NUREG/CR-5591
ORNL/TM-11568
Vol. 2, No. 2

Heavy-Section Steel Irradiation Program

Semiannual Progress Report for April – September 1991

Prepared by
W. R. Corwin

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

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

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555-0001
2. The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328
3. The National Technical Information Service, Springfield, VA 22161-0002

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

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

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

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

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

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

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

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

DISCLAIMER NOTICE

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, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product, or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

NUREG/CR-5591
ORNL/TM-11568
Vol. 2, No. 2

Heavy-Section Steel Irradiation Program

Semiannual Progress Report for April – September 1991

Manuscript Completed: September 1994
Date Published: October 1994

Prepared by
W. R. Corwin

Oak Ridge National Laboratory
Operated by Martin Marietta Energy Systems, Inc.

Oak Ridge National Laboratory
Oak Ridge, TN 37831-6285

Prepared for
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC FIN L1098
Under Contract No. DE-AC05-84OR21400

Abstract

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

For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established with its primary goal to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels as they relate to light-water reactor pressure-vessel integrity. The program includes the direct continuation of irradiation studies previously conducted within the Heavy-Section Steel Technology Program augmented by enhanced examinations 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. The HSSI Program is arranged into ten tasks: (1) program management, (2) K_{Ic} curve shift in high-copper welds, (3) K_{Ia} curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K_{Ic} and K_{Ia} curve shifts in low upper-shelf (LUS) welds, (6) irradiation effects in a commercial LUS weld, (7) microstructural analysis of irradiation effects, (8) in-service aged material evaluations, (9) correlation monitor materials, and (10) special technical assistance.

During this period, additional statistical analyses were performed to examine the effects of precleavage ductile tearing on the K_{Jc} curve shifts and shapes and the specimen size adjustment of K_{Jc} data using the β_{Ic} adjustment procedure on 72 and 73W welds. "Dummy" duplex crack-arrest specimens were examined to increase the likelihood of the successful testing of the 20 remaining irradiated crack-arrest specimens. A compilation of an annealing data base was initiated. The preliminary study to examine both local and global variations of chemical composition and transition temperature in the beltline and nozzle welds was completed, and based on the results, it was decided to treat beltline and nozzle weld metals as two separate materials. Fracture toughness testing with compact specimens from both beltline and nozzle welds was begun. The rate-theory-based model which has been developed was used to demonstrate that the pressure-vessel operating temperature of 288°C (550°F) is near a threshold below which the use of the steady state analysis is probably not appropriate. A detailed inventory of all remaining correlation monitor material was begun. Neutron transport calculations for various surveillance specimen locations of the High Flux Isotope Reactor at Oak Ridge National Laboratory were initiated to allow a better evaluation of the modified dpa exposure parameter being evaluated by the Nuclear Regulatory Commission.

Contents

Figures

Tables

Acknowledgments

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

Preface

The primary goal of the Heavy-Section Steel Irradiation (HSSI) Program is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels as they relate to light-water reactor pressure vessel (RPV) integrity. The program includes studies of the effects of irradiation on the degradation of mechanical and fracture properties of vessel materials augmented by enhanced examinations and modeling of the accompanying microstructural changes. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are being examined on a wide range of fracture properties. Results from the HSSI studies will be incorporated into codes and standards directly applicable to resolving major regulatory issues which involve RPV irradiation embrittlement such as pressurized-thermal shock, operating pressure-temperature limits, low-temperature overpressurization, and the specialized problems associated with low upper-shelf welds.

This HSSI Program progress report covers work performed from April through September 1991. The work performed by Oak Ridge National Laboratory (ORNL) is managed by the Metals and Ceramics (M&C) Division of ORNL. Major tasks at ORNL are carried out by the M&C, Computing Applications, and Engineering Technology Divisions.

Previous HSSI Progress Reports in this series are:

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

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

ORNL-4176
ORNL-4315
ORNL-4377
ORNL-4463
ORNL-4512
ORNL-4590
ORNL-4653
ORNL-4681
ORNL-4764
ORNL-4816
ORNL-4855
ORNL-4918
ORNL-4971
ORNL/TM-4655 (Vol. II)
ORNL/TM-4729 (Vol. II)
ORNL/TM-4805 (Vol. II)
ORNL/TM-4914 (Vol. II)
ORNL/TM-5021 (Vol. II)
ORNL/TM-5170
ORNL/NUREG/TM-3
ORNL/NUREG/TM-28
ORNL/NUREG/TM-49
ORNL/NUREG/TM-64
ORNL/NUREG/TM-94
ORNL/NUREG/TM-120
ORNL/NUREG/TM-147
ORNL/NUREG/TM-166
ORNL/NUREG/TM-194
ORNL/NUREG/TM-209
ORNL/NUREG/TM-239
NUREG/CR-0476 (ORNL/NUREG/TM-275)
NUREG/CR-0656 (ORNL/NUREG/TM-298)
NUREG/CR-0818 (ORNL/NUREG/TM-324)
NUREG/CR-0980 (ORNL/NUREG/TM-347)
NUREG/CR-1197 (ORNL/NUREG/TM-370)
NUREG/CR-1305 (ORNL/NUREG/TM-380)
NUREG/CR-1477 (ORNL/NUREG/TM-393)
NUREG/CR-1627 (ORNL/NUREG/TM-401)
NUREG/CR-1806 (ORNL/NUREG/TM-419)
NUREG/CR-1941 (ORNL/NUREG/TM-437)
NUREG/CR-2141, Vol. 1 (ORNL/TM-7822)
NUREG/CR-2141, Vol. 2 (ORNL/TM-7955)
NUREG/CR-2141, Vol. 3 (ORNL/TM-8145)
NUREG/CR-2141, Vol. 4 (ORNL/TM-8252)
NUREG/CR-2751, Vol. 1 (ORNL/TM-8369/V1)
NUREG/CR-2751, Vol. 2 (ORNL/TM-8369/V2)
NUREG/CR-2751, Vol. 3 (ORNL/TM-8369/V3)
NUREG/CR-2751, Vol. 4 (ORNL/TM-8369/V4)
NUREG/CR-3334, Vol. 1 (ORNL/TM-8787/V1)
NUREG/CR-3334, Vol. 2 (ORNL/TM-8787/V2)
NUREG/CR-333[illegible], Vol. 3 (ORNL/TM-8787/V3)
NUREG/CR-3744, Vol. 1 (ORNL/TM-9154/V1)
NUREG/CR-3744, Vol. 2 (ORNL/TM-9154/V2)
NUREG/CR-4219, Vol. 1 (ORNL/TM-9593/V1)
NUREG/CR-4219, Vol. 2 (ORNL/TM-9593/V2)
NUREG/CR-4219, Vol. 3, No. 1
(ORNL/TM-9593/V3&N1)
NUREG/CR-4219, Vol. 3, No. 2
(ORNL/TM-9593/V3&N2)
NUREG/CR-4219, Vol. 4, No. 1
(ORNL/TM-9593/V4&N1)
NUREG/CR-4219, Vol. 4, No. 2
(ORNL/TM-9593/V4&N2)

NUREG/CR-4219, Vol. 5, No. 1
(ORNL/TM-9593/V5&N1)
NUREG/CR-4219, Vol. 5, No. 2
(ORNL/TM-9593/V5&N2)
NUREG/CR-4219, Vol. 6, No. 1
(ORNL/TM-9593/V6&N1)
NUREG/CR-4219, Vol. 6, No. 2
(ORNL/TM-9593/V6&N2)

Summary

1. Program Management

The Heavy-Section Steel Irradiation (HSSI) Program is arranged into ten tasks: (1) program management, (2) K_{Ic} curve shift in high-copper welds, (3) K_{Ia} curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K_{Ic} and K_{Ia} curve shifts in low upper-shelf (LUS) welds, (6) irradiation effects in a commercial LUS weld, (7) microstructural analysis of irradiation effects, (8) in-service aged material evaluations, (9) correlation monitor materials, and (10) special technical assistance. Progress reports are issued on a semiannual basis, and the report chapters correspond to the tasks. The work is performed by the Oak Ridge National Laboratory (ORNL). During the report period, 18 program briefings, reviews, or technical presentations were made by the HSSI staff, and three technical papers were published.

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

The objectives of the Fifth Irradiation Series are to determine the K_{Ic} curve shifts and shapes for two irradiated high-copper, 0.23 and 0.31 wt %, submerged-arc welds (SAWs) [72W and 73W, respectively]. Additional statistical analyses have been performed to examine (1) the effects of precleavage ductile tearing on the K_{Jc} curve shifts and shapes, (2) the specimen size adjustment of K_{Jc} data using the β_{Ic} adjustment procedure, and (3) the effects of including data which violate the definition of "small-scale yielding." Limiting the results to those which did not experience significant stable ductile tearing, those satisfying the definition of small-scale yielding, and β_{Ic}-adjusted data did not change the results of the analyses regarding shifts and shape changes of the fracture toughness curves. The establishment of bounding curves using Weibull-based procedures has also been examined, and preliminary results are encouraging.

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

The objectives of the Sixth Irradiation Series are to determine the K_{Ia} curve shifts and shapes for two high-copper SAWs. The program is being conducted in two phases. In Phase I, 36 weld-embrittled-type crack-arrest specimens were tested, and detailed results with some preliminary conclusions have been published and a summary given in a previous semiannual progress report. In Phase II of the K_{Ia} program, 24 duplex-type crack-arrest specimens will be tested. The first four tested duplex specimens experienced arrest in the electron-beam fusion zone apparently due to lack of fusion. Based on test results from "dummy specimens" fabricated to intentionally duplicate the lack of fusion conditions, preparations for modifying the 20 remaining irradiated crack-arrest specimens are nearing completion.

4. Irradiation Effects in Cladding

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

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

Two irradiation series will be performed within this task. The primary objective of Series 8 is to examine the K_{Ic} and K_{Ia} for LUS high-copper weld metal irradiated at 288°C (550°F), with particular emphasis on the shift and change in shape of the American Society of Mechanical Engineers curves following irradiation. The purpose of the Ninth Irradiation Series is to evaluate the correlation between fracture toughness and CVN impact energy during irradiation, annealing, and reirradiation. During this reporting period, a review of the literature on the annealing response of reactor pressure vessel steels continued with the aim of determining additional research needs. In particular, the effects that various parameters have on the residual embrittlement and reirradiation rates will be considered. In addition, compilation of an annealing data base was initiated.

6. Irradiation Effects in a Commercial LUS Weld

The primary objective of the HSSI Program Tenth Irradiation Series is to investigate the postirradiation fracture toughness of the SAW from the Midland Unit 1 reactor vessel. The reactor is a PWR owned by Consumers Power Company that was canceled prior to startup. The weld from that vessel is of considerable interest because it carries the Babcock and Wilcox designation WF-70, an LUS SAW fabricated with a specific heat of weld wire (heat 72105) and specific lot of flux (lot 8669), which is the controlling material in five operating reactor vessels. Activity during this reporting period includes the completion of a preliminary study to examine both local and global variations of chemical composition and transition temperature in the beltline and nozzle welds. As a result of this preliminary survey, it was decided to treat beltline and nozzle weld metals as two separate materials. Fracture toughness testing with compact specimens from both beltline and nozzle welds is under way.

7. Microstructural Analysis and Modeling

The objective of this task is to provide an enhanced capability to interpolate and extrapolate the degree of radiation embrittlement beyond existing data bases. To accomplish this, a combination of studies examining ultrafine-scale radiation-induced damage and developing models based on fundamental mechanisms are being performed. During this reporting period, the rate-theory-based model that has been developed in this task was used to demonstrate that the light-water reactor (LWR) pressure vessel operating temperature of 288°C (550°F) is near a threshold below which the use of the steady state analysis is probably not appropriate. The fraction of point defects that are lost due to matrix recombination was used as a metric in this work. The influence of the displacement rate was also investigated in this transient regime, and the dose rate dependence of the recombination fraction during the transient was observed to be the inverse of that observed at steady state.

8. In-Service Aged Material Evaluations

The overall objective of this task is to assess the service-induced degradation of fracture resistance through examination of components exposed during in-nuclear-plant operation. The initial focus of this task is to augment the existing hot-cell testing capability available to the HSSI Program with remote machining capabilities for the fabrication of specimens from samples of activated steel obtained from service-exposed components. There was no significant activity in this task during this reporting period.

9. Correlation Monitor Materials

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

10. Special Technical Assistance

This task was established during the current reporting period to explicitly emphasize and provide performance and financial monitoring of various analytical and experimental investigations conducted to support the NRC in resolving short-term, high-priority regulatory and research issues. The current activity being performed as part of this task is to provide three-dimensional neutron transport calculations for various surveillance specimen locations within the vessel of the High Flux Isotope Reactor (HFIR) at ORNL to allow a better evaluation of the modified displacements per atom exposure parameter being evaluated by the NRC. Calculations are being performed for the HFIR locations designated keys 2, 4, 5, and 7. During this reporting period, the one-dimensional modeling of the HFIR was performed, and it was discovered that using only one thermal group with no upscatter would not meet the needed criteria. Therefore, a cross-section set with 64 total groups, with 12 thermal upscatter groups, was generated. A two-dimensional calculation was made with the DORT code. Cumulative scalar fluxes calculated previously are 15 to 25% higher than those from the current HFIR model.

Heavy-Section Steel Irradiation Program Semiannual Progress Report for April through September 1991[*],[†]

W. R. Corwin

1. Program Management

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

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

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

The program is broken down into one task responsible for overall program management and nine technical tasks: (1) program management, (2) K_{Ic} curve shift in high-copper welds, (3) K_{Ia} curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K_{Ic} and K_{Ia} curve shift in LUS welds, (6) irradiation effects in a commercial LUS weld, (7) microstructural analysis of irradiation effects, (8) in-service aged material evaluations, (9) correlation monitor materials, and (10) special technical assistance. Accordingly, the chapters of this progress report correspond to these ten tasks.

During this period, six program briefings, reviews, or presentations were made by the HSSI staff during program reviews and visits with NRC staff or others. Three technical papers[1-3] were published. In addition, 12 technical presentations were made: four[4-7] at the Annual Meeting of the International Group on Radiation Damage Mechanisms, Raleigh, North Carolina, April 22-26, 1991; one[8] to American Society for Testing and Materials (ASTM) Subcommittee E24.01, Indianapolis, Indiana, May 8, 1991; one[9] to American Society of Mechanical Engineers (ASME) Code Section XI, Orlando, Florida, May 21, 1991; one[10] at the

---

[*] 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-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

[†] The submitted manuscript has been authored by a contractor of the U.S. Government under contract DE-AC05-84OR21400. Accordingly, the U.S. Government retains a non-exclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes.

Workshop on Neutron Dose Rate Effects/Spectral Effects on Material Irradiation Degradation, Atlantic City, New Jersey, June 26, 1991; one[11] at the 11th International Conference on Structural Mechanics in Reactor Technology (SMiRT), Tokyo, Japan, August 22, 1991; one[12] at the SMiRT 11 Post-Conference Seminar No. 2, "Assuring Structural Integrity of Steel Reactor Pressure Boundary Components," Taipei, Taiwan, August 26-28, 1991; one[13] at Japan Atomic Energy Research Institute, Tokai Research Establishment, Tokai, Japan, August 30, 1991; one[14] at Mitsubishi Heavy Industries, Takasago, Japan, September 2, 1991; and one[15] at Japan Power Engineering and Inspection Corporation, Tokyo, Japan, September 3, 1991.

References

1. R. K. Nanstad and R. G. Berggren, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn. *Irradiation Effects on Charpy Impact and Tensile Properties of Low Upper-Shelf Welds, Heavy-Section Steel Irradiation Program, Irradiation Series 2 and 3*, USNRC Report NUREG/CR-5696 (ORNL/TM-11804), June 1991.*

2. W. R. Corwin, "Comparisons of Irradiation-induced Shifts in Fracture Toughness, Crack Arrest, and Charpy Impact Energy in High-Copper Welds," pp. 231-36 in *Transactions of the 11th International Conference on Structural Mechanics in Reactor Technology (SMiRT)*, Vol. F, August 1991.+

3. F. M. Haggag, W. R. Corwin, and R. K. Nanstad, "Effects of Irradiation on the Fracture Properties of Stainless Steel Weld Overlay Cladding," *Nucl. Eng. Des.* **124**, 129-41 (1990).+

4. R. E. Stoller, "A Proposed Matrix of Surveillance Steels for Fundamental Studies of Irradiation Embrittlement," presented at the Annual Meeting of the International Group on Radiation Damage Mechanisms, Raleigh, North Carolina, April 22-26, 1991.

5. R. E. Stoller, "Low-Temperature Embrittlement of Structural Steels," presented at the Annual Meeting of the International Group on Radiation Damage Mechanisms, Raleigh, North Carolina, April 22-26, 1991.

6. R. E. Stoller, "The UCSB-Harwell Matrix of Model and Commercial Steels," presented at the Annual Meeting of the International Group on Radiation Damage Mechanisms, Raleigh, North Carolina, April 22-26, 1991.

7. R. E. Stoller, "Overview of the Key Issues Concerning the Use of DPA as a Damage Correlation Parameter," presented at the Annual Meeting of the International Group on Radiation Damage Mechanisms, Raleigh, North Carolina, April 22-26, 1991.

8. R. K. Nanstad, "Evaluation of K_{Ic} Validity Limits for Pressure Vessel Steels," presented to ASTM Subcommittee E24.01, Indianapolis, Indiana, May 8, 1991.

9. R. K. Nanstad, "Comparison of K_{Ic} Curves and HSSI Program Data," presented to ASME Code Section XI, Orlando, Florida, May 21, 1991.

10. R. K. Nanstad, L. K. Mansur, K. Farrell, and R. E. Stoller, "Observations and Analyses Regarding Accelerated Neutron Embrittlement of Ferritic Steels," presented at the Workshop on Neutron Dose Rate Effects/Spectral Effects on Material Irradiation Degradation, Atlantic City, New Jersey, June 26, 1991.

11. W. R. Corwin, "Comparisons of Irradiation-Induced Shifts in Fracture Toughness, Crack Arrest, and Charpy Impact Energy in High-Copper Welds," presented at the 11th International Conference on Structural Mechanics in Reactor Technology (SMiRT), Tokyo, Japan, August 22, 1991.

12. W. R. Corwin, "Recent Results from the Heavy-Section Steel Irradiation Program," presented at SMiRT 11 Post-Conference Seminar No. 2, Assuring Structural Integrity of Steel Reactor Pressure Boundary Components, Taipei, Taiwan, August 26-28, 1991.

13. W. R. Corwin, "Recent Results from the Heavy-Section Steel Irradiation Program," presented at Japan Atomic Energy Research Institute, Tokai Research Establishment, Tokai, Japan, August 30, 1991.

14. W. R. Corwin, "Recent Results from the Heavy-Section Steel Irradiation Program," presented at Mitsubishi Heavy Industries, Takasago, Japan, September 2, 1991.

15. W. R. Corwin, "Recent Results from the Heavy-Section Steel-Irradiation Program," presented at Japan Power Engineering and Inspection Corporation, Tokyo, Japan, September 3, 1991.

---

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

+Available in public technical libraries.

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

R. K. Nanstad, D. E. McCabe, F. M. Haggag, and K. O. Bowman

The objectives of the Fifth Irradiation Series are to determine the K_{Ic} curve shifts and shapes for two irradiated high-copper, 0.23 and 0.31 wt %, submerged-arc welds (SAWs) [72W and 73W, respectively]. All planned unirradiated and irradiated testing for the Fifth Irradiation Series has been completed and the results presented previously. Also, previous progress reports discussed some of the statistical analyses; specimen size effects on cleavage fracture toughness, K_{Jc}; and specimen size effects on precleavage stable ductile tearing. Those results demonstrated that there is a specimen size effect on the cleavage fracture toughness but that the effect is not due to differences in ductile tearing behavior.

Additional statistical analyses have been performed to examine (1) the effects of precleavage ductile tearing on the K_{Jc} curve shifts and shapes, (2) the specimen size adjustment of K_{Jc} data using the β_{Ic} adjustment procedure,[1] and (3) the effects of including data which violate the definition of "small-scale yielding."

The fracture toughness data are also being analyzed using Weibull-based procedures. Wallin[2] has developed an analysis procedure that involves fitting a three-parameter Weibull distribution to the fracture toughness data at each temperature and also includes a size adjustment derived from a fixed Weibull slope concept. Wallin determined that the shape parameter (Weibull slope) is equal to four, which is near the value that makes the Weibull and normal distributions (Weibull slope of 3.25) almost indistinguishable from one another. He also determined that a reasonable value of K_{min}, at least for pressure-vessel steels, is about $20~\mathrm{MPa}\sqrt{\mathrm{m}}$. Following adjustment of a large body of data from different specimen sizes to the 1T size, Wallin then estimated the scale parameter at each temperature for which he had sufficient data (at least two data points) and then fit an exponential equation, with temperature $(T - T_o)$ as the independent variable, to the 63.2 percentile points estimated from the data. The temperature, T_o, is that at which the mean fracture toughness is $100~\mathrm{MPa}\sqrt{\mathrm{m}}$. To obtain bounding curves, then, one can fit the exponential to the desired percentile, e.g., the five or one percentile. Preliminary analyses to estimate the temperature dependence of the mean fracture toughness for 72 and 73W indicate good agreement with the analysis of a much larger data set by Wallin. Furthermore, bounding curves were obtained which adequately bound the data. These analyses are in the process of being finalized, and preparation of the final report is in progress.

REFERENCES

1. J. G. Merkle, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *An Examination of the Size Effects and Data Scatter Observed in Small-Specimen Cleavage Fracture Toughness Testing*, USNRC Report NUREG/CR-3672 (ORNL/TM-9088), April 1984.*

2. K. Wallin, "The Scatter in K_{Ic}-Results," *Eng. Frac. Mech.* **19(6)**, 1085-93 (1984).†

---

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

†Available from public technical libraries.

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

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

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

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

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

3.1 Testing of Duplex-Type Crack-Arrest Specimens - Phase II

Four of the 24 duplex-type specimens, two each from the 72 and 73W welds, have been previously tested. In all four specimens, the flaw arrested in the fusion zone between the 4340 crack-starter material and the weld metal test section. Porosity and lack of fusion were probably the major reasons for the crack arresting in that region. (The heat-affected zone in unirradiated duplex specimens sometimes arrests the flaw in the fusion region.) It is likely that this lack of fusion or porosity exists in the remaining 20 specimens. An attempt is being made to utilize these specimens by increasing the crack-driving force. The idea behind this modification is that a sufficient large crack-driving force may cause the propagating flaw to jump across the unfused, porous zone provided that the porous zone is not too large.

About ten unirradiated duplex crack-arrest specimens ("dummy specimens") have been manufactured with an intentionally unfused electron-beam (EB) weld zone. The unfused region is an approximate duplication of the condition found in the four irradiated weld metal duplex specimens previously tested. The initial crack-driving force is approximately proportional to the square root of the diameter of the crack-starter hole. At present, the diameter of the hole in the irradiated duplex specimens is approximately 4 mm. Specimens with two different hole diameters, 16 and 19 mm (5/8 and 3/4 in.), have been manufactured to determine the optimum hole diameter. Only two of the ten specimens tested were successful. (The specimens had 16-mm-diam crack-starter holes.) In the unsuccessful tests, the unfused region was too large to allow the crack to jump across. It is difficult to make an EB weld with a predetermined "defective zone." However, in the two successful tests, a fast-running crack propagated across the unfused EB weld region and well into the test section. It is believed that, if the unfused region is no larger than about one-third of the net section at the root of the side grooves (which is indeed the case in the four irradiated 72 and 73W specimens already tested and mentioned above), then increasing the crack-starting hole diameter to 16 mm increases the probabilities of obtaining useful data from the remaining 20 irradiated crack-arrest specimens. The dummy specimens were machined from A 533, grade B base metal whose crack-arrest behavior is already known. The detailed results of these tests were given in the previous semiannual report.

The preparations for modifying the 20 irradiated crack-arrest specimens are nearing completion. A specially adapted, remotely operable lathe has been successfully used on a dummy specimen outside the hot cell. As a dummy specimen, a monolithic crack-arrest specimen of 4340 steel and of the same geometry as the irradiated duplex ones is used, hardened to about HRC 45-48, the estimated hardness of the 4340 portion of the irradiated duplex specimens. The lathe will be moved into the hot cell shortly and tested again on a dummy specimen. Soon thereafter, the irradiated specimens will be modified and tested.

Ultrasonic nondestructive examination (NDE) of duplex crack-arrest specimens has been performed using newly acquired equipment. The examination was very encouraging as it revealed the extent of an unfused zone. Previous tests using standard techniques did reveal the presence of an unfused zone but failed to size it. Ultrasonic NDE of duplex crack-arrest specimens is being pursued to determine feasibility as a quality control technique on the EB weld zone. If successful, it can provide the opportunity for rewelding defective specimens for possible salvage.

3.2 Preparations for Testing ENEA Irradiated Crack-Arrest Specimens

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

The NRC has agreed to have the irradiated crack-arrest specimens tested at ORNL because of the usefulness and applicability of these data to the safety assessment of U.S. RPVs. ENEA has nine irradiated crack-arrest specimens manufactured from an ASTM A 508, class 3 forging. The in-plane dimensions of three of the specimens are 25 × 200 × 200 mm, and the remaining six are 13 × 100 × 100 mm. The fluence of the nine specimens varied between approximately 2 to 3.2×10^{19} neutrons/cm^2 (1 MeV), and the irradiation temperature varied from 240 to 280°C.

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

References

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

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

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

---

* Available in public technical libraries.

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

4. Irradiation Effects in Cladding

R. K. Nanstad

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

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

S. K. Iskander and R. K. Nanstad

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). A preliminary review of the literature on the IAR response of RPV steels has suggested additional research needs. The postanneal relationship between fracture toughness and CVN impact energy and the rate of fracture toughness degradation after annealing are among the properties that require further study.

A relationship between the residual postanneal transition temperature shift, ΔTT_{RES}, and an annealing effect parameter, AEP (see Figure 1), has been developed by MacDonald.[1] It is an empirically developed relationship that has been fitted to the ΔTT_{RES} of A 302, grade B; A 533, grade B plate materials; and A 533, grade B welds with a standard deviation of 11°C (20°F) on ΔTT_{RES}. One of its features is that it can predict ΔTT_{RES} from the transition temperature shift due to irradiation and the irradiation and annealing times and temperatures. The MacDonald relationship has been developed using a data base of 62 data points. Since that time, more data have become available.

In order to investigate in-depth the effect of various parameters on the IAR response of materials of interest, a compilation of an annealing data base has been initiated. It is being prepared by F. B. Kam's group to enable detailed studies of available CVN data on the reponse of RPV steels during IAR.

Reference

1. B. MacDonald, "Post-Irradiation Annealing Recovery of Commercial Pressure Vessel Steels," pp. 972-78 in *Effects of Radiation on Materials: Twelfth International Symposium, ASTM STP 870*, ed. F. A. Garner and J. S. Perrin, American Society for Testing and Materials, Philadelphia, 1985.[*]

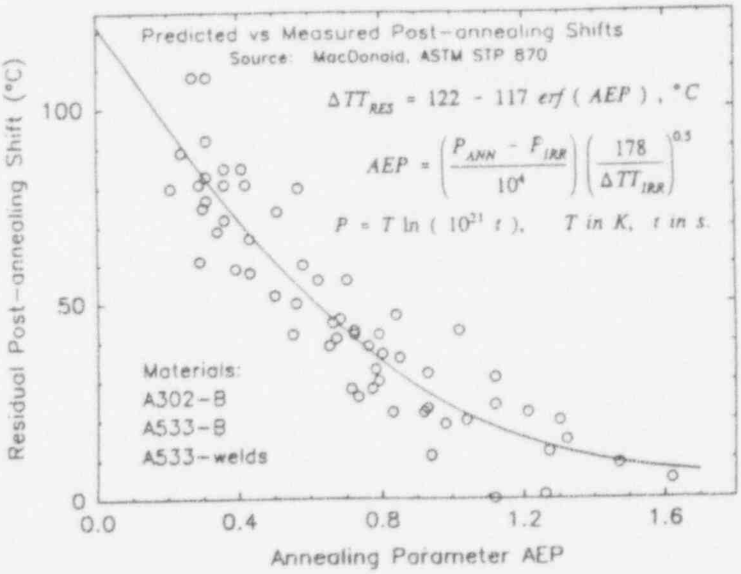

Figure 1. **Postanneal residual transition temperature shift versus annealing effect parameter, AEP. *Source*: B. MacDonald, "Post-Irradiation Annealing Recovery of Commercial Pressure Vessel Steels," pp. 972-78 in *Effects of Radiation on Materials: Twelfth International Symposium, ASTM STP 870*, ed. F. A. Garner and J. S. Perrin, American Society for Testing and Materials, Philadelphia, 1985.**

---

[*] Available in public technical libraries.

6. Irradiation Effects in a Commercial LUS Weld

D. E. McCabe, R. K. Nanstad, and R. L. Swain

The primary objective of the Tenth Irradiation Series is to investigate the postirradiation fracture toughness of the SAW from the Midland Unit 1 reactor vessel. A preliminary study to look at homogeneity of chemistry and transition temperature has been completed. Through-thickness variation was determined at four locations spaced at 90° intervals around the beltline and at two locations at 180° intervals around the nozzle belt. Transition temperature was measured by drop-weight NDT and by CVN transition curve. NDT varied from –40 to –60°C. RT_{NDT} was determined by the CVN 68-J temperature –33°C (60°F). There was no location that had better than 68 J (50 ft-lb) at NDT + 33°C (60°F). For 25 sets of CVN data, the RT_{NDT}s varied from –31 to 20°C, suggesting higher inhomogeneity than that indicated by drop-weight tests. Upper-shelf energies varied from 66 to 107 J (49 to 79 ft-lb). Chemical compositions were determined at every location where Charpy curves were determined. Copper content within the beltline weld varied from 0.21 to 0.34 wt %. Copper content ranged from 0.37 to 0.46 wt % in the nozzle weld. As a result of this preliminary survey, it was decided to treat beltline and nozzle weld metals as two separate materials. High variability of RT_{NDT} was considered to be perceived weakness in the use of Charpy data to determine RT_{NDT}.

7. Microstructural Analysis and Modeling

R. E. Stoller, K. Farrell, and L. K. Mansur

The low-temperature embrittlement of the ferritic steels used in the fabrication of LWR pressure vessels and support structures is believed to be a result of at least two mechanisms. The first is matrix hardening due to the formation of radiation-induced obstacles that impede the motion of dislocations, and the second is a non-hardening form of embrittlement due to the segregation of certain elements (e.g., P and S) to internal interfaces such as grain boundaries. The two primary defects suspected of causing the matrix hardening are fine precipitates and some sort of point defect cluster. The study of radiation-induced or radiation-modified precipitation in these steels has received considerable attention,[1,2] but the point defect cluster component, also called the "matrix defect," has been more difficult to characterize.

The modeling work initiated under this task was primarily intended to investigate this matrix defect. Irradiation temperature and displacement rate were identified as two key variables to be explored. The direction of the investigation was influenced by the unexpectedly high levels of embrittlement observed in the pressure vessel of the HFIR at ORNL after irradiation at 60°C (ref. 3). The displacement rate was a potentially significant variable in explaining the HFIR vessel embrittlement; the temperature was sufficiently low to warrant an investigation of point defect transient effects. Earlier modeling and analysis of a low-temperature creep experiment indicated that point defect transient effects could be significant below about 200 to 300°C (ref. 4).

The general behavior of the kinetic model developed for this work has been described in detail elsewhere.[5,6] The initial use of the model demonstrated that the LWR pressure vessel operating temperature of 288°C is near a threshold below which the use of the steady state analysis is probably not appropriate. The fraction of point defects that is lost due to matrix recombination was used as a metric in this work. The influence of the displacement rate was also investigated in this transient regime, and the dose rate dependence of the recombination fraction during the transient was observed to be the inverse of that observed at steady state.

The time required for the point defect concentrations to reach steady state is determined by the displacement rate and the sink structure.[5,6] The time dependence of the point defect concentrations at 60 and 285°C for a physically reasonable dislocation density of 1×10^{14} m^{-2} are compared in Figure 2 to highlight this point. The vacancy concentration reaches steady state after about 100 s at 285°C but not until after almost 30 years at 60°C. This result is particularly significant for the conditions of the HFIR pressure vessel, since it means that the entire irradiation has been conducted within the point defect transient regime. The very strong temperature dependence and long times observed below 300°C in Figure 2 indicate that models which invoke the assumption of steady state point defect concentrations to analyze low-temperature embrittlement data may be suspect. Fairly modest differences in irradiation temperature between different experiments could confound data correlation, and even limited temperature extrapolations in this domain need to consider the potential influence of the transient.

As a result of these experimental and theoretical observations, it was determined that the assumption of steady state defect concentrations would not be used in the development of the subsequent model that was used to investigate matrix defect hardening. The results obtained with this model indicate that both interstitial- and vacancy-type clusters could contribute significantly to RPV and support structure embrittlement. The relative importance of the two types of defects is determined by the irradiation temperature and displacement rate.

A dislocation barrier model has been used to calculate the matrix hardening due to the interstitial and vacancy clusters. As discussed by Bement, such models are typically derived from the theory developed by Orowan to describe hardening due to precipitates[7] and involve a calculation of the shear stress required to move a dislocation through an obstacle in its glide plane. An incremental change in the yield stress can be related to this shear stress. In its simplest form, the change in the shear stress, $\Delta\tau$, can be written as:

$$\Delta\tau = \frac{Gb}{\beta \bar{\ell}} , \qquad (1)$$

in which G is the shear modulus and b is the magnitude of the Burgers vector.

The average barrier spacing, $\bar{\ell}$, is computed from the number density (N) and the diameter (d) of the

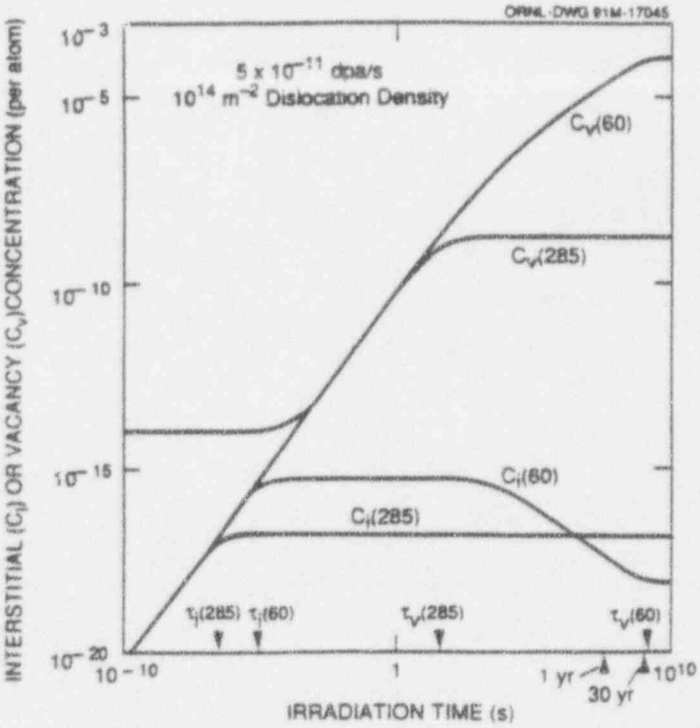

Figure 2. **Time dependence of the point defect concentrations at 60 and 285°C for typical sink strengths but without point defect clustering.**

obstacles, $\bar{\ell} = (Nd)^{-1/2}$. The factor β in Equation (1) is a function of the barrier strength. For example, β = 1 for a periodic array of strong barriers, such as incoherent precipitates. The Taylor factor can be used to convert the calculated shear stress to a change in the uniaxial yield strength, $\Delta\sigma_y \sim 3.1\Delta\tau$.

A number of potential values for the barrier strength term in Equation (1) are discussed in ref. 7. Depending on the barrier model chosen, the calculated hardening due to a specific type of obstacle can vary by more than a factor of ten. For example, if interstitial loops are assumed to be strong barriers, β = 3 to 4 in the Orowan model. Alternately, a weak barrier model that incorporates Friedel's effective barrier spacing gives:

$$\Delta\tau = \frac{Gb}{\beta} dN^{2/3} \tag{2}$$

with β = 8 for body-centered-cubic materials. A comparison of Equation (1) with β = 3 and Equation (2) is shown in Figure 3. The results shown in Figure 3 reflect loop densities of 1×10^{20} and $1 \times 10^{22}\ m^{-3}$. The difference between the two models is greater for smaller loops and for lower number densities. The figure includes a similar comparison for voids. In the case of voids, β = 1 in the Orowan (strong barrier) model. If Friedel's model is invoked for voids, the radius is used in Equation (2) instead of the diameter and β = 10. The calculated hardening due to voids is more sensitive to the choice of barrier model than is the case for loops. A model proposed by Weeks et al.[9] leads to values that lie between these two cases. The three void-hardening models are compared in Figure 4. The influence of the uncertainties related to the various barrier models will be discused further below.

The major material and irradiation parameters used in these calculations are listed in Table 1. The range of displacement rate values includes those characteristic of accelerated test reactor irradiations and power reactor surveillance and support structure locations. Where possible, the values of the material parameters represent best estimates obtained from the literature.[10-12] For some parameters, where it was difficult to determine a "best" value, a range of values was used to determine the sensitivity of the calculations to these parameters. Values for the parameters in the point defect clustering model were chosen to be consistent with the most recent molecular dynamics simulations.[13,14] The microstructural parameters such as the dislocation density and grain size were similarly varied within the range of values found in the literature.[15,16]

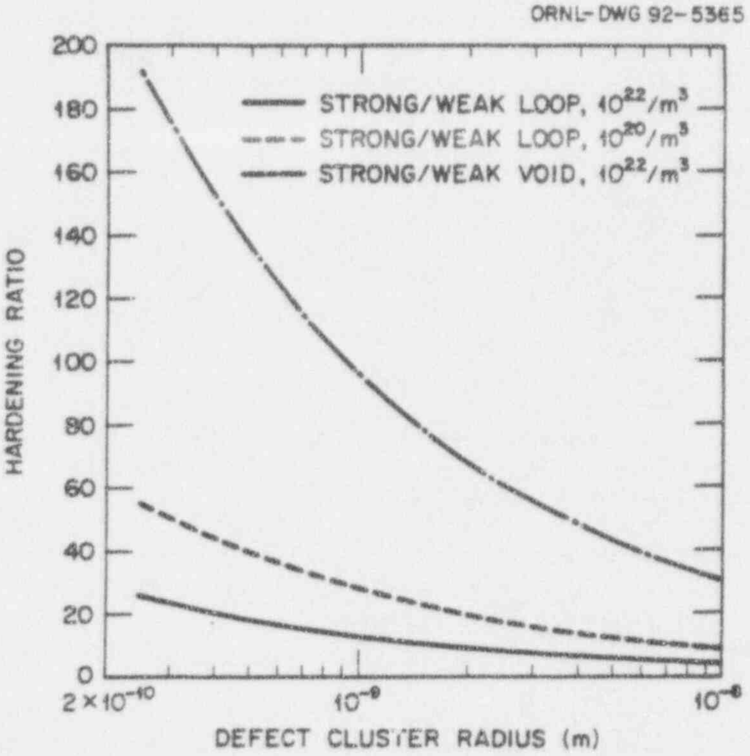

Figure 3. Ratio of hardening values obtained with "strong" and "weak" barrier models[7] for interstitial loops and voids.

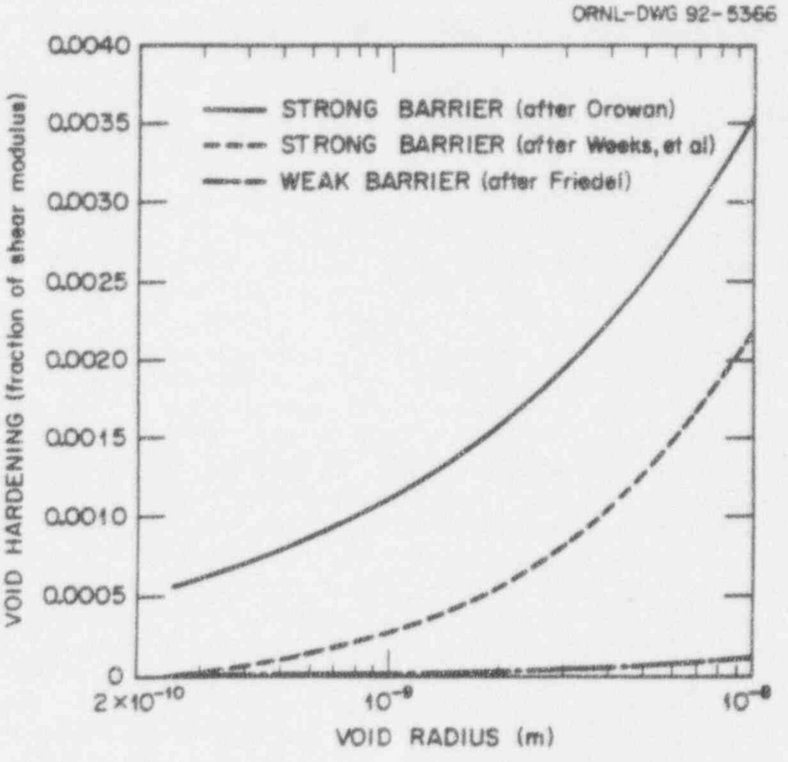

Figure 4. Comparison of void-hardening values obtained with various models.[7] Void number density is $1 \times 10^{22}\ m^{-3}$.

Table 1. Typical irradiation and material parameters

Irradiation temperature (°C)	60 to 288
Displacement rate (dpa/s)	1.0×10^{-11} to 1.0×10^{-7}
Cascade efficiency	0.1
Interstitial clustering fraction	0.0 to 0.6
Interstitial cluster binding energies (eV)	
di, tri, and tetra interstitial	0.5, 0.75, and 1.25
Vacancy clustering fraction	0.0 to 0.6
Vacancy cluster radius (nm)	0.25 to 1.0
Pre-exponential, interstitial diffusivity (m^2/s)	5.0×10^{-6}
Interstitial migration energy (eV)	0.25
Pre-exponential, vacancy diffusivity (m^2/s)	5.0×10^{-5}
Vacancy migration energy (eV)	1.25
Vacancy formation energy (eV)	1.55
Lattice parameter (nm)	0.287
Recombination radius (nm)	0.574 to 1.15
Surface free energy (J/m^2)	2.947 to $4.5 \times 10^{-4} * T(C)$
Interstitial-dislocation capture efficiency	1.25
Vacancy-dislocation capture efficiency	1.00
Network dislocation density (m^{-2})	1.0×10^{11} to 1.0×10^{15}
Effective grain size (μm)	10. to 1000

As discussed above, the predicted hardening will be strongly dependent on the assumed barrier strength, and it is not clear which barrier model is most appropriate for the small clusters that are formed. Therefore, a simple Orowan-like model was used to calculate the hardening. Both the interstitial loops and microvoids were assumed to be strong barriers, with $\beta = 3$ and $\beta = 1$, respectively. This approach should be adequate for the purpose of examining the potential degree of hardening that these defect clusters could cause.

The temperature dependence of the calculated hardening due to interstitial and vacancy clusters at a dose of 0.1 displacements per atom (dpa) is shown in Figure 5. Results are included for both a low and a high displacement rate. The higher point defect supersaturations that are obtained at lower temperatures lead to much higher cluster densities and, in turn, greater hardening. More hardening is predicted at the higher displacement rate for the same reason. The hardening tends to saturate below 100°C and has a fairly steep temperature dependence at higher temperatures. Vacancy clusters are responsible for more hardening than are interstitial clusters at a given temperature and damage rate. This is a reflection of the higher value of β used for the interstitial clusters.

The predicted hardening shown in Figure 5 appears to be somewhat higher than what is experimentally observed, particularly at the lowest temperatures and highest damage rates. Reported values of irradiation-induced changes in the yield strength of RPV steels at 50 to 300°C are in the range of 5 to 50 ksi (refs. 3, 16-18). In addition, a substantial component of the reported hardening is believed to be due to copper-rich precipitates[1,2,19] which are not accounted for here. The use of one of the alternate barrier models discussed in Section 2.3 would lead to lower calculated hardening values. Reference to Figures 2 and 3 shows that the use of the softest barrier models (i.e., those based on Friedel's effective barrier spacing) would lead to negligible hardening for the conditions shown in Figure 5.

However, the dose of 0.1 dpa used in Figure 5 is higher than most RPV irradiation data, and this could be responsible for some of the apparent discrepancy. The dose dependence of the predicted hardening is shown in Figure 6, along with typical test reactor data and the HFIR surveillance data. The A533B data shown in Figure 6(*a*) are for the HSST-02 plate, and the A302 data are the A302 correlation monitor. Both materials were irradiated in the University of Virginia reactor in an experiment conducted by researchers from the University of California at Santa Barbara.[16] The HFIR data include archive materials that were irradiated in the Oak Ridge Research Reactor.[3] Reasonable agreement is observed for the doses and displacement rates at which there are data. Some reduction in the barrier strength may be appropriate, particularly for the microvoids, but there does not appear to be a rationale for invoking one of the very soft barrier models.

Some interesting differences are observed between the behavior of the two cluster types at

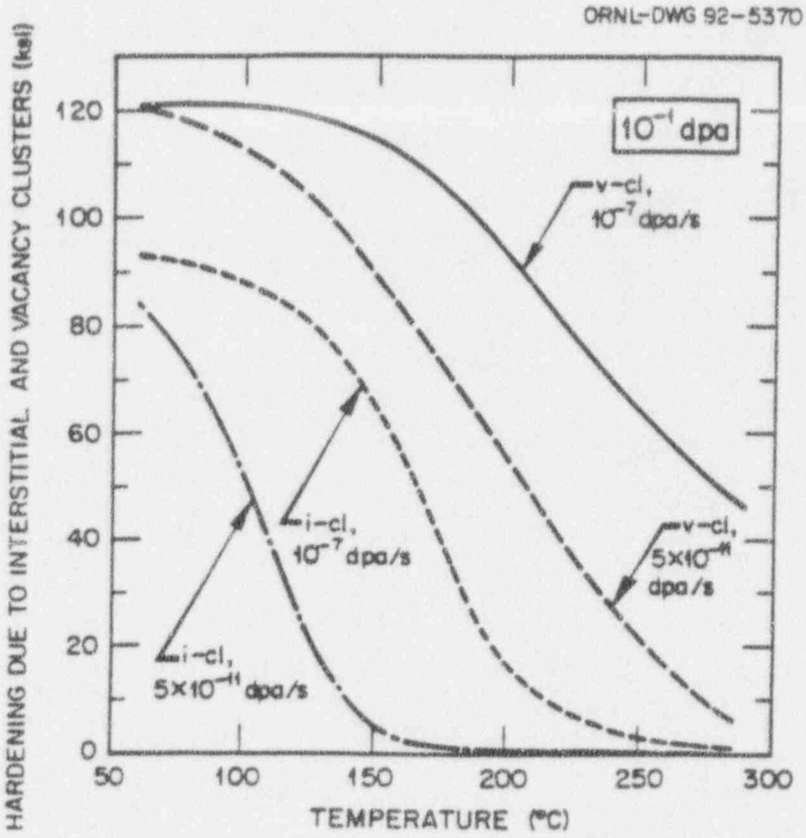

Figure 5. Temperature dependence of the calculated hardening due to interstitial and vacancy clusters at 0.1 dpa for displacement rates of 5×10^{-11} and 1×10^{-7}dpa/s.

285°C in Figure 6(*a*). The interstitial cluster contribution increases at a rate that is proportional to the square root of the dose until it saturates. The dose at which the embrittlement saturates is independent of the displacement rate, but the saturation level is proportional to the square root of the displacement rate. The hardening due to vacancy clusters is also initially proportional to the square root of the dose, but the dose at which saturation occurs is a function of the displacement rate. At 60°C, Figure 6(*b*) indicates that saturation has not yet occurred at a dose of 0.1 dpa. This is related to the low defect mobility that leads to the long times required for the point defect concentrations to reach steady state as shown in Figure 2. There is little displacement rate dependence observed. Hardening due to vacancy clusters exhibits the square root dose dependence, but the dose dependence of the interstitial clusters is initially higher. The interstitial cluster hardening has a dose dependence of about 0.75 at the lowest doses in Figure 6(*b*) and then undergoes a transition to square root dependence. The hardening curves appear to be diverging at 0.1 dpa, and the behavior is likely to be similar to that at 285°C if the calculations were carried out to much higher doses.

In order to better show the relative importance of the two types of defect clusters, the ratio interstitial cluster hardening to vacancy cluster hardening is shown as a function of irradiation temperature and displacement rate in Figures 7 and 8. Vacancy clusters are clearly more important at higher temperatures and lower displacement rates. However, both types of clusters can produce a significant amount of hardening. As mentioned above, there is probably some justification for reducing the barrier strength term for the vacancy clusters. For example, if a value of $\beta = 2$ was used for the vacancy clusters, the hardening increment from the interstitial clusters would be comparable to or greater than that of the vacancy clusters for essentially all the conditions examined here. Thus, it seems appropriate to investigate the behavior of both cluster types in greater detail. Since the two defects will almost certainly behave differently under thermal annealing, this investigation may be necessary to understand the postirradiation annealing behavior of ferritic steels.

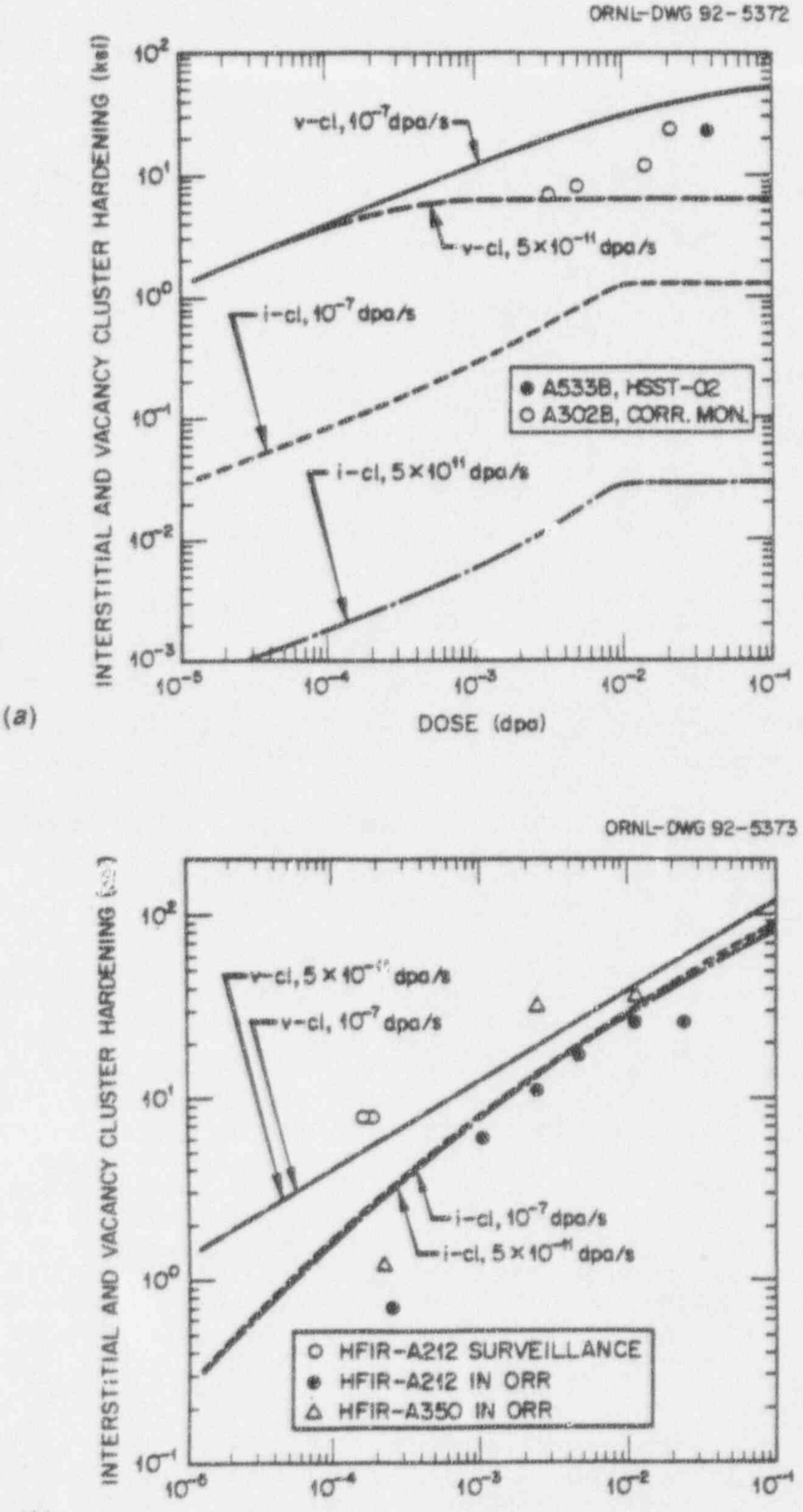

Figure 6. Dose dependence of the calculated hardening due to interstitial and vacancy clusters for displacement rates of 5×10^{-11} and 1×10^{-7} dpa/s at (*a*) 285°C and (*b*) 60°C.

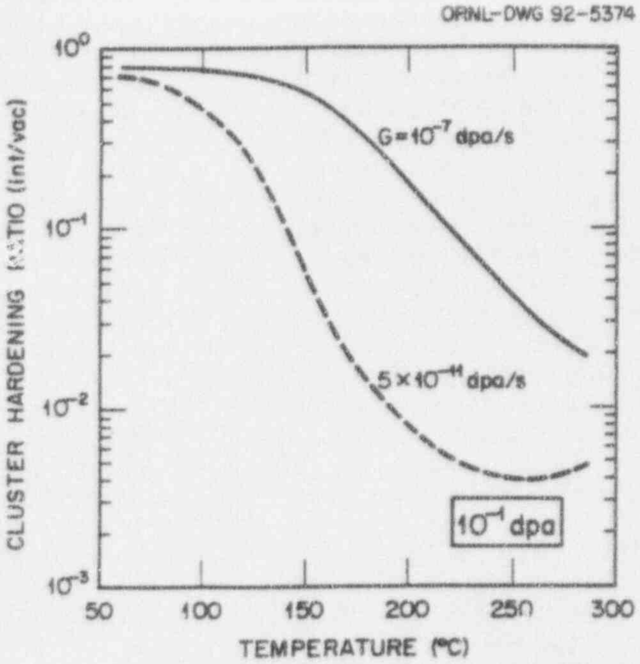

Figure 7. Temperature dependence of the ratio of interstitial cluster to vacancy cluster hardening at 0.1 dpa for displacement rates of 5×10^{-11} and 1×10^{-7} dpa/s.

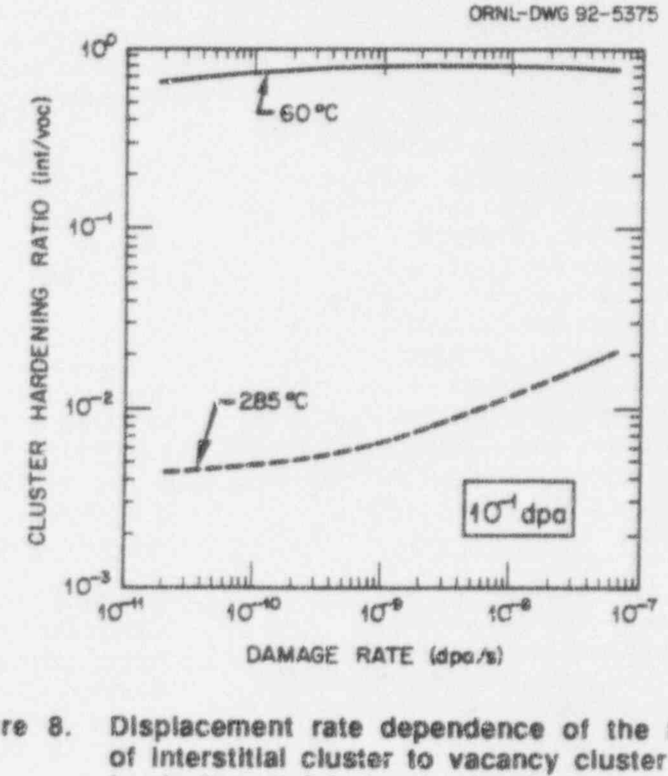

Figure 8. Displacement rate dependence of the ratio of interstitial cluster to vacancy cluster hardening at 0.1 dpa for irradiation temperatures of 60 and 285°C.

The duration of the point defect transient may not be significant at the highest RPV operating temperatures, but these calculations clearly show that the transient needs to be considered at lower temperatures. However, some caution may be required in the use and interpretation experiments conducted in test reactors, even at temperatures approaching 288°C. Test reactors typically have duty cycles in which the reactor operates for only several hours to a few days between shutdowns. Because of these frequent startups, experiments may be conducted mostly or entirely within the point defect transient, even at temperatures where the in-service component operates long enough for the point defects to be at steady state for most of its lifetime. Temperature and flux transients associated with reactor startup have been shown to influence the microstructure that evolves in elevated-temperature irradiation.[20] For components such as the HFIR pressure vessel or LWR support structures that operate at temperatures below 100°C, the point defect transient may exceed the operating lifetime. These components and experiments operating in this temperature range can not be analyzed on the basis of the steady state theory.

The time-dependent model has been used to investigate the potential contribution of interstitial and vacancy clusters to radiation-induced hardening of the ferritic steels. Although there is some uncertainty in determining the strength of these clusters as barriers to dislocation motion, the results indicate that both cluster types could induce similar levels of hardening. Using only the simplest barrier model, the calculated hardening is comparable to that observed experimentally. Since this work has not included any hardening contribution from radiation-induced precipitates, it appears that the cluster contribution obtained from the model is probably somewhat too large. However, it seems reasonable to conclude that these clusters play an important role in hardening and that they may be responsible for greater hardening than the copper-rich precipitates for some conditions of dose, displacement rate, and temperature. An understanding of their behavior under thermal annealing may be particularly important for interpreting postirradiation annealing studies and for predicting embrittlement under subsequent reirradiation.

While many of the trends shown in Figures 5 through 8 can be understood in terms of the various reactions between the point defect and the cluster populations, some of the details are not yet fully understood. Further work is under way to develop an explanation for all the observed dependencies on displacement rate and dose and to investigate the sensitivity of the model to variations in critical material parameters. Both the interstitial and interstitial clustering models require some further development to ensure that they are numerically well behaved for all the temperature and irradiation conditions of potential interest. The choice of the barrier strength terms used in Equations 1 and 2 needs additional attention because of their key role in converting defect cluster densities to yield strength changes. The range of uncertainty in these terms must be reduced to permit a determination of the relative importance of defect clusters and precipitates in irradiation hardening. Since the computing times required for the current model are already rather long, some attention must be given to improving the efficiency of the calculations as more detail is incorporated. Finally, a version of the computer code will be developed to simulate interrupted irradiations and thermal annealing because of some of the concerns that were discussed above.

References

1. G. R. Odette, *Scr. Metall.* 17, 1183-88 (1983).†

2. G. E. Lucas, G. R. Odette, P. M. Lombrozo, and J. W. Sheckherd, "Effects of Composition, Microstructure, and Temperature on Irradiation Hardening of Pressure Vessel Steels," pp. 900-930 in *Effects of Radiation on Materials, ASTM STP 870*, ed. F. A. Garner and J. S. Perrin, American Society of Testing and Materials, Philadelphia, 1985.

3. R. D. Cheverton, J. G. Merkle, and R. K. Nanstad, Oak Ridge Natl. Lab., Oak Ridge, Tenn., *Evaluation of HFIR Pressure Vessel Integrity Considering Radiation Embrittlement*, ORNL/TM-10444, 1988.*

4. R. E. Stoller, M. L. Grossbeck, and L. K. Mansur, "A Theoretical Model of Accelerated Irradiation Creep at Low Temperatures by Transient Interstitial Absorption," in *Effects of Radiation on Materials: 15th Conference, ASTM STP 1125*, ed. R. E. Stoller, A. S. Kumar, and D. S. Gelles, American Society of Testing and Materials, Philadelphia, 1992.

5. R. E. Stoller and L. K. Mansur, "The Influence of Displacement Rate on Damage Accumulation in Irradiated Materials," pp. 52-67 in *Radiation Materials Science, Proceedings of International Conference, Ukrainian Academy of Sciences*, Kharkov, U.S.S.R., 1990.†

6. R. E. Stoller, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., *Modeling the Influence of Irradiation Temperature and Displacement Rate on Radiation-Induced Hardening in Ferritic Steels*, USNRC Report NUREG/CR-5859 (ORNL/TM-12073), July 1992.[*]

7. A. L. Bement, Jr., "Fundamental Materials Problems in Nuclear Reactors," pp. 693-728 in *Strength of Metals and Alloys, Proceedings of Second International Conference*, American Society for Metals, Metals Park, Ohio, 1970.

8. Not used.

9. R. W. Weeks, S. R. Pati, M. F. Ashby, and P. Barrand, *Acta Metall.* **17**, 1403-10 (1969).[†]

10. *Interfacial Phenomena in Metals and Alloys*, L. E. Murr, Addison-Wesley, Reading, Mass., 1975.[†]

11. H. E. Schaefer, K. Maier, M. Weller, D. Herlach, H. Seeger, and J. Diehl, *Scr. Metall.* **11**, 803-9 (1977).[†]

12. G. Hettich, H. Mehrer, and K. Maier, *Scr. Metall.* **11**, 795-802 (1977).[†]

13. T. Diaz de la Rubia and M. W. Guinan, *J. Nucl. Mater.* **174**, 151-57 (1990).[†]

14. C. A. English, W. J. Phythian, and A. J. E. Foreman, *J. Nucl. Mater.* **174**, 135-40 (1990).[†]

15. J. C. Van Duysen, J. Bourgoin, C. Janot, and J. M. Penisson, "Evolution of the Microstructure of PWR Pressure Vessel Steels from the CHOOZ A Reactor Surveillance Programme After Long-Term Irradiations," in *Effects of Radiation on Materials: 15th Conference, ASTM STP 1125*, ed. R. E. Stoller, A. S. Kumar, and D. S. Gelles, American Society of Testing and Materials, Philadelphia, 1992.

16. G. R. Odette and G. E. Lucas, Electric Power Research Institute, Palo Alto, Calif., *Irradiation Embrittlement of LWR Pressure Vessel Steels*, EPRI NP-6114, 1989.

17. R. K. Nanstad, K. Farrell, D. N. Braski, and W. R. Corwin, *J. Nucl. Mater.* **158**, 1-6 (1988).[†]

18. G. E. Lucas, G. R. Odette, R. N. Maiti, and J. W. Sheckherd, "Tensile Properties of Irradiated Pressure Vessel Steels," pp. 379-94 in *Effects of Radiation on Materials (Part 2), ASTM STP 956*, ed. F. A. Garner, C. H. Henager, and N. Igata, American Society of Testing and Materials, Philadelphia, 1987.

19. G. R. Odette and C. K. Sheeks, "A Model for Displacement Cascade Induced Microvoid and Precipitate Formation in Dilute Iron-Copper Alloys," pp. 415-36 in *Phase Stability During Irradiation*, ed. J. R. Holland, L. K. Mansur, and D. I. Potter, The Metallurgical Society of AIME, Warrendale, Pa., 1981.

20. M. Kiritani, T. Yoshiie, S. Kojima, Y. Satoh, and K. Hamada, *J. Nucl. Mater.* **174**, 327-51 (1990).[†]

[†] Available in public technical libraries.

[*] Available for purchase from National Technical Information Services, Springfield, VA 22161.

8. In-Service Aged Material Evaluations

R. K. Nanstad

The overall objective of this task is to assess the service-induced degradation of fracture resistance through examination of components exposed during in-nuclear-plant operation. The initial focus of this task is to augment the existing hot-cell testing capability available to the HSSI Program with remote machining capabilities for the fabrication of specimens from samples of activated steel obtained from service-exposed components. There was no significant activity in this task during this reporting period.

9. Correlation Monitor Materials

W. R. Corwin

This is a new task that has been established with the explicit purpose of ensuring the continued availability of the pedigreed and extremely well-characterized material now required for inclusion in all additional and future surveillance capsules in commercial LWRs. Having recognized that the only remaining materials qualified for use as a correlation monitor in reactor surveillance capsules are the pieces remaining from the early HSST plates 01, 02, and 03, this task will provide for cataloging, archiving, and distributing the material on behalf of the NRC. The initial activities to be performed in this task will be to identify existing material and records in preparation of establishing a storage, monitoring, and disbursement facility. During this reporting period, a detailed inventory of all remaining correlation monitor material was begun.

10. Special Technical Assistance

J. V. Pace and W. R. Corwin

This task was established during the current reporting period to explicitly emphasize and provide performance and financial monitoring of various analytical and experimental investigations conducted to support the NRC in resolving short-term, high-priority regulatory and research issues. The current activity being performed as part of this task is to provide three-dimensional (3-D) neutron transport calculations for various surveillance specimen locations within the vessel of the HFIR at ORNL to allow a better evaluation of the modified dpa exposure parameter being evaluated by the NRC.

During this reporting period, work was initiated with the gathering of needed cross-section codes, setting up of the one-dimensional model of the HFIR, and checking input data. Updated nuclear cross sections were transferred to the K-25 Cray Computer in preparation for the 3-D calculations required to accurately obtain thermal and fast neutron exposure information for the surveillance results from the HFIR. After completing the first set of cross-section reduction calculations, it was discovered that using only one thermal group with no upscatter would not meet the needed criteria for reproducing the results with multiple, fine thermal energy groups. Therefore, a cross-section set with a minimal number of upscatter groups was generated. The final set had 64 total groups with 12 thermal upscatter groups. The next stage was to run the two-dimensional (2-D) calculations to obtain the directional fluxes. This was done in three steps: (1) run one iteration to obtain the fast groups, (2) run several iterations to obtain the thermal groups, and (3) run one last iteration to obtain all the directional fluxes. The directional fluxes are processed through three pre-processor codes to place them in the proper format for the 3-D code. A 2-D calculation was made with the DORT code. Selected fluxes from that run have been compared with fluxes from the previous calculations made several years ago. Cumulative scalar fluxes calculated previously are 15 to 25% higher than those from the current HFIR model. Additionally, the previous fluxes were 20% lower than the experimental values. This effort will be continued through the next reporting period.

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	$in.\text{-}lb/in.^2$	5.71015
$W \bullet m^{-3} \bullet K^{-1}$	$Btu/h \bullet ft^2 \bullet {}^\circ F$	1.176110
kg	lb	2.20462
kg/m^3	$lb/in.^3$	3.61273×10^{-5}
mm/N	$in./lb_f$	0.175127
$T(^\circ F) = 1.8(^\circ C) + 32$		

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

NUREG/CR-5591
ORNL/TM-11568/V2&N2
Vol. 2, No. 2
Dist. Category RF

INTERNAL DISTRIBUTION

1.	D. J. Alexander
2.	B. R. Bass
3.	K. O. Bowman
4.	R. D. Cheverton
5-15.	W. R. Corwin
16.	D. F. Craig
17.	T. L. Dickson
18.	K. Farrell
19.	F. M. Haggag
20-22.	H. W. Hayden, Jr.
23.	S. K. Iskander
24.	F. B. Kam
25.	J. Keeney
26.	L. K. Mansur
27.	W. J. McAfee
28.	D. E. McCabe
29.	J. G. Merkle
30.	M. K. Miller
31-33.	R. K. Nanstad
34.	W. E. Pennell
35.	C. E. Pugh
36.	A. F. Rowcliffe
37.	I. I. Siman Tov
38.	R. E. Stoller
39.	T. J. Theiss
40.	K. R. Thoms
41.	J. A. Wang
42.	ORNL Patent Section
43.	Central Research Library
44.	Document Reference Section
45-46.	Laboratory Records Department
47.	Laboratory Records (RC)
48-50.	M&C Records Office

EXTERNAL DISTRIBUTION

51. ABB-COMBUSTION ENGINEERING, Windsor, CT 60695

S. T. Byrne

52. ATI, Suite 160, 2010 Crow Canyon Place, San Ramon, CA 94583

W. L. Server

53. BABCOCK AND WILCOX, B&W R&D Division, 1562 Beeson St., Alliance, OH 44601

W. A. Van Der Sluys

54. BETTIS ATOMIC POWER LABORATORY, Westinghouse Electric Corp., P.O. Box 79,
West Mifflin, PA 15122

L. A. James

55. CAROLINA POWER AND LIGHT CO., P.O. Box 1551, Raleigh, NC 27602

S. P. Grant

56. EG&G IDAHO, INC., P.O. Box 1625, Idaho Falls, ID 83415-2406

V. Shah

57. GROVE ENGINEERING, Suite 218, 9040 Executive Park Drive, Knoxville, TN 37923

W. A. Pavinich

58. HANFORD ENGINEERING DEVELOPMENT LABORATORY, P.O. Box 1970, Richland, WA 99352

M. L. Hamilton

59-60. UNIVERSITY OF CALIFORNIA, Department of Chemical and Nuclear Engineering, Ward Memorial Drive, Santa Barbara, CA 93106

G. E. Lucas
G. R. Odette

61-62. UNIVERSITY OF MICHIGAN, Ford Nuclear Reactor, 2301 Bonisteel Blvd., Ann Arbor, MI 48109-2100

R. Fleming
P. A. Simpson

63. UNIVERSITY OF MISSOURI-ROLLA, Department of Nuclear Engineering, Rolla, MO 65401

A. S. Kumar

64. E. T. Wessel, Lake Region Mobile Home Village, 312 Wolverine Lane, Haines City, FL 33844

65-67. WESTINGHOUSE ELECTRIC CORP., P.O. Box 355, Pittsburgh, PA 15320

W. Bamford
T. Mager
R. C. Shogan

68. WESTINGHOUSE R&D CENTER, 1310 Beulah Rd., Pittsburgh, PA 15325

R. G. Lott

69. DOE OAK RIDGE OPERATIONS OFFICE, P.O. Box 2001, Oak Ridge, TN 37831-6269

Office of Deputy Assistant Manager for Energy Research and Development

70-71. DOE, OFFICE OF SCIENTIFIC AND TECHNICAL INFORMATION, P.O. Box 62, Oak Ridge, TN 37831

72-200. Given distribution as shown in category RF (NTIS-10)

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

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

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

NUREG/CR-5591
Vol. 2. No. 2
ORNL/TM-11568/V2&N2

2. TITLE AND SUBTITLE

Heavy-Section Steel Irradiation Program Semiannual
Progress Report for April 1991 through September 1991

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	1994

4. FIN OR GRANT NUMBER
L1098

5. AUTHOR(S)

W. R. Corwin

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

April-September 1991

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

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The primary goal of the Heavy-Section Steel Irradiation 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 integrity. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 10 tasks: (1) program management, (2) K_{Ic} curve shift in high-copper welds, (3) K_{Ia} curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K_{Ic} and K_{Ia} curve shifts in low upper-shelf welds, (6) irradiation effects in a commercial low upper-shelf weld, (7) microstructural analysis of irradiation effects, (8) in-service aged material evaluations, (9) correlation monitor materials, and (10) special technical assistance. This report provides an overview of the activities within each of these tasks from April to September 1991.

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

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

13. AVAILABILITY STATEMENT
Unlimited

14. SECURITY CLASSIFICATION
(This Page)
Unclassified
(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NUREG/CR-5591
Vol. 2, No. 2

HEAVY-SECTION STEEL IRRADIATION PROGRAM

OCTOBER 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

120555139531 1 1AN1R5
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS-PDR-NUREG
2WFN-6E7
WASHINGTON DC 20555

FIRST CLASS MAIL
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67