Heavy-Section Steel Irradiation Program

Semiannual Progress Report for
October 1996 - March 1997

Manuscript Completed: October 1997
Date Published: February 1998

Prepared by
T. M. Rosseel

Oak Ridge National Laboratory
Operated by Lockheed Martin Energy Research Corp.

Oak Ridge National Laboratory
Oak Ridge, TN 37831-6285

M. G. Vassilaros, NRC Project Manager

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

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED *ph*

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

**Portions of this document may be illegible
electronic image products. Images are
produced from the best available original
document.**

Abstract

Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. Because the RPV is the only key safety-related component of the plant for which a redundant backup system does not exist, it is imperative to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance that occurs during service. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established. Its primary goal is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels as they relate to light-water RPV integrity. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into seven tasks: (1) program management, (2) irradiation effects in engineering materials, (3) annealing, (4) microstructural analysis of radiation effects, (5) in-service irradiated and aged material evaluations, (6) fracture toughness curve shift method, (7) special technical assistance, and (8) foreign research interactions. The work is performed by the Oak Ridge National Laboratory.

1. The first part of the report
describes the general situation
of the country and the
state of the economy.
It also mentions the
main problems of the
country and the
state of the
economy.

Contents

Abstract	iii
List of Figures	vi
List of Tables	vii
Acknowledgments	ix
Preface	xi
Summary	xv
1. Program Management	1-1
2. Irradiation Effects in Engineering Materials	2-1
2.1 Fracture Toughness Shifts in High-Copper Weldments (Series 5 and 6)	2-1
2.2 Irradiation Effects in a Commercial LUS Weld	2-1
3. Annealing	3-1
3.1 Temper Embrittlement in Reactor Pressure Vessel Steel Heat-Affected Zones	3-1
3.2 Annealing Effects in LUS Welds (Series 9)	3-5
4. Microstructural Analysis of Radiation Effects	4-1
4.1 Introduction	4-1
4.2 Experimental Investigation of the Effects of Primary Knockon Atom Energy Spectrum	4-1
4.3 Modeling and Simulation of Neutron Energy Spectrum Effects	4-3
5. In-Service Irradiated and Aged Materials Evaluations	5-1
5.1 Remotely Operated Machining Center	5-1
6. Fracture Toughness Curve Shift Method	6-1
6.1 Introduction	6-1
6.2 Master Curve Technology	6-1
6.3 Comparison of Irradiation-Induced Charpy and Fracture Toughness Curve Shifts	6-7
7. Special Technical Assistance	7-1
7.1 Aging and Testing Methods	7-1
7.2 Correlation Monitor Materials	7-1
7.3 Transfer of Government-Furnished Equipment (GFE) and Materials	7-2
7.4 Test Reactor Irradiation Coordination	7-2

Contents (continued)

8.	Foreign Research Interactions	8-1
8.1	Japanese Power Demonstration Reactor Vessel Steel Examinations	8-1
8.2	Technical Assistance for Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Workings Groups 3 and 12	8-2
8.3	Belgian Interactions	8-3
8.4	AEA New Coordinated Research Program	8-3
8.5	Korean Interactions	8-4
Conversion Factors		CF-1

Figures

2.1	R curves for LUS WF-70 weld metal showing the effect of side grooving on resistance to ductile tearing	2-3
2.2	Truncation of master curve using 5% of stable crack growth as the limitation value	2-4
2.3	Comparison of R curves for WF-70 beltline and nozzle course weld metal	2-5
2.4	Load versus displacement test record of irradiated nozzle course weld metal	2-6
2.5	R curves of WF-70 weld metal before and after irradiation	2-7
2.6	Postirradiation R curve for WF-70 nozzle course weld metal	2-8
3.1	Prior austenite grain size of A 533 grade B after AEA cycle	3-3
3.2	Prior austenite grain size of A 533 grade B after Gleeble cycle	3-4
4.1	Comparison of tensile data on HSST Plate-02 and similar materials from commercial reactor surveillance programs and 2.5-MeV electron irradiation	4-2
4.2	Comparison of spectrally averaged damage production cross sections (per NRT dpa) for various irradiation environments; defect survival ratio is shown in (a) and the interstitial clustering fraction is shown in (b)	4-5
4.3	Normalized iron PKA spectra for PWR and BWR 1/4-T and 3/4-T positions and the HFIR PTP position	4-6
4.4	Variation of spectrally averaged damage production cross sections (per NRT dpa) through the RPV for PWR and BWR	4-7

Contents (continued)

Figures

6.1	Comparison of the HSST Plate 02 linear-elastic K_{Ic} database relative to the master curve with 5% margin-adjusted tolerance bound curve derived by testing several PCVN specimens	6-4
6.2	Comparison of the K_{Ic} EPRI database and the ASME lower-bound curve relative to the 5% margin-adjusted tolerance bound curve derived by testing several PCVN specimens of HSST Plate 02	6-5
6.3	Median fracture toughness values of specimens with different thicknesses (B) and widths (W)	6-6
6.4	Correlation between fracture toughness T_{100} and Charpy T_{41J} shifts for weld metals	6-8
6.5	Correlation between fracture toughness T_{100} and Charpy T_{41J} shifts for base metals	6-9
6.6	Fracture toughness data of the unirradiated base and weld metals adjusted to 1T (CT) size and normalized to T_{100}	6-10
6.7	Fracture toughness data of the irradiated base and weld metals adjusted to 1T (CT) size and normalized to T_{100}	6-11

Tables

3.1	Commercial materials selected	3-2
3.2	Prior austenite grain growth (two methods)	3-2
3.3	Program plan for aging cycles	3-2
3.4	Preliminary results	3-4
3.5	Transition temperature, T_0 (50% energy)	3-5
8.1	Chemistry of JPDR weld metal trepans remote from the core	8-1
8.2	Chemistry of ASTM A 302 grade B modified JPDR trepans remote from the core	8-2

Acknowledgments

The authors thank Julia Bishop for her contributions in the preparation of the draft manuscript for this report, Sandra Lyttle for the final manuscript preparation, Ralph Sharpe for editing the document, and Roxanne Raschke for her assistance in ensuring that all the details associated with this report were completed. The authors also gratefully acknowledge the continuing technical and financial contributions of the U.S. 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 October 1996 to March 1997. 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 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)
NUREG/CR-5591, Vol. 4, No. 1
(ORNL/TM-11568/V4&N1)
NUREG/CR-5591, Vol. 4, No. 2
(ORNL/TM-11568/V4&N2)
NUREG/CR-5591, Vol. 5, No. 1
(ORNL/TM-11568/V5&N1)
NUREG/CR-5591, Vol. 5, No. 2
(ORNL/TM-11568/V5&N2)
NUREG/CR-5591, Vol. 6, No. 1
(ORNL/TM-11568/V6&N1)
NUREG/CR-5591, Vol. 6, No. 2
(ORNL/TM-11568/V6&N2)
NUREG/CR-5591, Vol. 7, No. 1
(ORNL/TM-11568/V7&N1)
NUREG/CR-5591, Vol. 7, No. 2
(ORNL/TM-11568/V7&N2)

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 listed 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. 1 (ORNL/TM-8787/V1)
NUREG/CR-3334, Vol. 2 (ORNL/TM-8787/V2)
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)
NUREG/CR-4219, Vol. 7, No. 1 (ORNL/TM-9593/V7&N1)

Summary

1. Program Management

The Heavy-Section Steel Irradiation (HSSI) Program is arranged into 8 tasks: (1) program management, (2) irradiation effects in engineering materials, (3) annealing, (4) microstructural analysis of radiation effects, (5) in-service irradiated and aged materials evaluations, (6) fracture toughness curve shift method, (7) special technical assistance, and (8) foreign research interactions. Report chapters correspond to the tasks. The work is performed by the Oak Ridge National Laboratory (ORNL).

2. Irradiation Effects in Engineering Materials

The objective of this task is to develop data addressing the current method of shifting the American Society of Mechanical Engineers (ASME) fracture toughness (K_{Ic} , K_{Ia} , and K_{IR}) curves to account for irradiation embrittlement in high-copper welds. The specific activities to be performed in this task are the: (1) continuation of Phase 2 of the fifth irradiation series; (2) completion of the sixth irradiation series, including the testing of nine irradiated Italian crack-arrest specimens; and (3) evaluation of irradiation effects in a commercial low upper-shelf (LUS) weld metal. The continuation of Phase 2 of the fifth series includes irradiation of HSSI Weld 73W to a high fluence [5×10^{19} n/cm² (>1 MeV)] to determine whether the small K_{Ic} curve-shape change observed in the fifth series is exacerbated. A NUREG report with the detailed results of testing and analysis of the Italian crack-arrest specimens has been prepared and transmitted to the U.S. Nuclear Regulatory Commission (NRC). A presentation describing Oak Ridge National Laboratory (ORNL) experiences in crack-arrest fracture toughness of irradiated pressure vessel steels was given at the American Society for Testing and Materials (ASTM) Symposium on User's Experience in Crack-Arrest Testing, New Orleans, November 18, 1996; the corresponding paper has been prepared and will be published in a special issue of the *ASTM Journal of Testing and Evaluation*.

Preparation of a draft NUREG report that addresses how R curve behavior of LUS materials affects the transition temperature characteristics of such materials has been initiated. This information is relevant to the consideration of the master curve analyses to design applications.

3. Annealing

The purpose of the annealing task is to evaluate the correlation between fracture toughness and Charpy V-notch (CVN) impact energy during irradiation, annealing, and reirradiation (IAR).

Five representative reactor pressure vessel (RPV) steels have been selected to evaluate their respective propensity for temper embrittlement when they are austenitized to large, prior austenite grain size. One austenitization cycle simulates an AEA Technology treatment that resulted in severe embrittlement propensity after aging for 2000 h at 450°C. A second austenitization was a simulation of a typical heat-affected zone cycle from weld processes using the Gleeble. All prior austenitization treatments and subsequent postweld heat treatments have been completed. The transition temperature in most materials was improved by the prior austenitization treatment. All materials showed temper embrittlement sensitivity after aging for 2000 h at 450°C. None showed the extreme embrittlement sensitivity reported by AEA Technology on its materials.

Three minicapsules were designed, fabricated, and loaded into the ORNL hot cell with previously irradiated and annealed CVN specimens and wire dosimeters. Three containers with retrievable, fission-radiometric, dosimetric sets (FRDS) have also been fabricated. The minicapsules and containers with the retrievable FRDS have been shipped and loaded in the University of California, Santa Barbara (UCSB), capsule at the University of Michigan Ford Nuclear Reactor (FNR). The retrievable FRDS were removed from the capsule after about 1 month of exposure and returned to ORNL and counted. The data are being used to analyze the flux and fluence at the locations in the UCSB capsule where the FRDS were placed. Work on the new IAR facilities and verification capsules has progressed to the point at which the facilities and capsules have to be stress-relieved before final checkout at temperature can begin.

4. Microstructural Analysis of Radiation Effects

This section discusses theoretical and experimental work relevant to determining the effects of neutron energy spectrum and appropriate average damage production cross sections in commercial RPVs. The experimental work is an analysis of tensile data from electron-irradiated commercial and model alloys. Electron irradiation provides a simulation of a highly thermalized neutron energy spectrum, and a comparison of these results with data on neutron irradiated materials indicates that displacements per atom (dpa) continues to be a good correlation parameter. A detailed theoretical investigation of neutron energy spectrum effects based on molecular dynamics displacement cascade simulations also indicates that the effects of neutron energy spectrum differences should be modest for most fission reactors. In particular, spectrum-averaged defect production cross sections obtained for pressurized-water reactor (PWR) and boiling-water reactor (BWR) neutron spectra were not significantly different.

5. In-Service Irradiated and Aged Materials Evaluations

During this reporting period, many of the necessary modifications to the remotely operated mill and saw were completed. The mill and saw are now ready for testing and U.S. Department of Energy (DOE) certification of the lifting bars. However, before the mill and saw can be installed in the hot cell, it must be decontaminated. The ORNL Metals and Ceramics Division has allocated funds, in addition to those from NRC, for this decontamination, and work is progressing.

6. Fracture Toughness Curve Shift Method

Preparation of a draft NUREG report on comparison of irradiation-induced Charpy impact and fracture toughness curve shifts has been initiated. Analysis of available data from the literature shows that for weld metals, on the average, the Charpy transition temperature shift at 41 J is the same as the shift of fracture toughness with a 95% confidence interval of about $\pm 30^{\circ}\text{C}$. For base metal, on the average, the fracture toughness shift is 12°C greater than the Charpy transition temperature shift at 41 J with 95% confidence intervals of about $\pm 35^{\circ}\text{C}$. The examination of the size adjustment procedure in the master curve approach is continuing for precracked Charpy and smaller size specimens. Specimens as small as $5 \times 5 \times 27$ and $5 \times 10 \times 55$ mm have been tested and analyzed.

7. Special Technical Assistance

The section describes the special analytical and experimental investigations that support the NRC in resolving regulatory research issues related to irradiation effects on materials.

Regarding the effect of long-term aging of stainless steel welds at 343°C, (1) the J-R tests of the precracked CVN specimens were completed and analyzed, (2) some reversion heat treatments were conducted followed by CVN testing, and (3) a draft NUREG report was prepared.

The review of the Materials Engineering Associates equipment to determine serviceability of each piece after its receipt at ORNL has continued. Additionally, the archival storage of the correlation monitor material was maintained.

With the completion of the fabrication of the UCSB irradiation facility during this reporting period, the equipment was successfully installed at the University of Michigan FNR, tested, and placed into operation. This facility will also be used in conjunction with other irradiations being conducted for the HSSI Program.

8. Foreign Research Interactions

During this reporting period, a preliminary matrix of tests to be performed on the Japan Power Demonstration Reactor vessel core samples was developed. Two staff members from the Reactor Component Reliability Laboratory, Tokai Research Establishment, Japan Atomic Energy Research Institute (JAERI), Yutaka Nishiyama and Kuno Onizawa, visited ORNL on January 17, 1997, to exchange information and discuss the preliminary test plan. Data obtained by JAERI also indicate that the attenuation of hardness through the depth of the wall due to self-shielding is small. In the ORNL machining plan, the data obtained from two adjacent layers will be assumed to be from the average depth of these layers.

Preliminary results of the testing of VVER-440 and VVER-1000 welds in the unirradiated, irradiated and thermally annealed conditions were presented in summary form at the Water Reactor Safety Information Meeting in Bethesda, Maryland. Additional meetings were held with Working Group 3 members from the Russian Research Center-Kurchatov Institute. These meetings resulted in the completion of a draft paper for publication in the proceedings of the Eighth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems in August 1997.

The search for a suitable container for shipment of irradiated HSSI Welds 72W and 73W specimens to Belgium was solved with the proposed design of a concrete- and lead-lined container.

For the International Atomic Energy Agency (IAEA) New Coordinated Research Program (CRP), some testing of precracked Charpy specimens (PCVNs) of HSST Plate 02 were completed and compared with previous results from 1T compact specimens. Receipt of a block of the IAEA reference steel designated JRQ is expected in the third quarter of FY 1997; results of Charpy impact and PCVN testing of JRQ will be presented at the next meeting of the New CRP in Vienna, anticipated for October 1997.

Planning for the formal collaboration with the Korean Atomic Energy Research Institute (KAERI) was initiated during the previous reporting period. The basis for such initial planning is a list of technical topics of interest provided to ORNL by KAERI researchers. No further activity has been undertaken in this subtask.

Heavy-Section Steel Irradiation Program Semiannual Progress Report for October 1996 through March 1997[†]

W. R. Corwin

1. Program Management

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

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

This 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 seven technical tasks: (1) program management; (2) irradiation effects on engineering materials;

^{*}Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, under Interagency Agreement DOE 1886-8109-8L with the U.S. Department of Energy under contract DE-AC05-96OR22464 with Lockheed Martin Energy Research Corp.

[†]The submitted manuscript has been authored by a contractor of the U.S. Government under contract DE-AC05-96OR22464. Accordingly, the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes.

(3) annealing; (4) microstructural analysis of radiation effects; (5) in-service irradiated and aged materials evaluations; (6) fracture toughness curve shift method; (7) special technical assistance; and (8) foreign research interactions.

During this period, one program briefing was presented by the HSSI staff during a visit with NRC staff, seven technical papers,¹⁻⁷ were published, and eight technical presentations⁸⁻¹⁴ were made.

References

1. R. K. Nanstad, *Report of Foreign Travel to Austria, Ukraine, and Russia*, ORNL/FTR-5905, Oak Ridge National Laboratory, Oak Ridge, Tenn. Oct. 7, 1996.
2. S. K. Iskander, M. A. Sokolov, and R. K. Nanstad, "Effects of Annealing Time on the Recovery of Charpy V-Notch Properties of Irradiated High-Copper Weld Metal," pp. 277-293 in *Effects of Radiation on Materials: 17th International Symposium, ASTM STP 1270*, D. S. Gelles, R. K. Nanstad, A. S. Kumar, and E. A. Little, eds., American Society for Testing and Materials, 1996.
3. M. A. Sokolov and R. K. Nanstad, "On Impact Testing of Subsize Charpy V-Notch Type Specimens," pp. 384-414 in *Effects of Radiation on Materials: 17th International Symposium, ASTM STP 1270*, D. S. Gelles, R. K. Nanstad, A. S. Kumar, and E. A. Little, eds., American Society for Testing and Materials, 1996.
4. M. A. Sokolov, R. K. Nanstad, and S. K. Iskander, "Effects of Thermal Annealing on Fracture Toughness of Low Upper-Shelf Welds," pp. 690-705 in *Effects of Radiation on Materials: 17th International Symposium, ASTM STP 1270*, D. S. Gelles, R. K. Nanstad, A. S. Kumar, and E. A. Little, eds., American Society for Testing and Materials, 1996.
5. M. A. Sokolov, *Report of Foreign Travel to The Netherlands*, ORNL/FTR-5937, Nov. 18, 1996.
6. R. K. Nanstad et al., "Effects of Thermal Annealing and Reirradiation on Toughness of Reactor Pressure Vessel Steels," pp. 99-110 in *Proc. U.S. Nuclear Regulatory Commission, Twenty-Fourth Water Reactor Safety Information Meeting*, NUREG/CP-0157, Vol. 2, Washington, D.C., February 1997.
7. W. R. Corwin and B. L. Broadhead, "Collaborative Investigations of In-Service Irradiated Material from the Japan Power Demonstration Reactor Pressure Vessel," pp. 49-50 in *Proc. U.S. Nuclear Regulatory Commission, Twenty-Fourth Water Reactor Safety Information Meeting*, NUREG/CP-0157, Vol. 2, Washington, D.C., February 1997.
8. M. A. Sokolov and R. K. Nanstad, "Comparison of Irradiation-Induced Shifts of K_{Jc} and Charpy Curves," presented by M. A. Sokolov at the Aging Materials and Evaluation (AMES) Task Group TG1C First Workshop on Mechanical Property-Property Correlations, sponsored by the Institute for Advanced Materials of the Joint Research Centre of the Commission of European Communities, Petten, Netherlands, Oct. 1-2, 1996.
9. W. R. Corwin, "Collaborative Investigation of In-Service Irradiated Material From the Japan Power Demonstration Reactor," presented at the 24th Water Reactor Safety Information Meeting, Bethesda, Maryland, Oct. 21, 1996.

10. S. K. Iskander et al., "Effects of Thermal Annealing and Reirradiation on Toughness of Reactor Pressure Vessel Steels," presented by R. K. Nanstad at the 24th Water Reactor Safety Information Meeting, Bethesda, Maryland, Oct. 21, 1996.
11. S. K. Iskander, E. T. Manneschildt, P. Milella, and A. Pini, "Crack-Arrest Testing of Irradiated Nuclear Reactor Pressure Vessel Steels at the Oak Ridge National Laboratory," presented by S. K. Iskander at ASTM Committee E-8 Workshop on User's Experience in Crack-Arrest Toughness Testing, New Orleans, Nov. 18, 1996.
12. M. A. Sokolov, D. E. McCabe, R. K. Nanstad, and Y. A. Davidov, "The Use of Precracked Charpy and Smaller Specimens to Establish the Master Curve," presented by M. A. Sokolov at the ASTM Symposium on Small Specimen Test Techniques, New Orleans, Jan. 13-14, 1997.
13. D. J. Alexander, "Fracture Toughness Testing with Subsize Compact Specimens Through the Transition Temperature Region," presented by M. A. Sokolov at the ASTM Symposium on Small Specimen Test Techniques, New Orleans, Jan. 13-14, 1997.
14. S. K. Iskander, ORNL personal communication with Y. Nishiyama and K. Onizawa, Reactor Component Reliability Laboratory, Tokai Research Establishment, Japan Atomic Energy Research Institute, Jan. 17, 1997.

2. IRRADIATION EFFECTS IN ENGINEERING MATERIALS

D. E. McCabe

2.1 Fracture Toughness Shifts in High-Copper Weldments (Series 5 and 6) (S. K. Iskander)

The objective of this subtask is to develop data addressing the current method of shifting the American Society of Mechanical Engineers (ASME) fracture toughness (K_{Ic} , K_{Ia} , and K_{IR}) curves to account for irradiation embrittlement in high-copper welds. The specific activities to be performed in this task are: (1) the continuation of Phase 2 of the fifth irradiation series and (2) the completion of the sixth irradiation series, including testing of the nine irradiated Italian crack-arrest specimens. The continuation of Phase 2 of the Fifth Series includes irradiation of HSSI Weld 73W to a high fluence [5×10^{19} n/cm² (>1 MeV)] to determine whether the K_{Ic} curve shape change observed in the fifth series is exacerbated. No additional work is currently funded in this subtask.

Testing was previously completed on nine irradiated Italian crack-arrest specimens fabricated from a 0.06% Cu forging that conforms to the American Society for Testing and Materials (ASTM) Specification for Quenched and Tempered Vacuum-Treated Carbon and Alloy Steel Forgings for Pressure Vessels, Class 3 (A 508-81 Class 3). A significant result from the tests is that both Charpy and crack-arrest fracture toughness shifts because of neutron irradiation, to a level of 3.5×10^{-3} n/cm² (>1 MeV) at approximately 288°C, were similar and small, averaging about 10 K. A NUREG report with detailed results and analysis, *Results of Crack-Arrest Tests on Irradiated A 508 Class 3 Steel* [NUREG/CR-7477 (ORNL-6894)], by S. K. Iskander, P. P. Milella, and A. Pini, has undergone final editing preceding its transmittal to the NRC for printing.

A presentation was given by S. K. Iskander at the ASTM Symposium on User's Experience in Crack-Arrest Testing, New Orleans, November 18, 1996, "Crack-Arrest Testing of Irradiated Nuclear Reactor Pressure Vessel Steels at Oak Ridge National Laboratory," by S. K. Iskander, P. P. Milella, A. Pini, and E. T. Mannesmidt. The corresponding paper has been prepared for publication in a special issue of the *ASTM Journal of Testing and Evaluation*.

2.2 Irradiation Effects in a Commercial LUS Weld (D. E. McCabe)

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

The technology of establishing the transition range fracture toughness for low upper-shelf materials requires that some special attention be given to J-R curve effects. There is a point in the transition range, as upper shelf is approached, where J-R curve properties start to impact the K_{Jc} transition range toughness values. It is currently understood that specimen size effects and the weakest-link postulate that is used to model size effects operates within a restricted range within the transition. Upper-shelf R curves show minimal to no size effects. In addition, R curves show sensitivity to side grooving (Figure 2.1), whereas transition range K_{Jc} data have shown no side groove effects. Hence, side groove dependence in K_{Jc} data is expected to develop as upper-shelf temperatures are approached. In particular, when LUS materials are applied to master curve analysis, some difficulties in data development and interpretation are to be expected. The K_{Jc} data developed at or near the reference temperature, T_0 , for master curve placement ultimately suggest a temperature range of high fracture toughness to levels well beyond the maximum fracture toughness capability of the material. Therefore, it follows that some information on R curves should be introduced into master curve plots. One possible approach is suggested in Figure 2.2. In this case, K_R values selected from K_R curves developed at three test temperatures are plotted as filled squares. These are K_R values after 1.25 mm (0.05 in.) of slow-stable crack growth, which represents 5% of slow-stable crack growth in a 1T C(T) specimen with an initial crack aspect ratio, a_0/W , of 0.5. A line drawn through these points and projected through the master curve suggests a truncation fracture toughness level for a usable part of the master curve. Here, we can note a paradoxical situation with the K_{Jc} data plotted at 0°C. These data (at 0°C) have two K_{Jc} values from 1/2T C(T) specimens, nine from 1T C(T) specimens (six of which were not side grooved), and four from 2T C(T) specimens (none of which were side grooved). All data have been converted to 1T equivalence. The above data development was made long before the impact of the R curve was known, and the lack of side grooving means that most of these specimens were following a non-side-grooved R curve up to K_{Jc} instability. This elevated the toughness capability for these specimens above that of side-grooved specimens. Size adjustment of data from 2T specimens to 1T equivalence at a temperature at which the presence of weakest-link effects is somewhat questionable has resulted in the three highest values shown at 0°C. This same phenomenon is one of the subjects for discussion in the technical basis document for the ASTM test method on the determination of reference temperature, T_0 .

Irradiation Effects on the R Curve

Even though beltline and nozzle course welds were supposed to be identical WF-70 weld materials, the fracture-mechanics-based data had detected some difference. The reference temperatures differed by about 25°C and the R curves also differed marginally, as shown in Figure 2.3. Most R curve tests were made at 150°C, a temperature which is safely on the upper shelf for both unirradiated and irradiated [up to 1×10^{19} n/cm² (>1 MeV)] conditions. Before irradiation, side-grooved specimens of both the beltline and nozzle course weld material developed full R curves with no instabilities up to the discontinuation of loading. After irradiation at 1×10^{19} n/cm² (>1 MeV), three of nine specimens suffered crack instabilities just beyond maximum load. This developed even though both materials were about 100°C or more above the T_0 temperature (Figure 2.4). Judging by the path of unloading in load vs displacement, the evidence suggests crack instability resulting from fast ductile tearing, which was triggered by excessive stored elastic strain energy in the specimen. For this to happen, the K_R curves must have a flat upper plateau. Figure 2.5 shows the changed R curve in the beltline weld material and Figure 2.6 indicates the same for the nozzle course weld. Certainly, both materials have the type of R curve just described, and the upper-bound K_J value is low—low enough such as to trigger ductile tearing instability and maintain an uncontrolled running crack.

Figure 2.1. R curves for LUS WF-70 weld metal showing the effect of side grooving on resistance to ductile tearing.

ORNL DWG 97-128220

Figure 2.2. Truncation of master curve using 5% of stable crack growth as the limitation value.

Figure 2.3. Comparison of R curves for WF-70 beltline and nozzle course weld metal.

Figure 2.4. Load versus displacement test record of irradiated nozzle course weld metal. The test ended with ductile instability.

Figure 2.5. R curves of WF-70 weld metal before and after irradiation. Note the changed slope in the upper plateau of the K_R trend.

ORNL DWG 97-128224

Figure 2.6. Postirradiation R curve for WF-70 nozzle course weld metal. Note the flat upper plateau.

Conclusion

The R curve behavior of materials affects the shape of K_{Jc} data trends and specimen performance characteristics in the upper part of the transition range. The consequences of this behavioristic peculiarity of fracture-mechanics-based data have not been given adequate study.

3. ANNEALING

S. K. Iskander

3.1 Temper Embrittlement in Reactor Pressure Vessel Steel Heat-Affected Zones (D. E. McCabe)

Attention is now being given to determine if there is a potential problem with temperature embrittlement in RPV steels. An annealing experiment on laboratory heats made with steels having typical pressure vessel chemical compositions was conducted by AEA Technology, Harwell, United Kingdom. AEA showed quite clearly that there can be grain boundary embrittlement in RPV steels given large prior austenite grain size and high phosphorus on the order of 0.017 wt %. The first task in this project will be to reexamine the AEA Technology heat treatments using five commercially made RPV steels, representing A 302 grade B, A 533 grade B, A 508 class 2, and two modified A 302 grade B production heats (Table 3.1). The phosphorus content covers the range typical of commercial RPV production heats. The AEA Technology heat treatment accentuated the temper embrittlement phenomenon by creating a microstructure that optimizes embrittlement sensitivity. This is a preferred method for an experimental objective of screening materials.

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

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

This report will cover only the first task previously mentioned. The five commercial heats of RPV steel were obtained as archival materials left over from several previous projects. Each material was sectioned into four blocks of a convenient size for AEA Technology austenitization, nominally 51 × 56 × 127 mm (2 × 2.2 × 5 in.) (Table 3.2). All blocks of the five materials were austenitized at 1204°C (2200°F) with a 30-min soak time followed by an oil quench. Then all were given postweld heat treatments at 615°C (1140°F) for 24 h and again followed by an oil quench.

Three thermal embrittlement cycles are to be evaluated: (1) age for 2000 h at 450°C (482°F); (2) age for 168 h at 490°C (914°F); and (3) age for 2000 h at 450°C, then followed with heat maintained at 490°C for 168 h (Table 3.3).

The second austenitization treatment applied is a Gleeble simulation of the thermal cycle of heat-affected zone material. In this case, the specimens were 14-mm-diameter (0.564-in.) rods. The austenitizing temperature was 1260°C (2300°F), and the entire cycle was completed within 50 s. The target ASTM grain size was between 4 and 5, which is typical for the coarse-grain region of HAZs. All were postweld heat-treated at 615°C (1140°F) for 24 h. One thermal embrittlement cycle was applied; 450°C for 2000 h. It has been completed. An additional aging cycle at 450°C for 168 h is planned.

Table 3.1. Commercial materials selected

Material	Code	Content (wt %)			
		P	Ni	S	Cu
A 302 Grade B	Maine Yankee	0.015	0.2	0.017	0.14
A 508 Class 2	Midland	0.010	0.76	0.015	0.02
Modified A 302 Grade B	GE (Z5)	0.016	0.62	0.015	0.17
Modified A 302 Grade B	GE (Z7)	0.007	0.53	0.014	0.16
A 533 Grade B	HSST Plate 01	0.018	0.75	0.013	<i>a</i>
^a Copper content to be determined.					

Table 3.2. Prior austenite grain growth (two methods)

Method	Description
AEA Technology	Austenitize 1204°C, 30 min, oil quench
	PWHT 616°C, 24 h, oil quench (ASTM G.S. = 00-0)
Gleeble simulation of LBZ cycle	Austenitize 1260°C, 10 s, cool to 400°F, 40 s
	PWHT 616°C, 24 h, oil quench (ASTM G.S. = 4-5)

Table 3.3. Program plan for aging cycles

Aging temperature (°C)	Aging time (h)	Grain coarsening cycle	
		AEA	Gleeble
Initial	None	✓	✓
450	168		✓
450	2000	✓	✓
490	168	✓	
450/490	2000/168	✓	

The austenitization treatments shown in Table 3.2 for the five commercial steels have been completed. The resulting prior austenite grain sizes from the AEA Technology cycle were a mix of ASTM 0 to 00 grains and ASTM 4 to 5 by the Gleeble simulation. Examples are shown in Figures 3.1 and 3.2.

The CVN transition temperature curves were determined before and after the AEA Technology austenitization cycle. It was decided that the best measure of ΔT_T would be at the 50% transition energy level, which would eliminate false effects caused in some cases by a change in CVN upper-shelf energy. The transition temperature shift established on this basis is presented in Table 3.4. Despite an expected detriment in transition temperature resulting from increased grain size, the transition temperature was generally improved by the AEA Technology austenitization cycle. In this particular case, the severe oil quench from the austenitizing temperature produced beneficial prior martensite and/or lower bainite that more than offset the large grain effect.

An appreciable number of the aging treatments and subsequent CVN transition curves have been completed. Table 3.5 previews the findings to date. Of the five materials evaluated, only the modified A 302 grade B, designated "low phosphorus," had significantly lower phosphorus than the others. Nevertheless, this steel had suffered the greatest temper embrittlement damage.

All five materials were aged for 2000 h at 450°C in the as-received metallurgical condition; compare the two bottom rows in Table 3.5. Except for the A 302 grade B, the sensitivity to embrittlement is almost negligible when compared to the other grain-enlarged conditions of all other materials listed in the table. The two blank rows in Table 3.5 represent work in progress.

ORNL PHOTO 96-1250-2

Figure 3.1. Prior austenite grain size of A 533 grade B after AEA cycle (100X).

Figure 3.2. Prior austenite grain size of A 533 grade B after Gleeble cycle (100X).

Table 3.4. Preliminary results
(ΔTT at mid-transition, CVN energy) of effects
of AEA Technology austenitize cycle

Material	ΔTT (°C)
A 302B	9
A 533B	-87
A 508	4
Modified A 302B (high P)	-14
Modified A 302B (low P)	-64

Table 3.5. Transition temperature, T_0 (50% energy)

Austenitizing method	Aging temperature (°C)	Aging time (h)	Transition temperature (°C)			Modified A 302B (°C)	
			A 302B	A 533B	A 508 class 2	High P	Low P
AEA	Initial	Initial	9	-67	4	-23	-73
	450	2000	45	78	41	34	42
	490	168	44	65	16	16	-3
Gleeble	Initial	Initial					
	450	2000	43	66	-46	-13	-4
	450	168					
As-received	Initial	Initial	0	20	0	-9	-9
	450	2000	64	38	10	-4	10

3.2 Annealing Effects in LUS Welds (Series 9) (S. K. Iskander, M. A. Sokolov, D. W. Heatherly, R. L. Swain, D. Sparks, M. T. Hurst, and R. G. Sitterson)

The purpose of the ninth irradiation series is to evaluate the correlation between fracture toughness and CVN impact energy during irradiation, annealing, and reirradiation (IAR).

3.2.1 Reirradiation of Previously Irradiated and Annealed Charpy Specimens

Space became available in the University of California, Santa Barbara (UCSB), capsule at three fluence levels when it was installed in the University of Michigan Ford Nuclear Reactor (FNR). The highest fluence level was estimated to be approximately one order of magnitude greater than the lowest, and the intermediate fluence level approximately midway between the highest and lowest. These fluence levels are estimates since no dosimetry had yet been performed on the new capsule. In collaboration with Professors Odette and Lucas of UCSB, plans were made to place three sets of 10 CVN specimens, the maximum number possible in the space available, at each of these locations. HSSI Weld 73W CVN specimens that had been previously irradiated and annealed (IA) at 454°C were chosen since reirradiation in the UCSB capsule would provide data on the rate of reembrittlement at three different fluence levels in about 1 year.

Three minicapsules to contain, ship, and reirradiate the IA Charpy specimens and wire dosimeters have been designed and machined. The minicapsules were loaded in the ORNL hot cell with the 30 IA CVN specimens, and wire dosimeters were placed in the V-grooves of the Charpy specimens. Special holders were also designed and loaded with retrievable fission-radiometric dosimetric sets (FRDSs).

The minicapsules and retrievable FRDS were shipped to the FNR in early January 1997. The loading of the three minicapsules and retrievable FRDS into the UCSB capsule was performed by the FNR staff in mid-January 1997 with active HSSI staff participation. The FNR commenced irradiation of the UCSB capsule on Jan. 20, 1997.

One of the minicapsules with 10 of the CVN specimens and one of retrievable FRDS were placed in Cell Positions D1/D2 to be subjected to the relatively highest fluence. For the intermediate and low fluence, the two remaining mini-capsules each with 10 CVN specimens were loaded in Cell Position G1/G2 and H1/H2, respectively. The UCSB capsule was designed with logical cells of 50-mm (2-in.) height and width, and a 25-mm (1-in.) depth. The length of CVN specimens is 55 mm (2.164 in.); consequently, the minicontainer and the retrievable FRDS are greater than the length of the UCSB individual cells. The "spillover" due to the length of the packages was accommodated in Cell Positions I1 and I2.

The FRDSs were retrieved from the UCSB capsule after 4 weeks irradiation and shipped to ORNL for counting and analysis to determine flux and spectrum. This determination will also enable estimates to be made as to the required time for the HSSI IA CVN specimens to accumulate the target fluence. Three "dummy" inserts of low carbon steel with the same external dimensions as the three FRDS have been loaded in place of the ones removed. The three containers each with ten IA Charpy specimens of HSSI Weld 73W are still in the UCSB capsule.

3.2.2 New HSSI IAR Facility

A new facility, designed to IAR in situ, had been fabricated and assembled, and was awaiting stress-relief heat treatment and testing. The UCSB capsule was placed beside the IAR facility. Because of contractual time constraints, all resources were devoted to completing the UCSB capsule instead. A surrogate facility was fabricated instead of the IAR one to support the UCSB capsule. The surrogate facility is a dummy one that does not contain the heating elements and temperature controls. Drawings for the surrogate facility were prepared, purchase requisitions were placed, and a facility was fabricated. The surrogate facility relieved the Engineering Technology Division Irradiation Engineering Group (IEG) from having to work on both the UCSB capsule and the IAR facility at the same time. The IEG located surplus stainless steel for the fabrication of the dummy facility at the Y-12 Plant, saving time and money to purchase the required stainless steel from a vendor, which would have delayed fabrication of the dummy facility.

The IAR facilities and dosimetry and temperature verification capsules had been previously moved from the IEG assembly laboratory to the Instrumentation and Controls (I&C) Division. Then, the heaters were connected to the power supply, the thermocouples were connected to the data acquisition and control instrumentation, and the system was checked out. The IAR facility, together with the dosimetry and temperature verification capsules, are to be stress relieved at 460°C to ensure that no significant distortion occurs when the facility is heated up to this temperature during an in situ anneal. The clearances between the capsules and facility are such that if significant distortion occurs, then the capsules cannot be removed from the facility. Testing of the IAR facility required the preparation of a vessel and a heat exchanger to simulate conditions at the FNR site.

The design of a reusable capsule to contain fracture toughness and CVN specimens, either in the un- or irradiated conditions was also put on hold during this period. About 20 drawings have been prepared and are awaiting review.

3.2.3 Instrumentation for Control and Data Acquisition

An I&C engineer traveled to the FNR (a) to survey changes that have occurred at the reactor site that will affect electrical connections and (b) to determine power requirements. As a result of the changes at FNR, the HSSI IAR J-Box needs to be modified. The computer that was at the FNR site has been brought back and rebuilt with a new mother board, disk drives, memory, and power supply to accommodate new WindowsTM-based software.

4. Microstructural Analysis of Radiation Effects

R. E. Stoller and K. Farrell

4.1 Introduction

Differences in neutron energy spectra are manifested as differences in the energy spectra of the primary knockon atoms (PKAs) that are produced. Low-energy PKAs, such as those produced by thermal neutrons, are somewhat more efficient at net defect production than are high-energy PKAs. This suggests that these displacements should be more heavily weighted in dosimetry. Conversely, the lower energy PKAs produce fewer point defect clusters that can promote the formation of extended defects that are responsible for mechanical property changes. In order to derive an effective damage production cross section, a method of weighting the high- and low-energy contributions must be devised.

Current theoretical and experimental work relevant to determining the effects of neutron energy spectrum is described below, with a focus on the modeling. The experimental work is an analysis of tensile data from electron-irradiated commercial and model alloys. Electron irradiation provides a simulation of a highly thermalized neutron energy spectrum, and a comparison of these results with data on neutron irradiated materials indicates that dpa continues to be a good correlation parameter. A detailed theoretical investigation of neutron energy spectrum effects based on molecular dynamics displacement cascade simulations also indicates that the effects of neutron energy spectrum differences should be modest for most fission reactors. In particular, spectrum-averaged defect production cross sections obtained for PWR and BWR neutron spectra were not significantly different.

4.2 Experimental Investigation of the Effects of Primary Knockon Atom Energy Spectrum

ORNL standard SS3 minitensile specimens were irradiated at $\sim 290^\circ\text{C}$ with 2.5-MeV electrons to doses of 9.5×10^{-4} and 3.2×10^{-3} dpa, equivalent to fast neutron fluences of about 0.63 and 2.1×10^{22} n/m², respectively. The irradiations were done at no cost by staff at the Institut für Festkörperforschung, Jülich, Germany, as part of a collaborative effort to determine the relative importance of incascade clustering and point-defect formation. Two engineering and two model alloys were irradiated: the HSST Plate-02 correlation monitor and early RPV alloy A212B and pure iron and an Fe-0.28 Cu binary alloy.

A direct comparison of electron- and neutron-irradiated materials would provide information on neutron energy spectrum effects and input for the ORNL embrittlement model. At this time, the HSST Plate-02 provides the best such comparison. When compared on the basis of dpa, the change in yield strength observed in the HSST Plate-02 correlation monitor after 2.5-MeV electron irradiation is comparable to that obtained under neutron irradiation. This implies that cascades are not required to obtain hardening at low doses, but also that the overall damage efficiency of low-energy displacements is not significantly higher than high-energy displacements. This assumption is illustrated in Figure 4.1, where it shows tensile data for HSST Plate-02 irradiated by 2.5-MeV electrons and neutrons. The neutron irradiation data for Plate-02 (Cu = 0.14) shown in Figure 4.1, were obtained from previous HSST experiments in the Oak Ridge Research Reactor (ORR) and Bulk Shielding Reactor (BSR). Additional data are shown for similar plate steels irradiated in commercial reactor surveillance programs.

Figure 4.1. Comparison of tensile data on HSST Plate-02 and similar materials from commercial reactor surveillance programs and 2.5-MeV electron irradiation.

The data from the electron irradiations fall well within the scatter band of the neutron data. This conclusion should be confirmed by higher dose electron data.

4.3 Modeling and Simulation of Neutron Energy Spectrum Effects

The results of molecular dynamics (MD) simulations of displacement cascades with energies up to 40 keV¹⁻³ have been used to characterize the energy dependence of two measures of primary damage production: (1) the number of surviving point defects expressed as a fraction of the those predicted by the standard secondary displacement model by Norgett, Robinson, and Torrens (NRT)⁴, and (2) the fraction of the surviving interstitials contained in clusters that formed during the cascade event. The former ratio is important because it is only the surviving point defects that can contribute to radiation-induced microstructural evolution. The latter is significant because these small clusters provide nuclei for the growth of larger defects which can give rise to mechanical property changes. The formation of these small clusters directly within the cascade means that the extended defects can evolve more quickly than if the clusters could only be formed by the much slower process of classical nucleation. Then, the SPECOMP and SPECTER codes^{5,6} were used to compute effective cross sections for point defect survival and point defect clustering. PKA spectra for iron obtained from SPECTER were used to weight these effective cross sections in order to calculate spectrum-averaged values for various neutron irradiation environments, including several locations in both water- and sodium-moderated fission reactors.

The molecular dynamics code, MOLDY, and the interatomic potential for iron that was used in this work have been described in previous reports, and detailed descriptions can be found in Refs. 1-3 and 7-10. Notably, this work has identified for the first time the effect of subcascade formation on defect survival at high PKA energies^{2,3}. The surviving defect fraction is a smoothly decreasing function of PKA energy up to about 30 keV, above this energy the survival fraction begins to increase slightly and may saturate. This helps minimize the effect of differences in neutron energy spectra.

A nonlinear, least-squares fit to the energy dependence of the surviving defect fraction (η) and the interstitial clustering fraction (f_{icl}) from the MD simulations is given by Eqs. (1) and (2), respectively.

$$\eta = 0.5608 \cdot E_{MD}^{(-0.3029)} + 3.227 \times 10^{-3} \cdot E_{MD} \quad (4.1)$$

$$f_{icl} = [0.097 \cdot \ln(E_{MD} + 0.9)]^{(0.3859)} - 7 \times 10^{-6} \cdot E_{MD}^{2.5} \quad (4.2)$$

The second term in both equations reflects the effects of subcascade formation mentioned previously. This formation leads to a minimum in the energy dependence of defect survival, and a maximum in the interstitial clustering fraction. This reaction occurs at the MD cascade energy of about 20 keV, which corresponds to a PKA energy of about 30 keV. These changes occur because subcascade formation makes a single high-energy cascade appear to be the equivalent of several lower energy cascades. Thus, the defect survival fraction is slightly higher and the interstitial clustering fraction slightly lower at 40 keV than at 20 keV. No cascades with energies higher than 40-keV have been completed at this time. Based on the degree of subcascade formation observed in the 40 keV cascade simulations, it appears unlikely that η and f_{icl} will change significantly at higher cascade energies. Therefore, the

40-keV values obtained from the data fitting were applied for all the higher energy cascades in the SPECTER calculations.

The defect production cross sections for surviving MD defects and clustered interstitials were generated by modifying the SPECOMP⁵ and SPECTER⁶ computer codes. SPECOMP normally calculates displacement cross sections for compounds using the PKA recoil energy distributions contained in a 100-neutron-energy by 100-recoil-energy grid for each of 40 different elements. For the present defect and clustered interstitial calculations, Eqs. (1) and (2) were included in the codes as a factor multiplying the standard Norgett-Robinson-Torrens (NRT) displacement cross section. SPECOMP thus produced surviving defect and clustered interstitial cross sections on a 100-point neutron energy grid.

The SPECTER computer code contains libraries of calculated cross sections for displacements, gas production, total energy distribution, and atomic recoil energy distributions for over 40 elements and various compounds. For a given neutron energy spectrum and irradiation time, the code can calculate the net radiation damage effects, as given above. In the current case, the SPECOMP calculations for the surviving defect and clustered interstitial cross sections were added to the SPECTER libraries. SPECTER runs for various power and research reactor neutron spectra thereby produced spectral-averaged values for the point defect and interstitial clustering fractions.

The primary results of these calculations are summarized in Figure 4.2. The PKA-spectrum-averaged defect survival fraction is shown in Figure 4.2a, and the interstitial clustering fraction in Figure 4.2b. In both cases, the effective production cross section has been divided by the NRT dpa cross section. Values are shown for the 1/4-T and 3/4-T RPV positions from representative PWR and BWR neutron spectra. Four additional irradiation sites are also shown: positions located in the peripheral target position (PTP) and removable beryllium (RB*) reflector of the High Flux Isotope Reactor (HFIR) at ORNL, and the midcore and below-core (BC) positions in the Fast Flux Test Facility (FFTF) at the DOE Hanford Reservation.

Including the HFIR and FFTF positions provides a broad comparison of both hard and soft neutron spectra. The former is a very high-flux, water-moderated test reactor, and the latter is a liquid-metal-(Na) moderated fast reactor. In spite of the differences between the neutron energy spectra, only relatively small differences are observed in the two spectrum-averaged damage cross sections. This finding is consistent with the normalized iron PKA spectra shown in Figure 4.3 for these same environments. In the region wherein the PKA probability is highest (i.e., for PKA energies between 1 keV and 0.1 MeV), the spectra are relatively similar.

A slight difference between the PWR and BWR spectra can be seen in the way the spectra change as a function of thickness through the pressure vessel. This change is illustrated in Figure 4.4, which shows the spectrum-averaged defect production cross sections for four positions: the last water node point before the RPV, the 1/4 and 3/4 RPV locations, and the first node point in the cavity beyond the RPV. The values for the two reactor types are most similar at the pressure vessel wall and then diverge somewhat at deeper depths. The reason for this modest divergence is a subtle difference in the neutron energy spectra and the way in which neutron attenuation occurs. The BWR spectrum is widely considered to be softer than the PWR; therefore, it would be expected to give a higher defect survival fraction. However, the average PKA energy at the 1/4-T and deeper positions is actually somewhat higher than for the BWR. This difference can be rationalized by a comparison of the neutron spectra. If the energy-dependent fluxes are normalized using their respective peak values, the BWR spectrum shows both a higher relative thermal flux and a higher relative fast flux than does the PWR spectrum.

Figure 4.2. Comparison of spectrally averaged damage production cross sections (per NRT dpa) for various irradiation environments; defect survival ratio is shown in (a) and the interstitial clustering fraction is shown in (b).

Figure 4.3. Normalized iron PKA spectra for PWR and BWR 1/4-T and 3/4-T positions and the HFIR PTP position.

Figure 4.4. Variation of spectrally averaged damage production cross sections (per NRT dpa) through the RPV for PWR and BWR.

Although it is not yet possible to simulate the highest energy displacement cascades that are generated in the materials used in fission reactors, the analysis of MD cascade simulations in iron for energies up to 40 keV provides considerable insight. The primary damage parameters derived from the MD results exhibit a strong dependence on cascade energy up to 10 keV, but this dependence is diminished and slightly reversed between 20 and 40 keV. This reversal is due to the formation of well-defined subcascades in this energy region. Analysis of the cascades indicates that little further change should occur at higher energies, suggesting that the results reported herein should be relevant to an evaluation of neutron energy spectrum effects. Notably, the spectrally averaged defect production cross sections calculated for several fission reactor neutron spectra were all quite similar.

Acknowledgments

The assistance of L. R. Greenwood from the Materials and Chemistry Division of the Pacific Northwest National Laboratory in carrying out the SPECTER and SPECOMP calculations and of P. Jung and H. Ullmaier from the Institut für Festkörperforschung, Jülich, Germany, in carrying out the electron irradiations is gratefully acknowledged.

References

1. R. E. Stoller, *J. Nucl. Mater.* 233-237, 999-1003 (1996).
2. R. E. Stoller, *JOM* 48 23-27 (1996).
3. R. E. Stoller, G. R. Odette, and B. D. Wirth, "Primary Defect Formation in bcc Iron," presented at the International Workshop on Defect Production, Accumulation, and Materials Performance in Irradiation Environments, Davos, Switzerland, Oct. 2-8, 1996, submitted to *J. Nucl. Mater.*
4. M. J. Norgett, M. T. Robinson, and I. M. Torrens, *Nucl. Eng. and Des.* 33 50-54 (1975).
5. L. R. Greenwood, "SPECOMP Calculations of Radiation Damage in Compounds," pp. 598-602, *Reactor Dosimetry: Methods, Applications, and Standardization*, ASTM STP 1001, H. Farrar IV and E. P. Lippincott, eds., American Society of Testing and Materials, West Conshohocken, Penn., 1989.
6. L. R. Greenwood and R. K. Smither, *SPECTER: Neutron Damage Calculations for Materials Irradiations*, ANL/FPP/TM-197, Argonne National Laboratory, Argonne, Ill., January 1985.
7. M. W. Finnis, *MOLDY6-A Molecular Dynamics Program for Simulation of Pure Metals*, AERE R-13182, UK AEA Harwell Laboratory, 1988.
8. M. W. Finnis and J. E. Sinclair, "A Molecular Dynamic Study of Displacement Cascades in Iron," *Phil. Mag.* A50 45-55 (1984) and "Erratum," *Phil. Mag.* A53 161 (1986).
9. A. F. Calder and D. J. Bacon, "A Comparison of Displacement Cascades in Cu and Fe Molecular Dynamic and Its Application to Microstructural Evaluation," *J. Nucl. Mater.* 207 25-45 (1993).
10. W. J. Pythian, R. E. Stoller, A. J. E. Foreman, A. F. Calder, and D. J. Bacon, "A Simple Empirical-N-Body Potential Transition Metals," *J. Nucl. Mater.* 223 245-261 (1995).

5. IN-SERVICE IRRADIATED AND AGED MATERIALS EVALUATIONS

S. K. Iskander

5.1 Remotely Operated Machining Center (S. K. Iskander and L. E. Creech)

During this reporting period, modifications of the remotely operated mill and saw, preparation of the hot cell for these items, and the decontamination of cell 6, which is scheduled to receive the mill and saw, continued to progress.

Before the saw and computer numerically controlled (CNC) machine can be installed, the decontamination of cell 6 must be completed. Funds other than those from the NRC were allocated by the M&C Division for sharing the cost of decontaminating hot cell 6, which became contaminated from work performed some time ago. Preliminary time estimates for the decontamination are 12 to 13 weeks.

The following work was initiated and most of it completed:

1. Assemble the saw and stand, and install the electrical system.
2. Fabricate cable assemblies for the mill and saw.
3. Assemble the electrical cables and mechanical controls and tubing into feedthroughs.
4. Install the mill into the chip containment enclosure and mount the controls on the cart.
5. Relocate the mill wiring into the new enclosure.

The saw and stand assembly has been completed, and the electrical system has been installed and checked. The saw has been tested electrically and functions as intended; it is ready for operational development. A variety of saw blades has been received, and preliminary saw characterization for specimen production has started. Saw feed rates and blade speeds have been characterized for no load. The saw will be tested under operating conditions using a 75-mm-diam low-alloy carbon-steel bar, and at the same time the cutting time will be estimated.

Installation of the milling machine into the chip containment enclosure has been completed. A number of minor sheet-metal modifications were required. Lighting and temporary blowout solenoid mounting was also completed. The turret back plate was modified to fit. The mill terminal box has been mounted to the new chip enclosure. All components have been reconnected electrically and are ready to test. Preliminary operational testing has identified a problem with the "door open" interlock switch. The switch was removed, but simply shorting the input doesn't seem to satisfy the controller. Work on this item is continuing.

The cell 6 electrical enclosure has been installed, and the long "conduit-run" cables have been pulled. The cell "feed-through runs" are ready for installation as soon as cell decontamination is completed. The details of the enclosure back-plate component installation have been changed to allow nearly all cable terminations to be completed before the lube pump and solenoid valves are removed from the development setup. This step will reduce the time required to move the equipment into cell 6 after operational development is done. The Cell 6 rear-electrical-enclosure layout was modified so that (a) all components can be installed from the front and (b) panel removal will not be required. This enables conduit run and feedthrough run cable termination to proceed. The installation design was

changed to use existing conduit routing so that no additional conduit will be required. The long "conduit-run" cables have been terminated at the rear of cell 6. This activity was completed in less than the estimated time and has offset time overruns for the enclosure installation.

A suitable scale and a "balance bar" were used to determine the as-built centers of gravity and weights. The saw and mill were measured for weight and centers of gravity for lift fixture development. Stress analysis of preliminary lifting frame geometries for both saw and mill was completed. The mill was about 20% over the estimated weight, so the lifting bar stock, 1 1/4-in. pipe, was increased to Schedule 80. That and other drawing-dimensional details were updated for the measured center-of-gravity location, and fabrication was completed. The saw lift fixture has been completed and installed. Certification of lifting fixtures of both saw and mill, according to DOE and ORNL procedures, is still required. The lift fixtures will be tested at 125% of rated capacity when they are both completed.

A topical outline of manufacturing operational development activities has also been completed. A manufacturing operational development plan is necessary because of the extraordinary cost of hot cell time and the value of specimen material. Such a plan, based on manufacturing science methodologies for process quality control, is expected to be cost effective and is recommended. Also in progress is operational testing of the mill and saw before they are installed into the hot cell.

Work on the "as-built" drawings of the saw and mill is progressing. This is an important step, since once installed in the cell, any maintenance of the saw and mill must rely on the as-built drawings.

The craft work and engineering expenditure are slightly exceeding the plan estimate, but they are within the contingency and on schedule. Additional planning is needed for operational development.

6. FRACTURE TOUGHNESS CURVE SHIFT METHOD

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

6.1 Introduction

The purposes of this task are to (a) examine the technical basis for the currently accepted methods for shifting fracture toughness curves to account for irradiation damage and (b) to work through national codes and standards bodies to revise those methods, if a change is warranted. Specific activities of this task include (1) collection and statistical analysis of pertinent fracture toughness data to assess the shift and potential change in shape of the fracture toughness curves due to neutron irradiation, thermal aging, or both; (2) evaluation of methods for indexing fracture toughness curves to values that can be deduced from material surveillance programs required under the *Code of Federal Regulations* (10 CFR Part 50), Appendix H; (3) participation in the pertinent ASME *Boiler and Pressure Vessel Code*, Sect. XI, ASTM E-8 and E-10 committees to facilitate obtaining data and disseminating the results of the research; (4) interaction with other researchers in the national and international technical community addressing similar problems; and (5) frequent interaction, telephone conversations, and detailed technical meetings with the NRC staff to ensure that the results of the research and proposed changes to the accepted methods for shifting the fracture toughness curves reflect staff assessments of the regulatory issues.

6.2 Master Curve Technology

An important benefit of the master curve technology is the demonstration that small specimens can be used to characterize the fracture toughness of larger specimens with thicknesses that approach the thickness of materials used in actual applications. This work is aimed at determining if the precracked Charpy V-notch (PCVN) or even smaller size specimens can be used for master curve establishment.

The fracture toughness data scatter in the transition region that is characteristic of commercial grade steels and their weldments is handled by Weibull statistical modeling, and the concept of a universal-shaped curve (master curve) is applied to characterize fracture toughness in the transition region. The master curve is developed to describe the median K_{Jc} fracture toughness for 1T size compact specimens. For small specimen applications, size adjustment is a key point in this analysis procedure. The weakest-link theory is used¹ to explain statistical specimen size effects so that data, for example, equivalent to that for a 1T size specimen, $K_{Jc(1T)}$, can be calculated from data measured with specimens of different sizes, $K_{Jc(xT)}$:

$$K_{Jc(1T)} = 20 + [K_{Jc(xT)} - 20] \left[\frac{B_{(xT)}}{B_{(1T)}} \right]^{1/4}, \quad \text{MPa}\sqrt{\text{m}}, \quad (6.1)$$

where $B_{(xT)}$ and $B_{(1T)}$ are the test specimen and 1T specimen thicknesses, respectively. Statistical size adjustment is based on the fact that the cleavage fracture in the transition range is initiated by small microstructural defects that are always present in commercially produced reactor pressure vessel

steels. Thus, the thicker the specimen being tested, the higher the probability of encountering a trigger point of a critical size that is situated within the volume of material loaded to a critical stress state along the crack tip front. As a result, large specimens will display lower fracture toughness than specimens of smaller thickness. Equation (6.1) is the mathematical expression for these statistical effects. Finally, knowing all of the parameters of the distribution allows one to determine the median K_{Jc} toughness, $K_{Jc(\text{med})}$ (the K_{Jc} value at $P_f = 0.5$), for a specimen of chosen reference size, usually a 1T C(T), at a given temperature.

However, the application of this procedure to small specimens has some limitations. On the high-temperature side, small specimens are limited by specimen capacity to maintain constraint. The test specimen capacity currently set in the proposed ASTM draft standard is calculated using the following:

$$K_{Jc(\text{limit})} = \left(\frac{Eb_o\sigma_y}{30} \right)^{1/2}, \text{ MPa}\sqrt{\text{m}} \quad (6.2)$$

where E is elastic modulus, b_o is the initial remaining ligament dimension, and σ_y is the yield strength. According to the proposed ASTM draft standard, the invalid data point is censored and assigned the $K_{Jc(\text{limit})}$ toughness value. The remaining ligament size, b_o , is a critical parameter to satisfy the constraint limit for a given material. As the lower-shelf toughness at low temperatures is approached, Eq. (6.1) becomes inapplicable because the statistical-size effects diminish and the initiation criterion is no longer dominant. Fracture becomes more propagation-controlled. This means that the test-temperature range for small specimens is quite narrow in order to provide data acceptable for the master-curve analysis procedure.

Four types of specimens were tested in the present study. The first is the standard full-size Charpy specimen, 10 x 10 x 55 mm. The second was made by thin-wire electrodischarge machine cutting along the axes of the full-size CVN specimen. This specimen is 4.8 mm thick, 10 mm wide, and 55 mm long. Although this type is about half as thick as a full-size specimen, it has the same remaining ligament size, which is the key size parameter in the constraint limit equation [Eq. (6.2)]. The third specimen was made from the broken half of a full-size specimen; it has a 4.8- by 4.8-mm cross section and is about 27 mm long. Finally, the fourth type was a 0.2T compact specimen and it was selected because of its 5-mm thickness. The latter three types of specimens may be useful when the number of standard specimens are too limited for a good material evaluation, which is often the case in plant life extension and annealing evaluations. All specimens were fatigue-precracked to an $a:W$ ratio of about 0.5. Load vs load-point displacement was measured. The 10 x 10 x 55 and 4.8 x 10 x 55 mm specimens were tested with the span of 40 mm, while the 4.8 x 4.8 x 27 mm specimens were tested with a 20-mm span. All 4.8-mm-thick specimens were 20% side-grooved. The standard size Charpy specimens were tested without side grooves.

The ASTM A 533 grade B, class 1 plate, designated HSST Program Plate 02, was selected for the current study based on the existence of an extensive fracture toughness database for Plate 02 accumulated by testing large specimens. Seventy specimens of different sizes up to 11T thickness were tested by Westinghouse for the establishment of what is now known as the ASME lower-bound K_{Ic} curve. Additionally, 25 1T compact specimens of the plate had been previously tested in the transition range as a part of the HSST Program conducted at ORNL.²

Initially, these 1T data were analyzed by the master curve procedure, and the reference fracture-toughness-transition temperature, T_{100} , was determined to be -23°C . Based on this value of T_{100} , most of the tests were performed at -50°C . At the temperature of -50°C , the value of $K_{Jc(\text{med})}$ from 1T compact specimens is $71.9 \text{ MPa}\cdot\text{m}^{1/2}$, which for 5-mm-thick specimens converts to about $100 \text{ MPa}\cdot\text{m}^{1/2}$, based on the estimation by Eq. (6.1). The draft ASTM standard recommends performing tests at a temperature close to that at which median K_{Jc} values will be about $100 \text{ MPa}\cdot\text{m}^{1/2}$. Seven $10 \times 10 \times 55$, eight $4.8 \times 10 \times 55$, twelve $4.8 \times 4.8 \times 27 \text{ mm}$, and six 0.2T C(T) specimens were tested at -50°C . Additionally, seven $10 \times 10 \times 55$ and eight $4.8 \times 10 \times 55 \text{ mm}$ specimens were tested at -30°C .

For the full size CVN specimens, the reference transition temperatures, T_{100} , were -23 and -26°C after testing at -50 and -30°C , respectively. The difference between the two values is only 3°C , which indicates that median toughness values determined by PCVNs fit very well to the shape of the master curve. The average of these two values, -25°C , is used as the reference fracture toughness temperature determined by testing of PCVN specimens in following evaluations of HSST Plate 02 properties.

Data from PCVN specimens are in very good agreement with 1T compact K_{Jc} data ($T_{100} = -23^{\circ}\text{C}$). However, a question remains regarding the relevance of properties evaluated by the "master curve" procedure to the ASME lower-bound K_{Ic} curve. Obviously none of the K_{Jc} values from PCVN specimens could satisfy the validity requirements for linear-elastic K_{Ic} stated in ASTM E 399-90. Nevertheless, it was recently shown (see previous semiannual progress report for example) that the Weibull distribution function models very well the scatter in the ASME K_{Ic} data, while the temperature dependence is described by the master curve. Thus, the master curve evaluated by testing PCVN specimens can be compared to the linear-elastic K_{Ic} data of HSST Plate 02 or to all of the ASME K_{Ic} database.

Figure 6.1 shows the fracture toughness K_{Ic} data for HSST Plate 02 derived by the testing of 70 specimens of different sizes up to 11T thickness.³ The statistical size adjustment by Eq. (6.1) is applied to convert the data to 1T size equivalence. Finally, the master curve and lower, 5%, tolerance bound evaluated from the testing of PCVN specimens ($T_{100} = -25^{\circ}\text{C}$) are compared to these size-adjusted K_{Ic} data on the same plot. To cover uncertainty in T_{100} due to testing of only a few PCVN specimens, a margin, $\Delta T_{100} = 7^{\circ}\text{C}$, is added to the tolerance bound. Procedure and details of margin calculation are presented in the proposed ASTM Draft Standard.

Having success in describing the K_{Ic} database by the master curve from PCVN specimens of the same material, the next step is to make a direct comparison between the ASME lower-bound curve and the 5% margin-adjusted tolerance bound curve (Figure 6.2). All 174 K_{Ic} data from the Electric Power Research Institute database have been reexamined and checked for accuracy (ref. 4); these data are also plotted as on Figure 6.2. The top and right axes are in English units, which is a usual procedure when relative temperature, $T - RT_{NDT}$, is expressed in $^{\circ}\text{F}$. The first ASME curve was manually constructed as the lower boundary to K_{Ic} values only in the normalized temperature range of $T - RT_{NDT}$ from -100 to $+100^{\circ}\text{F}$. Figure 6.2 shows that the master curve and the 5% margin-adjusted tolerance bound derived from testing several PCVN specimens represents very well the large K_{Ic} database accumulated by testing of massive specimens.

The fracture toughness tests with 4.8-mm-thick specimens of HSST Plate 02 at -50°C exhibit some disparities in results. The median of 12 K_{Jc} data of $4.8 \times 4.8 \times 27 \text{ mm}$ specimens is $96.4 \text{ MPa}\sqrt{\text{m}}$, which yields $T_{100} = -21^{\circ}\text{C}$. This result is in very good agreement with PCVN and large compact specimen data. It needs to be pointed out, though, that only 8 of 12 specimens provided valid K_{Jc} values at this test temperature because of a very small remaining ligament of this type of specimen.

Figure 6.1. Comparison of the HSST Plate 02 linear-elastic K_{Ic} database relative to the master curve with 5% margin-adjusted tolerance bound curve derived by testing several PCVN specimens.

The median of eight K_{Jc} data of $4.8 \times 10 \times 55$ mm specimens is $75.8 \text{ MPa}\sqrt{\text{m}}$, which yields $T_{100} = 1^\circ\text{C}$. Results of six 0.2T compact specimens (5 mm thick) tested at the same temperature almost reproduce the data of $4.8 \times 10 \times 55$ mm three-point bend specimens. The median K_{Jc} of 0.2T C(T) is $73.8 \text{ MPa}\sqrt{\text{m}}$, which yields $T_{100} = 2^\circ\text{C}$. These two types of specimens have one common parameter: the ratio of width (W) to the thickness (B) is equal to 2, while the PCVN and its smaller equivalent $4.8 \times 4.8 \times 27$ mm specimens have W:B ratios of 1.

Figure 6.3 compares the median K_{Jc} values of different specimens tested at -50°C . The $K_{Jc(\text{med})}$ values are plotted against thickness of specimens, $B_{(xT)}$, relative to 1T compact specimen thickness, $B_{(1T)}$. The median K_{Jc} value of $71.9 \text{ MPa}\sqrt{\text{m}}$, derived from analysis of 1T specimens², is used as a reference point to plot as a solid line the median K_{Jc} trend prediction according to weakest-link size adjustment model by Eq. (6.1) covering the range of thicknesses studied. In addition to the specimens of the current study, the median fracture toughness of 2T compact specimens tested by Westinghouse³ at -46°C is also presented on Figure 6.3. As discussed earlier, PCVN and $4.8 \times 4.8 \times 27$ mm specimen data follow the weakest-link size adjustment model. However, the median data from $4.8 \times 10 \times 55$ mm and 0.2T C(T) specimens fell below this model trend. It indicates that the current mathematical expression for weakest-link theory by Eq. (6.1) may not apply to some specimen geometries as the thickness of specimens becomes well below 10 mm. Further investigation is needed to evaluate a size adjustment model for small thicknesses.

Figure 6.2. Comparison of the K_{Ic} Electric Power Research Institute (EPRI) database and the ASME lower-bound curve relative to the 5% margin-adjusted tolerance bound curve derived by testing several PCVN specimens of HSST Plate 02.

ORNL DWG 97-128233

Figure 6.3. Median fracture toughness values of specimens with different thicknesses (B) and widths (W).

6.3 Comparison of Irradiation-induced Charpy and Fracture Toughness Curve Shifts

In the previous semiannual report, preliminary analysis of a database of Charpy impact and fracture toughness curve shifts for RPV materials was presented. Currently, there are 39 data sets for welds and 50 for base metals. To compare the shifts of fracture toughness and Charpy curves resulting from irradiation, data sets of irradiated materials were grouped so that average values of neutron fluence of Charpy and fracture toughness data sets would match each other. Although the values of neutron fluences were not always identical, it was assumed that the differences were negligible.

Figure 6.4 presents a plot of fracture toughness T_{100} vs Charpy 41-J shifts for weld metals. Initially, a linear regression ($y = a + bx$) gave the following fit:

$$\Delta T_{100} = 4 + 0.94\Delta T_{41J}, (^\circ\text{C}), \quad (6.3)$$

where ΔT_{100} is the shift of the reference fracture toughness temperature and ΔT_{41J} is the shift of the Charpy transition temperature at the 41-J energy level. The correlation coefficient (r^2) was 0.86. Since the offset from the origin was only 4°C , the intercept coefficient was forced to be zero, and the resulting fit gives, basically, a 1:1 correlation:

$$\Delta T_{100} = 0.98\Delta T_{41J}, (^\circ\text{C}), \quad (6.4)$$

with about the same correlation coefficient. The dotted and dashed lines are 95% on the mean and predicted values, respectively. The 95% confidence interval on the mean value (dotted lines) is relatively narrow. The scatter of data reveals an interval of about $\pm 30^\circ\text{C}$ at 95% confidence level.

Figure 6.5 presents a plot of the fracture toughness T_{100} versus Charpy 41-J shifts for base metal. As in the case with weld metal, a linear regression analysis was performed and resulted in a 1:1 slope, but there is an offset of 12°C from the origin:

$$\Delta T_{100} = 12 + \Delta T_{41J}, (^\circ\text{C}), \quad (6.5)$$

with the correlation coefficient (r^2) of 0.72. Thus, the results of the current analysis show that, on average, the Charpy 41-J shift underestimates the fracture toughness shift by 12°C for base metal. The scatter (95% confidence interval on predicted value) is slightly higher for base metal ($\pm 35^\circ\text{C}$) than for weld metal ($\pm 30^\circ\text{C}$).

The preliminary analysis has been performed to examine the maintenance of the master curve shape for embrittled materials. Figures 6.6 and 6.7 summarize fracture toughness values from the present database for materials in the unirradiated and irradiated conditions, respectively. All fracture toughness data are adjusted to 1T size equivalent and test temperatures are normalized to T_{100} . The current data (for shifts up to $\sim 150^\circ$) support the master curve shape for both the unirradiated and irradiated materials.

Figure 6.4. Correlation between fracture toughness T_{100} and Charpy T_{41J} shifts for weld metals.

Figure 6.5. Correlation between fracture toughness T_{100} and Charpy T_{41J} shifts for base metals.

Figure 6.6. Fracture toughness data of the unirradiated base and weld metals adjusted to 1T C(T) size and normalized to T_{100} .

Figure 6.7. Fracture toughness data of the irradiated base and weld metals adjusted to 1T C(T) size and normalized to T_{100} .

References

1. K. Wallin, "Size Effect in K_{Ic} Results," *Engineering Fracture Mechanics*, **22**(1), 149–163 (1985).*
2. J. J. McGowan, R. K. Nanstad, and K. R. Thoms, *Characterization of Irradiated Current-Practice Welds and A533 Grade B Class 1 Plate for Nuclear Pressure Vessel Service*, NUREG/CR-4880, Vol. 1 (ORNL-6484/V1), Washington, D.C., July 1988.[†]
3. T. U. Marsten, ed., *Flaw Evaluation Procedures: Background and Application of ASME Section XI Appendix A*, EPRI NP-719-SR, Electric Power Research Institute, Palo Alto, Calif., August 1978.
4. R. K. Nanstad, J. A. Keeney, and D. E. McCabe, *Preliminary Review of the Bases for the K_{Ic} Curve in the ASME Code*, ORNL/NRC/LTR-93/15, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, Oak Ridge, Tenn., July 1993.[†]

* Available in public technical libraries.

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

7. SPECIAL TECHNICAL ASSISTANCE

W. R. Corwin

7.1 Aging and Testing Methods (R. K. Nanstad, S. K. Iskander, D. J. Alexander, and Y. A. Davidov)

The objectives of this subtask are to provide an assessment of new testing methods and specific material properties for irradiated components of nuclear reactors. Specific activities for this subtask are: (1) completing an interim report on effects of thermal aging on the impact and tensile testing in structural stainless steel welds that have received 50,000 h of thermal exposure; (2) completing the fracture toughness tests on the remaining CVN-type specimens of the structural stainless steel welds aged for 50,000 h and initiating the characterization of the material by metallography and electron microscopy and then preparing a final report; (3) completing the thermal aging at 288°C of three-wire weld-overlay stainless steel cladding to beyond 50,000 h, machining, and testing specimens; (4) preparing a report on thermally aged (50,000-h) weld-overlay stainless steel cladding; and (5) initiating the testing and evaluation of dynamic PCVN specimens.

Regarding the effect of long-term aging of stainless steel welds at 343°C, three multipass shielded metal-arc welds were prepared from type-304L base plate with type-308 filler material. The chemistry of the filler wire was adjusted to obtain different ferrite levels (4, 8, or 12%). Portions of these welds were aged for 3,000, 10,000, 20,000, or 50,000 h at 343°C. CVN and tensile specimens were machined from the welds after each aging sequence and tested at various temperatures. Subsequently, some PCVN specimens were prepared from selected cases for J-R testing. During this reporting period, (1) the J-R tests of the PCVN specimens were completed and analyzed, (2) some reversion heat treatments were conducted followed by CVN testing, and (3) a draft NUREG report was prepared. In addition to significant decreases in impact toughness which increased with aging time, examination of the material aged for 20,000 h showed that the ferrite phase undergoes spinodal decomposition, creating iron- and chromium-rich regions. Moreover, relatively large G-phase particles were observed heterogeneously associated with dislocations, as well as fine G-phase particles homogeneously distributed through the ferrite phase. A more detailed abstract will be presented in the next progress report following completion of the NUREG report.

7.2 Correlation Monitor Materials (W. R. Corwin and E. T. Manneschildt)

This subtask was established to ensure the continued availability of the pedigreed and extremely well-characterized material now required for inclusion in all additional and future surveillance capsules in commercial light-water reactors. Having recognized that the only original materials qualified for use as a correlation monitor in reactor surveillance capsules are the pieces remaining from the early HSST plates 01, 02, and 03, this task provides for cataloging, archiving, and distributing the material on behalf of the NRC. During this reporting period, the archival storage of the correlation monitor material was maintained.

7.3 Transfer of Government-Furnished Equipment (GFE) and Materials (W. R. Corwin)

This subtask was established to provide assistance to the NRC by assessing the GFE and materials currently located at Materials Engineering Associates (MEA)-controlled sites in Lanham, Maryland, and Latrobe, Pennsylvania, and arranging for transfer of that material to ORNL and its subsequent evaluation for use or disposal. The review of the MEA equipment to determine serviceability of each piece after its receipt at ORNL has continued.

7.4. Test Reactor Irradiation Coordination (D. W. Heatherly, M. T. Hurst, D. W. Sparks, and K. R. Thoms)

The objective of this subtask is to provide the support required to supply and coordinate irradiation services needed by NRC contractors other than ORNL. The services include the design and assembly of irradiation facilities and capsules and arranging for their exposure, disassembly, and a return of specimens. Currently, UCSB is the only other NRC contractor for which irradiations are being conducted. These irradiations will be conducted at the University of Michigan FNR in conjunction with other irradiations being conducted for the HSSI Program.

Early in this reporting period, the UCSB Irradiation Facility fabrication was completed. The instrumentation cabinets necessary for safe and efficient operation of the facility were completed and connected to the facility for a complete checkout to see that the facility and instrumentation would perform as expected. These initial heatup tests were performed at ORNL before the shipment of the facility and instrumentation to the FNR. The irradiation facility was suspended in a container of demineralized water to simulate operation in the FNR reactor pool. The facility was successfully heated using the electrical heaters. The desired irradiation temperatures of 260, 290, and 310°C were reached with minimal amounts of electrical power applied to the various heater regions of the facility. Operation of the facility at FNR would require even less electrical heat input because of the influence of gamma heating from the FNR core.

While the previously described tests were being conducted, several other items were being fabricated in order to begin irradiation work at FNR. Because the HSSI-IAR facilities were not going to be started up simultaneously with the UCSB facility, it was necessary to fabricate two prototype, or dummy, IAR facilities to occupy the spaces in the base and framework adjacent to the UCSB facility. Beginning the UCSB irradiations with the IAR dummies in place would ensure that the flux spectrum would not change very much when the real HSSI-IAR facilities were installed to begin the IAR irradiation work at a later date.

Before shipping the UCSB facility to FNR, a complete fit test was performed at ORNL. The fit tests included the thermal shield and base for the IAR/UCSB facilities, the boral shielding box, two IAR facilities, and the UCSB facility. The two IAR facilities were removed, and two IAR dummy facilities were installed in their places to simulate the initial startup conditions at FNR.

It was also necessary to fabricate a lead-shielded specimen-transfer cask to be used for loading and unloading specimens into the UCSB facility baskets. The cask was necessary for transporting the irradiated baskets and specimens to and from the FNR hot cells. The management at FNR agreed to fabricate a support (specimen transfer rack) at poolside in which to suspend the UCSB facility while specimen transfers are being made. All of the items were completed early in this reporting period and were ready for use at the time the UCSB facility was delivered to the FNR in early December 1996.

The UCSB facility and all necessary instrumentation and additional hardware were delivered and installed at FNR in the middle of December 1996. The facility was installed in the FNR pool and connected to the instrumentation. Several mock runs and fit tests were performed to determine that the facility could be operated as planned and that the UCSB specimen baskets could be removed and replaced using the specimen transfer cask. All tests and checkouts went very well. Before beginning of irradiation and heatup tests, the facility specimen baskets were filled with dummy specimen packages made of aluminum blocks to simulate the stacks of UCSB and ORNL specimen packages.

Tests were performed to determine the effects of gamma heating from the FNR core. The facility was cranked into the face of the core while the reactor was at 2 MW. No electrical heat was applied to the facility, and it had been backfilled and was being purged with dry helium. The net result of the gamma heating in the facility was temperature rises of approximately 75°C in the front or high fluence portion of the facility and a rise of only 10°C in the very back of the intermediate/low fluence portion. While performing the gamma heating rate test, it was also determined that the UCSB facility could be considered a movable experiment instead of a fixed experiment. This meant that the experiment could be cranked into and away from the reactor core face without making any adjustments to the reactor power.

It was proven in additional tests that the UCSB facility heaters could maintain control temperatures with or without gamma heating supplied by the FNR core. The facility could be heated to the desired temperature electrically before it was cranked into the FNR core. The electrical heaters were found to respond very quickly to set-point changes, making it possible to have very good control of temperatures in the different regions of the facility. The heater controllers and heaters responded so quickly to the additional gamma heat input that there was no visible change in specimen temperature as a result of cranking the heated facility into the face of the core.

The UCSB facility, loaded with dummy specimens, was irradiated for approximately 8 d, and the temperatures were found to be very stable during the cycle.

On January 20, 1997, with personnel from UCSB, ORNL, NRC, and FNR on-site, the UCSB specimen baskets were withdrawn from the irradiation facility into the specimen transfer cask. The cask was transported to the FNR hot cell for installation of the actual test specimens from UCSB and flux monitors from ORNL. Three packets of preirradiated Charpy specimens from ORNL were also loaded into the upper portion of the high-flux specimen basket. Flux monitors loaded in the specimen baskets consisted of 7 FRDSs, 16 sets of wires (Fe, Ni), and 2 removable dosimeter tubes (RDTs). The loaded specimen baskets were then reinstalled into the UCSB irradiation facility to begin irradiation of the specimens. It was decided by the UCSB experimenters to change the three irradiation temperatures of 260, 290, and 320°C to 270, 290, and 310°C, respectively.

The FNR shut down on Feb. 14, 1997, after one full cycle (20 d) of irradiation on the actual UCSB test specimens and flux monitors. On February 17, 1997, the specimen baskets from the facility were transported to the FNR hot cells and all of the flux monitors were removed. The 16 sets of Ni/Fe wires were replaced with new ones, and the removable dosimeter tubes in the thermal shield were replaced with new tubes. The dosimeters were packaged and shipped to ORNL, where they are currently being analyzed.

It was determined during the dosimeter removal that additional lead shielding would be required for future specimen removal because of the high activation of the stainless steel specimen baskets in the facility. It is anticipated that an additional 2 in. of lead added to the transfer cask would provide the

necessary shielding for the duration of the irradiation experiment. Work is underway to fabricate an additional 2-in.-thick layer of lead shielding to be added to the specimen transfer cask.

During the weekend of Mar. 14, 1997, the FNR was shut down for refueling. The UCSB facility was retracted from the core and placed in the "shutdown" mode. On Saturday, March 15, all electrical power to the UCSB facility control system was lost as a result of electrical line work being performed in the Phoenix Memorial Laboratory area. The total power outage lasted approximately 4 h. When electrical power was restored, the UCSB computer rebooted and came up in the "automatic" control mode. In this mode all of the heater zones of the UCSB facility have a target temperature (desired irradiation temperature) set point, and the system will control at that temperature. As a result of heater breakers not being opened as required and as had occurred during previous shutdowns, the facility was allowed to heat up to near the desired irradiation temperature even though the reactor was shut down. After ~30 min, the facility control system was inadvertently placed into the "setout" mode. In the "setout" mode, all heater zones are supplied a preset amount of electrical power to heat the specimens to near the desired operating temperature. Because the reactor was down and operators were not present, this condition lasted for approximately 45 h. While this was an undesirable event, experimenters have determined that the test specimens inside the facility have suffered little or no damage. The main concern is to see that this or a similar event cannot occur in the future. The UCSB facility remained in the shutdown mode for the remainder of this reporting period until the event could be evaluated.

It is anticipated that irradiation of the UCSB facility will resume during the first month of the next reporting period.

8. FOREIGN RESEARCH INTERACTIONS

R. K. Nanstad

This task consolidates all of the major collaborative research interactions into five subtasks. The specific objectives of each subtask are described within the individual subtask reports.

8.1 Japanese Power Demonstration Reactor Vessel Steel Examinations (S. K. Iskander and R. K. Nanstad)

During this reporting period, a preliminary matrix of tests to be performed on the JPDR vessel core samples was developed. These tests include dosimetry and chemistry through the wall. ORNL continues to coordinate all JPDR-related activities with Kunio Onizawa of the Japan Atomic Energy Research Institute (JAERI). There are four trepans each of weld and base metal from the corebelt region and the same number and materials from a region remote from the corebelt. All 16 trepans, 87 mm in diameter and 78 mm long, have stainless steel cladding, from which most of the activity of material originates. To preserve as much of the core trepans as possible, the use of saw blades thinner than the standard ones is being investigated. The present blades result in a ~2-mm-wide kerf.

Two staff members from the Reactor Component Reliability Laboratory, Tokai Research Establishment, JAERI, Yutaka Nishiyama and Kunio Onizawa, visited ORNL on January 17, 1997, for an information exchange meeting and discussion of the preliminary test plan.

The chemistries of the weld and base materials (provided by JAERI) are given in Tables 8.1 and 8.2, respectively.

**Table 8.1. Chemistry of JPDR weld metal trepans
remote from the core**
(provided by K. Onizawa, JAERI, Apr. 4, 1997)

Trepan identification	Composition (wt %)		
	Cu	P	Ni
RW-1	0.14	0.025	0.59
RW-2	0.11	0.030	0.59

In a recent communication, JAERI indicated that it plans to analyze the chemistry of the weld metal at 10-mm intervals through the wall thickness.

**Table 8.2. Chemistry of ASTM A 302 grade B modified
JPDR trepans remote from the core**

Material	Composition ^a (wt %)									
	C	S	Mn	P	S	Cu	Ni	Cr	Mo	Al
Shell	0.17	0.24	1.27	0.10	0.011	0.10	0.63	0.02	0.47	0.012
	0.19			0.017	0.013	0.11	0.65	0.03	0.54	0.016
STP ^b	0.19	0.28	1.33	0.010 0.016	0.006	0.15 0.17	0.60	0.07	0.48	<0.005

^aTwo values indicate a range.
^bSurveillance test piece made from top/bottom head material.
Source: M. Suzuki et al., "An Update of the JPDR Investigation," presented at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village, Japan, May 12-17, 1997.

The 41-J transition temperature shift of the base metal from the core region is likely to be small. The shift at the inside surface predicted using the NRC *Regulatory Guide 1.99*, Rev. 2, assuming 0.1% Cu and 0.64% Ni, is about 15 K. Preliminary data* indicate that the shift in the base metal, about 15 mm below the inside surface, appears to be about 20 K. Data obtained by JAERI also indicate that the attenuation of hardness through the depth of the wall due to self-shielding is small. These findings will allow data obtained from two adjacent layers to be assumed that of the average depth of both layers.

8.2 Technical Assistance for Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Workings Groups 3 and 12 (R. K. Nanstad and M. A. Sokolov)

The purpose of this subtask is to provide technical support for the efforts of the U.S.-Russian JCCCNRS Working Group 3 on radiation embrittlement and Working Group 12 on aging. Specific activities under this subtask 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; and (3) preparation for and participation in meetings of Workings Groups 3 and 12.

Charpy impact and tensile testing of VVER-440 and VVER-1000 welds in the unirradiated, irradiated, and thermally annealed conditions were completed during the previous reporting period. The results were reported by R. K. Nanstad to Working Group 3 in September 1996 (a foreign trip report by R. K. Nanstad, *Report of Foreign Travel to Austria, Ukraine, and Russia*, ORNL/FTR-5905, was issued in October 1996). The results were also presented in summary form by R. K. Nanstad at the Water Reactor Safety Information Meeting in Bethesda, Maryland, in October 1996. Analysis of the VVER-440 and -1000 weld testing results continued and, as such, included evaluation of the fast

*M. Suzuki et al., "An Update of the JPDR Investigation," presented at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village, Japan, May 12-17, 1997.

neutron fluences relative to conversion from $E > 1.0$ MeV to $E > 0.5$ MeV for the U.S. experiments and vice versa for the Russian experiments with U.S. steels.

R. K. Nanstad traveled to Hannover, Germany, and met with Working Group 3 members from the Russian Research Center-Kurchatov Institute to discuss details of the cooperative irradiation experiments and preparation of a joint paper and report. Details of the discussions will be provided in a foreign trip report, now in preparation. As a result of the meeting in Hannover, a draft paper has been completed entitled "Exploratory Study of Irradiation, Annealing, and Reirradiation Effects on American and Russian Reactor Pressure Vessel Steels," by A. C. Chernobaeva, M. A. Sokolov, R. K. Nanstad, A. M. Kryukov, Y. A. Nikolaev, and Yu. N. Korolev and will be submitted for publication in the proceedings of the Eighth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems, to be held August 10-14, 1997, in Amelia Island, Florida.

8.3 Belgian Interactions (R. K. Nanstad and E. T. Manneschildt)

The objective of this subtask is to provide a framework for informal collaborative studies on fracture properties of materials irradiated in the HSSI Program. In particular, these studies will focus on the high-copper weld metals, 72W and 73W, irradiated in the fifth and sixth irradiation series. ORNL is to supply researchers at the Nuclear Research Center (SCK-CEN) in Mol, Belgium, with both unirradiated and irradiated material as well as detailed testing records.

During this reporting period, the search for a suitable container for shipment of irradiated 72W and 73W specimens to Belgium was solved. A concrete- and lead-lined container has been proposed for design and construction at ORNL, and preliminary cost estimates were provided to SCK-CEN for comment. The container will be suitable for shipping a package of four 4T compact specimens to Belgium; the design, fabrication, and shipping costs will be paid by SCK-CEN.

8.4 IAEA New Coordinated Research Program (R. K. Nanstad and D. E. McCabe)

The objective of this subtask is to provide for the participation of ORNL as an official member of the International Atomic Energy Agency (IAEA) New CRP on behalf of the NRC. The focus of the New CRP will be to examine the current ability to assess the toughness of RPV materials in the unirradiated and irradiated conditions using surveillance-program-sized specimens. In particular, it will examine PCVN specimens tested according to Draft 13 of the ASTM Master Curve Test Standard and will compare the results with CVN test results and fracture toughness test results with larger specimens on the same materials. All participants will be required to test the new IAEA reference material, JRQ, and will be encouraged to examine another RPV steel as well.

R. K. Nanstad participated (as the U.S. Chief Scientific Investigator) in the planning meeting of this project at the IAEA headquarters in Vienna in September 1996. The details have been published in a foreign trip report (R. K. Nanstad, *Report of Foreign Travel to Austria, Ukraine, and Russia*, ORNL/FTR-5905, October 7, 1996). Testing of precracked Charpy specimens of HSST Plate 02 and Plate JRQ will be conducted during FY 1997. An agreement was received from the IAEA with a request to be signed by ORNL. The ORNL legal department rewrote parts of the agreement, sent it to the IAEA, and it was signed by IAEA and subsequently approved by ORNL and DOE.

Some testing of PCVN specimens of HSST Plate 02 have been completed and compared with previous results from 1T compact specimens. The PCVN results show very good agreement with those

from the larger specimens. Receipt of a block of the IAEA reference steel designated JRQ is expected in the third quarter of FY 1997. Upon receipt of that block, CVN and PCVN specimens will be fabricated and tested; the results will be presented at the next meeting of the New CRP in Vienna, anticipated for October 1997.

8.5 Korean Interactions (R. K. Nanstad and W. R. Corwin)

The objective of this subtask is to provide a framework for informal collaborative studies on fracture properties of materials irradiated in the HSSI Program. In particular, these studies will focus on the low upper-shelf weld metal. The RPV of the PWR Kori-1, in Korea, was fabricated by Babcock and Wilcox in the United States. Its major fabrication welds were made using the Linde-80 welding flux that is known to produce welds susceptible to significant loss in upper-shelf toughness. The focus of the informal collaboration planned within this subtask will likely be to examine rate effects on embrittlement and annealing by examining Kori-1 LUS weld metal exposed in test reactor and surveillance irradiations.

Planning for the formal collaboration with KAERI was initiated during the previous reporting period. The basis for such initial planning is a list of technical topics of interest provided to ORNL by KAERI researchers. No further activity has been undertaken in this subtask.

CONVERSION FACTORS^a

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

^aMultiply SI quantity by given factor to obtain English quantity.

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. 8, No. 1
ORNL/TM-11568/V8&N1

2. TITLE AND SUBTITLE

Heavy-Section Steel Irradiation Program Semiannual
Progress Report for October 1996 Through March 1997

3. DATE REPORT PUBLISHED

MONTH YEAR

February 1998

4. FIN OR GRANT NUMBER

L1098

5. AUTHOR(S)

T. M Rosseel

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

October 1996 - March 1997

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)

Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. Because the RPV is the only key safety-related component of the plant for which a redundant backup system does not exist, it is imperative to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance that occurs during service. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established. Its primary goal is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels as they relate to light-water RPV integrity. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into seven tasks: (1) program management, (2) irradiation effects in engineering materials, (3) annealing, (4) microstructural analysis of radiation effects, (5) in-service irradiated and aged material evaluations, (6) fracture toughness curve shift method, (7) special technical assistance, and (8) foreign research interactions. The work is performed by the Oak Ridge National Laboratory.

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

pressure vessels
ductile testing
irradiation
fracture mechanics
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