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Heavy-Section Steel Irradiation Program

Progress Report for April - September 1996

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Prepared by
W. R. Corwin

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

Oak Ridge National Laboratory
Oak Ridge, TN 37831-6285

M. G. Vassilaros, NRC Project Manager

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Division of Engineering Technology
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Abstract

Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents that have the potential for major contamination release. The RPV is the only key safety-related component of the plant for which a duplicate or redundant backup system does not exist. It is therefore imperative to understand and be able to predict the capabilities and limitations of the integrity inherent in the RPV. In particular, it is vital to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance that occurs during service because 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 was 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 (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, fracture mechanics testing of all specimens of the irradiated LUS Midland Weld WF-70 were completed and detailed analyses performed. Heat treatment of five different RPV plate materials, including A 302 grade B, was initiated to examine of the effects of phosphorus segregation on the fracture toughness of the HAZ of welds of plate materials typical of those used in fabricating older RPVs. Initial results show that all five materials exhibited very large prior austenite grain sizes, such as would be found in the HAZ, as desired and expected, as a consequence of the initial heat treatment. Irradiated and annealed 1-in. compact tension [1T C(T)] specimens of Midland LUS weld material were tested and analyzed. Four sets of Charpy V-notch (CVN) specimens were aged for 168 h at various temperatures and tested to examine the reason for "overrecovery" of upper-shelf energy that has been observed when CVN specimens are annealed at 454°C (850°F). Molecular dynamics cascade simulations, used to develop effective defect production cross sections, were extended to 40 keV, the highest value ever achieved. With these simulations, the cascade simulations have now provided information representative of most of the fast neutron spectrum, since ~90% of the neutron spectrum at a pressurized-water reactor RPV is below 1.8 MeV. Investigations of the correlation between microstructural changes and mechanical property (hardness) changes in ion-irradiated model alloys was completed during this period. Preliminary planning for test specimen machining for the JPDR materials was completed. Although the "hotter" of the specimens will need to be machined in a remote hot cell, it appears that specimens with lower levels of activation can be machined in a "hot shop" that contains machine tools with the power required for high feed rates and gang machining. A database of Charpy impact and fracture toughness data for RPV materials that have been tested in the unirradiated and irradiated conditions is being assembled and analyzed. Weld metal appears to have similar CVN and fracture toughness transition temperature shifts, whereas the fracture toughness shifts are greater than CVN shifts for base metals. Analyses of RT_{NDT} and T_{100} values for material

comprising the American Society for Mechanical Engineers fracture toughness data were shown to have little correlation with each other. Draft subcontractor reports on precracked cylindrical tensile specimens, by AEA Technology and SRI International, were completed, reviewed by HSSI staff, and are now being revised. Testing on precracked CVN (PCVN) specimens, both quasi-static and dynamic, was evaluated with regard to the draft American Society for Standards and Materials (ASTM) Standard on Fracture Toughness Testing in the Transition Region. Additionally, testing of 0.2T compact specimens [0.2T C(T)] was initiated as an experimental comparison of the constraint limitation in the ASTM draft standard for the PCVN specimen. Equipment and materials that were provided to Materials Engineering Associates (MEA) by the U.S. Nuclear Regulatory Commission were inventoried by MEA, reviewed by Oak Ridge National Laboratory, and shipped to Oak Ridge before final disposition. The JCCCNRS Working Groups 3 and 12 met in Moscow on September 15–20, 1996, where information was presented on irradiation, annealing, reirradiation, and reannealing of HSSI weld 73W and Heavy-Section Steel Technology Plate 02 showing that the annealing treatments resulted in virtually full recovery after irradiation and reirradiation. Moreover, the reembrittlement, although substantial, fell slightly below that predicted by the lateral shift method. The design of the University of California, Santa Barbara, irradiation facility was completed, and fabrication of the major components continued. An instrumentation and control system has been designed, and assembly of this system is nearing completion.

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Acknowledgments

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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 that involve RPV irradiation embrittlement such as pressurized thermal shock, operating pressure-temperature limits, low-temperature overpressurization, and the specialized problems associated with low upper-shelf welds.

This HSSI Program progress report covers work performed from April to September 1996. 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, Computational Physics and Engineering, and Engineering Technology Divisions.

Previous HSSI progress reports in this series are as follows.

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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 before 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. Therefore, the HSST Program progress reports issued before formation of the HSSI Program are also listed here as a convenience to the reader.

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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 Oak Ridge National Laboratory (ORNL).

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 (ASME) fracture toughness (K_{Ic}, K_{Ia}, and K_{IR}) curves to account for irradiation embrittlement in high-copper welds. The specific activities to be performed in this task are (1) the continuation of Phase 2 of the Fifth Irradiation Series and (2) completion of the Sixth Irradiation Series, including the testing of the nine irradiated Italian crack-arrest specimens. The continuation of Phase 2 of the Fifth Series includes irradiation of HSSI weld 73W to a high fluence [5×10^{19} n/cm^2 (>1 MeV)] to determine whether the K_{Jc} curve shape change observed in the Fifth Series is exacerbated. Testing of the Italian crack-arrest specimens and preparation of a NUREG report presenting the detailed results of testing and analysis has been completed. The report is undergoing final editing before transmittal to the U.S. Nuclear Regulatory Commission (NRC) for printing. ORNL experiences in crack-arrest fracture toughness of irradiated pressure-vessel steels were presented at the American Society for Testing and Materials (ASTM) Symposium on User's Experience in Crack-Arrest Testing, New Orleans, November 18, 1996. No work for this task was funded during this reporting period.

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 V-notch (CVN) upper-shelf energy (USE). This task was specifically designed to address questions raised by the Advisory Committee for Reactor Safeguards concerning the shape of the K_{Ic} curve for irradiated welds with a low USE. In particular, it will clarify whether the high concentration of inclusions in low USE welds results in a transition relationship and behavior significantly different from high USE welds. The information developed under this task will augment information obtained from other HSSI tasks performed on two high USE weldments under the Fifth and Sixth Irradiation Series and on a commercial low USE weldment under the Tenth Irradiation Series. The results will provide an expanded basis for accounting for irradiation-induced embrittlement in reactor pressure vessel (RPV) materials. No work for this task was funding during this reporting period.

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

The objective of this task is to evaluate the chemical, mechanical, and fracture properties of the WF-70 weld metal at the beltline and nozzle course locations in the Midland Unit 1 reactor vessel before and after irradiation. This vessel became available for test sampling and evaluation when Consumers Power aborted plans to operate the facility. Weld metal WF-70 was used in all girth welds; this designation indicates that a specific lot of Linde 80 weld flux was used that produces low CVN upper-shelf toughness. The following activities occurred during this reporting period.

Tests to establish the transition temperature for all conditions of embrittlement for WF-70 weld metal are now complete. The tests performed had two objectives: (1) determine the fracture toughness according to the currently used ASME *Boiler and Pressure Vessel Code* and (2) evaluate new methods for evaluating transition temperature. The full complement of properties needed for both types of evaluation have been developed. Nozzle course WF-70 and beltline WF-70 welds are treated as different materials because of a difference in copper content (nominally 0.40 vs 0.26 wt %, respectively). Compact specimens were tested to establish the fracture mechanics–based transition temperature. The methodology used is the master curve. Values of T_0 for the unirradiated condition are –54°C for the beltline material and –32°C for the nozzle course material. The magnitudes of the ΔTT shifts at 1×10^{19} n/cm^2 fluence are 81 and 94°C for the beltline and nozzle course welds, respectively. The differences in copper content for the two welds resulted in less differences in T_0 transition temperature shifts than predicted by *Regulatory Guide 1.99.* In fact, the TT_{41J} shifts at 1×10^{19} n/cm^2 are reversed in that the shift for the beltline weld is greater than that of the nozzle course weld.

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

The original objective of this task was to examine the effects of neutron irradiation on the fracture toughness of the HAZ of welds of A 302 grade B (A302B) plate materials typical of those used in fabricating older RPVs. However, emphasis is now being given to determine if there is a potential problem with temper embrittlement in RPV steels as a consequence of neutron irradiation and postirradiation thermal annealing. The following activities occurred during this reporting period.

AEA Technology, Harwell, United Kingdom, showed quite clearly that there can be grain boundary embrittlement in RPV steels, given large prior austenite grain size and high phosphorus on the order of 0.017 wt %. The first task in this project will be to reexamine the AEA Technology heat treatments using five commercially made RPV steels, representing A 302 grade B, A 533 grade B, A 508 class 2, and two modified A 302 grade B heats. The phosphorus content covers the range typical of commercial RPV production heats. A second task will evaluate the potential for local brittle zone (LBZ) development in multipass submerged-arc welds made in RPV joints. Finally, the third task will be to select two materials of highest interest resulting from the two preceding tasks and produce a simulated commercial submerged-arc weld HAZ. The purpose will be to (1) determine if there are LBZs and (2) if so, demonstrate the significance of these zones to the structural integrity performance of RPVs. An additional objective will be to determine if irradiation will promote temper embrittlement of LBZs. The five materials previously mentioned have been sectioned, and the heat treatments have been initiated. Initial results show that all five materials exhibited very large prior austenite grain sizes, as desired and expected, as a consequence of the initial heat treatment.

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 CVN impact energy during irradiation, annealing, and reirradiation. Twelve irradiated and annealed 1-in. LUS compact tension [1T C(T)] specimens were tested and analyzed. Four sets of CVN specimens were aged for 168 h at various temperatures and tested to shed some light on the reason for "overrecovery" of upper-shelf energy that has been observed when CVN specimens are annealed at 454°C (850°F). Design was begun on three containers for shipment of previously irradiated and annealed CVN specimens and wire dosimeters. Designing has also begun on retrievable fission dosimetric set containers. Work on the new irradiation, annealing, and reirradiation facilities and capsules has progressed to the point where the facilities and capsules have to be stress-relieved before final checkout at temperature can begin.

7. Microstructural and Microfracture Analysis of Irradiation Effects

The overall long-term goal of this task is to develop a physically based model that can be used to predict irradiation-induced embrittlement in reactor vessel steels over the full range of their service conditions. The model should be tethered soundly on the microstructural level by results from advanced microstructural analysis techniques and constrained at the macroscopic level to produce predictions consistent with the large array of macroscopic embrittlement measurements that are available.

During this reporting period, the energy range of the molecular dynamics displacement cascade simulations was extended to 40 keV, the highest value ever achieved. Six cascades were completed at this energy in simulations involving 1,024,000 atoms. Each such cascade required about three weeks of central processing unit time on a very fast workstation. The 40-keV simulation energy corresponds to the average iron primary knock-on atom in a collision with a 1.8-MeV neutron. Thus, the cascade simulations have now provided information representative of most of the fast neutron spectrum since ~90% of the neutron spectrum at a pressurized-water reactor RPV is below 1.8 MeV. The component of this task investigating the correlation of microstructural changes and mechanical property (hardness) changes in ion-irradiated model alloys was completed during this period. The final aspect involved obtaining quantitative measurements of the radiation-induced defect distributions that could be used to compute a predicted hardening value for comparison with the measured values. Because some of the problems inherent in imaging small defects in magnetic materials proved to be intractable, only qualitative agreement between the calculated and measured hardnesses could be achieved.

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. Four activities are in progress: (1) establishing a machining capability for contaminated or activated materials by completing procurement and installation of a computer-based milling machine in a hot cell; (2) machining and testing specimens from cladding materials removed from the Gundremmingen reactor to establish their fracture properties; (3) preparing an interpretive report on the effects of neutron irradiation on cladding; and (4) continuing the evaluation of long-term aging of austenitic structural stainless steel weld metal by metallurgically examining and testing specimens aged at 288 and 343°C and reporting the results, as well as by continuing the aging of the stainless steel cladding toward a total time of 50,000 h.

During this reporting period, all work on the setup of the computer numerically controlled, remotely operated machining center, with a mill, a saw, as well as preparation of the hot cell, was put on hold pending the outcome of a funding review. Several meetings were held between HSSI Program staff and hot cell staff, ORNL management, and the Plant and Equipment (P&E) Division to review the current status and funds required to complete installation. A detailed work breakdown structure (WBS) was completed, and time, schedule, and cost estimates were obtained. The P&E Division provided assurance that the costs were accurate, and no cost overruns are foreseen. The NRC requested that ORNL obtain cost estimates for machining Charpy, tensile, and 1/2T compact specimens from irradiated materials; the estimates were prepared and submitted to the NRC. The NRC has determined that such a center is viable and has instructed ORNL to proceed with installation. The project WBS and time allocations for installation of the remotely operable milling machine and saw, as well as installation of a crane inside the hot cell, have been mapped onto a list of 13 maintenance job requests to allow efficient tracking of progress vs costs.

9. Evaluation of Steel from the JPDR Pressure Vessel

The objectives of the Japan Atomic Energy Research Institute 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 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 preliminary planning for test specimen machining was completed and costs of machining the specimens from base and weld metal from each of four materials were estimated. There are four trepans each of the weld and base metal from both the beltline region and the remote region. All 16 trepans, 87 mm in diameter and 78 mm long, have stainless steel cladding, from which most of the radioactivity of the material originates. It appears that the machining of the specimens from the remote area could proceed in a "hot shop" that contains machine tools with the power required for high feed rates and gang machining. Estimates for such an option are being pursued.

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. 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 caused by neutron irradiation, thermal aging, or both; (2) evaluation of methods for indexing fracture toughness curves to values that can be deduced from material surveillance programs required under the *Code of Federal Regulations* (10 CFR 50), Appendix H; (3) participation in the pertinent ASME *Boiler and Pressure Vessel Code*, Section XI, and ASTM E-8 and E-10 committees; (4) interaction with other researchers in the national and international technical community addressing similar problems; and (5) frequent interactions and detailed technical meetings with the NRC staff.

A database of Charpy impact and fracture toughness data for RPV materials that have been tested in the unirradiated and irradiated conditions is being assembled. Currently, there are 39 data sets for welds and 50 for base metals. Preliminary results show some differences between the weld metal and base metal data sets. For weld metals, on average, the Charpy transition temperature shift at 41 J is the same as the shift of fracture toughness with 95% confidence intervals of about ±30°C. For base metals, on average, the fracture toughness shift is 12°C greater than the Charpy 41-J temperature shift with 95% confidence intervals of about ±35°C. Various issues associated with the preliminary analyses will be addressed in future evaluations of the database.

Another subtask involves application of the Weibull statistic/master curve analysis procedure to the linear-elastic K_{Ic} database that has been used to support the ASME lower-bound curve. The master curve represents well the median trend of this database, while the scatter of data is characterized by lower- and upper-bound curves. Current results provide additional support to the statistical nature of brittle fracture and the importance of using a statistical method to describe probabilities of such events. Thus, different materials can be compared based on fracture toughness level by means of the reference temperature T_{100} rather than the empirical correlation to RT_{NDT} values. There is little correlation between these two reference temperatures for the materials analyzed.

11. Special Technical Assistance

This task is included within the HSSI Program to provide a vehicle in which to conduct and monitor short-term, high-priority subtasks and to provide technical expertise and assistance in the review of national codes and standards that may be referenced in NRC regulations or guides related to nuclear reactor components. This task currently addresses two major areas: (1) providing technical expertise and assistance in the review of national codes and standards and (2) experimentally evaluating test specimens and practices and material properties. The following activities occurred during this reporting period.

Crack-arrest testing of irradiated specimens is both difficult and expensive. A preliminary study has been conducted to explore one possible alternative of using information already available from instrumented CVN testing. Various stages of the fracture process can be identified on the voltage vs time output trace, including an arrest point indicating arrest of brittle fracture. Preliminary results show that prediction of conservative values of K_a are possible.

Draft reports on precracked cylindrical tensile specimens, by G. Gage of AEA Technology, Harwell Laboratory, United Kingdom, and J. H. Giovanola and J. E. Crocker of SRI International, Menlo Park, California, were completed, reviewed by HSSI staff, and are now being revised.

Although testing on precracked CVN (PCVN) specimens, both quasi-static and dynamic testing, has been performed by ORNL and many other investigators for many years, the advent of the draft ASTM Standard on Fracture Toughness Testing in the Transition Region has motivated a more focused effort regarding PCVN specimens. A considerable amount of quasi-static testing was performed by ORNL with the PCVN specimen for a number of RPV materials, and the general observation is that the specimen has high potential as a viable test specimen for determining T_o for RPV steels. Moreover, testing of 0.2T compact specimens [0.2T C(T)] has been initiated as an experimental comparison of the constraint limitation in the ASTM draft standard for PCVN specimens.

Equipment and materials that were provided to Materials Engineering Associates (MEA) by the NRC were inventoried by MEA, reviewed by ORNL, and shipped to Oak Ridge. A final inventory and recommendations for disposition are in preparation and will be sent to the NRC for approval before final disposition.

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. Specific activities of this task are: (1) the supplying of materials and preparation of test specimens for collaborative irradiation, annealing, and reirradiation studies to be conducted in Russia; (2) capsule preparation and initiation of irradiation of Russian specimens within the United States; and (3) preparation for, and participation in, meetings of Working Groups 3 and 12. The following activities occurred during this reporting period.

The CVN and round tensile specimens of two Russian weld metals irradiated in HSSI capsule 10.06 at the University of Michigan Ford Nuclear Reactor (FNR) were returned to ORNL and removed from the capsule for testing. R. K. Nanstad presented the preliminary results of these tests to the JCCCNRS Working Group 3 meeting in Moscow in September 1996.

JCCCNRS Working Groups 3 and 12 met in Moscow on September 15–20, 1996. The seventh meeting of Working Group 3 was held at the Russian Research Center-Kurchatov Institute. Information was presented on irradiation, annealing, reirradiation, and reannealing of HSSI weld 73W and Heavy-Section Steel Technology (HSST) Plate 02 showing that the annealing treatments resulted in virtually full recovery after irradiation and reirradiation. Moreover, the reembrittlement, although substantial, fell slightly below that predicted by the lateral shift method. Details of these meetings were reported by R. K. Nanstad in ORNL/FTR-5905, October 7, 1996.

13. Correlation Monitor Materials

This task was 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. Because 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 was established to provide cataloging, archiving, and distribution of the material on behalf of the NRC. During this reporting period, archival storage of the correlation monitor material at ORNL was maintained.

14. Test Reactor Irradiation Coordination

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

During this reporting period, the design of the UCSB irradiation facility was completed, certified-for-construction drawings were issued, and fabrication of the major components continued. An instrumentation and control system has been designed, and assembly of this system is nearing completion. This system will automatically control the electrical heaters in the six independently controlled temperature regions of the facility. Before shipment to the FNR, the facility and associated instrumentation system will be integrated and tested to ensure that the facility will perform as intended.

Heavy-Section Steel Irradiation Program Semiannual Progress Report for April through September 1996*†

W. R. Corwin

1. Program Management

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

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

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

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The HSSI Program consists of 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), (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, the HSSI staff made four program briefings, reviews, or presentations during program reviews and visits with NRC staff or others. Ten technical papers[1–10] were published, and 36 technical presentations were made.[11–46]

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2. FRACTURE TOUGHNESS SHIFTS IN HIGH-COPPER WELDS

S. K. Iskander

The objective of this task is to develop data addressing the current method of shifting the American Society of Mechanical Engineers (ASME) fracture toughness (K_{Ic}, K_{Ia}, and K_{IR}) curves to account for irradiation embrittlement in high-copper welds. The specific activities to be performed in this task are (1) the 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. The continuation of Phase 2 of the Fifth Series includes irradiation of HSSI weld 73W to a high fluence [5×10^{19} n/cm^2 (>1 MeV)] to determine whether the K_{Jc} curve shape change observed is exacerbated. No work has been funded for this task.

Testing was previously completed on nine irradiated Italian crack-arrest specimens fabricated from a 0.06% copper forging that conforms to the American Society for Testing and Materials (ASTM) Specification for Quenched and Tempered Vacuum-Treated Carbon and Alloy Steel Forgings for Pressure Vessels, Class 3 (A 508-81). A significant result from the tests is that both Charpy and crack-arrest fracture toughness shifts caused by neutron irradiation to a level of 3.5×10^{19} n/cm^2 (>1 MeV) at approximately 288°C were similar and small, averaging about 10 K. A NUREG report presenting detailed results and analysis, *Results of Crack-Arrest Tests on Irradiated A508 Class 3 Steel* [NUREG/CR-7477 (ORNL-6894)], by S. K. Iskander, P. P. Milella, and A. Pini, has been completed and is undergoing final editing before transmittal to the NRC for printing.

An abstract, "Crack-Arrest Testing of Irradiated Nuclear Reactor Pressure Vessel Steels at Oak Ridge National Laboratory," by S. K. Iskander, P. P. Milella, A. Pini, and E. T. Manneschmidt, was prepared and submitted for presentation at the ASTM Symposium on User's Experience in Crack-Arrest Testing, held in New Orleans on November 18, 1996.

3. FRACTURE TOUGHNESS SHIFTS IN LOW UPPER-SHELF WELDS

S. K. Iskander

This task examines the fracture toughness curve shifts and changes in shape for irradiated welds with low CVN upper-shelf energy (USE). This task was specifically designed to address questions raised by the Advisory Committee for Reactor Safeguards concerning the shape of the K_{Ic} curve for irradiated welds with a low USE. In particular, it will clarify whether the high concentration of inclusions in low-USE welds results in a transition relationship and behavior significantly different from high-USE welds. The information developed under this task will augment information obtained from other HSSI tasks performed on two high-USE weldments under the Fifth and Sixth Irradiation Series and on a commercial low-USE weldment under the Tenth Irradiation Series. The results will provide an expanded basis for accounting for irradiation-induced embrittlement in RPV materials. No work was funded for this task during this reporting period.

To provide material for this task and for the annealing task (Series 9), three trial, low-USE welds were fabricated by ABB-Combustion Engineering (ABB-CE), Chattanooga, Tennessee, which had also fabricated the welds for the Fifth and Sixth Irradiation Series. Previous semiannual reports gave the chemical and mechanical properties obtained by ABB-CE and ORNL for all three welds. The test results show that there is no significant difference between specimens machined from 13 mm below the top of the weld and those through the depth of the weld. Furthermore, the USE for a weld made with Linde 80 flux is only 10% less than the HSSI welds 72W and 73W fabricated using the same weld wire and Linde 124 flux. This indicates that in this case the Linde 80 flux alone does not cause significant decreases in the USE. There was no significant difference in the 41-J transition temperatures of the Linde 80 and Linde 124 flux welds. The USE of this weld is significantly higher than other Linde 80 flux welds tested by ORNL as part of the Second, Third, Fourth, and Tenth Irradiation Series.

An abstract, "Estimation of NDT and Crack-Arrest Toughness from Charpy Force Displacement Traces," by M. A. Sokolov, was prepared and submitted for the ASTM Symposium on Small Specimen Test Techniques, held in New Orleans on January 13–14, 1997.

4. IRRADIATION EFFECTS IN A COMMERCIAL LOW UPPER-SHELF WELD

D. E. McCabe

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

Tests to establish the transition temperature for all conditions of embrittlement for WF-70 weld metal are now complete. The tests performed had two objectives: (1) determine the fracture toughness by the currently used ASME *Boiler and Pressure Vessel Code*[1] and (2) evaluate new methods for evaluating transition temperature. This section concentrates on the ΔTT shift caused by 1×10^{19} n/cm^2 (>1 MeV) at 288°C (550°F). The full complement of properties needed for both types of evaluation have been developed. Beltline WF-70 and nozzle course WF-70 are treated as different materials because of a difference in copper content (nominally 0.26 vs 0.40 wt %, respectively). The tensile properties are summarized in Table 4.1. Note that the room temperature tensile properties are about 6% higher in the nozzle course weld than in the beltline weld. Irradiation to 1×10^{19} n/cm^2 (>1 MeV) increases the strength of both weld metals by about 25%.

CVN impact tests are used on LUS materials to establish the ASME Code RT_{NDT} temperature, from which the lower-bound K_{Ic} curve of fracture toughness is established. It has been reported previously that the unirradiated RT_{NDT} temperatures from 19 CVN transition curves range from –20 to +37°C for the beltline weld metal and from –8 to +18°C based on six nozzle course CVN transition curves. Table 4.2 (first row) provides the average RT_{NDT} from the data scatter (–9°C for the beltline and –1°C for the nozzle course).

Title 10, Part 50, of The *Code of Federal Regulations* (10 CFR 50)[2] references ASTM Standard E 185 to quantify transition temperature shift. The magnitude of shift is measured at the CVN 41-J energy level. If no CVN curves are available, the magnitude of the shift can be estimated by using the NRC *Regulatory Guide 1.99*, Rev. 2.[3] Material chemistry and fluence are used in that case.

Compact specimens were tested to establish the fracture mechanics–based transition temperature. The methodology used is the master curve. In this case, the transition temperature, T_o, is referenced to a K_{Jc} fracture toughness level of 100 MPa√m for median fracture toughness from 1T specimens. The determination of reference temperature, T_o, can be accomplished at any test temperature and with any specimen size that is established to be within certain reasonable limits. Values of T_o for the unirradiated condition are also shown in Table 4.2 (second row), that is, –54°C for the beltline material and –32°C for the nozzle course material. This difference in toughness was not detected from CVN results.

Table 4.1. Yield and tensile strengths for the beltline and nozzle course WF-70 weld metal before and after irradiation

Test temperature (°C)	Unirradiated			Irradiated					
				0.5 × 10^{19} n/cm^2			1 × 10^{19} n/cm^2		
	Number of specimens	Strength [MPa (ksi)]		Number of specimens	Strength [MPa (ksi)]		Number of specimens	Strength [MPa (ksi)]	
		Yield	Ultimate		Yield	Ultimate		Yield	Ultimate
Beltline WF-70 weld metal									
288	2	468 (68.0)	610 (88.4)						
150	2	475 (69.0)	584 (84.7)	2	582 (84.4)	668 (97.0)	2	594 (86.2)	697 (101.1)
22	2	512 (74.3)	613 (88.9)	2	633 (91.9)	717 (104.0)	2	646 (93.7)	746 (108.3)
−50	2	569 (82.6)	693 (100.6)						
−100	2	625 (90.7)	763 (110.8)						
−150	1	736 (106.9)	850 (123.4)						
Nozzle course WF-70 weld metal									
288	2	484 (70.2)	613 (89.0)				2	628 (91.1)	716 (103.9)
150	2	485 (70.4)	586 (85.1)	2	503 (72.9)	636 (92.3)	2	634 (92.0)	720 (104.5)
22	2	544 (79.0)	654 (94.9)	2	596 (86.4)	709 (102.9)	1	701 (101.7)	791 (114.8)
−50	2	579 (84.0)	717 (104.1)						
−100	2	648 (94.0)	819 (118.9)						

Table 4.2. Transition temperature summary for beltline and nozzle course weld metal

Parameter	Irradiation ($\times 10^{19}$ n/cm^2)	Transition temperature (°C)	
		Beltline	Nozzle course
Average RT_{NDT}	None	–9	–1
T_o	None	–54	–32
ΔTT_{41J}	0.5	45 (81)[a]	63 (103)[a]
ΔT_o	0.5	78	–
ΔTT_{41J}	1.0	103 (100)[a]	90 (128)[a]
ΔT_o	1.0	81	94

[a]Numbers in parentheses are the shifts predicted by *Regulatory Guide 1.99.*

Table 4.2 also shows transition temperature shifts as measured by the two methods plus the calculated values obtained using *Regulatory Guide 1.99*. The *Regulatory Guide 1.99* equation follows:

$$\Delta TT = (CF)(f)^x \ , \tag{1}$$

where CF ≈ 100 and 128°C for the beltline and nozzle course welds, respectively, f = fluence, and $x = 0.28 - 0.1 \log_{10}(f)$.

The transition temperature shift indicated by master curves is shown in Figure 4.1. Note that there is a clear difference between beltline and nozzle course transition temperatures. The magnitudes of the ΔTT shifts at 1×10^{19} n/cm^2 fluence are 81 and 94°C for the beltline and nozzle course welds, respectively. Figures 4.2 and 4.3 provide a basis of comparison among various ΔTT methodologies. Measurement of ΔTT from ultimate tensile strength change is not a recognized methodology, but a convenient comparison can be made by normalizing strength with a fixed coefficient (0.725 × MPa). Considering first the beltline weld in Figure 4.2, it is apparent that the single CVN curve used to evaluate ΔTT at 0.5×10^{19} n/cm^2 must have been in the extreme part of the CVN scatter band. Otherwise, there is excellent agreement with the *Regulatory Guide 1.99* trend line. For the nozzle course weld (Figure 4.3), there is reasonable agreement among the three experimental ΔTT determinations; however, the value of the chemistry factor (CF) obtained from *Regulatory Guide 1.99* appears to have been too large. The differences in copper content for the two welds resulted in smaller differences in T_o transition temperature shifts than predicted by *Regulatory Guide 1.99*. In fact, the TT_{41J} shifts at 1×10^{19} n/cm^2 are reversed in that the shift for the beltline weld is more than that of the nozzle course weld.

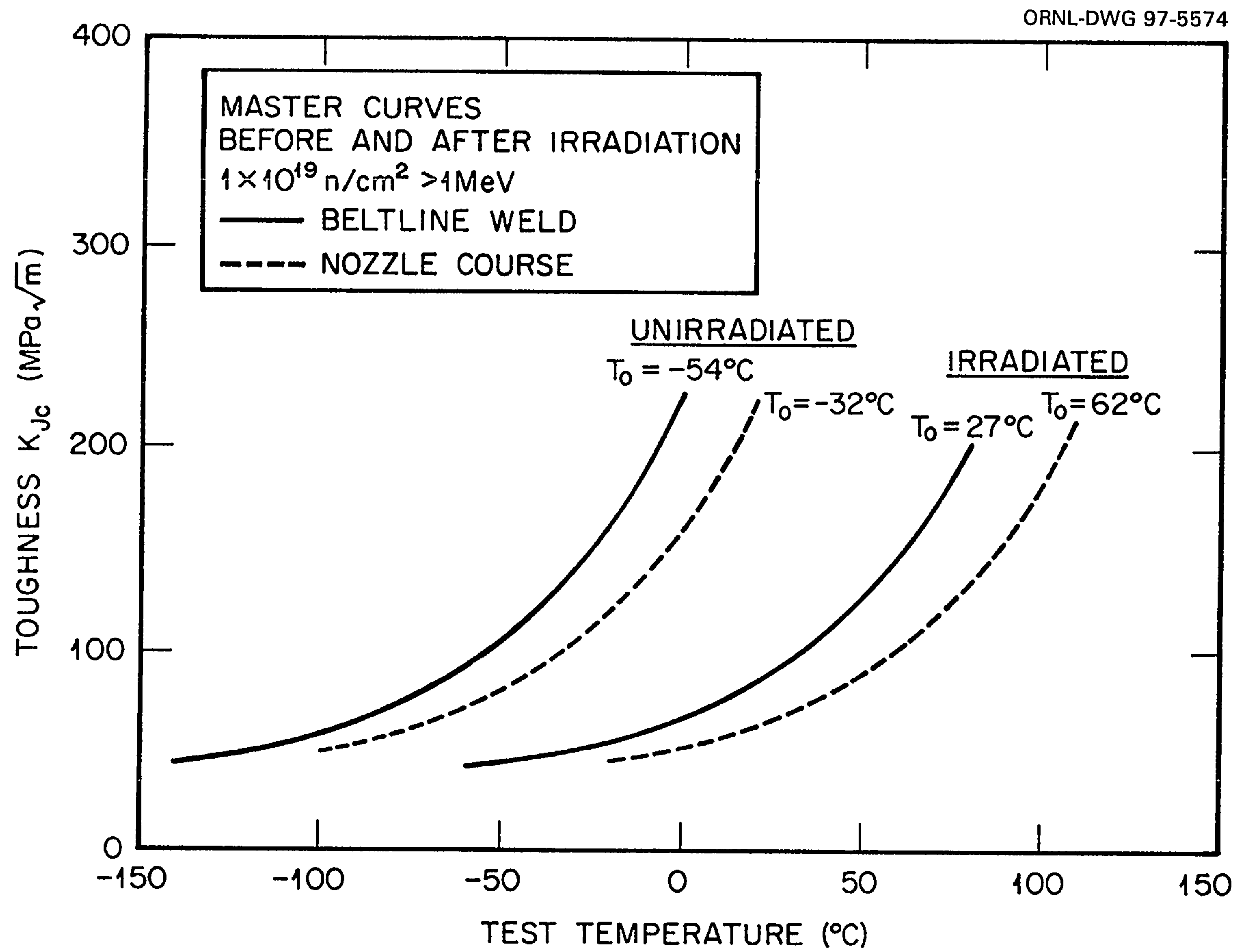

Figure 4.1 Master curves for the Midland beltline and nozzle course welds before and after irradiation to 1×10^{19} n/cm^2 (>1 MeV) at 288°C.

ORNL-DWG 97-5622

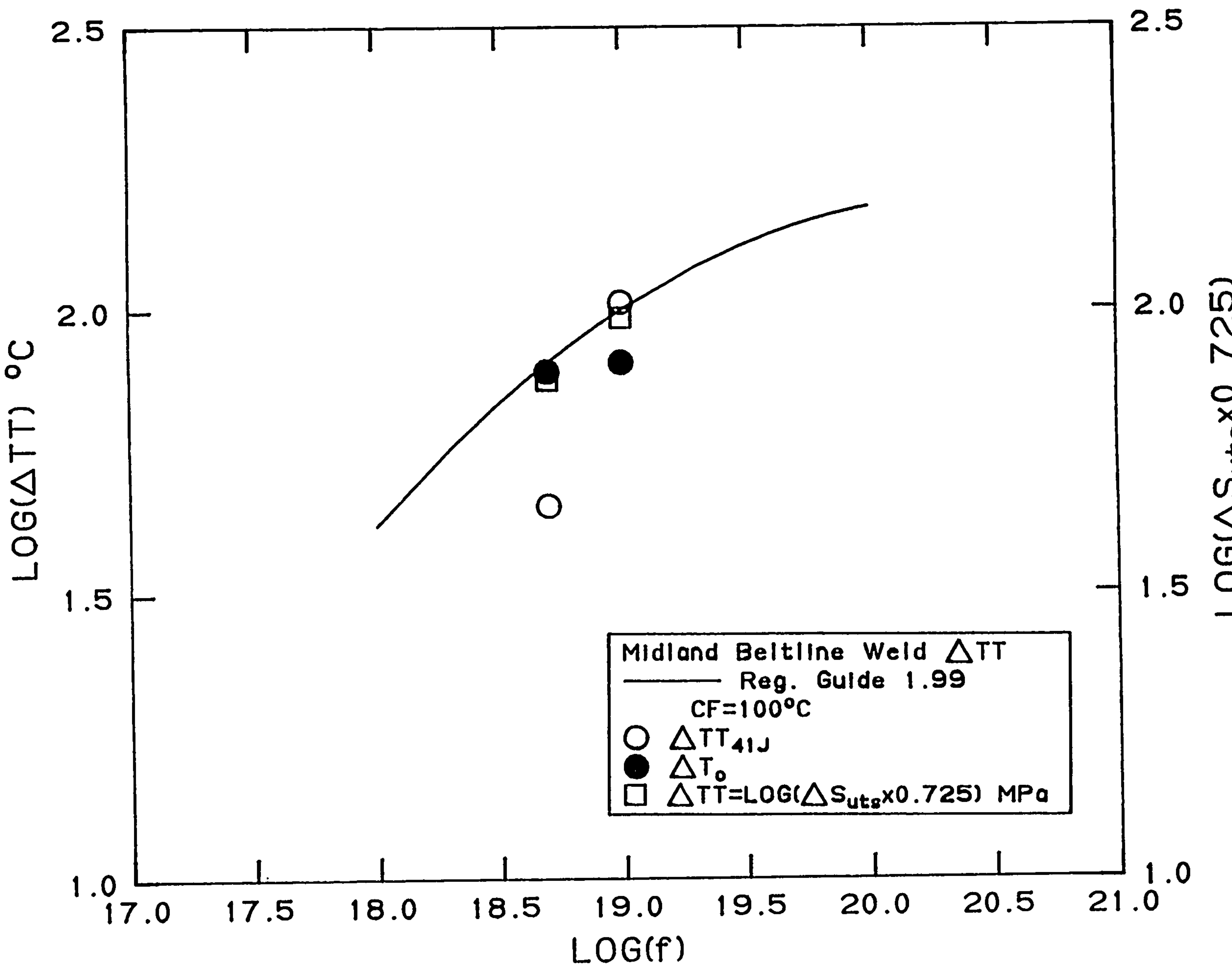

Figure 4.2. Comparison of the experimentally determined ΔTT shift with the *Regulatory Guide 1.99* prediction for the beltline weld metal.

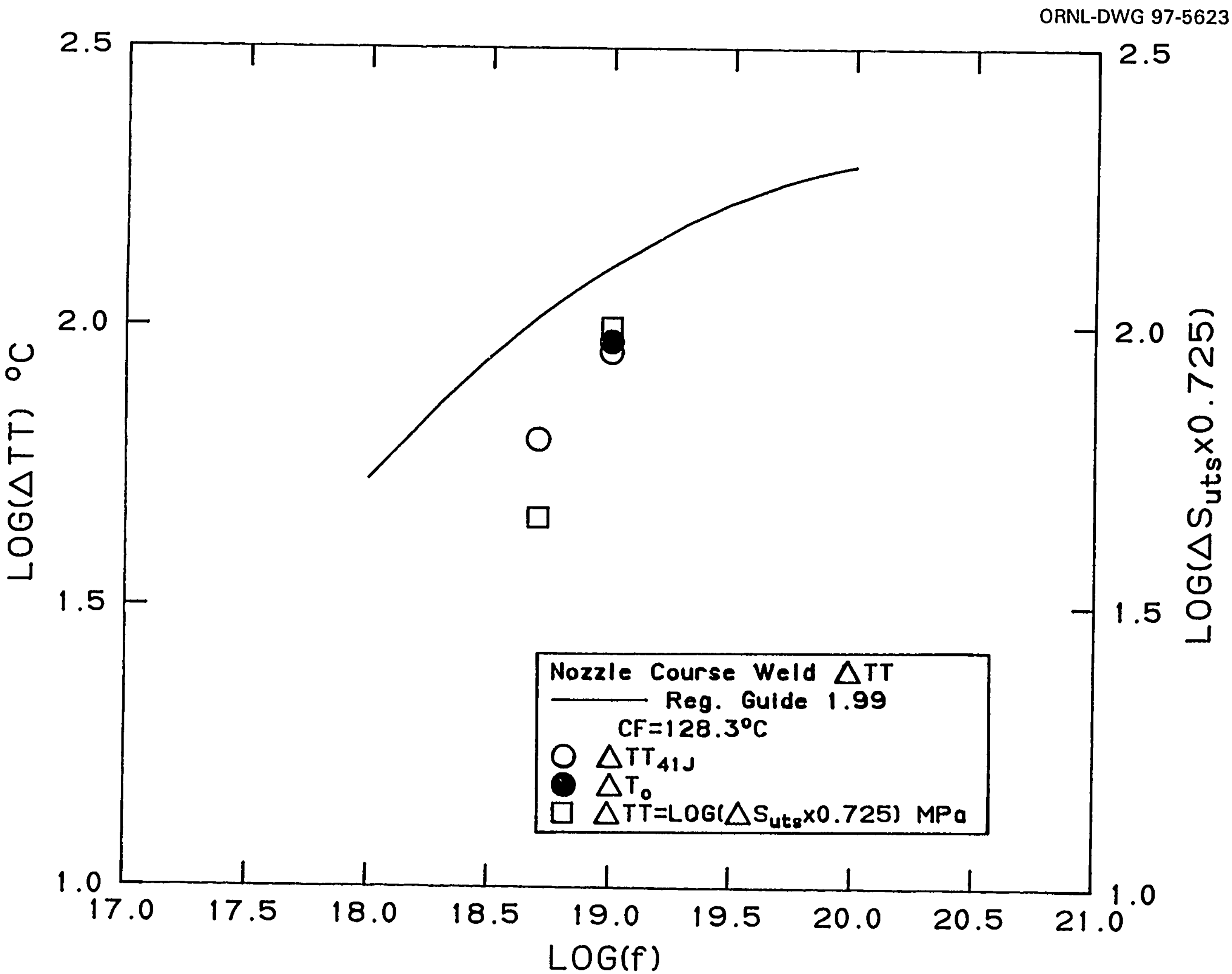

Figure 4.3. Comparison of the experimentally determined ΔTT shift with the *Regulatory Guide 1.99* prediction for the nozzle course weld metal.

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*Available in public technical libraries.

5. IRRADIATION EFFECTS ON HEAT-AFFECTED ZONES AND PLATE MATERIAL

D. E. McCabe

The task title for the Eleventh Irradiation Series has been changed to reflect some new additional objectives. Emphasis is now being given to determining if there is a potential problem with thermal embrittlement in RPV steels. An annealing experiment on laboratory heats made with steels having typical pressure vessel chemical compositions was conducted by AEA Technology, Harwell, United Kingdom. The experiment showed quite clearly that there can be grain boundary embrittlement in RPV steels given large prior austenite grain size and high phosphorus on the order of 0.017 wt %. The first task in this project will be to reexamine the AEA Technology heat treatments using five commercially made RPV steels, representing A 302 grade B, A 533 grade B, A 508 class 2, and two modified A 302 grade B heats. The phosphorus content covers the range typical of commercial RPV production heats. The AEA Technology heat treatment exaggerates the problem by creating a microstructure that optimizes embrittlement sensitivity. This is a recognized method for an experimental objective of screening materials.

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

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

This report covers only the first task previously mentioned. The five commercial heats of RPV steel were obtained as archival materials left over from several previous projects. Each material was sectioned into four blocks of a convenient size for heat treatment, nominally 51 × 56 × 127 mm (2 × 2.2 × 5 in.). The objective was to austenitize all materials in simulation of the AEA Technology heat treatments. All blocks of the five materials were austenitized at 1204°C (2200°F) with a 30-min soak time followed by an oil quench. Then all were given postweld heat treatments at 615°C (1140°F) for 24 h and again followed by an oil quench. The resulting prior austenite grain size was determined to be a mix of 0 to 00 ASTM standard grains.

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

A second austenitization treatment to be applied is a direct simulation of the thermal cycle of HAZ material. In this case, the specimens are 14-mm-diam (0.564-in.) rods. The Gleeble (electric resistance heating) system will provide the thermal cycle. The austenitizing temperature will be 1260°C (2300°F), and the entire cycle will be completed within 100 s. About 20 CVN specimens will be machined from these rods. The target ASTM grain size will be between 4 and 5, which is typical for local brittle zones. All will be postweld heat-treated at 615°C (1140°F) for 24 h.

6. ANNEALING EFFECTS IN LOW UPPER-SHELF WELDS

S. K. Iskander

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

6.1 Testing of Irradiated and Annealed Specimens (M. A. Sokolov, S. K. Iskander, R. L. Swain, E. T. Manneschmidt, and J. J. Henry)

Testing and data analysis has been performed on 12 irradiated and annealed 1-in. compact [1T C(T)] specimens. Preliminary results indicate that the shift in the master curve is approximately equal to that of the 41-J CVN transition temperature. Six Midland beltline weld specimens were annealed at 343°C, and six Midland nozzle course weld specimens annealed at 454°C, both for 168 h. Based on previous CVN results, test temperatures were selected to result in median initiation fracture toughness values (K_{Jc}) of about 100 Mpa$\sqrt{m}$.

To shed some light on the reason for "overrecovery" of USE that has been observed when CVN specimens are annealed at 454°C (850°F), sets of three to four unirradiated specimens were aged for 168 h at various temperatures. Overrecovery refers to the increase in USE of aged or annealed specimens with respect to the unaged (or annealed) value of the USE. Two slightly different specimen sizes were used, a so-called undersize specimen, in which the width (W) of the specimen is 95% of that of the standard 10-mm specimen and has a notch depth (a) of such that a/W = 0.191 rather than the standard a/W = 0.200. The smaller size was dictated by the space available in the Fifth Irradiation Series large specimen capsules. To account for the different sizes of the specimens, the increase in USE was referenced to the unaged USE. At a temperature of 300°C, no significant effect of aging was observed on the USE. At temperatures in the range of 375 to 525°C, the USE increased from about 117 to 120% over the unaged USE. Of interest is the large increase of 176% in USE observed at 607°C, which is also the temperature at which these welds were postweld heat-treated for 40 h when they were fabricated.

Design began on three containers for the previously irradiated and annealed Charpy specimens as well as wire dosimeters. Designing has also begun on retrievable fission dosimetric set containers. The specimens and dosimeters will be irradiated at the University of Michigan Ford Nuclear Reactor (FNR) in eight "cells" of the University of California, Santa Barbara (UCSB), capsule. The CVN specimens are HSSI weld 73W and have been previously irradiated and annealed at 454°C and will be reirradiated to three different fluence levels. This will provide data on the rate of reembrittlement for weld 73W. For the relatively high fluence, one package will be placed in positions D1/D2 of the capsule, with the intermediate and low fluences in positions G1/G2 and H1/H2, respectively. The "spillover" caused by the packages being longer than 2 in. could be accommodated in positions I1/I2.

The following papers were presented at the ASTM 18th International Symposium on Effects of Radiation on Materials, held in Hyannis, Massachusetts, on June 25–27, 1996:

S. K. Iskander, D. E. McCabe, M. A. Sokolov, R. K. Nanstad, J. T. Hutton, and D. L. Thomas, "Comparison Between Different Measures of Experimentally Determined Crack-Arrest Toughness"

S. K. Iskander, M. A. Sokolov, and R. K. Nanstad, "Comparison of Different Experimental and Analytical Measures of the Thermal Annealing Response of Neutron-Irradiated RPV Steels"

M. A. Sokolov, S. Spooner, G. R. Odette, B. D. Wirth, and G. E. Lucas, "SANS Study of High-Copper RPV Welds in Irradiated-and-Annealed Conditions"

6.2 Design, Fabrication, and Installation of New Irradiation Facilities

(D. W. Heatherly, C. A. Baldwin, D. W. Sparks, and G. E. Giles, Jr.)

Work on the two HSSI IAR facilities has progressed significantly. The facilities, along with the temperature verification capsule, have been sandblasted and polished. The two facilities were leak-checked and delivered to the Instrumentation and Controls (I&C) Division for installation of amphenol connectors. The temperature test capsule has also been sealed and leak-checked. The facilities and temperature capsule await stress relief to determine that no distortion will occur during operation. The facilities must then be checked at annealing temperatures.

Because of personnel constraints, a "dummy" surrogate facility was fabricated instead of the one now undergoing fitting and testing; the dummy does not contain heating elements or temperature controls. Drawings for this surrogate facility have been prepared, purchase requisitions have been placed, and fabrication is under way. The surrogate facility would relieve the Engineering Technology Division Irradiation Engineering Group (IEG) from working on the UCSB and IAR facility at the same time. The IEG located surplus stainless steel at the Y-12 Plant for the fabrication of the dummy facility; purchase of the required steel from a vendor would have delayed fabrication.

Design of the capsule (preirradiated specimen capsule) for reirradiating irradiated and annealed material is progressing. About 20 drawings have been prepared and will be reviewed and necessary changes made. Planning of the specimen complements for the first reirradiation capsules is under way; these will contain specimens that were previously irradiated and annealed. The target fluence for the first capsule is 0.5×10^{19} n/cm^2 (>1 MeV).

An I&C engineer traveled to the FNR at the University of Michigan to survey changes that have occurred at the reactor site that will affect electrical connections and to determine power requirements. As a result of the changes at the FNR, the HSSI IAR J-box needs modification. The computer that was at the FNR site has been brought back to ORNL and rebuilt with a new mother board, disk drives, memory, and power supply to accommodate new Windows-based software. Preparations that include a vessel and a heat exchanger have also been progressing to test the IAR facilities as mentioned previously.

7. Microstructural Analysis of Radiation Effects

R. E. Stoller and P. M. Rice

7.1 Primary Damage Simulation by Molecular Dynamics

The method of molecular dynamics (MD) has been used extensively to investigate the evolution of displacement cascades,[1–4] and the results of these cascade simulations have been used to develop improved estimates for the source terms used in the kinetic embrittlement model.[5,6] However, the highest energy results obtained previously were characteristic of collisions with neutrons somewhat below 1 MeV. A recent computer upgrade supported by the Office of Basic Energy Sciences of the Department of Energy has permitted the extension of these earlier cascade simulations to higher energies. Six 40-keV cascades were completed in simulations involving 1,024,000 atoms. Each such cascade required about three weeks of central processing unit time. These 40-keV simulations are the highest energy ever reported and correspond to the average iron primary knock-on atom in a collision with a 1.8-MeV neutron. Thus, the cascade simulations have now provided information representative of most of the fast neutron spectrum since ~90% of the neutron spectrum at a pressurized-water RPV is below 1.8 MeV.

The primary damage parameters extracted from the MD cascade simulations are (1) the total number of point defects that escape in-cascade recombination and (2) the fraction of the surviving point defects that are contained in clusters. Depending on the energy of the cascade, in-cascade recombination is typically complete after 10 to 20 ps. The number of surviving point defects can be conveniently expressed as a fraction of the number calculated using the standard Norgett-Robinson-Torrens (NRT) displacement model;[7] this fraction is here termed the cascade efficiency. The results discussed here are for simulations at 100 K, but little temperature dependence has been observed in the primary damage parameters for temperatures from 100 to 900 K.[2]

As shown in Figure 7.1, the values of the cascade efficiency for energies up to 20 keV indicate a smooth decrease with increasing cascade energy. The new results at 40 keV deviate from this trend, with the value increasing nearly to the that obtained at 10 keV. This is a result of what is termed subcascade formation at the higher energies. Figure 7.2 compares three cascades, one each at 10, 20, and 40 keV. The tendency of the cascade geometry to begin to break up into separate subcascades can clearly be seen. The 40-keV cascade gives the appearance of being composed of several regions that are similar to individual, lower-energy cascades. One impact of the cascade efficiency not continuing to decrease at the highest energies is reduction of the effect of differences in the neutron energy spectrum between different irradiation environments.

The degree of in-cascade point-defect clustering is important because such clusters "seed" the nucleation process and increase the overall level of radiation-induced hardening. Figure 7.3 compares the fraction of surviving vacancies and interstitials that are found in clusters as a function of the cascade energy. In the case of interstitial clustering, an average value of about 50% is observed for all energies above 5 keV. The results are somewhat more complicated for vacancies. Previous analysis of MD cascades in iron had indicated that little in-cascade vacancy clustering occurred.[1–3] This result arises when a strict clustering criterion is applied, namely, that the vacancies must occupy next-nearest-neighbor (nn) lattice positions to be considered clustered. The curve labeled nn in Figure 7.3 reflects this criterion. The value of about 10% is much lower than the interstitial clustering fraction obtained when the criterion is applied.

ORNL-DWG 97-5624

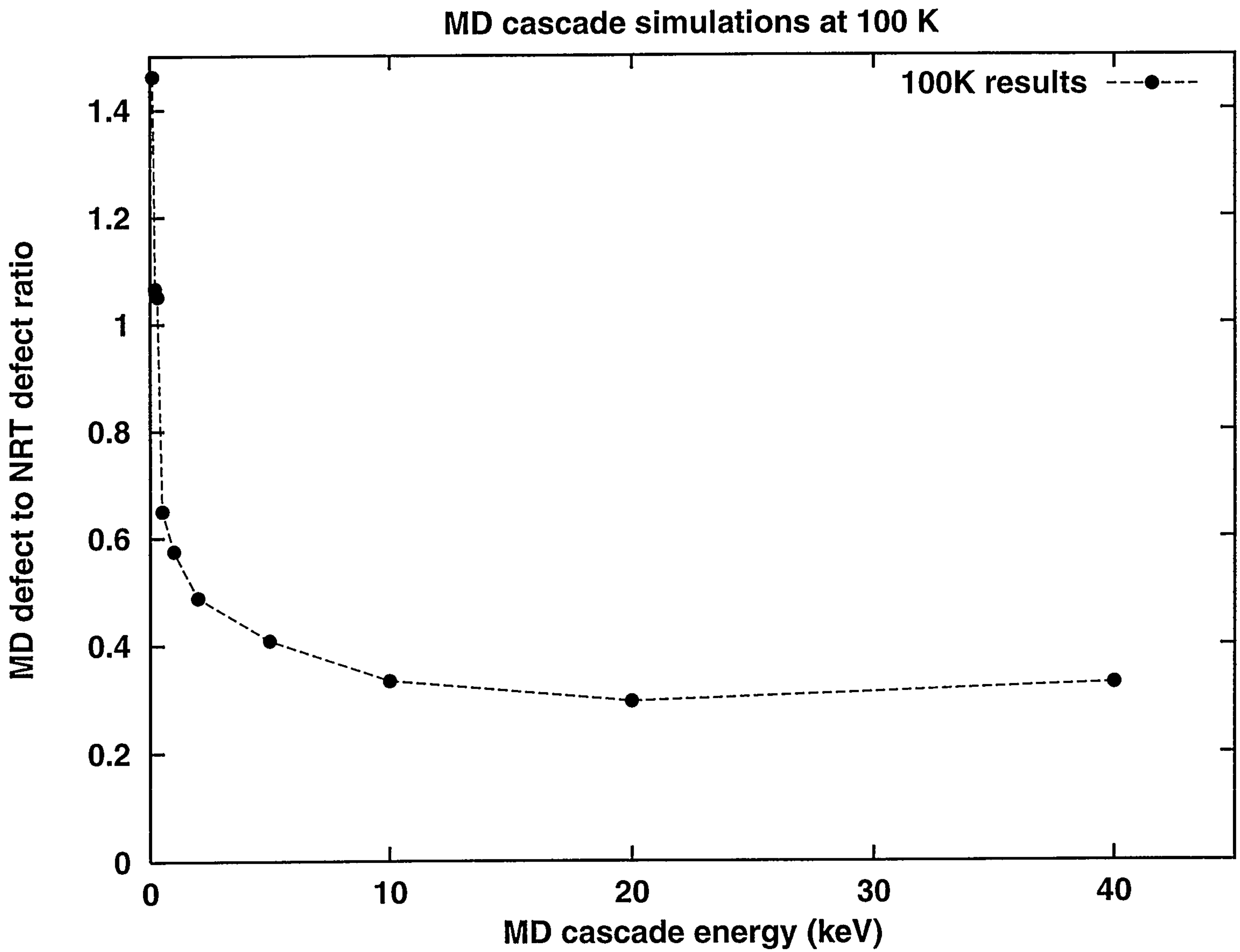

Figure 7.1. Ratio of surviving MD point defects to those calculated using the standard NRT displacement model.[7]

ORNL-DWG 97-5625

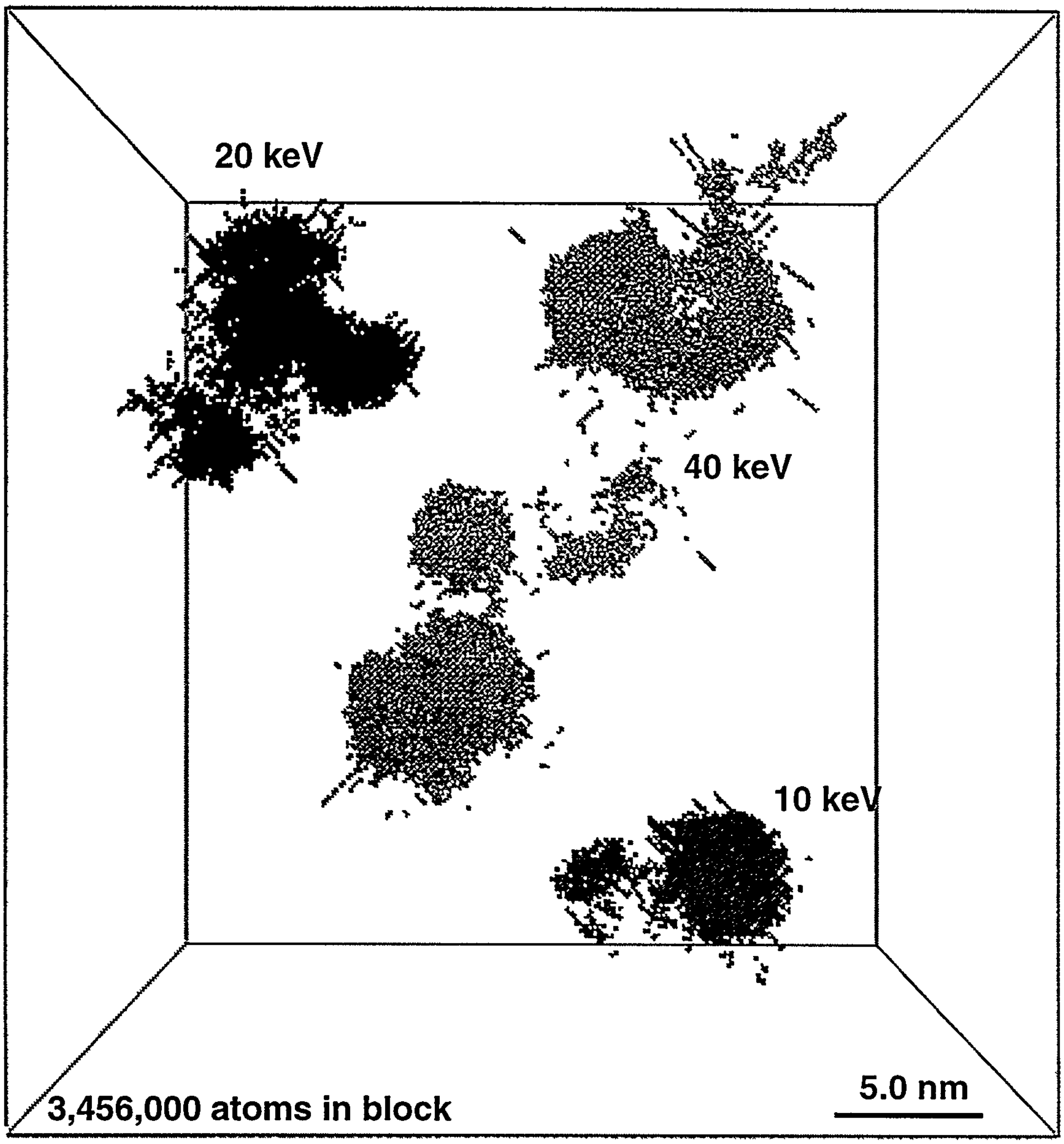

Figure 7.2. Comparison of cascade size and morphology for 10-, 20-, and 40-keV cascades at 100 K.

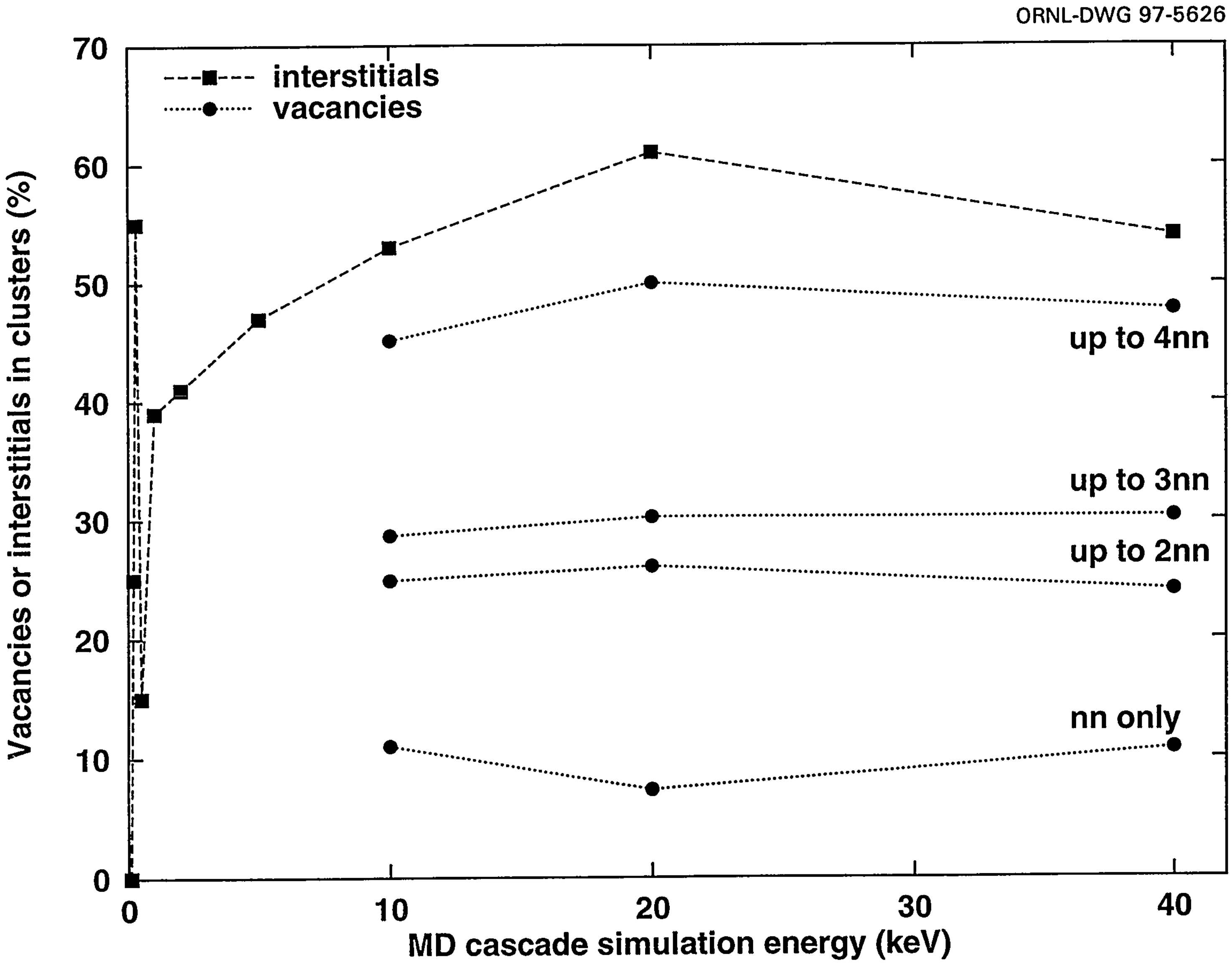

Figure 7.3. Fraction of surviving interstitials and vacancies found in clusters; vacancy cluster curves reflect different clustering criteria (see text).

However, vacancies are essentially immobile over the time span that is accessible by MD simulations, typically less that 100 to 200 ps. Therefore, the vacancy distributions in all the high-energy cascades were reanalyzed to look for somewhat longer-range spatial correlation. It was found that positions up to the fourth nearest neighbor appeared to be correlated, and the curves in Figure 7.3 labeled 2nn, 3nn, and 4nn reflect the clustering fraction if the criterion is relaxed to include up to second, third, or fourth nearest neighbors. These values can be considered to represent the potential for vacancy clustering when enough time has passed to permit a few vacancy jumps in the lattice. These results are significant because in-cascade vacancy clustering has long been assumed to be an important mechanism in embrittlement models,[5,6,8] and the apparent lack of vacancy clustering in the MD simulations was puzzling.

7.2 Comparison of Mechanical Property and Microstructural Changes

The results of these ion-irradiation experiments are discussed in detail in the last semiannual progress report covering October 1, 1995, to March 31, 1996. The only remaining component of this task involved obtaining quantitative measurements of the radiation-induced defect size distributions that could be used to compute a predicted hardening value for comparison with the measured values. This was accomplished by measuring the visible defects in a series of high-magnification transmission electron micrographs. Specimen doses were in the range of 0.05 to 0.1 displacements per atom (dpa), corresponding to fast neutron fluences of 3 to 6×10^{19} n/cm^2.

Typical defect size distributions are shown in Figure 7.4 for 2 Fe-N binary alloys. The defects remained quite small at doses up to 0.1 dpa; the diameter at the peak in the distribution was typically less than 10 nm. The lower cutoff in the distribution at 2 nm appears to represent a visibility limit in the TEM rather than an accurate reflection of the size of the defects present. These ferritic model alloys (and commercial pressure vessel steels) have two characteristics that limit TEM visibility. The first is that they are ferromagnetic. As such, they interact with the electromagnetic fields that form the lenses in a TEM and the electron beam itself. This leads to what can be considered virtual flaws in the lenses that require constant correction when the specimen is moved in the microscope. Because identification of such small defects requires extensive tilting to specific orientations, imaging the defects is inhibited. A second characteristic that limits TEM visibility is that these materials oxidize easily, and even very thin oxide layers can obscure small defects in the specimens. Thus, in spite of extensive effort, these problems proved to be somewhat intractable, and defects with diameters smaller than 2 nm could not be reliably imaged. As a result, only qualitative agreement between the calculated and measured hardnesses could be achieved. Our initial irradiation doses in this task were higher, up to 2 dpa. In this case, the defects were somewhat larger and the measured hardening could be accounted for if the small dislocation loops had an effective barrier strength of 0.21.[9] Applying this same factor to the distributions measured at these lower doses leads to an underprediction of the measured hardening, which is consistent with hardening by defects with sizes below the 2-nm visibility limit.

ORNL-DWG 97-5627

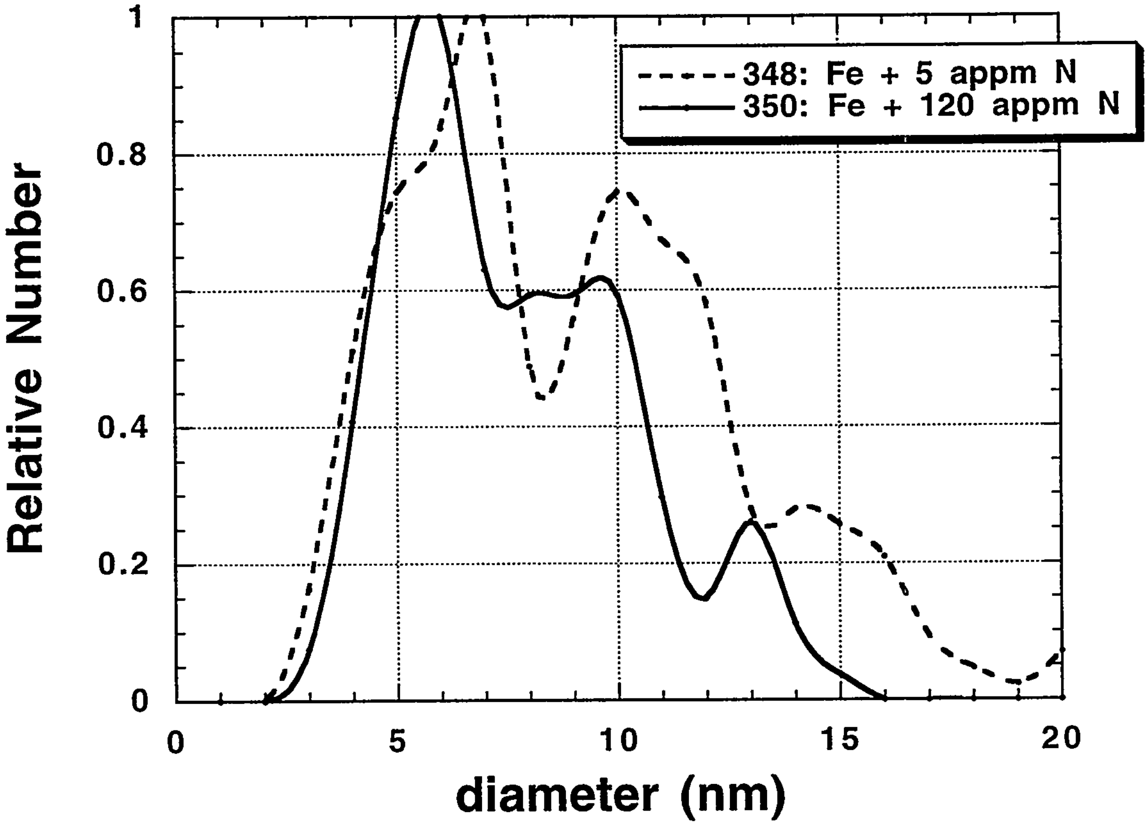

Figure 7.4. Comparison of defect size distributions observed at 0.1 dpa in ion-irradiated Fe-5 atomic parts per million (appm) N (alloy VM348) and Fe-120 appm N (alloy VM350).

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*Available in public technical libraries.

8. IN-SERVICE IRRADIATED AND AGED MATERIAL EVALUATIONS

S. K. Iskander, D. J. Alexander, R. K. Nanstad, J. J. Henry, Jr., and L. Creech

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. Four activities are currently in progress: (1) establishing a machining capability for contaminated or activated materials by completing procurement and installation of a computer-based milling machine in a hot cell; (2) machining and testing specimens from cladding materials removed from the Gundremmingen reactor to establish their fracture properties; (3) preparing an interpretive report on the effects of neutron irradiation on cladding; and (4) continuing the evaluation of long-term aging of austenitic structural stainless steel weld metal by metallurgically examining and testing specimens aged at 288 and 343°C and reporting the results, as well as by continuing the aging of the stainless steel cladding toward a total time of 50,000 h.

8.1 Remotely Operated Machining Center (S. K. Iskander, L. Creech, L. J. Turner, and R. K. Nanstad)

During this reporting period, all work on the setup of the computer numerically controlled (CNC), remotely operated machining center, with a mill, a saw, as well as preparation of the hot cell, was put on hold pending the outcome of a funding review. Several meetings were held between HSSI Program staff and hot cell staff to review the present status and funds required to complete installation. A significant effort was expended to determine as accurately as possible the funds required to complete installation. This necessitated an accurate work breakdown structure (WBS) of the remaining work on the saw, milling machine, and hot cells. The hot cell destined for installation of the CNC machine is contaminated from legacy work and must be cleaned. Since the contamination is from work performed many years ago, NRC and HSSI staff met with ORNL management to request that cleanup of the contamination be performed mostly with non-NRC funds. Hot cell decontamination cost estimates were prepared and submitted to ORNL management with a request for funds. Preliminary time estimates to decontaminate the hot cell are 12 to 13 weeks.

A detailed WBS was completed, and time, schedule, and cost estimates were obtained. The Plant and Equipment (P&E) Division provided assurance that the costs were accurate, and no cost overruns are foreseen.

To determine the viability of completing the machining center, the NRC requested that ORNL obtain cost estimates for machining Charpy, tensile, and 1/2T-size compact specimens from irradiated materials. The estimates were prepared and submitted to the NRC. Some of the estimates are based on machining the specimens at ORNL after the CNC machine is installed, and others are informal "order-of-magnitude" estimates provided by various other installations. The NRC has determined that such a center is viable and has instructed ORNL to proceed with installation.

The project WBS and time allocations for installation of the remotely operable milling machine and saw, as well as installation of a crane inside the hot cell, have been mapped onto a list of 13 maintenance job requests to allow efficient tracking of progress vs costs. The work was partitioned with considerations for schedule dependencies and equipment availability. The P&E Division has reviewed the list and has concurred. The assembly drawing bills of material have been reviewed and organized for an inventory check. The final major hardware items (a screw-driven slide and air cylinder) for the saw have been ordered.

9. JPDR VESSEL STEEL EXAMINATIONS

S. K. Iskander and R. K. Nanstad

The objectives of the Japan Atomic Energy Research Institute (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: (1) examination of material from the JPDR vessel in conjunction with a reevaluation of its exposure conditions, (2) new test reactor irradiations of archival and similar materials, and (3) 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 region 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 preliminary planning for test specimen machining was completed and costs of machining the specimens from base and weld metal from each of four materials were estimated. There are four trepans each of the weld and base metal from both the corebelt region and the remote region. All 16 trepans, 87 mm in diameter and 78 mm long, have stainless steel cladding, from which most of the radioactivity of the material originates.

It appears that the machining of the specimens from the remote area could proceed in a "hot shop" that contains machine tools with the power required for high feed rates and gang machining. Furthermore, in a hot shop, closer contact between the machinist (using tongs to handle the material) and the tooling is permitted, which leads to more efficient operations. This could result in considerable savings of time and money compared to machining in a hot cell with limited capabilities, where all handling is performed by manipulators.

10. FRACTURE TOUGHNESS CURVE SHIFT METHOD

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

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 caused by neutron irradiation, thermal aging, or both; (2) evaluation of methods for indexing fracture toughness curves to values that can be deduced from material surveillance programs required under the *Code of Federal Regulations* (10 CFR 50), Appendix H; (3) participation in the pertinent ASME *Boiler and Pressure Vessel Code*, Sect. XI, ASTM E-8 and E-10 committees to facilitate obtaining data and disseminating the results of the research; (4) interaction with other researchers in the national and international technical community addressing similar problems; and (5) frequent interaction, telephone conversations, and detailed technical meetings with the NRC staff to ensure that the results of the research and proposed changes to the accepted methods for shifting the fracture toughness curves reflect staff assessments of the regulatory issues.

10.1 Comparison of Irradiation-Inducted Charpy and Fracture Toughness Curve Shifts

A database of Charpy impact and fracture toughness data for RPV materials that have been tested in the unirradiated and irradiated conditions is being assembled. Currently, there are 39 data sets for welds and 50 for base metals. The majority of these base metal data are for plates, A 533 grade B and A 302 grade B; there are only seven data points for forgings. Each data set consists of Charpy impact data over a range of test temperatures to allow for curve fitting with a hyperbolic tangent (with fixed lower shelf) function, and cleavage fracture toughness data in sufficient quantity to allow for determination of the reference fracture toughness temperature, T_{100}, by Weibull statistic/master curve analysis. The T_{100} is the temperature at which a median fracture toughness equals to 100 MPa$\sqrt{}$m. Preliminary analyses have been performed with these data sets to investigate potential correlations between the radiation-induced temperature shifts from the Charpy impact tests and fracture toughness tests. The T_{100} temperature shift is being compared with various transition temperatures from the curve fit to the Charpy impact data. Preliminary results show some differences between the weld metal and base metal data sets. For weld metals, on average, the Charpy transition temperature shift at 41 J is the same as the shift of fracture toughness with 95% confidence intervals of about ±30°C. For base metals, on average, the fracture toughness shift is 12°C greater than the Charpy 41-J temperature shift with 95% confidence intervals of about ±35°C. The preliminary analysis was performed regardless of orientation, level of USEs, change in slope of the Charpy curve, etc. The analysis relies on the assumption that irradiation would not change the shape of the fracture toughness master curve as it does the Charpy impact transition curve. These issues will be addressed in future evaluations of the database.

10.2 Weibull Statistic/Master Curve Analysis of the ASME K_{Ic} Database

The objective of this evaluation is to apply the Weibull statistic/master curve analysis procedure to the linear-elastic K_{Ic} database that has been used to support the ASME lower-bound curve. All 174 K_{Ic} data on 11 materials from the Electric Power Research Institute database[1,2] were size-adjusted to 1T-specimen equivalence, and then T_{100} values for all materials were determined. Figure 10.1 shows the linear-elastic K_{Ic} data in the new fracture toughness coordinates. This is a plot of K_{Ic} values adjusted to 1T-size equivalence by the weakest-link theory equation vs temperature, T, normalized to the reference fracture toughness temperature, T_{100}. The individual T_{100} values for each material are presented in the same figure. In Figure 10.1, the master curve is the line designated "50%." For comparison, lower and upper curves that correspond to some lower- and upper-cumulative probability levels are also shown in Figure 10.1.

The following observations can be made based on the results in Figure 10.1. In Figure 10.1 the linear-elastic K_{Ic} data from different materials form a typical fracture toughness data trend, as if the data were a large data set from one material. The master curve represents well the median trend of this database, while the scatter of data is characterized by lower- and upper-bound curves. These results provide additional support to the statistical nature of brittle fracture and to the importance of using a statistical method to describe probabilities of such events.

Thus, different materials can be compared based on the fracture toughness level by means of the reference temperature T_{100} rather than the empirical correlation to RT_{NDT} values. Figure 10.2 shows a comparison of RT_{NDT} and T_{100} reference temperatures of all materials in the linear-elastic database. The RT_{NDT} values are as reported in Reference 1; drop-weight NDT values were not reported. There is little correlation between these two reference temperatures for the materials analyzed. The T_{100} temperatures spread evenly among steels in a temperature range from –150 to 30°F (–101 to –1°C). The RT_{NDT} values tend to form two clusters at about 0 and 50°F (–18 to 10°C). For 10 of the 11 materials shown in this database, RT_{NDT} is higher than T_{100}, suggesting that normalizing of material fracture toughness by RT_{NDT} may result in an underestimation of actual fracture toughness.

The plot in Figure 10.3(a) is the current representation of the ASME K_{Ic} database and lower-bound curve by normalizing to RT_{NDT}. The plot in Figure 10.3(b) is the same K_{Ic} database (not size adjusted) normalized by reference fracture toughness temperature T_{100} from the present work. These plots are shown with all of the ASME K_{Ic} data, including those below –160°F (–89°C). The first observation is that the ASME K_{Ic} curve is only a fitting function for the data in the normalized temperature range ($T - RT_{NDT}$) of about –130°F (–72°) and higher; hence, it is not a true lower bound to all of the data.[2] Second, in the transition region the shape of this curve was dictated entirely by data from one material, namely HSST Plate 02. As can be seen in Figure 10.2, for HSST Plate 02, as well as for HSST Plates 01 and 03, RT_{NDT} and T_{100} values are more or less comparable. For the rest of the materials, RT_{NDT} is greater than T_{100}. Hence, having an actual toughness comparable with HSST Plate 02, they do not contribute to construction of the transition region part of the ASME K_{Ic} lower-bound curve because they are shifted to the left too much. In Figure 10.3(b), fracture toughness values of the same materials are compared relative to the fracture toughness at 100 MPa√m (T_{100}). The master curve from Figure 10.1 is not shown in Figure 10.3 because the data shown are not size adjusted. However, lower tolerance bounds to the 1T master curve are presented in Figure 10.3(b). At low probability levels, the statistical size effect on tolerance bound is not so dominant; hence, as can be seen in Figure 10.3, tolerance bounds to the 1T-size master curve serve as lower bounds relatively well even for nonadjusted data.

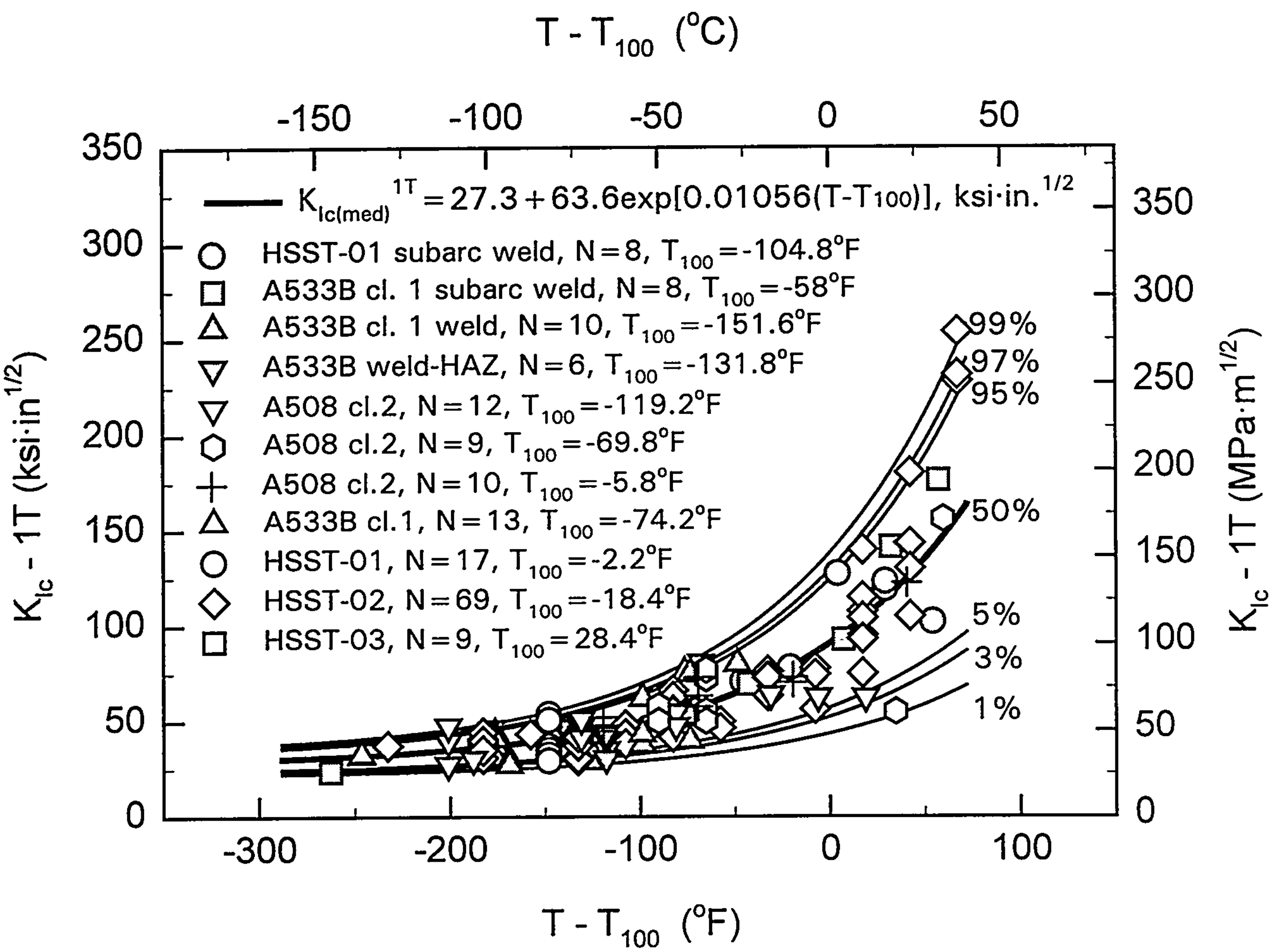

Figure 10.1. **The ASME K_{Ic} data statistically adjusted to 1T-size equivalence and described by the master curve and corresponding upper- and lower-tolerance bounds.**

ORNL-DWG 97-5629

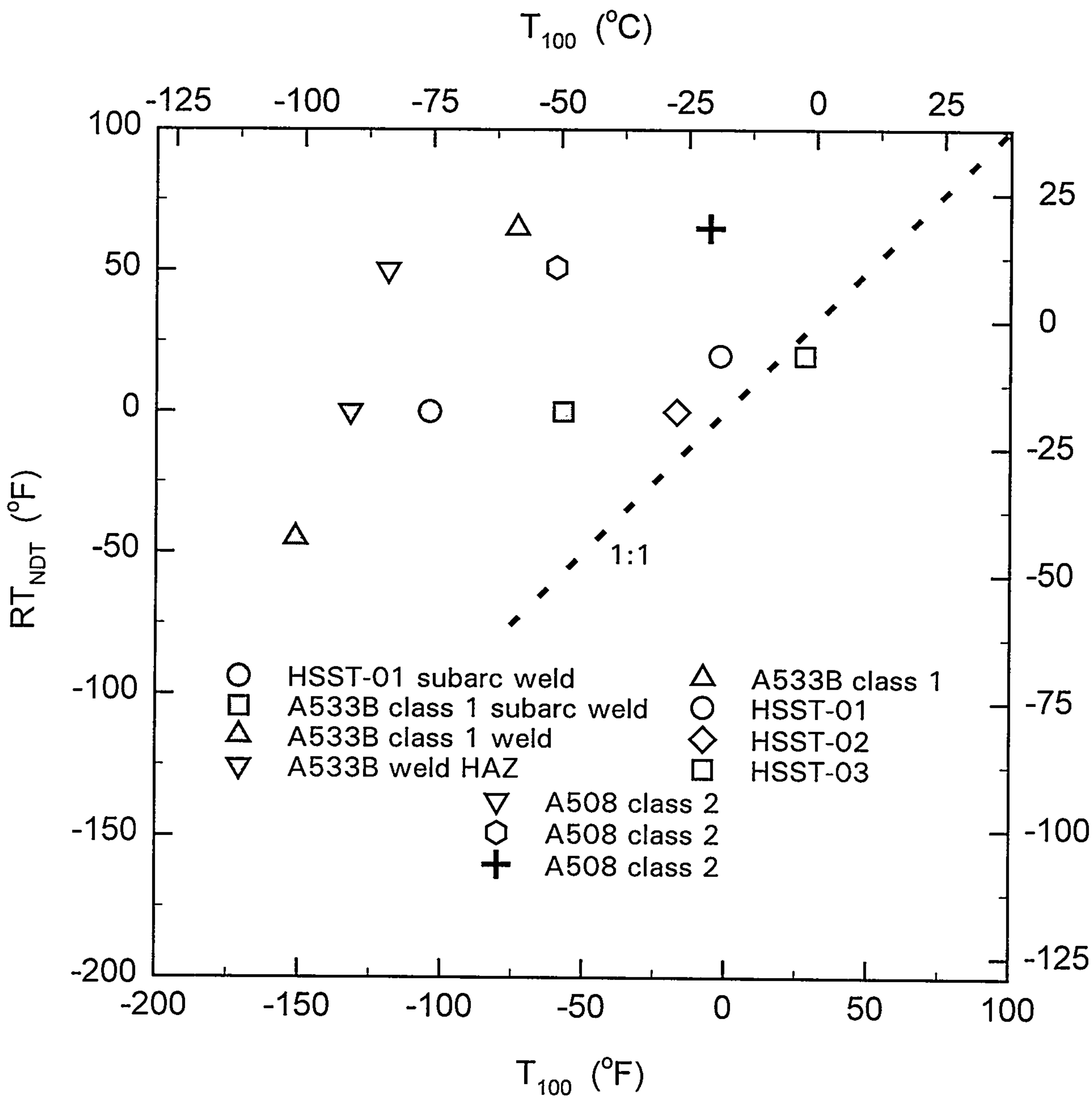

Figure 10.2. Comparison of RT_{NDT} and T_{100} for materials in the ASME K_{Ic} database.

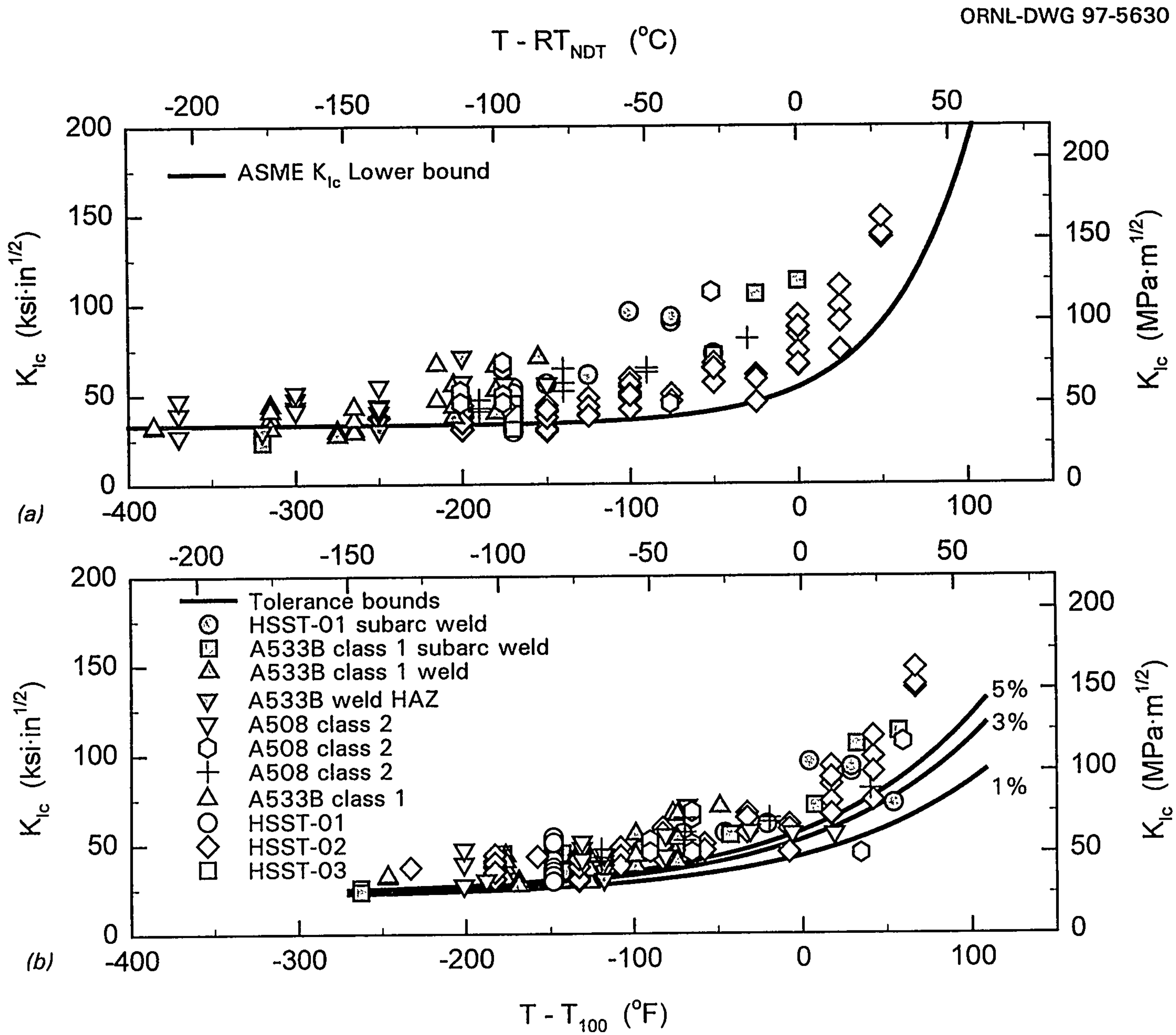

Figure 10.3 The ASME K_{Ic} database vs temperature normalized by (a) RT_{NDT} and (b) T_{100}.

References

1. T. U. Marsten, Ed., Electric Power Research Institute, *Flaw Evaluation Procedures, Background and Application of ASME Section XI, Appendix A*, EPRI NP-719-SR, August 1978.*

2. R. K. Nanstad, J. A. Keeney, and D. E. McCabe, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *Preliminary Review of the Bases for the K_{Ic} Curve in the ASME Code*, USNRC Report ORNL/NRC/LTR-93/15, July 1993.†

*Available in public technical libraries.

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

11. SPECIAL TECHNICAL ASSISTANCE

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

The purpose of this task is to perform various special analytical and experimental investigations to support the NRC in resolving regulatory research issues related to irradiation effects on materials. This task currently addresses two major areas: (1) providing technical expertise and assistance in the review of national codes and standards and (2) experimental evaluations of test specimens and practices and material properties.

11.1 Estimation of Crack-Arrest Toughness from Instrumented Charpy Impact Tests (S. K. Iskander, R. K. Nanstad, M. A. Sokolov, D. E. McCabe, J. T. Hutton, and D. L. Thomas)

In the fracture mechanics integrity analysis of RPVs, the initiation and arrest fracture toughness curves, the K_{Ic} and K_{Ia} curves, respectively, as described in Sect. XI of the ASME *Boiler and Pressure Vessel Code*,[1] are often used. These curves are used also for the normal operation of RPVs. The effects of neutron irradiation on toughness are accounted for by shifting the curves upward in temperature without changing the shape by an amount equal to the temperature shift of the CVN impact energy curve at the 41-J level. Such a procedure implies that the shifts in the fracture toughness curves are the same as those of the CVN 41-J shift and that irradiation does not change the shapes of the fracture toughness curves. Moreover, according to current procedures, the 34-K margin between the unirradiated ASME curves is assumed to be preserved after irradiation to the same fluence. There are, however, theoretical reasons and experimental evidence for the temperature difference between the two curves to decrease with radiation embrittlement. Crack-arrest testing of irradiated specimens is both difficult and expensive; thus, it is worthwhile to explore a proxy for the determination of crack-arrest toughness. One possible approach in this regard would be to use information already available from instrumented CVN testing. A preliminary study has been conducted to explore that approach and a paper was prepared for and presented at the ASTM 18th International Symposium on Effects of Radiation on Materials in June 1996. The following paragraph is a summary of the abstract and selected figures from that paper.

The objective of this investigation is to estimate the crack-arrest toughness, particularly of irradiated materials, from voltage vs time output of an instrumented tup during a test on a CVN specimen. This voltage vs time trace (which can be converted to force vs displacement), shown in Figure 11.1, displays events during fracture of the specimen. Various stages of the fracture process can be identified on the trace, including an arrest point indicating arrest of brittle fracture. The force at arrest, F_a, vs test temperature, T, relationship is examined to explore possible relationships to other experimental measures of crack-arrest toughness such as the DWT NDT, or crack-arrest toughness, K_a. For a wide range of weld and plate materials, the temperature at which $F_a = 2.45$ kN correlates with NDT with a standard deviation, σ, of about 11 K (see Figure 11.2). Excluding the so-called "low USE" welds from the analysis resulted in $F_a = 4.12$ kN and $\sigma = 6.6$ K (see Figure 11.3). As shown in Figure 11.4, the estimates of the correlation of the temperature for $F_a = 7.4$ kN with the temperature at 100 MPa·√m level for a mean ASME type K_{Ia} curve through crack-arrest toughness values shows that prediction of conservative values of K_a are possible.

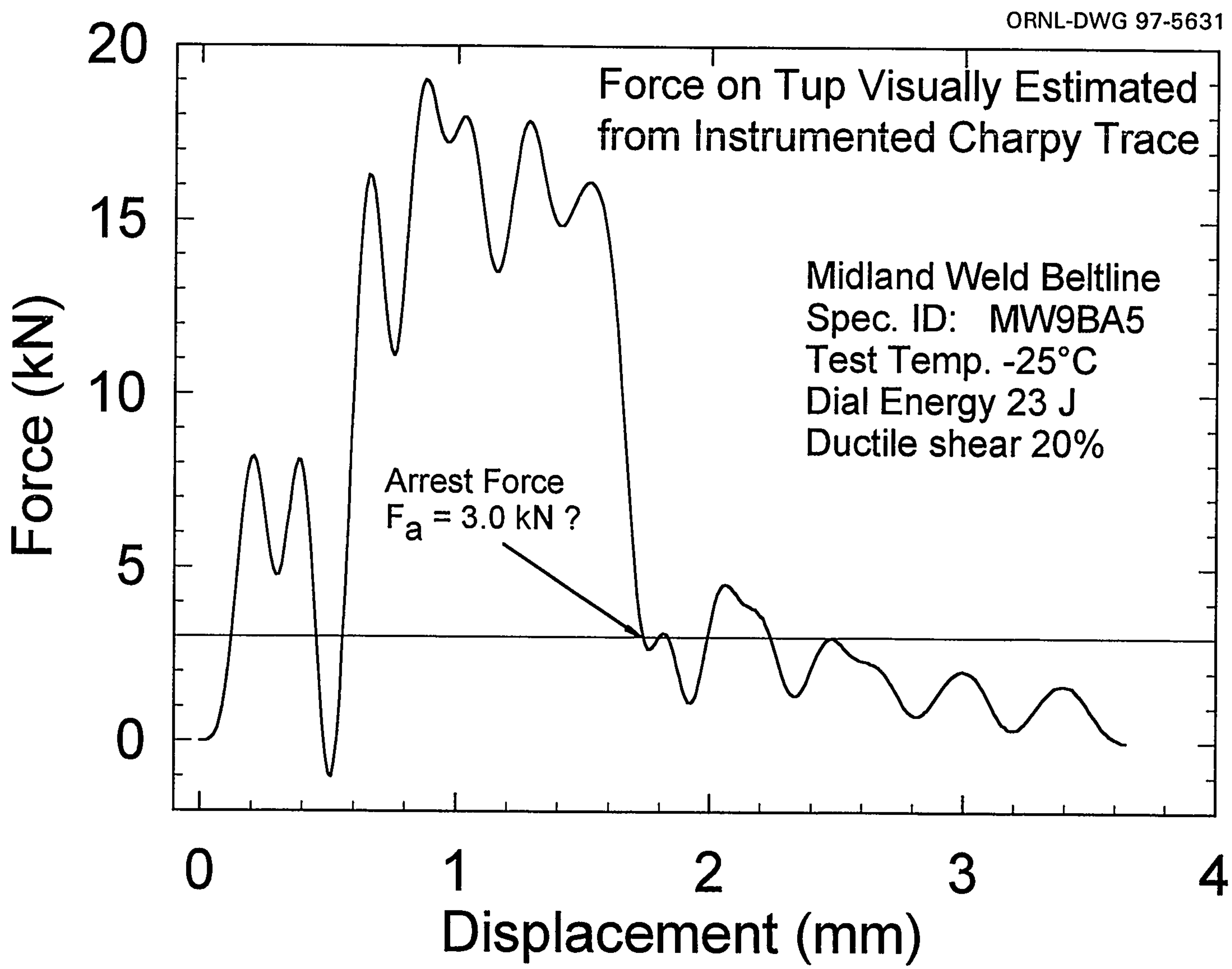

Figure 11.1. Load vs displacement trace deduced from voltage vs time output of an instrumented ASTM E 23 CVN tup.

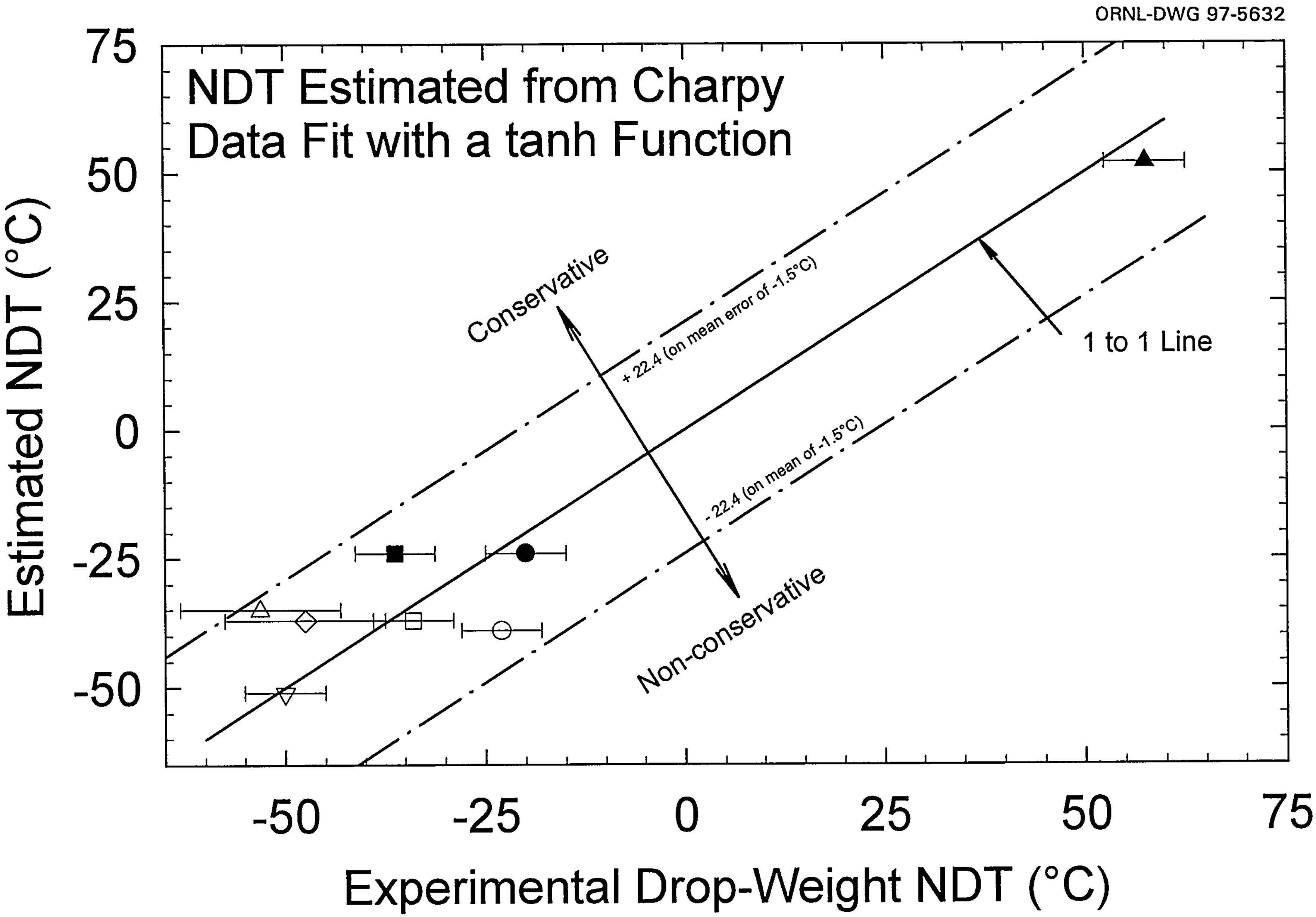

Figure 11.2. Comparison of estimated and actual NDT values for all the materials studied.

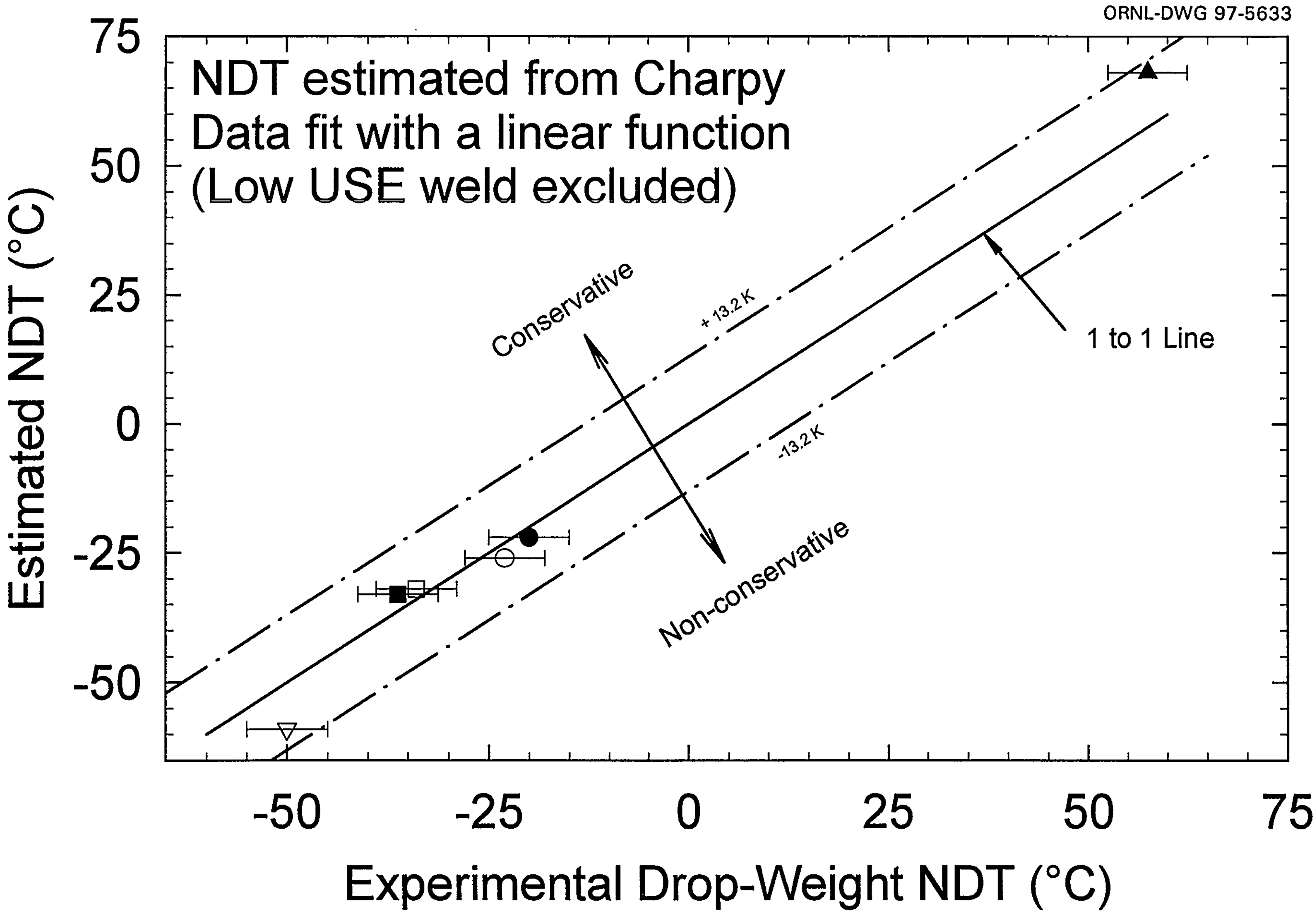

Figure 11.3. Comparison of estimated and actual NDT values for six of the materials studied (excluding two welds with anomalously low values of arrest force).

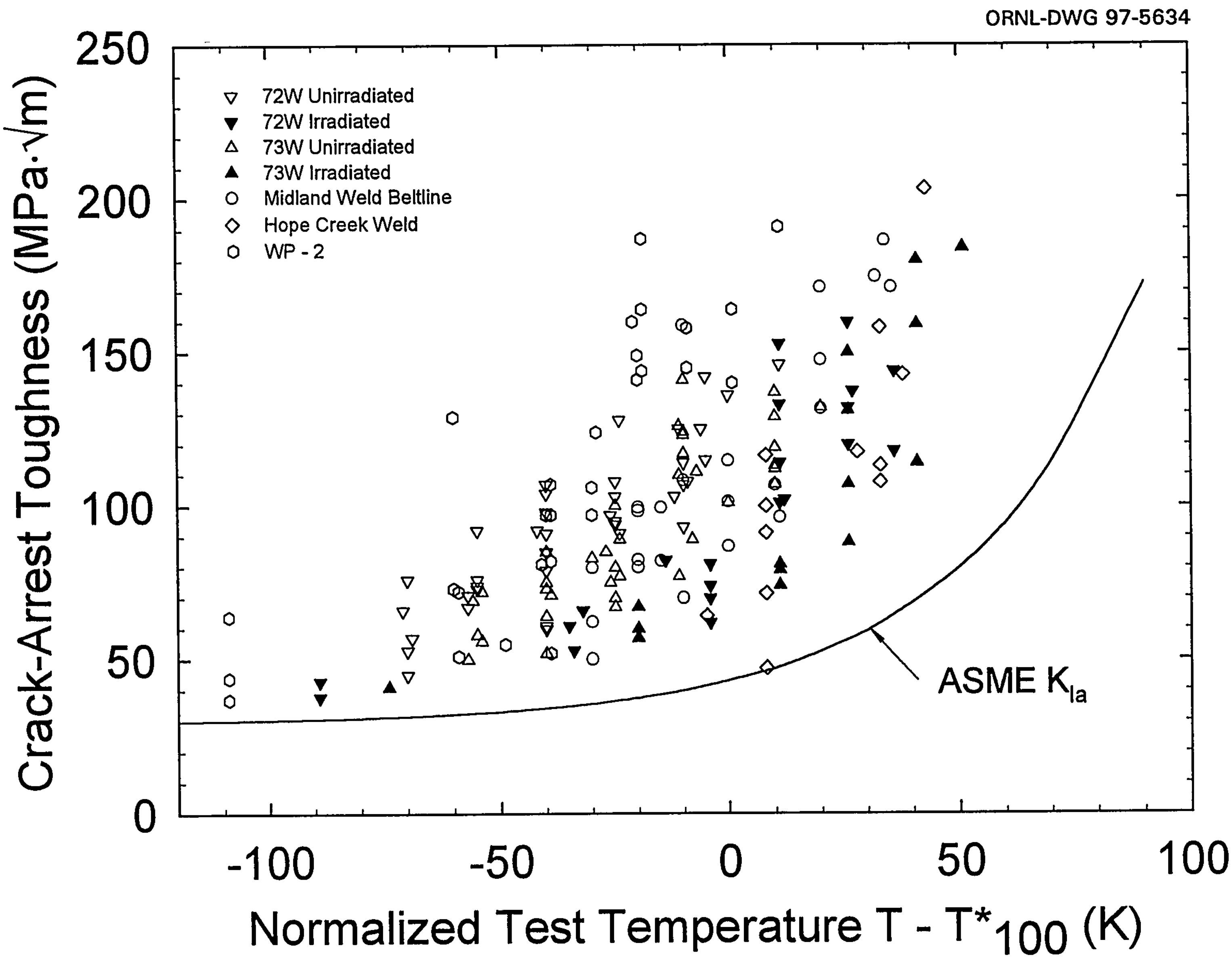

Figure 11.4. Crack-arrest toughness values of various materials normalized using a temperature estimated from the Charpy arrest force.

11.2 Fracture Toughness Testing With Precracked Cylindrical Tensile Specimens (D. J. Alexander, R. K. Nanstad, and J. G. Merkle)

A draft report, "Evaluation of Precracked Cylindrical Tensile Specimens for the Measurement of Fracture Toughness in the Transition Region of Reference Weld Metal 72W," by G. Gage of AEA Technology, Harwell Laboratory, United Kingdom, was reviewed and comments transmitted to the author for incorporation. AEA Technology machined and tested 4- and 8-mm-diam precracked cylindrical tensile (PCCT) specimens and compared the test results with those from the previous ORNL characterization using compact specimens of weld 72W performed in the HSSI Fifth Irradiation Series. The author concluded that, "The PCCT specimen is considered to offer potential for the measurement of fracture toughness in the transition region, however, the effect of specimen size and microstructural effects require further investigation."

The subcontractor report, *Fracture Toughness Testing with Cracked Round Bars: Feasibility Study* [NUREG/CR-6342 (ORNL/TM-12989, ORNL/SUB/94-DHK60)], by J. H. Giovanola and J. E. Crocker, SRI International, Menlo Park, California, was completed, reviewed by HSSI staff, and delivered to the NRC for publication. Subsequently, however, the authors requested that the report not be published in its present form because they determined the need to reanalyze some of the test results. That reanalysis is under way, and the report will be modified as needed before publication.

11.3 Evaluation of Precracked Charpy and Other Small Specimens for Fracture Toughness Testing (M. A. Sokolov, D. E. McCabe, R. K. Nanstad, D. J. Alexander, and S. K. Iskander)

The precracked CVN (PCVN) specimen has been used to estimate fracture toughness of pressure vessel steels for many years. ORNL and many other investigators have performed both quasi-static and dynamic testing. With the advent of the draft ASTM Standard on Fracture Toughness Testing in the Transition Region, however, a more focused effort regarding the PCVN specimen has been initiated within the HSSI Program. This effort involves evaluation of the PCVN specimen as a viable specimen for determination of the reference fracture toughness transition temperature (designated T_0 in the draft ASTM standard), as well as for potential use in estimation of dynamic fracture toughness by testing under impact loading. Such testing is motivated by the specimen's being relatively efficient with regard to the amount of material needed but even more so because the standard CVN impact specimen used in the surveillance programs of all commercial nuclear plants could be fatigue precracked and tested to obtain direct measurements of fracture toughness for a material in the postirradiation or postannealed conditions. A considerable amount of quasi-static testing has already been performed by ORNL with the PCVN specimen for a number of RPV materials, and the general observation is that the specimen has high potential as a viable test specimen for determining T_0 for RPV steels. In addition to interest in the PCVN specimen, the potential use of even smaller specimens has been expressed within the technical community. In cases where the surveillance specimens are limited, or have been exhausted and there is a need for additional data, the fabrication of miniature specimens from broken CVN specimens is a topic of interest. Moreover, testing of 0.2T compact specimens [0.2T C(T)] has been initiated as an experimental comparison of the constraint limitation in the ASTM draft standard for the PCVN specimen; the constraint limit is based on the remaining ligament dimension, which, for an a/W of 0.5, is the same for the PCVN and the 0.2T C(T) specimen. Bend specimens with width-to-thickness ratios of both 2 and 1 will also be tested for comparison. A summary of the results will be included in the next semiannual report.

11.4 Transfer of Government-Furnished Equipment and Materials

(W. R. Corwin, J. J. Henry, and R. K. Nanstad)

Government-furnished equipment and materials that were provided to Materials Engineering Associates (MEA) by the NRC were inventoried by MEA and reviewed by ORNL. An ORNL staff member was present at the MEA laboratories in Lanham, Maryland, during loading of the equipment and materials. The contents were shipped to Oak Ridge and were initially stored in a commercial warehouse. Subsequent evaluation of the equipment resulted in the identification of items appropriate for use on the HSSI Program, and these items were shipped to the ORNL site. Other items were identified for salvage and are being prepared for such disposition. A final inventory and recommendations for disposition are in preparation and will be sent to the NRC for approval before final disposition.

Reference

1. *ASME Boiler and Pressure Vessel Code. An American National Standard*, Sect. XI, "Rules for In-Service Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, 1992.*

*Available in public technical libraries.

12. TECHNICAL ASSISTANCE FOR JCCCNRS WORKING GROUPS 3 AND 12

R. K. Nanstad

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 irradiation, annealing, and reirradiation studies to be conducted in Russia; (2) capsule preparation and initiation of irradiation of Russian specimens within the United States; and (3) preparation for, and participation in, Working Groups 3 and 12 meetings.

12.1 Irradiation Experiments in Host Country (M. A. Sokolov and R. K. Nanstad)

The CVN and round tensile specimens of two Russian weld metals irradiated in HSSI capsule 10.06 at the University of Michigan FNR were returned to ORNL and removed from the capsule for testing. The weld metals are representative of those existing in either VVER-440 or VVER-1000 reactors. Most of the testing in the irradiated and postannealed conditions has been completed under a separately funded program (JCN 6715); the remaining specimens of the VVER-1000 weld will be reirradiated in a subsequent HSSI experiment. The remaining specimens of the VVER-440 weld will not be reirradiated but will instead be fatigue precracked and tested in their current conditions because they were irradiated at about 270°C and the new HSSI capsule is designed for irradiation at 288°C with no temperature gradient. The preliminary results of these tests were presented by R. K. Nanstad to the JCCCNRS Working Group 3 meeting in Moscow in September 1996 (see Sect. 12.2). The results were also included in an ORNL paper for the 24th Water Reactor Safety Information Meeting in October 1996. Further analyses of all the data will be performed with a view toward publication of a technical paper and possibly a joint NRC/Kurchatov Institute report.

12.2 JCCCNRS Working Groups 3 and 12 Meetings (R. K. Nanstad, M. A. Sokolov, S. K. Iskander, D. E. McCabe, D. J. Alexander, R. E. Stoller, K. Farrell, M. K. Miller, and P. M. Rice)

The JCCCNRS Working Groups 3 and 12 met in Moscow on September 15–20, 1996. Before those meetings, R. K. Nanstad accompanied M. E. Mayfield, Chief of the Chemical, Electrical, and Materials Engineering Branch, to meetings at the Ukraine State Scientific and Technical Centre for Nuclear and Radiation Safety in Kiev, Ukraine. This relatively new organization within the Ministry for Environmental Protection and Nuclear Safety (MEPNS) of Ukraine requested the meetings. The purpose of the meetings was ". . . to discuss reactor pressure vessel integrity for the Ukrainian VVER-type reactors, and in particular, to discuss methodologies for performing pressurized thermal shock transients, and areas in which the NRC can support MEPNS' efforts in this area under Priority 14 of the Lisbon Agreements." A number of topics for potential cooperation were identified during the meetings, and a proposed outline for a workshop on pressurized thermal shock was developed.

The seventh meeting of Working Group 3 was held at the Russian Research Center-Kurchatov Institute. The following are the five presentations made by R. K. Nanstad, with the authors indicated for each.

"Summary of Modeling and Mechanisms Research: Irradiated and Annealed RPV Steels at UCSB and ORNL," by R. E. Stoller, G. R. Odette, G. E. Lucas, K. Farrell, M. K. Miller, and P. M. Rice.

"Modeling of Embrittlement Recovery by Thermal Annealing and Summary of Experimental Results," by E. D. Eason, J. E. Wright, E. E. Nelson, G. R. Odette, E. V. Mader, A. Taboada, S. K. Iskander, M. A. Sokolov, and R. K. Nanstad.

"Summary of Experimental Results with Miniature Fracture Toughness Specimens," by R. K. Nanstad, M. A. Sokolov, D. E. McCabe, and D. J. Alexander.

"Comparison of Fracture Toughness, Crack-Arrest Toughness, and Charpy Impact Toughness for Unirradiated and Irradiated RPV Steels," by R. K. Nanstad, D. E. McCabe, S. K. Iskander, and M. A. Sokolov.

"ORNL Results on Effects of Irradiation and Thermal Annealing on Charpy Impact Toughness of VVER-440 and VVER-1000 Weld Metals," by R. K Nanstad and M. A. Sokolov.

A number of interesting presentations were made by various Russian scientists, with the most relevant being that by Dr. A. Chernobaeva regarding the results of irradiation, annealing, reirradiation (IAR), and reannealing (IARA) of HSSI weld 73W and HSST Plate 02. For the IAR and IARA results with HSSI weld 73W and HSST Plate 02, the annealing treatments resulted in virtually full recovery after irradiation and reirradiation. Moreover, the reembrittlement, although substantial, fell slightly below that predicted by the lateral shift method. *Regulatory Guide 1.99* (Rev. 2) predictions of irradiation-induced shifts for the two U.S. steels were about equal to the measured values for HSST Plate 02 and slightly overpredicted the results for HSSI weld 73W (these results are in general agreement with previous results from ORNL experiments). The results of the IAR and IARA experiments require much further study regarding the behavior vs predictions of the regulatory guide on thermal annealing, as well as from a fundamental materials science basis. Given the array of irradiation fluences and annealing temperatures and times used for experiments with HSSI weld 73W and HSST Plate 02, an atom probe-field ion microscope investigation with those steels would be very revealing and useful in understanding the kinetics of copper diffusion and role of matrix damage following annealing and reirradiation.

The sixth meeting of Working Group 12 and the joint meeting of Working Groups 3 and 12 were held following the Working Group 3 meeting. In the joint meeting, it was agreed that a combination of the working groups should be proposed to the JCCCNRS cochairmen. The proposed combined working group should be designated Working Group 3/12 and be titled, "Aging, Radiation Embrittlement, and Structural Integrity of Nuclear Power Plant Pressure Vessels and Other Components." The projects to be undertaken by the Working Group 3/12 would come from four general areas: (1) RPV embrittlement, (2) component aging, (3) methodology, and (4) codes and standards. Working Group 3/12 would meet annually at a date to be jointly determined. Unless otherwise agreed, the meetings would be held in Moscow.

Details of all these meetings were reported by R. K. Nanstad in ORNL/FTR-5905, dated October 7, 1996.

13. Correlation Monitor Materials

W. R. Corwin and E. T. Manneschmidt

This task was 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 original materials qualified for use as a correlation monitor in reactor surveillance capsules are the pieces remaining from the early HSST plates 01, 02, and 03, this task will provide for cataloging, archiving, and distribution of the material on behalf of the NRC.

During this reporting period, the transfer of the residual correlation monitor material found at Y-12 to ORNL was completed and archival storage of the correlation monitor material was maintained. Plans to initiate the construction of a weatherproof enclosure to protect the steel were further delayed because of the lack of overall program funding.

14. Test Reactor Irradiation Coordination

D. W. Heatherly, M. T. Hurst, D. W. Sparks, and K. R. Thoms

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

Early in this reporting period the design of the UCSB irradiation facility was completed and certified-for-construction drawings were issued. Details of the design were reported in the last semiannual progress report.* A vendor was chosen to fabricate the facility, which is about 80% complete. It is anticipated that the facility will be completed and ready for testing and use early in the next reporting period. Concurrently, the UCSB specimen container baskets have been designed and fabricated. The specimen container baskets have been inspected, test fitted, and cleaned and are ready for installation into the facility. A mockup test fit of UCSB specimen packets in the specimen container baskets will be performed before facility testing.

An instrumentation and control system has been designed, and assembly of this system is nearing completion. This system will automatically control the electrical heaters in the six independently controlled temperature regions of the facility. Before shipment to the FNR, the facility and associated instrumentation system will be integrated and tested to ensure that the facility will perform as intended.

The base and framework portion of the facility has been completed and is also ready for use. The base and framework will be used to provide a means to move the facility into and away from the FNR core. The design and fabrication of the base and framework will provide the support, thermal shielding, and mobility for both the UCSB and IAR facilities simultaneously. The design consists of a base plate that will be mounted to the existing moving trolley mechanism, a thermal shield between the facilities and the FNR core, and a boral shielding box to provide thermal neutron shielding for the UCSB facility.

A support structure (specimen transfer station) has been designed and is in the final stages of construction at the FNR. The support structure will suspend the UCSB facility so that the upper seal plug of the UCSB facility is just above the pool water surface, allowing specimen packets to be transferred in a dry environment. A specimen transfer scheme has been developed that will allow the individual UCSB specimen packets to be removed, rotated, or relocated within the facility during routine reactor shutdowns. This will be accomplished through the use of a lead-shielded transfer cask that has been designed and is being fabricated. A brief description of a typical specimen packet changeout is described in the following paragraph.

After the reactor is shut down, the UCSB facility will be retracted from the core using the trolley mechanism. Once retracted, the facility will be lifted and suspended from the transfer station at the pool water surface. The top seal plug of the facility will then be removed. The upper portion of the facility contains two steel shield plugs to provide shielding while the facility is at the surface of the pool. The specially designed shielding cask will then be lowered into place over the facility. The shielding

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*D. W. Heatherly, D. W. Sparks, and K. R. Thoms, "Test Reactor Irradiation Coordination," p. 14-1 in *Heavy-Section Steel Irradiation Program, Progress Report for October 1995-March 1996,* USNRC Report NUREG/CR-5591, Vol. 7, No. 1 (ORNL/TM-11568/V7&N1), submitted to the NRC for publication.

cask has been designed with vertical through holes that match the geometry of the facility. Lifting cables will be passed through the cask and fastened to the lifting cables at the top of the shield plugs. A schematic diagram of the facility in this configuration is shown in Figure 14.1. The steel shield plugs and lifting cables are designed such that the cables pass freely through the shields, lifting the specimen baskets, until the specimen baskets are out of the lower portion of the facility where they have the least amount of clearance. Once the baskets are above the low clearance area of the facility, the lifting cables will engage in the shield plugs and lift both the shield plugs and the specimen baskets. Both the shield plugs and the baskets will be raised until the shield plugs have passed through the vertical holes in the transfer cask, after which a slotted, lead shield plug will be installed into the top of the transfer cask. The specimen baskets will then be raised into the shielded cask and locked into it by a steel rod inserted through the lower part of the cask for the baskets to rest on. The steel shield plugs will then be removed from the lifting cables to be reinserted into the facility after the cask has been removed. The shielding cask will then be lifted from the facility and bolted onto a base that will provide lead shielding for the bottom of the cask. A schematic diagram of the specimen baskets in the shielded cask is shown in Figure 14.2. The cask containing the specimen baskets will then be delivered to the FNR hot cells for removal or insertion of specimen packets. After the cask has been placed inside the hot cell, the operator will remove the upper lead shield plug from the transfer cask, lift the specimen baskets out of the cask, and place them on a table. The lids will be removed from the specimen baskets, and changes in specimen packet loading can be completed. The specimens will be reinstalled into the reactor for further irradiation by simply reversing the operations just described.

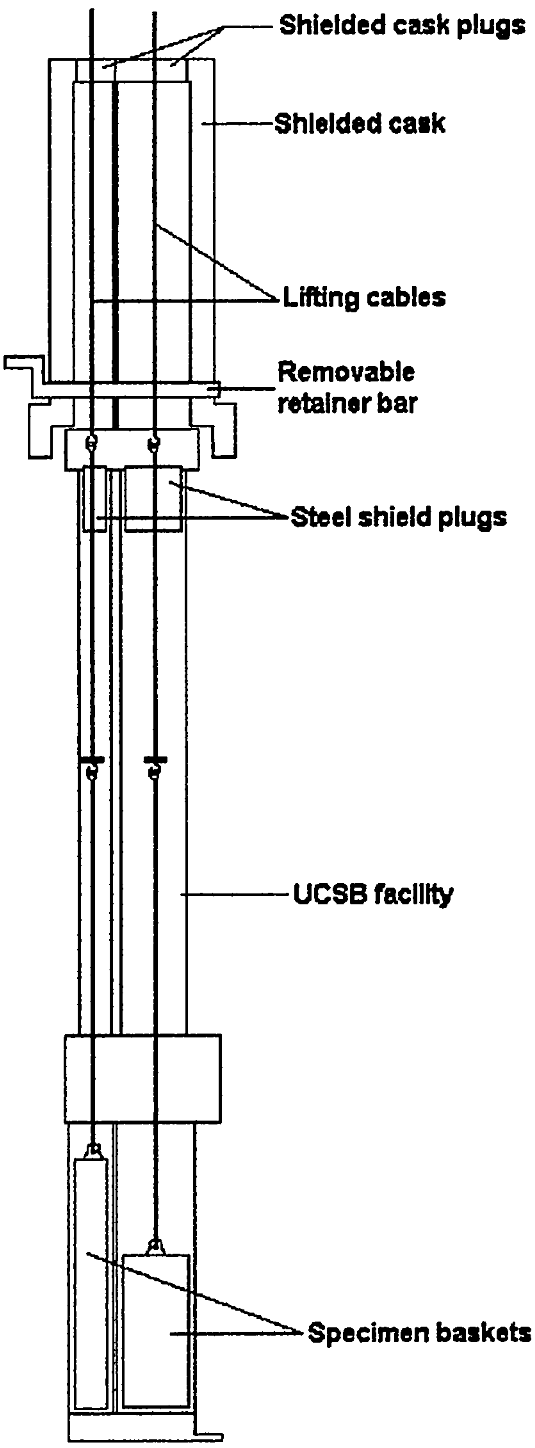

Figure 14.1. Shielded transfer cask positioned over the HSSI-UCSB facility.

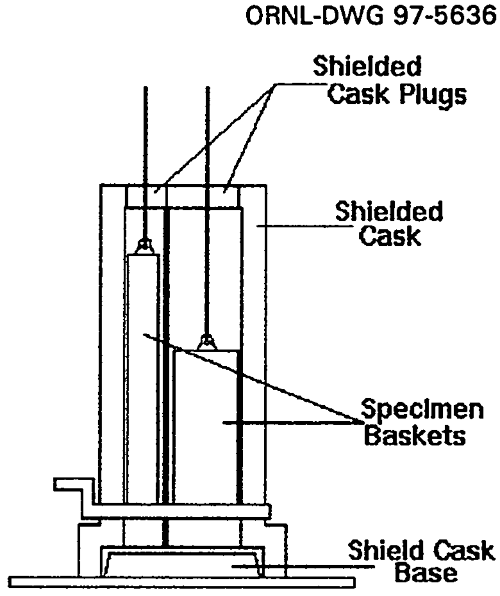

Figure 14.2. Loaded cask ready for transport to hot cells.

CONVERSION FACTORS[a]

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

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

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10. SUPPLEMENTARY NOTES

M. Vassilaros, NRC Project Manager

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 Through September 1996.

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

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