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ORNL/TM-11568/V5&N2
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Heavy-Section Steel Irradiation Program

Progress Report for
April 1994 – September 1994

Prepared by
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Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

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

For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established with its primary goal to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure-vessel steels as they relate to light-water RPV integrity. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 14 tasks: (1) program management, (2) fracture toughness curve shift in high-copper weldments (Series 5 and 6), (3) K_{Ic} and K_{Ia} curve shifts in low upper-shelf (LUS) welds (Series 8), (4) irradiation effects in a commercial LUS weld (Series 10), (5) irradiation effects on weld heat-affected zone and plate materials (Series 11), (6) annealing effects in LUS welds (Series 9), (7) microstructural and microfracture analysis of irradiation effects, (8) in-service irradiated and aged material evaluations, (9) Japan Power Development Reactor (JPDR) steel examination, (10) fracture toughness curve shift method, (11) special technical assistance, (12) technical assistance for Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, (13) correlation monitor materials, and (14) test reactor coordination.

During this period, all remaining irradiated Italian crack-arrest specimens were tested. The fabrication of three trial LUS scoping welds was begun to identify possible materials for studies on K_{Ic} shifts in LUS materials. A NUREG report detailing the unirradiated material properties of the LUS Midland Weld WF-70 was submitted to the Nuclear Regulatory Commission for publication that describes a disagreement between empirical fracture toughness test methods (e.g., Charpy impact and drop-weight tests) and fracture mechanics-based toughness evaluations. The irradiation of the second of the two large capsules containing Midland weld metal was completed. Testing of specimens from the Midland weld scoping capsules indicated that the measured Charpy 41-J temperature shifts are substantially less than predicted. The effects of annealing on recovery of the toughness were studied on three LUS welds, obtaining transition range fracture toughness master curves for welds 63W, 64W, and 65W. The major portion of the design of the new irradiation, annealing, and reirradiation facilities was completed. A combination of kinetic modeling, atom-probe field-ion microscopy data, and analysis of data from the Power Reactor Embrittlement Data Base (PR-EDB) were shown to indicate that copper in excess of the solubility limit at the final stress relief temperature is unlikely to be available to contribute to radiation-induced embrittlement. Analysis of the PR-EDB indicated that there is no significant difference between fluence dependence of embrittlement exhibited by pressurized-water reactors and boiling-water reactors, implying that there are no major spectral differences between the two reactor types. Results obtained from the ion-irradiation experiments on model alloys indicate a near-linear dependence of hardening on copper content. A computer numerically controlled machining center was procured by Oak Ridge National Laboratory (ORNL) for hot-cell machining operations. Tests of three-wire stainless steel cladding thermally aged for 20,000 h at 288 and 343°C were completed and show little effect of such aging. On the other hand, Charpy impact results for type 308 shielded metal-arc welds aged at 343°C for 20,000 h do show embrittlement. The formal collaborative research agreement between ORNL and the Japan Atomic Energy Research Institute was finalized, and the irradiated material from the wall of the JPDR was moved into temporary storage at the hot cells. Detailed plans were developed and work was initiated to evaluate the existing neutron dosimetry and transport calculations which have been performed for the JPDR by the Japanese. Acquisition of literature related to irradiation effects on fracture toughness of RPV steels was initiated, in support of a thorough examination of the technical basis for the current methods for shifting fracture toughness curves to account for irradiation damage. Portions of HSSI weld 72W have been supplied

to SRI, Inc., and AEA-Technology, Harwell, United Kingdom, for testing of precracked cylindrical tensile specimens. A large matrix of materials and specimen types has been completed, for subsize Charpy impact specimen evaluation. Normalization analyses for upper-shelf energies and transition temperatures of the subsize specimens relative to the full-size Charpy specimens have also been completed and will be used in development of a standard practice for subsize Charpy testing. Reconstituted Charpy specimens from six participating institutions have been tested and the preliminary results reported to the American Society for Testing and Materials. Charpy impact and round tensile specimens were fabricated from two Russian weld metals and irradiated. Some portions of the planned irradiations, annealing, and testing of U.S. steels have been completed in Russia. The construction of a controlled-access storage location at ORNL for correlation monitor materials was initiated and material transfer to it begun. Correlation monitor material from Heavy-Section Steel Technology Plate 02 was cut and shipped to Consumers Power Company in Michigan for use in surveillance capsules to be installed in the Palisades reactor vessel during a refueling outage in 1995. Design efforts were initiated to modify current ORNL irradiation capsules to facilitate the irradiation of University of California, Santa Barbara, specimens.

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Preface

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

This HSSI Program progress report covers work performed from April to September 1994. 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:

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

ORNL-4176
ORNL-4315
ORNL-4377
ORNL-4463
ORNL-4512
ORNL-4590
ORNL-4653
ORNL-4681
ORNL-4764
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NUREG/CR-1806 (ORNL/NUREG/TM-419)
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Summary

1. Program Management

The Heavy-Section Steel Irradiation (HSSI) Program is arranged into 14 tasks: (1) program management, (2) fracture toughness curve shift in high-copper weldments (Series 5 and 6), (3) K_{Ic} and K_{Ia} curve shifts in low upper-shelf (LUS) welds (Series 8), (4) irradiation effects in a commercial LUS weld (Series 10), (5) irradiation effects on weld heat-affected zone (HAZ) and plate materials (Series 11), (6) annealing effects in LUS welds (Series 9), (7) microstructural and microfracture analysis of irradiation effects, (8) in-service irradiated and aged material evaluations, (9) Japan Power Development Reactor (JPDR) steel examination, (10) fracture toughness curve shift method, (11) special technical assistance, (12) technical assistance for Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, (13) correlation monitor materials, and (14) test reactor coordination. Report chapters correspond to the tasks. The work is performed by the Oak Ridge National Laboratory (ORNL). During the report period, two technical papers and three foreign trip reports were published. In addition, 37 technical presentations were made.

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

The objective of this task is to develop data addressing the current method of shifting the American Society of Mechanical Engineers fracture toughness (K_{Ic}, K_{Ia}, and K_{IR}) curves to account for irradiation embrittlement in high-copper welds. The specific activities to be performed in this task are the (1) continuation of Phase 2 of the Fifth Irradiation Series and (2) completion of the Sixth Irradiation Series, including the testing of the nine irradiated Italian crack-arrest specimens. All nine irradiated Italian crack-arrest specimens have been successfully tested. The preliminary test results are presented here, and full details will be published in a NUREG report in preparation. In order to test the specimens with the ORNL clip gage, special clip gage blocks were designed and manufactured. The specially fabricated remote fixture was first set up in the Fracture Mechanics Laboratory and "shaken down" using surrogate unirradiated crack-arrest specimens. Potential problems were identified and mitigated before the fixture was moved into the hot cell to set up on the 500-kN MTS machine. The three larger specimens were particularly difficult to move into the hot cell because of their size. The fixture used with the irradiated Italian crack-arrest specimens will be decontaminated for use in the Fracture Mechanics Laboratory with the unirradiated crack-arrest specimens. A similar but smaller fixture capable of handling the crack-arrest specimen sizes generally used by ORNL is available for testing the irradiated Midland weld crack-arrest specimens.

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

This task examines the fracture toughness curve shifts and changes in shape for irradiated welds with low Charpy upper-shelf energy (USE). The information developed under this task is to augment that obtained in a similar irradiation experiment performed on two high Charpy USE weldments under the Fifth and Sixth Irradiation Series. The results will provide an expanded basis for accounting for irradiation-induced embrittlement in reactor pressure vessel (RPV) materials. A purchase order to fabricate three welds was issued to ABB-Combustion Engineering (ABB-CE), Chattanooga, Tennessee. This includes one weld using 73W wire and two others which will contain 0.31 and 0.45% Cu and impurities that are typical of older generation nuclear pressure vessels. ABB-CE fabricated the welds for the Fifth and Sixth Irradiation Series.

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

The objective of this task is to evaluate the transition temperature shift of the WF-70 weld metal at the beltline and nozzle course locations in the Midland Unit 1 reactor vessel. A NUREG report, *Unirradiated Material Properties of Midland Weld WF-70,* by D. E. McCabe, R. K. Nanstad, S. K. Iskander, and R. L. Swain [NUREG/CR-6249 (ORNL/TM-12777)], has been submitted to the Nuclear Regulatory Commission (NRC). An observation of particular interest is a basic disagreement between empirical fracture toughness test methods (e.g., Charpy impact and drop-weight tests) and fracture mechanics-based toughness evaluations (e.g., K_{Ic} and J-R tests). For these welds, the fracture toughness results indicate differences between the two welds which are not reflected by the Charpy results. Capsule 10.06, the second of two large capsules irradiated at the University of Michigan Ford Nuclear Reactor (FNR) completed irradiation about the first of September and is planned to be transported to ORNL about January of 1995. The specimens from capsules 10.01 and 10.02, irradiated in the University of Buffalo Reactor, are being tested in the hot cells at ORNL. Preliminary results indicate the measured Charpy 41-J temperature shifts are substantially less than predicted by *Regulatory Guide 1.99*, Rev. 2.

5. Irradiation Effects On Weld Heat-Affected Zone and Plate Materials (Series 11)

The purpose of this task is to examine the effects of neutron irradiation on the fracture toughness (ductile and brittle) of the HAZ of welds and of A302 grade B (A302B) plate materials typical of those used in fabricating older RPVs. Based on a survey of materials used for currently operating commercial reactors, the NRC decided to place initial emphasis on A302B steel, not the A302B modified with added nickel. Initial inquiries were made regarding procurement of A302B plate in sufficient quantities for the irradiation program. For the HAZ portion of the program, the intent is to examine HAZ material in A302B as well as in either A302B (modified) or A533, grade B, class 1 plate.

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

The purpose of the Ninth Irradiation Series is to evaluate the correlation between fracture toughness and Charpy V-notch impact energy during irradiation, annealing, and reirradiation (IAR). The effects of annealing on recovery of the toughness were studied on three LUS welds fabricated with materials and procedures used in early pressurized-water RPVs and remaining from the Second, Third, and Fourth Irradiation Series. Using the approach under development for transition range, fracture toughness master curves were derived for welds 63W, 64W, and 65W. Automated ball-indentation-derived yield strength and stress-strain curves were also obtained from tests performed on Heavy-Section Steel Technology (HSST) Plate 02 in the unirradiated and IAR conditions.

The major portion of the design of the new IAR facilities is complete. As it is being reviewed for fabrication, modifications may become necessary which, in turn, could involve a significant number of modifications to the design.

7. Microstructural and Microfracture Analysis of Irradiation Effects

The overall long-term goal of this task is to develop a physically based model which can be used to predict irradiation-induced embrittlement in reactor vessel steels over the full range of their service conditions. The model should be tethered soundly on the microstructural level by results from advanced microstructural analysis techniques and constrained at the macroscopic level to produce predictions consistent with the large array of macroscopic embrittlement measurements that are available. During this reporting period, a combination of kinetic modeling, atom-probe field-ion microscopy data, and analysis of data from the Power Reactor Embrittlement Data Base (PR-EDB) were shown to indicate that copper in excess of the solubility limit at the final stress relief temperature is unlikely to be available to contribute to radiation-induced embrittlement. Additional analysis of the PR-EDB indicates that there is no significant difference between fluence dependence of embrittlement exhibited by pressurized-water reactors and boiling-water reactors. This indicates that there are not major spectral differences between the two reactor types. Results obtained from the ion irradiation experiments on model alloys indicate a near-linear dependence of hardening on copper content with saturation occurring between 0.02 and 0.2 displacements per atom. Data from higher dose irradiations were somewhat anomalous, and the experiments are being repeated. High-magnification transmission electron microscopy did not reveal any differences between as-received and aged samples of weld 73W and a Russian weld in which the upper shelf was observed to increase during thermal annealing at 460°C. Significant grain boundary segregation of phosphorus was observed due to irradiation of Russian weld 28.

8. In-Service Irradiated and Aged Material Evaluations

The objective of this task is to provide a direct assessment of actual material properties in irradiated components of nuclear reactors, including the effects of irradiation and aging. A computer numerically controlled machining center was delivered to ORNL. It has been temporarily placed in a cold laboratory for initial verification, modification for remote operation inside a hot cell, and for operator training. Tests of three-wire stainless steel cladding thermally aged for 20,000 h at 288 and 343°C were completed and show little effect of such aging on tensile, Charpy impact, and fracture toughness properties. On the other hand, Charpy impact results for type 308 shielded metal-arc welds aged at 343°C for 20,000 h do show embrittlement, and results for the welds aged for 50,000 h show that the embrittlement continues with exposure time.

9. Evaluation of Steel from the JPDR Pressure Vessel

There is a need to validate the results of irradiation effects research by the examination of material taken directly from the wall of a pressure vessel which has been irradiated during normal service. This task has been included with the HSSI Program to provide just such an evaluation on material from the wall of the pressure vessel from the JPDR. During this reporting period, the formal collaborative research agreement between ORNL and the Japan Atomic Energy Research Institute was finalized, and the irradiated material from the wall of the JPDR was moved into temporary storage at the hot cells. A test matrix was developed to perform a cross comparison of Japanese and U.S. Charpy impact test machines that will be used in the JPDR studies. Detailed plans were developed and work was initiated to evaluate the existing neutron dosimetry and transport calculation, which have been performed for the JPDR by the Japanese, as well as to begin similar transport calculations at ORNL. The ORNL calculations will include the concrete in the biological shield as well as the pressure vessel.

10. Fracture Toughness Curve Shift Method

The purpose of this task is to examine the technical basis for the currently accepted methods for shifting fracture toughness curves to account for irradiation damage and to work through national codes and standards bodies to revise those methods, if a change is warranted. Acquisition of literature related to irradiation effects on fracture toughness of RPV steels was initiated. Also, fracture toughness and Charpy impact data for RPV steels in the unirradiated and irradiated conditions are being acquired and stored in a data base for evaluation and analysis. The data from all the relevant HSSI Programs are currently under evaluation, and the results will be presented in a letter report. Discussions have been held with researchers at some foreign institutions regarding their research activities in this area.

11. Special Technical Assistance

This task has been included with the HSSI Program to provide a vehicle in which to conduct and monitor short-term, high-priority subtasks and provide technical expertise and assistance in the review of national codes and standards that may be referenced in NRC regulations or guides related to nuclear reactor components. The following activities occurred during this reporting period. Portions of HSSI weld 72W have been supplied to SRI and AEA-Technology, Harwell, United Kingdom, for testing of precracked cylindrical tensile specimens. For the subsize Charpy impact specimen evaluation, a large matrix of materials and specimen types has been completed. Normalization analyses for USEs and transition temperatures of the subsize specimens relative to the full-size Charpy specimens have also been completed. To establish empirical values of ductile-to-brittle transition temperature shift due to specimen size, the results from this study and other published results were analyzed together and resulted in different temperature adjustments depending on the particular geometry of the subsize specimen. The results of these tests will be distributed to members of American Society for Testing and Materials (ASTM) Subcommittee E10.02 for their use in development of a standard practice for subsize Charpy testing. Reconstituted Charpy specimens from six participating institutions have been tested and the preliminary results reported to ASTM Subcommittee E10.02. Specimens from the four remaining participants will be tested as a group after they have all been received.

12. Technical Assistance for JCCCNRS Working Groups 3 and 12

The purpose of this task is to provide technical support for the efforts of the U.S.-Russian JCCCNRS Working Group 3 on radiation embrittlement and Working Group 12 on aging. Charpy impact and round tensile specimens were fabricated from two Russian weld metals supplied by the Russian National Research Center, Kurchatov Institute, and placed in HSSI capsule 10.06 for irradiation in the University of Michigan FNR. The capsule completed irradiation in early September 1994. The specimens will be returned to ORNL for testing in 1995. R. K. Nanstad met with Drs. A. Kryukov and A. Chernobaeva of Kurchatov Institute on September 29, 1994, regarding the irradiation and testing of U.S. steels provided to the Russian side as part of the cooperative test program. They have completed some portions of the planned irradiations, annealing, and testing. Some of the specimens of both U.S. materials which were irradiated and annealed are now being reirradiated.

13. Correlation Monitor Materials

This is a task that has been established with the explicit purpose of ensuring the continued availability of the pedigreed and extremely well-characterized material now required for inclusion in all additional and future surveillance capsules in commercial light-water reactors. Having recognized that the only remaining materials qualified for use as a correlation monitor in reactor surveillance capsules are the pieces remaining from the early HSST plates 01, 02, and 03, this task will provide for cataloging, archiving, and distributing the material on behalf of the NRC. During this reporting period, the transfer of the material from its site at the Y-12 Plant to a controlled-access storage location at ORNL was initiated. Detailed planning for the controlled-access location for storage of the remaining correlation monitor materials at ORNL was also continued. Construction of a concrete slab, as an extension of Building 7026, was begun. Additionally, correlation monitor material from HSST Plate 02 was cut and shipped to Consumers Power Company in Michigan for use in two surveillance capsules to be installed in the Palisades reactor vessel during a refueling outage in 1995.

14. Test Reactor Irradiation Coordination

The purpose of this task is to provide the support required to supply and coordinate irradiation services needed by NRC contractors other than ORNL. These services include the design and assembly of irradiation capsules as well as arranging for their exposure, disassembly, and return of specimens. Currently, the University of California, Santa Barbara (UCSB), is the only other NRC contractor for whom irradiations are to be conducted. During this reporting period, design efforts were initiated to modify current ORNL irradiation capsules to facilitate the irradiation of UCSB specimens. A preliminary analysis of the thermal performance of a basic concept was completed, showing that heated facilities will be required for the UCSB experiments. Hence, a more precise nuclear heat generation rate distribution calculation was initiated. The facility should consist of rectangular, heated core-edge positions that can be easily repositioned in the trolley to provide a variety of fluxes.

Heavy-Section Steel Irradiation Program Semiannual Progress Report for April through September 1994*,†

W. R. Corwin

1. Program Management

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

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

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

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

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(3) K_{Ic} and K_{Ia} curve shifts in LUS welds (Series 8), (4) irradiation effects in a commercial LUS weld (Series 10), (5) irradiation effects on weld heat-affected zone (HAZ) and plate materials (Series 11), (6) annealing effects in LUS welds (Series 9), (7) microstructural and microfracture analysis of irradiation effects, (8) in-service irradiated and aged material evaluations, (9) Japan Power Development Reactor (JPDR) steel examination, (10) fracture toughness curve shift method, (11) special technical assistance, (12) technical assistance for Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Groups 3 and 12, (13) correlation monitor materials, and (14) test reactor coordination.

During this period, 16 program briefings, reviews, or presentations were made by the HSSI staff during program reviews and visits with NRC staff or others. Two technical papers[1,2] and three foreign trip reports[3-5] were published. In addition, 37 technical presentations were made.[6-42]

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2. Fracture Toughness Shift in High-Copper Weldments

S. K. Iskander, C. W. Marschall,* A. Pini,† P. P. Millela,†
E. T. Manneschmidt, and K. W. Boling

2.1 Introduction

The objective of this task is to develop data addressing the current method of shifting the American Society of Mechanical Engineers (ASME) fracture toughness and crack-arrest toughness (K_{Ic}, K_{Ia}, and K_{IR}) curves to account for irradiation embrittlement in high-copper welds.

The specific activities to be performed in this task are (1) continuation of Phase 2 of the Fifth Irradiation Series and (2) completion of the Sixth Irradiation Series, including the testing of the nine irradiated Italian crack-arrest specimens. All nine irradiated Italian crack-arrest specimens have been successfully tested. The preliminary test results are presented here, and full details will be published in a NUREG report in preparation.

The U.S. NRC agreed to have ORNL test irradiated crack-arrest specimens belonging to the Italian Agenzia Nazionale per la Protezione dell'Ambiente (ANPA).‡ Testing was performed according to the American Society for Testing and Materials (ASTM) "Test for Determining Plane-Strain Crack-Arrest Fracture Toughness, K_{Ia}, of Ferritic Steels" (E 1221-88). The specimens were manufactured from a forging material whose specifications are in accordance with the ASTM "Specification for Quenched and Tempered Vacuum-Treated Carbon and Alloy Steel Forgings for Pressure Vessels, Class 3" (A 508 class 3). Thus, the results have usefulness and applicability to the safety assessment of U.S. RPVs. The HSSI Program has previously tested irradiated crack-arrest specimens using a specially designed and manufactured fixture.[1,2] However, this fixture was too small for the three large ANPA specimens, and a new, larger remote crack-arrest fixture was designed, manufactured, and tested.

As described in the two previous semiannual reports, the new, remote crack-arrest fixture was built and successfully tested using unirradiated specimens from the Midland weld. It was then used to test the large irradiated crack-arrest specimens from the ANPA. The results will have usefulness and applicability to the research performed for the HSSI Program. A method to measure the crack-mouth opening of the ANPA crack-arrest specimens was devised that necessitated the design and manufacture of special clip gage blocks.

The new, remote crack-arrest fixture was evaluated using unirradiated LUS energy weld metal crack-arrest specimens from the so-called Midland weld WF-70.[3] Potential problems were identified and mitigated before the fixture was moved into the hot cell to set up on the 500-kN MTS machine. During these tests, thermocouples (TCs) were tack-welded to the specimen to check the removable TC that will be used in the hot cells. A survey of the distribution of the temperature in the crack-arrest specimens was performed over the temperature range that the device will be used, and for both the "inverted" and "normal" crack-arrest specimen positions. The survey indicated that the maximum difference between the removable and tack-welded TCs was approximately $\pm 2°C$. The fixture used with the irradiated Italian crack-arrest specimens will be decontaminated for use in the Fracture Mechanics Laboratory with unirradiated crack-arrest specimens.

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‡Changes in 1994 have moved the Italian Nuclear Safety Authority, previously known as the Committee for Research and Development of Nuclear Energy and Alternative Energies (ENEA), to the ANPA.

Some time ago, ANPA started an extensive research program to characterize an ASTM A 508 class 3 forging produced in Italy.[4] 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. Several institutions are involved in the irradiation and testing of these specimens: ANPA CRE Casaccia Laboratories, Battelle Columbus Laboratories (BCL), and two laboratories of the French Commissariat a l'Energie Atomique (CEA). ANPA originally planned to test the irradiated crack-arrest specimens at BCL, but BCL has since then decommissioned their hot cell facilities.

2.2 Description of the Irradiated Specimens

There are nine ANPA irradiated crack-arrest specimens manufactured from an ASTM A 508 class 3 forging. The external dimensions of the three "large" specimens are 25 × 200 × 200 mm, and the six "small" ones are 13 × 100 × 100 mm. The fluence of the nine specimens varied between approximately 1.8 to 2.7 × 10^{19} neutrons/cm^2 (> 1 MeV), and the irradiation temperature varied from 240 to 280°C. The nine ANPA crack-arrest specimens were irradiated at the Ford Nuclear Reactor (FNR), Ann Arbor, Michigan, and then shipped to ORNL. The three larger specimens were particularly difficult to move into the hot cell because of their size.

The measurement of the crack-mouth opening displacement (CMOD) had originally been intended to be performed using a clip gage designed to be seated on knife edges located on the front face of the specimen. At ORNL, the CMOD of crack-arrest specimens is measured using clip gage blocks with conical recesses that receive a clip gage with conical points. In the case of unirradiated specimens, the clip gage blocks are attached by driving split pins in slightly undersized holes. If the specimens are to be irradiated, the clip gage blocks are attached in the same manner as unirradiated specimens, and then, to minimize the possibility of the clip gage blocks becoming detached, they are tack-welded to the specimens. According to ASTM E 1221-88, the CMOD is measured at a distance of 0.25W from the load line and on the opposite side of the crack tip, where W is the nominal width of the specimen.

Since ORNL has considerable experience with the conical points clip gage, after consultations with all interested parties, it was decided to use the ORNL method. Clip gage blocks were designed, manufactured, and remotely attached to the ANPA specimens. The CMOD measurements were made at distances of 0.29 and 0.27W in the small and large specimens, respectively, rather than the 0.25W location prescribed in the crack-arrest standard. An adjustment was made to correct for the slightly larger CMODs that would be measured. The correction was ~4 and ~2% for the small and large specimens, respectively.

2.3 Results of Tests on Irradiated Specimens

The results are shown in Figure 2.1 in which the test data have been "adjusted" to a reference temperature and fluence, as explained below. The ASME K_{Ia} equation* for the value of RT_{NDT} indicated has also been plotted, which, in SI units, is:

$$K_{Ia} = 29.4 + 13.675 \exp\left[0.0261\left(T - RT_{NDT}\right)\right] , \tag{2.1}$$

*The previous ASME equation was

$$K_{Ia} = 29.4 + 1.344 \exp[0.0261(T - RT_{NDT} + 89)].$$

The new ASME equation given in the Dec. 31, 1992, addenda of the *ASME Boiler and Pressure Vessel Code* appears to be a rewrite of the one given in WRC Bulletin 175 (August 1972).

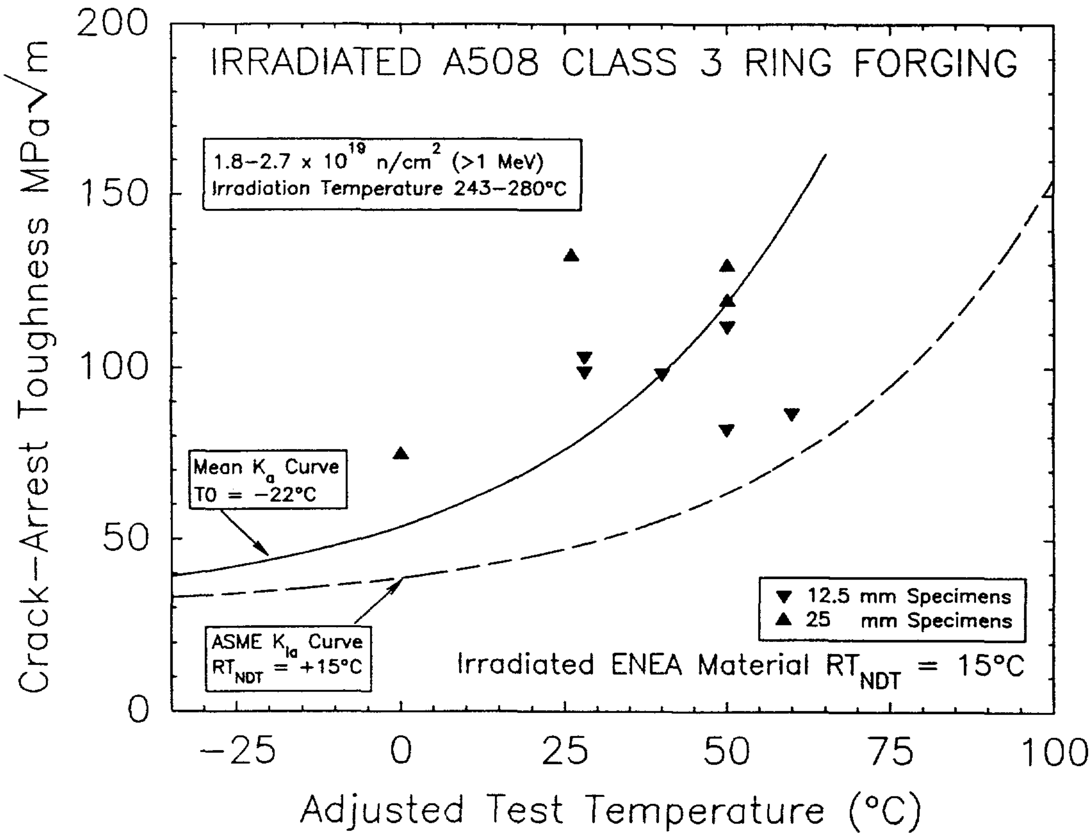

Figure 2.1. **Preliminary results of testing irradiated crack-arrest specimens from the Italian Agenzia Nazionale per la Protezione dell'Ambiente (ANPA). The crack-arrest toughness, K_a, has been adjusted to a single reference fluence of 3.2 × 10^{19} neutrons/cm^2 (> 1 MeV) and an irradiation temperature of 280° C.**

where K_{Ia} is the crack-arrest toughness in MPa√m, and T is the test temperature in °C. The second curve is based on the ASME curve and is as follows:

$$K_a = 29.4 + 13.675 \exp[0.0261(T - T_o)] \ , \qquad (2.2)$$

where T_o is a parameter obtained by nonlinear regression through all the data points and should not be confused with T_o used in the master curve concept. The value of T_o depends upon the material, number of experimental points available, etc. Thus, this curve may be interpreted as a mean curve in the least-squares sense.

The test data will be adjusted to account for the different exposure parameters of each specimen to a reference fluence and temperature of the CVN data irradiated in the same reactor as the crack-arrest specimens, thus reducing spectrum effects. The fluence for these CVN data is 3.2 × 10^{19} neutrons/cm^2 (> 1 MeV), and the irradiation temperature is 280° C. The test data were "adjusted" using the following method due to Odette.[5] The test temperature was increased 1 K for every degree that the irradiation temperature was less than the reference temperature, and vice versa, i.e., an adjustment of ±1 K to the test temperature per ∓1 K change in irradiation temperature.

The test temperature was also adjusted to the reference fluence by first determining the temperature shift at the 41-J impact energy level, $\Delta TT_{41\text{-}J}$, between the test temperature and the mean unirradiated K_a curve, as given in Equation (2.2). An adjusted shift was then calculated as follows:

$$\Delta TT'_{41-J} = \Delta TT_{41-J} \left(\frac{\Phi'}{\Phi} \right)^{0.5} , \tag{2.3}$$

where

$\Delta TT'_{41\text{-}J}$ = shift adjusted to the reference fluence, Φ',
$\Delta TT_{41\text{-}J}$ = shift between the test temperature and the mean unirradiated curve,
Φ' = the reference fluence, 3.2×10^{19} neutrons/cm^2 (> 1 MeV),
Φ = the fluence of the specimen.

The adjusted test temperature, T′, was then calculated as follows:

$$T' = T + (\Delta TT'_{41-J} - \Delta TT_{41-J}) + (T_{irr} - T_{ref}) , \tag{2.4}$$

where

T' = adjusted test temperature,
T = actual test temperature,
T_{irr} = temperature at which the specimen was irradiated,
T_{ref} = reference temperature, 280°C.

Analysis of the results is still under way, and a comprehensive report is being prepared.

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

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

This task examines the fracture toughness curve shifts and changes in shape for irradiated welds with low Charpy upper-shelf energy (USE). The information developed under this task is to augment that obtained in a similar irradiation experiment performed on two high Charpy USE weldments under the Fifth and Sixth Irradiation Series. The results will provide an expanded basis for accounting for irradiation-induced embrittlement in RPV materials. A purchase order to fabricate three welds was issued to ABB-Combustion Engineering (ABB-CE), Chattanooga, Tennessee. This includes one weld using 73W wire and two others which will contain 0.31 and 0.45% Cu and impurities that are typical of older generation nuclear pressure vessels. ABB-CE fabricated the welds for the Fifth and Sixth Irradiation Series.

The previous semiannual contained a discussion of the reasons for the above choices of copper content and welding wire. The first weld may be relatively easy to fabricate, since it uses the 73W weld wire, and no copper will be added during the welding. The other two welds are more difficult to fabricate since the weld wire is already copper coated, and the final copper content of the weld will depend on both the initial copper content of the weld wire as well as the feed rate of the supplemental copper to be added. It is expected that the delivery of the three welds will be completed by March 1995.

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

D. E. McCabe, S. K. Iskander, I. I. Siman-Tov, and D. W. Heatherly

4.1 Introduction

The objective of this task is to evaluate the transition temperature shift of the WF-70 weld metal at the beltline and nozzle course locations in the Midland reactor vessel. This material is classified as an LUS material, and the fracture toughness is to be characterized in the as-received condition as well as at three levels of irradiation fluence, 0.5, 1.0, and 5 $\times$ 10^{19} neutrons/cm^2. A complete survey of the material chemistries and mechanical properties has been made in the unirradiated condition.

4.2 Unirradiated Fracture Toughness Results

A NUREG report, *Unirradiated Material Properties of Midland Weld WF-70*, by D. E. McCabe, R. K. Nanstad, S. K. Iskander, and R. L. Swain [NUREG/CR-6249 (ORNL/TM-12777)], has been submitted to the NRC. An observation of particular interest is a basic disagreement between empirical fracture toughness test methods and fracture mechanics-based toughness evaluations. The empirical transition temperature methods are keyed by the temperature corresponding to 41 J of CVN energy and the no-break temperature of the DWT NDT test. These methods are recommended in the *Code of Federal Regulations* and the ASME Code to establish reactor operating limits. These methods did not indicate a difference in fracture toughness between the nozzle course and beltline WF-70 weld metals (see Table 4.1). However, fracture mechanics tests using

Table 4.1. Comparison of mechanical properties between nozzle course and beltline weld metal

Weld	Reference temperature (°C)			
	Charpy 41 J	Charpy upper-shelf energy	Average NDT[a]	T_o[b]
Beltline	–9	89.7	–53	–60
Nozzle	–1	88.6	–48	–33

[a]Nil-ductility transition.

[b]Temperature of 100-MPa√m fracture toughness.

1/2T compact specimens showed a significant difference. Here, a reference temperature, T_o, is used that is defined as the temperature where 1T compact specimens develop a mean fracture toughness value, $K_{Jc(mean)}$, of 100 MPa√m. Nozzle course and beltline specimens of 1/2T size were tested at –50°C, and the indicated reference temperatures were –60°C (–76°F) for the beltline weld and –33°C (–27°F) for the nozzle course weld. Tensile tests also indicated that there was some difference in the two weld metals (see Table 4.2).

Table 4.2. Tensile properties of unirradiated Midland WF-70 weld metal

Temperature		Strength			
		Yield		Ultimate tensile	
°C	°F	MPa	ksi	MPa	ksi
		Nozzle			
-100	-148	648	94.0	820	118.9
-75	-103	620	89.9	765	111.0
-50	-58	579	84.0	717	104.0
23	73	545	79.0	655	95.0
160	320	483	70.1	586	85.0
288	550	483	70.1	614	89.0
		Beltline			
-100	-148	548	79.5	758	110.0
-75	-103	483	70.1	710	103.0
-50	-58	465	67.4	682	98.9
23	73	407	59.0	586	85.0
288	550	427	61.9	558	80.9

Reference temperature, T_o, from fracture mechanics-type tests can be used to establish a universal transition curve that defines the median trend of fracture toughness versus test temperature. Replicate tests such as the six 1/2T specimens were used here to establish T_o on nozzle course and beltline welds and to characterize the variability of fracture toughness. Hence, confidence limits can be placed on data, as shown in Figure 4.1. The master curve and confidence limits were established from the 1/2T data only. The rest of the data shown are the balance of the tests performed on specimens of various sizes.

Another type of fracture mechanics toughness characterization is the J-R curve. These represent upper-shelf fracture toughness expressed in terms of the resistance to slow-stable crack growth. The ranges of specimen sizes and test temperatures covered are shown in Table 4.3. These data have provided an opportunity to evaluate a proposed CVN USE/J-R curve correlation model developed by Eason et al., and the comparison of this prediction to the Midland data is shown in Figure 4.2. In general, the predictions were found to be good, provided that the comparisons are based on results from side-grooved specimens and that test temperatures are above 100°C (210°F).

Again, J-R curves demonstrate that there is a difference in upper-shelf fracture toughness between beltline and nozzle course welds (see Figure 4.3). This conflicts with the CVN USEs shown in Table 4.1.

Use of modified J in J-R curve representation reduced specimen size effects. Figure 4.4 shows how a J-R curve from a large specimen is well approximated from a smaller specimen that would normally indicate lower toughness when using deformation theory J.

Although ASTM standards do not specify a specimen side-grooving practice, the effect of side grooves on J-R curve is enormous (see Figure 4.5). This was identified in the NUREG report mentioned earlier as a serious issue to be addressed, especially for the use of J-R curves as applied to plant licensing concerns.

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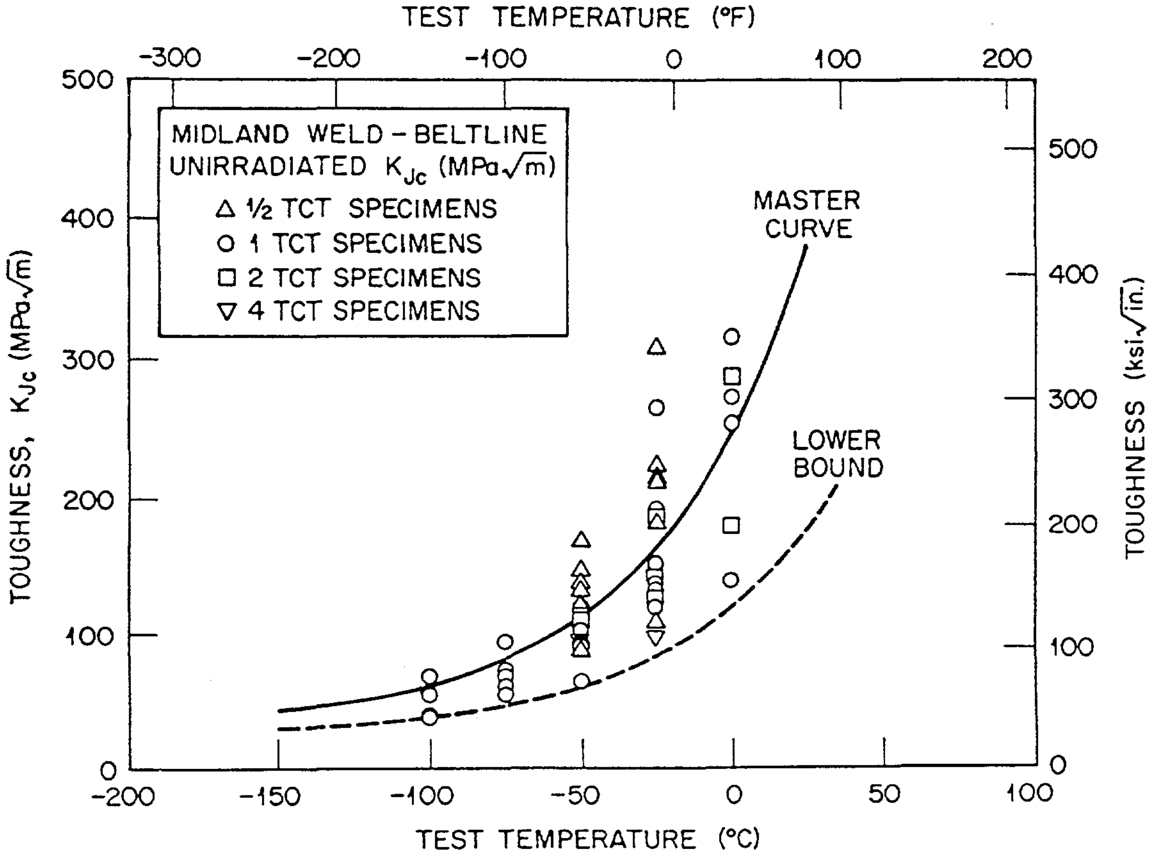

Figure 4.1. **Master curve and 5% confidence limit curve (dashed) adjusted 10°C for uncertainty in reference temperature, T_o.**

Table 4.3. J-R curve test matrix for Midland WF-70 weld metal

Specimen size	Number of specimens at various temperatures[a]		
	21°C (70°F)	150°C (302°F)	288°C (550°F)
	Beltline		
1/2T	2	2	2
1T	2	2	2
2T	-	-	2
4T	-	-	2
	Nozzle		
1/2T	-	-	2
1T	4[b]	2	2

[a]All specimens 20% side grooved unless noted otherwise.

[b]Two specimens not side grooved.

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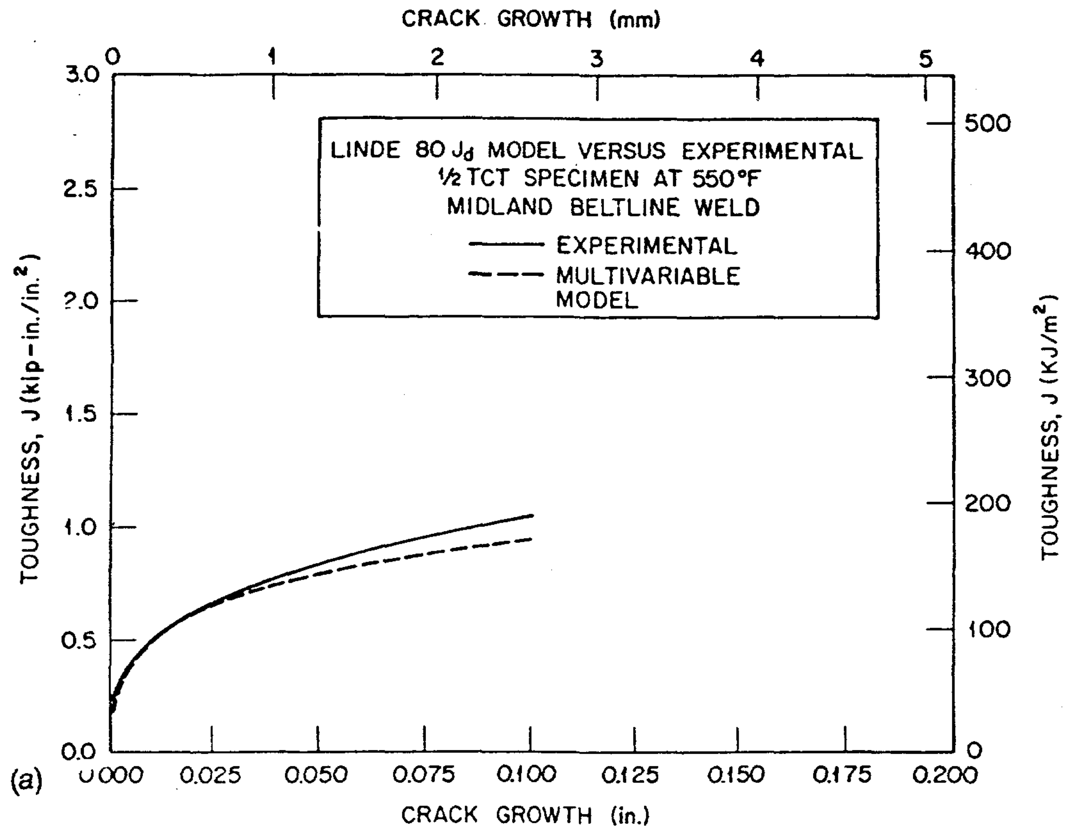

(a)

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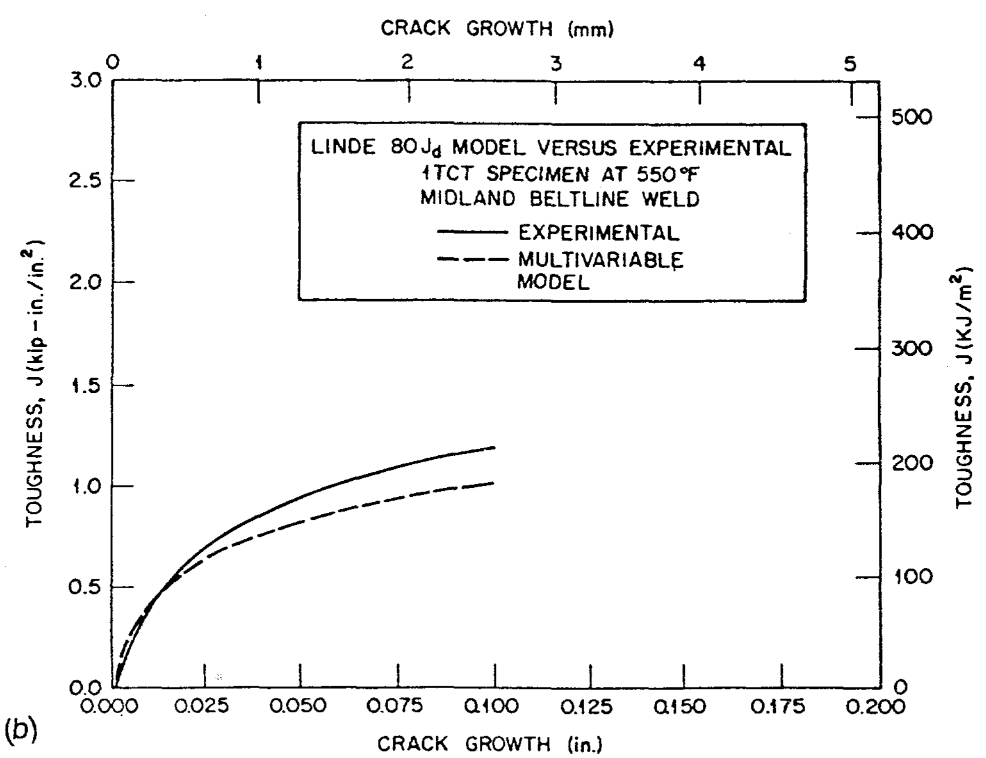

(b)

Figure 4.2 Experimental J-R curve and J-R curve calculated from a multivariable model for *(a)* 1/2T compact specimen and *(b)* 1T compact specimen.

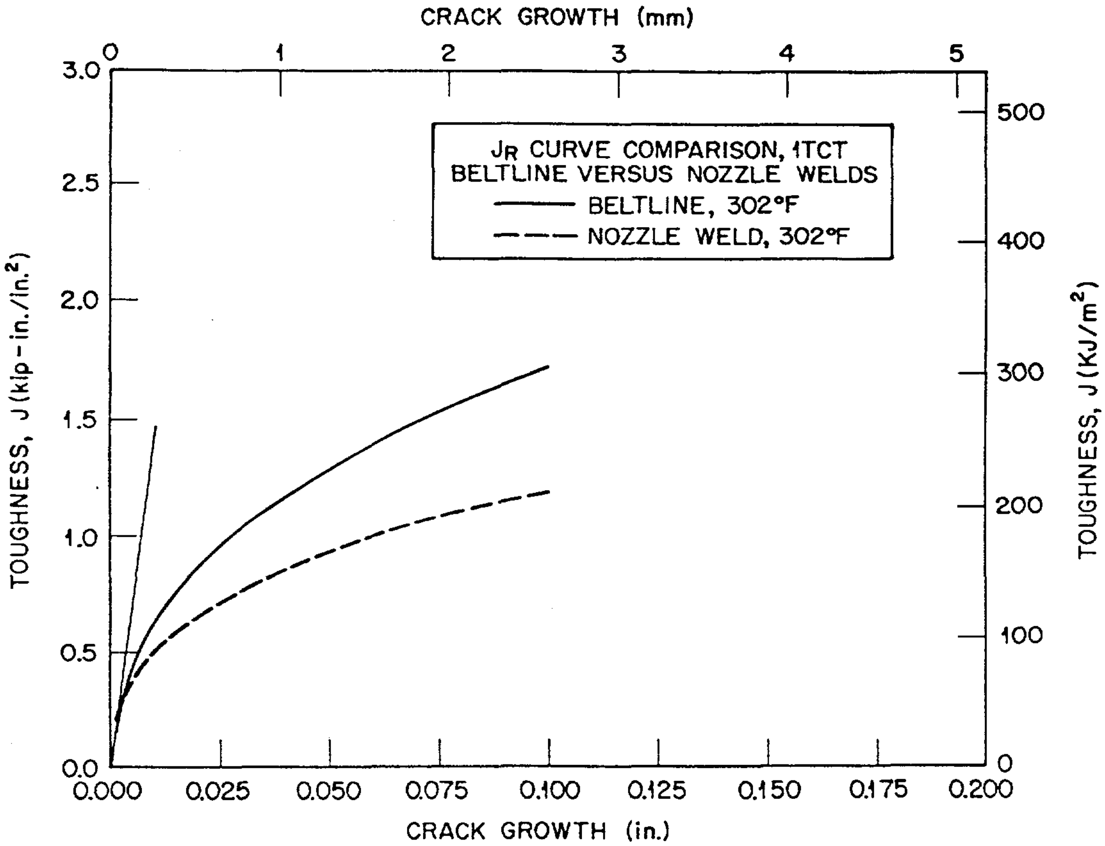

Figure 4.3. **J-R curves that compare the nozzle course versus the beltline weld metal ductile tearing resistance at 302° C.**

4.3 Status of Irradiation Capsules

Two large capsules, designated 10.05 and 10.06, contain the majority of irradiation exposure specimens in the Tenth Irradiation Series. The transfer and disassembly work on capsule 10.05 has been completed. Capsule 10.06 has been irradiated, and it is now being held in the reactor pool for cooldown before transport to ORNL.

4.4 Status of Testing Irradiated Specimens

Two small capsules, designated 10.01 and 10.02, that contain beltline and nozzle course weld specimens, respectively, have been disassembled and testing has begun. Both were irradiated by Materials Engineering Associates (MEA) in the University of Buffalo Reactor to 0.5×10^{19} neutrons/cm^2 at 288°C (550°F). The CVN 41-J transition temperature shifts were 43.3 and 65.8°C for the beltline and nozzle course welds, respectively. If we assume the average copper and nickel contents of the specimens are 0.26/0.57 and 0.40/0.57 for the beltline and nozzle course welds, respectively, *Regulatory Guide 1.99* (Rev. 2) predicts 41-J shifts of 79°C (142°F) and 102°C (183°F), respectively (calculations do not include the margin terms). Thus, in both cases, the actual shifts are substantially less than the *Regulatory Guide* predictions.

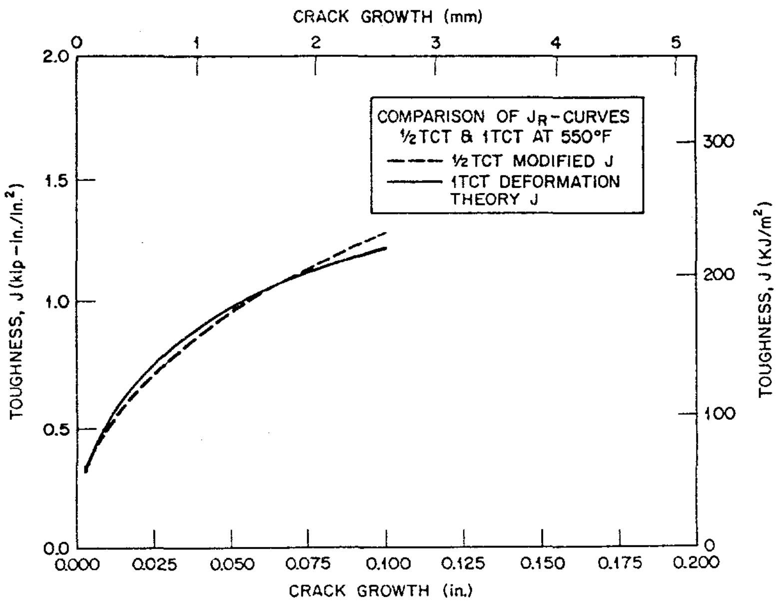

Figure 4.4. J-R curve comparison on beltline weld metal that shows how specimen size effects can be eliminated through the use of modified J.

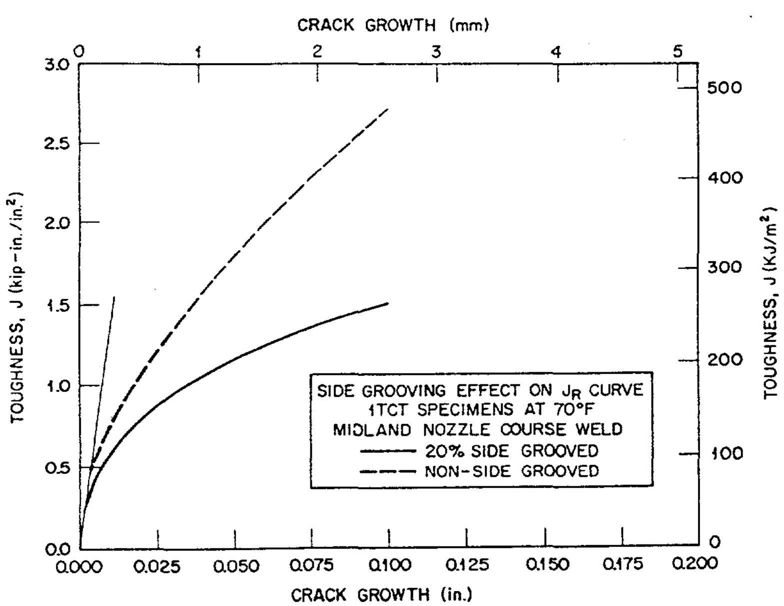

Figure 4.5. *(a)* J-R curves comparing the effect of side grooving on the nozzle course weld metal. Both are legitimate by ASTM standard E 1152-87. *(b)* Evaluation of the side-groove effect by the multivariable model showing the danger of misuse outside the range of data fit.

5. Irradiation Effects on Weld Heat-Affected Zone and Plate Materials (Series 11)

The purpose of this task is to examine the effects of neutron irradiation on the fracture toughness (ductile and brittle) of HAZ of welds and of A 302 grade B (A302B) plate materials typical of those used in fabricating older RPVs. The initial plate material of emphasis will be A302B steel, not the A302B modified with added nickel. This decision was made by NRC following a survey of the materials of construction for RPVs in operating U.S. nuclear plants. Reference 1 was used for the preliminary survey, and the information from that report was revised by NRC staff based on information contained in the licensee responses to Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)." The resulting survey showed a total of eight RPVs with A302B, ten with A302B (modified), and one with A302 grade A plate. Table 5.1 provides a summary of that survey. The numbers for the A 533 and A 508 steels are from ref. 1. During this reporting period, initial inquiries were made regarding procurement of A302B plate in sufficient quantities for the irradiation program. For the HAZ portion of the program, the intent is to examine HAZ material in A302B (i.e., with low nickel content), and in A302B (modified) or A533B-1 (i.e., with medium nickel content).

Reference

1. American Society of Mechanical Engineers Section XI Task Group on Reactor Vessel Integrity Requirements, *White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions*, EPRI TR-100251, Electric Power Research Institute, Palo Alto, California, January 1993.

Table 5.1. Number of U.S. nuclear reactor vessels with specific shell material

Vessel supplier	Vessel shell material[a]					
	A302B	A302B(Mod)	A302A	A533B-1	A508-2	A508-3
ABB Combustion Engineering (PWR)		1		14		
		Palisades (12-71)				
Westinghouse (PWR)	2	2	1	30	15	4
	Haddam Neck (1-68) Point Beach 1 (12-70)	Indian Point 2 (8-74) Indian Point 3 (8-76)	H.B. Robinson 2 (3-71)			
Babcock & Wilcox (PWR)	1	1		3	5[d]	
	Oconee 1 (7-73)[b]	Three-Mile Island 1(9-74)[c]				
General Electric (BWR)	5	6		25		
	Oyster Creek 1 (12-69) Dresden 2 (6-70) Millstone 1 (3-71) Browns Ferry 1 (8-74) Browns Ferry 2 (3-75)	Nine Mile Point 1 (12-69) Dresden 3 (11-71) Quad Cities 1 (2-73) Quad Cities 2 (3-73) Peach Bottom 2 (7-74) Peach Bottom 3 (12-74)				
Total	8	10	1	72	20	4

[a]Plant and date of commercial operation.

[b]Lower nozzle belt forging of A508-2.

[c]A302B Code Case 1339, with lower nozzle belt forging of A508-2.

[d]Includes Oconee 1 and Three-Mile Island 1.

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

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).

6.1 Effect of Annealing Irradiated Materials (M. A. Sokolov, R. L. Swain, and J. J. Henry)

Two sets of 1T compact specimens of irradiated HSST Plate 02 (Fourth Irradiation Series) were annealed at 343 and 454°C for 168 h, respectively. Fracture toughness testing of these specimens will be completed later. Automated ball-indentation (ABI) tests[1] were performed on HSST Plate 02 in the unirradiated, irradiated, and IAR conditions. The load and penetration depth of the indentor are recorded during these tests and are used to derive yield strength and true stress/true plastic strain curves. Up to five indentations were performed on each specimen in each condition. The average value of yield strength derived from indentation tests of an unirradiated specimen was 476 MPa, which corresponds well with the yield strength of 467 MPa reported in the Fourth HSSI Irradiation Series.[2] The yield strength derived from the indentation tests of the irradiated specimens is 568 MPa. Annealing at 343°C resulted in a slight recovery of the ABI yield strength (551 MPa), while annealing at 454°C indicated a significant recovery in the ABI yield strength (498 MPa). Figure 6.1 shows the average true stress/true plastic strain diagrams of Plate 02 in the conditions studied. Annealing at 454°C also completely recovered the plasticity of HSST Plate 02 as measured by the ABI-derived true stress/true plastic strain behavior.

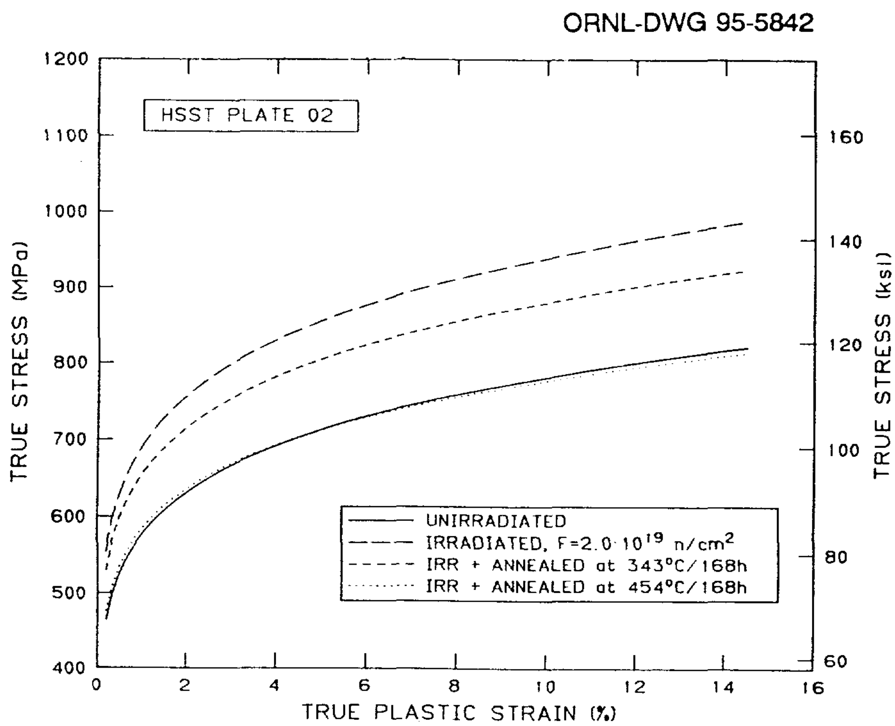

Figure 6.1. True stress-true plastic strain diagrams derived from automated ball-indentation tests of HSST Plate 02 in the unirradiated, irradiated, and irradiated/annealed conditions.

A study of the effect of thermal annealing on the recovery of fracture toughness of LUS submerged-arc welds was continued on specimens from the Second and Third Irradiation Series. Compact specimens, 0.5T thick, were used for fracture toughness testing in the transition region for characterization of irradiated LUS welds 63W, 64W, and 65W after annealing at 454°C for 168 h. All the compact specimens, none of which were side-grooved, were tested using the single-specimen unloading compliance procedure. The purpose of this characterization was the establishment of a data base of fracture toughness properties for these high-copper LUS welds in the annealed condition. These data will be used to evaluate the fracture toughness behavior after reirradiation of the annealed welds.

An analysis procedure based on earlier work by Wallin[3] and in the process of development into an ASTM standard by McCabe et al.[4] was used in this study. The procedure is based on the determination of the following transition temperature curve (or master curve) for 1T size compact specimens:

$$K_{Jc(med)} = 30 + 70 \exp\left[0.019(T - T_o)\right] , \tag{6.1}$$

where T is test temperature, and T_o is the temperature at the 100-MPa√m fracture toughness level. This method allows the development of a full transition curve based on testing six small [for example, 0.5T C(T)] specimens at one temperature. The data created from the 0.5T C(T) specimens are normalized to an equivalent 1T specimen size using the following relationship:

$$K_{Jc(1T)} = 20 + \left[K_{Jc(0.5T)} - 20\right] (B_{0.5T}/B_{1T}) , \tag{6.2}$$

where B is the thickness of the specimen size corresponding to its subscript.

Figures 6.2 through 6.4 show the transition temperature curves (solid lines) obtained for the welds studied in the annealed condition. In the cases where only one K_{Jc} result was obtained at a given test temperature, the datum was not used in positioning the master curve. As stated earlier, no fracture toughness tests were conducted in the transition region in either the unirradiated or irradiated conditions. Moreover, no compact specimens are available to perform tests in the unirradiated condition. To estimate positions of master curves for the welds in the unirradiated condition, a correlation proposed by Wallin[5] was used:

$$T_o^{unirr} = TT_{28J} - 18, \ ^\circ C , \tag{6.3}$$

where TT_{28J} is the temperature at CVN energy of 28 J. The standard deviation on this estimate is 15°C so that the temperature at 100 MPa√m can be expected to be within 30°C of the Equation (6.3) estimate. Based on Charpy impact data[6] of the welds, the values of TT_{28J} were determined as –24, –13, and –35°C for welds 63W, 64W, and 65W, respectively. Estimated master curves of the welds in the unirradiated condition were drawn by dashed lines in Figures 6.2 through 6.4. Finally, shifts of Charpy transition temperature at 41 J due to irradiation were used to estimate positions of the master curves for the welds in the irradiated condition. The Charpy curve shifts, adjusted to the irradiation temperatures of the tested compact specimens, were equal to 105, 88, and 59°C for welds 63W, 64W, and 65W, respectively.[6] It was found in ref. 6 that, for the Charpy impact data, the effect of irradiation temperature on the transition temperature shift is about –0.5°C/°C, meaning that a 1°C increase in irradiation temperature will cause a 0.5°C decrease in the transition temperature shift.

Taking into account the above considerations, the positions of the master curves on the temperature axis imply that the annealing of the welds at 454°C for 168 h has probably resulted in nearly full recovery of fracture toughness in the transition range.

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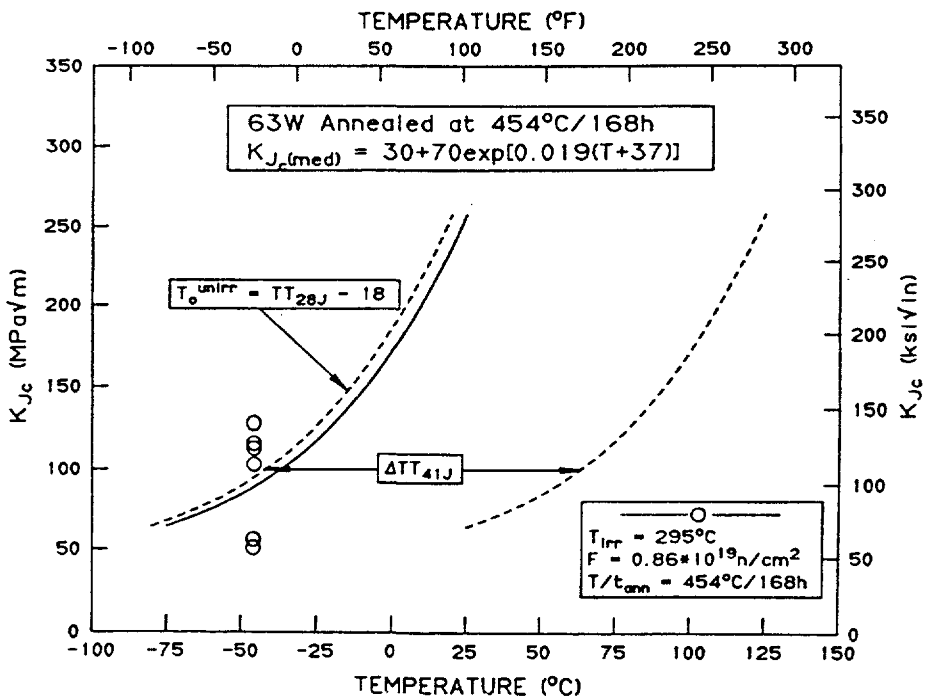

Figure 6.2. Master curve of weld 63W.

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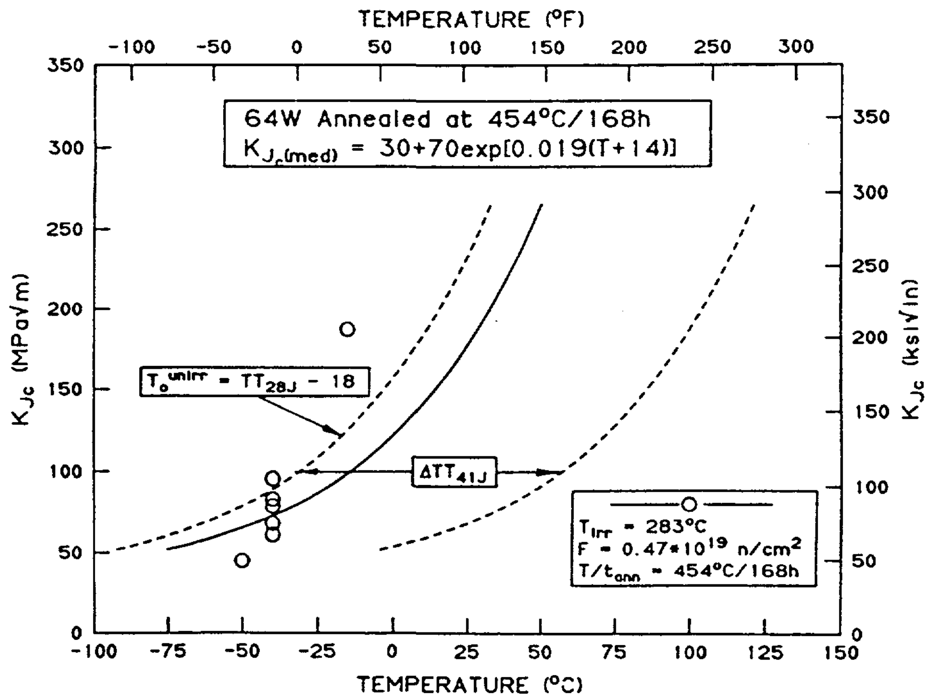

Figure 6.3. Master curve of weld 64W.

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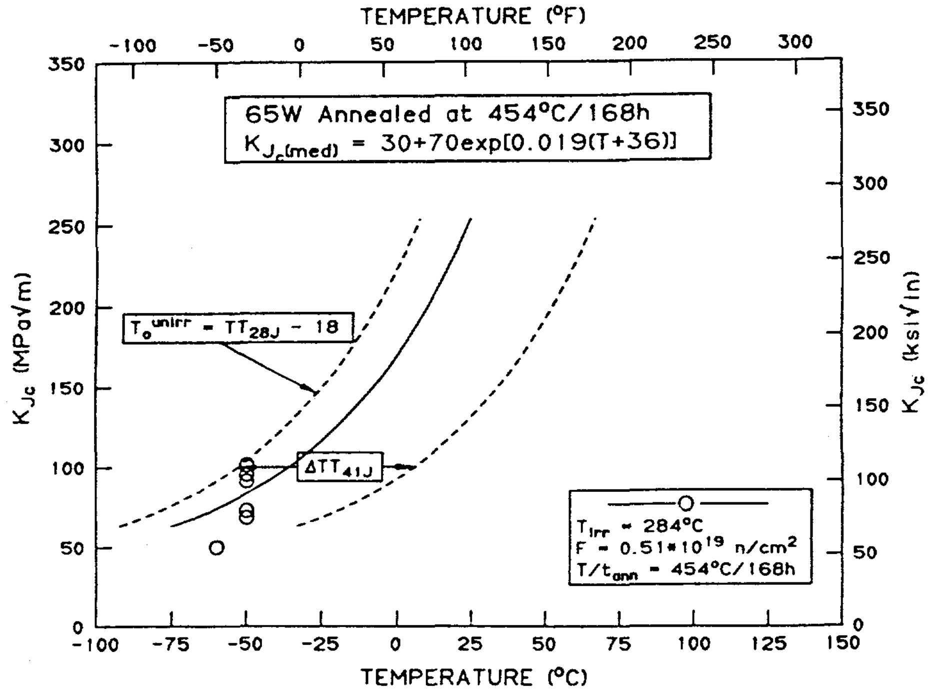

Figure 6.4. Master curve of weld 65W.

6.2 Design, Fabrication, and Installation of New Irradiation Facilities (I. I. Siman-Tov, D. W. Heatherly, G. E. Giles, Jr., and D. W. Sparks)

The major design work of the IAR facilities was completed, but, as the designs are reviewed, changes become necessary in all the interconnected parts. One change made was to insert extra shielding to protect the O-rings from radiation damage. An analysis to determine the shielding required to protect the O-rings that seal the facility and the capsules from the pool water was completed. The criterion was to stay below the activity level recommended by the manufacturer that causes deterioration of the O-rings. A 267-mm-long (10.5-in.) stainless steel plug is required for proper shielding. Analyses were also completed to determine the power required by the heaters for the annealing of the specimens at 450°C. A total power of 10 kW will be required for all 12 heater regions. The model showed that temperatures up to 650°C can be attained if desired. Heaters were ordered with a capacity to generate 1 kW per heater region. Since nuclear heating provides some heat during operation, the available heater power will be adequate to control the specimen temperatures during operation. The analysis was checked by G. E. Giles, Jr., of the ORNL Computational Physics and Engineering Division.

An analysis to determine the maximum temperature that the TCs will experience in the region leading from the capsule to the latching region was completed. Temperatures did not exceed the temperatures to be measured in the capsule. This heating effect should not impact the measurement of the specimen temperatures.

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2. J. J. McGowan, R. K. Nanstad, and K. R. Thoms, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Characterization of Irradiated Current-Practice Welds and A 533 Grade B Class 1 Plate for Nuclear Pressure Vessel Service*, USNRC Report NUREG/CR-4880, Vol. 1 (ORNL/TM-6484/V1), July 1988.†

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

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5. K. Wallin, "A Simple Theoretical Charpy V - K_{Ic} Correlation for Irradiation Embrittlement," pp. 93-100 in *Innovative Approaches to Irradiation Damage and Fracture Analysis*, PVP-Vol. 170, American Society of Mechanical Engineers, New York, 1989.*

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

*Available in public technical libraries.

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

7. Microstructural Analysis of Radiation Effects

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

7.1 Microstructural Modeling and Data Analysis

The microstructural modeling and experimental studies carried out under this task have raised a number of questions. Two of these have been further addressed by analyzing data from the Power Reactor Embrittlement Data Base (PR-EDB).[1] The first issue is related to the resolution of the High Flux Isotope Reactor (HFIR) embrittlement reported previously,[2] and the second is related to the fact that the copper dependence of embrittlement predicted by current models is more severe than that seen in the data.[3]

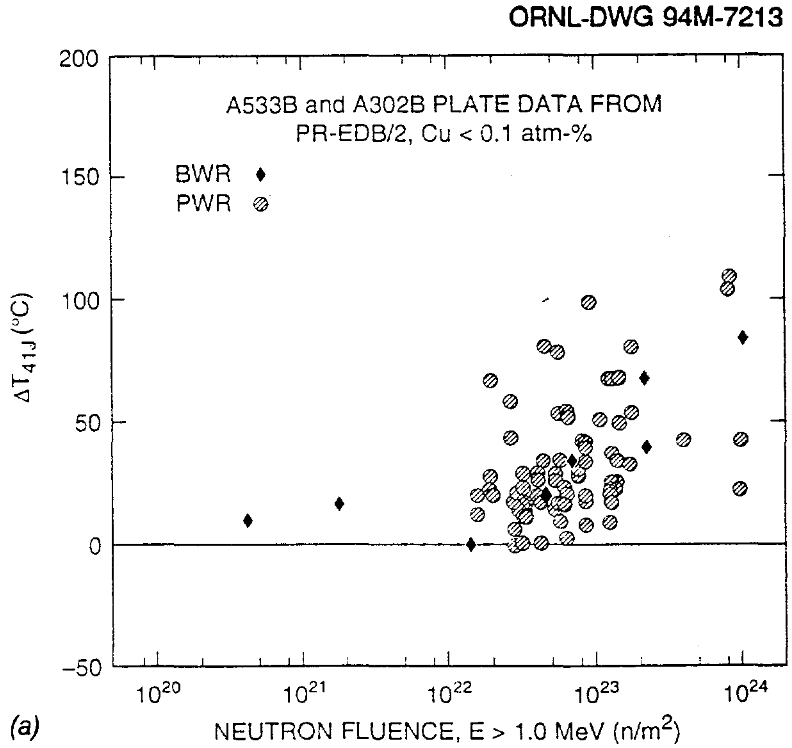
The HFIR embrittlement has been explained on the basis of uncounted atomic displacements that arose from high-energy gamma-rays.[2] These displacements had been neglected because they are usually insignificant compared to those generated by fast neutrons. The HFIR is unique in that it has a relatively long water path between the core and the pressure vessel. Since fast neutrons are rapidly thermalized in water while the high-energy gammas are transported with little attenuation, this leads to a very high ratio of fast gammas to fast neutron fluxes. Although the analogous water path is much shorter in commercial reactors, it is somewhat longer in boiling-water reactors (BWRs) than in pressurized-water reactors (PWRs). Thus, it seemed prudent to examine the PR-EDB to search for any systematic differences between surveillance data from the two reactor types arising from potential spectral differences.

A comparison of plate and weld material data from the PR-EDB for alloys A533B and A302B is shown in Figures 7.1 and 7.2. The high and low copper data are shown separately for both types of material. These figures demonstrate that no systematic differences can be observed. This indicates that any effect of the longer water path from the core to the pressure vessel in BWRs, relative to PWRs, is not significant.

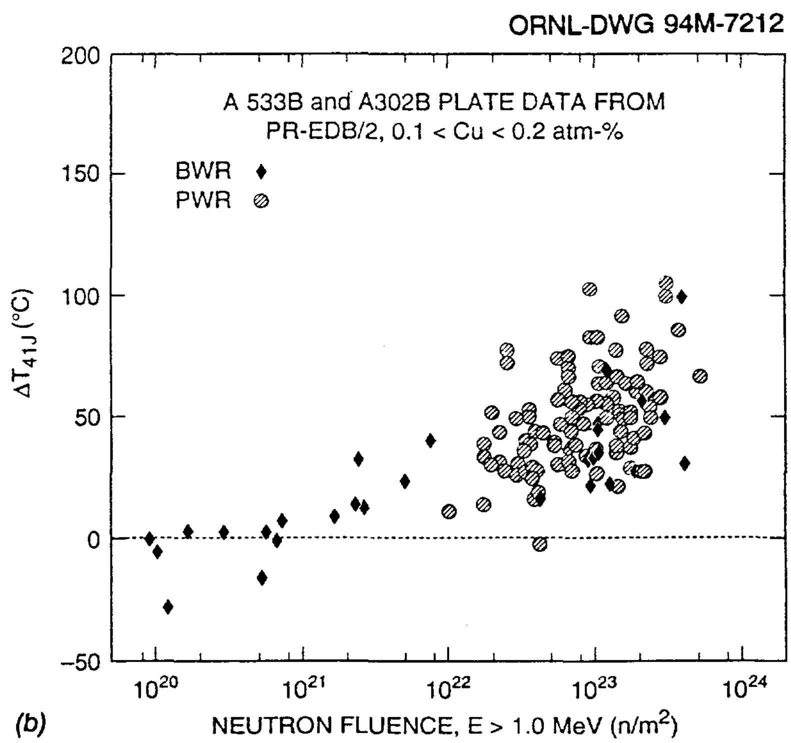
When typical material parameters are used in the microstructural embrittlement model developed in this task, the dependence of the predicted embrittlement on the assumed copper content is much stronger than observed in the surveillance data base. This is illustrated in Figure 7.3 for weld data from the PR-EDB and the model predictions from ref. 3. Notice that the data for copper values between 0.1 and 0.2 wt % largely overlap the data for copper contents above 0.3. This is consistent with arguments that have been advanced which maintain that the nominal bulk copper content of the material is not necessarily the appropriate value for predicting sensitivity to embrittlement.[4] If the copper content exceeds the copper solubility at the temperature of final stress relief, the excess copper is likely to form coarse precipitates prior to irradiation. In this case, the copper solubility should represent an upper limit to the copper available to participate in subsequent precipitation. APFIM measurements indicate that this value is unlikely to exceed 0.22 at. % (~ 0.25 wt %) after aging at 600 to 650°C.[4]

As a crude test of this hypothesis, the Charpy shift data for A533B and A302B plate and weld from the PR-EDB were plotted as a function of copper content. These data are shown in Figures 7.4 and 7.5 for plate and weld, respectively. The data have been broken into two broad fluence ranges in parts *(a)* and *(b)* of both figures. Although this analysis neglects important secondary variables such as nickel content, the data generally support a saturation of Charpy shift above 0.2 to 0.25 wt % copper.

Further analysis of the molecular dynamics (MD) cascade simulations has led to the development of final defect survival efficiency curves as a function of temperature and primary knock-on atom (PKA) energy. Analytical equations have been developed to fit the energy dependence of the point defect survival and clustering fractions obtained from the MD simulations. Iron PKA energy spectra are being obtained for neutron spectra typical of PWR and BWR RPV locations and several relevant test reactors. This information will be folded together with the MD survival curves to obtain the relevant spectrally averaged quantities needed in the kinetic embrittlement models.



(a)



(b)

Figure 7.1. Fluence dependence of Charpy transition temperature shift in A533B and A302B plate data from the Power Reactor Embrittlement Data Base from pressurized-water reactors and boiling-water reactors for *(a)* low and *(b)* high copper content.

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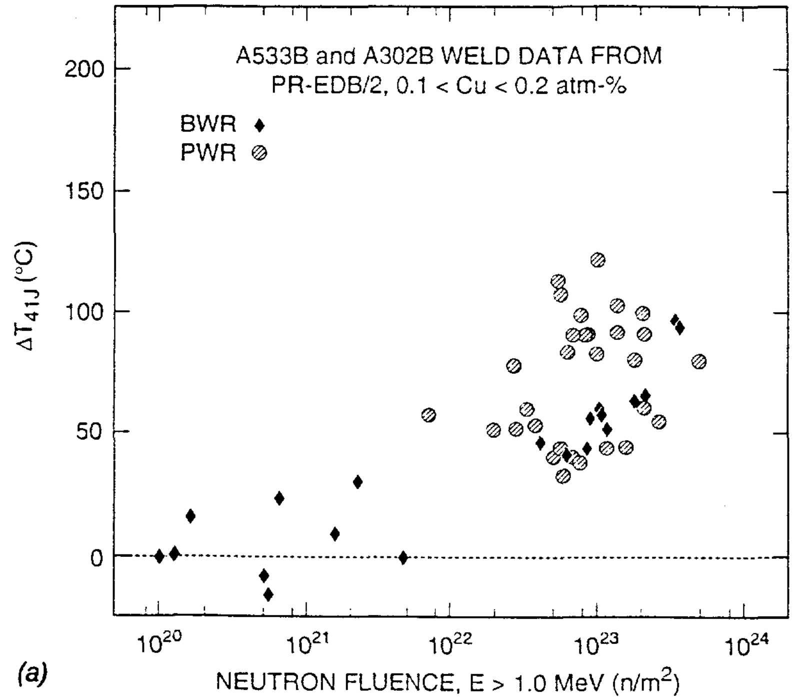

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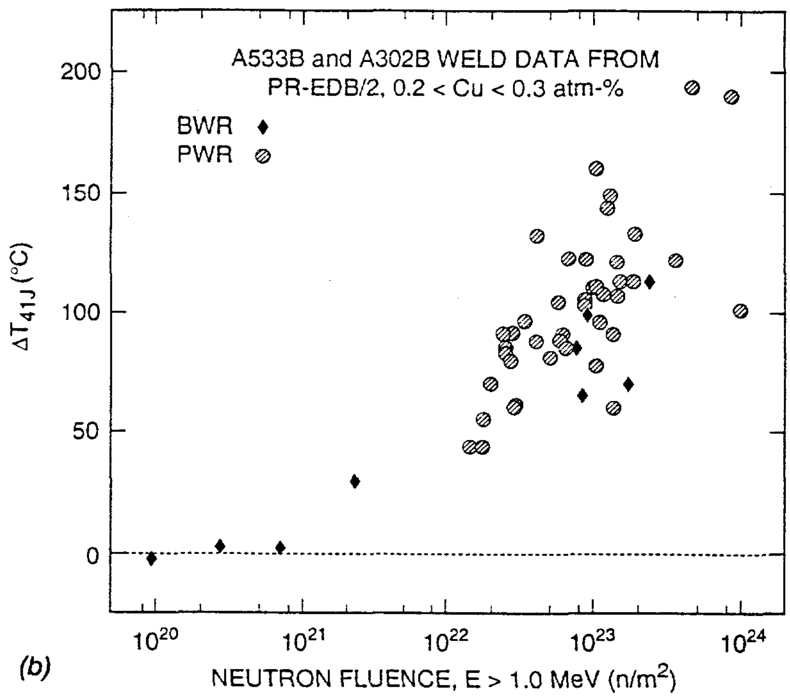

Figure 7.2. Fluence dependence of Charpy transition temperature shift in A533B and A302B weld data from the Power Reactor Embrittlement Data Base from pressurized-water reactors and boiling-water reactors for *(a)* low and *(b)* high copper content.

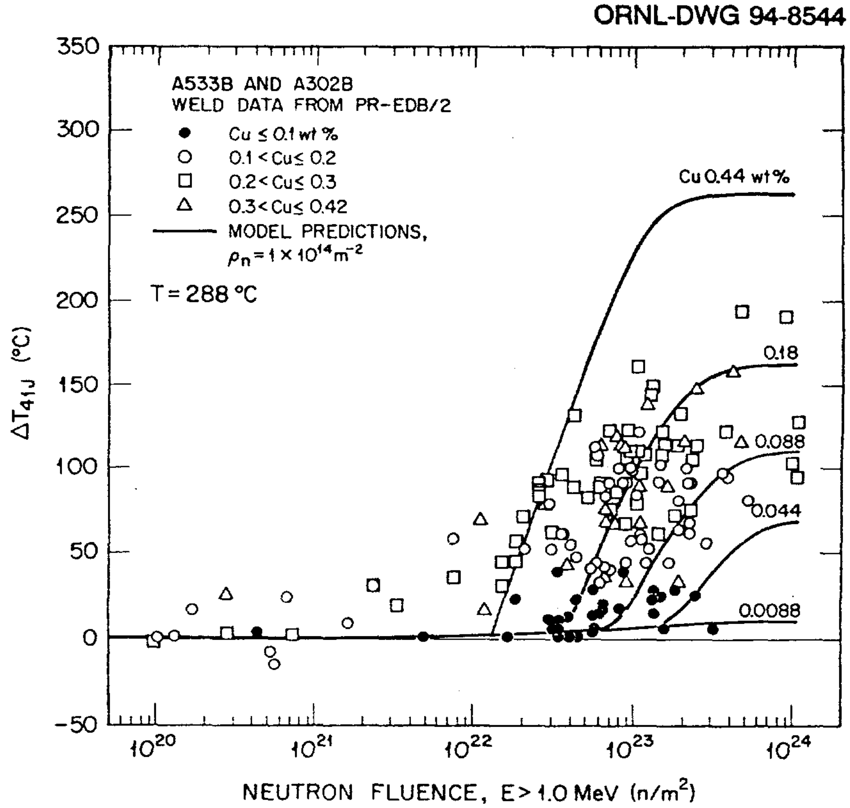

Figure 7.3. **Comparison between model predictions at 288°C and Charpy transition temperature shift data for A533B and A302B weld data from the Power Reactor Embrittlement Data Base. Copper content used in calculations as indicated.**

7.2 Experimental Investigations

Thermal and mechanical design of a new set of small irradiation capsules intended for use in the HFIR was completed, and engineering drawings have been prepared. These capsules will permit irradiation in the HFIR hydraulic tube at 288°C and will be used in IAR studies. The capsules will also be used to provide the highest dose rate data for a set of irradiations being planned in collaboration with the University of California, Santa Barbara (UCSB), to investigate displacement rate (flux) effects at 288°C. The high gamma heating rates in this position required the use of very small gas gaps that varied circumferentially to maintain eight SS-3 tensile samples within 5°C of the target temperature. Since it is not possible to place thermocouples in these capsules, the temperatures will be verified in initial runs using silicon-carbide thermal monitors. Measurement of the SiC lattice parameter change during post-irradiation annealing has been shown to provide a reliable measure of the irradiation temperature.

An initial comparison has been completed between the mechanical properties and microstructures in the ion-irradiated model alloys. Both nanoindentation hardness measurements and examination by TEM indicate that copper content has a strong effect on the size and density of point defect clusters that form under irradiation. In pure iron, little hardening is observed below 1 displacement per atom (dpa), whereas in the Fe-0.5Cu alloy, significant hardening is observed at 0.01 dpa. The formation of visible point defect clusters is consistent with the measured hardening. The measured hardness and the observed size distribution of point defect clusters were used to calculate the dislocation barrier strength for the defects. For doses up to 0.2 dpa, hardening appears to increase almost linearly with copper content up to 0.9 wt %. Little dose dependence is observed between 0.02 and 0.2 dpa, indicating a near saturation of the hardening associated

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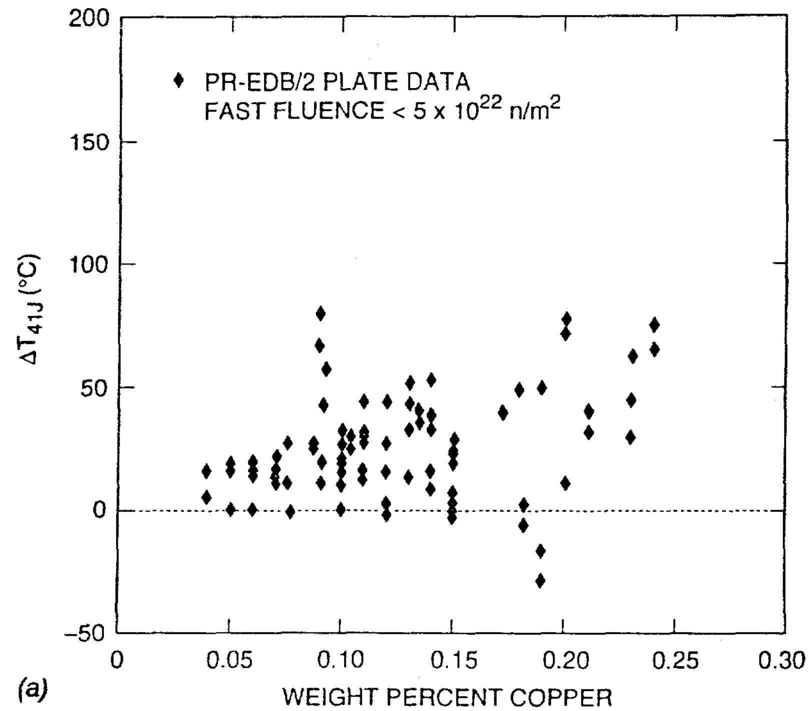

(a)

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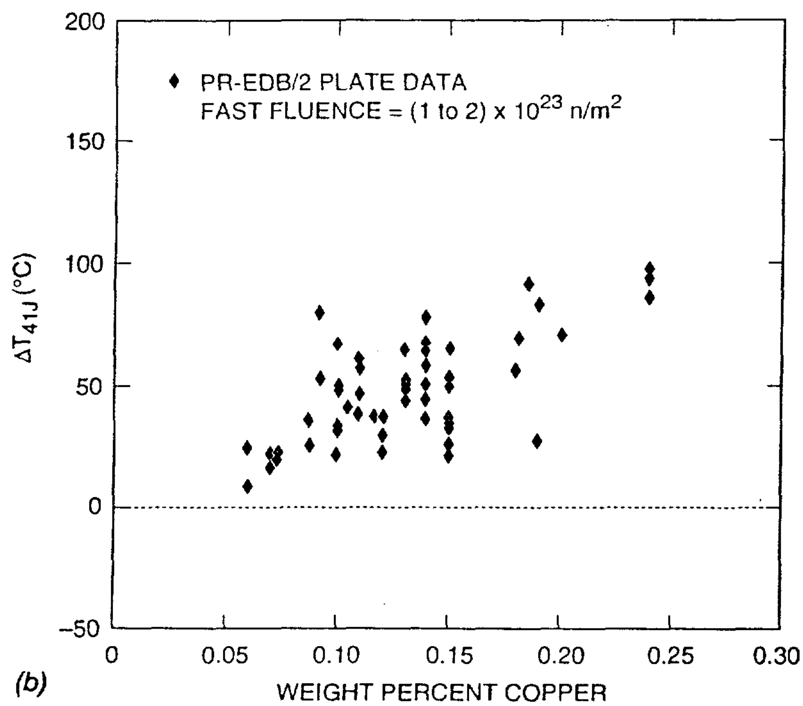

(b)

Figure 7.4. Dependence of Charpy transition temperature shift on copper content at low *(a)* and high *(b)* fluence for plate material from the Power Reactor Embrittlement Data Base.

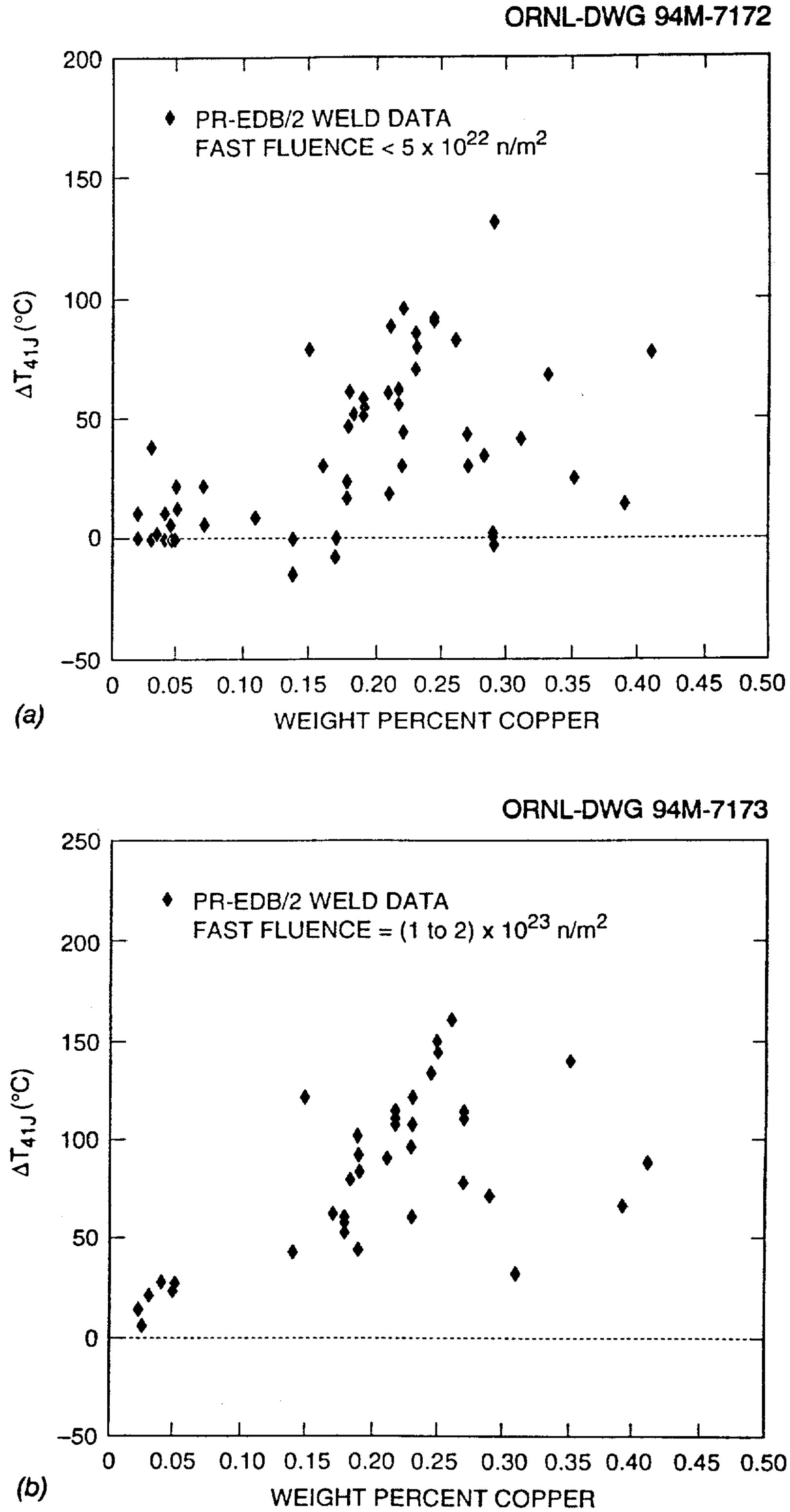

Figure 7.5. Dependence of Charpy transition temperature shift on copper content at low *(a)* and high *(b)* fluence for weld material from the Power Reactor Embrittlement Data Base.

with the point defect clusters. The data obtained at 2.0 dpa were inconsistent with the lower two doses, i.e., hardening appeared to decrease with increasing copper content. This behavior was associated with unusual hardness traces as a function of depth from the irradiated surface. The high-dose experiments are being repeated to verify the observations.

The USE of weld 73W and a Russian weld was observed to increase during thermal annealing at 460°C, in spite of the fact that the welds had been post-weld heat treated at 620 to 680°C. The fracture surfaces of Charpy specimens did not reveal any significant differences between the as-received and as-annealed material. TEM specimens were prepared from broken Charpy bars of both materials to determine if the upper shelf increase is associated with any microstructural change. TEM examination of the specimens was carried out, and the initial analysis of the precipitate and subgrain structure did not reveal any differences between the as-received and the aged specimens at high magnification.

An APFIM characterization has been completed on a Russian Cr-Ni-Mo-V pressure-vessel steel, designated as weld 28. This material had been irradiated in the Rovno Unit 1 reactor to a fluence of 1×10^{23} n/m^2. This is one of the steels that was examined by UCSB using small-angle neutron scattering. Both the irradiated material and unirradiated control specimens have been examined. The results on the irradiated material have shown that the lath boundaries are decorated with the same ultrafine precipitates previously observed in other, unirradiated VVER 440 and 1000 steels. The unirradiated material exhibited a fairly high degree of grain boundary coverage by phosphorus of ~ 13%. Fracture in this material remained transgranular down to about -190°C. Grain boundary coverage by phosphorus in the irradiated material increased to an average of 24%.

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4. M. K. Miller and M. G. Burke, *J. Nucl. Mater.* **195**, 68-82 (1992).†

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

†Available in public libraries.

8. In-Service Irradiated and Aged Material Evaluations

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

The computer numerically controlled (CNC) machining center (model VMC-100) was delivered to ORNL. It was temporarily placed in a cold laboratory (Building 4508) for initial verification, modification for remote operation inside a hot cell, and for operator training. An instructor from EMCO MAIER, Inc. (the vendor of the CNC machine), from Columbus, Ohio, came to ORNL in September 1994 and trained four employees. An action plan was prepared to complete installation of the CNC machine inside hot cell No. 6 of Building 3025E.

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

Full-size Charpy and subsize tensile specimens have been fabricated from the type 308 stainless steel weldments that were aged at 343°C for 20,000 and 50,000 h. The Charpy specimens have been tested. The results show that embrittlement continues with increased aging, with the 50,000-h specimens having higher transition temperatures and lower USE levels than the previous results for 20,000-h aging. Tensile tests and fractographic examination will be conducted to provide additional information.

9. JPDR Vessel Steel Examination

W. R. Corwin, B. L. Broadhead, and M. A. Sokolov

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

Even though an informal final agreement was reached some time ago with the Japan Atomic Energy Research Institute (JAERI) on the details of collaboration for research on the material from the vessel of the JPDR, there was very slow movement toward the production of a signed, formal agreement. However, following several minor revisions to the original document, the agreement was finalized during this reporting period.

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

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

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

During this reporting period, the irradiated material from the wall of the JPDR that will be examined at ORNL was moved into temporary storage at the hot cells where it is to be machined and examined. Also, in preparation for impact testing of the JPDR material, it will be necessary to perform a cross comparison of Charpy impact test machines that will be used in the JPDR studies. The Japanese machines use a tup which meets the specification of the Japan Industry Standards but which is different than the ASTM tup used at ORNL. This introduces the possibility of different impact results on the same material which has the likelihood of increasing at increasing energy levels. A test matrix was developed to assess the extent of the difference.

Detailed plans were developed and work was initiated to evaluate the existing neutron dosimetry and transport calculations which have been performed for the JPDR by the Japanese as well as to begin similar transport calculations at ORNL. The ORNL transport calculations will include the concrete in the biological shield as well as the pressure vessel. This will allow participation in the International Atomic Energy Agency exercise on concrete activation for decommissioning, as well as the original program aims to determine flux and fluence levels within the vessel itself.

10. Fracture Toughness Curve Shift Method

R. K. Nanstad

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

During this reporting period, acquisition of literature related to irradiation effects on fracture toughness of RPV steels was initiated. Also, fracture toughness and Charpy impact data for RPV steels in the unirradiated and irradiated conditions are being acquired and stored in a data base for evaluation and analysis. The data from all the relevant HSSI Programs is currently under evaluation, and the results will be presented in a letter report. R. K. Nanstad held discussions with researchers at AEA-Technology, Harwell, United Kingdom, and with French researchers from CEA, Electricité de France, and Framatome at the Framatome facilities in Paris, France. It is apparent that the subject of irradiation-induced fracture toughness shift is receiving increased attention within the European community. Information on the European Program on Reevaluation of the K_{Ic} Toughness Reference Curve was presented; the progress on that program will be followed, and results, as they become available, will be incorporated into this task.

11. Special Technical Assistance

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

The purpose of this task is to provide technical expertise and assistance in the review of national codes and standards that may be referenced in U.S. NRC regulations or guides related to nuclear reactor components. The specific activities to be performed include: (1) review of new materials and requirements proposed for inclusion into national codes and standards, of ASME code cases, and of potential deficiencies in proposed supporting technology and data; (2) performing detailed planning and initiating testing of low-alloy steam generator vessel materials with low-temperature postweld heat treatments; (3) initiating the evaluation of using small, round notched-bar specimens as a fracture specimen for potential surveillance applications; (4) evaluating the use of precracked Charpy specimens and subsize Charpy specimens in assessing material fracture toughness; (5) participating in the Charpy specimen reconstitution round robin coordinated by the ASTM by performing the testing of unirradiated specimens; and (6) coordinating the identification, inventorying, shipment, and examination and evaluation of current government-furnished materials and equipment from MEA-controlled sites to ORNL.

11.1 Evaluation of Precracked Cylindrical Tensile Specimen

Three slices have been cut from a portion of the 72W weldment to provide material for specimens for SRI. Bars have been fabricated from these slices and shipped to SRI for testing under the subcontract. Weld 72W was also supplied to AEA-Technology, Harwell, United Kingdom, for testing of precracked cylindrical tensile (PCCT) specimens. The SRI specimens will be of larger diameter than the Harwell specimens, which will allow for a comparison of size effects for the PCCT specimens as well as a comparison of results with the compact specimens previously tested in the Fifth Irradiation Series.

11.2 Initial Evaluation of Subsize CVN Testing (M. A. Sokolov and R. K. Nanstad)

The main issue in determining the feasibility of using subsize CVN specimens to determine properties of RPV steels is the correlation of transition temperature and upper-shelf toughness between subsize and full-size specimens. Four types of RPV steels were selected for this study. All of these steels were studied previously at ORNL using standard specimens under different tasks of the HSST and HSSI Programs. The USEs of the full-sized specimens varied from 73 to 330 J, the transition temperatures varied from –46 to 58°C, and the yield strengths varied from 410 to 940 MPa (see Table 11.1). Five designs of subsize specimens were chosen for the present study (see Figure 11.1).

The specimen V-notch dimensions (depth, angle, and notch radius) play important roles in the transition behavior due to their effects on constraint and, thus, the stress concentration developed beneath the notch during deformation. The USE is also affected by the specimen dimensions, in particular the ratio of the notch depth to specimen width (a/W). The sensitivity of USE to the relative notch depth was studied on type 3 specimens of HSST Plate 02 (see Figure 11.2). One set of specimens was made with a notch 1.7 mm (0.065 in.) deep compared with the common 1.0-mm depth. Increasing the depth significantly reduced the USE (from 31 J for a = 1.0 mm to 13 J for a = 1.7 mm).

The effect of a notch radius can be observed by comparing Charpy curves of HSST Plate 02, type 3 specimens with either 0.25- or 0.10-mm radii (see Figure 11.3). The sharpening of the notch leads to an increase of transition temperature by ~ 20°C and a decrease of the USE by ~ 5 J.

Table 11.1. Test matrix and mechanical properties of materials studied

Material	Upper-shelf energy (J)	$DBTT_{41}$[a] (°C)	Yield strength (MPa)	5 x 5 Type 1 U.S./Japan	3.3 x 3.3 Type 2 U.S./Japan	5 x 5 Type 3 Russian	3 x 4 Type 4 German	5 x 5 Type 5 ASTM[b]
A 533 wide plate, LT orientation	330	–43	422	X	X	X	X	-
A 533 wide plate, TL orientation	244	–46	410	X	X	X	X	-
A 508, as-quenched	115	58	634	X	X	X	X	-
A 508, quenched and tempered at 599°C	102	40	697	X	X	X	X	-
A 508, quenched and tempered at 677°C	116	18	605	X	X	X	X	-
A 508, quenched and tempered at 704°C	164	–32	500	X	X	X	X	-
HSST[c] Plate 02, TL orientation	141	0	432	X	X	X	X	X
HSST[c] Plate 014, quenched and tempered at 950°C	73	–34	940	X	X	X	X	-
15Kh2MFA, melt 103672	181	–40	630	X	X	X	X	-
HSSI[d] weld 72W	136	–28	500	X	X	X	X	X

X = 12 to 15 specimens.

[a]Ductile-to-brittle transition temperature.
[b]American Society for Testing and Materials.
[c]Heavy-Section Steel Technology Program.
[d]Heavy-Section Steel Irradiation Program.

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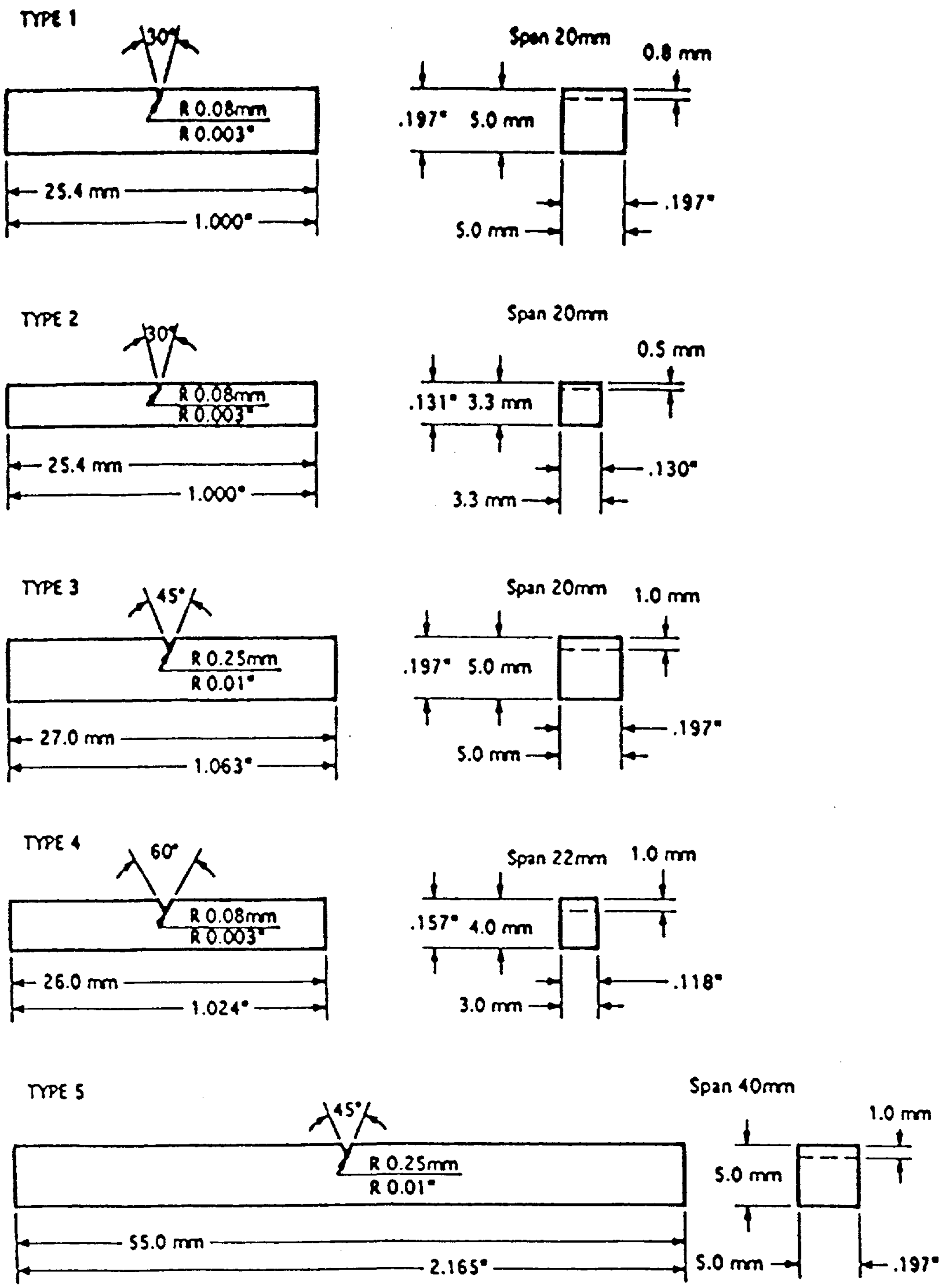

Figure 11.1. Dimensions of subsize specimens studied.

ORNL-DWG 95-5847

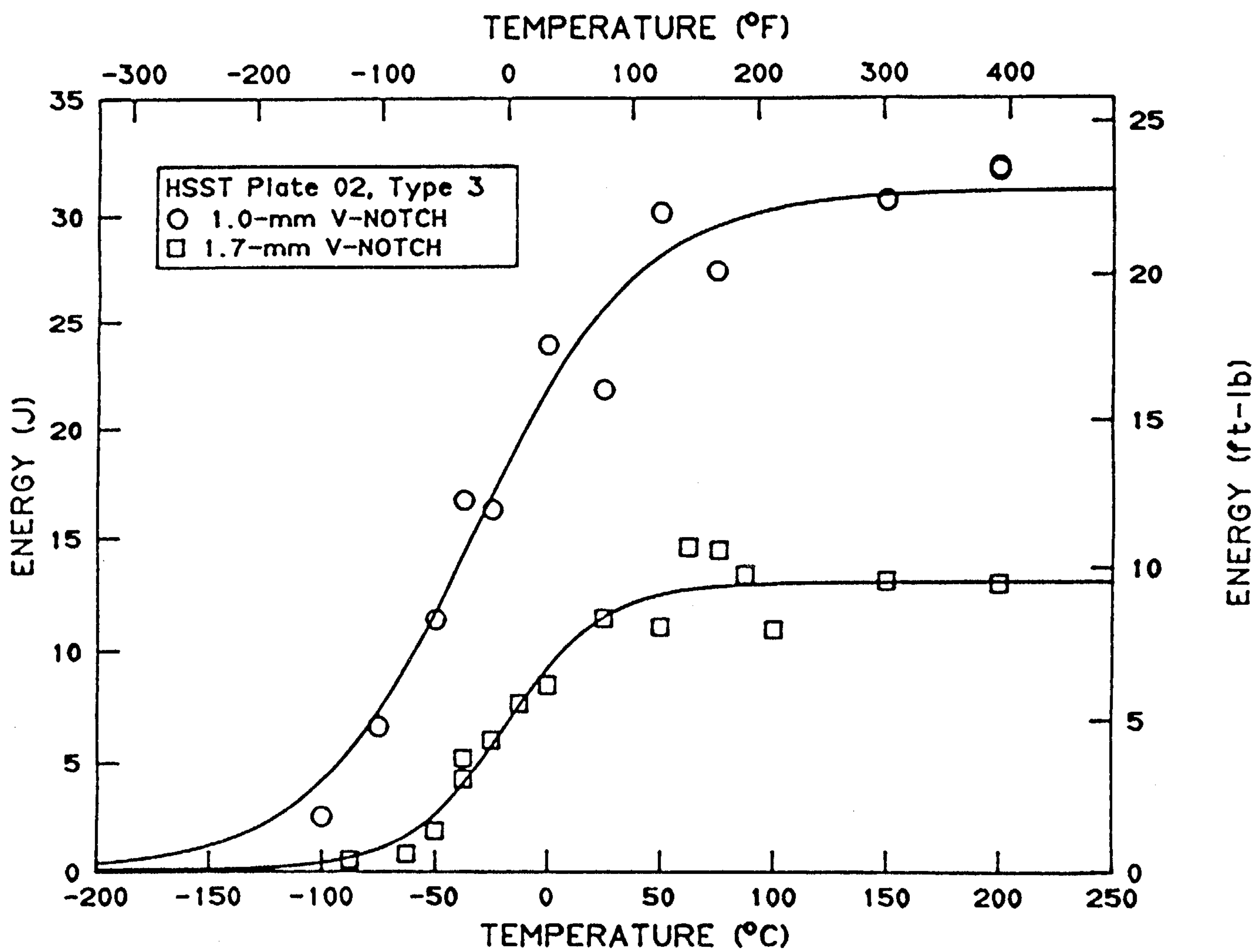

Figure 11.2. Impact curves of type 3 subsize specimens with 1.0- and 1.7-mm V-notch depth.

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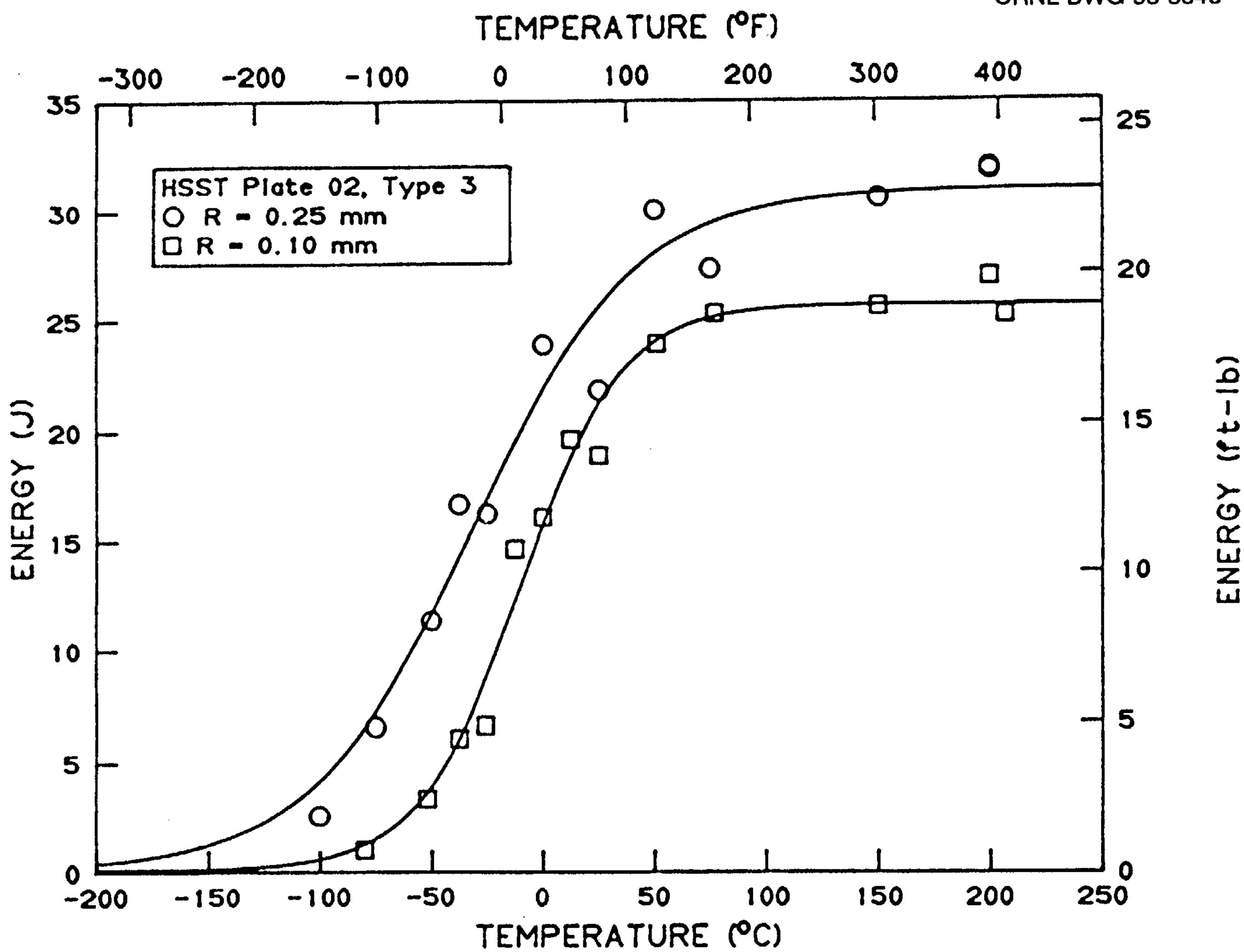

Figure 11.3. Impact curves of type 3 subsize specimens with 0.25- and 0.10-mm V-notch root radius.

Another important parameter is the distance between the anvil supports (span). Figure 11.4 shows energy versus test temperature curves of HSST Plate 02, type 4 specimens tested with a span of 22 and 20 mm. Data from specimens with the shorter span gave a slightly higher energy on the upper shelf, but this difference could be due to the scatter in the data. No difference was observed in the transition region. Another example of the effect of a span on impact properties is given by a comparison of impact curves of types 3 and 5 subsize specimens. The only difference between those specimens is that those of type 3 have one-half the span of type 5 (20 vs 40 mm). The energy versus test temperature curves of types 3 and 5 specimens of weld 72W are presented in Figure 11.5. Figure 11.6 presents the results for type 5 specimens of HSST Plate 02 cut from section "B" of the plate and type 3 specimens that were cut from the broken halves of tested type 5 specimens. In both examples these results do not show any difference between types 3 and 5 specimens.

Five type 4 specimens of weld 72W were tested at an impact velocity of 5.5 m/s while 13 were tested at 2.25 m/s, the typical impact velocity for subsize specimens in the present study. The results show no change in impact properties resulting from the increase of impact velocity from 2.25 to 5.5 m/s (see Figure 11.7).

For all types of subsize specimens, a linear dependence between the USE of full-size and subsize specimens is observed except for two points with USE of full-size specimens higher than 200 J. Material with a USE higher than 200 J for full-size specimens requires special consideration. Observation of the broken halves of specimens tested in the upper-shelf region shows large amounts of plastic deformation at the support points. These features are associated with the specimen "wrapping around" the striking edge and squeezing though the anvils. It is assumed that all interactions between the specimen, striker, and anvils would undoubtedly result in additional energy reflected by the machine dial. Additional investigations need to be performed to analyze these data. For purposes of this study, analyses of USE data were limited to 200 J for full-size specimens. Figure 11.8 presents the correspondence of the USE from full-size specimens to the USE from subsize specimen types 1 through 4 as well as the average ratio of the USE for full-size to the USE for sub-size specimens, which will be used as normalization factors for each specific geometry of subsize specimen.

It was assumed that the correlation between the transition temperatures of full- and subsize specimens could be presented as:

$$DBTT_{full\text{-}size} = DBTT_{subsize} + M \,, \tag{11.1}$$

where $DBTT_{full\text{-}size}$ and $DBTT_{subsize}$ are transition temperatures for full-size and subsize specimens, respectively, and M is a (constant) shift of the *ductile-to-brittle transition temperature (DBTT) due to specimen size*. Different specimen types will have different M values. The equivalent values of absorbed energy for determination of the DBTT of subsize specimens corresponding to 41 J for full-size specimens were determined by dividing 41 J by the normalization factor for each type of subsize specimen, as given in Figure 11.8. Thus, the equivalent levels corresponding to 41 J are 8.0 J for type 1 specimens, 2.2 J for type 2 specimens, 6.6 J for type 3 specimens, and 1.9 J for type 4 specimens. Figures 11.9 through 11.12 summarize the comparison of the DBTT for full-size and various different subsize specimens. The data show a linear dependence of DBTT. Data from the present study correspond very well with published data for the same subsize specimens but different materials.

To establish empirical values of DBTT shift due to specimen size, the results from this study and published data were analyzed together. The following equations were obtained:

$$DBTT_{full\text{-}size} = DBTT_{type\,1} + 26\ (\pm 37),\ ^{\circ}C\ ; \tag{11.2}$$

$$DBTT_{full\text{-}size} = DBTT_{type\,2} + 56\ (\pm 35),\ ^{\circ}C\ ; \tag{11.3}$$

$$DBTT_{full\text{-}size} = DBTT_{type\,3} + 49\ (\pm 44),\ ^{\circ}C\ ; \tag{11.4}$$

$$DBTT_{full\text{-}size} = DBTT_{type\,4} + 65\ (\pm 36),\ ^{\circ}C\ ; \tag{11.5}$$

where figures in parentheses are ±95% confidence intervals.

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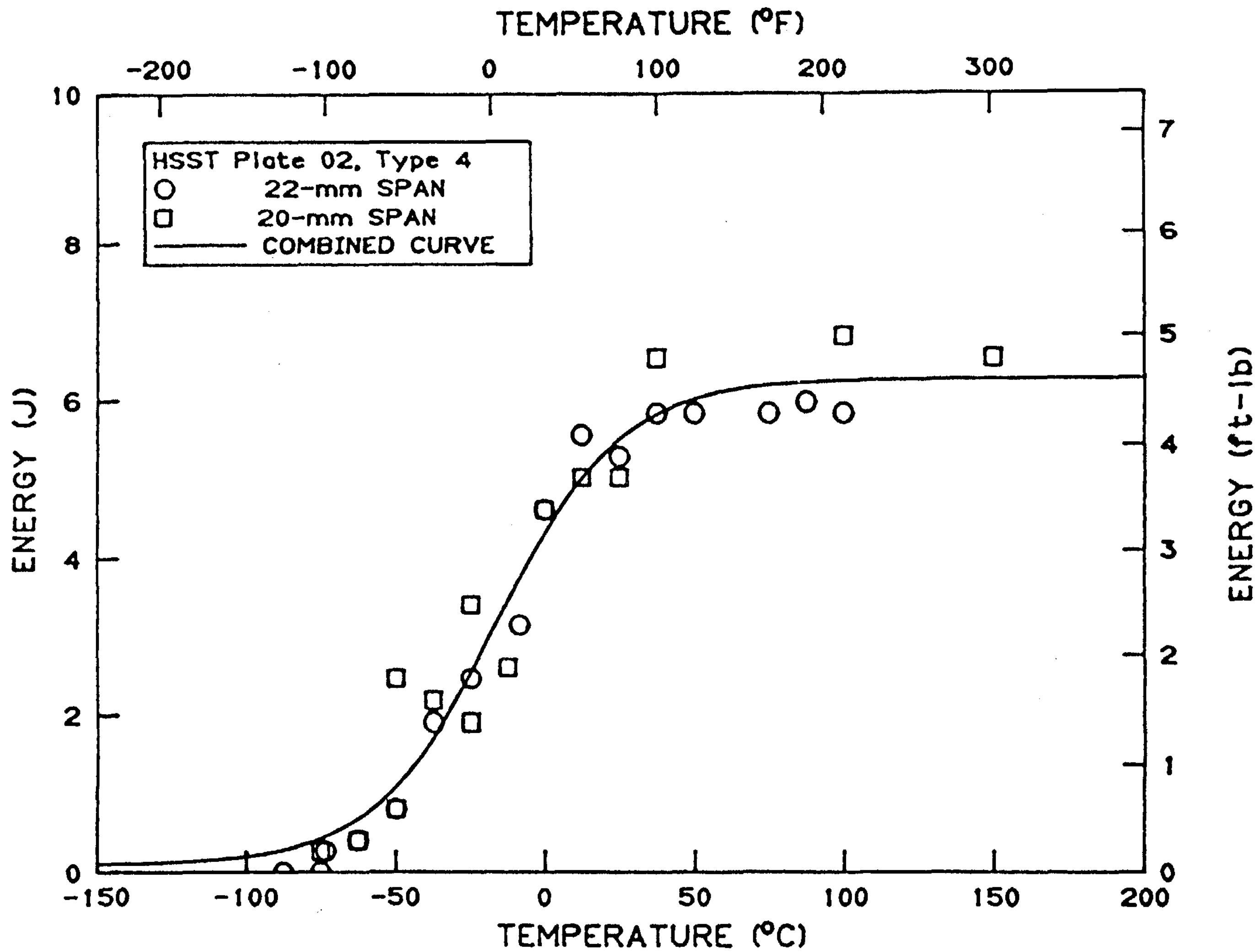

Figure 11.4. Impact curves of type 4 subsize specimens tested at 22- and 20-mm span.

ORNL-DWG 95-5850

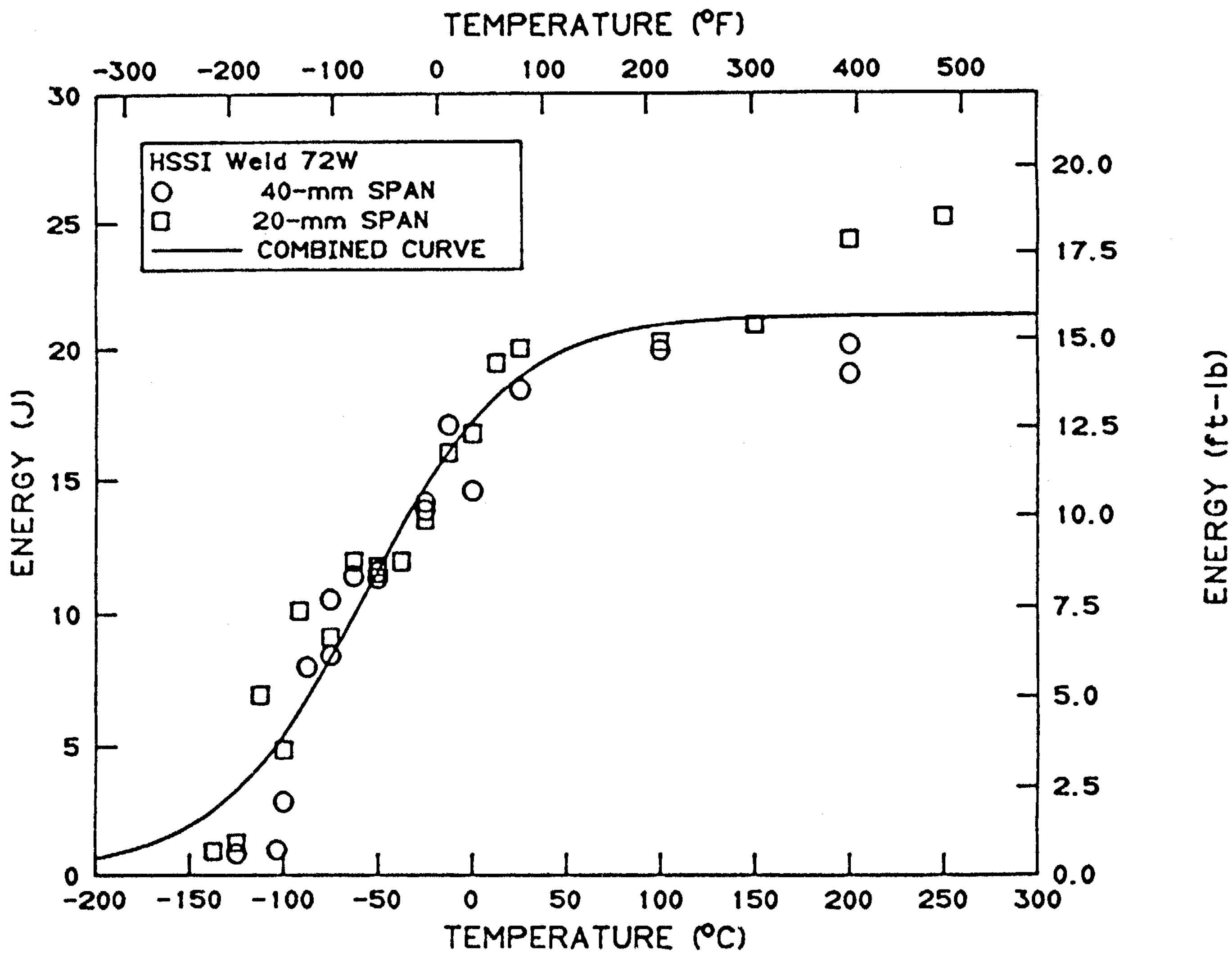

Figure 11.5. Impact curves of 45° angle notch specimens of weld 72W tested at 40- and 20-mm span.

ORNL-DWG 95-5851

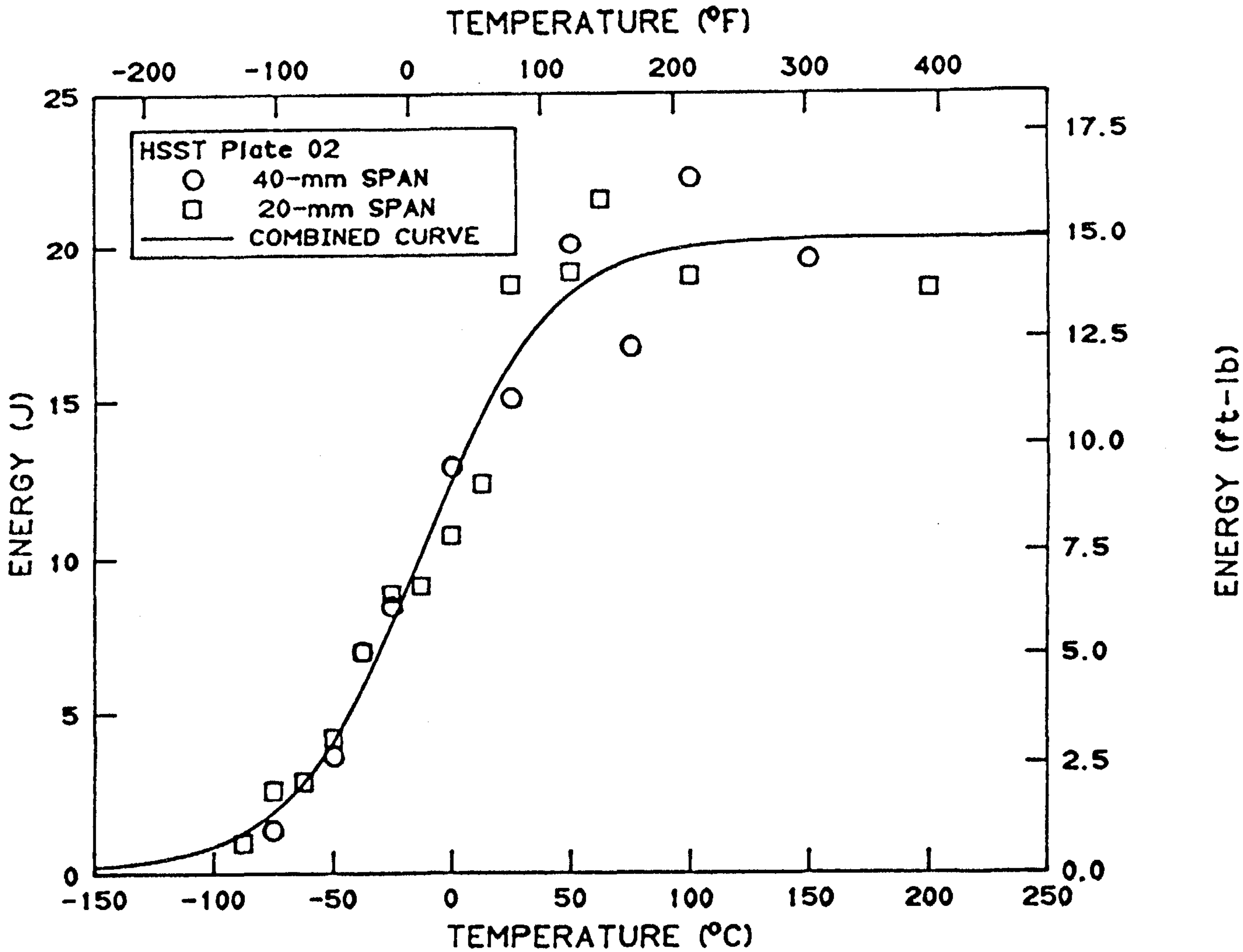

Figure 11.6. Impact curves of 45° angle notch specimens of HSST Plate 02 tested at 40- and 20-mm span.

ORNL-DWG 95-5852

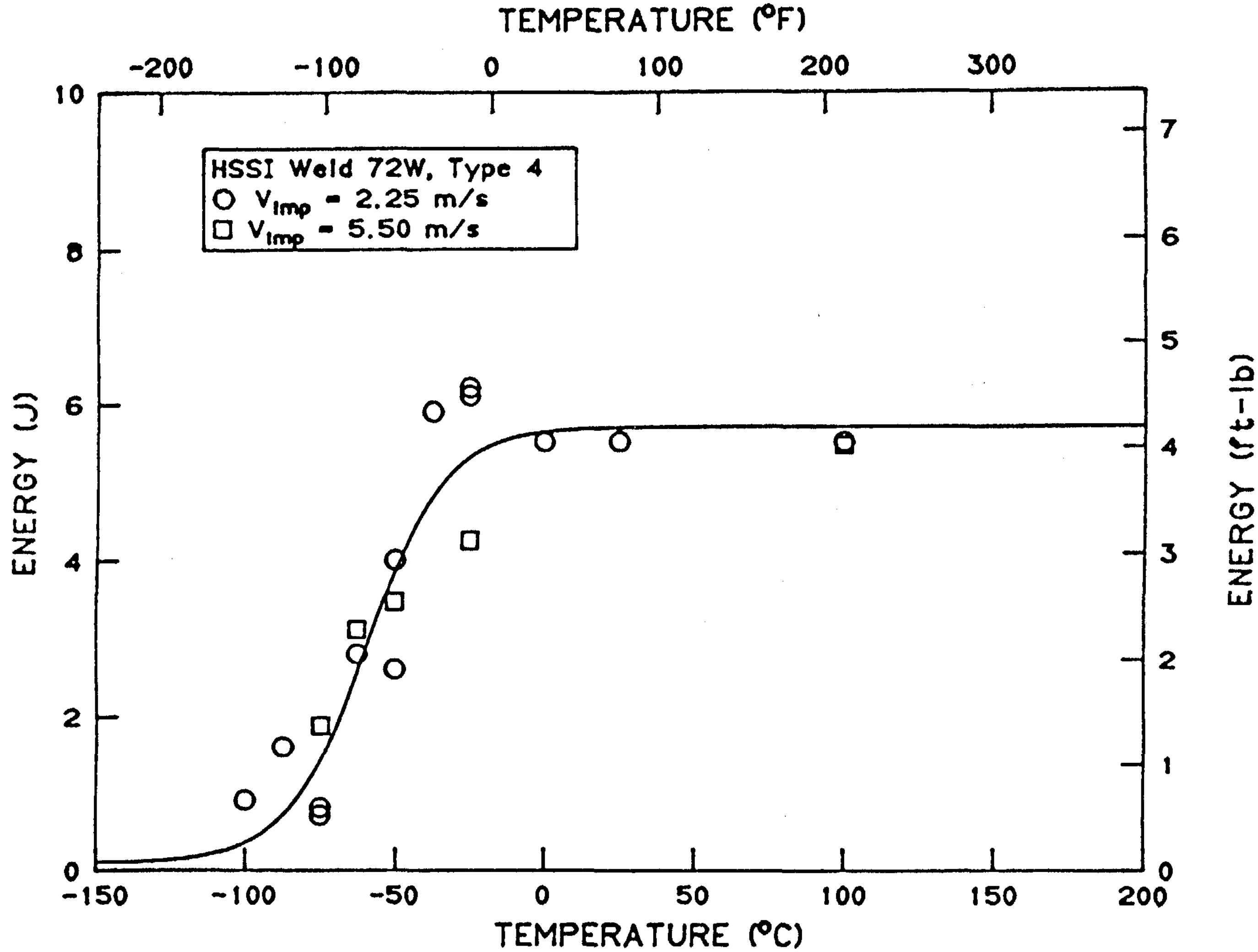

Figure 11.7. Impact curves of type 4 specimens of weld 72W tested at impact velocities of 5.5 and 2.25 m/s.

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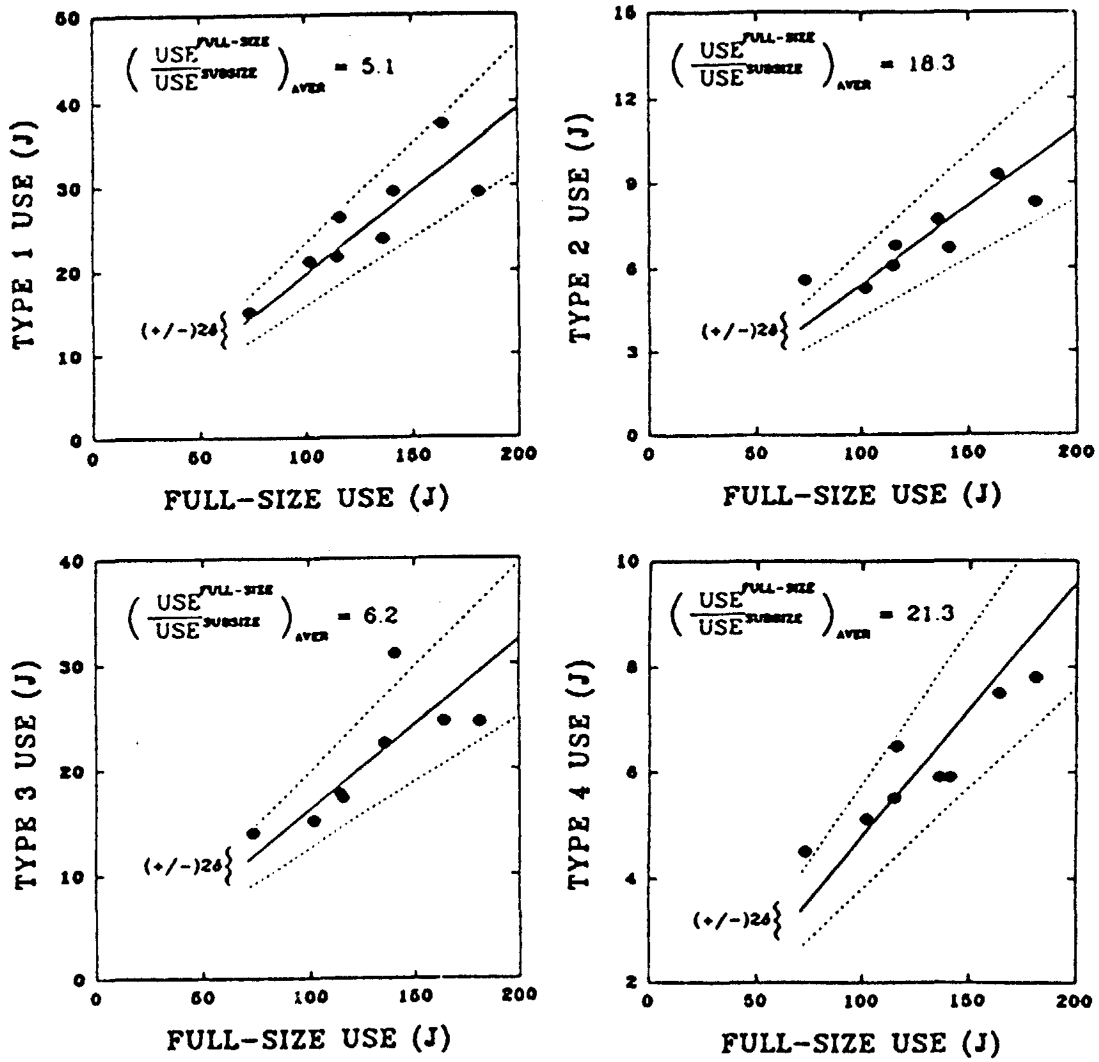

Figure 11.8. Correlations of upper-shelf energies between full- and subsize specimens.

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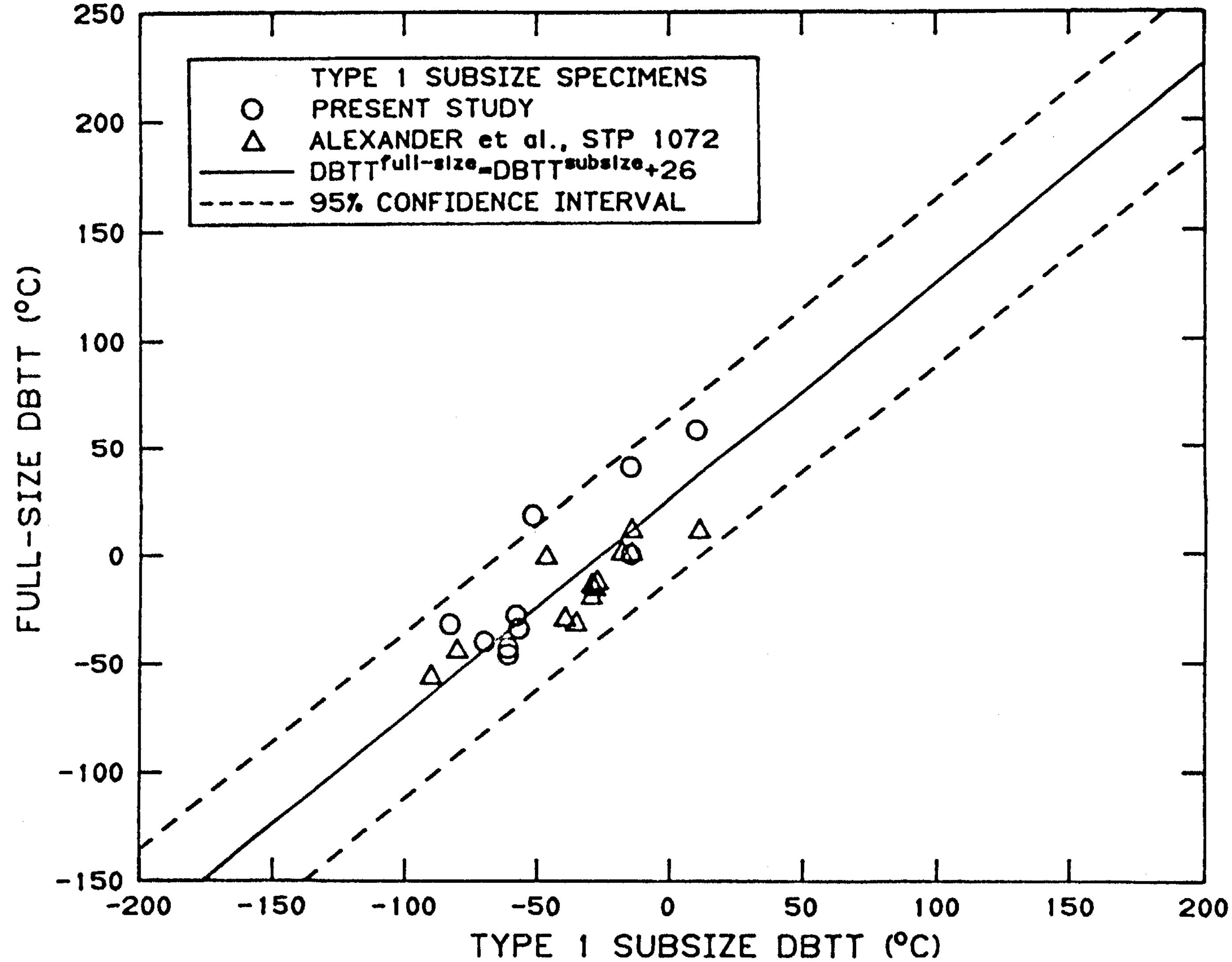

Figure 11.9. Correlation of ductile-to-brittle (DBTT) transition temperature between full-size and type 1 subsize specimens.

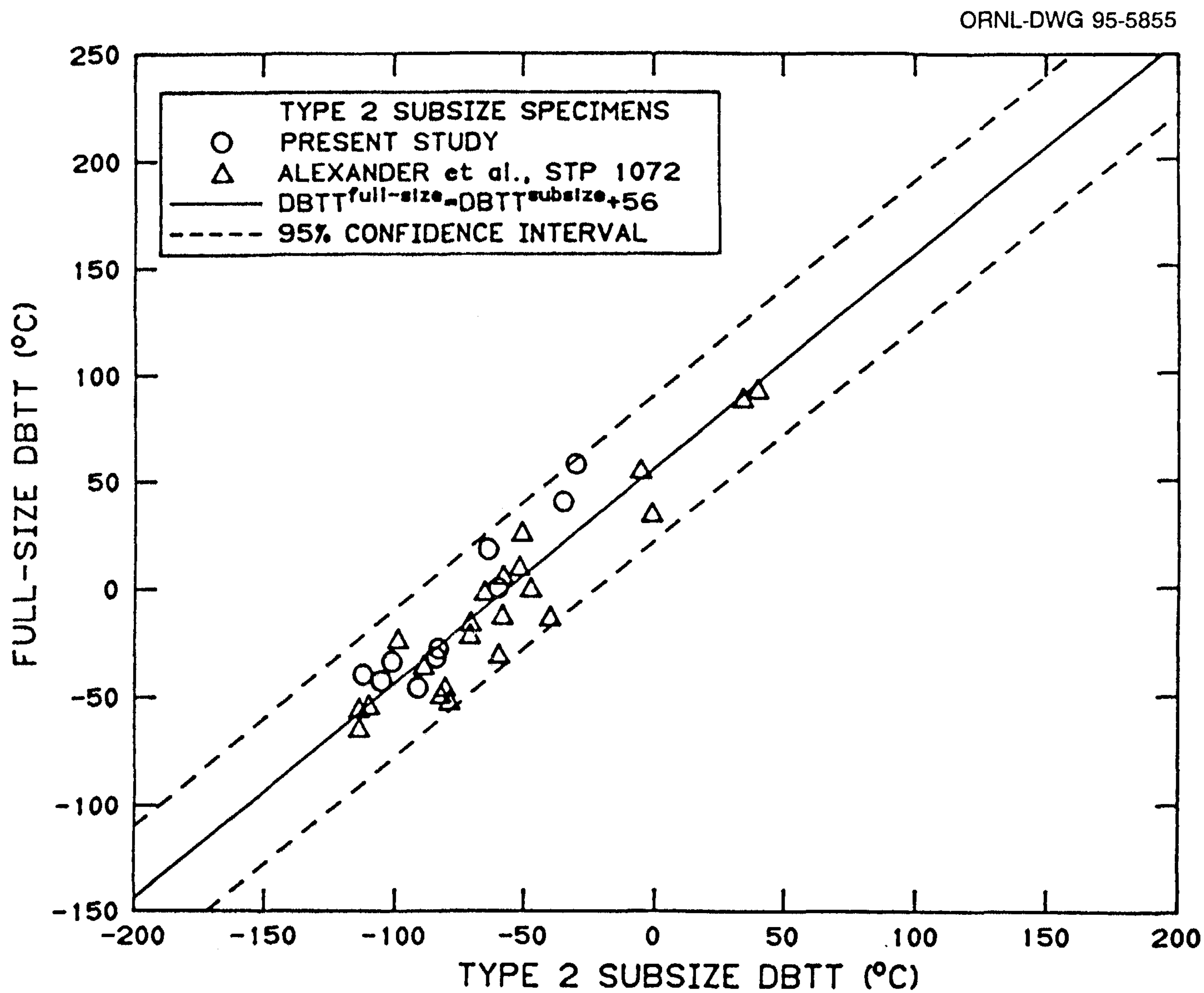

Figure 11.10. Correlation of DBTT between full-size and type 2 subsize specimens.

ORNL-DWG 95-5856

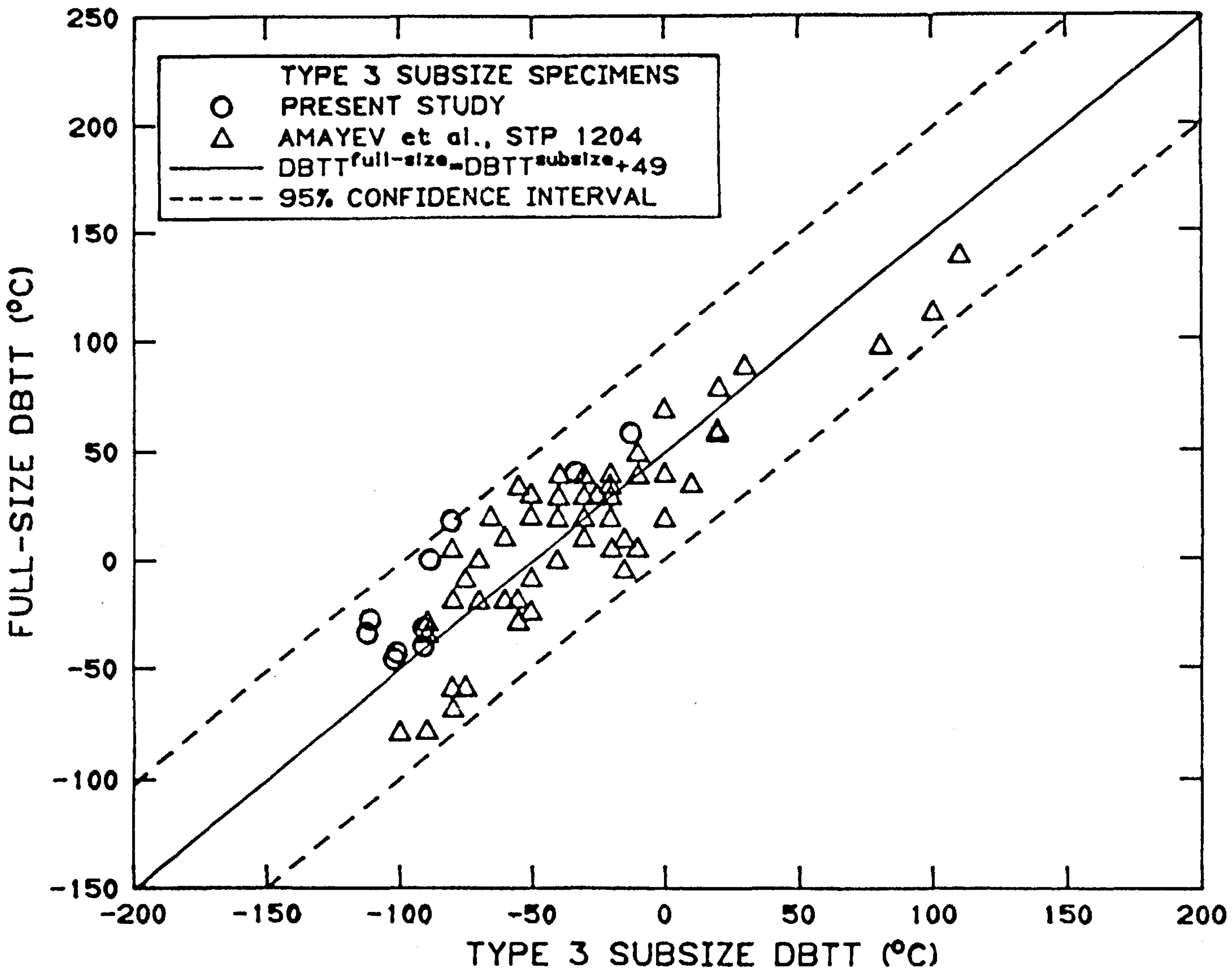

Figure 11.11. Correlation of DBTT between full-size and type 3 subsize specimens.

ORNL-DWG 95-5857

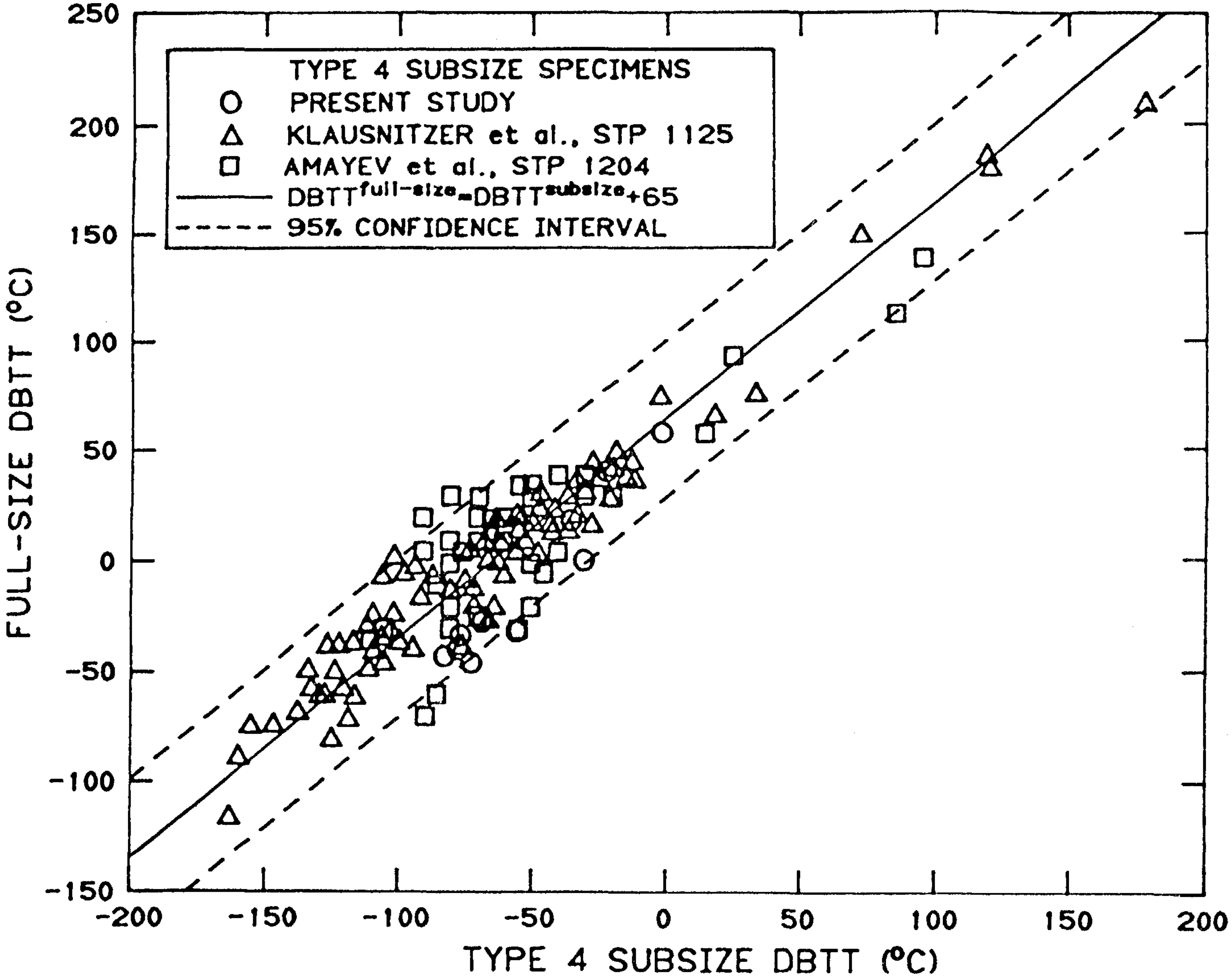

Figure 11.12. Correlation of DBTT between full-size and type 4 subsize specimens.

11.2.1 Summary and Conclusions

Five types of subsize specimens from ten materials were studied in the present work. The main results are as follows:

1. Subsize Charpy specimens may be useful for studies when material availability is limited. The broken halves of surveillance specimens can be remachined into subsize specimens to extend current surveillance programs or to monitor annealing response. The smallest specimen recommended by ASTM E-23 (5 × 5 × 55 mm) is too long for such an application.

2. It was found that (1) an increase of the relative notch depth (in the range studied) decreases USE but does not change the DBTT, (2) a decrease of the notch root radius (in the range studied) reduces USE and increases the DBTT, and (3) span and impact velocity (in the ranges studied) do not affect the USE and DBTT.

3. The 5 × 5 × 27 mm specimen with a 45° V-notch gives the same impact curve as the smallest specimen recommended by ASTM E-23 (5 × 5 × 55 mm).

4. The ratio between the USE of full-size and subsize specimens was determined for each type of subsize specimen. The appropriate ratios can be used as normalization factors for corresponding subsize specimen data.

5. The empirical correlations between DBTT of full-size and different subsize specimens were determined. Future understanding of the shift in the DBTT as a function of specimen size needs to be pursued.

6. Results obtained from the subsize specimens as well as the empirical correlations can be used for development of an ASTM standard practice for impact testing of subsize specimens for supplementary surveillance data in nuclear application.

11.3 ASTM Round Robin on Reconstituted Charpy Specimens

Reconstituted Charpy impact specimens from six participating institutions have been tested. Specimens with two insert sizes (10 and 14 mm) for each of two materials (HSST Plate 03 and an LUS weld) were tested with each of two strikers (2 and 8 mm radii) at each of two temperatures. The preliminary results were reported to ASTM Subcommittee E10.02 in June 1994. Specimens from the four remaining participants will be tested as a group later in 1994.

12. Technical Assistance for JCCCNRS Working Groups 3 and 12

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

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

12.1 Irradiation Experiments in Host Country

CVN and round tensile specimens were fabricated from two Russian weld metals supplied by the Russian National Research Center, Kurchatov Institute, and placed in HSSI capsule 10.06 for irradiation in the University of Michigan Ford Nuclear Reactor. The welds are identified (Russian designation) as weld 502 (wire SV-102KhMFT), a typical weld metal for VVER-440 reactors, and weld 260-11 (wire SV-12Kh2N2MAA), a typical weld for VVER-1000 reactors. The target neutron fluence of 1×10^{19} neutrons/cm^2 (> 1 MeV) at 288°C was achieved, and irradiation was completed in early September 1994. The specimens will be returned to ORNL for testing in 1995.

R. K. Nanstad met with Drs. A. Kryukov and A. Chernobaeva of Kurchatov Institute on September 29, 1994, regarding the irradiation and testing of U.S. steels provided to the Russian side as part of the cooperative test program. They have completed testing of the HSSI Program unirradiated weld 73W and the specimens annealed at 454°C for 72 h. They obtained essentially full recovery for this weld, and the unirradiated results appear to be in good agreement with the HSSI results, although a direct comparison has not yet been made. They have not completed the testing and dosimetry for the irradiated specimens, nor have they completed testing of the specimens annealed at 343°C for 150 h. Also, none of the irradiated or irradiated/annealed HSST Plate 02 specimens have yet been tested, although they have completed the testing of the unirradiated specimens. Nanstad agreed to send Dr. Chernobaeva details of the chemical composition and mechanical property information for both materials. Some of the specimens of both those materials which were irradiated and annealed are now being reirradiated. They have also tested unirradiated specimens of the Russian welds irradiated in capsule 10.06.

12.2 JCCCNRS Working Groups 3 and 12 Meetings

Meetings of the JCCCNRS were held in Kiev, Ukraine, on September 22-23 and in Moscow, Russia, on September 26-29, 1994. The visit to Ukraine was held as a separate workshop within the framework of the JCCCNRS. The Workshop on the Problem "Radiation Embrittlement, Structural Integrity and Life Extension of Reactor Pressure Vessel and Support," was held at the Institute for Problems of Strength. It was sponsored jointly by the Ukrainian State Committee for Nuclear Power Utilization, the Ukrainian State Committee for Nuclear and Radiation Safety, and Ukrainian National Academy of Science. A presentation was made by R. K. Nanstad on "A Summary of NRC Irradiation Effects Research," based on research results from the HSSI Program and information contributed by Professor G. R. Odette of the UCSB. The Working Group 3 meetings were held at the Russian National Research Center, Kurchatov Institute. Presentations for the HSSI Program were made by R. K. Nanstad in the areas of (1) J-integral resistance (J-R) round-robin testing, (2) embrittlement mechanisms and modeling, (3) thermal annealing experiments, and (4) subsize Charpy impact testing. A joint meeting of working Groups 3 and 12 (Life Extension) was held following the separate working group meetings, and other discussions were held subsequently with individual staff members of the Kurchatov Institute.

12.3 Personnel Interactions

The HSSI Program is sponsoring the sabbatical of Dr. Mikhail A. Sokolov at ORNL. Dr. Sokolov's areas of concentration are thermal annealing of irradiated RPVs and the use of subsize Charpy impact specimens for irradiated studies. The results of his research are presented within the particular technical tasks of HSSI semiannual progress reports and published technical reports and papers.

13. Correlation Monitor Materials

W. R. Corwin

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

During this reporting period, the transfer of the material, previously identified and inventoried as correlation monitor material, from its current site at the Y-12 Plant to a controlled-access storage location at ORNL was initiated. Almost all of the material has now been transferred to ORNL. A few pallets of steel remaining at Y-12 were not readily accessible, due to large plates of other materials that were in the way. The pallets will be transferred, once the large plates blocking their access are moved.

Detailed planning for the controlled-access location for storage of the remaining correlation monitor materials at ORNL was also continued. Construction of a concrete slab, as an extension of Building 7026, was begun. The slab will eventually be covered to shelter the samples from the weather to stop material deterioration. Information is being collected on types of available shelters and price ranges.

Additionally, correlation monitor material from HSST Plate 02 was cut and shipped to Consumers Power Company in Michigan for use in two surveillance capsules to be installed in the Palisades reactor vessel during a refueling outage in 1995.

14. Test Reactor Irradiation Coordination

I. I. Siman Tov and K. R. Thoms

This task was initiated during the current reporting period. It shall provide the support required to supply and coordinate irradiation services needed by NRC contractors other than ORNL. These services include the design and assembly of irradiation capsules as well as arranging for their exposure, disassembly, and return of specimens. Currently, the UCSB is the only other NRC contractor for whom irradiations are to be conducted. It is anticipated that these irradiations will be conducted at the University of Michigan in conjunction with other irradiations being conducted for the HSSI Program.

Design efforts were initiated to modify current ORNL irradiation capsules to facilitate the irradiation of UCSB specimens. A preliminary analysis of the thermal performance of a basic concept was completed. This analysis indicated that even for the highest specified flux, the nuclear heating was very marginal in providing adequate temperature control. Therefore, heated facilities will be required for the UCSB experiments, and a more precise nuclear heat generation rate distribution calculation was initiated. A heat transfer model was prepared to perform the analysis for the conceptual design to size the control gas gap, the heater power required, and the insulation needed to thermally isolate the three temperature regions for a given flux location.

In a review meeting with personnel from UCSB, information was exchanged on a possible conceptual design for a heated facility and the proposed experiment requirements of UCSB. UCSB agreed to develop specific specimen complements with the temperature and flux requirements that will have to be furnished by the facilities. The facility should consist of rectangular, heated core-edge positions that can be easily repositioned in the trolley to provide a variety of fluxes. The facility will allow rearranging the stacks of specimen complements within each heated facility. The heated regions will be divided such that the individual height of the specimen complements may vary as needed in a particular facility.

CONVERSION FACTORS*

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

*Multiply SI quanitty by given factor to obtain English quantity.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The goal of the Heavy-Section Steel Irradiation Program is to provide a thorough, quantitative assessment of effects of neutron irradiation on material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties. The HSSI Program is arranged into 14 tasks: (1) program management, (2) fracture toughness (K_{Ic}) curve shift in high-copper welds, (3) crack-arrest toughness (K_{Ia}) curve shift in high-copper welds, (4) irradiation effects on cladding, (5) K_{Ic} and K_{Ia} curve shifts in low upper-shelf welds, (6) annealing effects in low upper-shelf welds, (7) irradiation effects in a commercial low upper-shelf weld, (8) microstructural analysis of irradiation effects, (9) in-service aged material evaluations, (10) correlation monitor materials, (11) special technical assistance, (12) JPDR steel examination, (13) technical assistance for JCCCNRS Working Groups 3 and 12, and (14) additional requirements for materials. This report provides an overview of the activities within each of these tasks from April 1994 Through September 1994.

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