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Heavy-Section Steel Irradiation Program

Semiannual Progress Report for
October 1990 – March 1991

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Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. The RPV is the only key safety-related component of the plant for which a duplicate or redundant backup system does not exist. It is therefore imperative to understand and be able to predict the capabilities and limitations of the integrity inherent in the RPV. In particular, it is vital to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance that 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 nine 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) weld, (6) irradiation effects in a commercial LUS weld, (7) microstructural analysis of irradiation, (8) in-service aged material evaluations, and (9) correlation monitor materials.

During this period, additional analyses on the effects of precleavage stable ductile tearing on the toughness of high-copper welds 72W and 73W demonstrated that the size effects observed in the transition region are not due to substantial differences in ductile tearing behavior. Possible modifications to irradiated duplex crack-arrest specimens were examined to increase the likelihood of their successful testing. Characterization of a second batch of 72W and 73W welds was begun and results of the Charpy V-notch testing is provided. A review of literature on the annealing response of reactor pressure vessel steels was initiated. Drop-weight tests, Charpy impact tests, and chemical analyses through the thickness of both the beltline and nozzle course low upper-shelf welds from the Midland reactor were completed, showing large variations in bulk copper content, transition temperature, and upper-shelf energy. Atom probe field ion microscopy analyses revealed no evidence of fine copper precipitates or clusters in the unirradiated Midland welds but a substantial depletion in the copper concentration in the matrix. Experiments were initiated to examine the role of copper precipitation on embrittlement at the low temperatures relevant to reactor vessel support operations. A new task was established to assure the continued availability of the correlation monitor materials now required in all future surveillance capsules in commercial light water reactors and blocks of HSST Plate 03 were sent for use as correlation monitor materials for Units 3 and 4 of the Yong Gwang pressurized water reactor.

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ACKNOWLEDGMENTS

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

Preface

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

This HSSI Program progress report covers work performed from October 1990 through March 1991. The work performed by Oak Ridge National Laboratory (ORNL) is managed by the Metals and Ceramics Division (M&C) 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:

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  ORNL/TM-11568/V1&N1)
NUREG/CR-5591, Vol. 1, NO. 2
  (ORNL/TM-11568/V1&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 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
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ORNL/NUREG/TM-94
ORNL/NUREG/TM-120
ORNL/NUREG/TM-147
ORNL/NUREG/TM-166
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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)
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NUREG/CR-1305 (ORNL/NUREG/TM-380)
NUREG/CR-1477 (ORNL/NUREG/TM-393)
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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)
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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)
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(ORNL/TM-9593/V4&N2)
NUREG/CR-4219, Vol. 5, No. 1
(ORNL/TM-9593/V5&N1)
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(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 nine 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, (8) in-service aged material evaluations, and (9) correlation monitor materials. 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 this report period, 12 program briefings, reviews, or presentations were made; 5 technical documents 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 analyses were performed to examine the effects of precleavage stable ductile tearing, especially regarding effects of specimen size. There is no apparent effect of specimen thickness. These results are important because they demonstrate that the size effects observed in this study are not due to substantial differences in ductile tearing behavior. Additional activity on this task was associated with completion of the final report.

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 the previous semiannual progress report. In Phase II of the K_{Ia} program, 24 duplex-type crack-arrest specimens will be tested. This phase has been delayed due to some lack of fusion between the 4340 crack-arrest starter material and weld-metal test section that caused premature crack-arrest in the fusion zone in four specimens tested. Various modifications have been considered, and one that has been successful with two dummy unirradiated specimens is described. Characterization of a second batch of 72W and 73W welds is being conducted, and results of the Charpy V-notch (CVN) testing performed to date are given. A brief discussion is also given about activities involving testing of irradiated specimens of ENEA of Italy.

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, 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 Shifts 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, with particular emphasis on the shift and change in shape of the American Society of Mechanical Engineers (ASME) 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 has commenced

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.

6. Irradiation Effects in a Commercial LUS Weld

The primary objective of the HSSI Program Tenth Irradiation Series is to investigate the post-irradiation fracture toughness of the SAW from the Midland Unit 1 reactor vessel. The reactor is a PWR owned by Consumers Power Company and was cancelled prior to startup. The weld from that vessel is of considerable interest because it carries the Babcock & Wilcox (B&W) designation WF-70, an LUS SAW fabricated with a specific heat of weld wire (heat 72105) and specified lot of flux (lot 8669), which is the controlling material in several operating reactor vessels. Activity during this reporting period concentrated on completion of the drop-weight tests, Charpy impact tests, and chemical analyses through the thickness of both the beltline and nozzle course welds. The results showed that, although the variations in Charpy transition temperature and upper-shelf energy (USE) are large, they are consistent with other SAWs with high USE. The beltline and nozzle course welds exhibit similar Charpy impact toughness in the unirradiated condition. The mean 41-J temperature is about -7°C (19°F) with a 68% confidence bound (about one standard deviation) from -21 to 5°C (-5 to 41°F). The mean USE is 88 J (65 ft-lb) with 68% confidence bounds from 79 to 96 J (58 to 71 ft-lb). The average copper contents are about 0.26 and 0.40 wt % for the beltline and nozzle course welds, respectively. Atom-probe field-ion microscopy analyses were conducted and revealed no evidence of copper precipitates or clusters in the five samples examined. The atom-probe chemical analyses indicate that there was a substantial depletion in the copper in the matrix over the bulk chemical analysis, indicative of copper precipitation, but statistical analysis results were consistent with random precipitation rather than clustering.

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, experiments were initiated to examine the role of copper precipitation on embrittlement at the low temperatures relevant to reactor vessel support operations.

8. In-Service Aged Materials 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 light-water reactors. Having recognized that the only remaining materials qualified for use as a correlation monitor in reactor surveillance capsules are the pieces remaining from the early Heavy-Section Steel Technology (HSST) plates 01, 02, and 03, this task will provide for cataloging, archiving, and distributing the material on behalf of the Nuclear Regulatory Commission. During this reporting period, blocks of HSST plate 03 were sent to Korean Heavy Industries to meet their needs for correlation monitor materials for units 3 and 4 of the Yong Gwang PWR.

Heavy-Section Steel Irradiation Program Semiannual Progress Report for October 1990 - March 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 (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 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 overall the 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 eight 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

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effects on cladding, (5) K_{Ic} and K_{Ia} curve shifts in LUS welds, (6) irradiation effects in a commercial LUS weld, (7) microstructural analysis of irradiation effects, (8) in-service aged material evaluations, and (9) correlation monitor materials. Accordingly, the chapters of this progress report correspond to these nine 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. One topical report,[1] one foreign trip report,[2] and three technical papers[3-5] were published. In addition, six technical presentations were made: one[6] at the 16th MPA-Seminar, Stuttgart, Germany, October 4-5, 1990; one[7] at the Eighteenth Water Reactor Safety Information Meeting, Rockville, Maryland, October 22, 1990; one[8] at the American Society for Testing and Materials (ASTM) E-24 Committee Meetings, San Antonio, Texas, November 12-14, 1990; two[9,10] at the Joint VTT-ORNL Seminar on Structural Safety Assessment of Nuclear Power Plant Components, Oak Ridge, Tennessee, December 3, 1990; and one[11] at the meeting of the Midland Reactor Weld Coordination Group in San Diego, California, January 30, 1991.

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2. S. K. Iskander, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., Report of Foreign Travel, ORNL/FTR-3872, March 25, 1991.*

3. W. R. Corwin et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., "Heavy-Section Steel Irradiation Program Overview," pp. 2-7, 8 in *Transactions of the Eighteenth Water Reactor Safety Information Meeting*, USNRC Conference Proceeding, NUREG/CP-0113, October 1990.*

4. R. E. Stoller and L. K. Mansur, "The Influence of Displacement Rate on Damage Accumulation During the Point Defect Transient in Irradiated Materials," pp. 52-67 in *Proceedings of PM-90: International Conference on Radiation Materials Science, Vol. 1*, May 22-25, 1990, Alusha, The Crimea, U.S.S.R., October 1990.†

5. D. J. Alexander, R. K. Nanstad, W. R. Corwin, and J. T. Hutton, "A Semiautomated Computer-Interactive Dynamic Impact Testing System," pp. 83-94 in *Applications of Automation Technology to Fatigue and Fracture Testing*, ASTM STP 1092, ed. A. A. Braun, N. E. Ashbaugh, and F. M. Smith, American Society for Testing and Materials, Philadelphia, 1990.†

6. T. L. Dickson, "Effects of Irradiation on Initiation and Crack-Arrest Toughness of Two High-Copper Welds and on Stainless Steel Cladding," presented at the 16th MPA-Seminar, Stuttgart, Germany, October 4-5, 1990.

7. W. R. Corwin, "Heavy-Section Steel Irradiation Program Summary," presented at the Eighteenth Water Reactor Safety Information Meeting, Rockville, Md., October 22, 1990.

8. D. E. McCabe, "A Comparison of Weibull and β_{Ic} Analysis of Transition Range Data," presented at the ASTM E24 Committee Meetings, San Antonio, Tex., November 12-14, 1990.

9. W. R. Corwin, "Materials Irradiation Program," presented at the Joint VTT-ORNL Seminar on Structural Safety Assessment of Nuclear Power Plant Components, Oak Ridge, Tenn., December 3, 1990.

10. R. K. Nanstad, "Statistical Analysis of Fracture Toughness Results for Two Irradiated High-Copper Welds," presented at the Joint VTT-ORNL Seminar on Structural Safety Assessment of Nuclear Power Plant Components, Oak Ridge, Tenn., December 3, 1990.

11. R. K. Nanstad, "Heavy-Section Steel Irradiation Program 10th Irradiation Series: Midland Reactor Low Upper-Shelf Weld WF-70," presented at the meeting of the Midland Reactor Weld Coordination Group in San Diego, Calif., January 30, 1991.

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

†Available from public technical libraries.

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

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

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. The results from statistical analyses and curve fitting of the fracture toughness results, including specimen size effects, were reported in the previous semiannual.

Additional analyses were performed to examine the effects of precleavage stable ductile tearing, especially regarding effects of specimen size. Figure 1 shows a plot of K_{Jc} versus precleavage stable ductile tearing for specimens from 25.4 to 203.2 mm thick for the unirradiated weld 73W.

The ductile tearing was measured on the specimen fracture surface with a digital toolmaker's microscope. The data are plotted without regard to test temperature and, as shown, form a resistance curve (R-curve) of fracture toughness versus ductile tearing prior to cleavage. There is no apparent effect of specimen thickness. The same result was observed with weld 72W. These results are important because they demonstrate that the size effects observed in this study are not due to substantial differences in ductile tearing behavior.

Additional activity on this task was associated with completion of the final report.

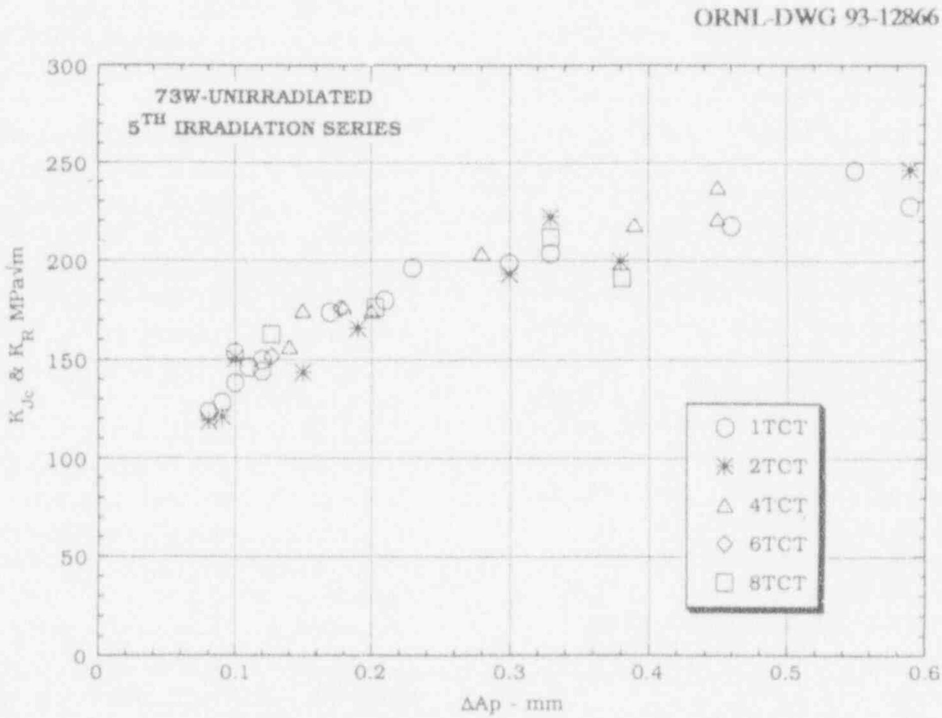

Figure 1. Results of fracture toughness versus precleavage stable ductile tearing for unirradiated HSSI weld 73W. The data from 1T through 8T compact specimens form a resistance curve with no apparent size effects.

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. The K_{IR} curve is used to define conditions 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 of the CVN 41-J energy level and that irradiation does not change the shapes of the fracture toughness curves.

The primary objective of the HSSI Sixth Irradiation Series (or, for brevity, the K_{Ia} program) is to determine the effect of irradiation on the shift and shape of the K_{Ia} versus (T - RT_{NDT}) curve. The objective of the HSSI Fifth Irradiation Series (the K_{Ic} program) is similar but determines the effect of irradiation on K_{Ic}. One of the significant results of the K_{Ic} program is that a change in shape of the lower bound to the irradiated initiation toughness, K_{cl}, curve for the 73W weldment has been observed, and for which a change in shape of the 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 been reported in the previous semiannual progress report. 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 72W and 73W welds, have been tested. In all four specimens, the flaw arrested in the fusion zone (FZ) 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. That would preclude successful testing in their present form at temperatures higher than those chosen for Phase I. The specimens could be tested at temperatures that are low with respect to RT_{NDT}, but that approach would not yield the desired information. In order to utilize these specimens, various modifications to the duplex specimens have been considered.

One modification that has been pursued is to increase the crack-driving force by increasing the diameter of the crack-starter hole. At present, the diameter of the hole in the 24 irradiated duplex specimens is approximately 4 mm. The idea behind this modification is that a sufficient crack-driving force may cause the propagating flaw to jump across the unfused, porous zone. Unirradiated duplex crack-arrest specimens with various hole sizes have been manufactured to determine the optimum hole diameter. To conserve the 72W and 73W weldment, the first set of specimens ("dummy specimens") was machined from A 533 grade B base metal whose crack-arrest behavior is already known. If the modifications look promising, they will be performed on three unirradiated 73W weld metal, duplex crack-arrest specimens that were previously rejected because of porosity.

The yield strengths at 85 and 95°C needed for the crack-arrest testing were estimated using equations developed by Irwin.[3] For loading rates approximately equal to those customarily used in "static" tensile testing, the following equation gives the yield strength at any temperature (t) if the yield strength is known at another temperature (which can be room temperature for convenience):

$$\sigma_{yt} = \sigma_{yt'} - A + \frac{55{,}000}{t + 273} , \qquad (1)$$

where

σ_{yt} = yield strength at temperature, t, in MPa,
t = temperature in °C,
$\sigma_{yt'}$ = the known yield strength at the temperature t',
A = *a calibration constant.*

When the value of the yield strength at room temperature is substituted, the equation becomes:

$$\sigma_{yt} = 395 + \frac{55{,}000}{t + 273} . \qquad (2)$$

Figure 2 shows Eq. (2) plotted together with the available data at various temperatures, which shows that the agreement is reasonable. This indicates that Eq. (1) gives useful results if the yield strength is required as a function of temperature and the yield strength is only known at one temperature. On the other hand, if the yield strength is known at several temperatures, an equation of the following form could be used to fit the data effectively:

$$\sigma_{yt} = A + \frac{B}{t + 273} . \qquad (3)$$

This gives the following equation (also shown in Fig. 2):

$$\sigma_{yt} = 442 + \frac{42{,}400}{t + 273} \qquad (4)$$

Three dummy duplex crack-arrest specimens with unfused regions have been tested. Two of the tests were successful, and a fast-running crack propagated beyond the unfused electron-beam (EB) weld region and into the test section. The results of these two successful tests are shown in Fig. 3. The open points are from previous tests on a specially heat-treated A 533 grade B base material, and the two closed points are from the dummy duplex crack-arrest specimens whose test sections were made from that same A 533 grade B base material. The NDT for the base metal used in these tests is 36°C and, thus, dummy duplex crack-arrest results have been obtained at temperatures of approximately 50 and 60°C above the NDT temperature.

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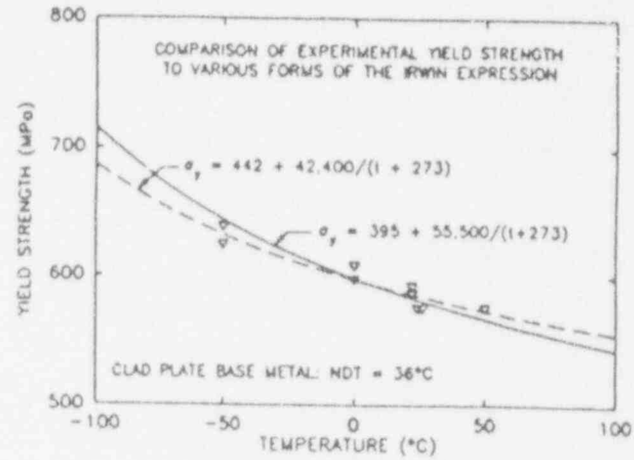

Figure 2. **Curve fitting used to extrapolate yield strength of the A 533 grade B base metal used in the test section of the "unfused" duplex crack-arrest specimens.**

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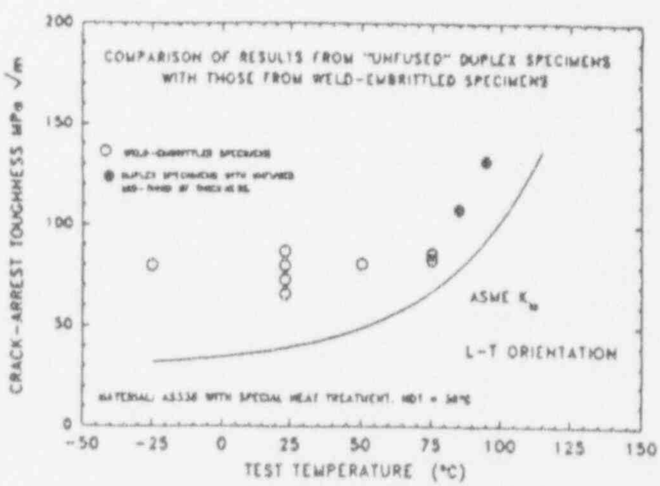

Figure 3. **Comparison of crack-arrest results obtained from two "unfused" middle-third duplex specimens with 16-mm crack-starter hole with results obtained with weld-embrittled-type specimens.**

In the third specimen, the unfused region unintentionally extended over almost two-thirds of the net section, and, presumably, this arrested the flaw in the weld fusion zone, but the crack did propagate for about 2 to 3 mm across the remaining fused one-third of the net section into the test section. These results show that, by increasing the crack-starter hole, the crack-driving force could be increased sufficiently for the crack to jump across a partially unfused EB zone, provided that the size of the unfused region is not greater than one-third the net section. Although the test sections of the dummy duplex crack-arrest specimens were base metal, the results are encouraging enough that it is believed that the same modification could be used successfully with the remaining 20 irradiated 72W and 73W weld metal, duplex crack-arrest specimens. The three dummy specimens tested have 16-mm-diam crack-starter holes. The middle third of the weld zones between the brittle 4340 crack-starter material and the test section was intentionally unfused. The unfused region is an approximate duplication of the condition found in the four irradiated weld metal duplex specimens previously tested.

Three more dummy duplex specimens with unfused middle-third sections are being manufactured. These three specimens will have 19-mm holes, which should increase the crack-driving force somewhat more than the 16-mm hole successfully used with the specimens mentioned above. If effective, the 19-mm crack-starter hole would increase the probability of successfully testing the remaining 20 irradiated duplex crack-arrest specimens.

Ultrasonic nondestructive examinations of the first set of three duplex crack-arrest specimens have been performed. The examinations revealed the presence of an unfused zone but failed to size it. More ultrasonic examinations using a different technique on one of the specimens from the second set of dummy specimens were also similarly unsuccessful. In the latter series of ultrasonic examinations, the specimen was immersed in fluid in an attempt to avoid some of the problems encountered with the previous batch. The anisotropy of the materials in the EB weld fusion zone seems to scatter the ultrasonic signal so that this type of test cannot be used to screen duplex-type crack-arrest specimens. These tests were performed in order to determine the feasibility of using ultrasonic examination as a quality control technique on the EB weld zone of duplex crack-arrest specimens.

Any eventual machining required to modify the diameter of the crack-starter hole in the hardened 4340 (approximately RC 45 to 48) crack-starter section of the remaining 20 irradiated duplex crack-arrest specimens is complicated by the high activity of the specimens. This is estimated to be approximately 35 R/h on contact. Furthermore, the crack-starter hole is slit open, with the slit running to the side of the specimen, to give an initial "crack length" of approximately 35% of the nominal specimen size, and this would complicate the machining process. Also, the profile of the new crack-starter hole must be approximately tangential to the already existing one in order to preserve the initial crack length-to-specimen size ratio of 35%. Procedures for modifying the 20 irradiated crack-arrest specimens are being developed. We are also continuing to prepare equipment that will be used to modify the 20 irradiated crack-arrest specimens using a dummy unirradiated specimen. For the dummy specimen, we are using a monolithic crack-arrest specimen of 4340 steel and of the same geometry as the irradiated duplex ones, hardened to the estimated hardness of the irradiated duplex specimens, about HRC 45 to 48.

3.2 Characterization of the Second Batch of 72W and 73W Welds

A second batch of 3.7 lin m (12 ft) each of the 72W and 73W welds was previously ordered from Combustion Engineering (CE) several years ago. This would be used to machine more specimens from welds 72W and 73W for either unirradiated or irradiated testing and for archival purposes. The first batch of 72W and 73W weldments, also fabricated by CE, was completely consumed by the specimen requirements of the Fifth and Sixth Irradiation Series. The welds of the second batch were manufactured in 235-mm-thick A 533 grade B class 1 plate using identical welding consumables and procedures as the first batch. The second batch was manufactured with HSST Plate 14A as the base metal, with the welds running transverse to the rolling direction, and were delivered as six slabs, each 1220 mm long. The individual blocks of 72W weld have been designated 72N, 72P, and 72Q, with a similar code for the blocks of the 73W weld. Some characterization was reported in the

previous semiannual progress report. In the present report, the results of CVN testing on material from two of the six slabs are given and compared to the results from the first batch. CVN specimens are being manufactured for testing from the remaining four slabs.

Figure 4 shows the comparison of the CVN impact energy for each of welds 72W and 73W. It may be seen that the CVN impact energy curves for the second batch are not significantly different from those of the first batch. The differences observed could be due to the five times as many specimens tested in the first batch (about 85 specimens) compared to those tested from the second batch. More testing is planned in order to provide a better statistical comparison between the two batches.

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

The Italian Committee for Research and Development of Nuclear Energy and Alternative Energies (ENEA), the Italian equivalent of the 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 ...oratories of the French Commissariat a l'Energie ...ue. ENEA originally planned to test the ...rack-arrest specimens at BCL, but BCL ...decommissioned their hot-cell

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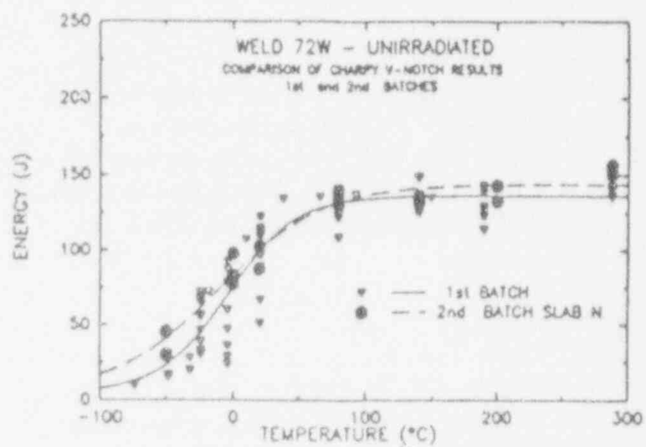

(a)

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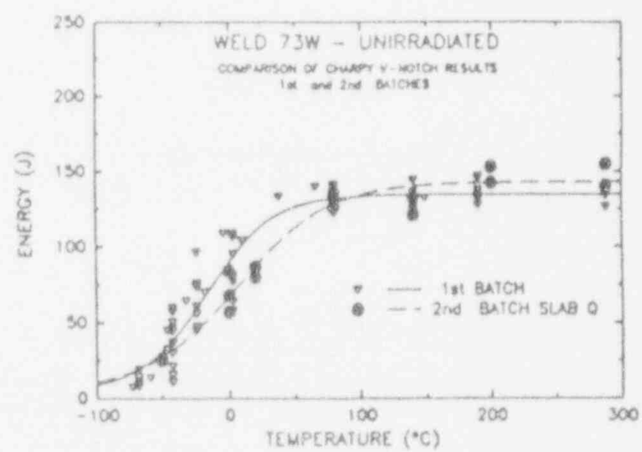

(b)

Figure 4. Comparison of results of Charpy V-notch impact energy obtained from the second batch of welds to those obtained with the first batch: (*a*) 72W welds and (*b*) 73W welds.

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.

Planning of the irradiated test program is under way. The remote fixture used in the testing of the HSSI Sixth Irradiation Series is too small to be

used with the three large specimens. There are other geometric details about the ENEA specimens time and cost to manufacture a new fixture.

We have requested that BCL move to ORNL the nine irradiated crack-arrest specimens belonging to the Italian NRC. BCL no longer has the shipping cask they intended to use, so they are making other arrangements. The specimens are located at the Ford Nuclear Reactor (FNR) in Michigan.

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. G. R. Irwin, "Linear Fracture Mechanics, Fracture Transition, and Fracture Control," *Eng. Fract. Mech.* 1, 241-57 (1968).*

*Available from public technical libraries.

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

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The NRC has agreed to test the irradiated crack-arrest specimens 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 the ASTM A 508 class 3 forging. The in-plane dimensions of three of the specimens are $25 \times 200 \times 200$ mm, and the remaining six are $13 \times 100 \times 100$ mm. The fluence of the nine that will also require modifications to the presently available equipment. It may be more effective in

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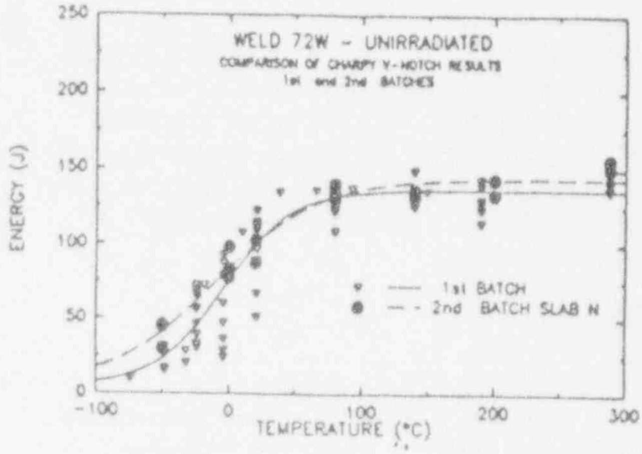

(a)

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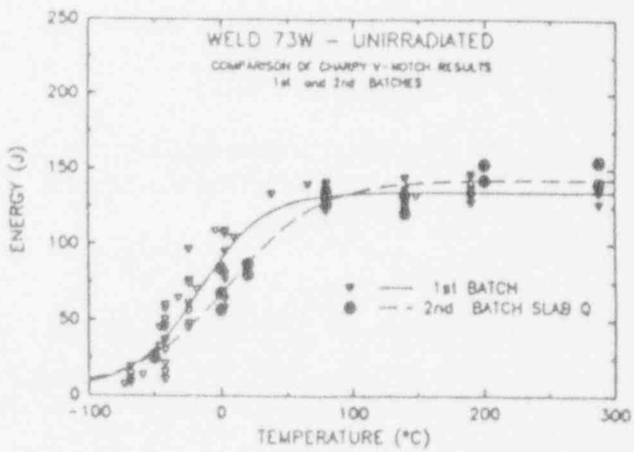

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4. IRRADIATION EFFECTS IN CLADDING

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

A review of the literature on the annealing response of RPV steels has commenced with the aim of determining if additional research is necessary. In particular, the effect that various parameters (e.g., chemical composition, irradiation temperature, annealing temperature, and time) have on the residual embrittlement will be considered. The reirradiation rate of annealed RPV steels will also be considered. The results of this literature review, as well as a survey of operating commercial reactor vessels predicted to approach or exceed the 10CFR50 toughness criteria, will be used to select materials for Irradiation Series 8 and 9.

6. IRRADIATION EFFECTS IN A COMMERCIAL LUS WELD

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

The primary objective of the HSSI Program Tenth Irradiation Series is to investigate the post-irradiation fracture toughness of the SAW from the Midland Unit 1 reactor vessel. The reactor is a PWR owned by Consumers Power Company and was cancelled prior to startup. The weld from that vessel is of considerable interest because it carries the Babcock and Wilcox (B&W) designation WF-70, an SAW fabricated with a specific heat of weld wire (heat 72105) and specific lot of flux (lot 8669). Welds with the WF-70 designation are the controlling materials (regarding irradiation effects) in at least five operating nuclear plants. The WF-70 weld was fabricated using copper-coated wire and Linde 80 flux and is known to be an LUS high-copper weld. The fact that this particular weld exists in numerous operating reactors is the primary motivation for this series. An important supporting objective is the determination of local and global copper variation. Another specific supporting objective is a complete unirradiated characterization to include determination of the RT_{NDT} and the CVN USE. Activities during this period concentrated on completion of the Charpy impact tests and chemical composition analyses to allow for development of the final irradiation plan and fabrication of specimens for the initial irradiation capsules.

6.1 Drop-Weight Tests

Drop-weight specimens were fabricated from the 1/4t (1/4 thickness from the inside vessel surface) and 3/4t locations in the four beltline sections and at the 3/4t and 7/8t locations in the two nozzle course sections. The specimens were fabricated using a single weld pass for the brittle weld crack starter and tested according to ASTM E208-86. The NDT temperatures range from -60 to -45°C (-76 to -49°F) in the beltline weld and from -55 to -40°C (-67 to -40°F) in the nozzle course weld. No clear pattern is evident regarding consistency of NDT temperatures with position in the weld thickness.

6.2 Charpy Impact Tests

CVN impact tests were conducted at various temperatures to obtain a full curve of absorbed energy versus test temperature. The emphases are on determination of the RT_{NDT} and the USE. Additionally, lateral expansion and fracture appearance were determined for all specimens. Initial CVN tests were conducted after the NDT temperatures had been determined so that tests could be performed at NDT + 60°F (33°C) as required by Sect. III of the *ASME Boiler and Pressure Vessel Code*[1] for determination of the RT_{NDT}. In all cases, tests of three CVN specimens at NDT + 33°C (60°F) did not meet the minimum requirements for the RT_{NDT} to equal the NDT. Thus, the RT_{NDT} is higher than the NDT in every case and is controlled by the CVN behavior. Charpy curves were obtained for five through-thickness locations from 1/4t to 7/8t for the beltline sections and from three locations from 1/2t to 7/8t for the nozzle course sections.

A summary of the Charpy impact results for beltline weld section 1-13 has been reported previously[2] and showed 41-J transition temperatures ranging from -17 to 6°C (1 to 43°F) through the thickness. The USEs for the same tests ranged from 89 to 107 J (65 to 79 ft-lb). The 41-J temperatures ranged from -19 to 15°C (-2 to 59°F) for the four beltline sections and from -10 to 6°C (14 to 43°F) for the two nozzle course sections. There was no discernible trend of behavior through the thickness in the beltline or the nozzle course sections. The USEs ranged from 78 to 107 J (57 to 79 ft-lb) for the beltline sections and from 84 to 90 J (62 to 66 ft-lb) for the nozzle course sections. Thus, for both the transition temperatures and USEs, the nozzle course weld results fall within the bounds exhibited by the beltline weld.

The Charpy impact results were also statistically analyzed to determine confidence bands (normal distribution is assumed) for each set of results. Additionally, similar analyses were done for combined data sets, including each section, each thickness location, all beltline tests, all nozzle course tests, and all tests. Figures 5 and 6 show plots of Charpy impact energy versus test temperature for all the tests performed with 68 and 95% confidence intervals (essentially 1 and 2 standard deviations), respectively. Those plots represent well over 200 individual Charpy impact tests. The 41-J temperature bounds range from -21 to 5°C (-5 to 41°F) for the 68% interval and from -36 to 17°C (-32 to 63°F) for the 95% interval. On the upper shelf, the variations are ±9 J (±7 ft-lb) for the 68% interval and ±18 J (±13 ft-lb) for the 95% interval. The variations shown by these confidence intervals were compared to those from welds 72W and 73W tested in the HSSI Fifth Irradiation Series,[3] a high upper-shelf SAW, and found to be very similar.

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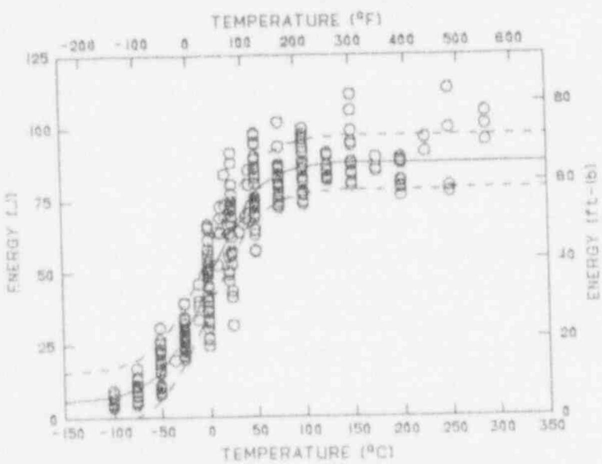

Figure 5. Unirradiated Charpy impact energy versus test temperature for Midland Unit 1 reactor vessel beltline and nozzle course welds. Plot shows mean (solid) hyperbolic tangent curve fit as well as upper and lower 68% confidence bounds (dashed.)

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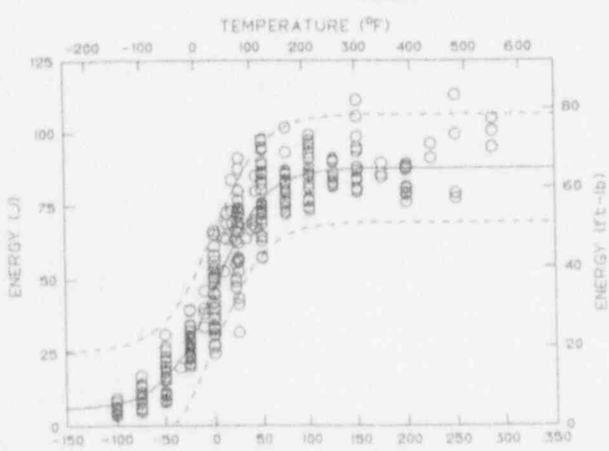

Figure 6. Unirradiated Charpy impact energy versus test temperature for Midland Unit 1 reactor vessel beltline and nozzle course welds. Plot shows mean (solid) hyperbolic tangent curve fit as well as upper and lower 95% confidence bounds (dashed).

The ASME Code[1] requires determination of the RT_{NDT} through a combination of drop-weight and Charpy impact tests. As mentioned earlier, Charpy tests were conducted to dete mine that the RT_{NDT} is controlled by the Charpy behavior in each case examined. However, sufficient specimens were not available from each set to continue testing for determination of the RT_{NDT} in that manner so that the RT_{NDT} determinations will be made by the alternative method of constructing a lower bound curve and subtracting 60°F (33°C) from the temperature at 68 J (50 ft-lb). These results will be reported later.

6.3 Chemical Composition

Regarding chemical composition, fracture surfaces were cut off CVN specimens from the various locations and submitted for chemical analyses to Combustion Engineering, Inc., (CE), Chattanooga, Tenn., or MQS Inspection, Inc., Woodlawn, Ohio. The first section to be analyzed was beltline weld section 1-13 and the results were reported previously.[2] The copper contents in section 1-13 range from 0.21 to 0.32 wt % (mean value of 0.25 wt %). Those values are substantially lower than those reported by B&W in BAW-1799 (ref. 4) for WF-70. They are, however, consistent with the copper content reported in the weld wire qualification record, 0.27 wt %.

Subsequent analyses of the other three beltline weld sections confirmed the relatively low mean copper contents in the beltline region, while analyses of the two nozzle course sections confirmed the higher copper contents reported by B&W. The B&W analyses were conducted with material removed from nozzle course dropouts, so

their analyses also represented the nozzle course weld. Table 1 provides a summary of the chemical analysis results for some of the major elements associated with radiation sensitivity. The elements other than copper are nominally the same in the beltline and nozzle course welds. The copper content ranges from 0.21 to 0.34 wt % in the beltline weld and from 0.37 to 0.46% in the WF-70 portion of the nozzle course weld. As Table 1 shows, the average copper contents are about 0.26 and 0.40% for the beltline and nozzle course welds, respectively. The overall average copper content is 0.29% with a standard deviation of about 0.07%. Interestingly, those average and standard deviation results for the WF-70 welds in the Midland Unit 1 reactor vessel are the same as those reported in BAW-1799 (ref. 4) for all Linde 80 welds made with copper-coated wire.

In addition to the optical microscopy and bulk chemical analyses, atom-probe field-ion microscopy (APFIM) analysis of the beltline weld in section 1-13 was performed at five through-thickness locations. The APFIM analyses were conducted to characterize the nature of precipitates or clusters, especially those of copper, in the microstructure and to compare the matrix copper content with the bulk content. Similar APFIM analyses will be conducted with the nozzle course welds and with both welds in the postirradiated condition. High-quality field-ion micrographs of the weld were obtained as shown in Fig. 7. No evidence of darkly imaging copper precipitates or clusters was observed in any of the weld samples. Some ultrafine brightly imaging refractory carbides were observed, as indicated by the arrow in Fig. 7; these carbides were observed in all five samples, but the field-ion micrographs that were printed did not always have them in the

Table 1. Summary of major radiation-sensitive elements for Midland Unit 1 reactor vessel welds

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

[a]Number of measurements.

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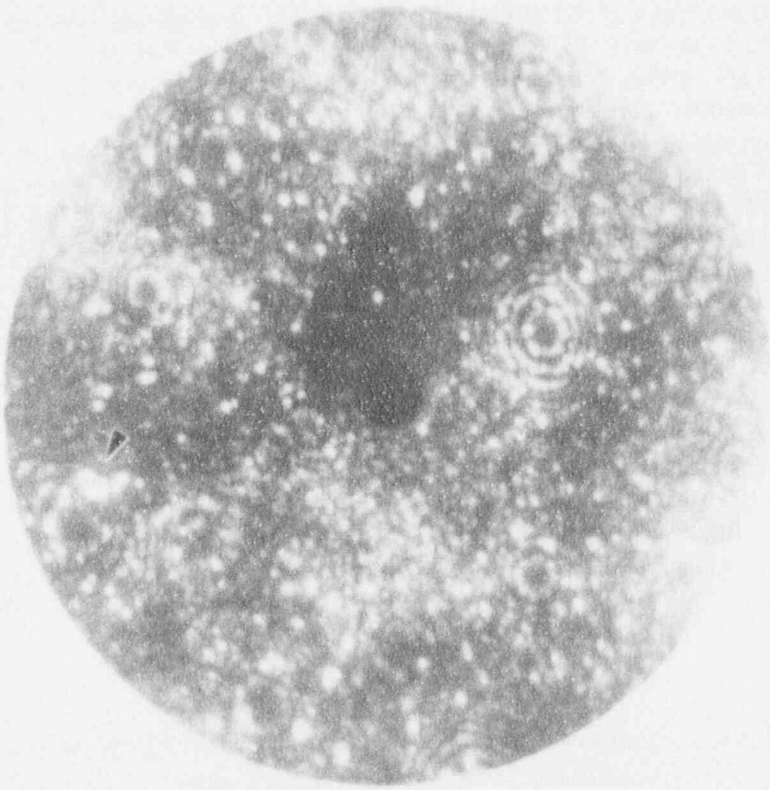

Figure 7. Field-ion micrograph of Midland Unit 1 reactor vessel beltline section 1-13 at the 3/4t location.

field of view. The carbides were roughly spherical and approximately 1 nm in diameter. Similar ultrafine carbides have been observed in irradiated A 302 grade B steel.[5] The average compositions of the five beltline weld samples were determined in the ORNL energy-compensated atom probe and are summarized in Table 2. The results indicate that there was a substantial depletion in the copper in the matrix over the bulk chemical analysis. This behavior has been observed previously in other reactor steels and is due to copper precipitation at grain boundaries and other microstructural features during the postweld heat treatment.[5] The atom-probe composition data were subjected to statistical analysis to determine whether the copper was in random solid solution in the matrix or whether clustering had occurred. The results of the significance of the mean separation of the copper atoms in the atom-probe ion-by-ion data chain indicated that no deviation from a random solid solution was detected. The atom-probe data were also analyzed in terms of the Johnston and Klotz order parameter, and, again, the results indicated no deviation from a random solid solution.

6.4 Unirradiated Testing and Irradiation Experiments

Preliminary fracture toughness tests of both beltline and nozzle course welds were reported in the last semiannual progress report.[2] Most of the fracture toughness specimens for the unirradiated test matrix have been fabricated and testing is under way. Additionally, Charpy impact, tensile, and 0.5TC(T) specimens have been fabricated for capsules 1 and 2 [5×10^{18} neutrons/cm^2 (>1 MeV)] and for capsules 3 and 4 [5×10^{19} neutrons/cm$_2$ (>1 MeV)]. Fabrication of the various specimens (drop-weight, Charpy, tensile, and fracture toughness) for capsule 5 [1×10^{19} neutrons/cm^2 (>1 MeV)] is under way. The final design of the large capsules is also under way, and insertion of capsule 5 in the FRN at the University of Michigan is anticipated in late fall of 1991.

Table 2. Atom-probe analysis of the matrix composition for Midland Unit 1 reactor vessel beltline section 1-13

	Element (at %)									
Location	Cu	Ni	Mn	Mo	Cr	Co	Si	P	C	Fe
1/4t	0.09	0.49	1.45	0.14	0.28	0.01	1.16	0.01	0.05	balance
Error	0.04	0.09	0.16	0.04	0.09	0.01	0.15	0.01	0.03	
1/2t(D)	0.11	0.59	1.39	0.23	0.08	0.02	1.57	0.06	0.02	balance
Error	0.02	0.06	0.09	0.04	0.02	0.01	0.09	0.02	0.01	
1/2t(E)	0.11	0.64	1.38	0.15	0.07	0.01	1.25	0.03	0.01	balance
Error	0.04	0.08	0.12	0.04	0.03	0.01	0.12	0.01	0.01	
5/8t	0.06	0.45	1.22	0.36	0.18	0.04	1.60	0.05	0.16	balance
Error	0.06	0.17	0.29	0.16	0.11	0.02	0.32	0.04	0.09	
3/4t	0.11	0.58	1.28	0.12	0.05	0.02	1.35	0.004	0.004	balance
Error	0.02	0.05	0.07	0.02	0.01	0.01	0.07	0.003	0.003	

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7. MICROSTRUCTURAL ANALYSIS AND MODELING

K. Farrell

7.1 Damage Mechanisms

7.1.1 Neutron Spectrum Effects

The study of neutron spectrum effects on embrittlement of three RPV steels, A212B, A350, and A533B, and one support structure material, A36, that was expected to begin producing results in February of this year has been unavoidably derailed for a second time by reactor closedown. This time, the National Research Universal (NRU) reactor at Chalk River, to which we had switched our experiments following closedown of the High-Flux Beam Reactor (HFBR) at Brookhaven National Laboratory, sprung a water leak. The operators of the NRU have worked quickly to investigate the leak, and the prognosis is that the leak is fixable and the reactor should be back on line by September of this year. In the meantime, there are signs that the HFBR may be restarted soon, but it will not be open for experiments until the fall. Because we have a great deal of effort invested in dosimetry, specimen packaging, and experiment approvals for the NRU, we will await restart of the NRU. We will hold the HFBR option in reserve in case the NRU suffers further delay.

7.1.2 Influence of Copper Impurity on Embrittlement

Radiation-induced precipitates of a body-centered cubic copper-rich phase have been linked to embrittlement of LWR pressure-vessel materials during irradiations at temperatures above 150°C. There is tentative evidence from the U.S.S.R. that copper might also acerbate embrittlement during irradiations at ~50°C. Such low-temperature effects might be a factor in the premature embrittlement of the High Flux Isotope Reactor (HFIR) pressure vessel and are becoming a major area of concern for LWR vessel support structures that typically operate in the temperature range 50 to 150°C. We have initiated some new experiments to ascertain whether copper impurity does accelerate embrittlement during irradiation at low temperature. The tests will be made with miniature tensile specimens irradiated in perforated capsules in the high flux hydraulic facility of the HFIR at a temperature of ~60°C. The materials are the model alloys used by the International Group on Radiation Damage Mechanisms (IGRDM), obtained from J. R. Hawthorne of ME Associates, and consisting of pure Fe, Fe-Cu, Fe-Cu-Ni, and Fe-Cu-Ni-P. The exposure fluences will span the range 10^{16} to 10^{19} n/cm^2. If accelerated embrittlement is found in any of these alloys, we will pursue the cause(s) with TEM and APFIM studies.

The copper effect experiments were described very briefly at the IGRDM third annual meeting in Raleigh, N.C., April 22-26, 1991, and they drew the attention of two parties who have asked to participate in the experiments. W. Phythian of Harwell wishes to include some hardness coupons and small-angle neutron scattering (SANS) coupons of the Electric Power Research Institute (EPRI)/University of California, Santa Barbara (UCSB)/Harwell model Fe-Cu-Ni-P alloys. F.M.D. Bowden, of the United Kingdom's Nuclear Installations Inspectorate, wants to irradiate tensile specimens of the vessel support structure material for the new Sizewell B reactor, England. Because most United Kingdom test reactors have been closed, Bowden is presently irradiating his material at Saclay, France. He would like to use the HFIR irradiations as an independent check of the French tests. We will accommodate these requests on a cooperative basis.

7.1.3 Baseline Low-Temperature Embrittlement Data of Pressure Vessel Materials

Materials test reactors for development of reference embrittlement data of pressure-vessel (PV) materials are becoming scarce due to closures. The return to service of the HFIR provides an opportunity to use its hydraulic facility to gather limited reference data. The limitations are a small irradiation volume (~3/8 in. ID by 2 in. per capsule) and a fixed temperature of ~60°C. These conditions exclude Charpy or fracture toughness specimens but are suitable for miniature tensiles, hardness coupons, SANS, TEM disks, etc. As described above, we will be using the facility to obtain low-temperature irradiation data for model

alloys and for PV support structure material. We also plan to use it to obtain reference tensile data on the four commercial steels (A212B, A350, A533B, and A36) on which our spectral effects work is focused. We already have reference tensile data on two of the steels, A212B and A350, obtained in the Oak Ridge Research Reactor (ORR) just before it was decommissioned. Comparison irradiation of these two steels in the HFIR hydraulic tube will show whether the hydraulic facility can be an acceptable low-temperature substitute for overall dwindling materials test reactor capabilities.

8. IN-SERVICE AGED MATERIALS 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 of 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, blocks of HSST plate 03 were sent to Korean Heavy Industries to meet their needs for correlation monitor materials for units 3 and 4 of the Yong Gwang PWR.

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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. Analyses of -ecleavage stable ductile tearing of high-copper welds 72W and 73W demonstrated that the size e..ects observed in the transition region are not due to substantial differences in ductile tearing behavior. Drop-weight tests, Charpy impact tests, and chemical analyses of the Midland reactor low upper-shelf welds were completed, showing large variations in bulk copper content, transition temperature, and upper-shelf energy. Atom probe field ion microscopy analyses revealed no evidence of fine copper precipitates or clusters in the unirradiated Midland welds but a substantial depletion in the copper concentration in the matrix.

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