WOLF CREEK
NUCLEAR OPERATING CORPORATION

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APR - 8 2003

RA 03-0041

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket No. 50-482: Submittal of Test Results for the Withdrawal of Surveillance Capsule X from the Reactor Vessel

Gentlemen:

Surveillance capsule X was withdrawn from the Wolf Creek Generating Station reactor vessel on April 12, 2002, at the end of Refuel 12. The capsule was withdrawn to determine the vessel integrity after being subjected to neutron radiation exposure equivalent to the peak end-of-life (extended) fluence on the inside surface of the vessel. Capsule X reached the peak vessel surface fluence equivalent to 54 effective full power years (EFPY) after an actual exposure of 13.83 EFPY, since the lead factor for the capsule is 4.3.

Appendix H to 10 CFR 50 requires that a report be submitted to the Nuclear Regulatory Commission for each capsule withdrawn. The report must describe the capsule and the test results for the capsule. The enclosure provides Westinghouse report WCAP-16028 Revision 0 for the analysis of capsule X.

There are no commitments contained in this correspondence. If you have any questions concerning this matter, please contact me at (620) 364-4038, or Ms. Jennifer Yunk at (620) 364-4272.

Very truly yours,

Karl A (Tony) Harris

Karl A. (Tony) Harris

KAH/rlg

Enclosure

A008

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Revision 0

March 2003

Analysis of Capsule X from Wolf Creek Nuclear Operating Corporation, Wolf Creek Reactor Vessel Radiation Surveillance Program

Westinghouse

WCAP-16028, Revision 0

Analysis of Capsule X from the Wolf Creek Nuclear Operating Corporation, Wolf Creek Reactor Vessel Radiation Surveillance Program

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March 2003

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PREFACE

This report has been technically reviewed and verified by:

Reviewer:

Sections 1 through 5, 7, 8, Appendices B, C and D	A.R. Rawluszki
Section 6 and Appendix A	D. M. Chapman

EXECUTIVE SUMMARY

The purpose of this report is to document the results of the testing of surveillance Capsule X from Wolf Creek Capsule X was removed at 13.8 EFPY and post irradiation mechanical tests of the Charpy V-notch and tensile specimens were performed. A fluence evaluation utilizing the recently released neutron transport and dosimetry cross-section libraries was derived from the ENDF/B-VI data-base. Capsule X received a fluence of 3.49×10^{19} n/cm^2 after irradiation to 13.8 EFPY. The peak clad/base metal interface vessel fluence after 13.8 EFPY of plant operation was 8.1×10^{18} n/cm^2.

This evaluation lead to the following conclusions: 1) The measured 30 ft-lb shift in transition temperature values of the lower shell plate R2508-3 contained in capsule X (longitudinal & transverse) is less than the Regulatory Guide 1.99, Revision 2[1], predictions. 2) The measured 30 ft-lb shift in transition temperature values of the weld metal contained in capsule X is greater than the Regulatory Guide 1.99, Revision 2, predictions. However, the shift value is less than two sigma allowance by Regulatory Guide 1.99, Revision 2. 3) The measured percent decrease in upper shelf energy for all the surveillance materials of Capsules X contained in the Wolf Creek surveillance program are less than the Regulatory Guide 1.99, Revision 2 predictions. 4) All beltline materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are predicted to maintain an upper shelf energy greater than 50 ft-lb throughout the extended life of the vessel (54 EFPY) as required by 10CFR50, Appendix G [2]. 5) The Wolf Creek surveillance data was found to be credible. This evaluation can be found in Appendix D.

Lastly, a brief summary of the Charpy V-notch testing can be found in Section 1. All Charpy V-notch data was plotted using a symmetric hyperbolic tangent curve fitting program.

1 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance Capsule X, the fourth capsule removed and tested from the Wolf Creek reactor pressure vessel, led to the following conclusions:

- The Charpy V-notch data presented in WCAP-15078, Rev. 1[3] were based on a re-plot of all capsule data from WCAP-10015[4], WCAP-11553[5] and WCAP-13365, Rev. 1[6] using CVGRAPH, Version 4.1, which is a symmetric hyperbolic tangent curve-fitting program. The results presented are only for the Capsule X test results, which are also based on using CVGRAPH, Version 4.1. This report also shows the composite plots that show the results from the previous capsule. Appendix C presents the CVGRAPH, Version 4.1, Charpy V-notch plots and the program input data

- Capsule X received an average fast neutron fluence (E> 1.0 MeV) of 3.49×10^{19} n/cm^2 after 13.8 effective full power years (EFPY) of plant operation.

- Irradiation of the reactor vessel lower shell plate R2508-3 (heat number C4935-2) Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction (longitudinal orientation), resulted in an irradiated 30 ft-lb transition temperature of 36.11°F and an irradiated 50 ft-lb transition temperature of 67.75°F. This results in a 30 ft-lb transition temperature increase of 61.06°F and a 50 ft-lb transition temperature increase of 67.64°F for the longitudinal oriented specimens. See Table 5-9.

- Irradiation of the reactor vessel lower shell plate R2508-3 (heat number C4935-2) Charpy specimens, oriented with the longitudinal axis of the specimen perpendicular to the major working direction (transverse orientation), resulted in an irradiated 30 ft-lb transition temperature of 55.97°F and an irradiated 50 ft-lb transition temperature of 97.04°F. This results in a 30 ft-lb transition temperature increase of 53.96°F and a 50 ft-lb transition temperature increase of 62.71°F for the longitudinal oriented specimens. See Table 5-9.

- Irradiation of the weld metal (heat number 90146) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 10.66°F and an irradiated 50 ft-lb transition temperature of 54.73°F. This results in a 30 ft-lb transition temperature increase of 68.36°F and a 50 ft-lb transition temperature increase of 75.38°F. See Table 5-9.

- Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of –74.34°F and an irradiated 50 ft-lb transition temperature of –52.92°F. This results in a 30 ft-lb transition temperature increase of 69.66°F and a 50 ft-lb transition temperature increase of 61.08°F. See Table 5-9.

- The average upper shelf energy of the lower shell plate R2508-3 (longitudinal orientation) resulted in an average energy decrease of 6 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 142 ft-lb for the longitudinal oriented specimens. See Table 5-9.

- The average upper shelf energy of the lower shell plate R2508-3 (transverse orientation) resulted in no energy decrease after irradiation. This results in an irradiated average upper shelf energy of 94 ft-lb for the longitudinal oriented specimens. See Table 5-9.

- The average upper shelf energy of the weld metal Charpy specimens resulted in an average energy decrease of 7 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 93 ft-lb for the weld metal specimens. See Table 5-9.

- The average upper shelf energy of the weld HAZ metal Charpy specimens resulted in an average energy decrease of 26 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 135 ft-lb for the weld HAZ metal. See Table 5-9.

- A comparison, as presented in Table 5-10, of the Wolf Creek reactor vessel surveillance material test results with the Regulatory Guide 1.99, Revision 2[1] predictions led to the following conclusions:

  — The measured 30 ft-lb shift in transition temperature values of the lower shell plate R2508-3 contained in capsule X (longitudinal & transverse) are less than the Regulatory Guide 1.99, Revision 2, predictions.

  — The measured 30 ft-lb shift in transition temperature value of the weld metal contained in capsule X is greater than the Regulatory Guide 1.99, Revision 2, predictions. However, the shift value is less than two sigma allowance by Regulatory Guide 1.99, Revision 2.

  — The measured percent decrease in upper shelf energy for all the surveillance materials of Capsules X contained in the Wolf Creek surveillance program are less than the Regulatory Guide 1.99, Revision 2 predictions.

- All beltline materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are predicted to maintain an upper shelf energy greater than 50 ft-lb throughout the extended life of the vessel (54 EFPY) as required by 10CFR50, Appendix G [2].

- The calculated end-of-license (54 EFPY) neutron fluence (E> 1.0 MeV) at the core midplane for the Wolf Creek reactor vessel using the Regulatory Guide 1.99, Revision 2 attenuation formula (i e., Equation #3 in the guide) are as follows:

<u>Calculated:</u> Vessel inner radius* = 3.51×10^{19} n/cm^2

Vessel 1/4 thickness = 2.09×10^{19} n/cm^2

Vessel 3/4 thickness = 7.42×10^{18} n/cm^2

*Clad/base metal interface. (From Table 6-2)

2 INTRODUCTION

This report presents the results of the examination of Capsule X, the fourth capsule removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Wolf Creek Nuclear Operating Corporation Wolf Creek reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Wolf Creek Nuclear Operating Corporation Wolf Creek reactor pressure vessel materials was designed and recommended by the Westinghouse Electric Corporation. A description of the surveillance program and the pre-irradiation mechanical properties of the reactor vessel materials are presented in WCAP-10015, "Kansas Gas and Electric Company Wolf Creek Generating Station Unit No. 1 Reactor Vessel Radiation Surveillance Program"[4]. The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-79, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Capsule X was removed from the reactor after 13.8 EFPY of exposure and shipped to the Westinghouse Science and Technology Department Hot Cell Facility, where the post-irradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

This report summarizes the testing of and the post-irradiation data obtained from surveillance capsule X removed from the Wolf Creek Nuclear Operating Corporation Wolf Creek reactor vessel and discusses the analysis of the data.

3 BACKGROUND

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy, ferritic pressure vessel steels such as SA533 Grade B Class 1 (base material of the Wolf Creek reactor pressure vessel beltline) are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness during high-energy irradiation.

A method for ensuring the integrity of reactor pressure vessels has been presented in "Fracture Toughness Criteria for Protection Against Failure," Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code [8]. The method uses fracture mechanics concepts and is based on the reference nil-ductility transition temperature (RT_{NDT}).

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT per ASTM E-208[7]) or the temperature 60°F less than the 50 ft-lb (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented perpendicular (transverse) to the major working direction of the plate. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{Ic} curve) which appears in Appendix G to the ASME Code[8]. The K_{Ic} curve is a lower bound of static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{Ic} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined using these allowable stress intensity factors

RT_{NDT} and, in turn, the operating limits of nuclear power plants can be adjusted to account for the effects of radiation on the reactor vessel material properties. The changes in mechanical properties of a given reactor pressure vessel steel, due to irradiation, can be monitored by a reactor vessel surveillance program, such as the Wolf Creek reactor vessel radiation surveillance program[4], in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the initial RT_{NDT}, along with a margin (M) to cover uncertainties, to adjust the RT_{NDT} (ART) for radiation embrittlement. This ART (RT_{NDT} initial + M + ΔRT_{NDT}) is used to index the material to the K_{Ic} curve and, in turn, to set operating limits for the nuclear power plant that take into account the effects of irradiation on the reactor vessel materials.

4 DESCRIPTION OF PROGRAM

Six surveillance capsules for monitoring the effects of neutron exposure on the Wolf Creek reactor pressure vessel core region (beltline) materials were inserted in the reactor vessel prior to initial plant start-up. The six capsules were positioned in the reactor vessel between the neutron pads and the vessel wall as shown in Figure 4-1. The vertical center of the capsules is opposite the vertical center of the core

Capsule X was removed after 13.8 effective full power years (EFPY) of plant operation. This capsule contained Charpy V-notch, tensile, and 1/2T-CT fracture mechanics specimens made from lower shell plate R2508-3 (heat number C4935-2) and submerged arc weld metal representative of all the reactor vessel beltline region weld seams. In addition, this capsule contained Charpy V-notch specimens from the weld Heat-Affected-Zone (HAZ) metal of plate R2508-3.

Test material obtained from the Lower Shell Plate (after thermal heat treatment and forming of the plate) was taken at least one plate thickness from the quenched edges of the plate. All test specimens were machined from the ¼ thickness location of the plate after performing a simulated post-weld stress-relieved weldment joining lower shell plate R2508-1 and adjacent lower shell plate R2508-3. All heat-affected-zone specimens were obtained from the weld heat-affected-zone of the lower shell plate R2508-3

Charpy V-notch impact specimens from lower shell plate R2508-3 were machined in the longitudinal orientation (longitudinal axis of the specimen parallel to the major working direction) and also in the transverse orientation (longitudinal axis of the specimen perpendicular to the major working direction). The core region weld Charpy impact specimens were machined from the weldment such that the long dimension of each Charpy specimen was perpendicular to the weld direction. The notch of the weld metal Charpy specimens was machined such that the direction of crack propagation in the specimen was in the welding direction

Tensile specimens from lower shell plate R2508-3 were machined in both the longitudinal and transverse orientations. Tensile specimens from the weld metal were oriented with the long dimension of the specimen perpendicular to the weld direction

Compact tension test specimens from lower shell plate R2508-3 were machined in the longitudinal and transverse orientations. Compact tension test specimens from the weld metal were machined perpendicular to the weld direction with the notch oriented in the direction of welding. All specimens were fatigue pre-cracked according to ASTM E399.

The chemical composition and heat treatment of the unirradiated surveillance materials are presented in Tables 4-1 and 4-2, respectively. The data in Table 4-1 and 4-2 was obtained from the unirradiated surveillance program report, WCAP-10015, Appendix A. Contained in Table 4-3 are the results of the chemical analysis performed on four Charpy specimens from Capsule X The results of the NBS certified standards are presented in Tables 4-4 and 4-5.

Capsule X contained dosimeter wires of pure iron, copper, nickel, and aluminum 0.15 weight percent cobalt (cadmium-shielded and unshielded). In addition, cadmium shielded dosimeters of neptunium (Np^{237}) and uranium (U^{238}) were placed in the capsule to measure the integrated flux at specific neutron energy levels.

The capsule contained thermal monitors made from two low-melting-point eutectic alloys and sealed in Pyrex tubes. These thermal monitors were used to define the maximum temperature attained by the test specimens during irradiation. The composition of the two eutectic alloys and their melting points are as follows:

2.5% Ag, 97.5% Pb	Melting Point: 579°F (304°C)
1.5% Ag, 1.0% Sn, 97.5% Pb	Melting Point: 590°F (310°C)

The arrangement of the various mechanical specimens, dosimeters and thermal monitors contained in Capsule X is shown in Figure 4-2.

Table 4-1 Chemical Composition (wt%) of the Wolf Creek Reactor Vessel Surveillance Materials (Unirradiated)[a]

Element	Lower Shell Plate R2508-3	Weld Metal [b]
C	0.20	0.11
Mn	1.45	1.46
P	0 008	0 005
S	0 010	0.011
Si	0.20	0 48
Ni	0.62	0 09
Mo	0.55	0.56
Cr	0.05	0.09
Cu	0.07	0.04
Al	0.032	0 009
Co	0.014	0.010
Pb	<0 001	<0.001
W	<0.01	<0.01
Ti	<0.01	<0.01
Zr	<0.001	<0.001
V	0 003	0.005
Sn	0.002	0 003
As	0.007	0.004
Cb	<0.01	<0.01
N_2	0.007	0 006
B	<0 001	<0.001

Notes:

(a) Data obtained from WCAP-10015 and duplicated herein for completeness.

(b) Weld wire Type B4, Heat Number 90146, Flux Type Linde 124, and Flux Lot Number 1061. Surveillance weldment is from a weld between the lower shell plates R2508-3 and R2508-1 and is identical to the intermediate to lower shell circumferential weld seam. In addition, this weld is made of the same weld wire heat as the longitudinal weld seams.

Table 4-2 Heat Treatment History of the Wolf Creek Reactor Vessel Surveillance Materials[(a)]

Material	Temperature (°F)	Time	Coolant
Lower Shell Plate R2508-3	Austenitized @ 1600 ± 25	4 hrs.	Water-Quench
	Tempered @ 1225 ± 25	4 hrs	Air-cooled
	Stress Relieved[(b)] @ 1150 ± 50	8 hrs. 30 min.	Furnace Cooled
Weld Metal (heat # 90146)	Stress Relieved[(b)] @ 1150 ± 50	10 hrs. 15 min.	Furnace Cooled

Notes:

(a) This table was taken from WCAP-10015[4].

(b) The stress relief heat treatment received by the surveillance test plate and weldment have been simulated.

Table 4-3 Chemical Composition (wt%) of four Charpy Specimens from Wolf Creek Capsule X				
	Concentration in Weight Percent			
	Weld Metal Specimens			Base Metal Specimen
Element	AW-59	AW-55	AW-54	AT-54
Al	0.016	0.008	0.008	0.015
Co	0.0144	0.01	0.01	0.0144
Cr	0.0681	0.116	0.112	0.0687
Cu	0.0511	0.05	0.0411	0.0747
Fe	95.0	96.8	94.5	95.6
Mn	1.34	1.45	1.41	1.34
Mo	0.502	0.545	0.527	0.511
Ni	0.589	0.112	0.108	0.591
P	0.0152	0.017	0.0142	0.0145
Si	0.189	0.326	0.310	0.179
Sn	0.004	0.004	0.003	0.003
Ti	0.006	0.006	0.004	0.004
V	0.008	0.008	0.007	0.006
Zr	<0.00001	<0.00001	<0.00001	<0.00001
C	0.12	0.12	0.12	0.22
S	0.013	0.013	0.013	0.013

Table 4-4 Chemical Results from Low Alloy Steel NIST Certified Reference Standards (wt%)

Element.	NIST 361		NIST-362	
	Certified	Measured	Certified	Measured
Al	0.021	0.020	0.083	0.078
Co	0.032	0.032	0.300	0.314
Cr	0.694	0 705	0.300	0.310
Cu	0 042	0.042	0.500	0.500
Fe	95.600	95.100	95.300	97.300
Mn	0.660	0.662	1.040	1.070
Mo	0.190	0.153	0.068	0.053
Ni	2.000	2 080	0.590	0 608
P	0.014	0.015	0.041	0.035
Si	0 222	0.179	0.390	0 212
Sn	0.010	0.011	0.016	0.018
Ti	0 020	0.021	0.097	0.039
V	0 011	0.013	0.040	0.040
Zr	0.009	0.003	0.190	0.120
C	0.383	0.380	0.160	0.160
S	0.014	0 013	0.036	0.035

Table 4-5 Chemical Results from Low Alloy Steel NIST Certified Reference Standards (wt%)

Element.	NIST 363		NIST-364	
	Certified	Measured	Certified	Measured
Al	0.240	0.306	0.008	0.013
Co	0.048	0.048	0.150	0.153
Cr	1.310	1.320	0.060	0.066
Cu	0.100	0.100	0.240	0.248
Fe	94.400	96.800	96.700	96.900
Mn	1.500	1.540	0.250	0.257
Mo	0.028	0.026	0.490	0.426
Ni	0.300	3.150	0.140	0.140
P	0.029	0.029	0.010	0.010
Si	0.740	0.485	0.060	0 058
Sn	0.104	0.101	0.008	0.008
Ti	0.050	0.054	0.240	0.237
V	0.310	0.313	0.100	0.107
Zr	0.049	0.067	0.068	0.034
C	0.620	0.610	0.870	0.850
S	0.007	0.006	0.025	0.023

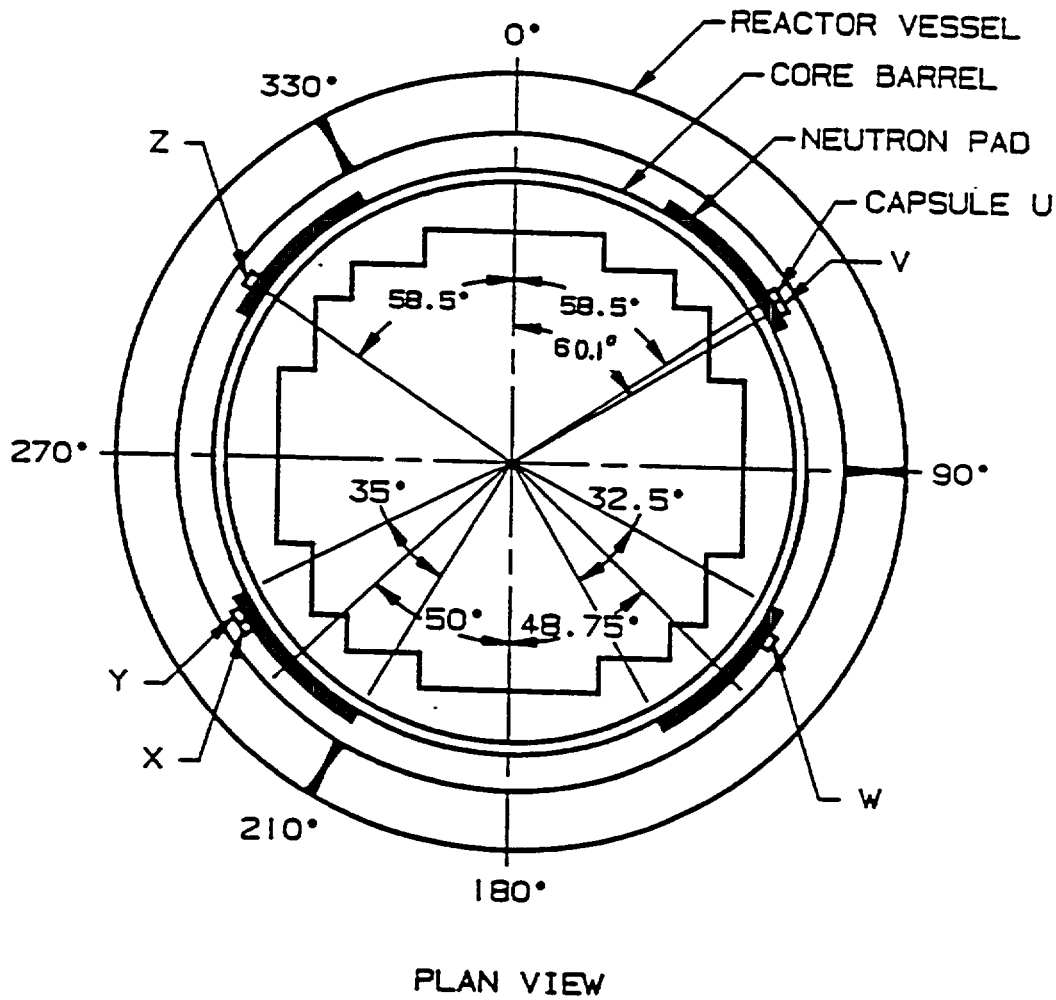


Figure 4-1 Arrangement of Surveillance Capsules in the Wolf Creek Reactor Vessel

LEGEND: AL - LOWER SHELL PLATE R2508-3 (LONGITUDINAL)
AT - LOWER SHELL PLATE R2508-3 (TRANSVERSE)
AW - WELD METAL
AH - HEAT-AFFECTED-ZONE MATERIAL

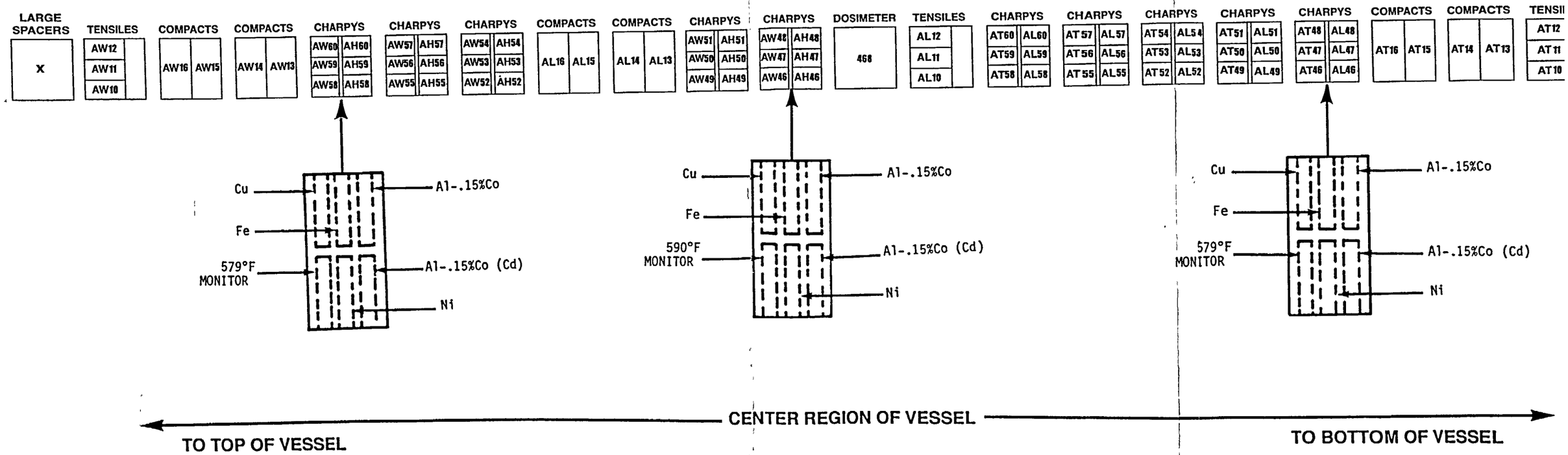


Figure 4-2 Capsule X Diagram Showing The Location of Specimens, Thermal Monitors, and Dosimeters

5 TESTING OF SPECIMENS FROM CAPSULE X

5.1 OVERVIEW

The post-irradiation mechanical testing of the Charpy V-notch impact specimens and tensile specimens was performed in the Remote Metallographic Facility (RMF) at the Westinghouse Science and Technology Center. Testing was performed in accordance with 10CFR50, Appendices G and H[2], ASTM Specification E185-82[9], and Westinghouse Procedure RMF 8402[10], Revision 2 as modified by Westinghouse RMF Procedures 8102[11], Revision 1, and 8103[12], Revision 1.

Upon receipt of the capsule at the hot cell laboratory, the specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in WCAP-10015[4]. No discrepancies were found

Examination of the two low-melting point 579°F (304°C) and 590°F (310°C) eutectic alloys indicated no melting of either type of thermal monitor. Based on this examination, the maximum temperature to which the test specimens were exposed was less than 579°F (304°C).

The Charpy impact tests were performed per ASTM Specification E23-98[13] and RMF Procedure 8103 on a Tinius-Olsen Model 74, 358J machine. The tup (striker) of the Charpy impact test machine is instrumented with a GRC 930-I instrumentation system, feeding information into an IBM compatible computer. With this system, load-time and energy-time signals can be recorded in addition to the standard measurement of Charpy energy (E_D). From the load-time curve (Appendix B), the load of general yielding (P_{GY}), the time to general yielding (t_{GY}), the maximum load (P_M), and the time to maximum load (t_M) can be determined Under some test conditions, a sharp drop in load indicative of fast fracture was observed The load at which fast fracture was initiated is identified as the fast fracture load (P_F), and the load at which fast fracture terminated is identified as the arrest load (P_A).

The energy at maximum load (E_M) was determined by comparing the energy-time record and the load-time record. The energy at maximum load is approximately equivalent to the energy required to initiate a crack in the specimen Therefore, the propagation energy for the crack (E_p) is the difference between the total energy to fracture (E_D) and the energy at maximum load (E_M).

The yield stress (σ_Y) was calculated from the three-point bend formula having the following expression·

$$\sigma_Y = (P_{GY} * L) / [B * (W - a)^2 * C] \qquad (1)$$

where:
- L = distance between the specimen supports in the impact machine
- B = the width of the specimen measured parallel to the notch
- W = height of the specimen, measured perpendicularly to the notch
- a = notch depth

The constant C is dependent on the notch flank angle (ϕ), notch root radius (ρ) and the type of loading (i.e., pure bending or three-point bending). In three-point bending, for a Charpy specimen in which $\phi = 45°$ and $\rho = 0.010$ inch, Equation 1 is valid with C = 1.21. Therefore, (for L = 4W),

$$\sigma_Y = (P_{GY} * L) / [B * (W - a)^2 * 1\ 21\ ] = (3.305 * P_{GY} * W) / [B * (W - a)^2] \quad (2)$$

For the Charpy specimen, B = 0 394 inch, W = 0.394 inch and a = 0.079 inch. Equation 2 then reduces to:

$$\sigma_Y = 33.3 * P_{GY} \quad (3)$$

where σ_y is in units of psi and P_{GY} is in units of lbs. The flow stress was calculated from the average of the yield and maximum loads, also using the three-point bend formula.

The symbol A in columns 4, 5, and 6 of Tables 5-5 through 5-8 is the cross-section area under the notch of the Charpy specimens:

$$A = B * (W - a) = 0.1241 \text{ sq.in.} \quad (4)$$

Percent shear was determined from post-fracture photographs using the ratio-of-areas methods in compliance with ASTM Specification E23-98 and A370-97a[14] . The lateral expansion was measured using a dial gage rig similar to that shown in the same specification.

Tensile tests were performed on a 20,000-pound Instron, split-console test machine (Model 1115) per ASTM Specification E8-99[15] and E21-92 (1998)[16], and Procedure RMF 8102. All pull rods, grips, and pins were made of Inconel 718. The upper pull rod was connected through a universal joint to improve axiality of loading. The tests were conducted at a constant crosshead speed of 0.05 inches per minute throughout the test.

Extension measurements were made with a linear variable displacement transducer extensometer. The extensometer knife-edges were spring-loaded to the specimen and operated through specimen failure. The extensometer gage length was 1.00 inch. The extensometer is rated as Class B-2 per ASTM E83-93[17].

Elevated test temperatures were obtained with a three-zone electric resistance split-tube furnace with a 9-inch hot zone. All tests were conducted in air. Because of the difficulty in remotely attaching a thermocouple directly to the specimen, the following procedure was used to monitor specimen temperatures. Chromel-Alumel thermocouples were positioned at the center and at each end of the gage section of a dummy specimen and in each tensile machine griper. In the test configuration, with a slight load on the specimen, a plot of specimen temperature versus upper and lower tensile machine griper and controller temperatures was developed over the range from room temperature to 550°F. During the actual testing, the grip temperatures were used to obtain desired specimen temperatures. Experiments have indicated that this method is accurate to ±2°F.

The yield load, ultimate load, fracture load, total elongation, and uniform elongation were determined directly from the load-extension curve. The yield strength, ultimate strength, and fracture strength were calculated using the original cross-sectional area. The final diameter and final gage length were determined from post-fracture photographs. The fracture area used to calculate the fracture stress (true stress at fracture) and percent reduction in area was computed using the final diameter measurement.

5.2 CHARPY V-NOTCH IMPACT TEST RESULTS

The results of the Charpy V-notch impact tests performed on the various materials contained in Capsule X, which received a fluence of 3.49×10^{19} n/cm^2(E> 1.0 MeV) in 13.8 EFPY of operation, are presented in Tables 5-1 through 5-11 and are compared with unirradiated results[4] as shown in Figures 5-1 through 5-12.

The transition temperature increases and upper shelf energy decreases for the Capsule X materials are summarized in Table 5-9 and led to the following results:

Irradiation of the reactor vessel lower shell plate R2508-3 (heat number C4935-2) Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction (longitudinal orientation), resulted in an irradiated 30 ft-lb transition temperature of 36.11°F and an irradiated 50 ft-lb transition temperature of 67.75°F. This results in a 30 ft-lb transition temperature increase of 61.06°F and a 50 ft-lb transition temperature increase of 67.64°F for the longitudinal oriented specimens.

Irradiation of the reactor vessel lower shell plate R2508-3 (heat number C4935-2) Charpy specimens, oriented with the longitudinal axis of the specimen perpendicular to the major working direction (transverse orientation), resulted in an irradiated 30 ft-lb transition temperature of 55.97°F and an irradiated 50 ft-lb transition temperature of 97.04°F. This results in a 30 ft-lb transition temperature increase of 53.96°F and a 50 ft-lb transition temperature increase of 62.71°F for the longitudinal oriented specimens.

Irradiation of the weld metal (heat number 90146) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 10.66°F and an irradiated 50 ft-lb transition temperature of 54.73°F. This results in a 30 ft-lb transition temperature increase of 68.36°F and a 50 ft-lb transition temperature increase of 75.38°F

Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of –74.34°F and an irradiated 50 ft-lb transition temperature of –52.92°F. This results in a 30 ft-lb transition temperature increase of 69.66°F and a 50 ft-lb transition temperature increase of 61.08°F.

The average upper shelf energy of the lower shell plate R2508-3 (longitudinal orientation) resulted in an average energy decrease of 6 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 142 ft-lb for the longitudinal oriented specimens.

The average upper shelf energy of the lower shell plate R2508-3 (transverse orientation) resulted in no energy decrease after irradiation. This results in an irradiated average upper shelf energy of 94 ft-lb for the longitudinal oriented specimens.

The average upper shelf energy of the weld metal Charpy specimens resulted in an average energy decrease of 7 ft-lb after irradiation This results in an irradiated average upper shelf energy of 93 ft-lb for the weld metal specimens.

The average upper shelf energy of the weld HAZ metal Charpy specimens resulted in an average energy decrease of 26 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 135 ft-lb for the weld HAZ metal.

A comparison, as presented in Table 5-10, of the Wolf Creek reactor vessel surveillance material test results with the Regulatory Guide 1.99, Revision 2[1] predictions led to the following conclusions:

— The measured 30 ft-lb shift in transition temperature values of the lower shell plate R2508-3 contained in capsule X (longitudinal & transverse) are less than the Regulatory Guide 1.99, Revision 2, predictions.

— The measured 30 ft-lb shift in transition temperature value of the weld metal contained in capsule X is greater than the Regulatory Guide 1.99, Revision 2, predictions. However, the shift value is less than two sigma allowance by Regulatory Guide 1.99, Revision 2.

— The measured percent decrease in upper shelf energy for all the surveillance materials of Capsules X contained in the Wolf Creek surveillance program are less than the Regulatory Guide 1.99, Revision 2 predictions.

All beltline materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are predicted to maintain an upper shelf energy greater than 50 ft-lb throughout the extended life of the vessel (54 EFPY) as required by 10CFR50, Appendix G [2].

The fracture appearance of each irradiated Charpy specimen from the various surveillance Capsule X materials is shown in Figures 5-13 through 5-16 and shows an increasingly ductile or tougher appearance with increasing test temperature.

The load-time records for individual instrumented Charpy specimen tests are shown in Appendix B.

The Charpy V-notch data presented in WCAP-15078, Rev. 1[3] were based on a re-plot of all capsule data from WCAP-10015[4], WCAP-11553[5] and WCAP-13365, Rev. 1[6] using CVGRAPH, Version 4.1, which is a symmetric hyperbolic tangent curve-fitting program. The results presented are only for the Capsule X test results, which are also based on using CVGRAPH, Version 4.1. This report also shows the composite plots that show the results from the previous capsule. Appendix C presents the CVGRAPH, Version 4.1, Charpy V-notch plots and the program input data.

5.3 TENSILE TEST RESULTS

The results of the tensile tests performed on the various materials contained in Capsule X irradiated to 3.49×10^{19} n/cm^2 (E> 1.0 MeV) are presented in Table 5-11 and are compared with unirradiated results[4] as shown in Figures 5-17 and 5-19.

The results of the tensile tests performed on the lower Shell Plate R2508-3 (longitudinal orientation) indicated that irradiation to 3.49×10^{19} n/cm^2 (E> 1.0 MeV) caused approximately a 9 to 11 ksi increase in the 0.2 percent offset yield strength and approximately a 8 ksi increase in the ultimate tensile strength when compared to unirradiated data[4]. See Figure 5-17.

The results of the tensile tests performed on the lower Shell Plate R2508-3 (Transverse orientation) indicated that irradiation to 3.49×10^{19} n/cm^2 (E> 1.0 MeV) caused approximately a 7 to 11 ksi increase in the 0.2 percent offset yield strength and approximately a 9 to 10 ksi increase in the ultimate tensile strength when compared to unirradiated data[4]. See Figure 5-18.

The results of the tensile tests performed on the surveillance weld metal indicated that irradiation to 3.49×10^{19} n/cm^2 (E> 1.0 MeV) caused approximately a 2 to 9 ksi increase in the 0.2 percent offset yield strength and approximately a 1 to 7 ksi increase in the ultimate tensile strength when compared to unirradiated data[4]. See Figure 5-19.

The fractured tensile specimens for the Lower Shell Plate R2508-3 material are shown in Figures 5-20 and 5-21, while the fractured tensile specimens for the surveillance weld metal are shown in Figure 5-22. The engineering stress-strain curves for the tensile tests are shown in Figures 5-23 through 5-25.

5.4 1/2T COMPACT TENSION SPECIMEN TESTS

Per the surveillance capsule testing contract, the 1/2T Compact Tension Specimens were not tested and are being stored at the Westinghouse Science and Technology Center Hot Cell facility.

Table 5-1 Charpy V-notch Data for the Wolf Creek Lower Shell Plate R2508-3 Irradiated to a Fluence of 3.49 x 10^{19} n/cm^2 (E> 1.0 MeV) (Longitudinal Orientation)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
AL48	-50	-46	2	3	0	0.00	2
AL57	0	-18	13	18	5	0.13	5
AL53	25	-4	21	28	12	0.30	10
AL52	40	4	37	50	24	0.61	15
AL56	50	10	53	72	33	0.84	20
AL55	75	24	43	58	29	0.74	30
AL59	110	43	74	100	47	1.19	50
AL50	135	57	108	146	67	1.70	65
AL60	150	66	133	180	74	1.88	90
AL49	175	79	100	136	64	1.63	75
AL51	190	88	122	165	67	1.70	85
AL46	225	107	150	203	71	1.80	100
AL54	250	121	146	198	75	1.91	100
AL58	275	135	135	183	75	1.91	100
AL47	300	149	137	186	75	1.91	100

Table 5-2 Charpy V-notch Data for the Wolf Creek Lower Shell Plate R2508-3 Irradiated to a Fluence of 3.49 x 10^{19} n/cm^2 (E> 1.0 MeV) (Transverse Orientation)

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	°F	°C	ft-lbs	Joules	mils	mm	%
AT46	-75	-59	5	7	0	0.00	2
AT50	-25	-32	11	15	4	0.10	5
AT60	15	-9	15	20	8	0.20	15
AT56	50	10	30	41	20	0.51	25
AT54	75	24	41	56	29	0.74	40
AT53	100	38	52	71	36	0.91	45
AT59	125	52	55	75	38	0.97	55
AT58	150	66	67	91	50	1.27	65
AT48	175	79	98	133	57	1.45	90
AT51	175	79	79	107	53	1.35	80
AT57	200	93	88	119	51	1.30	95
AT52	225	107	91	123	69	1.75	100
AT49	250	121	93	126	60	1.52	100
AT47	275	135	101	137	66	1.68	100
AT55	300	149	96	130	61	1.55	100

Table 5-3 Charpy V-notch Data for the Wolf Creek Surveillance Weld Metal Irradiated to a Fluence of 3.49 x 10^{19} n/cm^2 (E> 1.0 MeV)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
AW47	-75	-59	4	5	0	0.00	10
AW52	-35	-37	13	18	5	0.13	20
AW53	0	-18	23	31	15	0.38	45
AW51	25	-4	37	50	25	0.64	50
AW58	50	10	53	72	35	0.89	60
AW46	75	24	64	87	45	1.14	80
AW59	100	38	68	92	51	1.30	85
AW57	125	52	76	103	55	1.40	95
AW55	125	52	78	106	53	1.35	95
AW48	150	66	67	91	54	1.37	85
AW60	160	71	89	121	62	1.57	95
AW56	200	93	84	114	54	1.37	99
AW50	225	107	102	138	64	1.63	100
AW49	250	121	94	127	69	1.75	100
AW54	250	121	96	130	64	1.63	100

Table 5-4 Charpy V-notch Data for the Wolf Creek Heat-Affected-Zone (HAZ) Material Irradiated to a Fluence of 3.49 x 10^{19} n/cm^2 (E> 1.0 MeV)

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	°F	°C	Ft-lbs	Joules	mils	mm	%
AH53	-175	-115	6	8	0	0.00	0
AH51	-100	-73	15	20	5	0.13	2
AH50	-75	-59	34	46	12	0.30	5
AH49	-50	-46	88	119	42	1.07	30
AH58	-50	-46	29	39	12	0.30	15
AH52	-25	-32	50	68	21	0.53	20
AH47	0	-18	152	206	70	1.78	100
AH55	0	-18	92	125	44	1.12	50
AH59	25	-4	98	133	52	1.32	65
AH48	35	2	132	179	70	1.78	75
AH54	50	10	140	190	70	1.78	100
AH60	100	38	146	198	70	1.78	100
AH46	150	66	140	190	71	1.80	100
AH57	200	93	129	175	74	1.88	100
AH56	200	93	124	168	72	1.83	100

Table 5-5 Instrumented Charpy Impact Test Results for the Wolf Creek Lower Shell Plate R2508-3 Irradiated to a Fluence of 3.49 x 10^{19} n/cm^2 (E>1.0 MeV) (Longitudinal Orientation)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. t_M (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress σ_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_p/A								
AL48	-50	2	16	8	8	1068	0.10	1068	0.10	1068	0	36	36
AL57	0	13	105	54	51	3868	0.20	3868	0.20	3868	0	129	129
AL53	25	21	169	58	111	3079	0.14	3810	0.21	3783	41	103	115
AL52	40	37	298	223	75	3167	0.14	4231	0.53	4212	183	105	123
AL56	50	53	427	297	130	3078	0.14	4212	0.67	4058	0	102	121
AL55	75	43	346	205	141	3110	0.14	4153	0.50	4141	854	104	121
AL59	110	74	596	292	305	2952	0.14	4040	0.68	3803	1752	98	116
AL50	135	108	870	366	505	3045	0.15	4137	0.83	3240	1054	101	120
AL60	150	133	1072	292	779	2934	0.14	4116	0.69	1461	689	98	117
AL49	175	100	806	287	519	2869	0.14	4066	0.68	3786	1041	96	115
AL51	190	122	983	293	690	2885	0.14	4069	0.70	2687	1779	96	116
AL46	225	150	1209	353	856	2825	0.14	4059	0.82	N/A	N/A	94	115
AL54	250	146	1176	349	828	2788	0.14	4007	0.82	N/A	N/A	93	113
AL58	275	135	1088	279	809	2631	0.15	3911	0.70	N/A	N/A	88	109
AL47	300	137	1104	276	828	2714	0.14	3888	0.69	N/A	N/A	90	110

Table 5-6 Instrumented Charpy Impact Test Results for the Wolf Creek Lower Shell Plate R2508-3 Irradiated to a Fluence of 3.49 x 10^{19} n/cm^2 (E>1.0 MeV) (Transverse Orientation)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. t_M (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress σ_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_P/A								
AT46	-75	5	40	19	21	2391	0.13	2391	0.13	2391	0	80	80
AT50	-25	11	89	49	40	3839	0.19	3839	0.19	3839	0	128	128
AT60	15	15	121	58	63	3814	0.21	3814	0.21	3814	94	127	127
AT56	50	30	242	151	91	2922	0.14	3909	0.40	3877	263	97	114
AT54	75	41	330	159	171	3086	0.14	4001	0.41	3991	1021	103	118
AT53	100	52	419	209	209	2979	0.14	4066	0.52	4035	1530	99	117
AT59	125	55	443	200	243	2888	0.14	3873	0.51	3823	1335	96	113
AT58	150	67	540	197	343	2861	0.14	3864	0.51	3499	1191	95	112
AT48	175	98	790	288	501	2740	0.15	4089	0.70	3353	2066	91	114
AT51	175	79	637	273	363	2890	0.14	3906	0.66	3674	1106	96	113
AT57	200	88	709	276	433	2833	0.14	3947	0.67	3552	2362	94	113
AT52	225	91	733	268	465	2718	0.14	3865	0.67	N/A	N/A	90	110
AT49	250	93	749	264	485	2814	0.15	3948	0.66	N/A	N/A	94	113
AT47	275	101	814	287	527	2776	0.15	3999	0.70	N/A	N/A	92	113
AT55	300	96	774	258	515	2717	0.14	3842	0.65	N/A	N/A	90	109

Table 5-7 Instrumented Charpy Impact Test Results for the Wolf Creek Surveillance Weld Metal Irradiated to a Fluence of 3.49 x 10^{19} n/cm^2 (E>1.0 MeV)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. t_M (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress σ_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_p/A								
AW47	-75	4	32	14	18	1800	0.12	1800	0.12	1800	0	60	60
AW52	-35	13	105	42	63	3814	0.17	3814	0.17	3814	416	127	127
AW53	0	23	185	33	152	3201	0.16	3201	0.16	3201	1941	107	107
AW51	25	37	298	137	161	3104	0.14	4062	0.36	4000	1984	103	119
AW58	50	53	427	212	215	3286	0.15	4242	0.50	4177	1597	109	125
AW46	75	64	516	219	297	3344	0.14	4316	0.50	4188	2141	111	128
AW59	100	68	548	216	332	3151	0.14	4121	0.52	3736	2053	105	121
AW57	125	76	612	219	394	3070	0.14	4135	0.52	3812	2294	102	120
AW55	125	78	628	215	414	3174	0.15	4169	0.52	3486	2501	106	122
AW48	150	67	540	199	341	2934	0.14	3879	0.50	3693	2308	98	113
AW60	160	89	717	281	436	3027	0.14	4081	0.65	3389	2313	101	118
AW56	200	84	677	207	470	3000	0.14	4012	0.52	2679	2204	100	117
AW50	225	102	822	299	523	3098	0.15	4181	0.67	N/A	N/A	103	121
AW49	250	94	757	276	482	2941	0.14	3980	0.66	N/A	N/A	98	115
AW54	250	96	774	292	482	3009	0.17	4034	0.70	N/A	N/A	100	117

Table 5-8 Instrumented Charpy Impact Test Results for the Wolf Creek Heat-Affected-Zone (HAZ) Metal Irradiated to a Fluence of 3.49 x 10^{19} n/cm^2 (E>1.0 MeV)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. t_M (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress σ_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_P/A								
AH53	-175	6	48	23	25	2931	0.14	2931	0.14	2931	N/A	98	98
AH51	-100	15	121	70	51	4991	0.21	4991	0.21	4991	N/A	166	166
AH50	-75	34	274	224	50	3892	0.15	4782	0.47	4749	N/A	130	144
AH49	-50	88	709	344	365	3874	0.16	4764	0.68	4082	N/A	129	144
AH58	-50	29	234	176	58	3648	0.15	4483	0.40	4452	N/A	121	135
AH52	-25	50	403	329	74	3701	0.15	4675	0.66	4605	N/A	123	139
AH47	0	152	1225	329	895	3583	0.15	4569	0.68	N/A	N/A	119	136
AH55	0	92	741	322	419	3510	0.15	4559	0.67	3925	1008	117	134
AH59	25	98	790	321	468	3475	0.15	4495	0.68	4039	1337	116	133
AH48	35	132	1064	326	738	3499	0.15	4499	0.69	2002	404	117	133
AH54	50	140	1128	324	804	3395	0.15	4468	0.70	N/A	N/A	113	131
AH60	100	146	1176	317	860	3383	0.15	4430	0.68	N/A	N/A	113	130
AH46	150	140	1128	318	810	3203	0.15	4341	0.70	N/A	N/A	107	126
AH57	200	129	1039	302	737	3074	0.14	4252	0.69	N/A	N/A	102	122
AH56	200	124	999	304	695	3058	0.15	4212	0.70	N/A	N/A	102	121

Table 5-9 Effect of Irradiation to 3.49 x 10^{19} n/cm^2 (E>1.0 MeV) on the Capsule V Notch Toughness Properties of the Wolf Creek Reactor Vessel Surveillance Materials[c]

Material	Average 30 (ft-lb)[a] Transition Temperature (°F)			Average 35 mil Lateral[b] Expansion Temperature (°F)			Average 50 ft-lb[a] Transition Temperature (°F)			Average Energy Absorption[a] at Full Shear (ft-lb)		
	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔE
Lower Shell Plate R2508-3 (Long.)	-24.95	36.11	61.06	-0.4	72.95	73.36	0.11	67.75	67.64	148	142	-6
Lower Shell Plate R2508-3 (Trans.)	2.0	55.97	53.97	25.44	102.4	76.95	34.32	97.04	62.71	94	95	+1
Weld Metal (Heat # 90146)	-57.69	10.66	68.36	-27.06	53.49	80.56	-20.64	54.73	75.38	100	93	-7
HAZ Metal	-144.01	-74.34	69.66	-89.86	-27.62	62.24	-114	-52.92	61.08	161	135	-26

a. "Average" is defined as the value read from the curve fit through the data points of the Charpy tests (see Figures 5-1, 5-4, 5-7 and 5-10)

b. "Average" is defined as the value read from the curve fit through the data points of the Charpy tests (see Figures 5-2, 5-5, 5-8 and 5-11).

Table 5-10 Comparison of the Wolf Creek Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence[d] ($x 10^{19}$ n/cm^2, E > 1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%)[c]
Lower Shell Plate	U	0.316	34.88	36.46	14.5	2
R2508-3	Y	1.19	53.55	16.03	20	11
	V	2.22	62.22	52.03	23	13
(Longitudinal)	X	3.49	67.83	61.06	25	4
Lower Shell Plate	U	0.316	34.88	23.79	14.5	0
R2508-3	Y	1.19	53.55	35.39	20	0
	V	2.22	62.22	54.53	23	6
(Transverse)	X	3.49	67.83	53.96	25	0
Surveillance	U	0.316	33.24	27.21	16	8
Program	Y	1.19	51.03	45.09	22	6
Weld Metal	V	2.22	59.29	46.33	25	11
	X	3.49	64.64	68.36	28	7
Heat Affected Zone	U	0.316	---	58.41	---	13
Material	Y	1.19	---	12.98	---	0
	V	2.22	---	55.91	---	0
	X	3.49	---	69.66	---	16

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

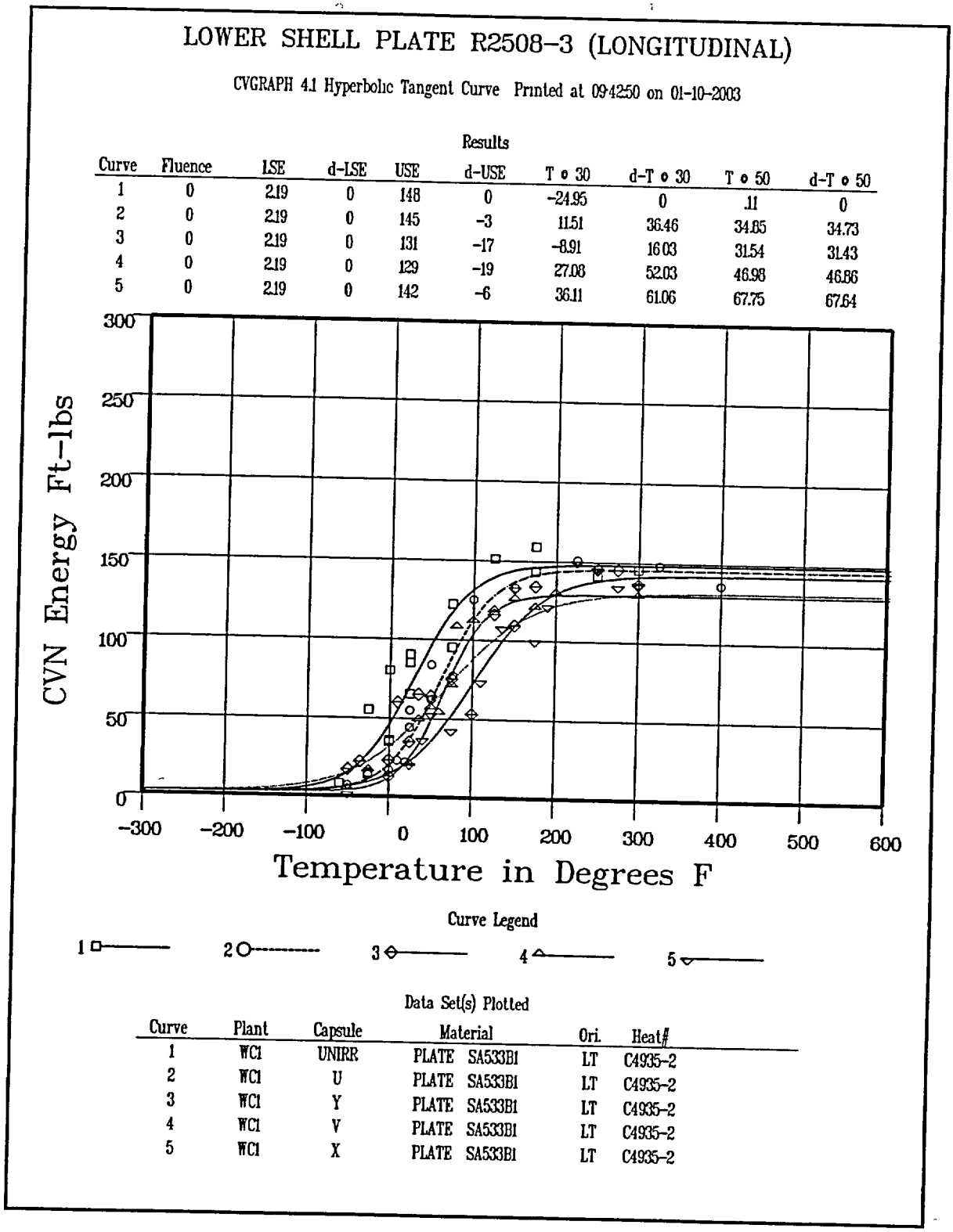
(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1 (See Appendix C)

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82

(d) The fluence values presented here are the calculated values, not the best estimate values.

Table 5-11 Tensile Properties of the Wolf Creek Capsule X Reactor Vessel Surveillance Materials Irradiated to 3.49 x 10^{19} n/cm^2 (E > 1.0 MeV)

Material	Sample Number	Test Temp. (°F)	0.2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Stress (ksi)	Fracture Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Lower Shell Plate R2508-3 (Long.)	AL-10	75	70.0	89.3	2.75	176.6	56.0	13.0	28.2	68
	AL-11	300	63.7	81.5	2.65	175.1	54.0	11.0	23.7	69
	AL-12	550	60.9	85.5	2.76	149.9	56.2	10.9	21.6	62
Lower Shell Plate R2508-3 (Trans.)	AT-10	75	71.3	90.2	3.10	168.4	63.2	12.8	25.3	62
	AT-11	300	63.7	82.3	2.80	140.7	57.0	11.3	20.8	59
	AT-12	550	58.6	86.3	0.35	15.9	7.1	11.3	18.1	55
Weld Metal	AW-10	75	81.0	96.4	3.15	180.7	64.2	11.5	25.1	64
	AW-11	300	70.8	86.0	2.81	165.8	57.2	9.5	21.1	65
	AW-12	550	70.8	92.8	3.20	178.7	65.2	11.0	22.2	64



Results

Curve	Fluence	USE	d-USE	USE	d-USE	T o 30	d-T o 30	T o 50	d-T o 50
1	0	2.19	0	148	0	-24.95	0	.11	0
2	0	2.19	0	145	-3	11.51	36.46	34.85	34.73
3	0	2.19	0	131	-17	-8.91	16.03	31.54	31.43
4	0	2.19	0	129	-19	27.08	52.03	46.98	46.86
5	0	2.19	0	142	-6	36.11	61.06	67.75	67.64

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WC1	UNIRR	PLATE SA533B1	LT	C4935-2
2	WC1	U	PLATE SA533B1	LT	C4935-2
3	WC1	Y	PLATE SA533B1	LT	C4935-2
4	WC1	V	PLATE SA533B1	LT	C4935-2
5	WC1	X	PLATE SA533B1	LT	C4935-2

Figure 5-1 Charpy V-Notch Impact Energy vs. Temperature for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Longitudinal Orientation)

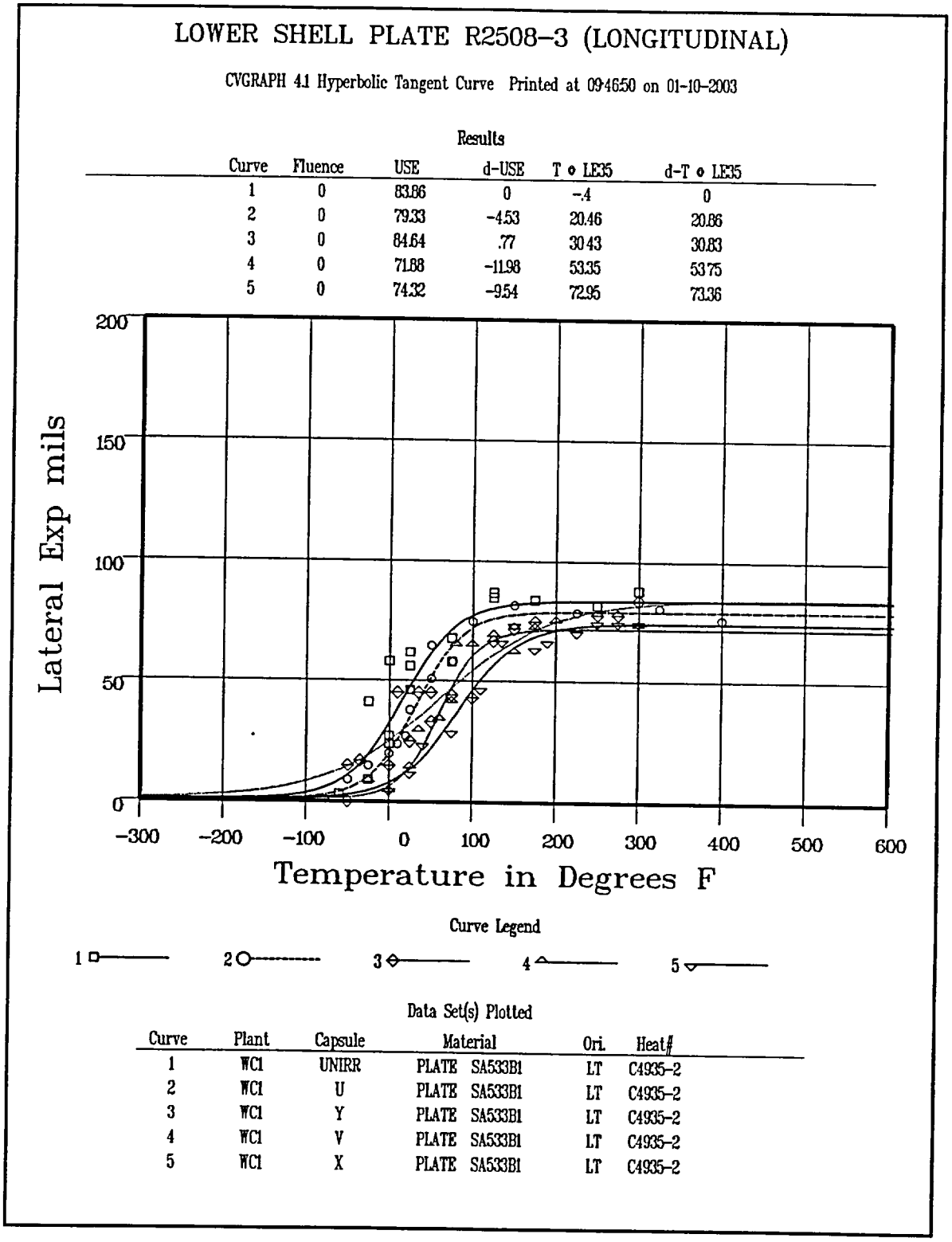


Figure 5-2 Charpy V-Notch Lateral Expansion vs. Temperature for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Longitudinal Orientation)

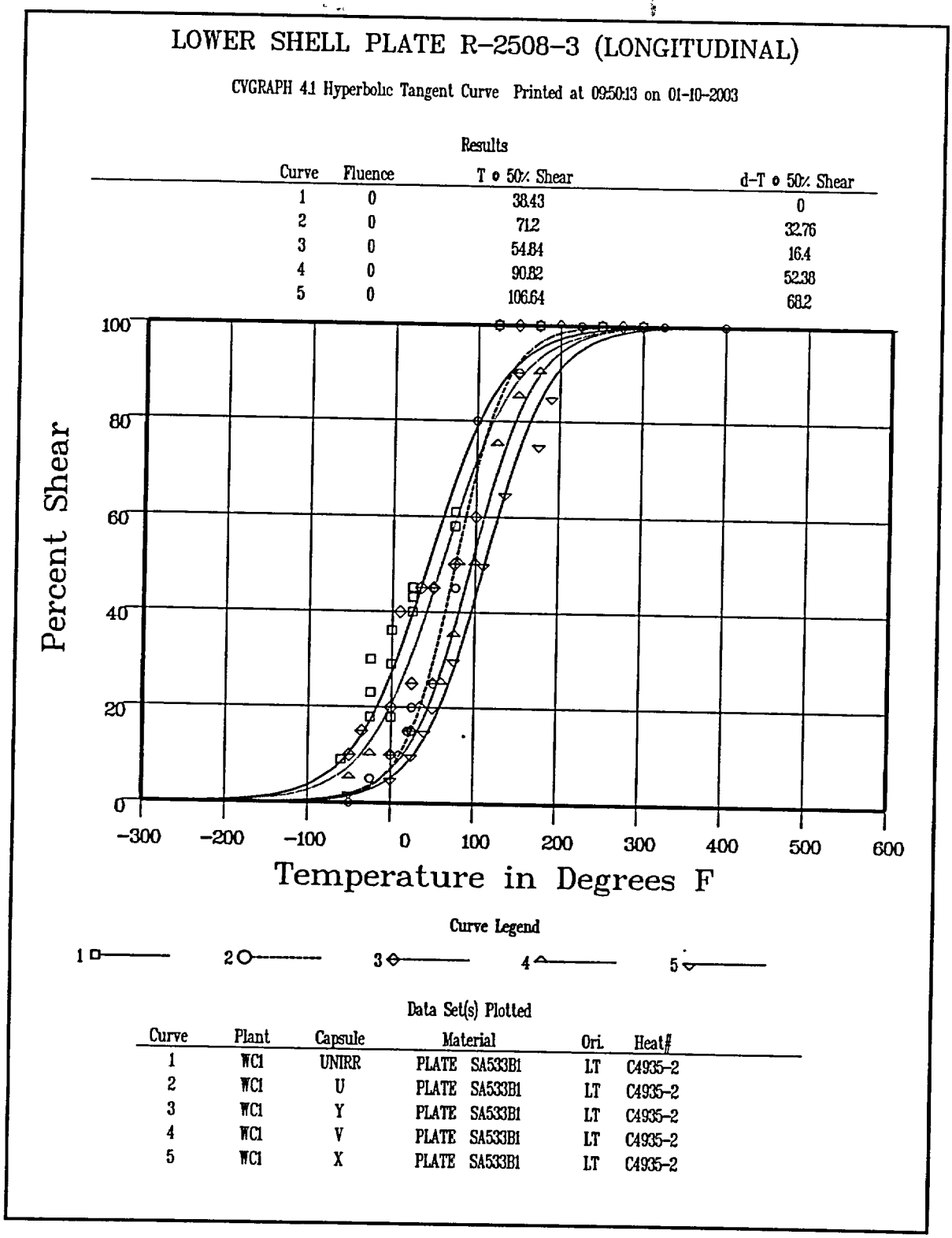
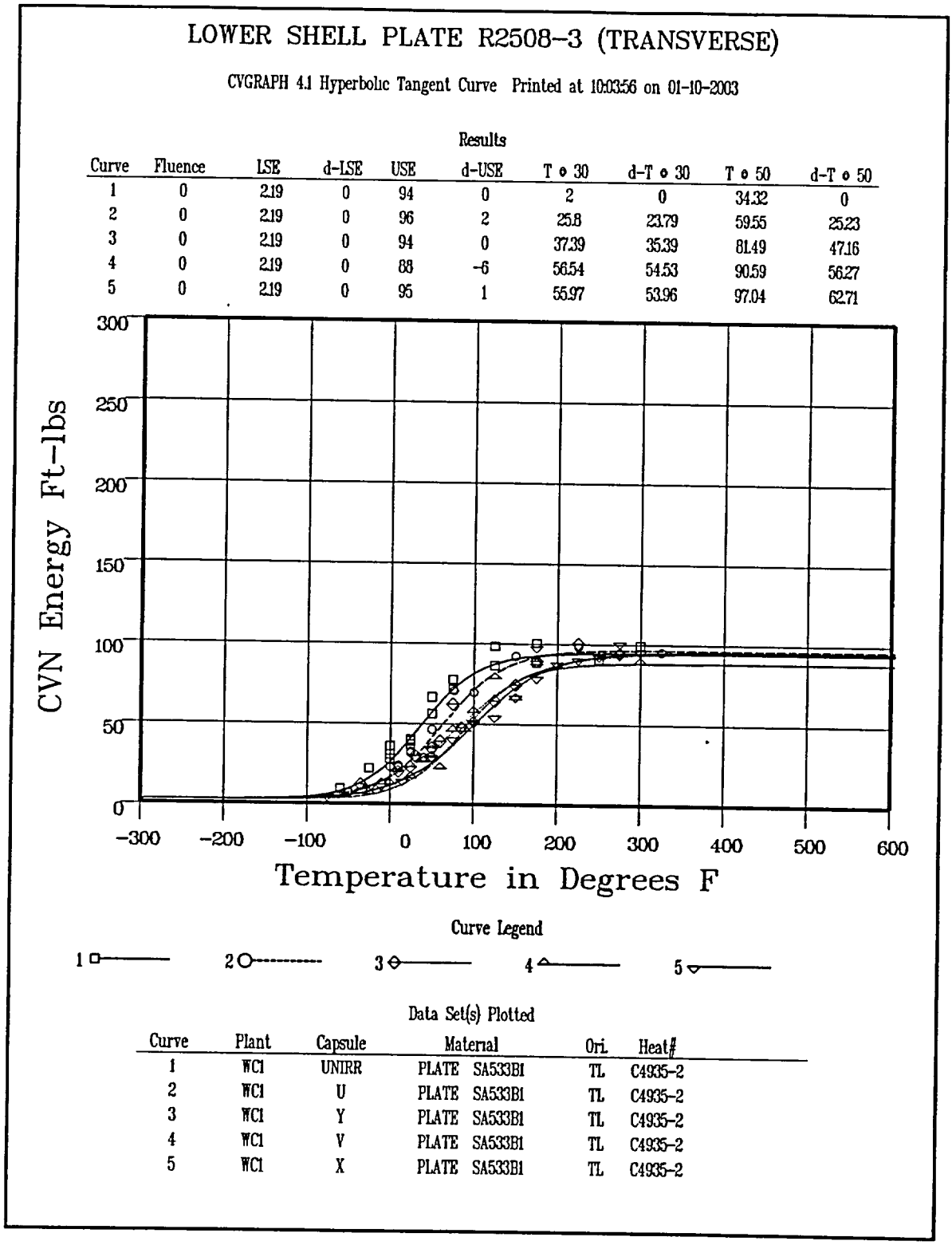


Figure 5-3 **Charpy V-Notch Percent Shear vs. Temperature for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Longitudinal Orientation)**



Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	94	0	2	0	34.32	0
2	0	2.19	0	96	2	25.8	23.79	59.55	25.23
3	0	2.19	0	94	0	37.39	35.39	81.49	47.16
4	0	2.19	0	88	-6	56.54	54.53	90.59	56.27
5	0	2.19	0	95	1	55.97	53.96	97.04	62.71

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WC1	UNIRR	PLATE SA533B1	TL	C4935-2
2	WC1	U	PLATE SA533B1	TL	C4935-2
3	WC1	Y	PLATE SA533B1	TL	C4935-2
4	WC1	V	PLATE SA533B1	TL	C4935-2
5	WC1	X	PLATE SA533B1	TL	C4935-2

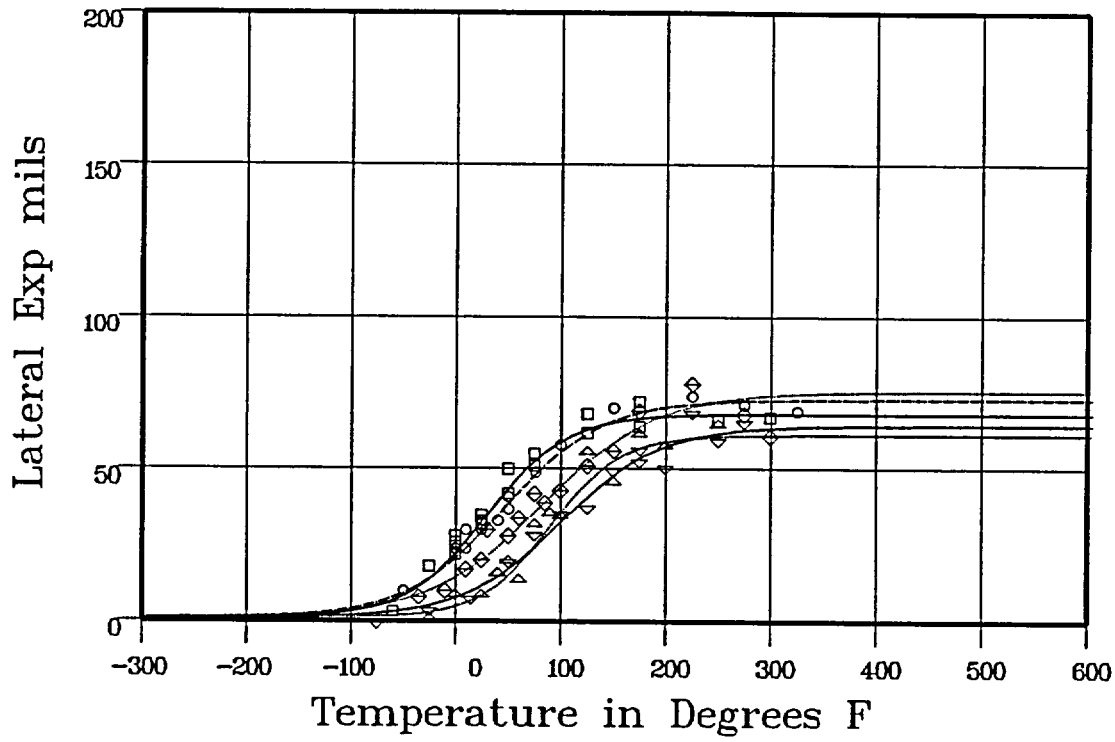
Figure 5-4 Charpy V-Notch Impact Energy vs. Temperature for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Transverse Orientation)

LOWER SHELL PLATE R2508-3 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:07:23 on 01-10-2003

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	68.05	0	25.44	0
2	0	72.86	4.81	35.7	10.25
3	0	75.48	7.43	67.84	42.39
4	0	61.41	-6.63	93.79	68.34
5	0	64.44	-3.6	102.4	76.95



Curve Legend

1 2 3 4 5

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WC1	UNIRR	PLATE SA533B1	TL	C4935-2
2	WC1	U	PLATE SA533B1	TL	C4935-2
3	WC1	Y	PLATE SA533B1	TL	C4935-2
4	WC1	V	PLATE SA533B1	TL	C4935-2
5	WC1	X	PLATE SA533B1	TL	C4935-2

Figure 5-5 Charpy V-Notch Lateral Expansion vs. Temperature for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Transverse Orientation)

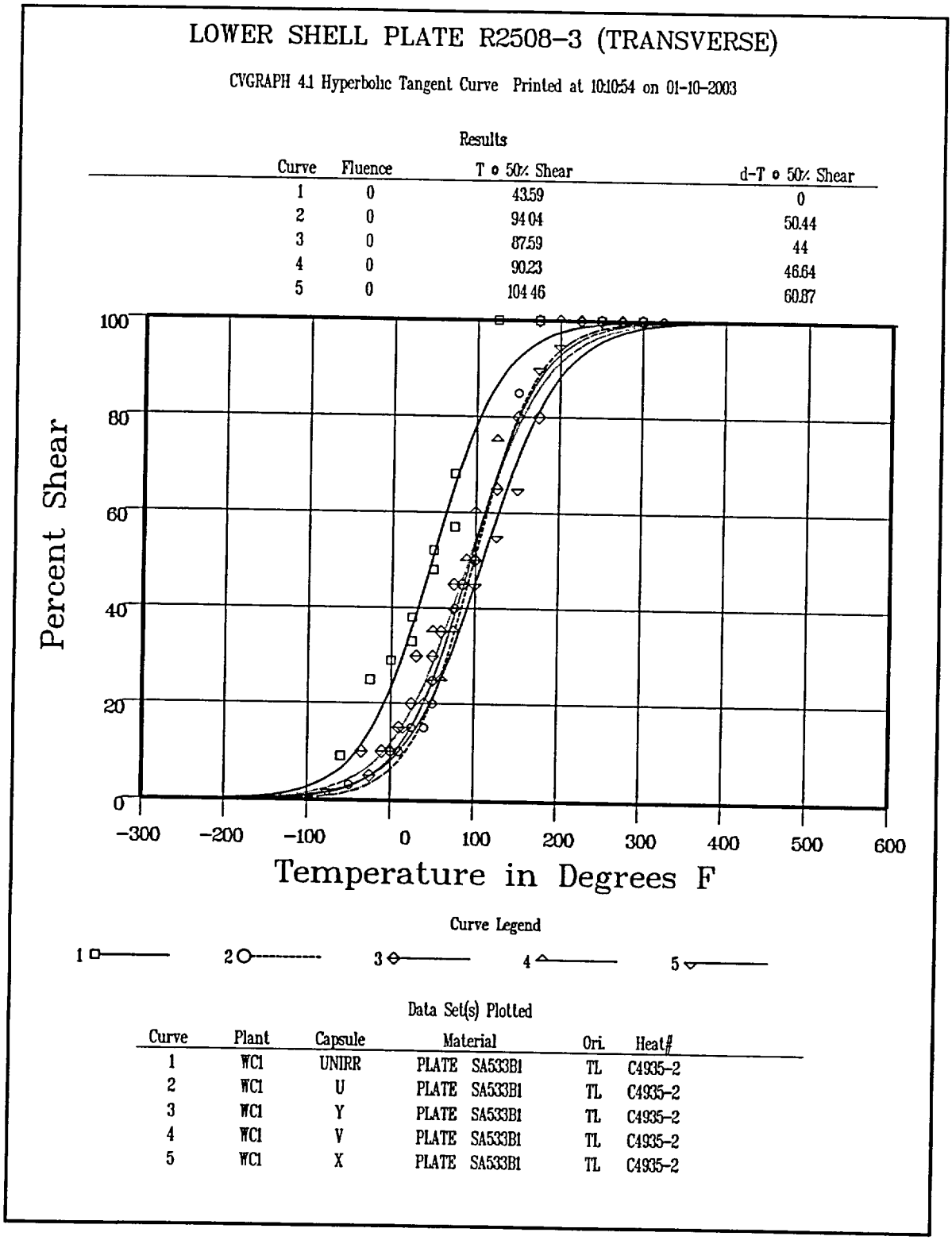


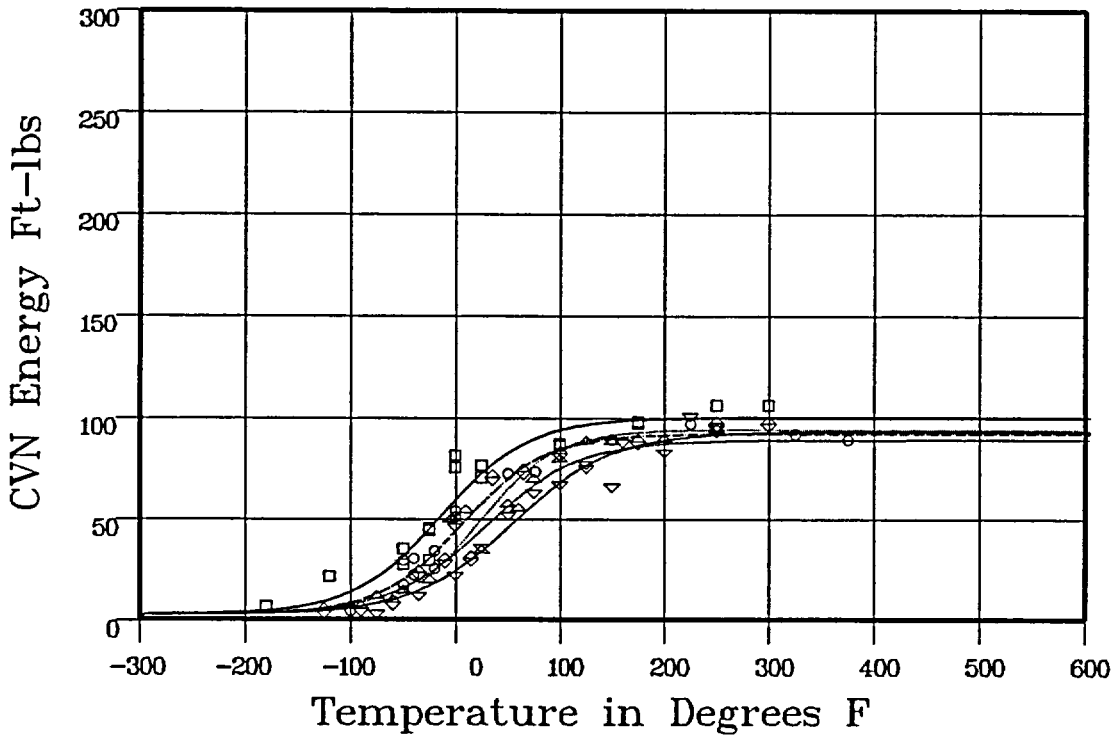
Figure 5-6 Charpy V-Notch Percent Shear vs. Temperature for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Transverse Orientation)

SURVEILLANCE PROGRAM WELD METAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:18:28 on 01-10-2003

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	100	0	-57.69	0	-20.64	0
2	0	2.19	0	92	-8	-30.47	27.21	6.44	27.09
3	0	2.19	0	94	-6	-12.59	45.09	20.82	41.47
4	0	2.19	0	89	-11	-11.36	46.33	31.79	52.44
5	0	2.19	0	93	-7	10.66	68.36	54.73	75.38



Curve Legend

1 □——— 2 ○- - - - - 3 ◇——— 4 △——— 5 ▽———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WC1	UNIRR	WELD	WIRE	HEAT NO.90146
2	WC1	U	WELD	WIRE	HEAT NO.90146
3	WC1	Y	WELD	WIRE	HEAT NO.90146
4	WC1	V	WELD	WIRE	HEAT NO.90146
5	WC1	X	WELD	WIRE	HEAT NO.90146

Figure 5-7 Charpy V-Notch Impact Energy vs. Temperature for Wolf Creek Reactor Vessel Weld Metal

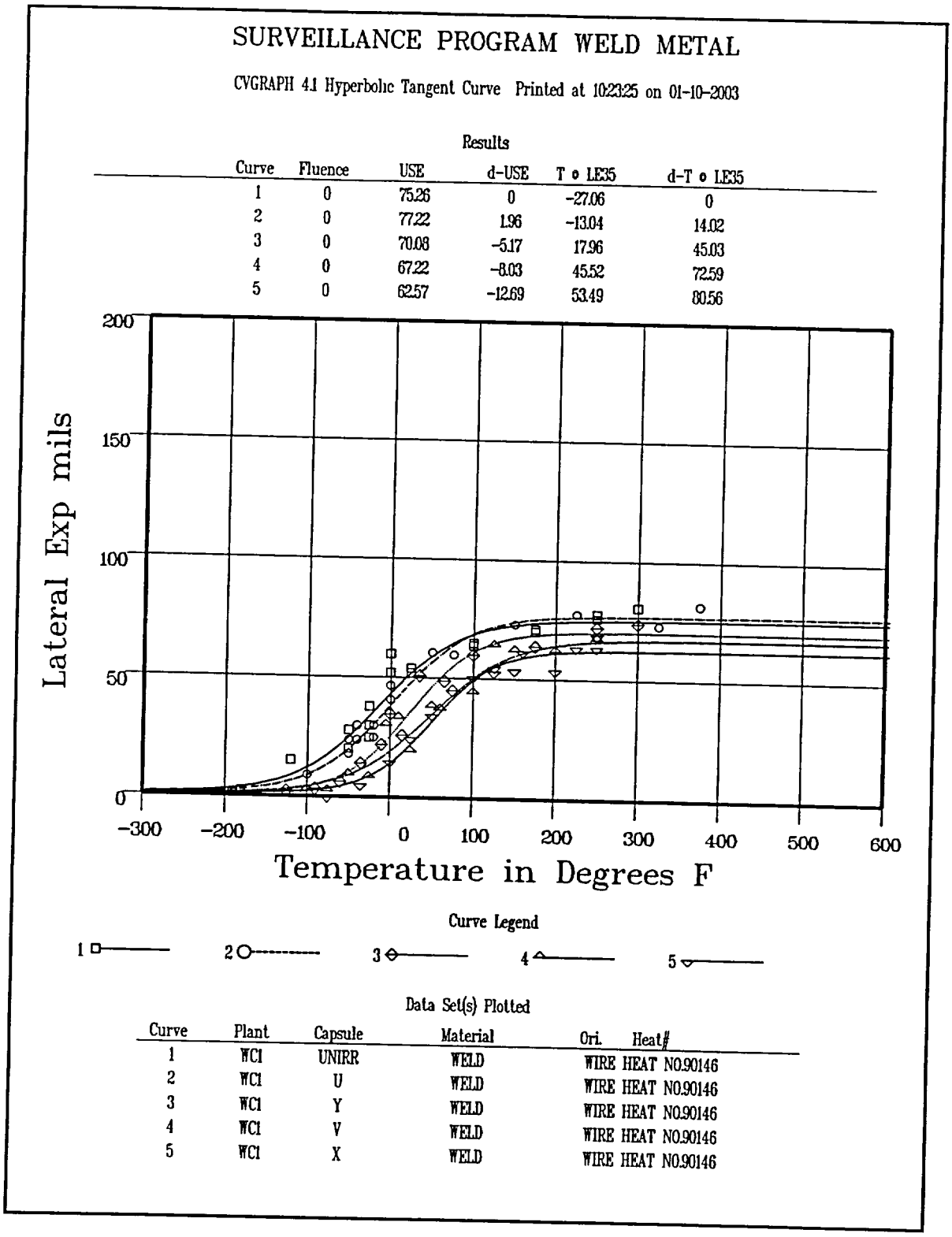


Figure 5-8 **Charpy V-Notch Lateral Expansion vs. Temperature for Wolf Creek Reactor Vessel Weld Metal**

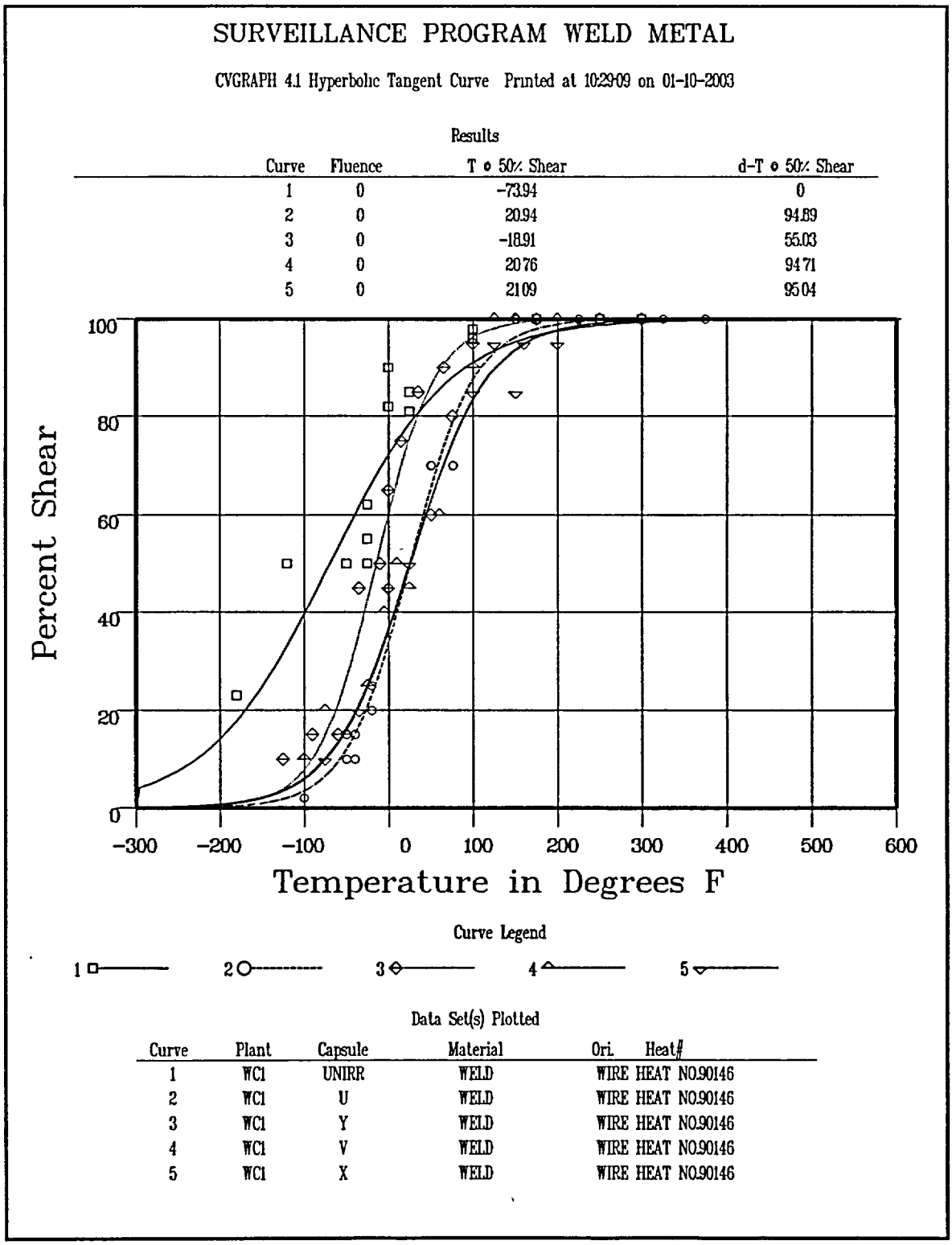


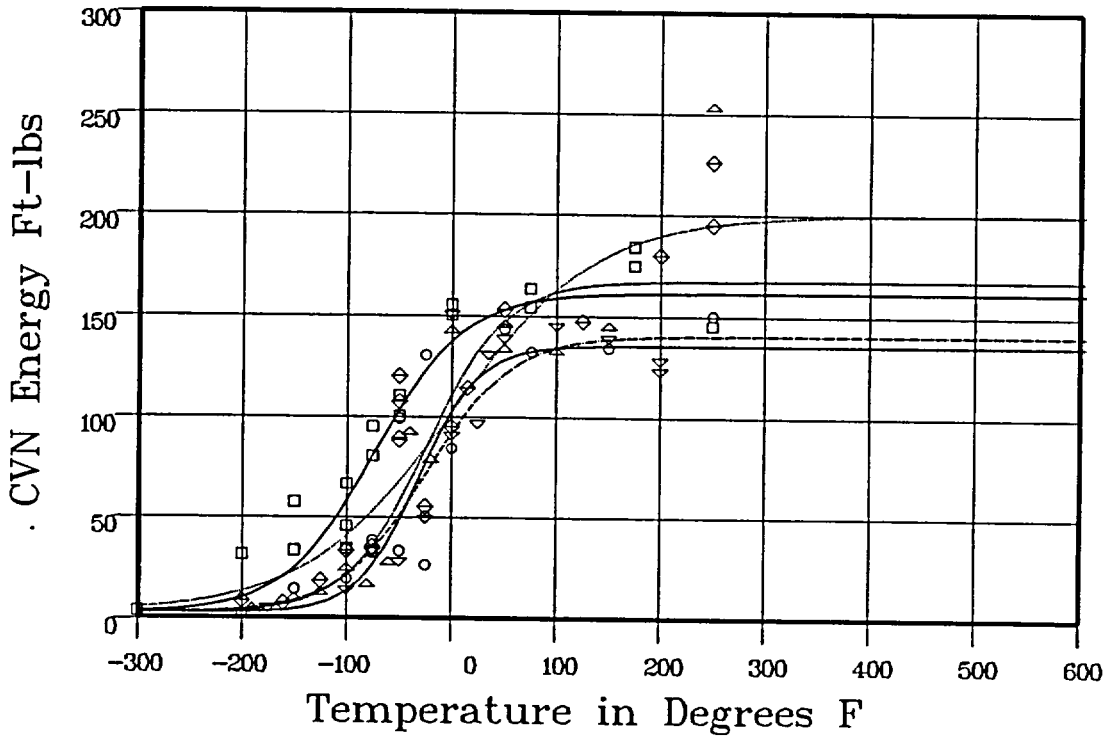
Figure 5-9 Charpy V-Notch Percent Shear vs. Temperature for Wolf Creek Reactor Vessel Weld Metal

HEAT AFFECTED ZONE

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:34:30 on 01-10-2003

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T o 30	d-T o 30	T o 50	d-T o 50
1	0	2.19	0	161	0	-144.01	0	-114	0
2	0	2.19	0	140	-21	-85.59	58.41	-55.32	58.68
3	0	2.19	0	200	39	-131.02	12.98	-84.82	29.18
4	0	2.19	0	167	6	-88.09	55.91	-61.99	52.01
5	0	2.19	0	135	-26	-74.34	69.66	-52.92	61.07



Curve Legend

1 □——— 2 ○- - - - - 3 ◇——— 4 △——— 5 ▽———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WC1	UNIRR	HEAT AFFD ZONE	WIRE	HEAT N0.90146
2	WC1	U	HEAT AFFD ZONE	WIRE	HEAT N0.90146
3	WC1	Y	HEAT AFFD ZONE	WIRE	HEAT N0.90146
4	WC1	V	HEAT AFFD ZONE	WIRE	HEAT N0.90146
5	WC1	X	HEAT AFFD ZONE	WIRE	HEAT N0.90146

Figure 5-10 Charpy V-Notch Impact Energy vs. Temperature for Wolf Creek Reactor Vessel Heat-Affected-Zone Material

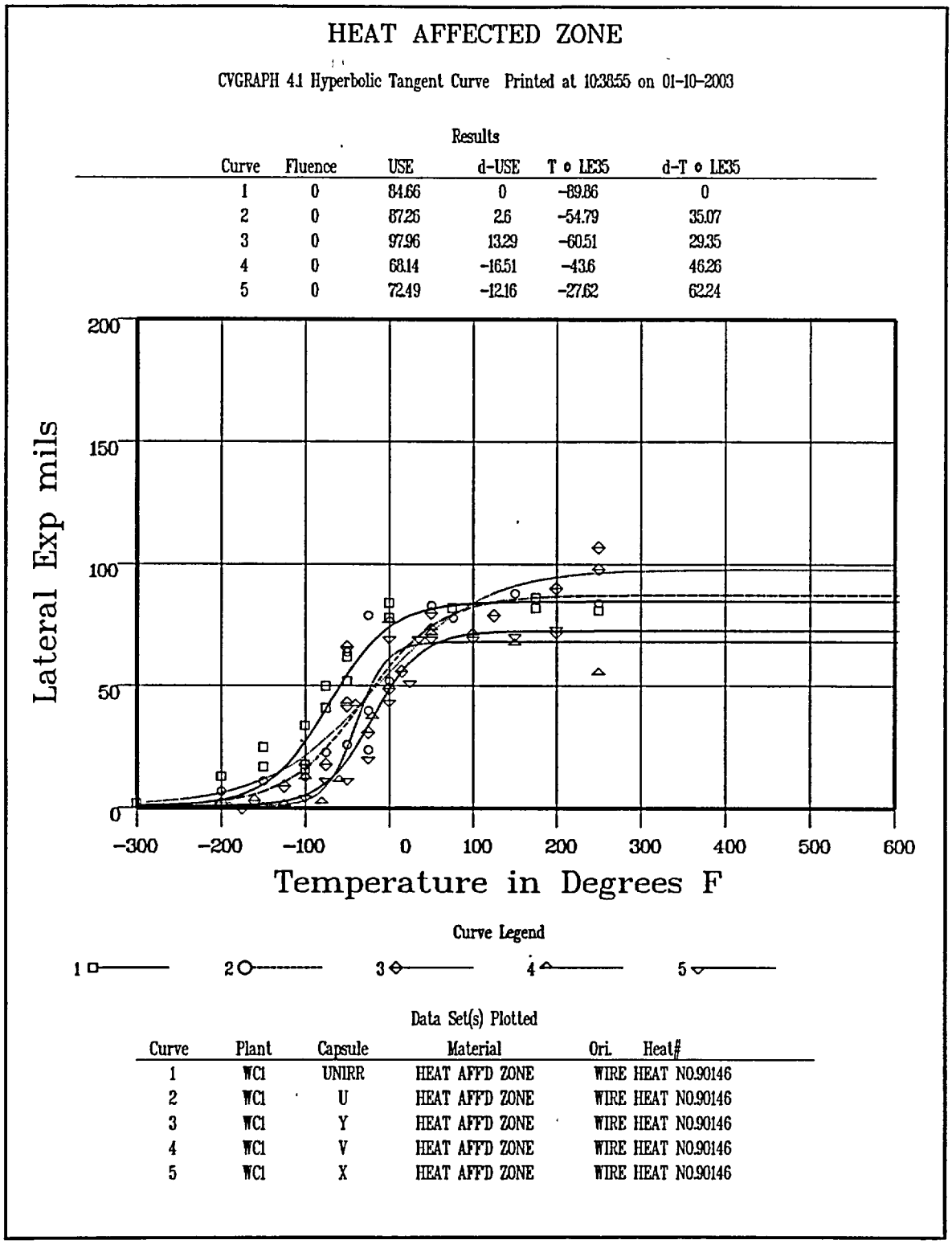


Figure 5-11 Charpy V-Notch Lateral Expansion vs. Temperature for Wolf Creek Reactor Vessel Heat-Affected-Zone Material

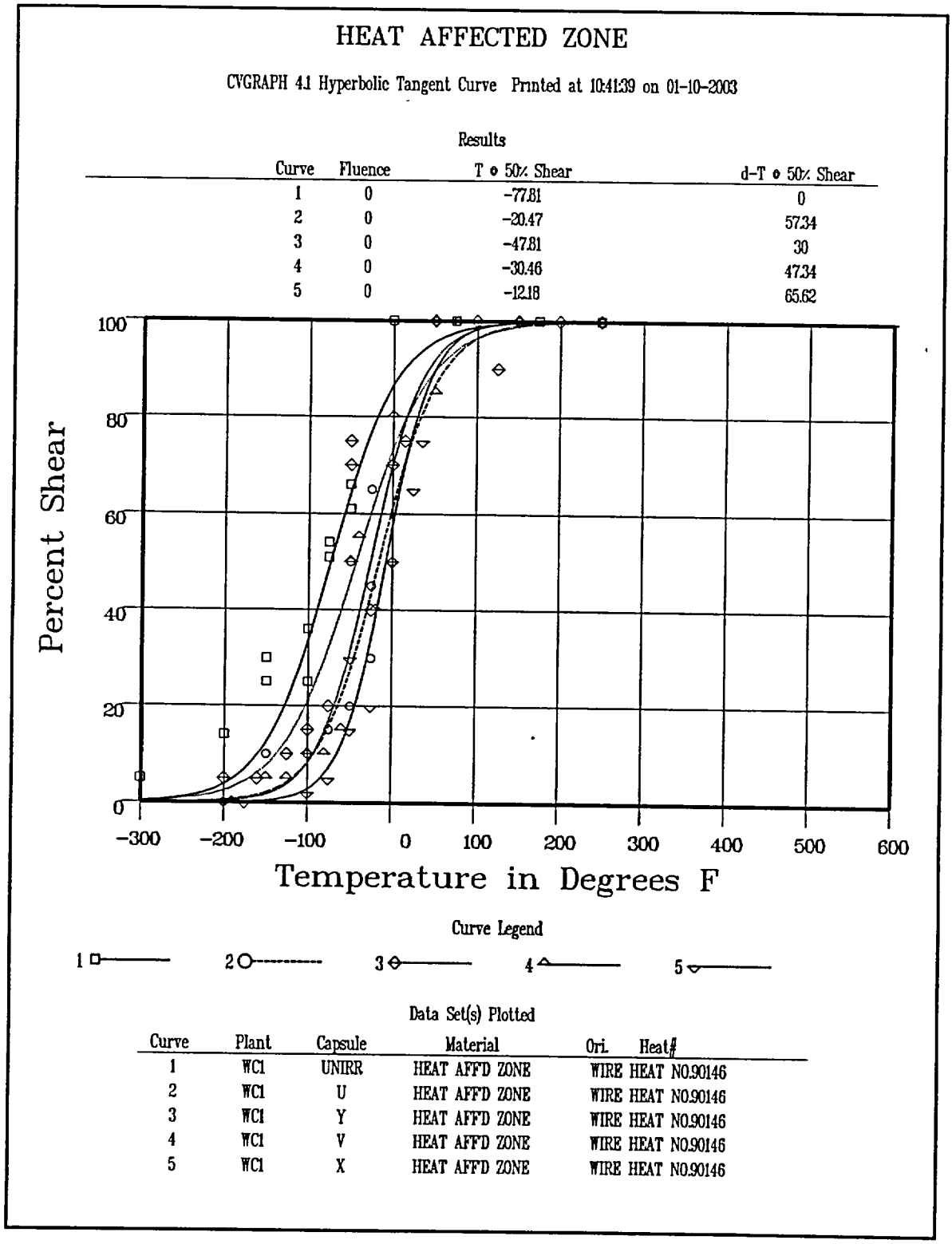


Figure 5-12 Charpy V-Notch Percent Shear vs. Temperature for Wolf Creek Reactor Vessel Heat-Affected-Zone Material

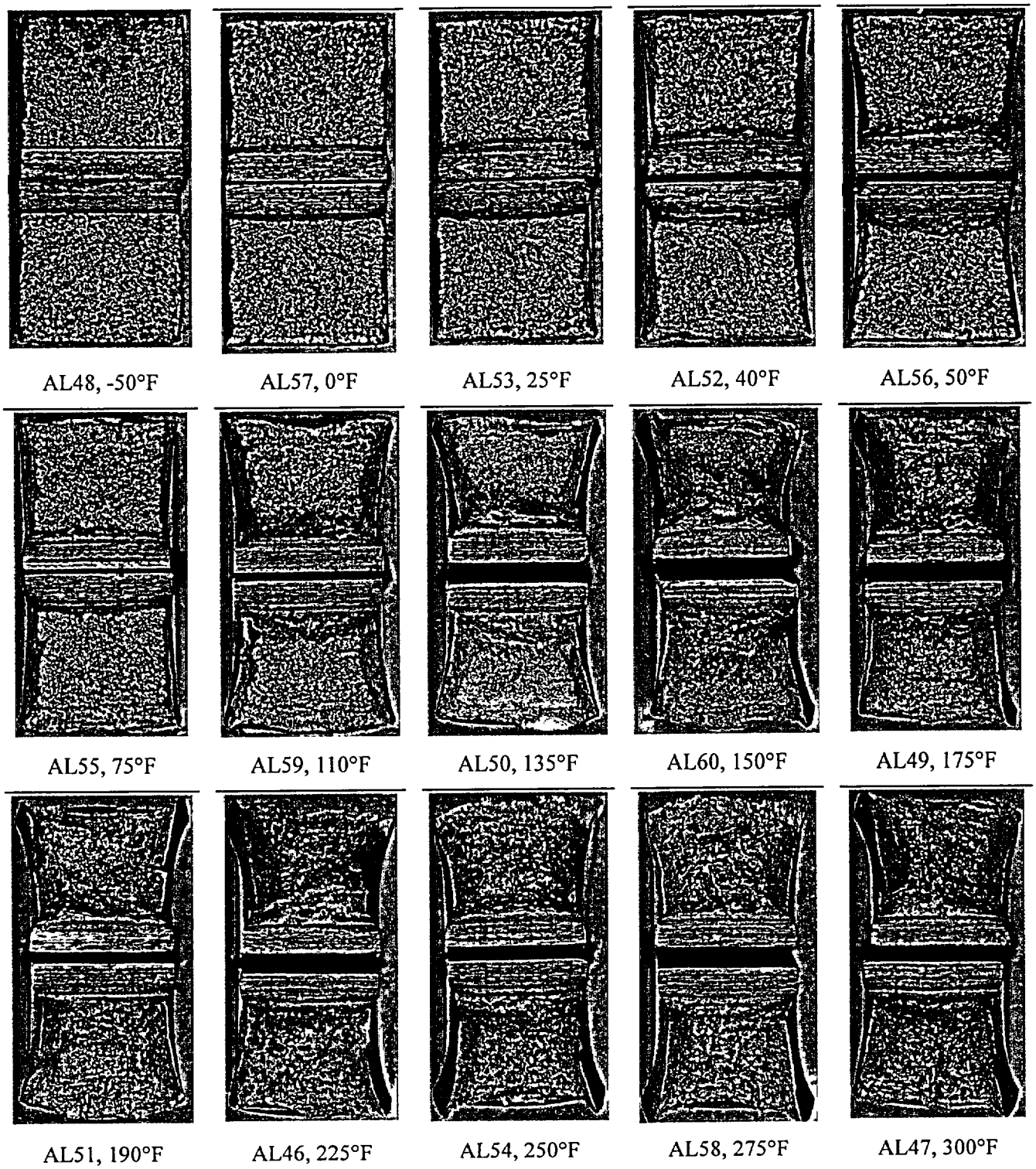


Figure 5-13 Charpy Impact Specimen Fracture Surfaces for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Longitudinal Orientation)

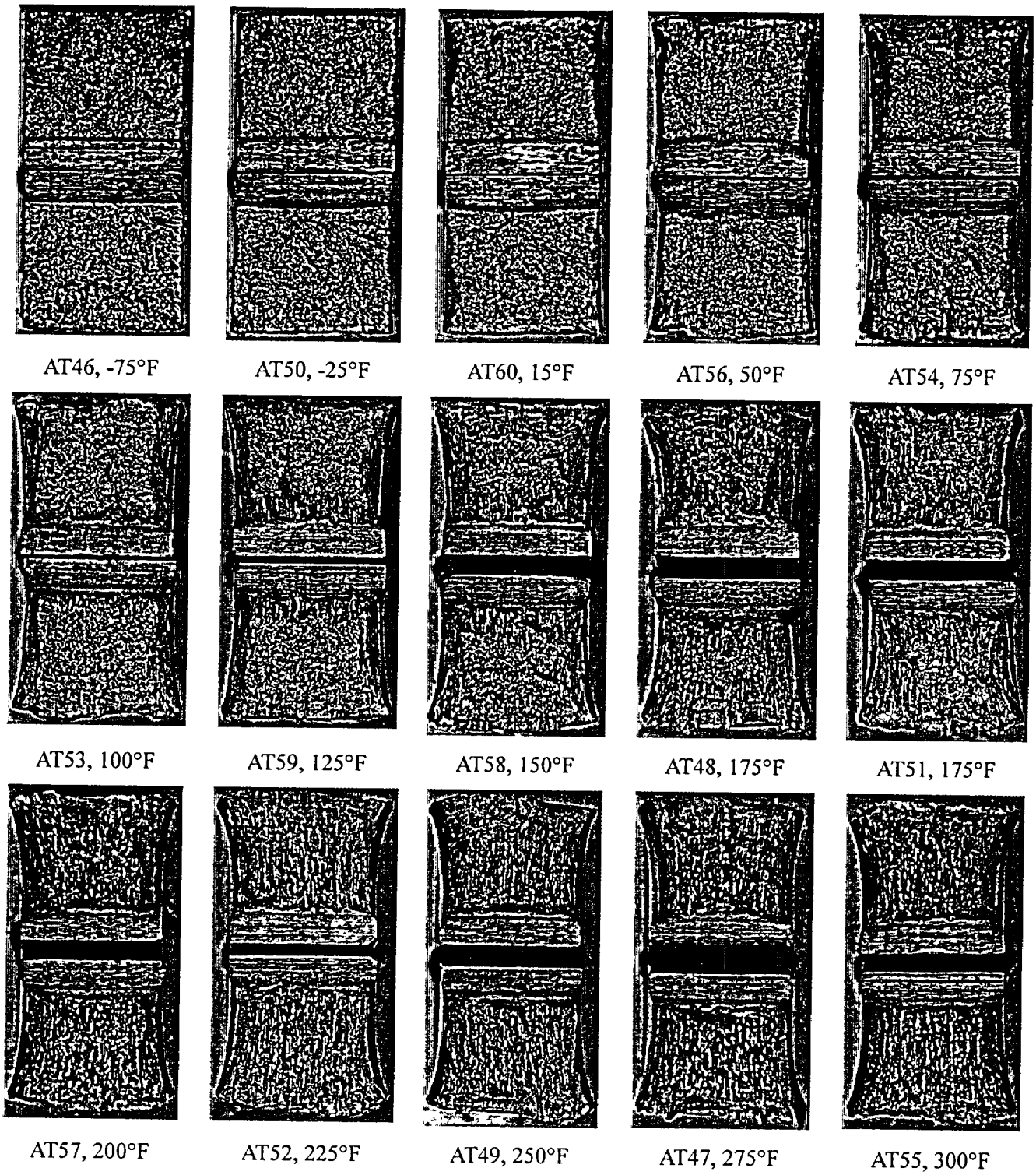


Figure 5-14 Charpy Impact Specimen Fracture Surfaces for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Transverse Orientation)

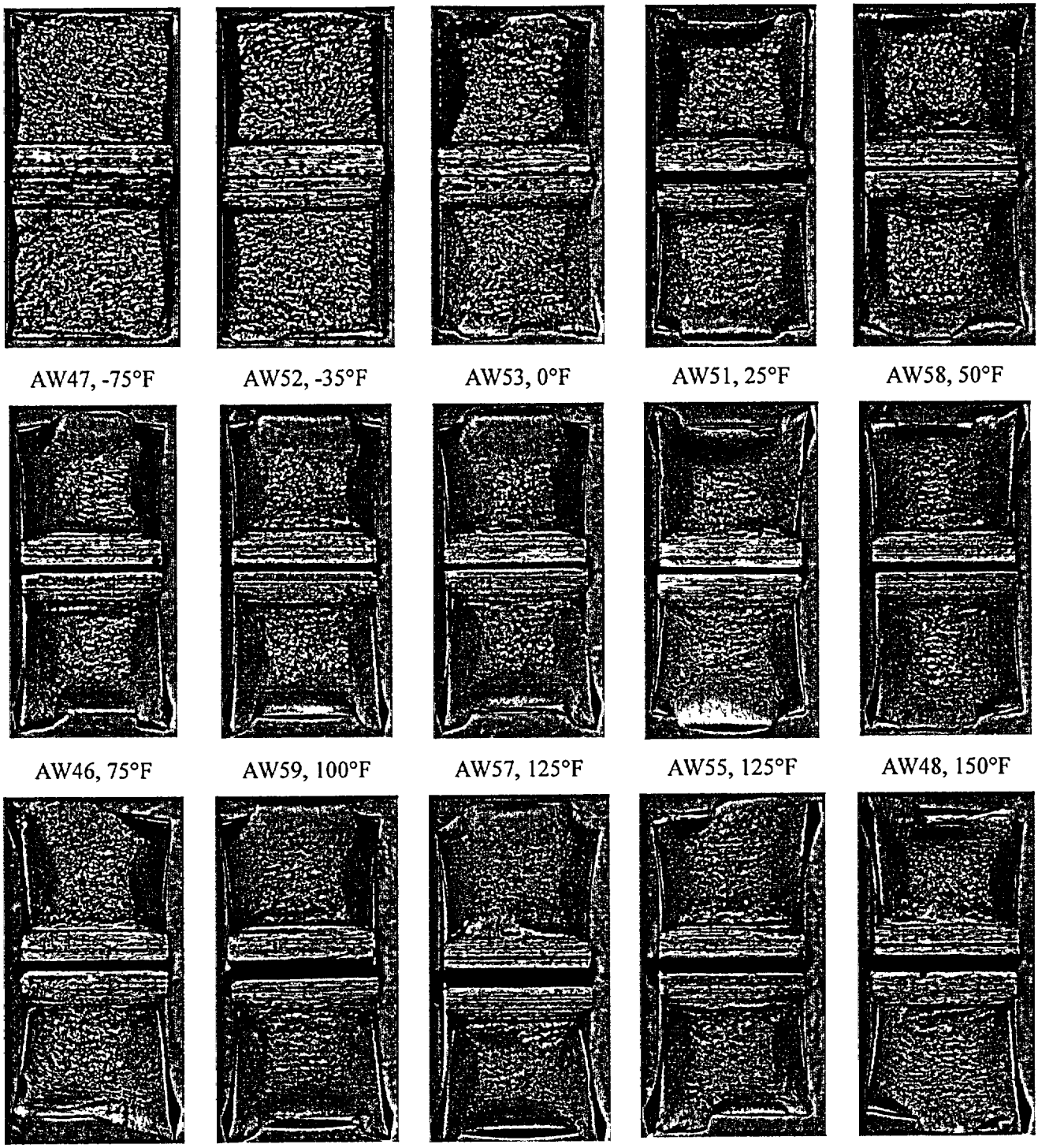


Figure 5-15 Charpy Impact Specimen Fracture Surfaces for Wolf Creek Reactor Vessel Weld Metal

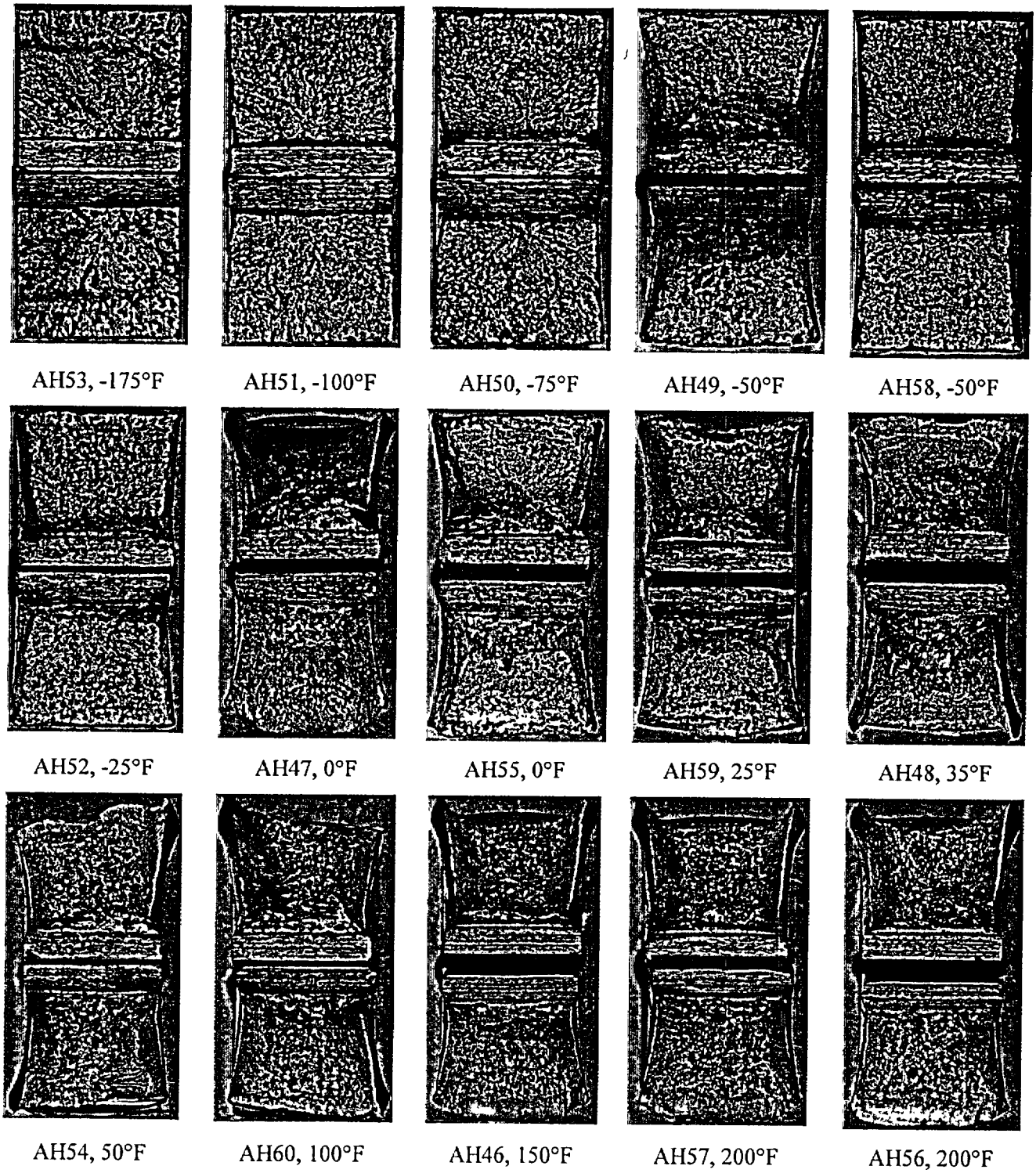


Figure 5-16 Charpy Impact Specimen Fracture Surfaces for Wolf Creek Reactor Vessel Heat-Affected-Zone Metal

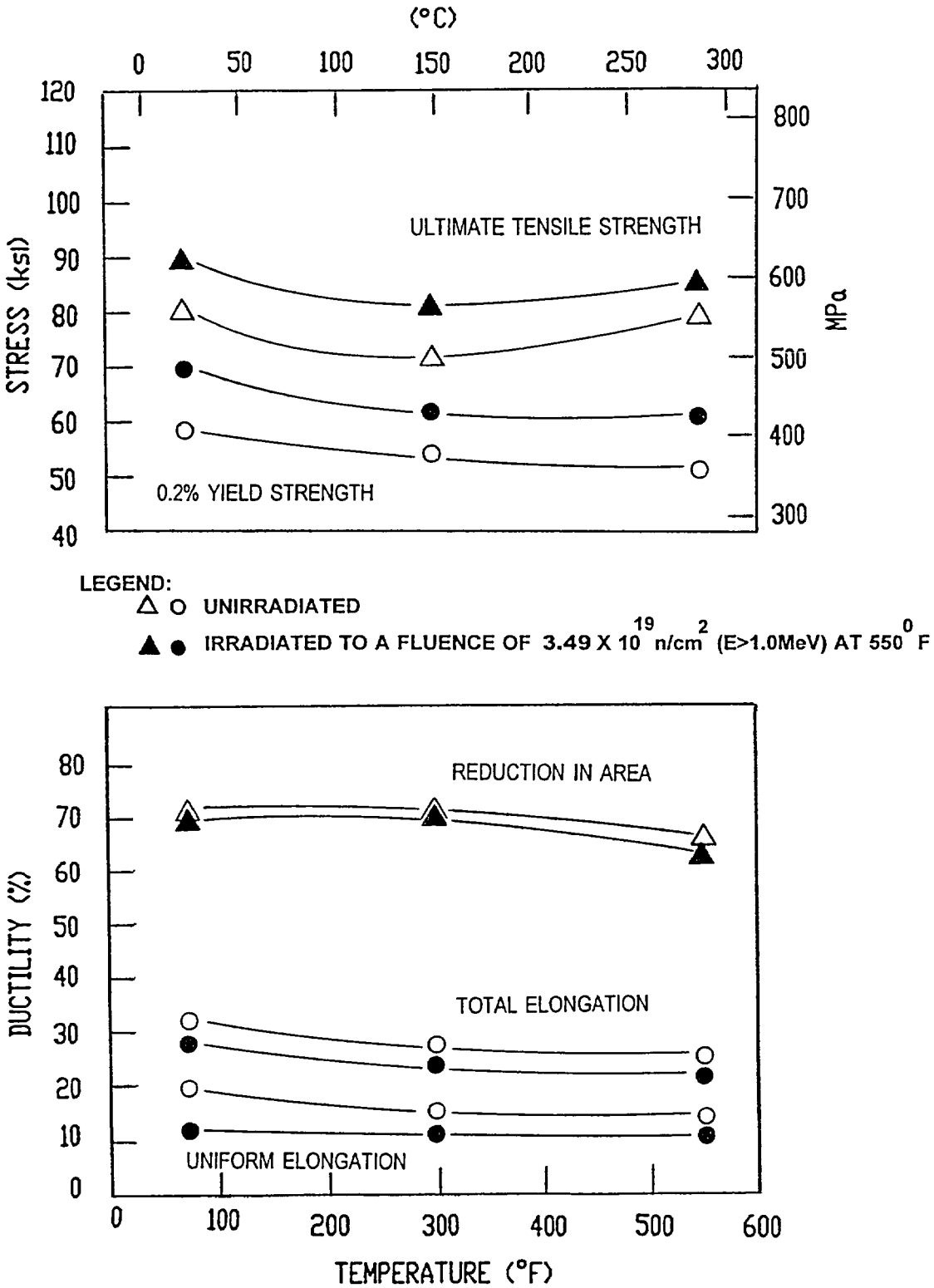


Figure 5-17 Tensile Properties for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Longitudinal Orientation)

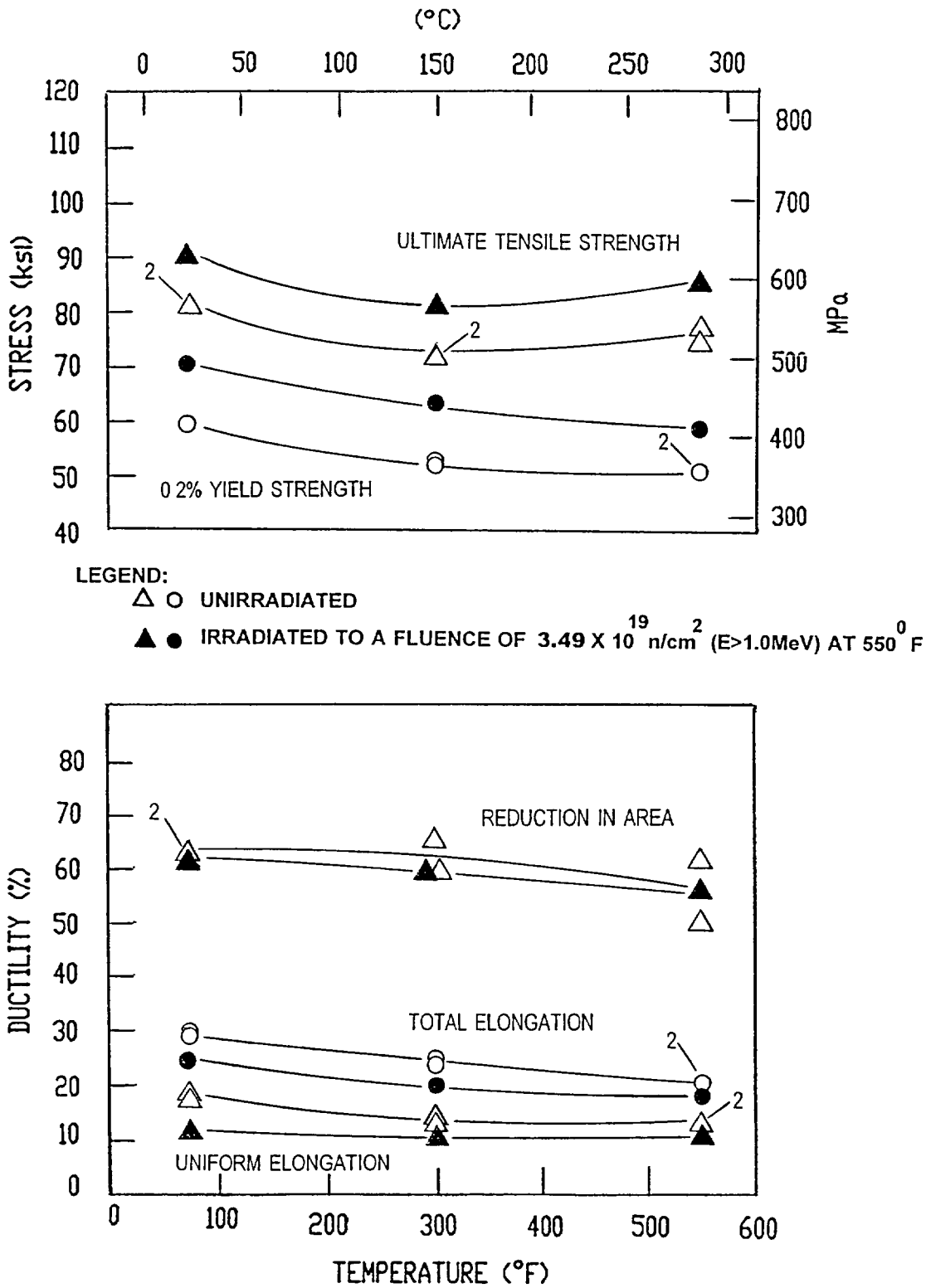


Figure 5-18 Tensile Properties for Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Transverse Orientation)

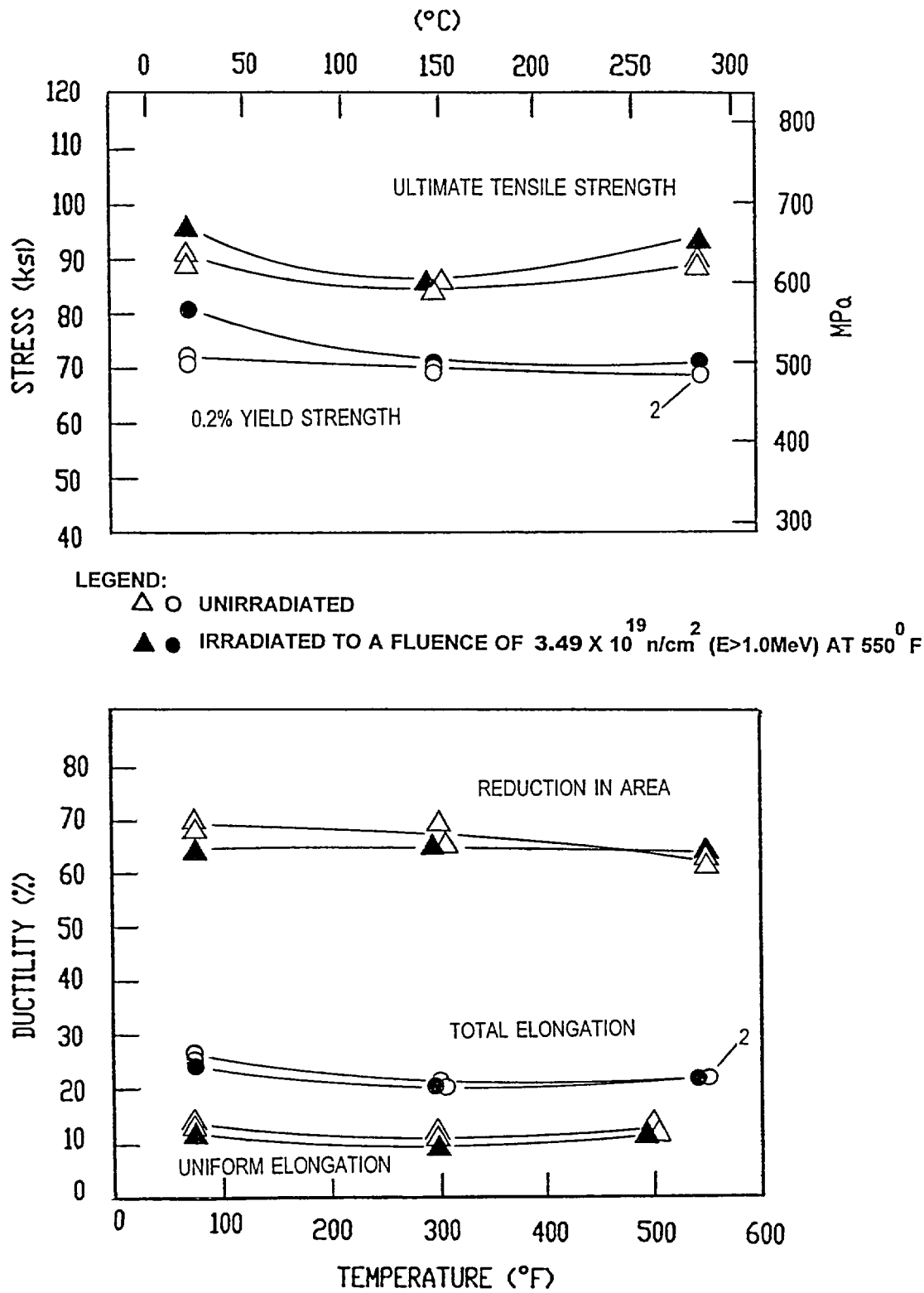
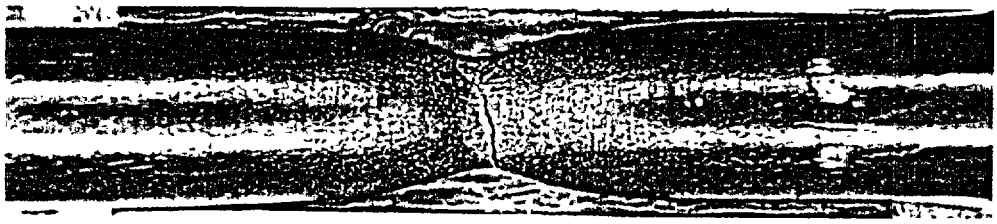
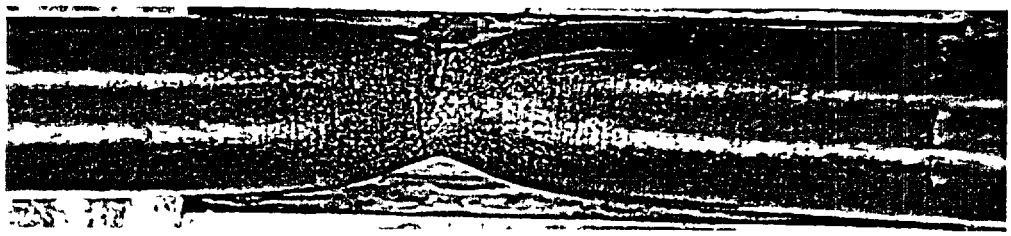


Figure 5-19 Tensile Properties for Wolf Creek Reactor Vessel Weld Metal



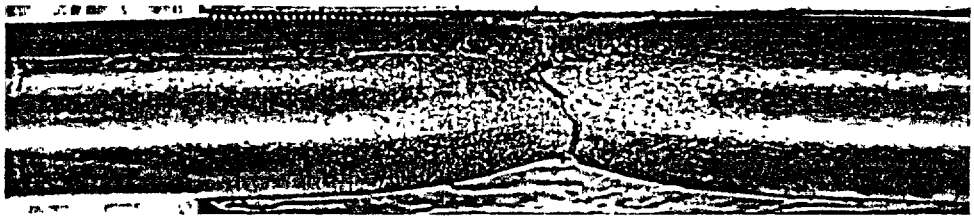
Specimen AL10 Tested at 75°F



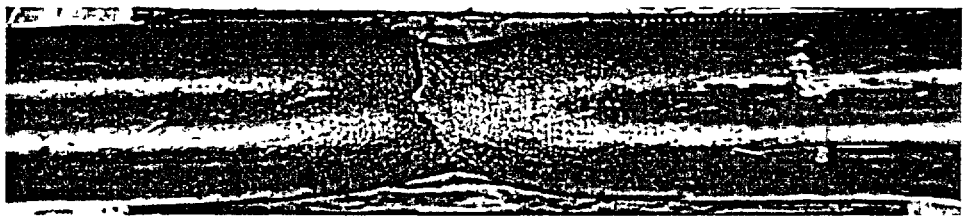
Specimen AL11 Tested at 300°F

Specimen AL12 Tested at 550°F

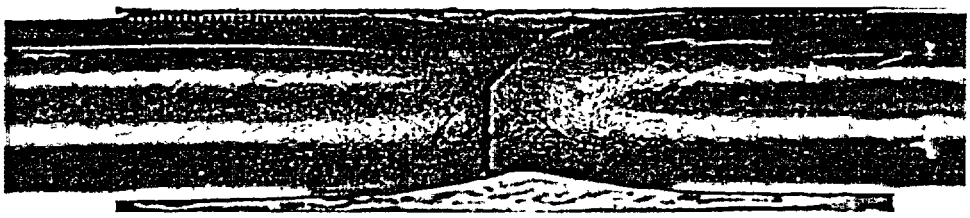
Figure 5-20 Fractured Tensile Specimens from Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Longitudinal Orientation)



Specimen AT10 Tested at 75°F

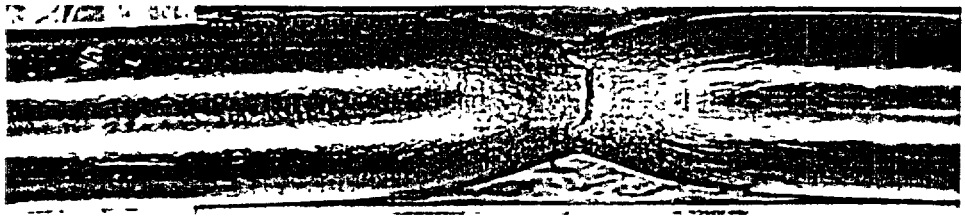


Specimen AT11 Tested at 300°F



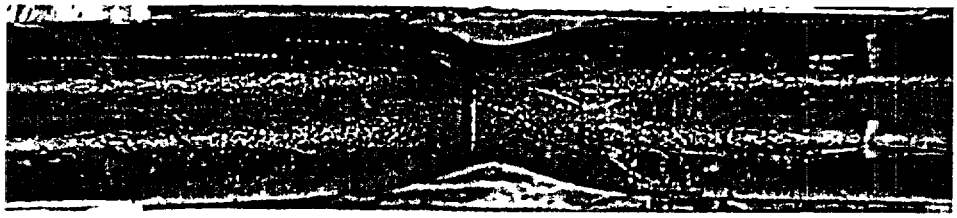
Specimen AT12 Tested at 550°F

Figure 5-21 Fractured Tensile Specimens from Wolf Creek Reactor Vessel Lower Shell Plate R2508-3 (Transverse Orientation)



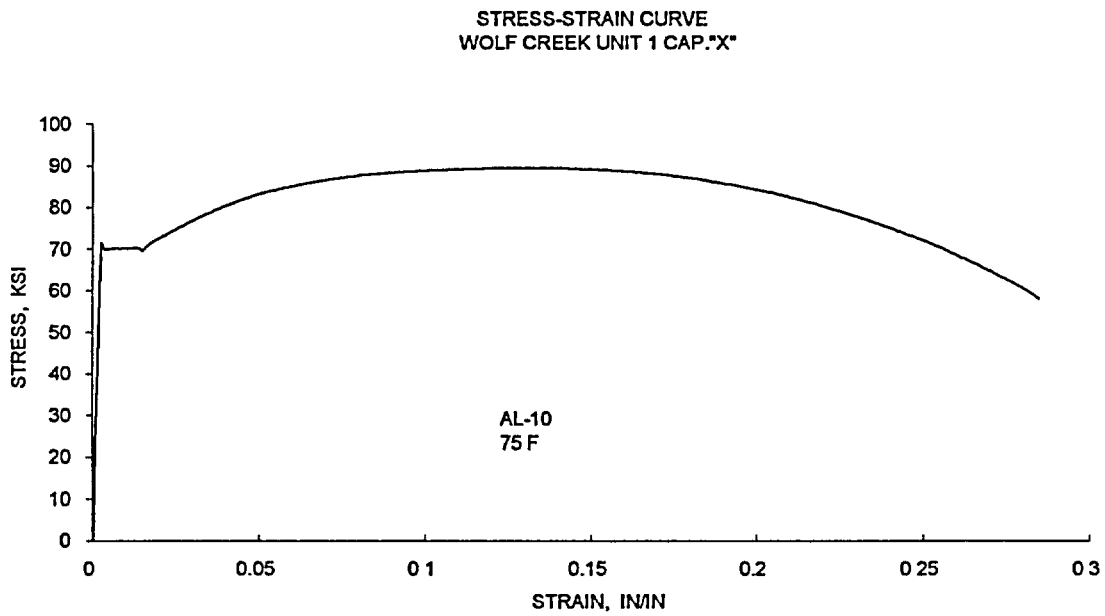
Specimen AW10 Tested at 75°F

Specimen AW11 Tested at 300°F



Specimen AW12 Tested at 550°F

Figure 5-22 Fractured Tensile Specimens from Wolf Creek Reactor Vessel Weld Metal



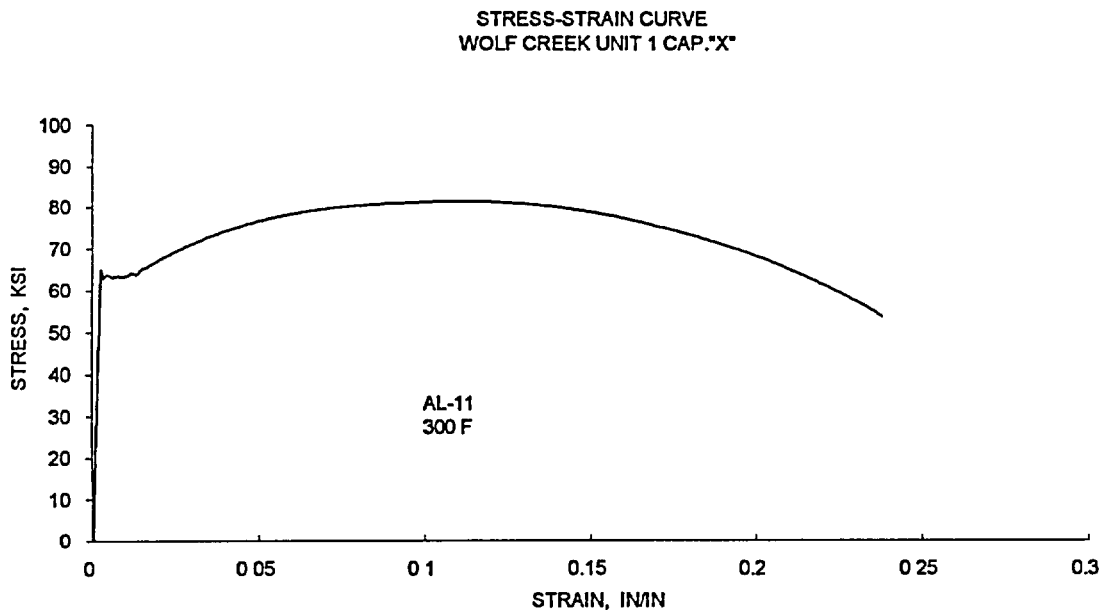


Figure 5-23 Engineering Stress-Strain Curves for Wolf Creek Lower Shell Plate R2508-3 Tensile Specimens AL-10, AL-11 and AL-12 (Longitudinal Orientation)

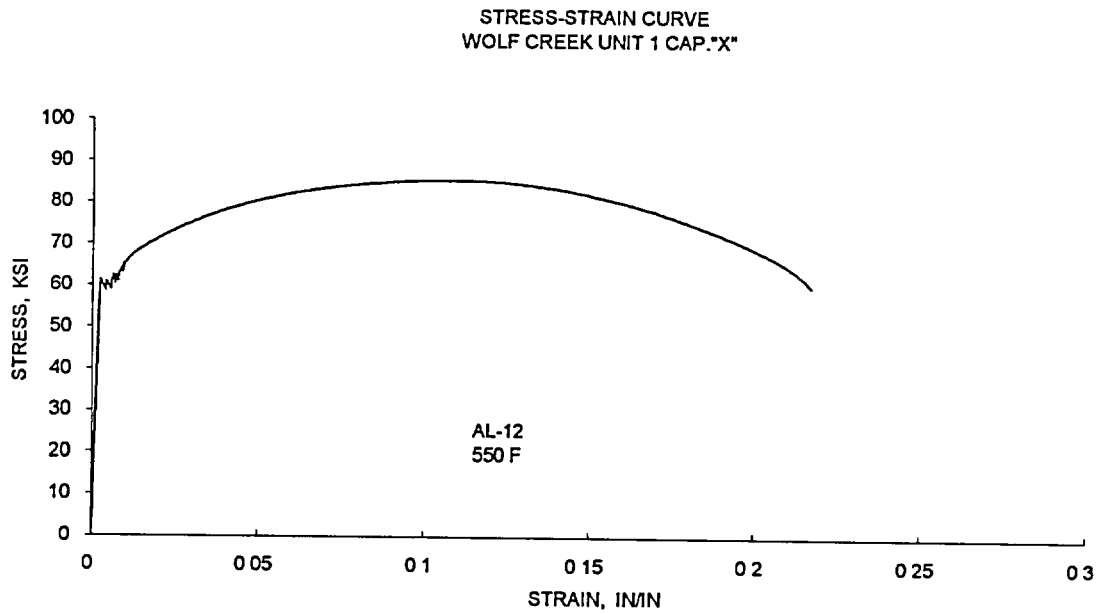
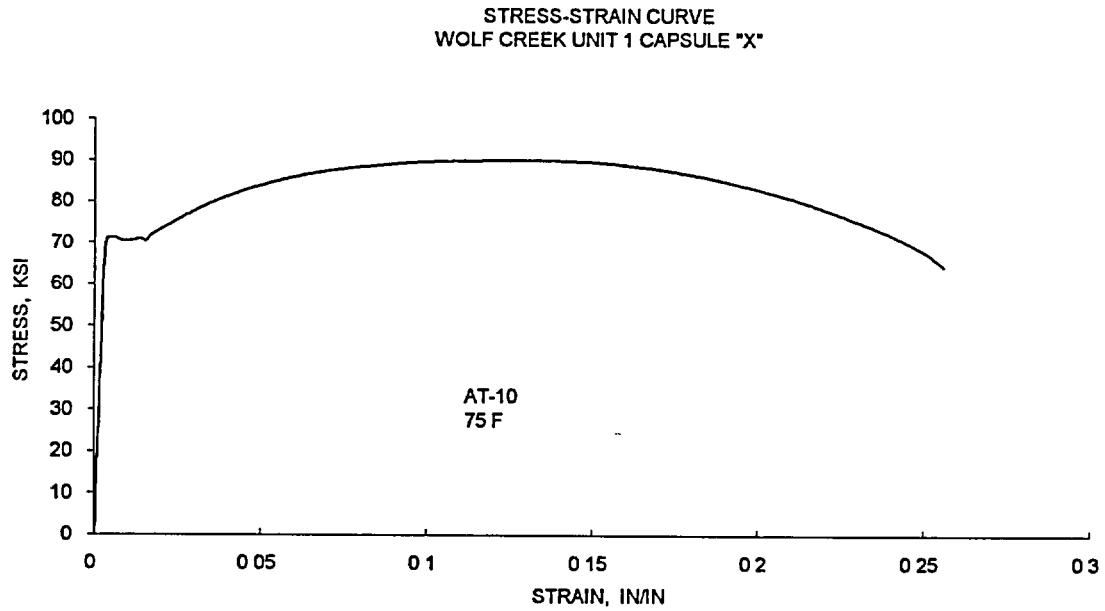


Figure 5-23 - Continued



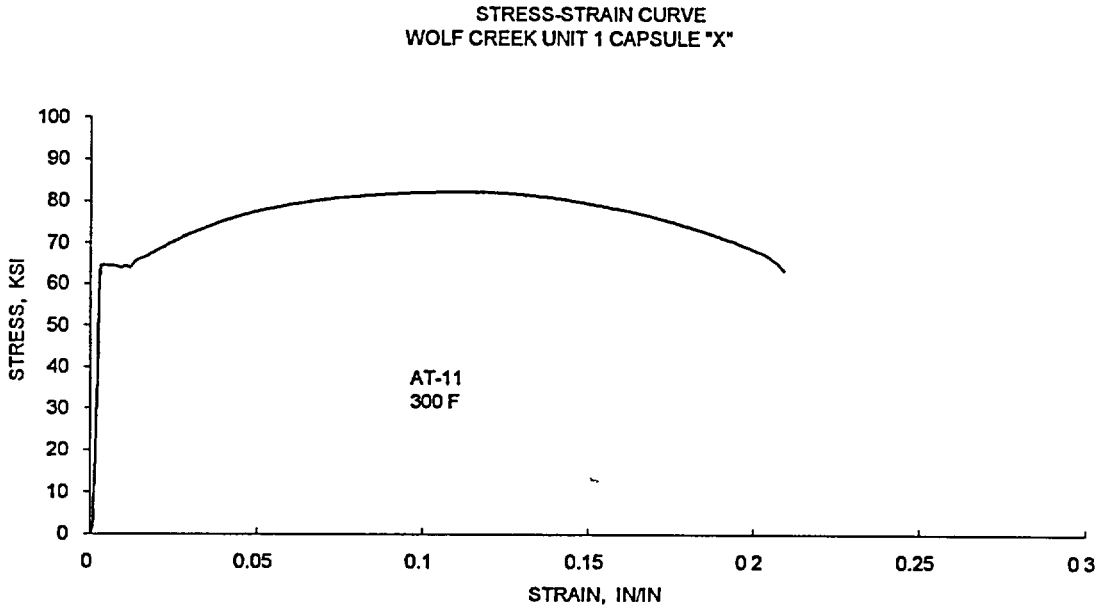


Figure 5-24 **Engineering Stress-Strain Curves for Wolf Creek Lower Shell Plate R2508-3 Tensile Specimens AT-10, AT-11 and AT-12 (Longitudinal Orientation)**

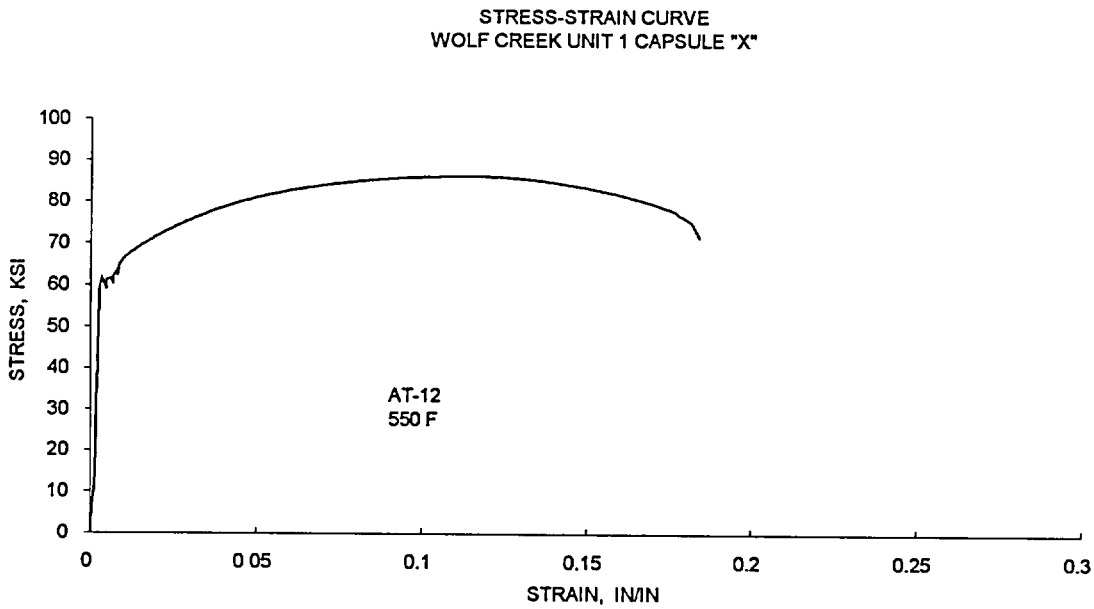
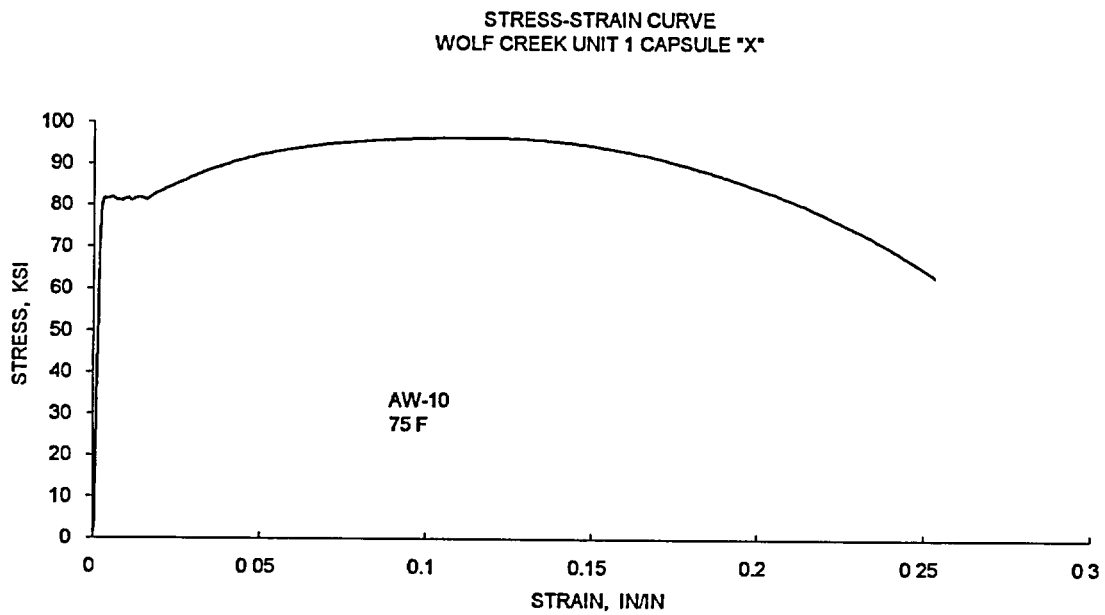


Figure 5-24 - Continued



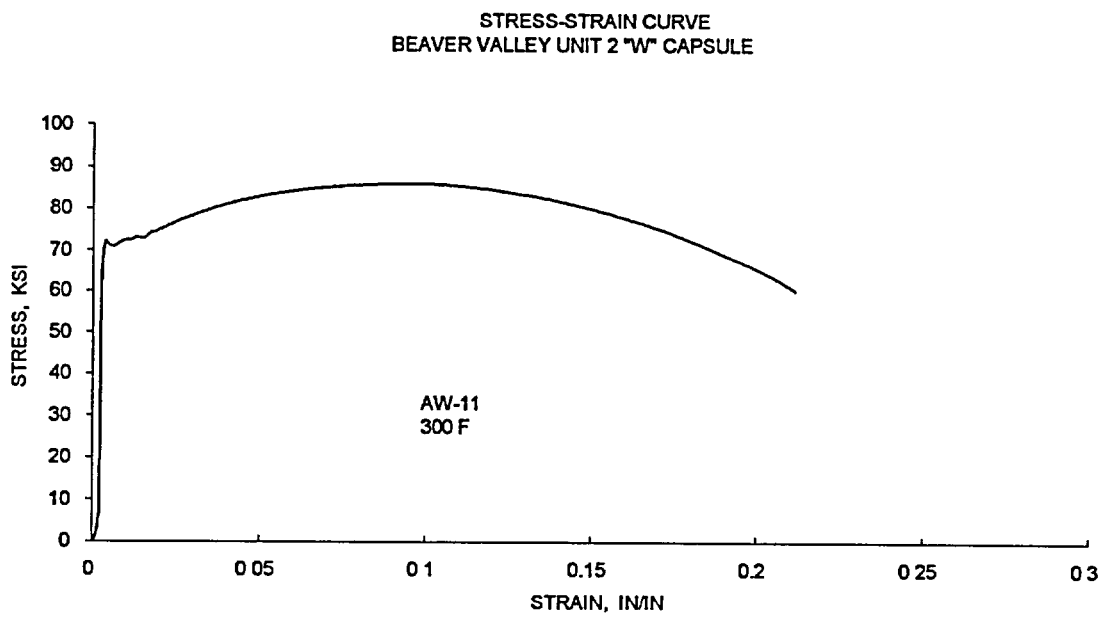


Figure 5-25 Engineering Stress-Strain Curves for Weld Metal Tensile Specimens AW-10, AW-11 and AW-12

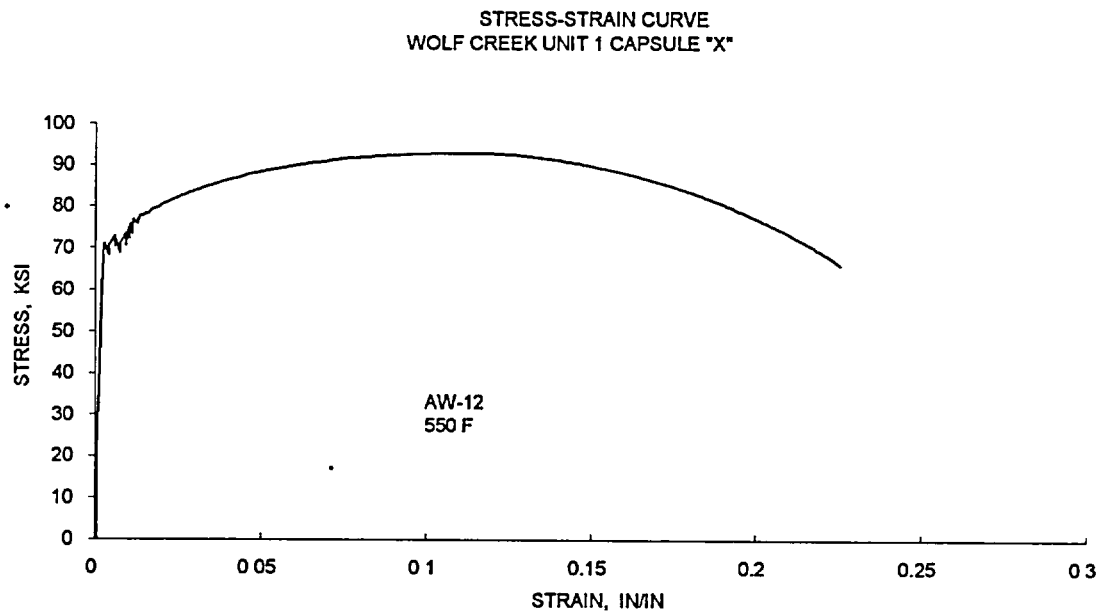


Figure 5-25 - Continued

6 RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 INTRODUCTION

This section describes a discrete ordinates S_n transport analysis performed for the Wolf Creek reactor to determine the neutron radiation environment within the reactor pressure vessel and surveillance capsules. In this analysis, fast neutron exposure parameters in terms of fast neutron fluence (E > 1.0 MeV) and iron atom displacements (dpa) were established on a plant and fuel cycle specific basis. An evaluation of the most recent dosimetry sensor set from Capsule X, withdrawn at the end of the twelfth plant operating cycle, is provided. In addition, in order to provide a complete measurement database applicable to Wolf Creek, results from prior in-vessel irradiations are included in Appendix A to this report. The data included in Appendix A were previously documented in Reference 3. Comparisons of the results from these dosimetry evaluations with the analytical predictions served to validate the plant specific neutron transport calculations. These validated calculations subsequently formed the basis for providing projections of the neutron exposure of the reactor pressure vessel for operating periods extending to 54 Effective Full Power Years (EFPY). These projections also account for a plant uprating, from 3411 MWt to 3565 MWt, which occurred during and post the seventh operating cycle.

The use of fast neutron fluence (E > 1.0 MeV) to correlate measured material property changes to the neutron exposure of the material has traditionally been accepted for the development of damage trend curves as well as for the implementation of trend curve data to assess the condition of the vessel. In recent years, however, it has been suggested that an exposure model that accounts for differences in neutron energy spectra between surveillance capsule locations and positions within the vessel wall could lead to an improvement in the uncertainties associated with damage trend curves and improved accuracy in the evaluation of damage gradients through the reactor vessel wall.

Because of this potential shift away from a threshold fluence toward an energy dependent damage function for data correlation, ASTM Standard Practice E853, "Analysis and Interpretation of Light-Water Reactor Surveillance Results," recommends reporting displacements per iron atom (dpa) along with fluence (E > 1.0 MeV) to provide a database for future reference. The energy dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693, "Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements per Atom." The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the reactor vessel wall has already been promulgated in Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

All of the calculations and dosimetry evaluations described in this section and in Appendix A were based on the latest available nuclear cross-section data derived from ENDF/B-VI and made use of the latest available calculational tools. Furthermore, the neutron transport and dosimetry evaluation methodologies follow the guidance and meet the requirements of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,"[20]. Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996[21]. The specific calculational methods applied are also consistent with those described in WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology."[22]

methods applied are also consistent with those described in WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology." [22]

6.2 DISCRETE ORDINATES ANALYSIS

A plan view of the Wolf Creek reactor geometry at the core midplane is shown in Figures 6-1a-c. Six irradiation capsules attached to the neutron pads are included in the reactor design that constitutes the reactor vessel surveillance program. The capsules are located at azimuthal angles of 61° and 241 (29° from the core cardinal axes) and 58.5°, 121.5°, 238.5°, and 301.5° (31.5° from the core cardinal axes) as shown in Figure 4-1. The stainless steel specimen containers are 1.182-inch by 1-inch and approximately 56-inches in height. The containers are positioned axially such that the test specimens are centered on the core midplane, thus spanning the central 5 feet of the 12-foot high reactor core.

From a neutronic standpoint, the surveillance capsules and associated support structures are significant. The presence of these materials has a marked effect on both the spatial distribution of neutron flux and the neutron energy spectrum in the water annulus between the neutron pads and the reactor vessel. In order to determine the neutron environment at the test specimen location, the capsules themselves must be included in the analytical model.

In performing the fast neutron exposure evaluations for the Wolf Creek reactor vessel and surveillance capsules, a series of fuel cycle specific forward transport calculations were carried out using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = \phi(r,\theta) * \frac{\phi(r,z)}{\phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation. This synthesis procedure was carried out for each operating cycle at Wolf Creek.

For the Wolf Creek transport calculations, the r,θ models depicted in Figures 6-1a-c were utilized since the reactor is octant symmetric. This r,θ model includes the core, the reactor internals, the neutron pad -- including explicit representations of the surveillance capsules at 29° and 31.5°, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological shield wall. These models formed the basis for the calculated results and enabled making comparisons to the surveillance capsule dosimetry evaluations. In developing this analytical model set, nominal design dimensions were employed for the various structural components. Likewise, water temperatures, and hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full power operating conditions. The coolant densities were treated on a fuel cycle specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, et cetera. The geometric mesh

description of the r,θ reactor model consisted of 183 radial by 99 azimuthal intervals. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,θ calculations was set at a value of 0.001.

The r,z model used for the Wolf Creek calculations that is shown in Figure 6-2 extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation approximately 1.5-feet below the active fuel to approximately 2.5-feet above the active fuel. As in the case of the r,θ model, nominal design dimensions and full power coolant densities were employed in the calculations. In this case, the homogenous core region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel former plates located between the core baffle and core barrel regions were also explicitly included in the model. The r,z geometric mesh description of the reactor model consisted of 153 radial by 107 axial intervals. As in the case of the r,θ calculations, mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,z calculations was also set at a value of 0.001.

The one-dimensional radial model used in the synthesis procedure consisted of the same 153 radial mesh intervals included in the r,z model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

The core power distributions used in the plant specific transport analysis were taken from the appropriate Wolf Creek fuel cycle design reports. The data extracted from the design reports represented cycle dependent fuel assembly enrichments, burnups, and axial power distributions. This information was used to develop spatial and energy dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of fuel cycle averaged neutron flux, which when multiplied by the appropriate fuel cycle length, generated the incremental fast neutron exposure for each fuel cycle. In constructing these core source distributions, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of individual fuel assemblies. From these assembly dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

All of the transport calculations supporting this analysis were carried out using the DORT discrete ordinates code Version 3.1[23] and the BUGLE-96 cross-section library[24]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and angular discretization was modeled with an S_{16} order of angular quadrature for the r and r,z models while an S_8 order of angular quadrature was used in the r,θ models. Energy and space dependent core power distributions, as well as system operating temperatures, were treated on a fuel cycle specific basis.

Selected results from the neutron transport analyses are provided in Tables 6-1 through 6-6. In Table 6-1, the calculated exposure rates and integrated exposures, expressed in terms of both neutron fluence (E > 1.0 MeV) and dpa, are given at the radial and azimuthal center of the two azimuthally symmetric surveillance capsule positions (29° and 31.5°). Also note that Table 6-1 presents calculated exposure rates and integrated exposures for Capsule X, which was irradiated at a 31.5° location during Cycles 1 through 12 until it was removed from service. These results, representative of the axial midplane of the active core, establish the calculated exposure of the surveillance capsules withdrawn to date as well as projected into the future. Similar information is provided in Table 6-2 for the reactor vessel inner radius. The vessel data given in Table 6-2 are representative of the axial location of the maximum neutron exposure at each of the four azimuthal locations. It is also important to note that the data for the vessel inner radius were taken at the clad/base metal interface, and thus, represent the maximum calculated exposure levels of the vessel forgings and welds.

Both calculated fluence (E > 1.0 MeV) and dpa data are provided in Tables 6-1 and 6-2. These data tabulations include both plant and fuel cycle specific calculated neutron exposures at the end of the twelfth operating fuel cycle as well as projections to 15.53, 20, 24, 32, 40, 48, and 54 effective full power years (EFPY). The projections were based on the assumption that the radial power distribution from fuel cycle 12 was representative of future plant operation (excluding cycle 13 projections). All remaining core parameters were obtained by utilizing cycle 12 (excluding cycle 13 projections). The future projections are also based on the current reactor power level of 3565 MWt.

Radial gradient information applicable to fast (E > 1.0 MeV) neutron fluence and dpa are given in Tables 6-3 and 6-4, respectively. The data, based on the cumulative integrated exposures from Cycles 1 through 12, are presented on a relative basis for each exposure parameter at several azimuthal locations. Exposure distributions through the vessel wall may be obtained by multiplying the calculated exposure at the vessel inner radius by the gradient data listed in Tables 6-3 and 6-4.

The calculated fast neutron exposures for the four surveillance capsules withdrawn from Wolf Creek reactor are provided in Table 6-5. These assigned neutron exposure levels are based on the plant and fuel cycle specific neutron transport calculations performed for the Wolf Creek reactor.

Updated lead factors for the Wolf Creek surveillance capsules are provided in Table 6-6. The capsule lead factor is defined as the ratio of the calculated fluence (E > 1.0 MeV) at the geometric center of the surveillance capsule to the corresponding maximum calculated fluence at the pressure vessel clad/base metal interface. In Table 6-6, the lead factors for capsules that have been withdrawn from the reactor (U, Y, V, and X) were based on the calculated fluence values for the irradiation period corresponding to the time of withdrawal for the individual capsules. For the capsules remaining in the reactor (W and Z), the lead factors correspond to the calculated fluence values at the end of cycle 12 operations.

6.3 NEUTRON DOSIMETRY

The validity of the calculated neutron exposures previously reported in Section 6.2 is demonstrated by a direct comparison against the measured sensor reaction rates and via a least squares evaluation performed for each of the capsule dosimetry sets However, since the neutron dosimetry measurement data merely serves to validate the calculated results, only the direct comparison of measured-to-calculated results for the most recent surveillance capsule removed from service is provided in this section of the report. For completeness, the assessment of all measured dosimetry removed to date, based on direct, best estimate, and least squares evaluation comparisons, is documented in Appendix A.

The direct comparison of measured versus calculated fast neutron threshold reaction rates for the sensors from Capsule X that was withdrawn from Wolf Creek at the end of the twelfth fuel cycle, is summarized below.

Reaction	Reaction Rates (rps/atom)		M/C Ratio
	Measured	Calculated	
$^{63}Cu(n,\alpha)^{60}Co$	4.65E-17	4.34E-17	1.07
$^{54}Fe(n,p)^{54}Mn$	4.72E-15	4.79E-15	0.99
$^{58}Ni(n,p)^{58}Co$	6.49E-15	6.71E-15	0.97
$^{238}U(n,p)^{137}Cs$ (Cd)	3.01E-14	2.56E-14	1.18
$^{237}Np(n,f)^{137}Cs$ (Cd)	2.56E-13	2.50E-13	1.02
		Average:	1.05
		% Standard Deviation:	8.0

The measured-to-calculated (M/C) reaction rate ratios for the Capsule X threshold reactions range from 0.97 to 1.18, and the average M/C ratio is 1.05 ± 8.0% (1σ). This direct comparison falls well within the ± 20% criterion specified in Regulatory Guide 1.190; furthermore, it is consistent with the full set of comparisons given in Appendix A for all measured dosimetry removed to date from the Wolf Creek reactor. As a result, these comparisons validate the current analytical results described in Section 6.2 which are deemed applicable for Wolf Creek.

6.4 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the Wolf Creek surveillance capsule and reactor pressure vessel is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology was carried out in the following four stages:

1 - Comparison of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL).

2 - Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment.

3 - An analytical sensitivity study addressing the uncertainty components resulting important input parameters applicable to the plant specific transport calculations used in the neutron exposure assessments.

4 - Comparisons of the plant specific calculations with all available dosimetry results from the Wolf Creek surveillance program.

The first phase of the methods qualification (PCA comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This phase, however, did not test the accuracy of commercial core neutron source calculations nor did it address uncertainties in operational or geometric variables that impact power reactor calculations. The second phase of the qualification (H. B. Robinson comparisons) addressed uncertainties in these additional areas that are primarily methods related and would tend to apply generically to all fast neutron exposure evaluations. The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations as well as to a lack of knowledge relative to various plant specific input parameters. The overall calculational uncertainty applicable to the Wolf Creek analysis was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with Wolf Creek measurements) was used solely to demonstrate the validity of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used in any way to modify the calculated surveillance capsule and pressure vessel neutron exposures previously described in Section 6.2. As such, the validation of the Wolf Creek analytical model based on the measured plant dosimetry is completely described in Appendix A.

The following summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in Reference 22.

	Capsule	Vessel IR
PCA Comparisons	3%	3%
H. B. Robinson Comparisons	3%	3%
Analytical Sensitivity Studies	10%	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%	5%
Net Calculational Uncertainty	12%	13%

The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was random and no systematic bias was applied to the analytical results.

The plant specific measurement comparisons described in Appendix A support these uncertainty assessments for Wolf Creek.

Figure 6-1a

Wolf Creek r,θ Reactor Geometry at the Core Midplane

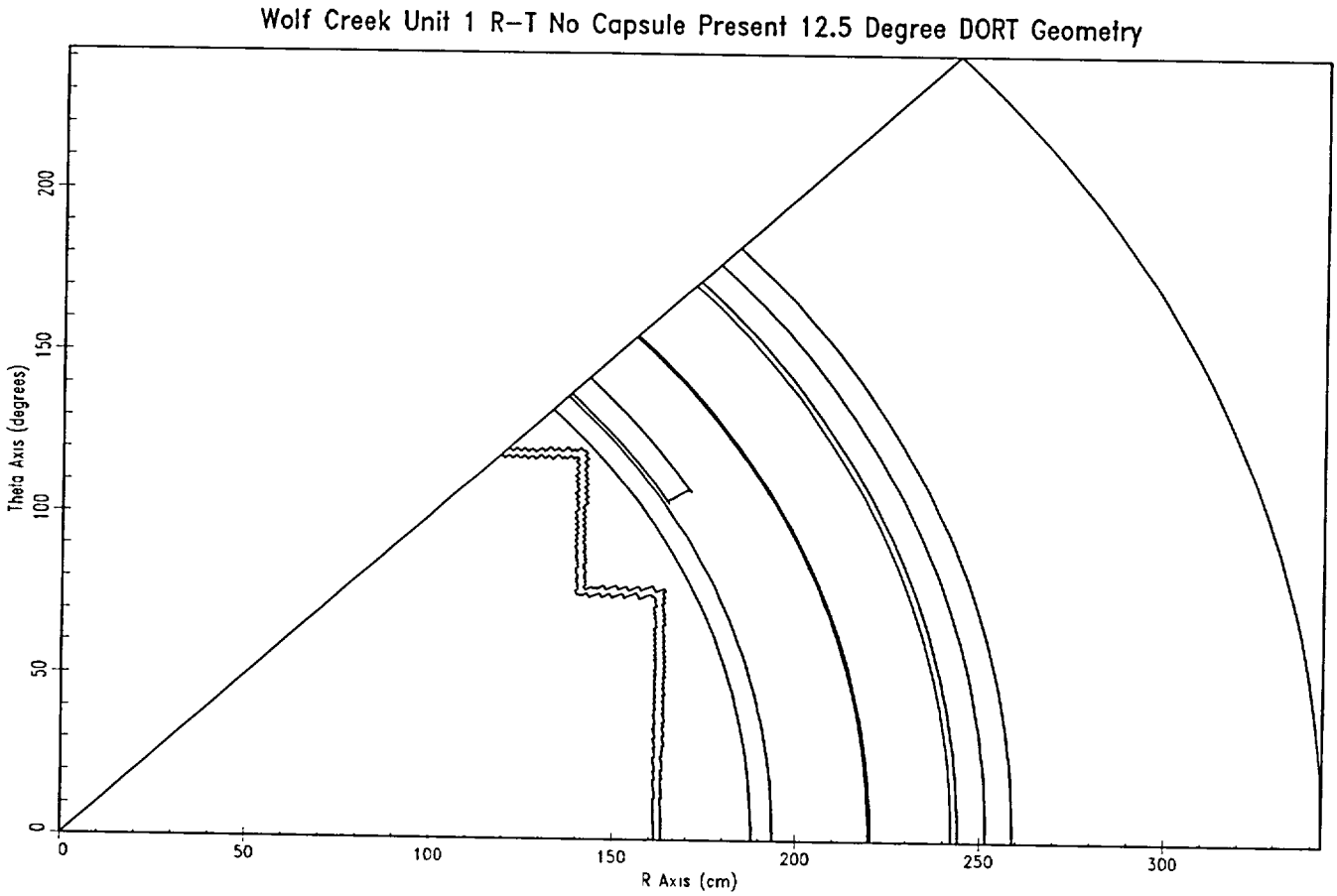


Figure 6-1b

Wolf Creek r,θ Reactor Geometry at the Core Midplane

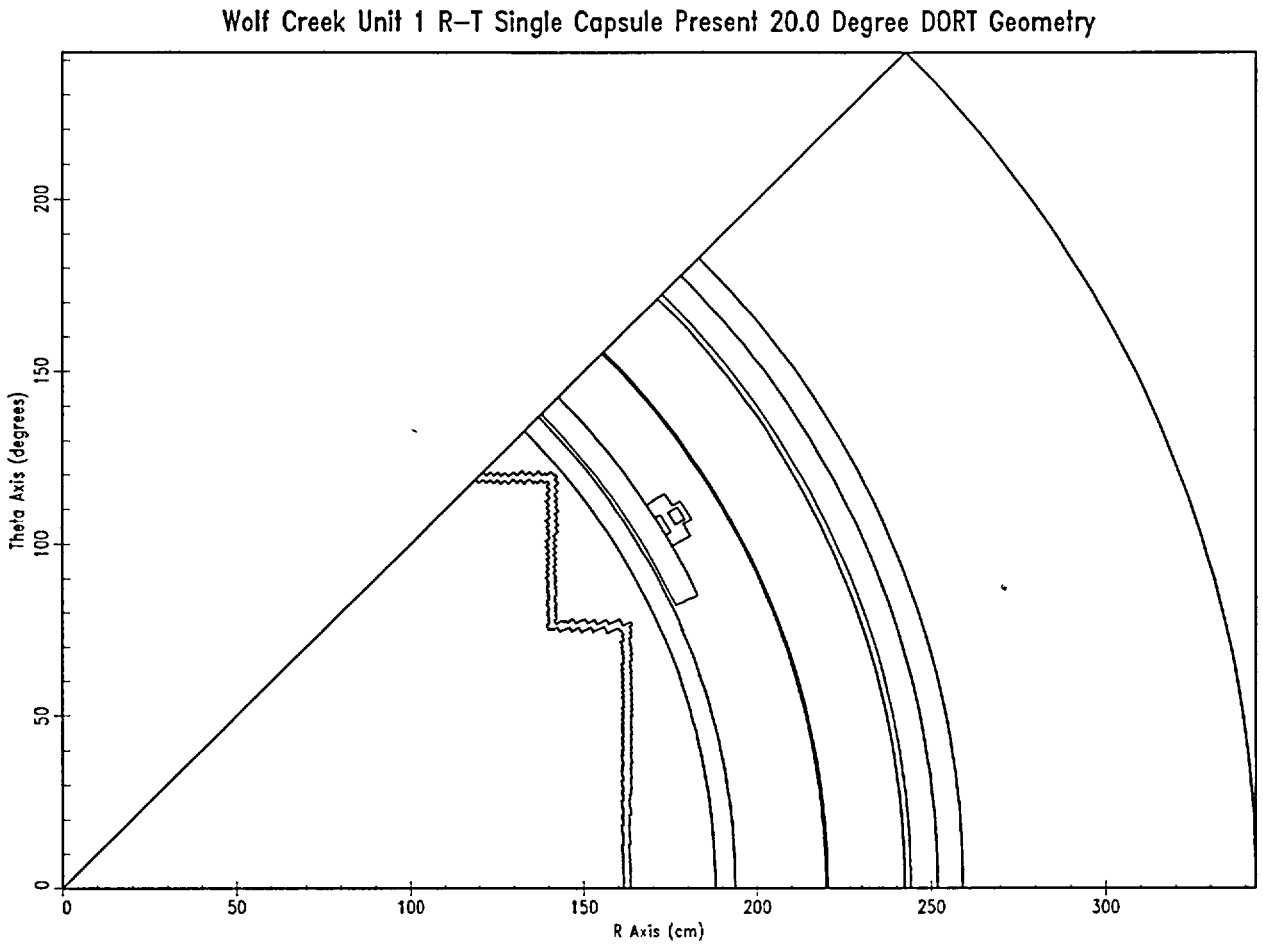


Figure 6-1c

Wolf Creek r,θ Reactor Geometry at the Core Midplane

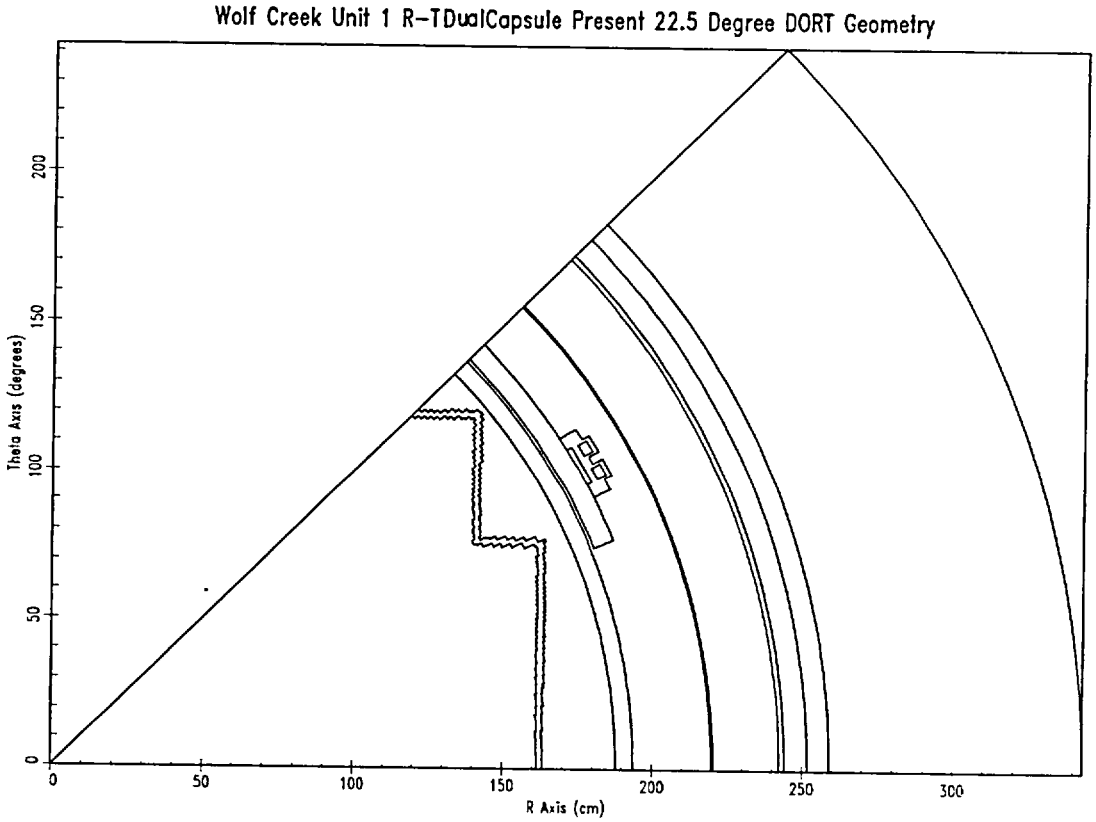


Figure 6-2

Wolf Creek r,z Reactor Geometry

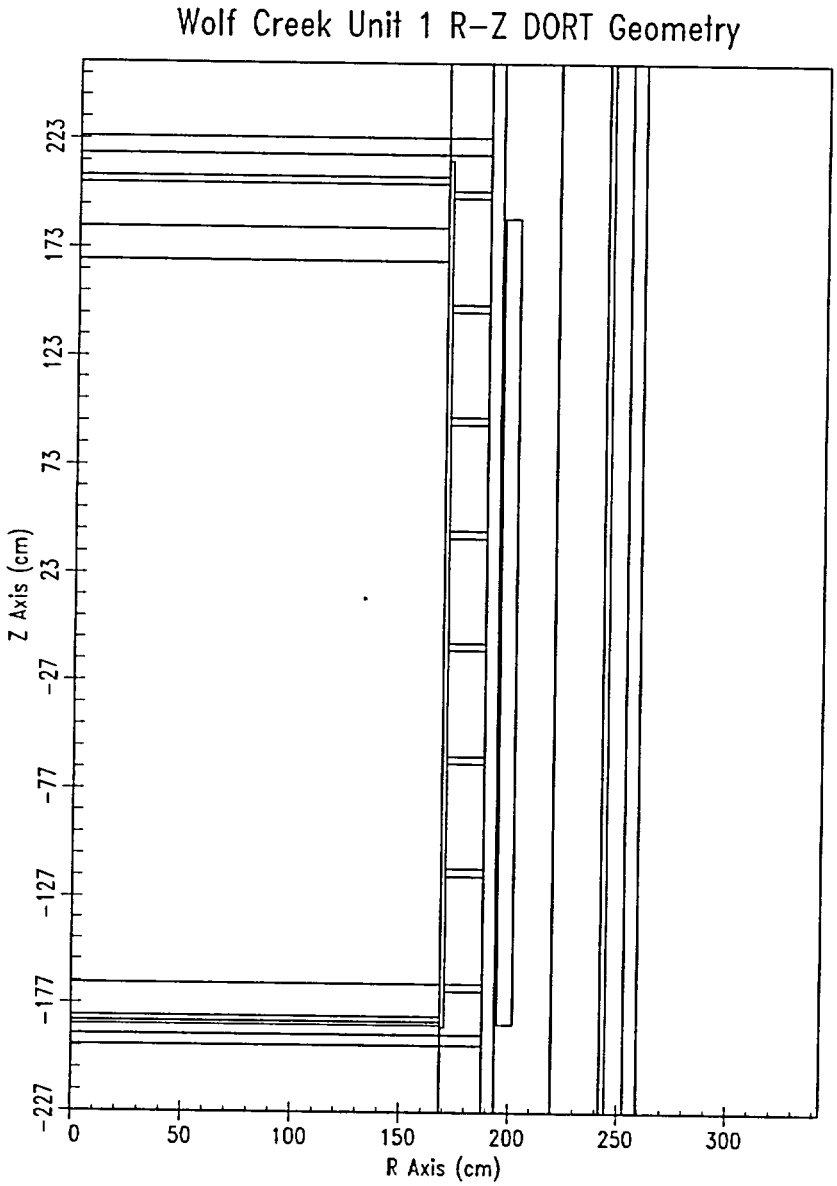


Table 6-1

Calculated Neutron Exposure Rates and Integrated Exposures
At The Surveillance Capsule Center

Neutrons (E > 1.0 MeV)

Cycle	Cumulative Operating Time [EFPY]	Flux [E>1.0 MeV] [n/cm^2-sec] 29 Deg	31.5 Deg
1	1.07	8.73E+10	9.33E+10
2	1.75	9.08E+10	1.00E+11
3	2.43	7.67E+10	8.33E+10
4	3.57	7.30E+10	8.04E+10
5	4.79	7.12E+10	7.60E+10
6	5.82	6.66E+10	7.05E+10
7	7.12	6.44E+10	6.98E+10
8	8.33	7.44E+10	7.91E+10
9	9.78	6.22E+10	7.10E+10
10	11.10	8.19E+10	8.64E+10
11	12.47	7.23E+10	8.25E+10
12	13.83	7.29E+10	8.29E+10
Projection	15.53	6.97E+10	7.79E+10
Projection	20.00	7.29E+10	8.29E+10
Projection	24.00	7.29E+10	8 29E+10
Projection	32.00	7.29E+10	8.29E+10
Projection	40.00	7.29E+10	8.29E+10
Projection	48.00	7.29E+10	8.29E+10
Projection	54.00	7.29E+10	8.29E+10

Note Neutron exposure values reported for the surveillance capsules are centered at the core midplane.

Table 6-1 cont'd

Calculated Neutron Exposure Rates and Integrated Exposures
At The Surveillance Capsule Center

Neutrons (E > 1.0 MeV)

Cycle	Cumulative Operating Time [EFPY]	Fluence [E>1.0 MeV] [n/cm^2-sec] 29 Deg	31.5 Deg
1	1.07	2.96E+18	3.16E+18
2	1.75	4.91E+18	5.32E+18
3	2.43	6.54E+18	7.09E+18
4	3.57	9.18E+18	1.00E+19
5	4.79	1.19E+19	1.29E+19
6	5.82	1.41E+19	1.52E+19
7	7.12	1.67E+19	1.81E+19
8	8.33	1.96E+19	2.11E+19
9	9.78	2.22E+19	2.42E+19
10	11.10	2.57E+19	2.78E+19
11	12.47	2.88E+19	3.13E+19
12	13.83	3.19E+19	3.49E+19
Projection	15.53	3.58E+19	3.93E+19
Projection	20.00	4.61E+19	5.10E+19
Projection	24.00	5.53E+19	6.14E+19
Projection	32.00	7.37E+19	8.24E+19
Projection	40.00	9.21E+19	1.03E+20
Projection	48.00	1.11E+20	1.24E+20
Projection	54.00	1.24E+20	1.40E+20

Note: Neutron exposure values reported for the surveillance capsules are centered at the core midplane

Table 6-1 cont'd

Calculated Neutron Exposure Rates and Integrated Exposures
At The Surveillance Capsule Center

IRON ATOM DISPLACEMENT RATES

Cycle	Cumulative Operating Time [EFPY]	Displacement Rate [dpa/sec] 29 Deg	31.5 Deg
1	1.07	1.71E-10	1.82E-10
2	1.75	1.78E-10	1.97E-10
3	2.43	1.49E-10	1.62E-10
4	3.57	1.42E-10	1.56E-10
5	4.79	1.38E-10	1.48E-10
6	5.82	1.29E-10	1.37E-10
7	7.12	1.25E-10	1 35E-10
8	8.33	1.44E-10	1.53E-10
9	9.78	1.21E-10	1.38E-10
10	11.10	1.59E-10	1.68E-10
11	12.47	1.40E-10	1.60E-10
12	13.83	1.42E-10	1.61E-10
Projection	15.53	1.35E-10	1.51E-10
Projection	20.00	1.42E-10	1.61E-10
Projection	24.00	1.42E-10	1.61E-10
Projection	32.00	1.42E-10	1.61E-10
Projection	40.00	1.42E-10	1.61E-10
Projection	48.00	1.42E-10	1.61E-10
Projection	54.00	1.42E-10	1.61E-10

Note: Neutron exposure values reported for the surveillance capsules are centered at the core midplane

Table 6-1 cont'd

Calculated Neutron Exposure Rates and Integrated Exposures
At The Surveillance Capsule Center

IRON ATOM DISPLACEMENTS

Cycle	Cumulative Operating Time [EFPY]	Displacements [dpa] 29 Deg	31.5 Deg
1	1.07	5.79E-03	6.18E-03
2	1.75	9.61E-03	1.04E-02
3	2.43	1.28E-02	1.39E-02
4	3.57	1.79E-02	1.95E-02
5	4.79	2.32E-02	2.52E-02
6	5.82	2.74E-02	2.96E-02
7	7.12	3.25E-02	3.52E-02
8	8.33	3.80E-02	4.10E-02
9	9.78	4.32E-02	4.69E-02
10	11.10	4.99E-02	5.40E-02
11	12.47	5.59E-02	6.09E-02
12	13.83	6.20E-02	6.78E-02
Projection	15.53	6.96E-02	7.63E-02
Projection	20.00	8.96E-02	9.91E-02
Projection	24.00	1.08E-01	1.19E-01
Projection	32.00	1.43E-01	1.60E-01
Projection	40.00	1.79E-01	2.01E-01
Projection	48.00	2.15E-01	2.42E-01
Projection	54.00	2.42E-01	2.72E-01

Note: Neutron exposure values reported for the surveillance capsules are centered at the core midplane.

Table 6-2

Calculated Azimuthal Variation of Maximum Exposure Rates
and Integrated Exposures at the Reactor Vessel
Clad/Base Metal Interface

Cycle	Cumulative Operating Time [EFPY]	Maximum Pressure Vessel Flux [E>1.0 MeV] [n/cm^2-sec]			
		0 Deg	15 Deg	30 Deg	45 Deg
1	1.07	1.26E+10	1.87E+10	2.17E+10	2.19E+10
2	1.75	1.39E+10	1.99E+10	2.31E+10	2.68E+10
3	2.43	1.13E+10	1.63E+10	1.91E+10	1.85E+10
4	3.57	1.22E+10	1.67E+10	1.83E+10	1.90E+10
5	4.79	1.15E+10	1.67E+10	1.77E+10	1.71E+10
6	5.82	9.47E+09	1.59E+10	1.69E+10	1.62E+10
7	7.12	8.16E+09	1.35E+10	1.67E+10	1.63E+10
8	8.33	9.32E+09	1.65E+10	1.90E+10	1.65E+10
9	9.78	7.78E+09	1 09E+10	1.57E+10	1.72E+10
10	11.10	9.92E+09	1.56E+10	1.99E+10	1.90E+10
11	12.47	9.15E+09	1.31E+10	1.79E+10	1.98E+10
12	13.83	9.00E+09	1.36E+10	1.81E+10	2.14E+10
Projection	15.53	9.30E+09	1.36E+10	1.73E+10	1.94E+10
Projection	20.00	9.00E+09	1.36E+10	1.81E+10	2.14E+10
Projection	24.00	9.00E+09	1.36E+10	1.81E+10	2.14E+10
Projection	32.00	9.00E+09	1.36E+10	1.81E+10	2.14E+10
Projection	40.00	9.00E+09	1.36E+10	1.81E+10	2.14E+10
Projection	48.00	9.00E+09	1.36E+10	1.81E+10	2.14E+10
Projection	54.00	9.00E+09	1.36E+10	1.81E+10	2.14E+10

Table 6-2 cont'd

Calculated Azimuthal Variation of Maximum Exposure Rates
And Integrated Exposures at the Reactor Vessel
Clad/Base Metal Interface

Cycle	Cumulative Operating Time [EFPY]	Maximum Pressure Vessel Fluence [E>1.0 MeV] [n/cm^2]			
		0 Deg	15 Deg	30 Deg	45 Deg
1	1.07	4.26E+17	6.32E+17	7.36E+17	7.44E+17
2	1.75	7.11E+17	1.04E+18	1.21E+18	1.29E+18
3	2.43	9.50E+17	1.39E+18	1.62E+18	1.69E+18
4	3.57	1.39E+18	1.99E+18	2.27E+18	2.37E+18
5	4.79	1.84E+18	2.63E+18	2.96E+18	3.03E+18
6	5.82	2.14E+18	3.14E+18	3.50E+18	3.55E+18
7	7.12	2.45E+18	3.66E+18	4.14E+18	4.18E+18
8	8.33	2.80E+18	4.28E+18	4.85E+18	4.79E+18
9	9.78	3.14E+18	4.74E+18	5.53E+18	5.54E+18
10	11.10	3.55E+18	5.40E+18	6.36E+18	6.33E+18
11	12.47	3.95E+18	5 96E+18	7.13E+18	7.19E+18
12	13.83	4.33E+18	6.54E+18	7.91E+18	8.10E+18
Projection	15.53	4.86E+18	7.31E+18	8.88E+18	9.20E+18
Projection	20.00	6.12E+18	9.23E+18	1.14E+19	1.22E+19
Projection	24.00	7.26E+18	1.09E+19	1.37E+19	1.49E+19
Projection	32.00	9.53E+18	1.44E+19	1.83E+19	2.03E+19
Projection	40.00	1.18E+19	1.78E+19	2.28E+19	2.57E+19
Projection	48.00	1.41E+19	2.12E+19	2.74E+19	3.11E+19
Projection	54.00	1.58E+19	2.38E+19	3.08E+19	3.51E+19

Table 6-2 cont'd

Calculated Azimuthal Variation of Fast Neutron Exposure Rates
And Iron Atom Displacement Rates At the Reactor Vessel
Clad/Base Metal Interface

Cycle	Cumulative Operating Time [EFPY]	Maximum Iron Atom Displacements [dpa/sec] 0 Deg	15 Deg	30 Deg	45 Deg
1	1.07	1.95E-11	2.87E-11	3.35E-11	3.47E-11
2	1.75	2.16E-11	3.06E-11	3.56E-11	4.22E-11
3	2.43	1.75E-11	2.51E-11	2.95E-11	2.93E-11
4	3.57	1.90E-11	2.58E-11	2.82E-11	3.01E-11
5	4.79	1.79E-11	2.56E-11	2.73E-11	2.70E-11
6	5.82	1.48E-11	2.44E-11	2.61E-11	2.57E-11
7	7.12	1.27E-11	2.08E-11	2.58E-11	2.57E-11
8	8.33	1.46E-11	2.54E-11	2.93E-11	2.61E-11
9	9.78	1.21E-11	1.68E-11	2.43E-11	2.73E-11
10	11.10	1.55E-11	2.40E-11	3.06E-11	3.00E-11
11	12.47	1.43E-11	2.02E-11	2.77E-11	3.13E-11
12	13.83	1.40E-11	2.10E-11	2.79E-11	3.37E-11
Projection	15.53	1.45E-11	2.10E-11	2.67E-11	3.07E-11
Projection	20.00	1.40E-11	2.10E-11	2.79E-11	3.37E-11
Projection	24.00	1.40E-11	2.10E-11	2.79E-11	3.37E-11
Projection	32 00	1.40E-11	2.10E-11	2.79E-11	3.37E-11
Projection	40 00	1.40E-11	2.10E-11	2.79E-11	3.37E-11
Projection	48.00	1.40E-11	2.10E-11	2 79E-11	3.37E-11
Projection	54.00	1.40E-11	2.10E-11	2.79E-11	3 37E-11

Table 6-2 cont'd

Calculated Azimuthal Variation of Maximum Exposure Rates
And Integrated Exposures at the Reactor Vessel
Clad/Base Metal Interface

Cycle	Cumulative Operating Time [EFPY]	Maximum Iron Atom Displacements [dpa] 0 Deg	15 Deg	30 Deg	45 Deg
1	1.07	6.62E-04	9.74E-04	1.14E-03	1.18E-03
2	1.75	1.11E-03	1.60E-03	1.87E-03	2.04E-03
3	2.43	1.48E-03	2.14E-03	2.49E-03	2.67E-03
4	3.57	2.16E-03	3.07E-03	3.51E-03	3.75E-03
5	4.79	2.85E-03	4.05E-03	4.56E-03	4.79E-03
6	5.82	3.33E-03	4.84E-03	5.41E-03	5.62E-03
7	7.12	3.82E-03	5.64E-03	6.40E-03	6.61E-03
8	8.33	4.36E-03	6.59E-03	7.50E-03	7.59E-03
9	9.78	4.89E-03	7.31E-03	8.54E-03	8.76E-03
10	11.10	5.53E-03	8.32E-03	9.82E-03	1.00E-02
11	12.47	6.15E-03	9.19E-03	1.10E-02	1.14E-02
12	13.83	6.75E-03	1.01E-02	1.22E-02	1.28E-02
Projection	15.53	7.57E-03	1.13E-02	1.37E-02	1.45E-02
Projection	20.00	9.55E-03	1.42E-02	1.77E-02	1.93E-02
Projection	24.00	1.13E-02	1.69E-02	2.12E-02	2.36E-02
Projection	32.00	1.49E-02	2.22E-02	2.82E-02	3.21E-02
Projection	40.00	1.84E-02	2.75E-02	3.53E-02	4.06E-02
Projection	48.00	2.20E-02	3.28E-02	4.23E-02	4.91E-02
Projection	54.00	2.46E-02	3.68E-02	4.76E-02	5.55E-02

Table 6-3

Relative Radial Distribution Of Neutron Fluence (E > 1 0 MeV)
Within The Reactor Vessel Wall

RADIUS (cm)	AZIMUTHAL ANGLE			
	0°	15°	30°	45°
220.35	1.00	1.00	1.00	1.00
225.87	0.56	0.56	0 55	0.550
231.39	0.28	0.27	0.26	0.26
236.90	0.13	0.13	0.12	0 12
242.42	0.06	0.06	0 06	0.05

Note:
Base Metal Inner Radius = 220.35 cm
Base Metal 1/4T = 225.87 cm
Base Metal 1/2T = 231.39 cm
Base Metal 3/4T = 236.90 cm
Base Metal Outer Radius = 242.42 cm

Table 6-4

Relative Radial Distribution of Iron Atom Displacements (dpa)
Within The Reactor Vessel Wall

RADIUS (cm)	AZIMUTHAL ANGLE			
	0°	15°	30°	45°
220.35	1.00	1.00	1.00	1.00
225.87	0.64	0 63	0 63	0.64
231.39	0.39	0.38	0.37	0.39
236.90	0.23	0.22	0.22	0.23
242.42	0.14	0.13	0.12	0.13

Note:
Base Metal Inner Radius = 220.35 cm
Base Metal 1/4T = 225.87 cm
Base Metal 1/2T = 231.39 cm
Base Metal 3/4T = 236.90 cm
Base Metal Outer Radius = 242.42 cm

Table 6-5

Calculated Fast Neutron Exposure of Surveillance Capsules
Withdrawn from Wolf Creek

Capsule	Irradiation Time [EFPY]	Fluence (E>1.0 MeV) [n/cm^2]	Iron Displacements [dpa]
U	1.07	3.16E+18	6.18E-03
Y	4.79	1.19E+19	2.32E-02
V	9.78	2.22E+19	4.32E-02
X	13.83	3.49E+19	6 78E-02

Table 6-6

Calculated Surveillance Capsule Lead Factors

Capsule ID And Location	Status	Lead Factor
U (31.5°)	Withdrawn EOC 1 (for analysis)	4.25
Y (29°)	Withdrawn EOC 5 (for analysis)	3.93
V (29°)	Withdrawn EOC 9 (for analysis)	4.02
X (31.5°)	Withdrawn EOC 12 (for analysis)	4.30
W (31.5°)	In Reactor	4.11
Z (31.5°)	In Reactor	4.11

Notes:
(1) Capsules U, Y, V, and X were contained in dual capsule holders, while Capsules W and Z are being irradiated in single capsule holders.
(2) Lead factors for capsules remaining in the reactor are based on exposure calculations through Cycle 12 operations for the single capsule holders.

7 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule meets the requirements of ASTM E185-82 and is recommended for future capsules to be removed from the Wolf Creek reactor vessel. This recommended removal schedule is applicable to 32 EFPY of operation.

Table 7-1 Recommended Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Lead Factor [a]	Withdrawal EFPY [b]	Fluence (n/cm^2) [a]
U	58.5°	4.25	1.07	3.16×10^{18} (c)
Y	241°	3.93	4.79	1.19×10^{19} (c)
V	60.1°	4.02	9.78	2.22×10^{19} (c)
X	238.5°	4.30	13.83	3.49×10^{19} (c)
W	121.5°	4.11	Standby	(d)
Z	301.5°	4.11	Standby	(d)

Notes:

(a) Updated in Capsule X dosimetry analysis

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) The standby capsules have already reached a peak vessel surface fluence, equivalent to 54 EFPY. They will reach two times this fluence at 26.8 EFPY. Thus, it is recommended that the standby capsules be removed and placed in storage, as recommended in NUREG-1801, to preserve meaningful metallurgical data

8 REFERENCES

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APPENDIX A

VALIDATION OF THE RADIATION TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

A.1 Neutron Dosimetry

Comparisons of measured dosimetry results to both the calculated and least squares adjusted values for all surveillance capsules withdrawn from service to date at Wolf Creek are described herein. The sensor sets from these capsules have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (Reference A-1). One of the main purposes for presenting this material is to demonstrate that the overall measurements agree with the calculated and least squares adjusted values to within ± 20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in Section 6.2 of this report. This information may also be useful in the future, in particular, as least squares adjustment techniques become accepted in the regulatory environment.

A.1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of the four neutron sensor sets withdrawn to date as a part of the Wolf Creek Reactor Vessel Materials Surveillance Program are presented. The capsule designation, location within the reactor, and time of withdrawal of each of these dosimetry sets were as follows:

Capsule ID	Azimuthal Location	Withdrawal Time	Irradiation Time [EFPY]
U	31.5°	End of Cycle 1	1.07
Y	29°	End of Cycle 5	4.79
V	29°	End of Cycle 9	9.78
X	31.5°	End of Cycle 12	13.83

The azimuthal locations included in the above tabulation represent the first octant equivalent azimuthal angle of the geometric center of the respective surveillance capsules.

The passive neutron sensors included in the evaluations of Surveillance Capsules U, Y, V, and X are summarized as follows:

Sensor Material	Reaction Of Interest	Capsule U	Capsule Y	Capsule V	Capsule X
Copper	$^{63}Cu(n,\alpha)^{60}Co$	X	X	X	X
Iron	$^{54}Fe(n,p)^{54}Mn$	X	X	X	X
Nickel	$^{58}Ni(n,p)^{58}Co$	X	X	X	X
Uranium-238	$^{238}U(n,f)^{137}Cs$	X	X	X	X
Neptunium-237	$^{237}Np(n,f)^{137}Cs$	X	X	X	X
Cobalt-Aluminum*	$^{59}Co(n,\gamma)^{60}Co$	X	X	X	X

* The cobalt-aluminum measurements for this plant include both bare wire and cadmium-covered sensors

Since all the dosimetry monitors were accommodated within the dosimeter block centered at the radial, azimuthal, and axial center of the material test specimen array, gradient corrections were not required for these reaction rates. Pertinent physical and nuclear characteristics of the passive neutron sensors are listed in Table A-1.

The use of passive monitors such as those listed above does not yield a direct measure of the energy dependent neutron flux at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The measured specific activity of each monitor,
- the physical characteristics of each monitor,
- the operating history of the reactor,
- the energy response of each monitor, and
- the neutron energy spectrum at the monitor location.

The radiometric counting of the neutron sensors from Capsules U and Y was carried out at the Westinghouse Analytical Services Laboratory at the Waltz Mill Site (Reference A-2). The radiometric counting of the sensors from Capsule V was completed at the Pace Analytical Laboratory, also located at the Waltz Mill Site. Capsule X's radiometric sensor counting was completed by Pace Analytical Services, located at the Waltz Mill Site. In all cases, the radiometric counting followed established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor was determined by means of a high-resolution gamma spectrometer. For the copper, iron, nickel, and cobalt-aluminum sensors, these analyses were performed by direct counting of each of the individual samples. In the case of the uranium and neptunium fission sensors, the analyses were carried out by direct counting preceded by dissolution and chemical separation of cesium from the sensor material.

The irradiation history of the reactor over the irradiation periods experienced by Capsules U, Y, V, and X was based on the reported monthly power generation of Wolf Creek from initial reactor startup through the end of the dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations. The irradiation history applicable to Capsules U, Y, V, and X is given in Table A-2.

Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full-power operation were determined from the following equation:

$$R = \frac{A}{N_0 \, F \, Y \sum \frac{P_j}{P_{ref}} \, C_j \, [1 - e^{-\lambda t_j}] \, [e^{-\lambda t_d}]}$$

where:

R	=	Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).
A	=	Measured specific activity (dps/gm).
N_0	=	Number of target element atoms per gram of sensor.
F	=	Weight fraction of the target isotope in the sensor material.
Y	=	Number of product atoms produced per reaction.
P_j	=	Average core power level during irradiation period j (MW).
P_{ref}	=	Maximum or reference power level of the reactor (MW).
C_j	=	Calculated ratio of $\phi(E > 1.0 \text{ MeV})$ during irradiation period j to the time weighted average $\phi(E > 1.0 \text{ MeV})$ over the entire irradiation period.
λ	=	Decay constant of the product isotope (1/sec).
t_j	=	Length of irradiation period j (sec).
t_d	=	Decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the equation describing the reaction rate calculation, the ratio $[P_j]/[P_{ref}]$ accounts for month-by-month variation of reactor core power level within any given fuel cycle as well as over multiple fuel cycles. The ratio C_j, which was calculated for each fuel cycle using the transport methodology discussed in Section 6.2, accounts for the change in sensor reaction rates caused by variations in flux level induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single-cycle irradiation, C_j is normally taken to be 1.0. However, for multiple-cycle irradiations, particularly those employing low leakage fuel management, the additional C_j term should be employed. The impact of changing flux levels for constant power operation can be quite significant for sensor sets that have been irradiated for many cycles in a reactor that has transitioned from non-low leakage to low leakage fuel management or for sensor sets contained in surveillance capsules that have been moved from one capsule location to another. The fuel cycle specific neutron flux values along with the computed values for C_j are listed in Table A-3. These flux values represent the cycle dependent results at the radial and azimuthal center of the respective capsules at the axial elevation of the active fuel midplane.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets, additional corrections were made to the ^{238}U measurements to account for the presence of ^{235}U impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. Corrections were also made to the ^{238}U and ^{237}Np sensor reaction rates to account for gamma ray induced fission reactions that occurred over the course of the capsule irradiations. The correction factors applied to the Wolf Creek fission sensor reaction rates are summarized as follows:

Correction	Capsule U	Capsule Y	Capsule V	Capsule X
^{235}U Impurity/Pu Build-in	0.87	0.84	0.80	0.76
$^{238}U(\gamma,f)$	0.97	0.97	0.97	0.97
Net ^{238}U Correction	0.84	0.81	0.78	0.73
$^{237}Np(\gamma,f)$	0.99	0.99	0.99	0.99

These factors were applied in a multiplicative fashion to the decay corrected uranium and neptunium fission sensor reaction rates.

Results of the sensor reaction rate determinations for Capsules U, Y, V, and X are given in Table A-4. In Table A-4, the measured specific activities, decay corrected saturated specific activities, and computed reaction rates for each sensor indexed to the radial center of the capsule are listed. The fission sensor reaction rates are listed both with the applied corrections for ^{238}U impurities, plutonium build-in, and gamma ray induced fission effects.

A.1.2 Least Squares Evaluation of Sensor Sets

Least squares adjustment methods provide the capability of combining the measurement data with the corresponding neutron transport calculations resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as $\phi(E > 1.0\ MeV)$ or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least squares methods, as applied to surveillance capsule dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross-section, σ_{ig}, each with an uncertainty δ. The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least squares evaluation of the Wolf Creek surveillance capsule dosimetry, the FERRET code (Reference A-3) was employed to combine the results of the plant specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters ($\phi(E > 1.0\ MeV)$ and dpa) along with associated uncertainties for the four in-vessel capsules withdrawn to date.

The application of the least squares methodology requires the following input:

1 - The calculated neutron energy spectrum and associated uncertainties at the measurement location.
2 - The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.
3 - The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the Wolf Creek application, the calculated neutron spectrum was obtained from the results of plant specific neutron transport calculations described in Section 6.2 of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section A.1.1. The dosimetry reaction cross-sections and uncertainties were obtained from the SNLRML dosimetry cross-section library (Reference A-4). The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)".

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum were input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties followed the guidance provided in ASTM Standard E 944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance."

The following provides a summary of the uncertainties associated with the least squares evaluation of the Wolf Creek surveillance capsule sensor sets.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, irradiation history corrections, and corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least squares evaluation:

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	5%
$^{54}Fe(n,p)^{54}Mn$	5%
$^{58}Ni(n,p)^{58}Co$	5%
$^{238}U(n,f)^{137}Cs$	10%
$^{237}Np(n,f)^{137}Cs$	10%
$^{59}Co(n,\gamma)^{60}Co$	5%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

The reaction rate cross-sections used in the least squares evaluations were taken from the SNLRML library. This data library provides reaction cross-sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross-sections and uncertainties are provided in a fine multigroup structure for use in least squares adjustment applications. These cross-sections were compiled from the most recent cross-section evaluations and they have been tested with respect to their accuracy and consistency for least squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources.

For sensors included in the Wolf Creek surveillance program, the following uncertainties in the fission spectrum averaged cross-sections are provided in the SNLRML documentation package.

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	4.08-4.16%
$^{54}Fe(n,p)^{54}Mn$	3.05-3.11%
$^{58}Ni(n,p)^{58}Co$	4.49-4.56%
$^{238}U(n,f)^{137}Cs$	0.54-0.64%
$^{237}Np(n,f)^{137}Cs$	10.32-10.97%
$^{59}Co(n,\gamma)^{60}Co$	0.79-3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum

The neutron spectra input to the least squares adjustment procedure were obtained directly from the results of plant specific transport calculations for each surveillance capsule irradiation period and location. The spectrum for each capsule was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties R_g and $R_{g'}$ specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\delta_{gg'} + \theta e^{-H}$$

where

$$H = \frac{(g-g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term) The value of δ is 1 0 when g = g', and is 0.0 otherwise

The set of parameters defining the input covariance matrix for the Wolf Creek calculated spectra was as follows

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	29%
(E < 0 68 eV)	52%
Short Range Correlation (θ)	
(E > 0 0055 MeV)	0 9
(0.68 eV < E < 0.0055 MeV)	0 5
(E < 0.68 eV)	0 5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0 68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

A.1.3 Comparisons of Measurements and Calculations

Results of the least squares evaluations of the dosimetry from the Wolf Creek surveillance capsules withdrawn to date are provided in Tables A-5 and A-6. In Table A-5, measured, calculated, and best-estimate values for sensor reaction rates are given for each capsule. Also provided in this tabulation are ratios of the measured reaction rates to both the calculated and least squares adjusted reaction rates. These ratios of M/C and M/BE illustrate the consistency of the fit of the calculated neutron energy spectra to the measured reaction rates both before and after adjustment. In Table A-6, comparison of the calculated and best estimate values of neutron flux (E > 1.0 MeV) and iron atom displacement rate are tabulated along with the BE/C ratios observed for each of the capsules.

The data comparisons provided in Tables A-5 and A-6 show that the adjustments to the calculated spectra are relatively small and well within the assigned uncertainties for the calculated spectra, measured sensor reaction rates, and dosimetry reaction cross-sections. Further, these results indicate that the use of the least squares evaluation results in a reduction in the uncertainties associated with the exposure of the surveillance capsules. From Section 6.4 of this report, it may be noted that the uncertainty associated with the unadjusted calculation of neutron fluence (E > 1.0 MeV) and iron atom displacements at the surveillance capsule locations is specified as 12% at the 1σ level. From Table A-6, it is noted that the corresponding uncertainties associated with the least squares adjusted exposure parameters have been reduced to 6% for neutron flux (E > 1.0 MeV) and 8% for iron atom displacement rate. Again, the uncertainties from the least squares evaluation are at the 1σ level.

Further comparisons of the measurement results with calculations are given in Tables A-7 and A-8. These comparisons are given on two levels. In Table A-7, calculations of individual threshold sensor reaction rates are compared directly with the corresponding measurements. These threshold reaction rate comparisons provide a good evaluation of the accuracy of the fast neutron portion of the calculated energy spectra. In Table A-8, calculations of fast neutron exposure rates in terms of ϕ(E > 1.0 MeV) and dpa/s are compared with the best estimate results obtained from the least squares evaluation of the capsule dosimetry results. These two levels of comparison yield consistent and similar results with all measurement-to-calculation comparisons falling well within the 20% limits specified as the acceptance criteria in Regulatory Guide 1.190.

In the case of the direct comparison of measured and calculated sensor reaction rates, the M/C comparisons for fast neutron reactions range from 0.93-1.30 for the 20 samples included in the data set. The overall average M/C ratio for the entire set of Wolf Creek data is 1.08 with an associated standard deviation of 8.2%.

Based on these comparisons, it is concluded that the calculated fast neutron exposures provided in Section 6.2 of this report are validated for use in the assessment of the condition of the materials comprising the beltline region of the Wolf Creek reactor pressure vessel.

Table A-9 has been included to address current and projected (through 54 EFPY) neutron fluences (E >1.0 MeV) experienced by each of the circumferential and vertical welds modeled for this project.

Table A-1
Nuclear Parameters Used In the Evaluation of Neutron Sensors

Monitor Material	Reaction of Interest	Target Atom Fraction	90% Response Range(MeV)	Product Half-life	Fission Yield (%)
Copper	^{63}Cu (n,α)	0.6917	4.9 – 11.8	5.271 y	
Iron	^{54}Fe (n,p)	0.0585	2.1 – 8.3	312.3 d	
Nickel	^{58}Ni (n,p)	0.6808	1.5 – 8.1	70.82 d	
Uranium-238	^{238}U (n,f)	1.0000	1.2 – 6.7	30.07 y	6.02
Neptunium-237	^{237}Np (n,f)	1.0000	0.4 – 3.5	30.07 y	6.17
Cobalt-Aluminum	^{59}Co (n,γ)	0.0015	non-threshold	5.271 y	

Notes: The 90% response range is defined such that, in the neutron spectrum characteristic of the Wolf Creek surveillance capsules, approximately 90% of the sensor response is due to neutrons in the energy range specified with approximately 5% of the total response due to neutrons with energies below the lower limit and 5% of the total response due to neutrons with energies above the upper limit.

Table A-2

Monthly Thermal Generation During the First Twelve Fuel Cycles
Of The Wolf Creek Reactor
(Reactor Power of 3411 MWt June 12, 1985 - May 17, 1993;
3450 MWt May 17, 1993 –November 2, 1994; and 3565 MWt thereafter)

Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)
1985	6	356676	1988	8	2533606	1991	10	0
1985	7	1025780	1988	9	2450165	1991	11	0
1985	8	1643803	1988	10	492163	1991	12	0
1985	9	2053023	1988	11	0	1992	1	1268945
1985	10	2086772	1988	12	0	1992	2	1524407
1985	11	2366472	1989	1	2095086	1992	3	321390
1985	12	2368666	1989	2	2113705	1992	4	2446580
1986	1	2480479	1989	3	2535552	1992	5	2534299
1986	2	2005668	1989	4	2454150	1992	6	2453249
1986	3	2513225	1989	5	2498149	1992	7	2535304
1986	4	933250	1989	6	2448863	1992	8	2531360
1986	5	2341310	1989	7	2493515	1992	9	2453478
1986	6	1670026	1989	8	2534633	1992	10	2534881
1986	7	2210358	1989	9	2453774	1992	11	2296524
1986	8	2439547	1989	10	2516573	1992	12	2535128
1986	9	2406802	1989	11	2450503	1993	1	2534643
1986	10	1219774	1989	12	2536033	1993	2	2288546
1986	11	0	1990	1	2534772	1993	3	282997
1986	12	650000	1990	2	2017613	1993	4	0
1987	1	1533313	1990	3	599723	1993	5	1124412
1987	2	2192444	1990	4	0	1993	6	2453687
1987	3	2471746	1990	5	1003923	1993	7	2535510
1987	4	2247475	1990	6	2442569	1993	8	2535563
1987	5	2436662	1990	7	2515109	1993	9	2453641
1987	6	2250313	1990	8	2534494	1993	10	2532824
1987	7	2066874	1990	9	2453417	1993	11	2435990
1987	8	2527262	1990	10	2533710	1993	12	2557464
1987	9	1954923	1990	11	2421081	1994	1	2051879
1987	10	0	1990	12	2531359	1994	2	2312015
1987	11	0	1991	1	2363291	1994	3	2561261
1987	12	0	1991	2	1840498	1994	4	2531248
1988	1	1216547	1991	3	1969185	1994	5	2597022
1988	2	956585	1991	4	1506284	1994	6	2520895
1988	3	2526972	1991	5	1692964	1994	7	2573456
1988	4	2452604	1991	6	2434282	1994	8	2577876
1988	5	2533966	1991	7	2534580	1994	9	1098290
1988	6	2451743	1991	8	2466385	1994	10	0
1988	7	2531412	1991	9	1221097	1994	11	2306676

Table A-2 cont'd

Monthly Thermal Generation during the First Twelve Fuel Cycles
Of The Wolf Creek Reactor
(Reactor Power of 3411 MWt June 12, 1985 - May 17, 1993;
3450 MWt May 17, 1993 –November 2, 1994; and 3565 MWt thereafter)

Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)
1994	12	2636320	1998	2	2394162	2001	4	2562211
1995	1	2639803	1998	3	2649959	2001	5	2631325
1995	2	2383600	1998	4	2564698	2001	6	2565829
1995	3	2020996	1998	5	2649737	2001	7	2651544
1995	4	2543603	1998	6	2561103	2001	8	2651448
1995	5	2634001	1998	7	2642611	2001	9	2565781
1995	6	2524794	1998	8	2649749	2001	10	2654261
1995	7	2632369	1998	9	2564818	2001	11	2565708
1995	8	2634129	1998	10	2649145	2001	12	2651478
1995	9	2468895	1998	11	2561237	2002	1	2650720
1995	10	2637097	1998	12	2643012	2002	2	2388288
1995	11	2549850	1999	1	2616340	2002	3	1836759
1995	12	2640215	1999.	2	2368213			
1996	1	2481700	1999	3	2642613			
1996	2	0	1999	4	104209			
1996	3	0	1999	5	1740059			
1996	4	1784195	1999	6	2564943			
1996	5	2622027	1999	7	2649843			
1996	6	2349911	1999	8	2514800			
1996	7	2642811	1999	9	2563711			
1996	8	2603985	1999	10	2652619			
1996	9	2564802	1999	11	2543480			
1996	10	2621302	1999	12	2624035			
1996	11	2564301	2000	1	2642844			
1996	12	2649864	2000	2	2468736			
1997	1	2648915	2000	3	2650259			
1997	2	2393586	2000	4	2560819			
1997	3	2650251	2000	5	2638828			
1997	4	2564428	2000	6	2564614			
1997	5	2218326	2000	7	2645033			
1997	6	2563505	2000	8	2650187			
1997	7	2648185	2000	9	2064968			
1997	8	2649138	2000	10	0			
1997	9	2563192	2000	11	1898243			
1997	10	18863	2000	12	2650536			
1997	11	0	2001	1	2649695			
1997	12	2490372	2001	2	2393593			
1998	1	2649875	2001	3	2516638			

Table A-3

Calculated C_j Factors at the Surveillance Capsule Center
Core Midplane Elevation

Fuel Cycle	ϕ(E > 1.0 MeV) [n/cm^2-s]				C_j			
	Capsule U	Capsule Y	Capsule V	Capsule X	Capsule U	Capsule Y	Capsule V	Capsule X
1	9.33E+10	8.73E+10	8.73E+10	9.33E+10	1.00	1.11	1.20	1.16
2		9.08E+10	9.08E+10	1.00E+11		1.15	1.25	1.25
3		7.67E+10	7.67E+10	8.33E+10		0.97	1.06	1.04
4		7.30E+10	7.30E+10	8.04E+10		0.93	1.01	1.00
5		7.12E+10	7.12E+10	7.60E+10		0.90	0.98	0.95
6			6.66E+10	7.05E+10			0.92	0.88
7			6.43E+10	6.98E+10			0.89	0.87
8			7.44E+10	7.91E+10			1.02	0.98
9			6.22E+10	7.10E+10			0.86	0.88
10				8.64E+10				1.08
11				8.25E+10				1.03
12				8.29E+10				1.03
Average	9.33E+10	7.88E+10	7.27E+10	8.04E+10	1.00	1.00	1.00	1.00

Table A-4
Measured Sensor Activities and Reaction Rates

Surveillance Capsule U

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Radially Adjusted Saturated Activity (dps/g)	Radially Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	4.44E+04	3.54E+05	3.54E+05	5 41E-17
	Center	4.40E+04	3 51E+05	3.51E+05	5.36E-17
	Bottom	4.75E+04	3.79E+05	3.79E+05	5.78E-17
	Average				**5.52E-17**
^{54}Fe (n,p) ^{54}Mn	Top	1.51E+06	3.50E+06	3.50E+06	5.55E-15
	Center	1.50E+06	3.48E+06	3.48E+06	5.52E-15
	Bottom	1.80E+06	4.18E+06	4.18E+06	6.62E-15
	Average				**5.90E-15**
^{58}Ni (n,p) ^{58}Co	Top	1.64E+07	5.43E+07	5.43E+07	7.77E-15
	Center	1.61E+07	5.33E+07	5.33E+07	7.62E-15
	Bottom	1.76E+07	5.82E+07	5.82E+07	8.33E-15
	Average				**7.91E-15**
^{238}U (n,f) ^{137}Cs (Cd)	Middle	1.43E+05	5.90E+06	5.90E+06	3.87E-14
^{238}U (n,f) ^{137}Cs (Cd)		Including ^{235}U, ^{239}Pu, and γ,fission corrections:			**3.26E-14**
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	1.24E+06	5.12E+07	5.12E+07	3 26E-13
^{237}Np (n,f) ^{137}Cs (Cd)			Including γ,fission correction.		**3.23E-13**
^{59}Co (n,γ) ^{60}Co	Top	1.04E+07	8.30E+07	8 30E+07	5.42E-12
	Middle	1.00E+07	7.98E+07	7.98E+07	5 21E-12
	Bottom	1.01E+07	8.06E+07	8.06E+07	5.26E-12
	Average				**5.30E-12**
^{59}Co (n,γ) ^{60}Co (Cd)	Top	5.27E+06	4.21E+07	4.21E+07	2.75E-12
	Middle	5.14E+06	4.10E+07	4.10E+07	2.68E-12
	Bottom	4 89E+06	3 90E+07	3.90E+07	2.55E-12
	Average				**2.66E-12**

Notes: 1) Measured specific activities are indexed to a counting date of February 2, 1987.
2) The average ^{238}U (n,f) reaction rate of 3.26E-14 includes a correction factor of 0.87 to account for plutonium build-in and an additional factor of 0.97 to account for photo-fission effects in the sensor
3) The average ^{237}Np (n,f) reaction rate of 3.23E-13 includes a correction factor of 0.99 to account for photo-fission effects in the sensor

Table A-4 cont'd

Measured Sensor Activities and Reaction Rates

Surveillance Capsule Y

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Radially Adjusted Saturated Activity (dps/g)	Radially Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.37E+05	3.38E+05	3.38E+05	5.16E-17
	Center	1.20E+05	2.96E+05	2.96E+05	4.52E-17
	Bottom	1.21E+05	2.99E+05	2.99E+05	4.56E-17
	Average				**4.75E-17**
^{54}Fe (n,p) ^{54}Mn	Top	1.66E+06	3.05E+06	3.05E+06	4.84E-15
	Center	1.49E+06	2.74E+06	2.74E+06	4.34E-15
	Bottom	1.48E+06	2.72E+06	2.72E+06	4.31E-15
	Average				**4.50E-15**
^{58}Ni (n,p) ^{58}Co	Top	8.04E+06	4.53E+07	4.53E+07	6.48E-15
	Center	7.38E+06	4.16E+07	4.16E+07	5.95E-15
	Bottom	7.33E+06	4.13E+07	4.13E+07	5.91E-15
	Average				**6.12E-15**
^{238}U (n,f) ^{137}Cs (Cd)	Middle	5.43E+05	5.33E+06	5.33E+06	3.50E-14
^{238}U (n,f) ^{137}Cs (Cd)		Including ^{235}U, ^{239}Pu, and γ,fission corrections:			**2.84E-14**
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	4.40E+06	4.32E+07	4.32E+07	2.76E-13
^{237}Np (n,f) ^{137}Cs (Cd)			Including γ,fission correction·		**2.73E-13**
^{59}Co (n,γ) ^{60}Co	Top	2.59E+07	6.39E+07	6.39E+07	4.17E-12
	Bottom	2.57E+07	6.34E+07	6.34E+07	4.14E-12
	Average				**4.16E-12**
^{59}Co (n,γ) ^{60}Co (Cd)	Top	1.30E+07	3.21E+07	3.21E+07	2.09E-12
	Middle	1.36E+07	3.36E+07	3.36E+07	2.19E-12
	Bottom	1.39E+07	3.43E+07	3.43E+07	2.24E-12
	Average				**2.17E-12**

Notes: 1) Measured specific activities are indexed to a counting date of February 19, 1992.
2) The average ^{238}U (n,f) reaction rate of 2.84E-14 includes a correction factor of 0.84 to account for plutonium build-in and an additional factor of 0.97 to account for photo-fission effects in the sensor.
3) The average ^{237}Np (n,f) reaction rate of 2.73E-13 includes a correction factor of 0.99 to account for photo-fission effects in the sensor.

Table A-4 cont'd

Measured Sensor Activities and Reaction Rates

Surveillance Capsule V

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Radially Adjusted Saturated Activity (dps/g)	Radially Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.64E+05	2.79E+05	2.79E+05	4.25E-17
	Center	1 61E+05	2.74E+05	2.74E+05	4.17E-17
	Bottom	1.85E+05	3.14E+05	3.14E+05	4.79E-17
	Average				**4.41E-17**
^{54}Fe (n,p) ^{54}Mn	Top	1.35E+06	2 67E+06	2.67E+06	4.24E-15
	Center	1 37E+06	2.71E+06	2.71E+06	4.30E-15
	Bottom	1.51E+06	2.99E+06	2.99E+06	4.74E-15
	Average				**4.43E-15**
^{58}Ni (n,p) ^{58}Co	Top	4.01E+06	4.38E+07	4.38E+07	6.27E-15
	Center	4.00E+06	4.37E+07	4.37E+07	6.25E-15
	Bottom	4.37E+06	4.77E+07	4.77E+07	6 83E-15
	Average				**6.45E-15**
^{238}U (n,f) ^{137}Cs (Cd)	Middle	1.14E+06	5.91E+06	5.91E+06	3.88E-14
^{238}U (n,f) ^{137}Cs (Cd)	Including ^{235}U, ^{239}Pu, and γ,fission corrections:				**3.01E-14**
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	8.16E+06	4.23E+07	4 23E+07	2.70E-13
^{237}Np (n,f) ^{137}Cs (Cd)	Including γ,fission correction:				**2.67E-13**
^{59}Co (n,γ) ^{60}Co	Top	2.78E+07	4.72E+07	4.72E+07	3 08E-12
	Middle	3.13E+07	5.32E+07	5.32E+07	3.47E-12
	Bottom	2 63E+07	4.47E+07	4.47E+07	2.92E-12
	Average				**3.16E-12**
^{59}Co (n,γ) ^{60}Co (Cd)	Top	1.66E+07	2.82E+07	2.82E+07	1.84E-12
	Middle	1.62E+07	2.75E+07	2.75E+07	1.80E-12
	Bottom	1.57E+07	2.67E+07	2.67E+07	1.74E-12
	Average				**1.79E-12**

Notes: 1) Measured specific activities are indexed to a counting date of May 18, 1998.

2) The average ^{238}U (n,f) reaction rate of 3.01E-14 includes a correction factor of 0.80 to account for plutonium build-in and an additional factor of 0.97 to account for photo-fission effects in the sensor

3) The average ^{237}Np (n,f) reaction rate of 2.67E-13 includes a correction factor of 0.99 to account for photo-fission effects in the sensor.

Table A-4 cont'd

Measured Sensor Activities and Reaction Rates

Surveillance Capsule X

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Radially Adjusted Saturated Activity (dps/g)	Radially Adjusted Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	2.39E+05	3.27E+05	3.27E+05	4.99E-17
	Center	2.17E+05	2.97E+05	2.97E+05	4.53E-17
	Bottom	2.13E+05	2.92E+05	2.92E+05	4 45E-17
	Average				**4.66E-17**
^{54}Fe (n,p) ^{54}Mn	Top	2.13E+06	3.19E+06	3.19E+06	5.06E-15
	Center	1.92E+06	2.88E+06	2.88E+06	4.56E-15
	Bottom	1.91E+06	2.86E+06	2.86E+06	4.54E-15
	Average				**4.72E-15**
^{58}Ni (n,p) ^{58}Co	Top	8.37E+06	4.75E+07	4.75E+07	6.81E-15
	Center	7.81E+06	4.44E+07	4.44E+07	6.35E-15
	Bottom	7.75E+06	4.40E+07	4.40E+07	6.30E-15
	Average				**6.49E-15**
^{238}U (n,f) ^{137}Cs (Cd)	Middle	1.65E+06	6.25E+06	6.25E+06	4.11E-14
^{238}U (n,f) ^{137}Cs (Cd)		Including ^{235}U, ^{239}Pu, and γ,fission corrections			**3.02E-14**
^{237}Np (n,f) ^{137}Cs (Cd)	Middle	1.07E+07	4.06E+07	4.06E+07	2.59E-13
^{237}Np (n,f) ^{137}Cs (Cd)			Including γ,fission correction		**2.56E-13**
^{59}Co (n,γ) ^{60}Co	Top	4.42E+07	6.05E+07	6.05E+07	3.95E-12
	Bottom	4.44E+07	6.08E+07	6.08E+07	3.96E-12
	Average				**3.96E-12**
^{59}Co (n,γ) ^{60}Co (Cd)	Top	2.46E+07	3.37E+07	3.37E+07	2.20E-12
	Middle	2.27E+07	3.11E+07	3.11E+07	2.03E-12
	Bottom	2.40E+07	3.28E+07	3.28E+07	2.14E-12
	Average				**2.12E-12**

Notes: 1) Measured specific activities are indexed to a counting date of September 20, 2002.
2) The average ^{238}U (n,f) reaction rate of 3.02E-14 includes a correction factor of 0.76 to account for plutonium build-in and an additional factor of 0.97 to account for photo-fission effects in the sensor.
3) The average ^{237}Np (n,f) reaction rate of 2.56E-13 includes a correction factor of 0.99 to account for photo-fission effects in the sensor.

Table A-5

Comparison of Measured, Calculated, and Best Estimate
Reaction Rates At the Surveillance Capsule Center

Capsule U

Reaction	Reaction Rate [rps/atom] Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$	5.52E-17	4 81E-17	5.39E-17	1.15	1.02
$^{54}Fe(n,p)^{54}Mn$	5.89E-15	5.45E-15	5.89E-15	1.08	1.00
$^{58}Ni(n,p)^{58}Co$	7.91E-15	7 65E-15	8.16E-15	1.03	0.97
$^{238}U(n,f)^{137}Cs$ (Cd)	3.26E-14	2.96E-14	3.16E-14	1.10	1 03
$^{237}Np(n,f)^{137}Cs$ (Cd)	3.23E-13	2.92E-13	3.17E-13	1.11	1.02
$^{59}Co(n,\gamma)^{60}Co$	5.29E-12	4.22E-12	5.20E-12	1.25	1.02
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.66E-12	2.92E-12	2.70E-12	0.91	0.98

Capsule Y

Reaction	Reaction Rate [rps/atom] Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$	4.74E-17	4.24E-17	4.53E-17	1.12	1.05
$^{54}Fe(n,p)^{54}Mn$	4.50E-15	4.68E-15	4.65E-15	0.96	0.97
$^{58}Ni(n,p)^{58}Co$	6.11E-15	6.56E-15	6.45E-15	0.93	0.95
$^{238}U(n,f)^{137}Cs$ (Cd)	2.84E-14	2.51E-14	2.52E-14	1.13	1.12
$^{237}Np(n,f)^{137}Cs$ (Cd)	2.73E-13	2.45E-13	2.62E-13	1.11	1.04
$^{59}Co(n,\gamma)^{60}Co$	4.15E-12	3.48E-12	4.08E-12	1.20	1.02
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.17E-12	2.42E-12	2.21E-12	0.90	0.98

Capsule V

Reaction	Reaction Rate [rps/atom] Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$	4.40E-17	3.98E-17	4.30E-17	1.11	1.02
$^{54}Fe(n,p)^{54}Mn$	4.43E-15	4.35E-15	4.64E-15	1.02	0.95
$^{58}Ni(n,p)^{58}Co$	6.45E-15	6.09E-15	6.56E-15	1.06	0.98
$^{238}U(n,f)^{137}Cs$ (Cd)	3.01E-14	2.32E-14	2.57E-14	1.30	1.18
$^{237}Np(n,f)^{137}Cs$ (Cd)	2.67E-13	2.26E-13	2.60E-13	1.18	1.03
$^{59}Co(n,\gamma)^{60}Co$	3.15E-12	3.18E-12	3.12E-12	0.99	1.01
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	1.79E-12	2.21E-12	1.82E-12	0.81	0.98

Capsule X

Reaction	Reaction Rate [rps/atom] Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$	4.65E-17	4.34E-17	4.53E-17	1.07	1.03
$^{54}Fe(n,p)^{54}Mn$	4.72E-15	4.79E-15	4.84E-15	0.99	0.98
$^{58}Ni(n,p)^{58}Co$	6.49E-15	6.71E-15	6.73E-15	0.97	0.96
$^{238}U(n,f)^{137}Cs$ (Cd)	3.01E-14	2.56E-14	2.62E-14	1.18	1.15
$^{237}Np(n,f)^{137}Cs$ (Cd)	2.56E-13	2.50E-13	2.57E-13	1.02	1.00
$^{59}Co(n,\gamma)^{60}Co$	3.99E-12	3.56E-12	3.93E-12	1.12	1.02
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.12E-12	2.47E-12	2.16E-12	0.86	0.98

Table A-6

Comparison of Calculated and Best Estimate Exposure Rates
At The Surveillance Capsule Center

Capsule ID	$\phi(E > 1.0\ MeV)$ [n/cm^2-s]			
	Calculated	Best Estimate	Uncertainty (1σ)	BE/C
U	9.40E+10	1.00E+11	6%	1 06
Y	7 93E+10	8.02E+10	6%	1 01
V	7.31E+10	8.20E+10	6%	1 12
X	8.09E+10	8.31E+10	6%	1.03

Notes: 1) Calculated results are based on the synthesized transport calculations taken at the core midplane following the completion of each respective capsules irradiation period.

Capsule ID	Iron Atom Displacement Rate [dpa/s]			
	Calculated	Best Estimate	Uncertainty (1σ)	BE/C
U	1.82E-10	1.94E-10	8%	1.07
Y	1 53E-10	1.58E-10	8%	1 03
V	1 40E-10	1.58E-10	8%	1 13
X	1.55E-10	1 61E-10	8%	1.04

Notes· 1) Calculated results are based on the synthesized transport calculations taken at the core midplane following the completion of each respective capsules irradiation period.

Table A-7

Comparison of Measured/Calculated (M/C) Sensor Reaction Rate Ratios Including all Fast Neutron Threshold Reactions

Reaction	M/C Ratio			
	Capsule U	Capsule Y	Capsule V	Capsule X
$^{63}Cu(n,\alpha)^{60}Co$	1.15	1.12	1.11	1.07
$^{54}Fe(n,p)^{54}Mn$	1.08	0.96	1.02	0.99
$^{58}Ni(n,p)^{58}Co$	1.03	0.93	1.06	0.97
$^{238}U(n,p)^{137}Cs$ (Cd)	1.10	1.13	1.30	1.18
$^{237}Np(n,f)^{137}Cs$ (Cd)	1.11	1.11	1.18	1.02
Average	1.09	1.05	1.13	1.05
% Standard Deviation	4.0	9.2	9.7	8.0

Notes: 1) The overall average M/C ratio for the set of 20 sensor measurements is 1.08 with an associated standard deviation of 8.2%.

Table A-8

Comparison of Best Estimate/Calculated (BE/C) Exposure Rate Ratios

Capsule ID	BE/C Ratio	
	$\phi(E > 1.0\ MeV)$	dpa/s
U	1.07	1.07
Y	1.02	1.03
V	1.14	1.12
X	1.04	1.03
Average	1.07	1.06
% Standard Deviation	4.9	4.1

Table A-9

Current and Projected Neutron Fluences (E > 1.0 MeV) Experienced by the Intermediate and Upper Circumferential Welds

Cumulative Operations Time (EFPY)	Fluences (n/cm^2-sec)			
	Circumferential		Vertical	
	Intermediate	Upper	0°	30°
13.83	7.97E+18	2.53E+17	4.33E+18	7.91E+18
15.53	9.05E+18	2.96E+17	4.86E+18	8.88E+18
20.00	1.20E+19	4.00E+17	6.12E+18	1.14E+19
24.00	1.47E+19	4.93E+17	7.26E+18	1.37E+19
32.00	2.01E+19	6.80E+17	9.53E+18	1.83E+19
40.00	2.54E+19	8.66E+17	1.18E+19	2.28E+19
48.00	3.07E+19	1.05E+18	1.41E+19	2.74E+19
54 00	3.47E+19	1.19E+18	1.58E+19	3.08E+19

1 Upper Circumferential weld location at 235.97 cm above core centerline and at an azimuth of 45° to document the maximum neutron fluence.
2. Intermediate Circumferential weld location at –38.35 cm below core centerline and at an azimuth of 45° to document the maximum neutron fluence.

A.2 Appendix A References

A-1. Regulatory Guide RG-1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

A-2. REAC-SAP-172, "Analysis of Neutron Dosimetry from Wolf Creek – Capsules U, Y, and V," Perock, J. D. April, 1998.

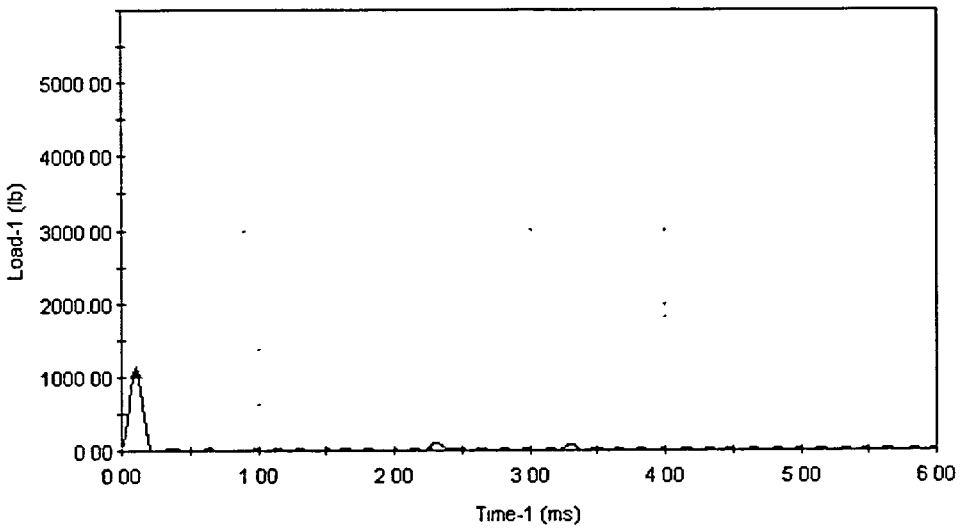
A-3. A. Schmittroth, *FERRET Data Analysis Core*, HEDL-TME 79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

A-4. RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium", July 1994.

APPENDIX B

LOAD-TIME RECORDS FOR CHARPY SPECIMEN TESTS

- Specimen prefix "AL" denotes Lower Plate, Longitudinal Orientation
- Specimen prefix "AT" denotes Lower Plate, Transverse Orientation
- Specimen prefix "AW" denotes Weld Material
- Specimen prefix "AH" denotes Heat-Affected Zone material
- Load (1) is in units of lbs
- Time (1) is in units of milli seconds



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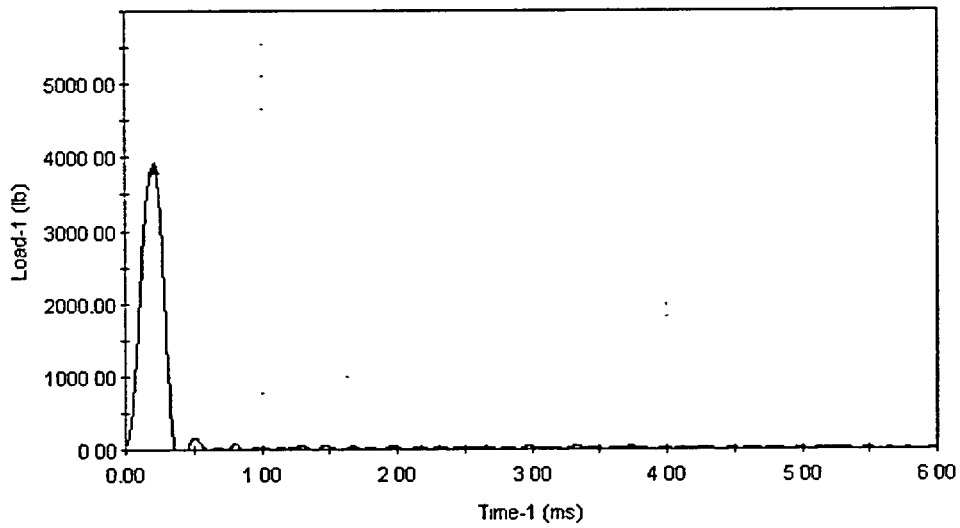

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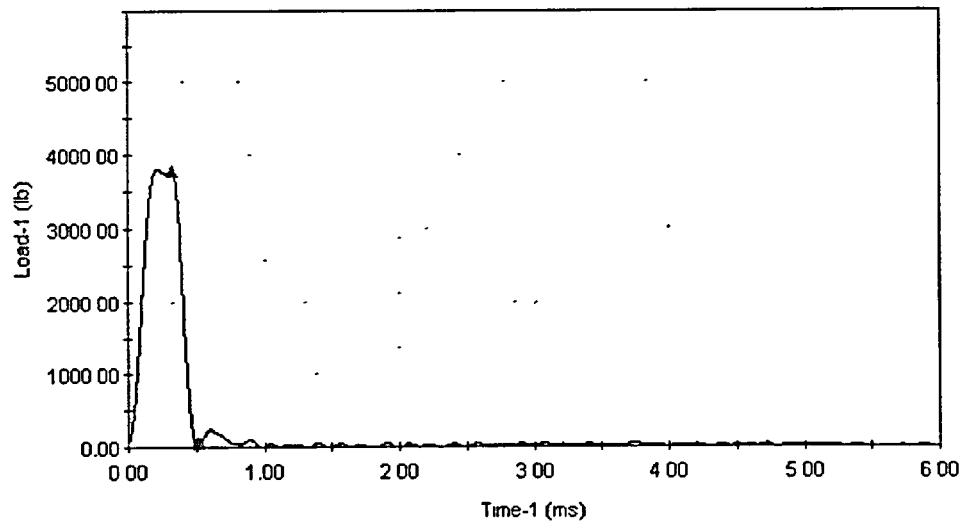
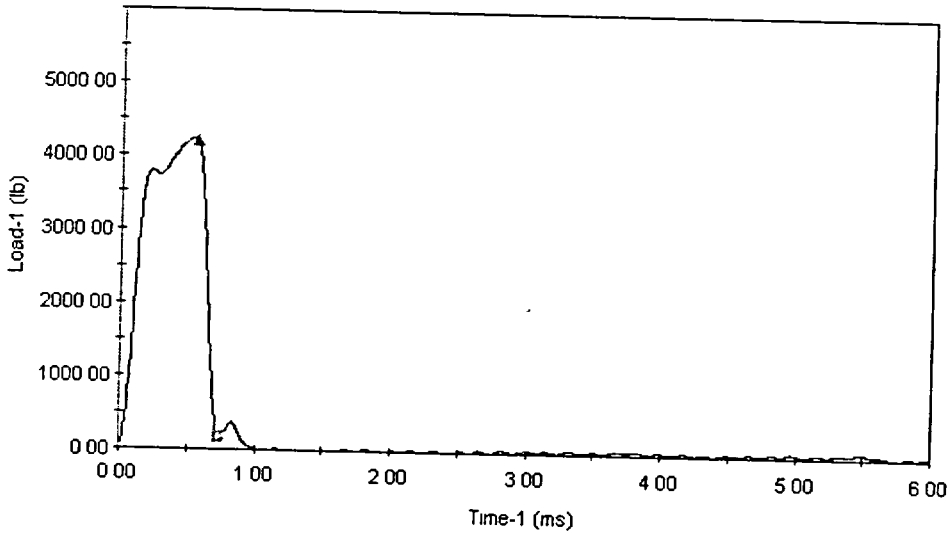


AL53, 25°F



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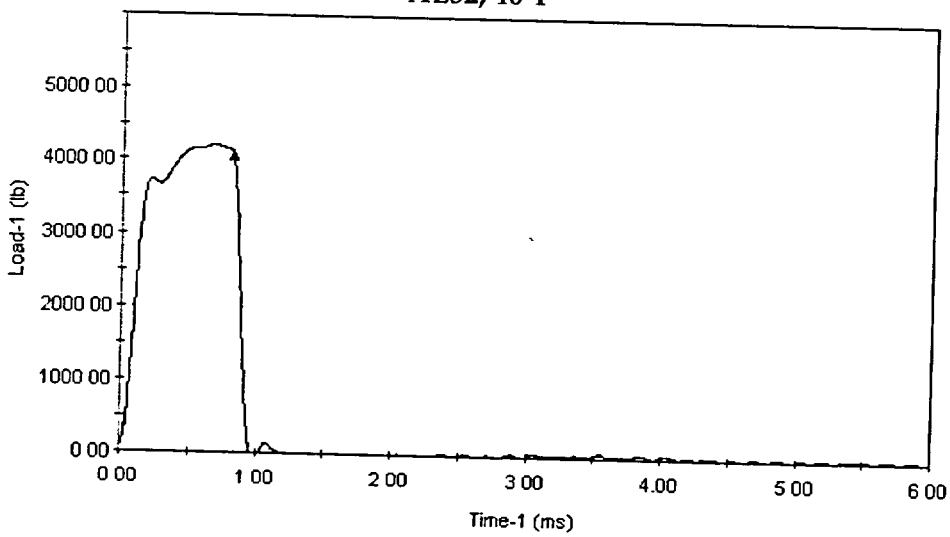

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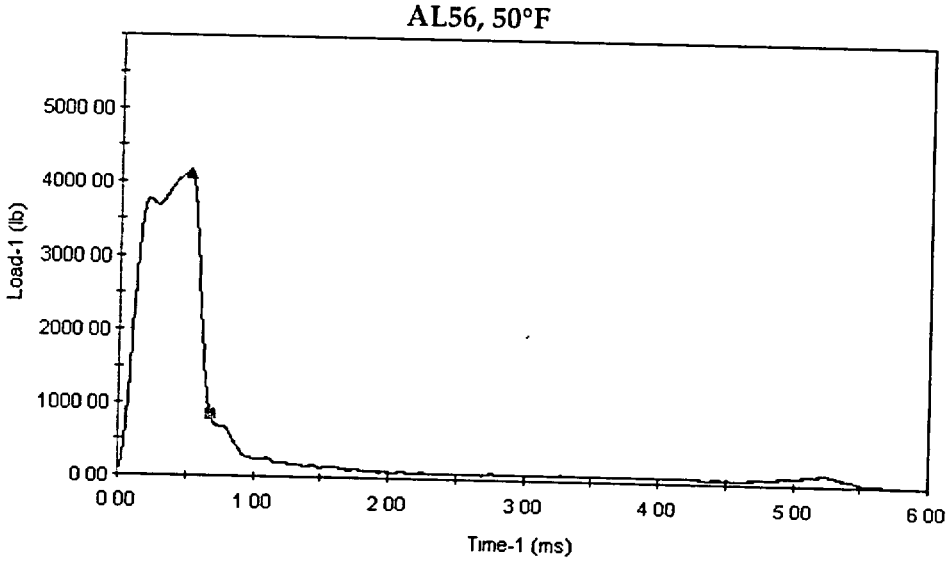

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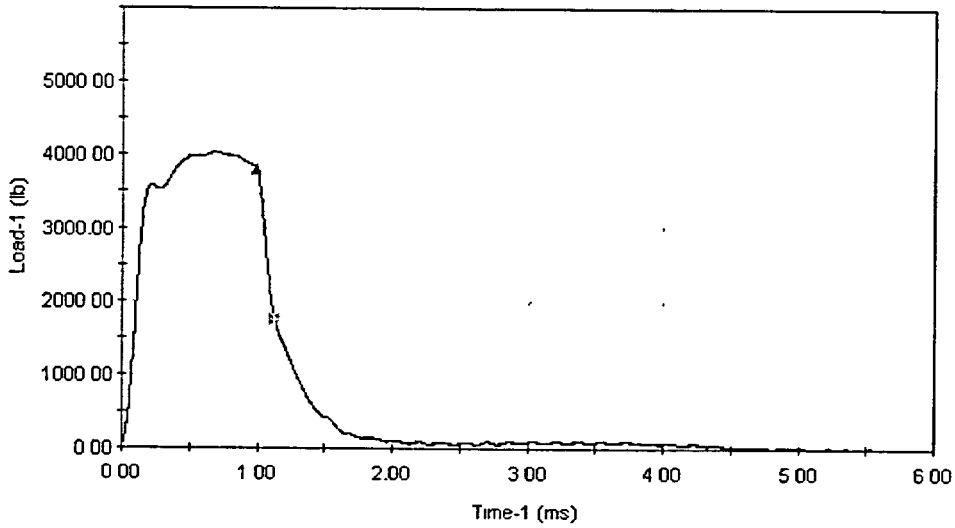
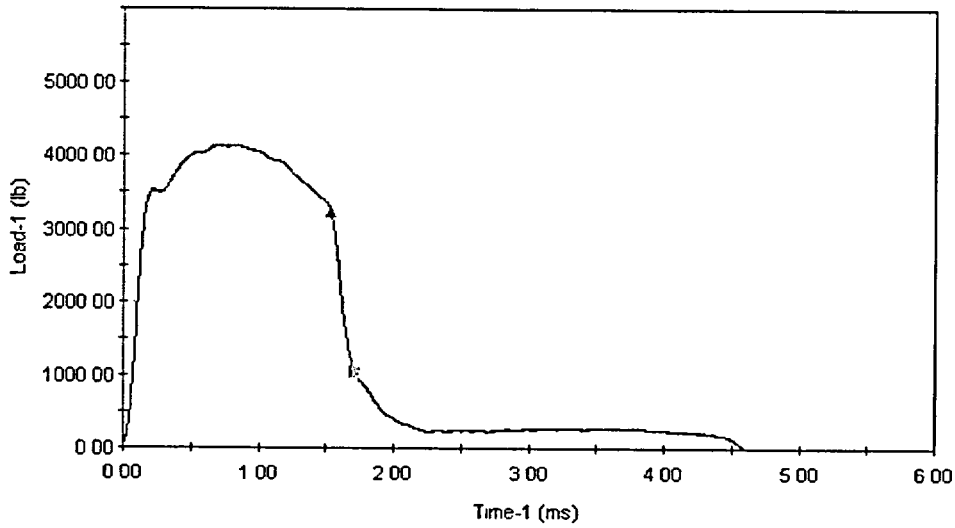
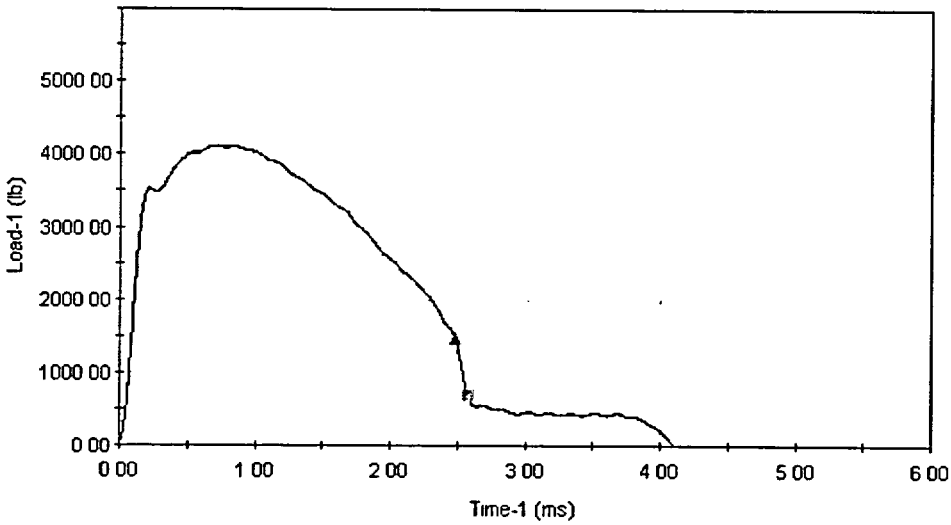


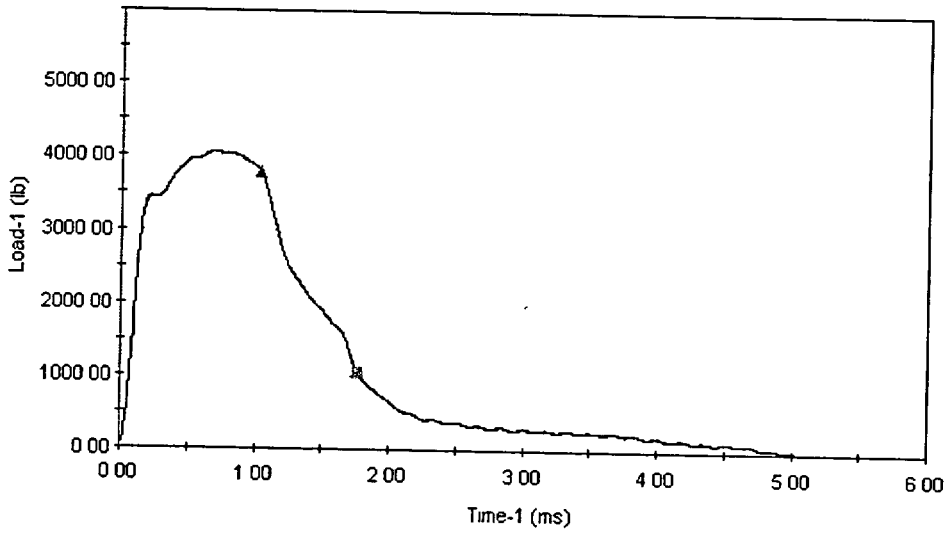
AL59, 110°F



AL50, 135°F



AL60, 150°F



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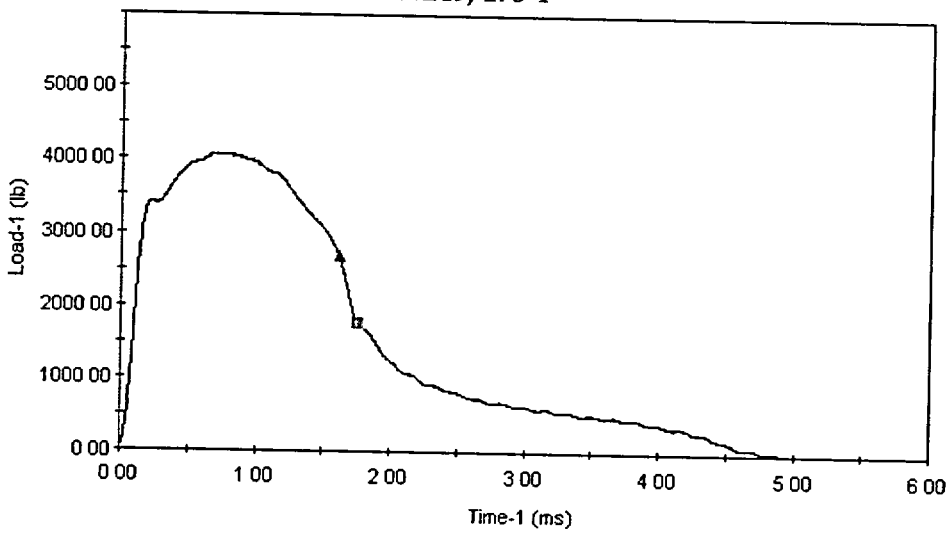
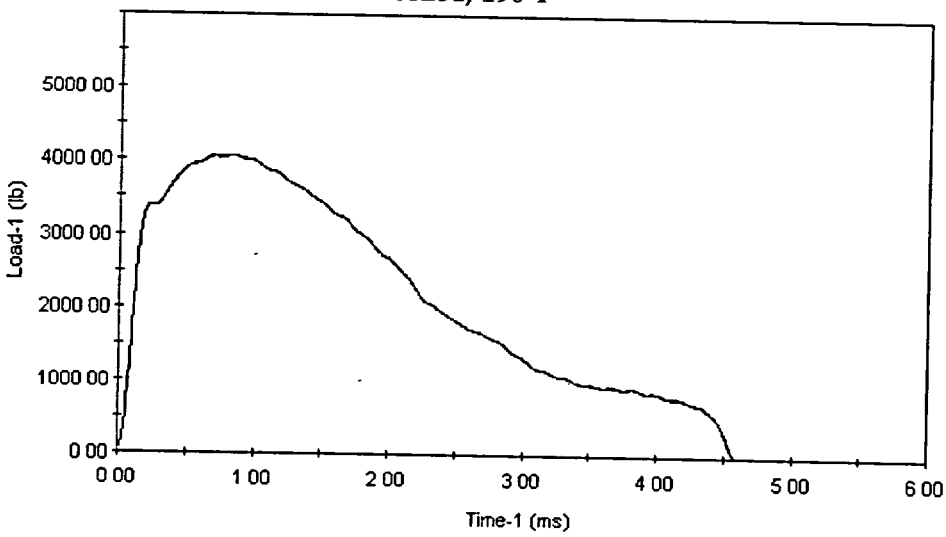
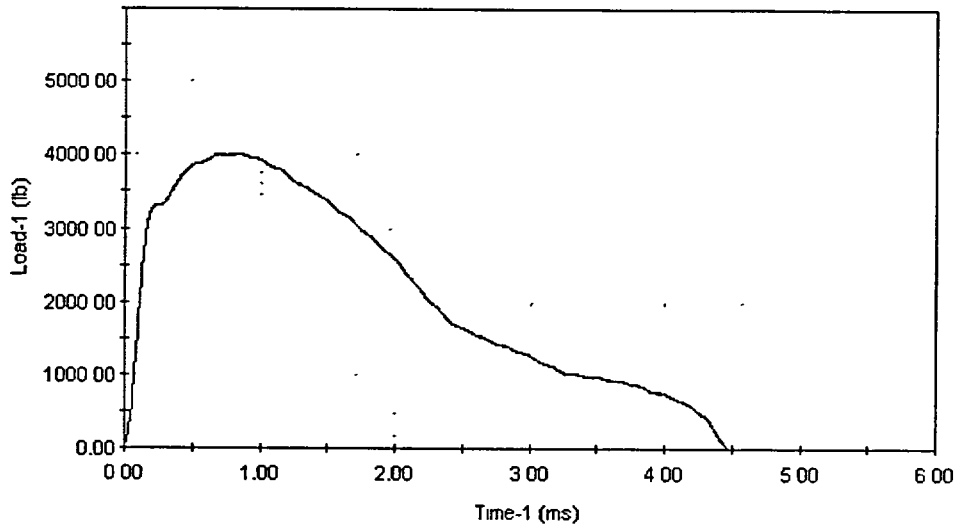


AL51, 190°F



AL46, 225°F



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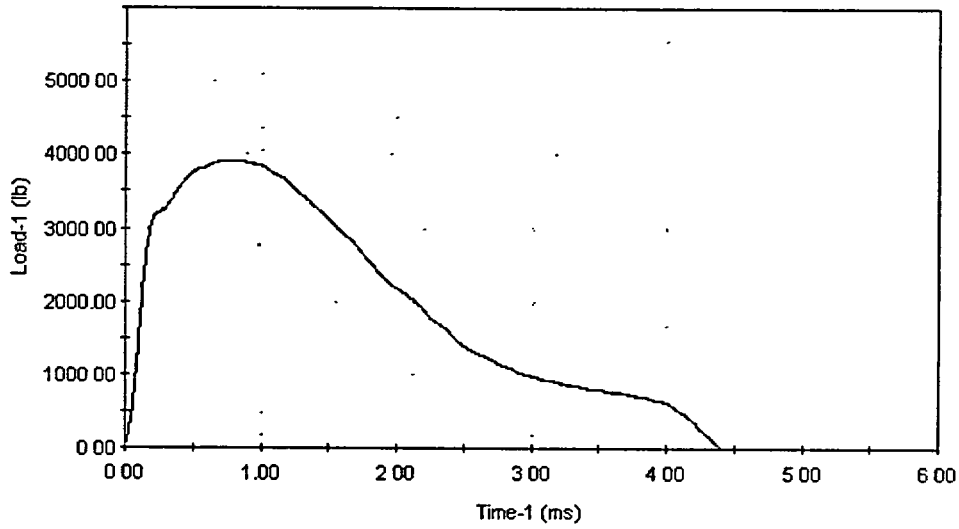
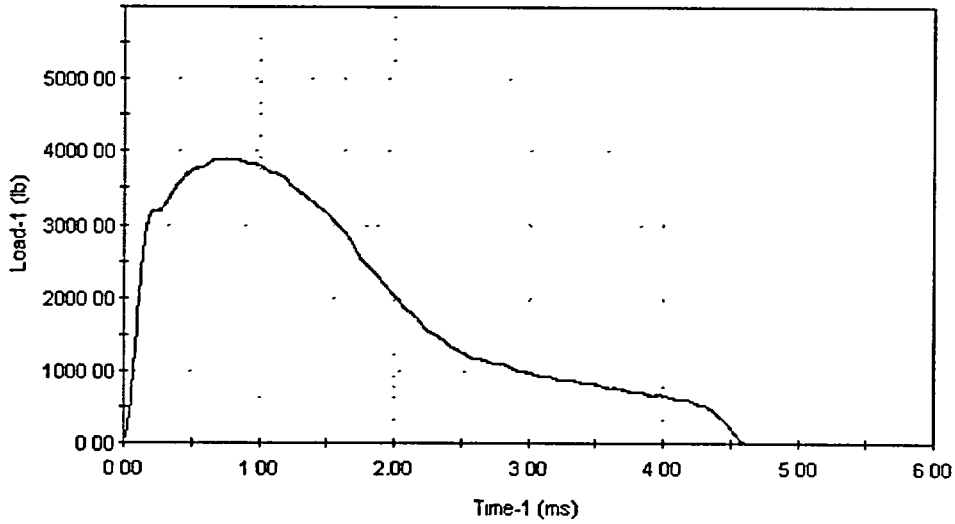


AL58, 275°F



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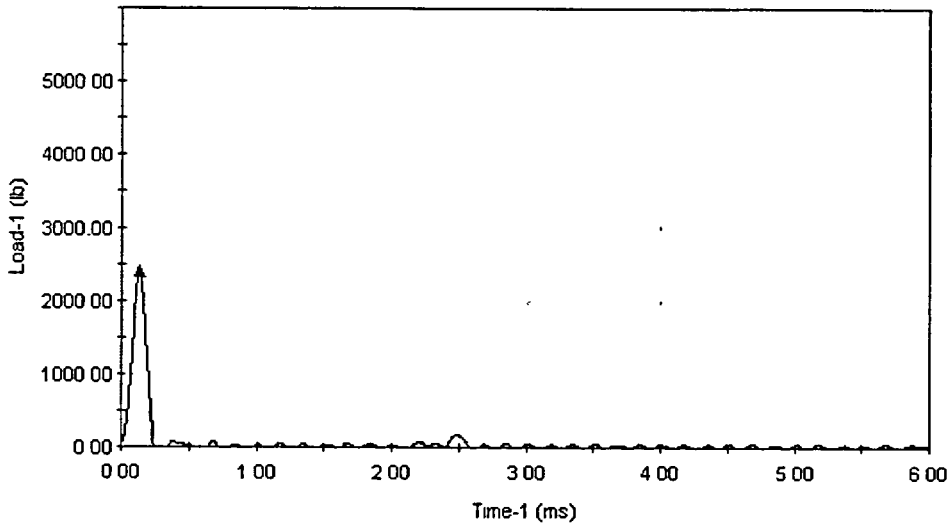

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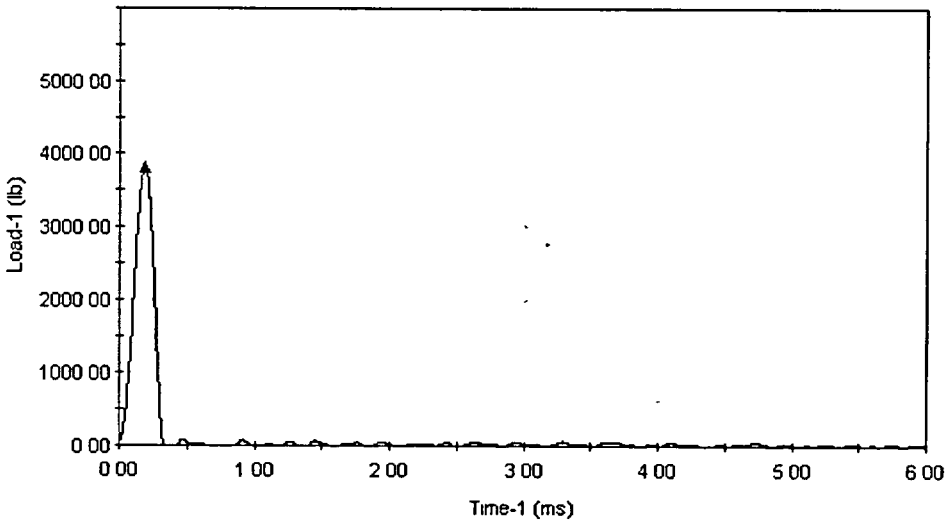

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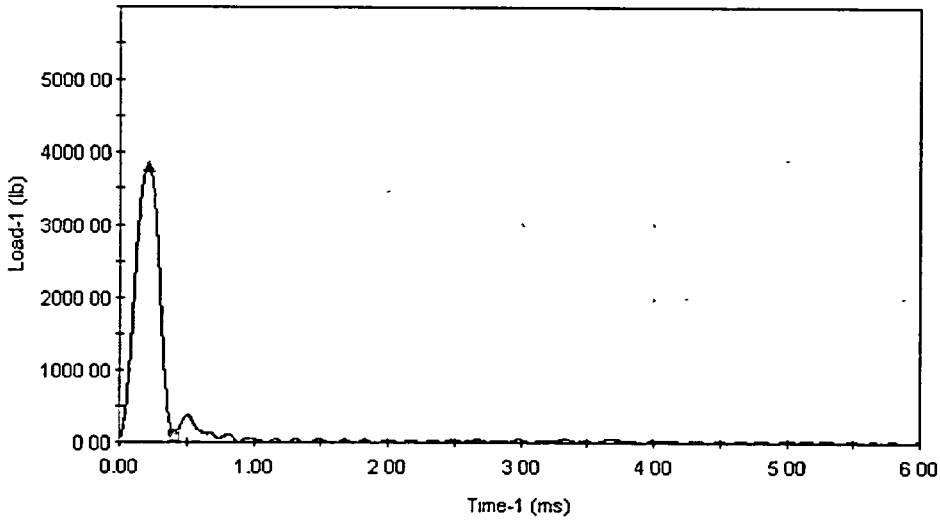
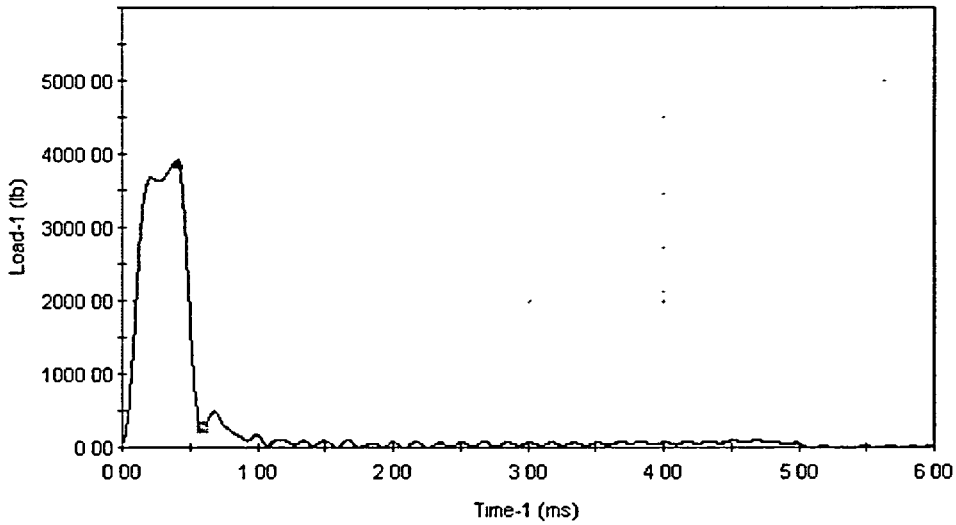


AT60, 15°F



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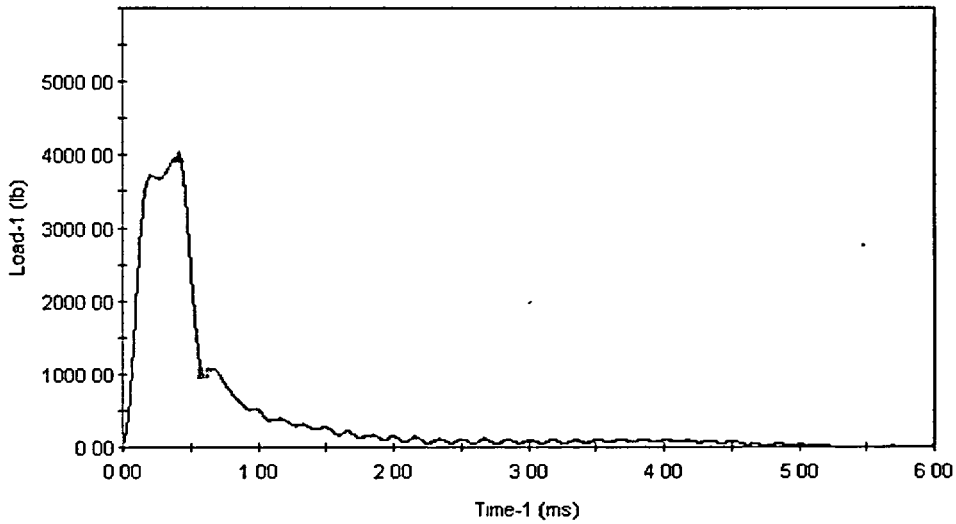
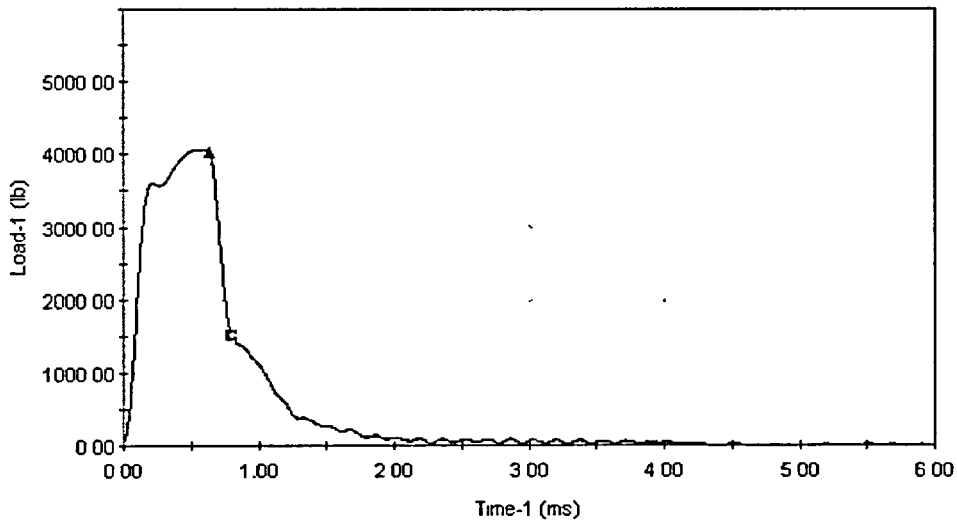


AT54, 75°F



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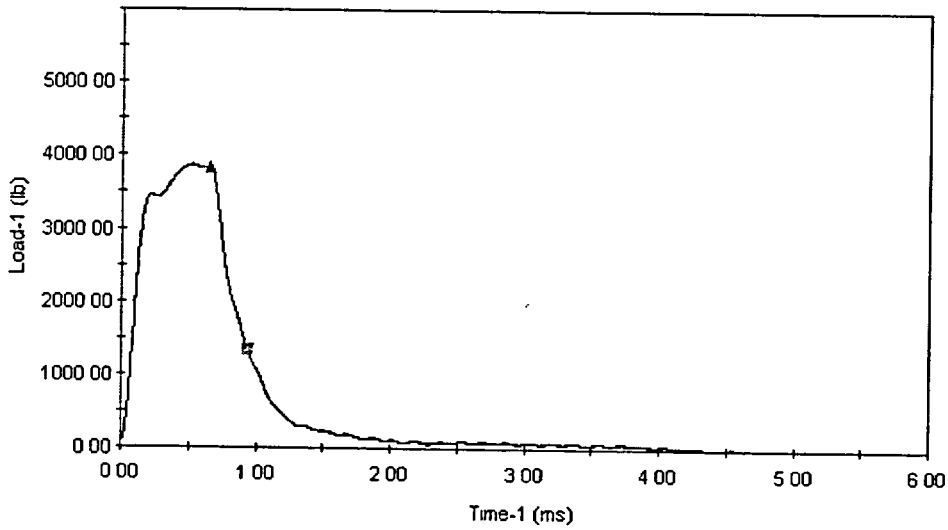

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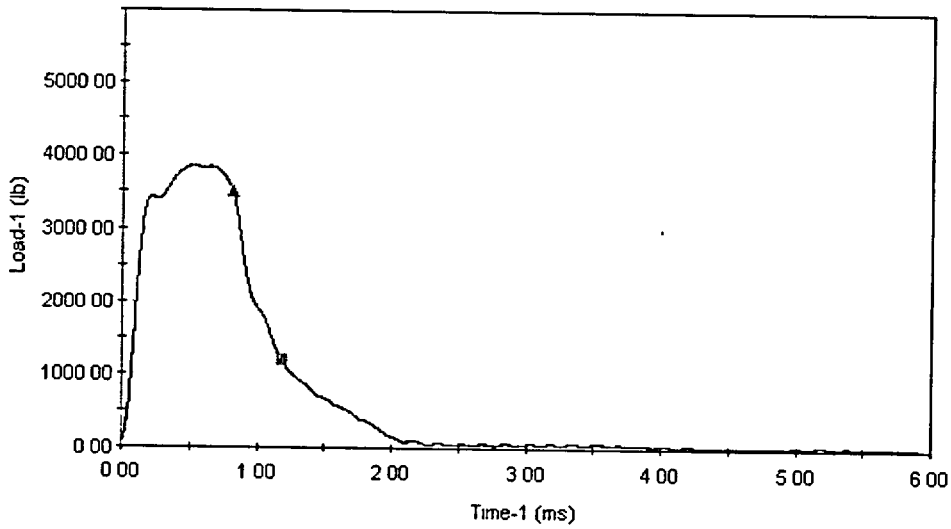

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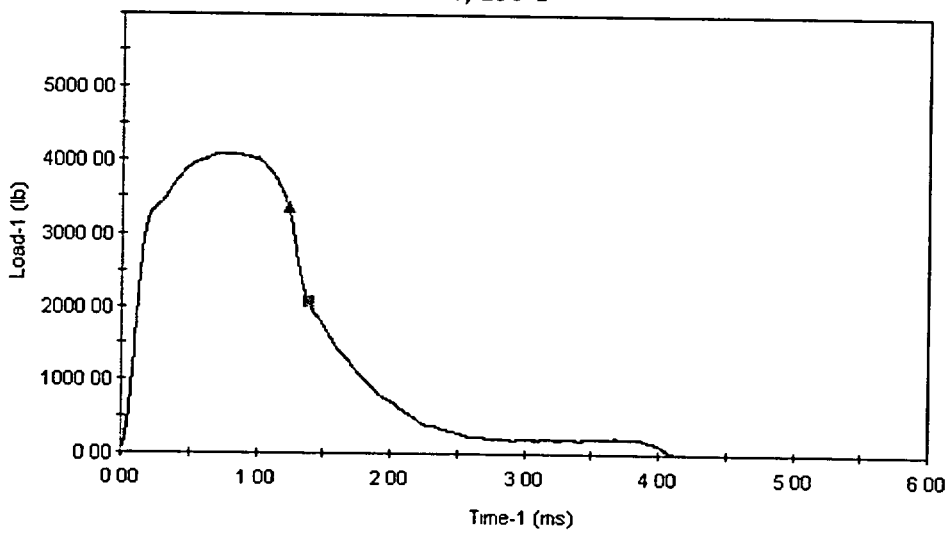

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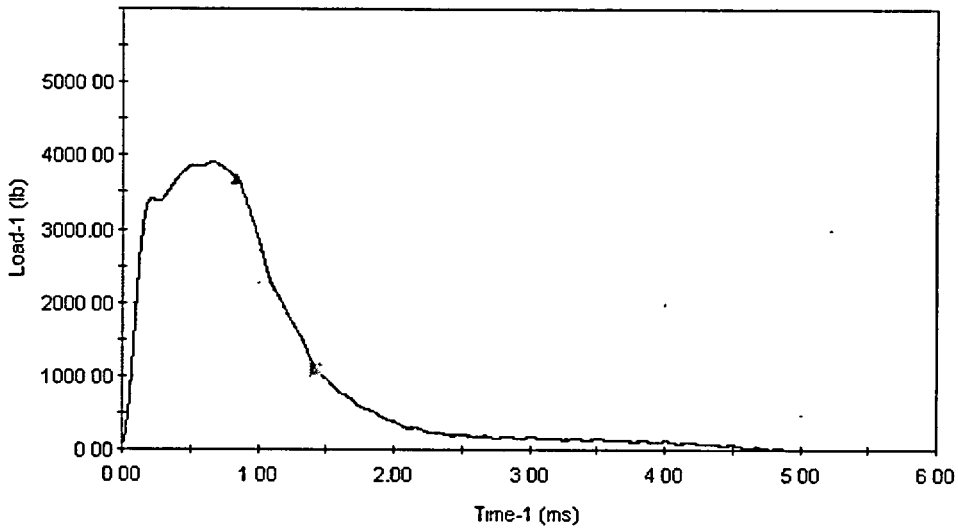

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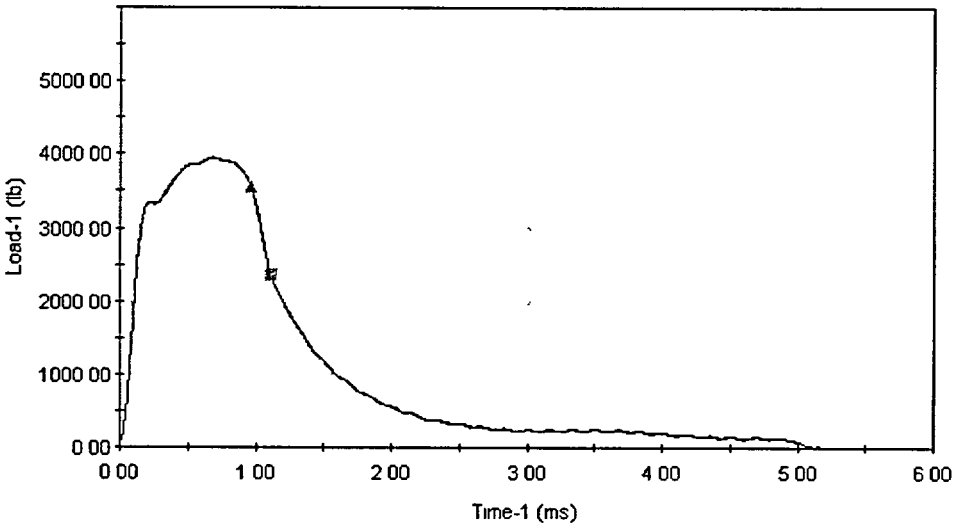

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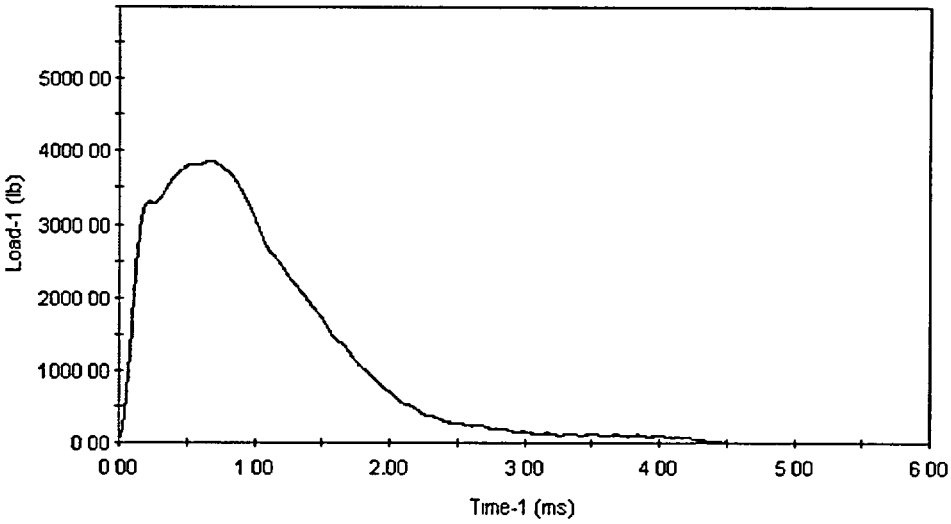

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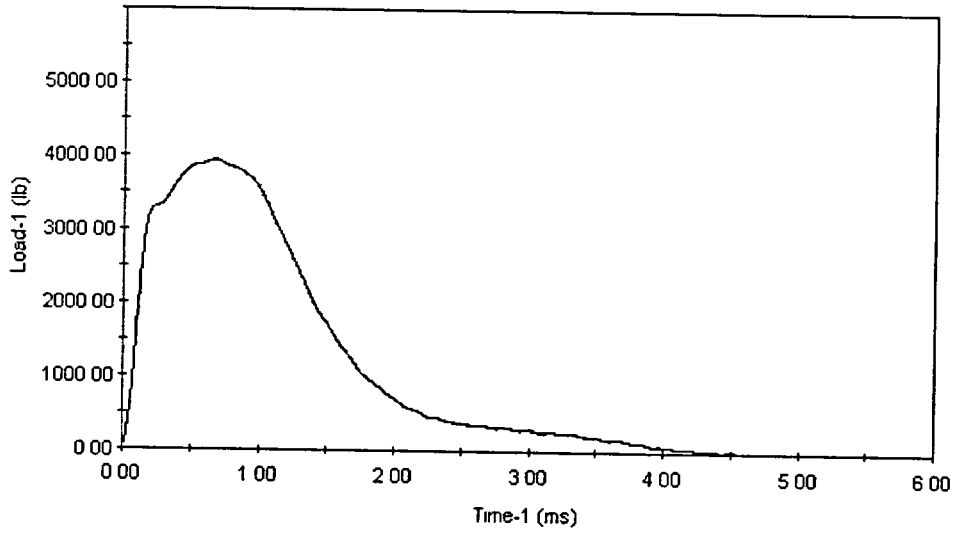

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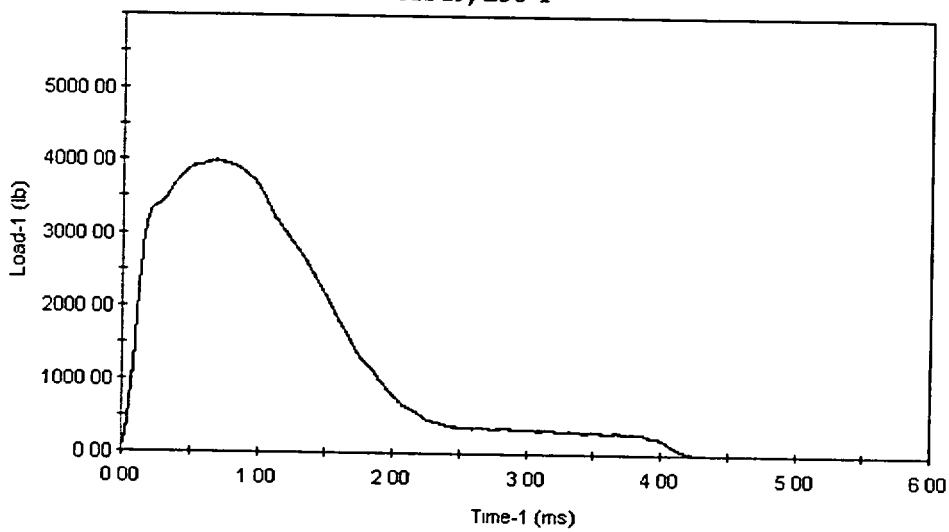

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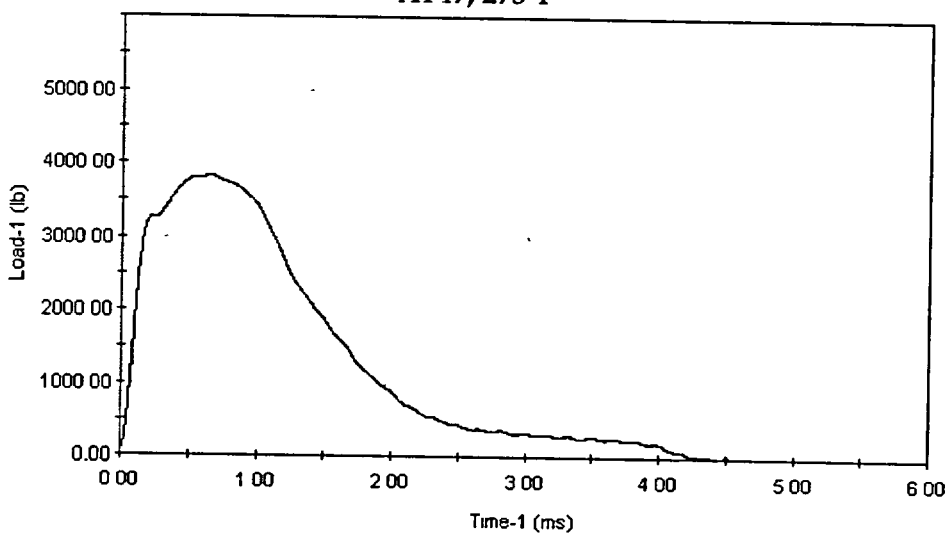

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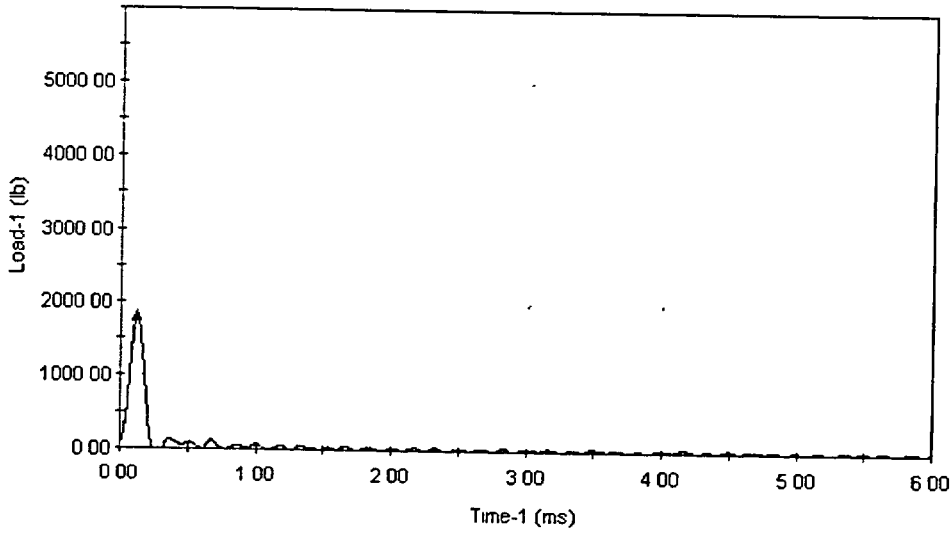

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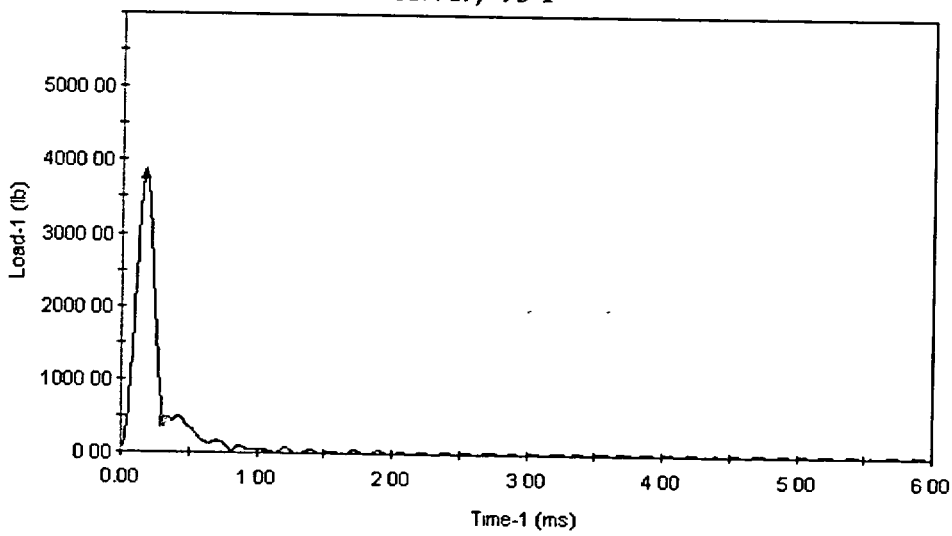

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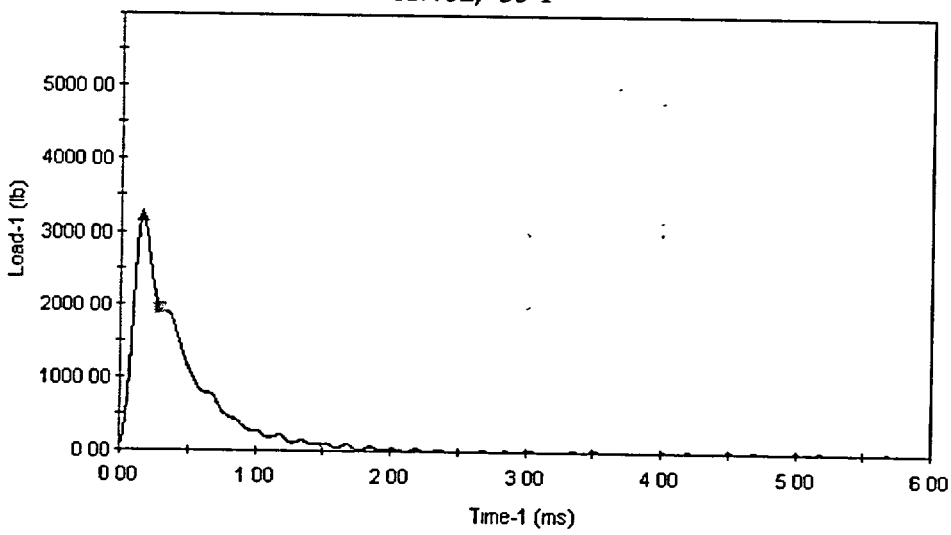
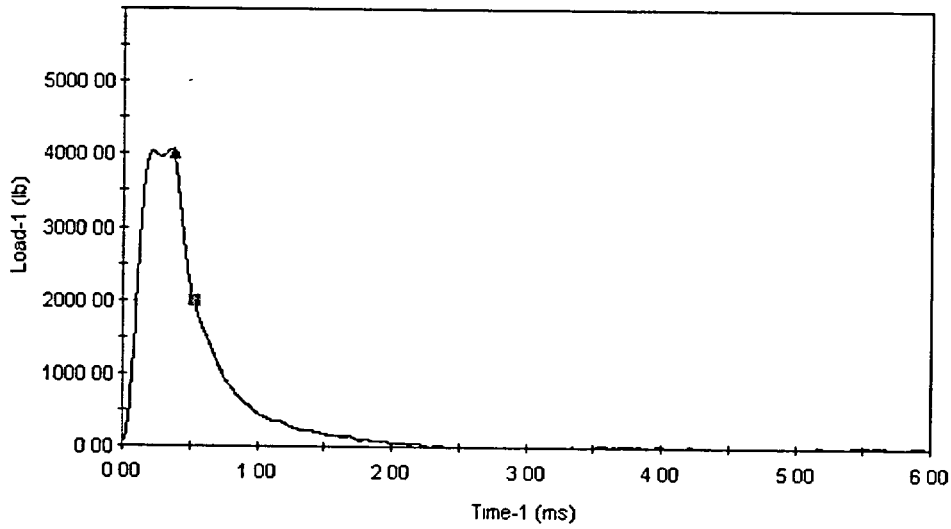
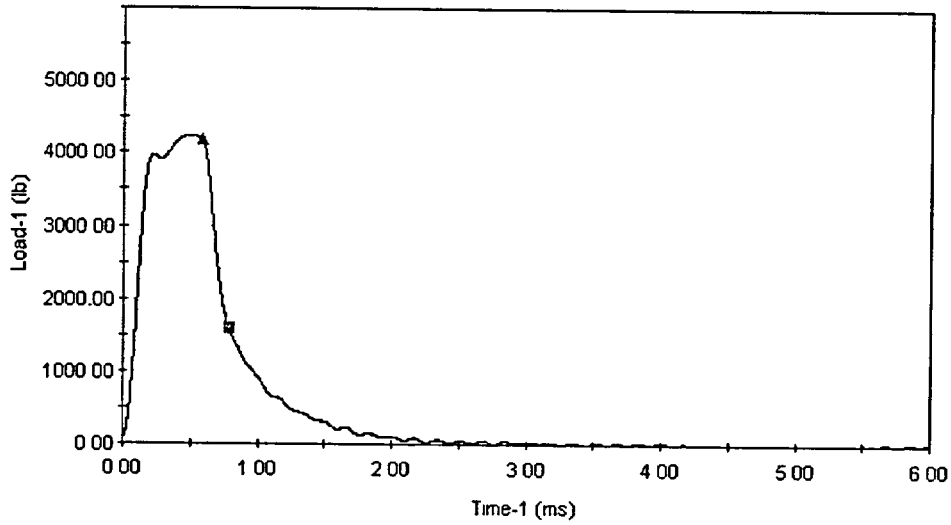


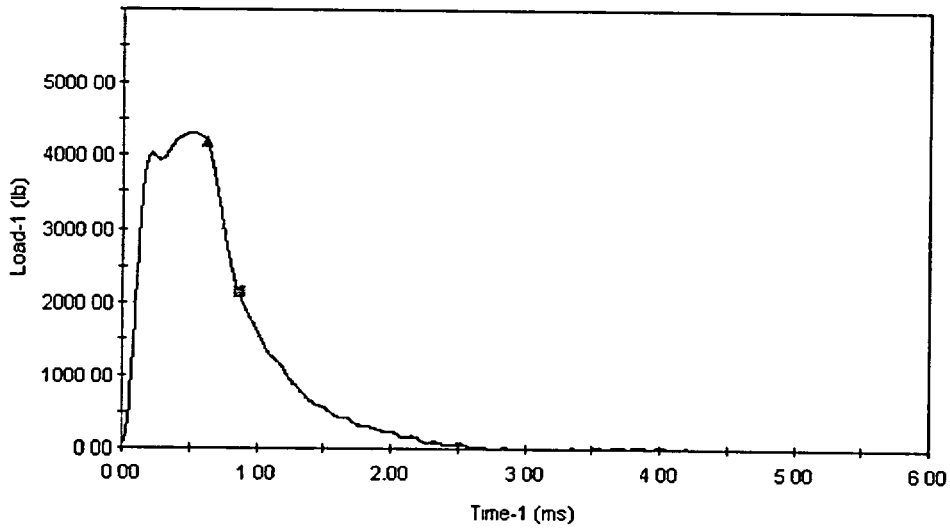
AW53, 0°F



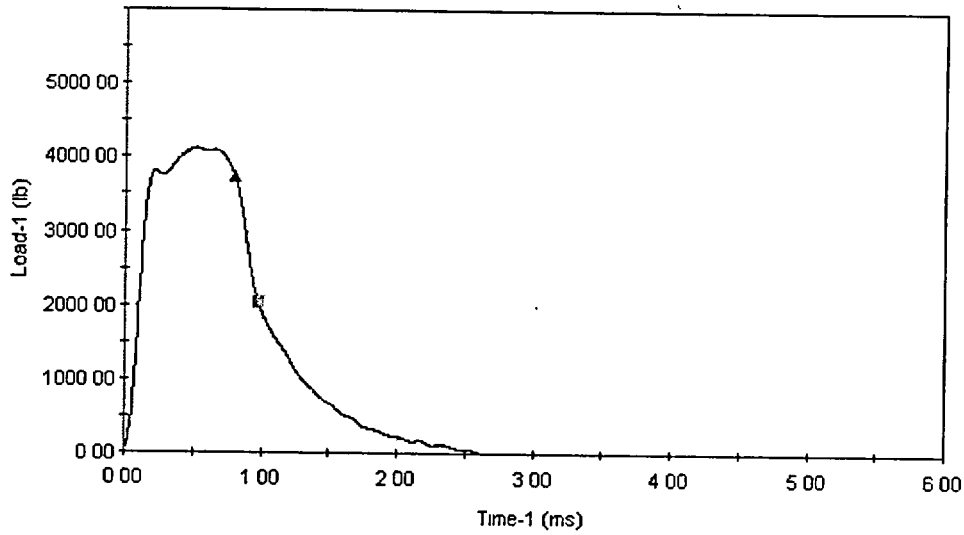
AW51, 25°F



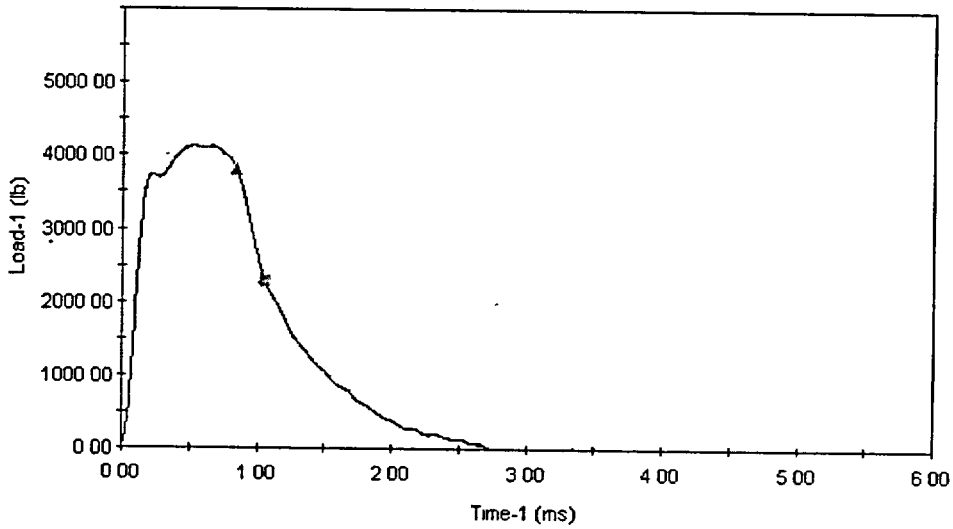
AW58, 50°F



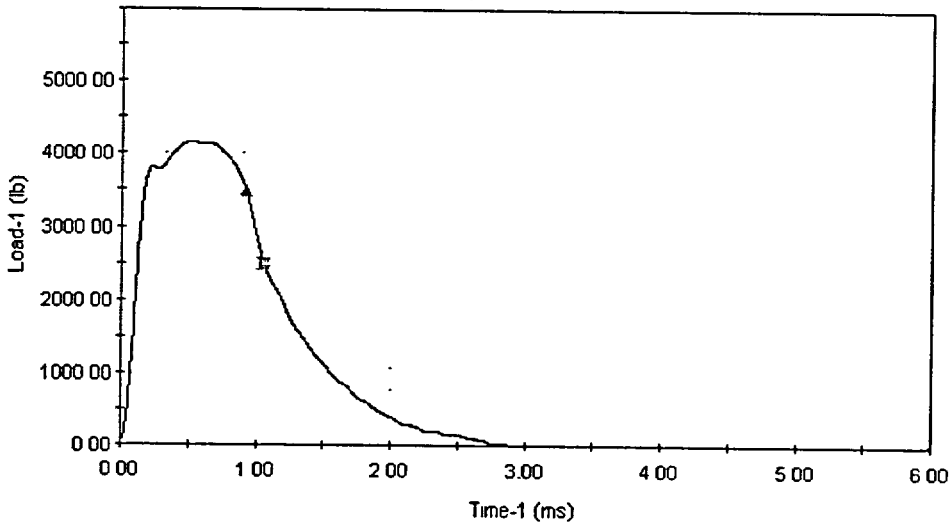
AW46, 75°F



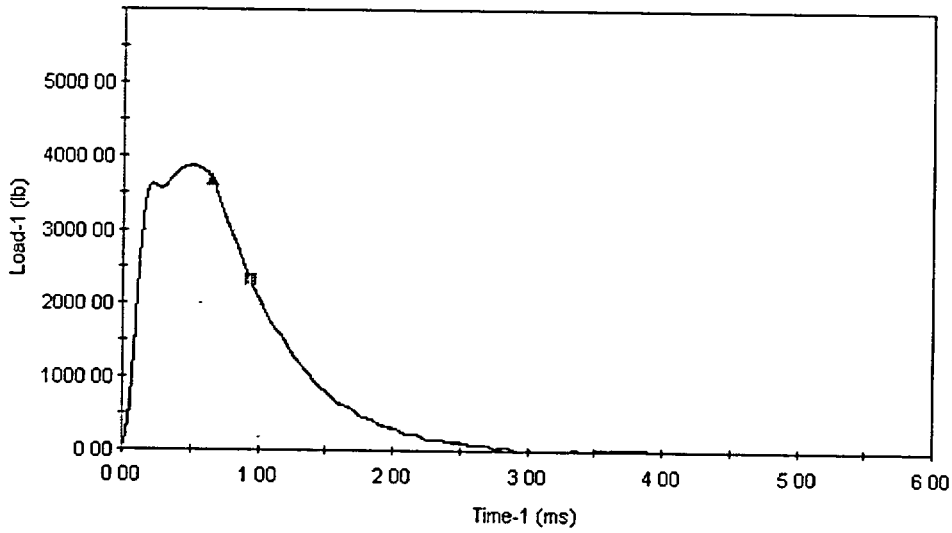
AW59, 100°F



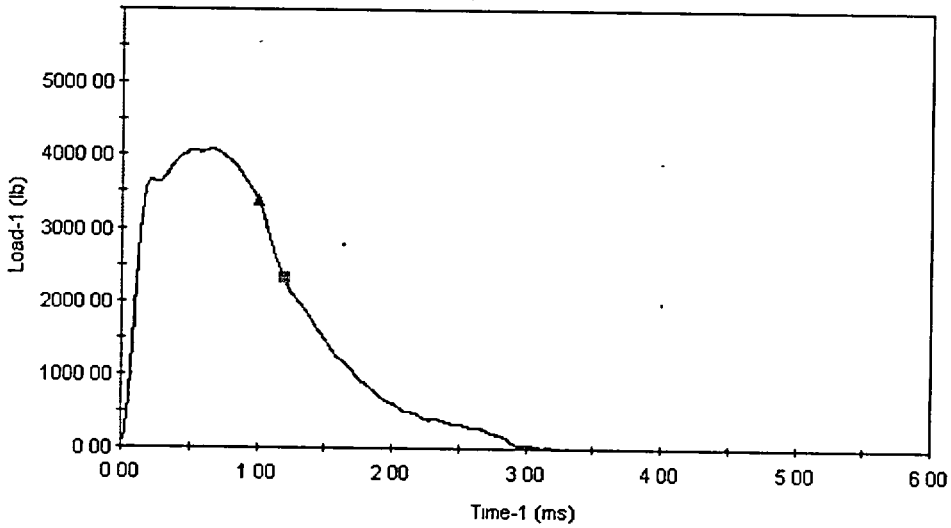
AW57, 125°F



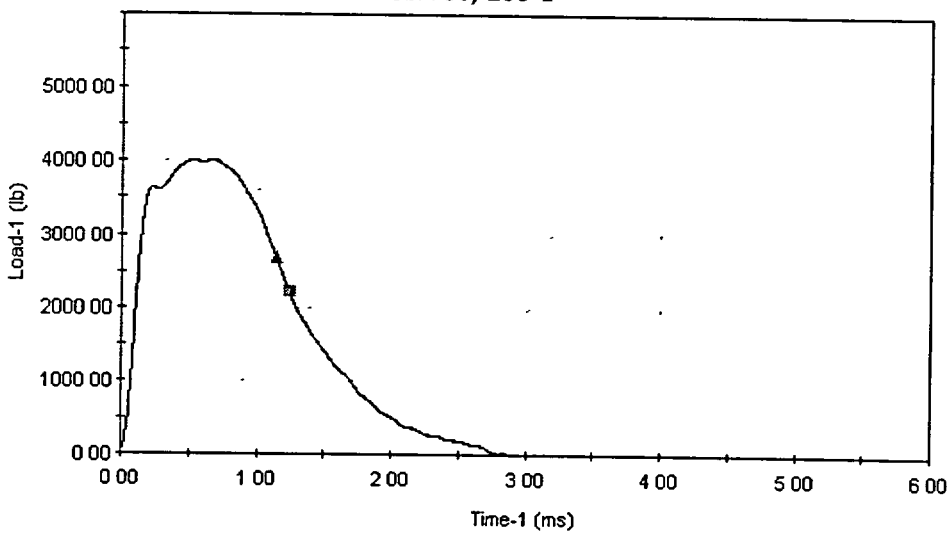
AW55, 125°F



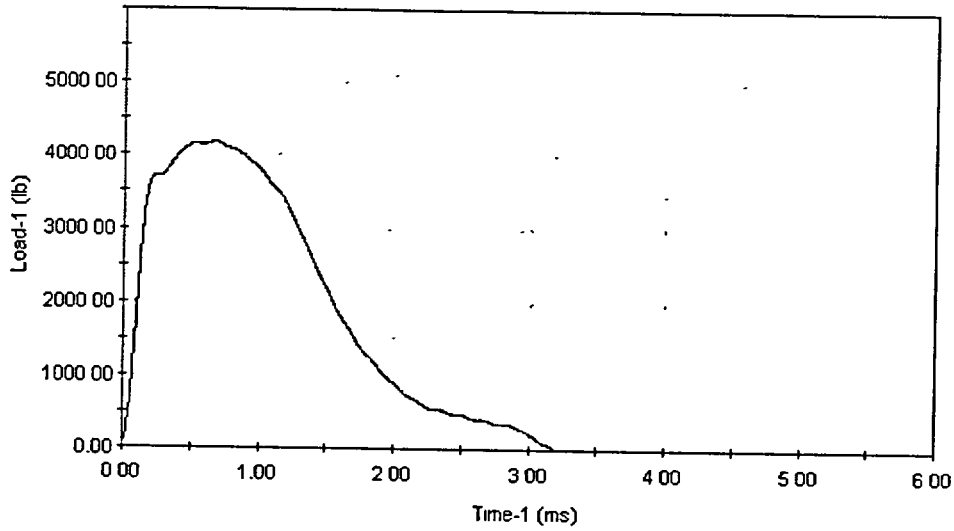
AW48, 150°F



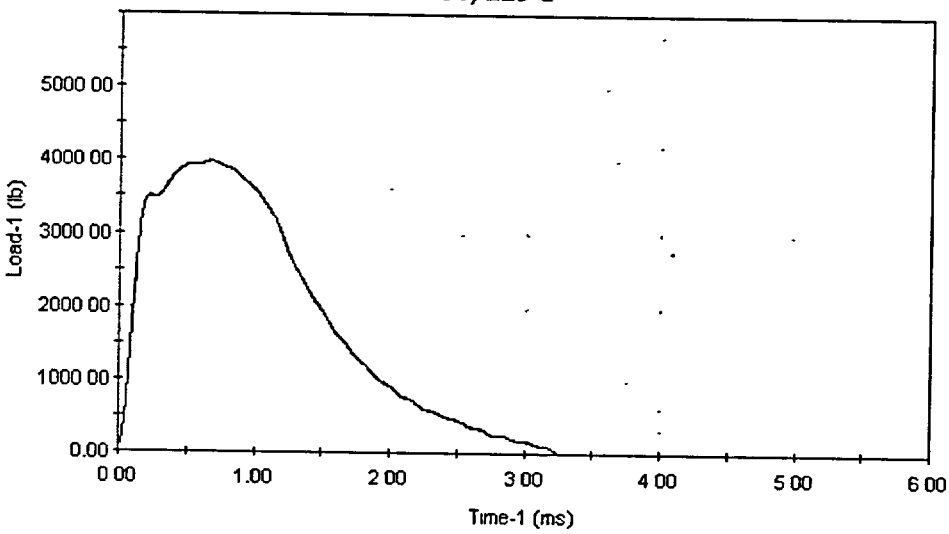
AW60, 160°F



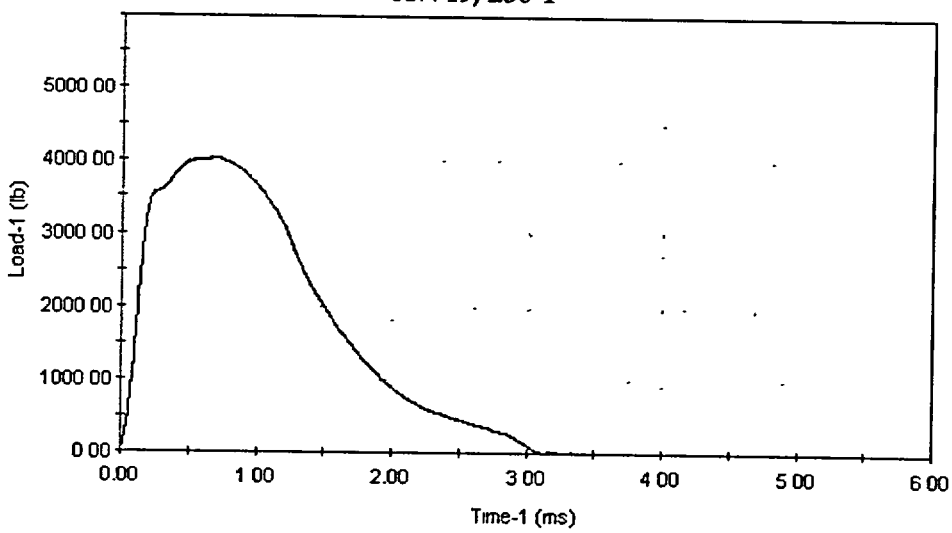
AW56, 200°F



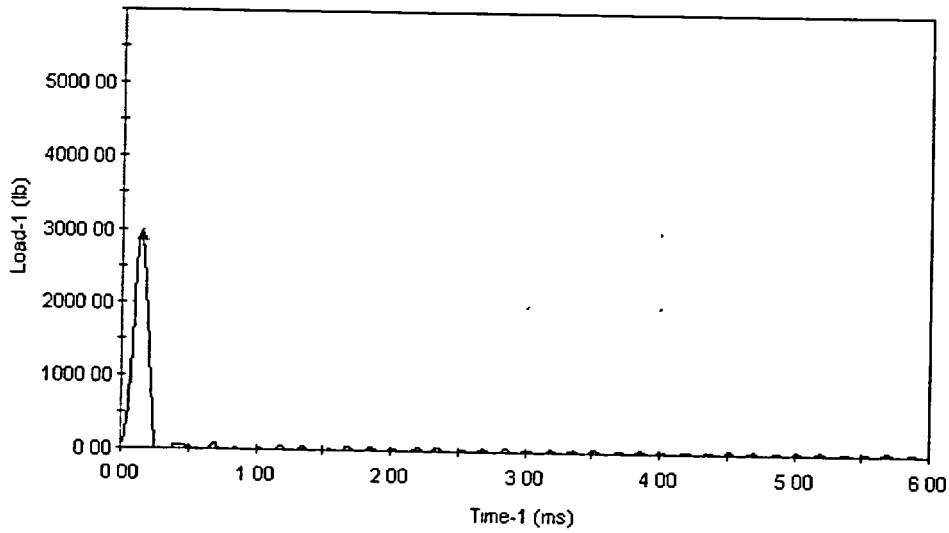
AW50, 225°F



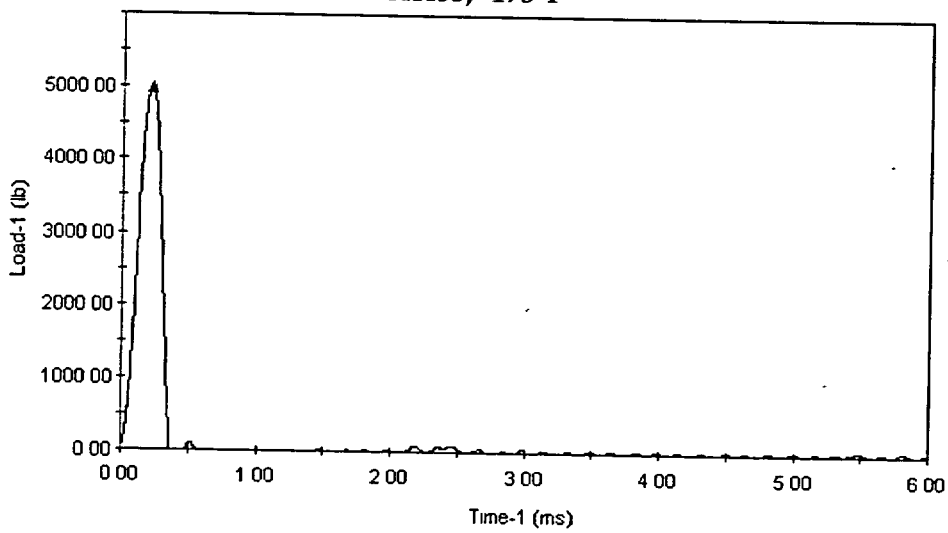
AW49, 250°F



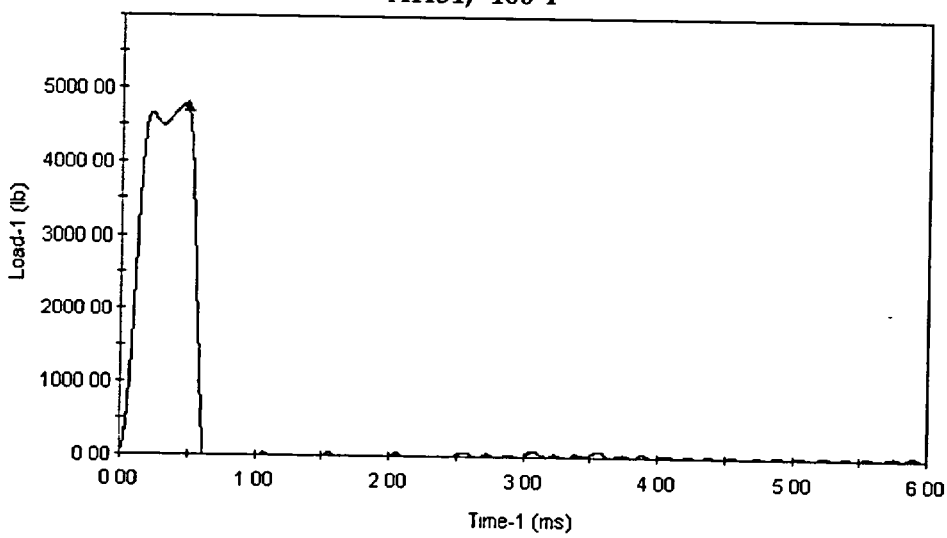
AW54, 250°F



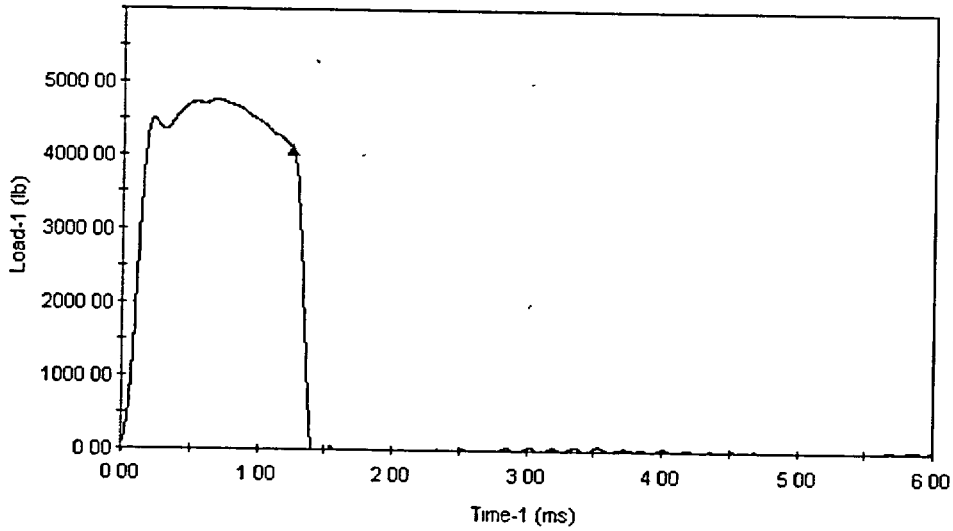
AH53, -175°F



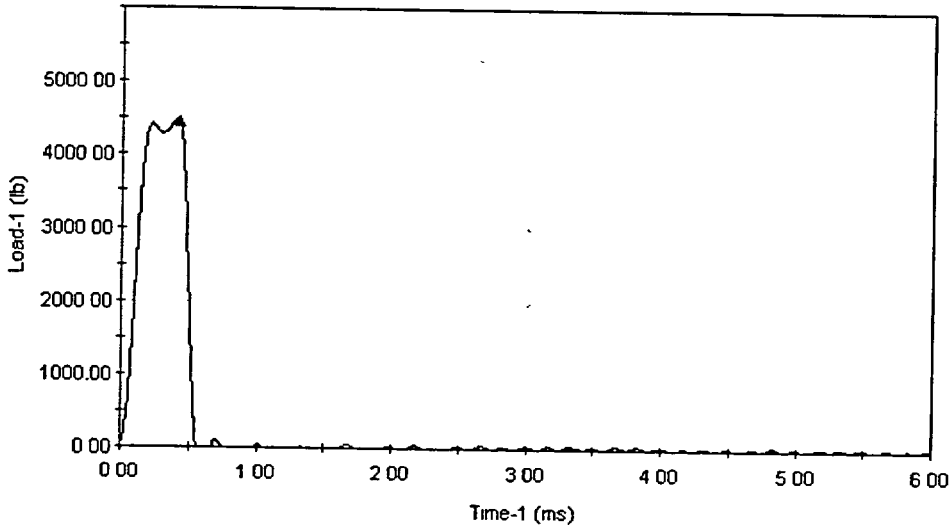
AH51, -100°F



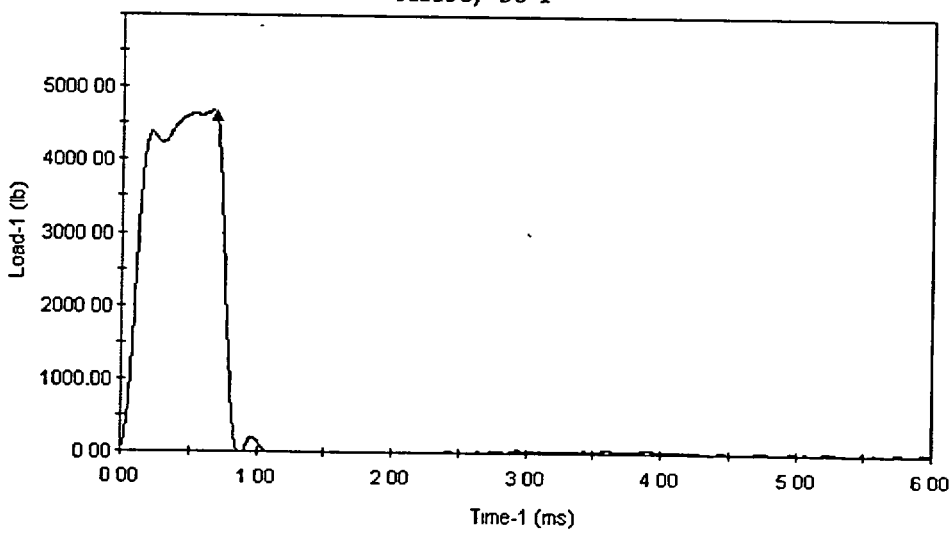
AH50, -75°F



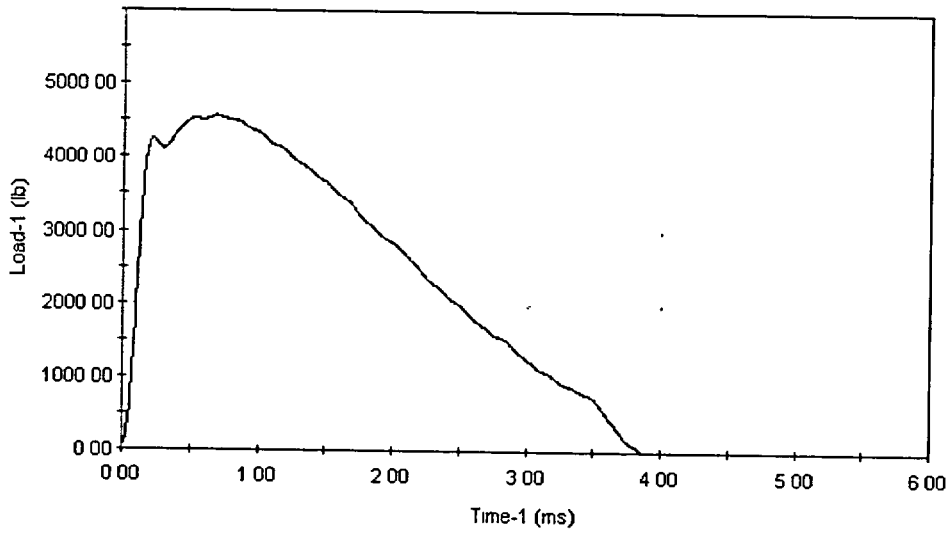
AH49, -50°F



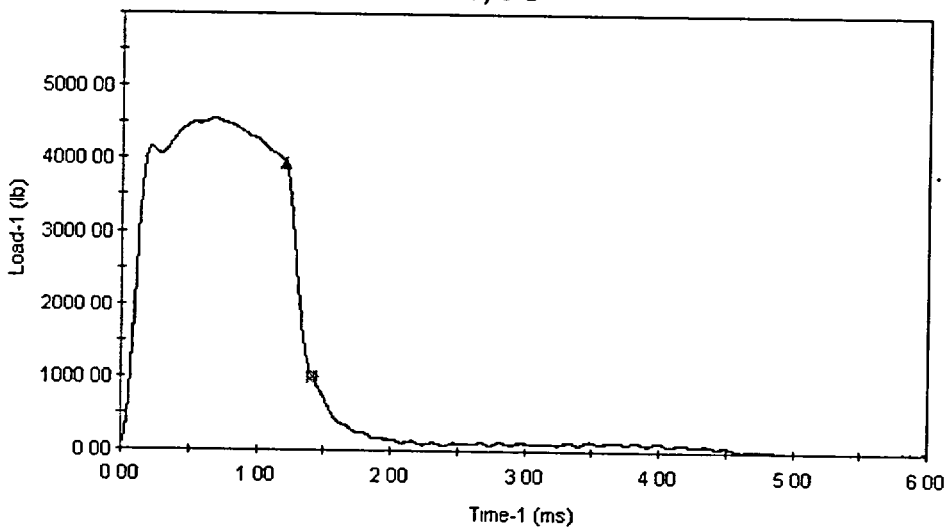
AH58, -50°F



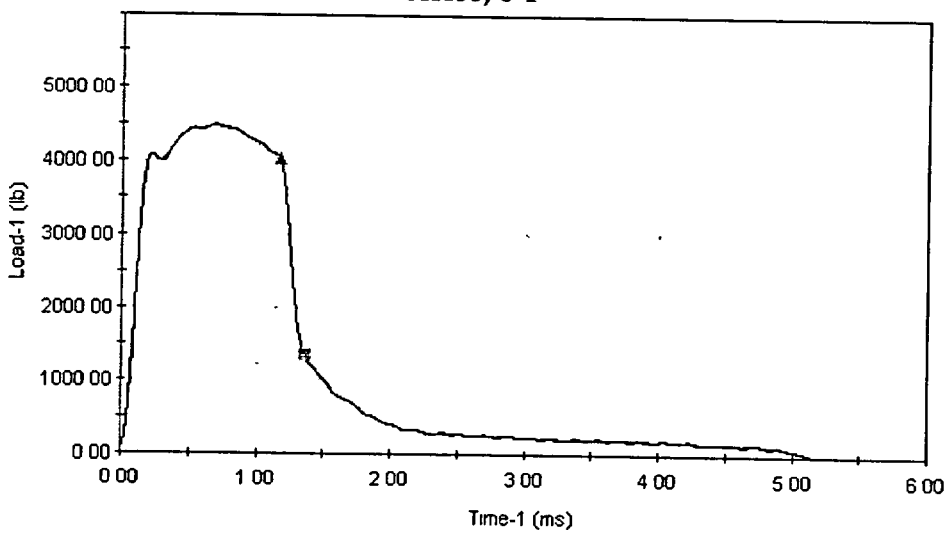
AH52, -25°F



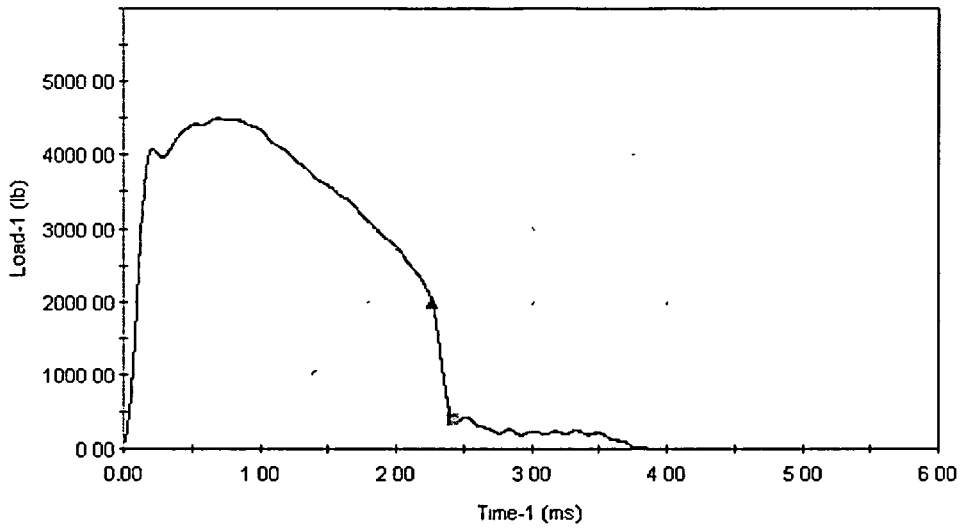
AH47, 0°F



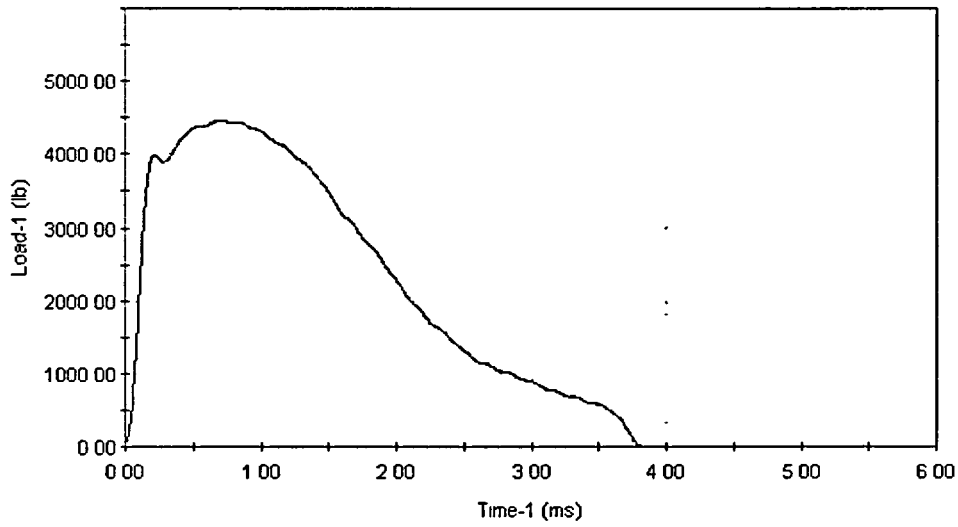
AH55, 0°F



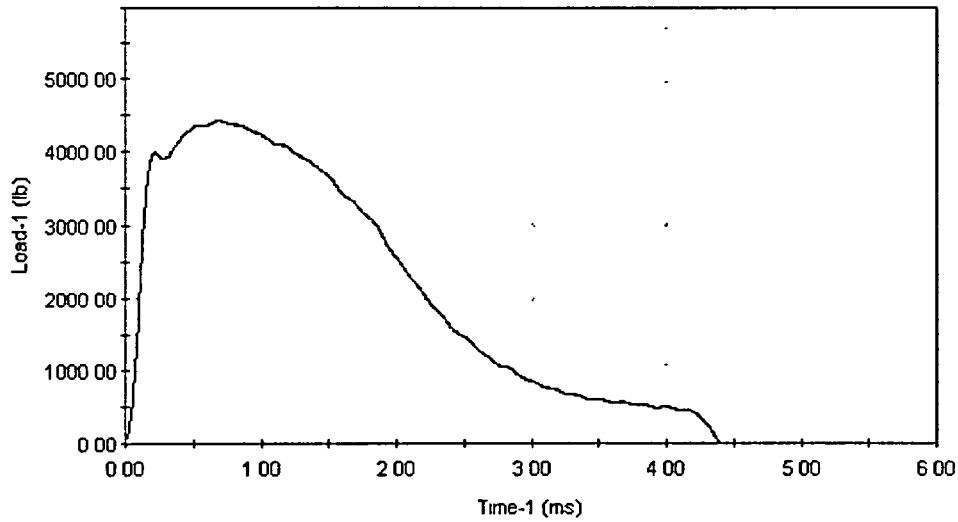
AH59, 25°F



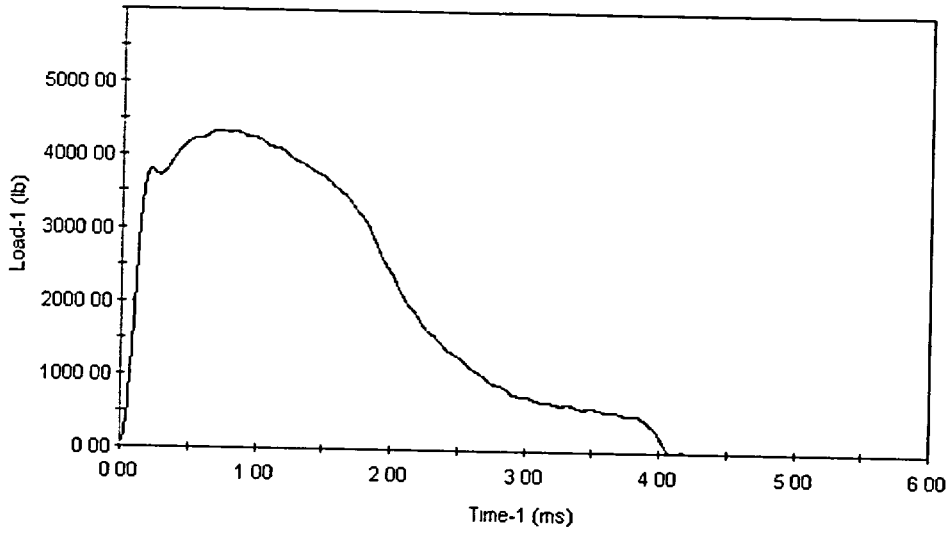
AH48, 35°F



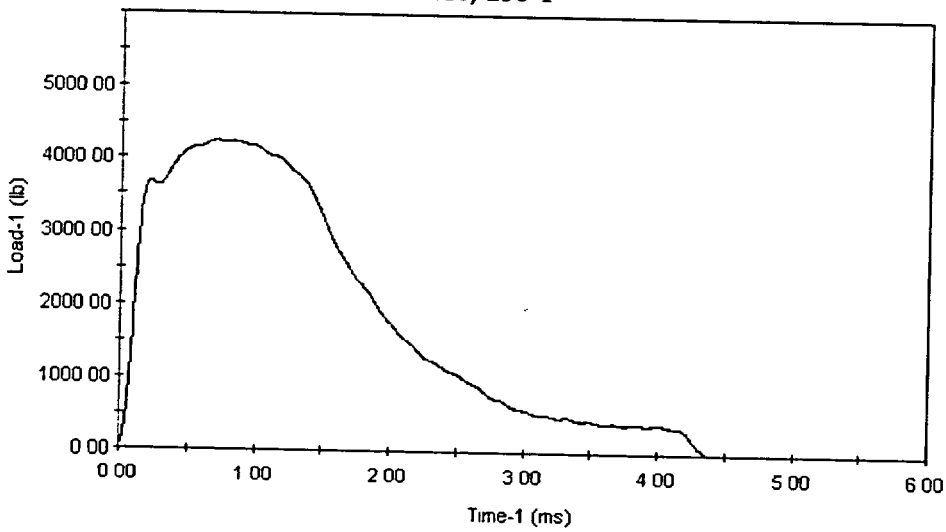
AH54, 50°F



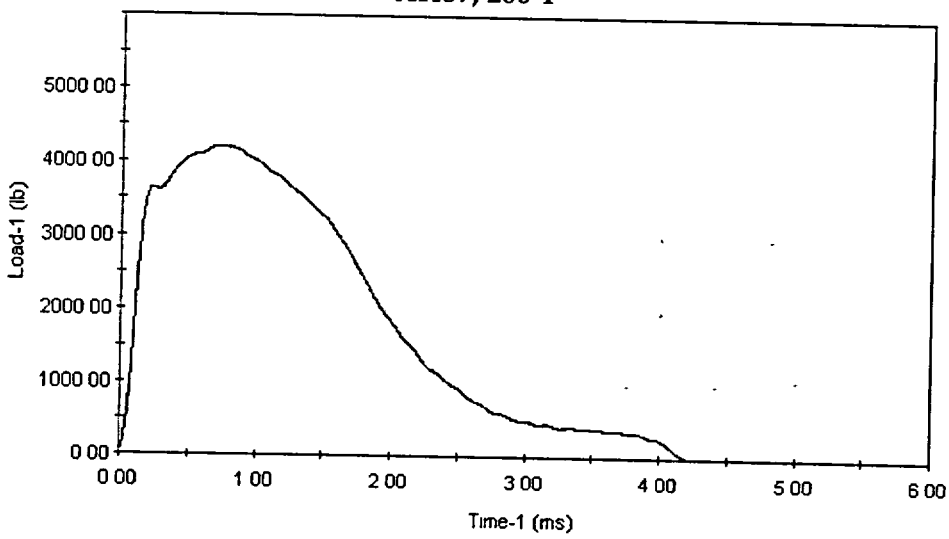
AH60, 100°F



AH46, 150°F



AH57, 200°F



AH56, 200°F

APPENDIX C

CHARPY V-NOTCH PLOTS FOR EACH CAPSULE USING SYMMETRIC HYPERBOLIC TANGENT CURVE-FITTING METHOD

Contained in Table C-1 are the upper shelf energy values used as input for the generation of the Charpy V-notch plots using CVGRAPH, Version 4.1. The definition for Upper Shelf Energy (USE) is given in ASTM E185-82, Section 4.18, and reads as follows:

> "*upper shelf energy level* – the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper shelf energy."

If there are specimens tested in set of three at each temperature Westinghouse reports the set having the highest average energy as the USE (usually unirradiated material). If the specimens were not tested in sets of three at each temperature Westinghouse reports the average of all 100% shear Charpy data as the USE. Hence, the USE values reported in Table C-1 and used to generate the Charpy V-notch curves were determined utilizing this methodology.

The lower shelf energy values were fixed at 2.2 ft-lb for all cases.

Table C-1 Upper Shelf Energy Values Fixed in CVGRAPH [ft-lb]

Material	Unirradiated	Capsule U	Capsule Y	Capsule V	Capsule X
Lower Shell Plate R2508-3 (Long.)	148	145	131	129	142
Lower Shell Plate R2508-3 (Trans.)	94	96	94	88	95
Weld Metal (heat # 90146)	100	92	94	89	93
HAZ Material	161	140	200	167	135

CAPSULE X (LONGITUDINAL ORIENTATION)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:13:31 on 12-12-2002

Page 1

Coefficients of Curve 1

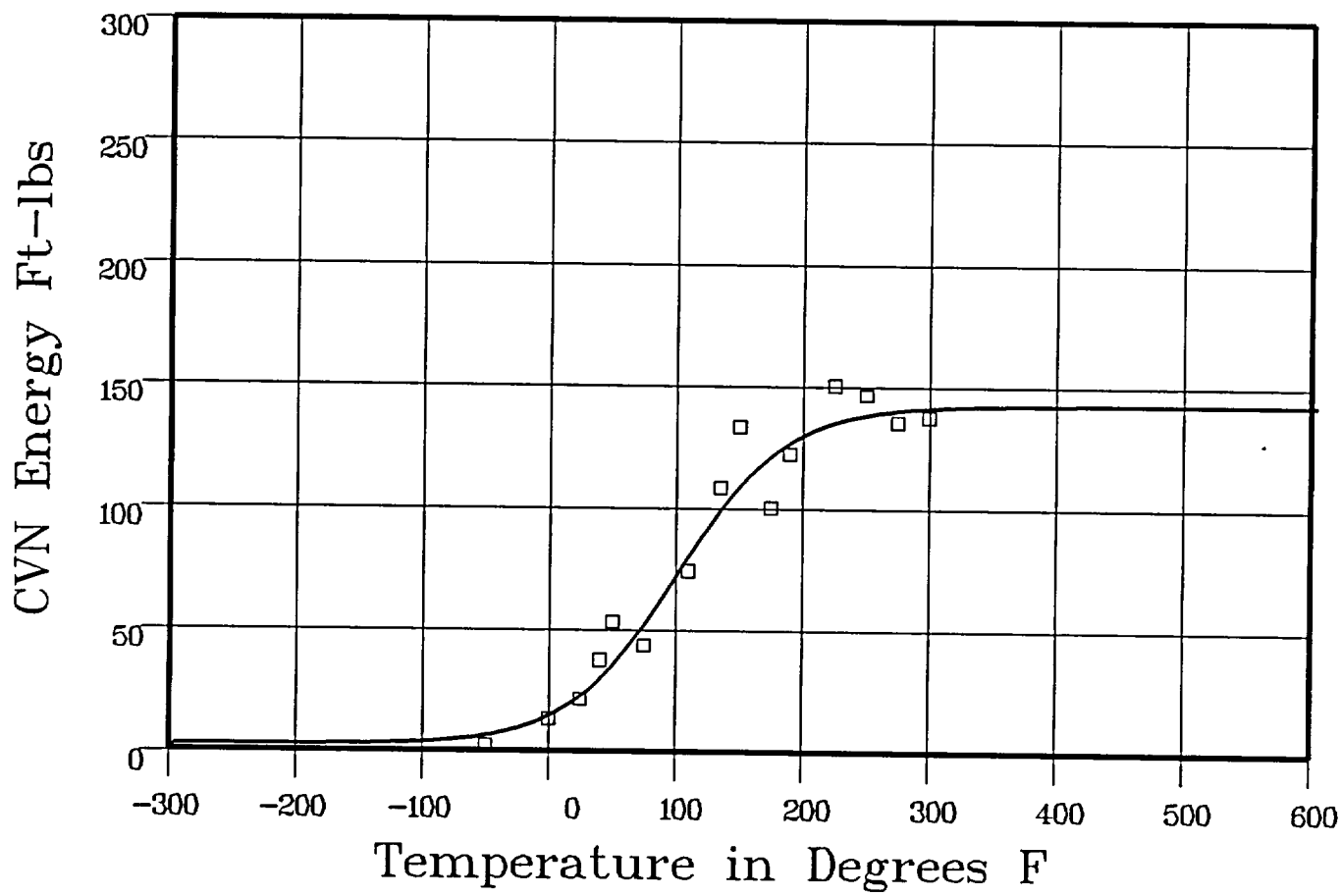
A = 72.09	B = 69.9	C = 85.66	T0 = 95.8

Equation is: CVN = A + B * [tanh((T - T0)/C)]

Upper Shelf Energy: 142 Fixed Temp. at 30 ft-lbs: 36.1 Temp. at 50 ft-lbs: 67.7 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: LT

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: PLATE SA533B1 Ori: LT Heat #: C4935-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-50	2	6.69	-4.69
0	13	15.69	-2.69
25	21	24.66	-3.66
40	37	32.07	4.92
50	53	37.92	15.07
75	43	55.45	-12.45
110	74	83.58	-9.58
135	108	102.02	5.97
150	133	111.23	21.76

**** Data continued on next page ****

CAPSULE X (LONGITUDINAL ORIENTATION)

Page 2

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: LT

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
175	100	122.98	-22.98
190	122	128.04	-6.04
225	150	135.47	14.52
250	146	138.28	7.71
275	135	139.9	-4.9
300	137	140.82	-3.82

SUM of RESIDUALS = -.86

CAPSULE X (LONGITUDINAL ORIENTATION)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:23:49 on 12-12-2002

Page 1

Coefficients of Curve 1

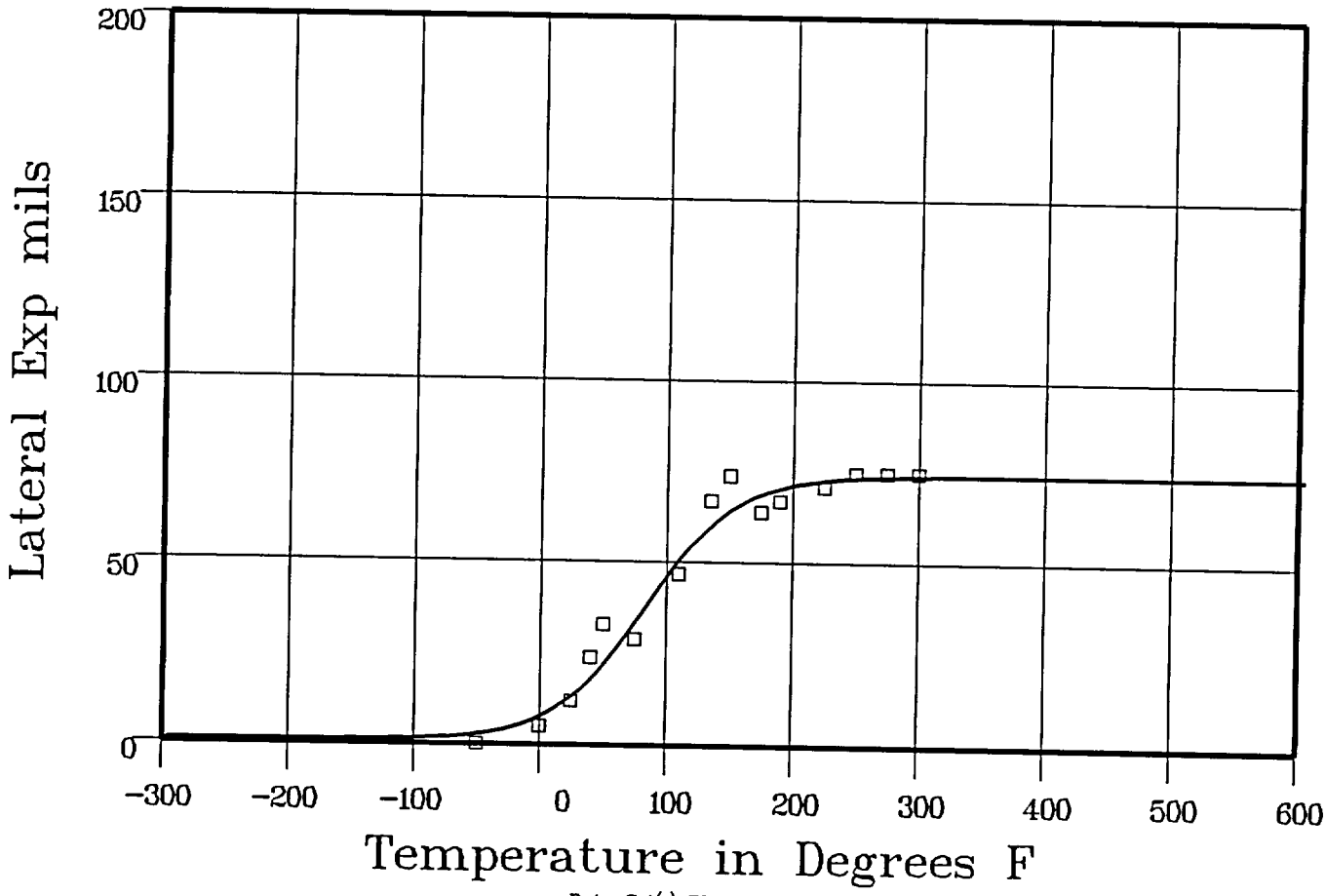
A = 37.66	B = 36.66	C = 72.74	T0 = 78.25

Equation is: L.E. = A + B * [tanh((T - T0)/C)]

Upper Shelf L.E.: 74.32 Temperature at L.E. 35: 72.9 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: LT

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap.: X Material: PLATE SA533B1 Ori.: LT Heat #: C4935-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-50	0	3.09	-3.09
0	5	8.64	-3.64
25	12	14.77	-2.77
40	24	19.98	4.01
50	33	24.09	8.9
75	29	36.02	-7.02
110	47	52.71	-5.71
135	67	61.59	5.4
150	74	65.37	8.62

**** Data continued on next page ****

CAPSULE X (LONGITUDINAL ORIENTATION)

Page 2

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: LT

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
175	64	69.52	-5.52
190	67	71.07	-4.07
225	71	73.04	-2.04
250	75	73.67	1.32
275	75	73.99	1
300	75	74.15	.84

SUM of RESIDUALS = -3.79

CAPSULE X (LONGITUDINAL ORIENTATION)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:25:58 on 12-12-2002

Page 1

Coefficients of Curve 1

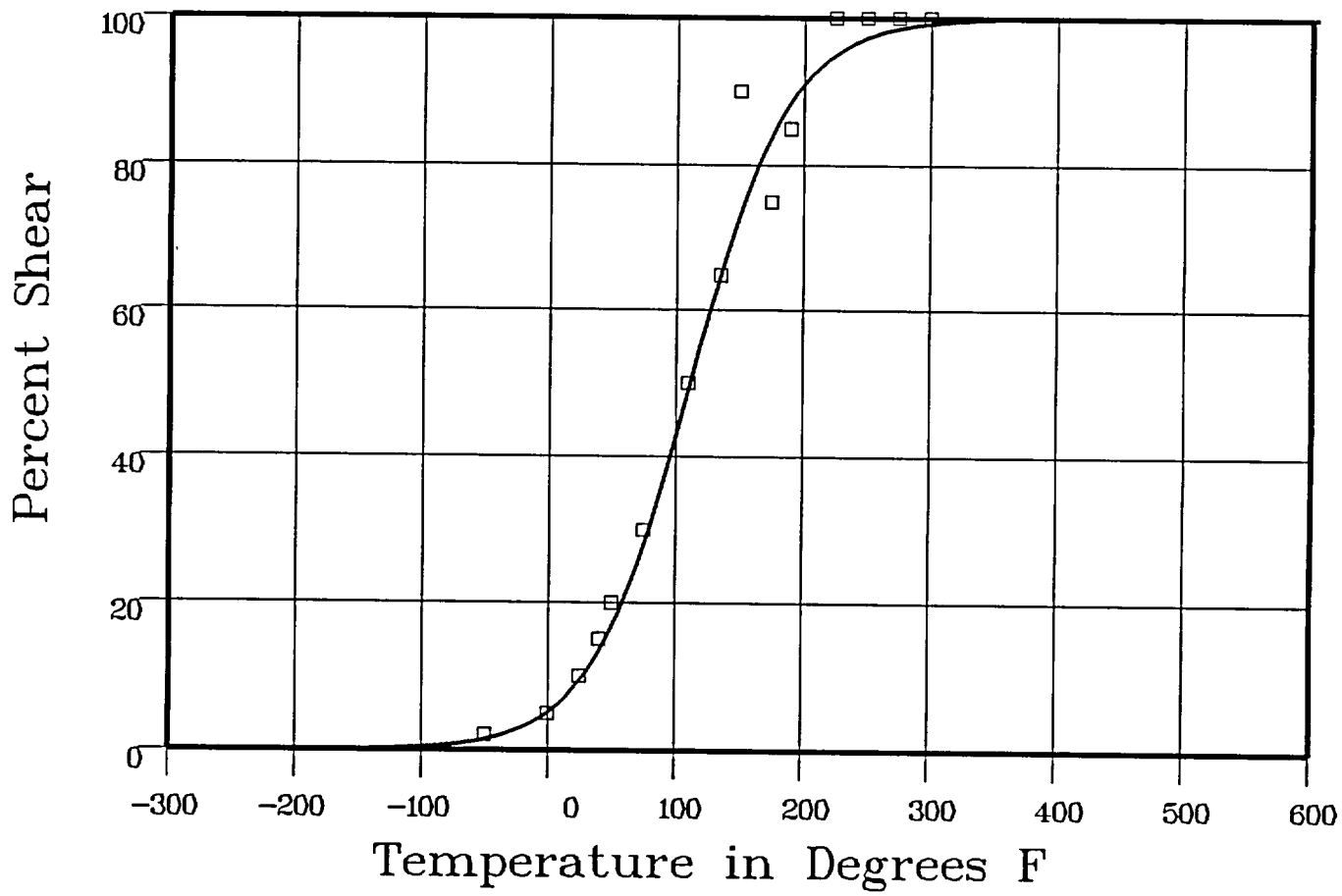
A = 50	B = 50	C = 76.71	T0 = 106.64

Equation is: Shear% = A + B * [tanh((T - T0)/C)]

Temperature at 50% Shear: 106.6

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: LT

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: PLATE SA533B1 Ori: LT Heat #: C4935-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-50	2	1.65	.34
0	5	5.84	-.84
25	10	10.63	-.63
40	15	14.96	.03
50	20	18.59	1.4
75	30	30.47	-.47
110	50	52.18	-2.18
135	65	67.68	-2.68
150	90	75.59	14.4

**** Data continued on next page ****

CAPSULE X (LONGITUDINAL ORIENTATION)

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: LT

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
175	75	85.59	-10.59
190	85	89.78	-4.78
225	100	95.62	4.37
250	100	97.67	2.32
275	100	98.77	1.22
300	100	99.35	.64

SUM of RESIDUALS = 2.55

CAPSULE X (TRANSVERSE ORIENTATION)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:13:45 on 01-10-2003

Page 1

Coefficients of Curve 1

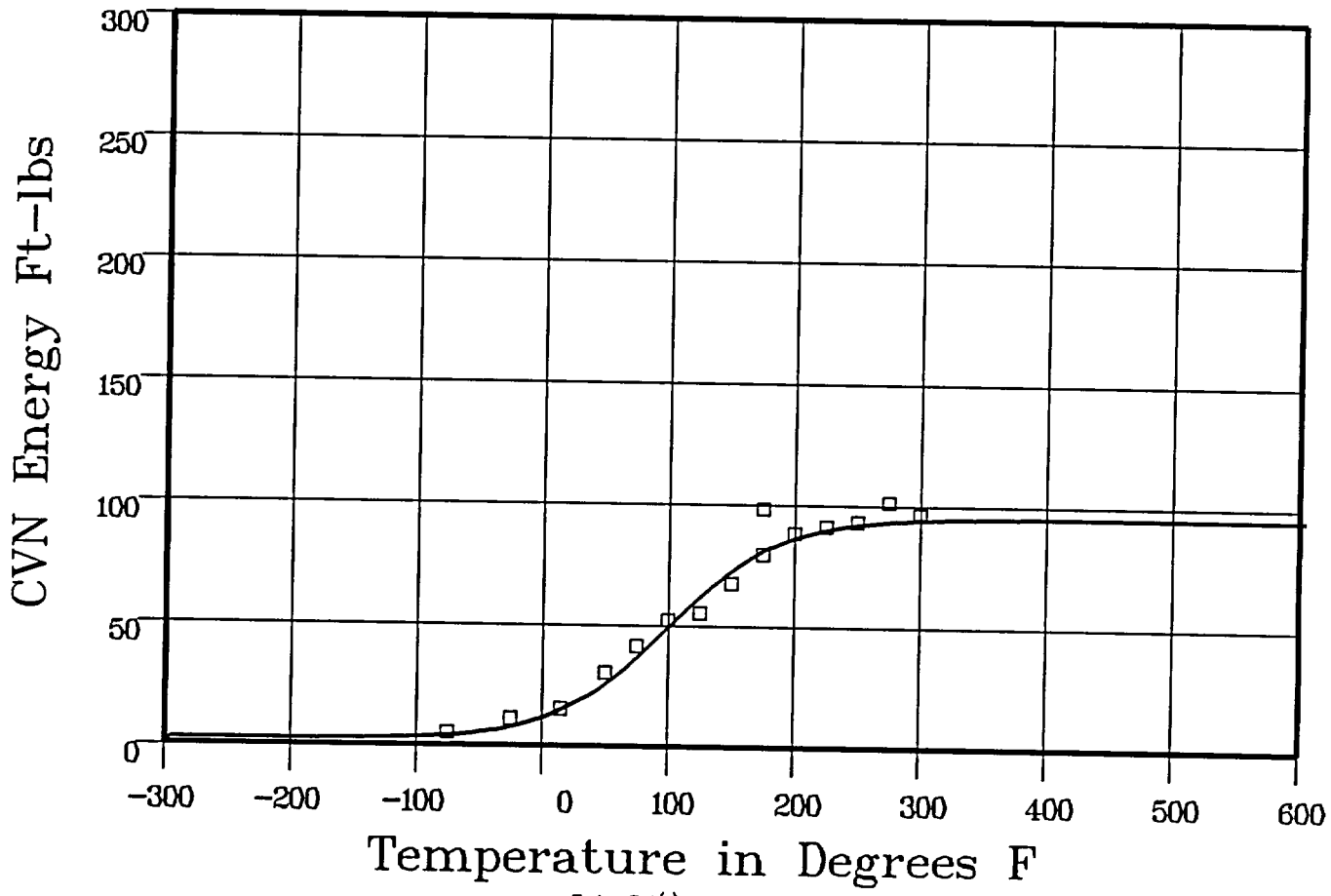
A = 48.59	B = 46.4	C = 90.28	T0 = 94.31

Equation is: CVN = A + B * [tanh((T - T0)/C)]

Upper Shelf Energy: 95 Fixed Temp. at 30 ft-lbs: 55.9 Temp. at 50 ft-lbs: 97 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: TL

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: PLATE SA533B1 Ori: TL Heat #: C4935-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-75	5	4.33	.66
-25	11	8.36	2.63
15	15	15.85	-.85
50	30	27.49	2.5
75	41	38.82	2.17
100	52	51.51	.48
125	55	63.78	-8.78
150	67	74.06	-7.06
175	79	81.69	-2.69

**** Data continued on next page ****

CAPSULE X (TRANSVERSE ORIENTATION)

Page 2

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: TL

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
175	98	81.69	16.3
200	88	86.85	1.14
225	91	90.13	.86
250	93	92.14	.85
275	101	93.33	7.66
300	96	94.03	1.96

SUM of RESIDUALS = 17.86

CAPSULE X (TRANSVERSE ORIENTATION)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:33:46 on 12-12-2002

Page 1

Coefficients of Curve 1

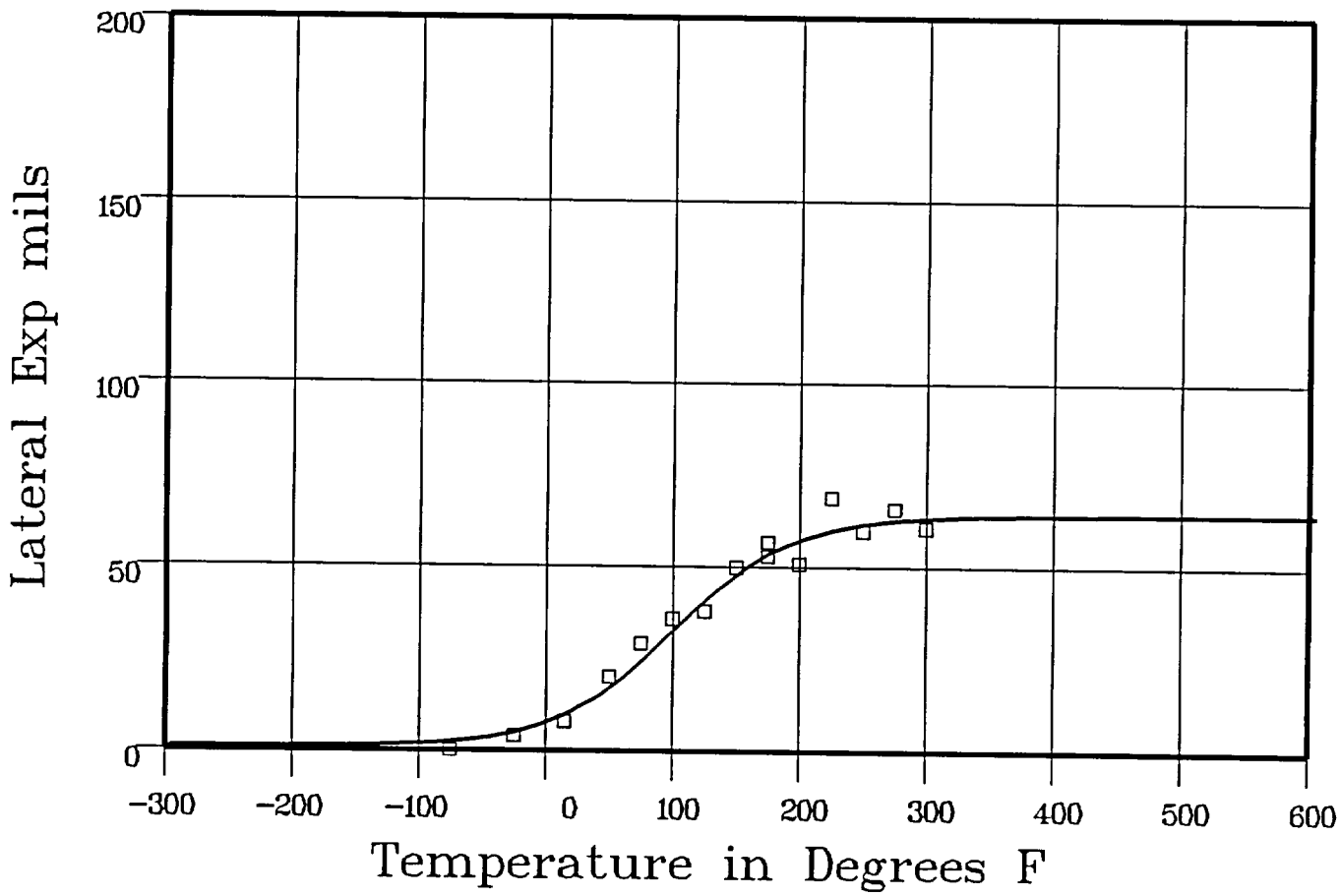
A = 32.72	B = 31.72	C = 94.2	T0 = 95.62

Equation is: L.E. = A + B * [tanh((T - T0)/C)]

Upper Shelf L.E.: 64.44 Temperature at L.E. 35: 102.4 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: TL

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: PLATE SA533B1 Ori.: TL Heat #: C4935-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-75	0	2.65	-2.65
-25	4	5.54	-1.54
15	8	10.7	-2.7
50	20	18.45	1.54
75	29	25.88	3.11
100	36	34.19	1.8
125	38	42.3	-4.3
150	50	49.23	.76
175	53	54.51	-1.51

**** Data continued on next page ****

CAPSULE X (TRANSVERSE ORIENTATION)

Page 2

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: TL

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
175	57	54.51	2.48
200	51	58.2	-7.2
225	69	60.61	8.38
250	60	62.13	-2.13
275	66	63.06	2.93
300	61	63.62	-2.62

SUM of RESIDUALS = -3.66

CAPSULE X (TRANSVERSE ORIENTATION)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:36:03 on 12-12-2002

Page 1

Coefficients of Curve 1

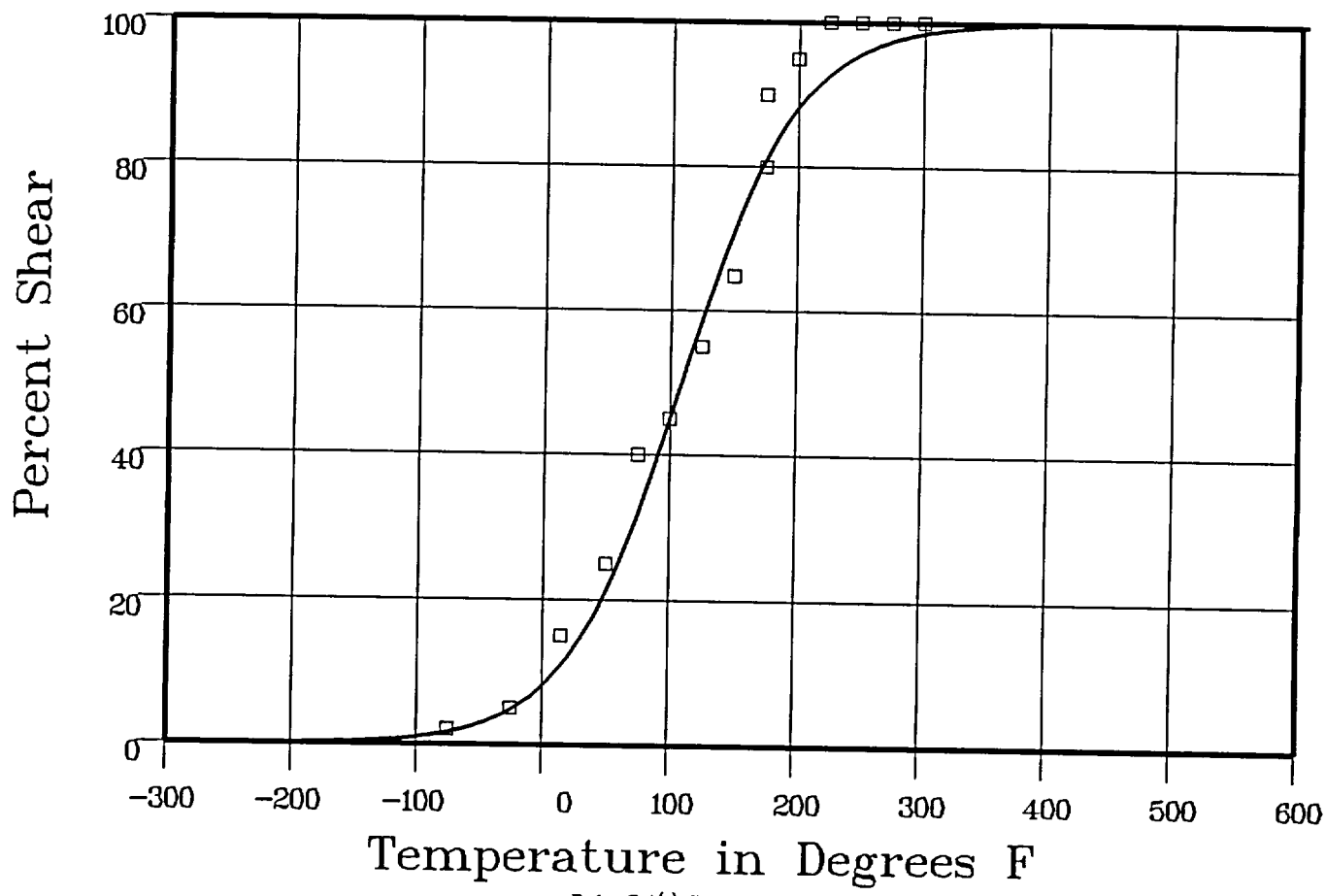
A = 50	B = 50	C = 90.18	T0 = 104.46

Equation is: Shear% = A + B * [tanh((T - T0)/C)]

Temperature at 50% Shear: 104.4

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: TL

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: PLATE SA533B1 Ori: TL Heat #: C4935-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-75	2	1.83	.16
-25	5	5.35	-.35
15	15	12.08	2.91
50	25	23	1.99
75	40	34.21	5.78
100	45	47.52	-2.52
125	55	61.18	-6.18
150	65	73.29	-8.29
175	80	82.69	-2.69

**** Data continued on next page ****

CAPSULE X (TRANSVERSE ORIENTATION)

Page 2

Material: PLATE SA533B1 Heat Number: C4935-2 Orientation: TL

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
175	90	82.69	7.3
200	95	89.26	5.73
225	100	93.54	6.45
250	100	96.18	3.81
275	100	97.77	2.22
300	100	98.7	1.29

SUM of RESIDUALS = 17.61

CAPSULE X (WELD)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:16:18 on 01-10-2003

Page 1

Coefficients of Curve 1

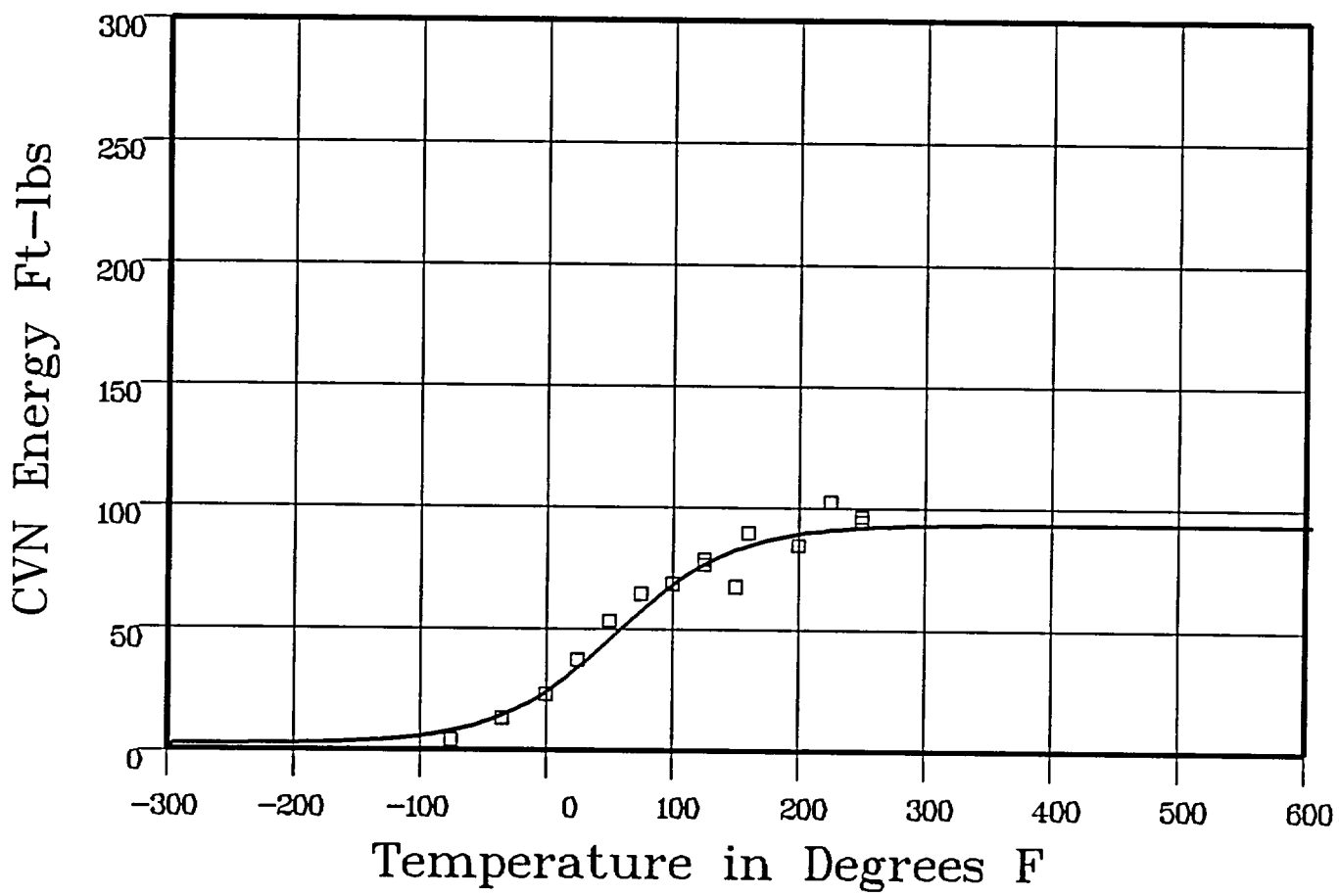
A = 47.59	B = 45.4	C = 95.38	T0 = 49.68

Equation is: CVN = A + B * [tanh((T - T0)/C)]

Upper Shelf Energy: 93 Fixed Temp. at 30 ft-lbs: 10.6 Temp. at 50 ft-lbs: 54.7 Lower Shelf Energy: 2.19 Fixed

Material: WELD Heat Number: WIRE HEAT N0.90146 Orientation:

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: WELD Ori: Heat #: WIRE HEAT N0.90146

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-75	4	8.39	-4.39
-35	13	15.35	-2.35
0	23	25.88	-2.88
25	37	36.1	.89
50	53	47.74	5.25
75	64	59.37	4.62
100	68	69.54	-1.54
125	78	77.47	.52
125	76	77.47	-1.47

**** Data continued on next page ****

CAPSULE X (WELD)

Page 2

Material: WELD Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
150	67	83.12	-16.12
160	89	84.82	4.17
200	84	89.27	-5.27
225	102	90.75	11.24
250	96	91.65	4.34
250	94	91.65	2.34

SUM of RESIDUALS = -.65

CAPSULE X (WELD)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:18:50 on 01-10-2003

Page 1

Coefficients of Curve 1

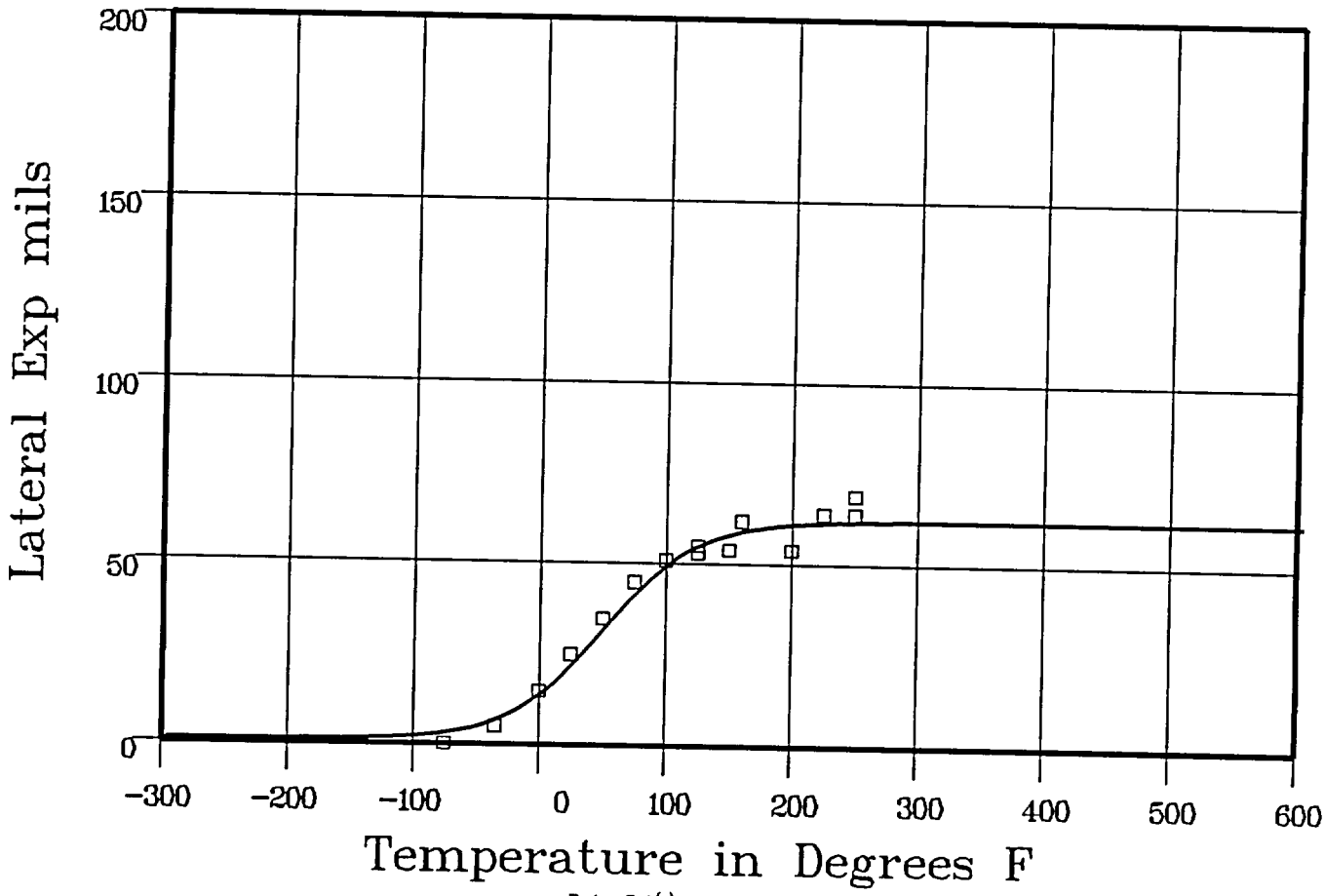
A = 31.67	B = 30.67	C = 75.77	T0 = 45

Equation is: L.E. = A + B * [tanh((T - T0)/C)]

Upper Shelf L.E.: 62.34 Temperature at L.E. 35: 53.2 Lower Shelf L.E.: 1 Fixed

Material: WELD Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: WELD Ori: Heat #: WIRE HEAT NO.90146

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-75	0	3.47	-3.47
-35	5	7.62	-2.62
0	15	15.33	-.33
25	25	23.76	1.23
50	35	33.69	1.3
75	45	43.22	1.77
100	51	50.7	.29
125	53	55.72	-2.72
125	55	55.72	-.72

**** Data continued on next page ****

CAPSULE X (WELD)

Page 2

Material: WELD Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
150	54	58.73	-4.73
160	62	59.53	2.46
200	54	61.34	-7.34
225	64	61.82	2.17
250	64	62.07	1.92
250	69	62.07	6.92

SUM of RESIDUALS = -3.86

CAPSULE X (WELD)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:20:43 on 01-10-2003

Page 1

Coefficients of Curve 1

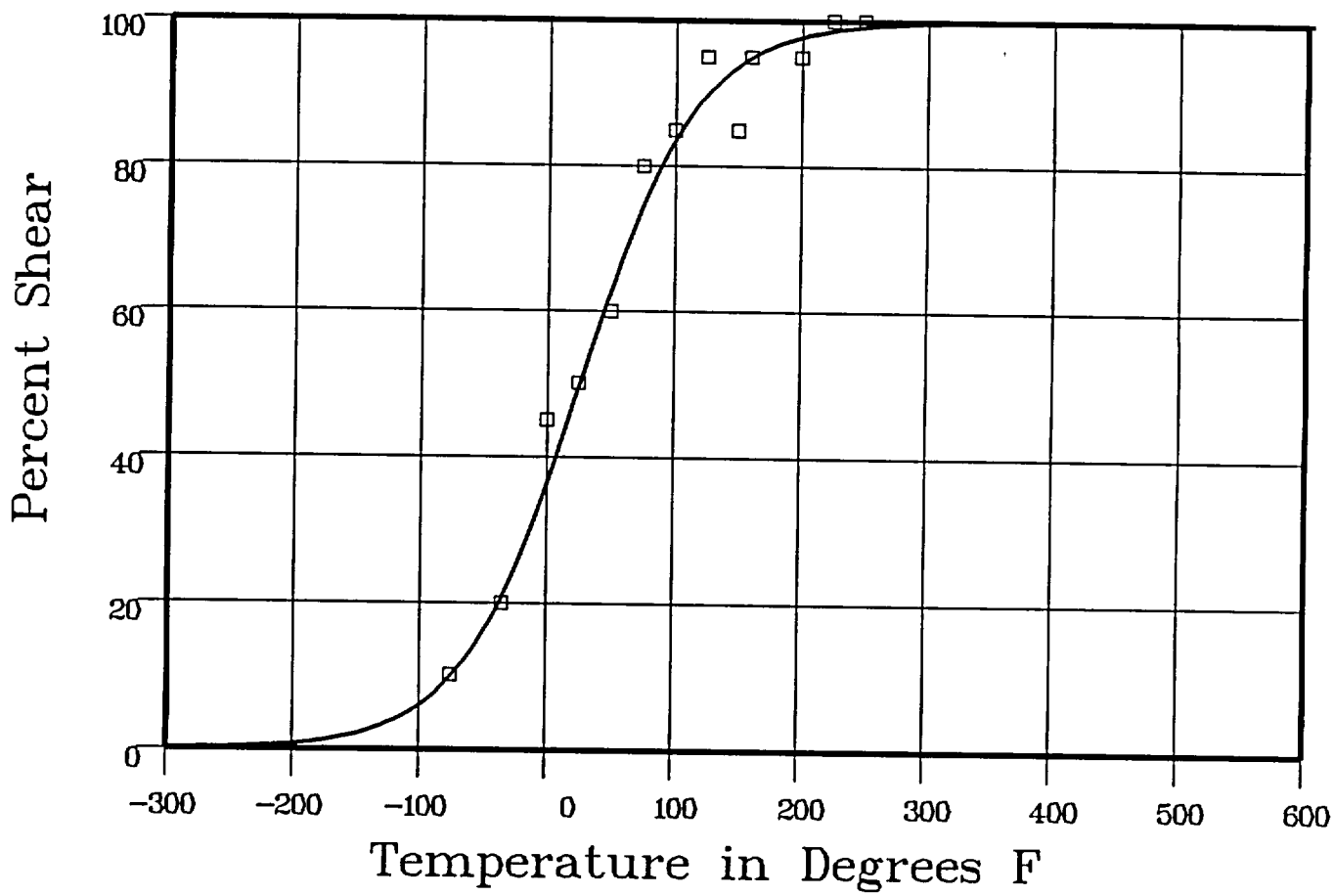
A = 50	B = 50	C = 91.24	T0 = 21.09

Equation is: Shear% = A + B * [tanh((T - T0)/C)]

Temperature at 50% Shear: 21

Material: WELD Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: WELD Ori: Heat #: WIRE HEAT NO.90146

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-75	10	10.84	-.84
-35	20	22.62	-2.62
0	45	38.64	6.35
25	50	52.13	-2.13
50	60	65.33	-5.33
75	80	76.52	3.47
100	85	84.93	.06
125	95	90.69	4.3
125	95	90.69	4.3

**** Data continued on next page ****

CAPSULE X (WELD)

Page 2

Material: WELD Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
150	85	94.4	-9.4
160	95	95.45	-.45
200	95	98.05	-3.05
225	100	98.86	1.13
250	100	99.34	.65
250	100	99.34	.65

SUM of RESIDUALS = -2.91

CAPSULE X (HAZ)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:32:26 on 12-12-2002

Page 1

Coefficients of Curve 1

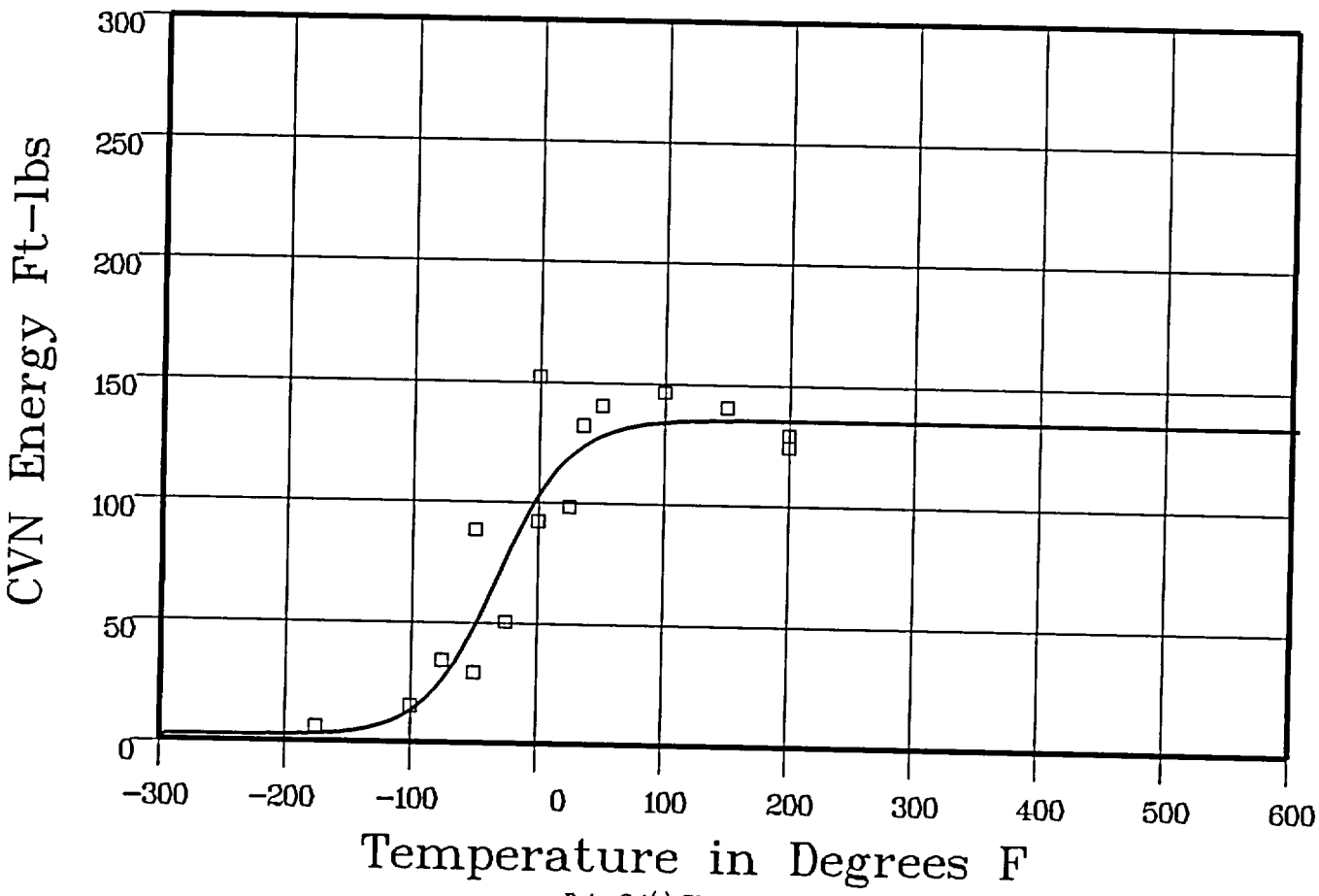
A = 68.59	B = 66.4	C = 56.86	T0 = -36.56

Equation is: CVN = A + B * [tanh((T - T0)/C)]

Upper Shelf Energy: 135 Fixed Temp. at 30 ft-lbs: -74.3 Temp. at 50 ft-lbs: -52.9 Lower Shelf Energy: 2.19 Fixed

Material: HEAT AFF'D ZONE Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: HEAT AFF'D ZONE Ori: Heat #: WIRE HEAT NO.90146

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-175	6	3.21	2.78
-100	15	15.07	-.07
-75	34	29.49	4.5
-50	29	53.19	-24.19
-50	88	53.19	34.8
-25	50	81.91	-31.91
0	92	106.24	-14.24
0	152	106.24	45.75
25	98	121.33	-23.33

**** Data continued on next page ****

CAPSULE X (HAZ)

Page 2

Material: HEAT AFF'D ZONE Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
35	132	125.08	6.91
50	140	128.96	11.03
100	146	133.91	12.08
150	140	134.81	5.18
200	124	134.96	-10.96
200	129	134.96	-5.96

SUM of RESIDUALS = 12.36

CAPSULE X (HAZ)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:33:28 on 12-12-2002

Page 1

Coefficients of Curve 1

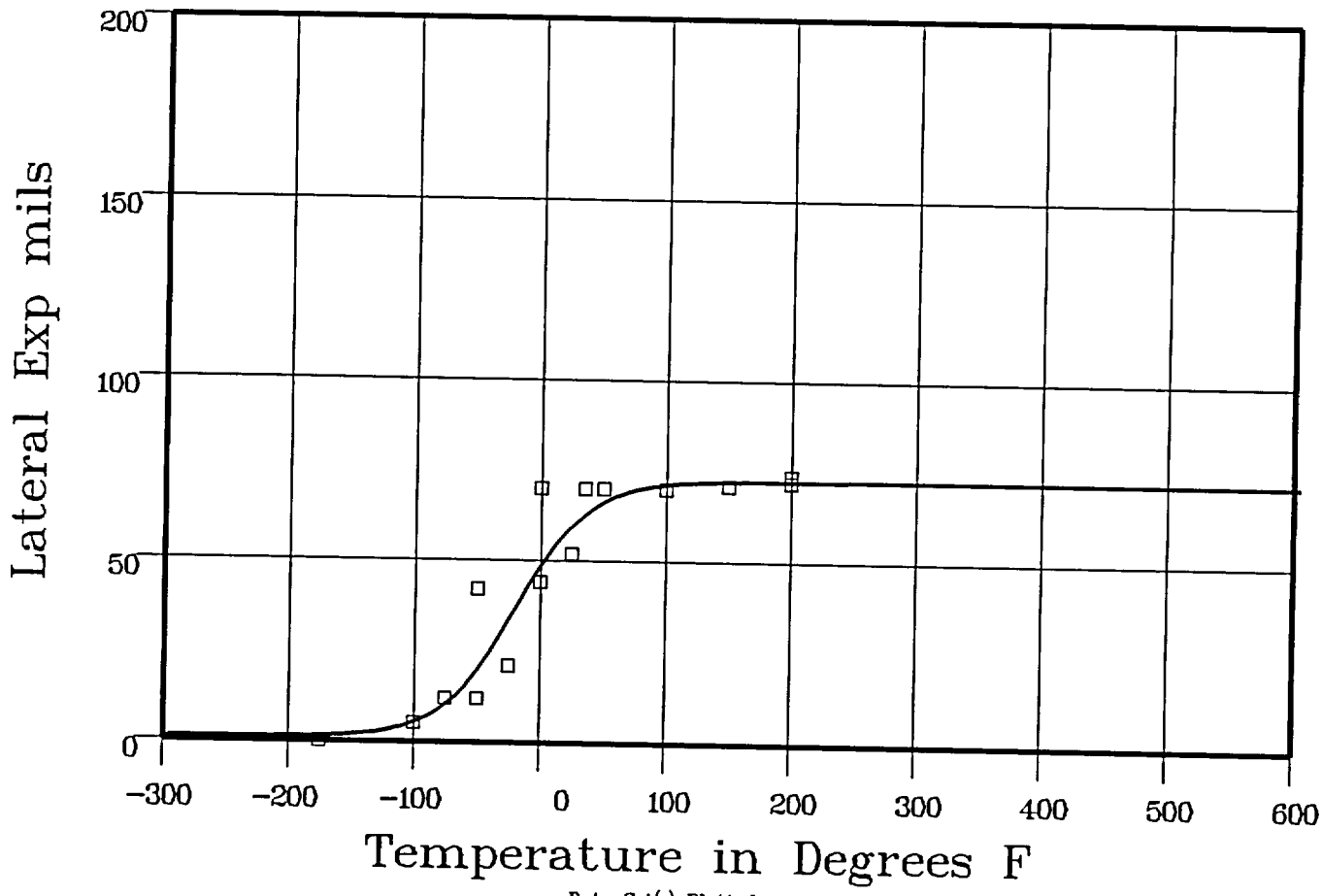
A = 36.74	B = 35.74	C = 58.51	T0 = -24.75

Equation is: L.E. = A + B * [tanh((T - T0)/C)]

Upper Shelf L.E: 72.49 Temperature at L.E. 35: -27.6 Lower Shelf L.E: 1 Fixed

Material: HEAT AFF'D ZONE Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap.: X Material: HEAT AFF'D ZONE Ori.: Heat #: WIRE HEAT NO.90146

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-175	0	1.41	-1.41
-100	5	6.07	-1.07
-75	12	11.88	.11
-50	12	22.21	-10.21
-50	42	22.21	19.78
-25	21	36.6	-15.6
0	44	51.03	-7.03
0	70	51.03	18.96
25	52	61.45	-9.45

**** Data continued on next page ****

CAPSULE X (HAZ)

Page 2

Material: HEAT AFFD ZONE Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:

Charpy V–Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
35	70	64.28	5.71
50	70	67.34	2.65
100	70	71.5	-1.5
150	71	72.31	-1.31
200	72	72.46	-.46
200	74	72.46	1.53

SUM of RESIDUALS = .67

CAPSULE X (HAZ)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:34:31 on 12-12-2002

Page 1

Coefficients of Curve 1

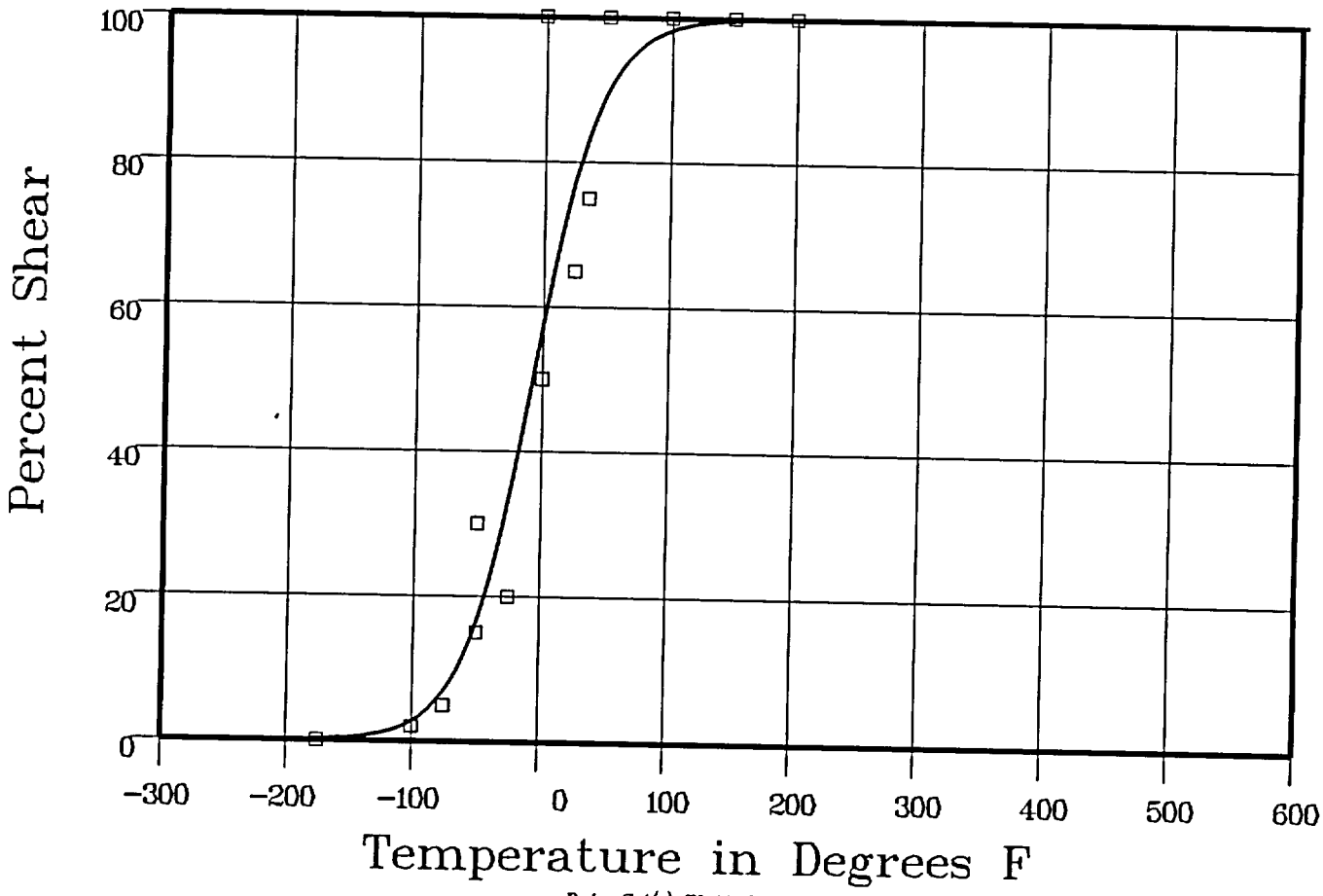
A = 50	B = 50	C = 52.52	T0 = -12.18

Equation is: Shear% = A + B * [tanh((T - T0)/C)]

Temperature at 50% Shear: -12.1

Material: HEAT AFF'D ZONE Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:



Data Set(s) Plotted

Plant: WC1 Cap: X Material: HEAT AFF'D ZONE Ori: Heat #: WIRE HEAT NO.90146

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-175	0	2	-2
-100	2	3.4	-1.4
-75	5	8.37	-3.37
-50	15	19.15	-4.15
-50	30	19.15	10.84
-25	20	38.03	-18.03
0	50	61.39	-11.39
0	100	61.39	38.6
25	65	80.47	-15.47

**** Data continued on next page ****

CAPSULE X (HAZ)

Page 2

Material: HEAT AFF'D ZONE Heat Number: WIRE HEAT NO.90146 Orientation:

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
35	75	85.77	-10.77
50	100	91.43	8.56
100	100	98.62	1.37
150	100	99.79	.2
200	100	99.96	.03
200	100	99.96	.03

SUM of RESIDUALS = -5.18

APPENDIX D

WOLF CREEK SURVEILLANCE PROGRAM CREDIBILITY EVALUATION

INTRODUCTION:

Regulatory Guide 1.99, Revision 2, describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels Position C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there has been four surveillance capsules removed from the Wolf Creek reactor vessel. To use these surveillance data sets, they must be shown to be credible. In accordance with the discussion of Regulatory Guide 1.99, Revision 2, there are five requirements that must be met for the surveillance data to be judged credible.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the Indian Point Unit 2 reactor vessel surveillance data and determine if the Indian Point Unit 2 surveillance data is credible.

EVALUATION:

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture

Toughness Requirements", as follows:

> "the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

The Wolf Creek reactor vessel consists of the following beltline region materials:

- Intermediate Shell Plates R2005-1, 2, 3
- Lower Shell Plates R2508-1, 2, 3
- Intermediate & Lower Shell Longitudinal Weld Seams (Heat # 90146),
- Intermediate to Lower Shell Circumferential Weld Seam (Heat # 90146).

Per WCAP-10015, the Wolf Creek surveillance program was based on ASTM E185-79. When the surveillance program material was selected it was believed that copper and phosphorus were elements most important to embrittlement of the reactor vessel steels. Lower shell plate R2508-3 had the highest initial RT_{NDT} and the lowest USE of all plate materials in the beltline region. In addition, lower shell plate R2508-3 had approximately the same copper and phosphorus content as the other beltline plate materials. Therefore, based on the highest initial RT_{NDT} and the lowest USE, lower shell plate was chosen for the surveillance program.

The weld material in the Wolf Creek surveillance program was made of the same wire as all the reactor vessel beltline welds, thus it was chosen as the surveillance weld material.

Hence, Criterion 1 is met for the Wolf Creek reactor vessel.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper shelf energy unambiguously.

Based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper shelf energy of the Wolf Creek surveillance materials unambiguously. Hence, the Wolf Creek surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for the plate.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2 In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to industry at a meeting held by the NRC on February 12 and 13, 1998. At this meeting the NRC presented five cases. Of the five cases Case 1 ("Surveillance data available from plant but no other source") most closely represents the situation listed above for Wolf Creek surveillance weld metal. Note, for the plate materials, the straight forward method of Regulatory Guide 1.99, Revision 2 will be followed.

TABLE D-1

Calculation of Chemistry Factors using Wolf Creek Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Plate R2508-3 (Longitudinal)	U	0.316	0.684	36.46	24.94	0 468
	Y	1.19	1.05	16.03	16.83	1.10
	V	2.22	1.22	52.03	63.48	1.49
	X	3.49	1.33	61.06	81.21	1.77
Lower Shell Plate R2508-3 (Transverse)	U	0 316	0 684	23 79	16.27	0 468
	Y	1.19	1.05	35.39	37.16	1.10
	V	2.22	1.22	54.53	66.53	1.49
	X	3.49	1.33	53.96	71.77	1.77
				SUM:	378.19	9.656
	$CF_{R2508-3} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (378\ 19) \div (9.656) =$ **39.1°F**					
Surveillance Weld Material	U	0.316	0 684	27.21	18 612	0.468
	Y	1.19	1.05	45.09	47.34	1.10
	V	2.22	1.22	46.3	56.49	1.49
	X	3.49	1.33	68.36	90.92	1.77
				SUM:	213.362	4.828
	$CF_{Surv\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (213.362) \div (4.828) =$ **44.1°F**					

Notes:

(a) f = fluence Calculated fluence from Section 6 of this report [x 10^{19} n/cm^2, E > 1.0 MeV]

(b) FF = fluence factor = $f^{(0\ 28 - 0\ 1 * \log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from Figures 5-1, 5-4 and 5.7, herein [°F]

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-2.

Table D-2:
Wolf Creek Surveillance Capsule Data Scatter about the Best-Fit Line for Surveillance Forging Materials.

Material	Capsule	CF ($Slope_{best\ fit}$)	FF	Measured ΔRT_{NDT}	Predicted ΔRT_{NDT}	Scatter ΔRT_{NDT} (°F)	<17°F (Base Metals) <28°F (Weld)
Lower Shell Plate R2508-3 (Longitudinal)	U	39.1	0.684	36.46	26.74	9.72	Yes
	Y	39.1	1.05	16.03	41.06	-25.03	No
	V	39.1	1.22	52.03	47.70	4.33	Yes
	X	39.1	1.33	61.06	52.00	9 06	Yes
Lower Shell Plate R2508-3 (Transverse)	U	39.1	0.684	23.79	26.74	-2.95	Yes
	Y	39.1	1.05	35.39	41.06	-5.67	Yes
	V	39.1	1.22	54.53	47.70	6.83	Yes
	X	39.1	1.33	53.96	52.00	1.96	Yes
Vessel Beltline Welds (Heat # 90146)	U	44.1	0.684	27.21	30.16	-2.95	Yes
	Y	44.1	1.05	45.09	46.31	-1.22	Yes
	V	44.1	1.22	46.3	53.80	-7.5	Yes
	X	44.1	1.33	68.36	58.65	9.71	Yes

Table D-2 indicates that only one data point falls outside the +/- 1σ of 17°F scatter band for the lower shell plate R2508-3 surveillance data. One out of 8 data point is still consider credible. No weld data points fall outside the +/- 1σ of 28°F scatter band for the surveillance weld data, therefore the weld data is also credible per the third criterion.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the neutron pad and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pad The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criterion is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Wolf Creek surveillance program does not contain correlation monitor material. Therefore, this criterion is not applicable to the Wolf Creek surveillance program.

CONCLUSION:

Based on the preceding responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B and 10 CFR 50.61, the Wolf Creek surveillance plate and weld data is credible.