ATTACHMENT 1

WCAP-16001, "ANALYSIS OF CAPSULE Y FROM DOMINION SURRY UNIT 2 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM"

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Analysis of Capsule Y from Dominion Surry Unit 2 Reactor Vessel Radiation Surveillance Program

Westinghouse

WCAP-16001, Revision 0

Analysis of Capsule Y from Dominion Surry Unit 2 Reactor Vessel Radiation Surveillance Program

T.J. Laubham
J. Conermann
S.L. Anderson

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Approved: _____
J.A. Gresham, Manager
Engineering & Materials Technology

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

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PREFACE

This report has been technically reviewed and verified by:

Reviewer:

Sections 1 through 5, 7, 8, Appendices A, B, and C	A.R. Rawluszki	
Section 6 and Appendix D	G.K. Roberts	

EXECUTIVE SUMMARY

The purpose of this report is to document the results of the testing of surveillance Capsule Y from Surry Unit 2. Capsule Y was removed at 20.3 EFPY and post irradiation mechanical tests of the Charpy V-notch and tensile specimens was performed. A fluence evaluation utilizing the recently released neutron transport and dosimetry cross-section libraries derived from the ENDF/B-VI data-base. Capsule Y received a fluence of 2.73 x 10^{19} n/cm^2 after irradiation to 20.3 EFPY. The peak clad/base metal interface vessel fluence after 20.3 EFPY of plant operation was 2.15 x 10^{19} 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 C4339-1 contained in capsule Y (longitudinal & transverse) is greater than the Regulatory Guide 1.99, Revision 2[1], predictions. However, the shift value is less than two sigma allowance by Regulatory Guide 1.99, Revision 2. 2) The measured 30 ft-lb shift in transition temperature values of the weld metal contained in capsule Y is less than the Regulatory Guide 1.99, Revision 2, predictions. 3) The measured percent decrease in upper shelf energy for all the surveillance materials of Capsules Y contained in the Surry Unit 2 surveillance program are less than the Regulatory Guide 1.99, Revision 2 predictions 4) All beltline materials, except the intermediate to lower shell girth weld, 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 life of the vessel (30.1 EFPY) as required by 10CFR50, Appendix G [2]. The low USE for the intermediate to lower shell girth weld is justified under the Equivalent Margin Analysis (EMA) in BAW-2192PA & BAW-2178PA.

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 Y, the fifth capsule removed from the Surry Unit 2 reactor pressure vessel, led to the following conclusions.

- The Charpy V-notch data presented in WCAP-8085[3], BMI-0975-/SU2[4] and WCAP-11499[5] were based on Charpy curves using a hyperbolic tangent curve-fitting routine. The results presented in this report are based on a re-plot of all capsule data using CVGRAPH, Version 4.1, which is a symmetric hyperbolic tangent curve-fitting program. Appendix B presents a comparison of the Charpy V-Notch test results for each capsule based on previous fit vs. symmetric hyperbolic tangent fit. Appendix C presents the CVGRAPH, Version 4.1, Charpy V-notch plots and the program input data.

- Capsule Y received an average fast neutron fluence (E> 1.0 MeV) of 2.73×10^{19} n/cm^2 after 20 3 effective full power years (EFPY) of plant operation.

- Irradiation of the reactor vessel lower shell plate C4339-1 (heat number 4339) 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 113.11°F and an irradiated 50 ft-lb transition temperature of 157.64°F. This results in a 30 ft-lb transition temperature increase of 114.22°F and a 50 ft-lb transition temperature increase of 124.26°F for the longitudinal oriented specimens. See Table 5-11.

- Irradiation of the reactor vessel lower shell plate C4339-1 (heat number 4339) 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 118.27°F and an irradiated 50 ft-lb transition temperature of 174.08°F. This results in a 30 ft-lb transition temperature increase of 106.81°F and a 50 ft-lb transition temperature increase of 122 18°F for the longitudinal oriented specimens. See Table 5-11.

- Irradiation of the weld metal (heat number 0227) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 157.84°F and an irradiated 50 ft-lb transition temperature of 247.25°F. This results in a 30 ft-lb transition temperature increase of 178.32°F and a 50 ft-lb transition temperature increase of 224.76°F. See Table 5-11.

- Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 75.07°F and an irradiated 50 ft-lb transition temperature of 144.55°F. This results in a 30 ft-lb transition temperature increase of 127.91°F and a 50 ft-lb transition temperature increase of 150.28°F. See Table 5-11.

- Irradiation of the Correlation Monitor Material Plate HSST02 Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 194.47°F and an irradiated 50 ft-lb transition temperature of 232.43°F. This results in a 30 ft-lb transition temperature increase of 148.02°F and a 50 ft-lb transition temperature increase of 154.12°F. See Table 5-11.

- The average upper shelf energy of the lower shell plate C4339-1 (longitudinal orientation) resulted in an average energy decrease of 17 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 111 ft-lb for the longitudinal oriented specimens. See Table 5-11.

- The average upper shelf energy of the lower shell plate C4339-1 (transverse orientation) resulted in an average energy decrease of 10 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 94 ft-lb for the longitudinal oriented specimens. See Table 5-11.

- The average upper shelf energy of the weld metal Charpy specimens resulted in an average energy decrease of 33 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 58 ft-lb for the weld metal specimens See Table 5-11.

- The average upper shelf energy of the weld HAZ metal Charpy specimens resulted in an average energy decrease of 22 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 94 ft-lb for the weld HAZ metal. See Table 5-11.

- The average upper shelf energy of the Correlation Monitor Material Plate HSST02 Charpy specimens resulted in an average energy decrease of 23 ft-lb after irradiation This results in an irradiated average upper shelf energy of 100 ft-lb for the weld correlation monitor metal. See Table 5-11.

- A comparison, as presented in Table 5-12, of the Surry Unit 2 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 C4339-1 contained in capsule Y (longitudinal & transverse) is greater than the Regulatory Guide 1.99, Revision 2, predictions. However, the shift value is within two sigma of the predicted value.

  — The measured 30 ft-lb shift in transition temperature values of the weld metal contained in capsule Y is less than the Regulatory Guide 1.99, Revision 2, predictions.

  — The measured percent decrease in upper shelf energy for all the surveillance materials of Capsules Y contained in the Surry Unit 2 surveillance program are less than the Regulatory Guide 1 99, Revision 2 predictions

- All beltline materials, except the intermediate to lower shell girth weld, 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 life of the vessel (30.1 EFPY) as required by 10CFR50, Appendix G [2]. The low USE for the intermediate to lower shell girth weld is justified under the Equivalent Margin Analysis (EMA) in BAW-2192PA & BAW-2178PA.

- The calculated end-of-license (30.1 EFPY) neutron fluence (E> 1.0 MeV) at the core midplane for the Surry Unit 2 reactor vessel using the Regulatory Guide 1.99, Revision 2 attenuation formula (i.e., Equation #3 in the guide) are as follows:

Calculated: Vessel inner radius* = 2.93×10^{19} n/cm^2

Vessel 1/4 thickness = 1.80×10^{18} n/cm^2

Vessel 3/4 thickness = 6.84×10^{18} n/cm^2

*Clad/base metal interface. Value was interpolated between 20.3 and 32 EFPY (From Table 6-2)

2 INTRODUCTION

This report presents the results of the examination of Capsule Y, the fifth capsule removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Dominion Generation Surry Unit 2 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Dominion Generation Surry Unit 2 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-8085, "Virginia Electric & Power Co. Surry Unit No. 2 Reactor Vessel Radiation Surveillance Program"[3]. The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-70, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." Capsule Y was removed from the reactor after 20.3 EFPY of exposure and shipped to the Westinghouse Science and Technology Center Hot Cell Facility, where the post-irradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

The Charpy V-notch data presented in WCAP-8085[3], BMI-0975-/SU2[4] and WCAP-11499[5] were based on a hand-fit using engineering judgment. The results presented in this report are based on a re-plot of all capsule data using CVGRAPH, Version 4.1, which is a symmetric hyperbolic tangent curve-fitting program. Appendix B presents a comparison of the Charpy V-Notch test results of previous curve fits vs the symmetric hyperbolic tangent fit. Appendix C presents the CVGRAPH, Version 4.1, Charpy V-notch plots and the program input 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 Surry Unit 2 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 [16]. 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[15]) 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[16]. 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 Surry Unit 2 reactor vessel radiation surveillance program[3], 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

Eight surveillance capsules for monitoring the effects of neutron exposure on the Surry Unit 2 reactor pressure vessel core region (beltline) materials were inserted in the reactor vessel prior to initial plant start-up. The eight capsules were positioned in the reactor vessel between the thermal shield 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 Y was removed after 20.3 effective full power years (EFPY) of plant operation This capsule contained Charpy V-notch, tensile, and Wedge Opening Loading (WOL) specimens made from lower shell plate C4339-1 (heat number 4339) and submerged arc weld metal fabricated with the same weld wire heat 0227 as used in the reactor vessel intermediate to lower shell circumferential weld seams. In addition, this capsule contained Charpy V-notch specimens from the weld Heat-Affected-Zone (HAZ) metal of plate C4339-1 and SA533 Grade B Class 1 plate correlation monitor material.

Test material obtained from the Lower Shell Plate (after stress relieving) was taken at least one plate thickness from the quenched edges of the plate. Charpy V-notch impact specimens from lower shell plate C4339-1 were machined in the longitudinal orientation (longitudinal axis of the specimen parallel to the major rolling direction) and also in the transverse orientation (longitudinal axis of the specimen perpendicular to the major rolling 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 C4339-1 were machined in the longitudinal orientation. Tensile specimens from the weld metal were oriented with the long dimension of the specimen perpendicular to the weld direction.

WOL specimens from lower shell plate C4339-1 and weld metal were machined such that the simulated crack in the specimen would propagate normal and parallel to the major working direction for the plate specimen and parallel to the weld direction

The heat treatment of the surveillance materials is presented in Table 4-1. The chemical composition of the unirradiated and irradiated surveillance material[3,5] is presented in Table 4-2. No chemical analyses were performed for the surveillance materials in capsule Y.

Capsule Y contained dosimeter wires of pure iron, copper, nickel, and aluminum 0.15 weight percent cobalt (cadmium-shielded and unshielded). In addition, cadmium shielded fission monitors 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)
1.75% Ag, 0.75% Sn, 97.5% Pb	Melting Point: 590°F (310°C)

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

Table 4-1 Heat Treatment History of the Surry Unit 2 Reactor Vessel Surveillance Materials[a]			
Material	**Temperature (°F)**	**Time (hrs.)**	**Coolant**
Lower Shell Plate B4339-1	1625	9	Brine-Quench
	1212	9	Brine-Quench
	1140	15-1/4	Furnace Cooled
Weld Metal (heat # 0227)	1140	15-1/4	Furnace Cooled
Correlation Monitor Material	1675 ± 25	4	Air Cooled
	1600 ± 25	4	Water quenched
	1225 ± 25	4	Furnace Cooled
	1150 ± 25	40	Furnace Cooled to 600°F

Notes

(a) This table was taken from WCAP-8085[3].

Table 4-2 Chemical Composition (wt%) of the Surry Unit 2 Reactor Vessel Surveillance Materials (Unirradiated and Irradiated)

Element	Lower Shell Plate C4339-1		Weld Metal[b]		Correlation Monitor
	Unirradiated[c]	Capsule V Specimen V-12[d]	Unirradiated[c]	Capsule V Specimen W-12[d]	Unirradiated[c]
C	0.23	0.258	0 09	0 096	0.22
Mn	1 30	1 38	1.51	1 56	1.48
P	0 012	0 0108	0.017	0 0176	0.012
S	0 014	0.014	0 016	0 012	0.018
Si	0.25	0.239	0.46	0.448	0.25
Ni	0.54	0 52	0.56	0 53	0.68
Cr	0 075	0 091	0.10	0 115	---
V	0 001	0.007	0.002	<0 002	---
Mo	0.54	0.533	0.41	0.411	0.52
Co	<0 001[a]	0 007	<0.001[a]	0 006	---
Cu	0 11	0 104	0 19	0 184	0 14
Sn	0 008	---	0 005	---	---
Zn	0.005	---	0 009	---	---
Al	0 035	---	0.015	---	---
N_2	0 007	---	0 010	---	---
Ti	<0.001[a]	<0.001	<0 001[a]	<0.001[a]	---
Zr	<0 001[a]	---	<0.001[a]	---	---
As	0 009	---	0 005	---	---
B	<0.003[a]	---	<0.001[a]	---	---
Sb	0.001	---	0.001	---	---

Notes

(a) Not detected, the number represents the minimum limit of detection.

(b) Surveillance weld fabricated from the same heat of weld wire (0227) and Grau Lo flux (LW320) as used in the vessel intermediate to lower shell girth weld seam.

(c) This information was obtained from WCAP-8085[3].

(d) This information was obtained from WCAP-11499[3]

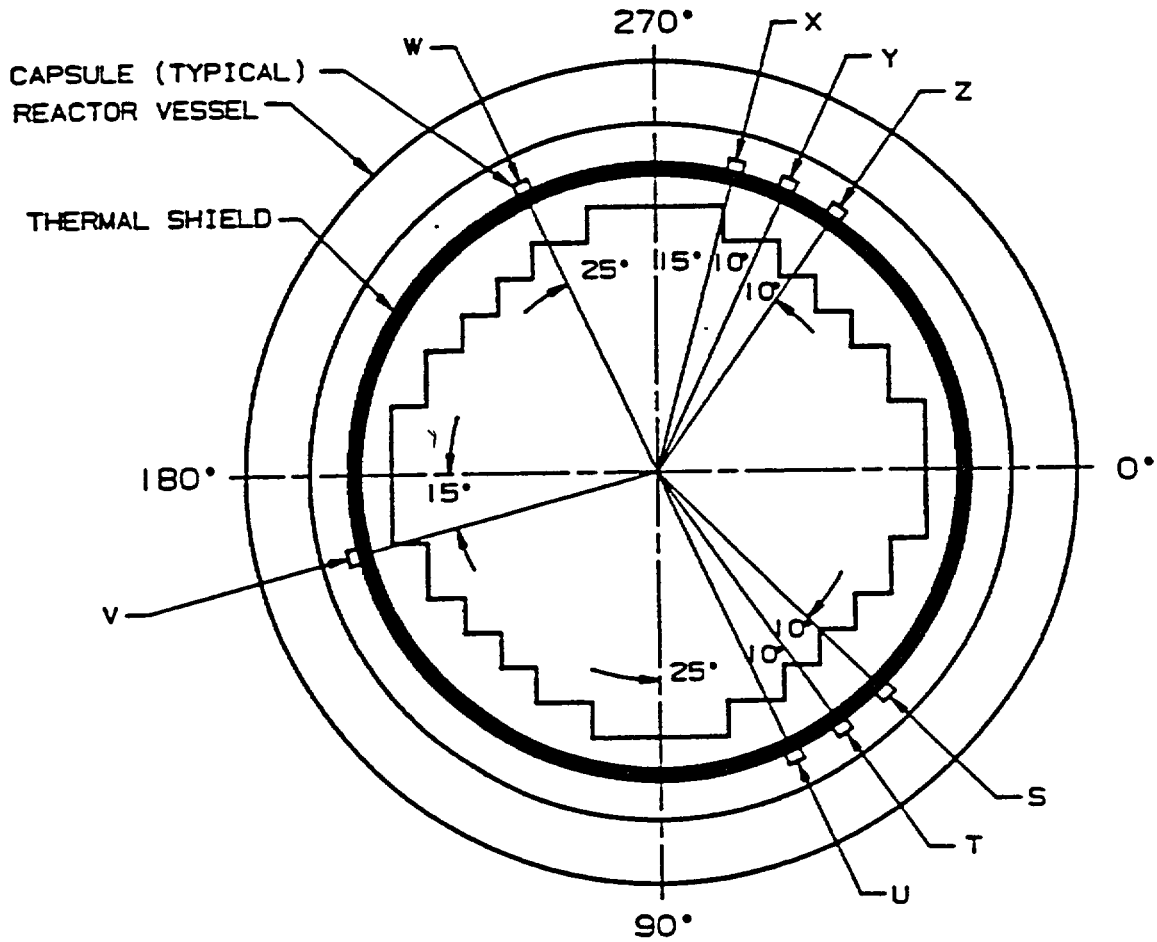

Figure 4-1 **Original Arrangement of Surveillance Capsules in the Surry Unit 2 Reactor Vessel**
[Note: Capsule Y was moved from 295° to 165°, Capsule Z was moved from 305° to 254° at the beginning of cycle 13 and Capsule T was moved from 55° to 165° at the beginning of cycle 18,]

SPECIMEN NUMBERING CODE:

V – SHELL PLATE C4339-1 (LONGITUDINAL ORIENTATION)
T – SHELL PLATE C4339-1 (TRANSVERSE ORIENTATION)
W – WELD METAL
H – HEAT AFFECTED ZONE
R – ASTM CORRELATION MONITOR MATERIAL

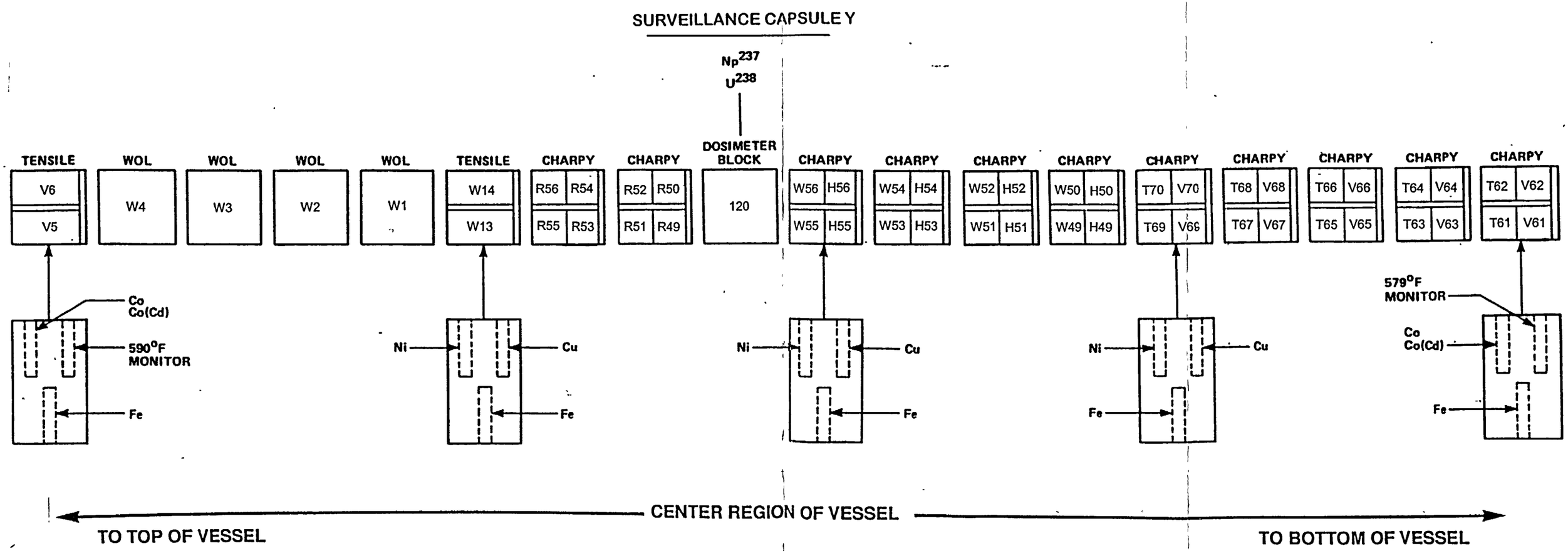

Figure 4-2 Capsule Y Diagram Showing The Location of Specimens, Thermal Monitors, and Dosimeters

5 TESTING OF SPECIMENS FROM CAPSULE Y

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[6], and Westinghouse Procedure RMF 8402[11], Revision 2 as modified by Westinghouse RMF Procedures 8102[12], Revision 1, and 8103[13], 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-8085[3]. 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[7] 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 A), 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:

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

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),

$$\sigma_Y = (P_{GY} * L) / [B * (W-a)^2 * 1.21] = (3.33 * P_{GY} * W) / [B * (W-a)^2] \quad (2)$$

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 5-6 through 5-10 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[8] . 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[9] and E21-92 (1998)[10], 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[18].

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 Y, which received a fluence of 2.73 x 10^{19} n/cm^2(E> 1.0 MeV) in 20.3 EFPY of operation, are presented in Tables 5-1 through 5-10 and are compared with unirradiated results[3] as shown in Figures 5-1 through 5-15.

The transition temperature increases and upper shelf energy decreases for the Capsule Y materials are summarized in Table 5-11 and led to the following results:

Irradiation of the reactor vessel lower shell plate C4339-1 (heat number 4339) 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 113.11°F and an irradiated 50 ft-lb transition temperature of 157.64°F. This results in a 30 ft-lb transition temperature increase of 114.22°F and a 50 ft-lb transition temperature increase of 124.26°F for the longitudinal oriented specimens. See Table 5-11.

Irradiation of the reactor vessel lower shell plate C4339-1 (heat number 4339) 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 118.27°F and an irradiated 50 ft-lb transition temperature of 174.08°F. This results in a 30 ft-lb transition temperature increase of 106.81°F and a 50 ft-lb transition temperature increase of 122.18°F for the longitudinal oriented specimens. See Table 5-11.

Irradiation of the weld metal (heat number 0227) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 157.84°F and an irradiated 50 ft-lb transition temperature of 247.25°F. This results in a 30 ft-lb transition temperature increase of 178.32°F and a 50 ft-lb transition temperature increase of 224.76°F. See Table 5-11.

Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 75 07°F and an irradiated 50 ft-lb transition temperature of 144.55°F. This results in a 30 ft-lb transition temperature increase of 127.91°F and a 50 ft-lb transition temperature increase of 150.28°F. See Table 5-11.

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The average upper shelf energy of the lower shell plate C4339-1 (longitudinal orientation) resulted in an average energy decrease of 17 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 111 ft-lb for the longitudinal oriented specimens. See Table 5-11.

The average upper shelf energy of the lower shell plate C4339-1 (transverse orientation) resulted in an average energy decrease of 10 ft-lb after irradiation This results in an irradiated average upper shelf energy of 94 ft-lb for the longitudinal oriented specimens. See Table 5-11.

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The average upper shelf energy of the weld HAZ metal Charpy specimens resulted in an average energy decrease of 22 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 94 ft-lb for the weld HAZ metal. See Table 5-11.

The average upper shelf energy of the Correlation Monitor Material Plate HSST02 Charpy specimens resulted in an average energy decrease of 23 ft-lb after irradiation. This results in an irradiated average upper shelf energy of 100 ft-lb for the weld correlation monitor metal. See Table 5-11.

A comparison, as presented in Table 5-12, of the Surry Unit 2 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 C4339-1 contained in capsule Y (longitudinal & transverse) is greater than the Regulatory Guide 1.99, Revision 2, predictions. However, the shift value is within two sigma of the predicted value.

— The measured 30 ft-lb shift in transition temperature values of the weld metal contained in capsule Y is less than the Regulatory Guide 1.99, Revision 2, predictions.

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

The fracture appearance of each irradiated Charpy specimen from the various surveillance Capsule V materials is shown in Figures 5-16 through 5-20 and shows an increasingly ductile or tougher appearance with increasing test temperature.

All beltline materials, except the intermediate to lower shell girth weld, 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 life of the vessel (30.1 EFPY) as required by 10CFR50, Appendix G [2]. The low USE for the intermediate to lower shell girth weld is justified under the Equivalent Margin Analysis (EMA) in BAW-2192PA & BAW-2178PA

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

The Charpy V-notch data presented in WCAP-8085[3], BMI-0975-/SU2[4] and WCAP-11499[5] were based on a hand-fit. The results presented in this report are based on a re-plot of all capsule data using CVGRAPH, Version 4.1[14], which is a symmetric hyperbolic tangent curve-fitting program. Appendix B presents a comparison of the Charpy V-Notch test results for each capsule based on previous fit vs. symmetric hyperbolic tangent fit. 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 Y irradiated to 2.73×10^{19} n/cm^2 (E> 1.0 MeV) are presented in Table 5-13 and are compared with unirradiated results[3] as shown in Figures 5-21 and 5-22.

The results of the tensile tests performed on the lower Shell Plate C4339-1 (longitudinal orientation) indicated that irradiation to 2.73×10^{19} n/cm^2 (E> 1.0 MeV) caused approximately a 14 ksi increase in the 0.2 percent offset yield strength and approximately a 10 to 13 ksi increase in the ultimate tensile strength when compared to unirradiated data[3]. See Figure 5-21.

The results of the tensile tests performed on the surveillance weld metal indicated that irradiation to 2.73×10^{19} n/cm^2 (E> 1.0 MeV) caused approximately a 19 to 21 ksi increase in the 0.2 percent offset yield strength and approximately a 12 to 21 ksi increase in the ultimate tensile strength when compared to unirradiated data[3]. See Figure 5-22.

The fractured tensile specimens for the Lower Shell Plate C4339-1 material are shown in Figure 5-23, while the fractured tensile specimens for the surveillance weld metal are shown in Figure 5-24. The engineering stress-strain curves for the tensile tests are shown in Figures 5-25 and 5-26.

5.4 WEDGE OPENING LOADING TESTS

Per the surveillance capsule testing contract, the Wedge Opening Loading (WOL) 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 Surry Unit 2 Lower Shell Plate C4339-1 Irradiated to a Fluence of 2.73 x 10^{19} n/cm^2 (E> 1.0 MeV) (Longitudinal Orientation)							
Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	°F	°C	ft-lbs	Joules	mils	mm	%
V70	0	-18	8	11	2	0.05	2
V61	10	-12	12	16	4	0 10	5
V65	100	38	33	45	22	0 56	15
V64	125	52	37	50	26	0.66	20
V68	175	79	52	71	38	0.97	45
V69	200	93	60	81	40	1.02	50
V66	250	121	100	136	69	1.75	95
V67	300	149	105	142	74	1.88	100
V63	325	163	111	151	71	1.80	100
V62	350	177	116	157	74	1 88	100

Table 5-2 Charpy V-notch Data for the Surry Unit 2 Lower Shell Plate C4339-1 Irradiated to a Fluence of 2.73 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	%
T62	0	-18	6	8	0	0.00	2
T67	10	-12	11	15	2	0.05	5
T70	100	38	28	38	19	0.48	15
T69	125	52	40	54	27	0.69	20
T61	175	79	47	64	33	0.84	35
T63	200	93	46	62	34	0.86	40
T68	250	121	74	100	57	1.45	90
T64	300	149	95	129	68	1.73	100
T66	325	163	95	129	69	1.75	100
T65	350	177	93	126	68	1.73	100

Table 5-3 Charpy V-notch Data for the Surry Unit 2 Surveillance Weld Metal Irradiated to a Fluence of 2.73 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	%
W56	72	22	11	15	4	0.10	5
W53	100	38	18	24	13	0.33	15
W54	150	66	24	33	15	0.38	25
W49	175	79	39	53	31	0.79	65
W55	200	93	36	49	27	0.69	55
W50	250	121	54	73	43	1.09	98
W51	350	177	54	73	40	1.02	100
W52	375	191	66	89	50	1.27	100

Table 5-4 Charpy V-notch Data for the Surry Unit 2 Heat-Affected-Zone (HAZ) Material Irradiated to a Fluence of 2.73 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	%
H56	-50	-46	18	24	7	0.18	15
H49	-25	-32	25	34	12	0.30	20
H54	0	-18	12	16	7	0.18	35
H52	100	38	32	43	27	0.69	55
H53	150	66	44	60	35	0.89	65
H55	200	93	66	89	48	1.22	75
H50	300	149	95	129	66	1.68	100
H51	325	163	93	126	66	1.68	100

Table 5-5 Charpy V-notch Data for the Surry Unit 2 Correlation Monitor Material Irradiated to a Fluence of 2.73 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	%
R51	72	22	7	9	0	0.00	10
R56	125	52	25	34	15	0.38	15
R54	175	79	22	30	15	0.38	20
R52	200	93	34	46	25	0.64	25
R49	225	107	32	43	23	0.58	30
R50	250	121	65	88	43	1.09	55
R53	325	163	93	126	69	1.75	100
R55	375	191	107	145	69	1.75	100

Table 5-6 Instrumented Charpy Impact Test Results for the Surry Unit 2 Lower Shell Plate C4339-1 Irradiated to a Fluence of 2.73 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 S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_P/A								
V70	0	8.3	67	36	30	3603	0.15	3632	0.16	3556	0	120	120
V61	10	12.1	98	54	44	3934	0.16	4433	0.19	4406	0	131	139
V65	100	33.4	269	213	56	3455	0.15	4560	0.48	4521	0	115	133
V64	125	36.0	290	224	66	3310	0.14	4483	0.51	4483	180	110	130
V68	175	51.7	416	303	114	3283	0.15	4396	0.66	4345	1159	109	128
V69	200	59.5	480	308	171	3212	0.15	4460	0.66	4301	1120	107	128
V66	250	99.2	799	312	487	3139	0.15	4474	0.67	3459	2648	105	127
V67	300	102.5	826	303	523	3038	0.15	4387	0.67	N/A	N/A	101	124
V63	325	108.9	878	311	566	3133	0.15	4423	0.68	N/A	N/A	104	126
V62	350	111.4	898	300	598	3105	0.15	4311	0.66	N/A	N/A	103	123

Table 5-7 Instrumented Charpy Impact Test Results for the Surry Unit 2 Lower Shell Plate C4339-1 Irradiated to a Fluence of 2.73 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 S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_P/A								
T62	0	6	52	27	25	3035	0.14	3040	0.14	3020	0	101	101
T67	10	10	84	45	39	3862	0.16	4108	0 18	3957	0	129	133
T70	100	28	228	166	61	3309	0.14	4420	0.40	4412	150	110	129
T69	125	40	323	234	89	3358	0.15	4467	0.53	4435	533	112	130
T61	175	46	372	226	146	3249	0.14	4419	0.51	4187	1187	108	128
T63	200	46	367	220	147	3192	0.15	4362	0.51	4308	1747	106	126
T68	250	74	599	219	380	3026	0.14	4245	0.52	3770	2730	101	121
T64	300	95	766	302	464	3025	0.14	4335	0.67	N/A	N/A	101	123
T66	325	94	760	295	465	3068	0.15	4259	0.67	N/A	N/A	102	122
T65	350	91	729	289	440	2874	0.15	4211	0.67	N/A	N/A	96	118

Table 5-8 Instrumented Charpy Impact Test Results for the Surry Unit 2 Surveillance Weld Metal Irradiated to a Fluence of 2.73 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 S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_P/A								
W56	72	11	92	54	39	3608	0.15	4237	0.19	4183	0	120	131
W53	100	20	158	73	85	3530	0.14	4425	0.23	4264	266	118	132
W54	150	24	195	69	127	3318	0.14	4243	0.22	4143	1033	110	126
W49	175	39	314	172	143	3211	0.14	4189	0.42	4165	2057	107	123
W55	200	36	290	175	115	3148	0.14	4109	0.43	4099	1530	105	121
W50	250	54	436	189	247	3259	0.14	4214	0.45	3935	2494	109	124
W51	350	53	428	162	266	3022	0.14	4016	0.41	N/A	N/A	101	117
W52	375	66	529	192	336	2956	0.15	4039	0.49	N/A	N/A	98	116

Table 5-9 Instrumented Charpy Impact Test Results for the Surry Unit 2 Heat-Affected-Zone (HAZ) Metal Irradiated to a Fluence of 2.73 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 S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_P/A								
H56	-50	18	144	74	70	3773	0.15	4637	0.23	4622	0	126	140
H49	-25	25	201	73	129	3805	0.15	4665	0.22	4507	0	127	141
H54	0	12	96	35	62	3408	0.14	3523	0.16	3508	689	113	115
H52	100	33	264	67	197	3343	0.14	4183	0.22	4107	1542	111	125
H53	150	44	357	178	179	3342	0.15	4282	0.43	4277	2263	111	127
H55	200	65	527	236	291	3321	0 15	4390	0.54	2540	879	111	128
H50	300	94	759	305	454	3079	0 15	4311	0.68	N/A	N/A	103	123
H51	325	90	723	227	496	3217	0.15	4293	0.53	N/A	N/A	107	125

Table 5-10 Instrumented Charpy Impact Test Results for the Surry Unit 2 Correlation Monitor Metal Irradiated to a Fluence of 2.73 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 S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_P/A								
R51	72	6	52	26	25	2914	0.13	2965	0.14	2956	0	97	98
R56	125	25	199	150	49	3205	0.14	4128	0.38	4094	0	107	122
R54	175	22	176	62	115	3122	0.14	3910	0.22	3854	502	104	117
R52	200	33	267	167	101	2989	0.14	4052	0.43	3928	1246	100	117
R49	225	32	255	154	101	2993	0.14	4073	0.40	4063	1212	100	118
R50	250	65	523	220	303	3066	0.15	4240	0.52	4030	2233	102	122
R53	325	91	729	294	435	3034	0.14	4213	0.66	N/A	N/A	101	121
R55	375	105	842	290	552	3037	0.16	4154	0.68	N/A	N/A	101	120

Table 5-11 Effect of Irradiation to 2.73 x 10^{19} n/cm^2 (E>1.0 MeV) on the Capsule V-Notch Toughness Properties of the Surry Unit 2 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 C4339-1 (Long)	-1.11	113.11	114.22	26.92	161.52	134.6	33.38	157.64	124.26	128	111	-17
Lower Shell Plate C4339-1 (Trans)	11.46	118.27	106.81	41.73	178.53	136.79	51.9	174.08	122.18	104	94	-10
Weld Metal (heat # 0227)	-20.48	157.84	178.32	-1.88	218.35	220.24	22.48	247.25	224.76	91	58	-33
HAZ Metal	-52 83	75.07	127.91	-8.73	144.04	152 77	-5 72	144.55	150 28	116	94	-22
Correlation Monitor Material	46.44	194.47	148.02	58.96	235.59	176.63	78.3	232.43	154.12	123	100	-23

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, 5-10 and 5-13).

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, 5-11 and 5-14).

c. Any difference in unirradiated properties reported in this Table in comparison to WCAP-8085, BMI-0975-/SU2 and WCAP-11499, are due to now using a symmetric hyperbolic tangent curve fitting of the Charpy data versus the method used in the previous analyses. See Appendix B.

Table 5-12 Comparison of the Surry Unit 2 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] ($\times 10^{19}$ 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 C4339-1 (Longitudinal)	X	0.297	49.03	59.08	15	5
	V	1.89	86.17	79.12	23	5
	Y	2.73	93.07	114.22	26	13
Lower Shell Plate C4339-1 (Transverse)	X	0.297	49.03	48 67	15	10
	V	1.89	86 17	63.6	23	10
	Y	2.73	93.07	106.81	26	10
Surveillance Program Weld Metal	X	0.297	99.73	95.65	25	22
	V	1.89	175.28	140.21	38	34
	Y	2 73	189.31	178.32	41	36
Heat Affected Zone Material	X	0.297	--	29.22	---	13
	V	1.89	--	51.23	---	19
	Y	2.73	--	127.91	---	19
Correlation Monitor Material	X	0.297	--	62.19	---	16
	V	1.89	--	116.55	---	17
	Y	2.73	--	148.02	---	19

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 The best estimate copper/nickel values for the surveillance plate and weld were 0.11/0 54 and 0.19/0.55, respectively.

(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 fluence values, not the best estimate. For best estimate values see Appendix D of this report

Table 5-13 Tensile Properties of the Surry Unit 2 Capsule V Reactor Vessel Surveillance Materials Irradiated to 2.73×10^{19} n/cm^2 (E > 1.0 MeV)

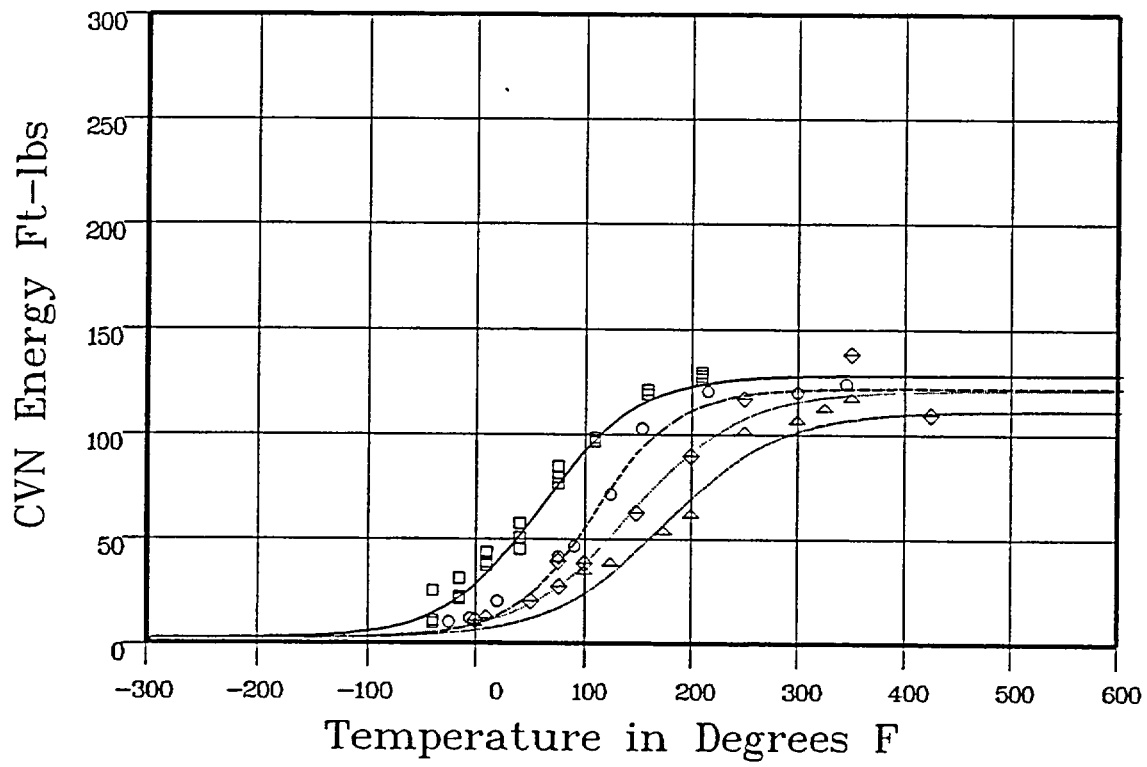
Material	Sample Number	Test Temp. (°F)	0.2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Stress (ksi)	Fracture Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Lower Shell Plate C4339-1 (Long.)	V 5	300	75.7	96.7	3.20	200.4	65.2	9.0	21.5	67
	V 6	550	71.3	98.6	3.50	159.8	71.3	10.5	19.5	55
Weld Metal (Heat # 0227)	W 13	300	84.5	98.8	3.58	167.2	72.8	9.0	18.5	56
	W 14	550	80.5	97.0	3.60	160.5	73.3	8.5	17.5	54

LOWER SHELL PLATE C4339-1 (LONG)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:08:19 on 07-19-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	128	0	-1.11	0	33.38	0
2	0	2.19	0	122	-6	57.97	59.08	89.01	55.63
3	0	2.19	0	121	-7	78.01	79.12	118.95	85.57
4	0	2.19	0	111	-17	113.11	114.22	157.64	124.26



Curve Legend

1 □——— 2 ○--------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori	Heat#
1	SU2	UNIRR	PLATE SA533B1	LT	C4339-1
2	SU2	X	PLATE SA533B1	LT	C4339-1
3	SU2	V	PLATE SA533B1	LT	C4339-1
4	SU2	Y	PLATE SA533B1	LT	C4339-1

Figure 5-1 Charpy V-Notch Impact Energy vs. Temperature for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Longitudinal Orientation)

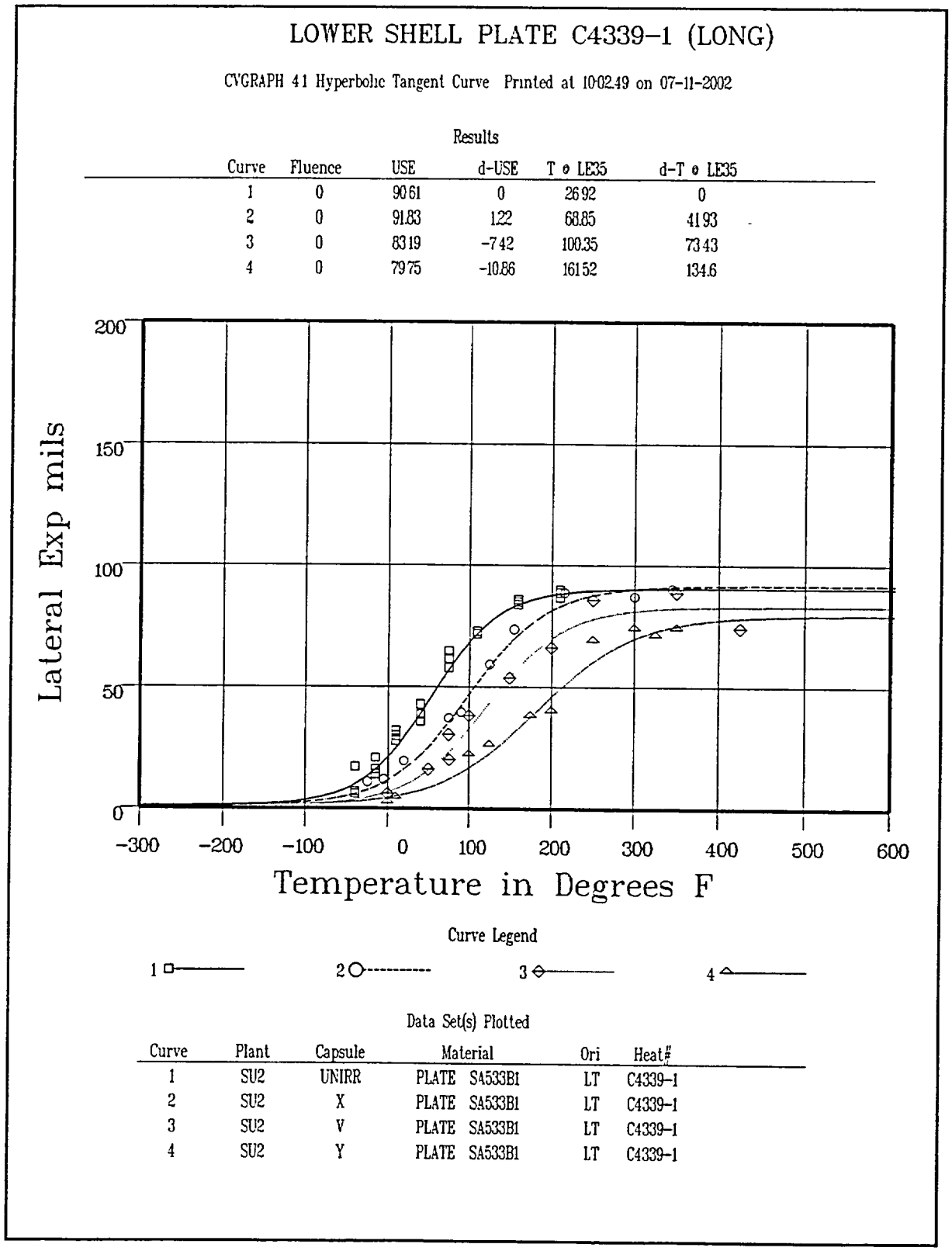


Figure 5-2 Charpy V-Notch Lateral Expansion vs. Temperature for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Longitudinal Orientation)

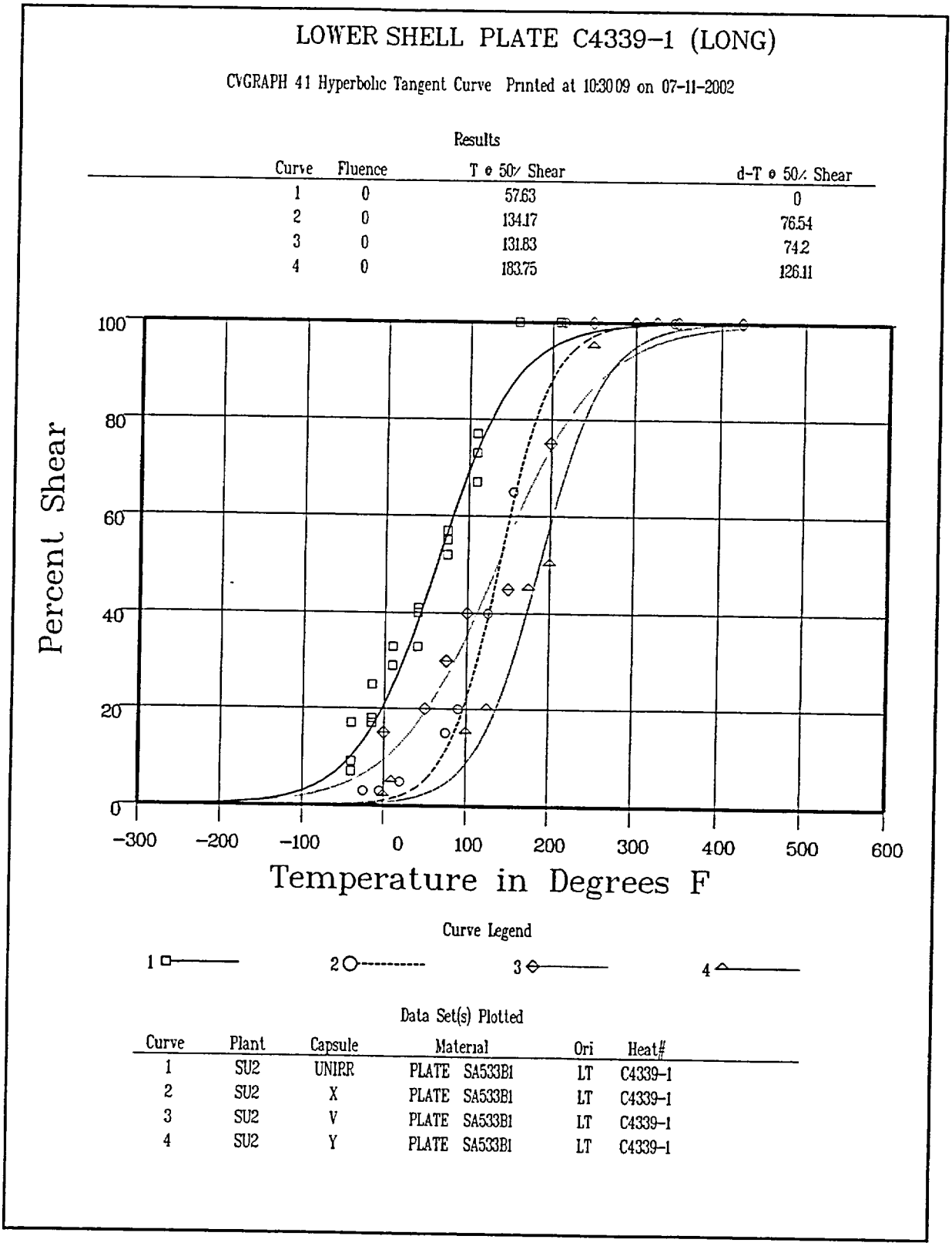


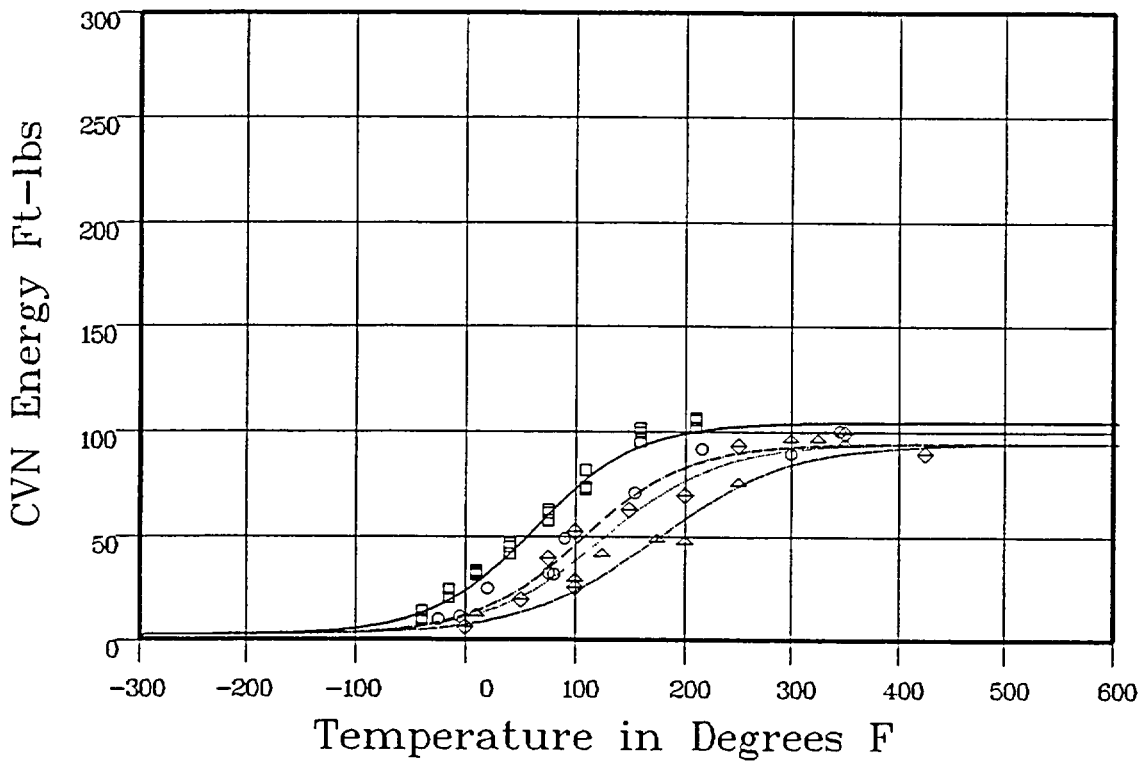
Figure 5-3 Charpy V-Notch Percent Shear vs. Temperature for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Longitudinal Orientation)

LOWER SHELL PLATE C4339-1 (TRANS)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:24:33 on 07-19-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	104	0	11.46	0	51.9	0
2	0	2.19	0	94	-10	60.13	48.67	104.81	52.91
3	0	2.19	0	94	-10	75.06	63.6	124.07	72.17
4	0	2.19	0	94	-10	118.27	106.81	174.08	122.18



Curve Legend

1 ▫——— 2 ○---------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori	Heat#
1	SU2	UNIRR	PLATE SA533B1	TL	C4339-1
2	SU2	X	PLATE SA533B1	TL	C4339-1
3	SU2	V	PLATE SA533B1	TL	C4339-1
4	SU2	Y	PLATE SA533B1	TL	C4339-1

Figure 5-4 Charpy V-Notch Impact Energy vs. Temperature for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Transverse Orientation)

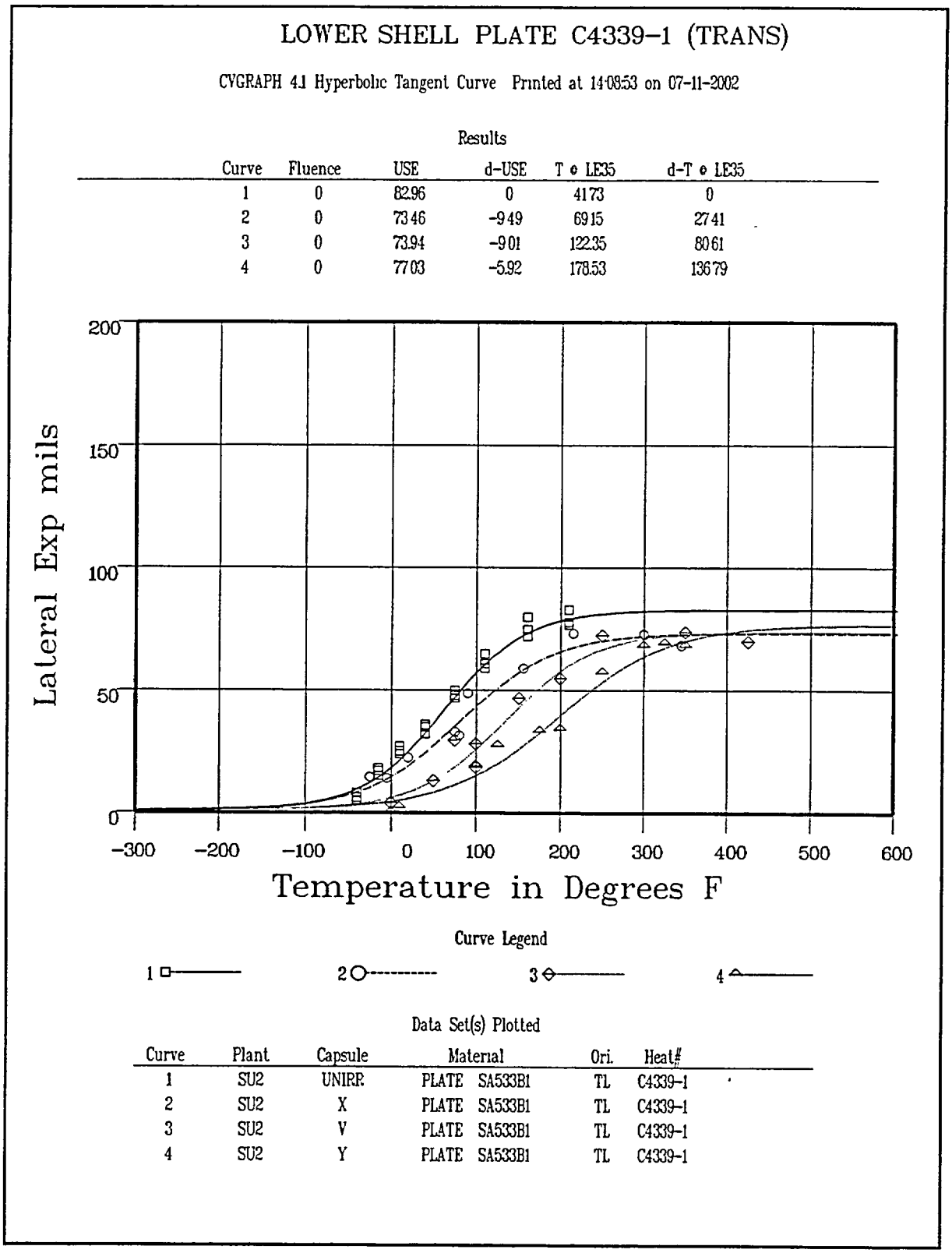


Figure 5-5 Charpy V-Notch Lateral Expansion vs. Temperature for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Transverse Orientation)

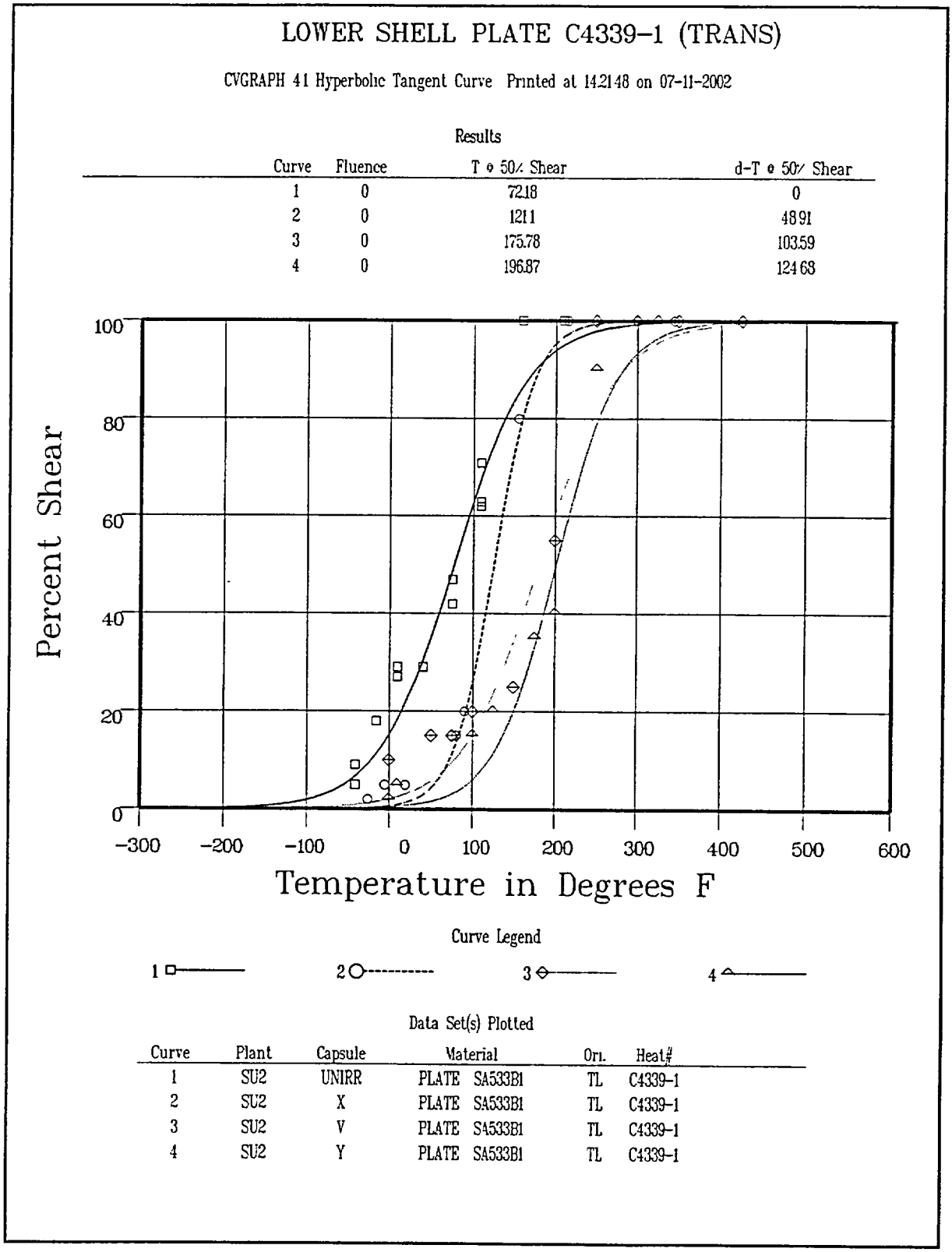


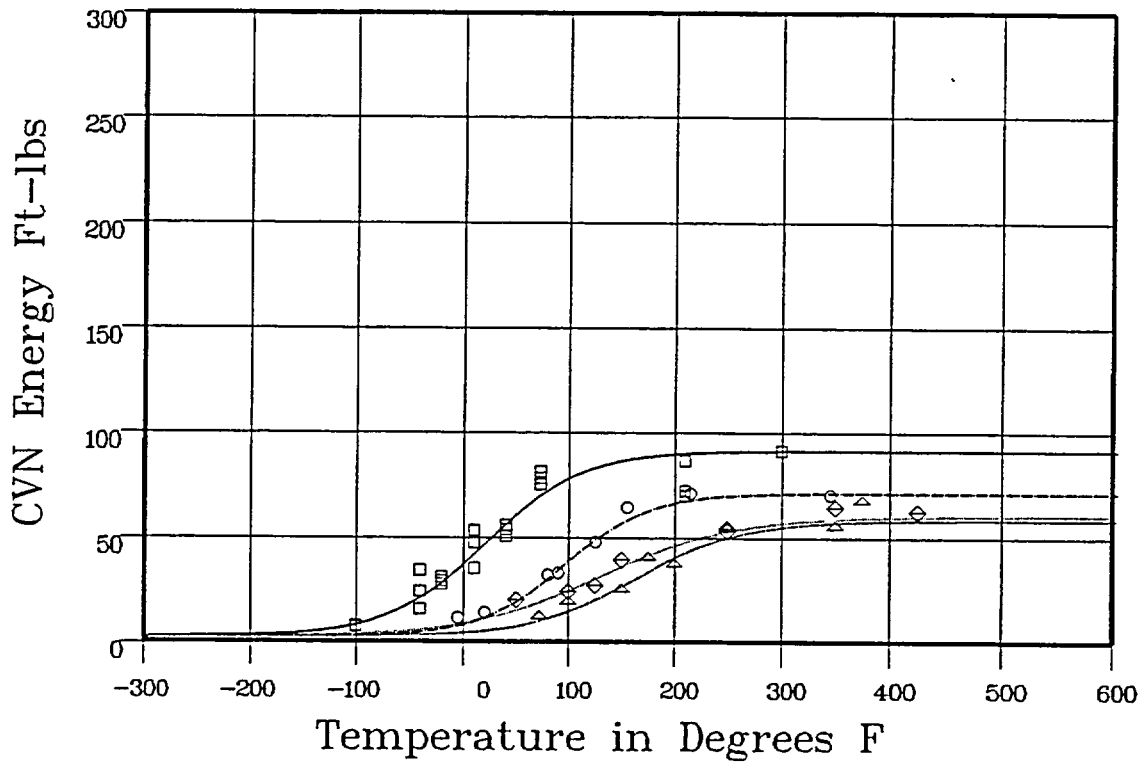
Figure 5-6 Charpy V-Notch Percent Shear vs. Temperature for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Transverse Orientation)

SURVEILLANCE PROGRAM WELD METAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:31:31 on 07-19-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	91	0	-20.48	0	22.48	0
2	0	2.19	0	71	-20	75.16	95.65	124.66	102.17
3	0	2.2	0	60	-31	119.73	140.21	222.71	200.22
4	0	2.2	0	58	-33	157.84	178.32	247.25	224.76



Curve Legend

1 □——— 2 ○- - - - - 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori	Heat#
1	SU2	UNIRR	WELD	WIRE	HEAT 0227
2	SU2	X	WELD	WIRE	HEAT 0227
3	SU2	V	WELD	WIRE	HEAT 0227
4	SU2	Y	WELD	WIRE	HEAT 0227

Figure 5-7 Charpy V-Notch Impact Energy vs. Temperature for Surry Unit 2 Reactor Vessel Weld Metal

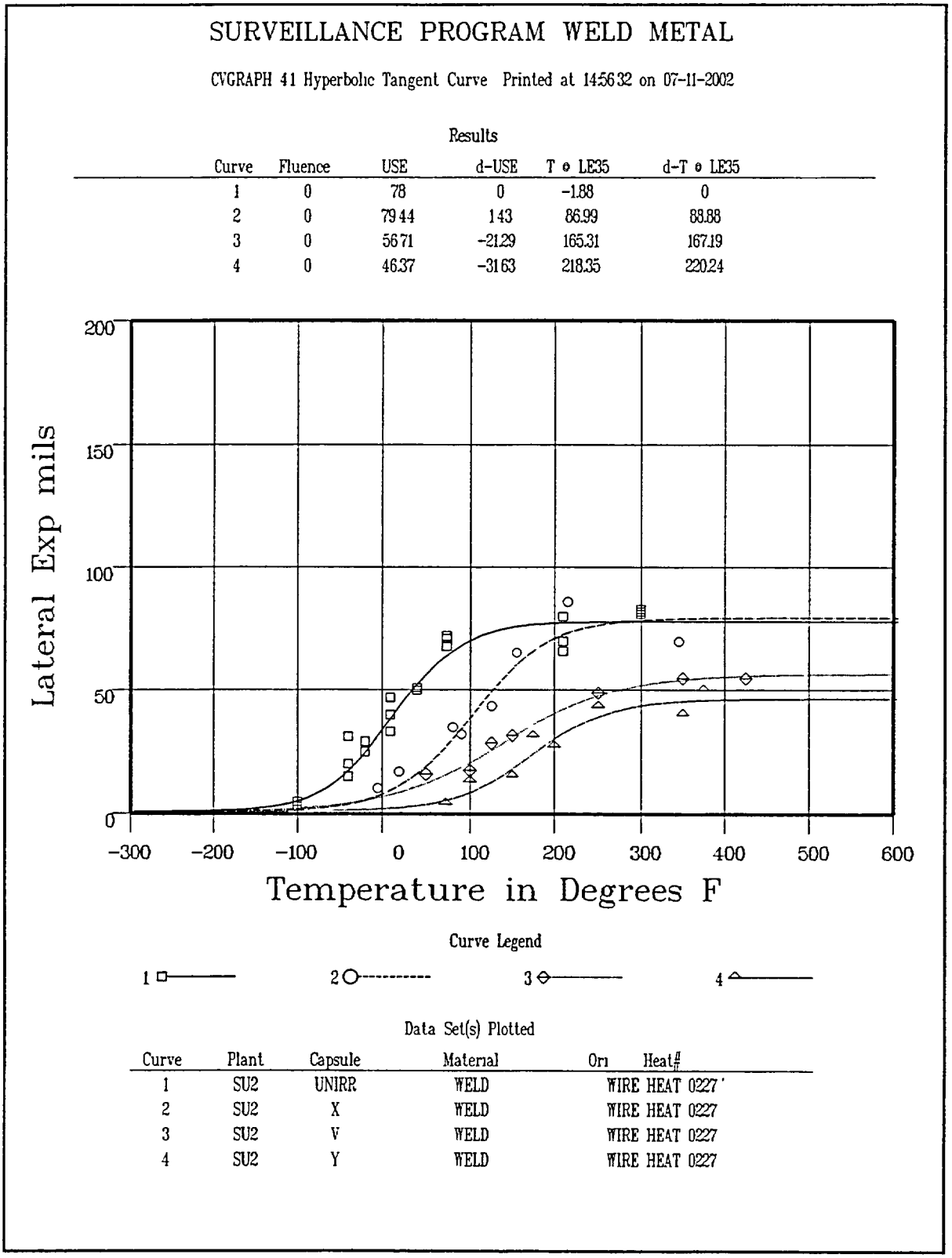


Figure 5-8 Charpy V-Notch Lateral Expansion vs. Temperature for Surry Unit 2 Reactor Vessel Weld Metal

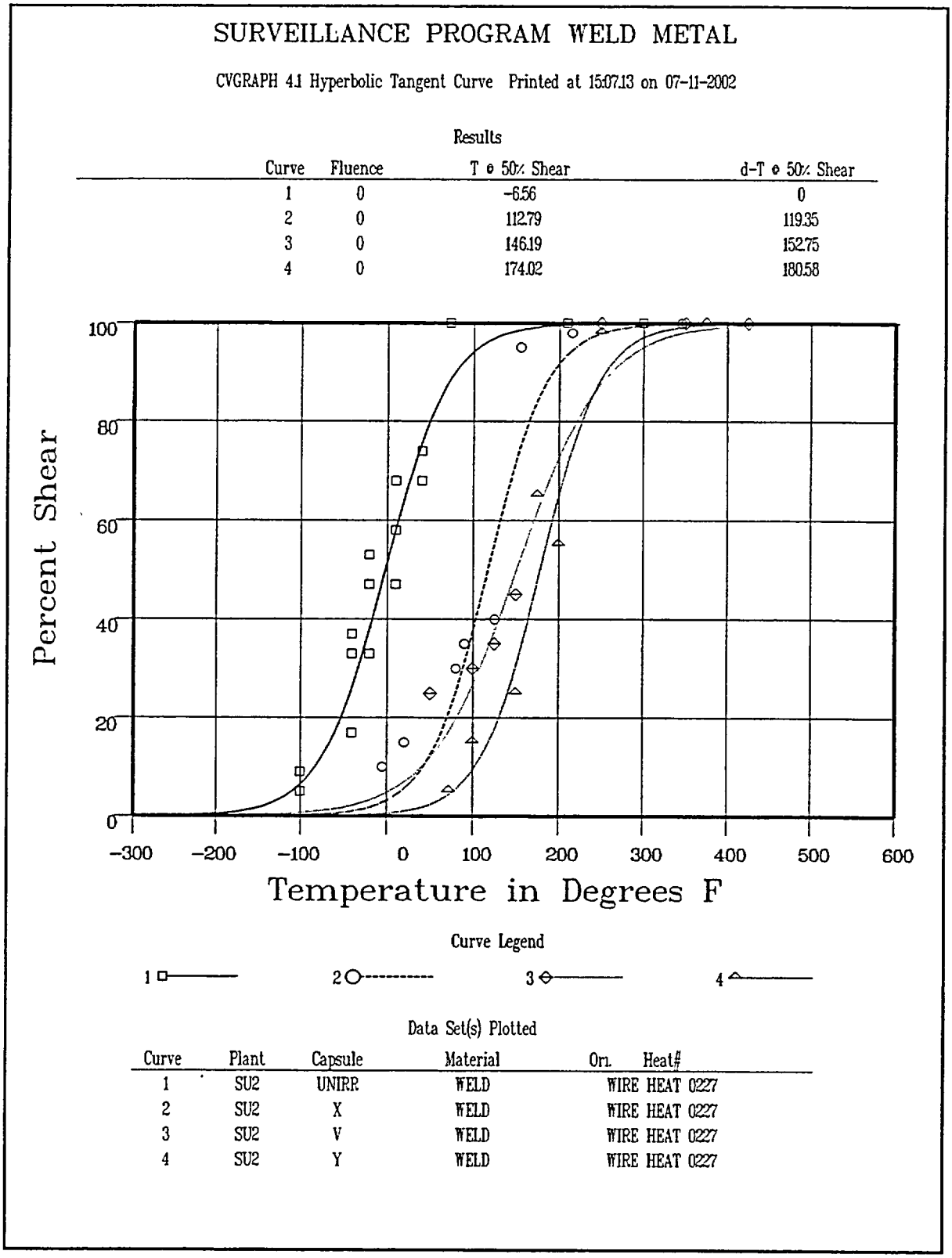


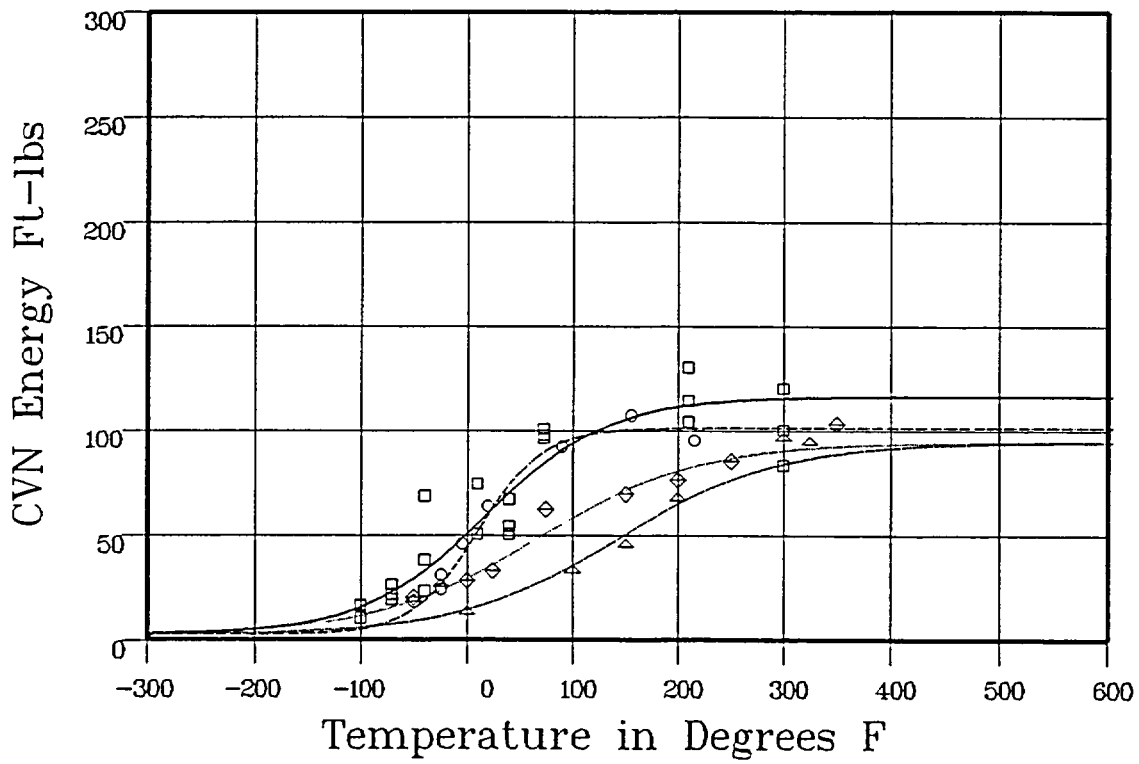
Figure 5-9 Charpy V-Notch Percent Shear vs. Temperature for Surry Unit 2 Reactor Vessel Weld Metal

SURVEILLANCE PROGRAM HEAT AFFECTED ZONE

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:39:11 on 07-19-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	116	0	-52.83	0	-5.72	0
2	0	2.19	0	101	-15	-23.61	29.22	3.6	9.32
3	0	2.19	0	94	-22	-1.6	51.23	68.18	73.9
4	0	2.19	0	94	-22	75.07	127.91	144.55	150.28



Curve Legend

1 □——— 2 ○---------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori	Heat#
1	SU2	UNIRR	HEAT AFF'D ZONE	C4339-1	SIDE OF WELD
2	SU2	X	HEAT AFF'D ZONE	C4339-1	SIDE OF WELD
3	SU2	V	HEAT AFF'D ZONE	C4339-1	SIDE OF WELD
4	SU2	Y	HEAT AFF'D ZONE	C4339-1	SIDE OF WELD

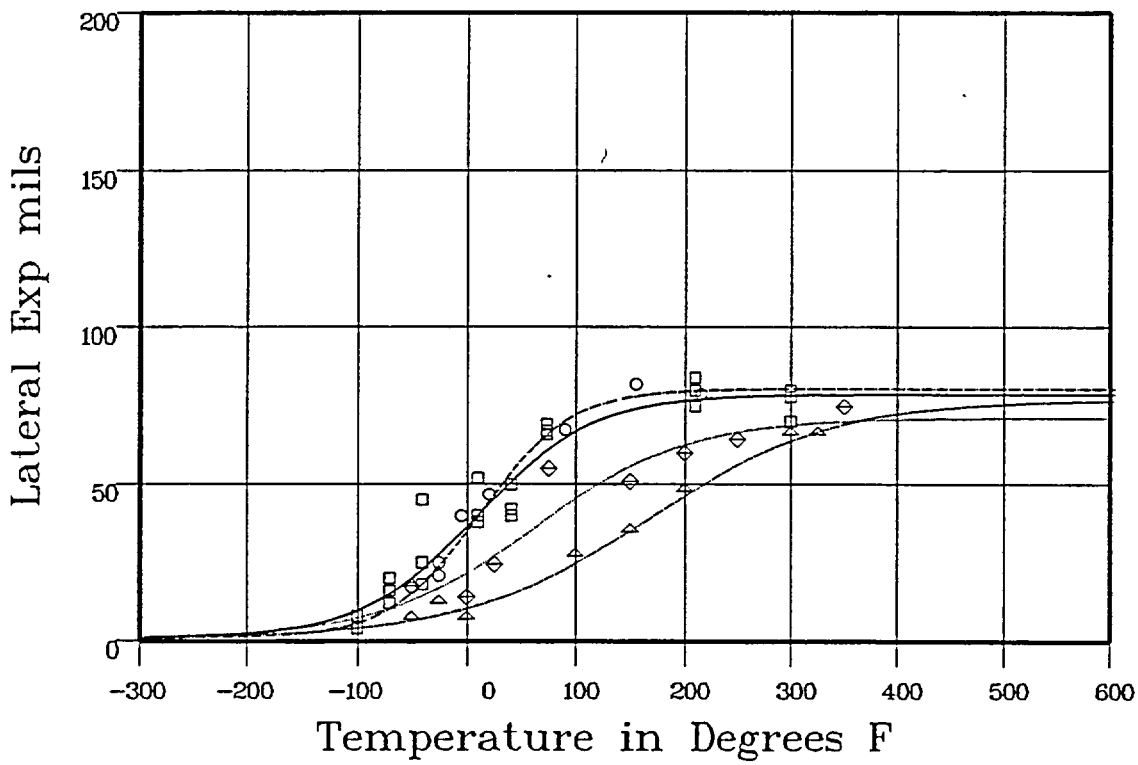
Figure 5-10 Charpy V-Notch Impact Energy vs. Temperature for Surry Unit 2 Reactor Vessel Heat-Affected-Zone Material

SURVEILLANCE PROGRAM HEAT AFFECTED ZONE

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:12:16 on 07-12-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	78.84	0	-8.73	0
2	0	80.55	1.71	-4.43	4.3
3	0	71.15	-7.69	51.95	60.68
4	0	77.08	-1.75	144.04	152.77



Curve Legend

1 □——— 2 ○---------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	SU2	UNIRR	HEAT AFF'D ZONE	C4339-1	SIDE OF WELD
2	SU2	X	HEAT AFF'D ZONE	C4339-1	SIDE OF WELD
3	SU2	V	HEAT AFF'D ZONE	C4339-1	SIDE OF WELD
4	SU2	Y	HEAT AFF'D ZONE	C4339-1	SIDE OF WELD

Figure 5-11 Charpy V-Notch Lateral Expansion vs. Temperature for Surry Unit 2 Reactor Vessel Heat-Affected-Zone Material

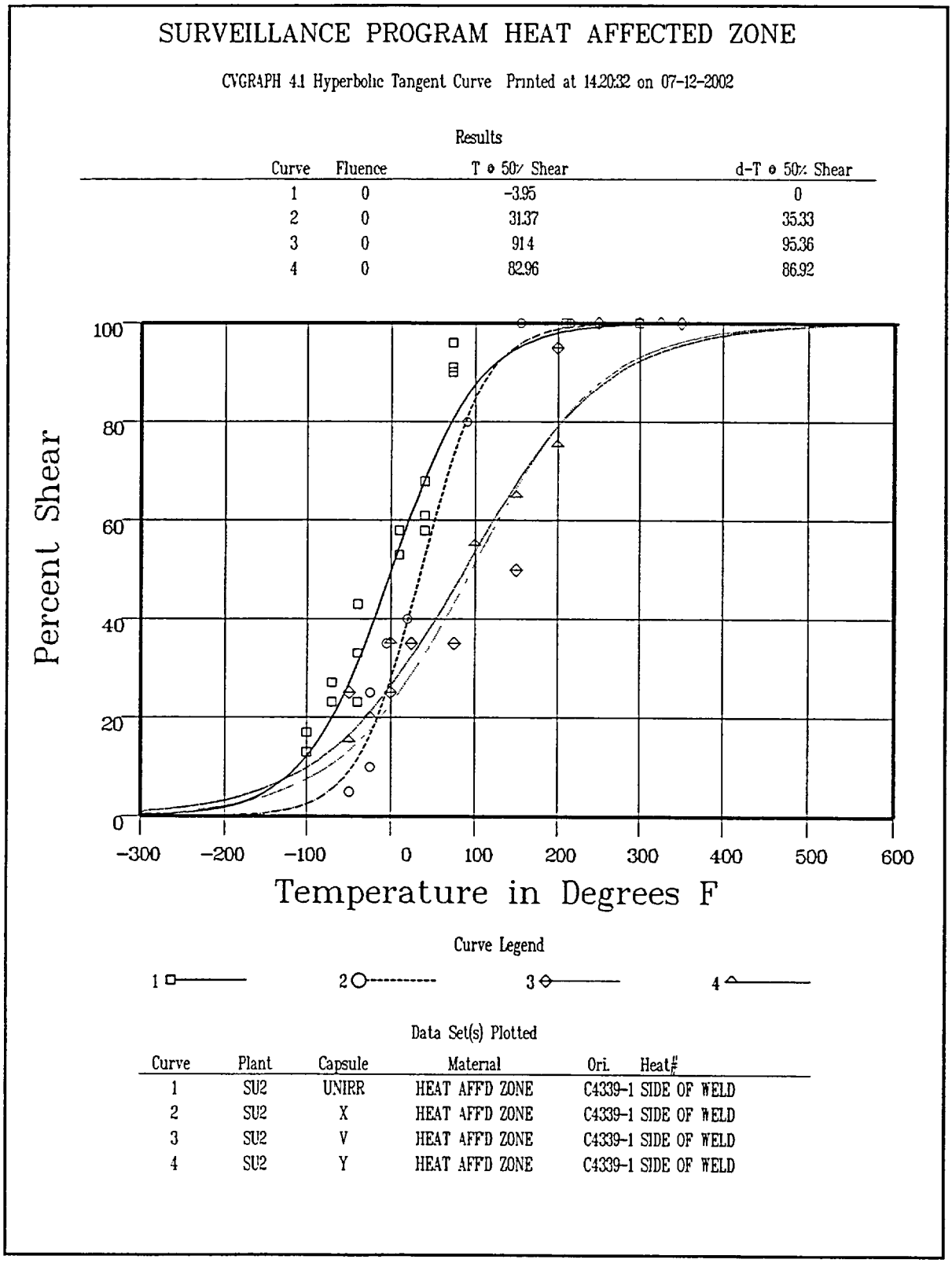


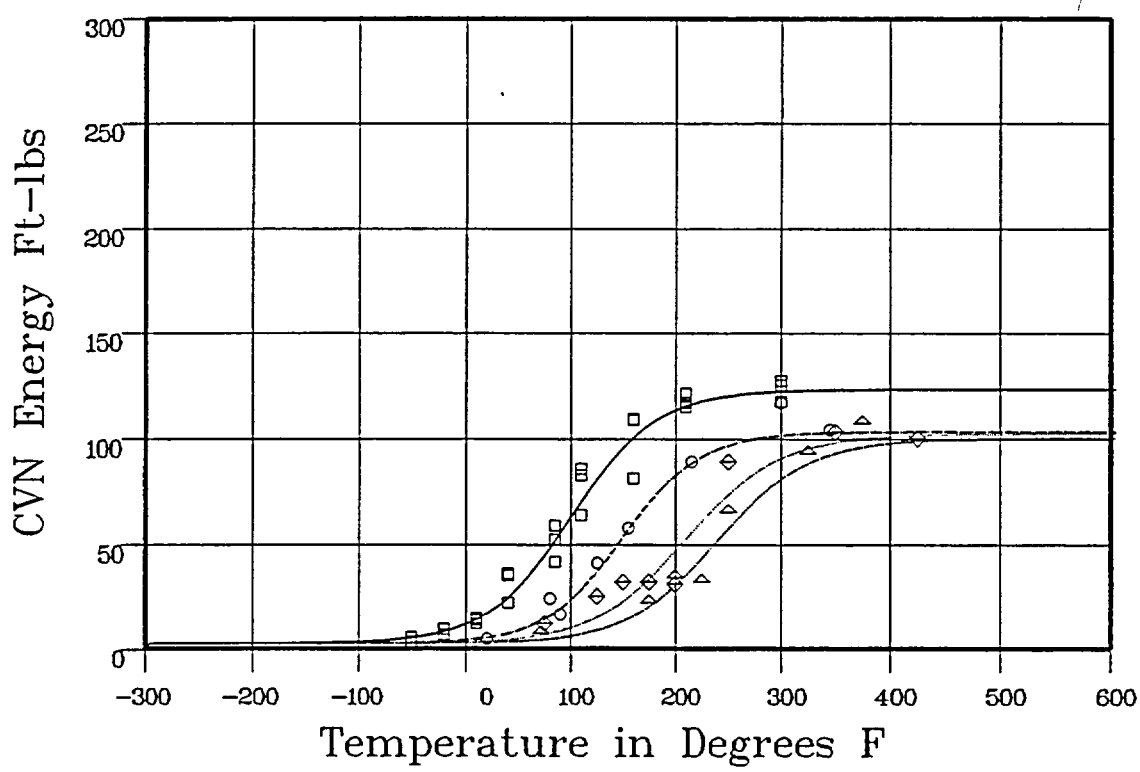
Figure 5-12 Charpy V-Notch Percent Shear vs. Temperature for Surry Unit 2 Reactor Vessel Heat-Affected-Zone Material

SURVEILLANCE PROGRAM MONITOR MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:48:20 on 07-19-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	123	0	46.44	0	78.3	0
2	0	2.19	0	103	-20	108.64	62.19	140.6	62.3
3	0	2.19	0	102	-21	162.99	116.55	201.2	122.89
4	0	2.19	0	100	-23	194.47	148.02	232.43	154.12



Curve Legend

1 □——— 2 ○--------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	SU2	UNIRR	SRM SA533B1	LT HSST PLATE	02
2	SU2	X	SRM SA533B1	LT HSST PLATE	02
3	SU2	V	SRM SA533B1	LT HSST PLATE	02
4	SU2	Y	SRM SA533B1	LT HSST PLATE	02

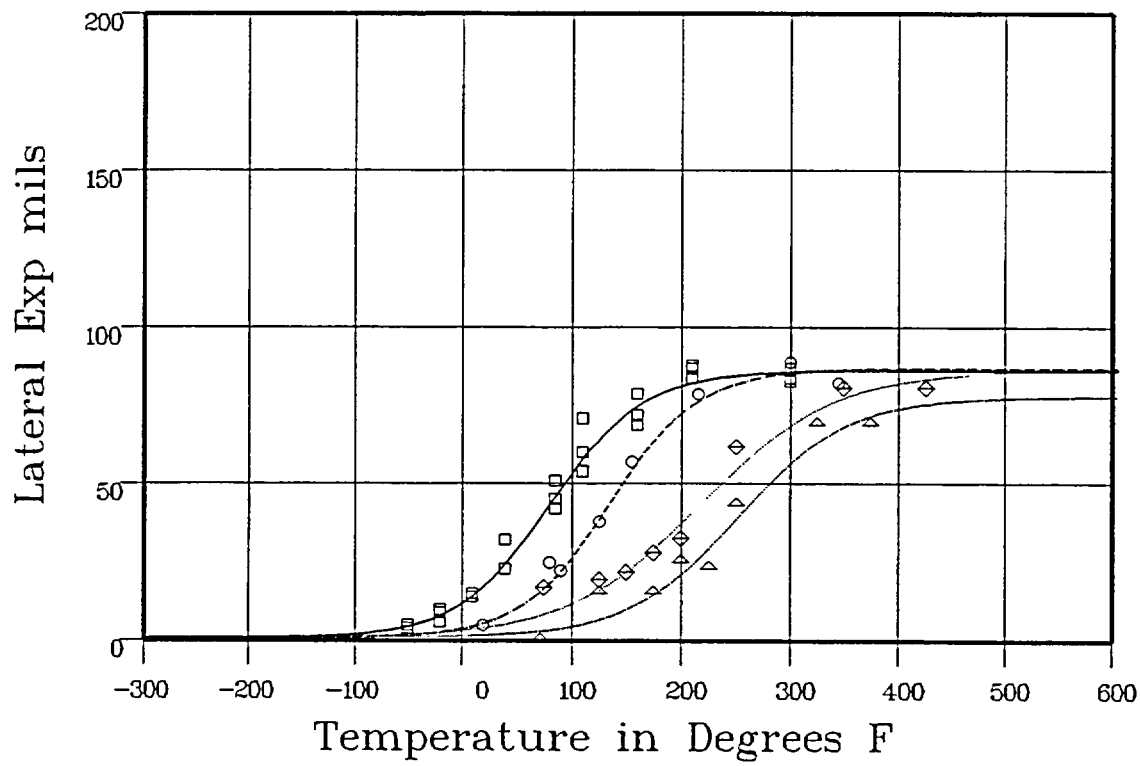
Figure 5-13 Charpy V-Notch Impact Energy vs. Temperature for Surry Unit 2 Reactor Vessel Correlation Monitor Material

SURVEILLANCE PROGRAM MONITOR MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 11:03:00 on 07-15-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	86.31	0	58.96	0
2	0	86.95	.64	113.61	54.65
3	0	86.5	.18	188.47	129.51
4	0	78.05	-8.26	235.59	176.63



Curve Legend

1 □——— 2 ○---------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Orl. Heat#
1	SU2	UNIRR	SRM SA533B1	LT HSST PLATE 02
2	SU2	X	SRM SA533B1	LT HSST PLATE 02
3	SU2	V	SRM SA533B1	LT HSST PLATE 02
4	SU2	Y	SRM SA533B1	LT HSST PLATE 02

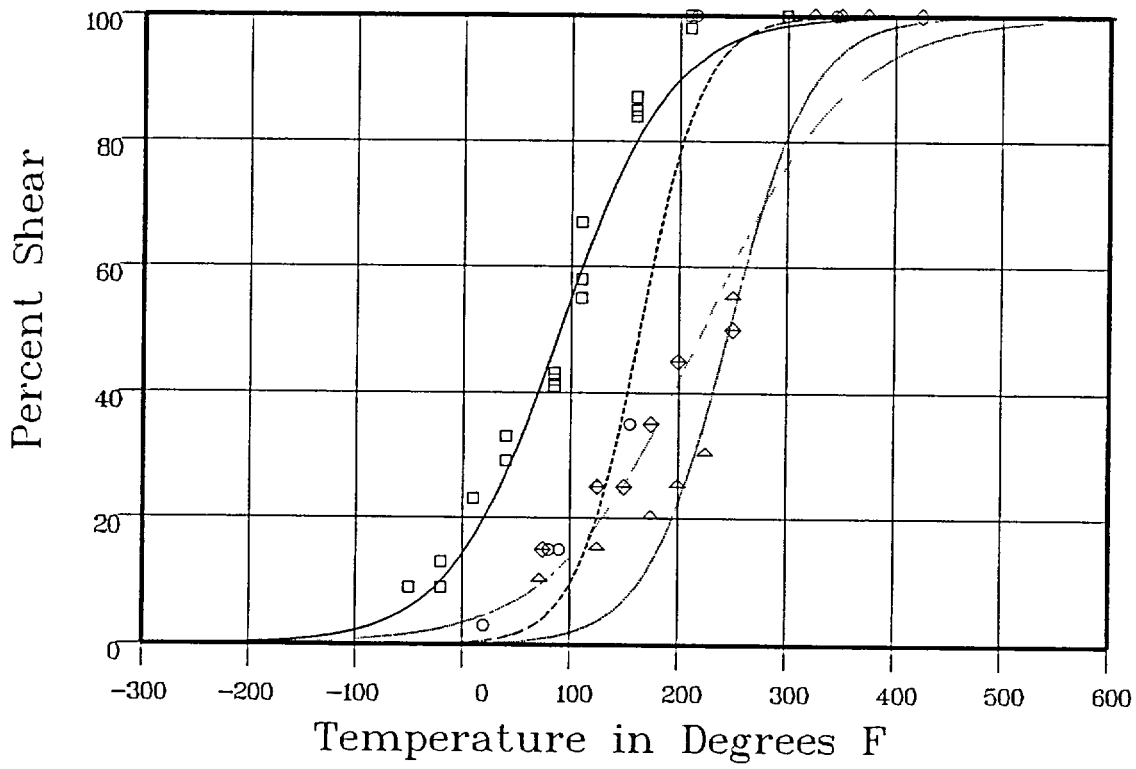
Figure 5-14 Charpy V-Notch Lateral Expansion vs. Temperature for Surry Unit 2 Reactor Vessel Correlation Monitor Material

SURVEILLANCE PROGRAM MONITOR MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 12:47:42 on 07-15-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	85.54	0
2	0	159.37	73.82
3	0	217.54	131.99
4	0	241.87	156.32



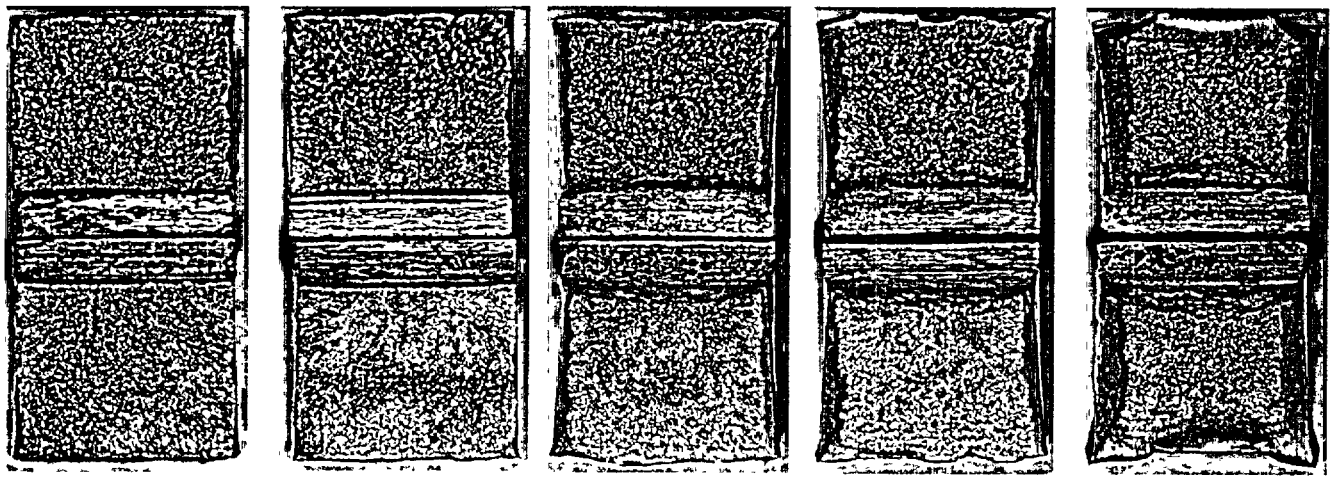
Curve Legend

1 ☐——— 2 ○---------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori	Heat#
1	SU2	UNIRR	SRM SA533B1	LT	HSST PLATE 02
2	SU2	X	SRM SA533B1	LT	HSST PLATE 02
3	SU2	V	SRM SA533B1	LT	HSST PLATE 02
4	SU2	Y	SRM SA533B1	LT	HSST PLATE 02

Figure 5-15 Charpy V-Notch Percent Shear vs. Temperature for Surry Unit 2 Reactor Vessel Correlation Monitor Material



V70, 0°F V61, 10°F V65, 100°F V64, 125°F V68, 175°F

V69, 200°F V66, 250°F V67, 300°F V63, 325°F V62, 350°F

Figure 5-16 Charpy Impact Specimen Fracture Surfaces for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Longitudinal Orientation)

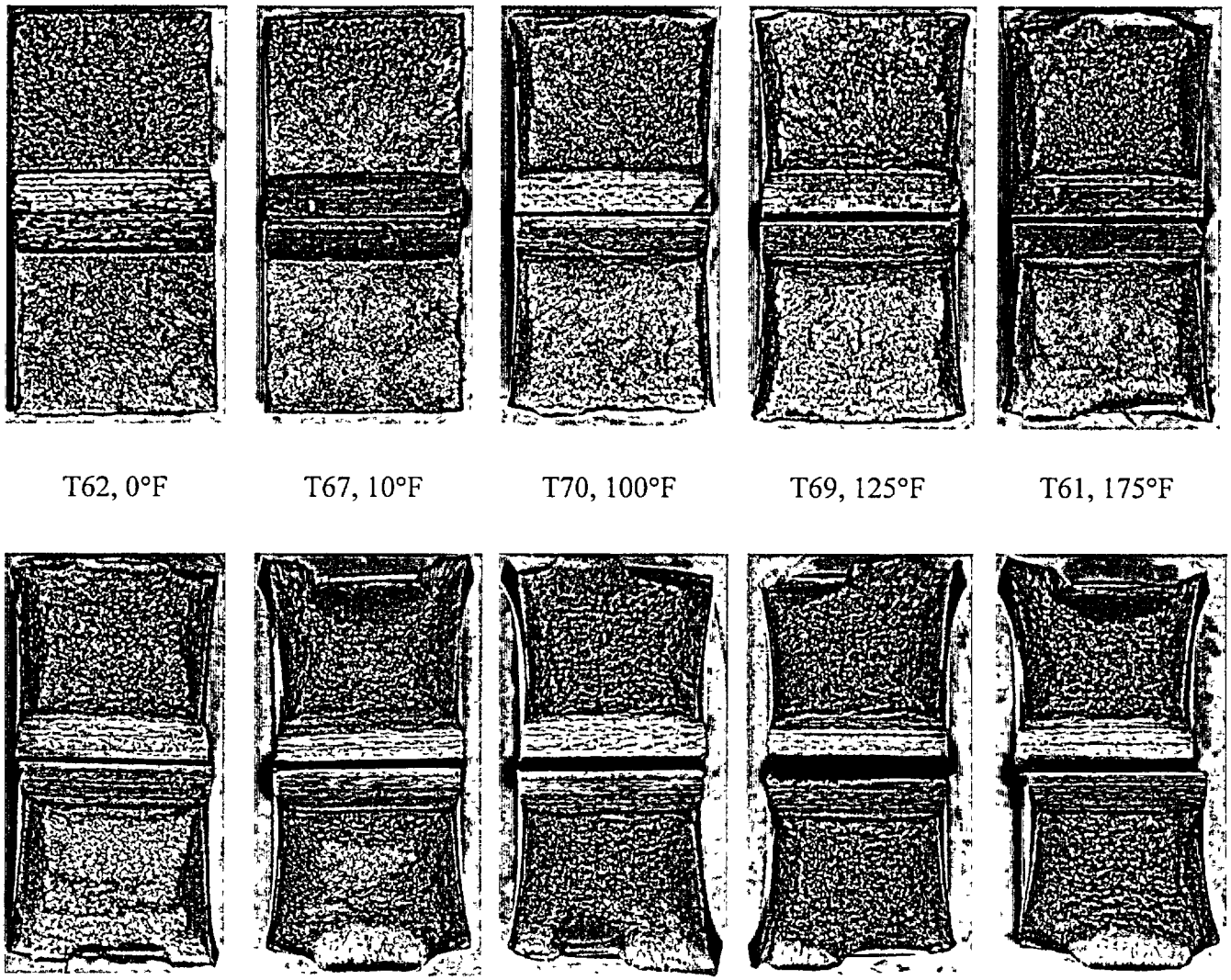


Figure 5-17 Charpy Impact Specimen Fracture Surfaces for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Transverse Orientation)

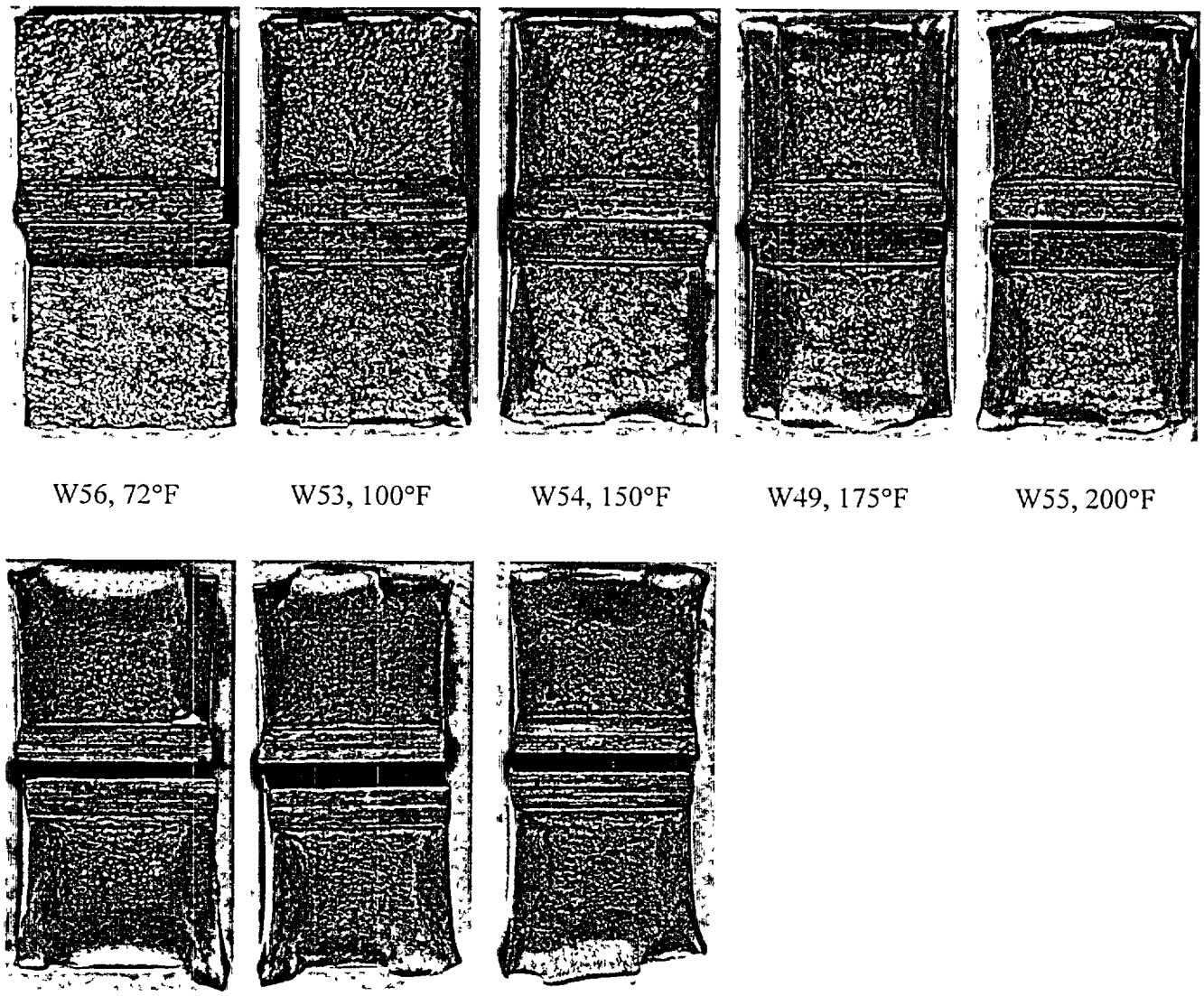


Figure 5-18 Charpy Impact Specimen Fracture Surfaces for Surry Unit 2 Reactor Vessel Weld Metal

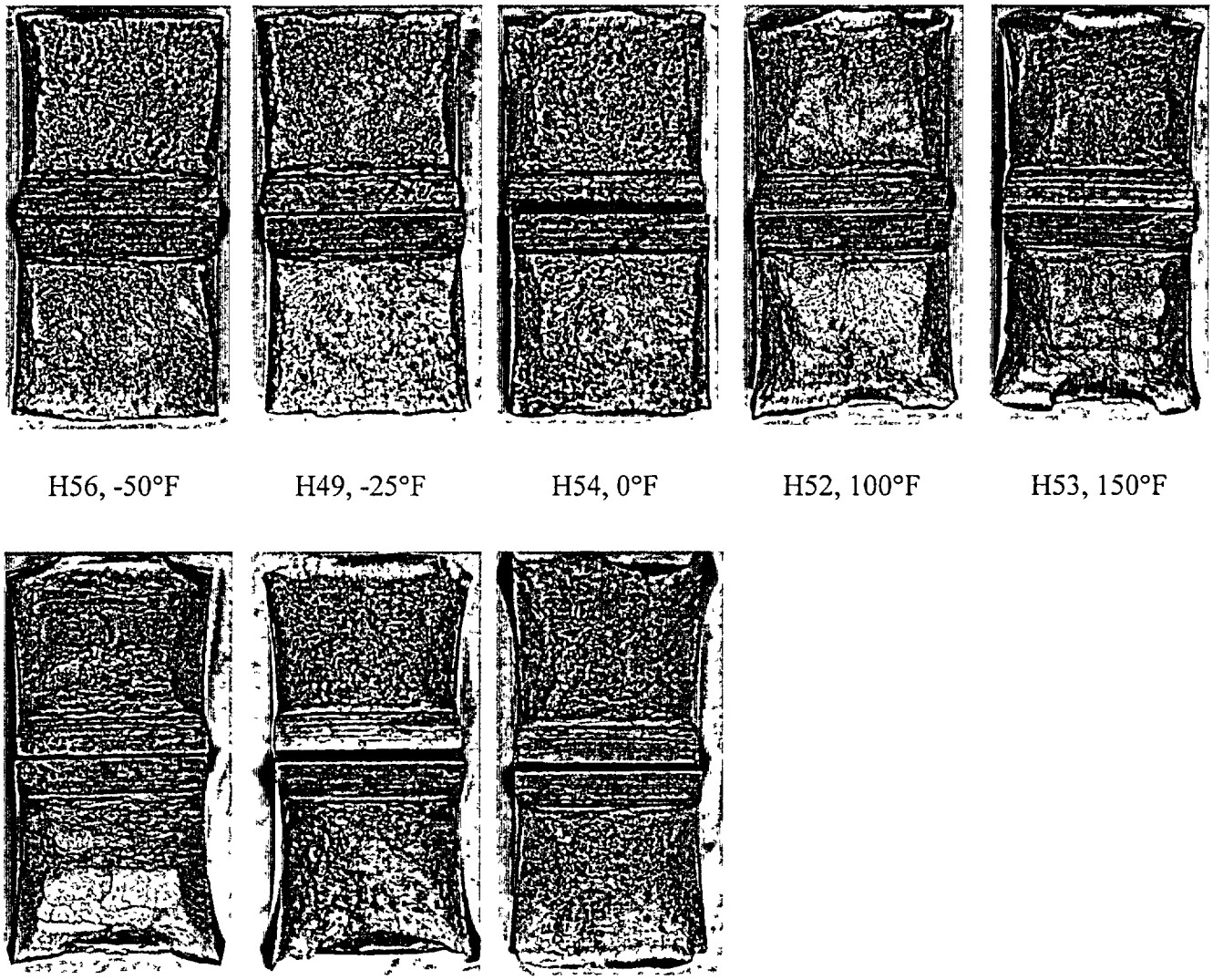


Figure 5-19 Charpy Impact Specimen Fracture Surfaces for Surry Unit 2 Reactor Vessel Heat-Affected-Zone Metal

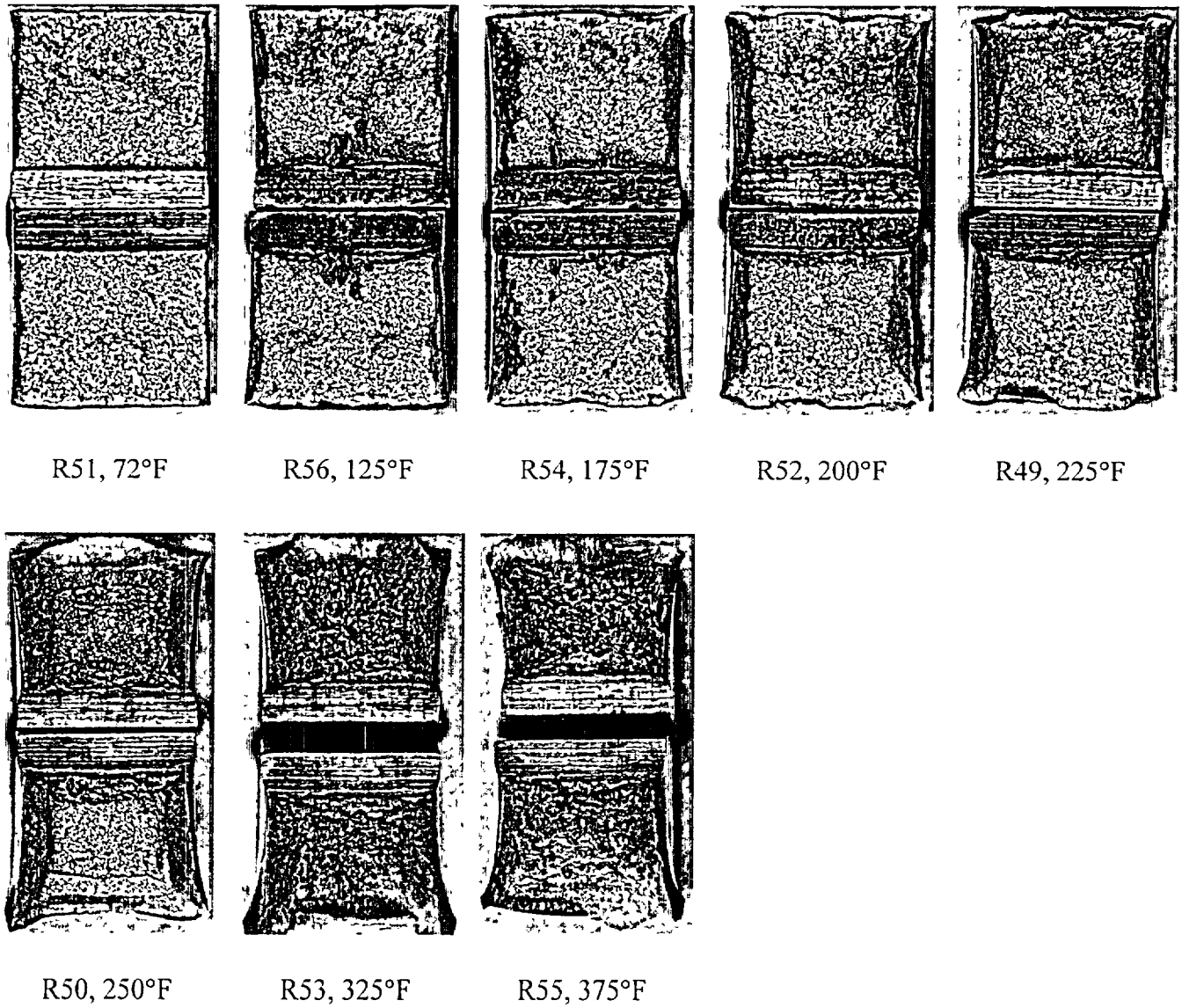


Figure 5-20 Charpy Impact Specimen Fracture Surfaces for Surry Unit 2 Reactor Vessel Correlation Monitor Material

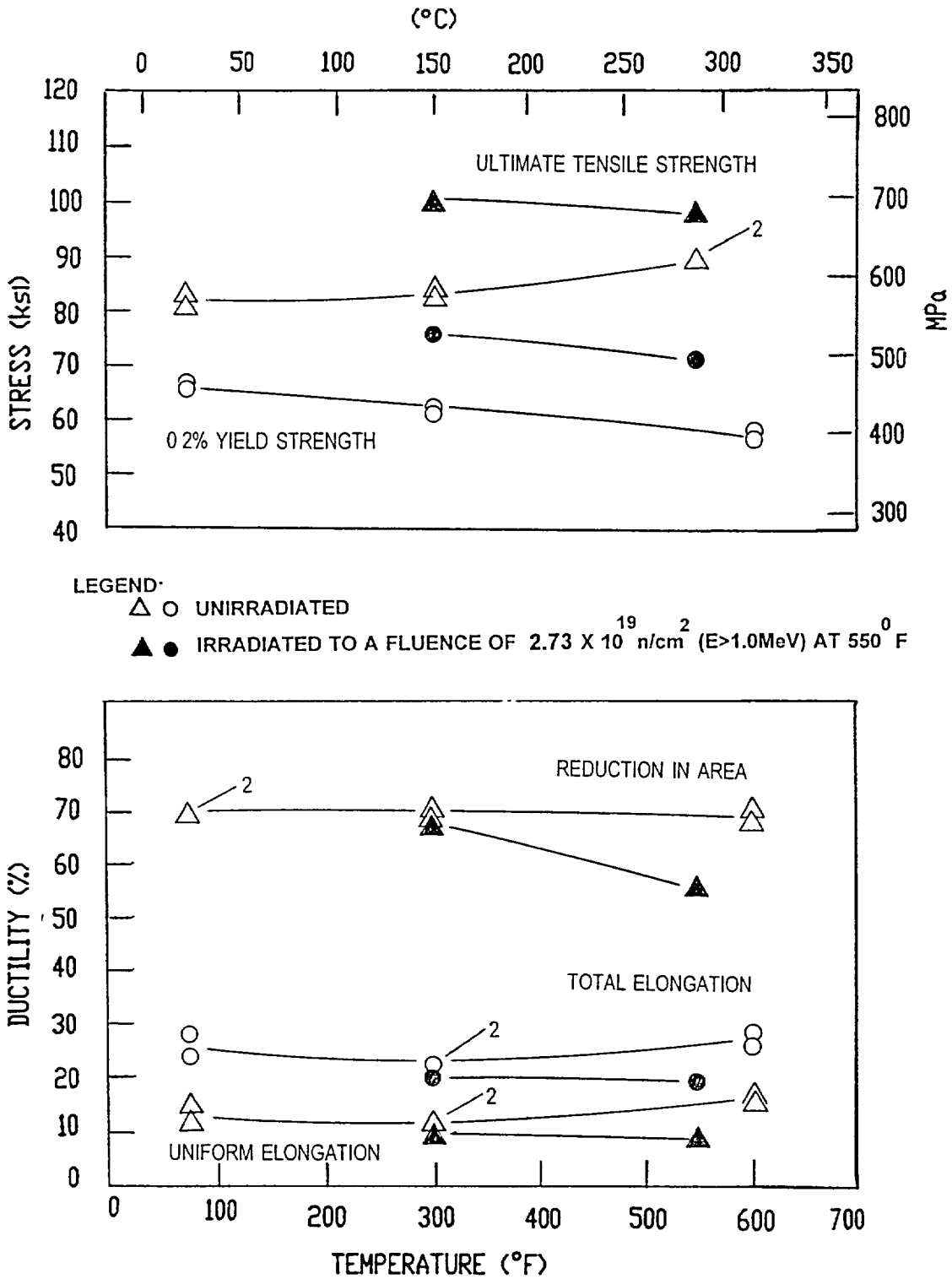


Figure 5-21 Tensile Properties for Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Longitudinal Orientation)

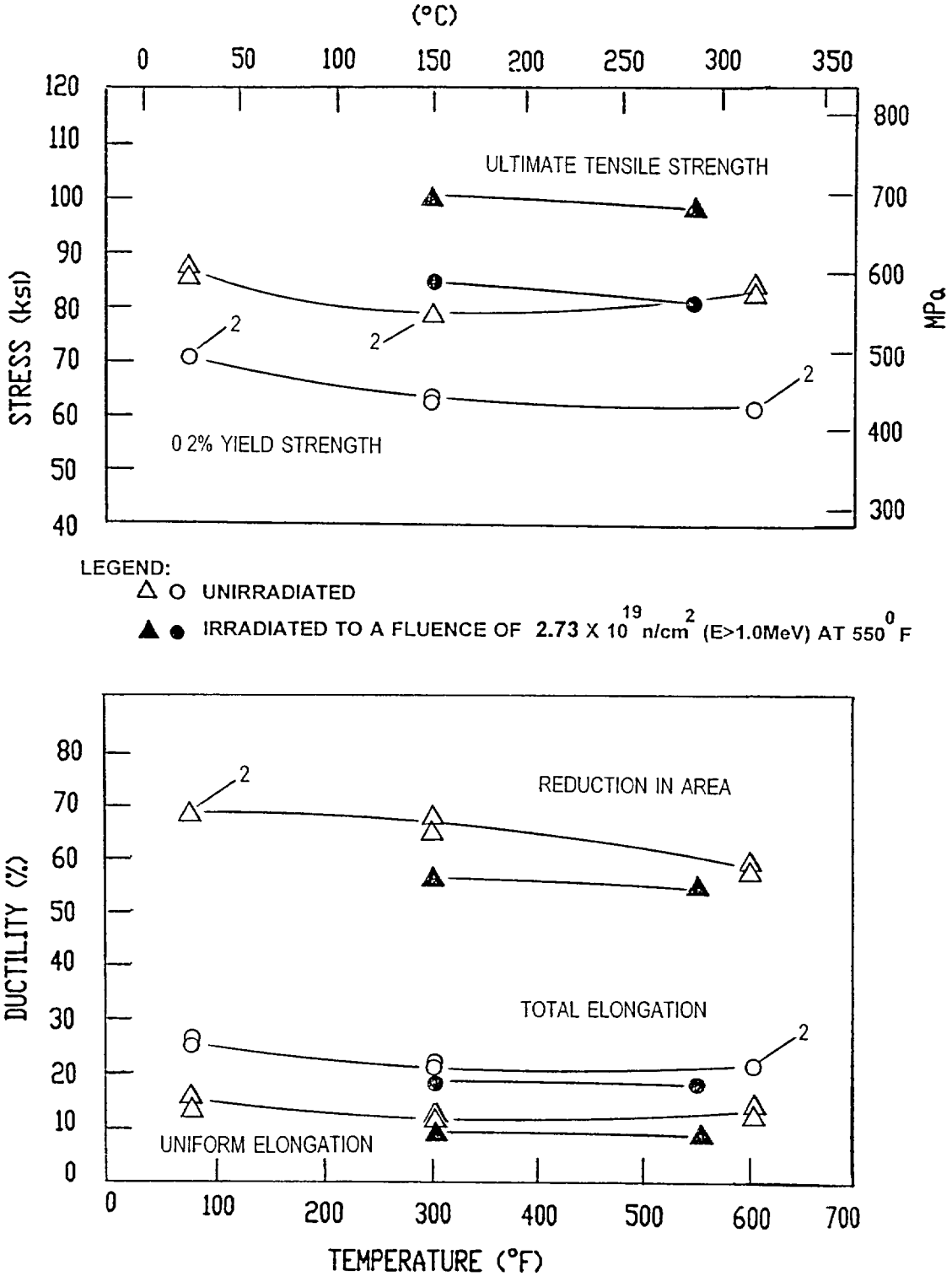
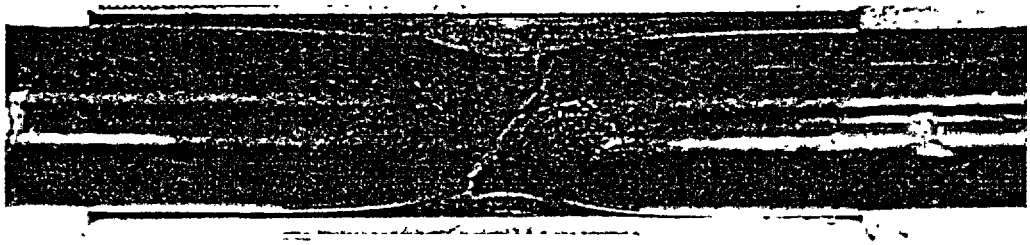


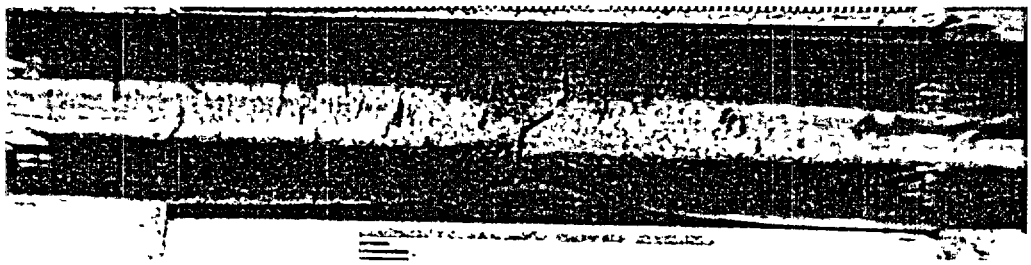
Figure 5-22 Tensile Properties for Surry Unit 2 Reactor Vessel Weld Metal

Specimen V5 Tested at 300°F

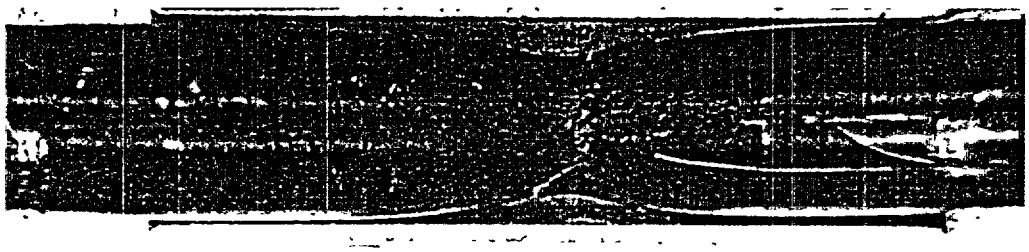


Specimen V6 Tested at 550°F

Figure 5-23 Fractured Tensile Specimens from Surry Unit 2 Reactor Vessel Lower Shell Plate C4339-1 (Longitudinal Orientation)

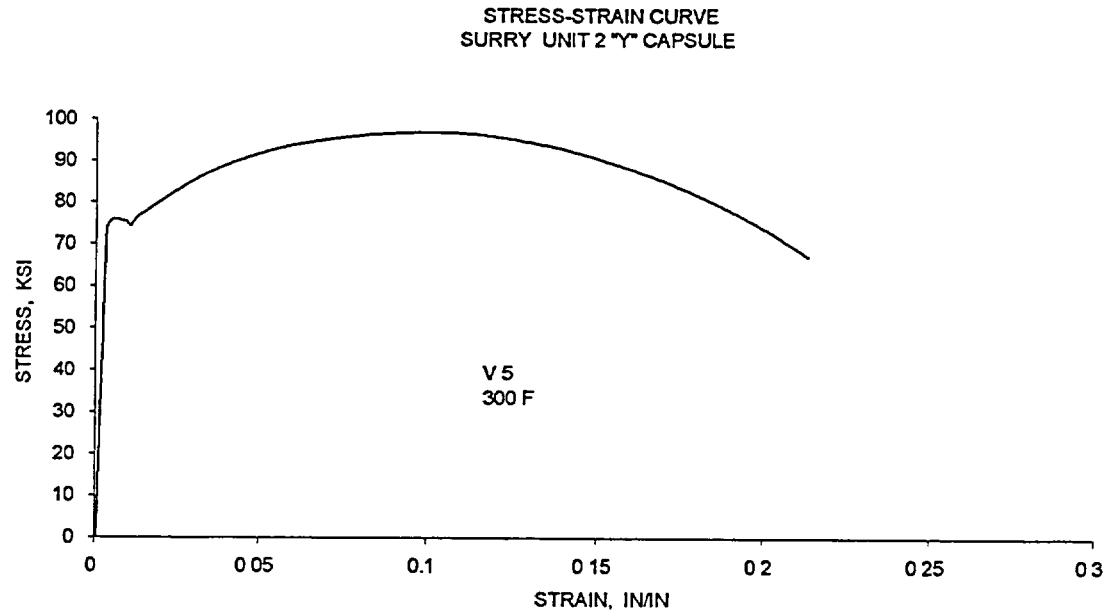


Specimen W13 Tested at 300°F



Specimen W14 Tested at 550°F

Figure 5-24 Fractured Tensile Specimens from Surry Unit 2 Reactor Vessel Weld Metal



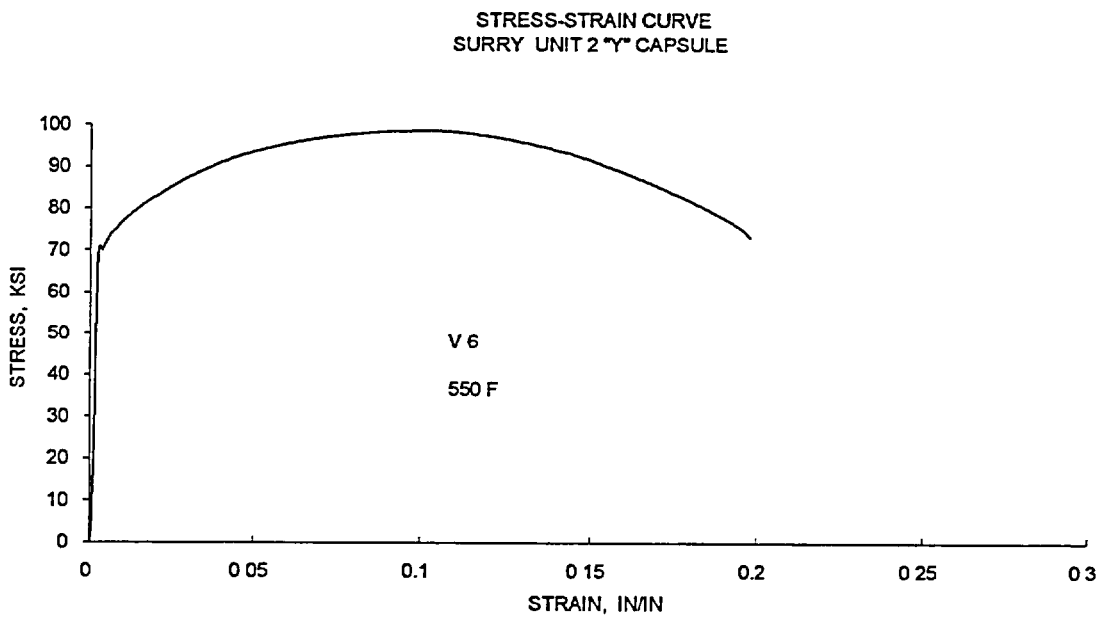
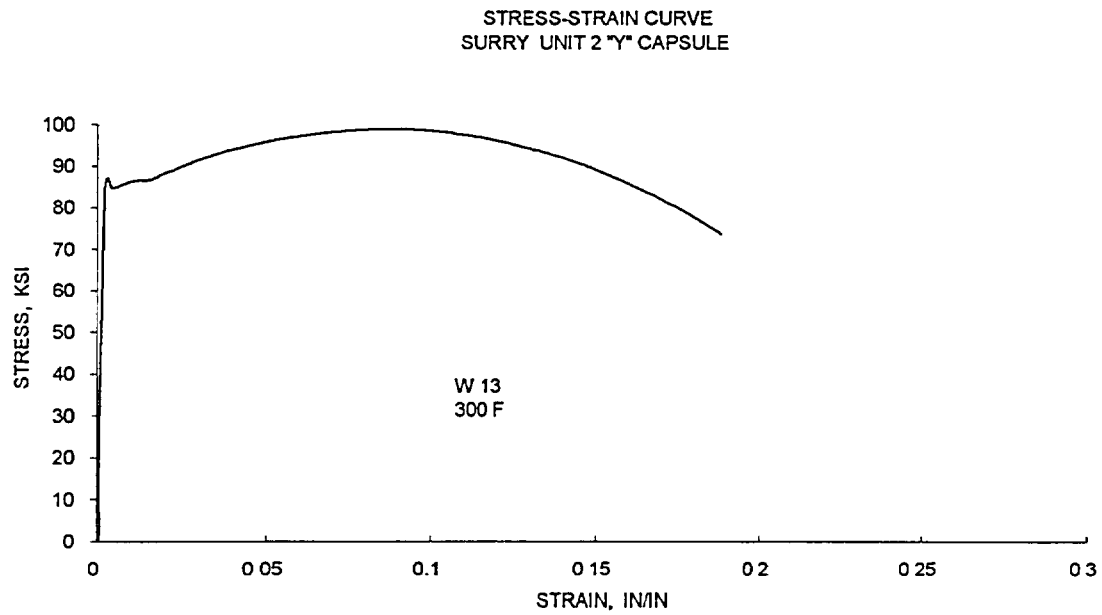


Figure 5-25 Engineering Stress-Strain Curves for Surry Unit 2 Lower Shell Plate C4339-1 Tensile Specimens V5 and V6 (Longitudinal Orientation)



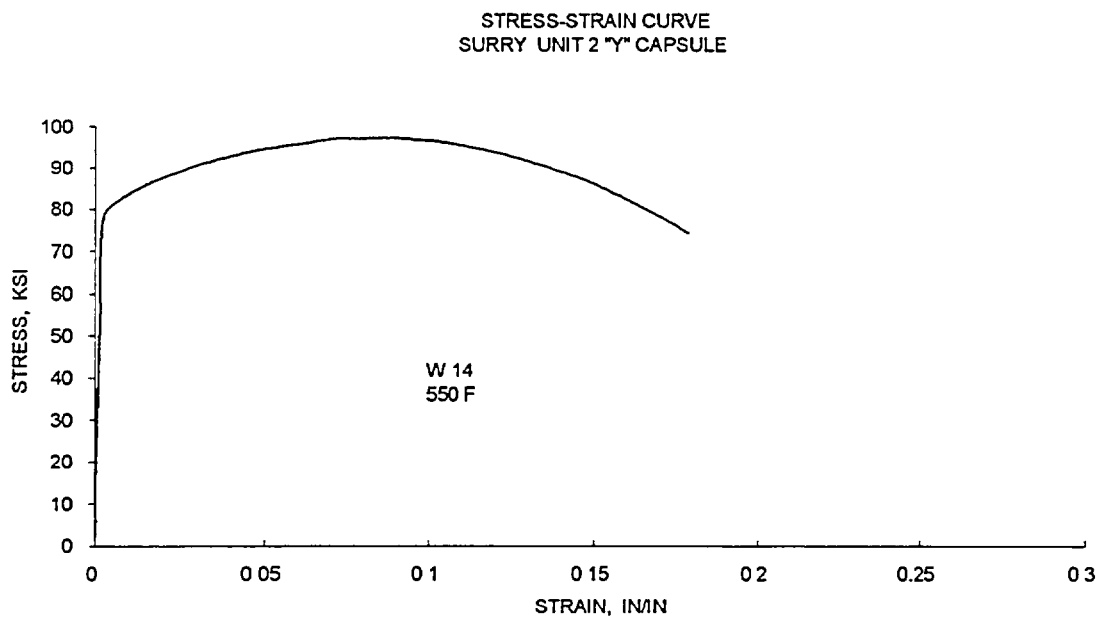


Figure 5-26 Engineering Stress-Strain Curves for Weld Metal Tensile Specimens W13 and W14

6 RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 INTRODUCTION

This section describes a discrete ordinates S_n transport analysis performed for the Surry Unit 2 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 Y, withdrawn at the end of the seventeenth plant operating cycle, is provided. In addition, to provide an up-to-date data base applicable to the Surry Unit 2 reactor, the sensor sets from the previously withdrawn capsules (X, W, V, and S) were re-analyzed using the current dosimetry evaluation methodology. These dosimetry updates are presented in Appendix D of this report. Comparisons of the results from all five of 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).

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 data base 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 D 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."[19] 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.[20] The specific calculational methods applied are also consistent with those described in WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology."[21]

6.2 DISCRETE ORDINATES ANALYSIS

A plan view of the Surry Unit 2 reactor geometry at the core midplane is shown in Figure 4-1. Eight irradiation capsules attached to the thermal shield are included in the reactor design to constitute the reactor vessel surveillance program. The capsule holders are located at azimuthal angles of 165° and 285° (15° from the core cardinal axes), 65°, 245° and 295° (25° from the core cardinal axes), 55° and 305° (35° from the core cardinal axes), and at 45° as shown in Figure 4-1. The stainless steel specimen containers are nominally 1 inch square and 56 inches in height The containers are positioned axially such that the test specimens are centered on the core midplane.

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 core barrel and the reactor vessel. In order to determine the neutron environment at the test specimen locations, the capsules themselves must be included in the analytical model

The fast neutron exposure evaluations for the Surry Unit 2 reactor vessel and surveillance capsules were based on a series of fuel cycle specific forward transport calculations that were combined using the following three-dimensional flux synthesis technique:

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

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 Surry Unit 2.

For the Surry Unit 2 transport calculations, the r,θ model depicted in Figure 6-1 was utilized since the reactor is octant symmetric (with the exception of the surveillance capsules). This r,θ model includes the core, the reactor internals and core barrel, explicit representations of the surveillance capsules at 15°, 25°, 35°, and 45°, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the shield tank external to the pressure vessel. This r,θ model was utilized in the synthesis procedure to perform the surveillance capsule dosimetry evaluations and subsequent comparisons with calculated results, in addition to calculating the maximum neutron exposure levels at the pressure vessel wall. In developing this analytical model, nominal design dimensions were employed for the various structural components.

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 156 radial by 83 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 Surry Unit 2 calculations (see Figure 6-2) extended radially from the centerline of the reactor core out to a location interior to the shield tank and over an axial span from an elevation 1-foot below the active fuel to approximately 1-foot 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 formers 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 148 radial by 90 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 148 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 supplied by Dominion Generation for each operating cycle of Surry Unit 2. The data used in the calculations 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[22] and the BUGLE-96 cross-section library.[23] 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. 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 irradiation capsules comprising the reactor vessel materials surveillance program

In the case of the Surry Unit 2 surveillance program, eight capsules were initially inserted into the reactor prior to plant startup as follows:

Capsules X and V 15°
Capsules U, W, and Y 25°
Capsules T and Z 35°
Capsule S 45°

At the conclusion of cycle 12, Capsule Y was moved from the 25° to the 15° azimuthal position and Capsule Z was repositioned from the 35° to the 25° azimuth Likewise, at the conclusion of cycle 17, Capsule T was moved from the 35° to the 15° holder location. The irradiation history of these eight surveillance caspules is summarized as follows:

Capsule	Location	Irradiation History
X	15°	Cycle 1 (Withdrawn)
W	25°	Cycles 1-4 (Withdrawn)
V	15°	Cycles 1-8 (Withdrawn)
S	45°	Cycles 1-13 (Withdrawn)
Y	25°/15°	Cycles 1-12 at 25°; Cycles 13-17 at 15° (Withdrawn)
U	25°	Cycles 1-17 (In reactor)
T	35°/15°	Cycles 1-17 at 35°; Future cycles at 15° (In reactor)
Z	35°/25°	Cycles 1-12 at 35°; Cycles 13-17 at 25° (In reactor)

The data listed in Table 6-1 provides the exposure history of all capsules located at each of the four azimuthally symmetric holder positions (15°, 25°, 35°, and 45°) throughout the entire course of irradiation. In addition, based on these irradiation histories, composite exposure projections for Capsules Y, T, and Z were also computed and are likewise included in Table 6-1. The results given in Table 6-1 are representative of the axial midplane of the active core and establish the calculated exposure of the surveillance capsules withdrawn to date as well as projected into the future.

In Table 6-2, cycle specific maximum integrated neutron exposures, expressed in terms of both neutron fluence (E > 1.0 MeV) and dpa, are given at the pressure vessel clad base metal interface at azimuthal angles of 0°, 15°, 30°, and 45° relative to the core major axis. These data are representative of the axial location of the maximum neutron exposure of the pressure vessel.

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 seventeenth operating fuel cycle as well as projections to 54 effective full power years of operation. The projections were based on the assumption that the reactor power level and spatial power distribution averaged over cycles 11 through 17 was representative of future operation. The average relative radial power distribution used in performing these projections is summarized as follows:

	8	9	10	11	12	13	14	15
H	1.111							
J	1.155	1.149						
K	1.174	1.267	1.224					
L	1.288	1.227	1.342	1.228				
N	1.185	1.333	1.236	1.272	0 762			
N	1.254	1.157	1.200	1.008	0.387			
P	0.846	0.995	0.548	0.324				
R	0.343	0.277						

As depicted, the average relative radial power distribution is shown in the southeast quadrant of the reactor.

Based on this average core power distribution, the maximum calculated fast neutron flux (E > 1.0 MeV) incident on the Surry Unit 2 reactor pressure vessel was 2.53e+10 n/cm^2 and occurred along the 0 degree azimuth. The contribution of individual fuel assemblies to that overall maximum is summarized as follows:

Fast Neutron Flux (E > 1.0 MeV) per Unit Assembly Power

	8	9	10	11	12	13	14	15
H	2.00e+04							
J	5.80e+04	8.34e+04						
K	2.52e+05	5.76e+05	2.27e+05					
L	1.53e+06	3.59e+06	1.61e+06	5.26e+05				
N	1.23e+07	2.51e+07	1.21e+07	4.05e+06	1.03e+06			
N	1.42e+08	2.56e+08	1.00e+08	2.96e+07	6.02e+06			
P	2.04e+09	2.92e+09	8.11e+08	2.25e+08				
R	2.85e+10	3.47e+10						

The data as presented include the effects of symmetry along both the 0° and 45° axes. These importance functions can be used in a multiplicative fashion with actual future fuel cycle designs in order to estimate the impact of these future designs on the maximum vessel exposure relative to the projections based on the average power distribution.

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 one through seventeen, 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 five surveillance capsules withdrawn from the Surry Unit 2 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 Surry Unit 2 reactor

Updated lead factors for the Surry Unit 2 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 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, the lead factors correspond to the calculated fluence values at the end of cycle seventeen.

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 both direct and least squares evaluation comparisons, is documented in Appendix D.

The direct comparison of measured versus calculated fast neutron threshold reaction rates for the sensors from Capsule Y, that was withdrawn from Surry Unit 2 at the end of the seventeenth fuel cycle, is summarized below.

Reaction	Reaction Rates (rps/atom)		M/C
	Measured	Calculated	Ratio
$^{63}Cu(n,\alpha)^{60}Co$ (Cd)	3.21E-17	3.27E-17	0.98
$^{54}Fe(n,p)^{54}Mn$	3.04E-15	3.35E-15	0.91
$^{238}U(n,f)^{137}Cs$ (Cd)	1.36E-14	1.54E-15	0.88
$^{237}Np(n,f)^{137}Cs$ (Cd)	1.28E-13	1.11e-13	1.15
		Average:	0.98
		Sample % Standard Deviation:	12.4

The measured-to-calculated (M/C) reaction rate ratios for the Capsule Y threshold reactions range from 0.88 to 1.15, and the average M/C ratio is 0.98 ± 12.4% (1σ).

The intent of this comparison along with the additional M/C data provided in Appendix D is to demonstrate that the plant specific dosimetry from the Surry Unit 2 surveillance program supports the overall calculational uncertainty estimate described in Section 6.4. In addition, based on discussions in Regulatory Guide 1.190, it is expected that fast neutron measurements fall within ± 20% of the qualified calculation. The full set of threshold foil M/C data for the five surveillance capsules withdrawn from Surry Unit 2 shows this to be true. As a result, these comparisons are considered to validate the current analytical results provided in Section 6.2.

6.4 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the Surry Unit 2 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 from 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 Surry unit 2 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 Surry unit 2 analysis was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with Surry Unit 2 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 Surry Unit 2 analytical model based on the measured plant dosimetry is completely described in Appendix D.

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 3.

	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 D support these uncertainty assessments for Surry Unit 2.The plant specific measurement comparisons described in Appendix D support these uncertainty assessments for Surry Unit 2.

Figure 6-1

Surry Unit 2 r,θ Reactor Geometry at the Core Midplane

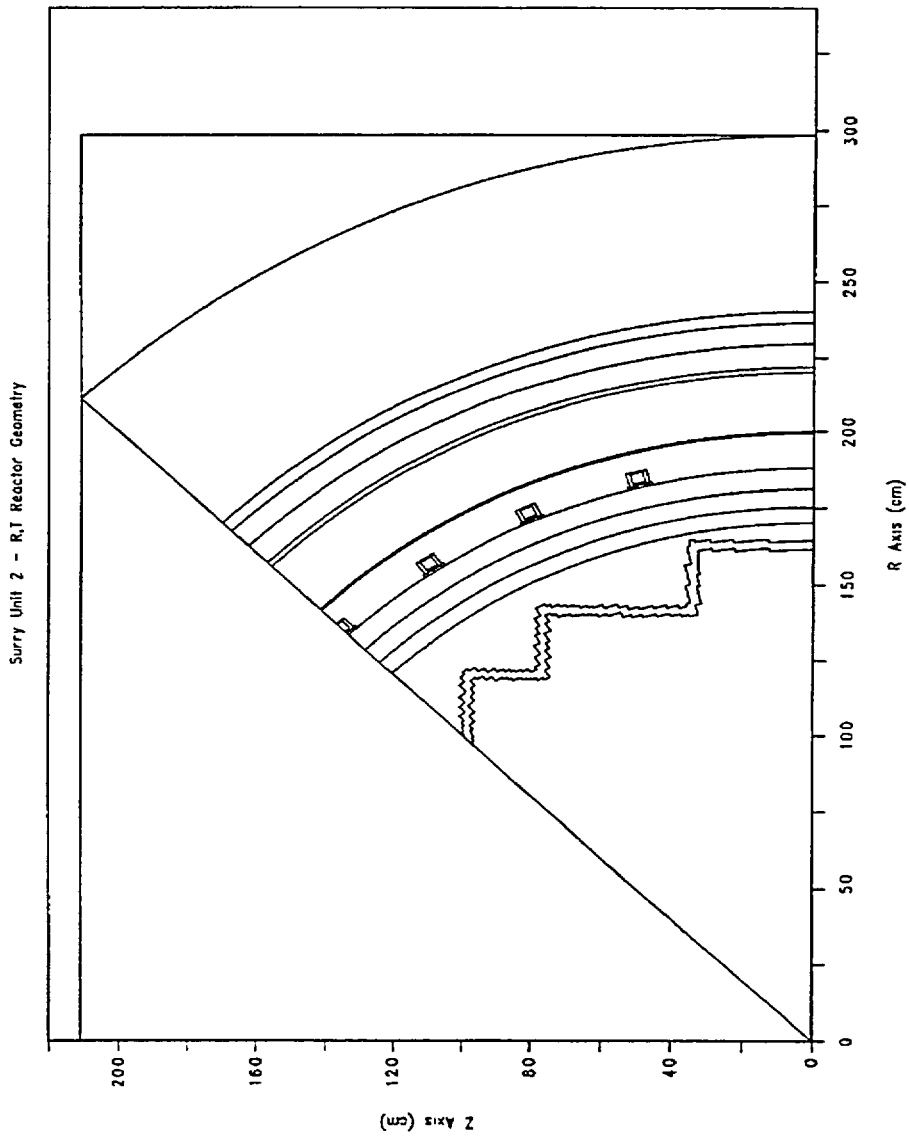


Figure 6-2

Surry Unit 2 r,z Reactor Geometry

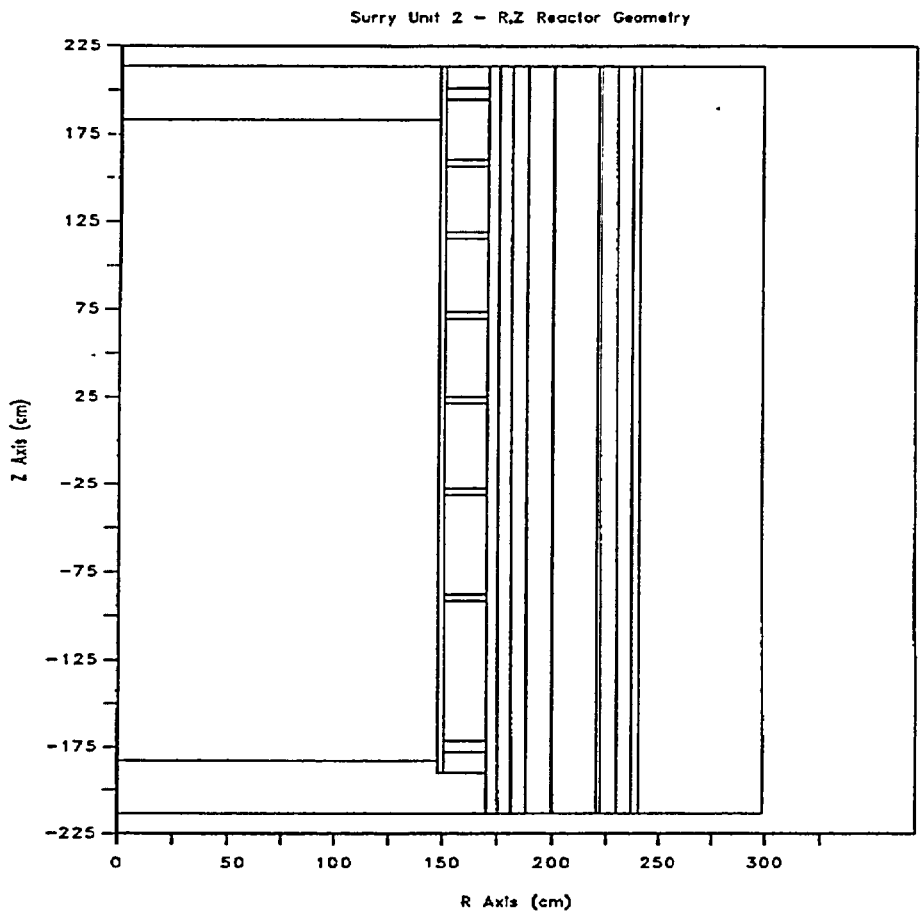


Table 6-1

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

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Flux (E > 1.0 MeV) [n/cm²-s]			
				15°	25°	35°	45°
1	3.71E+07	3.71E+07	1.2	8.01E+10	5.26E+10	3 57E+10	2 79E+10
2	2 28E+07	5.98E+07	1.9	8.24E+10	5.45E+10	3.80E+10	3.03E+10
3	2 37E+07	8.36E+07	2.6	8.06E+10	5.56E+10	3.85E+10	3.01E+10
4	3 48E+07	1.18E+08	3 8	8.09E+10	5.32E+10	3.62E+10	2.81E+10
5	3.55E+07	1.54E+08	4 9	6.58E+10	4.31E+10	3.00E+10	2.44E+10
6	4.07E+07	1.95E+08	6 2	6.46E+10	4.37E+10	2.96E+10	2.34E+10
7	3.77E+07	2.32E+08	7.4	6.63E+10	4.01E+10	2.82E+10	2.29E+10
8	3.40E+07	2.66E+08	8.4	5.49E+10	3.79E+10	2.70E+10	2.20E+10
9	4.01E+07	3.06E+08	9.7	5.34E+10	3.57E+10	2.46E+10	1.96E+10
10	3.82E+07	3.45E+08	10.9	5.57E+10	3.80E+10	2.60E+10	1.99E+10
11	4.76E+07	3.92E+08	12.4	4.50E+10	3.09E+10	2.30E+10	1.89E+10
12	4.75E+07	4.40E+08	13.9	4.72E+10	3.18E+10	2.27E+10	1.83E+10
13	3.25E+07	4.72E+08	15 0	4 98E+10	3.19E+10	2.10E+10	1.65E+10
14	4.01E+07	5.12E+08	16.2	4 55E+10	3.23E+10	2.28E+10	1 80E+10
15	4.47E+07	5.57E+08	17.7	4.42E+10	3.22E+10	2.36E+10	1.89E+10
16	4.09E+07	5.98E+08	18.9	4.38E+10	3.21E+10	2.31E+10	1.83E+10
17	4 15E+07	6.39E+08	20 3	4.25E+10	3.11E+10	2.18E+10	1.68E+10
Future	3 70E+08	1.01E+09	32.0	4.60E+10	3.23E+10	2.30E+10	1.84E+10
Future	5.06E+08	1.51E+09	48.0	4.60E+10	3.23E+10	2.30E+10	1.84E+10
Future	1 89E+08	1.70E+09	54.0	4.60E+10	3.23E+10	2.30E+10	1.84E+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

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Flux (E > 1.0 MeV) [n/cm^2-s]		
				Composite Capsule Y	Composite Capsule T	Composite Capsule Z
1	3.71E+07	3.71E+07	1.2	5.26E+10	3.57E+10	3.57E+10
2	2.28E+07	5.98E+07	1.9	5.45E+10	3.80E+10	3.80E+10
3	2.37E+07	8.36E+07	2.6	5.56E+10	3.85E+10	3.85E+10
4	3.48E+07	1.18E+08	3.8	5.32E+10	3.62E+10	3.62E+10
5	3.55E+07	1.54E+08	4.9	4.31E+10	3.00E+10	3.00E+10
6	4.07E+07	1.95E+08	6.2	4.37E+10	2.96E+10	2.96E+10
7	3.77E+07	2.32E+08	7.4	4.01E+10	2.82E+10	2.82E+10
8	3.40E+07	2.66E+08	8.4	3.79E+10	2.70E+10	2.70E+10
9	4.01E+07	3.06E+08	9.7	3.57E+10	2.46E+10	2.46E+10
10	3.82E+07	3.45E+08	10.9	3.80E+10	2.60E+10	2.60E+10
11	4.76E+07	3.92E+08	12.4	3.09E+10	2.30E+10	2.30E+10
12	4.75E+07	4.40E+08	13.9	3.18E+10	2.27E+10	2.27E+10
13	3.25E+07	4.72E+08	15.0	4.98E+10	2.10E+10	3.19E+10
14	4.01E+07	5.12E+08	16.2	4.55E+10	2.28E+10	3.23E+10
15	4.47E+07	5.57E+08	17.7	4.42E+10	2.36E+10	3.22E+10
16	4.09E+07	5.98E+08	18.9	4.38E+10	2.31E+10	3.21E+10
17	4.15E+07	6.39E+08	20.3	4.25E+10	2.18E+10	3.11E+10
Future	3.70E+08	1.01E+09	32.0	4.60E+10	4.60E+10	3.23E+10
Future	5.06E+08	1.51E+09	48.0	4.60E+10	4.60E+10	3.23E+10
Future	1.89E+08	1.70E+09	54.0	4.60E+10	4.60E+10	3.23E+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

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm²]			
				15°	25°	35°	45°
1	3.71E+07	3.71E+07	1 2	2.97E+18	1.95E+18	1.32E+18	1.03E+18
2	2.28E+07	5.98E+07	1.9	4.85E+18	3.19E+18	2.19E+18	1.72E+18
3	2.37E+07	8.36E+07	2.6	6.76E+18	4 51E+18	3.10E+18	2.44E+18
4	3.48E+07	1.18E+08	3.8	9.57E+18	6.36E+18	4.36E+18	3.42E+18
5	3.55E+07	1.54E+08	4.9	1 19E+19	7.90E+18	5 43E+18	4 28E+18
6	4 07E+07	1.95E+08	6.2	1.45E+19	9.67E+18	6 63E+18	5.23E+18
7	3.77E+07	2.32E+08	7.4	1.70E+19	1.12E+19	7.69E+18	6.10E+18
8	3.40E+07	2.66E+08	8 4	1.89E+19	1.25E+19	8.61E+18	6.84E+18
9	4.01E+07	3.06E+08	9.7	2.11E+19	1.39E+19	9.60E+18	7.63E+18
10	3.82E+07	3.45E+08	10.9	2 32E+19	1 54E+19	1.06E+19	8.39E+18
11	4.76E+07	3.92E+08	12.4	2 53E+19	1.68E+19	1.17E+19	9 29E+18
12	4.75E+07	4.40E+08	13.9	2.76E+19	1.83E+19	1.28E+19	1.02E+19
13	3.25E+07	4.72E+08	15.0	2.92E+19	1.94E+19	1 34E+19	1.07E+19
14	4.01E+07	5.12E+08	16.2	3.10E+19	2.07E+19	1.44E+19	1.14E+19
15	4.47E+07	5.57E+08	17.7	3.30E+19	2.21E+19	1.54E+19	1.23E+19
16	4.09E+07	5.98E+08	18.9	3.48E+19	2.34E+19	1.64E+19	1.30E+19
17	4.15E+07	6.39E+08	20.3	3 65E+19	2.73E+19	1.73E+19	1 37E+19
Future	3.70E+08	1 01E+09	32.0	5 36E+19	3 67E+19	2.58E+19	2 05E+19
Future	5.06E+08	1 51E+09	48.0	7.68E+19	5 30E+19	3.74E+19	2 98E+19
Future	1.89E+08	1.70E+09	54.0	8.55E+19	5.91E+19	4 18E+19	3.33E+19

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

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence(E > 1.0 MeV) [n/cm²]		
				Composite Capsule Y	Composite Capsule T	Composite Capsule Z
1	3.71E+07	3.71E+07	1.2	1.95E+18	1.32E+18	1.32E+18
2	2.28E+07	5.98E+07	1.9	3.19E+18	2.19E+18	2.19E+18
3	2.37E+07	8.36E+07	2.6	4.51E+18	3.10E+18	3.10E+18
4	3.48E+07	1.18E+08	3.8	6.36E+18	4.36E+18	4.36E+18
5	3.55E+07	1.54E+08	4.9	7.90E+18	5.43E+18	5.43E+18
6	4.07E+07	1.95E+08	6.2	9.67E+18	6.63E+18	6.63E+18
7	3.77E+07	2.32E+08	7.4	1.12E+19	7.69E+18	7.69E+18
8	3.40E+07	2.66E+08	8.4	1.25E+19	8.61E+18	8.61E+18
9	4.01E+07	3.06E+08	9.7	1.39E+19	9.60E+18	9.60E+18
10	3.82E+07	3.45E+08	10.9	1.54E+19	1.06E+19	1.06E+19
11	4.76E+07	3.92E+08	12.4	1.68E+19	1.17E+19	1.17E+19
12	4.75E+07	4.40E+08	13.9	1.83E+19	1.28E+19	1.28E+19
13	3.25E+07	4.72E+08	15.0	2.00E+19	1.34E+19	1.38E+19
14	4.01E+07	5.12E+08	16.2	2 18E+19	1.44E+19	1.51E+19
15	4.47E+07	5.57E+08	17.7	2.38E+19	1.54E+19	1.65E+19
16	4.09E+07	5.98E+08	18.9	2.56E+19	1.64E+19	1.78E+19
17	4.15E+07	6.39E+08	20.3	2.73E+19	1.73E+19	1.91E+19
Future	3.70E+08	1.01E+09	32.0	4.43E+19	3.43E+19	3.11E+19
Future	5.06E+08	1.51E+09	48.0	6.76E+19	5.75E+19	4.74E+19
Future	1.89E+08	1.70E+09	54.0	7.63E+19	6.62E+19	5.35E+19

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

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacement Rate [dpa/s]			
				15°	25°	35°	45°
1	3.71E+07	3 71E+07	1.2	1.34E-10	8.56E-11	5.77E-11	4.48E-11
2	2 28E+07	5.98E+07	1.9	1.38E-10	8.87E-11	6.14E-11	4.87E-11
3	2.37E+07	8.36E+07	2.6	1 35E-10	9.06E-11	6.23E-11	4.85E-11
4	3.48E+07	1.18E+08	3.8	1 35E-10	8.67E-11	5.85E-11	4.53E-11
5	3.55E+07	1.54E+08	4.9	1.10E-10	7.00E-11	4.84E-11	3.92E-11
6	4.07E+07	1.95E+08	6.2	1.08E-10	7.10E-11	4.77E-11	3.75E-11
7	3.77E+07	2.32E+08	7 4	1.11E-10	6.51E-11	4.55E-11	3.67E-11
8	3.40E+07	2 66E+08	8 4	9.14E-11	6.15E-11	4 36E-11	3.53E-11
9	4.01E+07	3.06E+08	9.7	8.89E-11	5.80E-11	3.97E-11	3.15E-11
10	3 82E+07	3.45E+08	10.9	9.27E-11	6.17E-11	4.19E-11	3.20E-11
11	4 76E+07	3.92E+08	12.4	7.48E-11	5.02E-11	3 71E-11	3 04E-11
12	4.75E+07	4.40E+08	13.9	7.85E-11	5.17E-11	3.66E-11	2.94E-11
13	3.25E+07	4.72E+08	15.0	8.28E-11	5.17E-11	3.39E-11	2 65E-11
14	4.01E+07	5.12E+08	16 2	7.56E-11	5.24E-11	3.68E-11	2.90E-11
15	4.47E+07	5.57E+08	17.7	7.34E-11	5.24E-11	3.80E-11	3.04E-11
16	4.09E+07	5.98E+08	18.9	7.26E-11	5 21E-11	3.73E-11	2.94E-11
17	4.15E+07	6.39E+08	20 3	7.04E-11	5.05E-11	3.51E-11	2.70E-11
Future	3.70E+08	1.01E+09	32.0	7 64E-11	5.24E-11	3.71E-11	2.95E-11
Future	5.06E+08	1 51E+09	48.0	7.64E-11	5.24E-11	3.71E-11	2.95E-11
Future	1.89E+08	1.70E+09	54.0	7.64E-11	5.24E-11	3.71E-11	2.95E-11

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

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacement Rate [dpa/s]		
				Composite Capsule Y	Composite Capsule T	Composite Capsule Z
1	3.71E+07	3.71E+07	1.2	8.56E-11	5.77E-11	5.77E-11
2	2.28E+07	5.98E+07	1.9	8.87E-11	6.14E-11	6.14E-11
3	2.37E+07	8.36E+07	2.6	9.06E-11	6.23E-11	6.23E-11
4	3.48E+07	1.18E+08	3.8	8.67E-11	5.85E-11	5.85E-11
5	3.55E+07	1.54E+08	4.9	7.00E-11	4.84E-11	4.84E-11
6	4.07E+07	1.95E+08	6.2	7.10E-11	4.77E-11	4.77E-11
7	3.77E+07	2.32E+08	7.4	6.51E-11	4.55E-11	4.55E-11
8	3.40E+07	2.66E+08	8.4	6.15E-11	4.36E-11	4.36E-11
9	4.01E+07	3.06E+08	9.7	5.80E-11	3.97E-11	3.97E-11
10	3.82E+07	3.45E+08	10.9	6.17E-11	4.19E-11	4.19E-11
11	4.76E+07	3.92E+08	12.4	5.02E-11	3.71E-11	3.71E-11
12	4.75E+07	4.40E+08	13.9	5.17E-11	3.66E-11	3.66E-11
13	3.25E+07	4.72E+08	15.0	8.28E-11	3.39E-11	5.17E-11
14	4 01E+07	5.12E+08	16 2	7.56E-11	3.68E-11	5.24E-11
15	4.47E+07	5.57E+08	17.7	7.34E-11	3.80E-11	5.24E-11
16	4.09E+07	5.98E+08	18.9	7.26E-11	3.73E-11	5.21E-11
17	4.15E+07	6.39E+08	20.3	7.04E-11	3.51E-11	5.05E-11
Future	3.70E+08	1.01E+09	32.0	7.64E-11	7.64E-11	5.24E-11
Future	5.06E+08	1.51E+09	48.0	7.64E-11	7.64E-11	5.24E-11
Future	1.89E+08	1.70E+09	54.0	7.64E-11	7.64E-11	5.24E-11

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

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]			
				15°	25°	35°	45°
1	3.71E+07	3.71E+07	1.2	4.97E-03	3.17E-03	2.14E-03	1.66E-03
2	2.28E+07	5.98E+07	1.9	8.10E-03	5.19E-03	3.54E-03	2.77E-03
3	2.37E+07	8.36E+07	2 6	1.13E-02	7.35E-03	5.02E-03	3.92E-03
4	3.48E+07	1.18E+08	3 8	1.60E-02	1.04E-02	7.05E-03	5.49E-03
5	3.55E+07	1.54E+08	4 9	1.99E-02	1.29E-02	8.77E-03	6.89E-03
6	4.07E+07	1.95E+08	6 2	2.43E-02	1.57E-02	1.07E-02	8.41E-03
7	3.77E+07	2.32E+08	7.4	2.85E-02	1.82E-02	1.24E-02	9.80E-03
8	3.40E+07	2.66E+08	8.4	3.16E-02	2.03E-02	1.39E-02	1.10E-02
9	4.01E+07	3.06E+08	9.7	3.51E-02	2.26E-02	1.55E-02	1.23E-02
10	3.82E+07	3.45E+08	10.9	3.87E-02	2.50E-02	1.71E-02	1.35E-02
11	4.76E+07	3.92E+08	12.4	4.22E-02	2.74E-02	1.89E-02	1 49E-02
12	4.75E+07	4.40E+08	13.9	4.60E-02	2.98E-02	2.06E-02	1.63E-02
13	3.25E+07	4.72E+08	15.0	4.87E-02	3.15E-02	2.17E-02	1 72E-02
14	4.01E+07	5.12E+08	16.2	5.17E-02	3.36E-02	2.32E-02	1.84E-02
15	4.47E+07	5.57E+08	17.7	5.50E-02	3.59E-02	2.49E-02	1.97E-02
16	4.09E+07	5.98E+08	18.9	5.79E-02	3.81E-02	2.64E-02	2.09E-02
17	4.15E+07	6.39E+08	20.3	6.09E-02	4.02E-02	2.79E-02	2.20E-02
Future	3.70E+08	1.01E+09	32.0	8.92E-02	5.96E-02	4.16E-02	3.30E-02
Future	5.06E+08	1.51E+09	48.0	1.28E-01	8.60E-02	6.04E-02	4.78E-02
Future	1.89E+08	1.70E+09	54.0	1.42E-01	9.59E-02	6.74E-02	5.34E-02

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

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]		
				Composite Capsule Y	Composite Capsule T	Composite Capsule Z
1	3.71E+07	3.71E+07	1.2	3.17E-03	2.14E-03	2.14E-03
2	2.28E+07	5.98E+07	1.9	5.19E-03	3.54E-03	3.54E-03
3	2.37E+07	8.36E+07	2.6	7.35E-03	5.02E-03	5.02E-03
4	3.48E+07	1.18E+08	3.8	1.04E-02	7.05E-03	7.05E-03
5	3.55E+07	1.54E+08	4.9	1.29E-02	8.77E-03	8.77E-03
6	4.07E+07	1.95E+08	6.2	1.57E-02	1.07E-02	1.07E-02
7	3.77E+07	2.32E+08	7.4	1.82E-02	1.24E-02	1.24E-02
8	3.40E+07	2.66E+08	8.4	2.03E-02	1.39E-02	1.39E-02
9	4.01E+07	3.06E+08	9.7	2.26E-02	1.55E-02	1.55E-02
10	3.82E+07	3.45E+08	10.9	2.50E-02	1.71E-02	1.71E-02
11	4.76E+07	3.92E+08	12.4	2.74E-02	1.89E-02	1.89E-02
12	4.75E+07	4.40E+08	13.9	2.98E-02	2.06E-02	2.06E-02
13	3.25E+07	4.72E+08	15.0	3.25E-02	2.17E-02	2.23E-02
14	4.01E+07	5.12E+08	16.2	3.55E-02	2.32E-02	2.44E-02
15	4.47E+07	5.57E+08	17.7	3.88E-02	2.49E-02	2.67E-02
16	4.09E+07	5.98E+08	18.9	4.18E-02	2.64E-02	2.89E-02
17	4.15E+07	6.39E+08	20.3	4.47E-02	2.79E-02	3.10E-02
Future	3.70E+08	1.01E+09	32.0	7.30E-02	5.62E-02	5.04E-02
Future	5.06E+08	1.51E+09	48.0	1.12E-01	9.47E-02	7.68E-02
Future	1.89E+08	1.70E+09	54.0	1.26E-01	1.09E-01	8.67E-02

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

Table 6-2

Calculated Maximum Neutron Exposure at the Pressure Vessel Inner Radius
Clad/Base Metal Interface

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm²]			
				0°	15°	30°	45°
1	3.71E+07	3.71E+07	1.2	1 86E+18	8.56E+17	4.98E+17	3.10E+17
2	2.28E+07	5.98E+07	1.9	2 99E+18	1.39E+18	8.12E+17	5.13E+17
3	2.37E+07	8.36E+07	2 6	4.11E+18	1.93E+18	1.15E+18	7.24E+17
4	3 48E+07	1.18E+08	3 8	5 87E+18	2.74E+18	1.62E+18	1.02E+18
5	3.55E+07	1.54E+08	4 9	7 37E+18	3.41E+18	2.01E+18	1.27E+18
6	4.07E+07	1.95E+08	6 2	9 03E+18	4.16E+18	2.46E+18	1.56E+18
7	3.77E+07	2.32E+08	7 4	1 07E+19	4.88E+18	2.84E+18	1.82E+18
8	3.40E+07	2.66E+08	8 4	1 18E+19	5.41E+18	3.18E+18	2.04E+18
9	4 01E+07	3.06E+08	9.7	1 29E+19	6.01E+18	3.54E+18	2.27E+18
10	3.82E+07	3.45E+08	10.9	1 42E+19	6.62E+18	3.90E+18	2.49E+18
11	4.76E+07	3.92E+08	12.4	1 55E+19	7.22E+18	4.28E+18	2.76E+18
12	4.75E+07	4.40E+08	13.9	1 67E+19	7.84E+18	4.66E+18	3.01E+18
13	3.25E+07	4.72E+08	15 0	1 76E+19	8.29E+18	4.91E+18	3.17E+18
14	4.01E+07	5.12E+08	16.2	1 86E+19	8.80E+18	5.24E+18	3.38E+18
15	4.47E+07	5.57E+08	17.7	1.97E+19	9.35E+18	5.61E+18	3.62E+18
16	4.09E+07	5.98E+08	18 9	2.06E+19	9.86E+18	5.95E+18	3.84E+18
17	4.15E+07	6.39E+08	20 3	2.15E+19	1.04E+19	6.28E+18	4.05E+18
Future	3.70E+08	1.01E+09	32 0	3 08E+19	1.51E+19	9.31E+18	6.04E+18
Future	5.06E+08	1.51E+09	48 0	4.35E+19	2.16E+19	1.34E+19	8.74E+18
Future	1.89E+08	1.70E+09	54 0	4.82E+19	2.40E+19	1.50E+19	9.76E+18

Table 6-2 cont'd

Calculated Maximum Neutron Exposure at the Pressure Vessel Inner Radius
Clad/Base Metal Interface

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]			
				0°	15°	30°	45°
1	3.71E+07	3.71E+07	1.2	2.99E-03	1.44E-03	8.03E-04	5.13E-04
2	2.28E+07	5.98E+07	1.9	4.81E-03	2.33E-03	1.31E-03	8.48E-04
3	2.37E+07	8.36E+07	2.6	6.61E-03	3.24E-03	1.85E-03	1.20E-03
4	3.48E+07	1.18E+08	3.8	9.43E-03	4.60E-03	2.61E-03	1.68E-03
5	3.55E+07	1.54E+08	4.9	1.18E-02	5.72E-03	3.24E-03	2.10E-03
6	4.07E+07	1.95E+08	6.2	1.45E-02	6.99E-03	3.96E-03	2.57E-03
7	3.77E+07	2.32E+08	7.4	1.72E-02	8.20E-03	4.59E-03	3.00E-03
8	3.40E+07	2.66E+08	8.4	1.89E-02	9.09E-03	5.12E-03	3.37E-03
9	4.01E+07	3.06E+08	9.7	2.08E-02	1.01E-02	5.70E-03	3.75E-03
10	3.82E+07	3.45E+08	10.9	2.28E-02	1.11E-02	6.29E-03	4.12E-03
11	4.76E+07	3.92E+08	12.4	2.49E-02	1.21E-02	6.91E-03	4.55E-03
12	4.75E+07	4.40E+08	13.9	2.69E-02	1.32E-02	7.52E-03	4.97E-03
13	3.25E+07	4.72E+08	15.0	2.83E-02	1.39E-02	7.93E-03	5.23E-03
14	4.01E+07	5.12E+08	16.2	2.99E-02	1.48E-02	8.45E-03	5.58E-03
15	4.47E+07	5.57E+08	17.7	3.16E-02	1.57E-02	9.05E-03	5.99E-03
16	4 09E+07	5 98E+08	18.9	3.31E-02	1.65E-02	9.60E-03	6.35E-03
17	4.15E+07	6 39E+08	20.3	3.46E-02	1.74E-02	1.01E-02	6.69E-03
Future	3.70E+08	1.01E+09	32.0	4.95E-02	2.53E-02	1.50E-02	9.97E-03
Future	5.06E+08	1.51E+09	48.0	6.99E-02	3.62E-02	2.17E-02	1.44E-02
Future	1.89E+08	1.70E+09	54.0	7.75E-02	4.03E-02	2.42E-02	1.61E-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°
199 95	1.000	1 000	1.000	1.000
204.95	0.576	0.593	0.578	0 592
209.95	0.293	0.312	0.297	0.313
214.95	0.143	0.158	0.147	0.159
219.95	0.066	0.078	0.073	0 081

Note:
Base Metal Inner Radius = 199.95 cm
Base Metal 1/4T = 204.95 cm
Base Metal 1/2T = 209.95 cm
Base Metal 3/4T = 214.95 cm
Base Metal Outer Radius = 219.95 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°
199.95	1.000	1.000	1 000	1.000
204.95	0.630	0.640	0.631	0.636
209.95	0.374	0.391	0.377	0.384
214.95	0.217	0.235	0.223	0.230
219.95	0.111	0.130	0.122	0.129

Note:
Base Metal Inner Radius = 199.95 cm
Base Metal 1/4T = 204.95 cm
Base Metal 1/2T = 209.95 cm
Base Metal 3/4T = 214.95 cm
Base Metal Outer Radius = 219.95 cm

Table 6-5

Calculated Fast Neutron Exposure of Surveillance Capsules
Withdrawn from Surry Unit 2

Capsule	Irradiation Time [EFPY]	Fluence (E > 1.0 MeV) [n/cm^2]	Iron Displacements [dpa]
X	1.2	2.97E+18	4.97E-03
W	3 8	6.36E+18	1.04E-02
V	8.4	1.89E+19	3.16E-02
S	15.0	1.07E+19	1.72E-02
Y	20.3	2.73E+19	4.47E-02

Table 6-6

Calculated Surveillance Capsule Lead Factors

Capsule ID And Location	Status	Lead Factor
X	Withdrawn EOC 1	1.60
W	Withdrawn EOC 4	1.08
V	Withdrawn EOC 8	1.61
S	Withdrawn EOC 13	0 61
Y	Withdrawn EOC 17	1 27
U	In Reactor	1.15
T	In Reactor	0.80
Z	In Reactor	0.89

Notes:(1) Lead factors for capsules remaining in the reactor are based on cycle specific exposure calculations through the last completed fuel cycle, i.e., Cycle 17.

7 REFERENCES

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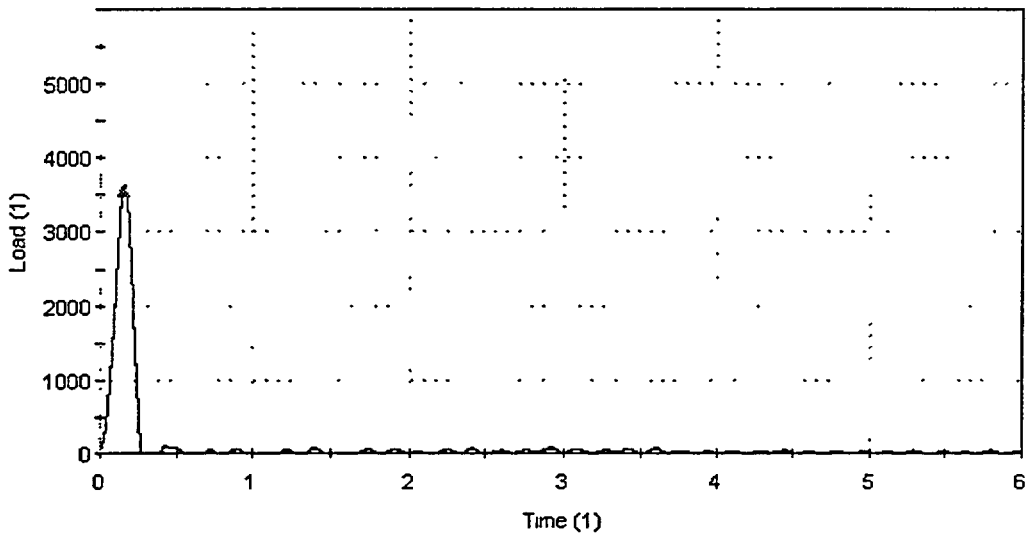
APPENDIX A

LOAD-TIME RECORDS FOR CHARPY SPECIMEN TESTS

- Specimen prefix "V" denotes Lower Plate, Longitudinal Orientation
- Specimen prefix "T" denotes Lower Plate, Transverse Orientation
- Specimen prefix "R" denotes Correlation Monitor Material
- Specimen prefix "W" denotes Weld Material
- Specimen prefix "H" denotes Heat-Affected Zone material
- Load (1) is in units of lbs
- Time (1) is in units of milli seconds



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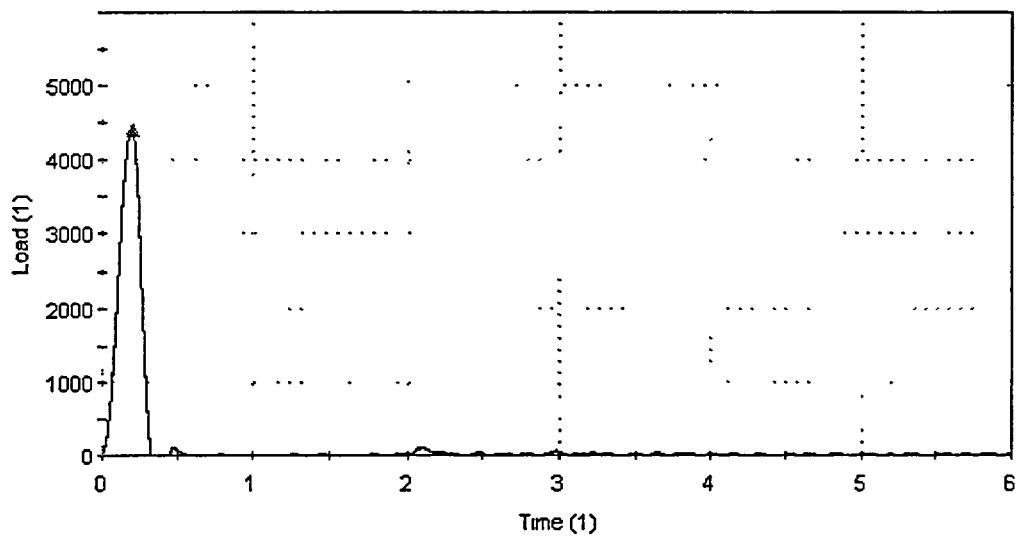

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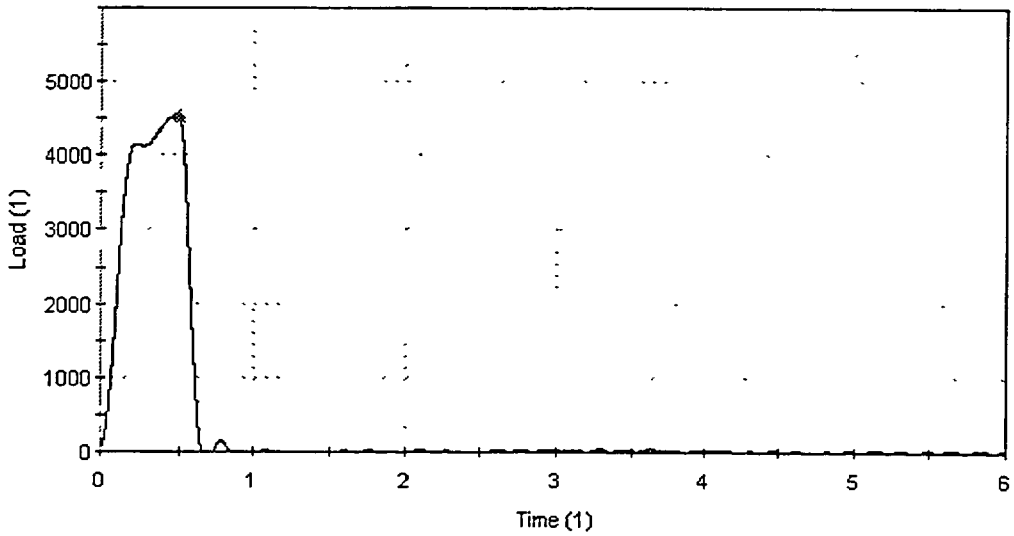

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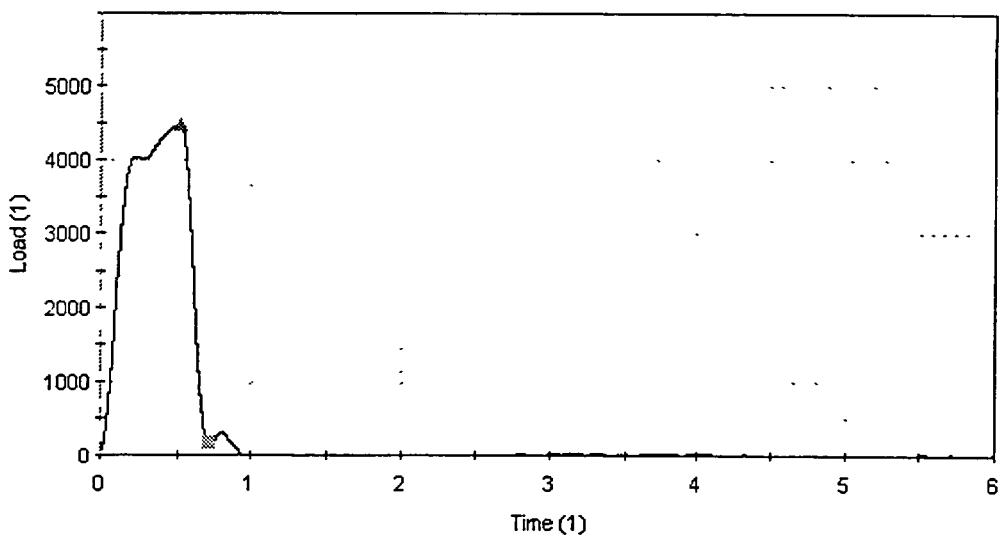

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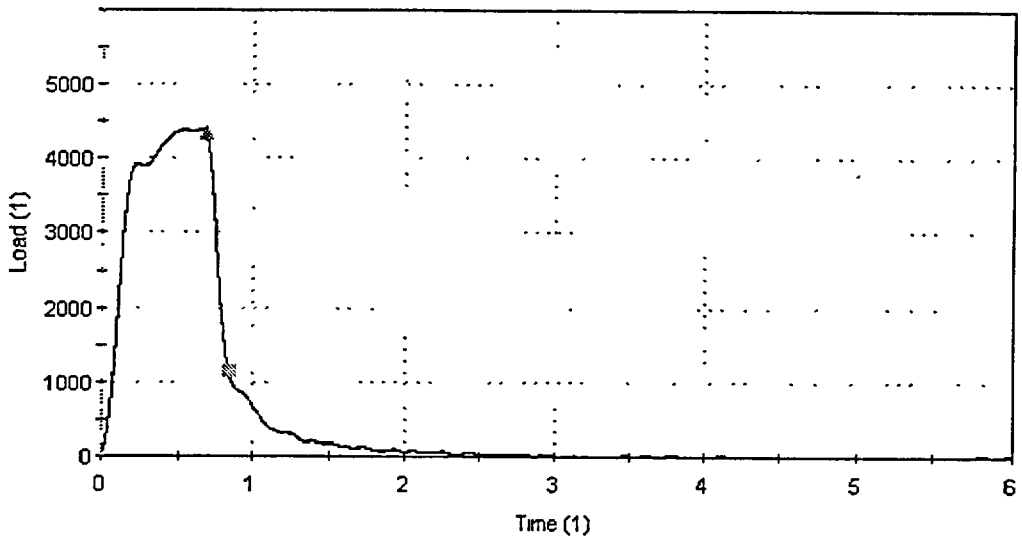

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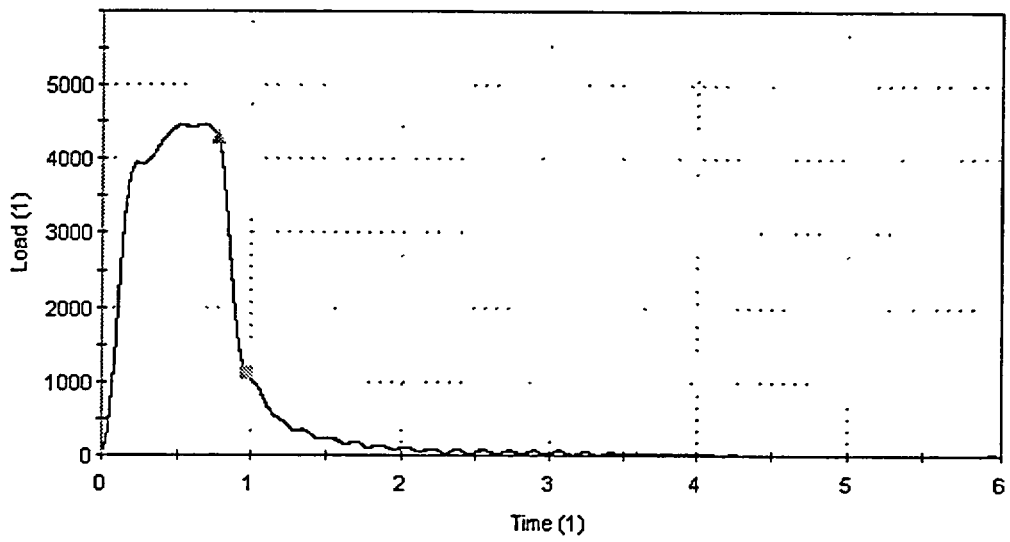

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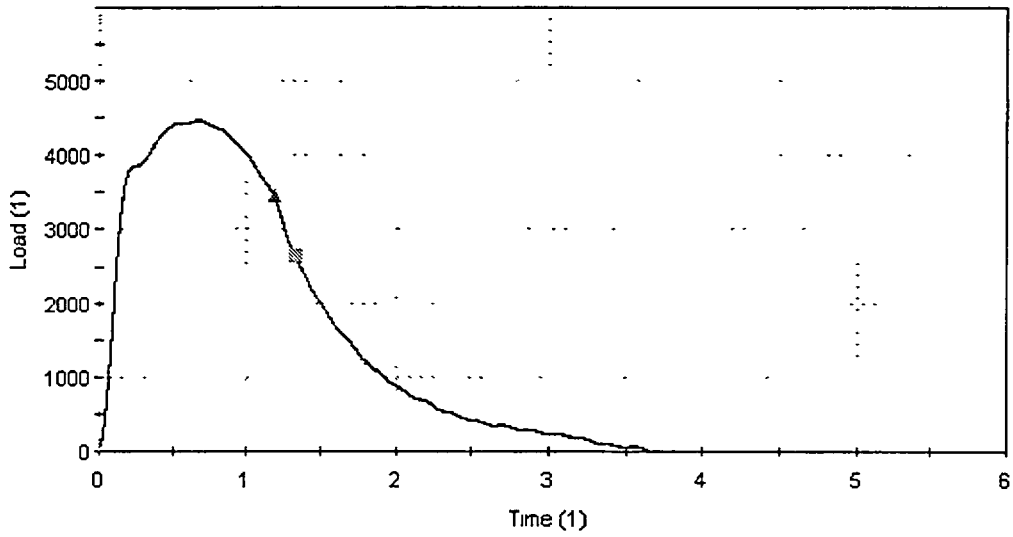
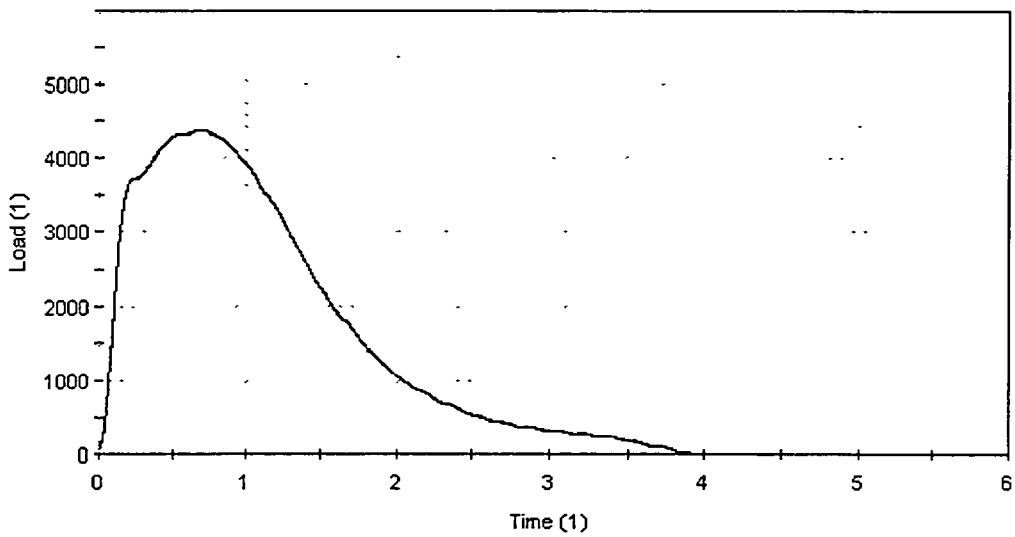
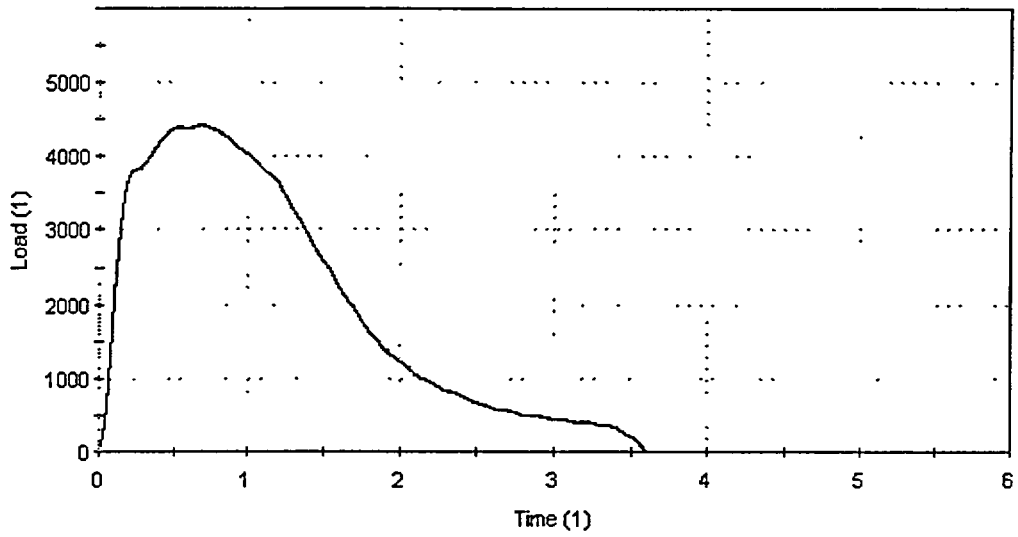


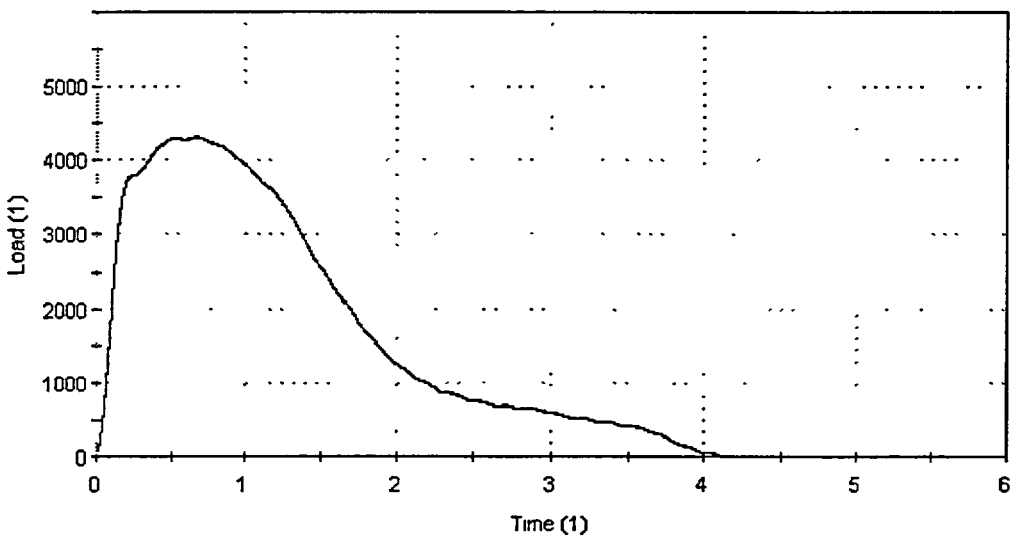
V66, 250°F



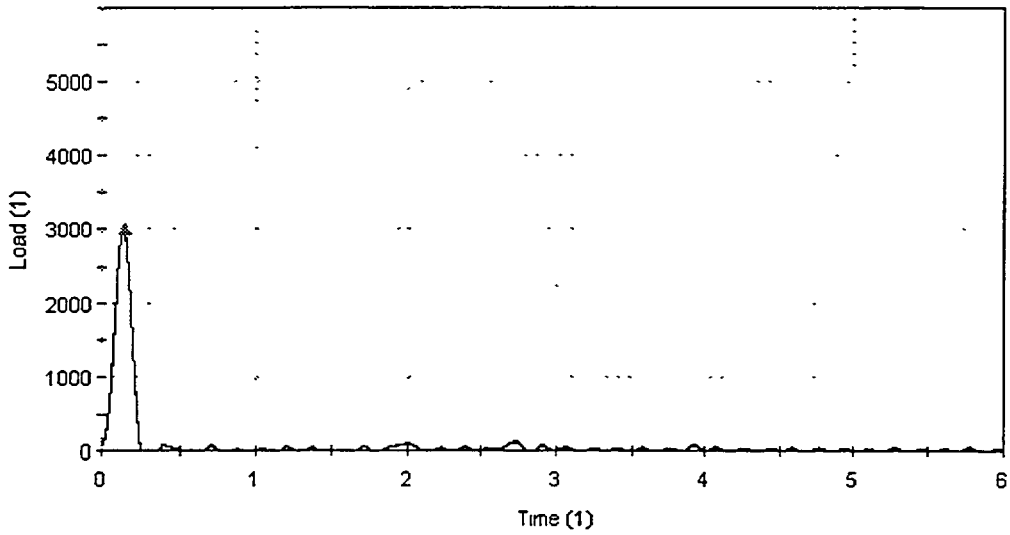
V67, 300°F



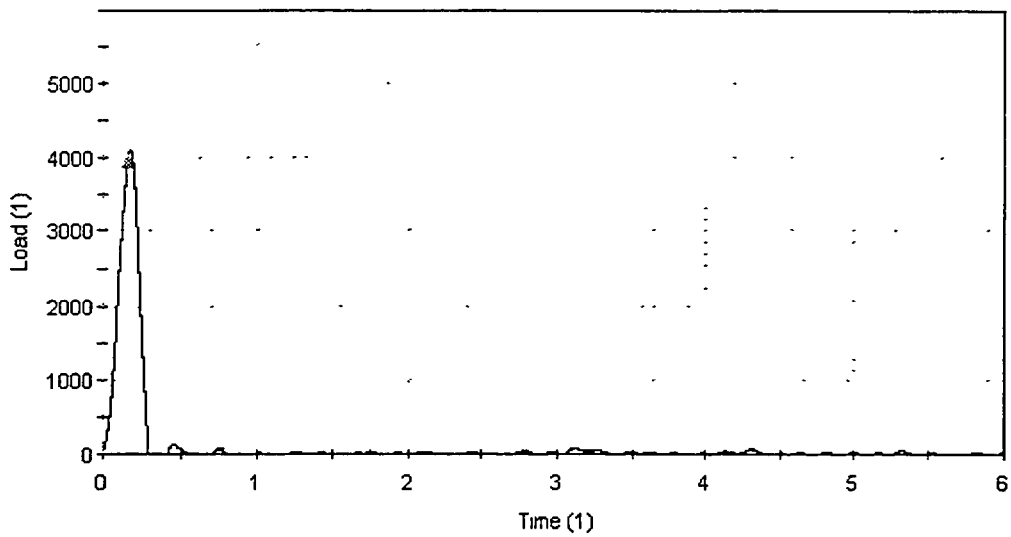
V63, 325°F



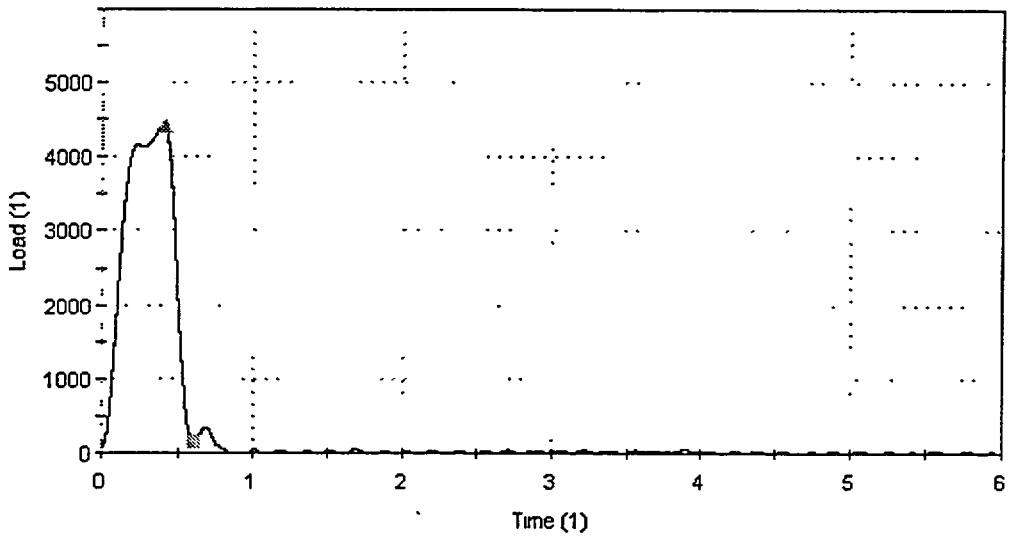
V62, 350°F



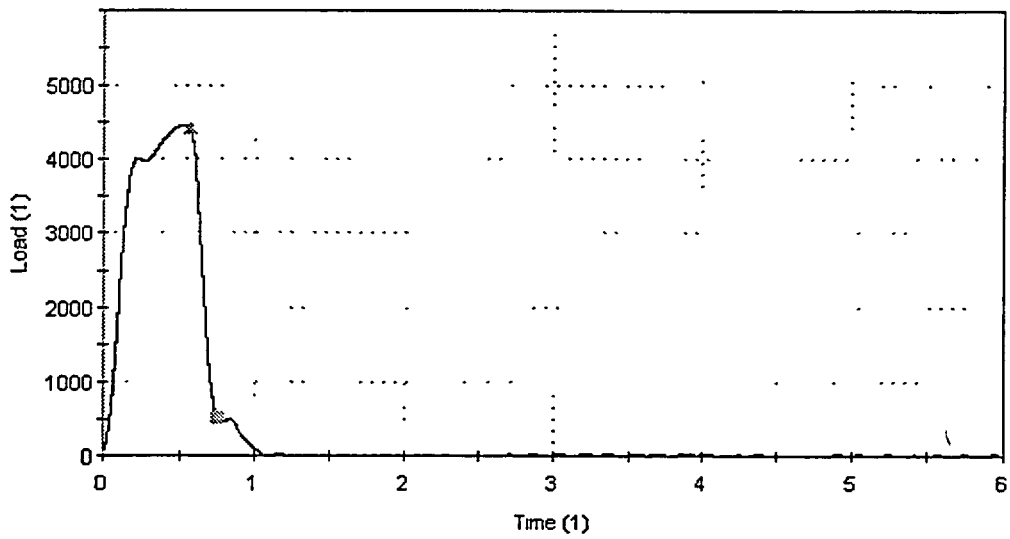
T62, 0°F



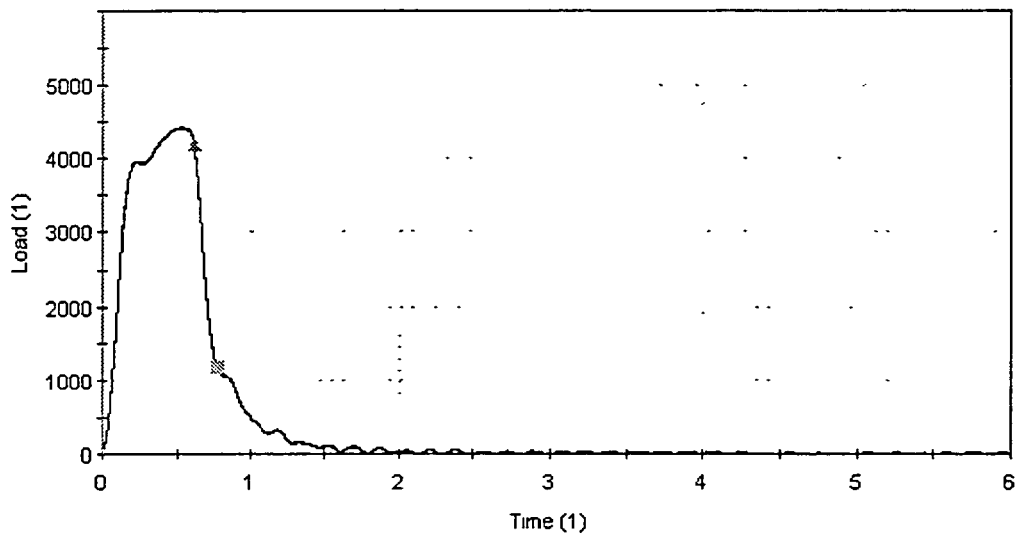
T67, 10°F



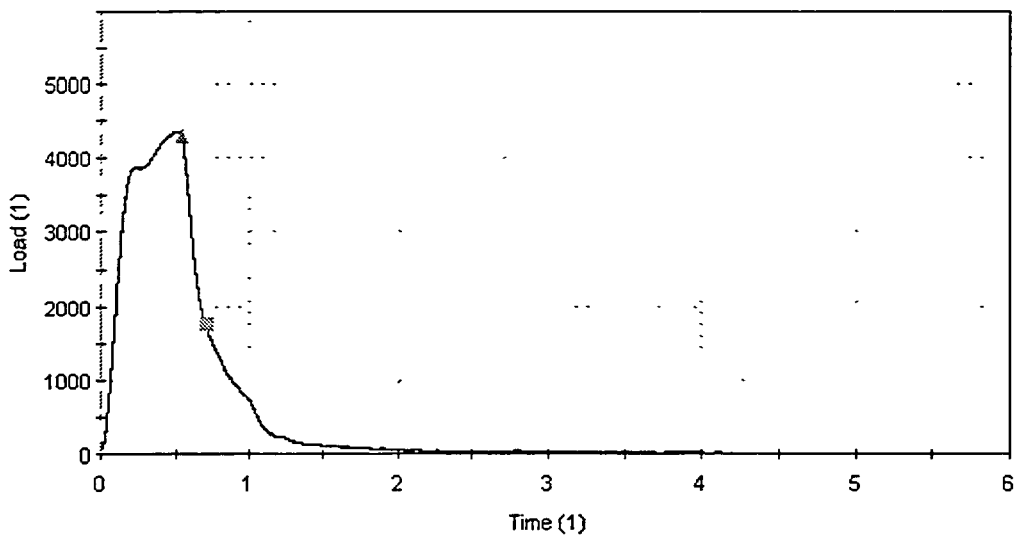
T70, 100°F



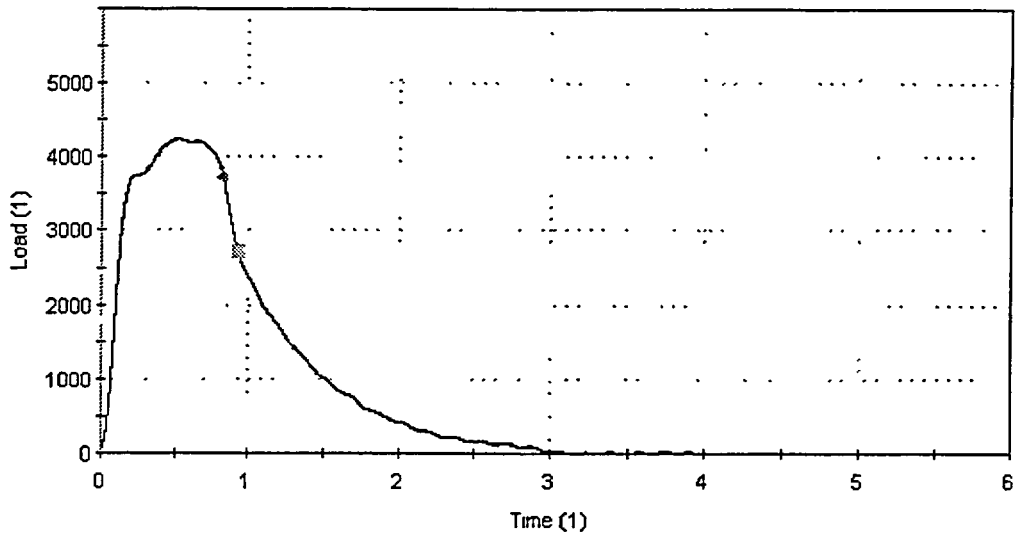
T69, 125°F



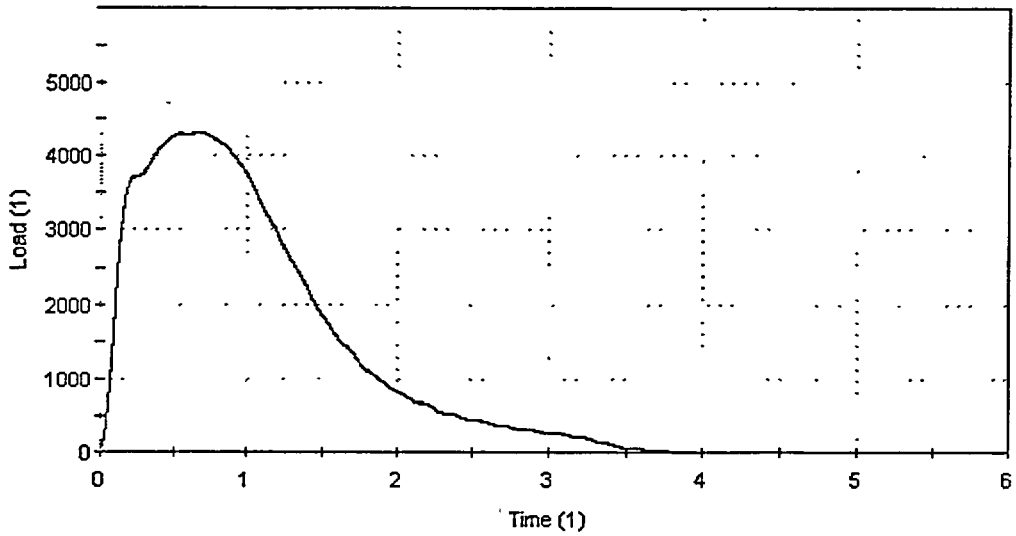
T61, 175°F



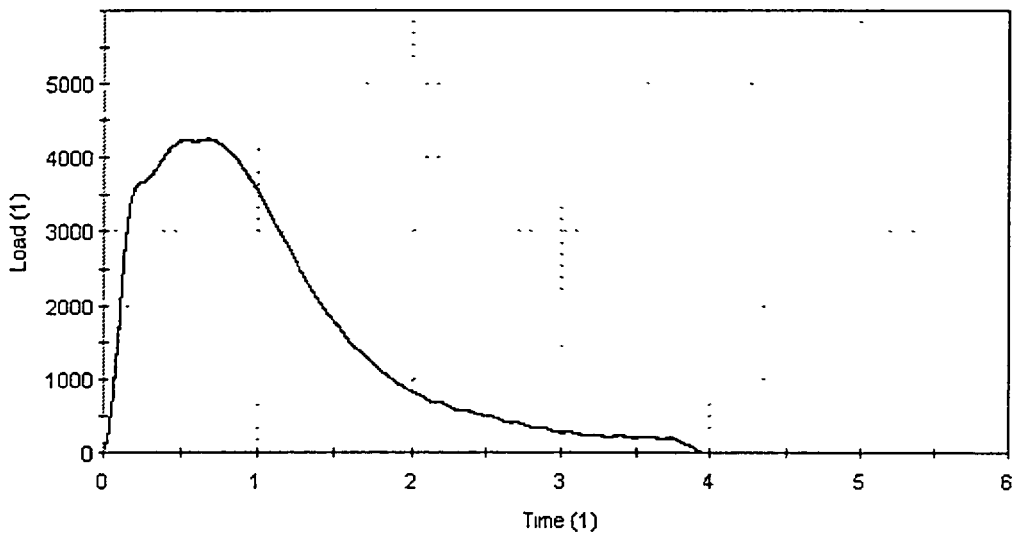
T63, 200°F



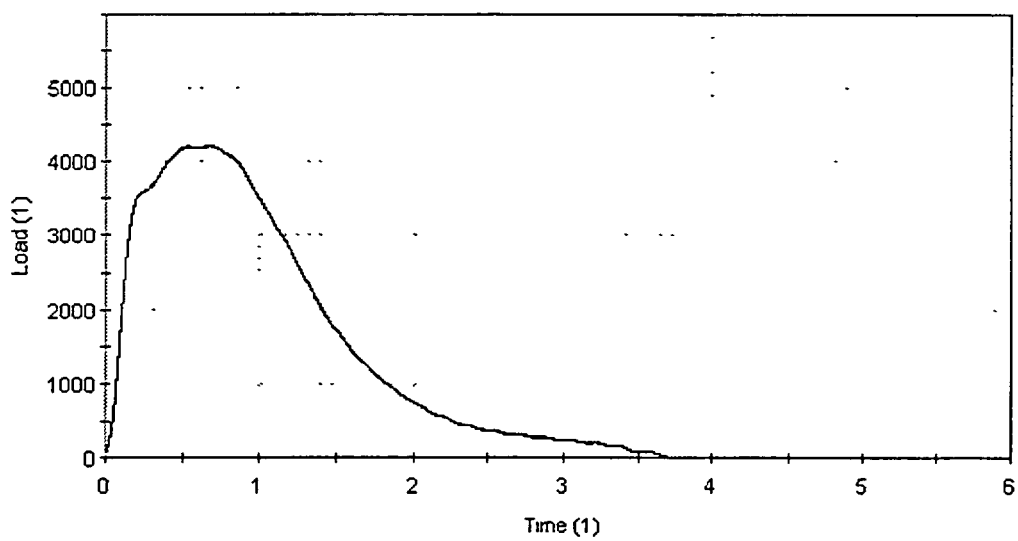
T68, 250°F



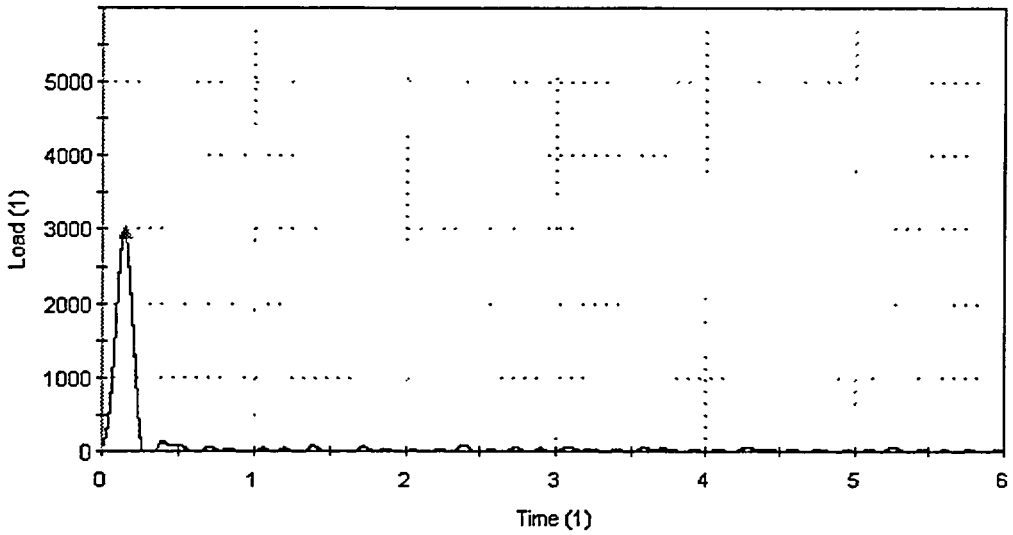
T64, 300°F



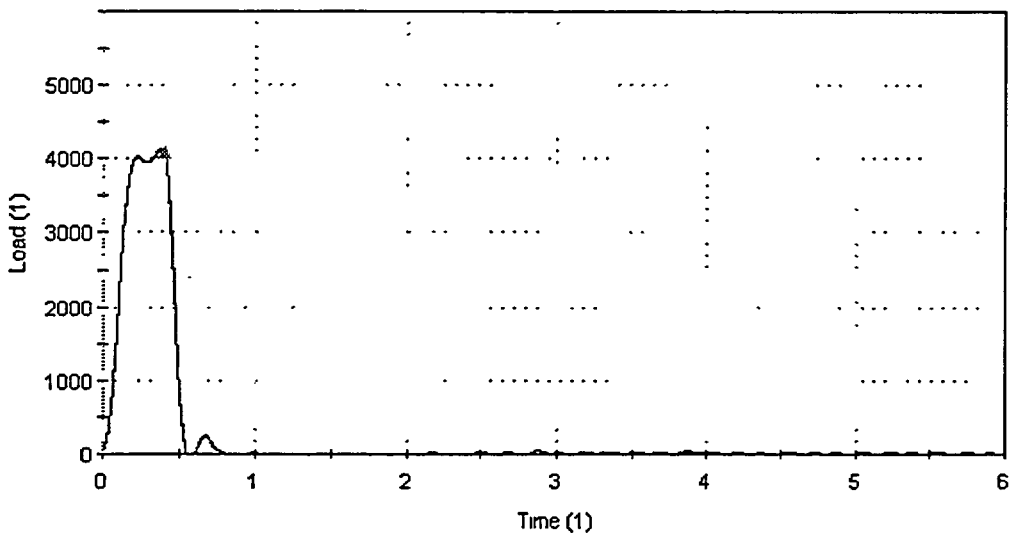
T66, 325°F



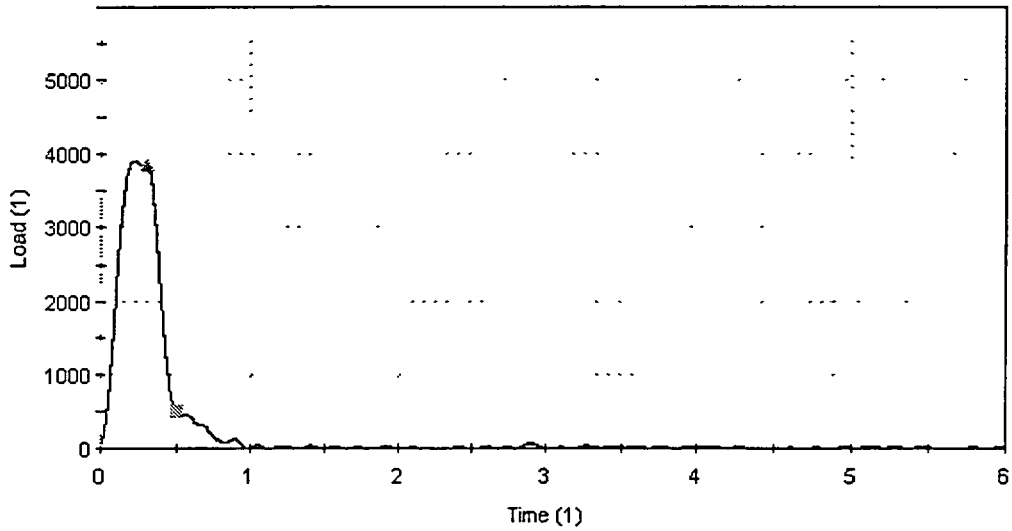
T65, 350°F



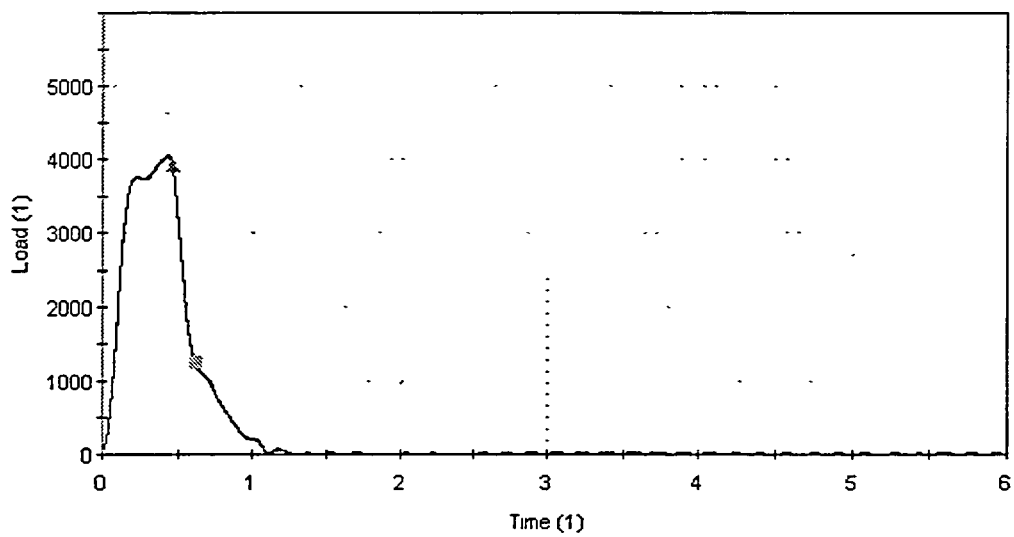
R51, 72°F



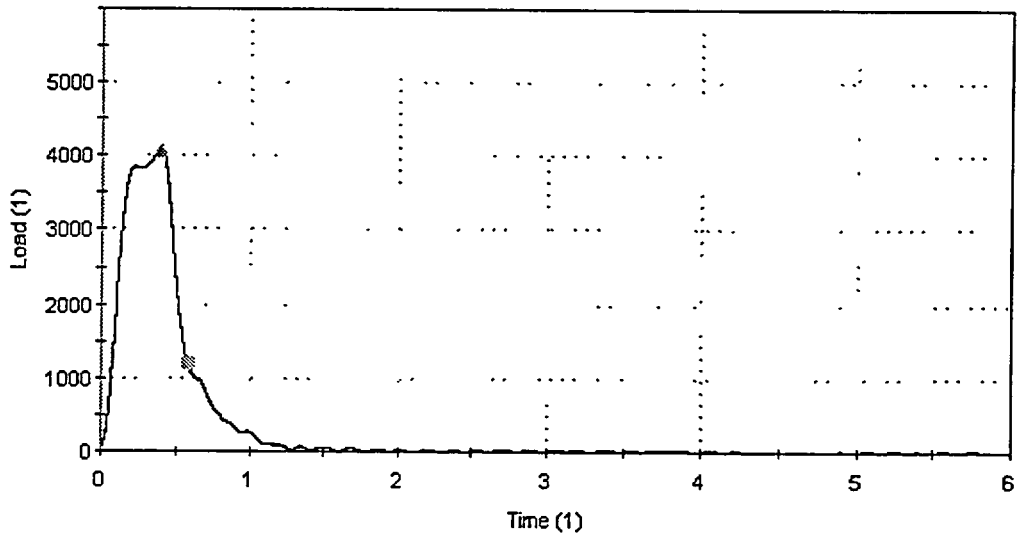
R56, 125°F



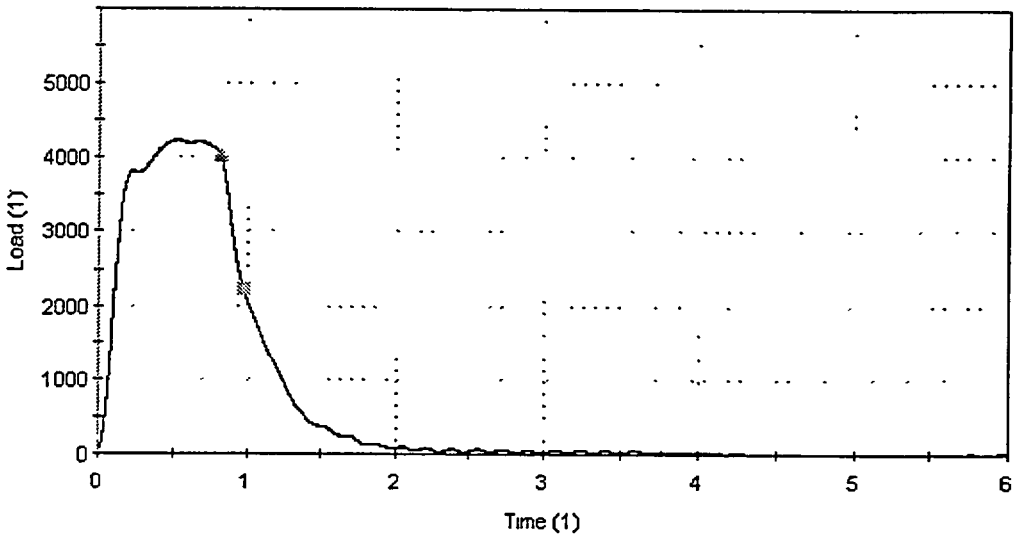
R54, 175°F



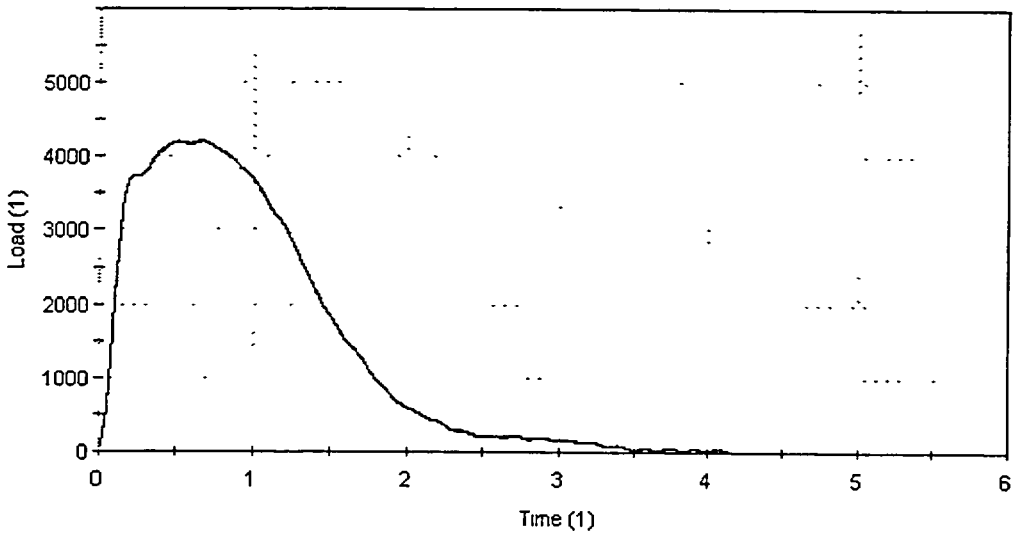
R52, 200°F



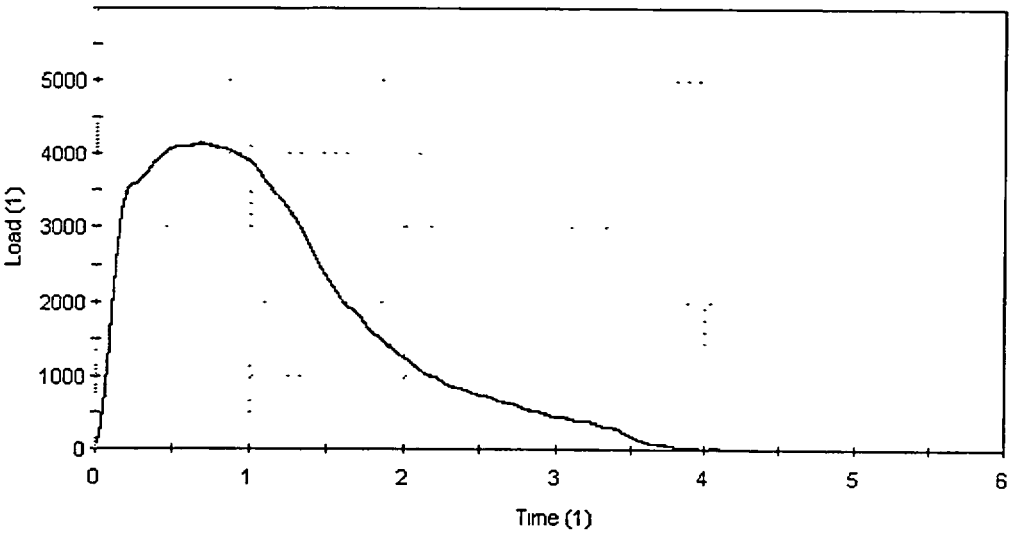
R49, 225°F



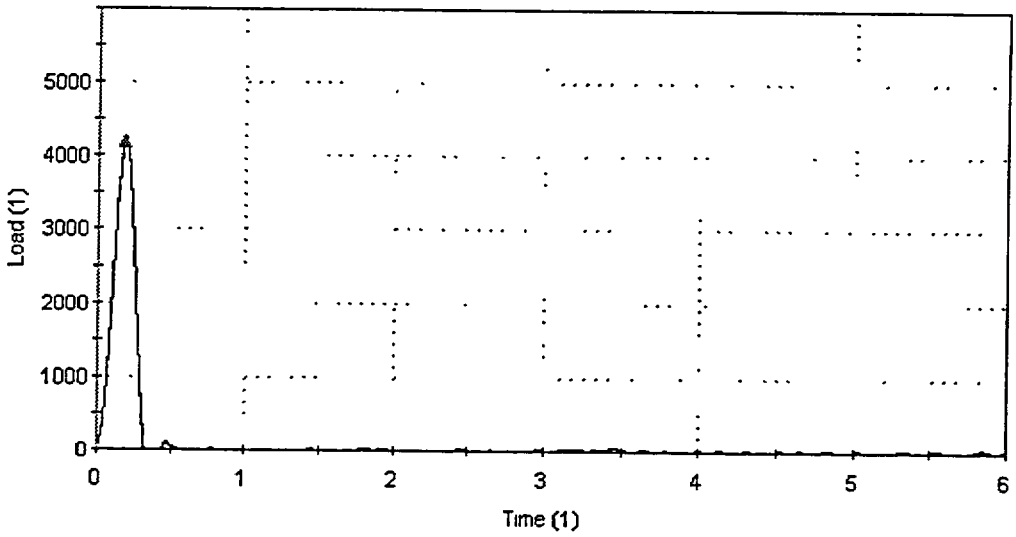
R50, 250°F



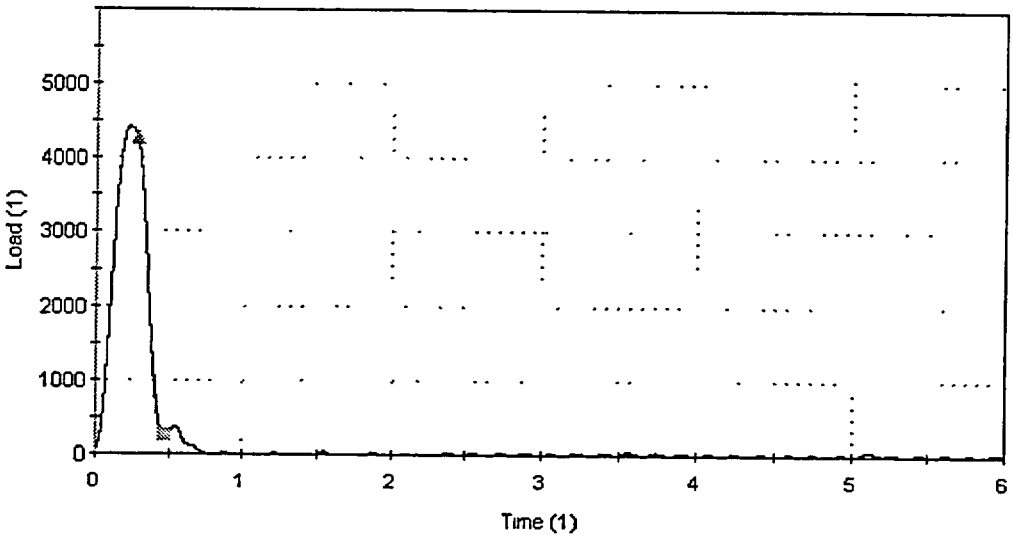
R53, 325°F



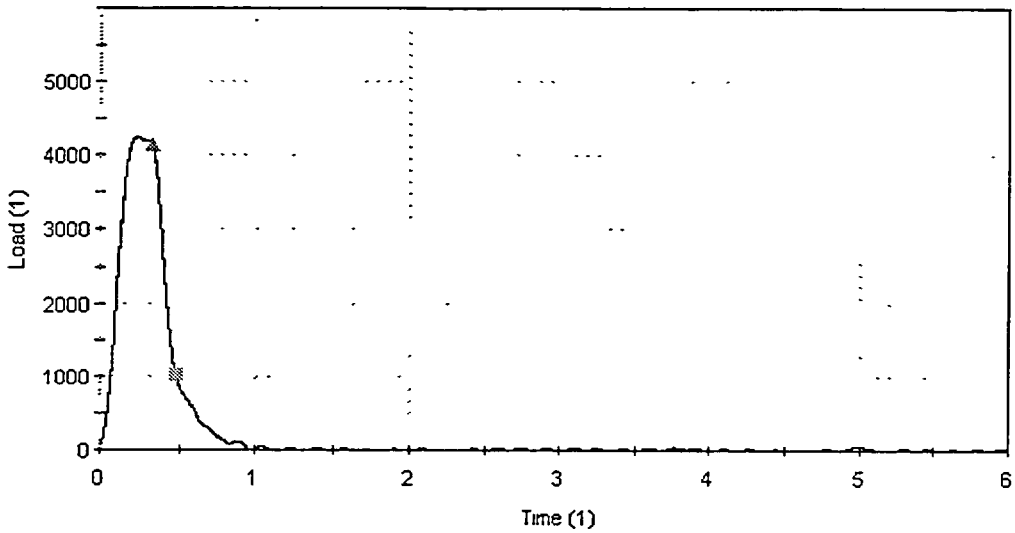
R55, 375°F



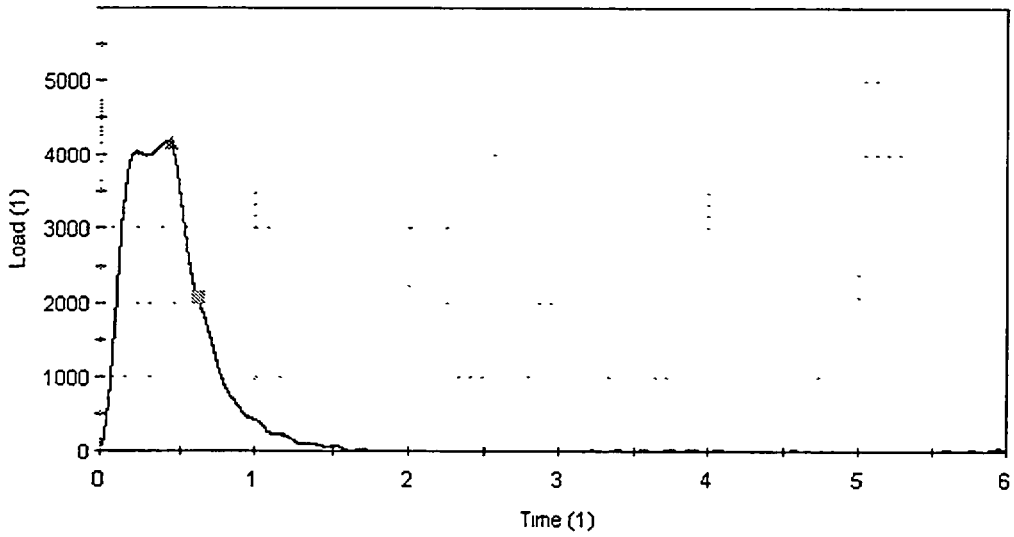
W56, 72°F



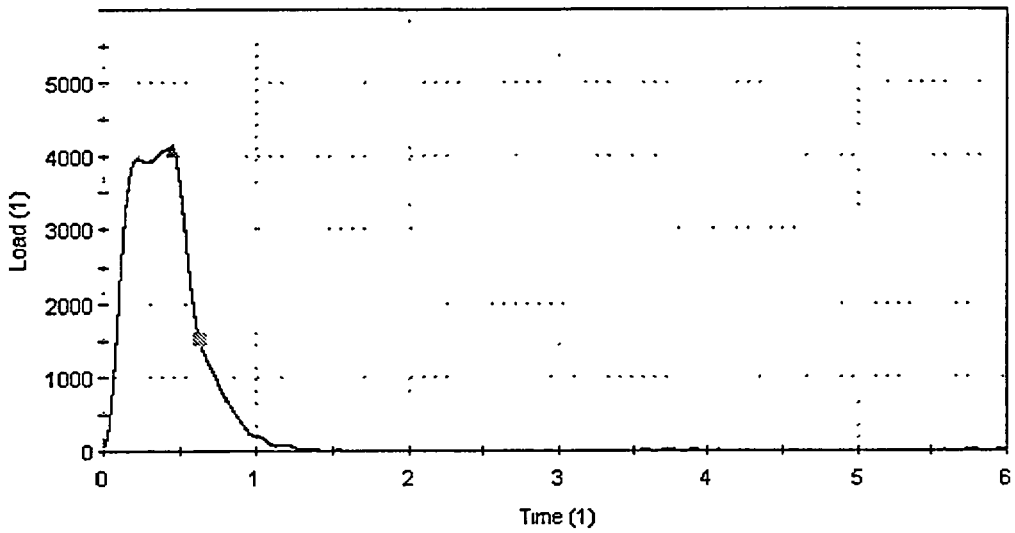
W53, 100°F



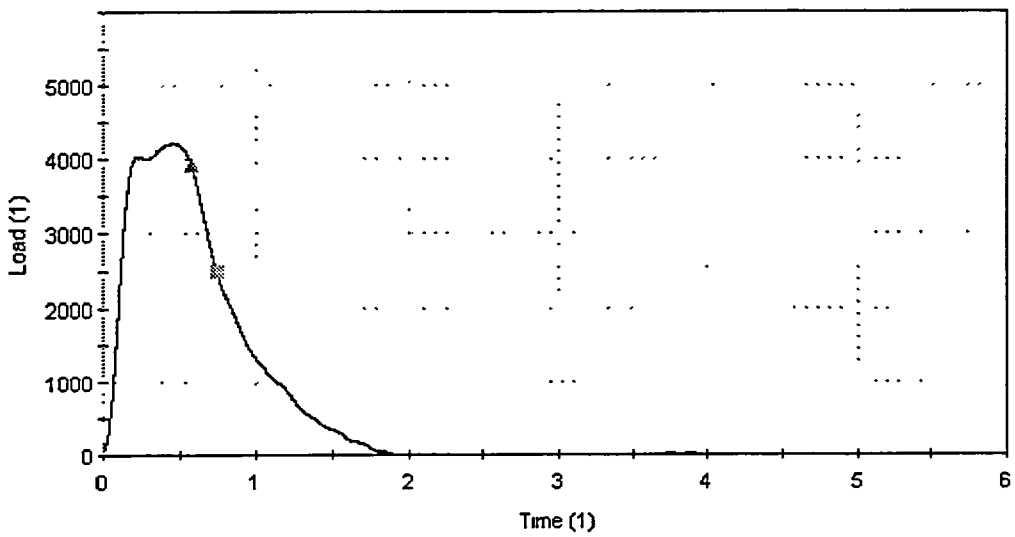
W54, 150°F



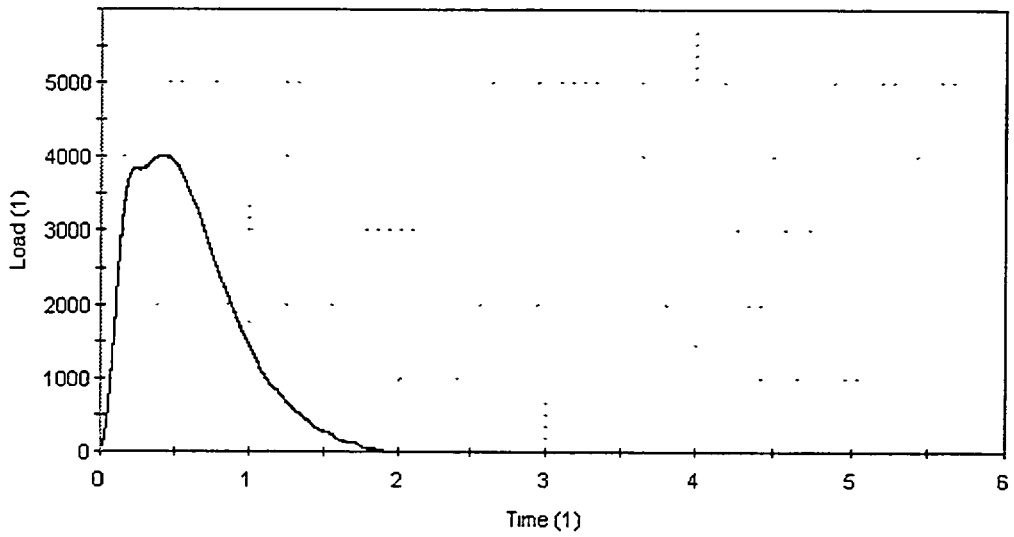
W49, 175°F



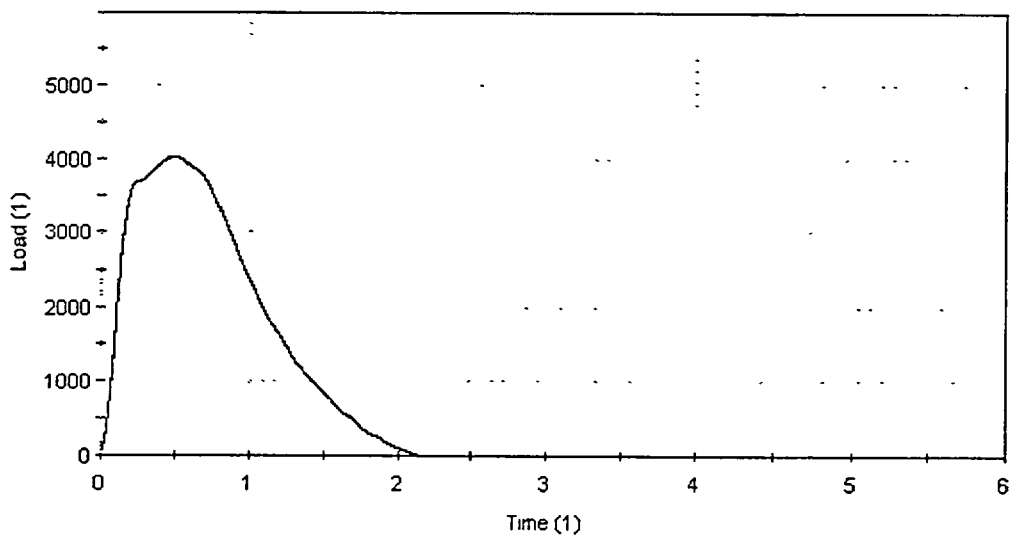
W55, 200°F



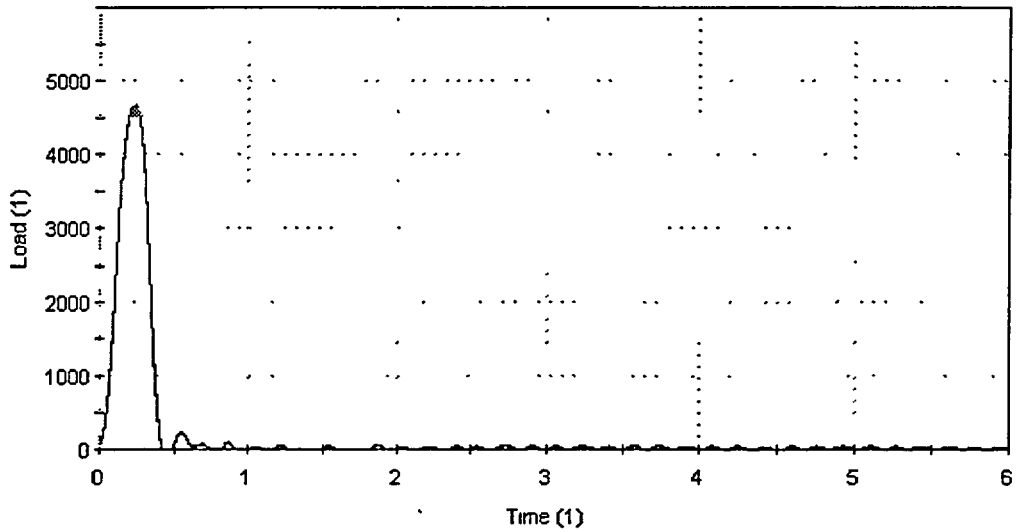
W50, 250°F



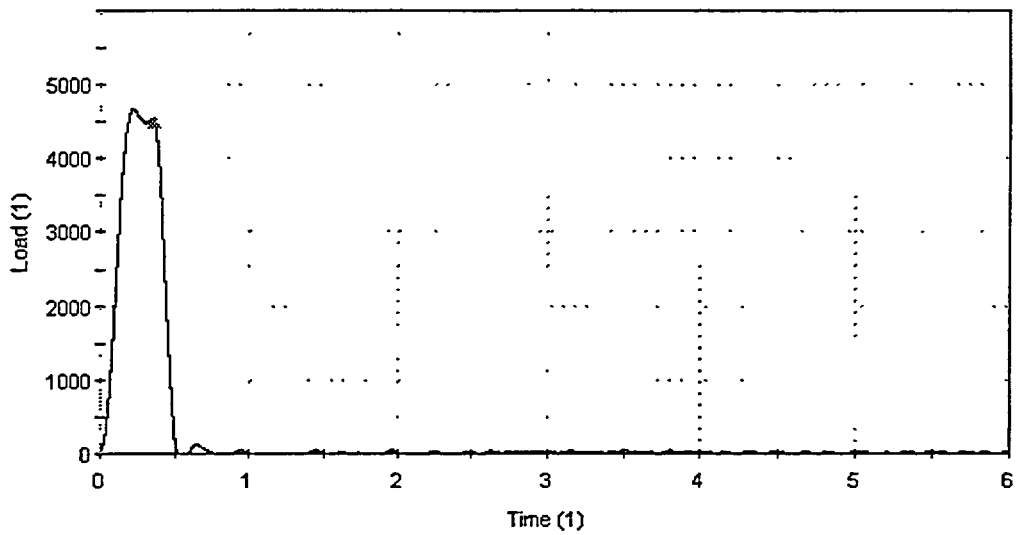
W51, 350°F



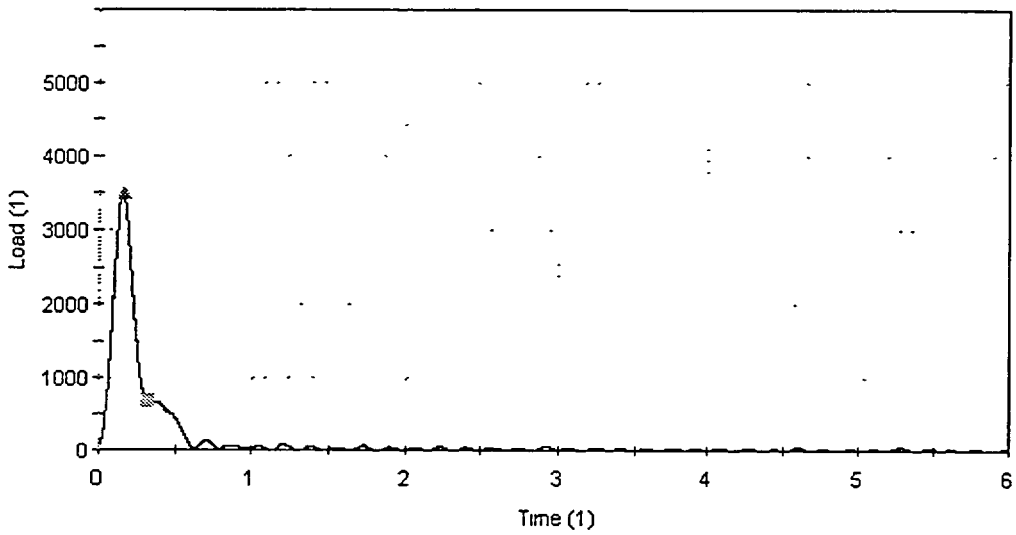
W52, 375°F



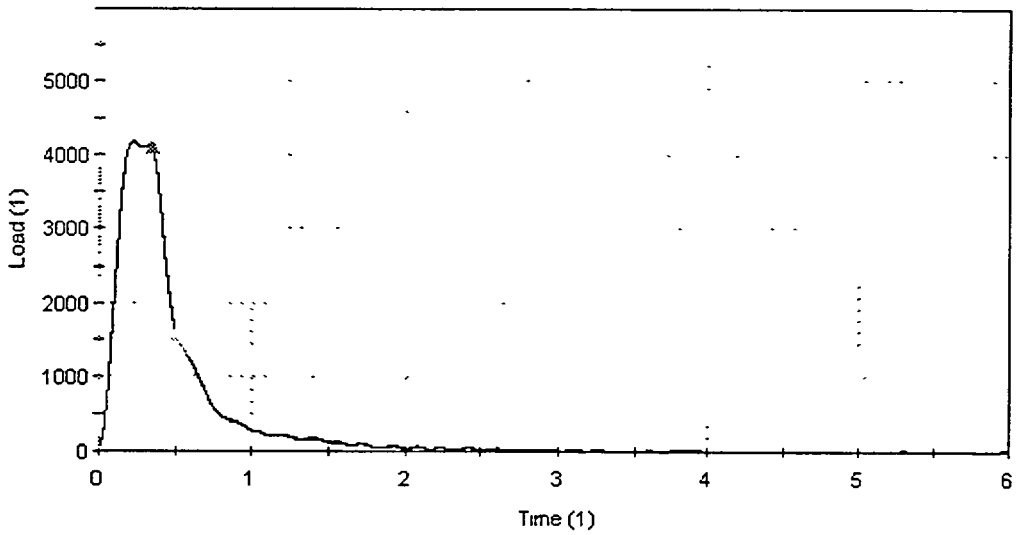
H56, -50°F



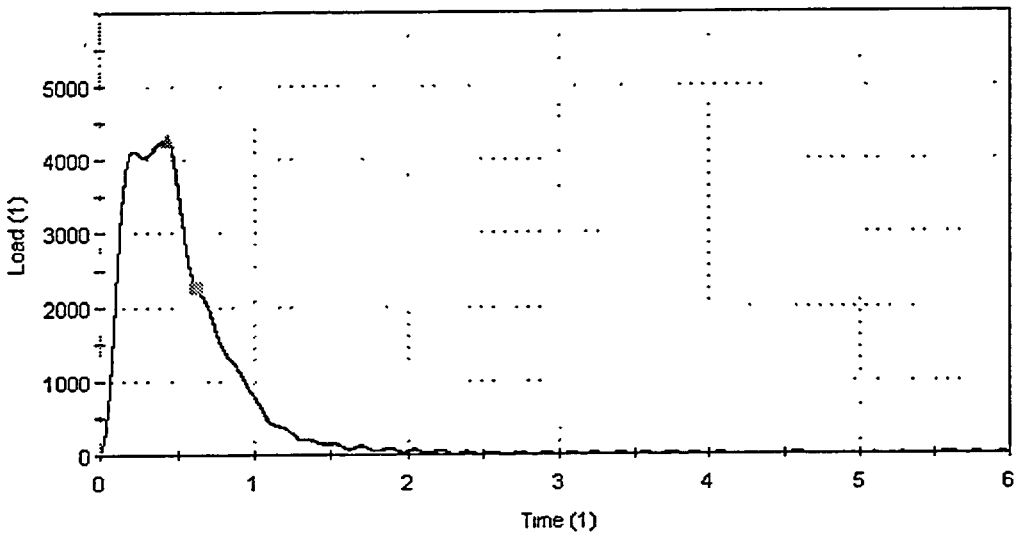
H49, -25°F



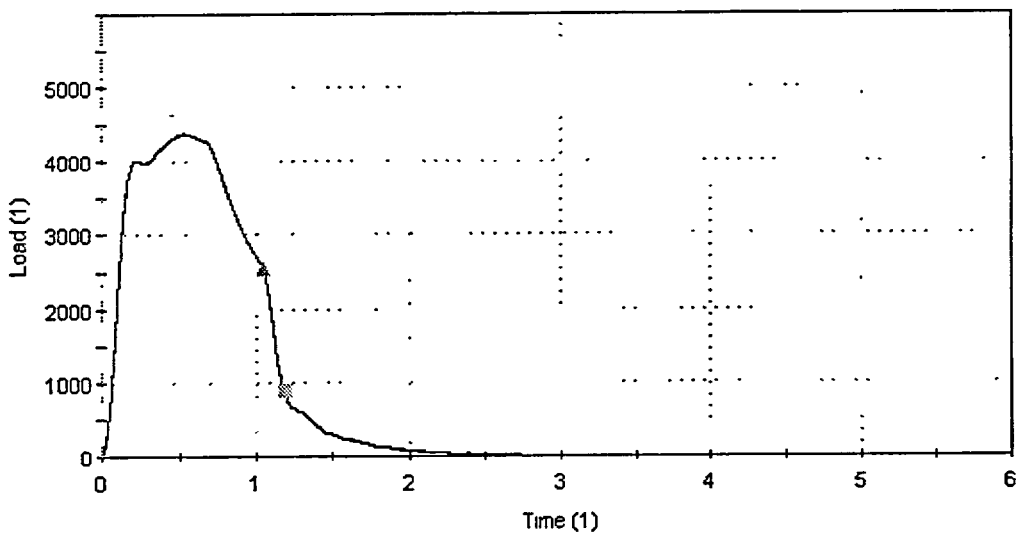
H54, 0°F



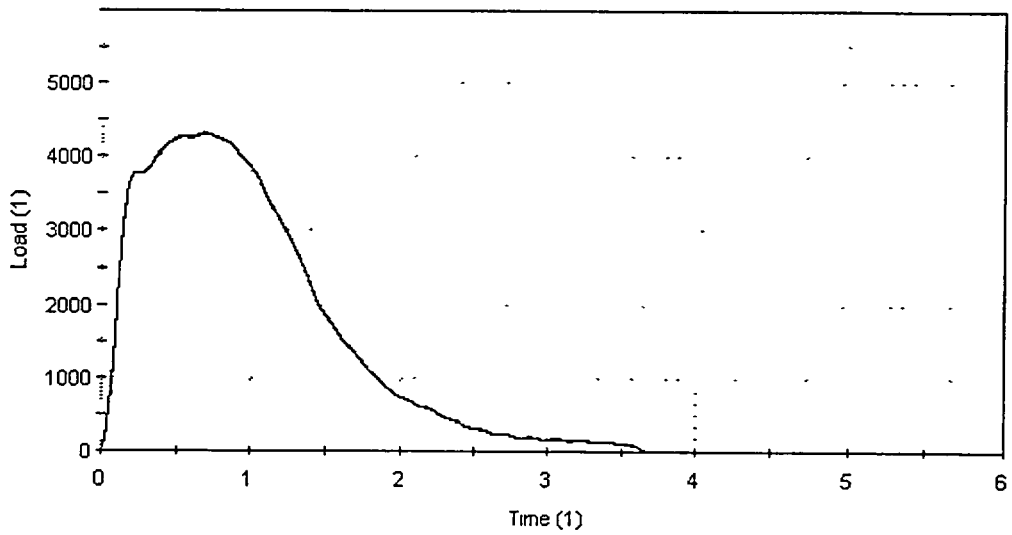
H52, 100°F



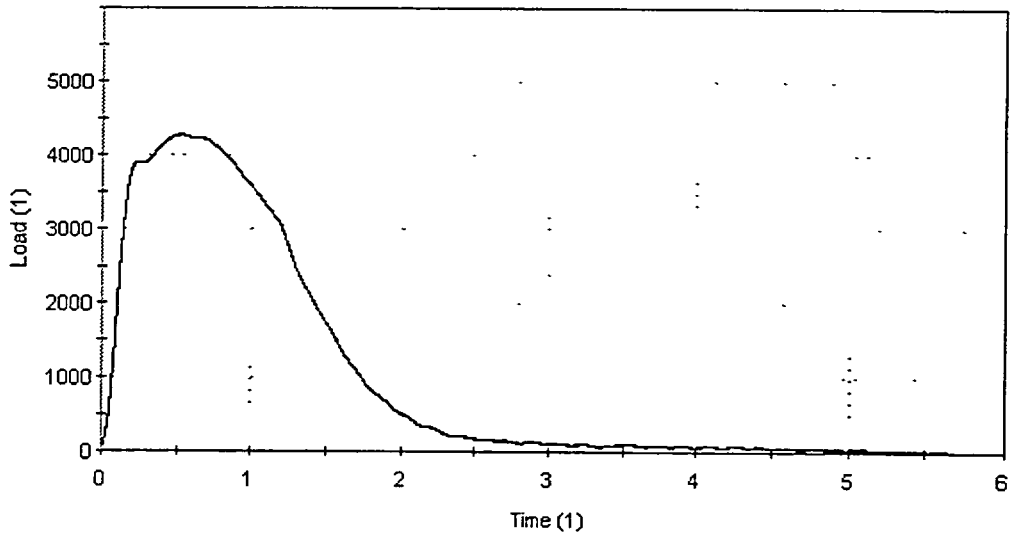
H53, 150°F



H55, 200°F



H50, 300°F



H51, 325°F

APPENDIX B

CHARPY V-NOTCH SHIFT RESULTS FOR EACH CAPSULE PREVIOUS FIT VS. SYMMETRIC HYPERBOLIC TANGENT CURVE-FITTING METHOD (CVGRAPH VERSION 4.1)

TABLE B-1
Changes in Average 30 ft-lb Temperatures for Lower Shell Plate C4339-1 (Longitudinal Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
X	-5°F	50°F	55°F	-1.11°F	57.97°F	59.08°F
V	-5°F	70°F	75°F	-1.11°F	78.01°F	79.12°F
Y	-5°F	---	---	-1.11°F	113.11°F	114.22°F

TABLE B-2
Changes in Average 50 ft-lb Temperatures for Lower Shell Plate C4339-1 (Longitudinal Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
X	35°F	90°F	55°F	33.38°F	89.01°F	55.63°F
V	35°F	115°F	80°F	33.38°F	118.95°F	85.57°F
Y	35°F	---	---	33.38°F	157.64°F	124.26°F

TABLE B-3
Changes in Average 35 mil Lateral Expansion Temperatures for Lower Shell Plate C4339-1
(Long. Orientation) Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
X	25°F	Not Reported	Not Reported	26.92°F	68.85°F	41.93°F
V	25°F	105°F	80°F	26.92°F	100.35°F	73.43°F
Y	25°F	---	---	26.92°F	161.52°F	134.6°F

TABLE B-4
Changes in Average Energy Absorption at Full Shear for Lower Shell Plate C4339-1
(Longitudinal Orientation) Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
X	125 ft-lb	120 ft-lb	-5 ft-lb	128 ft-lb	122 ft-lb	-6 ft-lb
V	125 ft-lb	121 ft-lb	-4 ft-lb	128 ft-lb	121 ft-lb	-7 ft-lb
Y	125 ft-lb	---	---	128 ft-lb	111 ft-lb	-17 ft-lb

TABLE B-5
Changes in Average 30 ft-lb Temperatures for Lower Shell Plate C4339-1 (Transverse Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
X	5°F	50°F	45°F	11.46°F	60.13°F	48.67°F
V	5°F	80°F	75°F	11 46°F	75.06°F	63 6°F
Y	5°F	---	---	11.46°F	118 27°F	106 81°F

TABLE B-6
Changes in Average 50 ft-lb Temperatures for Lower Shell Plate C4339-1 (Transverse Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
X	50°F	100°F	50°F	51 9°F	104.81°F	52.91°F
V	50°F	125°F	75°F	51.9°F	124 07	72.17°F
Y	50°F	---	---	51.9°F	174 08°F	122 18°F

TABLE B-7
Changes in Average 35 mil Lateral Expansion Temperatures for Lower Shell Plate C4339-1
(Transverse Orientation) Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
X	40°F	Not Reported	Not Reported	41.73°F	69.15°F	27.41°F
V	40°F	130°F	90°F	41.73°F	122.35°F	80.61°F
Y	40°F	---	---	41.73°F	178.53°F	136.79°F

TABLE B-8
Changes in Average Energy Absorption at Full Shear for Lower Shell Plate C4339-1
(Transverse Orientation) Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
X	105 ft-lb	95 ft-lb	-10 ft-lb	104 ft-lb	94 ft-lb	-10 ft-lb
V	105 ft-lb	94 ft-lb	-11 ft-lb	104 ft-lb	94 ft-lb	-10 ft-lb
Y	105 ft-lb	---	---	104 ft-lb	94 ft-lb	-10 ft-lb

TABLE B-9
Changes in Average 30 ft-lb Temperatures for Surveillance Weld Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
X	-20°F	75°F	95°F	-20.48°F	75.16°F	95.65°F
V	-20°F	125°F	145°F	-20.48°F	119.73°F	140.21°F
Y	-20°F	---	---	-20.48°F	157.84°F	178.32°F

TABLE B-10
Changes in Average 50 ft-lb Temperatures for Surveillance Weld Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
X	35°F	125°F	90°F	22 48°F	124 66°F	102.17°F
V	35°F	210°F	175°F	22.48°F	222.71°F	200.22°F
Y	35°F	---	---	22.48°F	247.25°F	224.76°F

TABLE B-11
Changes in Average 35 mil Lateral Expansion Temperatures for Surveillance Weld Material
Hand Fit vs. CVGRAPH 4 1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
X	5°F	Not Reported	Not Reported	-1 88°F	86.99°F	88 88°F
V	5°F	160°F	155°F	-1 88°F	165 31°F	167.19°F
Y	5°F	---	---	-1.88°F	218.35°F	220.24°F

TABLE B-12
Changes in Average Energy Absorption at Full Shear for Surveillance Weld Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
X	90 ft-lb	70 ft-lb	-20 ft-lb	91 ft-lb	71 ft-lb	-20 ft-lb
V	90 ft-lb	60 ft-lb	-30 ft-lb	91 ft-lb	60 ft-lb	-31 ft-lb
Y	90 ft-lb	---	---	91 ft-lb	58 ft-lb	-33 ft-lb

TABLE B-13
Changes in Average 30 ft-lb Temperatures for the Heat-Affected-Zone Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
X	-40°F	-30°F	10°F	-52.83°F	-23.61°F	29.22°F
V	-40°F	20°F	60°F	-52.83°F	-1.6°F	51.23°F
Y	-40°F	---	---	-52.83°F	75.07°F	127.91°F

TABLE B-14
Changes in Average 50 ft-lb Temperatures for the Heat-Affected-Zone Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
X	5°F	5°F	0°F	-5.72°F	3 6°F	9.32°F
V	5°F	80°F	75°F	-5.72°F	68 18°F	73.9°F
Y	5°F	---	---	-5.72°F	144.55°F	150.28°F

TABLE B-15
Changes in Average 35 mil Lateral Expansion Temperatures for the Heat-Affected-Zone Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
X	0°F	Not Reported	Not Reported	-8.73°F	-4.43°F	4.3°F
V	0°F	90°F	90°F	-8.73°F	51.95°F	60.68°F
Y	0°F	---	---	-8.73°F	144.04°F	152.77°F

TABLE B-16
Changes in Average Energy Absorption at Full Shear for the Heat-Affected-Zone Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
X	115 ft-lb	100 ft-lb	-5 ft-lb	116 ft-lb	101 ft-lb	-15 ft-lb
V	115 ft-lb	94 ft-lb	-21 ft-lb	116 ft-lb	94 ft-lb	-22 ft-lb
Y	115 ft-lb	---	---	116 ft-lb	94 ft-lb	-22 ft-lb

TABLE B-17
Changes in Average 30 ft-lb Temperatures for the Correlation Monitor Material
Hand Fit vs CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
X	40°F	100°F	60°F	46.44°F	108 64°F	62.19°F
V	45°F	165°F	120°F	46 44°F	162.99°F	116.55°F
Y	45°F	---	---	46 44°F	194 47°F	148 02°F

TABLE B-18
Changes in Average 50 ft-lb Temperatures for the Correlation Monitor Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
X	70°F	145°F	75°F	78.3°F	140 6°F	62.3°F
V	75°F	200°F	125°F	78.3°F	201.2°F	122.89°F
Y	75°F	---	---	78.3°F	232 43°F	154.12°F

TABLE B-19
Changes in Average 35 mil Lateral Expansion Temperatures for the Correlation Monitor Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
X	60°F	Not Reported	Not Reported	58.96°F	113.61°F	54.65°F
V	60°F	195°F	135°F	58.96°F	188 47°F	129.51°F
Y	60°F	---	---	58.96°F	235.59°F	176.63°F

TABLE B-20
Changes in Average Energy Absorption at Full Shear for the Correlation Monitor Material
Hand Fit vs CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
X	125 ft-lb	115 ft-lb	-10 ft-lb	123 ft-lb	103 ft-lb	-20 ft-lb
V	125 ft-lb	102 ft-lb	-23 ft-lb	123 ft-lb	102 ft-lb	-21 ft-lb
Y	125 ft-lb	---	---	123 ft-lb	100 ft-lb	-23 ft-lb