: weld metal and base metal, each in both the irradiated condition (from the beltline) and in nominally, thermally aged-only condition (from the upper flange). ORNL received four trepans of each material. JAERI has placed all the remaining vessel material in a hot warehouse on-site for long-term storage and currently has no plans to do anything else with it.

The objectives of the JAERI JPDR pressure vessel investigations are to obtain materials property information on the pressure-vessel steel actually exposed to in-service irradiation conditions and to help validate the methodology for aging evaluation and life prediction of RPVs. The Japanese research associated with the evaluation of irradiation effects is composed of three parts: examination of material from the JPDR vessel in conjunction with a reevaluation of its exposure conditions, new test reactor irradiations of archival and similar materials, and reevaluation of data from irradiation surveillance and related programs. The focus of the research to be performed by ORNL on the JPDR material is the determination of irradiation-induced damage through the thickness of the vessel in the beltline region and its comparison with the properties and microstructural evaluations of the same material following short, high-rate irradiations or with thermal damage only. This will be done by fabricating fracture and microstructural specimens from the trepans taken from the beltline and from the region remote from the beltline. Parallel determinations of exposure will be made by dosimetry measurements taken on the vessel material itself and by supporting neutron transport calculations.

During this reporting period, planning was initiated at ORNL for the machining of the JPDR vessel trepans in conjunction with the completion of the installation of the CNC to be used for hot-cell fabrication of material irradiated in service (see Task 8). Recent studies conducted at JAERI have focused on measurements of the level of neutron attenuation through the vessel using hardness and tensile testing through the full thickness of the vessel sections of base metal before and after their annealing. By comparing the results before and after annealing, as a function of through-wall position, and making the assumption that all of the change is due to recovery of irradiation damage, they deduce the through-wall attenuation in irradiation-induced hardening. They have obtained good agreement

between the tensile and hardness data for the base metal, showing a change of about 3.3 points in Vickers hardness for each 1 MPa change in yield strength. The data from the weld metal are not yet analyzed. Initial indications from the testing are that the attenuation is somewhat greater than would be predicted by the attenuation formula in *Regulatory Guide 1.99*. This trend will be further evaluated by both JAERI and ORNL in the continuation of this work.

10. Fracture Toughness Curve Shift Method

R. K. Nanstad, M. A. Sokolov, and D. E. McCabe

The purpose of this task is to examine the technical basis for the currently accepted methods for shifting fracture toughness curves to account for irradiation damage, and to work through national codes and standards bodies to revise those methods, if a change is warranted. Specific activities under this task include: (1) collection and statistical analysis of pertinent fracture toughness data to assess the shift and potential change in shape of the fracture toughness curves due to neutron irradiation, thermal aging, or both; (2) evaluation of methods for indexing fracture toughness curves to values that can be deduced from material surveillance programs required under the *Code of Federal Regulations* (10CFR50), Appendix H; (3) participation in the pertinent *ASME Boiler and Pressure Vessel Code*, Section XI, and ASTM E-8 and E-10 committees to facilitate obtaining data and disseminating the results of the research; (4) interaction with other researchers in the national and international technical community addressing similar problems; and (5) frequent interaction, telephone conversations, and detailed technical meetings with the NRC staff to ensure that the results of the research and proposed changes to the accepted methods for shifting the fracture toughness curves reflect staff assessments of the regulatory issues.

In a previous semiannual report,[1] preliminary analysis of all relevant HSSI Program data was presented and discussed. Using Weibull statistics, it was shown that the maximum likelihood approach gave good estimations of the fracture toughness reference temperature, T_0 (the temperature at 100 MPa$\sqrt{m}$), determined by rank method and could be used for analysis of data sets where application of the rank method did not prove to be feasible. It was shown by a linear least-squares fit to the data set that, on average, the fracture toughness shifts generally exceeded the Charpy 41-J shifts by about 15%. This result is dominated by the data with shifts greater than about 30°C. Results obtained from the adjustment procedures mentioned above are in a published letter report.[2] A database of Charpy impact and fracture toughness data for RPV materials which have been tested in the unirradiated and irradiated conditions is being assembled. Once the database is assembled, the raw data will be analyzed in a consistent manner, and the radiation-induced transition temperature shifts for Charpy and fracture toughness data will be compared.

Estimation of Nil-Ductility and Reference Crack-Arrest Toughness Temperatures from Charpy Force-Displacement Traces

As part of this evaluation, the evaluation of instrumented Charpy impact and dynamic precracked Charpy tests is being analyzed for potential use in the estimation of various toughness parameters from testing of small specimens in impact. A force-displacement trace from a Charpy impact test of an RPV steel in the transition range has a characteristic point, so-called "force at the end of unstable crack propagation," F_a. The end of unstable crack propagation in a Charpy impact test is the event that is somewhat common in nature with the DWT-NDT and the crack-arrest fracture toughness tests. This similarity was used to find correlations between NDT and a reference crack-arrest fracture toughness temperature (T_{a100}). The T_{a100} is defined as a temperature at median crack-arrest fracture toughness equal to 100 MPa$\sqrt{m}$ when the crack-arrest fracture toughness data are analyzed by the master curve procedure.

The two-parameter Weibull distribution function is suggested to model the scatter of the F_a in Charpy tests:

$$P[F_a \le F_{ia}] = 1 - \exp\left[-\left(\frac{F_{ia}}{F_o}\right)^b\right], \tag{1}$$

where P is the cumulative probability that the force at the end of unstable crack propagation, F_{ia}, is greater than or equal to the F_a value of interest. F_o and b are the scale and the slope parameters of the Weibull distribution function, respectively.

Forty-five specimens of Linde 80 weld were tested at 1.7°C (35°F) as a part of the ASTM reconstituted round-robin. Having such a large replicated data set, Charpy traces were evaluated in terms of F_a values and the Weibull function, Eq. (1), was used to model the scatter in these values. Figure 10.1 presents the Weibull distribution function fitting to the F_a data. The slope of regression line, parameter b in Eq. (1), is determined to be equal to four. The scale parameter, F_o, is illustrated in Figure 10.1 as an F_a value at $\ln\{\ln[1/(1 - P)]\} = 0$. Knowing parameters F_o and b allows one to determine the median F_a value, $F_{a(med)}$, for this data population as an F_a at P = 0.5.

Based on these results, the value of the Weibull slope in Eq. (1) was fixed at b = 4. The two-parameter Weibull probability function with b = 4 was used to model the distribution of the F_a in Charpy tests performed at ORNL on other RPV welds in the unirradiated and irradiated conditions. These data have a good replication at a given test temperature, thus, the statistical analysis was applicable. Figure 10.2 presents median F_a values versus temperature, T, normalized to NDT, namely T – NDT. It is shown that median F_a values of different RPV welds have a tendency to form the same shape of temperature dependence. It suggests a universal shape of the temperature dependence of F_a for different RPV steels:

$$F_{a(med)} = [1.79 + 0.027(T - NDT)]^2, \text{ kN} , \tag{2}$$

where T and NDT are in °C. According to Eq. (2), $F_{a(med)}$ equals 3.2 kN at T = NDT.

Although it is not yet clear that the above Weibull statistic/master curve approach is applicable to crack-arrest toughness data, crack-arrest fracture toughness data of HSSI 72W, HSSI 73W, and Midland beltline welds were analyzed by the same approach. The reference crack-arrest toughness temperatures, T_{a100}, were also used in this correlation study. In Figure 10.3, median F_a values are plotted against normalized temperature, $T - T_{a100}$, and a fit to the data is:

$$F_{a(med)} = [2.35 + 0.027(T - T_{a100})]^2 , \text{ kN} , \tag{3}$$

where T and T_{a100} are in °C. Thus, T_{a100} can be estimated from analysis of Charpy traces as a temperature at $F_{a(med)} = 5.5$ kN.

Dependencies (2) and (3) can be used for estimation of NDT or T_{a100} from any Charpy test results.

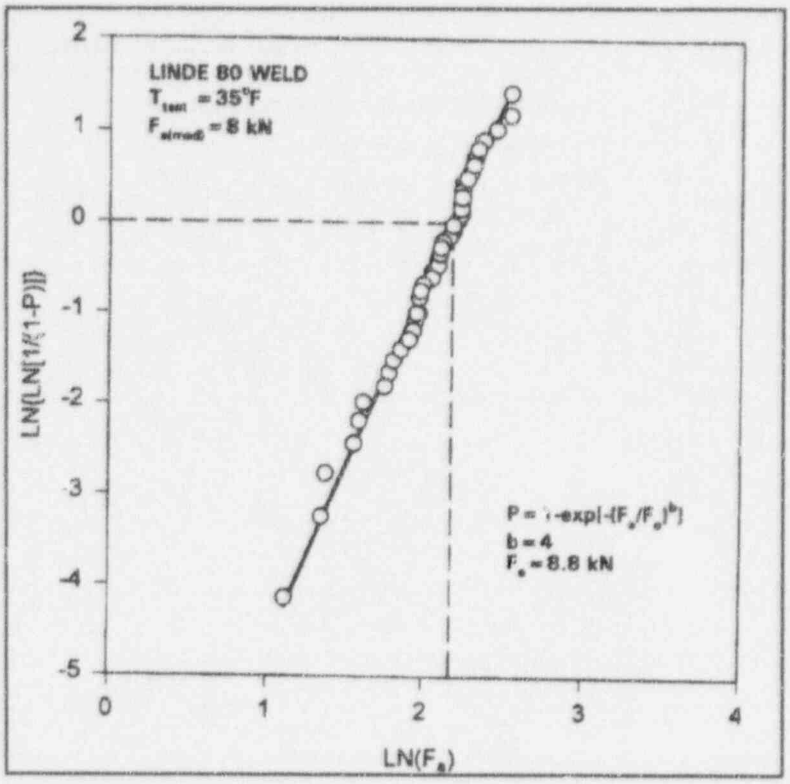

Figure 10.1. Weibull plot of F_a data from Linde 80 weld tested at 1.7°C (35°F).

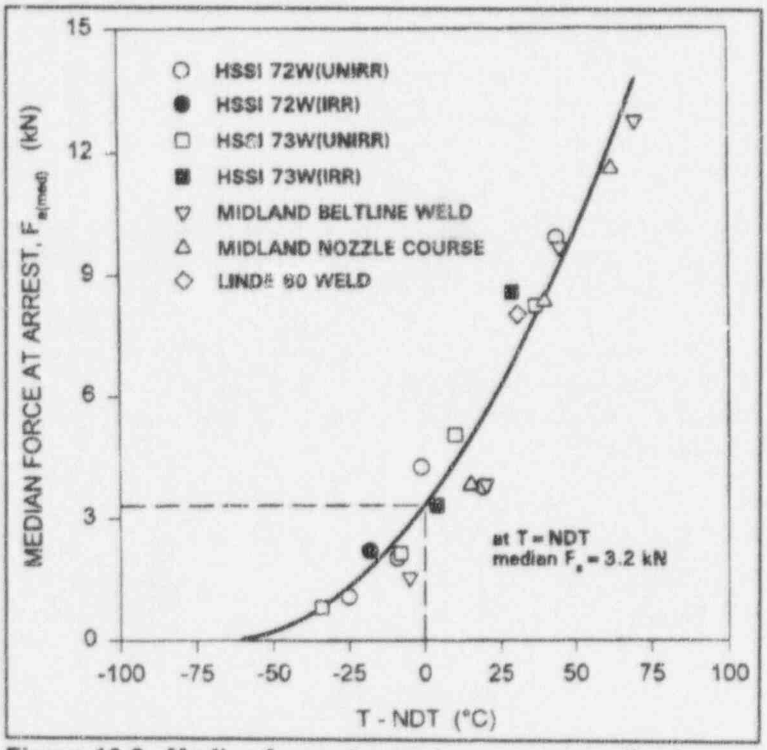

Figure 10.2. Median force at arrest versus normalized temperature T-NDT.

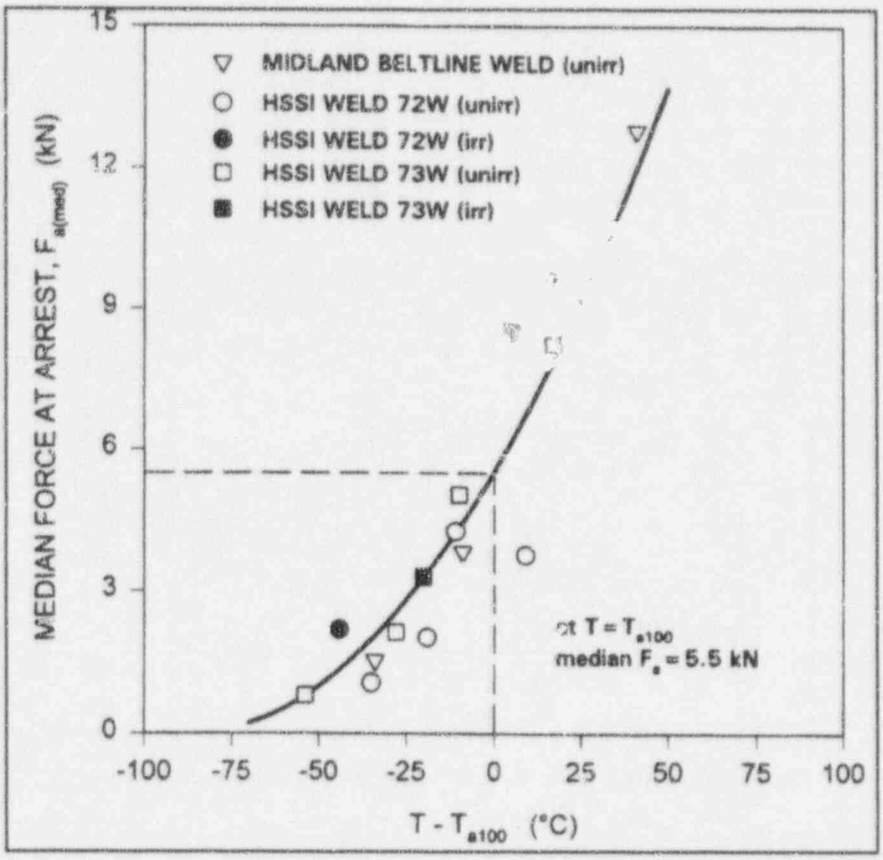

Figure 10.3. Median force at arrest versus normalized temperature, $T-T_{a100}$.

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*Available for purchase from National Technical Information Service, Springfield, VA 22161.

11. Special Technical Assistance

S. K. Iskander, D. J. Alexander, R. K. Nanstad, and M. A. Sokolov

The purpose of this task is to perform various special analytical and experimental investigations to support the NRC in resolving regulatory research issues related to irradiation effects on materials. This task currently addresses two major areas: (1) providing technical expertise and assistance in the review of national codes and standards and (2) experimental evaluations of test specimens and practices and material properties.

11.1 Plan to Survey Variability in Chemical, Mechanical, and Toughness Properties of the Pressure Vessel Research Users Facility (PVRUF) Reactor Vessel (S. K. Iskander, E. T. Manneschmidt, and J. J. Henry, Jr.)

Subtask 11.8 directs the HSSI Program to perform "a survey of the variability in chemical composition and fracture properties on the welds of the reactor vessel in the user facility at ORNL...". Furthermore, it requires that this survey include the determination of the variations of properties in the depth and circumferential directions within the beltline welds of the vessel. The mechanical properties will include CVN impact, DWT, and small specimen fracture toughness, up to 1T C(T).

Besides this task, the HSST Program and PNNL also have needs for material containing welds from this vessel. PNNL had previously performed nondestructive examination (NDE) of the vessel welds and would like material to verify the NDE indications that were recorded. The previous semiannual has extensive details on preparations for this task.[1]

During this reporting period, preparations continued for the characterization of the beltline weld from an unused RPV located at the K-25 site. Alternate material removal schemes are being considered to avoid the expense of handling a 375-ton vessel. Basically, smaller pieces within the capacity of the vendor's cranes will be removed from the vessel rather than the more optimum method of cutting an entire circumferential weld, etc. This will require more planning to avoid cuts at locations that have significant NDE indications of interest to PNNL. The HSST Program has also expressed interest in material from the vessel for their programmatic needs and to verify the residual stresses that are generally assumed to exist.

A meeting has been held at the K-25 site to discuss the work scope with an estimator and representatives for the welders and riggers. The K-25 personnel have a mobile 75-ton crane that could be used to aid in vessel cutup. There would be significant cost benefits to using an on-site crane rather than one supplied by a vendor. The estimate for K-25 personnel to perform the vessel cutup has been submitted to ORNL. A memo summarizing the progress made to date has been prepared.* Since that meeting, the estimator who prepared the estimate has relocated, and a new estimator is now reevaluating the cost estimates.

In a meeting in early February 1996 with the NRC HSSI Program Monitor, M. G. Vassilaros, ORNL was requested to proceed, as soon as possible, with the determination of the variation of chemistry through the thickness and circumferentially at four locations on the PVRUF vessel. Variations of radiation-sensitive chemical elements are one of the top priorities for the NRC. This will be performed by removal of "cores" from these four locations. This does not preclude, at some later date, characterization of the mechanical properties of the beltline weld if funds become available.

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*S. K. Iskander, Oak Ridge Natl. Lab., personal communication to R. K. Nanstad, Oak Ridge Natl. Lab., Nov. 27, 1995.

Vendors are being sought to remove "cores" from four equally spaced locations in one of the circumferential welds of the PVRUF vessel. ABB-CE, Chattanooga, Tennessee, has submitted a cost estimate to perform such an operation. No coring cutter capable of penetrating the entire wall thickness is currently available; ABB-CE proposes to perform the coring from both sides using a through-wall pilot hole.

The HSSI Program has contacted S. R. Doctor, PNNL, to ensure that the coring operation is not performed at locations that have given important ultrasonic indications when the vessel underwent NDE in the past for a separate NRC program. In April 1996, PNNL sent two staff members to place identifying punch marks at various locations on the inside of the vessel. The HSSI Program provided limited support to the PNNL staff during their visit to Oak Ridge. At the request of S. R. Doctor, ORNL has retrieved seven NDE blocks from the K-25 site and placed them in storage at ORNL.

At the time of preparation of this report, NRC decided to concentrate its resources with regard to characterization of RPV welds on material from the Shoreham vessel and transfer actual material acquisition from the PVRUF to the HSST Program.

References

1. S. K. Iskander et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., "Special Technical Assistance," in *Heavy-Section Steel Irradiation Program Semiannual Progress Report for April–September 1995*, USNRC Report NUREG/CR-5591, Vol. 6, No. 2 (ORNL/TM-11568/V6&N2), 1996.*

*Available for purchase from National Technical Information Service, Springfield, VA 22161.

12. Technical Assistance for JCCCNRS Working Groups 3 and 12

R. K. Nanstad, S. K. Iskander, M. A. Sokolov, and R. E. Stoller

The purpose of this task is to provide technical support for the efforts of the U.S.-Russian JCCCNRS Working Group 3 on radiation embrittlement and Working Group 12 on aging. Specific activities under this task are: (1) supply of materials and preparation of test specimens for collaborative IAR studies to be conducted in Russia; (2) capsule preparation and initiation of irradiation of Russian specimens within the United States; (3) preparation for, and participation in, Working Groups 3 and 12 meetings; and (4) sponsoring of the assignment at ORNL of a scientist from the Russian National Research Center, Kurchatov Institute.

Irradiation Experiments in Host Country

The CVN and round tensile specimens of two Russian weld metals irradiated in HSSI capsule 10.06 at the University of Michigan FNR were returned to ORNL. The capsule has been disassembled, specimens identified, and the specimens of Russian steels have been transferred from the disassembly hot cells (Building 3525) to the testing hot cells (Building 3025). Some of the specimens will be tested in the irradiated condition while the remainder will be thermally annealed and tested. This testing is anticipated to be completed by August 1996, depending on funding and scheduling of the next JCCCNRS Working Group 3 meeting.

13. Correlation Monitor Materials

W. R. Corwin and E. T. Manneschmidt

This task has been established with the explicit purpose of ensuring the continued availability of the pedigreed and extremely well-characterized material now required for inclusion in all additional and future surveillance capsules in commercial LWRs. Having recognized that the only original materials qualified for use as a correlation monitor in reactor surveillance capsules are the pieces remaining from the early HSST plates 01, 02, and 03, this task will provide for cataloging, archiving, and distributing the material on behalf of the NRC.

During this reporting period, the transfer of the residual correlation monitor material found at Y-12 to ORNL was completed, and archival storage of the correlation monitor material was maintained. Plans to initiate the construction of a weatherproof enclosure to protect the steel were delayed due to lack of overall program funding.

14. Test Reactor Irradiation Coordination

D. W. Heatherly, D. W. Sparks, and K. R. Thoms

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 the exposure, disassembly, and return of specimens.

Currently, the University of California at Santa Barbara (UCSB) is the only other NRC contractor for whom irradiations are to be conducted. These irradiations will be conducted at the University of Michigan in conjunction with other irradiations being conducted for the HSSI Program. When this project was initiated, the plan was to modify the current ORNL University of Michigan irradiation capsules to facilitate the irradiation of UCSB specimens. The request from UCSB was to have a high, intermediate, and low flux area in which to irradiate their specimen packets and to have the capability of removing and inserting specimen packets at given intervals in order to obtain desired fluences. It was also requested that each area for irradiation of specimen packets have 3 axial temperature zones of 260, 290, and 320°C (500, 554, and 608°F). Control of the zone temperatures was requested to be within 5° of desired temperature. After several iterations, it was determined that an entirely new facility and capsule design would be necessary to provide UCSB with the desired fluence, temperature, temperature control, and space for the specimen packets to be irradiated.

Work on this task was halted early in this reporting period while overall budgets and schedules for this activity were agreed upon by ORNL, UCSB, and NRC. About midway through the period it was decided that the UCSB irradiation facility and experiments would be funded as originally designed, and the irradiation experiments in the FNR at the University of Michigan would begin in the last quarter of 1996. As work resumed the first task was to complete the design drawings for the UCSB facility. The drawings were in the final review stage at the end of this reporting period. The entire drawing package consists of detailed drawings on the UCSB facility, thermocouples, electrical heaters, removable specimen container baskets, and a prototype (dummy) UCSB facility. The prototype facility will be used in the UCSB facility position in the event the UCSB facility is removed from the reactor.

The current design of the UCSB irradiation facility consists of a sealed container instrumented with electrical heaters and thermocouples for monitoring and controlling specimen temperatures. Inside the facility there are two removable specimen baskets which will contain the specimen packets supplied by UCSB. The UCSB specimen packets consist of various types of small specimens encapsulated in unsealed aluminum containers. The outer dimensions of the specimen packets are 50.8 x 50.8 x 12.7 mm (2 x 2 x 0.5 in.). The packets of specimens can be placed inside the facility in various locations to provide the desired temperature and fluence.

The high-flux portion of the facility consists of a removable specimen basket which can contain 2 specimen packets at 260°C (500°F), 2 specimen packets at 320°C (608°F), and up to 14 specimen packets at 290°C (554°F). The intermediate/low-flux portion of the UCSB facility consists of a removable specimen basket which can contain up to 9 specimen packets at 260°C (500°F), 9 specimen packets at 320°C (608°F), and 18 specimen packets at 290°C (554°F). There is also space for four half-size specimen packets in the intermediate/low-flux portion of the facility due to a thermocouple blade inserted into the center of the basket to measure specimen temperatures at the center of the basket. The high-flux and the intermediate/low-flux baskets can be removed from the facility during reactor shutdowns, and the specimen packets in the various regions of the facility can be removed and replaced according to the UCSB irradiation plan. Since the FNR typically runs on a 10-d cycle at 2 MW (20 MWd), the specimens can be irradiated to various levels of radiation damage in 20-MWd increments. After irradiation the specimen packets will be shipped to UCSB for disassembly and testing.

CONVERSION FACTORS[a]

SSI Unit	English Unit	Factor
mm	in.	0.0393701
cm	in.	0.393701
m	ft	3.28084
m/s	ft/s	3.28084
kN	lb_f	224.809
kPa	psi	0.145038
MPa	ksi	0.145038
MPa•√m	ksi•√in.	0.910048
J	ft•lb	0.737562
K	°F or °R	1.8
kJ/m^2	in.-lb/in.^{2}	5.71015
$W \cdot m^{-3} \cdot K^{-1}$	Btu/h•ft^2•°F	1.176110
kg	lb	2.20462
kg/m^3	lb/in.^{3}	3.61273×10^{-5}
mm/N	in./lb	0.175127
T(°F) = 1.8(°C) + 32		

[a]Multiply SI quantity by given factor to obtain English quantity.

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M. G. Vassilaros, NRC Project Manager

11. ABSTRACT (200 words or less)

The goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of effects of neutron irradiation on material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 14 tasks: (1) program management, (2) fracture toughness (K_{Ic}) curve shift in high-copper welds, (3) crack-arrest toughness (K_{Ia}) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K_{Ic} and K_{Ia} curve shifts in low upper-shelf welds, (6) annealing effects in low upper-shelf welds, (7) irradiation effects in a commercial low upper-shelf weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) JPDR steel examination, (13) technical assistance for JCCCNRS Working Groups 3 and 12, and (14) additional requirements for materials. This report provides an overview of the activities within each of these tasks from October 1995 Through March 1996.

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

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irradiation
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