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Heavy-Section Steel Irradiation Program

Progress Report for April 1997 – March 1998

Oak Ridge National Laboratory

U.S. Nuclear Regulatory Commission
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Abstract

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

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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 1997 to March 1998. The work is performed by Oak Ridge National Laboratory (ORNL) and is managed by the Metals and Ceramics (M&C) Division of ORNL. Major tasks at ORNL are carried out by the M&C and Engineering Technology divisions.

Previous HSSI progress reports in this series are:

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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 separation of the two programs in 1989. Previous HSST Program progress reports contain much information on the irradiation assessments being continued by the HSSI Program as well as earlier related studies. The HSST Program progress reports issued before formation of the HSSI Program are also listed here as a convenience to the reader.

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

1. Program Management

The purpose of the program management task is to ensure that the overall objectives are achieved. The major activities include program planning and resource allocation, program monitoring and control, and documentation and technology transfer. The Heavy-Section Steel Irradiation (HSSI) Program is arranged into eight tasks as described by the project and budgetary proposal. Each task, (1) program management, (2) irradiation effects in engineering materials, (3) annealing, (4) microstructural analysis of radiation effects, (5) in-service irradiated and aged materials evaluations, (6) fracture-toughness curve shift method, (7) special technical assistance, and (8) foreign research interactions, is described in a corresponding report chapter and is summarized in the following. The work is performed by Oak Ridge National Laboratory (ORNL).

2. Irradiation Effects in Engineering Materials

The objective of this task is to develop data addressing the current method of shifting the American Society of Mechanical Engineers fracture-toughness (K_{Ic}, K_{Ia}, and K_{IR}) curves to account for irradiation embrittlement in high-copper welds. The Tenth Irradiation Series has produced a complete characterization of WF-70 weld metal that was taken from the nozzle course and beltline positions in the Midland Unit 1 reactor pressure vessel (RPV). The characteristics of low upper-shelf steels evaluated by fracture-mechanics test methods have been improved as a result of this project.

3. Annealing

The purpose of the annealing task is to evaluate the correlation between fracture-toughness and Charpy V-notch (CVN)-impact energy during irradiation, annealing, and reirradiation. Reirradiation continues of previously irradiated and annealed HSSI Weld 73W CVN specimens in the HSSI/University of California at Santa Barbara (UCSB) capsule at the University of Michigan Ford Nuclear Reactor (FNR). The retrievable fission-radiometric dosimetric sets have been removed from the capsule after about a 1-month exposure and have been returned to ORNL and counted. The data have been analyzed and will be used to estimate the time to retrieve the containers. Work on the new irradiation, annealing, and reirradiation facilities and retrievable capsules is progressing, and the computer facilities are being upgraded for acquisition, transmission, and storage. Attention is also being given to the evaluation of a potential problem with temper embrittlement within the heat-affected zone (HAZ) of RPV welds, otherwise identified as local brittle zones. The grain-coarsened material in multipass weldments of the HAZ is being created in bulk volume using electrical-resistance heating. Five commercially made RPV steels are being evaluated. All materials showed a tendency to embrittle when aged at 450°C. However, the initial austenitize and postweld heat treatment condition of all materials gave extremely low

transition temperatures. Aging at 450°C for 168 h did not impart sufficient embrittlement to cause concern.

4. Microstructural Analysis of Radiation Effects

This section discusses three components of work in the area of microstructural characterization and modeling. First, the results of an atom-probe investigation of copper solubility in a model Fe-Cu-Ni alloy is described. Next is a comparison of the effects of fast neutron and 2.5-MeV electron irradiation on radiation-induced increases in the yield strength of the HSST-02 correlation monitor. Finally, the embrittlement model developed under this task is used to investigate the recent suggestion that interstitial cluster mobility may have a significant impact on the predictions of radiation damage models.

5. In-Service Irradiated and Aged Materials Evaluations

The mill and saw were installed in cell 6 of Building 3025E at ORNL and were made operational. Critical as-built drawings have been made as funding permitted. New grounding for the shielding on all the cables has been completed. A microphone and associated amplifier and speaker have been purchased and installed. Radiation shielding for the feedthrough conduits for all (new and old) cabling will be installed to facilitate operation of this hot cell.

6. Fracture-Toughness Curve Shift Method

A test matrix has been developed to perform irradiation of a statistically meaningful number of 1T compact specimens of a radiation-sensitive material with Charpy shifts up to 200°C. Specimens of smaller sizes are also included into the matrix. Contacts were initiated with investigators at the KORPUS facility in Dimitrovgrad, Russia; the LYRA facility in Petten, The Netherlands; HANORO in South Korea; the NRI facility in Rez, Czech Republic; and the RRC-Kurchatov Institute in Moscow, Russia; regarding technical aspects, availability, and cost relative to irradiation experiments with 1T compact specimens to validate the shape of the master curve for highly embrittled material. The possibility of in-core irradiations at the University of Michigan FNR is also under consideration. Contacts were initiated with investigators at the AEA-Harwell, United Kingdom (G. Gage); MPA, Stuttgart, Germany (J. Fohl); and the International Atomic Energy Agency (IAEA), Vienna, Austria (V. Lyssakov and M. Brumovsky); regarding the availability of radiation-sensitive material in sufficient amounts to perform irradiation experiments with 1T compact specimens to validate the shape of the master curve for highly embrittled material. ABB-CE in Chattanooga, Tennessee, was also contacted regarding technical aspects and costs relative to the manufacture of such welds for this experiment. In response to these contacts, investigators from MPA, Stuttgart (J. Fohl), Germany, offered a radiation-sensitive material in sufficient amounts to perform irradiation experiments with 1T compact specimens. This weld, designated KS-01, has been investigated to a limited degree by the MPA. The MPA results demonstrate that this weld exhibits a 41-J shift of about 200°C at 2.13×10^{19} n/cm^2, making it promising for our program.

Preparation continues on a draft NUREG report on comparison of irradiation-induced Charpy impact and fracture-toughness curve shifts.

7. Special Technical Assistance

The section describes the special analytical and experimental investigations that support the U.S. Nuclear Regulatory Commission in resolving regulatory research issues related to irradiation effects on materials.

A draft NUREG report on the effect of long-term thermal aging of stainless steel welds at 343°C was completed. The welds were produced by shielded metal-arc welding with type 308 filler metal with the chemical composition adjusted to obtain different ferrite levels (4, 8, or 12%). Portions of the welds were aged for 3,000; 10,000; 20,000; or 50,000 h. The significant decreases in toughness with increasing aging time appear to be the result of spinodal decomposition of the ferrite, as well as precipitation of both large and fine G-phase particles.

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

The UCSB facility, which was successfully installed at the University of Michigan FNR, tested, and placed into operation during the last reporting period, received neutrons on more than 70% of the available reactor days. The lost irradiation time was caused by the overtemperature event that occurred during the last period, the broken trolley crank, and the reactor operating in conditions undesirable for UCSB irradiation. A review of the overtemperature event revealed that the test specimens were not damaged. However, procedures were instituted and safeguards were installed to ensure that similar incidents would not happen again. During this period, 5,717 specimens had been inserted into the facility for irradiation and 1405 specimens had reached the desired fluence and were removed for testing. This facility is also being used in conjunction with other irradiations being conducted for the HSSI Program.

8. Foreign Research Interactions

During this reporting period, two staff members of the Japan Atomic Energy Research Institute, Masahide Suzuki and Kunio Onizawa, visited ORNL on February 24–25, 1998, to exchange information and discuss the test plan. Specimen orientation, location in the vessel wall, and schedules were discussed. Considerations are being given to the testing of CVN and precracked CVN specimens.

Charpy impact and tensile testing of both VVER-440 and VVER-1000 welds in the unirradiated, irradiated, and thermally annealed conditions were completed previously. A meeting with Working Group 3 members from the Russian Research Center-Kurchatov Institute resulted in completion of a paper, "Exploratory Study of Irradiation, Annealing, and Reirradiation Effects on American and Russian Reactor Pressure Vessel Steels," by A. A. Chernobaeva, M. A. Sokolov, R. K. Nanstad, A. M. Kryukov, Y. A. Nikolaev, and Yu. N. Korolev. The paper was presented by A. A. Chernobaeva at a symposium in Amelia Island, Florida, in August 1997 and was published in the proceedings of the symposium.

Concrete and lead-lined containers, designed by ORNL, were fabricated. Two such containers with eight irradiated 4T compact specimens of HSSI Welds 72W and 73W were shipped to SCK-CEN in September 1997.

For the IAEA New Coordinated Research Program (CRP), testing of precracked Charpy specimens of JRQ steel were completed and the test results were reported at the CRP meeting in October 1997. Nine participants had completed testing of the JRQ steel, and the results compare favorably.

Planning for the formal collaboration with the Korean Atomic Energy Research Institute was initiated previously and consisted of a list of technical topics of mutual interest. No further activity has been undertaken in this subtask.

Heavy-Section Steel Irradiation Program Progress Report for April 1997 Through March 1998*†

T. M. Rosseel

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 nil-ductility transition, and tensile properties are included. Models based on observations of radiation-induced microstructural changes using the atom-probe field-ion microscope and the high-resolution transmission electron microscope are being developed to provide a firm basis for extrapolating the measured changes in fracture properties to wide ranges of irradiation conditions. Archival storage and disbursement of correlation monitor materials provide support for commercial light-water reactor irradiation surveillance programs. Collaborative research activities with other domestic and foreign research programs provide a wider experimental base for understanding embrittlement and postirradiation annealing effects. The principal materials examined within the HSSI Program are high-copper welds because their postirradiation properties frequently limit the continued safe operation of commercial RPVs.

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.

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

The program is divided into one program management task and seven technical tasks. These tasks are (1) program management; (2) irradiation effects on engineering materials; (3) annealing; (4) microstructural analysis of radiation effects; (5) in-service irradiated and aged material evaluations; (6) fracture-toughness curve shift method; (7) special technical assistance; and (8) foreign research interactions.

During this period, 21 technical papers[1–21] were published and 23 technical presentations[22–44] were made. Additionally, the HSSI program manager presented five program briefings during visits and reviews by the acting director of the Office of Nuclear Regulatory Research, NRC; the director of the Engineering Technology Division, NRC; the RPV Integrity Peer Review Committee; and staff of the Japan Atomic Energy Research Institute and the South African Council for Nuclear Safety.

References

1. W. R. Corwin, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Heavy-Section Steel Irradiation Program Semiannual Progress Report for October 1995 - March 1996*, USNRC Report NUREG/CR-5591, Vol. 7, No. 1 (ORNL/TM-11568/V7&N1), April 1997.

2. S. K. Iskander and R. E. Stoller, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Results of Charpy V-Notch Impact Testing of Structural Steel Specimens Irradiated at ~30 °C to 1×10^{16} neutrons/cm^2 in a Commercial Reactor Cavity*, USNRC Report NUREG/CR-6399 (ORNL-6886), April 1997.

3. M. A. Sokolov and D. J. Alexander, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *An Improved Correlation Procedure for Subsize and Full-Size Charpy Impact Specimen Data*, USNRC Report NUREG/CR-6379 (ORNL-6888), April 1997.

4. F. M. Haggag and R. K. Nanstad, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Effects of Thermal Aging and Neutron Irradiation on the Mechanical Properties of Three-Wire Stainless Steel Weld Overlay Cladding*, USNRC Report NUREG/CR-6363 (ORNL/TM-13047), May 1997.

5. Randy K. Nanstad and William R. Corwin, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Report of Foreign Travel to Germany*, ORNL/FTR-6076, June 23, 1997.

6. S. K. Iskander, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab, *Report of Foreign Travel to Finland, Belgium, and Germany*, ORNL/FTR-6226, July 14, 1992.

7. R. K. Nanstad, M. A. Sokolov, R. E. Stoller, and H. W. Hayden, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Report of Foreign Travel to Japan and Russia*, ORNL/FTR-6145, August 25, 1997.

8. R. E. Stoller, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., "Non-steady-state Conditions and Incascade Clustering in Radiation Damage Modeling," *Journal of Nuclear Materials* **244,** 195–204 (1997).

9. P. M. Rice and R. E. Stoller, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., "The Effect of Solutes on Defect Distributions in Ion-Irradiated Model LWR Pressure Vessel Steels," *Journal of Nuclear Materials* **244**, 219–226 (1997).

10. P. J. Pareige, R. E. Stoller, K. F. Russell, and M. K. Miller, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., "Atom Probe Characterization of the Microstructure of Nuclear Pressure Vessel Surveillance Materials After Neutron Irradiation and After Annealing Treatments," *Journal of Nuclear Materials* **249**, 165–174 (1997).

11. P. J. Pareige, K. F. Russell, R. E. Stoller, and M. K. Miller, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., "Influence of Long-Term Thermal Aging on the Microstructural Evolution of Nuclear Reactor Pressure Vessel Materials: An Atom Probe Study," *Journal of Nuclear Materials* **250**, 176–183 (1997).

12. R. E. Stoller, G. R. Odette, and B. D. Wirth, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., "Primary Defect Formation in bcc Iron," *Journal of Nuclear Materials* **251**, 49–60 (1997).

13. M. K. Miller and K. F. Russell, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Atom Probe Characterization of Copper Solubility in the Midland Weld After Neutron Irradiation and Thermal Annealing*, ORNL/NRC/97-16 (Letter Report), August 1997.

14. W. R. Corwin, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Heavy-Section Steel Irradiation Program Semiannual Report for October 1996 - March 1997*, USNRC Report, NUREG/CR-5591, Vol. 7, No. 2, (ORNL/TM-11568/V7&N2), September 1997.

15. D. E. McCabe, *Report of Foreign Travel to Vienna, Austria*, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., ORNL/FTR-6328, November 12, 1997.

16. R. K. Nanstad, *Report of Foreign Travel to Russia*, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., ORNL/FTR-6287, November 19, 1997.

17. A. A. Chernobaeva, M. A. Sokolov, R. K. Nanstad, A. M. Kryukov, Y. A. Nikolaev, and Yu. N. Korolev, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., "Exploratory Study of Irradiation, Annealing, and Reirradiation Effects on American and Russian Reactor Pressure Vessel Steels," pp. 871–882 in *Proceedings of the Eighth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August 10–14, 1997, Amelia Island, Florida*, American Nuclear Society, La Grange Park, Illinois, 1997.

18. T. M. Rosseel, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Heavy-Section Steel Irradiation Program Semiannual Report for October 1996 - March 1997*, USNRC Report, NUREG/CR-5591, Vol. 8, No. 1, (ORNL/TM-11568/V8&N1), February 1998.

19. S. K. Iskander, P. P. Milella, and A. Pini, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Results of Crack-Arrest Tests on Irradiated A 508 Class 3 Steel*, USNRC Report NUREG/CR-6447 (ORNL-6894), February 1998.

20. M. A. Sokolov, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., "Statistical Analysis of the ASME K_{Ic} Database," *Journal of Pressure Vessel Technology, Transactions of the ASME*, **120**, 24–28 (February 1998).

21. P. J. Pareige, K. F. Russell, R. E. Stoller, and M. K. Miller, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., *Influence of Long-Term Thermal Aging on the Microstructural Evolution of Nuclear Reactor Pressure Vessel Materials*, USNRC Report, NUREG/CR-6537 (ORNL/TM-13406), March 1998.

22. D. E. McCabe and R. K. Nanstad, "Evaluate the Potential of Reactor Pressure Vessel Steel Heat-Affected Zone for Temper Embrittlement," presented by D. E. McCabe at the NRC Materials Engineering Contractors Workshop, Naval Surface Warfare Center, Carderock, Maryland, April 17–18, 1997.

23. M. A. Sokolov, D. E. McCabe, and R. K. Nanstad, "Fracture Toughness Curve Shift and Small Specimen Studies with Master Curve Technology," presented by M. A. Sokolov at the NRC Materials Engineering Contractors Workshop, Naval Surface Warfare Center, Carderock, Maryland, April 17–18, 1997.

24. D. E. McCabe, R. K. Nanstad, and L. Heatherly, "Evaluation of the Potential of RPV Steels HAZ for Temper Embrittlement," presented by R. K. Nanstad at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village Center, Japan, May 11–16, 1997.

25. R. K. Nanstad and D. E. McCabe, "Evaluation of Variability in Fracture Toughness and Chemical Composition for Midland Reactor Weld WF-70," presented by R. K. Nanstad at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village Center, Japan, May 11–16, 1997.

26. R. K. Nanstad and D. E. McCabe, "Progress and Current Activities for Use of the Master Curve to Establish Fracture Toughness of RPV Steels," presented by R. K. Nanstad at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village Center, Japan, May 11–16, 1997.

27. M. A. Sokolov, D. E. McCabe, Y. A. Davidov, and R. K. Nanstad, "Use of Precracked Charpy and Smaller Specimens for Establishing Transition Temperature Master Curve," presented by M. A. Sokolov at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village Center, Japan, May 11–16, 1997.

28. A. C. Chernobaeva, M. A. Sokolov, R. K. Nanstad, A. M. Kryukov, Y. A. Nikolaev, and Yu. N. Korolev, "Exploratory Study of Irradiation, Annealing, and Reirradiation of U.S. and Russian RPV Steels," presented by M. A. Sokolov at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village Center, Japan, May 11–16, 1997.

29. M. A. Sokolov and R. K. Nanstad, "Comparison of Irradiation-Induced Charpy Impact and Fracture Toughness Transition Temperature Shifts," presented by M. A. Sokolov at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village Center, Japan, Mav 11–16, 1997.

30. M. A. Sokolov, S. Spooner, W. A. Pavinich, G. R. Odette, B. D. Wirth, and G. E. Lucas, "Small-Angle Neutron Scattering Investigations of Irradiated and Annealed High-Copper Welds," presented by M. A. Sokolov at the meeting of the International Group on Radiation Damage Mechanisms, Shonan Village Center, Japan, May 11–16, 1997.

31. M. K. Miller, K. F. Russell, P. J. Pareige, M. J. Starink, and R. C. Thomson, "Low Temperature Copper Solubilities in Fe-Cu-Ni," presented by M. K. Miller at the 44th International Field Emission Symposium, Tsukuba, Japan, 7–11 June 1997.

32. A. A. Chernobaeva, M. A. Sokolov, R. K. Nanstad, A. M. Kryukov, Y. A. Nikolaev, and Yu. N. Korolev "Exploratory Study of Irradiation, Annealing, and Reirradiation Effects on American and Russian Reactor Pressure Vessel Steels," presented by A. A. Chernobaeva at the Eighth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Florida, August 10–14, 1997.

33. D. W. Heatherly, M. T. Hurst, K. R. Thoms, G. E. Lucas, G. R. Odette, and P. A. Simpson, "A New Versatile, Materials Irradiation Facility," presented by W. R. Corwin at the 14th International Structural Conference on Mechanics in Reactor Technology (SMiRT), Lyon, France, August 17–22, 1997.

34. S. K. Iskander, M. A. Sokolov, and R. K. Nanstad, "Response of Neutron-Irradiation RPV Steels to Thermal Annealing," presented by W. R. Corwin at the 14th International Conference on Structural Mechanics in Reactor Technology (SMiRT), Lyon, France, August 17–22, 1997.

35. D. E. McCabe, M. A. Sokolov, and R. K. Nanstad, "Fracture Toughness Evaluation of a Low Upper-Shelf Weld Metal from the Midland Reactor Using the Master Curve," presented by W. R. Corwin at the 14th International Conference on Structural Mechanics in Reactor Technology (SMiRT), Lyon, France, August 17–22, 1997.

36. R. E. Stoller, "Evaluation of Neutron Energy Spectrum Effects and RPV Thru-Wall Attenuation Based on Molecular Dynamics Displacement Cascade Simulations," presented by W. R. Corwin at the 14th International Conference on Structural Mechanics in Reactor Technology (SMiRT-14): Post-Conference Seminar; Current Issues in the Evolution of Risk-Informed Integrity Engineering, Lyon, France, August 25–26, 1997.

37. R. K. Nanstad, S. K. Iskander, D. E. McCabe, and M. A. Sokolov, "Irradiation, Annealing, and Reirradiation Research in the ORNL Heavy-Section Steel Irradiation Program," presented by R. K. Nanstad at the International Atomic Energy Agency Meeting on Irradiation Effects and Mitigation, Vladimir, Russia, September 14–19, 1997.

38. M. A. Sokolov, D. E. McCabe, D. J. Alexander, and R. K. Nanstad, "Applicability of the Fracture Toughness Master Curve to Irradiated Reactor Pressure Vessels," presented by R. K. Nanstad at the International Atomic Energy Agency Meeting on Irradiation Effects and Mitigation, Vladimir, Russia, September 14–19, 1997.

39. M. A. Sokolov, R. K. Nanstad, D. E. McCabe, D. J. Alexander, and S. K. Iskander, "Use of Small Specimens for Evaluating Structural Integrity of Nuclear Reactor Pressure Vessels," presented by M. A. Sokolov at the Symposium on Testing and Inspection Techniques for Structural Integrity Evaluation and Power Plant Life Extension, Indianapolis, Indiana, September 15, 1997.

40. D. E. McCabe, "Master Curve Concept Applied in an ASTM Standard Method," presented by D. E. McCabe at the IAEA New Coordinated Research Program on Assuring Structural Integrity of Reactor Pressure Vessels, Vienna, October 8–10, 1997.

41. D. E. McCabe, "Round Robin Report on JRQ, HSST Plate 02, and HSSI Fifth Irradiation Series Weld," presented by D. E. McCabe at the IAEA New Coordinated Research Program on Assuring Structural Integrity of Reactor Pressure Vessels, Vienna, October 8–10, 1997.

42. R. K. Nanstad, W. R. Corwin, and M. A. Sokolov, "Review of Small Size Specimen Toughness Testing in the Nuclear Industry," presented by R. K. Nanstad at the Pressure Vessel Research Council Workshop on Subsize Specimen and Nondestructive Testing for Fracture Toughness Determination, San Diego, February 2, 1998.

43. R. K. Nanstad, D. E. McCabe, M. A. Sokolov, D. J. Alexander, and S. K. Iskander, "Use of Small Specimens for Evaluating Structural Integrity of Nuclear Reactor Pressure Vessels," presented by M. A. Sokolov at the Pressure Vessel Research Council Workshop on Subsize Specimen and Nondestructive Testing for Fracture Toughness Determination, San Diego, February 2, 1998.

44. D. E. McCabe, "The Technical Basis for the Master-Curve Approach to Transition Temperature Definition," presented by D. E. McCabe at the NRC/Industry Meeting on RPV Integrity Issues, Rockville, Maryland, February 12, 1998.

2. Irradiation Effects on Engineering Materials

D. E. McCabe

2.1 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 was 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 heat of weld wire and specific lot of Linde 80 weld flux were used that produce 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.

The final report on the Tenth Irradiation Series, which provides an overview perspective on the postirradiation property changes, is nearing completion. Two irradiation levels were used: 0.5 and 1.0×10^{19} n/cm^2 (>1 MeV). At 0.5×10^{19} n/cm^2, the principal method of damage assessment was CVN transition curve shift and tensile property strength increase. The more comprehensive effort was applied to the 1.0×10^{19} n/cm^2 fluence level with both basic properties and fracture-mechanics tests.

Table 2.1 lists tensile properties for unirradiated and the two irradiated conditions. The tensile results were about what might have been expected, except for the nozzle course weld at 0.5×10^{19} n/cm^2. The reason for the seemingly insignificant damage was sought, but a logical explanation could not be produced. Table 2.2 provides the results from the CVN tests.

The before-and-after 1.0×10^{19} n/cm^2 irradiation master curves (MCs) that represent fracture-mechanics tests are shown in Figure 2.1. This type of transition temperature evaluation indicated that the beltline WF-70 weld and nozzle course WF-70 weld were two different materials. Conventional evaluations by CVN and drop-weight tests, however, did not distinguish between the two materials. For both fluences, the increments in ΔT_o and CVN ΔTT_{41J} are compared in Tables 2.2 and 2.3. There is some irregularity in the CVN results. The copper content in the WF-70 beltline material is nominally 0.25 wt % ($\Delta T_o = 81°C$, $\Delta T_{41J} = 103°C$, and $\Delta T_{LE} = 85°C$). The copper content in the nozzle course material is

Table 2.1. Before-and-after irradiation yield and tensile strengths

Test temperature (°C)	Unirradiated			Irradiated					
				0.5×10^{19} n/cm^2			1×10^{19} n/cm^2		
	Number of specimens	Strength (ksi)		Number of specimens	Strength (ksi)		Number of specimens	Strength (ksi)	
		Yield	Ultimate tensile		Yield	Ultimate tensile		Yield	Ultimate tensile
Beltline WF-70 weld metal									
288	2	68.0	88.4						
150	2	69.0	84.7	2	84.4	97.0	2	86.2	101.1
22	2	74.3	88.9	2	91.9	104.0	2	93.7	108.3
−50	2	82.6	100.6						
−100	2	90.7	110.8						
−150	1	106.9	123.4						
Nozzle course WF-70 weld metal									
288	2	70.2	89.0				2	91.1	103.9
150	2	70.4	85.1	2	72.9	92.3	2	92.0	104.5
22	2	79.0	94.9	2	86.4	102.9	1	101.7	114.8
−50	2	84.0	104.1						
−100	2	94.0	118.9						

Table 2.2. Features of Charpy transition curve indices

Energy criteria						
Material	41-J temperature (°C)			Charpy upper-shelf energy (J)		
	Unirradiated	Irradiated to 0.5×10^{19} n/cm^2	Irradiated to 1×10^{19} n/cm^2	Unirradiated	Irradiated to 0.5×10^{19} n/cm^2	Irradiated to 1×10^{19} n/cm^2
Beltline	–9	36	94	88.5	80.8	80.4
Nozzle course	–1	62	89	87.7	69.7	68.2

Material	Irradiated	Transition temperature change (°C)			Change in upper-shelf properties (%)	
		ΔTT_{41J}	$\Delta TT_{50\%}$[a]	ΔTT 50% lateral expansion[a]	Joules	Lateral expansion
Beltline	1×10^{19} n/cm^2	103	100	85	–10	–46
Nozzle course	1×10^{19} n/cm^2	90	72	65	–23	–40

[a]50% represents the mid-transition curve by energy.

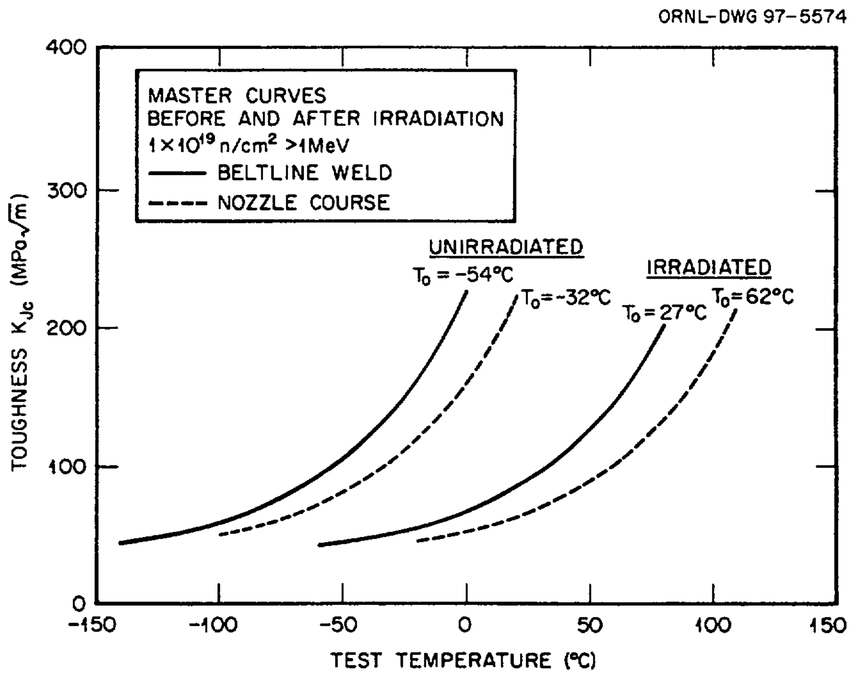

Figure 2.1. Master curves for Midland reactor beltline and nozzle course welds before and after irradiation.

Table 2.3. Property changes caused by irradiation

Fluence (n/cm^2)	ΔTT_{41J} (°C)	ΔT_o (°C)	Regulatory guide (°C)	$\Delta\sigma_{UTS}$ at room temperature (MPa)	$\Delta\sigma_{YS}$ at room temperature (MPa)	$\Delta TT_{41J}/\Delta\sigma_{YS}$ (°C/MPa)
Beltline						
0.5×10^{19}	45	78	81	104	121	0.37
1.0×10^{19}	103	81	100	134	134	0.77
Nozzle course						
0.5×10^{19}	63	NA	103	55	51	1.24
1.0×10^{19}	90	94	128	137	156	0.58

nominally 0.40 wt % ($\Delta T_o = 94°C$, $\Delta T_{41J} = 90°C$, and $\Delta T_{LE} = 65°C$). The CVN curve-shape change with irradiation damage has a detrimental influence on the accuracy of ΔTT_{41J} shift, often resulting in an extra transition-temperature shift. Table 2.3 also provides the changes in yield and ultimate strengths as well as the ratio of ΔTT_{41J} and $\Delta\sigma_{YS}$. The average ratio is 0.74, which compares well with the average determined for seven LUS welds in HSSI Series 2 and 3,[1] although the range of values in the current study is much greater than previously observed. Figures 2.2 and 2.3 show data points that represent the median of data scatter distributions. These figures make it clear that there is no curve-shape change in MC after irradiation for these welds.

Data scatter has always been an unwelcome problem associated with testing in the transition range. The MC method uses a three-parameter Weibull model to fit the data scatter and to establish data-bounding curves with assigned probability levels based on defensible statistical methods. Figures 2.4 and 2.5 show that all of the unirradiated fracture-mechanics data are bounded by 2% tolerance bounds on the data scatter. After about 200 tests, the lower 2% tolerance bound encompassed 199 data points.

Reference

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

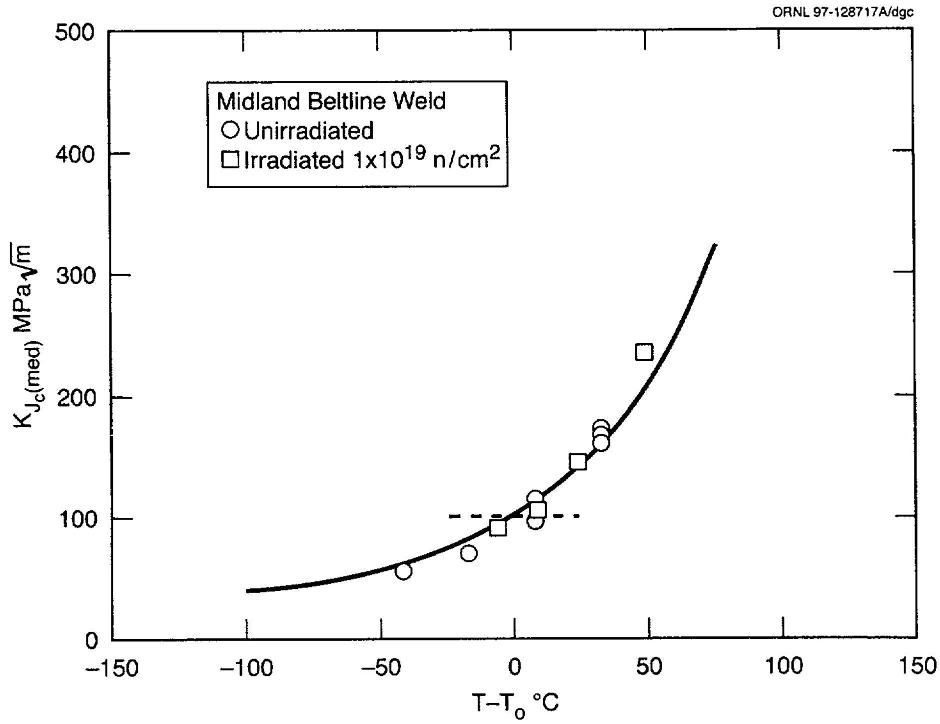

Figure 2.2. **Median fracture toughness plotted against the master curve for WF-70 beltline weld metal.**

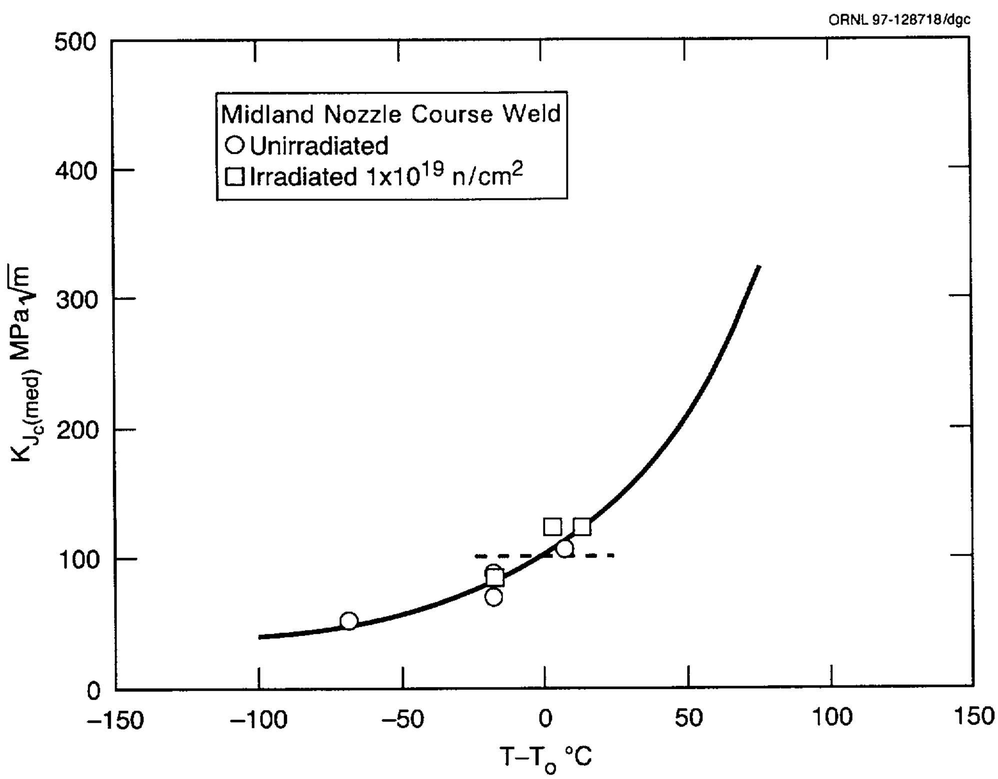

Figure 2.3. **Median fracture toughness plotted against the master curve for WF-70 nozzle course weld metal.**

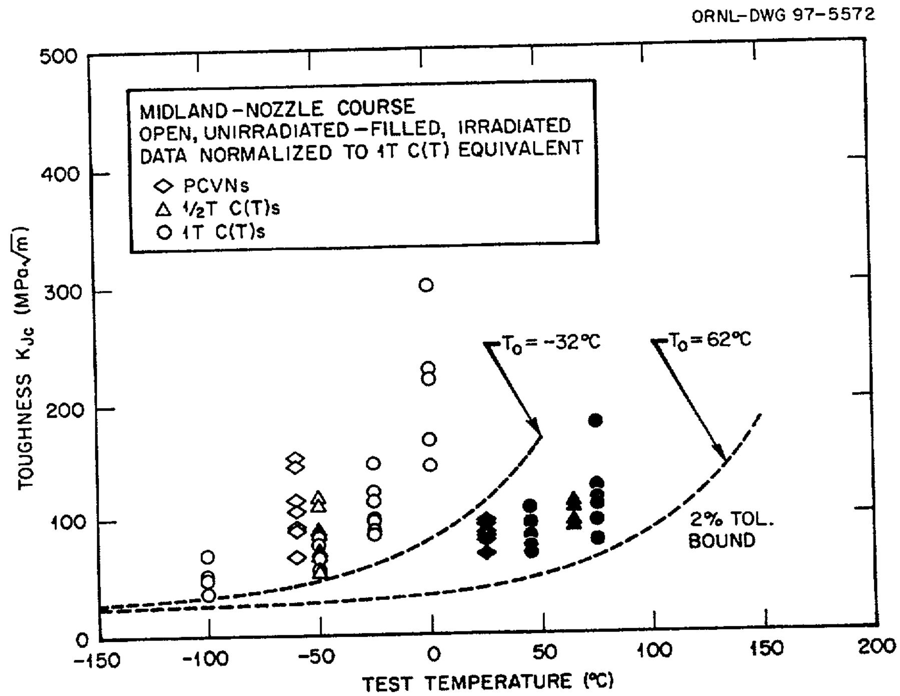

Figure 2.4. **All beltline weld metal data as normalized to 1T specimen size equivalence; 2% tolerance bound curves.**

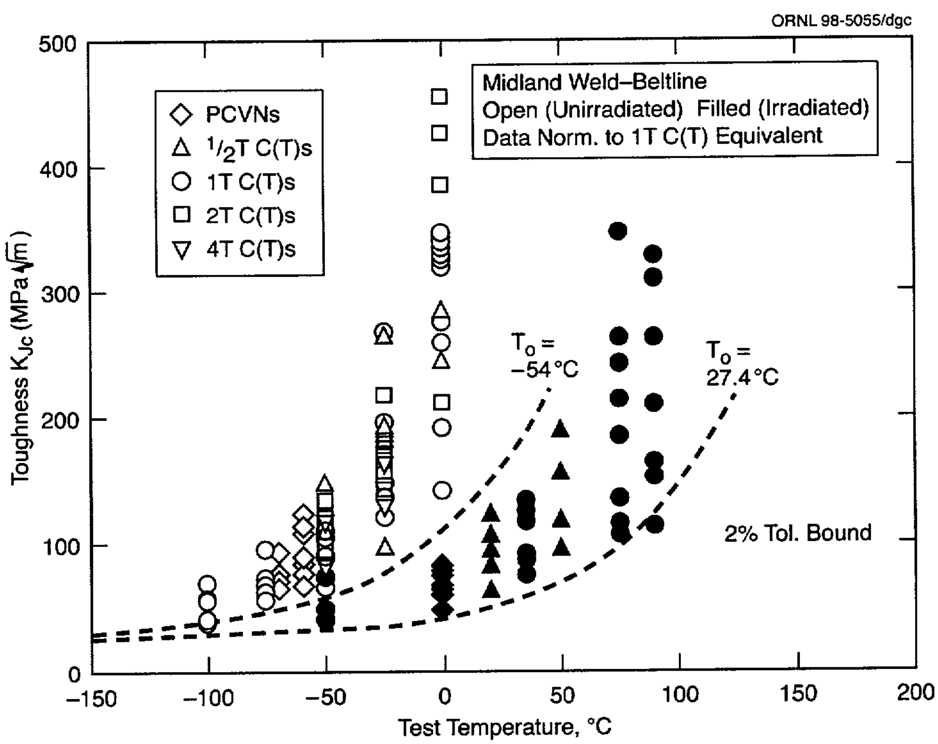

Figure 2.5. **All nozzle course weld metal data as normalized to 1T specimens size equivalence; 2% tolerance bound curves.**

3. Annealing

S. K. Iskander

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

The objective of this subtask is to determine if there is a potential problem associated with temperature embrittlement in RPV steels. An annealing experiment on laboratory heats made with steels having typical pressure-vessel chemical compositions was conducted by AEA Technology, Harwell, United Kingdom. This 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 was to reexamine the AEA Technology heat treatments using five commercially made RPV steels, representing A 302 grade B, A 533 grade B, A 508 class 2, and two modified A 302 grade B production heats (Table 3.1). The phosphorus content covers the range typical of commercial RPV production heats (Figure 3.1). Typical production steels have less than 0.02 wt % phosphorus. The AEA Technology heat treatment accentuated the temper-embrittlement phenomenon by creating a microstructure that optimizes embrittlement sensitivity. This is a preferred method when there is an experimental objective of screening materials.

The second austenitization treatment applied is a Gleeble simulation of the thermal cycle of heat-affected zone (HAZ) material. In this case, the specimens were 14-mm-diam (0.564-in.) rods. Using an austenitizing temperature of 1260°C (2300°F), the entire cycle was completed within 50 s. The target American Society for Testing and Materials (ASTM) grain size was between 4 and 5, which is typical for the coarse-grain region of HAZs. All specimens were postweld heat treated at 615°C (1140°F) for 24 h. One thermal embrittlement cycle of 450°C for 2000 h and two additional aging cycles were completed during this reporting period.

Fractography results demonstrated that the embrittlement of grain-coarsened and aged steels is associated with phosphorus segregation to grain boundaries. Auger analysis of the A 533 grade B material given AEA Technology austenitization treatment and aging at 450°C for 2000 h showed that phosphorus increased at the grain boundaries by an order of magnitude over that of the bulk material composition. When this material is tested on the lower shelf of the transition range, the mode of fracture was found to be 100% intergranular (Figures 3.2 and 3.3).

Table 3.1. Selected commercial materials

Material	Code	Content (wt %)			
		P	Ni	S	Cu
A 302 grade B	Maine Yankee	0.015	0.2	0.017	0.14
A 508 class 2	Midland	0.013	0.73	0.010	0.05
Modified A 302 grade B	GE (Z5)	0.016	0.60	0.017	0.17
Modified A 302 grade B	GE (Z7)	0.010	0.53	0.014	0.17
A 533 grade B	HSST Plate 01	0.010	0.75	0.013	0.16

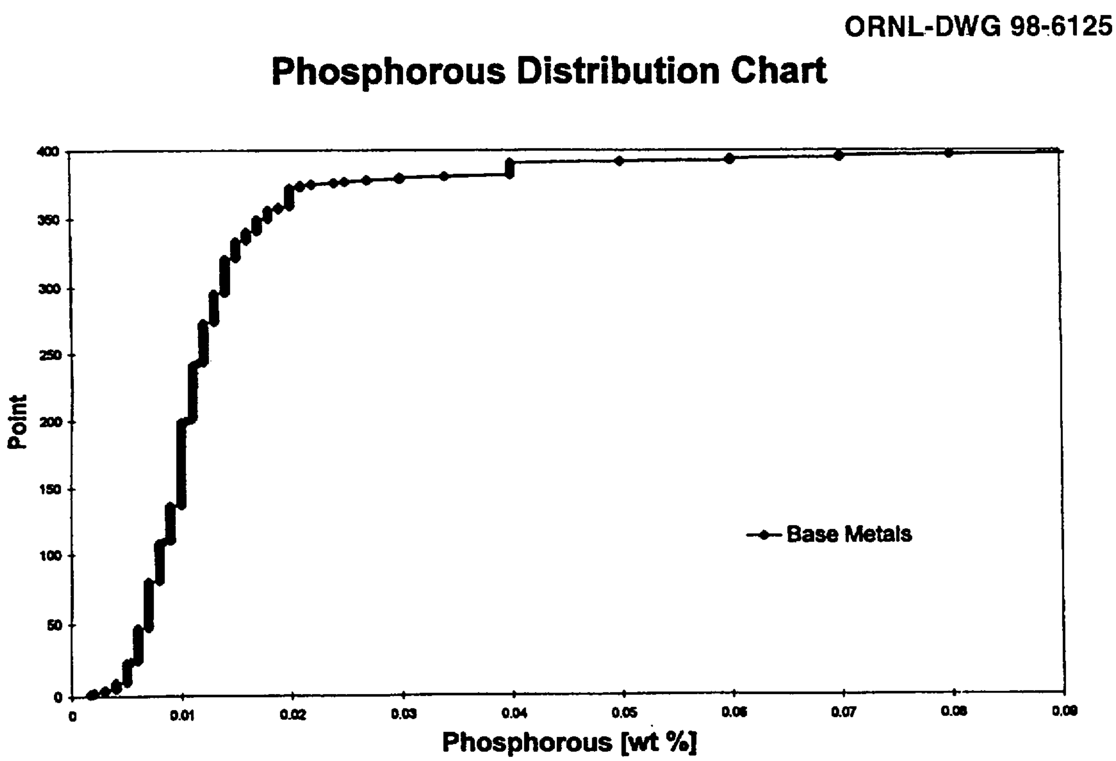

Figure 3.1. **Phosphorus content of production heats of RPV steels obtained from the Power-Reactor-Embrittlement Database.**

ORNL PHOTO 2558-98

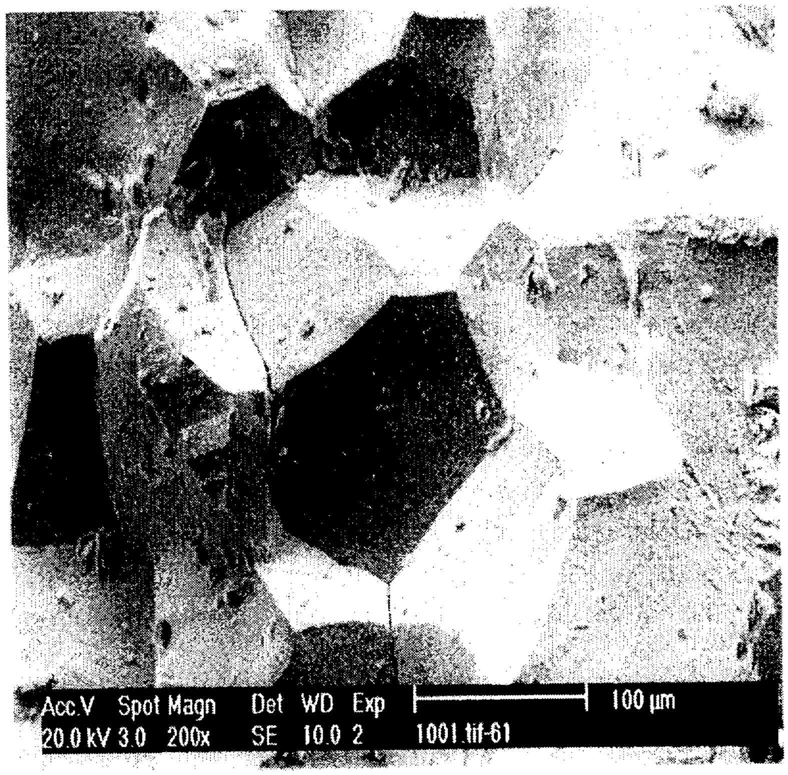

Figure 3.2. Modified A 302 grade B, low phosphorus (magnification: 200X).

ORNL PHOTO 2559-98

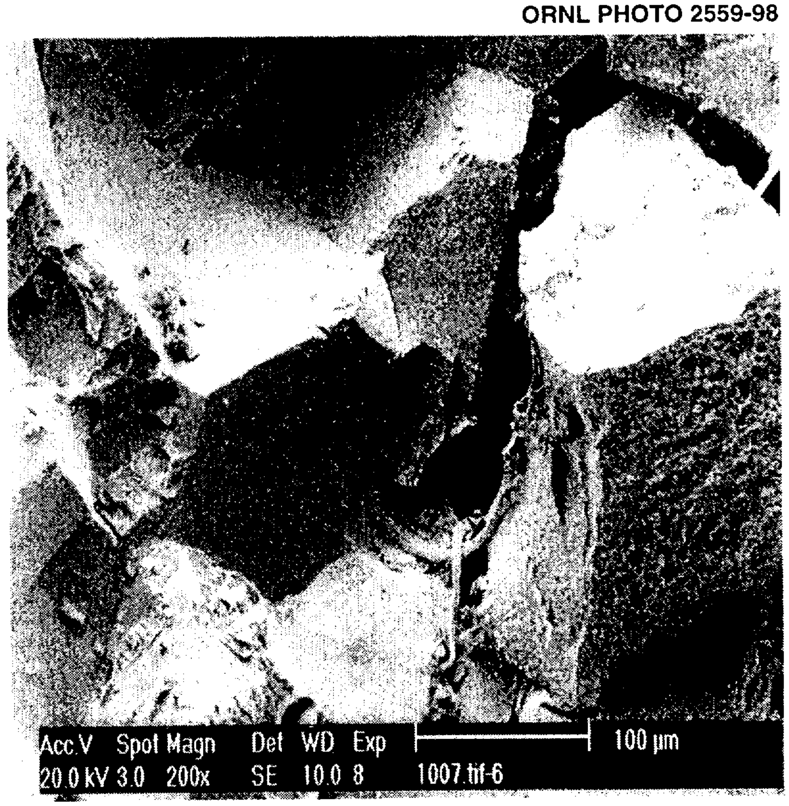

Figure 3.3. Modified A 302 grade B, high phosphorus (magnification: 200X).

The initial phase of creating coarse prior austenitic grains by the AEA Technology method and aging by three different cycles has been described previously.[1] During this reporting period, some fracture-mechanics transition-temperature determinations were made on three of the materials (Table 3.2) to compare the transition temperature by CVN curve and the MC method. The term T_0 is defined as the mid-transition temperature for the CVN test, and T_0 from the fracture-mechanics determination is the temperature where median $K_{Jc} = 100$ MPa$\sqrt{m}$. The dynamic CVN tests make the material appear to be far more damaged than it is in the case of the quasi-static fracture-toughness tests.

The work of simulating the grain-coarsened HAZ material by austenitization in the Gleeble has also continued with the addition of an evaluation of two conditions: (1) the unaged transition temperatures after Gleeble austenitization and postweld heat treatment and (2) the same as (1) but with aging at 490°C for only 168 h added. Table 3.3 includes these two conditions in rows 4 and 6. Note that the initial transition temperatures are excellent despite the coarse prior austenite grains. Aging at 450°C for 1 week causes insufficient damage to declare that there is an embrittlement problem in the local brittle zone. However, as stated previously, the very substantial embrittlement after aging 2000 h at 450°C provides motivation for investigating the combined effects of irradiation followed by thermal annealing at 450°C for 1 week.

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

(S. K. Iskander, D. W. Heatherly, C. A. Baldwin, Doug Sparks, M. T. Hurst, and R. G. Sitterson)

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

Table 3.2. Fracture mechanics plus CVN T_0 temperatures

Material	Age	T_0 (°C)		ΔT_0
		CVN	FM	
A302B	2000 h at 450°C	45	–34	79
	168 h at 490°C	44.5	–38	83
A508	2000 h at 450°C	41	–82	123
	168 h at 490°C	16	–35	51
A533B	2000 h at 450°C	78		
	168 h at 490°C	65	–48	113

Table 3.3. Charpy transition temperature, T_0 (50% energy)

Austenitizing method	Aging temperature (°C)	Aging time (h)	Transition temperature (°C)				
			A302B	A533B	A508 class 2	Modified A302B	
						High P	Low P
AEA	Initial	Initial	10	−67	4	−23	−73
	450	2000	45	78	41	34	42
	490	168	44.5	65	16	16	−3
Gleeble	Initial	Initial	−87	−67	−102		−67
	450	2000	43	66	−46	−18	−4
	490	168	−44	−28	−80		−47
As-received	Initial	Initial	0	20	0	−9	−9
	450	2000	64	38	10	−4	10

3.2.1 Reirradiation of Previously Irradiated and Annealed Charpy Specimens

As described in the previous report,[1] three containers, each containing 10 CVN specimens, were installed in the HSSI/UCSB capsule in the University of Michigan FNR. The CVN specimens were fabricated from HSSI weld 73W that had previously been irradiated and annealed at 454°C. The purpose of the reirradiation is to provide data on the rate of reembrittlement at three different fluence levels. The highest fluence level was chosen to be approximately one order of magnitude greater than the lowest and the intermediate fluence level midway between the highest and lowest. Wire dosimeters were placed in the V-grooves of Charpy specimens, and three containers with retrievable fission-radiometric dosimetric sets (FRDSs) were placed below, above, and in between the containers. The containers and retrievable FRDSs were shipped to the FNR in early January 1997 and were loaded into the HSSI/UCSB capsule by University of Michigan FNR staff with active HSSI staff participation in mid-January 1997. The FNR commenced irradiation of the capsule on January 20, 1997.

The FRDSs were retrieved from the UCSB capsule after 4 weeks of irradiation and were shipped to ORNL for counting and analysis to determine flux and spectrum. Three "dummy" inserts of low carbon steel with the same external dimensions as the three FRDSs have been loaded in place of the ones removed. Preliminary results from the FRDSs are now available, and estimates will soon be made as to the time required for the CVN specimens to accumulate the target fluence. The containers were recently rotated so that all the specimens in each container will be exposed to approximately the same dose.

3.2.2 New HSSI Irradiation, Annealing, and Reirradiation Facility

The new IAR facility has been fabricated and assembled. This facility will replace a stainless steel surrogate (dummy) facility that is presently in the FNR. A reusable capsule has also been designed and will be fabricated soon. The reusable capsule will allow either unirradiated or irradiated specimens to be loaded and specific specimens to be retrieved during refueling outages. Besides the cost savings of fabricating a new capsule for each new batch of specimens, as has been the custom, significant savings will also be achieved by avoiding the disposal cost of "hot" capsule parts. A special cask design is also being pursued that will avoid contact of the facility heater plates with the reactor pool water. The reusable capsule is to be fabricated of ferritic steel, which has the benefit of not becoming as hot as a stainless steel capsule when exposed to neutron irradiation. Because the life of a reusable capsule is limited by its activation to maximum levels that allow safe handling by reactor personnel, stainless steel reusable capsules would have to be replaced and disposed of relatively more often than ferritic capsules when their activation reaches a level that prohibits further use.

3.2.3 Instrumentation for Control and Data Acquisition

The data acquisition computer (DAC) system that is currently attached to the UCSB capsule is also attached to the Internet. The Internet connection allows the capsules at the FNR to be monitored remotely from ORNL to ensure proper operation and that operational data will be backed up on the ORNL computers in addition to the normal DAC system records at the FNR. To operate the new IAR facilities, the current computer system has been expanded. A separate server has also been added to isolate the FNR system from possible Internet problems.

Reference

1. S. K. Iskander et al., Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., pp. 3-1 to 3-6 in *Heavy-Section Steel Irradiation Program Semiannual Progress Report for October 1996-March 1997*, USNRC Report NUREG/CR-5591, Vol. 8, No. 1 (ORNL/TM-11568), February 1998.

4. Microstructural Analysis of Radiation Effects

R. E. Stoller, M. K. Miller, K. Farrell, and P. M. Rice

4.1 Introduction

Atom-probe characterization of the copper concentration in the matrix of a model Fe–1.1 at. % Cu–1.4% Ni alloy was performed after isothermal aging for various extended times at low temperatures. The matrix copper concentrations from material annealed at 500, 550, and 600°C are approximately 10% lower than the equilibrium values predicted by the Thermocalc™ program with the Kaufman database. Isothermal annealing for 4000 h at either 400 or 300°C was not sufficient to attain the equilibrium copper concentration.

The results of a comparison of fast neutron and 2.5-MeV electron irradiation at 288°C indicate that similar radiation-induced increases in yield strength are obtained in RPV steels for both types of irradiation. Data obtained for the Heavy-Section Steel Technology (HSST-02) correlation monitor plate are compared.

Recent molecular dynamics simulations of displacement cascades in iron indicate that small interstitial clusters may have a very low activation energy for migration and that their migration is one-dimensional, rather than three-dimensional. The mobility of these clusters can have a significant impact on the predictions of radiation damage models, particularly at the relatively low temperatures typical of commercial, light-water RPVs and other out-of-core components. A kinetic model used to investigate RPV embrittlement has been modified to permit an evaluation of the mobile interstitial clusters.

4.2 Atom-Probe Measurements of Low-Temperature Copper Solubilities in Fe-Cu-Ni Alloys*

One of the most important parameters in the embrittlement of pressure-vessel steels used in nuclear reactors is the copper concentration in the body-centered cubic a-iron matrix. Atom-probe field-ion microscopy results demonstrate that some of the copper in solid solution, after the standard stress-relief treatment, forms ultrafine copper-enriched clusters during neutron irradiation.[1,2] These copper-enriched clusters are associated with other solute atoms such as Ni, Mn, Si, and P. The presence of neutron-irradiation-induced clusters has also been shown to correlate with hardening and embrittlement of the

*The authors acknowledge the contributions of Dr. P. J. Pareige from the Faculté des Sciences de l'Université de Rouen and Drs. R. C. Thomson and M. J. Starink from Loughborough University. We also thank Dr. M. L. Jenkins of Oxford University for supplying the alloy used in this study. This research was supported in part by the Division of Materials Sciences, U.S. Department of Energy, under contract DE-AC05-96OR22464 with Lockheed Martin Energy Research Corp. and was conducted using the Shared Research Equipment User Program facilities at ORNL.

steel. In addition, the copper remaining in solid solution produces a small amount of solid solution hardening. Although the equilibrium level of copper in a-iron can be predicted by thermodynamic-based programs, no experimental determinations of copper solubility have been performed in the low-temperature regime of interest to nuclear power reactors.

The issue of copper solubility is also relevant to the long-term benefit gained by attempts to reduce or eliminate embrittlement by annealing the irradiated pressure vessel at low temperature (340 to 454°C).[3,4] However, few experiments have been performed to determine whether these annealing treatments dissolve the neutron-irradiation-induced copper-enriched precipitates or whether they coarsen the precipitates.[5] In addition, little data are available on the time required to reduce the influence of these copper-enriched precipitates on embrittlement at low temperatures. Therefore, it is important to estimate the copper concentration in the matrix of these alloys as a function of both temperature and time.

To realistically model the influence of copper on mechanical properties, it is important to distinguish between copper in solid solution and copper associated with copper-containing embryos, clusters, and precipitates. Atom-probe field-ion microscopy is the only analytical technique with sufficient spatial resolution to perform this type of characterization.

In this study, the matrix copper concentrations were determined by atom-probe field-ion microscopy in a model Fe–1.1 at. % Cu–1.4% Ni alloy after isothermal aging for various extended times at low temperatures. This material is a simplified, high-copper model of the steel used in the welds of the pressure vessel of a nuclear reactor. The bulk composition of this alloy was determined by wet chemical analysis to be Fe–1.13 at. % Cu, 1.36% Ni, 0.009% P, 0.05% C, and 0.031% Al. After an initial solution treatment of 0.5 h at 1000°C, the materials were aged for 8 h at 850°C and were water quenched [aqueous quenched (AQ)]. The specimens were then isothermally aged at 300, 400, 500, 550, and 600°C at times up to 1,000 h. Longer times up to 4,000 h were used for the 300 and 400°C annealing treatments because of the slower diffusion at lower temperatures. The specimens were quenched to room temperature in water after each isothermal aging treatment. The microhardness of each aging condition was determined with a Shimadzu hardness tester with a 25-g load. Short-term (100 h) aging studies of this material in the temperature range of 400 to 550°C have been reported previously.[6,7]

Atom-probe characterization of this material was performed in the ORNL energy-compensated atom probe. Field-ion specimens were fabricated with the standard two-stage electropolishing procedure. The electrolytes used were a mixture of 25% perchloric acid in glacial acetic acid in the first stage and a solution of 2% perchloric acid in 2-butoxythanol in the second stage.[8] Field-ion images were recorded with neon as the image gas. Atom-probe analyses were performed with a specimen temperature of 50 K and a pulse fraction of 20%. Pulse repetition rates between 50 and 1500 Hz were used. These conditions have previously been shown to minimize the possibility of preferential evaporation and retention of atoms during field evaporation in this type of material. To eliminate the possibility of the surface layers of the specimen having a different local composition arising from redeposition of copper during the electropolishing process, the probe aperture was placed in the central region of the field-ion image and the surface layers of the specimen were not used in these measurements. All compositions are given in atomic percent.

A summary of the copper levels measured in the matrix for all aging treatments examined is shown in Table 4.1. The copper concentration in the matrix of the material annealed at 850°C (0.91 at. % Cu) is lower than the bulk value (1.13 at. % Cu) because of segregation and the formation of copper precipitates at the grain boundaries and dislocations.

Table 4.1. Change in copper content of the matrix as a function of aging time and temperature

Time (h)	Copper concentration, at. %		
	30°C	400°C	500°C
0.5	–	–	0.47 (±0.11)
1	–	0.92 (±0.08)	0.31 (±0.10)
4	–	–	0.13 (±0.03)
10	–	0.91 (±0.09)	0.13 (±0.03)
100	–	0.54 (±0.08)	0.073 (±0.02)
168	0.93 (±0.06)	–	–
1000	0.82 (±0.06)	0.12 (±0.03)	0.082 (±0.03)
4000	0.80 (±0.05)	0.085 (±0.01)	–
	550°C	600°C	850°C (AQ)
5	–	–	0.91 (±0.04)
1000	0.12 (±0.03)	0.18 (±0.03)	–

These atom-probe results indicate that the equilibrium solubility limit is reached in less than 100 h at 500°C. This result is in agreement with the microhardness data. Because it is unlikely that the copper solubility is higher at lower temperatures, these results also indicate that the equilibrium concentration has not been attained after aging for 4000 h at 400°C. Only a small but significant decrease in the copper level was found after aging for 4000 h at 300°C. This lack of decomposition correlates with the microhardness results.

These results indicate that the typical time of 168 h (at ~450°C) proposed for postirradiation annealing of RPVs at low temperature may not be sufficient to remove all the copper from solid solution. Therefore, a slight improvement in the rate of reembrittlement on exposure to further neutron irradiation could be achieved by adopting a longer annealing time. In addition, the use of a temperature at the high end of the range or even slightly higher would significantly improve the diffusion of the copper and reduce reembrittlement by the mechanism of copper precipitation. Because solid-solution hardening is typically proportional to the square root of the solute concentration in low solute materials, there is a decrease of ~70% in the contribution from the copper in the matrix in the longest-aged materials at 400 and 500°C compared with the unaged material. This decrease more than offsets the contribution from precipitation hardening; therefore, the overall microhardness is higher in the annealed materials than in the material subjected only to the 850°C annealing. As the copper-enriched precipitates coarsen, their average size increases but their number density decreases. Therefore, these precipitates present less of an obstacle to the passage of dislocations; hence, the microhardness decreases as the material over-ages.

These atom-probe results are compared with thermodynamic predictions in the phase diagram shown in Figure 4.1. The matrix copper concentrations from material annealed at the longest times at higher temperatures (500, 550, and 600°C) are approximately 10% lower than the equilibrium values predicted from the Thermocalc™ program with the Kaufman database. Experimental data from the material after annealing at the two lower aging temperatures of 300 and 400°C exhibited significantly higher copper concentrations than predicted. This behavior is caused by the slow diffusion of copper at these low temperatures; therefore, the equilibrium level has not been attained, making comparison of these conditions inappropriate. A more detailed description of this work will be published soon.*

4.3 Electron Irradiation Hardening of Ferritic Alloys at 288°C†

Irradiations with energetic electrons can cause atomic displacements and radiation hardening in metals.[9,10] In nuclear reactors, some of the atomic displacements in the RPV are caused by electrons generated via the processes of Compton scattering and electron-positron pair production by high-energy gamma rays emanating from materials in the core region. The contribution of gamma (electron)-induced displacements to the total displacements responsible for hardening and embrittlement in most RPVs is negligible compared to the displacements from fast neutrons.[11] The atomic displacement cross section for 1-MeV gamma rays in iron alloys is only about 1 b vs about 1500 b for 1-MeV neutrons. Because, in most reactors, gammas and neutrons with energies higher than about 1 MeV are produced in roughly equal quantities, the damage from gamma displacements to the RPV is relatively small. However, if there is a long path of water between the core and the RPV, the neutrons will be attenuated more strongly than the gamma rays and the fraction of damage contributed from gamma displacements in the RPV will increase and in exceptional cases may exceed the damage from the neutrons. Such was the case[12,13] for the High Flux Isotope Reactor (HFIR) RPV. The irradiation strengthening efficiency of the gamma (electron) displacements, per dpa, in the HFIR RPV seemed to be the same as that of fast neutrons. This latter point was confirmed by subjecting tensile specimens of the archive steel of the HFIR vessel and other ferritic alloys to electron bombardments in a particle accelerator and comparing their properties with those for the same materials irradiated to the same displacement levels ($1.4–5.3 \times 10^{-3}$ dpa) with neutrons.[14]

However, the HFIR RPV-related electron hardening experiments on ferritic alloys were performed at a temperature of about 60°C, in keeping with the low operating temperature of the HFIR RPV. For the RPVs of commercial power reactors, a temperature of 288°C is more relevant. Hence it is advantageous to know the hardening efficiency of electron displacements at this temperature as well. An initial experiment using 2.5-MeV electrons at 288°C has been completed. The materials included two model alloys, pure iron and Fe–0.28% Cu, the A212B steel used in the HFIR RPV and the A533-B reference HSST-Plate 02 correlation monitor. Two flat SS-3 type tensile specimens of each material, with gauge

*M. K. Miller, K. F. Russell, P. J. Pareige, M. J. Starink, and R. C. Thomson, "Low Temperature Copper Solubilities in Fe-Cu-Ni," presented at the 44th International Field Emission Symposium, Tsukuba, Japan, June 7–11, 1997, to be published in *Materials Science and Engineering* (1998).

†The authors acknowledge the contributions of Drs. P. Jung and H. Ullmaier, Forschungszentrum Julich GmbH, Postfach 1913, D-5170 Julich, Germany.

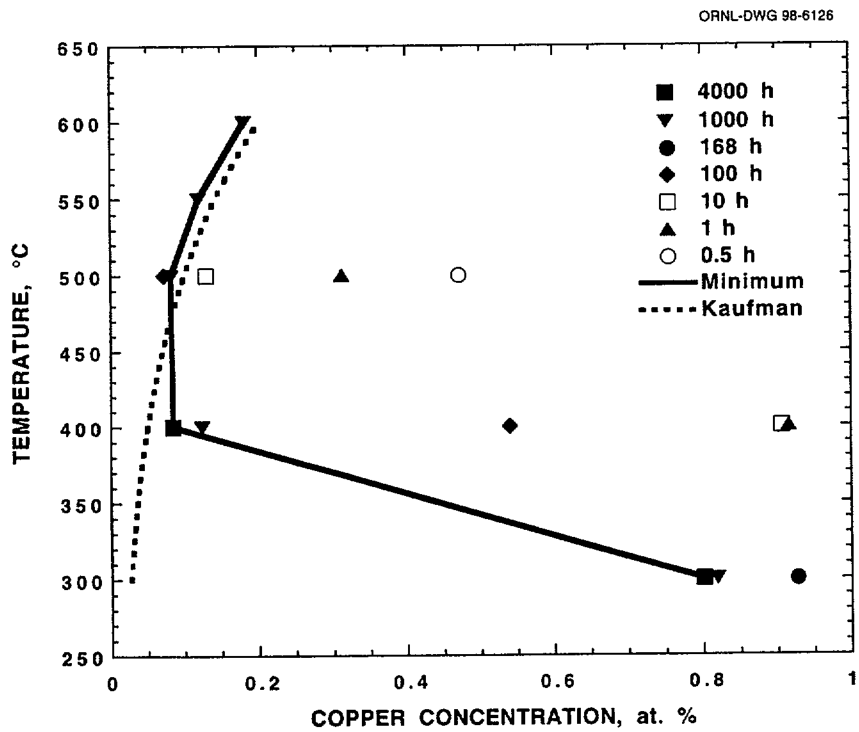

Figure 4.1. Comparison of atom-probe experimental data for matrix copper concentrations and predictions of the Thermocalc program.

sections of 1.52 × 0.76 mm and gauge length of 7.6 mm, were mounted in an upright array in the target chamber of the Van de Graaff accelerator at the Institut für Festkörperforschung, KFA, Julich. The specimens were irradiated with 2.5-MeV electrons for 57 and 222 h to electron doses of 2.82×10^{19} e-/cm^2 and 9.35×10^{19} e-/cm^2, corresponding to displacement doses of 1.2×10^{-3} and 4.0×10^{-3} dpa. The average current density in the few millimeter diameter beam spot was 0.2 A/m^2, and the beam spot was rastered horizontally and vertically over the specimen gauge areas at a rate of 500 scans per second. Cooling was provided by water-chilled helium gas pumped across the faces of the specimens via five nozzles located on each vertical edge of the specimen assembly. Temperatures were measured with a scanning infrared pyrometer normalized to the readings from a thermocouple welded to the gauge section of one of the specimens and were held in the range of 278 to 298°C. To homogenize the small gradient in damage through the thickness of the specimens caused by their attenuation of the beam, the specimen assembly was rotated 180° halfway through each run. Unirradiated control specimens were sealed in evacuated glass ampules and were furnace-heated at 288°C. Tensile tests were made in a screw-driven Tinius-Olsen machine under computer control at room temperature and at a strain rate of 1.1×10^{-3}/s. Tensile yield strengths were read at the lower yield point inflection or at 0.2% strain offset when there was no yield inflection.

Although the full set of data will be published elsewhere,* the key results are summarized in Figure 4.2, where the electron-induced increases in yield strengths for the HSST-02 plate steel are compared with the yield strength changes in neutron-irradiated specimens made from the same plate material and similar A533B alloy irradiated with neutrons at 288°C. For the other three materials, there are insufficient tensile data for 288°C neutron irradiations to make a satisfactory comparison. In Figure 4.2, the electron hardening is the same as the neutron hardening at the same dpa levels. This agrees with the observation made for irradiations of steels at temperatures <100°C.[14]

It is also obvious from Figure 4.2 that the range of dpa for which the agreement between electron hardening and neutron hardening is demonstrated is quite narrow and is at low levels of dpa. The latter is a regrettable shortcoming of electron irradiations and places a severe limitation on the generality of the seeming one-to-one relationship of electron hardening and neutron hardening. It is possible that the observed equality of hardening efficiency of electron and neutron damage represents presaturation hardening and may not hold at higher atomic displacement doses. There is evidence that saturation of electron damage will occur at higher doses,[15] which could lead to neutron hardening to outpace electron hardening. Further electron irradiations at higher doses are needed to explore this aspect.

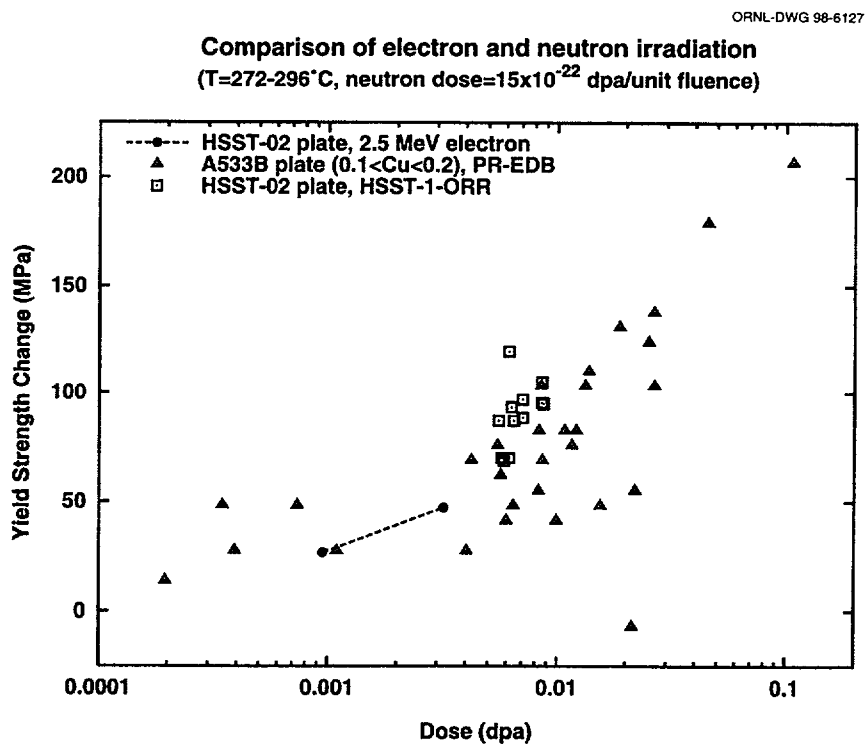

Figure 4.2. Comparison of measured yield-strength changes in A533B steel following either fast-neutron or 2.5-MeV electron irradiation.

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*K. Farrell, R. E. Stoller, P. Jung, and H. Ullmaier, "Electron Irradiation Hardening of Ferritic Alloys at 288°C," to be submitted to Journal of Nuclear Materials, 1998.

4.4 Effect of Point Defect Cluster Mobility in a Kinetic Embrittlement Model

Results of recent molecular dynamics simulations of displacement cascades in iron indicate that small interstitial clusters may have a very low activation energy for migration and that their migration is one-dimensional rather than three-dimensional. The mobility of these clusters can have a significant impact on predictions of radiation damage models, particularly at the relatively low temperatures typical of commercial, light-water RPVs and other out-of-core components. It has been suggested that the potential impact of such cluster mobility implies that a complete reexamination of the rate theory models used to simulate microstructural evolution is warranted.[16] The kinetic model developed in this task to investigate RPV embrittlement has been modified to permit an evaluation of the mobile interstitial clusters. Sink strengths appropriate to both one- and three-dimensional motion of the clusters have been evaluated.

An extensive comparison of the effects of interstitial cluster mobility is being published elsewhere;* however, the overall results can be summarized as follows. High cluster mobility leads to a reduction in the amount of predicted embrittlement caused by interstitial clusters because they are lost to sinks rather than built up in the microstructure. This increases the sensitivity of the predictions to displacement rate. The magnitude of this effect is greater at low temperatures where point defect clusters contribute significantly to embrittlement[17] and is somewhat reduced if the migration is one-dimensional because the corresponding sink strengths are lower than those for three-dimensional diffusion. Cluster mobility can also affect the evolution of copper-rich precipitates in the model since the radiation-enhanced diffusion coefficient increases because of the lower interstitial-cluster sink strength.

An example of the model predictions is shown in Figure 4.3, where results are shown for an irradiation temperature of 288°C. Values were calculated for two displacement rates. The lower is typical of commercial reactor surveillance programs, and the higher is representative of test reactor irradiations. Data are shown from irradiations in the University of Virginia Reactor.[18] The effects of cluster mobility are modest at this higher temperature, where most of the strengthening is a result of copper-rich precipitates rather than point defect clusters.[17,19] Cluster mobility leads to less strengthening at doses between $\sim 10^{-3}$ and 10^{-2} dpa. At the higher displacement rate, this leads to somewhat poorer agreement with the dose dependence observed in the test reactor data.

It is possible that this loss of agreement between model predictions and experimental data could be compensated for by justifiable adjustment of material parameters in the calculations. Specific parameter choices that were required in the initial model to obtain agreement with the data could have acted as a surrogate for interstitial cluster mobility. For example, relatively modest changes in the activation energy for self-diffusion or interstitial migration, or the binding energies of small interstitial clusters, can have a large impact on steady-state cluster concentrations. However, the increased dose-rate dependence arising

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*R. E. Stoller, "The Impact of Mobile Point Defect Clusters in a Kinetic Model of Pressure Vessel Embrittlement," presented at the 18th International Symposium on the Effects of Radiation on Materials, Hyannis, Mass., June 25–27, 1996, submitted for publication to the American Society of Testing and Materials.

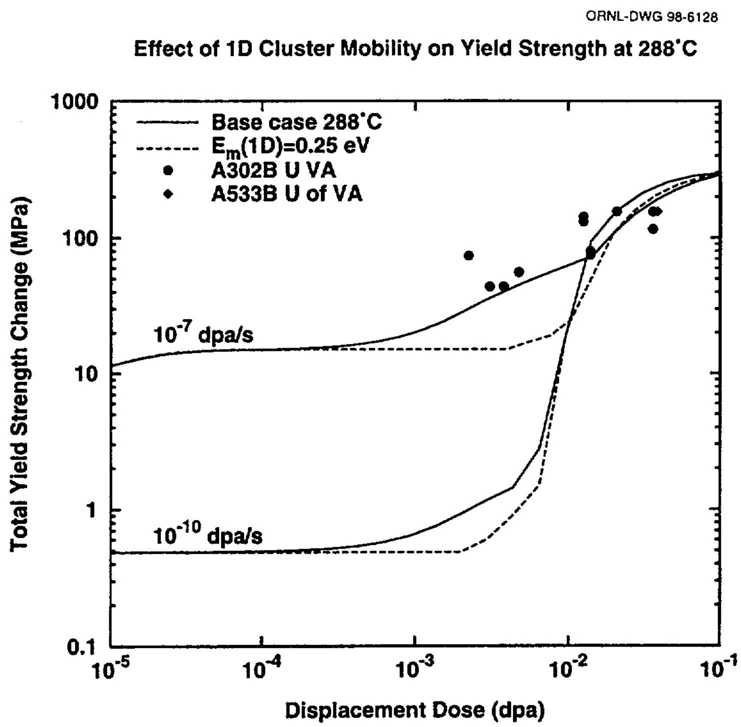

Figure 4.3. Comparison of calculated yield-strength changes in experimental data for 288°C irradiation at indicated displacement rates; one-dimensional interstitial cluster migration energy is 0.25 eV.

from cluster mobility is unlikely to be reduced by simple parameter changes. Although partitioning the small clusters into sessile and glissile components would reduce rate dependence, accounting for the mobility of interstitial clusters larger than 4 is likely to increase the rate dependence. Further calculations with more detailed models are needed to investigate these issues.

References

1. M. K. Miller, M. G. Hetherington, and M. G. Burke, *Metallurgical Transactions*, **20A**, 2651 (1989).

2. M. K. Miller and M. G. Burke, *Journal of Nuclear Materials*, **195**, 68 (1992).

3. J. R. Hawthorne, H. E. Watson, and F. L. Loss, p. 278 in *Effects of Radiation on Structural Materials*, ASTM STP 683, J. A. Sprague and D. Kramer, Eds., American Society for Testing and Materials, Philadelphia, 1978.

4. R. G. Lott, T. R. Mager, R. P. Shogan, and S. E. Yanichko, p. 242 in *Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels, An International Review*, Vol. 2, ASTM STP 909, L. E. Steele, Ed., American Society for Testing and Materials, Philadelphia, 1986.

5. P. J. Pariege, R. E. Stoller, K. F. Russell, and M. K. Miller, "Atom Probe Characterization of the Microstructure of Nuclear Pressure Vessel Steel Surveillance Materials After Neutron Irradiation and After Annealing Treatments," *Journal of Nuclear Materials* **249**, 165–174 (1997).

6. G. M. Worrall, J. T. Buswell, C. A. English, M. G. Hetherington, W. J. Phythian, and G. D. W. Smith, *Journal of Nuclear Materials*, **148**, 107 (1987).

7. P. J. Pareige, K. F. Russell, and M. K. Miller, *Applied Surface Science*, **94/95**, 362 (1996).

8. M. K. Miller, A. Cerezo, M. G. Hetherington, and G. D. W. Smith, *Atom Probe Field Ion Microscopy*, Oxford University Press, Oxford, England, 1996.

9. C. E. Dixon, pp. 475–487 in *Progress in Nuclear Energy*, Series V, Vol. 2, Pergamon Press, 1959.

10. M. J. Makin and T. H. Blewitt, *Acta Metallurgica* **10**, 241 (1962).

11. R. Gold, J. H. Roberts, and D. G. Doran, p. 603 in *Reactor Dosimetry: Methods, Applications, and Standardization*, ASTM STP 1001 (1989).

12. K. Farrell, S. T. Mahmood, R. E. Stoller, and L. K. Mansur, *Journal of Nuclear Materials* **210**, 268 (1994).

13. I. Remec, J. A. Wang, F. B. K. Kam, and K. Farrell, *Journal of Nuclear Materials* **217**, 258 (1994).

14. D. E. Alexander, L. E. Rehn, K. Farrell, and R. E. Stoller, *Journal of Nuclear Materials* **228**, 68 (1996).

15. P. Auger, P. J. Pareige, M. Akamatsu, and J.-C. Van Duysen, *Journal of Nuclear Materials* **211**, 194 (1994).

16. C. H. Woo and B. N. Singh, "The Concept of Production Bias and Its Possible Role in Defect Accumulation under Cascade Damage Conditions," *Physica Status Solidi (B)* **159**, 609–616 (1990).

17. R. E. Stoller, "Pressure Vessel Embrittlement Predictions Based on a Composite Model of Copper Precipitation and Point Defect Clustering," pp. 25–59 in *Effects of Radiation on Materials*, ASTM STP 1270, D. S. Gelles, R. K. Nanstad, A. S. Kumar, and E. A. Little, Eds., American Society of Testing and Materials, Philadelphia, 1996.

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

19. G. R. Odette, "On the Dominant Mechanism of Irradiation Embrittlement of Reactor Pressure Vessel Steels," *Scripta Metallurgica* **17**, 1183–1188 (1983).

5. In-Service Irradiated and Aged Material Evaluations

S. K. Iskander

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

During this reporting period, the mill and saw were installed and made operational in cell 6 in Building 3025E at ORNL. Before the saw and computer numerically controlled (CNC) machine were installed, hot cell 6 was decontaminated. In retrospect, this task was monumental in every aspect. Special stands for the saw and mill had to be designed, fabricated, and assembled. In particular, the mill and controller had to be separated into three units. The mill was placed inside the cell, and one control unit (the "front controller"), consisting of an on/off switch, emergency shutoff button, keyboard, monitor, and disk drive, was assembled into a separate unit that was placed in front of the cell beside the cell 6 window. The circuitry, with various plug-in boards that control motion in the three axes, spindle motor that also rotates the tool turret, etc., were mounted on a specially fabricated enclosure in the rear of the cell to minimize the cable length between these boards and the mill inside the cell. More than 30 drawings were prepared with details of every aspect of the project. Representative drawings, shown in Figures 5.1 through 5.3, convey the magnitude of installing this equipment in the hot cell. Although too small to be legible in every aspect, these drawings do convey a sense of the size and complexity of this task. Figure 5.1 shows the mill and saw as installed in the hot cell. The wiring diagram is shown in Figure 5.2, where each line represents a cable with up to 20 separate strands. Detailed connection diagrams for each strand were also prepared. All cables from inside the cell must be attached to Amphenol connectors, which allow the equipment to be connected or disconnected. The cabling for these Amphenol connectors must pass through radiation stainless/lead shielding plugs to a connection box or a connector on the outside of the cell and then to either the "rear-control" or the "front-control" box. Figure 5.3 shows some of the circuitry for the mill, saw, and associated lubrication and coolant pumps that was assembled in a specially fabricated enclosure.

Another example of the details that had to be designed and checked were lifting bars for the mill and saw. The top of the lift bar had to be located over the center of gravity, which had to be determined experimentally. Note that neither piece of equipment was designed to be lifted in the manner needed for installation in the hot cell. Stress analyses of the preliminary lifting frame geometries for both the saw and the mill also had to be performed. The lifting fixtures for the saw and the mill were successfully certified according to U.S. Department of Energy and ORNL procedures by testing at 125% of rated capacity.

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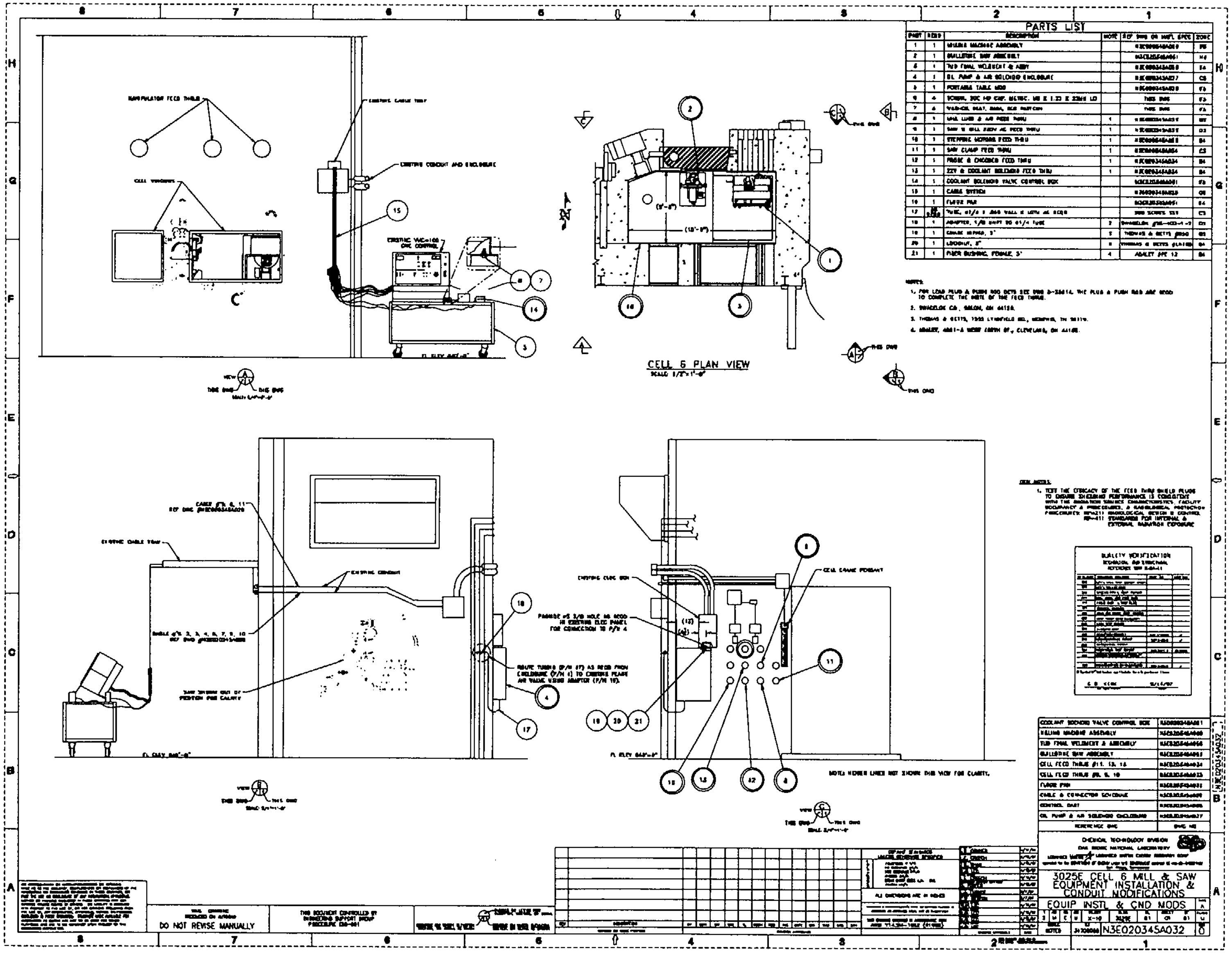

Figure 5.1. Drawing of mill and saw as installed in the hot cell.

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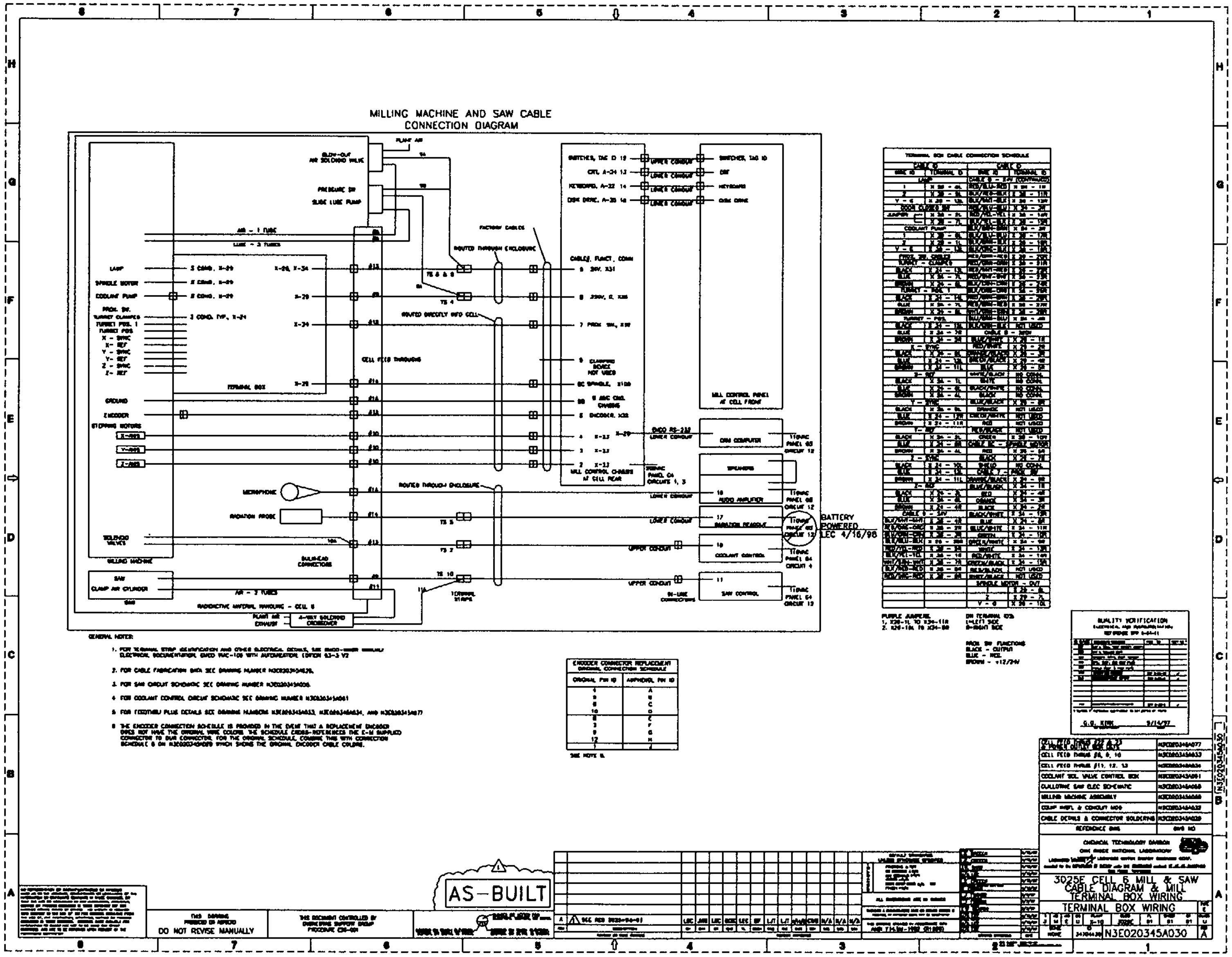

Figure 5.2. Wiring diagram in which each line represents a cable with up to 20 separate strands.

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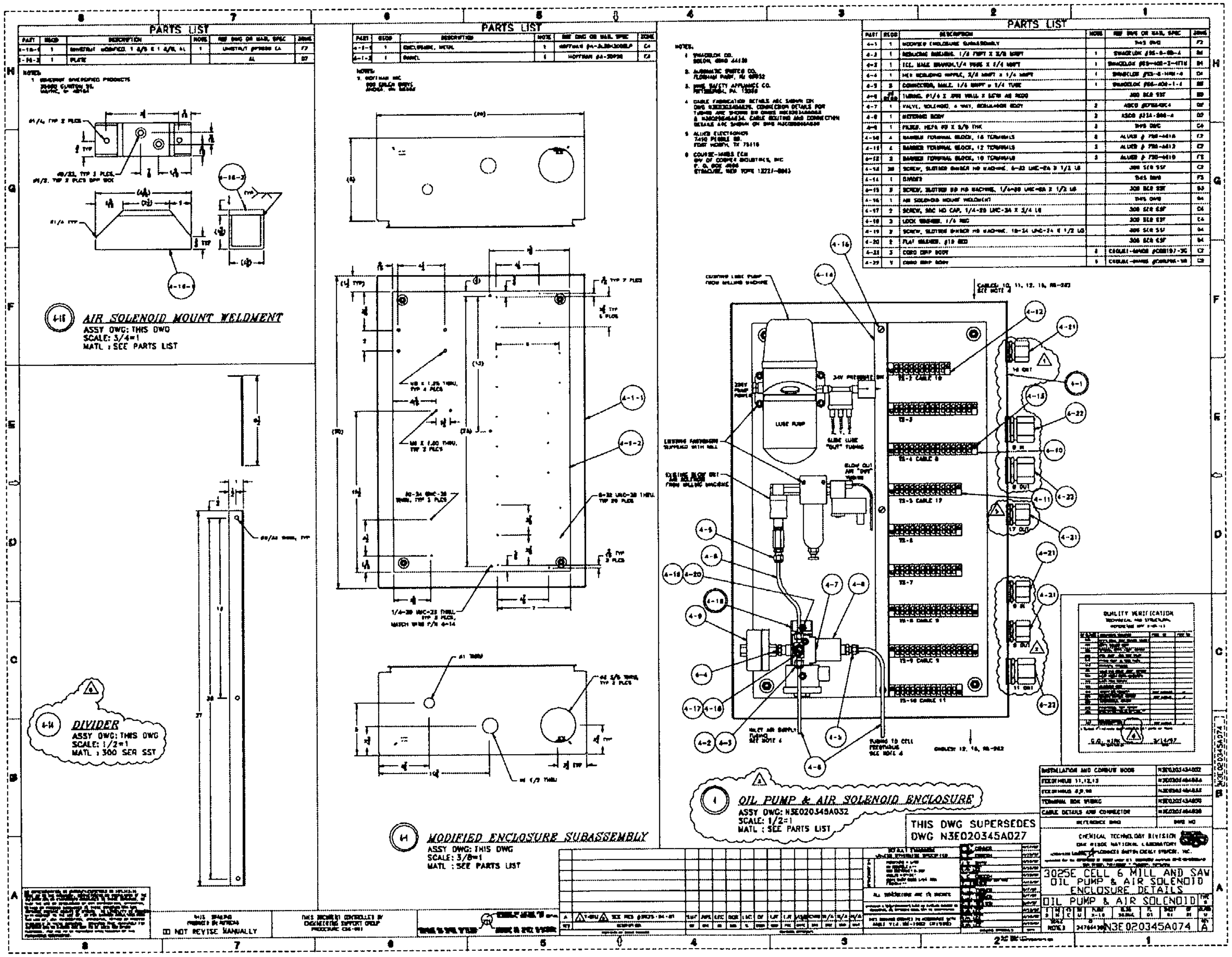

Figure 5.3. Details of new enclosure for circuitry for the mill, saw, and lubrication and coolant pumps.

Another difficult aspect of this task was the manufacture and installation of a single stainless steel "tub" for both the mill and the saw for support and to catch the cutting fluids used in machining. Installation of this large pan through the limited openings into a hazardous environment was successful only because of the perseverence and ingenuity of those involved.

Before installation into the hot cell, two experts from the mill manufacturer, EMCO-MAIER, were contracted to inspect the mill and replace marginally operating components. A computer-aided design/computer-aided manufacturing (CAD-CAM) code, WINCAM, was purchased and installed on a computer dedicated to mill operation. A special cable connecting the computer to the front controller through the RS 232 post was also purchased from EMCO-MAIER. Because of poor documentation and difficulties communicating with technical support at EMCO-MAIER, this phase was also difficult. Contrary to the original understanding, the WINCAM code is apparently the only viable communication tool for the EMCO-MAIER "Emcotronic TM02" mill controller. The mill will not accept control codes generated by other CAD-CAM codes such as the industry standard control codes (the so-called "FANUC" control). ORNL, however, was able to generate test machining instructions, export them from the computer to the mill computer control, and perform some sample test machining.

A limited amount of experience was gained using the WINCAM code to generate machining instructions. Machine instructions are highly specific to the task at hand, depending on the form of the raw material and the desired final product form. Task-specific fixtures also have to be designed and fabricated, the purpose of which is to hold the raw stock and intermediate products in place during machining.

A second EMCO-MAIER VMC-100 CNC machine has been purchased and installed in the Fracture Mechanics Laboratory at ORNL. According to EMCO-MAIER, this machine, which cost less than one-half the first machine, was used for demonstrations. The machine model is identical to the one installed in the hot cell and will be used for prototyping activities, namely the development of the fixtures, tooling, and computer commands destined to be used for machining irradiated material. Because of the accessibility of the second machine, it will be significantly cheaper to undertake the necessary developmental work that would otherwise have had to be performed on the hot cell machine. This will also save wear and tear on the hot cell machine. This model is no longer being manufactured; hence, the second machine could possibly provide parts for the hot cell machine in an emergency situation.

After installation of the mill and saw in the hot cell, various modifications were undertaken to optimize operation. For example, a manufacturing consultant strongly recommended placing a microphone near the mill to monitor operation. Sounds generated during machining can convey whether the operation is proceeding satisfactorily. A microphone, amplifier, and speakers have been procured and installed. Separation of the spindle motor power from other power cables has also been completed, which will enable more reliable operation of the mill. A separate ground cable attached to all the outer shields on the cables has also been installed. Many of these steps were taken to eliminate random signals observed during operation of the mill and are assumed to relate to cross talk between the various cables. Placement of these cables through radiation shields is progressing.

6. Fracture-Toughness Curve Shift Method

M. A. Sokolov

6.1 Introduction

The objectives of this task are to (a) examine the technical basis for the currently accepted methods for shifting fracture-toughness curves to account for irradiation damage and (b) work through national codes and standards bodies to revise those methods if revision is warranted. Specific activities under this task include

- collection and statistical analysis of pertinent fracture-toughness data to assess the shift and potential change in the shape of the fracture-toughness curves caused by neutron irradiation, thermal aging, or both;
- 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 Pt. 50), Appendix H;
- participation in the pertinent American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Sect. XI, ASTM E-8 and E-10, committees to facilitate obtaining data and disseminating results of the research;
- interaction with other researchers in the national and international technical community who are addressing similar problems; and
- 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 regulatory issues.

6.2 Master Curve Technology

An important benefit of MC technology is the ability to employ small specimens to characterize the fracture toughness of larger specimens with thicknesses that approach the thickness of materials used in actual applications. This work is aimed at determining whether precracked Charpy V-notch (PCVN) or even smaller specimens can be used for MC establishment. In the previous report,[1] we presented preliminary results of such a study on HSST Plate 02 material with different specimen geometries. The results showed that data from PCVN and its smaller equivalent specimens (4.8 × 4.8 × 27 mm) follow the weakest-link size adjustment procedure. However, results from 4.8 × 10 × 55 mm and 0.2T (5-mm-thick) compact specimens fell below this model trend. These two types of specimens have one common parameter—the ratio of width (W) to the thickness (B) equal to 2, while the PCVN and its smaller equivalent specimens (4.8 × 4.8 × 27 mm) specimens have W/B equal to 1.

The disparity in results for the three different three-point bend specimen sizes is not evident from the Weibull probability plots. At least for all three specimen sizes, the scatter of data follows the same Weibull slope of 4. Load-displacement curves of three-point bend specimens were normalized into so-called "key curves," and these key curves are compared in Figure 6.1. The key curves from 1T compact specimens tested at –40°C as a part of the HSSI Fourth Irradiation Series are also presented in Figure 6.1. Each type of specimen exhibited different levels of plastic deformation before cleavage. However, all three specimens tend to have the same deformation characteristics, suggesting that stress-strain conditions along the crack plane were comparable for all three-point bend specimens. The key curves of 1T compact specimens also followed the same trend, but cleavage in these considerably larger specimens occurred at significantly lower plastic deformation.

A test matrix is being developed to perform further investigation for evaluating a range of applicability for the current size adjustment model. This matrix includes tests of different specimen geometries with well-characterized materials like HSSI Weld 72W and International Atomic Energy Agency (IAEA) Plate JRQ, as well as some additional tests with HSST Plate 02.

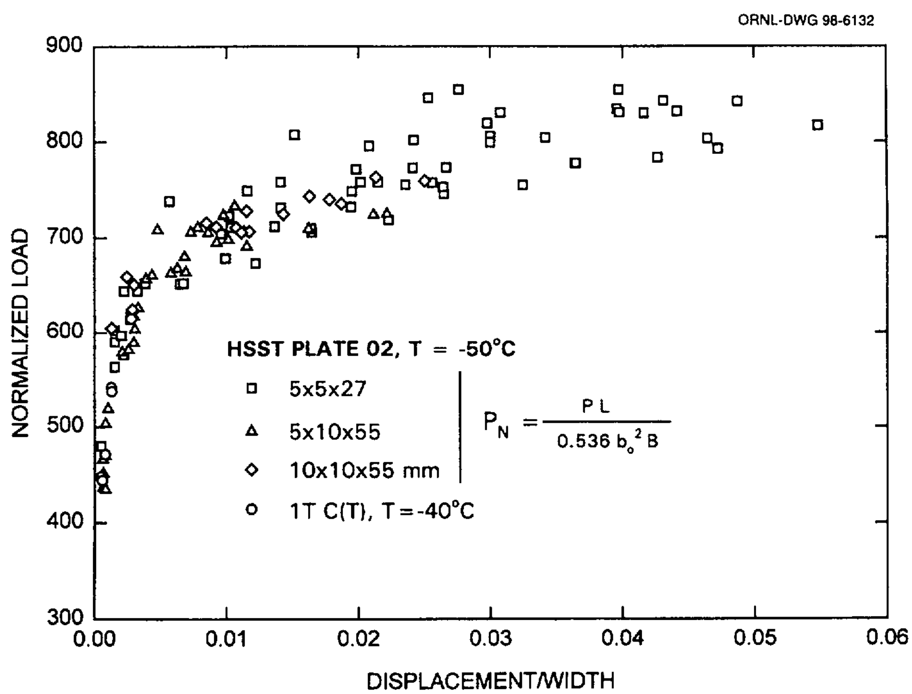

Figure 6.1. Key curves of different specimen types.

Another portion of this subtask deals with investigation of the MC shape stability for highly irradiated materials. A test matrix has been developed to perform irradiation of a statistically meaningful number of 1T compact specimens of a radiation-sensitive material with Charpy shifts up to 200°C. Specimens of smaller sizes are also included into the matrix. Contacts were initiated with investigators at the KORPUS facility in Dimitrovgrad, Russia; the LYRA facility in Petten, The Netherlands; HANORO in South Korea; the NRI facility in Rez, Czech Republic; and the RRC-Kurchatov Institute in Moscow, Russia; regarding the technical aspects, availability, and cost relative to irradiation experiments with 1T compact specimens to validate the shape of the MC for highly embrittled material. The possibility of in-core irradiation at the University of Michigan FNR was also considered. The IAR facility at the FNR was identified as the worst scenario because of the long irradiation time required to reach the desired neutron fluence of 2–5 × 10^{19} n/cm^2 and the resulting delay irradiating other critical materials such as the HAZ specimen, etc. Responses from HANARO in South Korea and the RRC-Kurchatov Institute in Russia indicated that such experiments are not feasible at those sites. On the other hand, the KORPUS, NRI, and LYRA facilities are interested, capable, and available to start such an experiment in the fall of 1998. The FNR in-core irradiation cannot be performed on a large number of 1T compact specimens.

Contacts were also initiated with investigators at AEA-Harwell, United Kingdom (G. Gage); MPA, Stuttgart, Germany (J. Fohl); and IAEA, Vienna, Austria (V. Lyssakov and M. Brumovsky); regarding the availability of radiation-sensitive material in sufficient amounts to perform irradiation experiments with 1T compact specimens to validate the shape of the MC for highly embrittled material. ABB-CE, Chattanooga, Tennessee, was also contacted regarding technical aspects and costs relative to the manufacture of such welds for this experiment. In response to these contacts, investigators from MPA, Stuttgart (J. Fohl), Germany, offered a radiation-sensitive material in sufficient amounts to perform irradiation experiments with 1T compact specimens. This weld, designated KS-01, has been investigated to a limited degree by MPA. From the MPA results, it was determined that this weld exhibits a 41-J shift of about 200°C at 2.13 × 10^{19} n/cm^2, making it very promising for our program. A comparison was also made between the average composition of KS-01 weld metal and that from U.S. RPV materials in the Power-Reactor-Embrittlement Database (data provided by J. A. Wang). The comparison did not reveal any significant deviation of the major chemical elements from our desired ranges.

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

In the previous report,[1] we presented a preliminary analysis of a database of Charpy impact and fracture-toughness curve shifts for RPV materials. Establishing a database with both the reference fracture toughness and Charpy-impact transition temperatures for a wide range of RPV steels in the unirradiated and irradiated conditions makes a logical initial request to establish a correlation between two transition temperatures, T_{100} and T_{41J}. Figure 6.2 illustrates such correlation with all the data available in the current database. The linear regression provides the following fit to the data:

$$T_{100} = T_{41J} - 24°C\ , \qquad \sigma = 20°C\ , \tag{6.1}$$

with a relatively high correlation coefficient, $r^2 = 0.90$. The data have a very wide range of transition temperatures, providing good means for application of this correlation. For example, the reference fracture-toughness temperature varies from about −150 to 150°C. Because there is a large number of RPV steel Charpy T_{41J} data available, Eq. (6.1) could serve as a first approximation for the static fracture-toughness reference temperature, including those irradiated. However, that data scatter is about ±40°C.

Reference

1. M. A. Sokolov, D. E. McCabe, R. K. Nanstad, and D. J. Alexander, "Fracture Toughness Curve Shift Method," pp. 6-1 to 6-12 in *Heavy-Section Steel Irradiation Program Semiannual Progress Report for October 1996 to March 1997*, NUREG/CR-5591 (ORNL/TM-11568), Vol. 8, No. 1, February 1998.

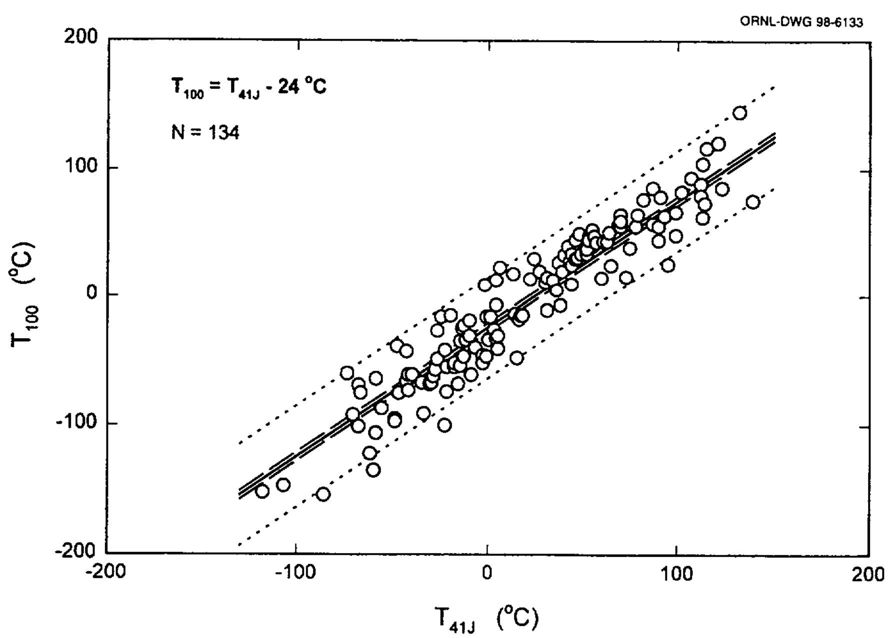

Figure 6.2. Correlation between T_{100} and T_{41J} for both base and weld metals in the unirradiated and irradiated conditions.

7. Special Technical Assistance

T. M. Rosseel

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

The objectives of this subtask are to provide an assessment of new testing methods and specific material properties for irradiated components of nuclear reactors. Specific activities for this subtask are

- completing an interim report on the effects of thermal aging on the impact and tensile testing in structural stainless steel welds that have received 50,000 h of thermal exposure;
- completing the fracture-toughness tests on the remaining CVN-type specimens of the structural stainless steel welds aged for 50,000 h, initiating characterization of the material by metallography and electron microscopy, and then preparing a final report;
- completing thermal aging at 288°C of three-wire, weld-overlay stainless steel cladding to beyond 50,000 h, machining, and testing specimens;
- preparing a report on thermally aged (50,000 h), weld-overlay stainless steel cladding; and
- initiating the testing and evaluation of dynamic PCVN specimens.

A draft NUREG report was completed regarding the effect of long-term thermal aging of stainless steel welds at 343°C. The welds were produced by shielded metal-arc welding with type 308 filler metal with the chemical composition adjusted to obtain different ferrite levels (4, 8, or 12%). Portions of the welds were aged for 3,000; 10,000; 20,000; or 50,000 h at 343°C. All tensile, CVN, and fracture-toughness test results are discussed in the draft NUREG report, including the results of CVN tests of specimens given reversion heat treatments. Figures 7.1 and 7.2 show the effects of aging time on tensile properties and Charpy-impact toughness. The significant decreases in toughness with increased aging time appear to be the results of spinodal decomposition of the ferrite, as well as precipitation of both large and fine G-phase particles. Note that the decrease in toughness increases with aging time for both the 8 and 12% ferrite levels.

7.2 Correlation Monitor Materials (T. M. Rosseel and E. T. Manneschmidt)

This subtask was established to ensure the continued availability of the pedigreed and extremely well-characterized material now required for inclusion in all additional and future surveillance capsules in commercial light-water reactors. Having recognized that the only original materials qualified for use

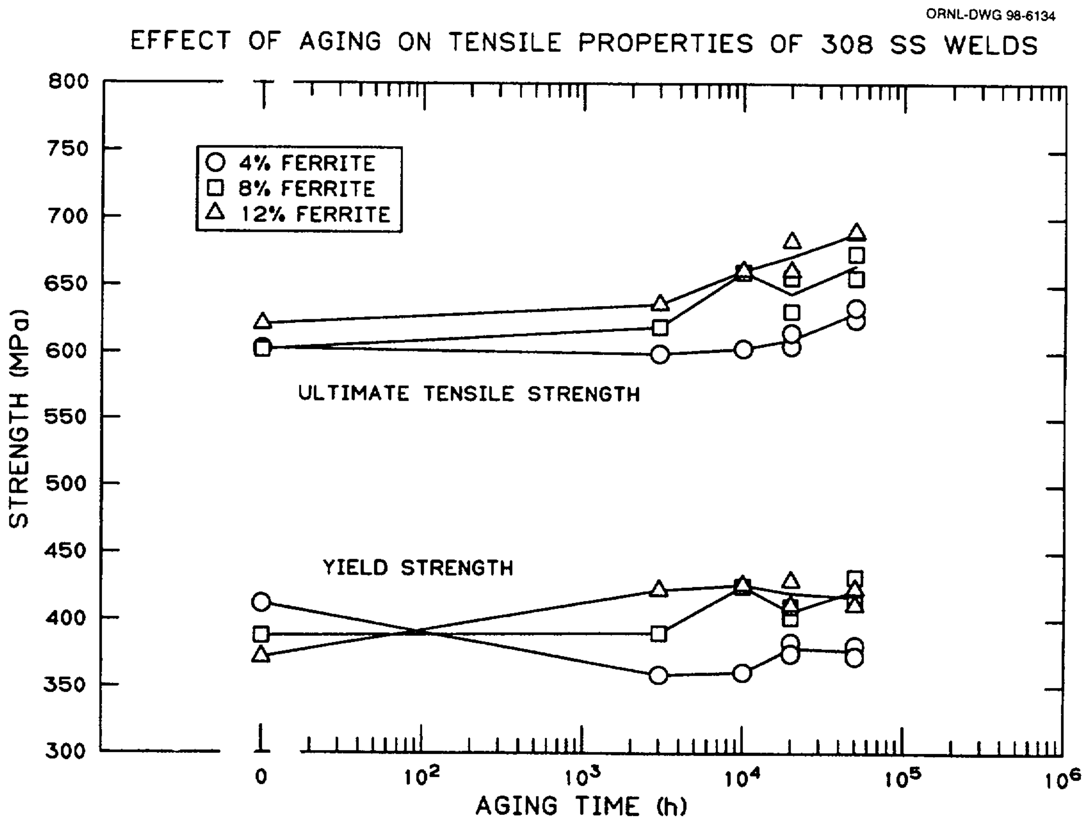

Figure 7.1. **Effect of aging at 343°C on the tensile properties of type 308 stainless steel welds.** There is little change in tensile properties.

as a correlation monitor in reactor surveillance capsules are the pieces remaining from the early HSST plates 01, 02, and 03, this task provides for cataloging, archiving, and distributing the material on behalf of the NRC. During this reporting period, archival storage of the correlation monitor material was maintained.

7.3 Transfer of Government-Furnished Equipment and Materials (T. M. Rosseel)

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

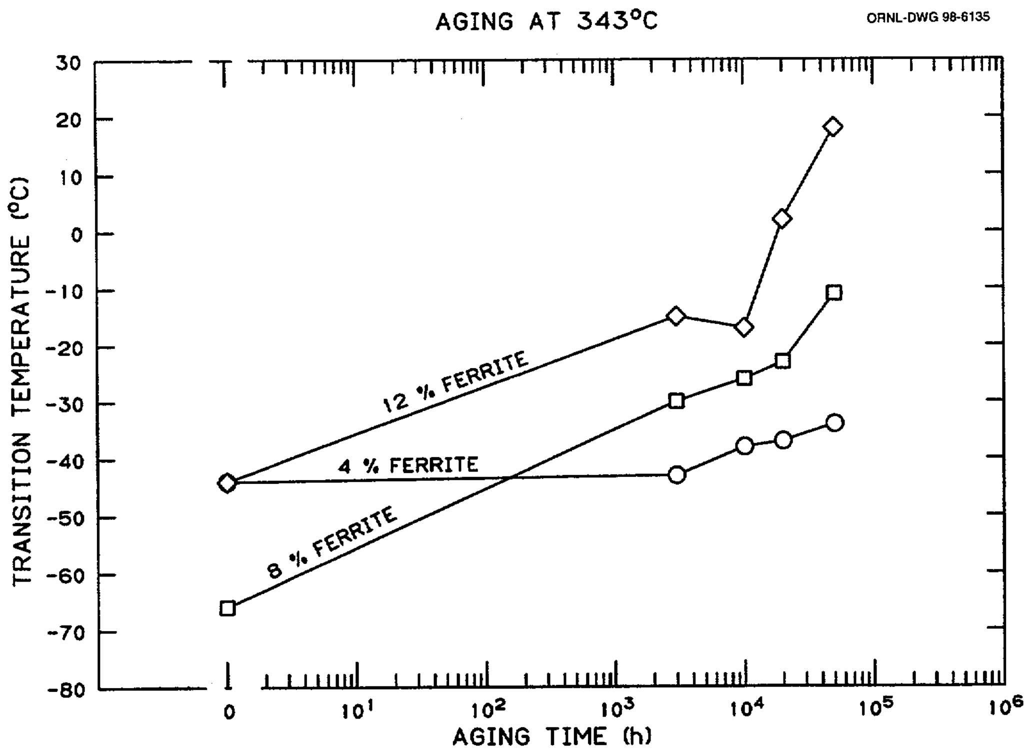

Figure 7.2. Effect of aging at 343°C on the ductile-to-brittle transition temperature for type 308 stainless steel welds. The transition temperature increases with increased aging time, and the effect is greater for welds with higher ferrite contents.

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

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

At the beginning of this reporting period, the UCSB irradiation facility remained shutdown because of an overtemperature event near the end of the last reporting period. Although the overtemperature event was undesirable, all experimenters determined that the test specimens inside the facility suffered no damage. The event was investigated, and the results were reviewed by quality assurance engineers within ORNL and FNR to develop operational guidelines that would ensure that the same or similar incidents would not happen in the future. As a result of the findings, several decisions were made with regard to the operation of the UCSB facility at FNR. A list of action items was assembled after consultation with all parties involved in the operation of the experiment. These actions included (1) preparing a list of operational requirements agreed to by all parties concerned; (2) preparing detailed procedures for startup, operation, shutdown, and specimen replacement; (3) changing the computer program and installing hardware to provide a more fail-safe system to operate the HSSI-UCSB facility; and (4) developing a new and more detailed alarm/response list. Irradiation of the UCSB and ORNL specimens in the UCSB facility resumed on May 22, 1997. Since that date, all of the operating and control mechanisms associated with the facility have performed as expected.

At the beginning of this reporting period, the UCSB facility had been irradiated for a total of 870 effective full-power hours (EFPH) and still contained all of the original specimen packets that were initially installed. The initial loading consisted of 22 specimen packets that contained a total of 3,995 specimens. After 2154 EFPH of irradiation, 2 packets containing 335 specimens were removed and 3 packets containing 876 specimens were inserted. A second specimen changeout was performed after 4157 EFPH; at that time, 2 packets containing 464 specimens were removed. The third and most recent specimen changeout occurred after 4593 EFPH. During that changeout, 2 packets containing 606 specimens were removed and 2 packets containing 864 specimens were inserted. At the end of this reporting period, a total of 5,717 specimens had been inserted into the facility for irradiation and 1405 specimens had reached the desired fluence and were removed for testing. Specimens removed from the facility are allowed to decay for a period of time before shipment to the experimenter. The first shipment of irradiated specimens to UCSB for testing is now under way. All of the preirradiated ORNL Charpy specimens installed initially have obtained approximately 60% of the desired fluence.

Although the computer and controls system for the UCSB facility has performed as expected and has not caused any irradiation time to be lost, approximately 20 days of irradiation time was lost because of a broken trolley system used to crank the facility into and away from the FNR core. The trolley crank mechanism had been used for earlier HSSI irradiations at FNR and had been in service for more than six years. The mechanism was replaced with a spare, and the replacement is expected to last for the duration of the UCSB irradiation campaign. During this 12-month reporting period, the FNR was operated for a total of 242 days and the UCSB irradiation facility was irradiated for 166 of those days. A total of 76 days of irradiation time was lost because of the overtemperature event, the broken trolley crank, and the reactor operating in conditions undesirable for UCSB irradiation. During undesirable reactor conditions, the UCSB facility is simply left shutdown and is cranked away from the FNR core.

8. Foreign Research Interactions

R. K. Nanstad

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

8.1 Japan Power Development Reactor Vessel Steel Examinations
(S. K. Iskander)

During this reporting period, Masahide Suzuki and Kunio Onizawa, of the Japan Atomic Energy Research Institute (JAERI), visited ORNL on February 24–25, 1998. This was the second visit by JAERI representatives, the first being in January 1997. Discussions were held concerning CVN testing on both unirradiated and irradiated welds by ORNL (pending NRC approval) and a new memorandum of understanding to extend the previous cooperative agreement.

The visitors made presentations on the future of nuclear electric power plants. Japan now has 50 operating plants, which generate about 30% of the electric power used in Japan. Two more plants are under construction, and five more are in the planning stages; all are expected to be in operation by 2010. Valuable unpublished JAERI results were given on recently completed fracture-toughness testing of irradiated steels and welds. Permission was given to ORNL to incorporate the data into the ORNL database. The database will be used to develop correlations between the shifts as determined using the MC approach and the CVN 41-J energy level. Other data presented by JAERI for an RPV steel showed that no significant effects of thermal aging at 288 °C could be determined on the CVN transition temperature. ORNL staff members also made several presentations on the MC approach, fracture-toughness determinations using small specimens, and atom-probe studies.

JAERI representatives also discussed unpublished test results on weld material from the Japan Power Development Reactor (JPDR). Although some unexplained variations in weld chemistry through the thickness of one of the remote trepans were observed, distribution of the radiation-sensitive elements was reasonably uniform through the thickness. The shift of 41-J CVN transition temperature of both the corebelt base and weld material is about 20 K when compared with the materials from the remote region. The 20-K shift is about the value expected for the chemistry and fluence exposure of the material. The JAERI representatives mentioned a problem they encountered while machining specimens from material nearest to the inside wall. Because of the J-type geometry of the weld, the width of the weldment is smallest near the inside. This limits positioning of the Charpy specimen to more than 25 mm from the inside vessel surface. ORNL will etch the welds as soon as possible to determine the geometry of the welds in the JPDR trepans at ORNL. Discussions about the proposed scope and schedules for ORNL machining and testing of the JPDR material call for testing only weldments. Estimated times for completion of the first phase (subject to NRC approval and availability of funds) would require a letter of understanding or some other means to extend the present agreement (which expires in March 1999).

JAERI has essentially completed testing of the base metal. It is estimated that eight Charpy or four 0.5T C(T) weld material specimens can be machined from each trepan at each "depth" in the wall. One possibility is to machine sixteen Charpy and eight 0.5T C(T) specimens from each of the two depths to be investigated. A major decision is which specimen orientation to test. The T-L is the orientation preferred by ORNL, although JAERI prefers the T-S orientation. Note that 0.5T C(T) specimens in the T-S orientation would sample slightly more material that is subjected to fluence gradient than the T-L orientation. A second possibility would be to machine 32 Charpy specimens from each depth, use approximately one-half for Charpy tests, and precrack the remainder. The precracked specimens would be tested in slow bend to obtain MC fracture-toughness values [similar to the 0.5T C(T) specimens]. If Charpy-sized specimens are used for the whole investigation, there would be a small number of specimens available for testing in the orientation different from that used for the predominant number of specimens. This could confirm the expectation that, for welds, Charpy-impact test results in either T-L or T-S orientations are approximately the same.

Other possible tasks that can be performed by ORNL include a limited number of tests on SS-3 type tensile specimens that will be used in conjunction with automated ball indentation (ABI) testing to verify the correlation between tensile properties deduced from ABI and those from conventional testing. These tests would facilitate the performance of annealing studies because they could then be performed on already available (either tested or untested) Charpy specimens. The completion date for all such tests is estimated to be about the end of 2001. Other studies contemplated beyond the year 2001 could include atom-probe studies of the distribution of copper in the welds, before and after annealing, and J-R tests on the three-pass, 12-mm-thick irradiated and unirradiated cladding.

8.2 Technical Assistance for JCCCNRS Working Groups 3 and 12
(R. K. Nanstad and M. A. Sokolov)

The purpose of this subtask is to provide technical support for the efforts of the U.S.-Russian Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Group 3 on radiation embrittlement and Working Group 12 on aging. Specific activities under this subtask are (1) supply of materials and preparation of test specimens for collaborative irradiation, annealing, and reirradiation studies to be conducted in Russia; (2) irradiation, annealing, and testing of Russian specimens within the United States; and (3) preparation for and participation in meetings of Workings Groups 3 and 12.

Charpy-impact and tensile testing of VVER-440 and VVER-1000 welds in the unirradiated, irradiated, and thermally annealed conditions were completed previously. Following completion of data analysis, a meeting was held in Hannover, Germany, with two researchers from the Russian Research Center-Kurchatov Institute (RRC-KI), to discuss analyses of the ORNL results with Russian specimens and the

RRC-KI results with U.S. materials. As a result of that meeting, a paper, "Exploratory Study of Irradiation, Annealing, and Reirradiation Effects on American and Russian Reactor Pressure Vessel Steels," by A. A. Chernobaeva, M. A. Sokolov, R. K. Nanstad, A. M. Kryukov, Y. A. Nikolaev, and Yu. N. Korolev, was completed and published in *Proceedings of the Eighth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors.* The paper was presented in August 1997 by A. A. Chernobaeva at the Symposium in Amelia Island, Florida. Discussions have continued with the RRC-KI staff regarding preparation of a NUREG report on the ORNL/RRC-KI irradiation project.

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

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

During this reporting period, a concrete- and lead-lined container was designed by ORNL and two such containers were fabricated for shipment of eight irradiated 4T compact specimens of HSSI Welds 72W and 73W. The containers were each placed in a steel box for shipment to SCK-CEN in Mol, Belgium. The design, fabrication, and shipping costs for this activity were funded by SCK-CEN. The irradiated specimens were shipped in September and arrived in SCK-CEN about October 1, 1997.

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

This subtask involves participation of ORNL as the official U.S. member of the IAEA New Coordinated Research Program (CRP) on behalf of the NRC. The focus of the New CRP is to examine the current ability to assess the toughness of RPV materials in the unirradiated and irradiated conditions using surveillance-size specimens.

The subject task group meets annually; the most recent meeting was held October 8–10, 1997, in Vienna. About 25 persons attended, representing 17 countries. Participants presented status reports on their respective program commitments. In particular, the focus was on the round-robin K_{Jc} data and T_o temperatures obtained in static three-point bend testing of precracked Charpy specimens of JRQ pressure vessel steel. Nine participants had completed this work. The grand average T_o value was −69°C, and all of the reporting laboratories were within 10°C of this value. A third annual meeting is scheduled for November 18–20, 1998, in Vienna.

8.5. Korean Interactions (R. K. Nanstad and T. M. Rosseel)

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

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

CONVERSION FACTORS[a]

SI unit	English unit	Factor
mm	in.	0.0393701
cm	in.	0.393701
m	ft	3.28084
m/s	ft/s	3.28084
kN	lb_f	224.809
kPa	psi	0.145038
MPa	ksi	0.145038
MPa•√m	ksi•√in.	0.910048
J	ft•lb	0.737562
K	°F or °R	1.8
kJ/m^2	$in.\text{-}lb/in.^2$	5.71015
$W{\bullet}m^{-3}{\bullet}K^{-1}$	$Btu/h{\bullet}ft^2{\bullet}°F$	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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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

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

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
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