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REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

Analysis of Capsule W-104
Entergy Operations, Inc.
Arkansas Nuclear One Unit 2 Power Plant

FRAMATOME ANP

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-- Reactor Vessel Material Surveillance Program --

by

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Executive Summary

This report describes the results of the examination of the second capsule (Capsule W-104) of the Entergy Operations, Inc. Arkansas Nuclear One Unit 2 Power Plant as part of their reactor vessel surveillance program (RVSP). The objective of the program is to monitor the effects of neutron irradiation on the mechanical properties of the reactor vessel materials by testing and evaluation of tensile and Charpy V-notch impact specimens. The Arkansas Nuclear One Unit 2 RVSP was designed and furnished by Combustion Engineering, Inc. and was based on ASTM Standard E 185-73.

Capsule W-104 was removed from the Arkansas Nuclear One Unit 2 reactor vessel at the end-of-cycle 14 (EOC-14) for testing and evaluation. The capsule received an average fast fluence of 2.937×10^{19} n/cm^2 (E > 1.0 MeV). Based on an extrapolated flux incorporating power uprated conditions, the projected end-of-life (32 EFPY) peak fast fluence of the Arkansas Nuclear One Unit 2 reactor vessel beltline region clad surface is 3.791×10^{19} n/cm^2 (E > 1.0 MeV).

The results of the tension tests indicated that the Arkansas Nuclear One Unit 2 surveillance materials exhibited normal behavior relative to the neutron fluence exposure. The Charpy impact data for the Arkansas Nuclear One Unit 2 surveillance materials exhibited the characteristic behavior of an increase in transition temperature and a decrease in upper-shelf energy as a result of neutron irradiation.

Table of Contents

		Page
1.	Introduction	1-1
2.	Background	2-1
3.	Surveillance Program Description	3-1
4.	Tests of Unirradiated Material	4-1
5.	Post-Irradiation Testing	5-1
	5.1. Capsule Disassembly and Inventory	5-1
	5.2. Thermal Monitors	5-1
	5.3. Tension Test Results	5-1
	5.4. Charpy V-Notch Impact Test Results	5-2
6.	Neutron Fluence	6-1
	6.1. Introduction	6-1
	6.2. Fluence Results	6-3
	6.3. Dosimetry Activity	6-10
7.	Discussion of Capsule Results	7-1
	7.1. Copper and Nickel Chemical Composition Data	7-1
	7.2. Unirradiated Material Property Data	7-1
	7.3. Irradiated Property Data	7-1
	7.3.1. Tensile Properties	7-2
	7.3.2. Impact Properties	7-2
8.	Summary of Results	8-1
9.	Certification	9-1
10.	References	10-1

Table of Contents (cont.)

Appendices		Page
A.	Reactor Vessel Surveillance Program Background Data and Information	A-1
B.	Instrumented Charpy V-Notch Specimen Test Results Load-Time Traces	B-1
C.	Unirradiated and Irradiated Tensile Data for the Arkansas Nuclear One Unit 2 RVSP Materials	C-1
D.	Unirradiated and Irradiated Charpy V-Notch Impact Surveillance Data for the Arkansas Nuclear One Unit 2 RVSP Materials Using Hyperbolic Tangent Curve-Fitting Method	D-1
E.	Charpy V-Notch Shift Comparison: Hand-Drawn Curve Fitting vs. Hyperbolic Tangent Curve Fitting	E-1
F.	Fluence Analysis Methodology	F-1

List of Tables

Table		Page
3-1.	Test Specimens Contained in Arkansas Nuclear One Unit 2 Capsule W-104	3-3
3-2.	Location of Test Specimens Contained in Arkansas Nuclear One Unit 2 Capsule W-104	3-3
3-3.	Chemical Composition of Arkansas Nuclear One Unit 2 Capsule W-104 Surveillance Materials	3-4
5-1.	Tensile Properties of Arkansas Nuclear One Unit 2 Capsule W-104 Reactor Vessel Surveillance Materials, Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)	5-4
5-2.	Charpy V-Notch Impact Results for Arkansas Nuclear One Unit 2 Capsule W-104 Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV) Transverse Orientation	5-5
5-3.	Charpy V-Notch Impact Results for Arkansas Nuclear One Unit 2 Capsule W-104 Weld Metal, Heat 10120, Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)	5-6
5-4.	Charpy V-Notch Impact Results for Arkansas Nuclear One Unit 2 Capsule W-104 Heat-Affected-Zone Material, Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)	5-7
5-5.	Charpy V-Notch Impact Results for Arkansas Nuclear One Unit 2 Capsule W-104 Standard Reference Material, HSST Plate 01 Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)	5-8

List of Tables (cont.)

Table		Page
5-6.	Hyperbolic Tangent Curve Fit Coefficients for the Arkansas Nuclear One Unit 2 Capsule W-104 Surveillance Materials	5-9
5-7.	Instrumented Charpy V-Notch Properties of Arkansas Nuclear One Unit 2 Capsule W-104, Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 2.937×10^{19} n/cm^2 (E>1.0 MeV) Transverse Orientation	5-10
5-8	Instrumented Charpy V-Notch Properties of Arkansas Nuclear One Unit 2 Capsule W-104, Weld Metal, Heat 10120, Irradiated to 2.937×10^{19} n/cm^2 (E > 1.0 MeV).	5-11
5-9	Instrumented Charpy V-Notch Properties of Arkansas Nuclear One Unit 2 Capsule W-104, Heat-Affected-Zone Material, Irradiated to 2.937×10^{19} n/cm^2 (E > 1.0 MeV)	5-12
5-10	Instrumented Charpy V-Notch Properties of Arkansas Nuclear One Unit 2 Capsule W-104, Standard Reference Material, HSST Plate 01, Irradiated to 2.937×10^{19} n/cm^2 (E > 1.0 MeV)	5-13
6-1.	Arkansas Nuclear One Unit 2 Fast Fluence Dosimetry	6-4
6-2.	3D Coordinates for Arkansas Nuclear One Unit 2 Points of Interest	6-5
6-3.	3D Synthesized Fluxes	6-6
6-4.	Cumulative Fluence Projections	6-9
6-5.	C/M Ratios	6-10
7-1.	Copper and Nickel Mean Chemical Composition Data for Arkansas Nuclear One Unit 2 Reactor Vessel Surveillance Materials	7-4
7-2.	Summary of Arkansas Nuclear One Unit 2 Reactor Vessel Surveillance Capsules Tensile Test Results	7-5
7-3.	Measured vs. Predicted 30 ft-lb Transition Temperature Changes for Arkansas Nuclear One Unit 2 W-104 Capsule Surveillance Materials – 2.937×10^{19} n/cm^2	7-6
7-4	Measured vs. Predicted Upper-Shelf Energy Decreases for Arkansas Nuclear One Unit 2 W-104 Capsule Surveillance Materials – 2.937×10^{19} n/cm^2	7-7
7-5	Summary of Arkansas Nuclear One Unit 2 Reactor Vessel Surveillance Capsules Charpy Impact Test Results	7-8
A-1.	Arkansas Nuclear One Unit 2 Surveillance Capsule Identifications and Original Locations	A-3

List of Tables (cont.)

Table		Page
A-2.	Description of the Arkansas Nuclear One Unit 2 Reactor Vessel Beltline Region Materials	A-4
C-1.	Tensile Properties of Unirradiated Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation	C-2
C-2.	Tensile Properties of Unirradiated Weld Metal, Heat 10120	C-2
C-3.	Tensile Properties of Unirradiated Heat Affected Zone Metal	C-3
C-4.	Tensile Properties of Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.33 x 10^{18} n/cm^2 (E>1.0 MeV) Transverse Orientation	C-3
C-5.	Tensile Properties of Weld Metal, Heat 10120 Irradiated to 3.35 x 10^{18} n/cm^2 (E>1.0 MeV)	C-4
C-6.	Tensile Properties of Heat Affected Zone Metal Irradiated to 3.56 x 10^{18} n/cm^2 (E>1.0 MeV)	C-4
D-1.	Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation	D-2
D-2.	Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.33 x 10^{18} n/cm^2 (E>1.0 MeV) Transverse Orientation	D-3
D-3.	Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation	D-4
D-4.	Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Longitudinal Orientation	D-5
D-5.	Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.41 x 10^{18} n/cm^2 (E>1.0 MeV) Longitudinal Orientation	D-6
D-6.	Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Longitudinal Orientation	D-7
D-7.	Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Weld Metal, Heat 10120	D-8
D-8.	Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Weld Metal, Heat 10120 Irradiated to 3.34 x 10^{18} n/cm^2 (E>1.0 MeV)	D-9

List of Tables (cont.)

Table		Page
D-9.	Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2 Weld Metal, Heat 10120	D-10
D-10.	Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Heat-Affected-Zone Material	D-11
D-11.	Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Heat-Affected-Zone Material, Irradiated to 3.49 x 10^{18} n/cm^2 (E>1.0 MeV)	D-12
D-12.	Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2 Heat-Affected-Zone Material	D-13
D-13.	Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Standard Reference Material	D-14
D-14.	Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2 Standard Reference Material	D-15
E-1.	Comparison of Curve Fit Transition Temperature Shifts for Arkansas Nuclear One Unit 2 Surveillance Material, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation	E-2
E-2.	Comparison of Curve Fit Transition Temperature Shifts for Arkansas Nuclear One Unit 2 Surveillance Material, Base Metal Plate C-8009-3, Heat C8182-2, Longitudinal Orientation	E-3
E-3.	Comparison of Curve Fit Transition Temperature Shifts for Arkansas Nuclear One Unit 2 Surveillance Material, Weld Metal, Heat 10120	E-4
E-4.	Comparison of Curve Fit Transition Temperature Shifts for Arkansas Nuclear One Unit 2 Surveillance Material, Heat-Affected-Zone Material	E-5
F-1.	Bias Correction Factors	F-7
F-2.	Calculational Fluence Uncertainties	F-10

List of Figures

Figure		Page
3-1.	Reactor Vessel Cross Section and Elevation Views Showing the Locations of RVSP Capsules in Arkansas Nuclear One Unit 2 Reactor Vessel	3-5
3-2.	Surveillance Capsule Assembly Showing Locations of Specimens and Monitors for Arkansas Nuclear One Unit 2 Capsule W-104	3-6
5-1.	Photographs of Thermal Monitors Removed from the Arkansas Nuclear One Unit 2 RVSP Capsule W-104 Top Compartment	5-14
5-2.	Photographs of Thermal Monitors Removed from the Arkansas Nuclear One Unit 2 RVSP Capsule W-104 Middle Compartment	5-15
5-3.	Photographs of Thermal Monitors Removed from the Arkansas Nuclear One Unit 2 RVSP Capsule W-104 Bottom Compartment	5-16
5-4.	Tension Test Stress-Strain Curve for Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation, Specimen No. 2JE, Tested at 70°F	5-17
5-5.	Tension Test Stress-Strain Curve for Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation, Specimen No. 2L3, Tested at 250°F	5-17
5-6.	Tension Test Stress-Strain Curve for Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation, Specimen No. 2JT, Tested at 550°F	5-18
5-7.	Tension Test Stress-Strain Curve for Weld Metal, Heat 10120, Specimen No. 3J7, Tested at 70°F	5-18
5-8.	Tension Test Stress-Strain Curve for Weld Metal, Heat 10120, Specimen No. 3KP, Tested at 250°F	5-19
5-9.	Tension Test Stress-Strain Curve for Weld Metal, Heat 10120, Specimen No. 3K5, Tested at 550°F	5-19
5-10.	Tension Test Stress-Strain Curve for Heat-Affected-Zone Material, Specimen No. 4K4, Tested at 70°F	5-20
5-11.	Tension Test Stress-Strain Curve Heat-Affected-Zone Material, Specimen No. 4JM, Tested at 250°F	5-20
5-12.	Tension Test Stress-Strain Curve for Heat-Affected-Zone Material, Specimen No. 4KU, Tested at 550°F	5-21
5-13.	Photographs of Tested Tension Test Specimens and Corresponding Fracture Surfaces – Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation	5-22
5-14.	Photographs of Tested Tension Test Specimens and Corresponding Fracture Surfaces – Weld Metal, Heat 10120	5-23

List of Figures (cont.)

Figure		Page
5-15.	Photographs of Tested Tension Test Specimens and Corresponding Fracture Surfaces – Heat-Affected-Zone Material	5-24
5-16.	Charpy Impact Data for Irradiated Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation	5-25
5-17.	Charpy Impact Data for Irradiated Weld Metal, Heat 10120	5-26
5-18.	Charpy Impact Data for Irradiated Heat-Affected-Zone Material	5-27
5-19.	Charpy Impact Data Irradiated Standard Reference Material, HSST Plate 01	5-28
5-20.	Photographs of Charpy Impact Specimen Fracture Surfaces, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation	5-29
5-21.	Photographs of Charpy Impact Specimen Fracture Surfaces, Weld Metal, Heat 10120	5-30
5-22.	Photographs of Charpy Impact Specimen Fracture Surfaces, Heat-Affected-Zone Material	5-31
5-23	Photographs of Charpy Impact Specimen Fracture Surfaces, Standard Reference Material, HSST Plate 01	5-32
6-1.	Arkansas Nuclear One Unit 2 Core Overview	6-3
6-2.	Arkansas Nuclear One Unit 2 104 Dosimetry Locations	6-4
B-1.	Load-Time Trace for Charpy V-Notch Impact Specimen 25L	B-2
B-2.	Load-Time Trace for Charpy V-Notch Impact Specimen 227	B-2
B-3.	Load-Time Trace for Charpy V-Notch Impact Specimen 246	B-3
B-4	Load-Time Trace for Charpy V-Notch Impact Specimen 24L	B-3
B-5	Load-Time Trace for Charpy V-Notch Impact Specimen 23C	B-4
B-6	Load-Time Trace for Charpy V-Notch Impact Specimen 251	B-4
B-7	Load-Time Trace for Charpy V-Notch Impact Specimen 22A	B-5
B-8.	Load-Time Trace for Charpy V-Notch Impact Specimen 24K	B-5
B-9.	Load-Time Trace for Charpy V-Notch Impact Specimen 236	B-6
B-10.	Load-Time Trace for Charpy V-Notch Impact Specimen 23B	B-6
B-11.	Load-Time Trace for Charpy V-Notch Impact Specimen 252	B-7
B-12.	Load-Time Trace for Charpy V-Notch Impact Specimen 24Y	B-7

List of Figures (cont.)

Figure — Page

B-13. Load-Time Trace for Charpy V-Notch Impact Specimen 342 ..... B-8
B-14. Load-Time Trace for Charpy V-Notch Impact Specimen 31B B-8
B-15. Load-Time Trace for Charpy V-Notch Impact Specimen 36M B-9
B-16. Load-Time Trace for Charpy V-Notch Impact Specimen 357 ..... B-9
B-17. Load-Time Trace for Charpy V-Notch Impact Specimen 33P B-10
B-18. Load-Time Trace for Charpy V-Notch Impact Specimen 37J B-10
B-19. Load-Time Trace for Charpy V-Notch Impact Specimen 32C B-11
B-20. Load-Time Trace for Charpy V-Notch Impact Specimen 32A B-11
B-21. Load-Time Trace for Charpy V-Notch Impact Specimen 35B B-12
B-22. Load-Time Trace for Charpy V-Notch Impact Specimen 36L B-12
B-23. Load-Time Trace for Charpy V-Notch Impact Specimen 36B B-13
B-24. Load-Time Trace for Charpy V-Notch Impact Specimen 43L B-13
B-25. Load-Time Trace for Charpy V-Notch Impact Specimen 47P B-14
B-26. Load-Time Trace for Charpy V-Notch Impact Specimen 421 ..... B-14
B-27. Load-Time Trace for Charpy V-Notch Impact Specimen 475 ..... B-15
B-28. Load-Time Trace for Charpy V-Notch Impact Specimen 44M B-15
B-29. Load-Time Trace for Charpy V-Notch Impact Specimen 41L B-16
B-30. Load-Time Trace for Charpy V-Notch Impact Specimen 467 ..... B-16
B-31. Load-Time Trace for Charpy V-Notch Impact Specimen 462 ..... B-17
B-32. Load-Time Trace for Charpy V-Notch Impact Specimen 45C B-17
B-33. Load-Time Trace for Charpy V-Notch Impact Specimen 456 ..... B-18
B-34. Load-Time Trace for Charpy V-Notch Impact Specimen 424 ..... B-18
B-35. Load-Time Trace for Charpy V-Notch Impact Specimen 413 ..... B-19
B-36. Load-Time Trace for Charpy V-Notch Impact Specimen 77D B-19
B-37. Load-Time Trace for Charpy V-Notch Impact Specimen 777 ..... B-20
B-38. Load-Time Trace for Charpy V-Notch Impact Specimen 77E B-20
B-39. Load-Time Trace for Charpy V-Notch Impact Specimen 76Y B-21
B-40. Load-Time Trace for Charpy V-Notch Impact Specimen 77U B-21
B-41. Load-Time Trace for Charpy V-Notch Impact Specimen 776 ..... B-22

List of Figures (cont.)

Figure — Page

B-42. Load-Time Trace for Charpy V-Notch Impact Specimen 773 ..... B-22

B-43. Load-Time Trace for Charpy V-Notch Impact Specimen 77P B-23

B-44. Load-Time Trace for Charpy V-Notch Impact Specimen 77M B-23

B-45. Load-Time Trace for Charpy V-Notch Impact Specimen 772 ..... B-24

B-46. Load-Time Trace for Charpy V-Notch Impact Specimen 77Y B-24

B-47. Load-Time Trace for Charpy V-Notch Impact Specimen 77A B-25

D-1. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation Refitted Using Hyperbolic Tangent Curve-Fitting Method D-16

D-2. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.33 x 10^{18} n/cm^2 (E>1.0 MeV) Transverse Orientation - Refitted Using Hyperbolic Tangent Curve-Fitting Method D-17

D-3. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Longitudinal Orientation Refitted Using Hyperbolic Tangent Curve-Fitting Method D-18

D-4. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.33 x 10^{18} n/cm^2 (E>1.0 MeV) Longitudinal Orientation - Refitted Using Hyperbolic Tangent Curve-Fitting Method D-19

D-5. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Weld Metal, Heat 10120 - Refitted Using Hyperbolic Tangent Curve-Fitting Method D-20

D-6. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Weld Metal, Heat 10120 Irradiated to 3.34 x 10^{18} n/cm^2 (E>1.0 MeV) - Refitted Using Hyperbolic Tangent Curve-Fitting Method D-21

D-7. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Heat-Affected-Zone Material - Refitted Using Hyperbolic Tangent Curve-Fitting Method D-22

D-8. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Heat-Affected-Zone Material, Irradiated to 3.49 x 10^{18} n/cm^2 (E>1.0 MeV) - Refitted Using Hyperbolic Tangent Curve-Fitting Method D-23

List of Figures (cont.)

Figure — Page

D-9. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Standard Reference Material Refitted Using Hyperbolic Tangent Curve-Fitting Method D-24

F-1. Fluence Analysis Methodology for Arkansas Nuclear One Unit 2 Surveillance Capsule F-4

1. Introduction

This report presents the test and evaluation results of the second reactor vessel surveillance capsule (Capsule W-104) removed from the Arkansas Nuclear One Unit 2 reactor vessel. The capsule was removed and the contents evaluated after being irradiated in the Arkansas Nuclear One Unit 2 reactor as part of the reactor vessel surveillance program (RVSP) as documented in CEN-15(A)[1]. This report describes the testing and the post-irradiation results obtained from Arkansas Nuclear One Unit 2 Capsule W-104 after receiving an average fluence of 2.937×10^{19} n/cm^2 (E>1.0 MeV). This data are compared to the previous Arkansas Nuclear One Unit 2 RVSP results from Capsule W-97[2]. This report meets the reporting requirements of 10CFR50, Appendix H.

The objective of the program is to monitor the effects of neutron irradiation on the mechanical properties of reactor vessel materials under actual plant operating conditions. The program was planned to monitor the effects of neutron irradiation on the reactor vessel materials for the 40-year design life of the reactor pressure vessel. The Arkansas Nuclear One Unit 2 RVSP was designed and furnished by Combustion Engineering, Inc., based on American Society for Testing and Materials (ASTM) Standard E 185-73.[3]

2. Background

The ability of the reactor vessel to resist fracture is a primary factor in ensuring the safety of the primary coolant system in light water reactors. The reactor vessel beltline region is the most critical region of the vessel since it is exposed to the highest level of neutron-irradiation. The general effects of fast neutron-irradiation on the mechanical properties of low-alloy ferritic steels used in the fabrication of reactor vessels are well characterized and documented. The low-alloy ferritic steels used in the beltline region of reactor vessels exhibit an increase in ultimate and yield strength properties with a corresponding decrease in ductility after irradiation. The most significant mechanical property change in reactor vessel steels is the increase in the ductile-to-brittle transition temperature accompanied by a reduction in Charpy upper-shelf energy (C_vUSE).

Code of Federal Regulation, Title 10, Part 50, (10 CFR 50) Appendix G, *"Fracture Toughness Requirements,"* [4] specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of commercial light water reactors and provides specific guidelines for determining the pressure-temperature limitations for operation of the RCPB. The fracture toughness and operational requirements are specified to provide adequate safety margins during normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The requirements of 10 CFR 50, Appendix G, became effective on August 16, 1973. These requirements are applicable to all boiling and pressurized light water nuclear power reactors, including those under construction or in operation on the effective date.

10 CFR 50, Appendix H, *"Reactor Vessel Material Surveillance Program Requirements,"* [5] defines the material surveillance program required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water reactors resulting from exposure to neutron-irradiation and the thermal environment. Fracture toughness test data are obtained from material specimens contained in capsules that are periodically withdrawn from the reactor vessel. These data permit determination of the conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life.

A method for guarding against non-ductile fracture in reactor vessels is described in Appendix G to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, *"Nuclear Power Plant Components"* [6] and Section XI, *"Rules for Inservice*

Inspection"[7]. This method uses fracture mechanics concepts and the reference nil-ductility temperature, RT_{NDT}, which is defined as the greater of the drop weight nil-ductility transition temperature (in accordance with ASTM E 208-81[8]) or the temperature that is 60°F below that at which the material exhibits 50 ft-lbs and 35 mils lateral expansion. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve), which appears in Appendix G of ASME B&PV Code Section III and Section XI. The K_{IR} curve is a lower bound of dynamic and crack arrest fracture toughness data obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for the material as a function of temperature. The operating limits can then be determined using these allowable stress intensity factors.

The RT_{NDT} and, in turn, the operating limits (pressure/temperature limits) of a nuclear power plant, are adjusted over the life of the plant to account for the effects of irradiation on the fracture toughness of the reactor vessel materials. Irradiation embrittlement and the resultant changes in mechanical properties of a given pressure vessel steel are monitored by each plant's surveillance program. The surveillance capsules, as part of the surveillance program, contain prepared specimens of the reactor vessel materials, which are irradiated in the reactor vessel. A surveillance capsule is periodically removed from the operating nuclear reactor and the specimens are tested to determine the changes in mechanical properties. The increase in the Charpy V-notch 30 ft-lb temperature is added to the original RT_{NDT} to adjust it for irradiation embrittlement. The adjusted RT_{NDT} is used to index the material to the K_{IR} curve which, in turn, is used to set operating limits for the nuclear power plant. These new limits take into account the effects of irradiation on the reactor vessel materials.

10 CFR 50, Appendix G, also requires a minimum initial C_vUSE of 75 ft-lbs for all beltline region materials unless it is demonstrated that lower values of upper-shelf fracture energy will provide an adequate margin of safety against fracture equivalent to those required by ASME Section XI, Appendix G. No action is required for a material that does not meet the initial 75 ft-lbs requirement provided that the irradiation embrittlement does not cause the C_vUSE to drop below 50 ft-lbs. The regulations specify that if the C_vUSE drops below 50 ft-lbs, it must be demonstrated, in a manner approved by the Office of Nuclear Reactor Regulation, that the lower values will provide adequate margins of safety.

3. Surveillance Program Description

The reactor vessel surveillance program for Arkansas Nuclear One Unit 2 includes six capsules designed to monitor the effects of neutron and thermal environment on the materials of the reactor pressure vessel core region. The capsules, which were inserted into the reactor vessel before initial plant startup, were positioned inside the reactor vessel between the core support barrel and the vessel wall at the locations shown in Figure 3-1. CEN-15(A)[1] includes a full description of the capsule locations and design. Capsule W-104 (capsule location code = 3) was irradiated in the 104° position in the reactor vessel from the start of cycle 1 to the end of cycle 14.

Capsule W-104 was removed during the 2R14-refueling shutdown of Arkansas Nuclear One Unit 2. The capsule contained Charpy V-notch (CVN) impact test specimens fabricated from one heat of base metal plate (SA-533, Grade B, Class 1), heat-affected-zone (HAZ) material, a weld metal representative of the Arkansas Nuclear One Unit 2 reactor vessel beltline region intermediate and lower shell longitudinal welds, and a Standard Reference Material (SRM). The SRM is a standard heat of SA-533, Grade B, Class 1 made available by the USAEC sponsored Heavy Section Steel Technology (HSST) program. The tensile test specimens were also fabricated from the same base metal plate, HAZ, and weld metal. The number of specimens of each material contained in Capsule W-104 is described in Table 3-1, and the location of the individual specimens within the capsule is shown in Table 3-2 and Figure 3-2.[1] The chemical compositions of the surveillance materials within Capsule W-104, obtained from the original surveillance program report,[1] are described in Table 3-3.

All base metal CVN and tensile specimens were machined from the ¼-thickness (¼T) location of the plate material. The base metal specimens were oriented such that the longitudinal axis of each specimen was perpendicular to the principal working direction of the plate (transverse orientation). The HAZ and weld metal specimens were oriented such that the longitudinal axis of each specimen was perpendicular to the weld seam. The CVN HAZ and weld metal specimens had the notch oriented parallel to the weld seam.

Capsule W-104 contained dosimeter wires of copper, iron, nickel, titanium, sulfur, aluminum-0.17 weight percent cobalt (cadmium-shielded and unshielded), and uranium-238 (^{238}U). The location of these dosimeters within Capsule W-104 is shown in Figure 3-2.

Thermal monitors fabricated from four low-melting alloys were included in the capsule. The thermal monitors were sealed in quartz tubes and inserted in spacers located in Figure 3-2. The eutectic alloys and their melting points are listed below:[1]

80% Ag, 20% Sn	Melting Point 536°F
5.0% Ag, 5.0% Sn, 90.0% Pb	Melting Point 558°F
2.5% Ag, 97.5% Pb	Melting Point 580°F
1.75% Ag, 0.75% Sn, 97.5% Pb	Melting Point 590°F

Table 3-1. Test Specimens Contained in Arkansas Nuclear One Unit 2 Capsule W-104

Material Description	Number of Test Specimens	
	Tensile	Charpy V-notch
Base Metal Plate C-8009-3 Heat C8182-2: Transverse	3	12
HAZ Metal	3	12
Weld Metal Heat 10120 C-8009-1/C-8009-2	3	12
SRM*	--	12
Total	9	48

* Standard Reference Material: SA-533, Grade B, Class 1, HSST Plate 01

Table 3-2. Location of Test Specimens Contained in Arkansas Nuclear One Unit 2 Capsule W-104

Compartment Position (from top)	Material	Specimen Type	Compartment ID
1	HAZ	Tensile	D314
2	HAZ	CVN	D324
3	SRM	CVN	D337
4	BM-T	Tensile	D342
5	BM-T	CVN	D352
6	Weld	CVN	D363
7	Weld	Tensile	D373

BM-T = Base Metal – Transverse
CVN = Charpy V-notch
HAZ = Heat Affected Zone
SRM = Standard Reference Material

Table 3-3. Chemical Composition of Arkansas Nuclear One Unit 2 Capsule W-104 Surveillance Materials

Element	Chemical Composition, wt%	
	Base Metal Plate C-8009-3 Heat C8182-2	Weld Metal Heat 10120
Si	0.21	0.14
S	0.011	0.009
P	0.009	0.004
Mn	1.40	1.33
C	0.22	0.13
Cr	0.15	0.02
Ni	0.60	0.08
Mo	0.63	0.62
V	0.006	0.006
Cb	< 0.01	< 0.01
B	< 0.001	< 0.001
Co	0.012	0.003
Cu	0.08	0.04
Al	0.034	0.001
W	<0.01	<0.01
Ti	<0.01	<0.01
As	0.010	0.006
Sn	0.005	0.001
Zr	0.001	0.001
N_2	0.008	0.005

Figure 3-1. Reactor Vessel Cross Section and Elevation Views Showing the Locations of RVSP Capsules in Arkansas Nuclear One Unit 2 Reactor Vessel

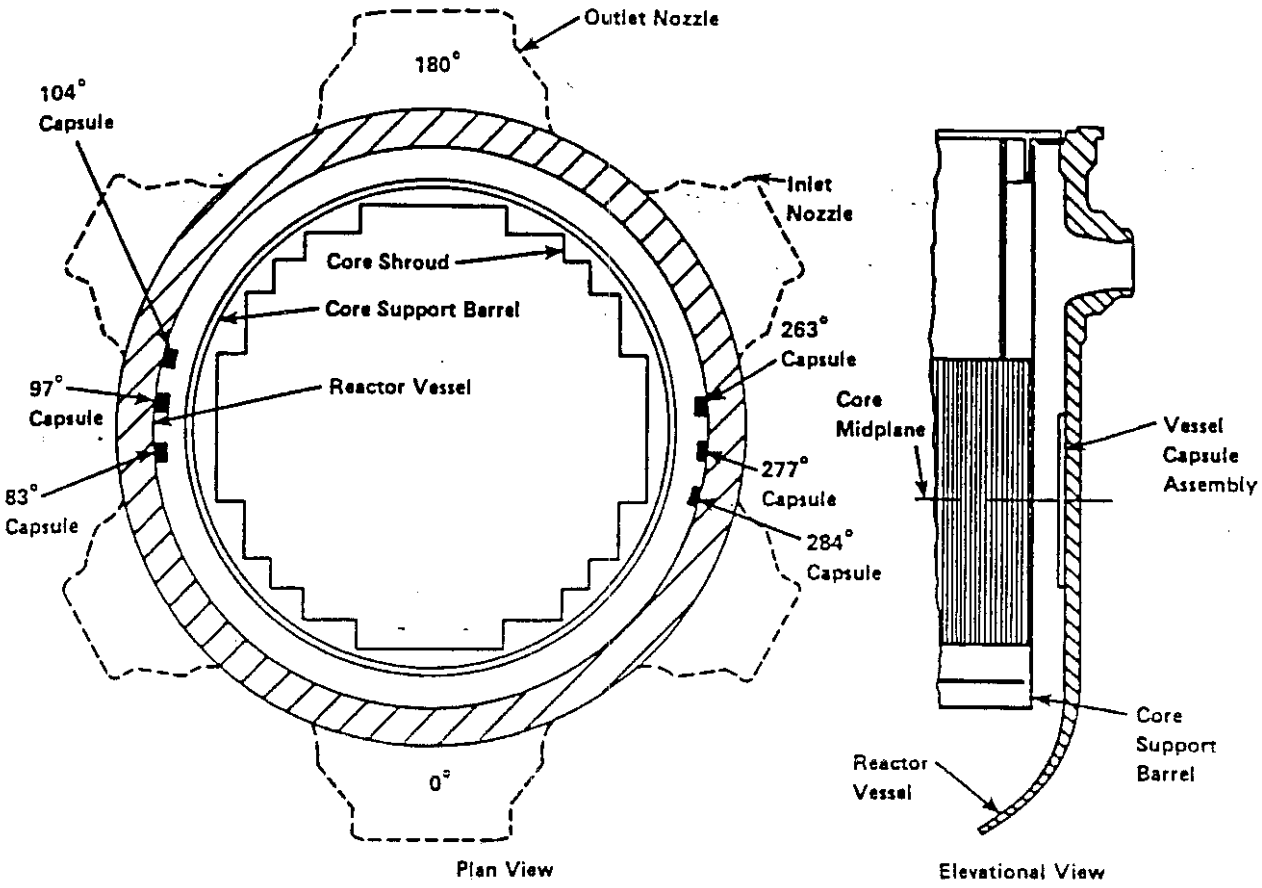
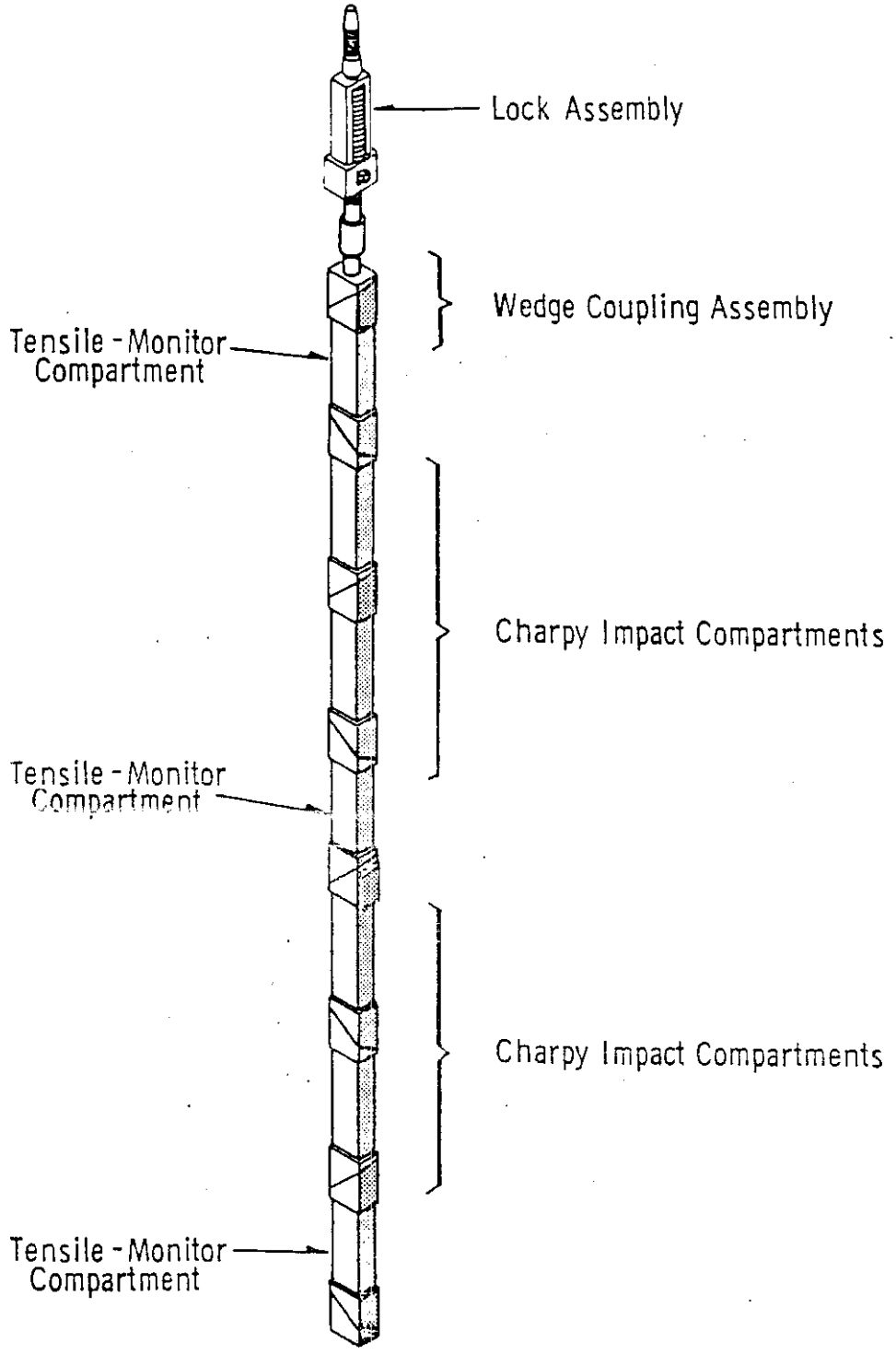


Figure 3-2. Surveillance Capsule Assembly Showing Locations of Specimens and Monitors for Arkansas Nuclear One Unit 2 Capsule W-104



4. Tests of Unirradiated Material

Unirradiated material was evaluated for two purposes: (1) to establish baseline data to which irradiated properties data could be compared; and (2) to determine those material properties as required for compliance with 10 CFR 50, Appendices G and H.

Combustion Engineering, Inc., as part of the development of the Arkansas Nuclear One Unit 2 RVSP, performed the testing of the unirradiated surveillance material. The details of the testing procedures are described by Combustion Engineering, Inc.[9]. The unirradiated mechanical properties for the Arkansas Nuclear One Unit 2 RVSP materials are summarized in Appendices C and D of this report.

The original unirradiated Charpy V-notch impact data were evaluated based on hand-fit Charpy curves generated using engineering judgment[9]. These data were re-evaluated herein using a hyperbolic tangent curve-fitting program, and the results of the re-evaluation are presented in Appendix D. In addition, Appendix E contains a comparison of the Charpy V-notch shift results for each surveillance material, hand-fit versus hyperbolic tangent curve-fit.

5. Post-Irradiation Testing

The post-irradiation testing of the tensile specimens, the Charpy V-notch impact specimens, thermal monitors, and dosimeters for the Arkansas Nuclear One Unit 2 capsule W-104 was performed at the BWX Technologies, Lynchburg Technology Center (LTC).[10]

5.1. Capsule Disassembly and Inventory

After capsule disassembly, the contents of Capsule W-104 were inventoried and found to be consistent with the surveillance program report inventory (CEN-15(A)). The capsule contained a total of 48 standard Charpy V-notch specimens, nine (9) tensile specimens, six (6) dosimetry blocks, and three (3) sets of temperature monitors.

5.2. Thermal Monitors

The low-melting point (536°F, 558°F, 580°F and 590°F) eutectic alloys contained in Capsule W-104 were encapsulated in quartz tubes. Each set of thermal monitors (capsule top, middle, and bottom) were photographed to reveal the shape of the monitors and examined for evidence of melting. In all cases the 536°F and 558°F temperature monitors showed evidence of partial melting, and the 580°F and 590°F monitors did not show any evidence of melting. The middle compartment 590°F temperature monitor appeared to show evidence of partial melting (See Figure 5-2), but upon opening the quartz tube, the temperature monitor was found intact and unmelted. There was a film on the inside of the quartz tube. Based on this examination, the maximum temperature that the capsule test specimens were exposed to was between 558°F and 580°F. Figures 5-1 through 5-3 contain photographs of the temperature monitors.

5.3. Tension Test Results

The capsule contained nine specimens, three specimens made of base metal with a transverse orientation, three made of HAZ material, and three made of weld metal. The specimens were of the standard round type with a gage length of 1.0 inch and a nominal gage diameter of 0.250 inch. Tensile tests were conducted at 70°F, 250°F and 550°F. The results of the post-irradiation tension tests are presented in Table 5-1, and the stress-strain curves are presented in Figures 5-4 through 5-12. The tests were performed using a MTS servohydraulic test machine using stroke control with an initial actuator travel rate of 0.0075 inch per minute. Following specimen yielding, an actuator speed of 0.03 inch per minute was used. The tension testing was performed in accordance with the

applicable requirements of ASTM Standard E 21-92.[11] Photographs of the tension test specimen fractured surfaces are shown in Figures 5-13 through 5-15.

5.4. Charpy V-Notch Impact Test Results

The Charpy V-notch impact testing was performed in accordance with the applicable requirements of ASTM Standard E 23-91.[12] Impact energy, lateral expansion, and percent shear fracture were measured at numerous test temperatures and recorded for each specimen. The impact energy was measured using a certified Satec S1-1K Impact tester (traceable to NIST Standard) with a striker velocity of 16.90 ft/sec and 240 ft-lb of available energy. The lateral expansion was measured using a certified dial indicator. The specimen percent shear was estimated by video examination and comparison with the visual standards presented in ASTM Standard E 23-91.

The results of the Charpy V-notch impact testing are shown in Tables 5-2 through 5-5 and Figures 5-16 through 5-19. The curves were generated using a hyperbolic tangent curve-fitting program to produce the best-fit curve through the data. The hyperbolic tangent (TANH) function (test response, i.e., absorbed energy, lateral expansion, and percent shear fracture, "R," as a function of test temperature, "T") used to evaluate the surveillance data is as follows:[13]

$$R = A + B * \tanh\left[\frac{(T - To)}{C}\right]$$

The Charpy V-notch data was entered, and the coefficients *A, B, To,* and *C* are determined by the program minimizing the sum of the errors squared (least-squares fit) of the data points about the fitted curve. Using these coefficients and the above TANH function, a smooth curve is generated through the data for interpretation of the material transition region behavior. The coefficients determined for irradiated materials in Capsule W-104 are shown in Table 5-6. When performing the TANH fit for the absorbed energy, the lower shelf was fixed at 2.2 ft-lbs and the upper shelf was fixed at the upper shelf value defined by ASTM E185-82. For the lateral expansion data, the lower shelf was fixed at 1 mil and the upper shelf was not fixed for the TANH fit. For the percent shear fracture data, the lower shelf was fixed at 0% and the upper shelf at 100% shear for the TANH fit.

Photographs of the Charpy V-notch specimen fracture surfaces are presented in Figures 5-20 through 5-23. The photograph of the base metal specimen number 251 fracture surfaces shows elongated features on both surfaces (See Figure 5-20). The location and shape of the feature is the same on both halves and is relatively large. The orientation of the elongated feature is

consistent with the rolling direction. Consequently, it is suspected that the elongated features are MnS inclusions. It is not unusual to find such inclusions in low alloy steel plate, however, they tend to be rather small and isolated.[9] Our experience suggests that MnS stringers could be of the order of 5-10 mils in length. The possible stringers in this particular specimen are considerably larger than this and, given, their size could have affected the impact toughness and MLE values exhibited by this specimen. It is also important to note that this feature is not seen in any of the other specimens and is suspected to be an isolated feature.

Due to the isolated feature in specimen 251, the impact data from this specimen was not used when determining the hyperbolic tangent curve fit. The inclusion or exclusion of the specimen 251 data in the tanh fit is inconsequential for the following reasons:

- When the data from specimen 251 is included in the tanh fit, the shift in RT_{NDT} (Table 7-3) is still within one standard deviation of the Regulatory Guide 1.99, Revision 2[18] predicted shift.
- The base metal plate C-8009-3 (specimen 251) that is included in the RVSP is not the limiting material in the reactor vessel, therefore the limiting material (plate C-8010-1) adjusted reference temperature and the pressurized thermal shock reference temperature are unaffected by this surveillance data.

All Charpy V-notch impact testing was performed using instrumentation to record a load-versus-time trace and energy-versus-time trace for each impact event. The load-versus-time traces were analyzed to determine time, load, and impact energy for general yielding, maximum load, fast fracture, and crack arrest properties during the test. The dynamic yield stress is calculated from the three-point bend formula:[14]

$$\sigma_y = 33.33 * (general\ yielding\ load)$$

The dynamic flow stress is calculated from the average of the yield and maximum loads, also using the three-point bend formula:[14]

$$\sigma_{flow} = 33.33 * \left(\frac{(general\ yielding\ load + maximum\ load)}{2} \right)$$

The instrumented Charpy V-notch impact properties are presented in Tables 5-7 through 5-10.

Table 5-1. Tensile Properties of Arkansas Nuclear One Unit 2 Capsule W-104 Reactor Vessel Surveillance Materials, Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)*

Material	Specimen No.	Test Temp. (°F)	Strength		Fracture Properties			Elongation		Reduction in Area (%)
			Yield (ksi)	Ultimate (ksi)	Load (lb)	Stress (ksi)	Strength (ksi)	Uniform (%)	Total (%)	
Base Metal Plate C-8009-3 Heat C8182-2 Transverse	2JE	70	78.0	100.0	3253	179	66.3	9.96	22.48	63.0
	2L3	250	74.1	93.3	3041	176	62.0	8.88	20.66	64.7
	2JT	550	68.3	94.1	3161	166	64.4	7.08	19.75	61.3
Weld Metal Heat 10120 C-8009-1/C-8009-2	3J7	70	78.0	92.9	2851	204	58.1	8.42	22.48	71.5
	3KP	250	72.9	85.0	2640	177	53.8	7.31	20.22	69.5
	3K5	550	68.8	87.2	2625	158	53.5	8.38	21.55	66.1
HAZ Metal	4K4	70	75.0	90.5	2632	188	53.6	6.81	21.81	71.5
	4JM	250	73.3	88.7	2795	184	57.0	6.89	20.21	69.1
	4KU	550	67.3	85.2	2679	159	54.6	6.50	19.75	65.7

* The fluence calculation is described in Section 6.0 of this report.

Table 5-2. Charpy V-Notch Impact Results for Arkansas Nuclear One Unit 2
Capsule W-104

Base Metal Plate C-8009-3, Heat C8182-2,
Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)
Transverse Orientation

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
25L	0	8.5	7	0
227	30	13.5	13	5
246	50	28.5	26	25
24L	60	26	24	20
23C	70	29.5	26	30
251[(a)]	100	19.5	17	55
22A	100	46.5	41	45
24K	125	57	49	60
236	200	80.5	69	85
23B	300	93.5*	78	100
252	350	88.5*	77	100
24Y	400	93.5*	82	100

(a) There is an anomaly in the fracture surface of the specimen.

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[15]

Table 5-3. Charpy V-Notch Impact Results for Arkansas Nuclear One Unit 2
Capsule W-104
Weld Metal, Heat 10120,
Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
342	-30	10.5	11	5
31B	0	22.5	23	30
36M	15	27.5	28	35
357	30	23.5	26	50
33P	30	66.5	56	55
37J	40	80	66	70
32C	50	22.5	25	40
32A	50	74	60	65
32M	70	88	72	85
35B	125	118*	81	95
36L	200	124.5*	96	100
36B	400	131*	95	100

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[15]

Table 5-4. Charpy V-Notch Impact Results for Arkansas Nuclear One Unit 2
Capsule W-104
Heat-Affected-Zone Material,
Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
43L	-60	27	19	20
47P	-30	22	17	30
421	-15	35	24	40
475	0	49.5	32	55
44M	30	57.5	39	40
41L	30	59.5	40	50
467	50	62.5	42	65
462	50	56.5	40	50
45C	70	111	75	95
456	125	133.5*	82	100
424	200	131.5*	81	100
413	400	123.5*	81	100

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[15]

Table 5-5. Charpy V-Notch Impact Results for Arkansas Nuclear One Unit 2
Capsule W-104
Standard Reference Material, HSST Plate 01
Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV)

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
77D	70	6	5	5
777	100	10.5	13	10
77E	130	22.5	22	35
76Y	150	24	23	40
77U	150	23.5	22	40
776	170	26	27	50
773	170	37.5	34	45
77P	200	48	43	55
77M	250	65	60	70
772	300	86*	75	100
77Y	350	89*	81	100
77A	400	87*	80	100

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[15]

Table 5-6. Hyperbolic Tangent Curve Fit Coefficients for the Arkansas Nuclear One Unit 2 Capsule W-104 Surveillance Materials

Material Description	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Base Metal Plate C-8009-3 Heat C8182-2 Transverse	A: 47.0 B: 44.8 C: 90.3 T0: 103.8	A: 40.2 B: 39.2 C: 96.6 T0: 101.6	A: 50.0 B: 50.0 C: 80.7 T0: 109.9
Weld Metal Heat 10120 C-8009-1/C-8009-2	A: 63.4 B: 61.2 C: 61.6 T0: 50.0	A: 47.9 B: 46.9 C: 73.9 T0: 41.9	A: 50.0 B: 50.0 C: 61.6 T0: 31.3
HAZ Metal	A: 65.9 B: 63.7 C: 82.9 T0: 36.0	A: 43.0 B: 42.0 C: 91.4 T0: 32.9	A: 50.0 B: 50.0 C: 101.3 T0: 16.0
Standard Reference Material, HSST Plate 01	A: 44.8 B: 42.6 C: 85.7 T0: 194.8	A: 42.3 B: 41.3 C: 103.0 T0: 199.3	A: 50.0 B: 50.0 C: 103.2 T0: 178.5

Table 5-7 Instrumented Charpy V-Notch Properties of Arkansas Nuclear One Unit 2 Capsule W-104, Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 2.937 x 10^{19} n/cm^2 (E>1.0 MeV) Transverse Orientation

Specimen ID	Test Temp. (F)	Charpy Energy (ft-lbf)	Yield Properties			Maximum Load Properties			Fast Fracture Properties			Crack Arrest Properties			Propagation Load Properties		Total Energy Properties		Yield Stress (ksi)	Flow Stress (ksi)
			Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Energy (ft-lbf)		
25L	0	8.5	146	3535	4.6	146	3535	4.6	146	3535	4.6	212	0	7.1	3540	2.5	212	7.1	117.8	117.8
227	30	13.5	151	3436	4.7	235	3701	9.6	235	3701	9.6	291	0	11.7	3698	2.0	291	11.7	114.5	118.9
246	50	28.5	147	3402	4.7	440	4161	23.5	458	4138	24.8	522	4.6	27.1	4133	3.6	522	27.1	113.4	126.0
24L	60	26.0	145	3314	4.5	398	4009	20.2	398	4009	20.2	465	0	22.6	4014	2.4	465	22.6	110.5	122.0
23C	70	29.5	144	3335	4.3	434	4108	22.4	434	4108	22.4	522	0	25.9	4108	3.5	522	25.9	111.2	124.0
251	100	19.5	144	3061	3.8	163	3098	4.8	170	3084	5.2	285	1304	8.5	1780	12.6	2388	17.4	102.0	102.6
22A	100	46.5	156	3181	4.6	614	4221	34.1	682	4175	38.9	789	1228	42.6	2946	10.7	1321	44.7	106.0	123.4
24K	125	57.0	164	3142	4.2	630	4216	33.8	788	4085	44.9	900	1548	49.1	2537	22.5	3290	56.3	104.7	122.6
236	200	80.5	172	2939	5.0	714	4014	38.8	1052	3376	60.4	1126	2792	64.3	584	43.5	3150	82.4	98.0	115.9
23B	300	93.5	162	2753	4.7	616	3772	30.9	NA	NA	NA	NA	NA	NA	0	63.2	3928	94.0	91.8	108.7
252	350	88.5	138	2620	3.4	618	3671	30.3	NA	NA	NA	NA	NA	NA	0	59.1	3862	89.5	87.3	104.8
24Y	400	93.5	172	2668	4.6	710	3662	35.3	NA	NA	NA	NA	NA	NA	0	57.8	4056	93.1	88.9	105.5

Table 5-8. Instrumented Charpy V-Notch Properties of Arkansas Nuclear One Unit 2 Capsule W-104, Weld Metal, Heat 10120, Irradiated to 2.937 x 10^{19} n/cm^2 (E > 1.0 MeV)

Specimen ID	Test Temp. (F)	Charpy Energy (ft-lbf)	Yield Properties			Maximum Load Properties			Fast Fracture Properties			Crack Arrest Properties			Propagation Load Properties		Total Energy Properties		Yield Stress (ksi)	Flow Stress (ksi)
			Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Energy (ft-lbf)		
342	-30	10.5	151	3823	4.9	151	3823	4.9	151	3823	4.9	242	0	8.5	3825	3.7	242	8.5	127.4	127.4
31B	0	22.5	147	3627	4.7	318	3979	15.5	318	3979	15.5	430	782	18.2	3197	4.9	1237	20.4	120.9	126.8
36M	15	27.5	147	3542	4.6	304	3813	14.2	304	3813	14.2	418	1350	17.6	2463	9.9	1505	24.1	118.1	122.6
357	30	23.5	152	3480	4.6	246	3705	10.1	246	3705	10.1	368	1286	14.1	2420	11.6	2950	21.7	116.0	119.7
33P	30	66.5	148	3445	4.8	704	4170	41.8	950	3901	58.6	1062	1274	62.5	2627	24.7	2062	66.5	114.8	126.9
37J	40	80.0	143	3399	4.6	700	4092	40.9	1129	3505	69.0	1240	1375	72.5	2130	39.2	3206	80.1	113.3	124.8
32C	50	22.5	154	3439	4.6	242	3673	9.8	242	3673	9.8	360	1187	13.3	2486	9.0	1448	18.8	114.6	118.5
32A	50	74.0	144	3413	4.4	698	4177	40.8	1088	3728	67.1	1200	929	70.7	2799	33.4	2300	74.2	113.8	126.5
32M*	70	88.0	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
35B	125	118.0	154	3084	4.6	704	3956	38.8	1220	3358	71.2	1310	2654	75.7	704	83.5	4888	122.3	102.8	117.3
36L	200	124.5	168	2847	4.8	710	3738	36.4	NA	NA	NA	NA	NA	NA	0	94.8	4456	131.2	94.9	109.7
36B	400	131.0	170	2555	4.5	710	3443	33.2	NA	NA	NA	NA	NA	NA	0	103.5	4755	136.7	85.2	100.0

* The data for this specimen was inadvertently erased.

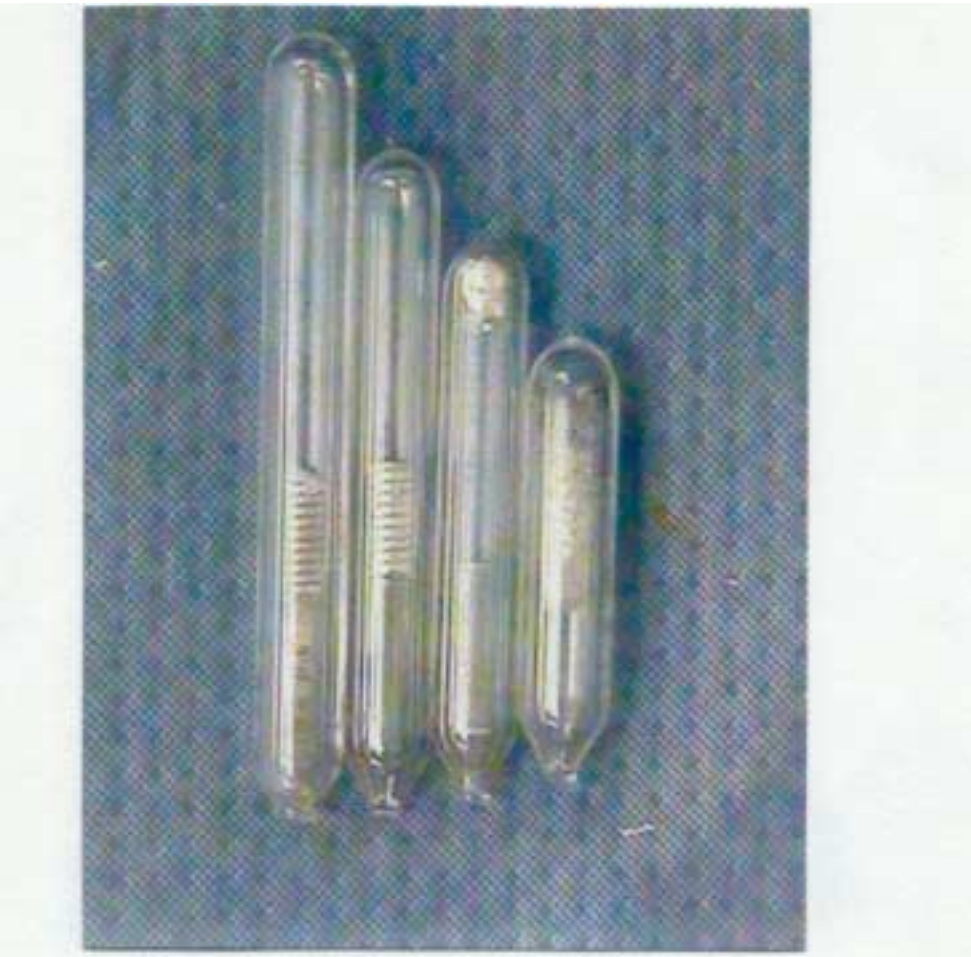
Table 5-9. Instrumented Charpy V-Notch Properties of Arkansas Nuclear One Unit 2 Capsule W-104, Heat-Affected-Zone Material, Irradiated to 2.937 x 10^{19} n/cm^2 (E > 1.0 MeV)

Specimen ID	Test Temp. (F)	Charpy Energy (ft-lbf)	Yield Properties			Maximum Load Properties			Fast Fracture Properties			Crack Arrest Properties			Propagation Load Properties		Total Energy Properties		Yield Stress (ksi)	Flow Stress (ksi)
			Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Energy (ft-lbf)		
43L	-60	27.0	150	4030	5.2	390	4421	22.3	390	4421	22.3	454	0	25.0	4428	2.7	454	25.0	134.3	140.8
47P	-30	22.0	147	3832	4.9	320	4214	16.5	320	4214	16.5	387	0	18.9	4216	2.4	387	18.9	127.7	134.1
421	-15	35.0	150	3809	5.1	506	4455	30.3	506	4455	30.3	578	0	33.0	4460	2.7	578	33.0	127.0	137.7
475	0	49.5	151	3694	5.0	612	4453	37.2	643	4347	39.6	756	1564	44.1	2783	11.3	1751	48.5	123.1	135.8
44M	30	57.5	154	3593	4.7	614	4492	36.7	882	4214	56.6	956	0	59.3	4244	22.6	956	59.3	119.8	134.7
41L	30	59.5	156	3574	4.9	708	4469	43.7	844	4278	53.8	958	1063	58.2	3215	16.4	1320	60.1	119.1	134.0
462	50	56.5	156	3595	4.7	616	4425	36.3	778	4253	48.2	882	1672	52.9	2581	20.2	1318	56.5	119.8	133.7
467	50	62.5	148	3540	4.7	610	4347	36.2	860	4131	54.2	976	1136	58.5	2995	26.0	2072	62.2	118.0	131.4
45C	70	111.0	154	3508	4.5	708	4395	42.7	1450	3197	91.3	1532	2509	95.2	688	74.5	3460	117.2	116.9	131.7
456	125	133.5	162	3232	4.9	712	4296	41.5	NA	NA	NA	NA	NA	NA	0	102.4	4416	144.0	107.7	125.5
424	200	131.5	170	3061	4.7	720	4175	39.9	NA	NA	NA	NA	NA	NA	0	101.5	4160	141.4	102.0	120.6
413	400	123.5	170	2742	4.3	720	3779	36.1	NA	NA	NA	NA	NA	NA	0	93.2	4165	129.3	91.4	108.7

Table 5-10. Instrumented Charpy V-Notch Properties of Arkansas Nuclear One Unit 2 Capsule W-104, Standard Reference Material, HSST Plate 01, Irradiated to 2.937 x 10^{19} n/cm^2 (E > 1.0 MeV)

Specimen ID	Test Temp. (F)	Charpy Energy (ft-lbf)	Yield Properties			Maximum Load Properties			Fast Fracture Properties			Crack Arrest Properties			Propagation Load Properties		Total Energy Properties		Yield Stress (ksi)	Flow Stress (ksi)
			Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Energy (ft-lbf)		
77D	70	6.0	116	3071	3.0	116	3071	3.0	116	3071	3.0	168	0	4.3	3080	1.3	168	4.3	102.4	102.4
777	100	10.5	129	3245	3.6	156	3337	5.1	165	3326	5.6	241	0	7.6	3312	2.4	241	7.6	108.2	109.7
77E	130	22.5	163	3252	5.0	347	3901	16.2	355	3882	16.7	467	787	19.3	3096	4.8	995	20.9	108.4	119.2
76Y	150	24.0	158	3206	4.9	340	3857	15.9	345	3846	16.2	462	966	19.2	2880	5.9	991	21.7	106.9	117.7
77U	150	23.5	154	3209	4.6	358	3885	16.9	358	3885	16.9	481	1010	20.4	2875	5.4	909	22.3	107.0	118.2
776	170	26.0	160	3158	4.7	342	3823	15.5	350	3804	16.1	463	1474	19.6	2330	8.5	1160	24.0	105.3	116.3
773	170	37.5	163	3151	4.8	539	4195	28.8	548	4177	29.4	666	1113	33.2	3064	6.7	1192	35.5	105.0	122.4
77P	200	48.0	164	3105	4.5	624	4163	33.7	662	4071	36.3	776	1978	41.3	2093	13.8	1764	47.5	103.5	121.1
77M	250	65.0	166	3004	4.8	614	4018	32.7	714	3968	39.5	834	2988	46.3	980	32.9	2640	65.6	100.1	117.0
772	300	86.0	166	2933	4.7	706	3942	38.1	NA	NA	NA	NA	NA	NA	0	49.7	3126	87.8	97.8	114.6
77Y	350	89.0	166	2857	4.8	616	3836	31.5	NA	NA	NA	NA	NA	NA	0	60.0	3286	91.6	95.2	111.5
77A	400	87.0	168	2795	4.6	536	3756	25.5	NA	NA	NA	NA	NA	NA	0	63.6	3318	89.1	93.2	109.2

Figure 5-1. Photographs of Thermal Monitors Removed from the Arkansas Nuclear One Unit 2 RVSP Capsule W-104 Top Compartment



TEMPERATURE MONITORS IN TOP SEGMENT #1
(TEMPERATURES INCREASE WITH LENGTH)

Figure 5-2. Photographs of Thermal Monitors Removed from the Arkansas Nuclear One Unit 2 RVSP Capsule W-104 Middle Compartment

TEMPERATURE MONITORS IN MIDDLE SEGMENT #4
(TEMPERATURES INCREASE WITH LENGTH)

Figure 5-3. Photographs of Thermal Monitors Removed from the Arkansas Nuclear One Unit 2 RVSP Capsule W-104 Bottom Compartment

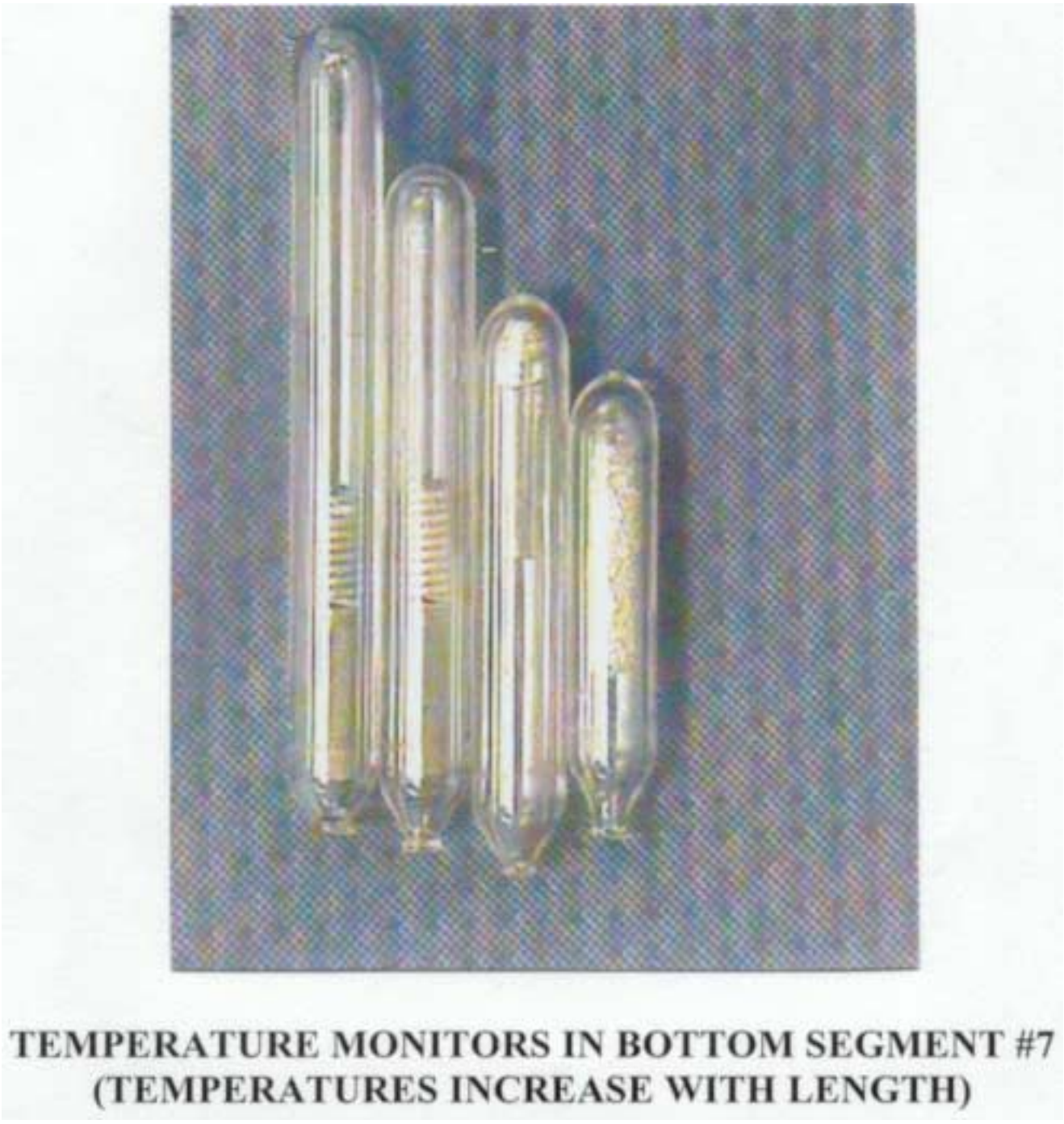


Figure 5-4. Tension Test Stress-Strain Curve for Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation, Specimen No. 2JE, Tested at 70°F

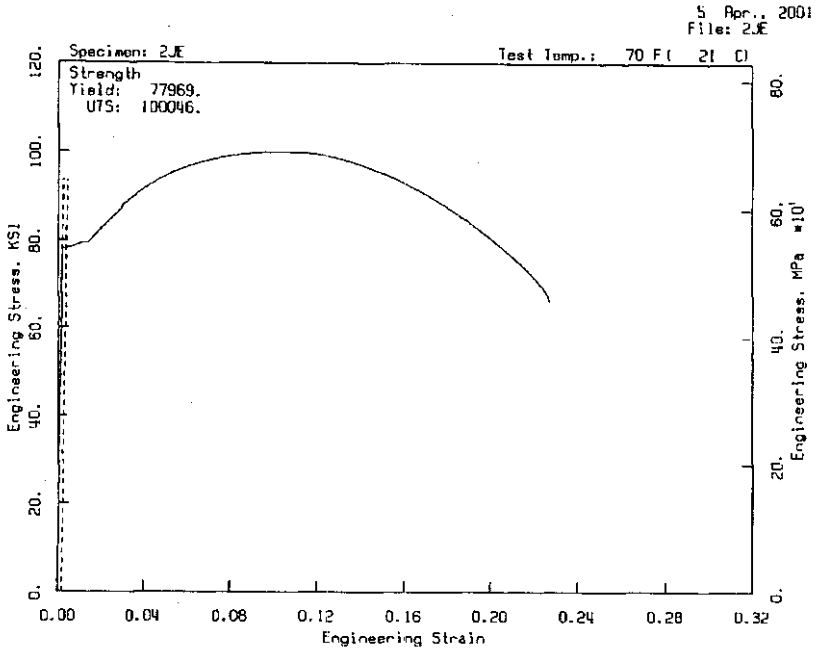


Figure 5-5. Tension Test Stress-Strain Curve for Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation, Specimen No. 2L3, Tested at 250°F

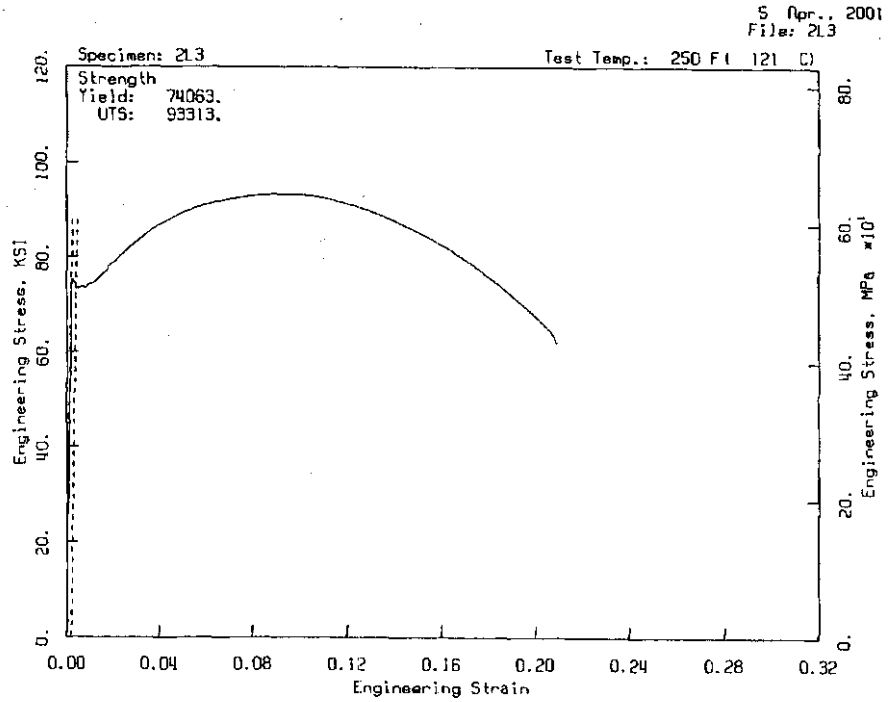


Figure 5-6. Tension Test Stress-Strain Curve for Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation, Specimen No. 2JT, Tested at 550°F

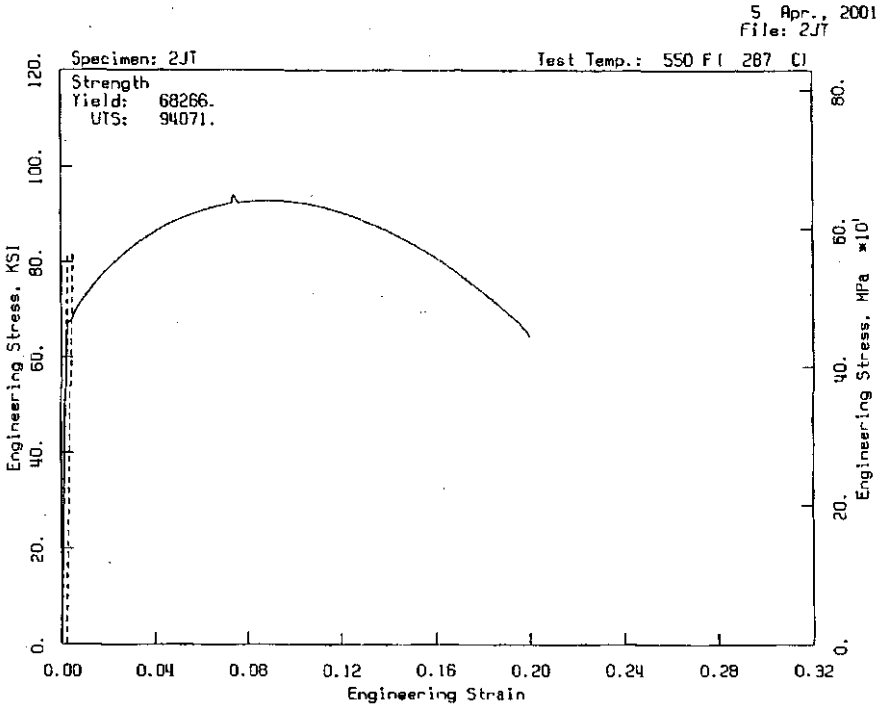


Figure 5-7. Tension Test Stress-Strain Curve for Weld Metal, Heat 10120, Specimen No. 3J7, Tested at 70°F

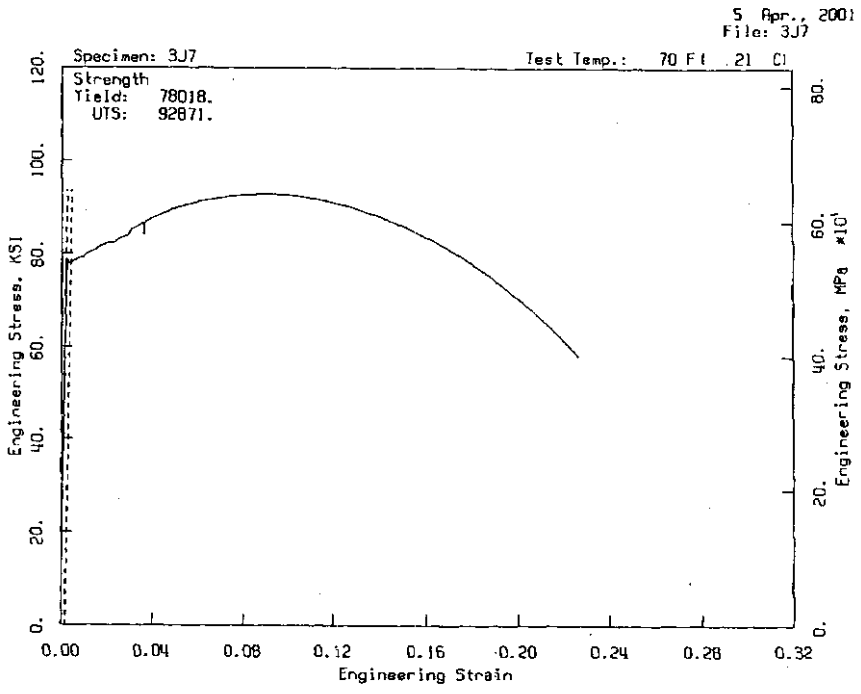


Figure 5-8. Tension Test Stress-Strain Curve for Weld Metal, Heat 10120, Specimen No. 3KP, Tested at 250°F

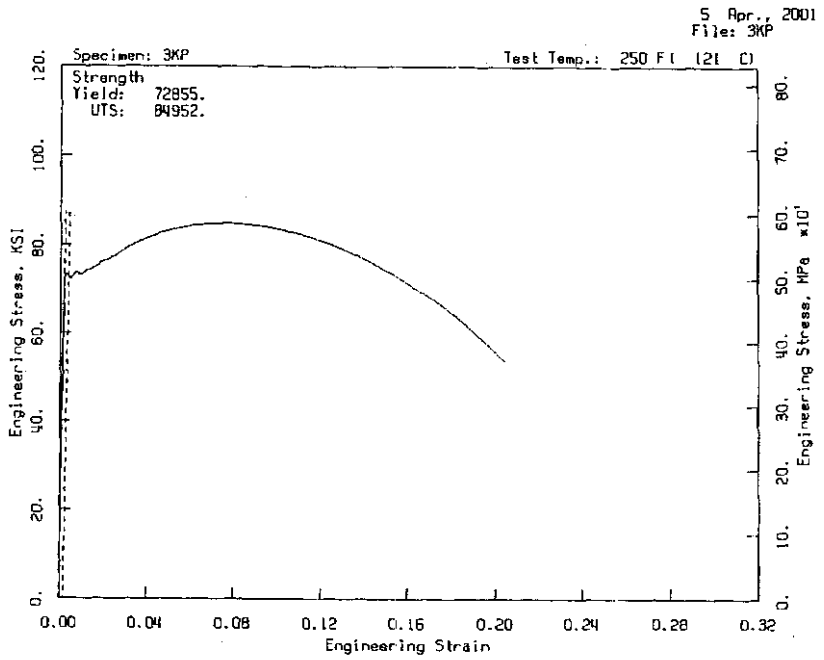


Figure 5-9. Tension Test Stress-Strain Curve for Weld Metal, Heat 10120, Specimen No. 3K5, Tested at 550°F

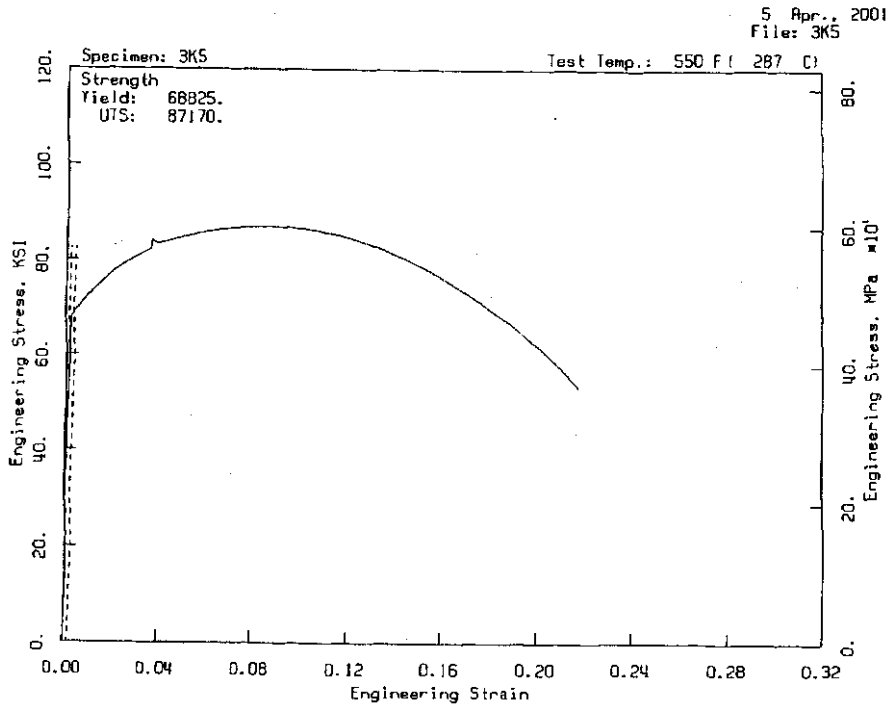


Figure 5-10. Tension Test Stress-Strain Curve for Heat-Affected-Zone Material, Specimen No. 4K4, Tested at 70°F

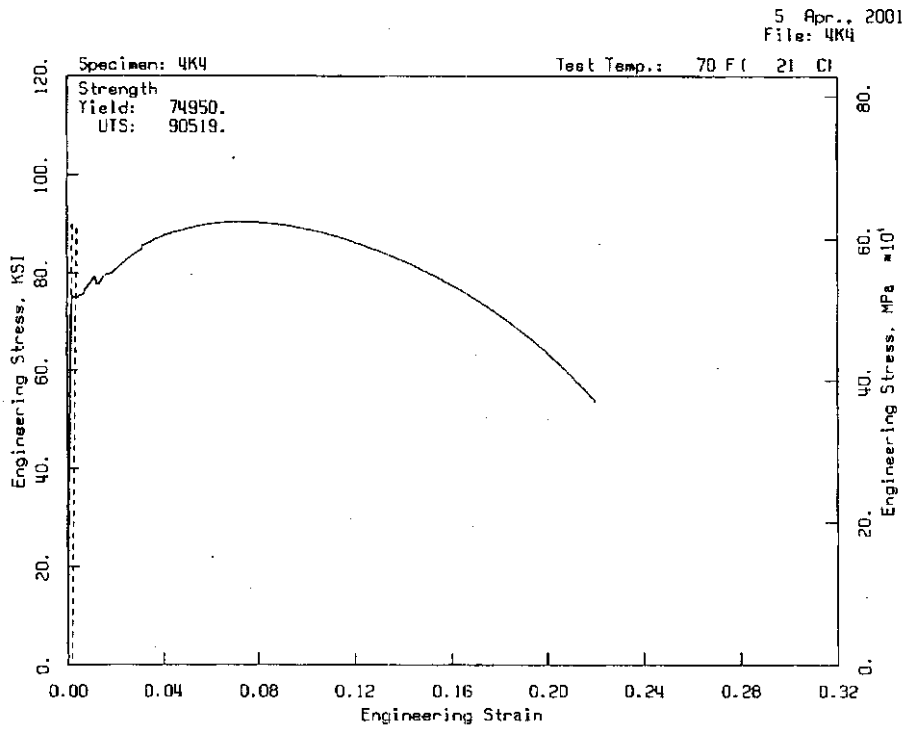


Figure 5-11. Tension Test Stress-Strain Curve Heat-Affected-Zone Material, Specimen No. 4JM, Tested at 250°F

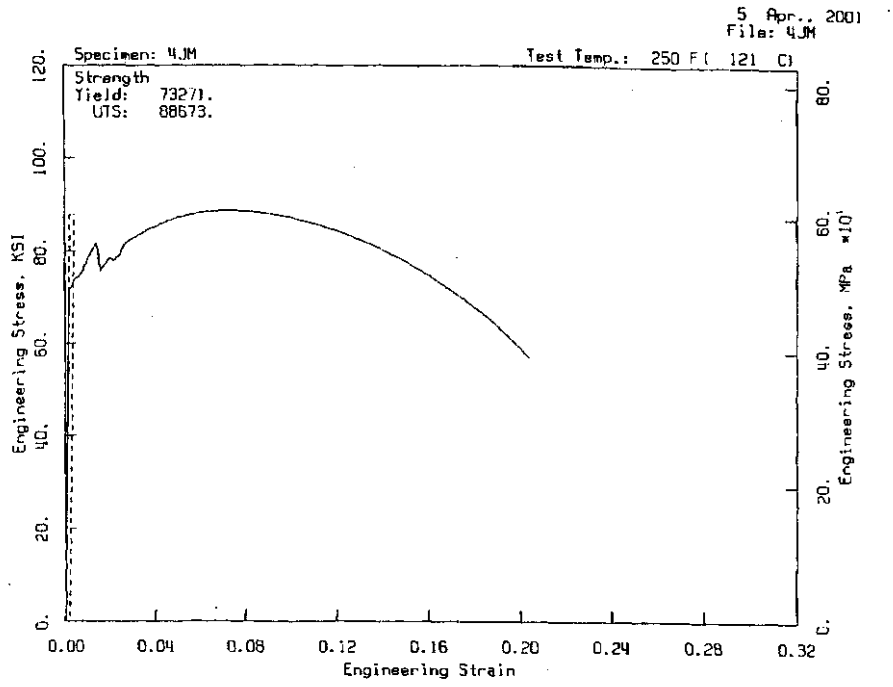


Figure 5-12. Tension Test Stress-Strain Curve for Heat-Affected-Zone Material, Specimen No. 4KU, Tested at 550°F

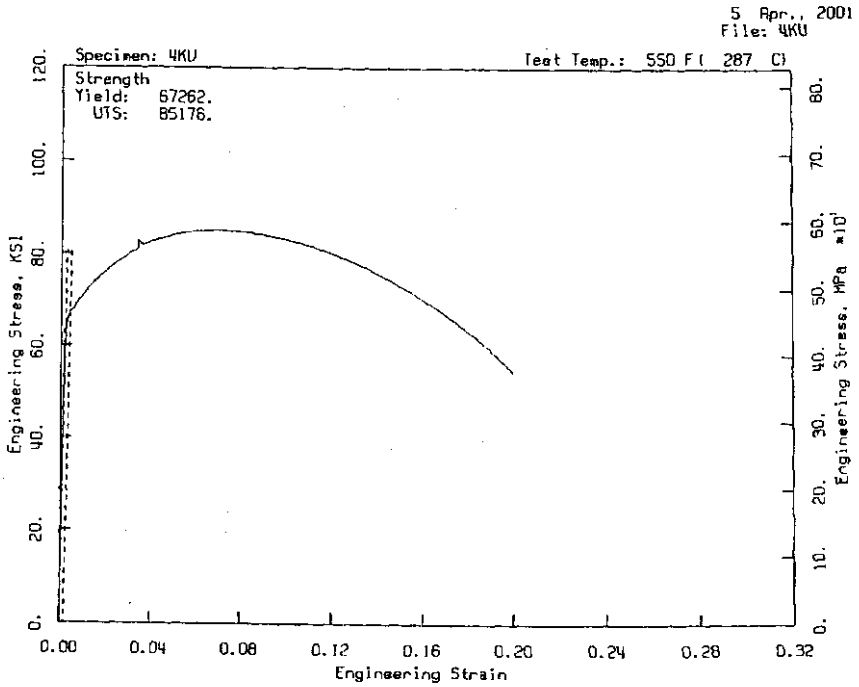


Figure 5-13. Photographs of Tested Tension Test Specimens and Corresponding Fracture Surfaces – Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation

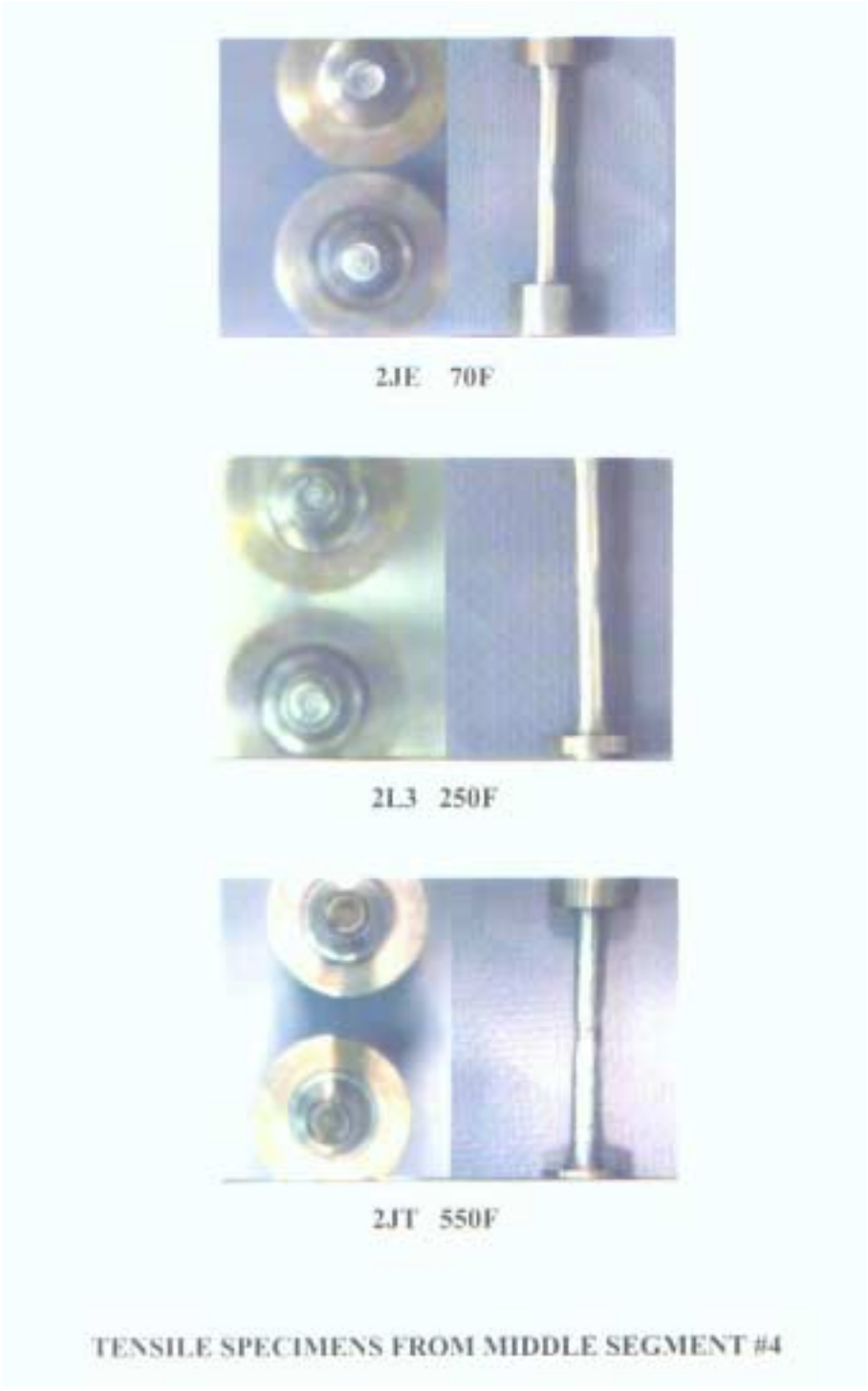


Figure 5-14. Photographs of Tested Tension Test Specimens and Corresponding Fracture Surfaces – Weld Metal, Heat 10120

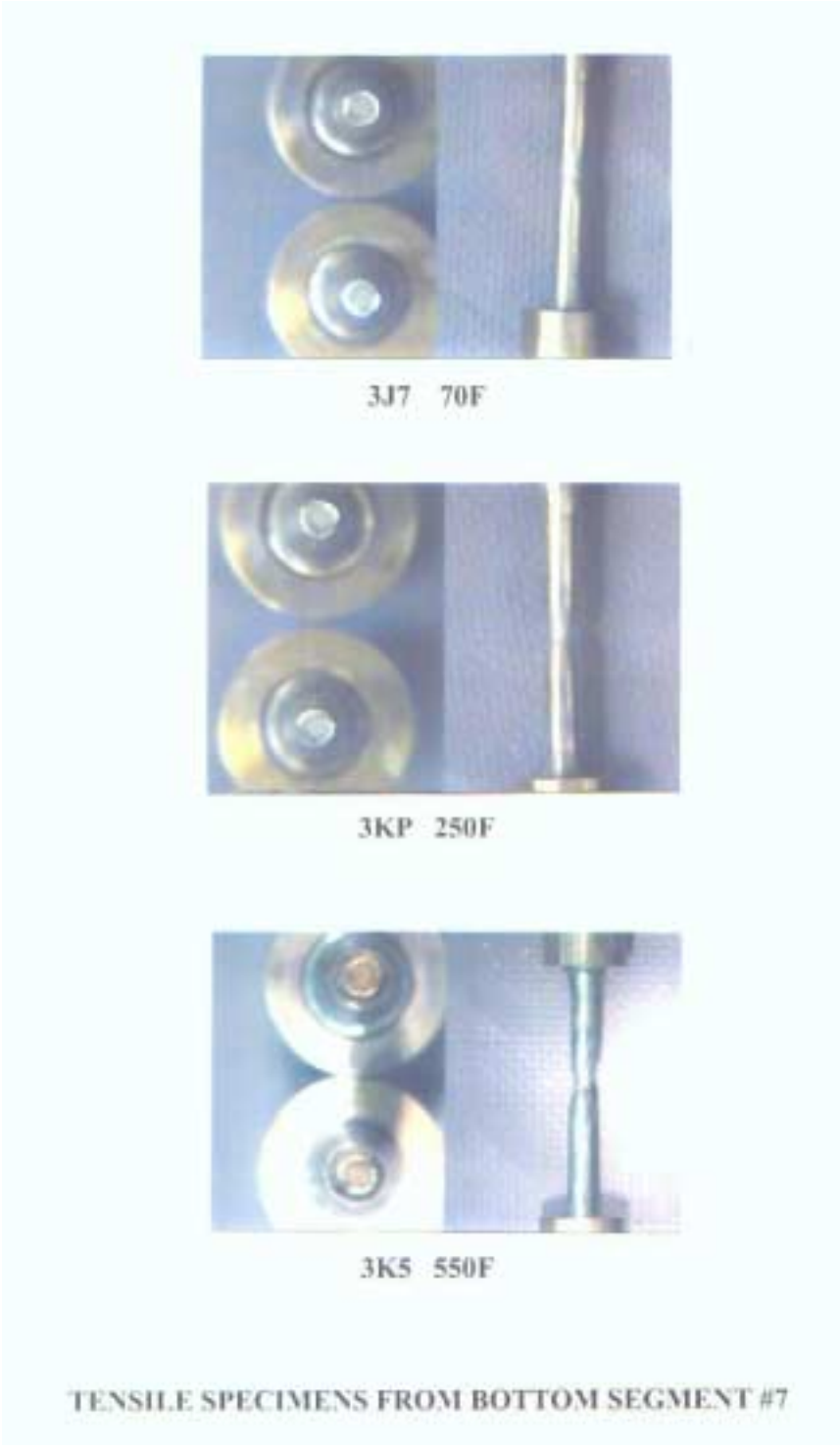


Figure 5-15. Photographs of Tested Tension Test Specimens and Corresponding Fracture Surfaces – Heat-Affected-Zone Material

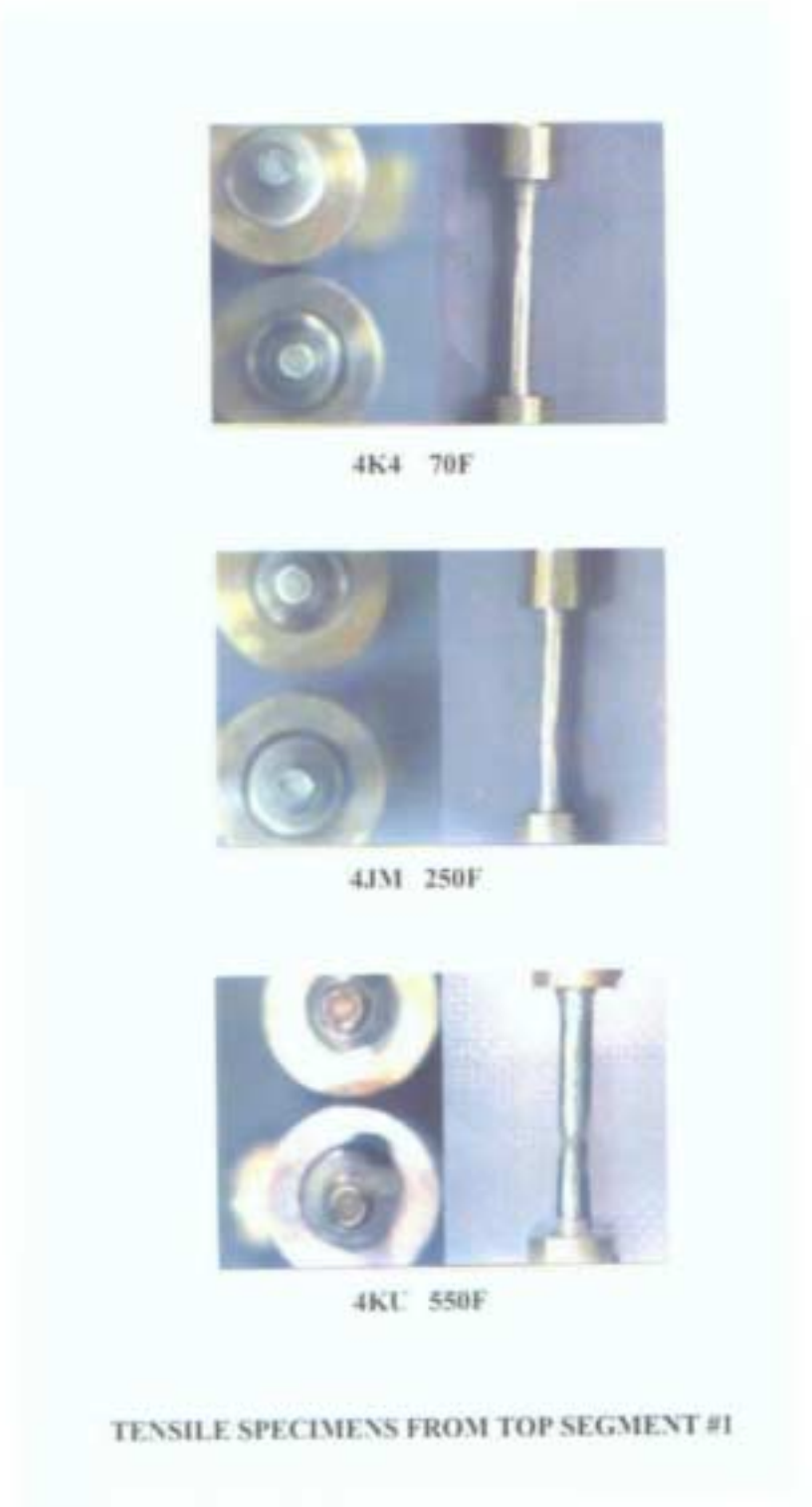
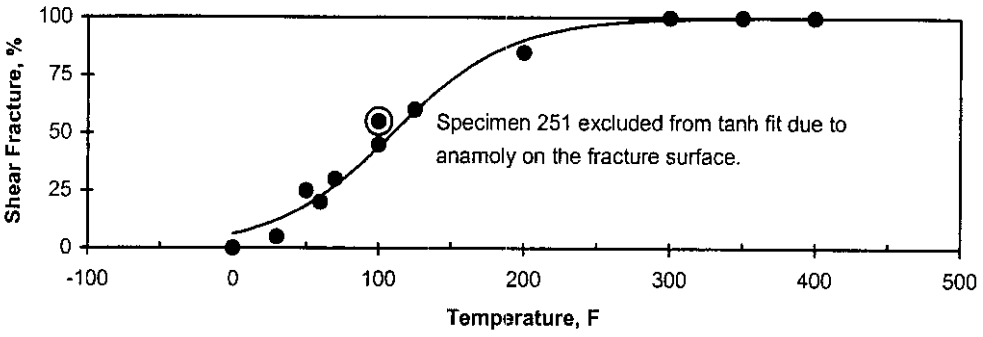
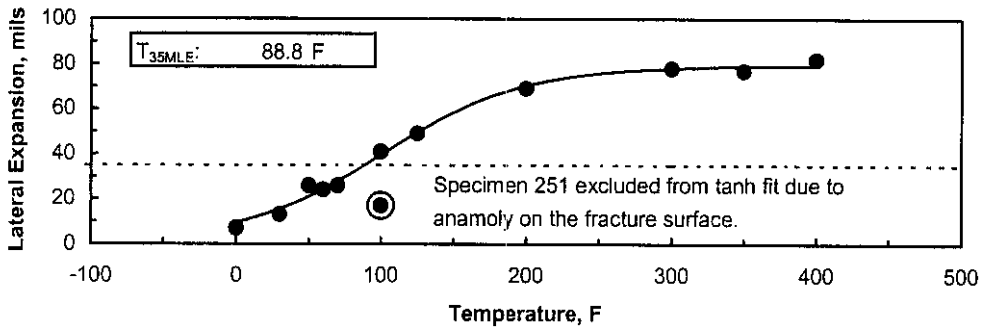


Figure 5-16. Charpy Impact Data for Irradiated Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation





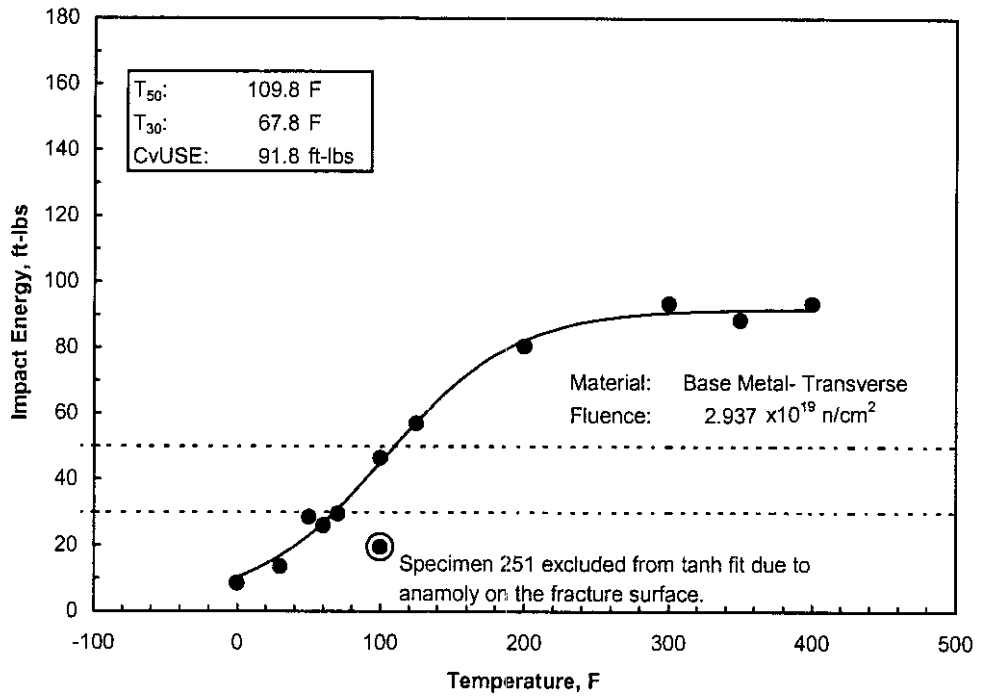
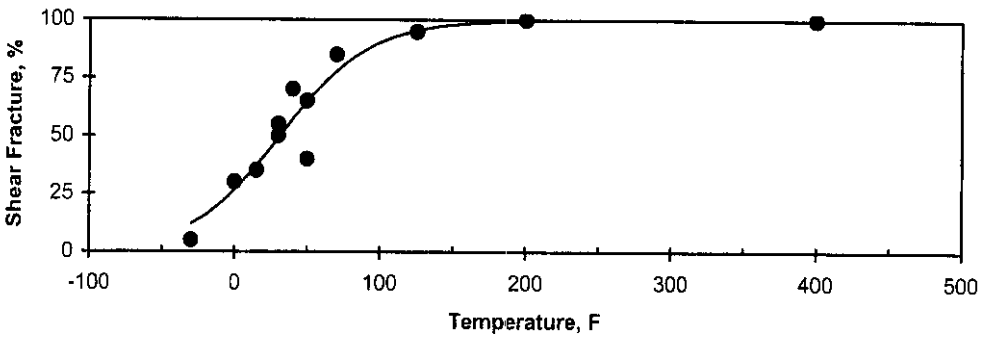
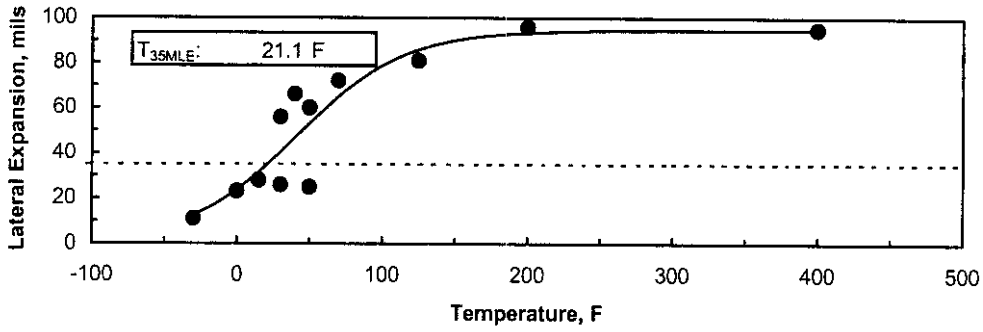


Figure 5-17. Charpy Impact Data for Irradiated Weld Metal, Heat 10120





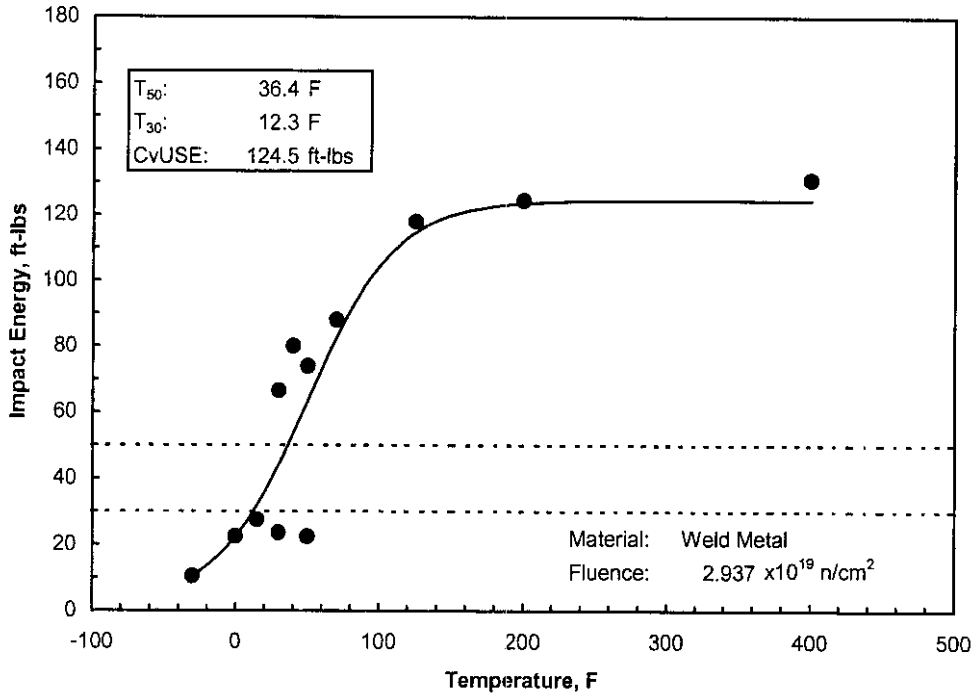
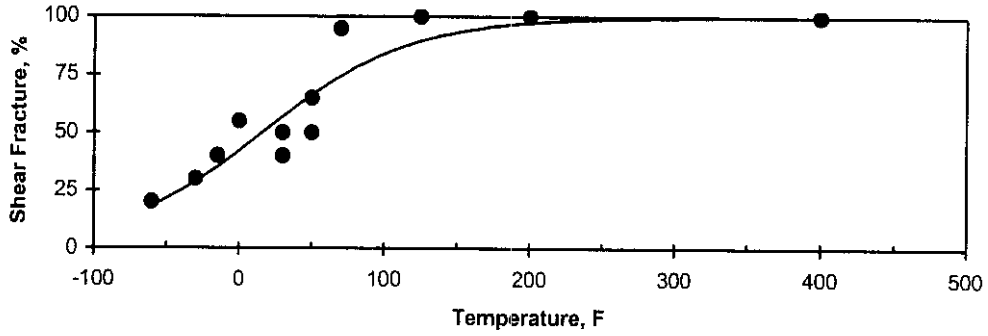
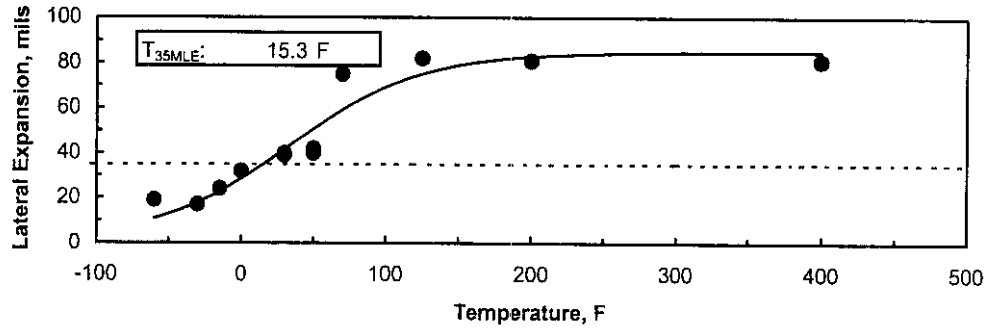


Figure 5-18. Charpy Impact Data for Irradiated Heat-Affected-Zone Material





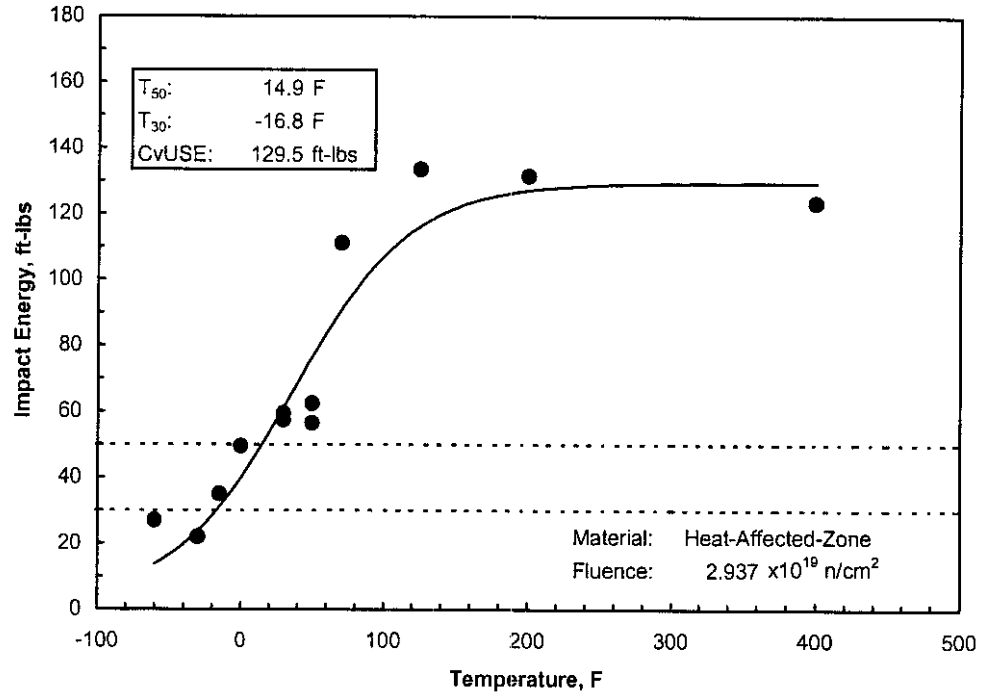
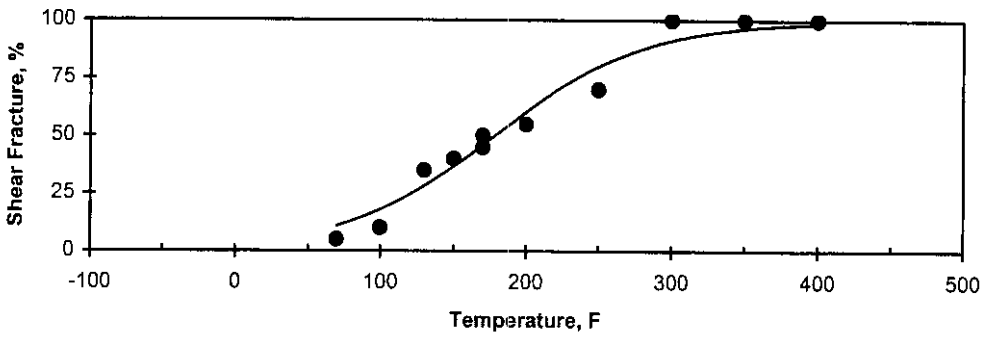
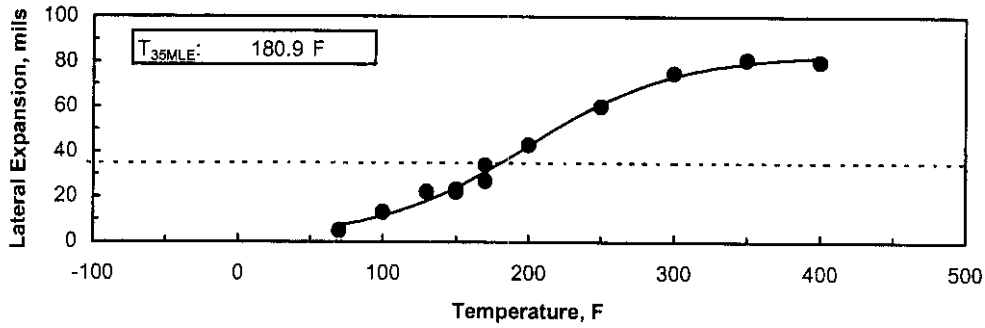


Figure 5-19. Charpy Impact Data for Irradiated Standard Reference Material, HSST Plate 01





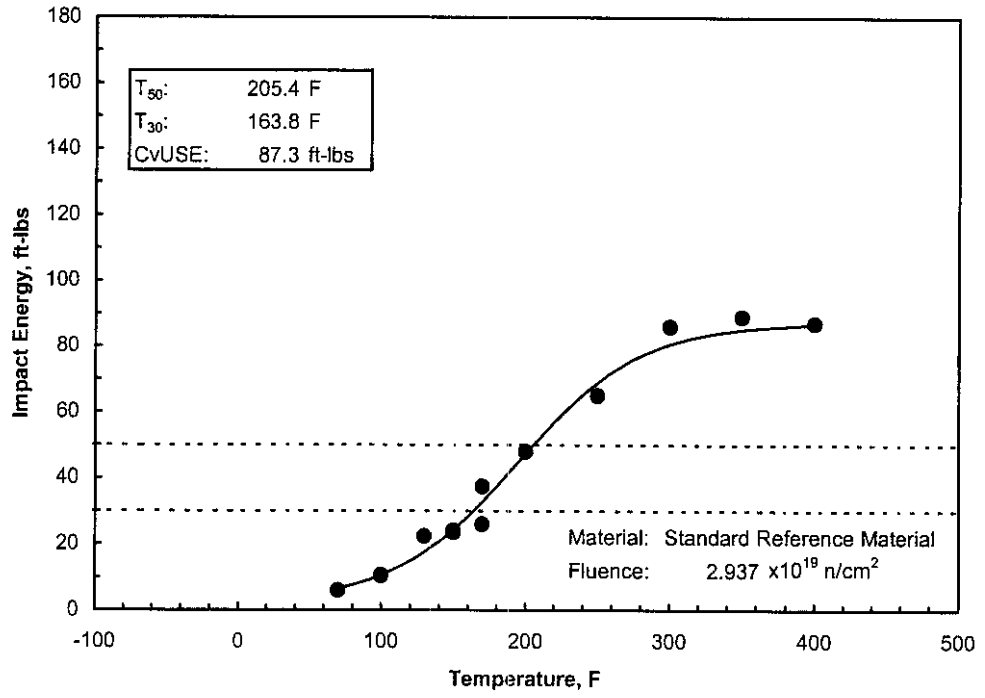


Figure 5-20. Photographs of Charpy Impact Specimen Fracture Surfaces, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation

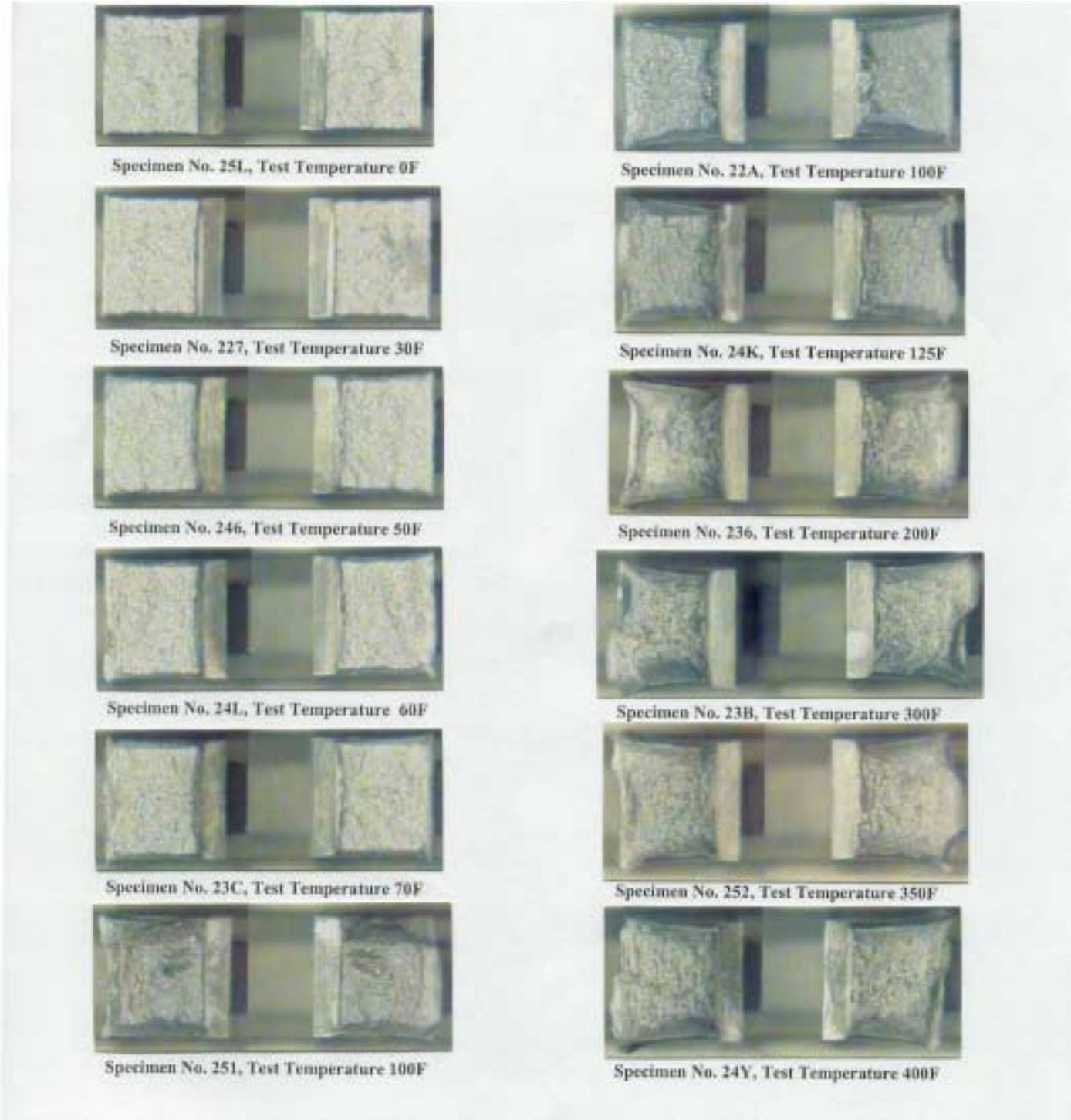


Figure 5-21. Photographs of Charpy Impact Specimen Fracture Surfaces, Weld Metal, Heat 10120

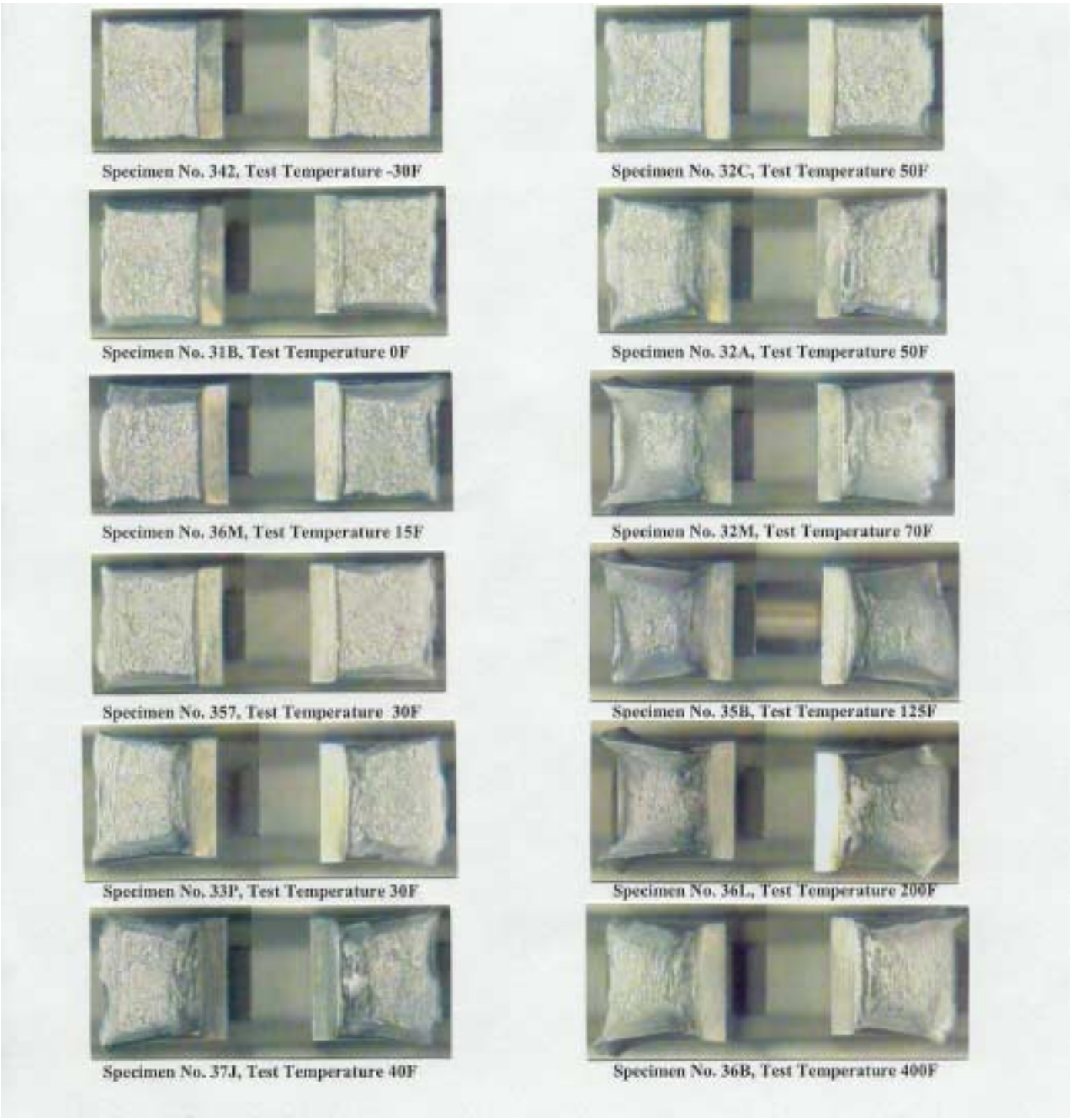


Figure 5-22. Photographs of Charpy Impact Specimen Fracture Surfaces, Heat-Affected-Zone Material

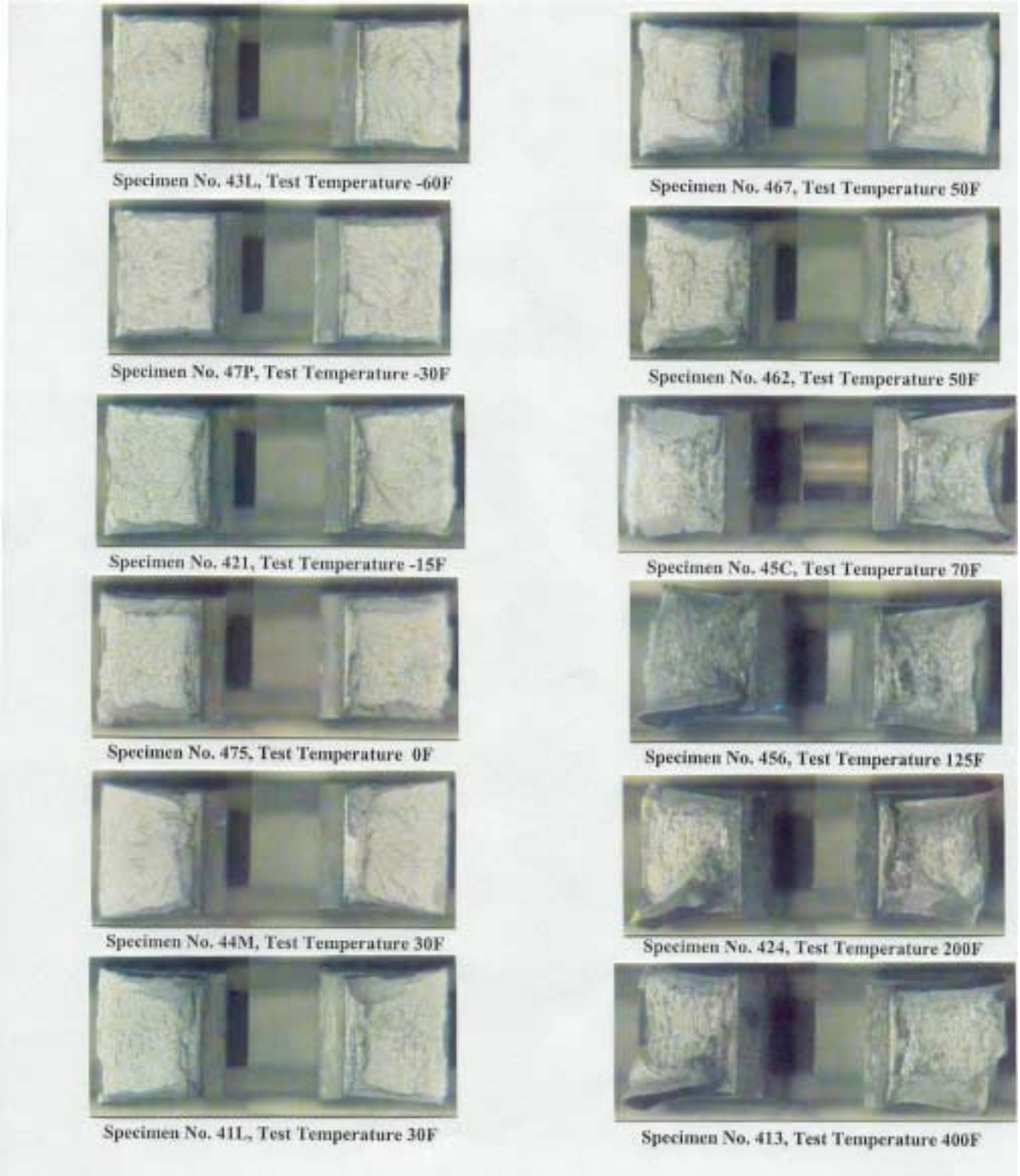
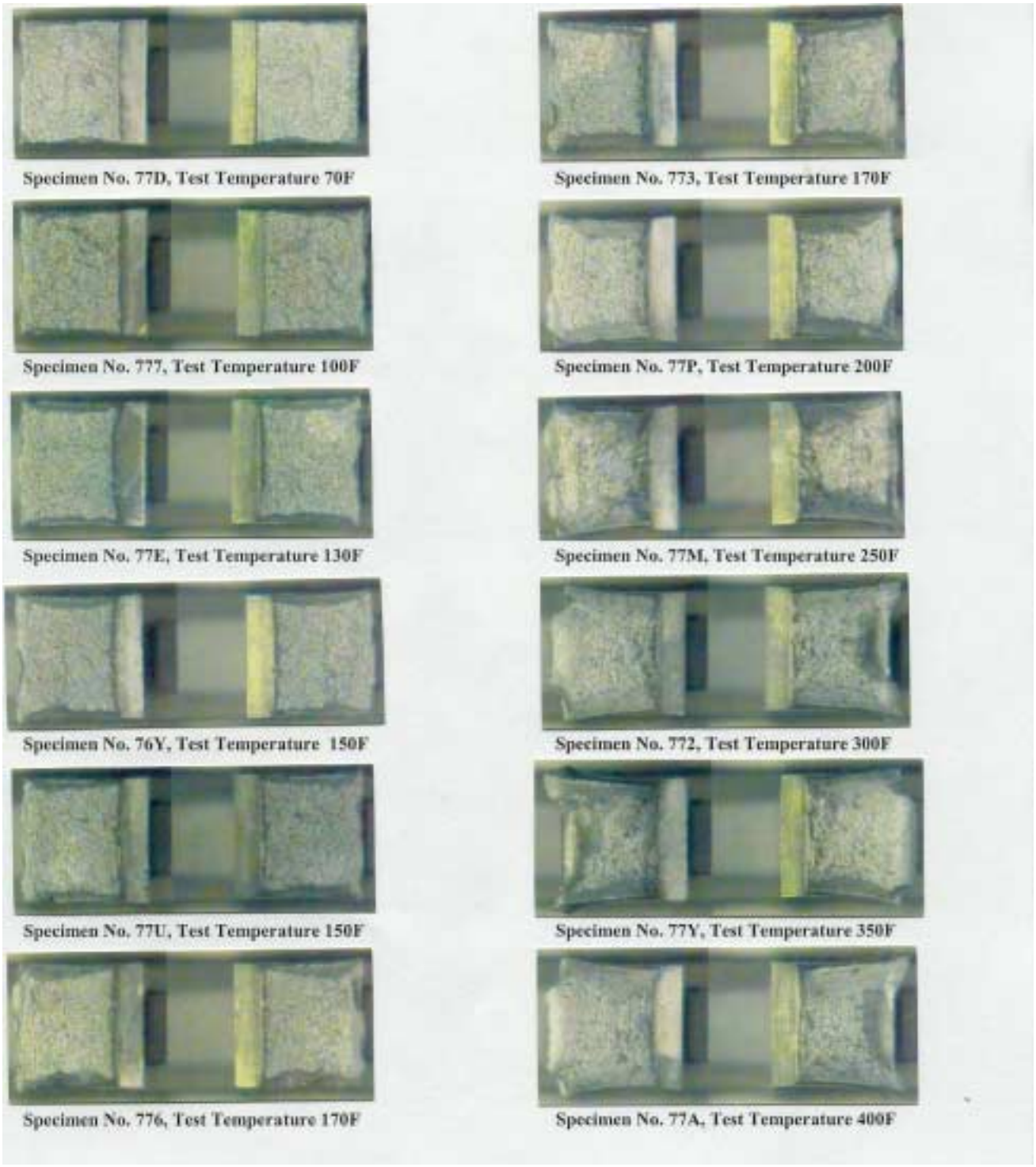


Figure 5-23. Photographs of Charpy Impact Specimen Fracture Surfaces, Standard Reference Material, HSST Plate 01



6. Neutron Fluence

6.1. Introduction

Over the last fifteen years, Framatome ANP, Inc. has developed a calculational based fluence analysis methodology,[16] that can be used to accurately predict the fast neutron fluence in the reactor vessel using surveillance capsule dosimetry or cavity dosimetry (or both) to verify the fluence predictions. This methodology was developed through a full-scale benchmark experiment that was performed at the Davis-Besse Unit 1 reactor,[16] and the methodology is described in detail in Appendix F. The results of the benchmark experiment demonstrated that the accuracy of a fluence analysis that employs the Framatome ANP methodology would be unbiased and have a precision well within the U.S Nuclear Regulatory Guide 1.190 limit of 20%.[16, 17]

The SER to BAW-2241P-A, Revision 1[16] concluded that "the proposed methodology is acceptable for referencing in licensing applications for determining the pressure vessel fluence of Westinghouse, CE, and B&W designed reactors." In addition, there were three limitations imposed on the SER. These limitations involved analysis of reactor designs not included in the BAW-2241P-A database (e.g. partial length fluence assembly designs), changes in cross sections from those reviewed by the Staff, and any other changes in methodology. None of these limitations are applicable to the analysis performed for Arkansas Nuclear One Unit 2.

The Framatome ANP methodology was used to calculate the neutron fluence exposure to the 104° (W-104) capsule of the Arkansas Nuclear One Unit 2 nuclear reactor. The methodology was also used to determine fluences on the inner surface of the reactor vessel, as well as at specified weld locations on the pressure vessel surface. The fast neutron fluence (E > 1 MeV) at each location was calculated in accordance with the requirements of U.S. Nuclear Regulatory Guide 1.190.[17]

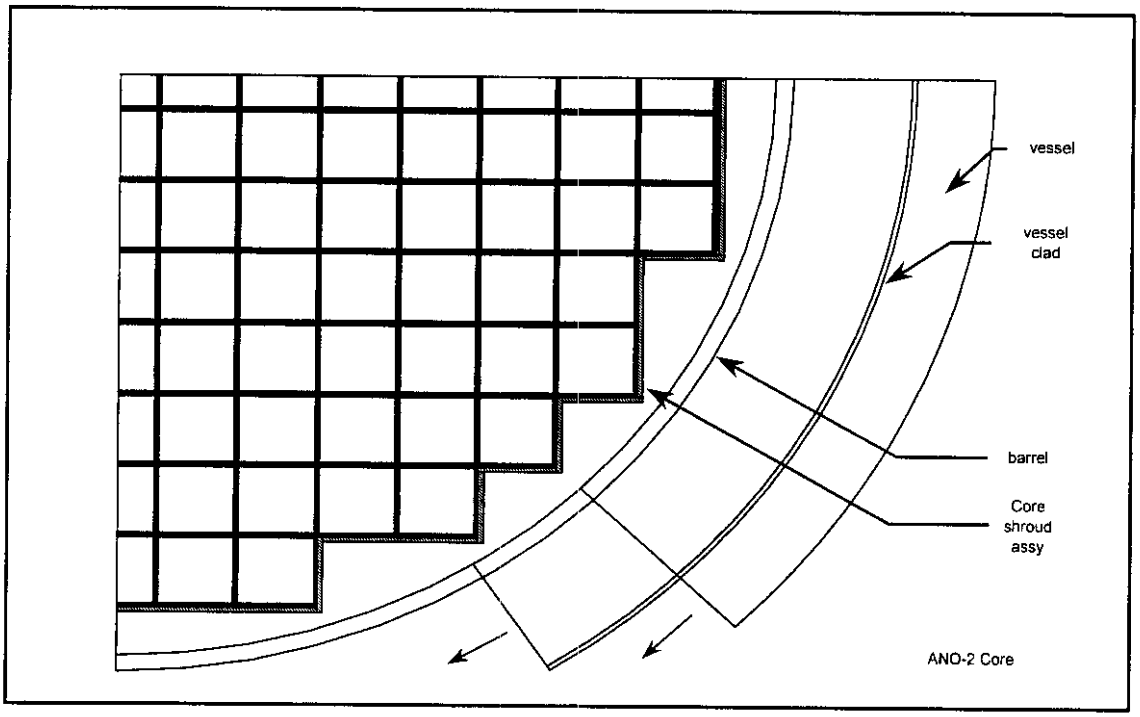
The energy-dependent flux on the capsule was used to determine the calculated activity of each dosimeter. Neutron transport calculations in two-dimensional geometry were used to obtain energy dependent flux distributions throughout the core. Reactor conditions were representative of an average over the cycle 1-9 and 10-14 irradiation periods. The cycle 10-14 results also reflect the use of lower leakage fuel management core designs. These periods were separated in order to adequately represent a water temperature reduction that occurred between cycles 9 and 10. Geometric detail was selected to explicitly represent the dosimeter holder and the reactor vessel. A more detailed discussion of the calculational procedure is given in Appendix F. The calculated activities were adjusted for known biases (photofission, short-half-life, U-235 impurity, and non-saturation), and compared to measured activities directly. It is noted that these measurements are not used in any way to determine the magnitude of the flux or the fluence. The measurements are used only to show that the calculational results are reasonable, and to show that the results for Arkansas Nuclear One Unit 2 cycles 1-14 are consistent with the Framatome ANP benchmark database of uncertainties.

6.2. Fluence Results

The Arkansas Nuclear One Unit 2 dosimeter holder is located in the downcomer region of the reactor, 14° off of the major axis. The dosimetry of the 104° capsule was located in the reactor for a total irradiation time of 5726 effective full power days (EFPDs) for cycles 1-14. The rated thermal power for the fourteen cycles was 2815 MWt.

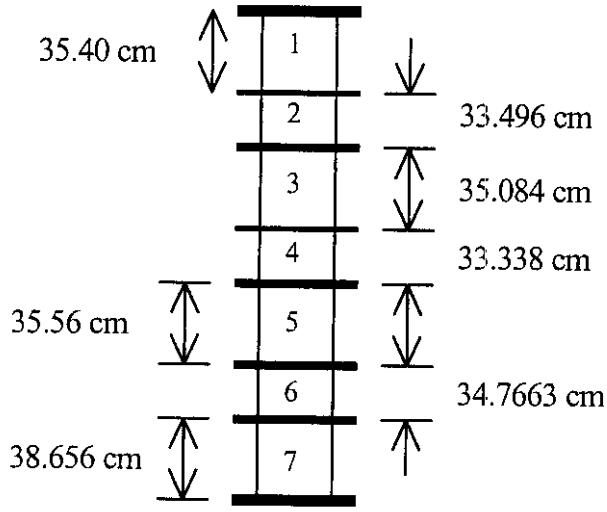
The incident fast fluence (E > 1.0 MeV) was calculated on the inner surface of the reactor vessel. The layout of the reactor vessel is shown in Figure 6-1.

Figure 6-1. Arkansas Nuclear One Unit 2 Core Overview



The capsule is located between the core barrel and pressure vessel cladding, and is mounted on the pressure vessel wall, with the centerline of the capsule at 198.025 cm. The capsule is divided into 7 compartments, 3 of which contain dosimetry. The dimensions and compartment identification numbers are shown in Figure 6-2.

Figure 6-2. Arkansas Nuclear One Unit 2 104° Dosimetry Locations



The three dosimeter compartments (1, 4, and 7) are labeled in accordance with their position in the holder, and dosimeters are labeled according to their holder relative position. For example, a Fe-54 dosimeter in compartment 1 would be designated as Top Fe-54. The dosimeters, which measure the fast fluence (E > 1.0 MeV) inside of the Arkansas Nuclear One Unit 2 reactor, are listed in Table 6-1. The axial positions of each dosimeter are also listed relative to the bottom of the DORT model.

Table 6-1. Arkansas Nuclear One Unit 2 Fast Fluence Dosimetry

Dosimeter	Can	Axial Position (cm)
Fe-54	Top	350.199
Fe-54	Middle	247.2495
Fe-54	Bottom	144.101
Cu-63	Top	350.199
Cu-63	Middle	247.2495
Cu-63	Bottom	144.101
Ni-58	Top	350.199
Ni-58	Middle	247.2495
Ni-58	Bottom	144.101
U-238	Top	350.199
U-238	Middle	247.2495
U-238	Bottom	144.101

The fluence on the center of the capsule must be determined for the center of the dosimetry capsule in order to allow for analysis of the Charpy and Tensile specimens. Doing this calculation for the entire cycle 1-14 analysis results in a maximum capsule fluence (E > 1.0 MeV) of 2.937E+19 n/cm^2.

Flux estimates were also made for several points on the inner surface of the reactor vessel and the vessel/clad interface. These estimates are of particular importance in determining the effect of neutron fluence on the properties of the vessel surface and welds. The points of interest, and their respective three-dimensional coordinates, relative to the DORT origin, are shown in Table 6-2, with the inner surface radius given first.

Table 6-2. 3D Coordinates for Arkansas Nuclear One Unit 2 Points of Interest

Point of Interest	R Position (cm)	θ Coordinate (°)	Z Coordinate (cm)
2-203 A	201.93/202.49	90	204.943 to 453
2-203 B	201.93/202.49	210	204.943 to 453
2-203 C	201.93/202.49	330	204.943 to 453
3-203 A	201.93/202.49	90	0 to 204.943
3-203 B	201.93/202.49	210	0 to 204.943
3-203 C	201.93/202.49	330	0 to 204.943
9-203	201.93/202.49	0 to 45	204.943
Intermediate Shell	201.93/202.49	0 to 45	204.943 to 453
Lower Shell	201.93/202.49	0 to 45	0 to 204.943
Inside Surface Max	201.93/202.49	0 to 45	0 to 453

The three-dimensionally synthesized fluxes at the inside surface of the vessel and vessel/clad interface are given in Table 6-3 for each point of interest over the cycle 1-9 and 10-14 irradiation periods. The azimuthal angles for the circumferential welds and the upper and lower shells shown in Table 6-3 were determined from the maximum fluxes of Table 6-3.

Table 6-3. 3D Synthesized Fluxes

Reactor Vessel Inside Surface				
Cycle 1-9 Flux Values (n/cm^2/s)				
Weld	**R (cm)**	**Theta (°)**	**Z (cm)**	**3D Flux**
2-203 A	201.93	90	204.943 to 453	4.4111E+10
2-203 B	201.93	210	204.943 to 453	3.3866E+10
2-203 C	201.93	330	204.943 to 453	3.3866E+10
3-203 A	201.93	90	0 to 204.943	4.4334E+10
3-203 B	201.93	210	0 to 204.943	3.4038E+10
3-203 C	201.93	330	0 to 204.943	3.4038E+10
9-203	201.93	12.12516	204.943	4.8222E+10
Intermediate Shell Max	201.93	12.12516	204.943	4.8222E+10
Lower Shell Max	201.93	12.12516	193.91	4.8471E+10
Inside Surface Max	201.93	12.12516	193.91	4.8471E+10

Cycle 10-14 Flux Values (n/cm^2/s)				
Weld	**R (cm)**	**Theta (°)**	**Z (cm)**	**3D Flux**
2-203 A	201.93	90	204.943 to 453	2.8885E+10
2-203 B	201.93	210	204.943 to 453	2.2492E+10
2-203 C	201.93	330	204.943 to 453	2.2492E+10
3-203 A	201.93	90	0 to 204.943	2.8985E+10
3-203 B	201.93	210	0 to 204.943	2.2570E+10
3-203 C	201.93	330	0 to 204.943	2.2570E+10
9-203	201.93	12.12516	204.943	3.0046E+10
Intermediate Shell Max	201.93	12.12516	204.943	3.0050E+10
Lower Shell Max	201.93	12.12516	195.7305	3.0153E+10
Inside Surface Max	201.93	12.12516	195.7305	3.0153E+10

Table 6-3. 3D Synthesized Fluxes (Continued)

Reactor Vessel/Clad Interface				
Cycle 1-9 Flux Values (n/cm^2/s)				
Weld	**R (cm)**	**Theta (°)**	**Z (cm)**	**3D Flux**
2-203 A	202.49	90	204.943 to 453	4.2496E+10
2-203 B	202.49	210	204.943 to 453	3.2606E+10
2-203 C	202.49	330	204.943 to 453	3.2606E+10
3-203 A	202.49	90	0 to 204.943	4.2700E+10
3-203 B	202.49	210	0 to 204.943	3.2763E+10
3-203 C	202.49	330	0 to 204.943	3.2763E+10
9-203	202.49	12.12516	204.943	4.6039E+10
Intermediate Shell Max	202.49	12.12516	204.943	4.6044E+10
Lower Shell Max	202.49	12.12516	192.0895	4.6266E+10
Inside Surface Max	202.49	12.12516	192.0895	4.6266E+10

Cycle 10-14 Flux Values (n/cm^2/s)				
Weld	**R (cm)**	**Theta (°)**	**Z (cm)**	**3D Flux**
2-203 A	202.49	90	204.943 to 453	2.7758E+10
2-203 B	202.49	210	204.943 to 453	2.1592E+10
2-203 C	202.49	330	204.943 to 453	2.1592E+10
3-203 A	202.49	90	0 to 204.943	2.7859E+10
3-203 B	202.49	210	0 to 204.943	2.1671E+10
3-203 C	202.49	330	0 to 204.943	2.1671E+10
9-203	202.49	10.89209	204.943	2.8788E+10
Intermediate Shell Max	202.49	10.89209	204.943	2.8790E+10
Lower Shell Max	202.49	10.89209	197.5515	2.8895E+10
Inside Surface Max	202.49	10.89209	197.5515	2.8895E+10

The fluence for the reactor pressure vessel can also be extrapolated to longer time periods in order to estimate total fluences on the points of interest. This extrapolation is performed by estimating the flux based on anticipated 7.5% power uprate occurring at the beginning of cycle 16. The uprated flux was determined based on uprated core designs and includes changes in the power distributions and thermal-hydraulic conditions. End of life fluences are determined by taking the cumulative fluence and then extrapolating forward. The cumulative fluence values for cycles 1-9 and 10-14 are shown in Table 6-4, along with the extrapolated EOL fluence, for both the vessel inside surface and the vessel/clad interface position. The end of life (32 and 48 EFPY) fluences are calculated using the following formula:

$$\phi t(EOL) = \phi t(EOC14) + \left(\phi_{Power\ Uprate} * (t_{EOL}(s) - t_{EOC14}(s))\right),$$

where $\phi t(EOL)$ is the fluence estimate at the end of life (32 and 48 EFPY), $\phi t(EOC14)$ is the fluence at the end of cycle 14, $\phi_{Power\ Uprate}$ is the anticipated power uprate flux through EOL, $t_{EOL}(s)$ is the total number of EFPS at the 32 EFPY end of life (1.01E9 s), and $t_{EOC14}(s)$ is the total number of EFPS accumulated through the end of cycle 14 (4.95E8 s). 48 EFPY (1.51E9 s) values are extrapolated using the same methodology.

As previously mentioned, the 32 and 48 EFPY fluences are determined from power uprate conditions. The values in Table 6-4 are documented and shown to be valid and acceptable for use from Framatome ANP document number 86-5012082-01, "ANO Unit 2 104 Capsule Fluence Analysis Report."

Table 6-4. Cumulative Fluence Projections

Vessel Inside Surface				
	Cumulative Fluence (n/cm^2)			
Flux Location	EOC Cycle 9	EOC Cycle 14	Extrapolated Fluence (n/cm^2)	
EFPY's	8.6 EFPY	15.7 EFPY	32 EFPY	48 EFPY
2-203 A	1.1971E+19	1.8535E+19	3.5682E+19	5.2817E+19
2-203 B	9.1912E+18	1.4302E+19	2.7540E+19	4.0824E+19
2-203 C	9.1912E+18	1.4302E+19	2.7540E+19	4.0824E+19
3-203 A	1.2032E+19	1.8618E+19	3.5823E+19	5.3016E+19
3-203 B	9.2377E+18	1.4366E+19	2.7648E+19	4.0978E+19
3-203 C	9.2377E+18	1.4366E+19	2.7648E+19	4.0978E+19
9-203	1.3087E+19	1.9914E+19	3.7756E+19	5.5578E+19
Intermediate Shell Max	1.3087E+19	1.9938E+19	3.7760E+19	5.5585E+19
Lower Shell Max	1.3155E+19	2.0006E+19	3.7910E+19	5.5795E+19
Inside Surface Max	1.3155E+19	2.0006E+19	3.7910E+19	5.5795E+19

Vessel/Clad Interface				
	Cumulative Fluence (n/cm^2)			
Flux Location	EOC Cycle 9	EOC Cycle 14	Extrapolated Fluence (n/cm^2)	
EFPY's	8.6 EFPY	15.7 EFPY	32 EFPY	48 EFPY
2-203 A	1.1533E+19	1.7840E+19	3.4318E+19	5.0784E+19
2-203 B	8.8492E+18	1.3755E+19	2.6462E+19	3.9216E+19
2-203 C	8.8492E+18	1.3755E+19	2.6462E+19	3.9216E+19
3-203 A	1.1589E+19	1.7919E+19	3.4455E+19	5.0981E+19
3-203 B	8.8918E+18	1.3816E+19	2.6568E+19	3.9367E+19
3-203 C	8.8918E+18	1.3816E+19	2.6568E+19	3.9367E+19
9-203	1.2495E+19	1.9036E+19	3.6126E+19	5.3199E+19
Intermediate Shell Max	1.2496E+19	1.9038E+19	3.6130E+19	5.3204E+19
Lower Shell Max	1.2556E+19	1.9122E+19	3.6274E+19	5.3410E+19
Inside Surface Max	1.2556E+19	1.9122E+19	3.6274E+19	5.3410E+19

6.3. Dosimetry Activity

The ratio of the specified activities to the measured specific activities (C/M) is presented in Table 6-5 for cycles 1-14. In this table, the target averaged C/M represents the average of all the individual target dosimeters and the overall average is the average C/M for the entire capsule.

Table 6-5. C/M Ratios

Cycles 1-14			
Dosimeter	**Calculated**	**Measured**	**Capsule Average**
Top Fe	1245.278	1258.000	1.00149
Middle Fe	1156.886	1168.000	
Bottom Fe	1196.933	1214.000	
Top Ni	1481.328	1413.000	
Middle Ni	1378.868	1322.000	
Bottom Ni	1425.947	1352.000	
Top Cu	11.274	11.050	
Middle Cu	10.510	10.520	
Bottom Cu	10.952	10.660	
Top Sh U238	28.525	28.520	
Middle Sh U238	27.716	29.790	
Bottom Sh U238	28.739	30.960	

7. Discussion of Capsule Results

7.1. Copper and Nickel Chemical Composition Data

To date, several chemical analyses have been performed on the Arkansas Nuclear One Unit 2 RVSP base metal plate and weld metal. These analyses have been performed on the unirradiated surveillance materials, and broken Charpy specimens tested as part of the Arkansas Nuclear One Unit 2, W-97 capsule analysis. The mean copper and nickel contents for the Arkansas Nuclear One Unit 2 RVSP base metal plate and weld metal represent the best-estimate chemical contents for these surveillance materials. The copper and nickel chemical content means are presented in Table 7-1.

7.2. Unirradiated Material Property Data

The base metal and weld metal were selected for inclusion in the Arkansas Nuclear One Unit 2 surveillance program in accordance with the regulations in effect at the time the program was designed. The applicable selection criterion was based on the unirradiated properties of the Arkansas Nuclear One Unit 2 reactor vessel beltline region materials only.

The unirradiated mechanical properties for the Arkansas Nuclear One Unit 2 RVSP materials are summarized in Appendices C and D of this report. The original unirradiated Charpy impact data were evaluated based on hand-fit Charpy curves generated using engineering judgement[9]. These data were re-plotted and re-evaluated herein using a hyperbolic tangent curve-fitting program in order to be consistent with the W-104 capsule Charpy curves. Appendix E contains a comparison of the Charpy V-notch shift results for each surveillance material: original-fit versus current hyperbolic tangent curve-fit for the unirradiated and W-97 capsule data.

7.3. Irradiated Property Data

In addition to the W-104 capsule mechanical test data, surveillance data are also available from the Arkansas Nuclear One Unit 2 RVSP W-97 capsule. Battelle, Columbus Laboratories performed the testing and evaluation for the W-97 capsule.[2]

7.3.1. Tensile Properties

Table 7-2 compares the irradiated and unirradiated tensile properties. Review of the surveillance tensile test data indicates that the ultimate strength and yield strength changes in the base metal plate

as a result of irradiation and the corresponding changes in ductility are within the ranges observed for similar irradiated materials. The changes in tensile properties for the surveillance weld metal, as a result of irradiation, are also within the observed ranges for similar irradiated materials. The general behavior of the tensile properties as a function of neutron irradiation is an increase in both ultimate and yield strength and a decrease in ductility as measured by both total elongation and reduction in area, except there was an increase in reduction of area for the W-97 weld metal.

The HAZ tensile properties are similar to the weld properties (See Table 5-1). In examining the tested tensile specimens (See Figure 5-15), it can be seen that the failure did not occur in the center of the reduced section. The HAZ is located in the center of the tensile specimen.[1] Since the base metal yield and ultimate strengths are higher than the weld, it is concluded that the HAZ tensile specimens probably failed in the weld metal. Therefore, the HAZ tensile specimen tests likely resulted in measuring weld metal properties.

7.3.2. Impact Properties

Tables 7-3 and 7-4 compare the measured changes in irradiated Charpy V-notch impact properties from the W-104 capsule with the predicted changes in accordance with Regulatory Guide 1.99, Revision 2.[18]

The measured 30 ft-lb transition temperature shift for the surveillance base metal plate is slightly less than the shift predicted using Regulatory Guide 1.99, Revision 2, Position 1.1. The measured 30 ft-lb transition temperature shift for the weld is less (about 1 standard deviation less) than the shift predicted using Regulatory Guide 1.99, Revision 2, Position 1.1. The measured 30 ft-lb transition temperature shift for the heat affected zone material is greater than (about 3 standard deviations greater) the shift predicted using Regulatory Guide 1.99, Revision 2, Position 1.1. The 30 ft-lb transition temperature for the HAZ is about the same as the weld, but the shift is considerably higher due to the low unirradiated 30 ft-lb transition temperature. The measured 30 ft-lb transition temperature shift for the standard reference material is considerably less (about 2 standard deviations less) than the shift predicted using Regulatory Guide 1.99, Revision 2, Position 1.1 (see Table 7-3).

The measured upper-shelf energies for the Arkansas Nuclear One Unit 2 the W-104 capsule surveillance materials do not fall below the required 50 ft-lb limit. The measured percent decrease in C_vUSE for the surveillance base metal plate, weld metal, HAZ, and SRM are within reasonable agreement with the values predicted using Regulatory Guide 1.99, Revision 2 (see Table 7-4).

The radiation-induced changes in toughness of the Arkansas Nuclear One Unit 2 surveillance materials are summarized in Table 7-5. The original irradiated Charpy impact data for the W-97

capsule were evaluated based on hand-fit Charpy curves generated using engineering judgement. These data were re-plotted and re-evaluated herein using a hyperbolic tangent curve-fitting program to be consistent with the W-104 capsule Charpy curves. The results of the re-evaluation are presented in Appendix D. In addition, Appendix E contains a comparison of the Charpy V-notch shift results for each surveillance material, hand-fit versus hyperbolic tangent curve-fit.

Table 7-1. Copper and Nickel Mean Chemical Composition Data for Arkansas Nuclear One Unit 2 Reactor Vessel Surveillance Materials

Surveillance Material	Data Source	Copper (% Weight)	Nickel (% Weight)
Weld Metal, C-8009-1/C-8009-2 Heat 10120	Baseline[9]	0.04	0.08
	Charpy 315 from Capsule W-97[2]	0.046	0.090
	Charpy 344 from Capsule W-97[2]	0.047	0.088
	Charpy 37J from Capsule W-97[2]	0.045	0.075
	Charpy 356 from Capsule W-97[2]	0.044	0.082
	Charpy 33J from Capsule W-97[2]	0.045	0.081
	Mean	**0.045**	**0.083**
Base Metal Plate C-8009-3, Heat C8182-2	Baseline[9]	0.08	0.60
	Charpy 13T from Capsule W-97[2]	0.078	0.559
	Charpy 14M from Capsule W-97[2]	0.077	0.548
	Charpy 21U from Capsule W-97[2]	0.075	0.545
	Charpy 264 from Capsule W-97[2]	0.078	0.555
	Charpy 257 from Capsule W-97[2]	0.078	0.559
	Mean	**0.078**	**0.561**

Table 7-2. Summary of Arkansas Nuclear One Unit 2 Reactor Vessel Surveillance Capsules Tensile Test Results

Material	Fluence, 10^{19} n/cm^2	Test Temp., F	Strength, ksi				Ductility, %			
			Yield	%[a]	Ultimate	%[a]	Total Elong.	%[a]	Reduction of Area	%[a]
Base Metal Plate C-8009-3, Heat C8182-2 (Transverse Orientation)	0.00	71	69.7[b]	---	90.8[b]	---	25.0[b]	---	68.7[b]	---
		250	65.5[b]	---	84.5[b]	---	22.0[b]	---	69.2[b]	---
		550	57.1[b]	---	83.8[b]	---	22.7[b]	---	64.5[b]	---
	0.333	77	73.0	4.7	95.9	5.6	21.7	-13.2	67.2	-2.1
		250	71.7	9.5	88.0	4.1	20.8	-5.5	67.4	-2.6
		550	66.2	16.0	89.0	6.2	17.8	-21.5	66.4	2.9
	2.937	70	78.0	11.9	100	10.1	22.5	-10.1	63	-8.3
		250	74.1	13.1	93.3	10.4	20.7	-6.1	64.7	-6.5
		550	68.3	19.7	94.1	12.2	19.8	-12.9	61.3	-5.0
Weld Metal, Heat 10120, C-8009-1/C-8009-2	0.00	71	75.6[b]	---	88.4[b]	---	24.0[b]	---	72.3[b]	---
		250	71.6[b]	---	82.2[b]	---	22.0[b]	---	72.1[b]	---
		550	65.9[b]	---	85.4[b]	---	22.0[b]	---	69.0[b]	---
	0.335	77	75.6	0.0	88.4	0.1	22.1	-7.9	79.3	9.6
		250	71.3	-0.3	81.7	-0.6	19.2	-12.7	73.6	2.1
		550	69.0	4.7	85.8	0.4	20.2	-8.2	73.8	7.0
	2.937	70	78.0	3.2	92.9	5.1	22.5	-6.3	71.5	-1.2
		250	72.9	1.9	85.0	3.4	20.2	-8.1	69.5	-3.6
		550	68.8	4.4	87.2	2.1	21.6	-2.0	66.1	-4.2

(a) Change relative to unirradiated material property.

(b) Mean value of available test data.

Table 7-3. Measured vs. Predicted 30 ft-lb Transition Temperature Changes for Arkansas Nuclear One Unit 2 W-104 Capsule Surveillance Materials – 2.937 x 10^{19} n/cm^2

Material	Measured 30 ft-lb Transition Temperature, F			30 ft-lb Transition Temperature Shift Predicted in Accordance With Regulatory Guide 1.99, Revision 2				
	Unirradiated	Irradiated	Difference	Chemistry Factor	ΔRT_{NDT}[(a)]	Margin (σ)	$\Delta RT_{NDT} - \sigma$	$\Delta RT_{NDT} + \sigma$
Base Metal Plate C-8009-3, Heat C8182-2 (Transverse Orientation)	15	68	53	49.6[(b)]	63.8	17	46.8	80.8
Weld Metal, Heat 10120, C-8009-1/C-8009-2	-5	12	17	33.7[(b)]	43.3	28	15.3	71.3
Heat-Affected-Zone Material	-130	-17	113	49.6[(b)]	63.8	17	46.8	80.8
Standard Reference Material, HSST Plate 01	32	164	132	131.7[(c)]	169.4	17	152.4	186.4

(a) ΔRT_{NDT} = Chemistry Factor * fluence factor (using the W-104 Capsule fluence).

(b) Chemistry factor based on mean copper and nickel contents as shown in Table 7-1.

(c) Chemistry factor based on copper and nickel contents as shown in NUREG/CR-6551.[20]

Table 7-4. Measured vs. Predicted Upper-Shelf Energy Decreases for Arkansas Nuclear One Unit 2 W-104 Capsule Surveillance Materials – 2.937 x 10^{19} n/cm^2

Material	Measured Upper-Shelf Energy, ft-lb			% Decrease Predicted In Accordance With Regulatory Guide 1.99, Rev. 2 Figure 2[(a)]
	Unirradiated	Irradiated	% Decrease	
Base Metal Plate C-8009-3, Heat C8182-2 (Transverse Orientation)	133.8	91.8	31.4	21.6[(b)]
Weld Metal, Heat 10120, C-8009-1/C-8009-2	154.0	124.5	19.2	23.8[(b)]
Heat-Affected-Zone Material	161.1	129.5	19.6	21.6[(b)]
Standard Reference Material, HSST Plate 01	145.5	87.3	40.0	33.9[(c)]

(a) Calculated using equation reported in NUREG/CR-5799.[21]

(b) Based on mean copper content as shown in Table 7-1.

(c) Chemistry factor based on copper content as shown in NUREG/CR-6551.[20]

Table 7-5. Summary of Arkansas Nuclear One Unit 2 Reactor Vessel Surveillance Capsules Charpy Impact Test Results

Material	Capsule	Fluence, 10^{19} n/cm^2	Measured Transition Temperature		Measured Upper-Shelf	
			ΔCv30, F	ΔCv50, F	Energy, ft-lb	% Decrease
Base Metal Plate C-8009-3, Heat C8182-2 (Transverse Orientation)	Baseline	---	---	---	134	---
	97	0.333	33	43	120	10.5
	104	2.937	53	68	92	31.4
Weld Metal, C-8009-1/C-8009-2	Baseline	---	---	---	154	---
	97	0.334	14	15	147	4.6
	104	2.937	17	26	125	19.2
Heat-Affected-Zone Material	Baseline	---	---	---	161	---
	97	0.349	51	35	138	14.1
	104	2.937	113	103	130	19.6
Standard Reference Material, HSST Plate 01	Baseline	---	---	---	146	---
	104	2.937	132	150	87	40.0

8. Summary of Results

The analysis of the reactor vessel material contained in the second surveillance capsule, Capsule W-104, removed for evaluation as part of the Arkansas Nuclear One Unit 2 Reactor Vessel Surveillance Program, led to the following conclusions:

1. The capsule received an average fast neutron fluence of 2.937×10^{19} n/cm^2 (E > 1.0 MeV).

2. Based on an extrapolated flux incorporating power uprate conditions, the projected end-of-life (32 EFPY) peak fast fluence of the Arkansas Nuclear One Unit 2 reactor vessel beltline region clad surface is 3.791×10^{19} n/cm^2 (E > 1.0 MeV).

3. The 30 ft-lb transition temperature for the Base Metal Plate C-8009-3, Heat C8182-2, in the transverse orientation, increased 53°F after irradiation to 2.937×10^{19} n/cm^2 (E > 1.0 MeV). In addition, the C_vUSE for this material decreased 31%.

4. The 30 ft-lb transition temperature for the Weld Metal, C-8009-1/C-8009-2, Heat 10120 increased 17°F after irradiation to 2.937×10^{19} n/cm^2 (E > 1.0 MeV). In addition, the C_vUSE for this material decreased 19%.

5. The 30 ft-lb transition temperature for the heat affected zone increased 113°F after irradiation to 2.937×10^{19} n/cm^2 (E > 1.0 MeV). In addition, the C_vUSE for this material decreased 20%.

6. The 30 ft-lb transition temperature for the Standard Reference Material, HSST Plate 01, increased 132°F after irradiation to 2.937×10^{19} n/cm^2 (E > 1.0 MeV). In addition, the C_vUSE for this material decreased 40%.

7. The measured upper-shelf energies for the Arkansas Nuclear One Unit 2 Capsule W-104 surveillance materials do not fall below the required 50 ft-lbs limit after the irradiation to 2.937×10^{19} n/cm^2 (E > 1.0 MeV).

9. Certification

The specimens obtained from the Entergy Operations, Inc. Arkansas Nuclear One Unit 2 reactor vessel surveillance capsule (Capsule W-104) were tested and evaluated using accepted techniques and established standard methods and procedures in accordance with the requirements of 10 CFR 50, Appendices G and H.

J. B. Hall 9-28-01

J. B. Hall (Materials Analysis) Date
Materials & Structural Analysis Unit

J. W. Newman 9-28-01

J. W. Newman, Jr. (Fluence Analysis) Date
Performance Analysis Unit

This report has been reviewed for technical content and accuracy.

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Verification of independent review.

D. R. Cofflin 9/28/01

D. R. Cofflin, Manager Date
Materials & Structural Analysis Unit

This report is approved for release.

D. L. Howell 9/28/01

D. L. Howell Date
Program Manager

10. References

1. A. D. Emery, *"Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Arkansas Nuclear One Unit 2 Reactor Vessel Materials,"* CEN-15(A), Combustion Engineering, Inc., Windsor, Connecticut, May 1975.

2. L. M. Lowry, et al., *"Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the Arkansas Nuclear One Unit 2 Generating Plant,"* Battelle Columbus Laboratories, Columbus, Ohio, May 1984.

3. ASTM Standard E 185-73, *"Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

4. Code of Federal Regulation, Title 10, Part 50, *"Domestic Licensing of Production and Utilization Facilities,"* Appendix G, Fracture Toughness Requirements.

5. Code of Federal Regulation, Title 10, Part 50, *"Domestic Licensing of Production and Utilization Facilities,"* Appendix H, Reactor Vessel Material Surveillance Program Requirements.

6. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, *"Nuclear Power Plant Components,"* Appendix G, Protection Against Nonductile Failure, 1989 Edition.

7. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components,"* Appendix G, Fracture Toughness Criteria for Protection Against Failure, 1989 Edition.

8. ASTM Standard E 208-81, *"Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

9. A. Ragl, *"Arkansas Nuclear One Unit 2 Evaluation of Baseline Specimens Reactor Vessel Materials Irradiation Surveillance Program,"* TR-MCD-002, Combustion Engineering, Inc., Windsor, Connecticut, February 1976.

* - Available from Framatome ANP, Lynchburg, Virginia.

10. S. M. Jensen, *"Evaluation of ANO Unit 2 Reactor Vessel Surveillance Capsule W-104,"* 00:1150-016-00-44:01 (Framatome ANP Document No. 31-1182299-00), BWX Technologies, Inc., Lynchburg, Virginia, May 2001.*

11. ASTM Standard E 21-92, *"Standard Test Methods for Elevated Temperature Tension Tests of Metallic Materials,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

12. ASTM Standard E 23-91, *"Standard Test Methods for Notched Bar Impact Testing of Metallic Materials,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

13. W. Oldfield, "*Curve Fitting Impact Test Data: A Statistical Procedure,*" ASTM Standardization News, November 1975.

14. W. L. Server and D. R. Ireland, "*Nonstandard Test Techniques Utilizing the Instrumented Charpy and Izod Tests*," Instrumented Impact Testing, American Society for Testing and Materials, ASTM STP 563, pp. 74-91, 1974.

15. ASTM Standard E 185-82, *"Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

16. Worsham, J.R., et al., *"Fluence and Uncertainty Methodologies,"* BAW-2241P-A, Revision 1, Framatome ANP, Lynchburg, Virginia, April 1999.

17. U.S. Nuclear Regulatory Commission, "*Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,*" Regulatory Guide 1.190, March 2001.

18. U.S. Nuclear Regulatory Commission, *"Radiation Embrittlement of Reactor Vessel Materials,"* Regulatory Guide 1.99, Revision 2, May 1988.

19. D. C. Mims, *"Response to Generic Letter 92-01, Revision 1, Supplement 1 Reactor Vessel Structural Integrity for ANO-2 TAC Nos. M92642 and M77399,"* Letter No. 2CAN069709 to the USNRC from Entergy Operations, Inc, Russellville, Arkansas, June 1997.

20. E. D. Eason, J. E. Wright, and G. R. Odette, *"Improved Embrittlement Correlations for Reactor Vessel Steels,"* NUREG/CR-6551, November 1998.

21. R. D. Cheverton, T. L. Dickson, J. G. Merkle, and R. K. Nanstad, "*Review of Reactor Pressure Vessel Evaluation Report for Yankee Row Nuclear Power Station,*" NUREG/CR-5799, March 1992.

* - Available from Framatome ANP, Lynchburg, Virginia.

APPENDIX A

Reactor Vessel Surveillance Program
Background Data and Information

A.1. Capsule Identification

The capsules, used in the Arkansas Nuclear One Unit 2 reactor vessel surveillance program, are identified in Table A-1 by identification and location. The capsule locations within the Arkansas Nuclear One Unit 2 reactor vessel are shown in Figure 3-1.

A.2. Arkansas Nuclear One Unit 2 Reactor Pressure Vessel

The Arkansas Nuclear One Unit 2 reactor vessel beltline region consists of two shells (intermediate and lower), containing three base metal plates each with associated longitudinal welds and one circumferential weld seam. Table A-2 presents a description of the Arkansas Nuclear One Unit 2 reactor vessel beltline materials including their copper and nickel chemical contents and their unirradiated RT_{NDT} values.[17]

A.3. Surveillance Material Selection Data

The plate material that was selected for the surveillance program was selected based on an estimate of the highest end of life RT_{NDT}, in accordance with ASTM Standard E 185-73. The Arkansas Nuclear One Unit 2 RVSP capsules include the predicted limiting reactor vessel beltline plate C-8009-3, heat no. C8182-2 at the time of the development of the surveillance program.[1] Through analysis of additional sources of chemistry data, it has been determined that the limiting plate is now C-8010-1, heat no. C8161-2.[19] The surveillance weld used in the Arkansas Nuclear One Unit 2 RVSP was fabricated using the wire heat 10120 which is the same wire heat used for the intermediate and lower shell longitudinal welds in the Arkansas Nuclear One Unit 2 reactor vessel.[19]

Table A-1. Arkansas Nuclear One Unit 2 Surveillance Capsule Identifications and Original Locations

Capsule Identification	Location Code	Capsule Location[(a)]
W-83	1	83°
W-97	2	97°
W-104	3	104°
W-263	4	263°
W-277	5	277°
W-284	6	284°

(a) Reference irradiation capsule locations as shown in Figure 3-1.

Table A-2. Description of the Arkansas Nuclear One Unit 2 Reactor Vessel Beltline Region Materials

Beltline Region Location	Material Identification	Material Type	Material Heat No.	Chemical Composition		Unirradiated RT_{NDT}, F
				Cu, wt%	Ni, wt%	
Inter. Shell Long. Welds	2-203	Subarc Weld	10120	0.046	0.082	-56
Lower Shell Long. Welds	3-203	Subarc Weld	10120	0.046	0.082	-56
Lower/Inter. Shell Girth Weld	9-203	Subarc Weld	83650	0.045	0.087	-10
Intermediate Shell	C-8009-1	SA-533B Cl.1	C8161-3	0.098	0.605	-26
Intermediate Shell	C-8009-2	SA-533B Cl.1	C8161-1	0.085	0.600	0
Intermediate Shell	C-8009-3	SA-533B Cl.1	C8182-2	0.096	0.580	0
Lower Shell	C-8010-1	SA-533B Cl.1	C8161-2	0.085	0.585	12
Lower Shell	C-8010-2	SA-533B Cl.1	B2545-1	0.083	0.668	-28
Lower Shell	C-8010-3	SA-533B Cl.1	B2545-2	0.080	0.653	-30

APPENDIX B

Instrumented Charpy V-Notch Specimen Test Results
Load-Time Traces

Figure B-1. Load-Time Trace for Charpy V-Notch Impact Specimen 25L

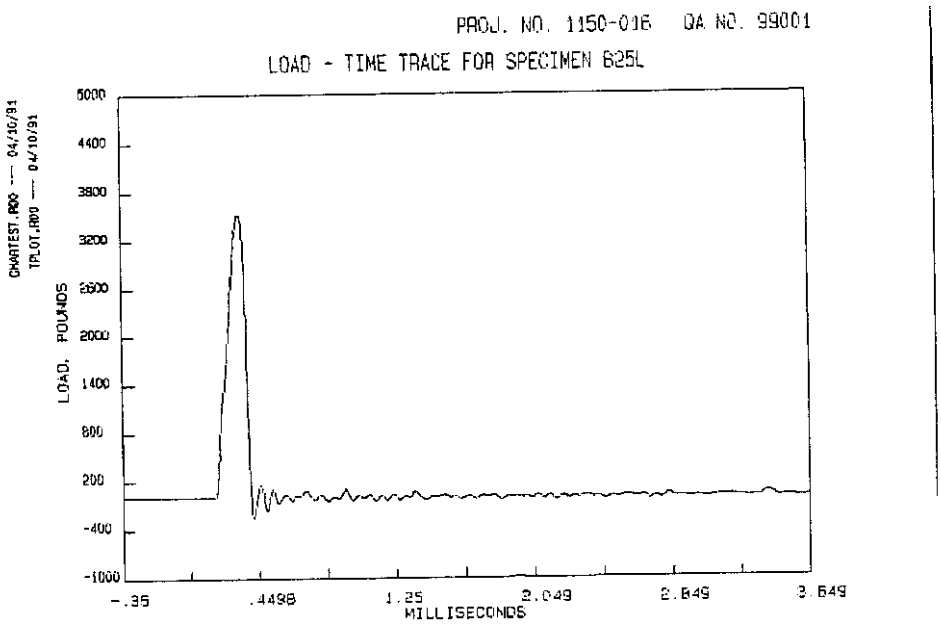


Figure B-2. Load-Time Trace for Charpy V-Notch Impact Specimen 227

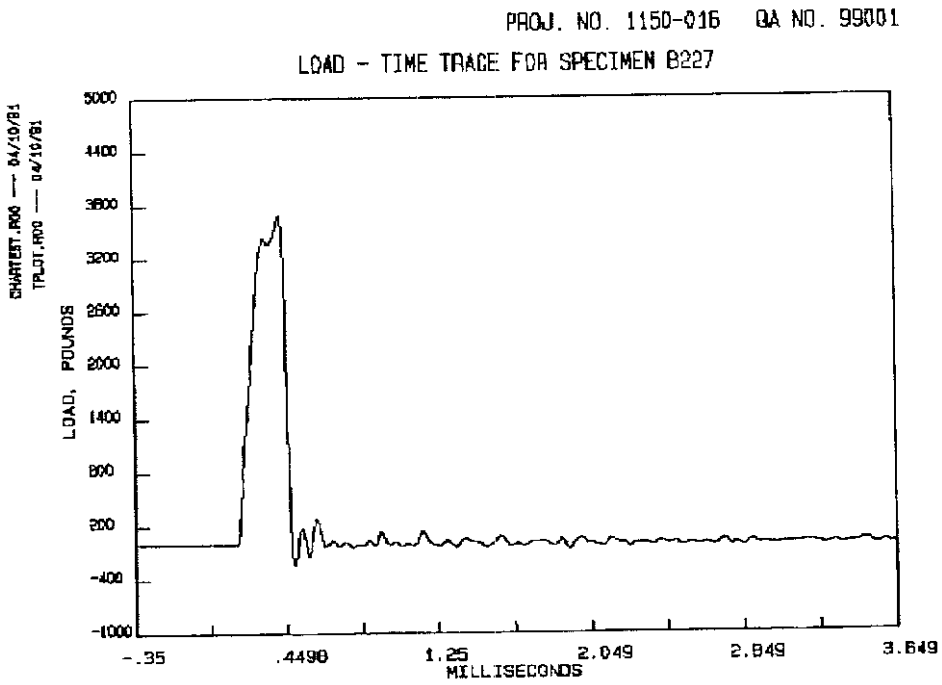


Figure B-3. Load-Time Trace for Charpy V-Notch Impact Specimen 246

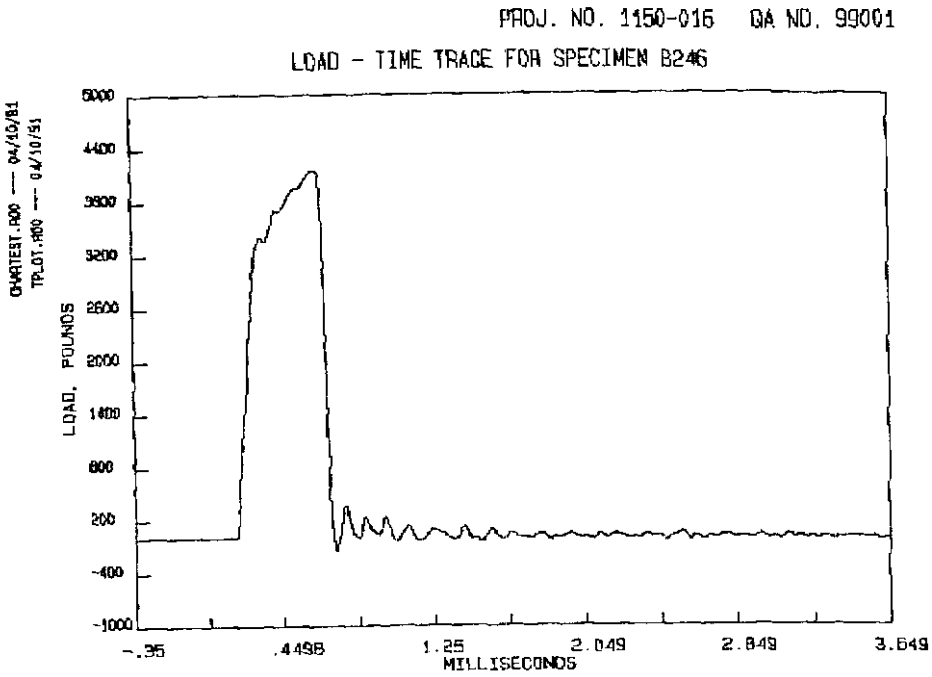


Figure B-4. Load-Time Trace for Charpy V-Notch Impact Specimen 24L

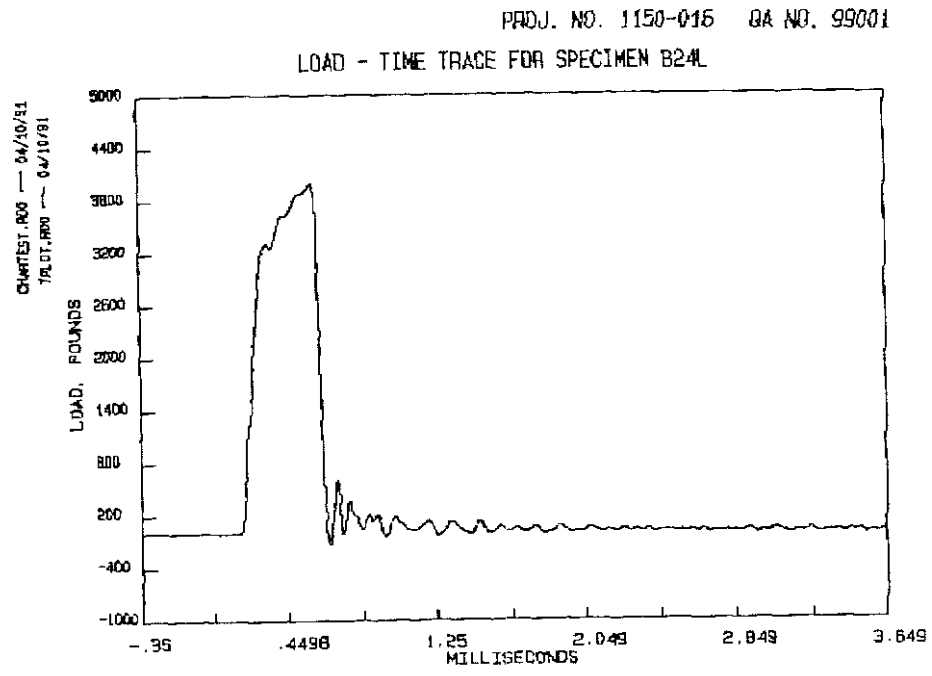


Figure B-5. Load-Time Trace for Charpy V-Notch Impact Specimen 23C

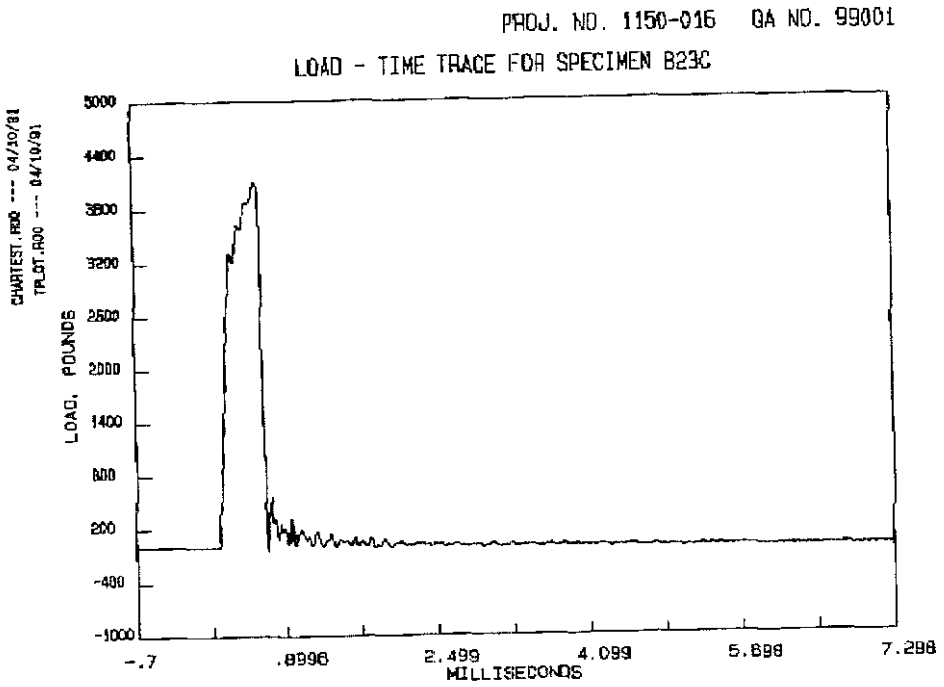


Figure B-6. Load-Time Trace for Charpy V-Notch Impact Specimen 251

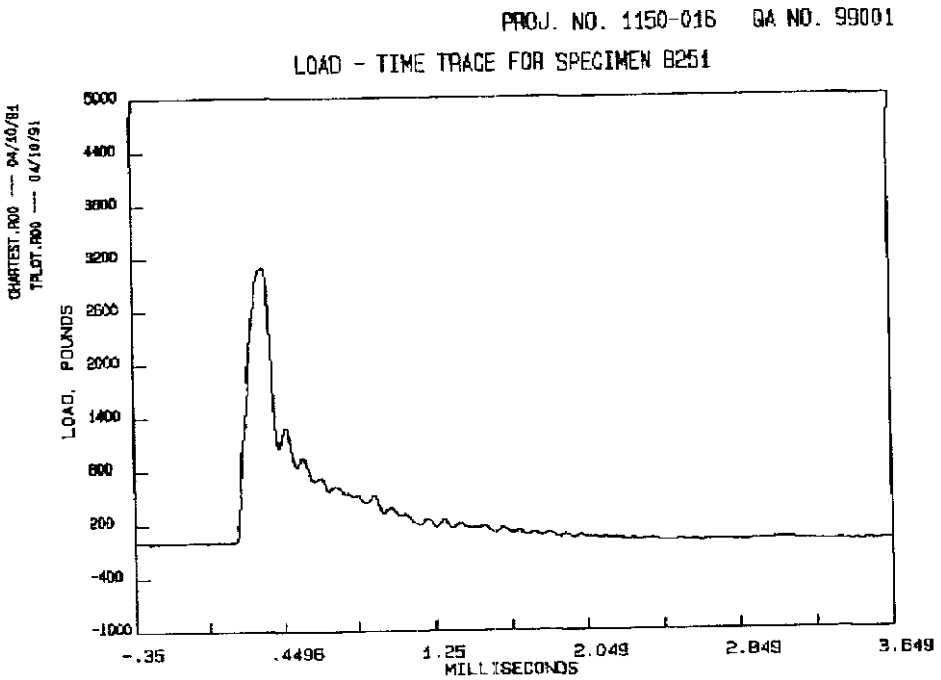


Figure B-7. Load-Time Trace for Charpy V-Notch Impact Specimen 22A

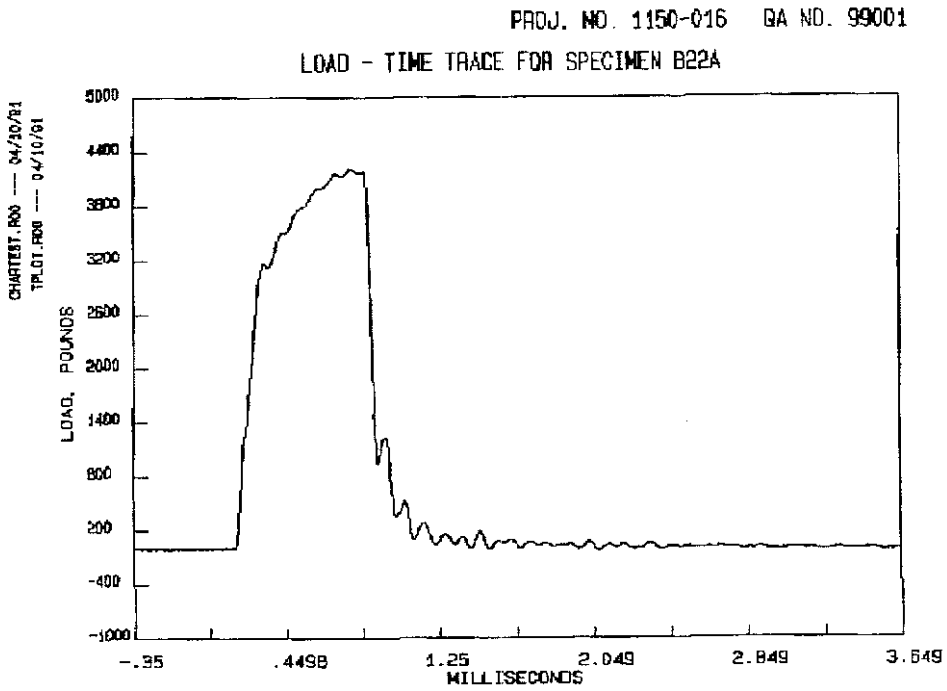


Figure B-8. Load-Time Trace for Charpy V-Notch Impact Specimen 24K

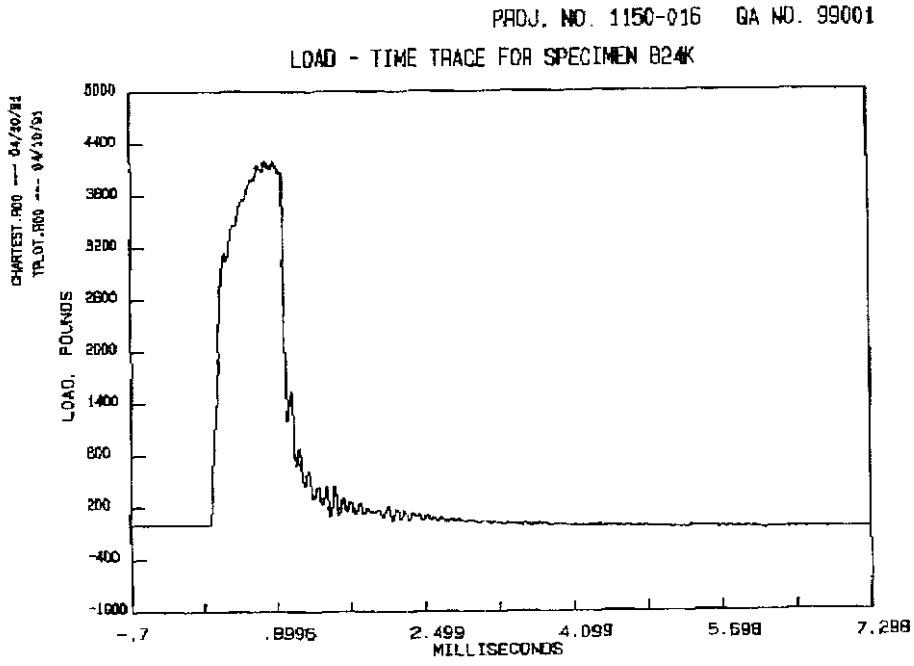


Figure B-9. Load-Time Trace for Charpy V-Notch Impact Specimen 236

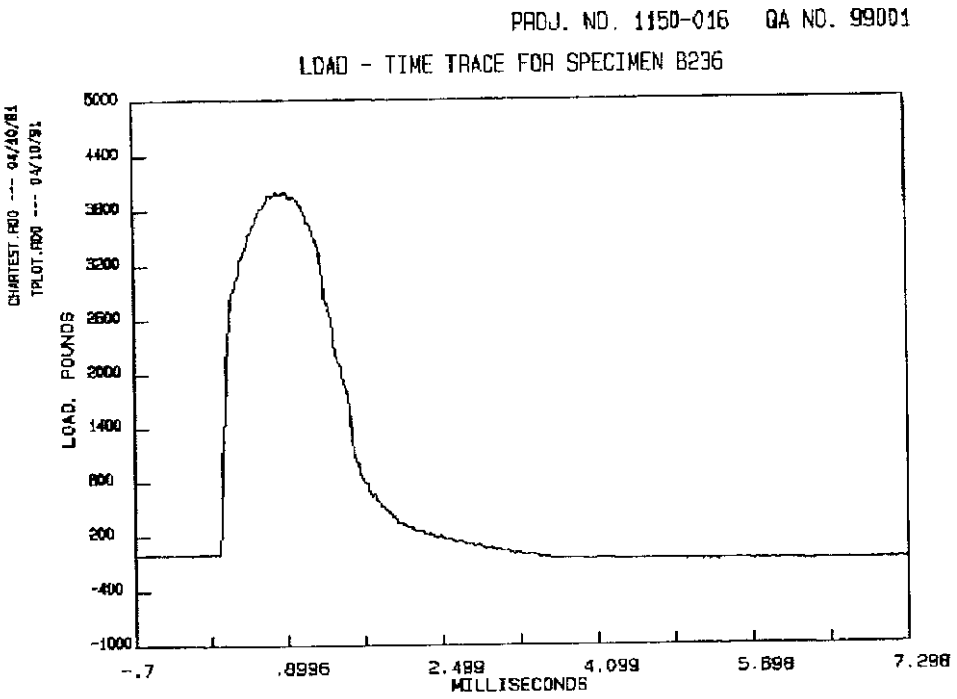


Figure B-10. Load-Time Trace for Charpy V-Notch Impact Specimen 23B

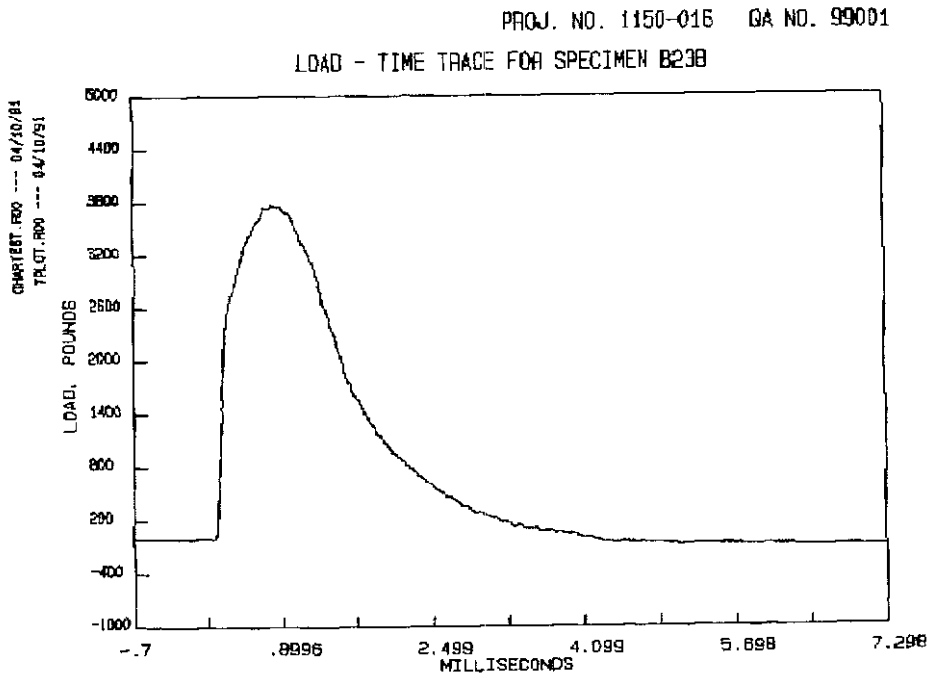


Figure B-11. Load-Time Trace for Charpy V-Notch Impact Specimen 252

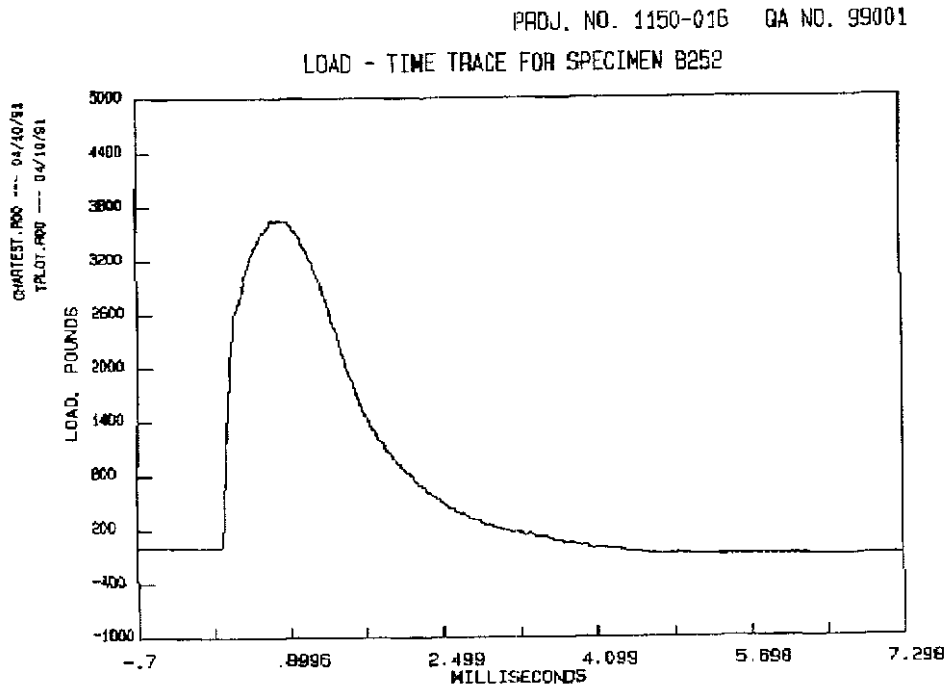


Figure B-12. Load-Time Trace for Charpy V-Notch Impact Specimen 24Y

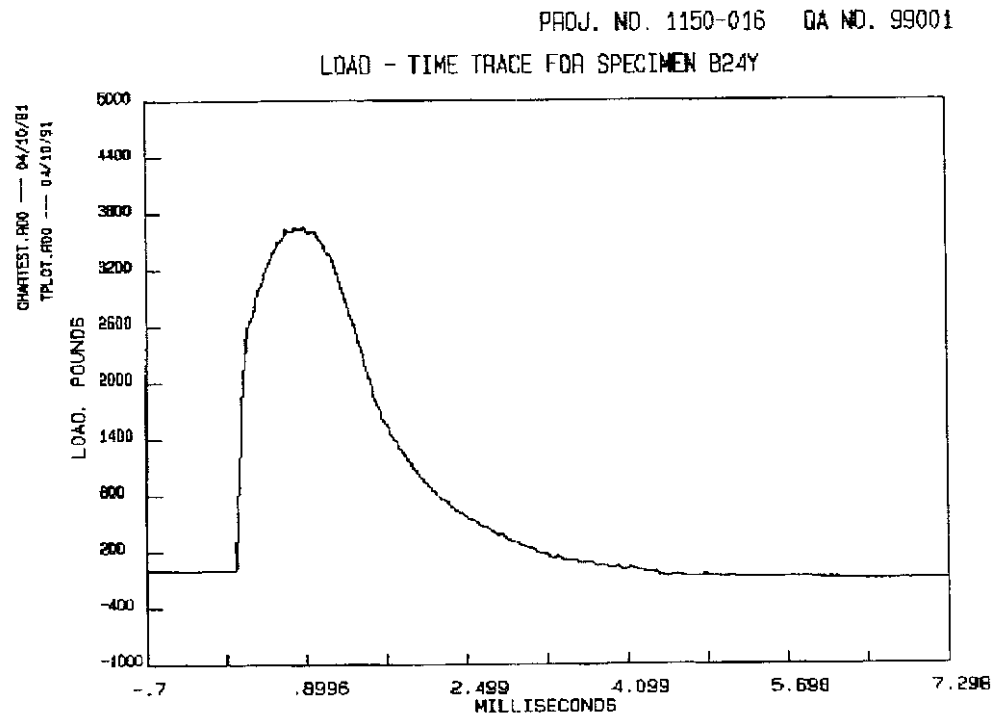


Figure B-13. Load-Time Trace for Charpy V-Notch Impact Specimen 342

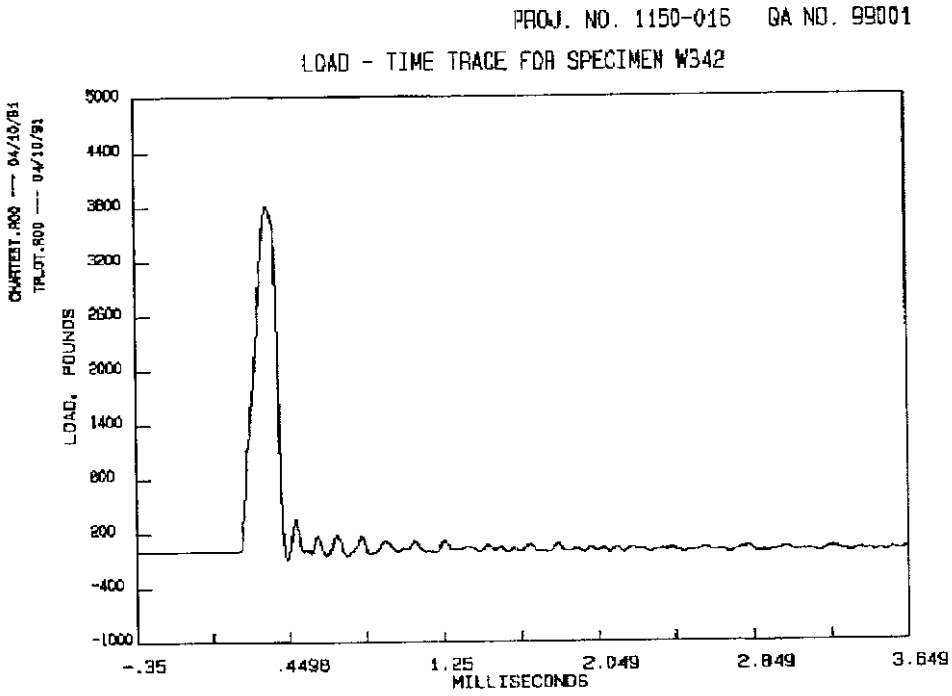


Figure B-14. Load-Time Trace for Charpy V-Notch Impact Specimen 31B

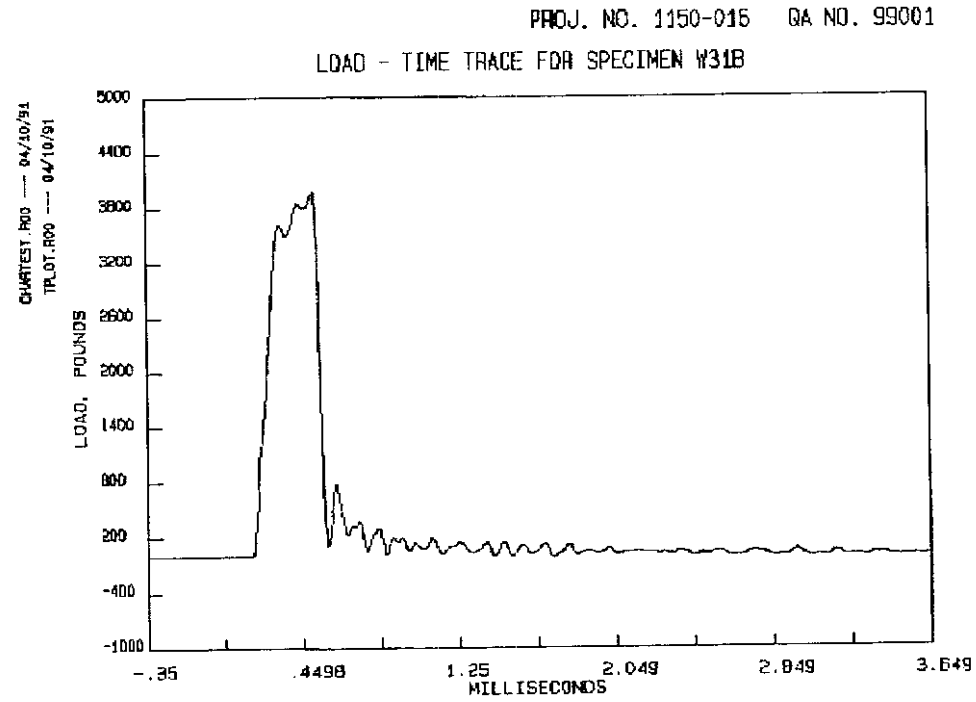


Figure B-15. Load-Time Trace for Charpy V-Notch Impact Specimen 36M

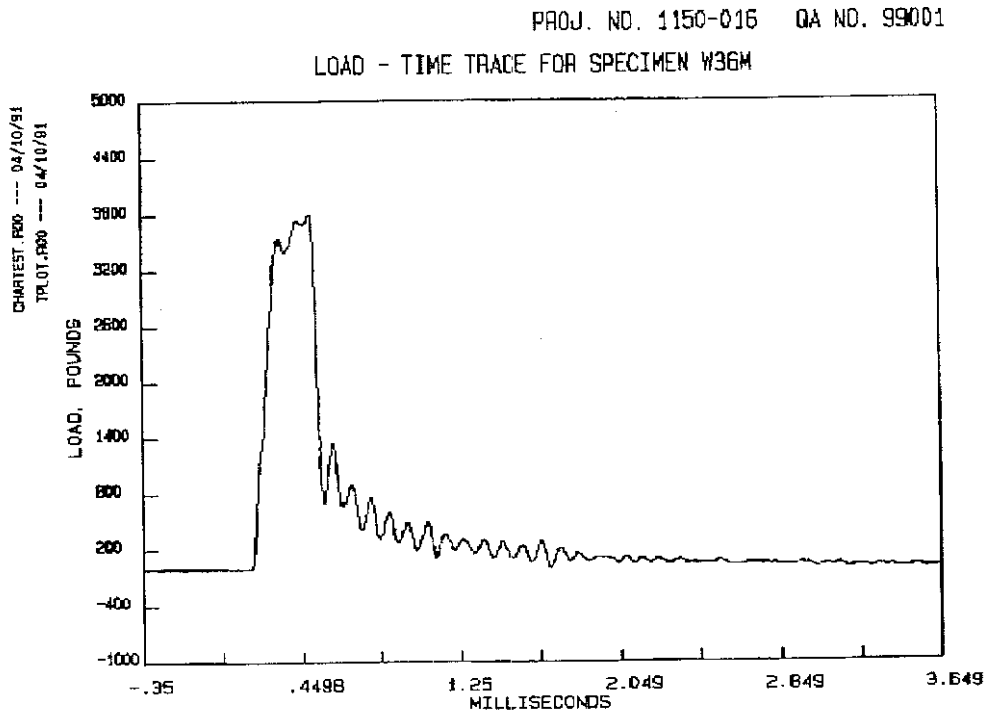


Figure B-16. Load-Time Trace for Charpy V-Notch Impact Specimen 357

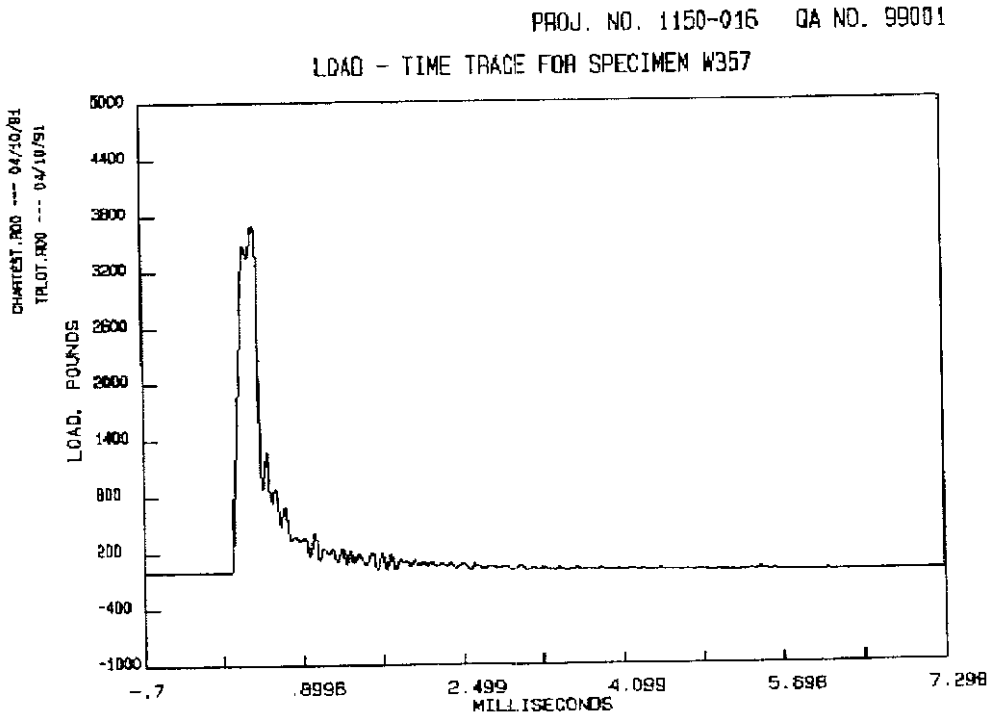


Figure B-17. Load-Time Trace for Charpy V-Notch Impact Specimen 33P

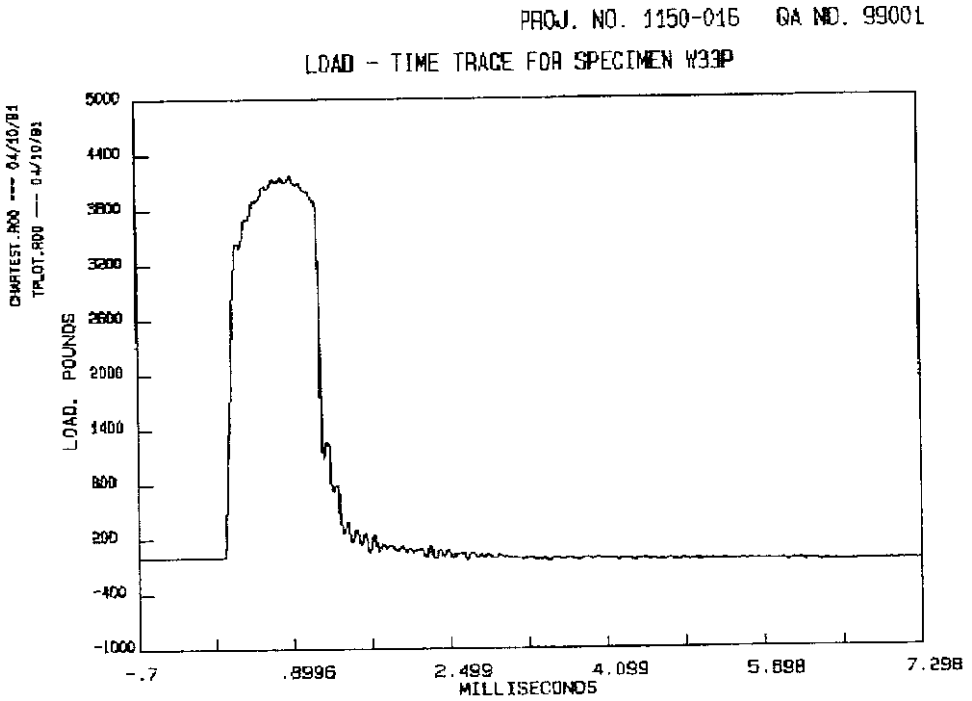


Figure B-18. Load-Time Trace for Charpy V-Notch Impact Specimen 37J

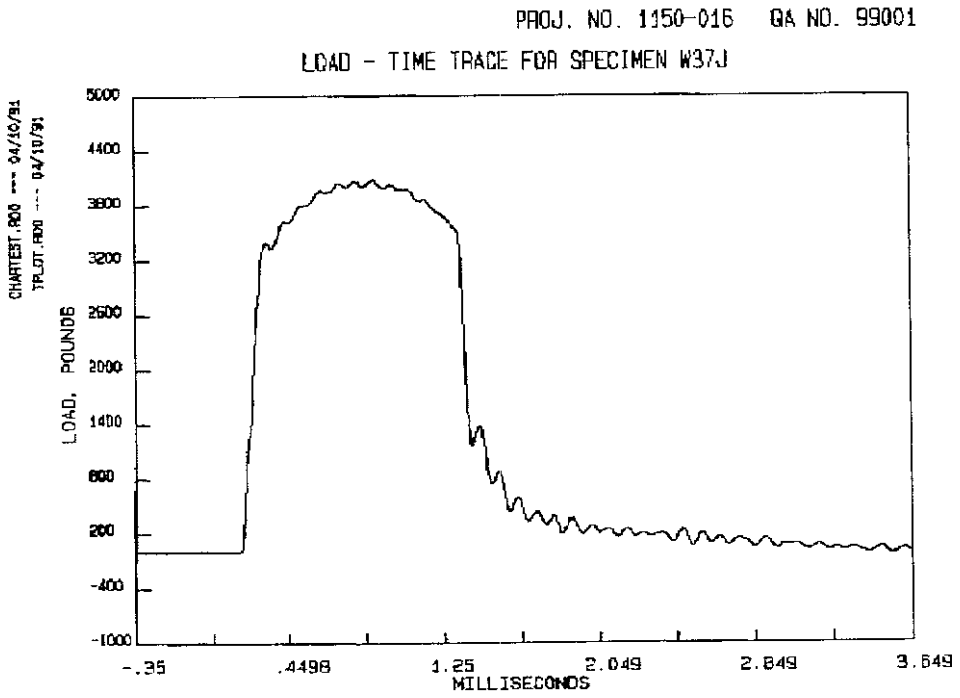


Figure B-19. Load-Time Trace for Charpy V-Notch Impact Specimen 32C

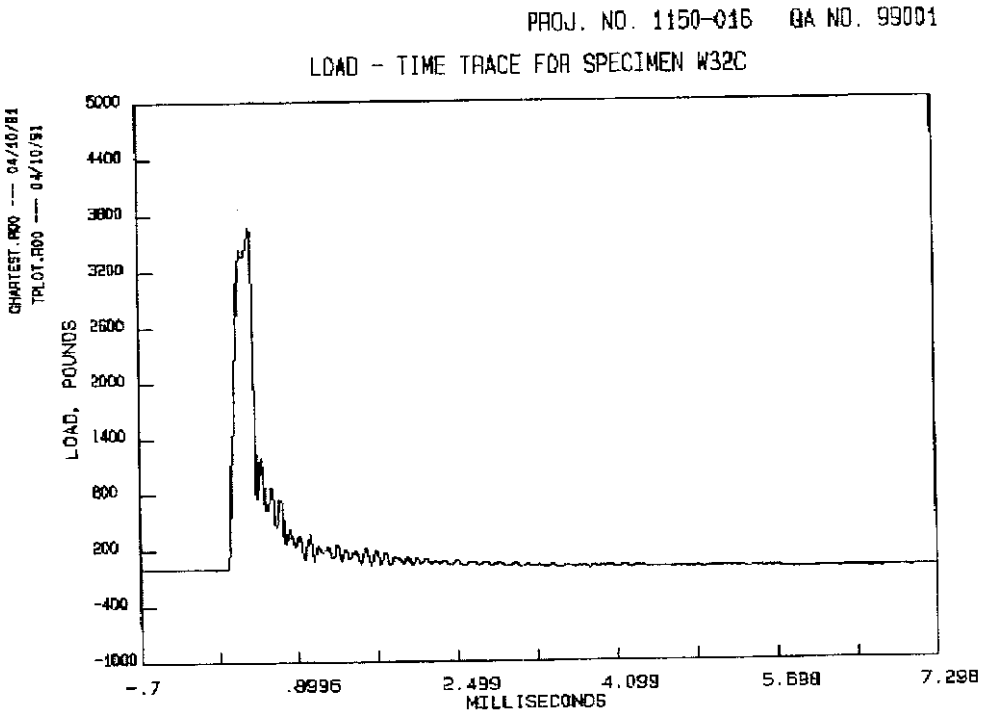


Figure B-20. Load-Time Trace for Charpy V-Notch Impact Specimen 32A

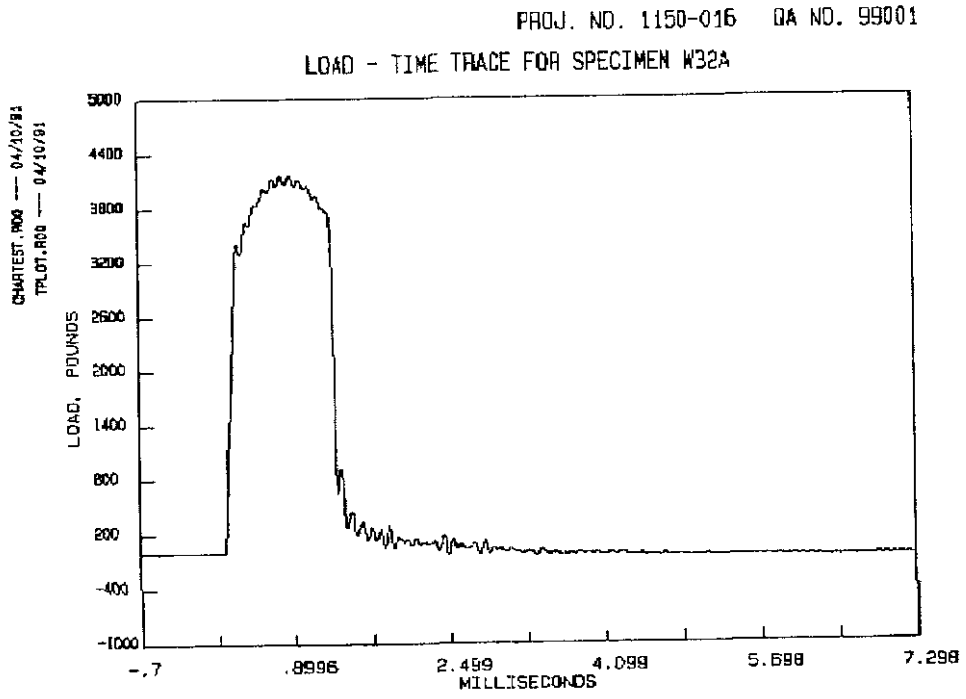


Figure B-21. Load-Time Trace for Charpy V-Notch Impact Specimen 35B

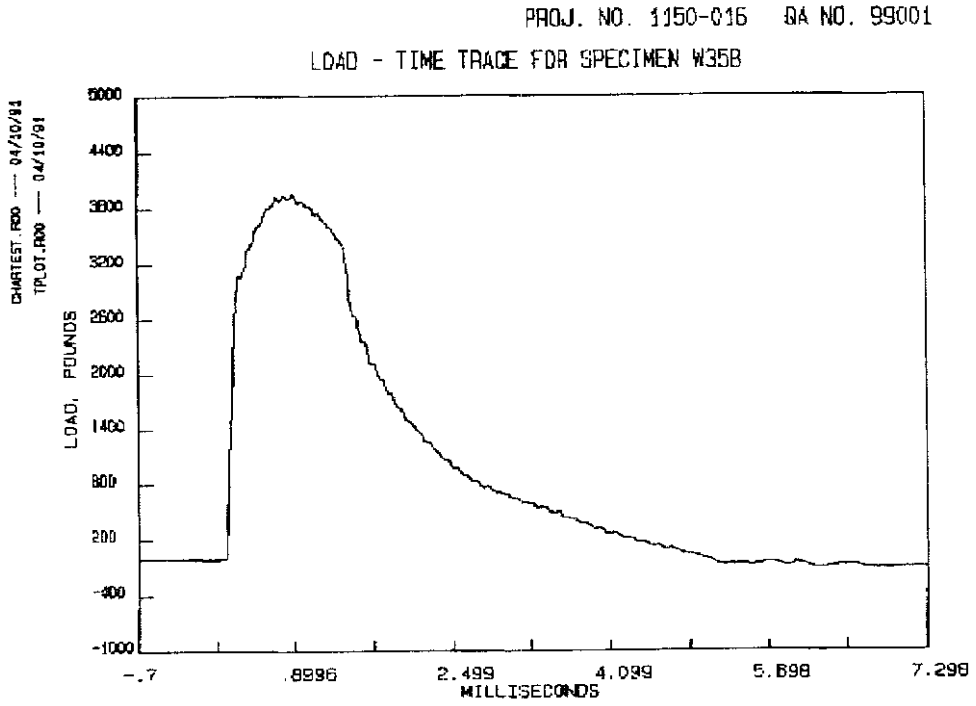


Figure B-22. Load-Time Trace for Charpy V-Notch Impact Specimen 36L

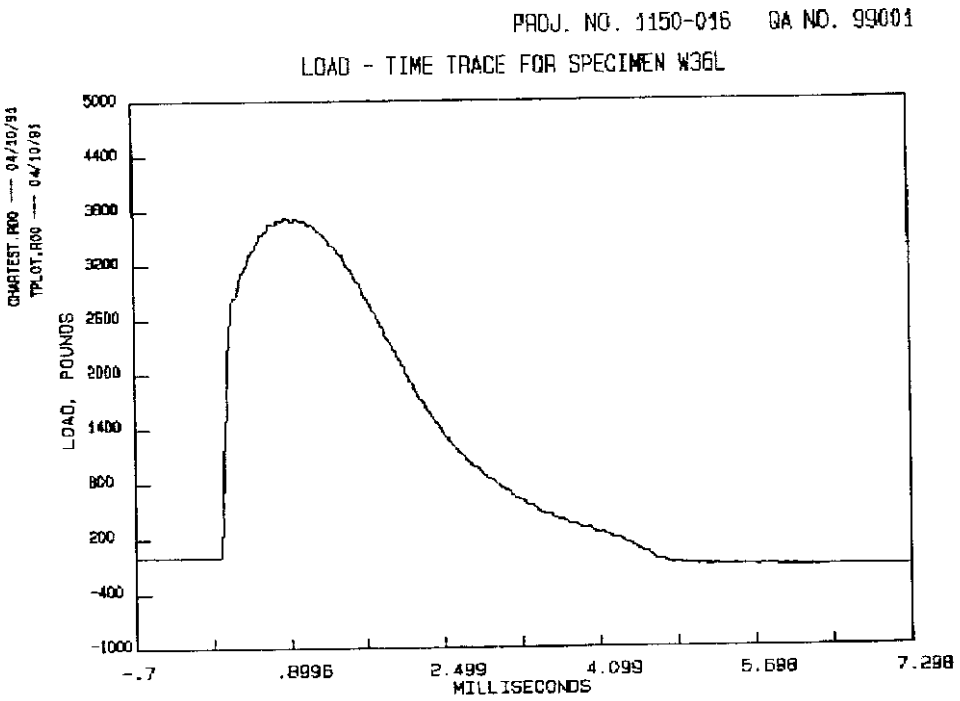


Figure B-23. Load-Time Trace for Charpy V-Notch Impact Specimen 36B

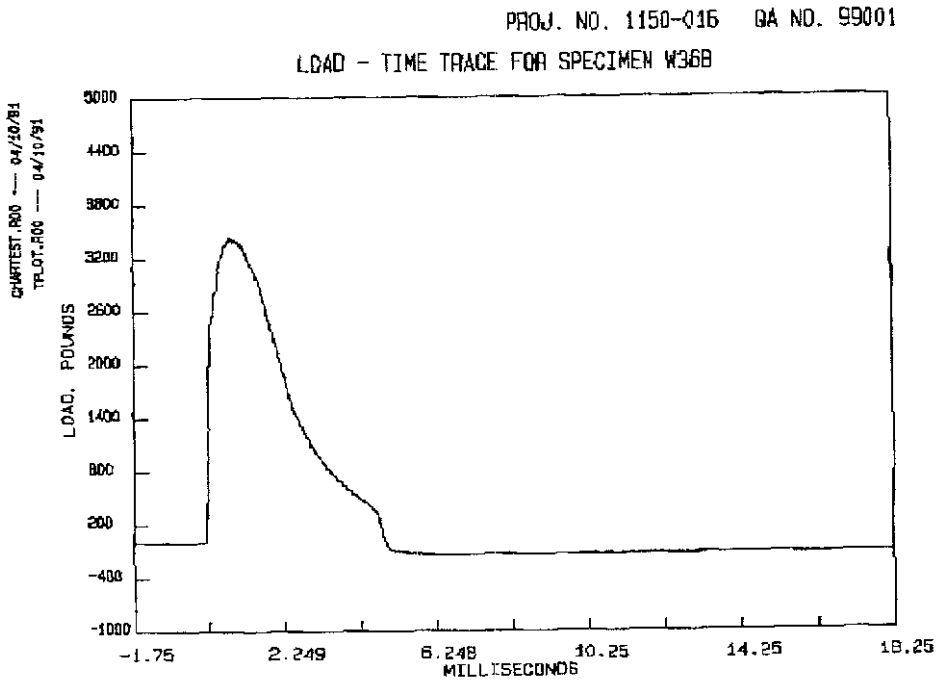


Figure B-24. Load-Time Trace for Charpy V-Notch Impact Specimen 43L

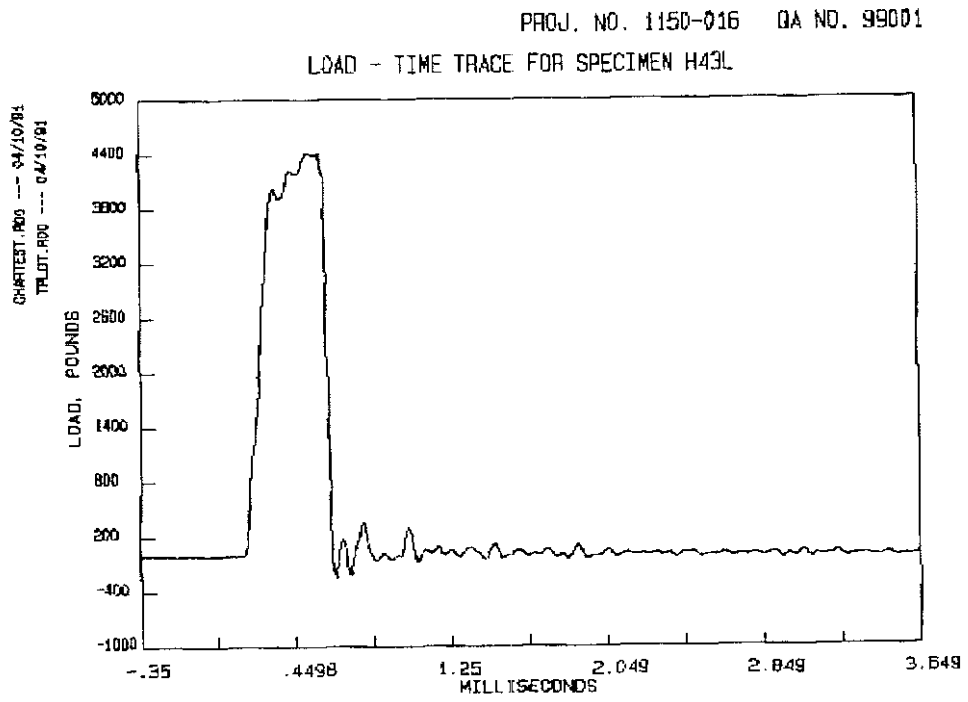


Figure B-25. Load-Time Trace for Charpy V-Notch Impact Specimen 47P

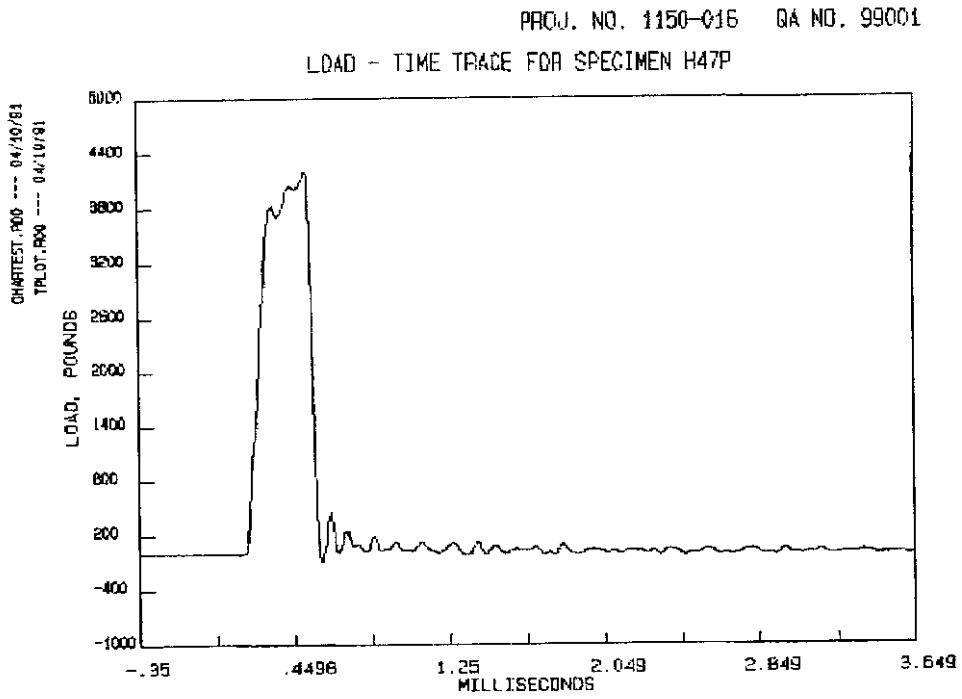


Figure B-26. Load-Time Trace for Charpy V-Notch Impact Specimen 421

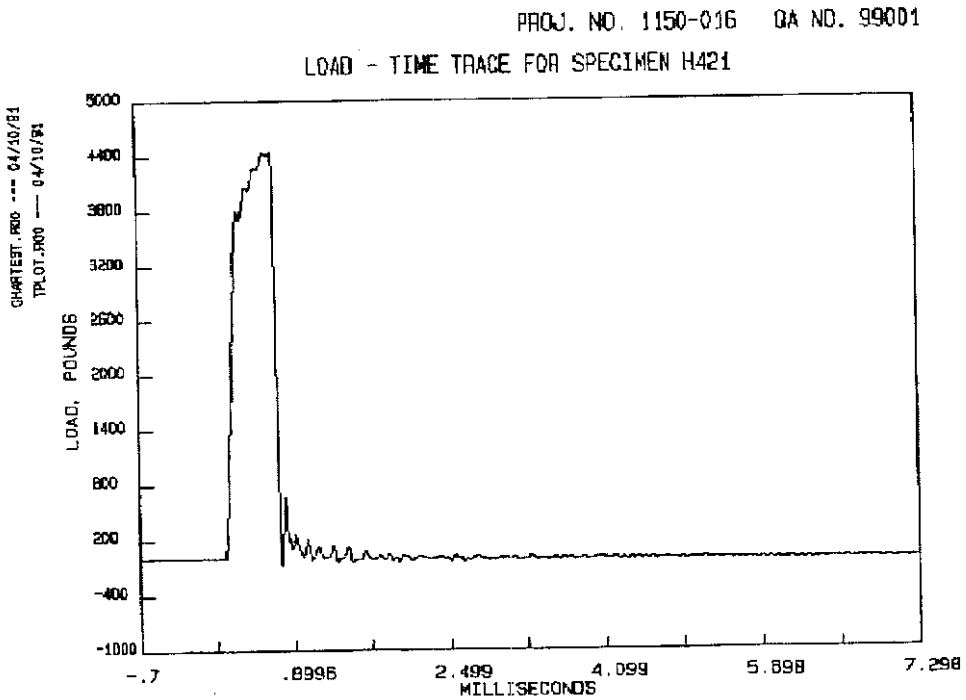


Figure B-27. Load-Time Trace for Charpy V-Notch Impact Specimen 475

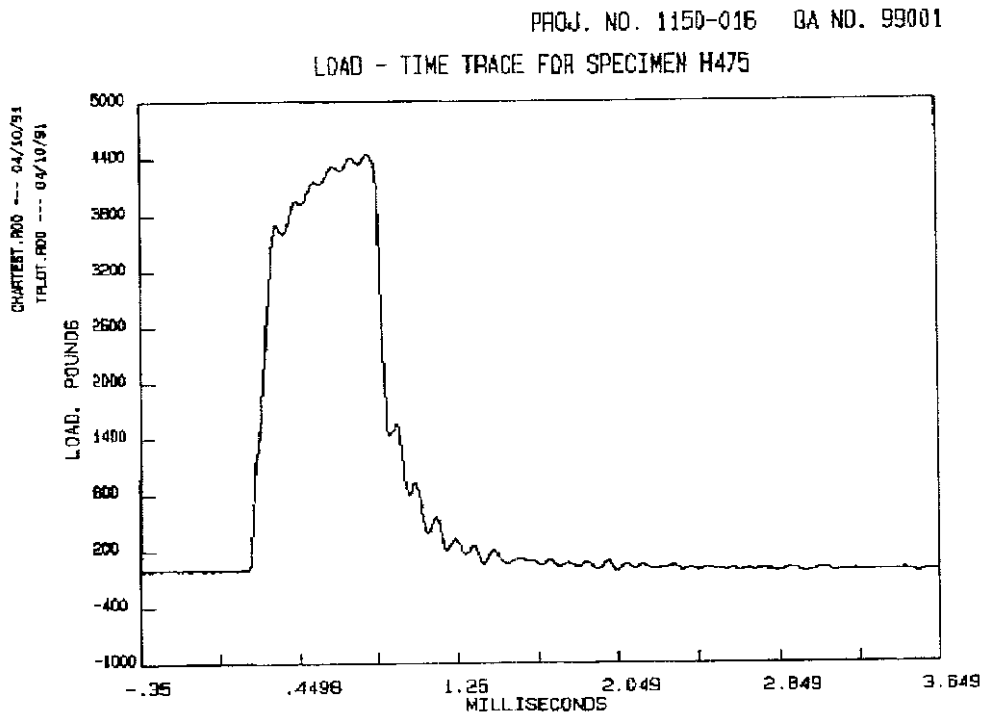


Figure B-28. Load-Time Trace for Charpy V-Notch Impact Specimen 44M

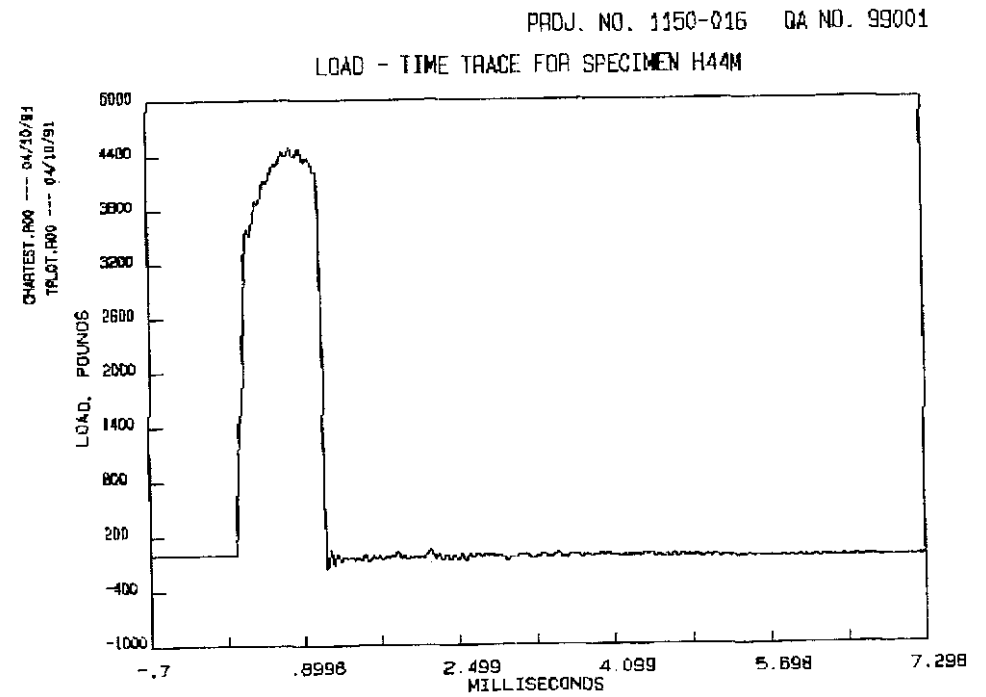


Figure B-29. Load-Time Trace for Charpy V-Notch Impact Specimen 41L

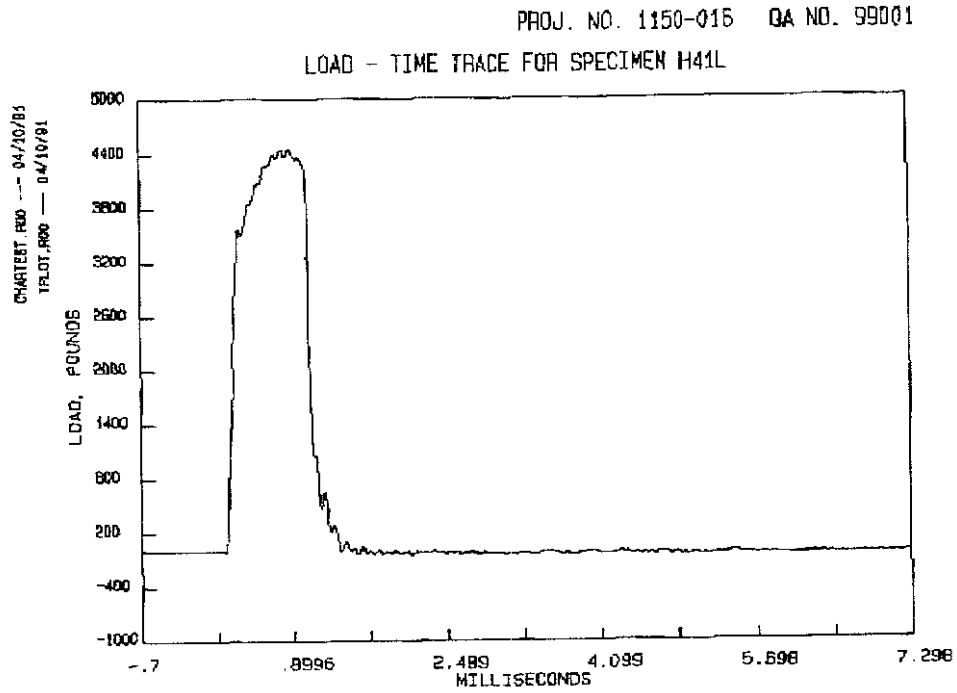


Figure B-30. Load-Time Trace for Charpy V-Notch Impact Specimen 467

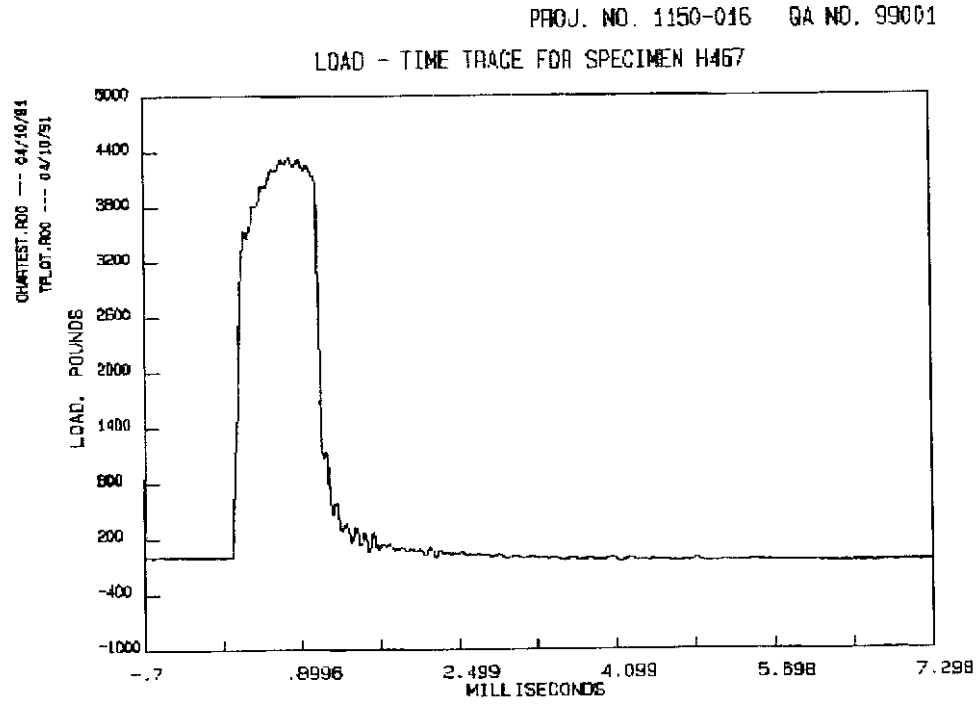


Figure B-31. Load-Time Trace for Charpy V-Notch Impact Specimen 462

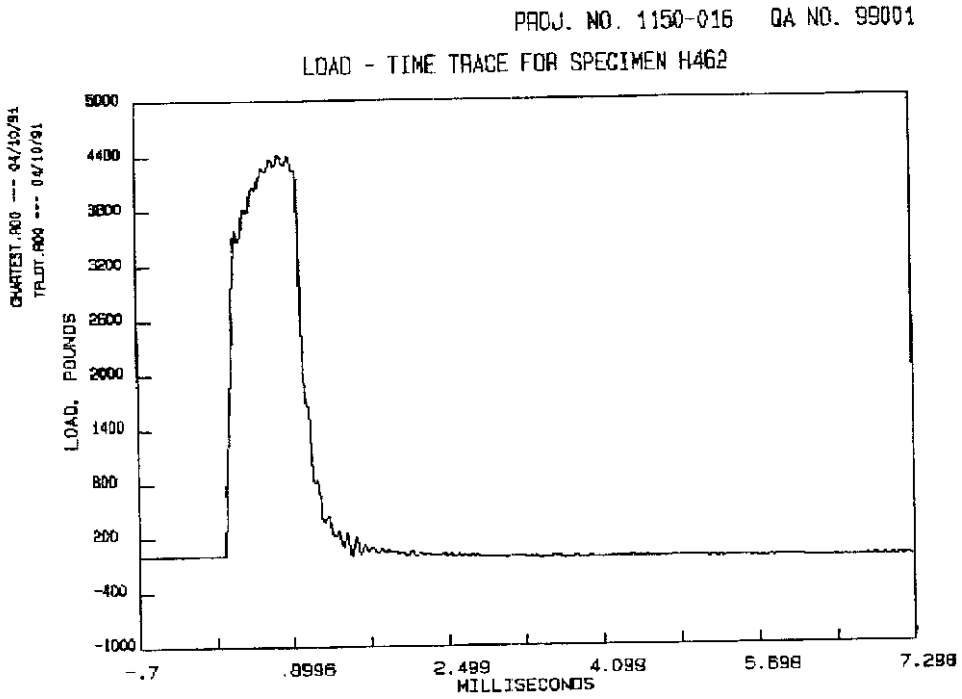


Figure B-32. Load-Time Trace for Charpy V-Notch Impact Specimen 45C

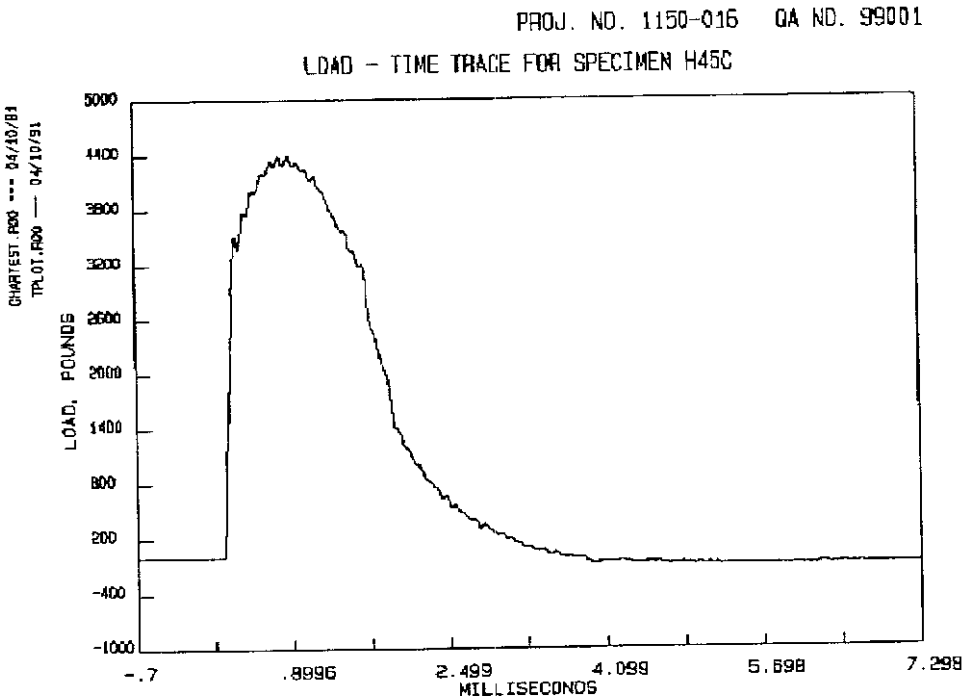


Figure B-33. Load-Time Trace for Charpy V-Notch Impact Specimen 456

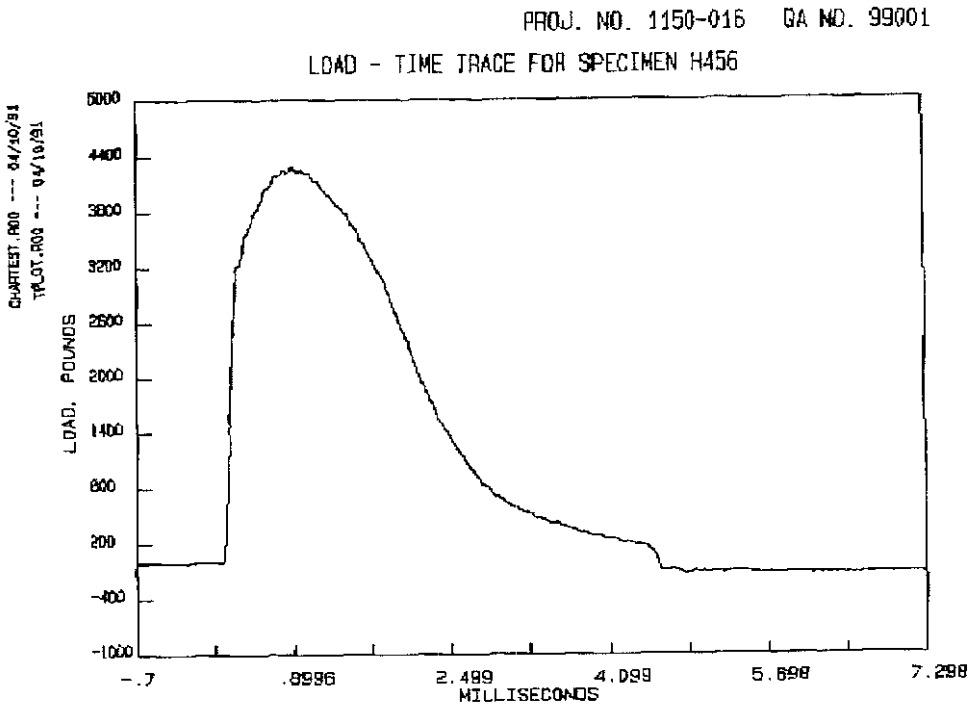


Figure B-34. Load-Time Trace for Charpy V-Notch Impact Specimen 424

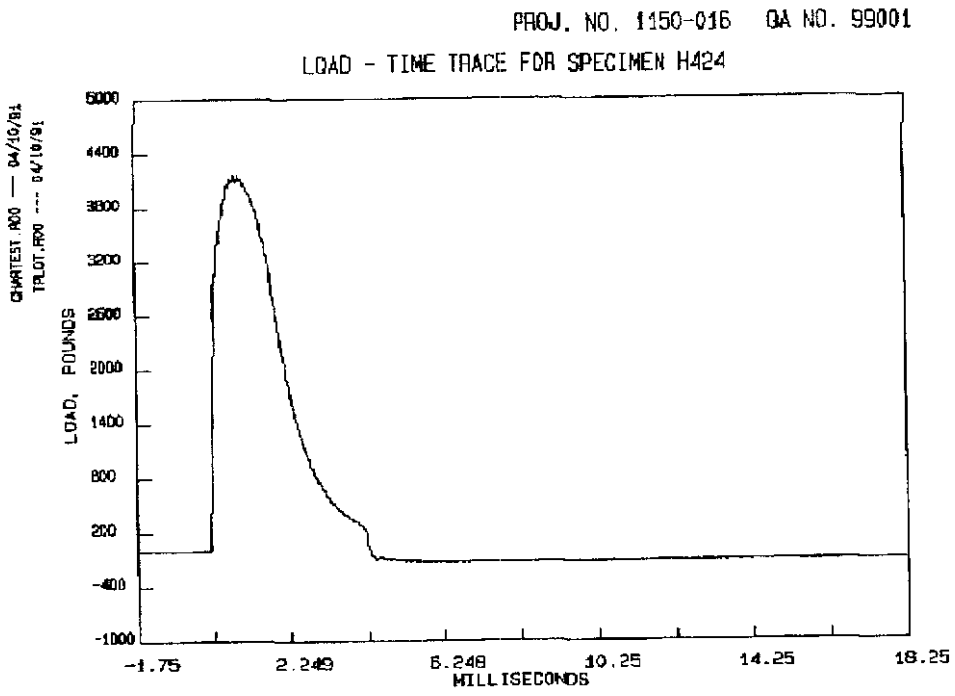


Figure B-35. Load-Time Trace for Charpy V-Notch Impact Specimen 413

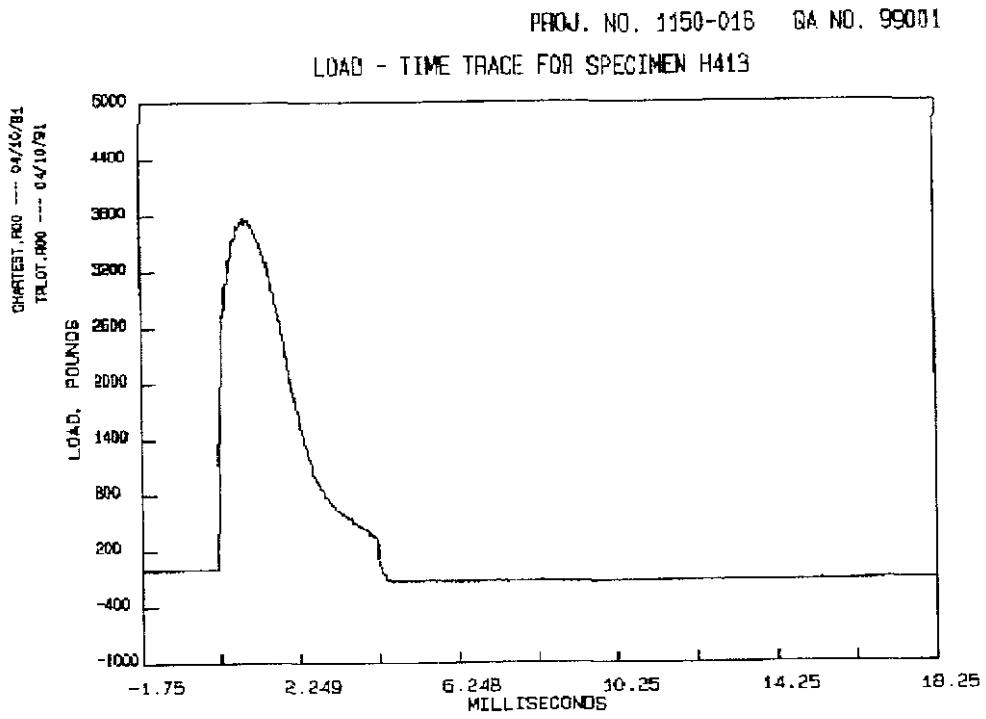


Figure B-36. Load-Time Trace for Charpy V-Notch Impact Specimen 77D

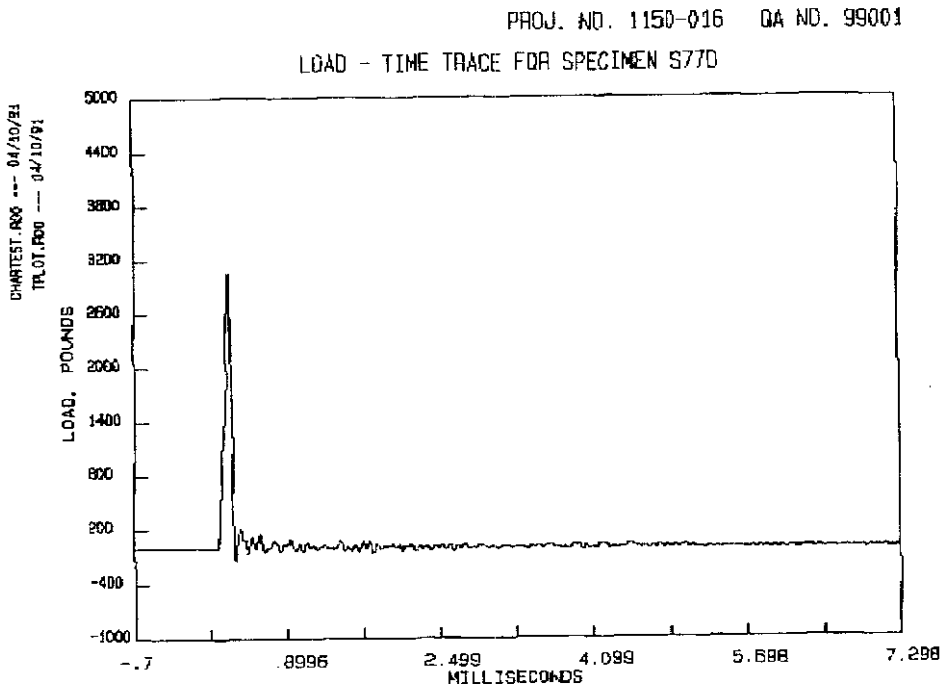


Figure B-37. Load-Time Trace for Charpy V-Notch Impact Specimen 777

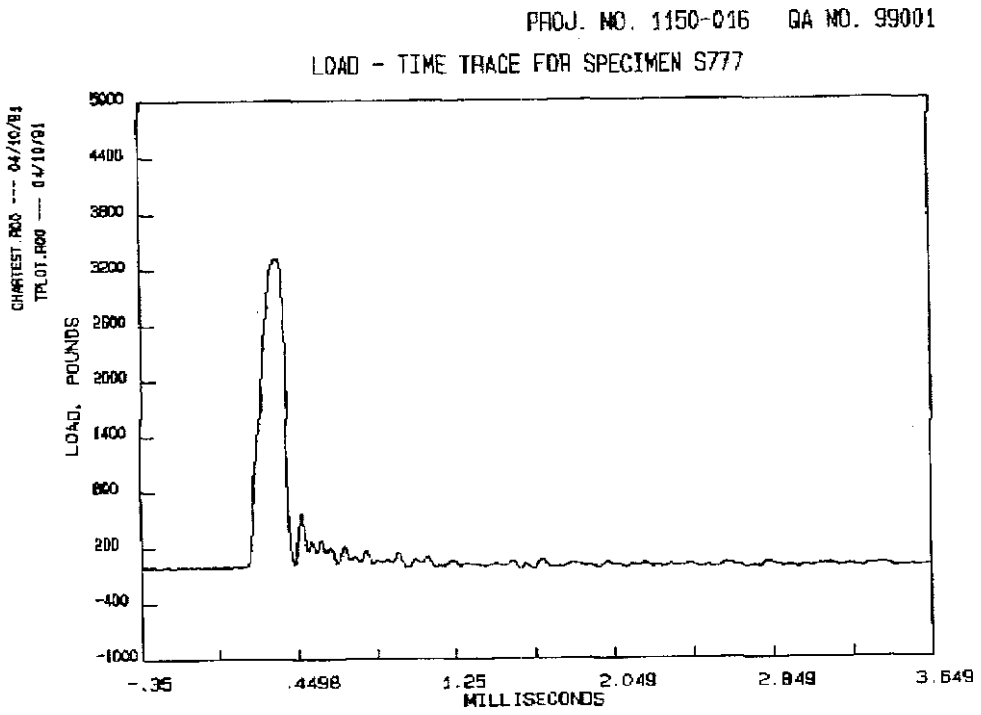


Figure B-38. Load-Time Trace for Charpy V-Notch Impact Specimen 77E

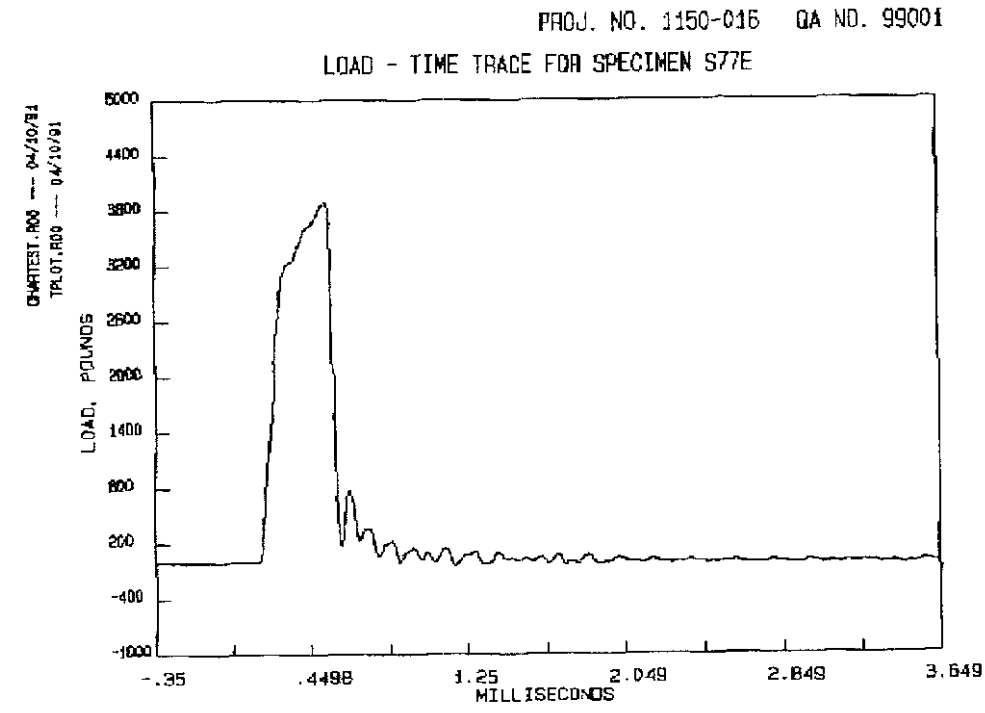


Figure B-39. Load-Time Trace for Charpy V-Notch Impact Specimen 76Y

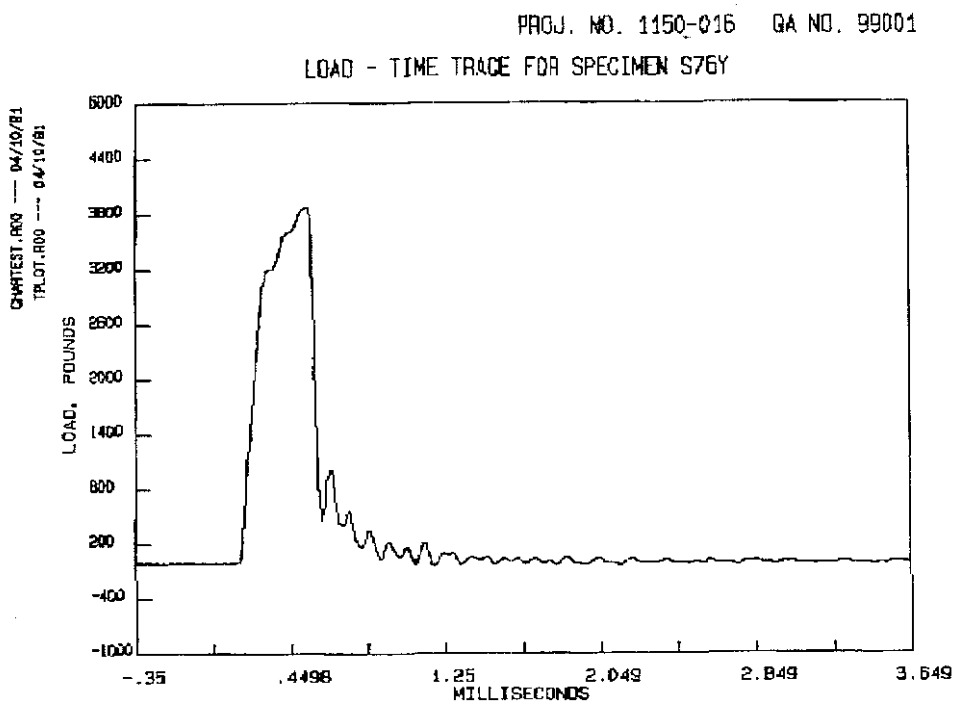


Figure B-40. Load-Time Trace for Charpy V-Notch Impact Specimen 77U

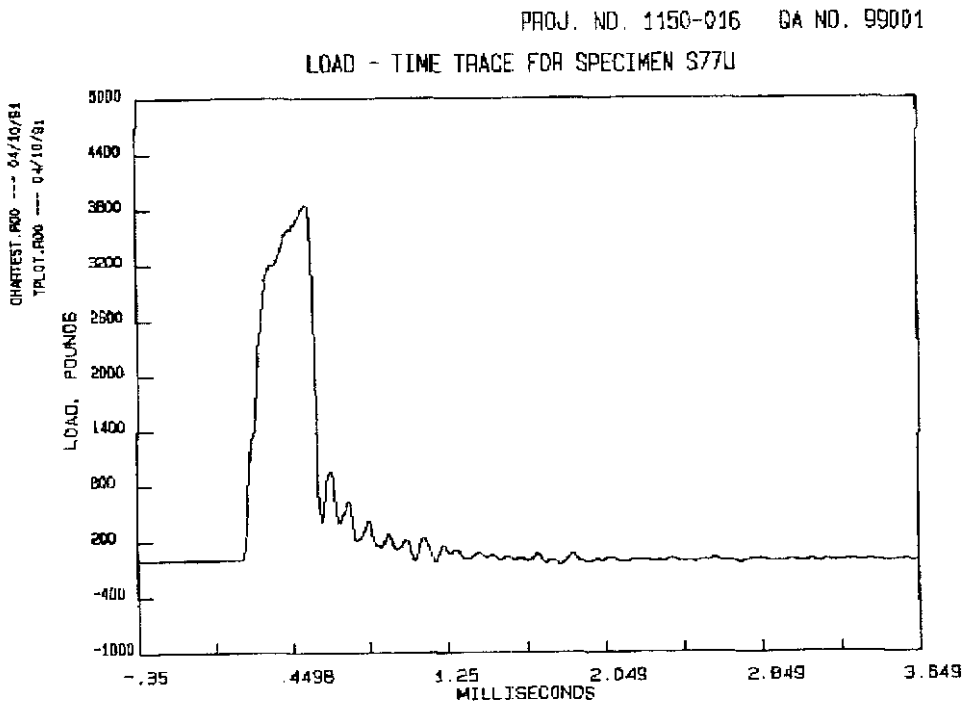


Figure B-41. Load-Time Trace for Charpy V-Notch Impact Specimen 776

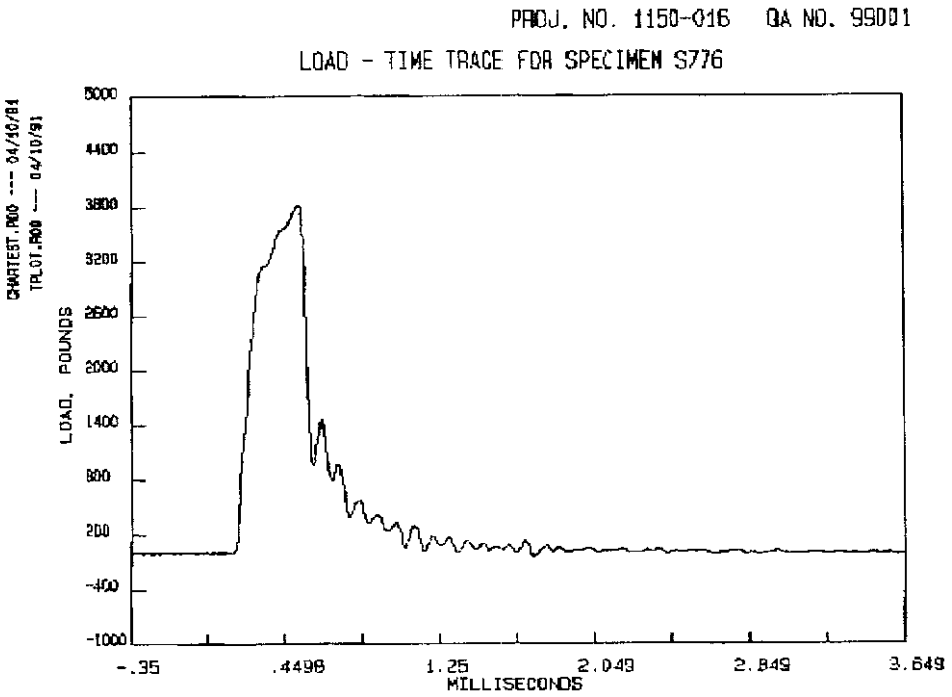


Figure B-42. Load-Time Trace for Charpy V-Notch Impact Specimen 773

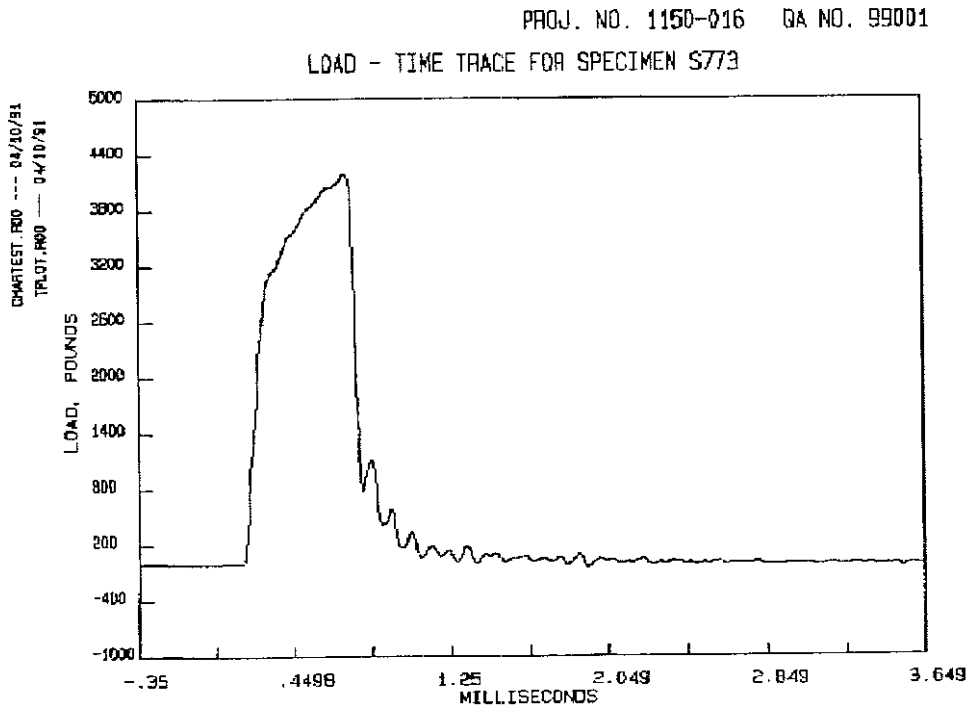


Figure B-43. Load-Time Trace for Charpy V-Notch Impact Specimen 77P

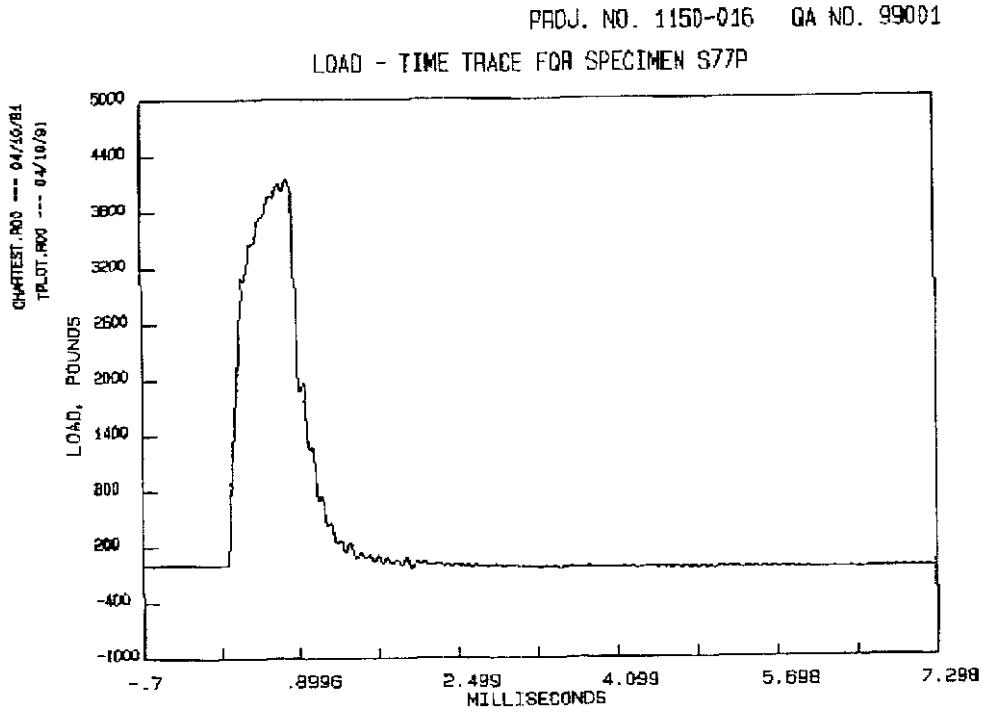


Figure B-44. Load-Time Trace for Charpy V-Notch Impact Specimen 77M

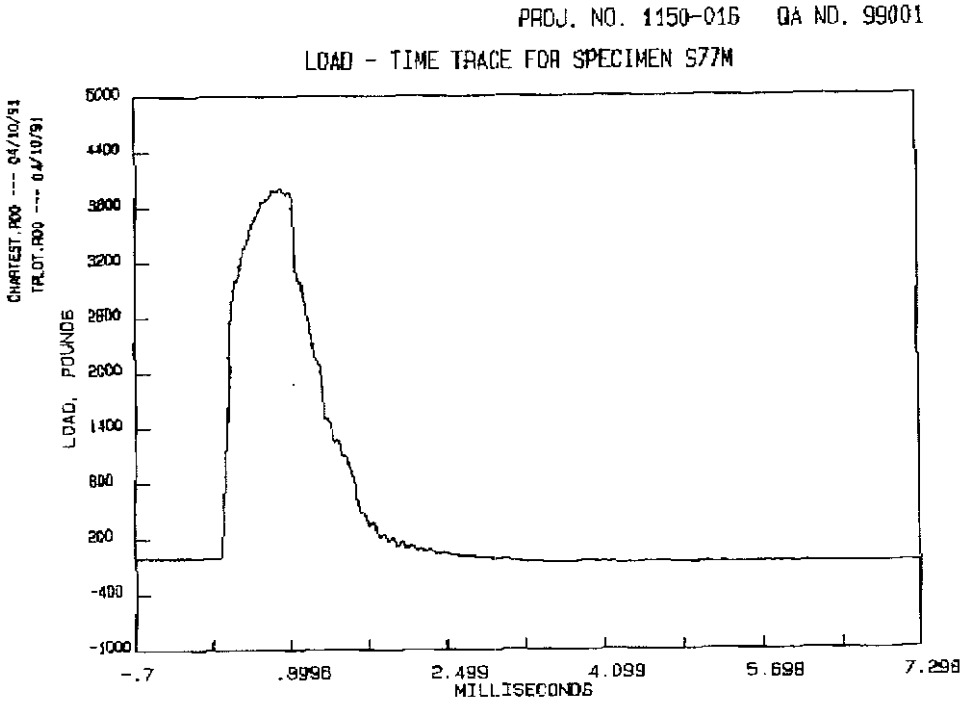


Figure B-45. Load-Time Trace for Charpy V-Notch Impact Specimen 772

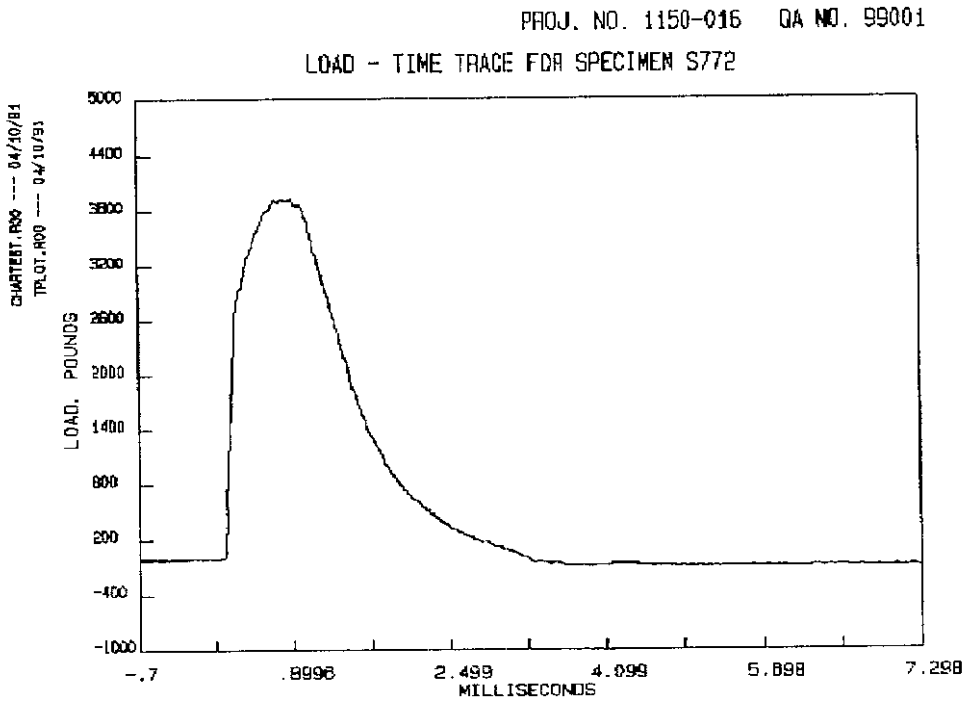


Figure B-46. Load-Time Trace for Charpy V-Notch Impact Specimen 77Y

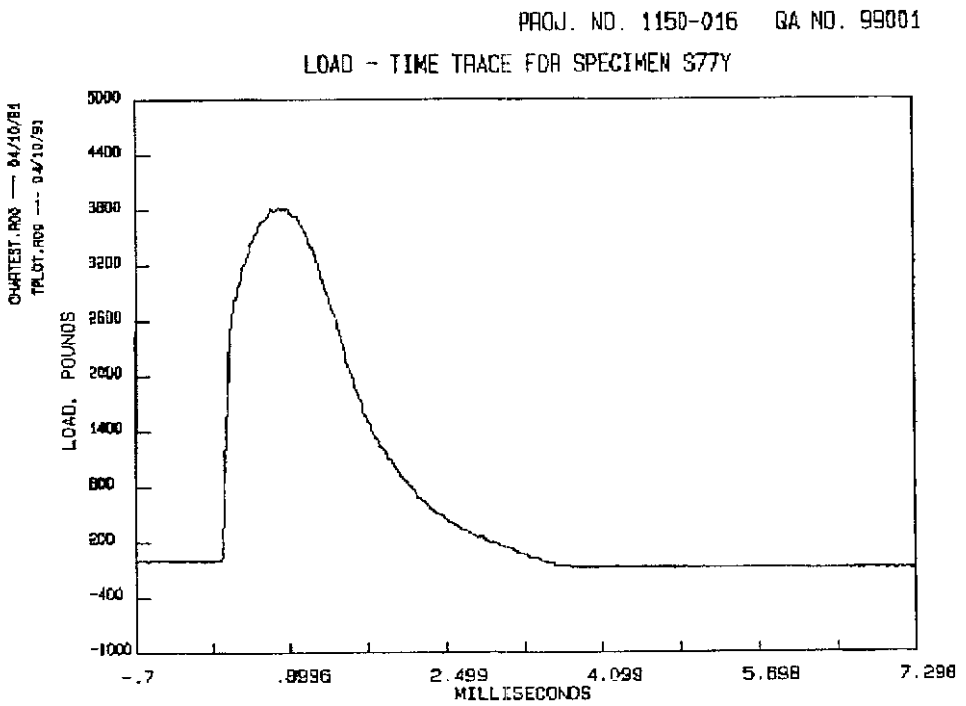
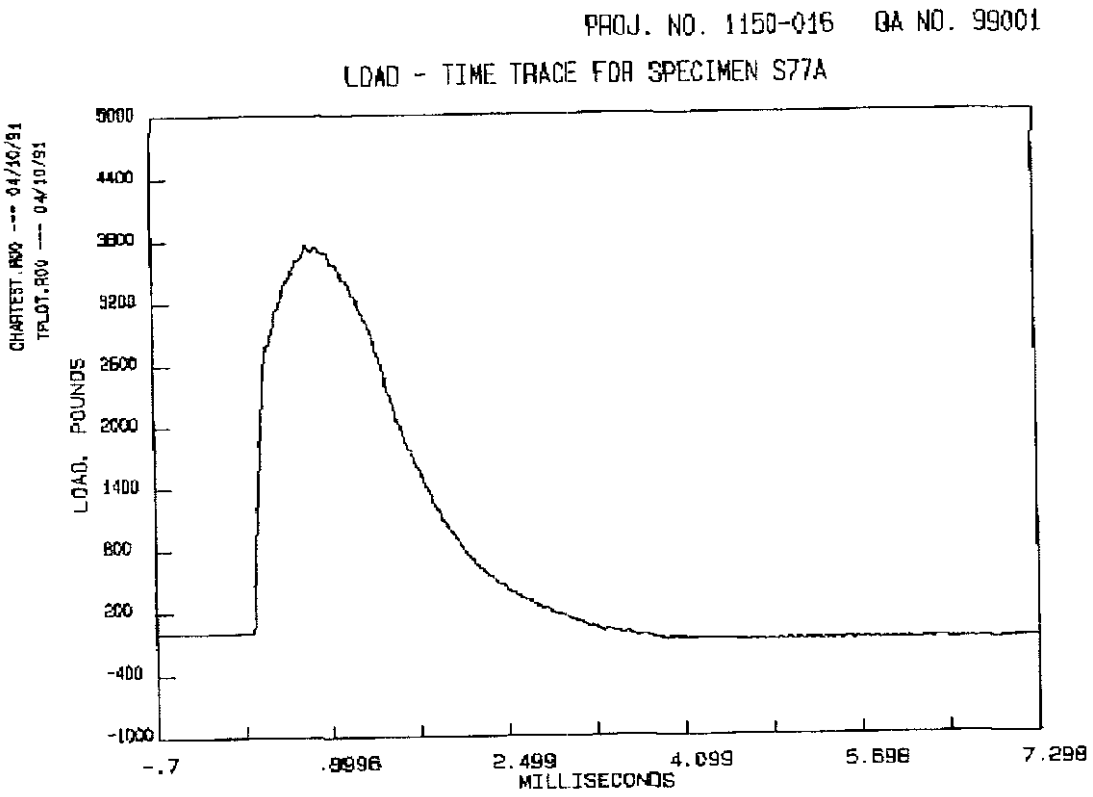


Figure B-47. Load-Time Trace for Charpy V-Notch Impact Specimen 77A



APPENDIX C

Unirradiated and Irradiated Tensile Data for the Arkansas Nuclear One Unit 2 RVSP Materials

Table C-1. Tensile Properties of Unirradiated Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation

Specimen No.	Test Temp. (F)	Strength, Ksi		Elongation, %		Reduction of Area, %
		0.2% Yield or Upper/Lower	Ultimate	Uniform	Total	
2JK	71	69.8/67.4	91.1	9.2	25	66.0
2L6	71	70.3/67.9	90.6	9.5	25	69.0
2J3	71	69.1/67.9	90.7	9.3	25	71.0
2J2	250	65.5/64.3	84.4	8.5	22	67.3
2L5	250	65.5/64.9	84.9	8.0	22	71.2
2KL	250	65.5/64.2	84.2	7.1	22	69.0
2K1	550	56.3/55.1	83.5	8.8	22	66.3
2KA	550	56.4/55.8	83.5	8.8	22	65.3
2K7	550	58.5/56.9	84.5	7.5	24	62.0

Table C-2. Tensile Properties of Unirradiated Weld Metal, Heat 10120

Specimen No.	Test Temp. (F)	Strength, Ksi		Elongation, %		Reduction of Area, %
		0.2% Yield or Upper/Lower	Ultimate	Uniform	Total	
3J6	71	75.2/74.0	88.6	9.2	25	73.0
3J5	71	75.2/74.0	87.8	8.3	22	71.0
3J4	71	76.4/74.0	88.7	8.0	25	73.0
3JK	250	70.3/67.9	80.2	7.4	23	71.1
3JP	250	70.9/69.7	82.1	7.0	20	72.5
3KC	250	73.5/70.4	84.2	6.6	23	72.6
3J3	550	64.9	88.1	7.2	22	70.7
3JY	550	66.7	84.5	7.2	22	66.7
3K6	550	66.1	83.6	7.3	22	69.6

Table C-3. Tensile Properties of Unirradiated Heat Affected Zone Metal

Specimen No.	Test Temp. (F)	Strength, Ksi		Elongation, %		Reduction of Area, %
		0.2% Yield or Upper/Lower	Ultimate	Uniform	Total	
4JY	71	68.6/67.4	91.1	9.9	26	71.0
4JJ	71	67.9/66.7	90.8	8.6	22	69.9
4KC	71	66.6/65.4	85.0	7.2	25	72.8
4KA	250	62.4/61.2	78.6	6.8	24	72.4
4JE	250	63.7	84.3	7.3	21	71.8
4KK	250	61.2/60.0	78.7	6.7	23	71.0
4J7	550	58.2/57.6	85.6	7.2	20	69.6
4JU	550	56.6/56.0	83.0	7.2	21	68.7
4JT	550	58.8/58.2	85.0	6.0	21	70.0

Table C-4. Tensile Properties of Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.33 x 10^{18} n/cm^2 (E>1.0 MeV) Transverse Orientation

Specimen No.	Test Temp. (F)	Strength, Ksi		Elongation, %		Reduction of Area, %
		0.2% Yield	Ultimate	Uniform	Total	
2L2	77	73.01	95.93	9.4	21.7	67.2
2J7	250	71.69	87.98	9.5	20.8	67.4
2JC	550	66.19	89.00	7.6	17.8	66.4

Table C-5. Tensile Properties of Weld Metal, Heat 10120 Irradiated to 3.35 x 10^{18} n/cm^2 (E>1.0 MeV)

Specimen No.	Test Temp. (F)	Strength, Ksi		Elongation, %		Reduction of Area, %
		0.2% Yield	Ultimate	Uniform	Total	
3J1	77	75.61	88.42	8.3	22.1	79.3
3K3	250	71.34	81.71	6.9	19.2	73.6
3KJ	550	69.00	85.77	7.1	20.2	73.8

Table C-6. Tensile Properties of Heat Affected Zone Metal Irradiated to 3.56 x 10^{18} n/cm^2 (E>1.0 MeV)

Specimen No.	Test Temp. (F)	Strength, Ksi		Elongation, %		Reduction of Area, %
		0.2% Yield	Ultimate	Uniform	Total	
4K7	77	71.34	93.09	7.6	22.6	72.8
4JA	250	66.97	86.69	7.7	19.5	63.0
4JC	550	68.27	88.78	8.4	19.8	68.2

APPENDIX D

Unirradiated and Irradiated Charpy V-Notch Impact Surveillance Data for the Arkansas Nuclear One Unit 2 RVSP Materials Using Hyperbolic Tangent Curve-Fitting Method

Table D-1. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
25E	-120	4	4	0
243	-80	6	4	0
231	-40	7.5	8	10
23J	-40	14.5	14	10
26D	0	24	24	20
24D	0	27.5	25	20
22Y	40	36.5	32	25
241	40	58.5	45	35
261	60	57	46	40
232	60	60	46	45
225	60	72.5	57	50
222	80	85.5	69	60
25B	120	107	77	70
25Y	120	110	75	80
22P	160	125.5	83	100
242	160	140	90	100
22M	210	127.5	85	100
23E	210	142	89	100

Table D-2. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.33 x 10^{18} n/cm^2 (E>1.0 MeV) Transverse Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
24T	-40	4.5	4.0	5
257	0	11.7	24.2	10
26B	35	20.5	24.4	25
22L	55	35.0	32.0	30
264	80	33.5	34.2	35
214	90	75.5	62.6	60
21K	110	66.7	56.8	50
24A	130	74.0	63.6	70
21U	160	88.0	87.0	80
253	240	113.3	86.4	100
25M	280	124.2	86.8	100
226	320	122.0	77.2	100

Table D-3. Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation

	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 68.0 B: 65.8 C: 72.7 T0: 62.8	A: 45.8 B: 44.8 C: 83.9 T0: 49.2	A: 50.0 B: 50.0 C: 78.7 T0: 66.9
W-97 Capsule	A: 61.0 B: 58.8 C: 91.7 T0: 102.1	A: 43.5 B: 42.5 C: 85.7 T0: 74.2	A: 50.0 B: 50.0 C: 92.5 T0: 94.2

Table D-4. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Longitudinal Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
11L	-80	6	8	5
12M	-40	8.5	8	10
11Y	-40	11.5	10	10
15L	0	18	18	25
134	0	31	26	25
11D	40	64	49	35
151	40	86	67	40
13M	60	105	72	60
15C	60	100	69	65
14Y	60	105	74	60
144	80	99	68	65
12A	80	108	72	65
14U	120	145	95	90
11K	120	154.5	96	100
11C	160	154	92	100
13P	160	166.5	98	100
115	210	159	92	100

Table D-5. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.41 x 10^{18} n/cm^2 (E>1.0 MeV) Longitudinal Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
112	-40	7.0	8.0	5
14M	0	21.3	20.0	15
126	15	21.0	20.8	15
131	30	34.2	30.8	20
154	55	46.0	39.8	35
13T	80	80.3	66.8	50
12J	120	92.3	72.4	60
11T	160	111.0	82.0	75
11A	200	142.5	98.2	100
124	241	141.2	98.2	100
13U	280	135.3	94.6	100
14A	320	141.0	94.8	100

Table D-6. Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Longitudinal Orientation

	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 80.4 B: 78.2 C: 60.6 T0: 47.0	A: 48.2 B: 47.2 C: 59.9 T0: 31.8	A: 50.0 B: 50.0 C: 70.7 T0: 48.0
W-97 Capsule	A: 71.1 B: 68.9 C: 86.1 T0: 83.2	A: 48.8 B: 47.8 C: 84.1 T0: 63.8	A: 50.0 B: 50.0 C: 94.5 T0: 89.7

Table D-7. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Weld Metal, Heat 10120

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
34D	-120	4.5	4	0
34L	-80	6.5	6	10
33B	-40	5	4	10
323	-40	10	10	15
321	0	24.5	23	30
35L	0	30.5	27	30
31K	20	80	61	65
31P	40	95.5	66	60
361	50	126.5	85	80
373	50	110.5	76	70
364	50	118	82	75
36T	80	128.5	86	85
375	80	140	91	90
377	120	140	91	95
35E	120	155	94	99
333	160	146	93	100
371	160	161	93	100

Table D-8. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Weld Metal, Heat 10120 Irradiated to 3.34 x 10^{18} n/cm^2 (E>1.0 MeV)

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
36D	-50	5.0	6.8	5
33J	0	26.9	27.8	25
37T	10	21.5	29.2	25
35D	15	26.5	29.4	35
32E	25	38.0	34.6	55
37K	35	89.0	68.8	60
312	60	100.0	80.6	75
344	78	117.0	89.3	90
31C	120	126.5	93.0	90
356	160	159.0	94.6	100
353	200	143.0	91.0	100
315	240	138.8	91.6	100

Table D-9. Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2 Weld Metal, Heat 10120

	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 78.1 B: 75.9 C: 42.5 T0: 27.1	A: 46.7 B: 45.7 C: 37.0 T0: 15.6	A: 50.0 B: 50.0 C: 60.1 T0: 18.4
W-97 Capsule	A: 74.6 B: 72.4 C: 45.7 T0: 42.2	A: 47.1 B: 46.1 C: 37.3 T0: 25.7	A: 50.0 B: 50.0 C: 49.0 T0: 27.9

Table D-10. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Heat-Affected-Zone Material

Specimen No.	Test Temp. (°F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
425	-160	12	7	5
47A	-120	16.5	9	10
41U	-80	50.5	31	25
42Y	-80	81	49	40
41J	-40	51	36	35
453	-40	106.5	63	70
45U	0	102	65	55
46C	0	116.5	71	80
441	40	147	81	90
43K	40	168	85	95
45J	50	55	44	50
44J	50	111.5	73	85
45A	50	127	73	75
415	80	122.5	80	90
41K	80	170	88	100
46B	120	156.5	81	100
447	120	164.5	84	100
46U	160	151.5	85	100
47U	160	163	87	100

Table D-11. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Heat-Affected-Zone Material, Irradiated to 3.49 x 10^{18} n/cm^2 (E>1.0 MeV)

Specimen No.	Test Temp. (°F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
41M	-80	14.0	19.0	15
461	-40	27.5	33.6	25
44C	-40	87.5	57.6	60
44U	-30	95.9	60.4	65
42L	-25	125.5	74.0	75
45M	-15	58.0	44.0	45
45P	0	55.3	44.2	45
476	35	160.0	88.4	100
45E	55	95.5	65.2	100
477	80	148.0	85.0	100
465	140	150.0	88.0	100
46E	200	138.5	91.0	100

Table D-12. Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2 Heat-Affected-Zone Material

Weld Metal	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 81.7 B: 79.5 C: 119.7 T0: -37.2	A: 42.3 B: 41.3 C: 94.4 T0: -60.3	A: 50.0 B: 50.0 C: 103.8 T0: -37.8
W-97 Capsule	A: 70.3 B: 68.1 C: 69.6 T0: -31.9	A: 45.3 B: 44.3 C: 97.5 T0: -40.4	A: 50.0 B: 50.0 C: 68.9 T0: -31.3

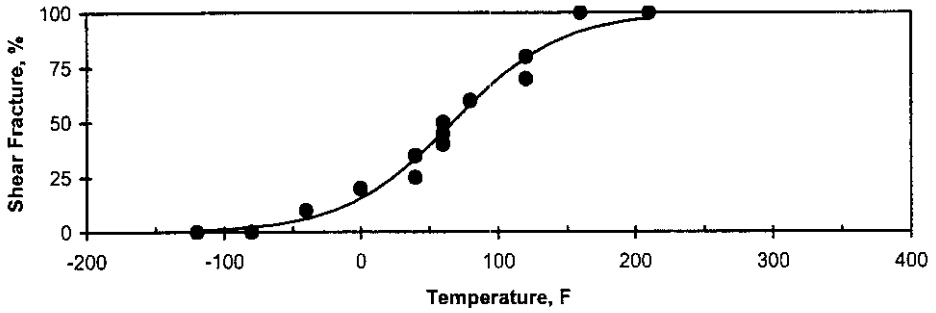
Table D-13. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Standard Reference Material

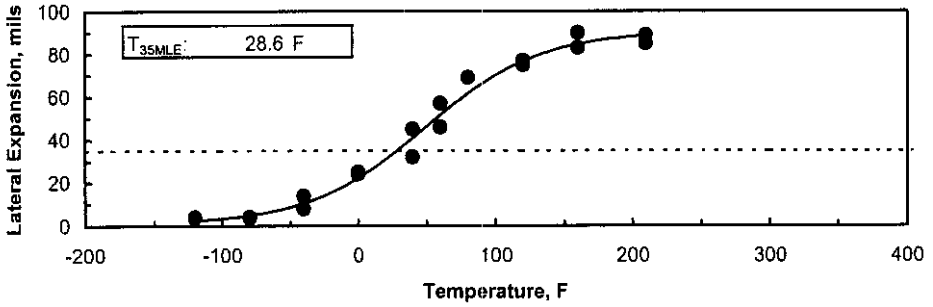
Specimen No.	Test Temp. (°F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
7A3	-80	4.5	2	0
7AL	-40	7	6	5
7AA	-40	8	7	5
7AK	0	14.5	14	15
7A1	0	16.5	17	15
7A2	40	25	26	25
7A7	40	38	33	25
7AC	80	85	68	50
7A5	80	81.5	63	40
7A4	120	110	78	70
7AJ	120	116	84	70
7AD	160	131	85	90
7A6	160	139	91	95
7AB	210	144.5	91	100
7AE	210	153	84	100

Table D-14. Hyperbolic Tangent Curve Fit Coefficients for Arkansas Nuclear One Unit 2 Standard Reference Material

Weld Metal	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 73.9 B: 71.7 C: 62.9 T0: 76.4	A: 45.2 B: 44.2 C: 57.3 T0: 55.4	A: 50.0 B: 50.0 C: 77.5 T0: 83.3

Figure D-1. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation
- Refitted Using Hyperbolic Tangent Curve-Fitting Method -





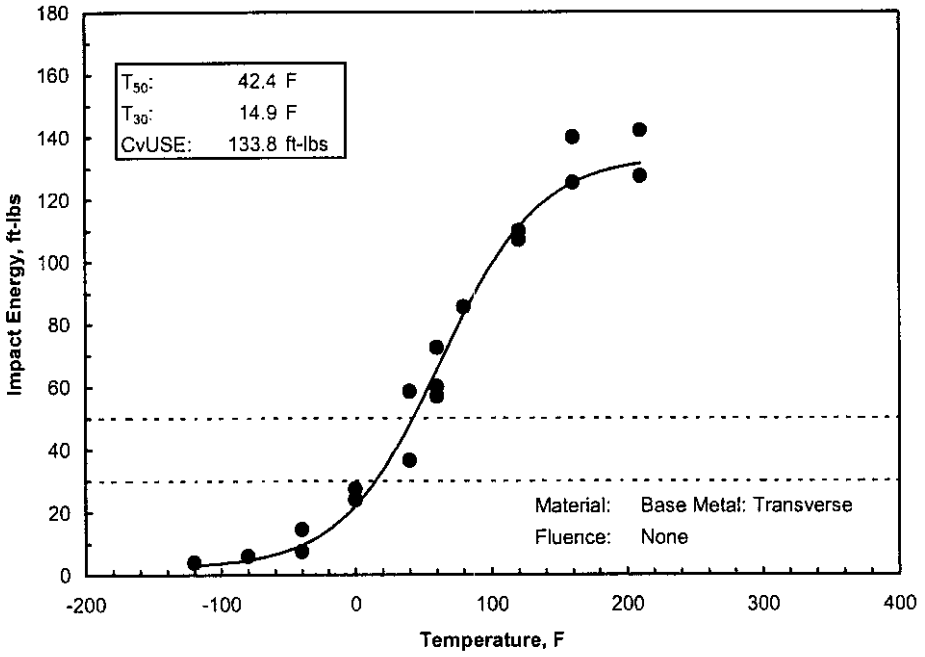
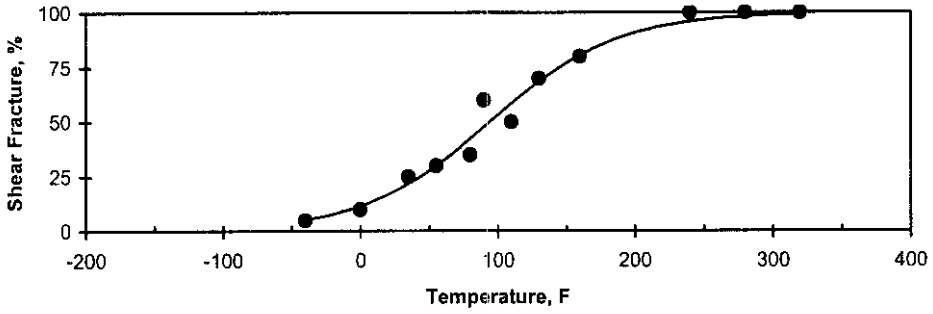
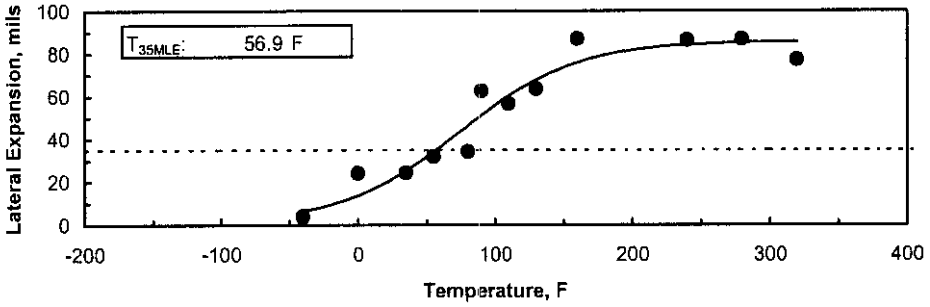


Figure D-2. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.33 x 10^{18} n/cm^2 (E>1.0 MeV) Transverse Orientation - Refitted Using Hyperbolic Tangent Curve-Fitting Method -





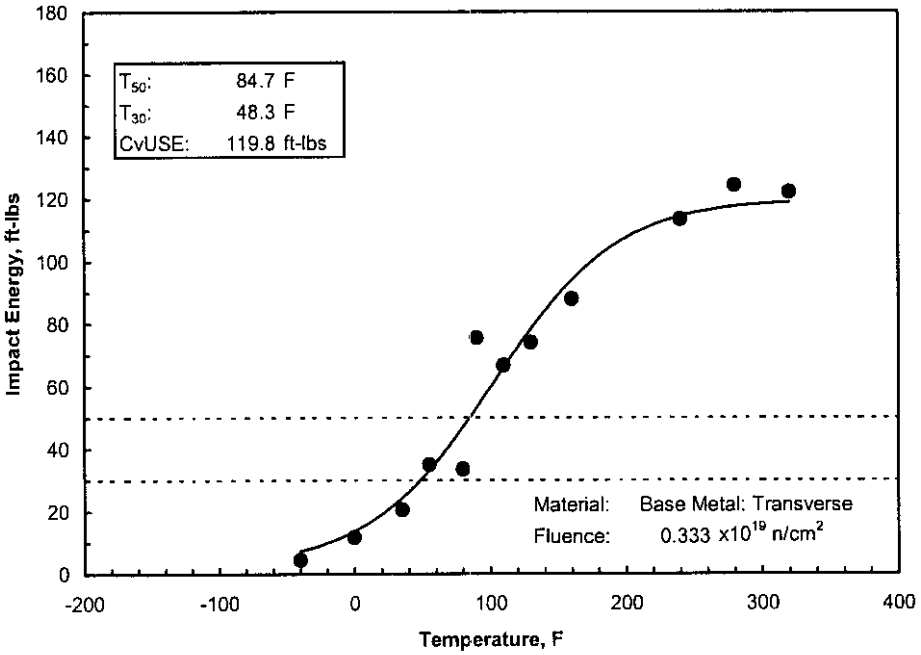
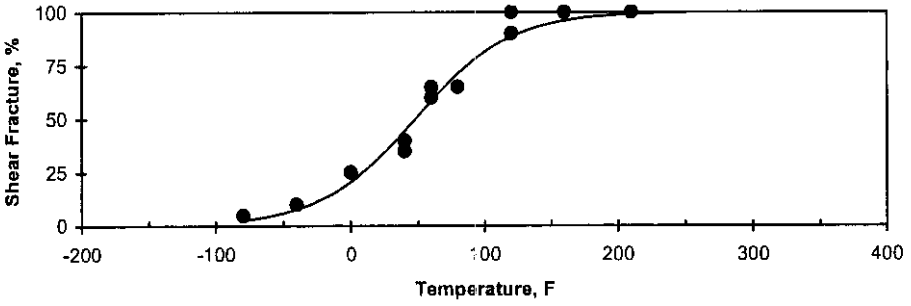
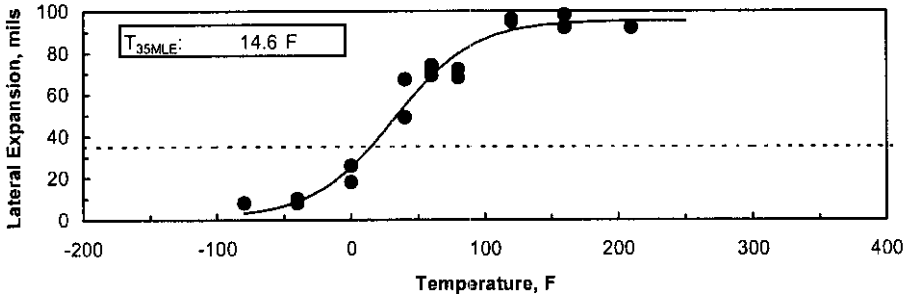


Figure D-3. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Base Metal Plate C-8009-3, Heat C8182-2, Longitudinal Orientation
- Refitted Using Hyperbolic Tangent Curve-Fitting Method -





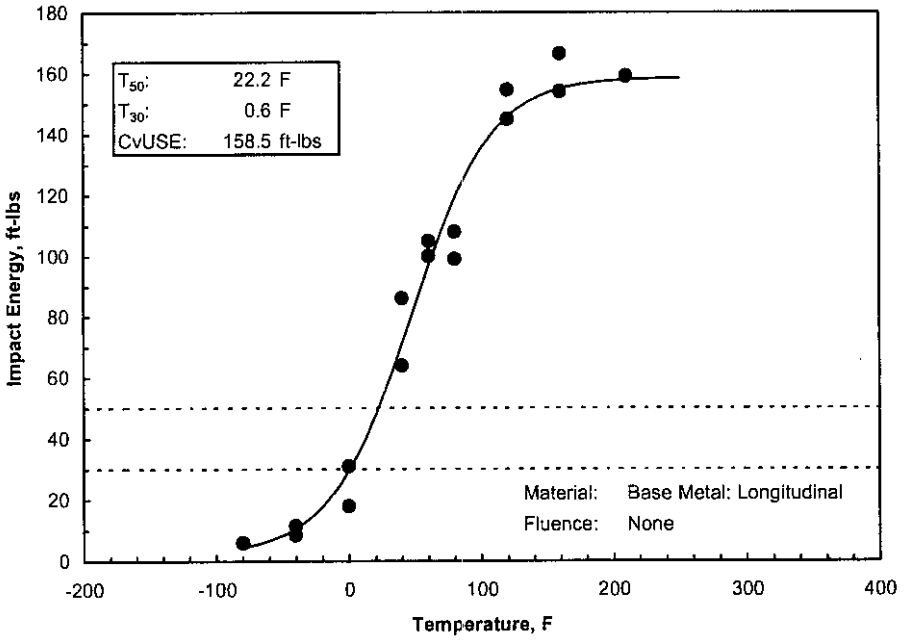
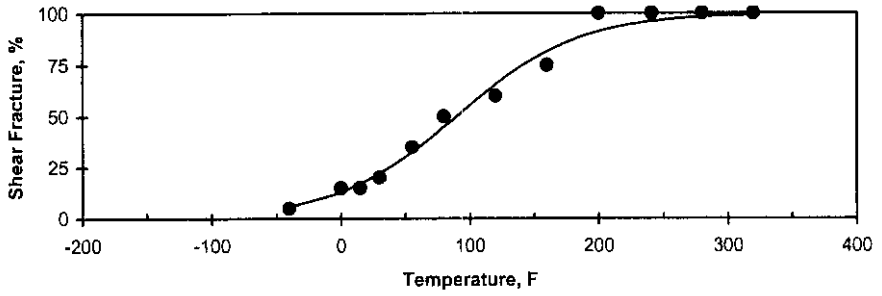
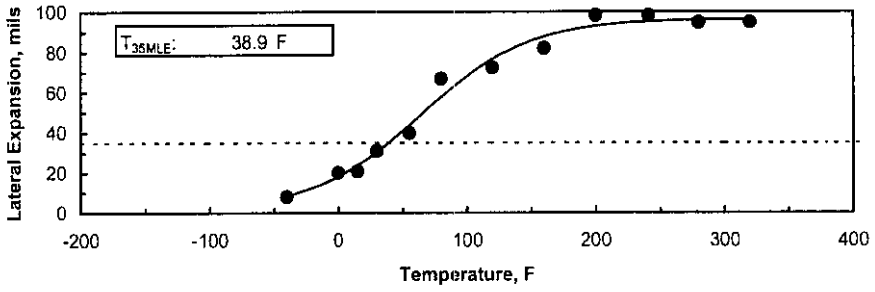


Figure D-4. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Base Metal Plate C-8009-3, Heat C8182-2, Irradiated to 3.41 x 10^{18} n/cm^2 (E>1.0 MeV) Longitudinal Orientation - Refitted Using Hyperbolic Tangent Curve-Fitting Method -





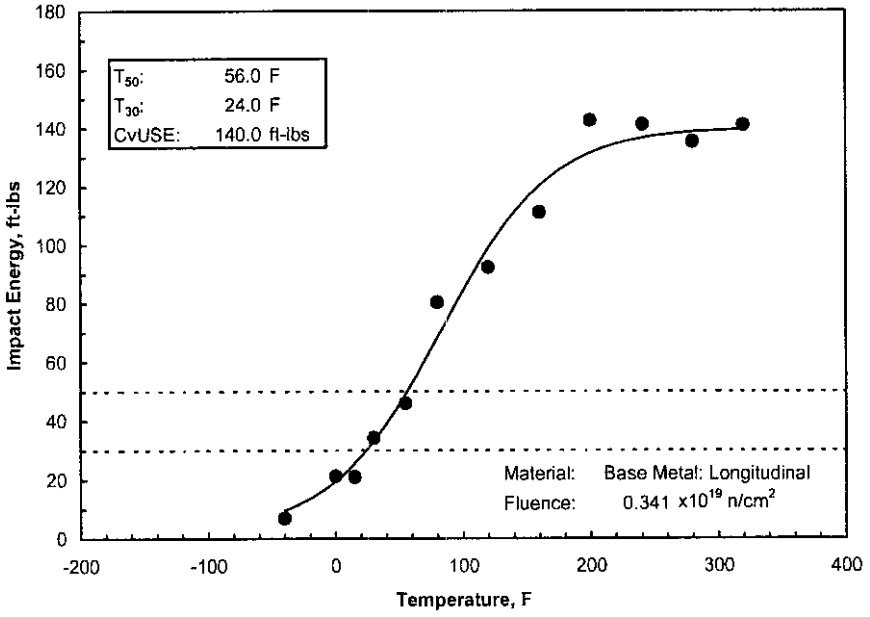


Figure D-5. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Weld Metal, Heat 10120 - Refitted Using Hyperbolic Tangent Curve-Fitting Method -

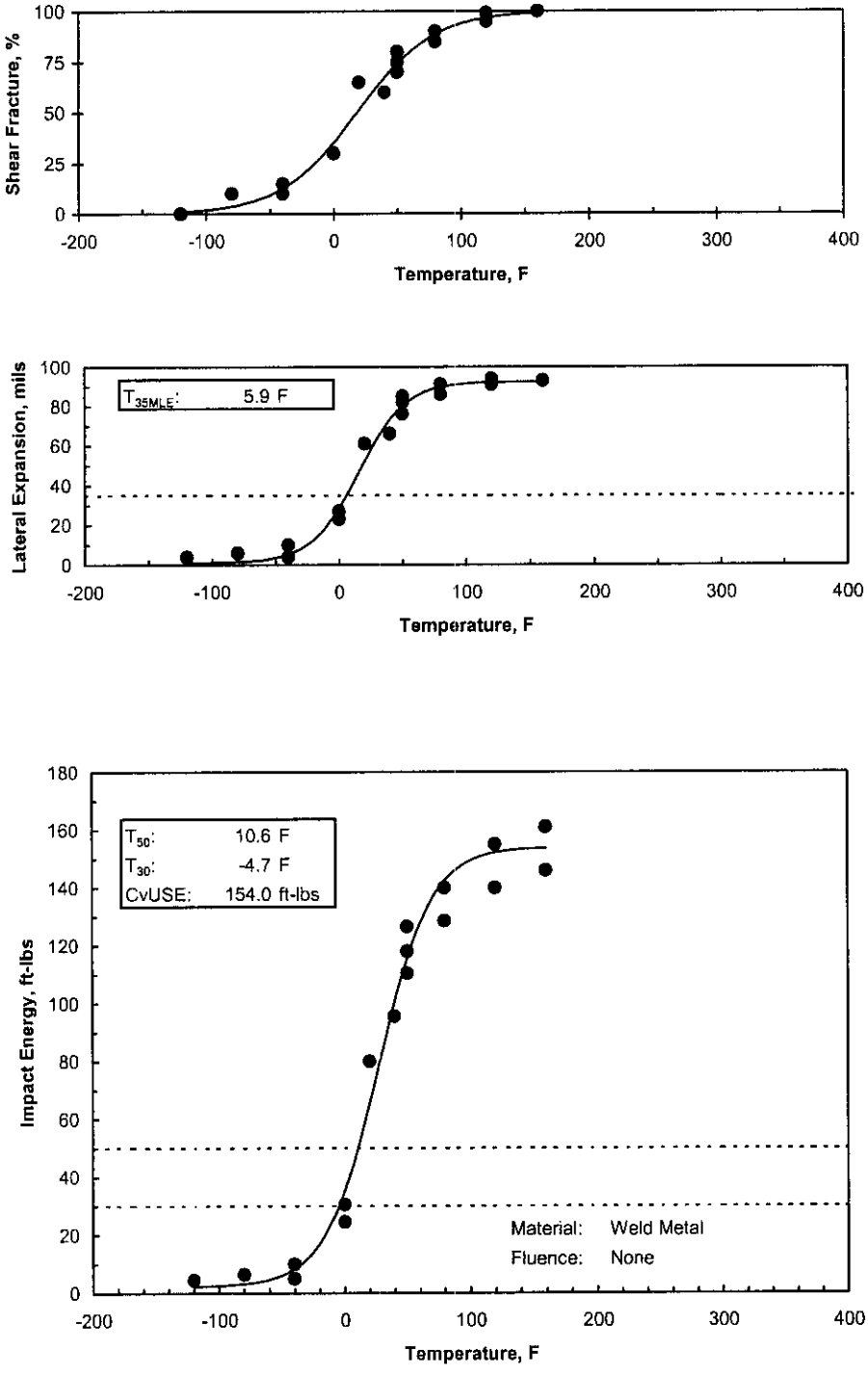
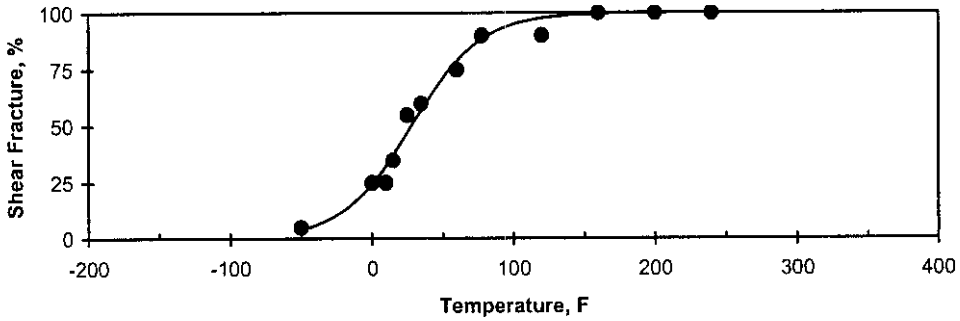
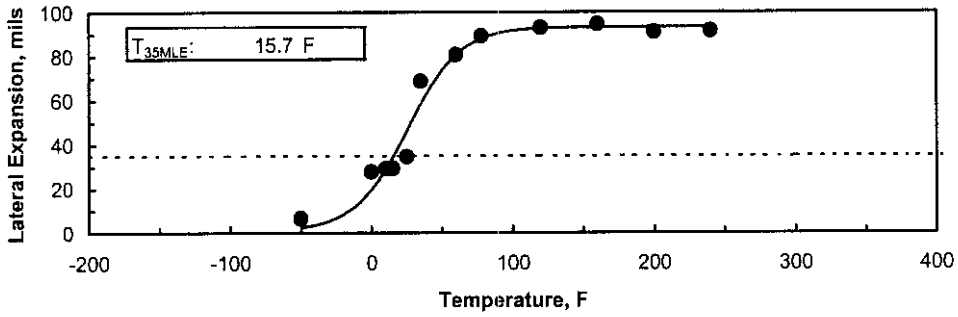


Figure D-6. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Weld Metal, Heat 10120 Irradiated to 3.34 x 10^{18} n/cm^2 (E>1.0 MeV) - Refitted Using Hyperbolic Tangent Curve-Fitting Method -





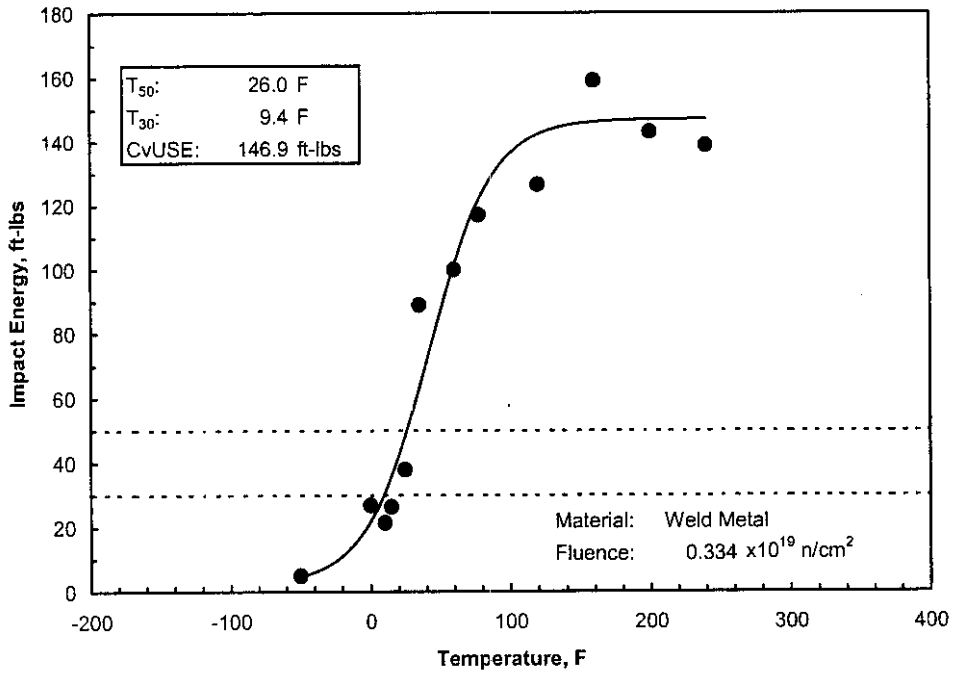
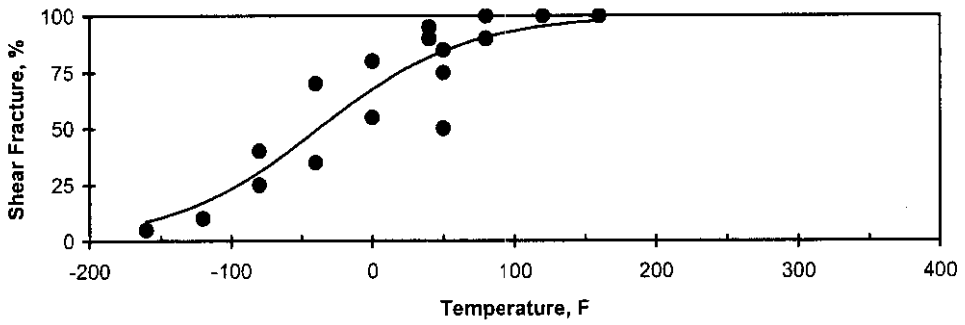
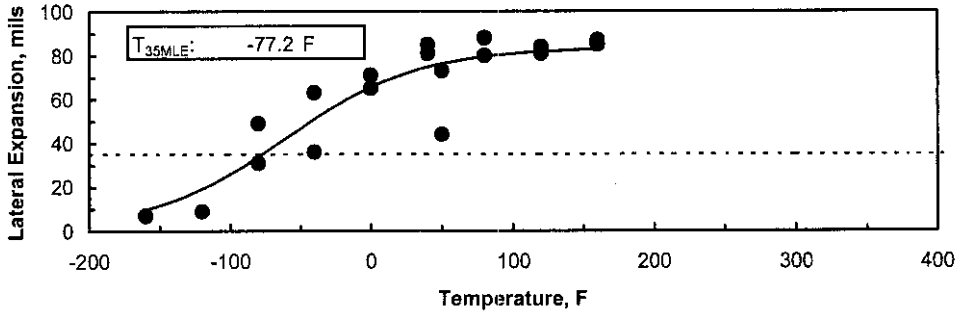


Figure D-7. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Heat-Affected-Zone Material - Refitted Using Hyperbolic Tangent Curve-Fitting Method -





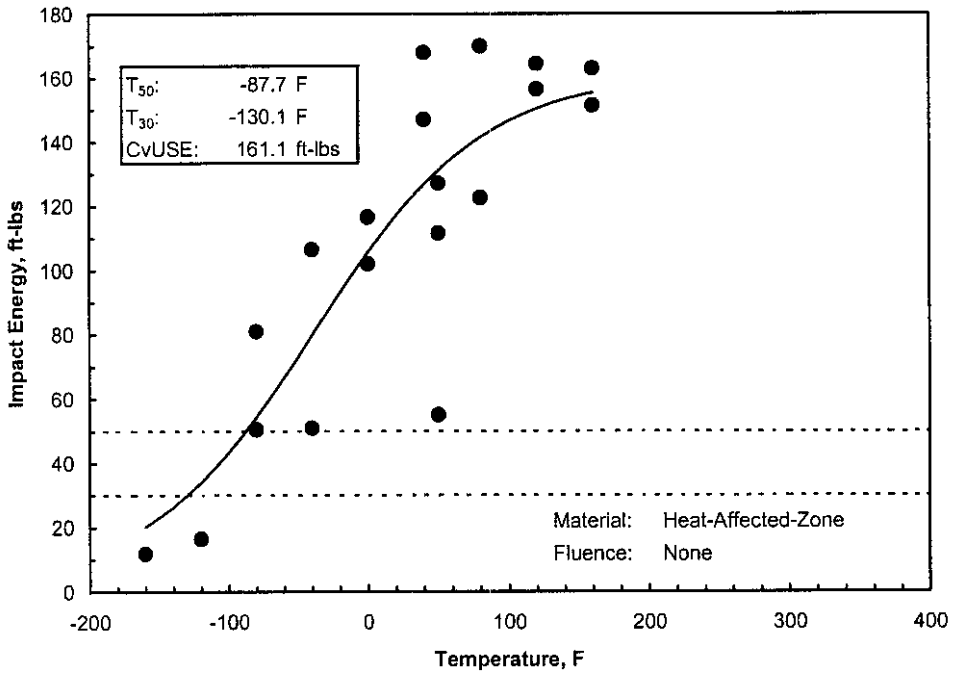
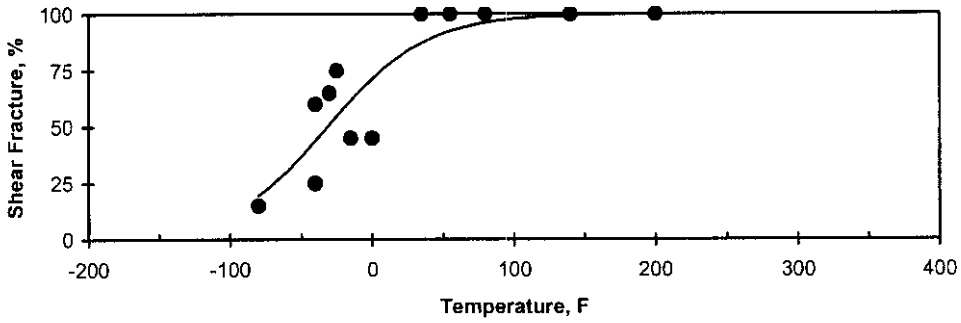
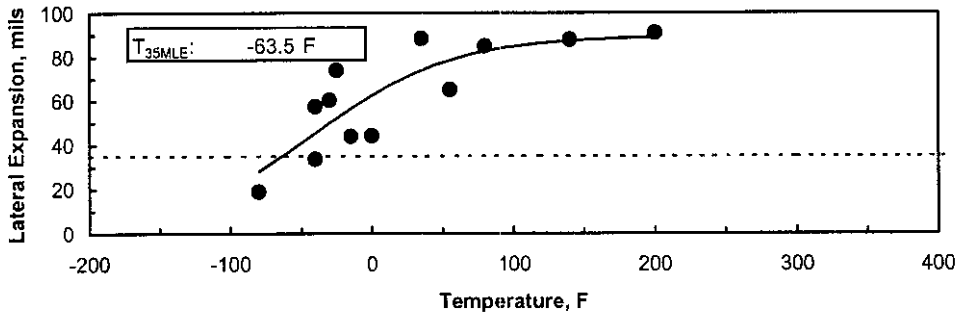


Figure D-8. Arkansas Nuclear One Unit 2 W-97 Capsule Surveillance Charpy Impact Data for Heat-Affected-Zone Material, Irradiated to 3.49 x 10^{18} n/cm^2 (E>1.0 MeV) - Refitted Using Hyperbolic Tangent Curve-Fitting Method -





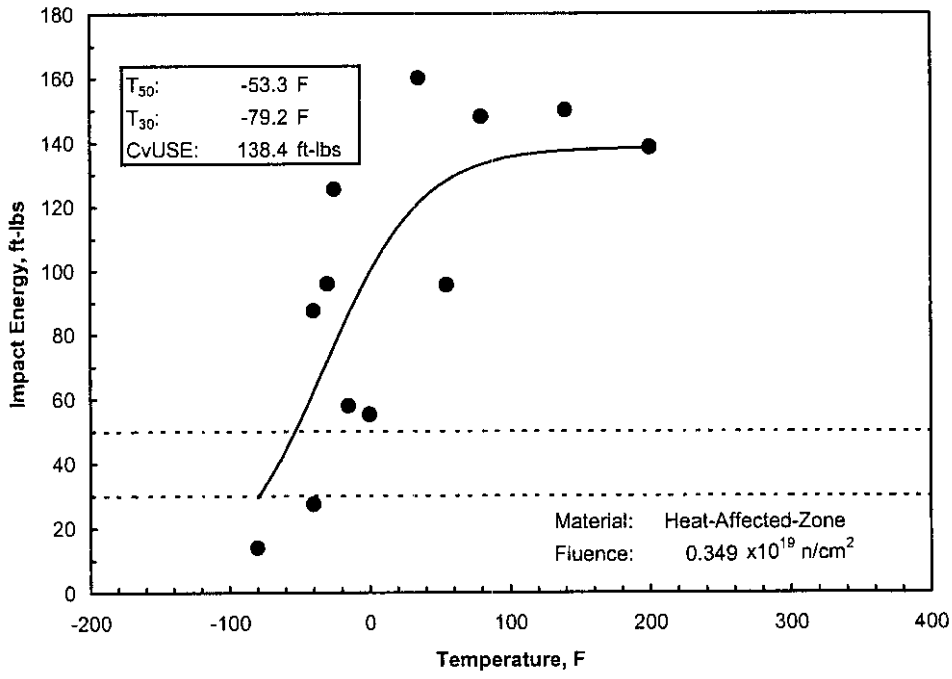
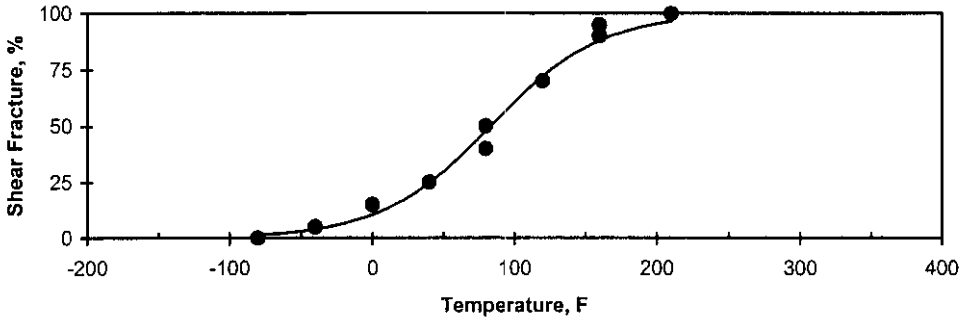
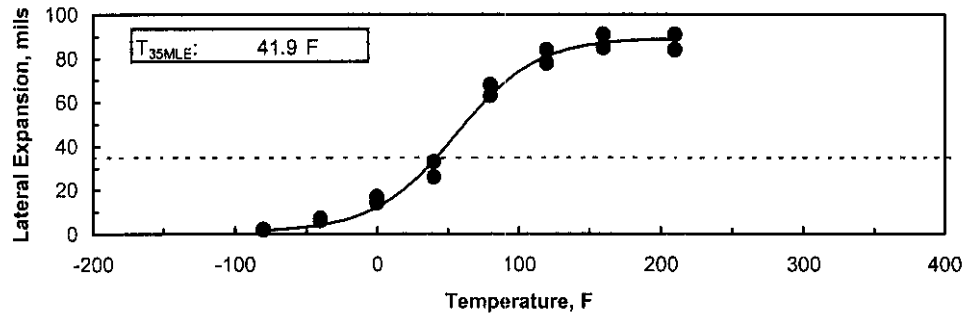
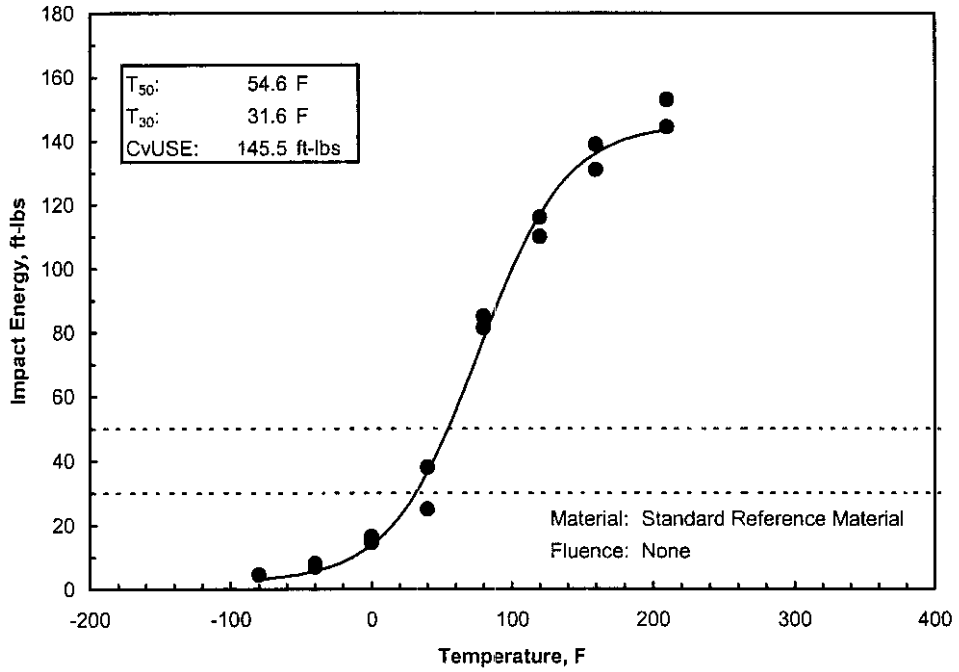


Figure D-9. Unirradiated Surveillance Charpy V-Notch Impact Data for Arkansas Nuclear One Unit 2, Standard Reference Material - Refitted Using Hyperbolic Tangent Curve-Fitting Method -







APPENDIX E

Charpy V-Notch Shift Comparison: Hand-Drawn Curve Fitting vs. Hyperbolic Tangent Curve Fitting

Table E-1. Comparison of Curve Fit Transition Temperature Shifts for Arkansas Nuclear One Unit 2 Surveillance Material, Base Metal Plate C-8009-3, Heat C8182-2, Transverse Orientation

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	30 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	10	---	15	---
97°	0.333	60	50	48	33

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	50 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	40	---	42	---
97°	0.333	90	50	85	43

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	35 MLE Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	30	---	29	---
97°	0.333	57	27	57	28

Table E-2. Comparison of Curve Fit Transition Temperature Shifts for Arkansas Nuclear One Unit 2 Surveillance Material, Base Metal Plate C-8009-3, Heat C8182-2, Longitudinal Orientation

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	30 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	3	---	1	---
97°	0.341	24	21	24	23

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	50 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	24	---	22	---
97°	0.341	57	33	56	34

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	35 MLE Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	20	---	15	---
97°	0.341	39	19	39	24

Table E-3. Comparison of Curve Fit Transition Temperature Shifts for Arkansas Nuclear One Unit 2 Surveillance Material, Weld Metal, Heat 10120

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	30 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	0	---	-5	---
97°	0.334	10	10	9	14

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	50 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	14	---	11	---
97°	0.334	25	11	26	15

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	35 MLE Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	10	---	6	---
97°	0.334	20	10	16	10

Table E-4. Comparison of Curve Fit Transition Temperature Shifts for Arkansas Nuclear One Unit 2 Surveillance Material, Heat-Affected-Zone Material

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	30 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	-80	---	-130	---
97°	0.349	-80	0	-79	51

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	50 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	-50	---	-88	---
97°	0.349	-52	-2	-53	35

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	35 MLE Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	-63	---	-77	---
97°	0.349	-63	0	-63	14

APPENDIX F

Fluence Analysis Methodology

The primary tool used in the determination of the flux and fluence exposure to the surveillance capsule dosimeters is the two-dimensional discrete ordinates transport code DORT.[F-3]

The ANO Unit 2 capsule was located at 14.0° (off of the major axis) for cycles 1 through 14. The power distributions in the 14 irradiation cycles were symmetric both in θ and Z. That is, the axial power shape is roughly the same for any angle and, conversely, that the azimuthal power shape is the same for any height. This means that the neutron flux at some point (R, θ, Z) can be considered to be a separable function of (R, θ) and (R, Z). Therefore, the cycle 1-14 irradiations can be modeled using the standard Framatome ANP synthesis procedures. [F-1]

Figure F–1 depicts the analytical procedure that is used to determine the fluence accumulated over each irradiation period. As shown in the figure, the analysis is divided into seven tasks: (1) generation of the neutron source, (2) development of the DORT geometry models, (3) calculation of the macroscopic material cross sections, (4) synthesis of the results, and (5-7) estimation of the calculational bias, the calculational uncertainty, and the final fluence. Each of these tasks is discussed in greater detail in the following sections.

F.1. Generation of the Neutron Source

The time-averaged space and energy-dependent neutron sources for cycles 1-14 were calculated using the SORREL[F-4] code. The effects of burnup on the spatial distribution of the neutron source were accounted for by calculating the cycle average fission spectrum for each fissile isotope on an assembly-by-assembly basis, and by determining the cycle-average specific neutron emission rate. This data was then used with the normalized time weighted average pin-by-pin relative power density (RPD) distribution to determine the space and energy-dependent neutron source. The azimuthally averaged, time averaged axial power shape in the peripheral assemblies was used with the fission spectrum of the peripheral assemblies to determine the neutron source for the axial DORT run. These two neutron source distributions were input to DORT as indicated in Figure F–1. Two separate sources (1-9 and 10-14) were developed in order to account for a change in the average reactor coolant system temperature that occurred between cycles 9 and 10.

F.2. Development of the Geometrical Models

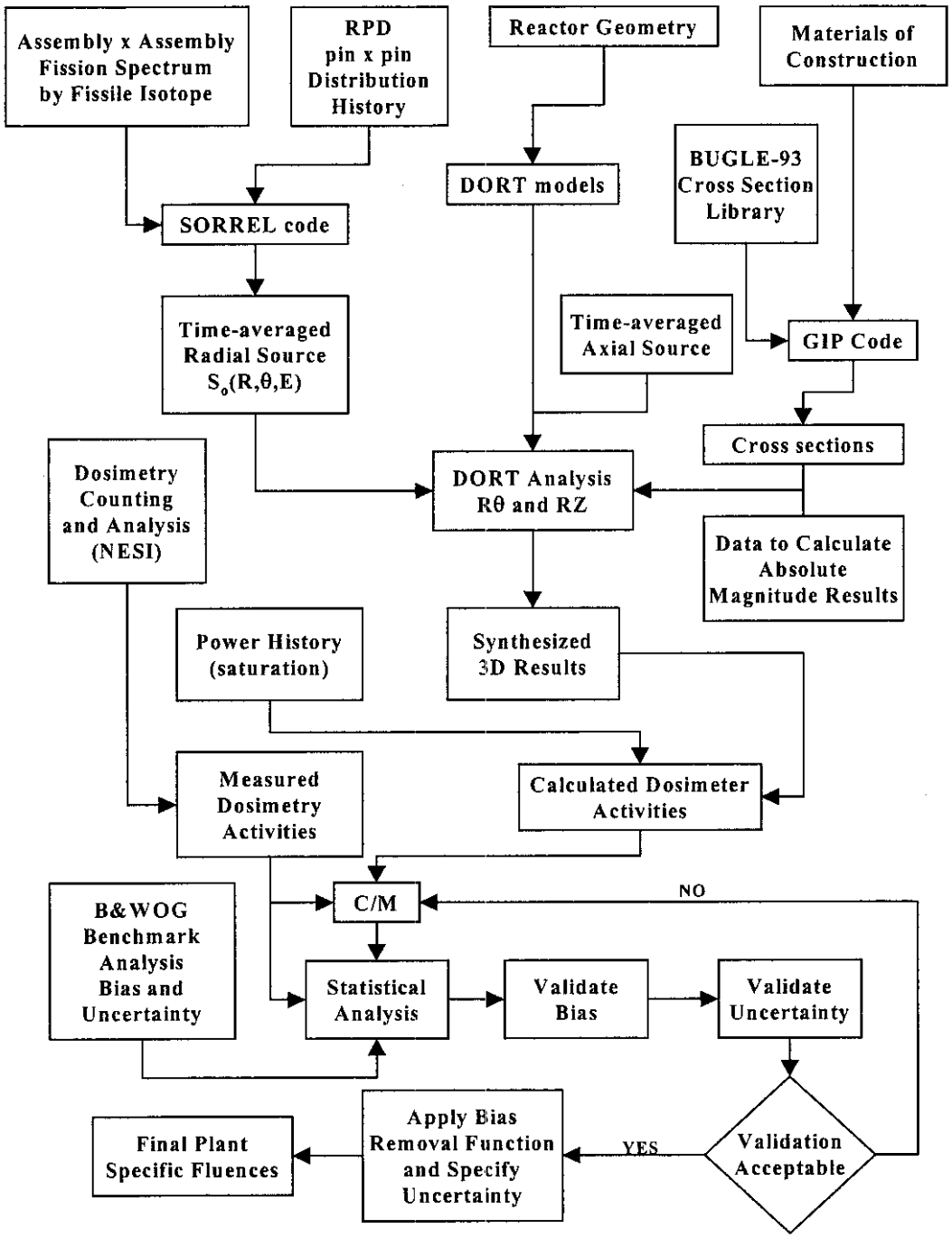
The system geometry models for the mid-plane (R, θ) DORT were developed using standard Framatome ANP interval size and configuration guidelines. The Rθ model for the cycles 1-9 and 10-14 analysis extends radially from the center of the core to the outer surface of the

pressure vessel, and azimuthally from the major axis to 45°. The axial model extended from 35 cm below the active core region to 35 cm above the active core region. The geometrical models either met or exceeded all guidance criteria concerning interval size that are provided in Regulatory Guide 1.190.[F-2] In all cases, cold dimensions were used. The geometry models were input to the DORT code as indicated in Figure F–1. These models will be used in all subsequent pressure-temperature curve analyses that may be performed by Framatome ANP for ANO Unit 2.

F.3. Calculation of Macroscopic Material Cross Sections

In accordance with Regulatory Guide 1.190, the BUGLE-93[F-5] cross section library was used. The GIP code[F-6] was used to calculate the macroscopic energy-dependent cross sections for all materials used in the analysis – from the core out through the cavity and into the concrete and from core plate to core plate. The ENDF/B-VI dosimeter reaction cross sections were used to generate the response functions that were used to calculate the DORT-calculated "saturated" specific activities.

Figure F-1. Fluence Analysis Methodology for ANO Unit 2 Surveillance Capsule



F.4. DORT Analyses

The cross sections, geometry, and appropriate source were combined to create a set of DORT models (Rθ and RZ) for the cycles 1-9 and 10-14 analyses. Each Rθ DORT run utilized a cross section Legendre expansion of three (P_3), seventy directions (S_{16}), with the appropriate boundary conditions. The RZ models used a cross section Legendre expansion of three (P_3), forty-eight directions (S_8), with the appropriate boundary conditions. A theta-weighted flux extrapolation model was used, and all other requirements of Regulatory Guide 1.190 that relate to the various DORT parameters were either met or exceeded for all DORT runs.

F.5. Synthesized Three Dimensional Results

The DORT analyses produce two sets of two-dimensional flux distributions, one for a vertical cylinder and one for the radial plane for each set of dosimetry. The vertical cylinder, which will be referred to as the RZ plane, is defined as the plane bounded 35 cm above and below the active core region and radially by the center of the core to the outside surface of the reactor pressure vessel. The horizontal plane, referred to as the Rθ plane, is defined as the plane bounded radially by the center of the core and the outside surface of the pressure vessel, and azimuthally by the major axis and the adjacent 45° radius. The vessel flux, however, varies significantly in all three cylindrical-coordinate directions (R, θ , Z). This means that if a point of interest is outside the boundaries of both the R-Z DORT and the R-θ DORT, the true flux cannot be determined from either DORT run. Under the assumption that the three-dimensional flux is a separable function, [F-1] both two-dimensional data sets were mathematically combined to estimate the flux at all three-dimensional points (R, θ, Z) of interest. The synthesis procedure outlined in Regulatory Guide 1.190 is identical to the basis used for the Framatome ANP flux-synthesis process.

F.6. Calculated Activities and Measured Activities

The calculated activities for each dosimeter type "d" for each irradiation period were determined using the following equation:

$$C_d = \sum_{g=1}^{G} \phi_g(\bar{r}_d) \times RF_g^d \times B_d \times NSF \qquad (1)$$

where

C_d ... calculated specific activity for dosimeter "d" in μCi of product isotope per gram of target isotope

$\phi_g(\bar{r}_d)$ ... three dimensional flux for dosimeter "d" at position $\bar{r}_d$ for energy group "g"

RF_g^d ... dosimeter response function for dosimeter "d" and energy group "g"

B_d ... bias correction factors for dosimeter "d"

NSF ... non-saturation correction factor (NSF).

For this analysis, two separate sets of activities will be calculated, therefore a combination of the calculated activities must be performed. The EOC 14 total calculated activity, $C^{14}{}_i$, will be accomplished using the equation, for dosimeter "i",

$$\left(C^{14}{}_{DORT}\right)_i = \left(C_{d(1-9)}\right)_i + \left(C_{d(10-14)}\right)_i \qquad (2)$$

Each calculated activity in equation 2, $C_{d(cycle)}$, is calculated using equation 1, however each set of data, i.e. 1-9 and 10-14, will be calculated using a cycle specific $NSF_{(cycle)}$ factor.

The bias correction factors (B_d) in the specific activity calculation above are listed in Table F–1.

Table F-1. Bias Correction Factors

Dosimeter Type	Bias
Activation	Short Half Life
Fission	Photofission
	Impurities

A photofission factor was applied to correct for the fact that some of the ^{137}Cs atoms present in the dosimeter were produced by (γ, f) reactions and were not accounted for in DORT analysis. Likewise, an impurity factor was included to account for U-235 content in the U-238 dosimetry. The short half life was insignificant and therefore was not applied.

F.7. C/M Ratios

The following explanations will define the meanings of the terms "measurements" (M) and "calculations" (C) as used in this analysis: [F-1]

- Measurements: The meaning of the term "measurements" as used by Framatome ANP is the measurement of the physical quantity of the dosimeter (specific activity) that responded to the neutron fluence, not to the "measured fluence." For the example of an iron dosimeter, a reference to the measurements means the specific activity of ^{54}Mn in $\mu Ci/g$, which is the product isotope of the dosimeter reaction:

$$^{54}Fe + {}^{1}_{0}n \rightarrow {}^{54}Mn + p^{+}$$

- Calculations: The calculational methodology produces two primary results – the calculated dosimeter activities and the neutron flux at all points of interest. The meaning of the term "calculations" as used by Framatome ANP is the calculated dosimeter activity. The calculated activities are determined in such a way that they are directly comparable to the measurement values, but without recourse to the measurements. That is, the calculated values are determined by the DORT calculation and are directly comparable to the measurement values. ENDF/B-VI based dosimeter reaction cross sections[F-7] and response functions were used in determining the calculated values for each individual dosimeter. In summary, it should be stressed that the calculation values in the Framatome ANP approach[F-1] are independent of the measurement values.

F.8. Uncertainty

The ANO Unit 2 cycles 1 through 14 fluence predictions are based on the methodology described in the Framatome ANP "Fluence and Uncertainty Methodologies" topical report, BAW-2241P-A.[F-1] The time-averaged fluxes, and thereby the fluences throughout the reactor and vessel, are calculated with the DORT discrete ordinates computer code using three-dimensional synthesis methods. The basic theory for synthesis is described in Section 3.0 of the topical and the DORT three-dimensional synthesis results are the bases for the fluence predictions using the Framatome ANP "Semi-Analytical" (calculational) methodology.

The uncertainties in the ANO Unit 2 fluence values have been evaluated to ensure that the greater than 1.0 MeV calculated fluence values are accurate (with no discernible bias) and have a mean standard deviation that is consistent with the Framatome ANP benchmark database of uncertainties. Consistency between the fluence uncertainties in the updated calculations for ANO Unit 2 cycles 1-14 and those in the Framatome ANP benchmark database ensures that the vessel fluence predictions are consistent with the 10 CFR 50.61, Pressurized Thermal Shock (PTS) screening criteria and the Regulatory Guide 1.99[F-8] embrittlement evaluations.

The verification of the fluence uncertainty for the ANO Unit 2 reactor includes:

- estimating the uncertainties in the cycles 1 through 14 dosimetry measurements,
- estimating the uncertainties in the cycles 1 through 14 benchmark comparison of calculations to measurements,
- estimating the uncertainties in the cycles 1 through 14 pressure vessel fluence, and
- determining if the specific measurement and benchmark uncertainties for cycles 1–14 are consistent with the Framatome ANP database of generic uncertainties in the measurements and calculations.

The embrittlement evaluations in Regulatory Guide 1.99 and those in 10 CFR 50.61 for the PTS screening criteria apply a margin term to the reference temperatures. The margin term includes the product of a confidence factor of 2.0 and the mean embrittlement standard deviation. The factor of 2.0 implies a very high level of confidence in the fluence uncertainty as well as the uncertainty in the other variables contributing to the embrittlement. The dosimeter measurements from the ANO Unit 2 analysis would not directly support this high level of confidence. However, the dosimeter measurement uncertainties are consistent with the Framatome ANP database. Therefore, the calculational uncertainties in the updated fluence predictions for ANO Unit 2 are supported by 728 additional dosimeter measurements and thirty-

nine benchmark comparisons of calculations to measurements as shown in Appendix A of the topical.[F-1] The calculational uncertainties are also supported by the fluence sensitivity evaluation of the uncertainties in the physical and operational parameters, which are included in the vessel fluence uncertainty.[F-1] The dosimetry measurements and benchmarks, as well as the fluence sensitivity analyses in the topical are sufficient to support a 95 percent confidence level, with a confidence factor of ±2.0, in the fluence results from the "Semi-Analytical" methodology.

The Framatome ANP generic uncertainty in the dosimetry measurements has been determined to be unbiased and has an estimated standard deviation of 7.0 percent for the qualified set of dosimeters. The ANO Unit 2 cycles 1-14 dosimetry measurement uncertainties were evaluated to determine if any biases were evident and to estimate the standard deviation. The dosimetry measurements were found to be appropriately calibrated to standards traceable to the National Institute of Standards and Technology and are thereby unbiased by definition. The mean measurement uncertainties associated with cycles 1-14 are as follows:

$$\sigma_M = 6.26\% .$$

This value was determined from Equation 7.6 in the topical and indicates that there is consistency with the Framatome ANP database. Consequently, when the Framatome ANP database is updated, the ANO Unit 2 cycles 1-14 dosimetry measurement uncertainties may be combined with the other 728 dosimeters. Since the cycles 1-14 measurements are consistent with the Framatome ANP database, it is estimated that the ANO Unit 2 dosimeter measurement uncertainty may be represented by the Framatome ANP database standard deviation of 7.0 percent. Based on the Framatome ANP database, there appears to be a 95 percent level of confidence that 95 percent of the ANO Unit 2 dosimetry measurements, for fluence reactions above 1.0 MeV, are within ±14.2 percent of the true values.

The Framatome ANP generic uncertainty for benchmark comparisons of dosimetry calculations relative to the measurements indicates that any benchmark bias in the greater than 1.0 MeV results is too small to be uniquely identified. The estimated standard deviation between the calculations and measurements is 9.9 percent. This implies that the root mean square deviation between the Framatome ANP calculations of the ANO dosimetry and the measurements should be approximately 9.9 percent in general and bounded by ±20.04 percent for a 95 percent confidence interval with thirty-nine independent benchmarks.

The weighted mean values of the ratio of calculated dosimeter activities to measurements (C/M) for cycles 1-14 have been statistically evaluated using Equation 7.15 from the topical. The standard deviation in the benchmark comparisons is as follows:

$$\sigma_{C/M} = 0.149\% .$$

This standard deviation indicates that the benchmark comparisons are consistent with the Framatome ANP database. Consequently, when the Framatome ANP database is updated, the cycles 1-14 benchmark uncertainties may be included with the other thirty-nine benchmark uncertainties in the topical. The consistency between the cycles 1-14 benchmark uncertainties and those in the Framatome ANP database indicates that the ANO Unit 2 fluence calculations for cycles 1-14 have no discernible bias in the greater than 1.0 MeV fluence values. In addition, the consistency indicates that the fluence values can be represented by the Framatome ANP reference set which includes a calculational standard deviation of 7.0 percent at dosimetry locations. That is:

Table F-2. Calculational Fluence Uncertainties

Type of Calculation	Uncertainty (%)	
	Standard Deviation (σ)	**95% / 95% Confidence ($\approx \pm 2\sigma$)**
Capsule	7.0	14.2
Pressure Vessel (maximum location)	10.0	20.0
Pressure Vessel (extrapolation)	11.4	22.8

F.9. References

F-1. Worsham, J. R., et al., "*Fluence and Uncertainty Methodologies*," BAW-2241P-A, Revision 1, Framatome ANP, Lynchburg, Virginia, April 1999.

F-2. U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, "*Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*," March 2001.

F-3. Rutherford, M. A., N. M. Hassan, et. al., Eds., *"DORT, Two Dimensional Discrete Ordinates Transport Cod,*." BWNT-TM-107, Framatome ANP, Lynchburg, Virginia, May 1995.*

F-4. Hassler, L. A., and N. M. Hassan, *"SORREL, DOT Input Generation Code User's Manual*," NPGD-TM-427, Revision 10, Framatome ANP, Lynchburg, Virginia, May 2001.*

F-5. Ingersoll, D. T., et. al., "*BUGLE-93, Production and Testing of the VITAMIN-B6 Fine Group and the BUGLE-93 Broad Group Neutron/photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data*," ORNL-DLC-175, Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, April 1994.

F-6. Hassler, L. A. and N. M. Hassan, "*GIP User's Manual for B&W Version, Group Organized Cross Section Input Program*," NPGD-TM-456, Revision 11, Framatome ANP, Lynchburg, Virginia, August 1994.*

F-7. Worsham, J. R., "*BUGLE-93 Response Functions*," FRA-ANP Document Number 32-1232719-00, Revision 0, Framatome ANP, Lynchburg, Virginia, June 1995.*

F-8. U.S. Nuclear Regulatory Commission, *"Radiation Embrittlement of Reactor Vessel Materials*," Regulatory Guide 1.99, Revision 2, May 1998.

*- Available from Framatome ANP, Inc., Lynchburg, Virginia