NUREG/CR-5591, Vol. 4, No. 1
ORNL/TM-11568

Heavy-Section Steel Irradiation Program

Progress Report for October 1992 – March 1993

Prepared by
W. R. Corwin

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

UNITED STATES NUCLEAR REGULATORY COMMISSION

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555-0001

2. The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328

3. The National Technical Information Service, Springfield, VA 22161-0002

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the Government Printing Office: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grantee reports, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852-2738, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018-3308.

NUREG/CR-5591, Vol. 4, No. 1
ORNL/TM-11568

Heavy-Section Steel Irradiation Program

Progress Report for
October 1992 – March 1993

Manuscript Completed: December 1997
Date Published: April 1998

Prepared by
W. R. Corwin

Oak Ridge National Laboratory
Operated by Lockheed Martin Energy Research Corp.
Oak Ridge, TN 37831-6285

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

M.G. Vassilaros, NRC Project Manager

MASTER

Prepared for
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code L1098

DISCLAIMER

Abstract

Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. The RPV is one of only two major safety-related components 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 that occurs during service. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established at Oak Ridge National Laboratory (ORNL) under sponsorship of the Nuclear Regulatory Commission (NRC). The primary goal of this major safety program is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior (in particular, the fracture toughness properties) of typical pressure-vessel steels as they relate to light-water-reactor pressure-vessel integrity. The program centers on experimental assessments of irradiation-induced embrittlement (including the completion of certain irradiation studies previously conducted by the Heavy-Section Steel Technology Program) 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.

During this reporting period, a preliminary matrix has been developed for high-fluence irradiations of HSSI weld 73W, the higher copper of the two welds examined in the first phase of the Fifth Series; the duplex-type crack-arrest specimens from Phase II of the irradiated K_{Ia} program were tested, and a new, remote crack-arrest fixture required for collaborative testing of large irradiated Italian K_{Ia} specimens was built; an initial assessment was made of the welding consumables to be used in the fabrication of the high-copper, low upper-shelf (LUS) weld to examine the shift and change in shape of K_{Ic} and K_{Ia} curves for an LUS high-copper weld metal; Charpy V-notch (CVN) specimens of high-copper weld metal that were irradiated in the Fifth Irradiation Series were annealed and tested; the tests for the fracture mechanics evaluation of the Midland LUS weld in the unirradiated condition were nearly completed; a new, brittle weld crack-starter bead material (McKay DWT) was qualified to replace the MUREX Hardex-N that had been widely used in the past for crack-arrest testing but is no longer available; the irradiation of the first large capsule containing the Midland LUS weld was completed; an evaluation was made of the results of refined calculations and detailed experimental measurements of the exposure parameters at the various surveillance capsule locations of the High Flux Isotope Reactor (HFIR) and detailed measurements made; experiments were begun to investigate the effects of a very wide range of flux levels on embrittlement at low temperatures as was the design of a high-temperature irradiation capsule for similar experiments; in-cavity irradiations of vessel support materials were completed; unirradiated microstructural characterization of a Russian VVER-440 reactor vessel steel was completed; collaborative investigations of in-cascade, irradiation-induced point-defect generation were initiated; a bid was placed for a computer numerically controlled vertical milling machine to provide remote machining capabilities of service-exposed components; arrangements were initiated to test round, notched and precracked fracture toughness specimens to evaluate their potential for reactor surveillance applications; the baseline testing for an American Society for Testing and Materials round robin on reconstituted CVN specimens was completed; informal agreement was reached on the details of the collaborative research agreement on material from the wall of the pressure vessel from the Japan Power Development Reactor with the Japan Atomic Energy Research Institute; the 1-year assignment of M. A. Sokolov of the Russian National Research Center-Kurchatov Institute to ORNL was initiated; CVN and tensile specimens of two U.S. reactor vessel materials were encapsulated and placed into irradiation positions in the Novovoronezh Unit 5 reactor vessel in Russia, and two weld materials of Russian vessel steels were machined into similar specimens and encapsulated for irradiation as part of the Tenth Irradiation Series; technical participation was provided at the fourth information exchange of Working Group 3 of the Joint Coordinating Committee for Civilian

Nuclear Reactor Safety, held in St. Petersburg and Moscow, Russia; and tensile and CVN specimens were tested for three stainless steel welds with varying ferrite contents, which have been aged at 343°C for up to 20,000 h.

Contents

Figures

Tables

Acknowledgments

The authors would like to thank Julia Bishop for her contributions in the preparation of the draft manuscript for this report, Marcia Hodges for the final manuscript preparation, and Kathy Spence for editing. The authors also gratefully acknowledge the continuing technical and financial contributions of the Nuclear Regulatory Commission to the Heavy-Section Steel Irradiation Program.

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 that 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 September 1992 through March 1993. 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 Applications, and Engineering Technology Divisions.

Previous HSSI Progress Reports in this series are:

NUREG/CR-5591, Vol. 1, No. 1 (ORNL/TM-11568/V1&N1)
NUREG/CR-5591, Vol. 1, No. 2 (ORNL/TM-11568/V1&N2)
NUREG/CR-5591, Vol. 2, No. 1 (ORNL/TM-11568/V2&N1)
NUREG/CR-5591, Vol. 2, No. 2 (ORNL/TM-11568/V2&N2)
NUREG/CR-5591, Vol. 3 (ORNL/TM-11568/V3)

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.

ORNL-4176
ORNL-4315
ORNL-4377
ORNL-4463
ORNL-4512
ORNL-4590
ORNL-4653
ORNL-4681
ORNL-4764
ORNL-4816
ORNL-4855
ORNL-4918
ORNL-4971
ORNL/TM-4655 (Vol. II)
ORNL/TM-4729 (Vol. II)
ORNL/TM-4805 (Vol. II)
ORNL/TM-4914 (Vol. II)
ORNL/TM-5021 (Vol. II)
ORNL/TM-5170
ORNL/NUREG/TM-3
ORNL/NUREG/TM-28
ORNL/NUREG/TM-49
ORNL/NUREG/TM-64
ORNL/NUREG/TM-94
ORNL/NUREG/TM-120
ORNL/NUREG/TM-147
ORNL/NUREG/TM-166
ORNL/NUREG/TM-194
ORNL/NUREG/TM-209
ORNL/NUREG/TM-239
NUREG/CR-0476 (ORNL/NUREG/TM-275)
NUREG/CR-0656 (ORNL/NUREG/TM-298)
NUREG/CR-0818 (ORNL/NUREG/TM-324)
NUREG/CR-0980 (ORNL/NUREG/TM-347)
NUREG/CR-1197 (ORNL/NUREG/TM-370)
NUREG/CR-1305 (ORNL/NUREG/TM-380)
NUREG/CR-1477 (ORNL/NUREG/TM-393)
NUREG/CR-1627 (ORNL/NUREG/TM-401)
NUREG/CR-1806 (ORNL/NUREG/TM-419)
NUREG/CR-1941 (ORNL/NUREG/TM-437)
NUREG/CR-2141, Vol. 1 (ORNL/TM-7822)
NUREG/CR-2141, Vol. 2 (ORNL/TM-7955)
NUREG/CR-2141, Vol. 3 (ORNL/TM-8145)
NUREG/CR-2141, Vol. 4 (ORNL/TM-8252)
NUREG/CR-2751, Vol. 1 (ORNL/TM-8369/V1)
NUREG/CR-2751, Vol. 2 (ORNL/TM-8369/V2)
NUREG/CR-2751, Vol. 3 (ORNL/TM-8369/V3)
NUREG/CR-2751, Vol. 4 (ORNL/TM-8369/V4)

NUREG/CR-3334, Vol. 3 (ORNL/TM-8787/V3)
NUREG/CR-3744, Vol. 1 (ORNL/TM-9154/V1)
NUREG/CR-3744, Vol. 2 (ORNL/TM-9154/V2)
NUREG/CR-4219, Vol. 1 (ORNL/TM-9593/V1)
NUREG/CR-4219, Vol. 2 (ORNL/TM-9593/V2)
NUREG/CR-4219, Vol. 3, No. 1 (ORNL/TM-9593/V3&N1)
NUREG/CR-4219, Vol. 3, No. 2 (ORNL/TM-9593/V3&N2)
NUREG/CR-4219, Vol. 4, No. 1 (ORNL/TM-9593/V4&N1)
NUREG/CR-4219, Vol. 4, No. 2 (ORNL/TM-9593/V4&N2)
NUREG/CR-4219, Vol. 5, No. 1 (ORNL/TM-9593/V5&N1)
NUREG/CR-4219, Vol. 5, No. 2 (ORNL/TM-9593/V5&N2)
NUREG/CR-4219, Vol. 6, No. 1 (ORNL/TM-9593/V6&N1)
NUREG/CR-4219, Vol. 6, No. 2 (ORNL/TM-9593/V6&N2)

Summary

1. Program Management

The Heavy-Section Steel Irradiation (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 (LUS) welds, (6) annealing effects in LUS welds, (7) irradiation effects in a commercial LUS weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) Japan Power Development Reactor (JPDR) steel examination, (13) technical assistance for Joint Coordinating Committee for Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, and (14) additional requirements for materials. Progress reports are issued on a semiannual basis, and the report chapters correspond to the tasks. The work is performed by Oak Ridge National Laboratory (ORNL). During the report period, 19 program briefings, reviews, and presentations were made, and 11 technical documents were published.

2. K_{Ic} Curve Shift in High-Copper Welds

The overall objectives of the Fifth Irradiation Series are to determine the K_{Ic} curve shifts and shapes for two irradiated high-copper, submerged-arc welds (SAWs). All planned unirradiated and irradiated testing for Phase I of the Fifth Irradiation Series has been completed and the results reported. The objective of Phase II of this series is to obtain postirradiation fracture toughness data to a neutron fluence of 5 x 10^{19} neutrons/cm^2 (>1 MeV) to examine if the trends observed at lower fluences are exacerbated for greater exposures. During this reporting period, a preliminary matrix has been developed for conduct of high-fluence irradiations on HSSI weld 73W, the higher copper of the two welds examined in the first phase of the Fifth Series.

3. K_{Ia} Curve Shift in High-Copper Welds

The objectives of the Sixth Irradiation Series are to determine the K_{Ia} curve shifts and shapes for two high-copper SAWs. The results of Phase I of the irradiated testing are reported, and the data were compared with unirradiated crack-arrest data. During this reporting period, 24 duplex-type crack-arrest specimens from Phase II of the K_{Ia} program were tested and preparation of a detailed report describing them begun. In addition, a new, remote crack-arrest fixture required for collaborative testing of large irradiated specimens from the Italian Committee for Research and Development of Nuclear Energy and Alternative Energies (ENEA) was built.

4. Irradiation Effects in Cladding

The objective of this series is to obtain toughness properties of stainless steel cladding in the unirradiated and irradiated conditions. The properties obtained include tensile, Charpy V-notch (CVN) impact, and J-integral toughness. The goal is to evaluate the fracture resistance of irradiated weld-metal cladding representative of that used in early pressurized-water reactors. The fracture properties are needed for detailed integrity analyses of vessels during overcooling situations. There was no significant activity within this task during this reporting period.

5. K_{Ic} and K_{Ia} Curve Shifts in LUS Welds

The primary objective of Series 8 is to examine the K_{Ic} and K_{Ia} for LUS high-copper weld metal irradiated at 288°C, with particular emphasis on the shift and change in shape of the American Society of Mechanical Engineers curves following irradiation. During the current reporting period, an initial assessment was made of the welding consumables to be used in the fabrication of the high-copper LUS weld to be examined, and plans were made to produce a test weld to ascertain its experimental suitability.

6. Annealing in LUS Welds

The purpose of the Ninth Irradiation Series is to evaluate the correlation between fracture toughness and CVN impact energy during irradiation, annealing, and reirradiation. During the current reporting period, the 72 CVN specimens that were irradiated with the Fifth Irradiation Series were annealed, and 18 of them have been tested. The upper-shelf energy (USE) of irradiated/annealed CVN specimens was shown to be approximately 25% greater than the unirradiated USE, and the 41-J transition temperature recovered approximately 90%.

7. Irradiation Effects in a Commercial LUS Weld

The primary objective of Series 10 is to investigate the postirradiation fracture toughness of the LUS high-copper SAW from the beltline region of the Midland Unit 1 reactor vessel. The weld from that vessel is of considerable interest because it carries the Babcock and Wilcox designation WF-70, an SAW fabricated with a specific heat of weld wire and specific lot of flux. Welds with the WF-70 designation are the controlling material (regarding irradiation effects) in several operating nuclear plants. During the current reporting period, the tests for fracture mechanics evaluation of the subject materials in the unirradiated condition have been nearly completed. There are a few more crack-arrest K_{Ia} tests to be completed. The static fracture mechanics data have been evaluated using an analysis procedure developed for ductile-brittle behavior under the auspices of the American Society for Testing and Materials (ASTM). This analysis method indicated that the nozzle material had about 17°C higher transition temperature than the beltline material. For crack-arrest K_{Ia} testing, a new, brittle weld crack-starter bead material (McKay DWT) has been qualified to replace the MUREX Hardex-N that had been widely used in the past but is no longer available. Irradiation work has been progressing on schedule. The exposure of capsule 10.05 (1T) is complete, and capsule 10.06 (2T) is awaiting exposure in the Ford Nuclear Reactor site in Michigan. Disassembly of capsule 10.05 will commence following identification of the location for that activity.

8. Microstructural Analysis of Irradiation Effects

The overall, long-term goal of this task is to develop a physically based model that 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. During this reporting period, detailed measurements were made of the spectrum at the surveillance specimen locations of the High Flux Isotope Reactor (HFIR); experiments were begun to investigate the effects of a very wide range of flux levels on embrittlement at low temperatures as was the design of a high-temperature irradiation capsule for similar experiments; in-cavity irradiations of vessel support materials were completed; unirradiated microstructural characterization of a Russian VVER-440 reactor vessel steel was completed; and collaborative investigations of in-cascade, irradiation-induced point-defect generation were initiated.

9. In-Service Aged Material Evaluations

The overall objective of this task is to assess the service-induced degradation of fracture resistance through examination of components exposed during in-nuclear-plant operation. The initial focus of this task is to augment the existing hot cell testing capability available to the HSSI Program with remote machining capabilities for the fabrication of specimens from samples of activated steel obtained from service-exposed components. Toward that end, the specification for a computer numerically controlled vertical milling machine was issued for bid during this reporting period. It is anticipated that the machine will be delivered and evaluated during the next reporting period.

10. Correlation Monitor Materials

The purpose of this task is to provide for the long-term, archival quality storage of the remaining correlation monitor materials at ORNL and to establish a means for their Nuclear Regulatory Commission (NRC)-approved disbursement to qualified recipients. No significant activity has occurred within this task since the detailed inventory was made of all available materials late during the previous reporting period.

11. Special Technical Assistance

This task has been included within the HSSI Program to provide a vehicle in which to conduct and monitor short-term, high-priority subtasks. During the current reporting period, an evaluation was made of the results of refined calculations and detailed experimental measurements of the exposure parameters that existed at the various surveillance capsule locations of the HFIR; the two organizations were selected and arrangements initiated for them to test round, notched and precracked fracture toughness specimens to evaluate their potential for reactor surveillance applications; the evaluation of annealing technology and materials studies from Russia continued; and the baseline testing for an ASTM round robin on reconstituted CVN specimens was completed.

12. 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. Informal agreement was reached during the current reporting period on the details of the agreement under which all the HSSI research will be conducted in collaboration with related studies being performed by the Japan Atomic Energy Research Institute. It is anticipated that the agreement will be signed and work initiated during the next reporting period.

13. Technical Assistance for JCCCNRS Working Groups 3 and 12

The purpose of this task is to provide technical support for the efforts of the Joint U.S.-Russian Coordinating Committee on Civilian Nuclear Reactor Safety 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 irradiation, annealing, and reirradiation 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) initiation of the 1-year assignment of M. A. Sokolov of the Russian National Research Center-Kurchatov Institute at ORNL, which began in late December 1992. Regarding irradiation experiments in the host country, ORNL had supplied CVN and tensile specimens of two materials which have been encapsulated and placed into irradiation positions in the Novovoronezh Unit 5 reactor vessel. Additionally, the Russians provided two weld materials which were machined into similar specimens for inclusion in the second large irradiation capsule of the Tenth Irradiation Series (see Task 7). During the period from September 24 to October 2, 1992, R. K. Nanstad participated in the fourth information exchange of Working Group 3 of the JCCCNRS. The meetings were held in St. Petersburg and Moscow, Russia.

14. Additional Requirements for Materials

The purpose of this task is to 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 includes proposed modifications and supplementary information as required in the areas of design, materials, fabrication, testing, and inspection. Activity during the current reporting period was focused on the stainless steel (SS) weld aging task. Three SS welds with varying ferrite contents have been aged at 343°C for up to 20,000 h. Additional material is currently being aged to reach 50,000 h. Tensile and CVN specimens were tested for each of the welds for aging times up to 20,000 h.

Heavy-Section Steel Irradiation Program Semiannual Progress Report for October 1992 - March 1993*

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 (LWR) pressure-vessel integrity. The program centers on experimental assessments of irradiation-induced embrittlement [including the completion of certain irradiation studies previously conducted by the Heavy-Section Steel Technology (HSST) Program] 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 nil-ductility transition (NDT), and tensile properties are included. Models based on observations of radiation-induced microstructural changes using the field-ion microprobe and the high-resolution transmission electron microscope 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 reactor pressure vessels (RPVs). In addition, a limited effort will focus on stainless steel (SS) 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 that 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 both the aging behavior and the potential for plant life extension of LWR pressure vessels.

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 one task responsible for overall program management and 13 technical 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 LUS welds, (6) annealing effects in LUS welds, (7) irradiation effects in a commercial LUS weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) Japan Power Development Reactor (JPDR) steel examination, (13) technical assistance for Joint Coordinating

*Research sponsored by the Office of Nuclear Regulatory Research, Division of Engineering, U.S. Nuclear Regulatory Commission, under Interagency Agreement DOE 1886-8109-8L with the U.S. Department of Energy under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

Committee for Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, and (14) additional requirements for materials. Accordingly, the chapters of this progress report correspond to these 14 tasks.

During this period, 11 program briefings, reviews, or presentations were made by the HSSI staff during program reviews and visits with NRC staff or others. Two topical reports,[1,2] two foreign trip reports,[3,4] and six technical papers[5-10] were published, and one letter report was submitted to NRC.[11] In addition, eight technical presentations were made: two[12,13] at the Twentieth Water Reactor Safety Information Meeting in Rockville, Maryland, October 21, 1993; five[14-18] at the International Group on Radiation Damage Mechanisms, Fountainebleau, France, November 16-20, 1992; and one[19] at Berkeley Nuclear Laboratories, Berkeley, United Kingdom, November 23, 1992.

References

1. R. K. Nanstad, F. M. Haggag, D. E. McCabe, S. K. Iskander, K. O. Bowman, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and B. H. Menke, Material Engineering Associates, Lanham, Md., *Irradiation Effects on Fracture Toughness of Two High-Copper Submerged-Arc Welds, HSST Series 5*, NUREG/CR-5913, Vols. 1 and 2 (ORNL/TM-12156/V1&V2), October 1992.*

2. R. K. Nanstad, D. E. McCabe, R. L. Swain, and M. K. Miller, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Chemical Composition and* RT_{NDT} *Determinations for Midland Weld WF-70*, NUREG/CR-5914 (ORNL-6740), December 1992.*

3. R. K. Nanstad, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Report of Foreign Travel*, ORNL/FTR-4399, October 19, 1992.*

4. R. K. Nanstad and R. E. Stoller, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Report of Foreign Travel*, ORNL/FTR-4486, December 28, 1992.*

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

5. W. R. Corwin, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., "Managing Irradiation Embrittlement in Aging Reactor Pressure Vessels," pp. 2-1 to 2-2 in *Transactions of the U.S. Nuclear Regulatory Commission Twentieth Water Reactor Safety Information Meeting*, USNRC Report NUREG/CP-0125, October 1992.*

6. K. Farrell et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., "Investigation of Low-Temperature Neutron Embrittlement of Ferritic Steels," pp. 2-7 to 2-8 in *Transactions of the U.S. Nuclear Regulatory Commission Twentieth Water Reactor Safety Information Meeting*, USNRC Report NUREG/CP-0125, October 1992.*

7. P. A. Simpson, "A Large-Scale Irradiation Facility for the HSSI Program," pp. 161-63 in *Transactions of the International Conference on Fifty Years of Controlled Nuclear Chain Reaction: Past, Present, and Future,* Vol. 66, TANSAO 66 1-626, American Nuclear Society, Hinsdale, Ill., 1992.+

8. W. A. Pavinich, W. L. Server, and R. K. Nanstad, "Application of Reference Toughness Curves to the HSSI Fifth Irradiation Series," pp. 238-50 in *Effects of Radiation on Materials, 15th International Symposium, ASTM STP 1125*, ed. R. E. Stoller, A. S. Kumar, and D. S. Gelles, American Society for Testing and Materials, Philadelphia, 1992.+

9. S. K. Iskander, W. R. Corwin, and R. K. Nanstad, "Effects of Irradiation on Crack-Arrest Toughness of Two High-Copper Welds," pp. 251-69 in *Effects of Radiation on Materials, 15th International Symposium, ASTM STP 1125*, ed. R. E. Stoller, A. S. Kumar, and D. S. Gelles, American Society for Testing and Materials, Philadelphia, 1992.+

10. R. K. Nanstad, D. E. McCabe, F. M. Haggag, K. O. Bowman, and D. J. Downing, "Statistical Analyses of Fracture Toughness Results for Two Irradiated High-Copper Welds," pp. 270-91 in *Effects of Radiation on Materials, 15th International Symposium, ASTM STP 1125*, ed. R. E. Stoller, A. S. Kumar, and D. S. Gelles, American Society for Testing and Materials, Philadelphia, 1992.+

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

+Available in public technical libraries.

11. Letter report, "The DOS 1 Neutron Dosimetry Experiment at the HB-4-A Key 7 Surveillance Site on the HFIR Pressure Vessel," by K. Farrell et al., from W. R. Corwin, Oak Ridge Natl. Lab., to M. E. Mayfield, NRC, February 12, 1993.**

12. K. Farrell, "Investigations of Low-Temperature Neutron Embrittlement of Ferritic Steels," presented at the 20th Water Reactor Safety Information Meeting, Bethesda, Maryland, October 21, 1992.

13. W. R. Corwin, "Managing Irradiation Embrittlement in Aging Reactor Pressure Vessels," presented at the 20th Water Reactor Safety Information Meeting, Bethesda, Maryland, October 21, 1992.

14. K. Farrell, F. B. Kam, C. A. Baldwin, F. W. Stallmann, L. Robinson, F. F. Dyer, J. V. Pace, III, F. M. Haggag, and B. M. Oliver, "Studies Regarding Neutron Spectrum Characterization for High Flux Isotope Reactor Surveillance Program," presented by R. K. Nanstad at the meeting of the International Group on Radiation Damage Mechanisms, Fountainebleau, France, November 16-20, 1992.

15. K. Farrell and S. T. Mahmood, "Low-Temperature Irradiation of RPV Steels and Model Alloys in the HFIR Hydraulic Tube," presented by R. E. Stoller at the meeting of the International Group on Radiation Damage Mechanisms, Fountainebleau, France, November 16-20, 1992.

16. R. K. Nanstad and S. K. Iskander, "Statistical Analysis of Fracture Toughness and Crack-Arrest Toughness Results for Two Irradiated High-Copper Welds," presented by R. K. Nanstad at the meeting of the International Group on Radiation Damage Mechanisms, Fountainebleau, France, November 16-20, 1992.

17. R. E. Stoller, "Modeling the Effects of Displacement Rates under Irradiation," presented at the meeting of the International Group on Radiation Damage Mechanisms, Fountainebleau, France, November 16-20, 1992.

18. R. E. Stoller, "Modeling the Effects of Transients under Irradiation," presented at the meeting of the International Group on Radiation Damage Mechanisms, Fountainebleau, France, November 16-20, 1992.

19. R. E. Stoller, "Modeling Embrittlement in Ferritic Steels: Effects of Atomic Displacement Rate and Point Defect Transients," presented at Berkeley Nuclear Laboratories, Berkeley, United Kingdom, November 23, 1992.

**Available in NRC PDR for inspection and copying for a fee.

2. K_{Ic} Curve Shift in High-Copper Welds

R. K. Nanstad

The objectives of the Fifth Irradiation Series are to determine the K_{Ic} curve shifts and shapes for two irradiated high-copper, 0.23 and 0.31 wt %, submerged-arc welds (SAWs) [72W and 73W, respectively]. All planned unirradiated and irradiated testing for Phase I of the Fifth Irradiation Series has been completed. The results from statistical analyses and curve fitting of the fracture toughness results, including specimen size effects and effects of precleavage stable ductile tearing, have been reported in *Irradiation Effects on Fracture Toughness of Two High-Copper Submerged-Arc Welds, HSSI Series 5*, NUREG/CR 5913 (ORNL/TM-12156/V1), published in October 1992.

The objective of Phase II of this series is to obtain postirradiation fracture toughness data to a neutron fluence of 5×10^{19} neutrons/cm^2 (>1 MeV). A preliminary matrix has been developed for conduct of irradiations with HSSI weld 73W. The plan envisions multiple testing of 25.4-mm-thick (1-in.) compact specimens in both the unirradiated and irradiated conditions at multiple temperatures in the transition temperature region. The detailed plan for such testing will take advantage of statistical design concepts. CVN and tensile tests will also be tested in the irradiated condition. The results will be used to compare transition temperature shifts and determine curve shape changes as a consequence of the higher fluence compared to that used in Phase I.

3. K_{Ia} Curve Shift in High-Copper Welds

S. K. Iskander, R. K. Nanstad, and E. T. Manneschmidt

In the fracture mechanics integrity analysis of RPVs, the initiation and arrest fracture toughness curves described in Sect. XI of the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code* are often used. These curves are also used for the normal operation of RPVs. The effects of neutron irradiation on toughness are accounted for by shifting the curves upward in temperature without change in shape by an amount equal to the shift of the CVN impact energy curve at the 41-J level. Such a procedure implies that the shifts in the fracture toughness curves are the same as those of the CVN 41-J energy level and that irradiation does not change the shapes of the fracture toughness curves.

The primary objective of the HSSI Sixth Irradiation Series (for brevity, the K_{Ia} program) is to determine the effect of irradiation on the shift and shape of the K_{Ia} versus ($T - RT_{NDT}$) curve. The objective of the HSSI Fifth Irradiation Series (the K_{Ic} program) is similar but determines the effect of irradiation on K_{Ic}. One of the significant results of the K_{Ic} program is that a decrease in slope of the irradiated initiation toughness curve for the 73W weldment has been observed, and for which a decrease in slope of CVN impact energy curve was also observed.[1]

3.1 Testing of the Sixth Series Irradiated Crack-Arrest Specimens

There were 36 weld-embrittled and 24 duplex-type specimens irradiated for the K_{Ia} program. The 36 weld-embrittled specimens have already been tested in Phase I of the K_{Ia} program, and a detailed report has been published.[2] A summary of the objectives of the program, the materials, the specimens used, and the results have also been reported.[3,4] In Phase II of the K_{Ia} program, 24 duplex-type crack-arrest specimens were tested, and a detailed report is under preparation.

3.2 Preparations for Testing Irradiated Crack-Arrest Specimens Supplied by ENEA

A new, remote crack-arrest fixture for testing the large irradiated specimens from the Italian Committee for Research and Development of Nuclear Energy and Alternative Energies (ENEA) has been built and is currently being tested. The impetus for building the new fixture is that the NRC has agreed to test the irradiated crack-arrest specimens for ENEA at ORNL. The results will have usefulness and applicability to the safety assessment of U.S. RPVs.

ENEA started an extensive research program some time ago to characterize an American Society for Testing and Materials (ASTM) A 508 class 3 forging produced in Italy. The research program encompassed both unirradiated and irradiated mechanical property data from the following types of specimens: tensile, Charpy impact (both standard and precracked), compact tensile, and crack arrest. The testing and irradiation of these specimens was performed at several locations: ENEA CRE Casaccia Laboratories, Battelle Columbus Laboratories (BCL), and two laboratories of the French Commissariat a l'Energie Atomique. ENEA originally planned to test the irradiated crack-arrest specimens at BCL, but BCL has recently decommissioned their hot cell facilities.

ENEA has nine irradiated crack-arrest specimens manufactured from the ASTM A 508 class 3 forging. The dimensions of three of the specimens are 25 × 200 × 200 mm, and the remaining six are 13 × 100 × 100 mm. The fluence of the nine specimens varied between approximately 2 to 3.2×10^{19} neutrons/cm^2 (>1 MeV), and the irradiation temperature varied from 240 to 280°C.

The nine irradiated ENEA crack-arrest specimens have been received and stored at ORNL. The specimens were located at the Ford Nuclear Reactor (FNR) in Michigan. The remote fixture used in the testing of the HSSI Sixth Irradiation Series is too small to be used with the three large specimens. There are other details about the ENEA specimens that will also require modifications to the presently available equipment. One example of such a difference is that these specimens have been designed to use knife edges located on the front face for measuring the crack-mouth opening displacements (CMOD). These knife edges are not an integral part of the specimen and will have to be attached remotely. With irradiated specimens, ORNL has previously used integrally attached clip gage blocks with conical recesses that receive a clip gage with conical points. In order to test the ENEA specimens, if a conical-point CMOD gage is used, the CMOD measurements cannot be made at the a/W = 0.25 location prescribed in the crack-arrest standard. It is still possible to obtain crack-arrest data by using an appropriate compliance relationship. The other possibility is to use specially machined knife-edge blocks. Both approaches are being considered. ORNL is in touch with BCL to keep them apprised of progress with the ENEA specimens.

References

1. R. K. Nanstad, F. M. Haggag, and S. K. Iskander, "Radiation-Induced Temperature Shift of the ASME K_{Ic} Curve," pp. 143-48 in *Transactions of the 10th International Conference on Structural Mechanics in Reactor Technology (SMiRT)*, Vol. S, ed. A. H. Hadgjian, American Association for Structural Mechanics in Reactor Technology, Los Angeles, 1989.*

2. S. K. Iskander, W. R. Corwin, and R. K. Nanstad, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Results of Crack-Arrest Tests on Two Irradiated High-Copper Welds*, USNRC Report NUREG/CR-5584 (ORNL/TM-11575), December 1990.†

3. S. K. Iskander et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab, "K_{Ia} Curve Shift in High-Copper Welds," pp. 11-16 in *Heavy-Section Steel Irradiation Program Semi-annual Progress Report October 1989-March 1990*, USNRC Report NUREG/CR-5591, Vol. 1, No. 1 (ORNL/TM-11568/V1&N1), August 1990.†

4. S. K. Iskander, W. R. Corwin, and R. K. Nanstad, "Effects of Irradiation on Crack-Arrest Toughness of Two High-Copper Welds," pp. 251-69 in *Effects of Radiation on Materials: 15th International Symposium, ASTM STP 1125*, ed. R. E. Stoller, A. S. Kumar, and D. S. Gelles, American Society for Testing and Materials, Philadelphia, 1992.*

*Available in public technical libraries.

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

4. Irradiation Effects on Cladding

F. M. Haggag

The objective of this series is to obtain toughness properties of stainless steel cladding in the unirradiated and irradiated conditions. The properties obtained include tensile, Charpy V-notch (CVN) impact, and J-integral toughness. The goal is to evaluate the fracture resistance of irradiated weld-metal cladding representative of that used in early pressurized-water reactors. The fracture properties are needed for detailed integrity analyses of vessels during overcooling situations. There was no significant activity within this task during this reporting period.

5. K_{Ic} and K_{Ia} Curve Shifts in LUS Welds

R. K. Nanstad, D. E. McCabe, and S. K. Iskander

The objectives of the HSSI Eighth Irradiation Series are to evaluate the irradiation-induced temperature shifts and shape changes of the K_{Ic} and K_{Ia} curves for high-copper, low CVN upper-shelf welds. These objectives are similar to those of the Fifth and Sixth Irradiation Series,[1,2] which were conducted with high-copper SAWs with relatively high CVN upper-shelf energies (USEs). The welds of the Fifth Series, designated 72W and 73W, contained copper contents of 0.23 and 0.31 wt %, respectively, and gave CVN USEs of about 135 J (100 ft-lb) in the unirradiated condition. A number of SAWs fabricated by Babcock and Wilcox (B&W) with Linde 80 welding flux and copper-coated welding wire exhibit CVN USEs of about 100 J (75 ft-lb) and less in the unirradiated condition. The low USEs of these welds derive from the very high number of small, nonmetallic inclusions contained in the welds. Furthermore, because of the relatively high copper contents, many of those welds exhibit relatively high irradiation-induced CVN transition temperature shifts and USE decreases.

In the unirradiated condition, the high CVN USE welds 72W and 73W gave reference temperatures (RT_{NDT}s) that were equal to the drop-weight nil-ductility temperatures (NDTs), meaning that the CVN energies and lateral expansions at NDT + 60°F (33°C) exceeded 50 ft-lb (68 J) and 0.035 in. (0.89 mm) in the manner prescribed by the *ASME Boiler and Pressure Vessel Code.*[3] In general, the LUS welds tend to give RT_{NDT}s which are controlled by the CVN behavior, meaning that the energy and lateral expansion requirements described above are met at higher temperatures above the drop-weight NDTs. In those cases, RT_{NDT} for each weld is determined by subtracting 60°F (33°C) from the temperature at which both the energy and lateral expansion requirements are met.[4] Because of the lower USEs, the CVN energy versus temperature curves for these materials tends to have lower slopes. Because of the high inclusion content and the relatively lower CVN curve slopes, the concern is that the fracture toughness versus temperature curves exhibited by these welds will also be of lower slopes than those exhibited by SAWs with higher CVN USEs.

Initial consideration of the low USE WF-70 weld from the Midland Unit 1 reactor vessel (see Task 7) was rejected because of the wide variation in copper content exhibited by that weld. For the irradiated condition, the concern is that such variations in copper content would confound determination of fracture toughness curve shape changes. Thus, the current plan for the Eighth Series envisions the fabrication of a weld with bare high-copper wire (no copper coating) and Linde 80 flux. One potential option under consideration is to use the same weld wire used for weld 73W. Review of the existing inventory of that weld wire is under way. If sufficient weld wire is available, a trial weld will be fabricated using Linde 80 welding flux. The trial weld will be tested to evaluate its tensile and impact properties. As part of this evaluation, the fracture toughness, crack-arrest toughness, and CVN test results from the unirradiated Midland reactor welds (beltline and nozzle welds are both designated WF-70) will be evaluated. This evaluation will compare the relationships between the various test results for the low USE welds with those shown by the high USE welds 72W and 73W. The plan for this task is under development with a view toward preparation of a specification for procurement of a specially fabricated low USE weld.

References

1. R. K. Nanstad, F. M. Haggag, D. E. McCabe, S. K. Iskander, K. O. Bowman, and B. H. Menke, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Irradiation Effects on Fracture Toughness of Two High-Copper Submerged-Arc Welds, HSSI Series 5*, USNRC Report NUREG/CR-5913, Vol. 1 (ORNL/TM-12156/V1), August 1992.*

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

2. S. K. Iskander, W. R. Corwin, and R. K. Nanstad, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Results of Crack-Arrest Tests on Two Irradiated High-Copper Welds*, USNRC Report NUREG/CR-5584 (ORNL/TM-11575), December 1990.*

3. *ASME Boiler and Pressure Vessel Code. An American National Standard*, Sect. III, American Society of Mechanical Engineers, New York, 1989.†

4. R. K. Nanstad, D. E. McCabe, R. L. Swain, and M. K. Miller, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab, *Chemical Composition and RT_{NDT} Determinations for Midland Weld WF-70*, USNRC Report NUREG/CR-5914 (ORNL/TM 12157), December 1992.*

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

†Available in public technical libraries.

6. Annealing Effects in LUS Welds

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.

Eighteen irradiated/annealed (I/A) CVN specimens were recently tested. The USE of I/A CVN specimens is approximately 25% greater than the unirradiated USE. To determine if this increase in USE is due to thermal aging, unirradiated specimens will be aged at 454°C (850°F) for 168 h. The 41-J energy level transition temperature, TT_{41J}, of the I/A undersize CVN specimens has recovered approximately 90%, and the USE has recovered 125%. Some background and more detailed results are given below.

The material used in this study is the HSSI 73W weld metal that has been used in two other major NRC HSSI studies: the Fifth and Sixth Irradiation Series. There were approximately 200 undersize CVN specimens machined for this task. The thickness of the undersize specimens is approximately 95% of the full-size ones, i.e., the total thickness of the specimen normal to the notch is approximately 9.5 mm rather than 10 mm of the "standard" or full-size specimen. Also, the ratio of the notch-to-specimen depth is 0.191 instead of 0.2. All other dimensions are the same as those of a standard CVN specimen. The CVN specimens were machined undersized so they would fit, for purposes of irradiation, in the grooves of the 4T C(T) specimens of the Fifth Irradiation Series.

Figure 1 compares the results of Charpy energy tests for both the undersized and normal specimens. There are apparently three differences between the results of the two specimen sizes. The most prominent difference is that the USE of the undersize specimens is approximately 87% of that of the full-sized ones. It also appears that the slope of the undersize specimens is steeper than the full-sized ones. However, the 41-J transition temperature of both sizes of specimens is about the same.

Seventy-two of the 120 specimens that were irradiated together with the Fifth Irradiation Series were annealed at approximately 454°C (850°F) for 168 h (1 week). Eighteen of these specimens were tested, and the results are compared to the unirradiated results in Fig. 2. One noticeable result is that the USE of the I/A specimens is approximately 25% higher than the unirradiated specimens. In fact, the USE of the undersize specimens is approximately 10% higher than the unirradiated full-sized ones.

The TT_{41J} has also recovered to within approximately 7 K of the unirradiated value. Test results from irradiated undersize specimens are not yet available, but the TT_{41J} percent recovery can be estimated as follows (bearing in mind the differences between full-size and undersize specimens mentioned above). The TT_{41J} shift of the Fifth Series full-sized specimens is 82 K. The TT_{41J} of the undersized specimens in the unirradiated and I/A conditions is −38 and −31°C, respectively, and therefore the TT_{41J} percent recovery is approximately 90%.

References

1. R. K. Nanstad, F. M. Haggag, D. E. McCabe, S. K. Iskander, K. O. Bowman, and B. H. Menke, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Irradiation Effects on Fracture Toughness of Two High-Copper Submerged-Arc Welds, HSSI Series 5*, USNRC Report NUREG/CR-5913, Vol. 1 (ORNL/TM-12156/V1), August 1992.*

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

ORNL-DWG 93-5853

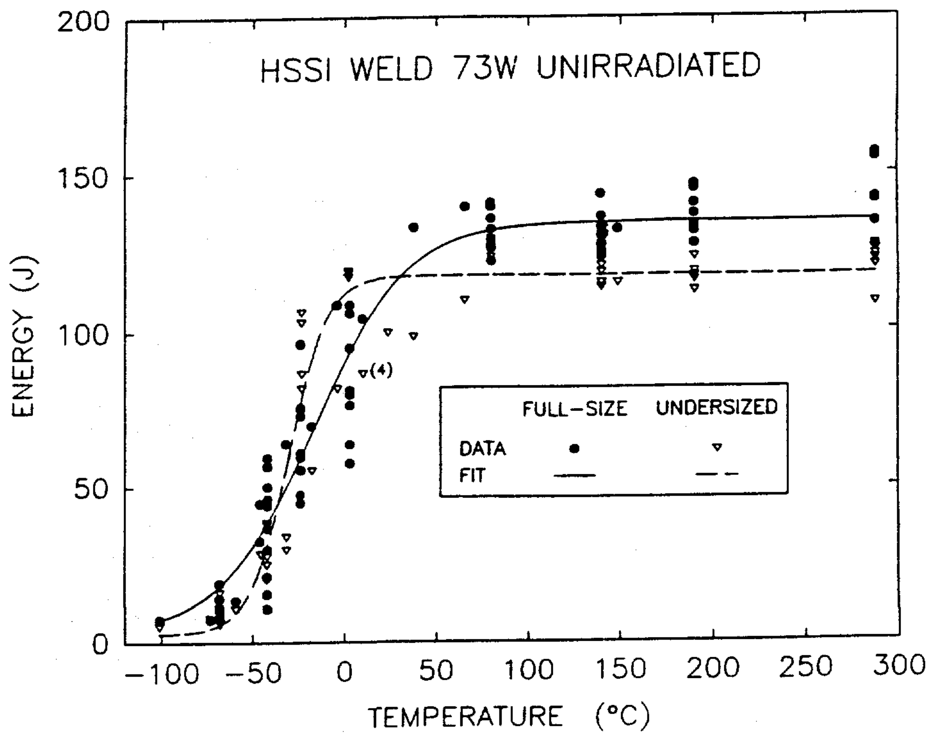


Figure 1. Charpy V-notch impact energy of unirradiated Heavy-Section Steel Irradiation 73W weld of full-size (10 by 10 mm) and undersize (9.5 by 10 mm) specimens.

ORNL-DWG 93-5854

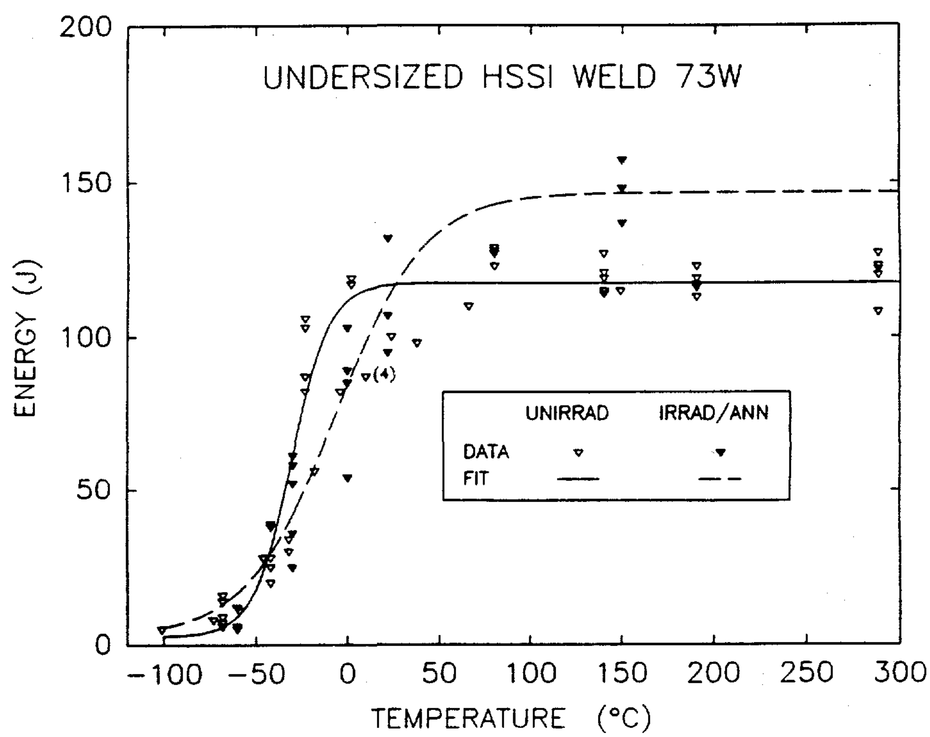


Figure 2. Comparison of the unirradiated and irradiated/annealed Charpy V-notch impact energy of undersize specimens.

7. Irradiation Effects in a Commercial LUS Weld

D. E. McCabe and S. K. Iskander

A report, *Chemical Composition and RT_{NDT} Determinations for Midland Weld WF-70*, NUREG/CR-5914 (ORNL/TM-12157) by R. K. Nanstad, D. E. McCabe, R. L. Swain, and M. K. Miller has been completed. This report documents the basic characterization of a commercially produced LUS pressure-vessel weld material, WF-70, using routine evaluation methods. Weld WF-70 is a B&W designation given to a specific heat of weld wire (72105) and weld flux lot (Linde 80 lot 8669). The material comes from an abandoned reactor construction project of Consumers Power Utility of Midland, Michigan. The vessel was sampled at one-quarter circumference positions around the beltline course weld and at about one-half circumference positions around the nozzle course weld for chemical composition, CVN toughness, and drop-weight NDT properties. The gradient of properties in the through-thickness direction was also determined. The nozzle course weld was WF-67 filler on the inside of a double-V weld and WF-70 filler on the outside half, the former of which was not of interest to the present project.

Table 1 shows the typical through-thickness chemical composition for the beltline weld material. There is excellent uniformity of all chemical elements except for copper. Copper variability is affected by the two welding parameters of filler metal/base metal mixing within the weld pool and the thickness of the copper coating on the electrode wire. Table 2 summarizes the copper variation plus the content of other radiation-sensitive elements for the various sampling positions about the vessel. The beltline weld metal had an average copper content of 0.26 wt %, varying between 0.21 to 0.34%. The nozzle weld was nominally 0.40 wt %, varying between 0.37 and 0.46%.

Charpy transition temperature toughness was evaluated at three energy levels: 41 J, 68 J, and upper shelf. These results are summarized for the beltline weld in Table 3. Codes 1-9, 1-11, 1-13, and 1-15 represent the one-quarter girth positions. Note the wide scatter in RT_{NDT} transition temperature (right-hand box). Because WF-70 is an LUS material, the RT_{NDT} is determined from the lower bound of CVN data scatter at the 68-J toughness level minus 33°C. The RT_{NDT} values listed in the last four columns of Table 3 show the lack of transition temperature definition from this method. There is a 57°C spread between the lowest and highest values. The same evaluation applied to the nozzle course weld metal had a 35°C spread, but here there were far fewer Charpy curves generated.

Drop-weight NDT temperatures were determined at the 1/4t and 3/4t (through-thickness) positions in beltline welds and 3/4t in nozzle weld WF-70. In both welds, the NDT temperature varied from a high of –40°C to a low of –60°C, giving an average overall determination of –50°C.

These results were compared to those of the Fifth Irradiation Series, where the welds had been prepared with extreme care. It was concluded that the scatter of toughness properties of the commercially produced weld WF-70 was about the same as that seen in the Fifth Series weld metal.

Other activities during the current reporting period were as follows: (1) completion of baseline unirradiated fracture mechanics tests, (2) baseline unirradiated tensile tests, (3) preparation and testing of crack-arrest specimens, (4) construction of capsule 10.06, (5) the irradiation exposure of capsule 10.05, and (6) preparation of a report on all baseline (unirradiated) properties.

Tensile properties have been determined on nozzle and beltline weld metals at six temperatures ranging from –100 to 288°C (see Table 4). It was interesting to note that the yield strength of the nozzle weld metal was significantly higher than that of the beltline weld, on the order of 100 MPa (15 ksi). However, it was noted that the point of reduction of area for the beltline weld was always at the same location, in the region of the weld fusion line. All tensile specimens were sampled with the parallel section transverse to the weld line. Since there was a possibility that this indicates a local weak zone that may have affected the weld metal strength characterization, a second set of

Table 1. Chemical composition of Midland reactor beltline section 1-9

Location	Chemical composition (wt %)																				
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	V	Nb	Ti	Co	Al	B	W	Cb	As	Sn	Zr	N	O
1/8t	0.082	1.67	0.015	0.006	0.57	0.11	0.58	0.40	0.24	0.005	-	<0.001	0.015	0.017	<0.001	<0.01	0.003	0.008	0.003	<0.001	0.016	0.062
	0.087	1.65	0.015	0.006	0.61	0.09	0.58	0.41	0.22	0.001	-											
1/4t	0.080	1.59	0.016	0.006	0.62	0.11	0.57	0.40	0.27	0.005	-	<0.001	0.015	0.015	<0.001	<0.01	0.003	0.007	0.003	<0.001	0.015	0.078
	0.076	1.66	0.018	0.005	0.62	0.09	0.57	0.41	0.22	0.002	-											
1/2t	0.095	1.51	0.014	0.008	0.55	0.15	0.62	0.44	0.22	0.005	-	<0.001	0.014	0.015	<0.001		0.003	0.005	0.003	<0.001	0.013	0.075
3/4t	0.078	1.64	0.016	0.007	0.61	0.11	0.58	0.40	0.30	0.005	-	<0.001	0.015	0.015	<0.001	<0.01	0.003	0.004	0.003	<0.001	0.018	0.060
	0.089	1.65	0.018	0.006	0.66	0.09	0.56	0.41	0.27	0.004	-											
7/8t	0.086	1.66	0.016	0.007	0.60	0.10	0.57	0.39	0.34	0.005	-	<0.001	0.015	0.016	<0.001	<0.01	0.003	0.006	0.003	<0.001	0.018	0.060

Source: Combustion Engineering, Inc.

Table 2. Statistical analyses of Midland reactor vessel shows widely varying copper contents in beltline and nozzle course welds

Section number	n[a]	Element (wt % ± 1σ)				
		Cu[b]	Ni	P	Mn	Si
Beltline weld						
1-9	8	0.26 ± 0.041 (0.22 to 0.34)	0.566 ± 0.031	0.016 ± 0.0013	1.629 ± 0.050	0.605 ± 0.031
1-11	8	0.258 ± 0.027 (0.23 to 0.31)	0.57 ± 0.007	0.016 ± 0.0014	1.615 ± 0.015	0.62 ± 0.029
1-13	5	0.248 ± 0.039 (0.21 to 0.32)	0.604 ± 0.016	0.018 ± 0.002	1.55 ± 0.067	0.62 ± 0.041
1-15	7	0.254 ± 0.026 (0.22 to 0.29)	0.567 ± 0.009	0.018 ± 0.0013	1.614 ± 0.014	0.644 ± 0.016
Average	28	0.256 ± 0.034 (0.21 to 0.34)	0.574 ± 0.023	0.017 ± 0.0019	1.607 ± 0.049	0.622 ± 0.033
Nozzle course weld						
3-1	4	0.398 ± 0.034 (0.37 to 0.46)	0.576 ± 0.021	0.015 ± 0.001	1.59 ± 0.045	0.548 ± 0.051
3-4	5	0.392 ± 0.016 (0.38 to 0.42)	0.567 ± 0.008	0.015 ± 0.002	1.61 ± 0.018	0.55 ± 0.043
Average	9	0.396 ± 0.028 (0.37 to 0.46)	0.572 ± 0.017	0.015 ± 0.002	1.59 ± 0.037	0.55 ± 0.048
Total average	37	0.290 ± 0.068 (0.21 to 0.46)	0.574 ± 0.022	0.016 ± 0.002	1.604 ± 0.046	0.605 ± 0.048

[a]Number of measurements.

[b]Range of copper shown in parentheses.

Table 3. Summary of Charpy impact results for Midland Unit 1 reactor vessel beltline weld sections

Through-thickness position	Charpy V-notch tests: 41-J temperature, °C (°F), at weld section				Charpy V-notch tests: 68-J temperature, °C (°F), at weld section				Charpy V-notch tests: Upper-shelf energy, J (ft-lb), at weld section				RT_{MI},[a] °C (°F), at weld section				RT_{NDT},[b] °C (°F), at weld section			
	1-13	1-9	1-11	1-15	1-13	1-9	1-11	1-15	1-13	1-9	1-11	1-15	1-13	1-9	1-11	1-15	1-13	1-9	1-11	1-15
1/4t	−11 (12)	−6 (21)	−13 (8)	4 (39)	21 (69)	37 (98)	25 (76)	50 (122)	101 (74)	77 (57)	91 (57)	82 (60)	−9 (15)	3 (37)	−9 (16)	16 (61)	−12 (9)	14 (57)	−9 (16)	16 (61)
1/2t	−16 (3)	−11 (13)	−4 (25)	−9 (15)	29 (84)	25 (77)	23 (74)	17 (63)	104 (77)	83 (61)	91 (67)	88 (65)	−5 (24)	−8 (17)	−10 (14)	−16 (3)	2 (36)	−8 (17)	−10 (14)	−15 (5)
5/8t	−22 (−7)	−18 (0)	−10 (13)	3 (37)	9 (48)	18 (64)	17 (63)	49 (121)	108 (80)	88 (65)	90 (66)	85 (62)	−25 (−12)	−16 (3)	−16 (3)	15 (60)	−20 (−3)	−16 (3)	−16 (3)	8 (47)
3/4t	−2 (27)	3 (38)	14 (57)	−6 (21)	37 (98)	53 (128)	58 (136)	28 (82)	90 (66)	81 (60)	84 (62)	89 (66)	3 (37)	20 (68)	24 (76)	−6 (21)	6 (43)	20 (68)	37 (99)	−6 (22)
7/8t		−3 (26)	−13 (8)	−8 (18)		46 (116)	30 (86)	22 (72)		78 (57)	79 (58)	83 (61)		13 (55)	−4 (25)	−12 (11)		13 (56)	18 (65)	−3 (26)

[a]Determined from T_{30} − 60°F (T_{68} − 33°C) using average curve fit, where T_{50} is the temperature corresponding to 50 ft-lb.

[b]Determined from T_{30} − 60°F (T_{68} − 33°C) using minimum curve fit, where T_{50} is the temperature corresponding to 50 ft-lb.

Table 4. Tensile properties of Midland weld metals

Temperature (°C)	Average value, tensile/automated ball indentation, MPa (ksi)	
	Yield strength	Ultimate tensile strength
	Nozzle	
-100	648 (94)	820 (119)
-75	620 (90)	765 (111)
-50	579 (84)	717 (104)
23	544/537 (79/78)	655/841 (95/122)
160	483 (70)	586 (85)
288	483 (70)	586 (85)
	Beltline	
-100	548 (79.5)	758 (110)
-75	483 (70)	710 (103)
-50	465 (67.5)	682 (99)
23	407/441 (59/64)	586/689(85/100)
288	427 (62)	558 (81)

beltline tensile specimens was made, oriented longitudinally to the weld line. These results will be presented in the upcoming unirradiated materials report.

All the fracture mechanics tests to characterize unirradiated material, with the exception of K_{Ia} tests, were completed. Compact specimen sizes ranged from 1/2T to 4T, and test temperatures ranged from −100 to 288°C. Data to characterize ductile-to-brittle transition temperature (DBTT) were evaluated using a test standard currently under development. Here, it was of interest to determine the position of a median fracture toughness transition curve (master curve) using only the data from six 1/2T C(T) specimens. The proposed practice provides guidelines for selection of an optimum test temperature from Charpy data. Since the nozzle and beltline materials had shown the same Charpy curve, there was only one test temperature selection for both materials, -50°C. The six 1/2T C(T) specimens of each material were tested at this temperature, and the "reference temperature" for the master curve was found to be -60°C for the beltline weld and -43°C for the nozzle weld. Figures 3 and 4 show the master curves along with the 5% probability curves that can be easily calculated using the proposed practice. All of the source data are plotted for comparison to the computed master and 5% confidence curves.

Thirty weld-embrittled crack-arrest specimens were prepared from the Midland weld. One-half of these specimens will be irradiated, and the other half are being tested as controls. A new stick electrode had to be qualified prior to preparation of these specimens. The MUREX Hardex-N stick electrode that was widely used for preparation of crack-starter beads of weld-embrittled crack-arrest specimens has not been available for many years. Previous experience with the MUREX Hardex-N stick electrode has shown that it takes time until a welder has the experience necessary to prepare successful crack-arrest specimens. The purpose of the qualification was to ensure that successful crack-starter beads can be made before committing considerable amounts of the scarce Midland material to the preparation of 30 specimens (one-half irradiated, one-half control). The orientation of crack-arrest and CVN specimens is such that the crack runs parallel to the welding direction (T-L orientation).

A number of trial crack-arrest specimens were prepared and tested with a qualification material using the McKay DWT stick electrode. The material is from a Hope Creek RPV circumferential

ORNL-DWG 93-5855

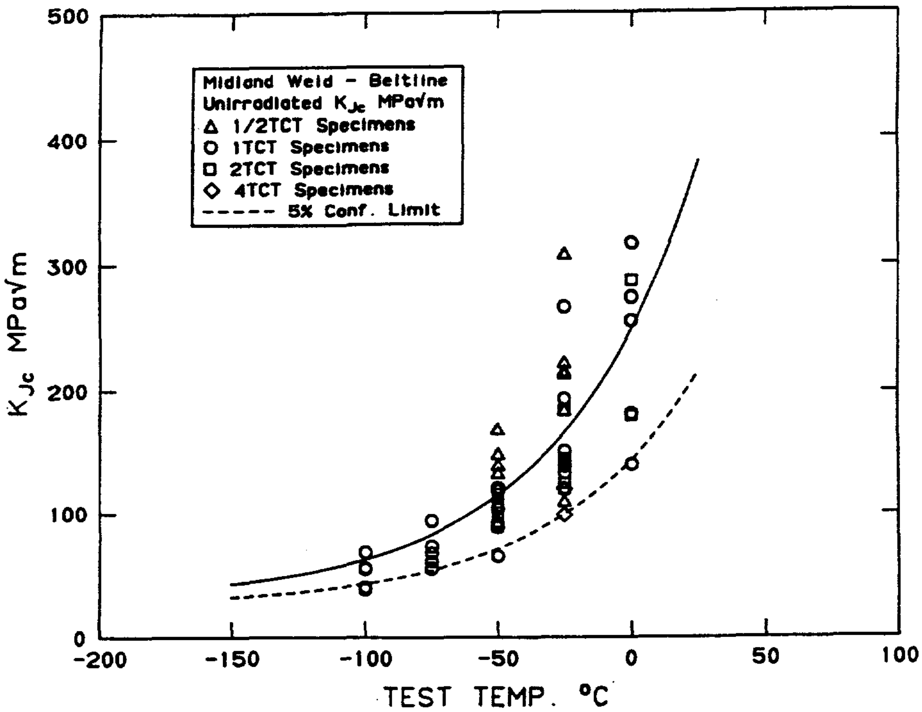


Figure 3. Master curve (solid line) and 5% confidence bound (dashed line) for beltline weld and data from four compact specimen sizes.

ORNL-DWG 93-5856

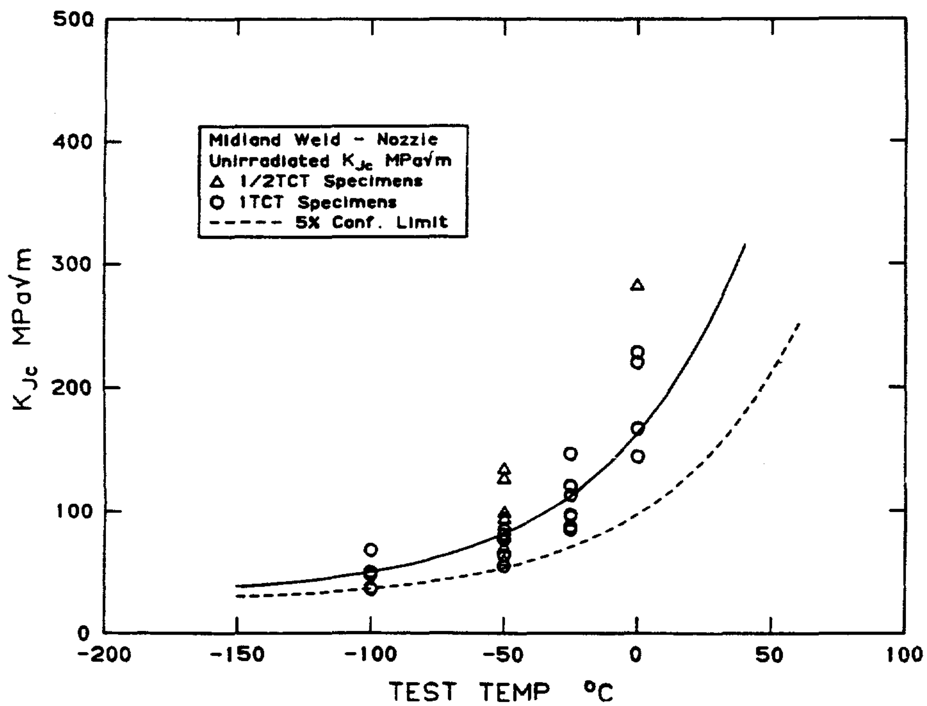


Figure 4. Master curve (solid line) and 5% confidence bound (dashed line) for beltline weld and data from four compact specimen sizes.

weld; this material was used in a nondestructive examination program to characterize flaws of commercial welds.[1] During preparation of the first several specimens, some porosity was detected in the fusion zone between the crack-starter weld bead and the weld-metal test section. This porosity prevents initiation of fast-running cracks. The porosity is now avoided by using longer runoff tabs but at the expense of a larger heat-affected zone.

The drop-weight NDT and reference temperatures, RT_{NDT}, for the Hope Creek trial weld are -50 and -23°C, respectively, the former of which is approximately the same as that of the Midland weld. Moreover, crack-arrest data obtained using the trial specimens will be a useful addition to the sparse crack-arrest data base for SAWs from commercially fabricated RPVs. The CVN impact energy data for specimens prepared from weld metal of the Hope Creek vessel are shown in Fig. 5. Together with the NDT, these data were used to determine the RT_{NDT}. The root area for the Hope Creek weld was approximately in the center of the wall thickness and was not used for preparation of any of the trial specimens.

ORNL-DWG 93-5857

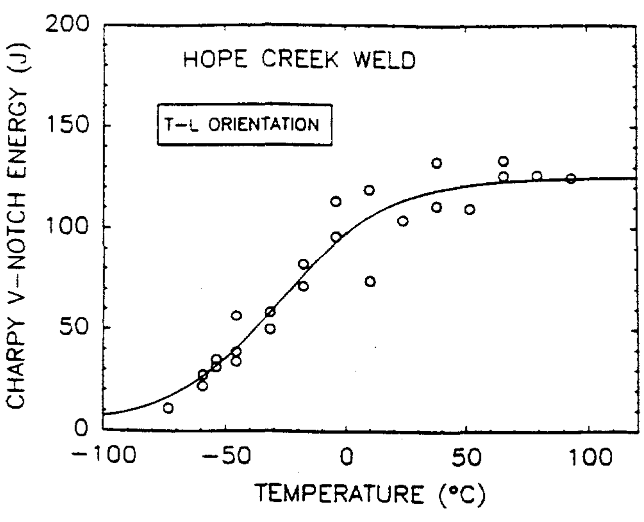


Figure 5. Charpy V-notch impact energy of material from Hope Creek submerged-arc weld. Specimens are oriented so that crack propagation is in the welding direction.

The results of crack-arrest tests on the Midland material obtained to date are shown in Fig. 6. For comparison, the results obtained from testing the Hope Creek material are also shown. Testing of the remaining unirradiated weld-embrittled Midland crack-arrest specimens and preparation of duplex-type crack-arrest specimens is under way. The duplex-type specimens will be used to obtain crack-arrest toughness, K_a, at a higher temperature than can be obtained using weld-embrittled specimens. A new, remote crack-arrest fixture, destined to be installed in the hot cells for use with irradiated specimens, is being used to test these unirradiated specimens.

A report on the unirradiated properties of the two weld materials is currently under preparation. It will restate the important findings of the chemical analyses, Charpy toughness survey, and drop-weight NDT tests. Transition temperature, as determined strictly from fracture mechanics-related tests, will be presented, viz., toughness expressed in terms of K_{Jc} versus test temperature. J_R-curves will be presented for test temperatures ranging from 0 to 288°C. These results will be used to compare to the CVN USE predictive models developed by Eason et al.[2] Two of the favorable comparisons are shown in Fig. 7. However, there were other cases where the comparison was not as favorable. A detailed evaluation of this multivariable model is planned.

The irradiation exposure of specimens is proceeding at a good pace. Scoping capsules 10.01 and 10.02 have been assembled and are expected to be inserted into the University of Buffalo research reactor in June 1993.

A preliminary analysis of the 1T dosimetry capsule, in simulation of capsule 10.05, has been made. The target fluence of 1.0×10^{19} neutrons/cm^2 has been exceeded by about 10% in the high-fluence region of the capsule. This suggests the average dose may be about right.

The irradiation of capsule 10.05 was completed in May 1993. This work was performed in the Ford Nuclear Reactor (FNR), but their capsule transfer setup is not capable of handling large capsules. This disassembly situation is made complex because so many irradiation facilities are being closed. However, we are currently studying various alternative solutions. Capsule

ORNL-DWG 93-5858

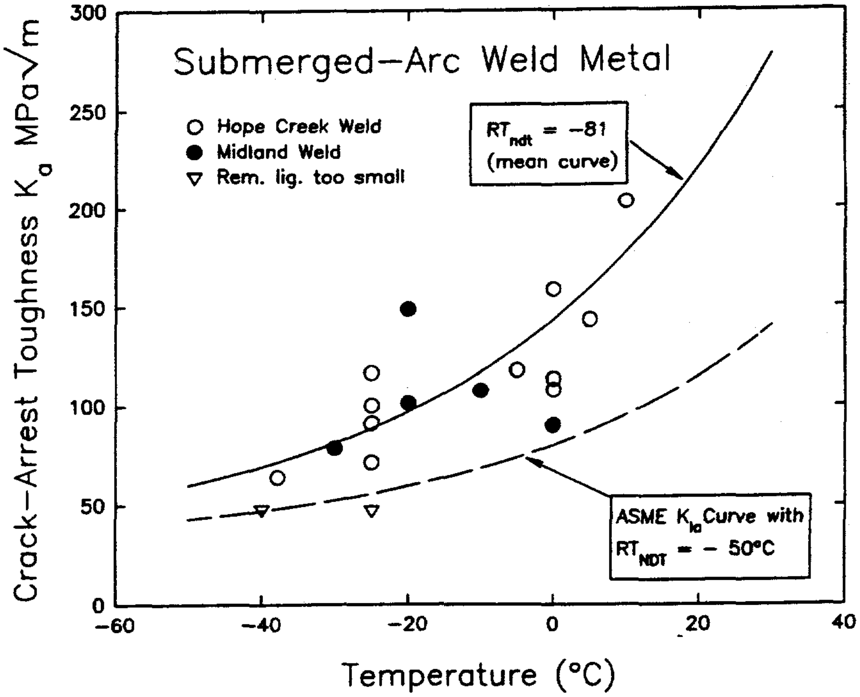


Figure 6. Crack-arrest toughness, K_a, of specimens machined from submerged-arc welds of Midland and Hope Creek reactor pressure vessels. Specimens are oriented so that crack propagation is in the welding direction.

ORNL-DWG 93-10947

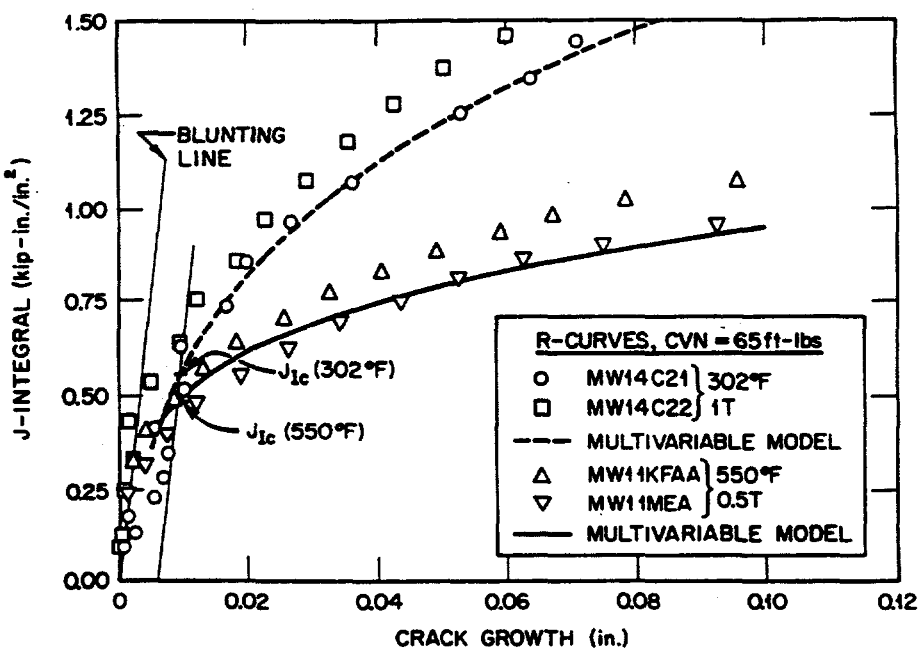


Figure 7. Example comparison of J_R-curve predictions by multivariable method and test data from Midland beltline weld.

10.06 is now being irradiated in the FNR and will be completed in 1994. Scoping capsules 10.03 and 10.04 (fluence of 5×10^{19} neutrons/cm^2) have been delayed pending the development of a new capsule design. The specimens needed for the assemblies are ready now.

References

1. K. V. Cook and R. W. McClung, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab, *Flaw Density Examinations of a Clad Boiling Water Reactor Pressure Vessel Segment*, USNRC Report NUREG/CR-4860 (ORNL/TM-10364), April 1987.*

2. E. D. Eason, J. E. Wright, and E. E. Nelson, Modeling and Computing Services, *Multivariable Modeling of Pressure Vessel and Piping J-R Data*, USNRC Report NUREG/CR-5729 (MCS 910401), May 1991.*

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

8. Microstructural Analysis of Irradiation Effects*

R. E. Stoller, K. Farrell, S. T. Mahmood, M. K. Miller, R. Jayaram, L. K. Mansur,
P. J. Othen (Department of Materials, Oxford University, Oxford, United Kingdom)
and G. Brauer (Research Center Rossendorf Inc., Dresden, Germany)

The overall, long-term goal of this task is to develop a physically based model that 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 status of several activities, conducted as a part of this task, are described below.

8.1 Experimental Investigations of Radiation-Induced Strengthening Under Low-Temperature Neutron Irradiation

8.1.1 HFIR Pressure-Vessel Embrittlement

The prime suspects in the embrittlement of the HFIR pressure vessel were a thermalized neutron spectrum, a low atomic displacement rate, and the impurities copper and boron. Each of these has now been systematically exonerated, in the process of which considerable neutron dosimetry studies have been made at the vessel [1]. These dosimetry data show that the neutron spectrum at one of the major embrittlement surveillance sites is not strongly thermalized. They also show that two fast neutron monitors, Be and Np-237, produce exceptionally high levels of reaction products compared with those for Ni and stainless steel monitors. Repeat dosimetry has now substantiated the unusual responses of the Be and Np monitors. F. B. Kam attributes these responses to photo nuclear events in the Be and Np due to a large flux of high-energy gamma rays (>2 MeV) at the HFIR vessel. The gamma field will not cause additional activation of Ni and Fe monitors. To account for the data, the ratio of gamma flux to fast flux must be greater than $\sim 10^3$. Such a ratio is attainable because the thick beryllium reflector and the long water path between the core and the vessel will attenuate neutrons but will be relatively transparent to gamma rays. In addition, (n,γ) reactions in the reflector and water will boost the flux of high- energy gammas. Preliminary, one-dimensional transport calculations indicate a ratio of $\phi_\gamma/\phi_{fast\text{-}n} > 10^3$ at the vessel.

High-energy gamma rays can cause atomic displacements in a pressure-vessel steel by transmitting energy to free electrons, which then bombard the metal lattice, creating Frenkel pairs. The displacement cross section is about 1/1000 of that for fast neutron displacements, so an irradiation spectrum with a $\phi_\gamma/\phi_{fast\text{-}n}$ ratio of 10^3 will produce roughly equal numbers of point defects from gamma rays and fast neutrons. However, most of the point defects generated by the fast neutrons will be annihilated by in-cascade recombination, whereas most of those created by gamma rays will avoid in-cascade recombination. Hence, the apparently accelerated embrittlement of the HFIR pressure vessel may be dominated by uncounted point defects introduced by gamma irradiation. This scenario is being investigated. Although gamma-induced embrittlement is feasible in principle, it has never been demonstrated. It would be highly desirable to perform specific experiments to measure embrittlement by gamma irradiation.

8.1.2 Irradiation Experiments Investigating the Effects of Flux Spectrum and Displacement Rate

Final assembly of the irradiation capsules for the first phase of the joint University of California, Santa Barbara, and ORNL experiment was completed during a visit to the University of Michigan, and irradiation of the first seven capsules began when the assemblies were loaded

*Work described within this task is supported in part by the Division of Materials Science, U.S. Department of Energy.

into row 60 of the experimental grid at the FNR on October 16, 1992. In addition to complete internal dosimetry packages, an external package consisting of iron and nickel wires was attached to the capsules. This external package was intended to permit a check of the neutron fluxes before the early capsules were disassembled. In the event that the target fluences were not reached, the capsules could be replaced and the exposure time adjusted for the longer time capsules.

When the first irradiation capsules were removed from the FNR, the external dosimeter packets were removed and the fluences calculated based on spectrally averaged cross sections obtained from the National Institute of Standards and Technology (NIST). The results indicated that the fluxes were lower than anticipated, significantly so in two of the positions. An explanation for the lower flux values was obtained through discussions with the staff at the University of Michigan. The discrepancy was largely due to two factors. First, the expected flux values were based on a set of x-y-z flux measurements done by the NIST in 1990. The zero reference point, perpendicular to the south face of the core for these measurements, was offset from the zero point of the experimental grid positions. In addition, a new experimental grid plate was subsequently installed somewhat further from the core. This led to a total offset of the actual capsule positions of 1.2 in. from the expected positions. The exposure received by three of the capsules (ORNL-F0, -F2, and -F5) was sufficiently close to the target values that they will be shipped back to ORNL for disassembly. The fourth capsule (F8) was reinserted for further irradiation, and new exposure times were calculated for the remaining three capsules (F1, F4, and F7).

The hydraulic tube (HT) of HFIR has been used to conduct a series of irradiations to investigate the embrittlement behavior of several engineering and model alloys at 60°C. A comprehensive data set has been obtained relating changes in tensile properties to fast fluence for five commercial ferritic steels and five model iron-based alloys irradiated at 50 to 60°C. The fast flux (E > 1.0 MeV) in the HFIR-HT is about 4.5×10^{14} n/cm^2. These data will provide the highest dose rate points for a larger experiment that spans fluxes down to 2×10^{10} n/cm^2. Data have been obtained up to a total fast fluence of 4.0×10^{19} n/cm^2, and are presently undergoing analyses.

As part of the HFIR-HT irradiation experiment, a detailed set of neutron dosimetry measurements has been made in the HT. A range of fission, activation, and helium accumulation monitors were used; they included: Al-17, Ag-109, Au-197, B-10, Be-9, Co-59, Cu-63, Fe-54, Li-6, Ni-58, Np-237, and Ti-46. The cobalt and silver monitors were irradiated both bare and cadmium covered. Good agreement was obtained among the various monitors, and the results indicate that the fast flux (E > 0.1 MeV) within the central 25-cm section of the tube is 8(±1)e14 n/cm^2/s. The fast flux falls to 3.3×10^{14} at the extreme end of the HT (±25 cm from the centerline). The thermal-to-fast (E > 0.1 MeV) ratio is 3.1 ± 0.3 over the entire length of the HT.

The HFIR-HT irradiations have demonstrated that the HT has the potential to be a valuable tool for radiation-anneal-reirradiation experiments. As a result, we have begun the design of a gas-gapped capsule that will permit irradiations to be conducted at temperatures up to 300°C. The use of this capsule will also permit high flux data to be obtained at 288°C to help investigate the effects of displacement rate at this temperature.

8.1.3 Trojan Cavity Irradiations

In order to investigate the effects of neutron irradiation at the low fluxes typical of LWR vessel support structures, a set of mechanical test specimens was irradiated in the cavity of Portland General Electric's Trojan reactor. Specimens were fabricated from archive material from the HFIR A212B and A350-3 alloys and a typical A36 structural steel. The specimen matrix included CVN specimens from the A36 (5) and A212B (6), and ORNL SS-3 minitensiles of A36 (14), A212B (15), and A350-3 (15). These irradiations have been completed and the specimens returned to ORNL.

The preliminary analysis of the dosimetry included in the Trojan cavity experiment indicates that the total fast fluence was lower than planned, between 1 and 2×10^{16} n/cm^2. Little hardening or DBTT shift is expected at this dose. As a result, tensile data will be obtained to screen the materials prior to scheduling the testing of the Charpy specimens.

8.2 Atom Probe Field Ion Microscopy (APFIM) Characterizations of VVER 440 and VVER 1000 Steels

An APFIM and transmission electron microscopy (TEM) characterization of Soviet types 15Kh2MFA Cr-Mo-V and 15Kh2NMFA Ni-Cr-Mo-V pressure-vessel steels that are used in VVER 440 and VVER 1000 reactors has been performed. Prior to examination in the APFIM, the 8-kg laboratory melts were heat treated. The 15Kh2MFA (VVER 440) steels were aged for 1 h at 1000°C, oil quenched and then aged for 10 h at 700°C, and air cooled to room temperature. The 15Kh2NMFA (VVER 1000) steel was austenitized at 920°C, water quenched and then aged at 650°C, air cooled, aged 25 h at 620°C plus 20 h at 650°C, and finally furnace cooled to room temperature.

The microstructure of the 15Kh2MFA Cr-Mo-V steels was found to be tempered martensite with approximately 10% ferrite, whereas the 15Kh2NMFA Ni-Cr-Mo-V steel was a mixture of ferrite and bainite. TEM revealed the presence of coarse (60- to 500-nm), blocky chromium-rich M_7C_3 carbides in both steels. Energy dispersive X-ray analysis with the use of TEM revealed that the metallic content of these chromium-rich carbides in the VVER 440 steel was approximately Fe - 40 to 60 at. % Cr, 3.1 to 3.5% V, 1.5 to 2.6% Mo. An atom probe analysis of one of these precipitates in the VVER 440 steel revealed a composition of 37.6 ± 1.7 at. % Cr, 22.7 ± 1.4% Fe, 3.5 ± 0.6% Mo, 2.7 ± 0.6% V, 0.6 ± 0.3% Mn, and 33.0 ± 1.6% C. No nitrogen, nickel, silicon, or copper was observed in this precipitate. In addition, some 10-nm-diam MC carbides were observed distributed throughout the matrix of the 15Kh2MFA Cr-Mo-V steel. These 10-nm-thick precipitates were also observed at the lath boundaries in both alloys and were lenticular in shape and had their major axis in the plane of the boundary. Atom probe selected area analysis identified these precipitates as vanadium carbonitrides with a carbon-to-nitrogen ratio of approximately 2:1. In addition to vanadium, the metallic content of these precipitates contained significant amounts of molybdenum, chromium, and iron.

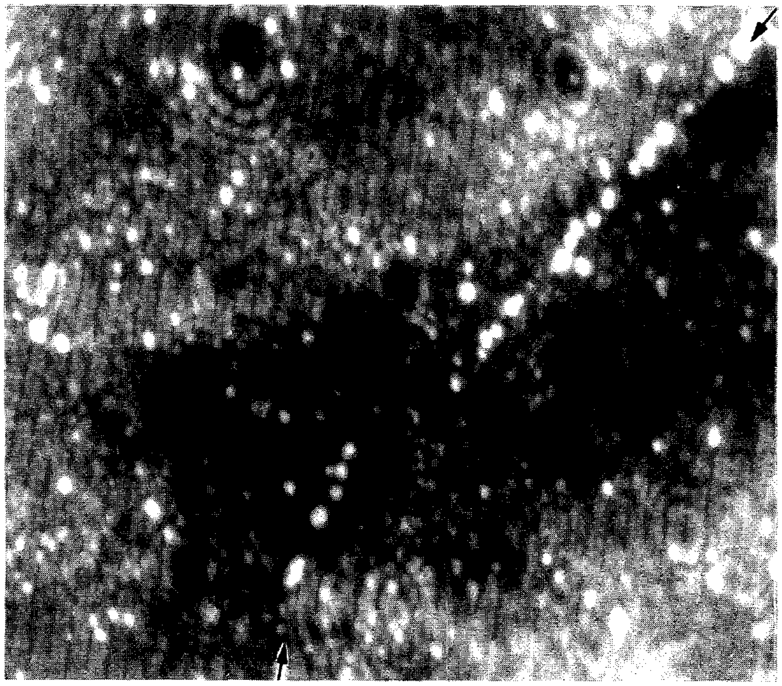
Field ion microscopy has revealed that the lath boundaries in both these unirradiated steels are decorated with an ultra-thin (<0.5 nm) film of brightly imaging molybdenum carbonitride precipitates as shown in Fig. 8. Atom probe analysis of this boundary in the VVER 440 steel revealed a phosphorus enrichment of ~52 times that of the matrix. Local enrichments of phosphorus of ~15.6 times the matrix analysis were measured for the boundary in the VVER 1000 steel. However, a more extensive analysis of this boundary revealed some local inhomogeneities in the phosphorus level and yielded a slightly lower average enrichment of ~8.6 times the matrix level. The difference in enrichment measured between the two VVER alloys is probably related to the different concentrations of phosphorus in the initial bulk materials—the VVER 1000 steel having both a lower enrichment and a lower bulk level of phosphorus. These levels of phosphorus were substantially higher than those previously measured (typically 3.5x) in unirradiated A533B steels. These enrichment factors assume that the phosphorus was constant over the entire volume analyzed. However, the volume analyzed will contain a substantial contribution from the matrix. If a width of a monolayer is assumed for the thickness of the segregated layer, then the actual enrichment factors should be ~5 times these values. These substantial phosphorus enrichments should be a serious cause for concern, since it has been well established that phosphorus is a primary cause of temper embrittlement in steels.

In anticipation of examining the irradiated VVER steels, two standard operating procedures have been written to permit the examination of neutron-irradiated pressure-vessel steels in the ORNL atom probe. The first procedure (MET-MMS-SOG-012) permits the preparation of field ion needle-shaped specimens from small blanks by standard electropolishing procedures. The second procedure (MET-MMS-SOP-011) details the loading and operation of the atom probe with low-level radioactive specimens.

8.3 Theoretical Modeling and Analysis

A new collaboration was initiated between ORNL staff and researchers at the Harwell Laboratory and the University of Liverpool in the United Kingdom. The purpose of the collaboration is to investigate high-energy cascades in iron with the intent of determining whether there are any

L916-35



L921-7

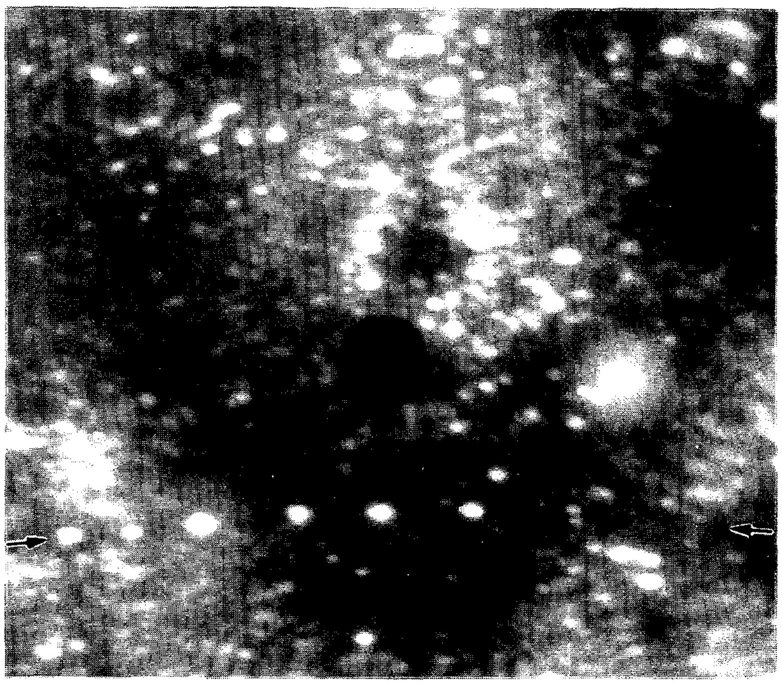


Figure 8. Field ion micrographs of lath boundaries in (*a*) VVER 440 and (*b*) VVER 1000 steels; boundary paths are indicated by arrows. Brightly imaging precipitates on boundaries are molybdenum carbonitrides.

inherent differences between extended defect formation in body-centered cubic (bcc) iron and what has been observed in face-centered cubic (fcc) metals such as copper. The work is being carried out using the MOLDY molecular dynamics simulation (MDS) code and a newly developed interatomic potential for iron. A number of low-energy (<5 keV) cascades were completed at Liverpool, and the first series of 10-keV cascades was completed at ORNL. This involved seven cascades at a temperature of 100 K. Each of the 10-keV cascades required about 150 h of central processing unit (CPU) time on a high-speed workstation, or about 85 CPU hours on a Cray Y-MP C90. Additional simulations are under way to investigate the effect of irradiation temperature; 10-keV cascades will be run at 600 K, and 5-keV cascades will be run at 900 K. An initial analysis of the results of MDS of 10-keV cascades in iron at 100 K has been completed. The residual defect distribution after more than 10 ps has been determined for each of the seven cascades. The results of this analysis are summarized in Tables 5 and 6. The bcc iron results are both similar to and different from those that have been obtained on fcc metals such as copper [2]. As in the fcc metals, only a fraction of the calculated NRT displacements per atom (dpa) remain after the cascade has cooled. The cascade efficiency appears to be somewhat higher in the case of iron, and the fraction of clustered defects appears to be lower. The fact that point defect clusters are observed at much lower doses in copper than in iron may, in part, be a result of the differences in their in-cascade clustering behavior. No vacancy clusters were observed in any of the cascades.

References

1. K. Farrell et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., "Investigation of Low-Temperature Neutron Embrittlement of Ferritic Steels," pp. 151-78 in *Proceedings of the U.S. Nuclear Regulatory Commission Twentieth Water Reactor Safety Information Meeting*, USNRC Report NUREG/CP-0126 Vol. 3, March 1993.*

2. C. A. English, A. J. E. Foreman, W. J. Phythian, D. J. Bacon, and M. L. Jenkins, *Materials Science Forum* **97-99**, 1-22 (1992).+

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

Table 5. Total point defect production in iron, 10-keV cascades at 100 K

Cascade number	Peak damage time (sec)	Vacancies[a] at peak	Frenkel pair[b] after ~11 ps	Cascade[c] efficiency
1	3.66×10^{-13}	2599	43	0.43
2	5.57×10^{-13}	3984	29	0.29
3	3.70×10^{-13}	2227	38	0.38
4	5.14×10^{-13}	3723	42	0.42
5	6.53×10^{-13}	7698	41	0.41
6	3.83×10^{-13}	2570	46	0.46
7	5.91×10^{-13}	4394	37	0.37
average	----	3885	39	0.39

[a]Vacancy count at the peak not corrected for those associated with the split (dumb-bell) interstitials.
[b]Includes both clustered and mono-defects.
[c]Cascade efficiency based on the number of expected NRT displacements.

$dpa_{NRT} = 0.8\ T_{dam}/2/E_d$, where $T_{dam}=10$ keV and $E_d=40$ eV

Table 6. Interstitial cluster production in iron, 10-keV cascades at 100 K

Cascade number	Interstitial 2-i	3-i	Cluster 4-i	Information 5-i	8-i
1	5	-	1	-	-
2	3	1	-	-	-
3	2	2	-	-	-
4	4	1	1	-	-
5	-	3	-	5	-
6	2	1	1	-	-
7	2	-	-	-	1
average	2.6	1.1	0.43		0.13

9. In-Service Aged Material Evaluations

F. M. Haggag, S. K. Iskander, R. K. Nanstad, and P. Arakawa

A specification was prepared for an official request for bids to purchase a computer numerically controlled (CNC) vertical milling machine. The available space for installing the CNC machine in the hot cell in Building 3525 is approximately 1 × 1 × 1 m (3 × 3 × 3 ft). Arakawa and Haggag will witness the machining of three specimens (compact fracture toughness, CVN, and a miniature flat tensile) at the manufacturer's site. This action was included in the official request for a bid, as the acceptance testing required, before delivery of the CNC machine to ORNL. It is expected that this new approach would be better than a sole-source acquisition since it should eliminate those vendors who cannot fabricate, using the proposed small CNC machine, the above three specimens according to specified machining tolerances.

10. Correlation Monitor Materials

W. R. Corwin

The use of correlation monitor materials is required by the most recent revision of the *ASTM Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels*, E185-93 (ref. 1). These well characterized pressure-vessel materials are to be included within each surveillance capsule to help establish the credibility of the individual data sets obtained from the specimens of interest within the capsule. The test for surveillance data credibility, as described in E185, is that the irradiation-induced shift of the transition temperature of the accompanying correlation monitor materials falls within the established scatter band for the data base for the particular monitor material. Very little of the original steel qualified as correlation monitor materials remains outside of ORNL. The purpose of this task is to provide for the long-term, archival quality storage of the remaining correlation monitor materials at ORNL and to establish a means for their NRC-approved disbursement to qualified recipients.

No significant activity has occurred within this task since the detailed inventory was made of all available materials late during the previous reporting period. Current plans are to begin the transfer of the appropriate material, which was identified during the recent inventory, from its current outdoor location at the Y-12 Plant in Oak Ridge, Tennessee, to a controlled access location at the X-10 site (ORNL) later this year.

References

1. "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," E 706 (IF), ASTM Designation E 185-93, *1993 Annual Book of ASTM Standards Vol. 12.02, Nuclear(II) Solar and Geothermal Energy*, American Society for Testing and Materials, Philadelphia, 1993.

+Available in public technical libraries.

11. Special Technical Assistance

William R. Corwin

This task has been included with the HSSI Program to provide a vehicle in which to conduct and monitor short-term, high-priority subtasks. A summary of the current activities within this task is contained, by subtask, in the following sections.

11.1 HFIR Neutron Exposure Parameters

W. R. Corwin and K. Farrell

The determination that the amount of irradiation-induced embrittlement in the pressure vessel surveillance specimens exposed in the HFIR at about 60°C was significantly higher than expected has led to several investigations to determine the cause of the acceleration. During the current reporting period, an evaluation was made of the results of refined calculations and detailed experimental measurements of the exposure parameters which existed at the various surveillance capsule locations. The results of these investigations showed conclusively that the very high ratios of thermal-to-fast neutrons that had been previously reported did not exist for the vast majority of the surveillance locations and, hence, were obviously not the cause of the accelerated embrittlement. The results leading to this conclusion were documented in a letter report to the NRC.[1] The recognition that the low-temperature, irradiation-induced embrittlement of the HFIR specimens could not have come from low-energy neutrons has led to a continued search for the cause. Further discussion of this topic is contained in Sect. 8.1.1 as part of the investigations within the HSSI subtask on low-temperature embrittlement.

11.2 Yankee Rowe Reactor Vessel Integrity Assessments

This subtask was completed during the previous reporting period.

11.3 Life Extension

R. K. Nanstad

The purpose of this subtask is to evaluate Russian research related to thermal annealing of RPVs. This activity is related, of course, to Task 6 on Annealing Effects in LUS Welds and to Task 13 on Technical Assistance for JCCCNRS Working Groups 3 (radiation embrittlement) and 12 (aging). As part of the activities in Task 13, irradiation, thermal annealing, and reirradiation will be performed on Russian steels. Furthermore, Task 12 on JPDR Steel Examination includes a subtask on thermal annealing. Thus, because of the substantial amount of thermal annealing research and actual in-service thermal annealing of about 11 reactor vessels by the Russians, a continuing evaluation of their experience is under way in this program. Moreover, M. A. Sokolov, a Russian researcher with expertise in thermal annealing, is on sabbatical at ORNL, and interactions with him on this topic are conducted on a continuing basis.

11.4 Notched Bar Surveillance Specimen Evaluations

W. R. Corwin

Significant limitations exist in the volume available within LWR surveillance capsules for the specimens used to predict the amount of embrittlement that the pressure vessel will experience. The type of surveillance specimens typically included in the capsules are CVN and tensile specimens, which are then used to predict the changes in the fracture toughness of the vessel by correlation. It would be desirable to directly measure the fracture toughness of the vessel materials, but the relatively large size of traditional fracture toughness specimens has limited their inclusion to special capsules and small numbers. The need to minimize the volume of specimens used for the direct measurement of irradiated fracture toughness has led to the recent examination of the round, notched and precracked tensile (RNPT) specimen. This specimen has a very high degree of constraint at the tip of the crack with respect to the overall volume of material required for the specimen and should therefore provide a relatively high specific measurement capability for fracture toughness. Hence, it is potentially very appealing for use in directly measuring fracture toughness for irradiated materials.

To examine whether this potential benefit can be exploited, the HSSI Program has initiated limited research activities with two research organizations, the Harwell Laboratories of AEA Technology and SRI, Inc. The two organizations were selected because they are currently involved in the development and evaluation of detailed testing procedures for the RNPT specimen and have shown very promising results on internally generated tests. It is planned that both organizations will prepare and test somewhat different versions of the RNPT specimen from materials whose fracture toughness properties have already been extensively examined with the HSSI Program. The comparison of the results from the existing, large fracture toughness data bases with those from the RNPT specimens will form the basis for a decision on the more detailed evaluation of the RNPT specimen for surveillance applications.

11.5 CVN Reconstitution Round Robin

R. K. Nanstad

The HSSI Program is participating in an ASTM Committee E-10 (Subcommittee E10.02) project on reconstitution of CVN specimens. Part of the project involves a round-robin program with the participation of 11 organizations, both foreign and domestic. All the participants will reconstitute broken CVN specimens according to specifications developed by the subcommittee. The ORNL participation involves the testing of all baseline and reconstituted specimens. We are not involved in the reconstitution exercise of the round robin. The two materials chosen for the study are HSST Plate 03 and an LUS weld. All baseline specimens were fabricated by another organization and delivered to ORNL.

Scoping tests were conducted with each material to allow for determination of two test temperatures for each material, one at about the 41-J (30-ft-lb) level and one on the upper shelf. As the result of those tests, baseline tests were conducted with the LUS weld at 1.7 and 93.3°C (35 and 200°F) and with HSST Plate 03 at −12 and 93.3°C (10 and 200°F). Tests were conducted with both the ASTM and International Standards Organization (ISO) strikers. For each striker and each material, 46 tests were conducted at the low temperature, and 23 tests were conducted at the high temperature. Successful choices of the test temperatures are indicated by the average results obtained. For example, with the ASTM striker, 46 tests of the weld at 1.7°C (35°F) showed an average energy of 43.9 J (32.4 ft-lb), while 46 tests of the plate at −12°C (10°F) showed an average energy of 40.9 J (30.2 ft-lb). At the high temperature, all specimens exhibited 100% ductile fracture; 23 tests of the weld showed an average energy of 94.5 J (69.7 ft-lb), while 23 tests of the plate showed an average energy of 121 J (164 ft-lb). Comparison of the ISO and ASTM striker results is under way. The specimens and results were given to the ASTM round-robin task leader for distribution to the participants.

References

1. Letter report, "The DOS 1 Neutron Dosimetry Experiment at the HB-4-A Key 7 Surveillance Site on the HFIR Pressure Vessel," by K. Farrell et al., from W. R. Corwin, Oak Ridge Natl. Lab., to M. E. Mayfield, NRC, February 12, 1993.*

*Available in NRC PDR for inspection and copying for a fee.

12. Evaluation of Steel from the JPDR Pressure Vessel

W. R. Corwin

By necessity, virtually all irradiation research is performed on LWR vessel materials, which are irradiated in environments that, at best, only approximate the actual exposure conditions experienced by material within the wall of a pressure vessel. There is a need to validate the results of such irradiation effects research by the examination of material taken directly from the wall of a pressure vessel that 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. The work conducted within this task will be performed in collaboration with related studies being performed by the Japan Atomic Energy Research Institute (JAERI).

The JPDR is a small boiling water reactor which began operation in 1963. It operated until 1976, accumulating ~17,000 h of operation, of which a little over 14,000 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 2.8×10^{18} n/cm^2 (>1 MeV). Best estimates of its impurity contents in the base metal are 0.16% Cu and 0.010% P with a nickel content of 0.63 to 0.65%. Chemical composition of the weld metal is unknown.

The principle focus of the existing Japanese program on the JPDR is to establish and demonstrate means of decommissioning and dismantling nuclear reactors. The studies on the irradiated material from the pressure vessels are a relatively small side project. The current status of the JPDR is that the pressure vessel and all of the piping have been dismantled and removed. The material from the pressure vessel is in pieces, roughly 800 by 800 mm x the original local wall thickness, and has been placed in a hot warehouse on-site for long-term storage. A number of full-thickness trepans of both the weld and the base metal have been removed from the vessel for eventual examination within the HSSI Program. The objectives of the JAERI JPDR pressure-vessel investigations are also to obtain materials property information on the pressure vessel steel exposed to in-service irradiation conditions and to establish a methodology for aging evaluation and life prediction of RPVs. Their research associated with the evaluation of irradiation effects is composed of three pieces: examination of material from the JPDR vessel, new test reactor irradiations of archival and similar materials, and reevaluation of data from irradiation surveillance and related programs. The overall objectives of this task within the HSSI Program are to provide: (1) a detailed examination and evaluation of the state of embrittlement of the base metal and weldment in the beltline and remote regions of the pressure vessel of the JPDR, including an evaluation of the mechanisms controlling the embrittlement; (2) a benchmark for through-wall attenuation of irradiation embrittlement for typical RPVs; and (3) a benchmark for the embrittlement trend analyses contained in U.S. and Japanese regulatory documents.

Informal agreement was reached during the current reporting period on the details of the research agreement under which all the collaborative research on the JPDR material will be conducted. It is anticipated that the agreement will be signed and work initiated during the next reporting period.

13. Technical Assistance for JCCCNRS Working Groups 3 and 12

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

The purpose of this task is to provide technical support for the efforts of the Joint U.S.-Russian Coordinating Committee on Civilian Nuclear Reactor Safety 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 irradiation, annealing, and reirradiation 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) initiation of the 1-year assignment of M. A. Sokolov of the Russian National Research Center-Kurchatov Institute at ORNL.

13.1 Irradiation Experiments in Host Country

Regarding irradiation experiments in the host country, ORNL supplied 100 CVN and 30 tensile specimens each of HSST Plate 02 and of weld 73W to the Russian representatives on Working Group 3. Some of those specimens have been encapsulated and placed into irradiation positions in the Novovoronezh Unit 5 reactor vessel. Additionally, the Russians provided two weld materials, for VVER-440 and VVER-1000 reactor vessels, from which we machined CVN and tensile specimens for inclusion in irradiation capsule 10.06 (see Task 7 on the Tenth Irradiation Series). Eighteen CVN and three tensile specimens of each weld were placed in a capsule location where the irradiation temperature is anticipated to be 288°C (550°F), while 29 CVN and 2 tensile specimens of each weld were prepared for an anticipated irradiation temperature of 270°C (518°F). Some of the specimens will be tested in the irradiated condition, while others will be tested following annealing or after annealing and reirradiation.

13.2 Personnel Interactions

The HSSI Program is sponsoring the sabbatical of Dr. Mikhail A. Sokolov at ORNL. Dr. Sokolov arrived at ORNL at the end of December 1992 and is expected to spend 1 year working at ORNL with staff members on the HSSI Program. He will specifically concentrate on activities in the area of thermal annealing of irradiated steels and the use of subsize Charpy impact specimens for irradiated studies.

During the period from September 24 to October 2, 1992, R. K. Nanstad participated in the fourth information exchange of Working Group 3 of the JCCCNRS. The meetings were held in St. Petersburg and Moscow, Russia. He participated in the meetings on radiation embrittlement, structural integrity, and life extension of RPVs and supports. He presented results of research on the effects of irradiation on K_{Ic} and K_{Ia} curves with consideration of statistical analyses, mechanisms of radiation embrittlement in pressure-vessel steels, and results of the U.S.-Russia round-robin program on Charpy impact and fracture toughness testing of Russian and U.S. steels. The meetings resulted in a detailed exchange of technical progress between participants, identification of specific areas requiring continued research, agreements regarding exchange of additional information, agreements regarding further exchange of reactor vessel materials for mutual irradiation and annealing studies, and agreements regarding additional proposals for other comparative studies. A foreign trip report (ORNL/FTR-4399) was prepared and submitted to NRC describing details of the discussions. Dr. Sokolov also attended those meetings.

14. Additional Requirements for Materials

D. J. Alexander

The purpose of this task is to 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 shall include proposed modifications and supplementary information as required in the areas of design, materials, fabrication, testing, and inspection. In addition, specific information shall be developed and analyzed to support recommendations related to proposed regulations, guides, or referenced codes and standards. The specific activities to be performed include: (1) review of new materials and requirements proposed for inclusion into national codes and standards, of ASME code cases, and of potential deficiencies in proposed supporting technology and data; (2) continuation of long-term aging of SS welds; and (3) continuation of the low-temperature postweld heat treatment study of low-alloy steels.

Present activity is focused on the SS weld aging task. Three SS welds with ferrite contents of 4, 8, or 12% have been aged at 343°C (650°F) for 3000, 10,000, or 20,000 h. Additional material is currently being aged to reach 50,000 h. Tensile and CVN specimens have been tested for each of the welds for aging times up to 20,000 h, and a report is being written to summarize the findings to date.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5591
Vol. 4, No. 1
ORNL/TM-11568/V4&N1

2. TITLE AND SUBTITLE

Heavy -Section Steel Irradiation Program Semiannual
Progress Report for October 1992 Through March 1993

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1998

4. FIN OR GRANT NUMBER

L1098

5. AUTHOR(S)

W.R. Corwin

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

October 1992 - March 1993

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Oak Ridge National Laboratory
Oak Ridge, TN 37831-6285

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

Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

M.G. Vassilaros, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The primary 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. During this reporting period, irradiated crack-arrest specimens were tested; charpy V-notch specimens of high-cooper weld metal were annealed and tested; a fracture mechanics evaluation of the unirradiated Midland low upper-shelf weld was nearly completed; irradiation of the first large Midland capsule was completed; refined calculations and detailed experimental measurements of the exposure parameters in the High Flux Isotope Reactor were evaluated; in-cavity irradiation of vessel support materials were completed; unirradiated microstructural characterization of a Russian reactor vessel steel was completed; collaborative investigations of in-cascade point-defect generation experiments and investigations of a very wide range of flux levels on low-temperature embrittlement were initiated; baseline testing for an ASTM round robin on reconstituted Charpy V-notch specimens was completed; informal agreement was reached on collaboration on pressure vessel material from the Japan Power Demonstration Reactor; impact and tensile specimens of two U.S. reactor vessels materials were encapsulated and irradiation begun in a Russian reactor; and tensile and impact specimens were tested for three stainless steel welds aged for up to 20,000 h.

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

pressure vessels
ductile testing
irradiation
fracture mechanics
cladding
embrittlement
LUS weld metal
crack arrest

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)
unclassified

(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE