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Ken Peters
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2003-0020

March 28, 2003

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

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket No. 50-382
Submittal of Second Reactor Vessel Surveillance Capsule Report

Dear Sir or Madam:

The reactor vessel material irradiation surveillance specimens inserted in the Waterford Steam Electric Station, Unit 3 (Waterford 3) reactor vessel prior to initial plant startup are required to be removed and examined to determine changes in material properties, in accordance with the surveillance program. The second capsule, 4/W-263, was removed on April 1, 2002, during the eleventh refueling outage after 13.83 effective full power years (EFPY). In accordance with 10 CFR 50 Appendix H, a summary report is required to be submitted within one year of the date of capsule withdrawal Entergy Operations, Inc. (Entergy) hereby submits the attached report summarizing the post irradiation testing and fluence analysis results associated with capsule 4/W-263. (All nomenclature or identification denoted as capsule 263° or W-263 in the attached report pertains to capsule 4/W-263.)

Weld metal specimen 3J7, tested at 550 °F, was shown to have failed outside the gauge length resulting in the clip gauge falling off. Therefore the stress-strain curve is incomplete. A review of the fabrication records by Entergy indicates that the failure probably occurred at or very close to the fusion line of the surveillance weld Therefore Entergy believes the data from the test conducted on specimen 3J7 is not representative for the weld metal tensile properties. Hence the tensile data from specimen 3J7 will be thoroughly evaluated prior to being used for historical or correlation purposes.

The current Waterford 3 pressure-temperature limits are valid through 16 EFPY. New pressure-temperature limit curves are currently being developed and Entergy plans to submit a license amendment request in August 2003 to establish pressure-temperature limits for operation beyond 16 EFPY.

A008

The proposed change does not include any new commitments.

If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

Sincerely,

K.J. Peters
Director, Nuclear Safety Assurance
Waterford Steam Electric Station, Unit 3

KJP/DBM/cbh

Attachment: WCAP-16002, Revision 0, Analysis of Capsule 263° from the Entergy Operations Waterford Unit 3 Reactor Vessel Radiation Surveillance Program

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith (w/o attachment)
N.S. Reynolds (w/o attachment)
NRC Resident Inspectors Office

Attachment To

W3F1-2003-0020

WCAP-16002, Revision 0, Analysis of Capsule 263° from the Entergy Operations Waterford Unit 3 Reactor Vessel Radiation Surveillance Program

Westinghouse Non-Proprietary Class 3

WCAP-16002
Revision 0

March, 2003

Analysis of Capsule 263° from the Entergy Operations Waterford Unit 3 Reactor Vessel Radiation Surveillance Program

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-16002

Analysis of Capsule 263° from the Entergy Operations Waterford Unit 3 Reactor Vessel Radiation Surveillance Program

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March 2003

Verified by: Chris Hoffmann 3-6-03
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EXECUTIVE SUMMARY

The purpose of this report is to document the results of the testing of surveillance capsule 263° from Waterford Unit 3. Capsule 263° was removed at 13.83 EFPY and post-irradiation mechanical testing of the Charpy V-notch and tensile specimens was performed. A fluence evaluation was also performed based on methodology and nuclear data including neutron transport and dosimetry cross-section libraries derived from the ENDF/B-VI database. The calculated peak clad base metal vessel fluence after 13.83 EFPY of plant operation was 1.23 x 10^{19} n/cm^2 and the surveillance Capsule 263° calculated fluence was 1.45 x 10^{19} n/cm^2. A brief summary of the Charpy V-notch testing results can be found in Section 1 and the updated capsule removal schedule can be found in Section 7. A credibility evaluation was performed of the Waterford Unit 3 surveillance data in accordance with Regulatory Guide 1.99, Revision 2; it can be found in Appendix D.

1 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance capsule 263°, the second capsule to be removed from the Waterford Unit 3 reactor pressure vessel, led to the following conclusions:

- The capsule received an average fast neutron calculated fluence (E > 1.0 MeV) of 1.45 x 10^{19} n/cm^2 after 13.83 effective full power years (EFPY) of plant operation.

- The reactor vessel lower shell plate M-1004-2 Charpy specimens in the transverse orientation were irradiated to 1.45 x 10^{19} n/cm^2 (E> 1.0MeV). This resulted in a 30 ft-lb transition temperature decrease of 9.1°F and a 50 ft-lb transition temperature increase of 8.1°F, with an irradiated 30 ft-lb transition temperature of -33.6°F and an irradiated 50 ft-lb transition temperature of 2.9°F for the transversely oriented specimens

- The HSST Plate O1MY correlation monitor material Charpy specimens in the longitudinal orientation were irradiated to 1.45 x 10^{19} n/cm^2 (E> 1.0 MeV). This resulted in a 30 ft-lb transition temperature increase of 150.5°F and a 50 ft-lb transition temperature increase of 151.3°F, with an irradiated 30 ft-lb transition temperature of 184.9°F and an irradiated 50 ft-lb transition temperature of 211.4°F for the longitudinally oriented specimens.

- The weld metal Charpy specimens were irradiated to 1.45 x 10^{19} n/cm^2 (E> 1.0 MeV). This resulted in a 30 ft-lb transition temperature increase of 6.9°F and a 50 ft-lb transition temperature increase of 13.8°F, with an irradiated 30 ft-lb transition temperature of -77.7°F and an irradiated 50 ft-lb transition temperature of -51.4°F.

- The weld heat-affected-zone (HAZ) metal Charpy specimens were irradiated to 1.45 x 10^{19} n/cm^2 (E > 1.0 MeV). This resulted in a 30 ft-lb transition temperature increase of 25.1°F and a 50 ft-lb transition temperature increase of 27.9°F. The irradiated 30 ft-lb transition temperature is -92.0°F and the irradiated 50 ft-lb transition temperature is -62.1°F.

- Based on the average values, the upper shelf energy of the lower shell plate M-1004-2 (transverse orientation) decreased 10 ft-lb after irradiation to 1.45 x 10^{19} n/cm^2 (E> 1.0 MeV). This resulted in an irradiated average upper shelf energy of 131 ft-lb for the transversely oriented specimens.

- Based on the average values, the upper shelf energy of the HSST Plate O1MY correlation monitor material (longitudinal orientation) decreased 20 ft-lb after irradiation to 1.45 x 10^{19} n/cm^2 (E > 1.0 MeV). This results in an irradiated average upper shelf energy of 113 ft-lb for the longitudinally oriented specimens.

- Based on the average values, the upper shelf energy of the weld metal Charpy specimens decreased 11 ft-lb after irradiation to 1.45 x 10^{19} n/cm^2 (E> 1.0 MeV). This results in an irradiated average upper shelf energy of 145 ft-lb for the weld metal specimens.

- Based on the average values, the average upper shelf energy of the weld HAZ Charpy specimens decreased 7 ft-lb after irradiation to 1.45 x 10^{19} n/cm^2 (E > 1.0 MeV). This results in an irradiated average upper shelf energy of 163 ft-lb for the weld HAZ.

- A comparison of the Waterford Unit 3 reactor vessel beltline 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 for all the surveillance program weld and plate materials from Capsule 263° is less than or comparable to the Regulatory Guide 1.99, Revision 2 predictions.

  - The measured percent decrease in upper shelf energy of the Capsule 263° surveillance material is less than the Regulatory Guide 1.99, Revision 2 predictions.

- The peak end-of-license (32 EFPY) neutron fluence (E> 1.0 MeV) at the core midplane for the Waterford Unit 3 reactor vessel is given below. One value is given corresponding to the clad base metal interface. A second value is given for the vessel inner wetted surface back-calculated from the clad base metal interface fluence through the 1/8 inch clad using the Regulatory Guide 1.99, Revision 2 attenuation formula (i.e., Equation 3 in the Guide; $f_{(depth\ x)} = f_{surface} * e^{(-0.24x)}$). Also provided are the calculated fluence at the vessel ¼ and ¾ thickness locations including the cladding, where thickness is 8.625 inches calculated using the Regulatory Guide 1.99, Revision 2 attenuation formula.

Vessel Clad Base Metal Interface:	2.48×10^{19} n/cm^2
Vessel Inner Wetted Surface:	2.56×10^{19} n/cm^2
RG 1.99 Attenuated Fluence:	Vessel ¼ thickness = 1.48×10^{19} n/cm^2 Vessel ¾ thickness = 5.25×10^{18} n/cm^2

  The preceding values of neutron fluence were based on a 107% RCS flow rate. Projections of neutron fluence beyond cycle 11 were based on a 1.5% uprate (3441 MWt) at the start of Cycle 12 and a 8% uprate (3716 MWt) at the start of Cycle 14.

- A credibility evaluation was performed of the Waterford Unit 3 surveillance materials data in accordance with Regulatory Guide 1.99, Revision 2, and is given in Appendix D of this report. The evaluation demonstrates that the surveillance results are credible for the transverse orientation plate and for the weld metal. Therefore, the Chemistry Factor derived in Appendix D for the surveillance plate and weld metal can be used for predicting shift.

- All beltline materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are expected to maintain an upper shelf energy greater than 50 ft-lb throughout the life of the vessel (32 EFPY) as required by 10CFR50, Appendix G[2].

2 INTRODUCTION

This report presents the results of the examination of the Capsule located at 263°, the second capsule to be removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Waterford Unit 3 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Waterford Unit 3 reactor pressure vessel materials was designed by Combustion Engineering. A description of the surveillance program and the pre-irradiation mechanical properties of the reactor vessel materials is presented in Reference 3. The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-73, "Standard Practice for Conducting Surveillance for Light-Water Cooled Nuclear Power Reactor Vessels". Capsule 263° was removed from the reactor after 13.83 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.

This report summarizes the testing of and the post-irradiation data obtained from surveillance capsule located at 263°, removed from the Waterford Unit 3 reactor vessel and discusses the analysis of the data.

3 BACKGROUND

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy, ferritic pressure vessel steels such as A533 Grade B Class 1 (base material of the Waterford Unit 3 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[4]. 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[5]) 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_{IR} or K_{Ic} curve) that appears in Appendix G to the ASME Code[4]. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. The K_{Ic} curve is a lower bound of crack initiation fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} or K_{Ic} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Operating limits can then be determined utilizing these allowable stress intensity factors. (Code Case N-640 allows the use of the K_{Ic} curve as an alternative to the K_{IR} curve.)

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 surveillance program, such as the Waterford Unit 3 reactor vessel radiation surveillance program[6], 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_{Ia} curve and, in turn, to set operating limits for the nuclear power plant that take into account the effects of irradiation on the reactor vessel materials.

4 DESCRIPTION OF PROGRAM

Six surveillance capsules for monitoring the effects of neutron exposure on the Waterford Unit 3 reactor pressure vessel core region (beltline) materials were inserted in the reactor vessel prior to initial plant start-up. The capsules were positioned in the reactor vessel between the core barrel and the vessel wall at locations shown in Figure 4-1. The vertical center of the capsule coincides with the vertical center of the core.

Capsule 263° was removed after 13.83 effective full power years (EFPY) of plant operation. This capsule contained Charpy V-notch impact and tensile specimens made from reactor vessel lower shell course plate M-1004-2, submerged arc weld metal identical to the beltline region girth weld seam and heat-affected-zone (HAZ) metal. The surveillance program weld and the reactor vessel girth seam weld were fabricated using weld wire heat 88114 using Linde 0091 flux[18]. Standard Reference Material from HSST-01MY Plate was included within capsule 263° in addition to the reactor vessel materials.

Test specimens obtained from lower shell plate M-1004-2 (after the heat treatment and forming of the plate) were taken at least one plate thickness from the quenched ends of the plate. All plate and HAZ test specimens were machined from the 1/4 thickness location of the plate. All specimens were removed after performing a simulated post-weld stress-relieving treatment on the test material. All heat-affected-zone specimens were obtained from the weld heat-affected-zone of plate M-1004-2. (The HAZ metal specimens were obtained adjacent to the weldment joining plates M-1004-1 and M-1004-2. The surveillance program weld specimens were obtained from the weldment joining plates M-1004-1 and M-1004-3.)

Charpy V-notch impact specimens from plate M-1004-2 were machined in two orientations. One set of specimens was machined with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation). The other set of specimens from plate M-1004-2 was machined with the transverse axis of the specimen perpendicular to the major working direction of the plate (transverse orientation). The Charpy V-notch specimens from the weld metal were machined with the longitudinal axis of the specimen transverse to the weld direction with the notch oriented in the direction of the weld. Tensile specimens from plate M-1004-2 were machined in with the longitudinal axis of the specimen normal to the major working direction of the plate (transverse orientation). Tensile specimens from the weld metal were oriented with the longitudinal axis of the specimen transverse to the weld direction. Capsule 263° contained neutron flux monitors of sulfur, iron, titanium, nickel (cadmium-shielded), aluminum-cobalt (cadmium-shielded and unshielded), copper (cadmium shielded) and uranium (cadmium-shielded and unshielded).

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

80% Au, 20% Sn	Melting Point 536°F (280°C)
90% Pb, 5% Sn, 5% Ag	Melting Point 558°F (292°C)
2.5% Ag, 97.5% Pb	Melting Point 580°F (304°C)
1.75% Ag, 0.75% Sn, 97.5% Pb	Melting Point 590°F (310°C)

The arrangement of the various mechanical test specimens, dosimeters and thermal monitors contained in capsule 263° is shown in Figure 4-2. A typical Waterford Unit 3 surveillance capsule Charpy impact compartment assembly is shown in Figure 4-3. A typical Waterford Unit 3 surveillance capsule tensile and flux-monitor compartment assembly is shown in Figure 4-4.

The heat treatment for the plate material consisted of austenitization at 1575°F ±50°F for 4 hours, water quenched, and tempered at 1220 °F ±25°F for 4 hours. The surveillance plates received a 40 hour stress relief at 1150°F ±25°F followed by furnace cooling to 600 °F. The weldment received a final 40 hour and 30 minute stress relief at 1100 to 1175°F as documented in Reference 6.

The copper and nickel contents (in weight percent) of the surveillance plate and weld materials and for the correlation monitor material are as follows:

Surveillance Material	Copper Content	Nickel Content	Data Source
Plate M-1004-2	0.03	0.58	Waterford Unit 3 FSAR
Weld	0.05	0.16	Waterford Unit 3 FSAR
Correlation Monitor Material	0.174	0.665	NUREG/CR-6551

The sources are detailed below:

a) Waterford Unit 3 Final Safety Analysis Report, through Revision 12-A, January 2003.
b) Database in NUREG/CR-6551, Improved Embrittlement Correlations for Reactor Pressure Vessel Steels, November 1998.

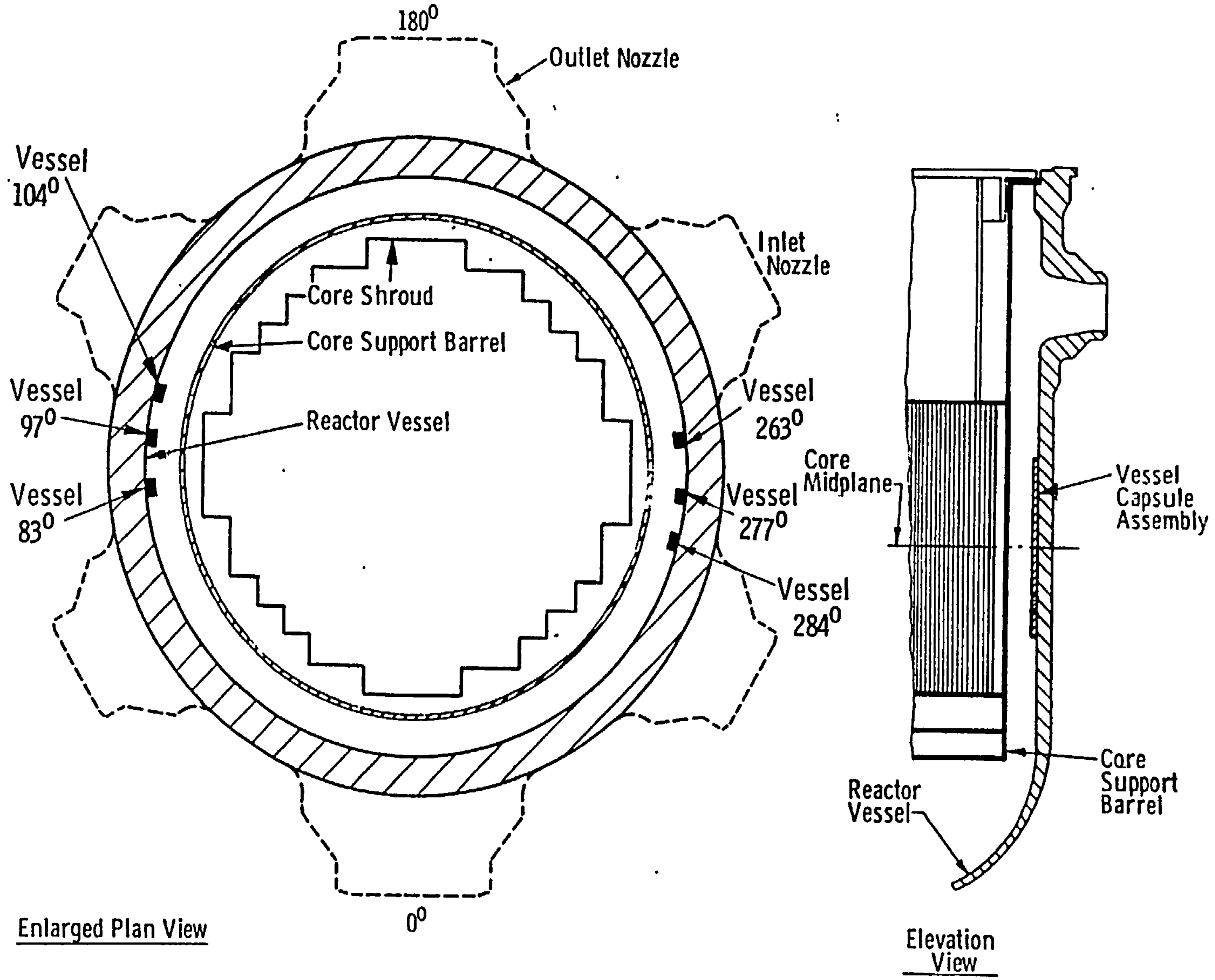


Figure 4-1. Arrangement of Surveillance Capsules in the Waterford Unit 3 Reactor Vessel

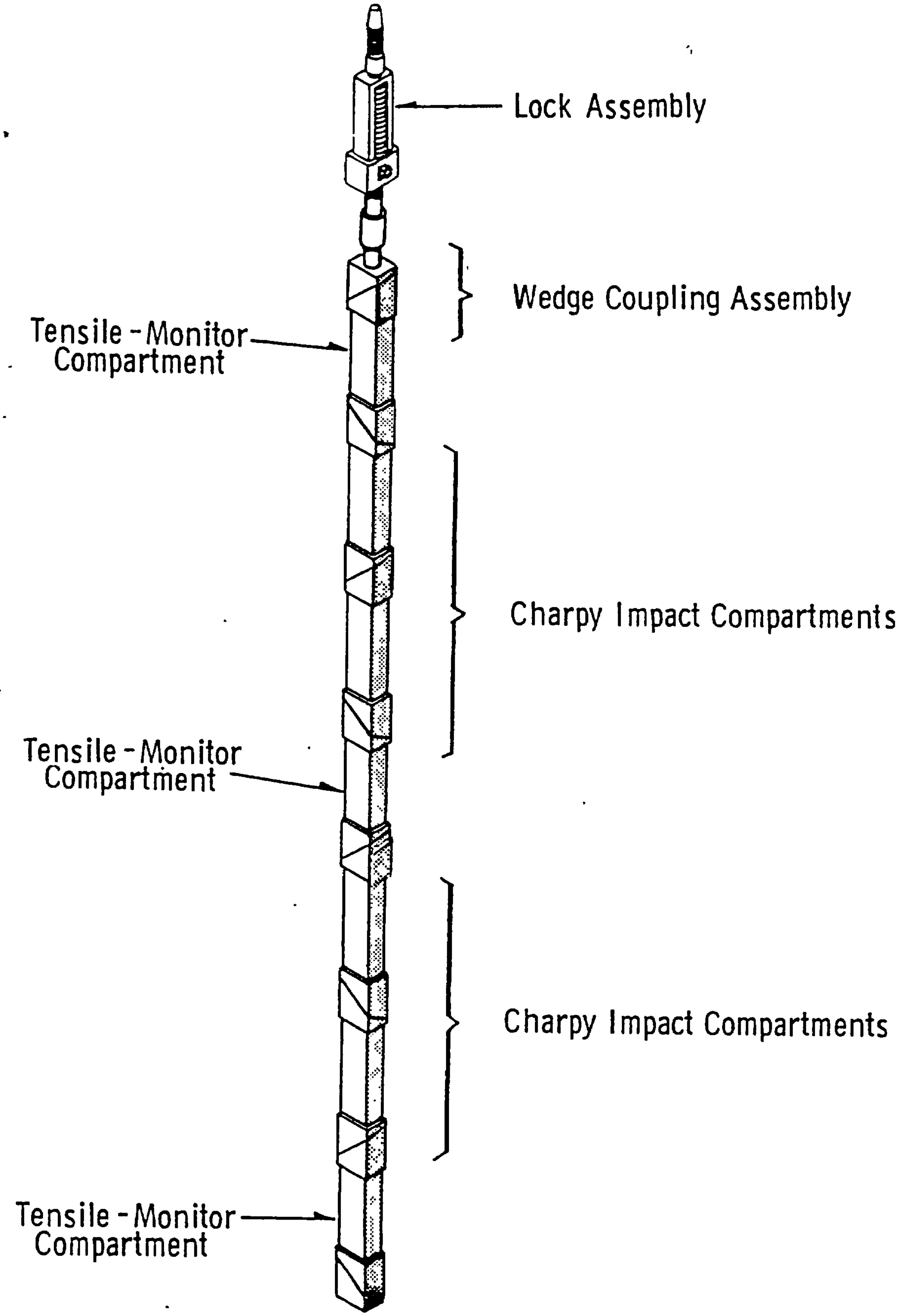


Figure 4-2 Typical Waterford Unit 3 Surveillance Capsule Assembly

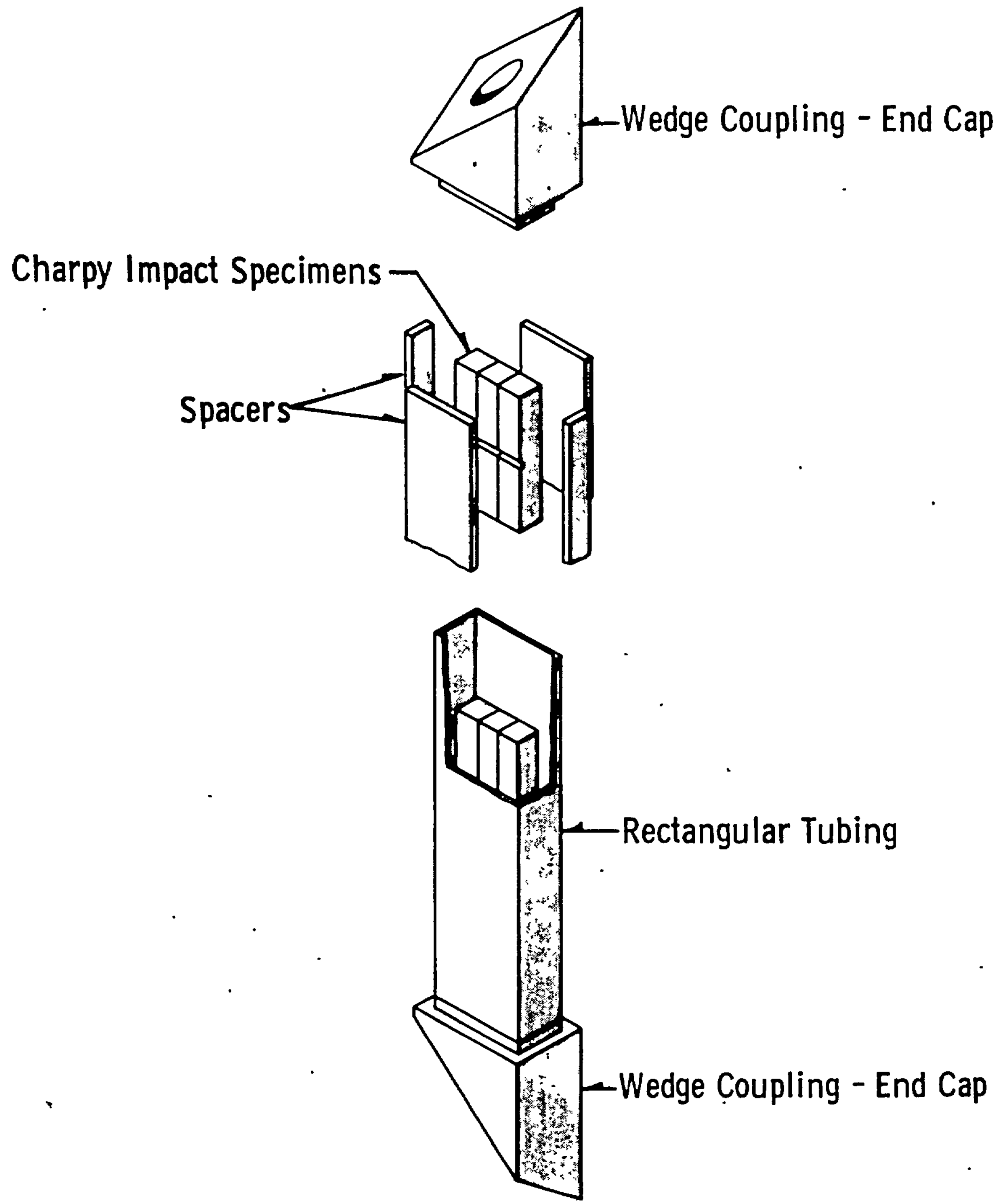


Figure 4-3 Typical Waterford Unit 3 Surveillance Capsule Charpy Impact Compartment Assembly

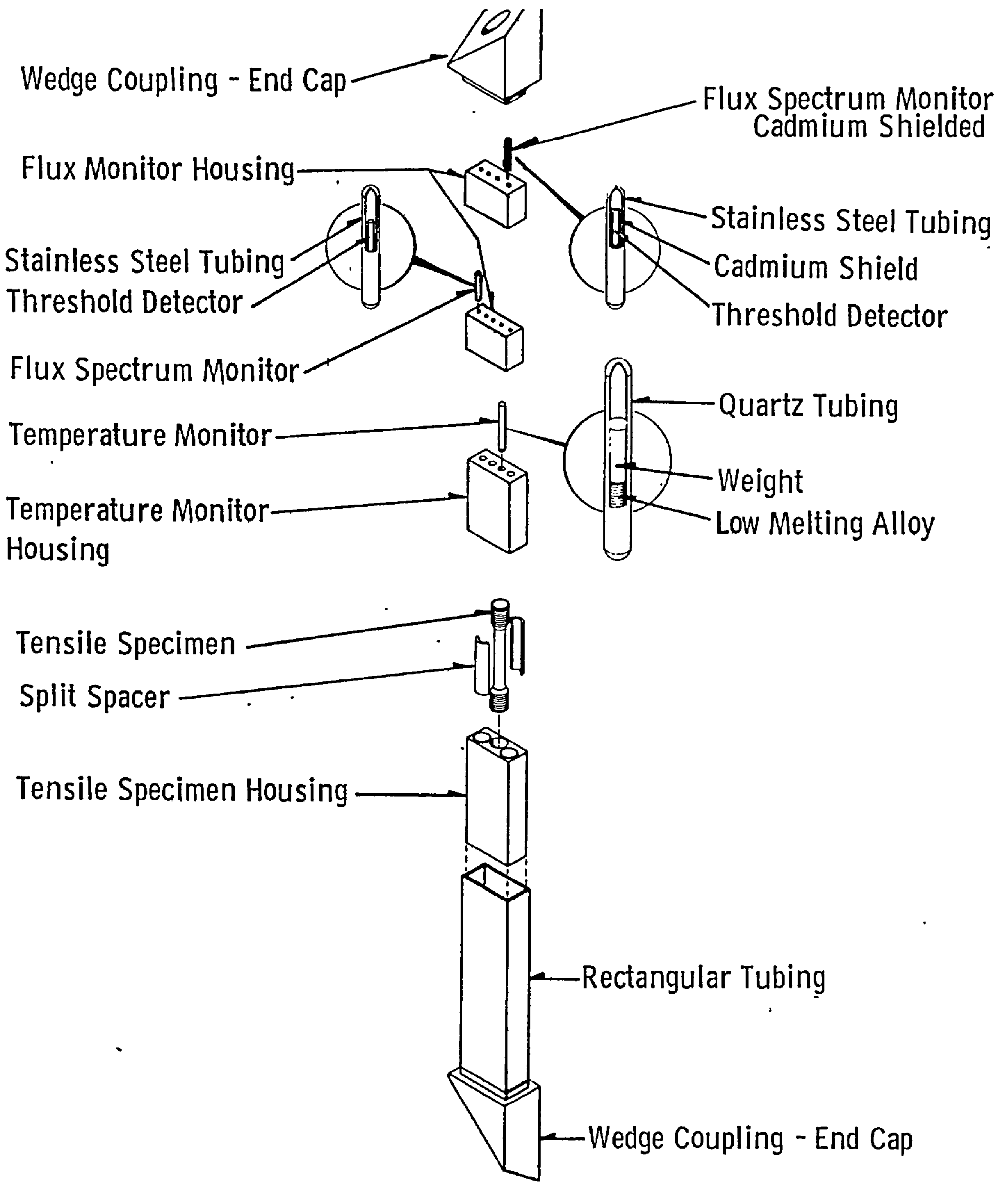


Figure 4-4 Typical Waterford Unit 3 Surveillance Capsule Tensile and Flux-Monitor Compartment Assembly

5 TESTING OF SPECIMENS FROM CAPSULE 263°

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 Standard Practice E185-82[7], and Westinghouse Procedure RMF 8402, Revision 2 as modified by Westinghouse RMF Procedures 8102, Revision 1, and 8103, 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 lists in TR-C-MCS-001[3]. No discrepancies were found.

Examination of the four low-melting, eutectic alloy thermal monitors indicated that the two lowest melting point monitors melted. Based on this examination, the maximum temperature to which the test specimens were exposed to was between 559°F and 579°F.

The Charpy impact tests were performed per ASTM Standard Test Method E23-98[8] and RMF Procedure 8103, Revision 1, on a Tinius-Olsen Model 74, 358J machine. The tup (striker) of the Charpy impact test machine is instrumented with an Instron Dynatup Impulse 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:

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

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

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

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

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

$$\sigma_y = 33.3 * P_{GY} \qquad (3)$$

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

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

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

Percent shear was determined from post-fracture photographs using the ratio-of-areas methods in compliance with ASTM Standard Test Method A370-97[9] . 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 Standard Test Methods E8-99[10] and E21-92(1998)[11], and RMF Procedure 8102, Revision 1. 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 cross-head speed of 0.05 inches per minute throughout the test.

Extension measurements were made with a linear variable displacement transducer (LVDT) 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[12].

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 gripper. In the test configuration, with a slight load on the specimen, a plot of specimen temperature versus upper and lower tensile machine gripper 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 263°, which received a fluence of 1.45×10^{19} n/cm^2 (E > 1.0 MeV) in 13.83 EFPY of operation, are presented in Tables 5-1 through 5-8 and are compared with unirradiated results from TR-C-MCS-002-P[6] as shown in Figures 5-1 through 5-12.

The transition temperature increases and upper shelf energy decreases for the capsule 263° materials are summarized in Table 5-9. These results led to the following observations:

The reactor vessel lower shell plate M-1004-2 Charpy specimens, oriented with the longitudinal axis of the specimen normal to the major working direction of the plate (transverse orientation), was irradiated to 1.45×10^{19} n/cm^2 (E> 1.0MeV). This resulted in a 30 ft-lb transition temperature decrease of 9.1°F and a 50 ft-lb transition temperature increase of 8.1°F, with an irradiated 30 ft-lb transition temperature of -33.6°F and an irradiated 50 ft-lb transition temperature of 11.0°F for the transversely oriented plate specimens. The 30 ft-lb transition temperature change was taken as 0°F rather than assume a negative shift.

The HSST Plate 01MY correlation monitor material Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation), was irradiated to 1.45×10^{19} n/cm^2 (E> 1.0 MeV). This resulted in a 30 ft-lb transition temperature increase of 150.5°F and a 50 ft-lb transition temperature increase of 151.3°F, with an irradiated 30 ft-lb transition temperature of 184.9°F and an irradiated 50 ft-lb transition temperature of 211.4°F for the longitudinally oriented plate specimens.

Irradiation of the weld metal Charpy specimens to 1.45×10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 6.9°F and a 50 ft-lb transition temperature increase of 13.8°F. This results in an irradiated 30 ft-lb transition temperature of -77.7°F and an irradiated 50 ft-lb transition temperature of -51.4°F for the surveillance weld material.

Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens to 1.45×10^{19} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 25.1°F and a 50 ft-lb transition temperature increase of 27.9°F. This results in an irradiated 30 ft-lb transition temperature of -92.0°F and an irradiated 50 ft-lb transition temperature of 62.1°F.

Irradiation of the lower shell plate M-1004-2 (transverse orientation) to 1.45×10^{19} n/cm^2 (E> 1.0 MeV) resulted in an average upper shelf energy decrease of 10 ft-lb after irradiation. This gives an irradiated average upper shelf energy of 131 ft-lb for the transversely oriented plate specimens.

Irradiation of the correlation monitor material (longitudinal orientation) to 1.45×10^{19} n/cm^2 (E > 1.0 MeV) resulted in an average upper shelf energy decrease of 20 ft-lb after irradiation. This gives an irradiated average upper shelf energy of 113 ft-lb for the longitudinal oriented plate specimens.

Irradiation of the weld metal Charpy specimens to 1.45 x 10^{19} n/cm^2 (E> 1.0 MeV) resulted in an average energy decrease of 11 ft-lb after irradiation. This gives an irradiated average upper shelf energy of 145 ft-lb for the weld metal specimens.

Irradiation of the weld HAZ metal Charpy specimens to 1.45 x 10^{19} n/cm^2 (E > 1.0 MeV) resulted in an average energy decrease of 7 ft-lb after irradiation. This gives an irradiated average upper shelf energy of 163 ft-lb for the weld HAZ metal.

A comparison is presented in Table 5-10 of the Waterford Unit 3 reactor vessel beltline material test results with the Regulatory Guide 1.99, Revision 2[1] predictions. The following observations are made:

- The measured 30 ft-lb shift in transition temperature values for all the surveillance plate and weld materials from capsule 263° is less than the Regulatory Guide 1.99, Revision 2, predictions. This is indicative of the excellent controls (both copper content and cleanliness of the plate) that were placed on the Waterford Unit 3 reactor vessel materials.

- The measured 30 ft-lb shift in transition temperature value for the HSST Plate O1MY correlation monitor material was within 5° F of the Regulatory Guide 1.99 prediction. This excellent agreement indicates that the irradiation environment has been accurately defined for both the correlation monitor material and the surveillance materials.

- The measured percent decrease in upper shelf energy of the materials from the 263° surveillance capsule is less than the Regulatory Guide 1.99, Revision 2 predictions.

- A similar analysis is provided in Table 5-10 for the results of capsule 97°. The measured 30 ft-lb shift in transition temperature values for all the surveillance plate and weld materials from capsule 97° is less or comparable to the Regulatory Guide 1.99, Revision 2 predictions.

- Further comparisons are made in the credibility evaluation presented in Appendix D

The fracture appearance of each irradiated Charpy specimen from the various surveillance capsule 263° materials is shown in Figures 5-13 through 5-16. The fracture surfaces show an increasingly ductile (i.e., tougher) appearance with increasing test temperature. The load-time records for individual instrumented Charpy specimen tests are shown in Appendix A.

All beltline materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are expected to maintain an upper shelf energy of no less than 50 ft-lb throughout the life of the vessel (32 EFPY) as required by 10CFR50, Appendix G.

The Charpy V-notch data presented in this report is based on a plot of all capsule data using CVGRAPH, Version 4.1, which is a hyperbolic tangent curve-fitting program. Appendices B and C contain the CVGRAPH, Version 4.1, Charpy V-notch plots and the program input data, and the Charpy V-notch shift results for each surveillance material from the hyperbolic tangent curve-fitting.

5.3 TENSILE TEST RESULTS

The results of the tensile tests performed on the various materials contained in capsule 263° irradiated to 1.45×10^{19} n/cm^2 (E > 1.0 MeV) are presented in Table 5-11 and are compared with unirradiated results from TR-C-MCS-002-P[6] as shown in Figures 5-17 through 5-19.

The results of the room temperature (70 to 75 °F) tensile tests performed on the lower shell plate M-1004-2 (transverse orientation) indicated that irradiation to 1.45×10^{19} n/cm^2 (E> 1.0 MeV) caused an approximate increase of 2 ksi in the 0.2 percent offset yield strength and approximately a 4 ksi increase in the ultimate tensile strength when compared to unirradiated data[6] (Figure 5-17).

The results of the room temperature tensile tests performed on the surveillance weld metal indicated that irradiation to 1.45×10^{19} n/cm^2 (E > 1.0 MeV) caused no significant change in the 0.2 percent offset yield strength and a 4 ksi increase in the ultimate tensile strength when compared to unirradiated data[6] (Figure 5-18).

The results of the tensile tests performed on the surveillance HAZ metal indicated that irradiation to 1.45×10^{19} n/cm^2 (E > 1.0 MeV) caused a 1 ksi increase in the 0.2 percent offset yield strength and 2 ksi increase in the ultimate tensile strength when compared to unirradiated data[6] (Figure 5-19).

The fractured tensile specimens for the lower shell plate M-1004-2 material are shown in Figure 5-20. The fractured tensile specimens for the surveillance weld metal and heat-affected-zone material are shown in Figures 5-21 and 5-22, respectively. The engineering stress-strain curves for the tensile tests are shown in Figures 5-23 through 5-25.

Table 5-1 Charpy V-notch Data for the Waterford Unit 3 Plate M-1004-2 Irradiated to a Fluence of 1.45 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	%
25E	-40	-40	19	26	14	0.36	10
23D	-30	-34	14	19	9	0.23	5
25J	-10	-23	55	75	37	0.94	15
246	0	-18	46	62	32	0.81	15
215	25	-4	75	102	50	1.27	45
23E	50	10	74	100	51	1.30	50
24U	75	24	66	89	49	1.24	45
22U	125	52	105	142	70	1.78	75
232	160	71	111	151	73	1.85	80
22Y	200	93	128	174	77	1.96	100
243	225	107	135	183	82	2.08	100
24M	350	177	131	178	75	1.91	100

Table 5-2 Charpy V-notch Data for the HSST Plate 01MY Correlation Monitor Material Irradiated to a Fluence of 1.45 x 1019 n/cm2 (E> 1.0 MeV), Longitudinal Orientation

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	Ft-lbs	Joules	mils	mm	%
A47	-30	-34	4	5	3	0.08	2
A4E	50	10	9	12	6	0.15	10
A46	125	52	12	16	8	0.20	20
A3L	175	79	23	31	17	0.43	25
A4C	200	93	32	43	25	0.64	30
A3P	240	116	92	125	60	1.52	65
A4A	275	135	77	104	58	1.47	75
A41	320	160	116	157	69	1.75	100
A43	350	177	108	146	69	1.75	100
A4K	375	191	117	159	74	1.88	100
A3E	425	218	114	155	68	1.73	100
A3Y	460	238	108	146	67	1.70	100

Table 5-3 Charpy V-notch Data for the Waterford Unit 3 Surveillance Weld Metal Irradiated to a Fluence of 1.45 x 1019 n/cm2 (E> 1.0 MeV)

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
333	-175	-115	4	5	0	0.00	5
332	-125	-87	7	9	1	0.03	10
363	-75	-59	29	39	16	0.41	20
34B	-50	-46	57	77	39	0.99	35
35Y	-25	-32	76	103	49	1.24	65
361	0	-18	94	127	61	1.55	75
34K	25	-4	121	164	77	1.96	90
33D	50	10	130	176	82	2.08	90
33U	75	24	134	182	83	2.11	90
365	100	38	143	194	87	2.21	100
37A	150	66	139	188	85	2.16	100
35L	200	93	153	207	83	2.11	100

Table 5-4 Charpy V-notch Data for the Waterford Unit 3 Heat Affected Zone Metal Irradiated to a Fluence of 1.45 x 1019 n/cm2 (E> 1.0 MeV)							
Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
4IP	-175	-115	4	5	1	0.03	5
41D	-125	-87	29	39	15	0.38	10
43E	-75	-59	31	42	23	0.58	45
436	-25	-32	83	113	47	1.19	60
42T	0	-18	114	155	67	1.70	75
41U	25	-4	118	160	75	1.91	90
45B	50	10	144	195	80	2.03	100
471	75	24	152	206	69	1.75	100
46L	110	43	142	193	74	1.88	100
473	150	66	182	247	74	1.88	100
446	225	107	163	221	77	1.96	100
44A	325	163	192	260	69	1.75	100

Table 5-5 Instrumented Charpy Impact Test Results for the Waterford Unit 3 Plate M-1004-2

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in²)			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								
25E	-40	19	153	67	86	3455	0.15	4199	0.22	4051	0	115	127
23D	-30	14	113	60	53	3231	0.14	4097	0.21	4087	0	108	122
25J	-10	55	443	325	118	3478	0.15	4477	0.68	4334	0	116	132
246	0	46	371	311	59	3327	0.15	4425	0.67	4369	0	111	129
215	25	75	604	316	289	3269	0.15	4413	0.68	3908	324	109	128
23E	50	74	596	308	288	3185	0.15	4282	0.69	4055	744	106	124
24U	75	105	846	308	538	3220	0.15	4303	0.69	3898	620	107	125
22U	125	66	532	295	237	2943	0.15	4164	0.69	3096	1211	98	118
232	160	111	894	290	605	2913	0.14	4079	0.68	2556	1340	97	116
22Y	200	128	1031	289	742	2900	0.14	4078	0.69	n/a	n/a	97	116
243	225	135	1088	292	796	2858	0.14	4116	0.69	n/a	n/a	95	116
24M	350	131	1056	265	790	2594	0.14	3732	0.69	n/a	n/a	86	105

Table 5-6 Instrumented Charpy Impact Test Results for the HSST Plate 01MY Correlation Monitor Material Irradiated to a Fluence of 1.45 x 1019 n/cm2 (E> 1.0 MeV), Longitudinal Orientation

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in²) Charpy E_D/A	Max. E_M/A	Prop. E_p/A	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)
A47	-30	4	32	16	16	1948	0.13	1948	0.13	1944	0	65	65
A4E	50	9	73	38	34	3504	0.15	3648	0.17	3638	0	117	119
A46	125	12	97	39	58	3318	0.15	3564	0.17	3559	345	110	115
A3L	175	23	185	66	120	3103	0.14	3938	0.22	3889	864	103	117
A4C	200	32	258	144	113	3052	0.14	4073	0.38	4039	1097	102	119
A3P	240	92	741	309	433	3230	0.15	4398	0.67	3540	1304	108	127
A4A	275	77	620	222	398	3079	0.14	4257	0.52	3889	2708	103	122
A41	320	116	935	305	630	3208	0.15	4310	0.68	n/a	n/a	107	125
A43	350	108	870	265	605	3057	0.17	4225	0.63	n/a	n/a	102	121
A4K	375	117	943	299	644	3083	0.15	4200	0.67	n/a	n/a	103	121
A3E	425	114	919	290	629	3024	0.15	4117	0.67	n/a	n/a	101	119
A3Y	460	108	870	272	598	2985	0.16	4103	0.64	n/a	n/a	99	118

Table 5-7 Instrumented Charpy Impact Test Results for the Waterford Unit 3 Surveillance Weld Metal

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2) Charpy E_D/A	Max. E_M/A	Prop. E_p/A	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)
333	-175	4	31	15	16	1915	0.11	1968	0.12	1963	0	64	65
332	-125	8	63	32	30	3578	0.15	3587	0.15	3580	0	119	119
363	-75	29	237	74	163	3766	0.16	4672	0.23	4633	0	125	140
34B	-50	56	450	240	210	3923	0.15	4621	0.51	4408	337	131	142
35Y	-25	74	596	248	348	3616	0.15	4551	0.53	4212	1240	120	136
361	0	92	739	332	407	3641	0.15	4524	0.68	3673	1513	121	136
34K	25	121	975	326	649	3549	0.15	4460	0.68	3420	2099	118	133
33D	50	125	1006	333	673	3457	0.15	4475	0.70	1915	931	115	132
33U	75	129	1037	330	707	3448	0.15	4479	0.70	2153	1356	115	132
365	100	140	1126	309	816	3317	0.15	4224	0.69	n/a	n/a	110	126
37A	150	133	1074	309	765	3226	0.14	4184	0.69	n/a	n/a	107	123
35L	200	147	1187	303	884	3065	0.14	4152	0.69	n/a	n/a	102	120

Table 5-8 Instrumented Charpy Impact Test Results for the Waterford Unit 3 Heat Affected Zone Material

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2) Charpy E_D/A	Max. E_M/A	Prop. E_p/A	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)
41P	-175	5	39	19	20	2325	0.12	2403	0.13	2396	0	77	79
41D	-125	31	248	79	169	4110	0.15	5107	0.22	4953	0	137	153
43E	-75	30	244	71	173	3817	0.15	4543	0.22	4461	464	127	139
436	-25	80	647	337	310	3566	0.16	4514	0.71	4034	817	119	135
42T	0	111	895	347	548	3583	0.15	4649	0.71	3859	584	119	137
41U	25	116	934	335	599	3512	0.16	4514	0.71	3585	2542	117	134
45B	50	139	1123	327	796	3448	0.15	4506	0.69	n/a	n/a	115	132
471	75	146	1179	334	845	3566	0.15	4609	0.69	n/a	n/a	119	136
46L	110	138	1108	322	786	3322	0.15	4465	0.69	n/a	n/a	111	130
473	150	175	1410	320	1090	3251	0.14	4423	0.70	n/a	n/a	108	128
44A	325	184	1485	310	1175	2805	0.15	4231	0.71	n/a	n/a	93	117

Table 5-9 Effect of Irradiation to 1.45 x 1019 n/cm2 (E>1.0 MeV) on the Notch Toughness Properties of the Waterford Unit 3

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 M-1004-2 (Transverse)	-24.44	-33.57	0 (-9.1)	-6.73	9.81	16.5	2.89	11.01	8.1	141	131	-10
Correlation Monitor Material (Longitudinal)	34.31	184.87	150.5	41.94	208.53	166.6	60.02	211.36	151.3	133	113	-20
Weld Metal	-84.58	-77.71	6.9	-68.31	-46.80	21.5	-65.19	-51.38	13.8	156	145	-11
HAZ Metal	-117.09	-91.96	25.1	-89.55	-56.57	33.0	-90.08	-62.15	27.9	170	163	-7

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

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

Table 5-10 Comparison of the Waterford Unit 3 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decrease with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence (x 10^{19} n/cm^2)$^{(a)}$	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F)	Measured (°F)	Predicted (%)$^{(b)}$	Measured (%)
Lower Shell Plate M-1004-2 (Longitudinal)^{c}	97°	0.647	18	6	17	8.9
Lower Shell Plate M-1004-2 (Transverse)	97°	0.647	18	28	17	12
	263°	1.45	22	0 (-9)	20.7	7
Surveillance Program Weld Metal	97°	0.647	39	28	17	8
	263°	1.45	49	7	20.7	7
Heat Affected Zone Material	97°	0.647	---	14	---	8
	263°	1.45	---	25	---	4
Correlation Monitor Material	97°	c	c	c	c	c
	263°	1.45	145	150	32	15

Notes:
(a) Calculated Fluences from 97° capsule analysis (BAW-2177) and 263° capsule analysis (section 6 of this report); results (E > 1.0 MeV)
(b) From Figure 2 of Regulatory Guide 1.99, Revision 2, using the Cu values given in Section 4 and the cited capsule fluence values.
(c) No correlation monitor material specimens in 97° capsule. No longitudinal plate specimens in 263° capsule.

Table 5-11 Tensile Specimens From Lower Shell Course Plate M-1004-2, Weld, and Heat Affected Zone Material

Sample Number	Sample Material	Test Temperature (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 (%)
2J2	Plate	75	71.3	93.2	2.89	223.0	58.9	12.0	29.3	74
2KK	Plate	250	66.2	85.5	2.98	186.9	60.7	10.5	22.9	68
2KL	Plate	550	61.1	88.0	3.15	152.8	64.2	9.5	19.9	58
3K3	Weld	75	83.5	96.0	2.83	225.9	57.7	10.5	25.8	74
3JD	Weld	250	75.4	88.5	2.66	198.9	54.2	9.0	22.9	73
3J7[1]	Weld	550	71.3	87.7	3.29	166.2	67.0	-	-	60
4JB	HAZ	75	70.3	93.8	2.95	200.4	60.1	7.0	19.0	70
4J1	HAZ	250	65.7	86.5	4.74	365.5	96 6	6.0	18.1	74
4KA[2]	HAZ	550	67.2	88.8	3.34	156.0	67.9	-	-	56

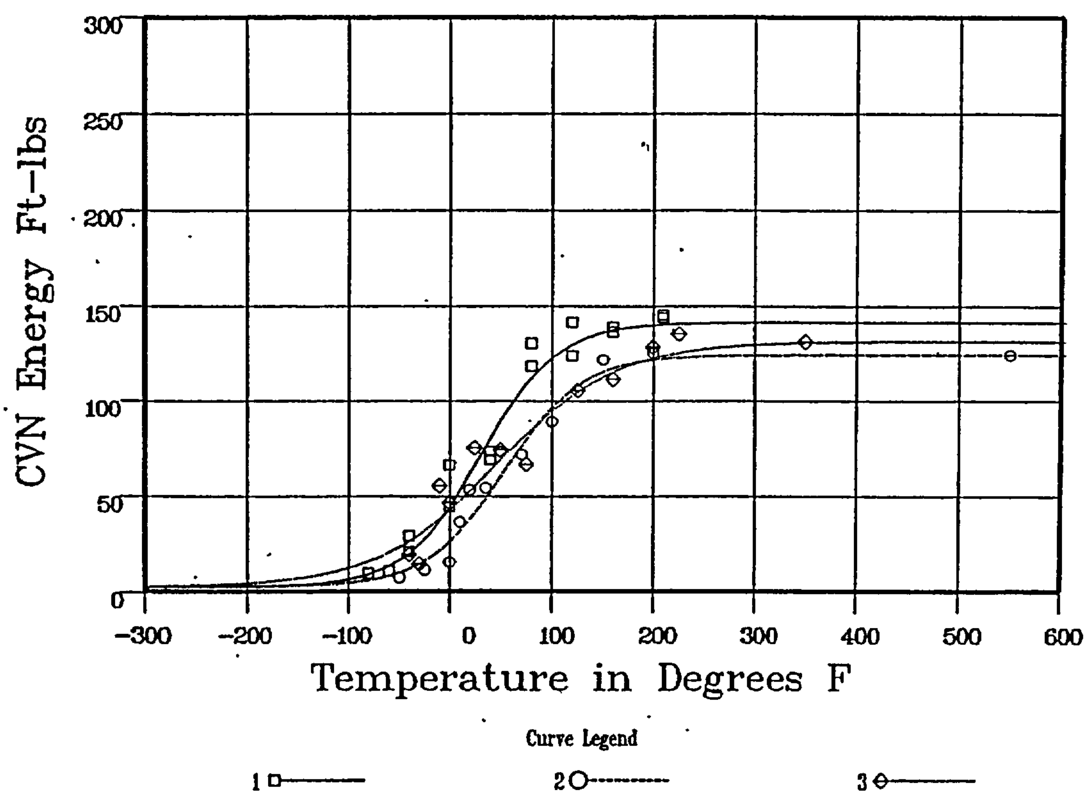
1) Specimen broke outside of the gage section
2) Specimen broke in knife edge of clip gage

SHELL PLATE M-1004-2 (TRANS)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:51:35 on 09-30-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	141	0	-24.44	0	2.89	0
2	0	2.19	0	124	-17	3.45	27.89	33.1	30.2
3	0	2.19	0	131	-10	-33.57	-9.13	11.01	8.12



Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	PLATE SA533B1	TL	M-1004-2
2	WF3	W-97	PLATE SA533B1	TL	M-1004-2
3	WF3	W-263	PLATE SA533B1	TL	M-1004-2

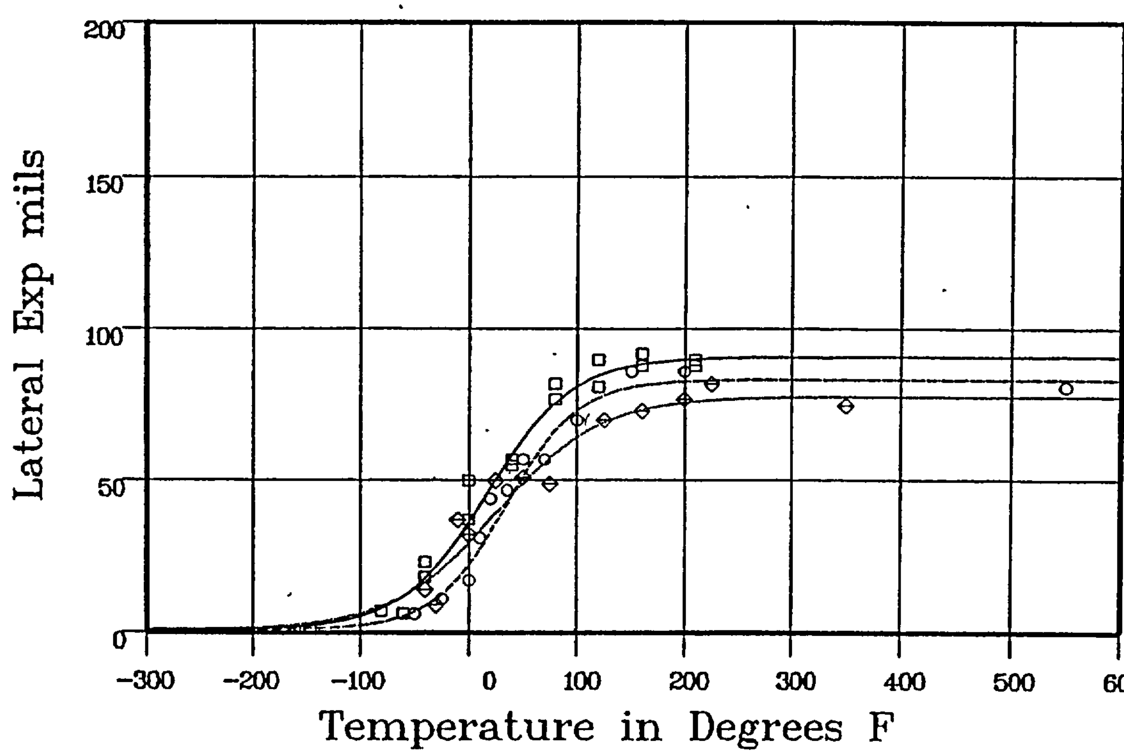
Figure 5-1 Charpy V-Notch Impact Energy vs. Temperature for Waterford Unit 3 Reactor Vessel Lower Shell Plate M-1004-2 (Transverse Orientation)

SHELL PLATE M-1004-2 (TRANS)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:37:03 on 10-01-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	90.9	0	-6.73	0
2	0	83.4	-7.5	18.83	25.57
3	0	77.86	-13.03	9.81	16.55



Curve Legend

1 □———— 2 ○-------- 3 ◇————

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	PLATE SA533B1	TL	M-1004-2
2	WF3	W-97	PLATE SA533B1	TL	M-1004-2
3	WF3	W-263	PLATE SA533B1	TL	M-1004-2

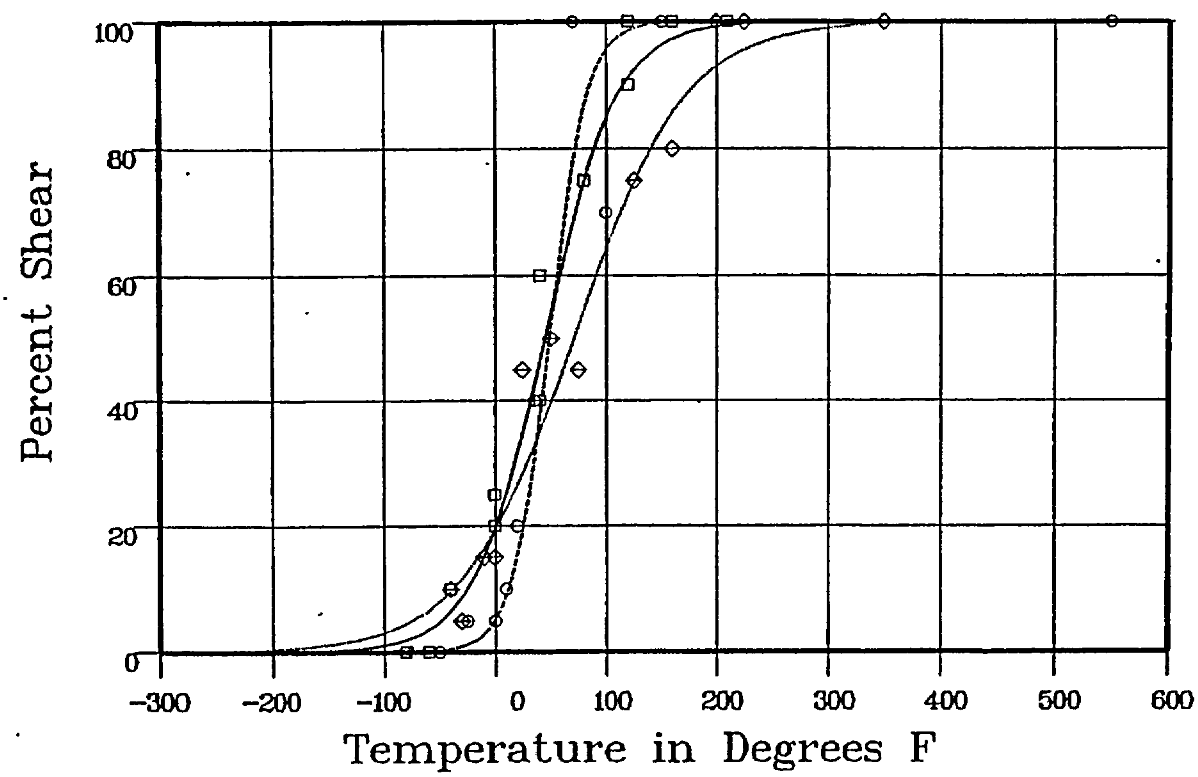
Figure 5-2 Charpy V-Notch Lateral Expansion vs. Temperature for Waterford Unit 3 Reactor Vessel Lower Shell Plate M-1004-2 (Transverse Orientation)

SHELL PLATE M-1004-2 (TRANS)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:41:58 on 10-01-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	40.31	0
2	0	44.06	3.75
3	0	66.3	25.98



Curve Legend

1 □——— 2 ○--------- 3 ◇———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	PLATE SA533B1	TL	M-1004-2
2	WF3	W-97	PLATE SA533B1	TL	M-1004-2
3	WF3	W-263	PLATE SA533B1	TL	M-1004-2

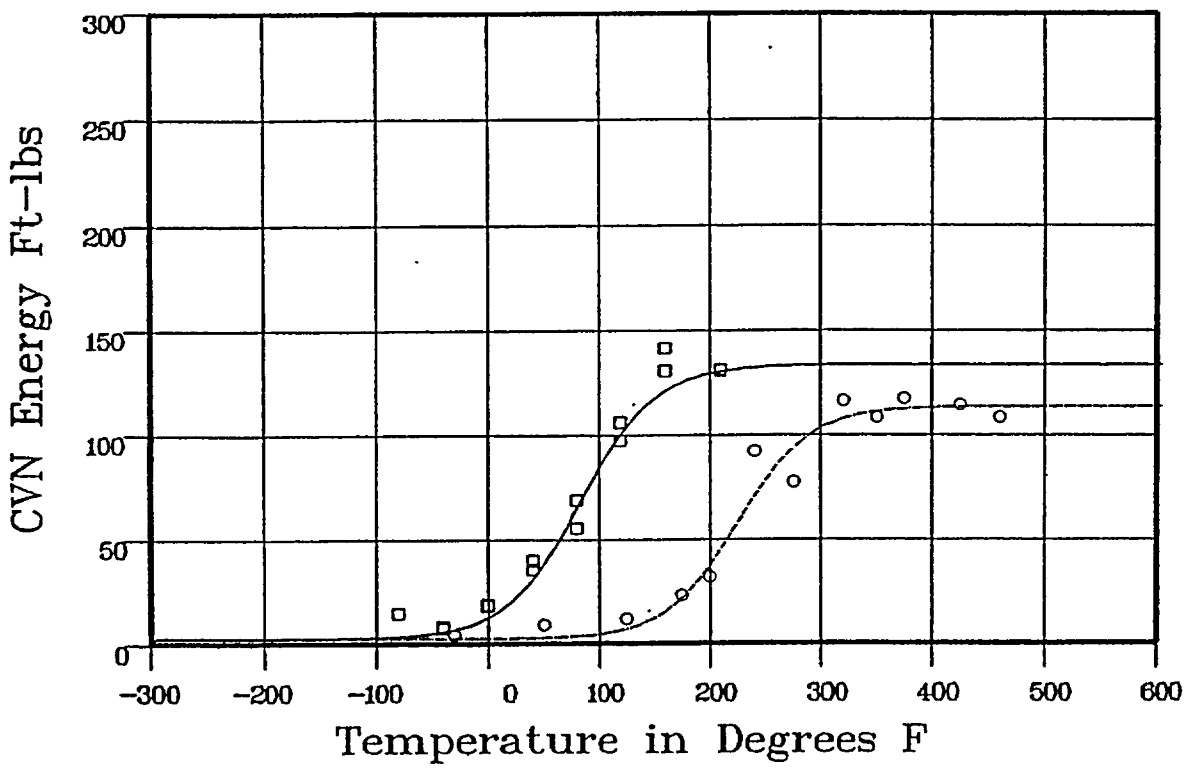
Figure 5-3 Charpy V-Notch Percent Shear vs. Temperature for Waterford Unit 3 Reactor Vessel Lower Shell Plate M-1004-2 (Transverse Orientation)

STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:20:51 on 10-01-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	133	0	34.31	0	60.02	0
2	0	2.19	0	113	-20	184.87	150.55	211.36	151.33



Curve Legend

1 □———— 2 ○----------

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	SRM HSST01	LT	A1008-1
2	WF3	W-263	SRM SA533B1	LT	A1008-1

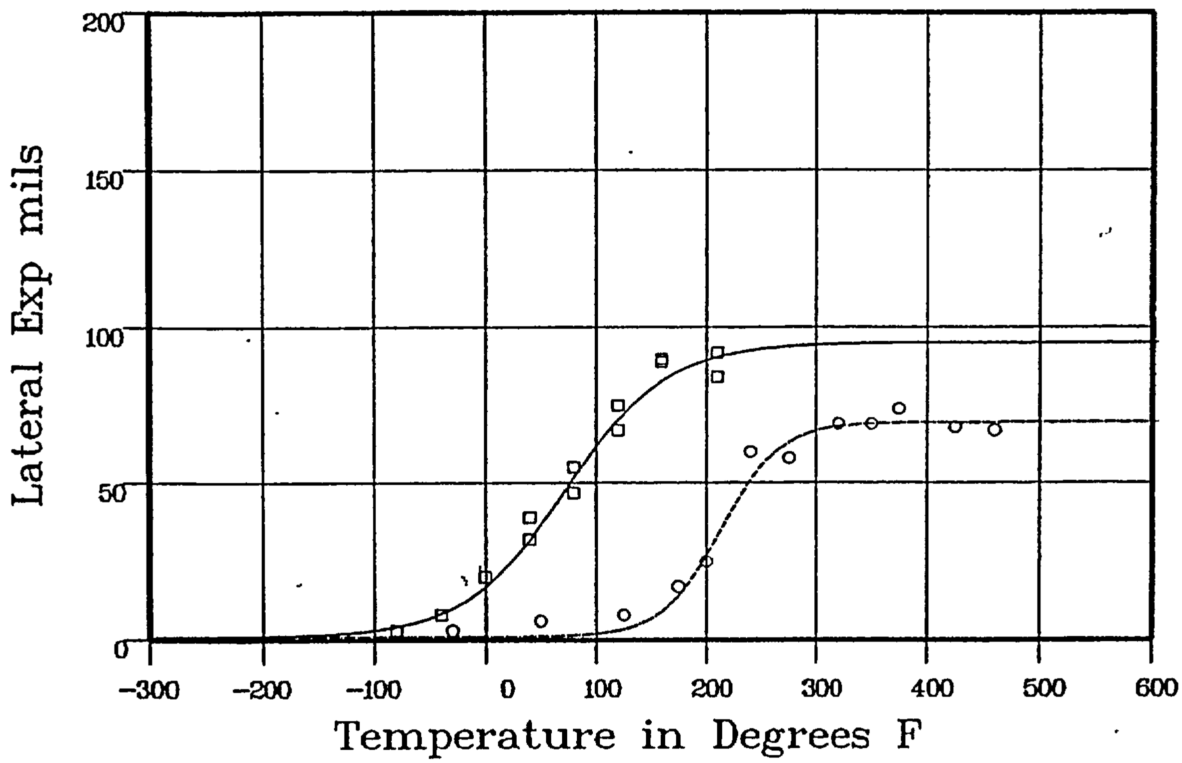
Figure 5-4 Charpy V-Notch Impact Energy vs. Temperature for HSST Plate O1MY Correlation Monitor Material (Longitudinal Orientation)

STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:23:23 on 10-01-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	95.25	0	41.94	0
2	0	69.45	-25.79	208.53	166.58



Curve Legend

1 □———— 2 ○---------

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	SRM HSST01	LT	A1008-1
2	WF3	W-263	SRM SA533B1	LT	A1008-1

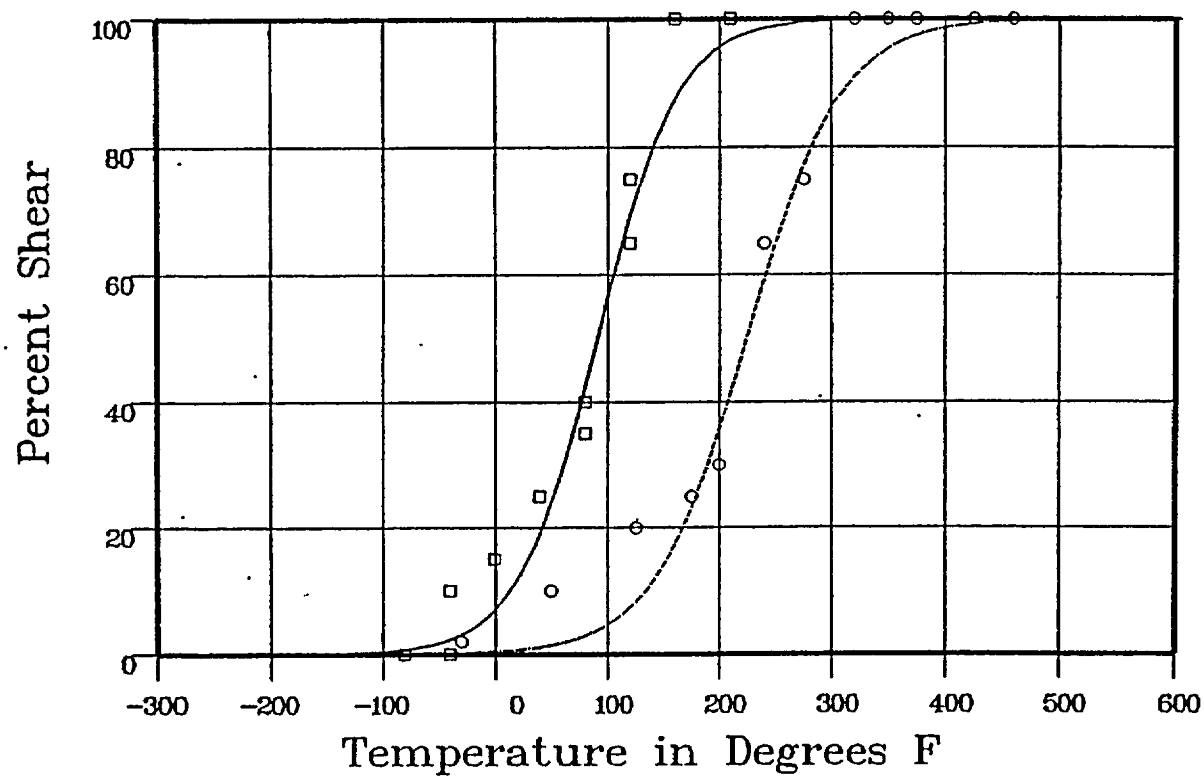
Figure 5-5 Charpy V-Notch Lateral Expansion vs. Temperature for HSST Plate 01MY Correlation Monitor Material (Longitudinal Orientation)

STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:25:54 on 10-01-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	86.38	0
2	0	219.84	133.46



Curve Legend

1 □———— 2 O----------

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	SRM HSST01	LT	A1008-1
2	WF3	W-263	SRM SA533B1	LT	A1008-1

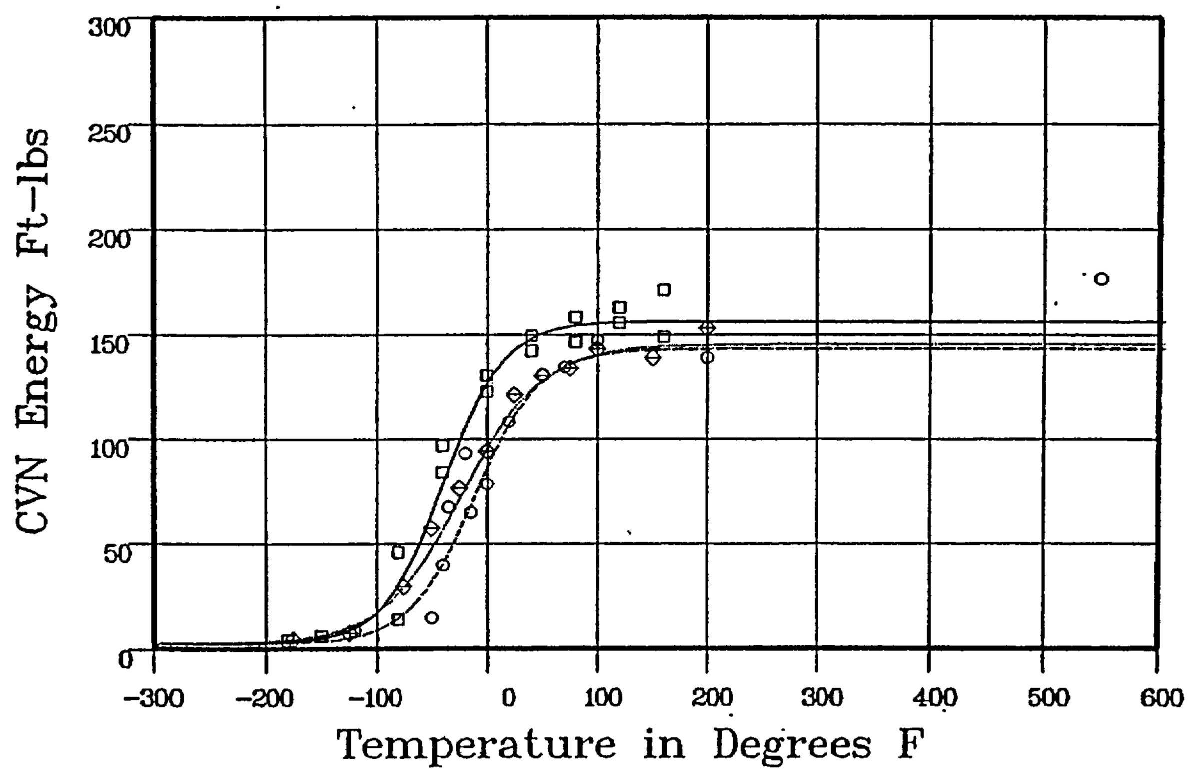
Figure 5-6 Charpy V-Notch Percent Shear vs. Temperature for HSST Plate 01MY Correlation Monitor Material (Longitudinal Orientation)

SURVEILLANCE PROGRAM WELD METAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:13:21 on 10-01-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	156	0	-84.58	0	-65.19	0
2	0	2.19	0	143	-13	-56.36	28.22	-34.87	30.31
3	0	2.19	0	145	-11	-77.71	6.87	-51.38	13.8



Curve Legend

1 □——— 2 ○-------- 3 ◇———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	WELD L 124/0091		88114/0145
2	WF3	W-97	WELD L 124/0091		88114/0145
3	WF3	W-263	WELD L 124/0091		88114/0145

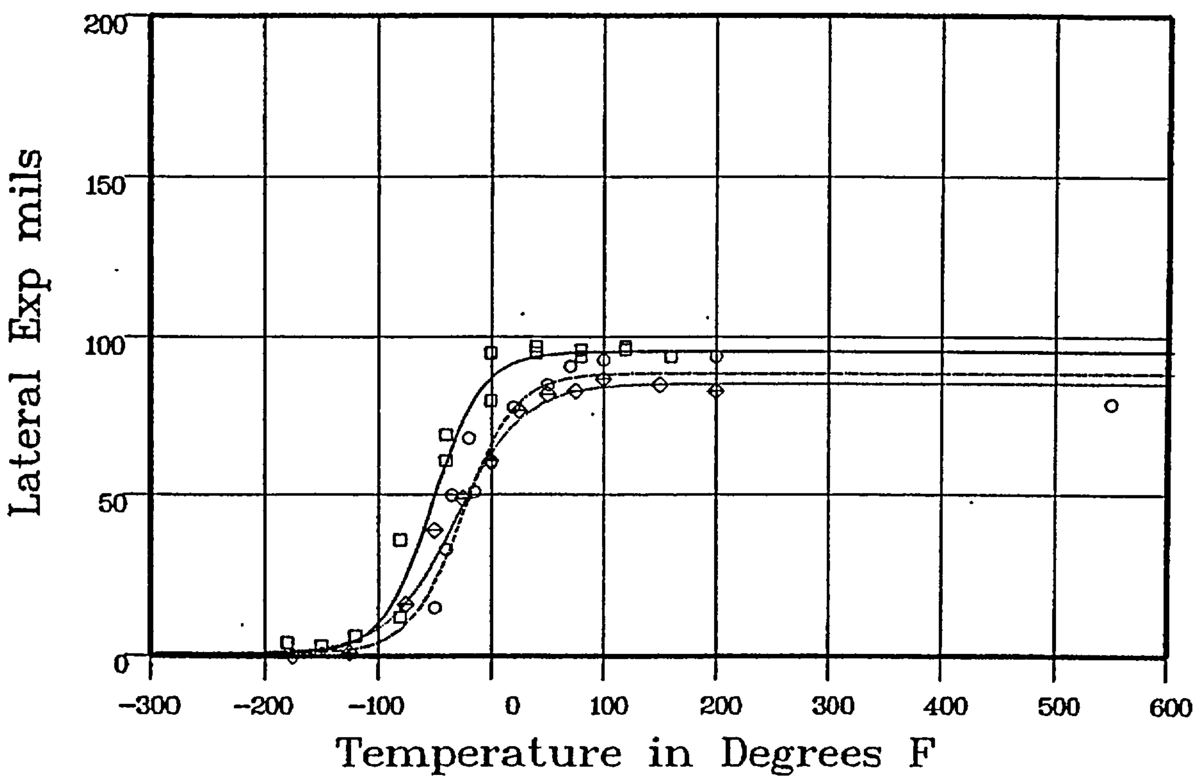
Figure 5-7 Charpy V-Notch Impact Energy vs. Temperature for Waterford Unit 3 Reactor Vessel Surveillance Weld Material

SURVEILLANCE PROGRAM WELD METAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:17:12 on 10-01-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	95.49	0	-68.31	0
2	0	88.67	-6.81	-39.63	28.68
3	0	85.35	-10.13	-46.8	21.51



Curve Legend

1 □——— 2 ○--------- 3 ◇———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	WELD L 124/0091		88114/0145
2	WF3	W-97	WELD L 124/0091		88114/0145
3	WF3	W-263	WELD L 124/0091		88114/0145

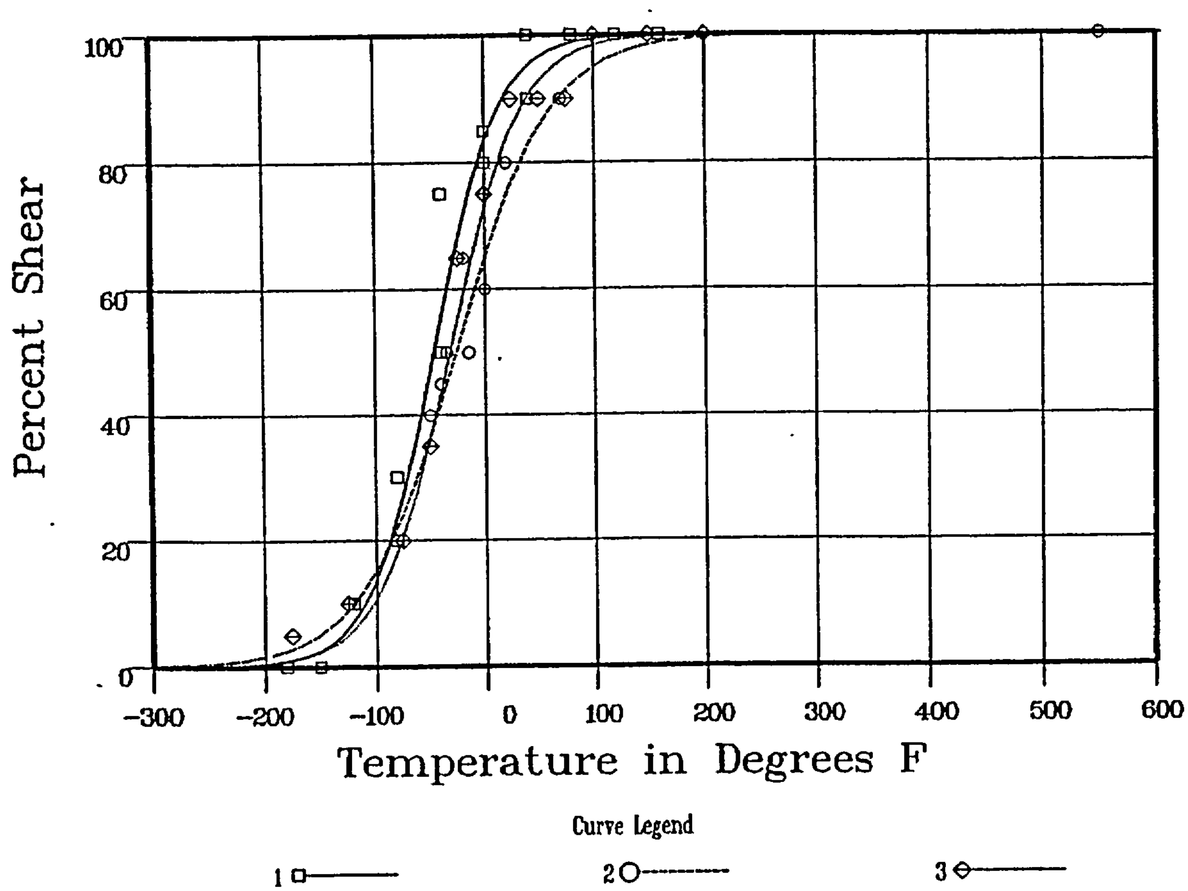
Figure 5-8 Charpy V-Notch Lateral Expansion vs. Temperature for Waterford Unit 3 Reactor Vessel Surveillance Weld Metal

SURVEILLANCE PROGRAM WELD METAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:21:23 on 10-01-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	-51.09	0
2	0	-30.93	20.15
3	0	-36.37	14.71



Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	WELD L 124/0091		88114/0145
2	WF3	W-97	WELD L 124/0091		88114/0145
3	WF3	W-263	WELD L 124/0091		88114/0145

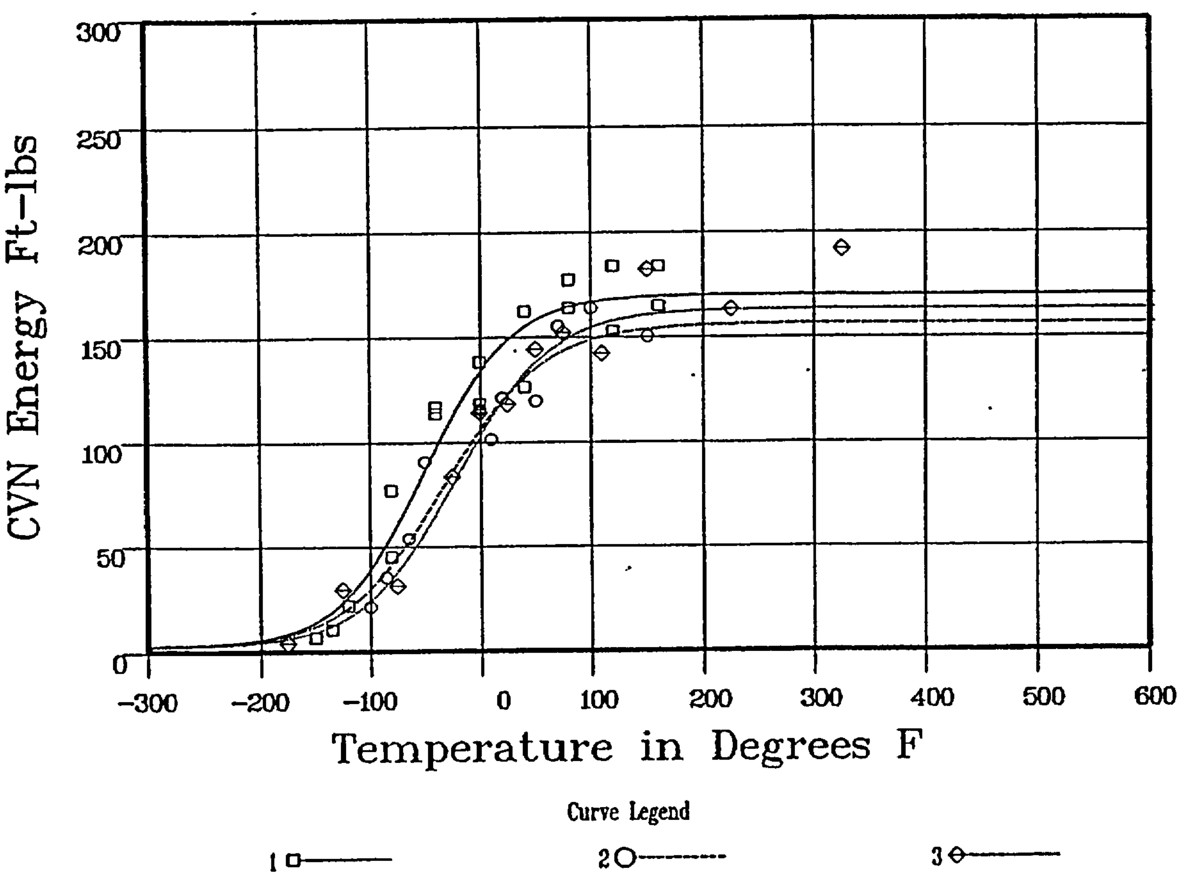
Figure 5-9 Charpy V-Notch Percent Shear vs. Temperature for Waterford Unit 3 Reactor Vessel Surveillance Weld Metal

HEAT AFFECTED ZONE

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:35:22 on 10-01-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	170	0	-117.09	0	-90.08	0
2	0	2.19	0	156	-14	-103.49	13.6	-71.83	18.25
3	0	2.19	0	163	-7	-91.96	25.13	-62.15	27.92



Data Set(s) Plotted

Curve	Plant	Capsule	Material		Ori.	Heat#
1	WF3	UNIRR	HEAT AFFD ZONE	SA533B1		M-1004-2
2	WF3	W-97	HEAT AFFD ZONE	SA533B1		M-1004-2
3	WF3	W-263	HEAT AFFD ZONE	SA533B1		M-1004-2

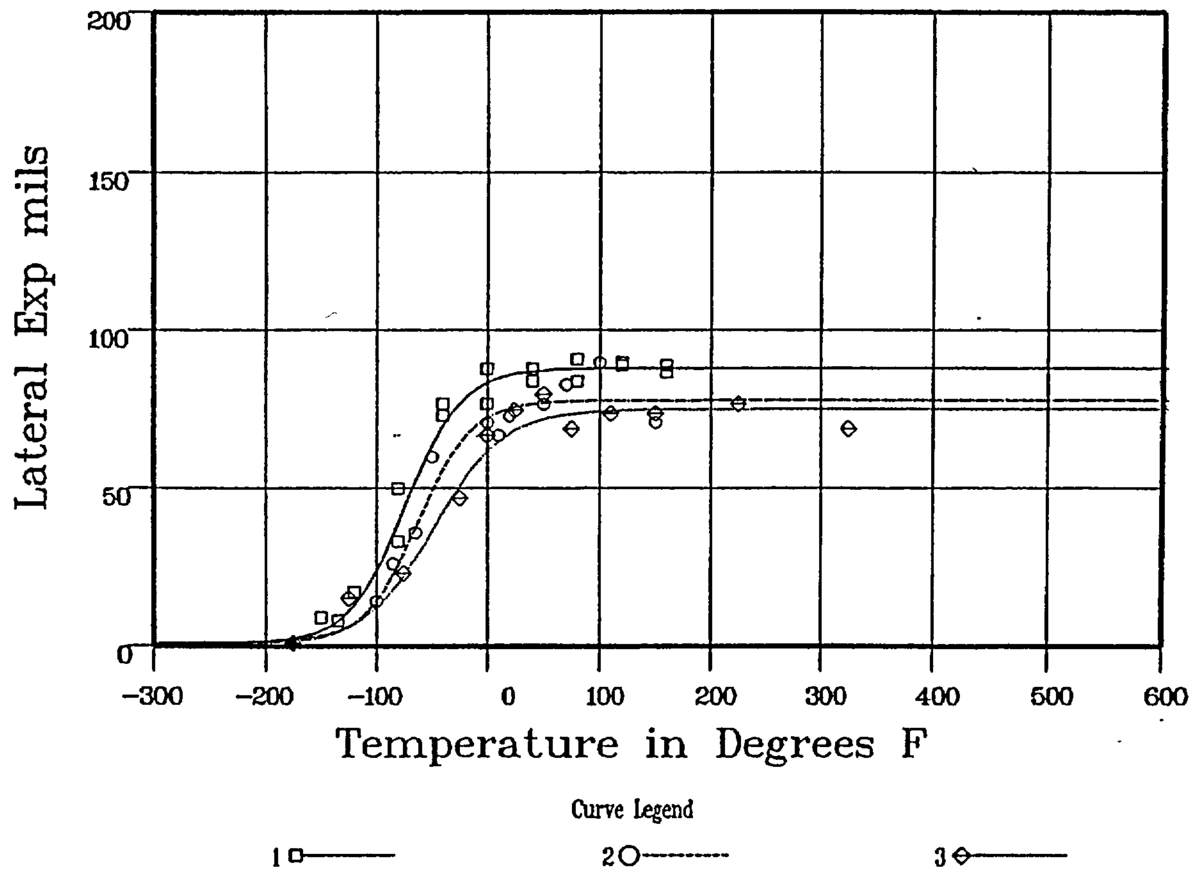
Figure 5-10 Charpy V-Notch Impact Energy vs. Temperature for Waterford Unit 3 Reactor Vessel Heat Affected Zone Material

HEAT AFFECTED ZONE

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:42:12 on 10-01-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	88.13	0	-89.55	0
2	0	78.06	-10.07	-72.24	17.3
3	0	75.19	-12.94	-56.57	32.98



Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	HEAT AFFD ZONE SA533B1		M-1004-2
2	WF3	W-97	HEAT AFFD ZONE SA533B1		M-1004-2
3	WF3	W-263	HEAT AFFD ZONE SA533B1		M-1004-2

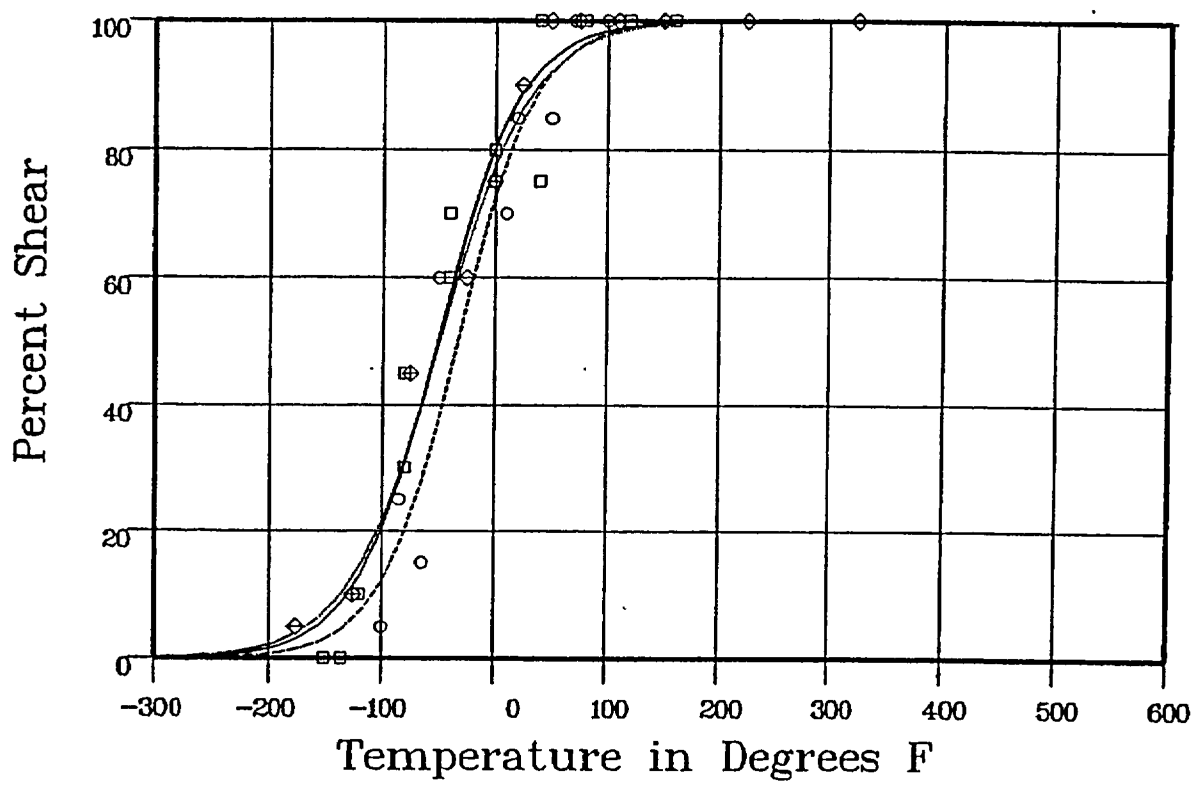
Figure 5-11 Charpy V-Notch Lateral Expansion vs. Temperature for Waterford Unit 3 Reactor Vessel Heat Affected Zone Material

HEAT AFFECTED ZONE

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:47:52 on 10-01-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	-55.31	0
2	0	-37.5	17.81
3	0	-53.43	1.87



Curve Legend

1 □———— 2 ○--------- 3 ◇————

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	WF3	UNIRR	HEAT AFFD ZONE SA533B1		M-1004-2
2	WF3	W-97	HEAT AFFD ZONE SA533B1		M-1004-2
3	WF3	W-263	HEAT AFFD ZONE SA533B1		M-1004-2

Figure 5-12 Charpy V-Notch Percent Shear vs. Temperature for Waterford Unit 3 Reactor Vessel Heat Affected Zone Material

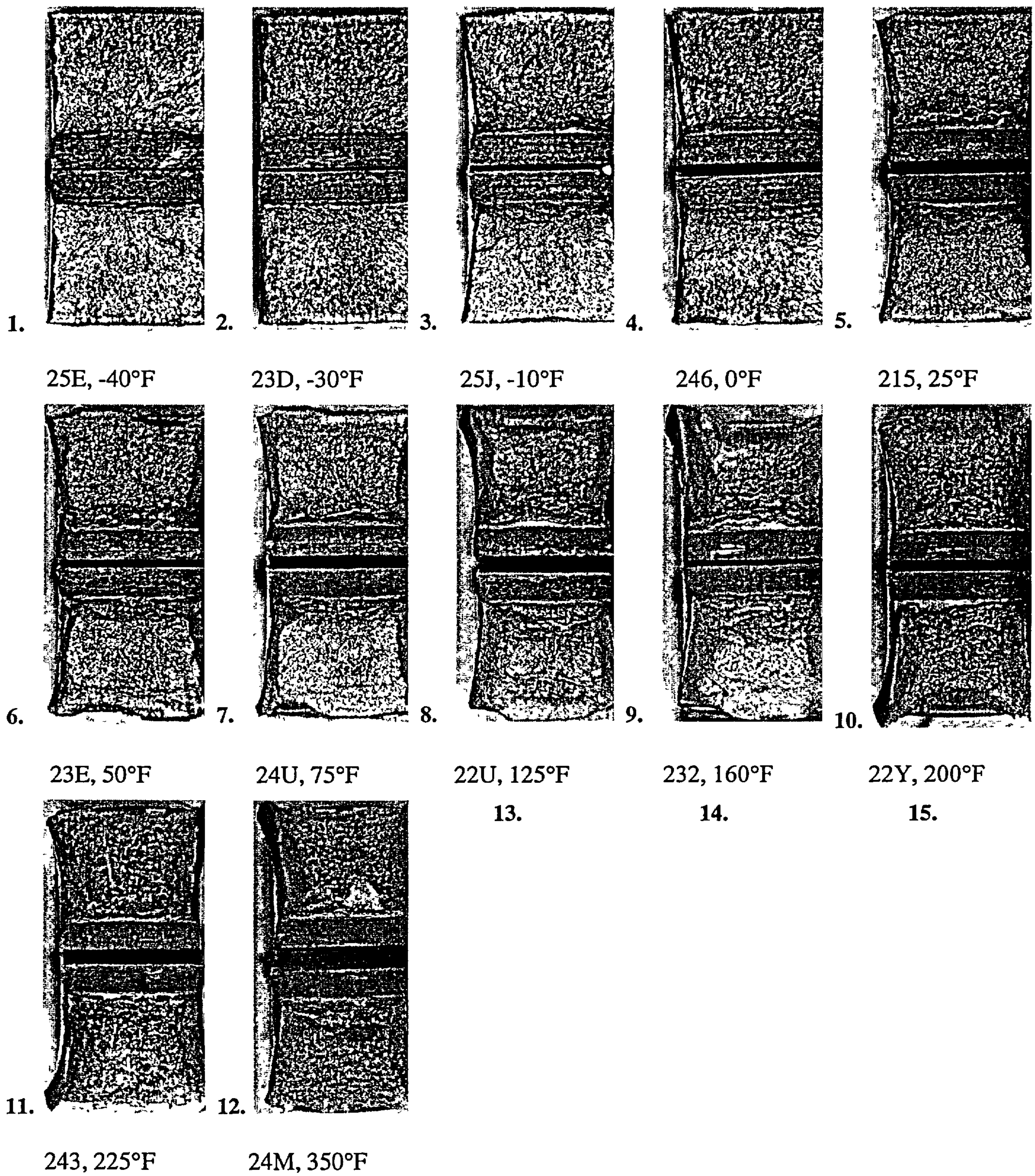


Figure 5-13 Charpy Impact Specimen Fracture Surfaces for Waterford Unit 3 Reactor Vessel Lower Shell Plate M-1004-2 (Transverse Orientation)

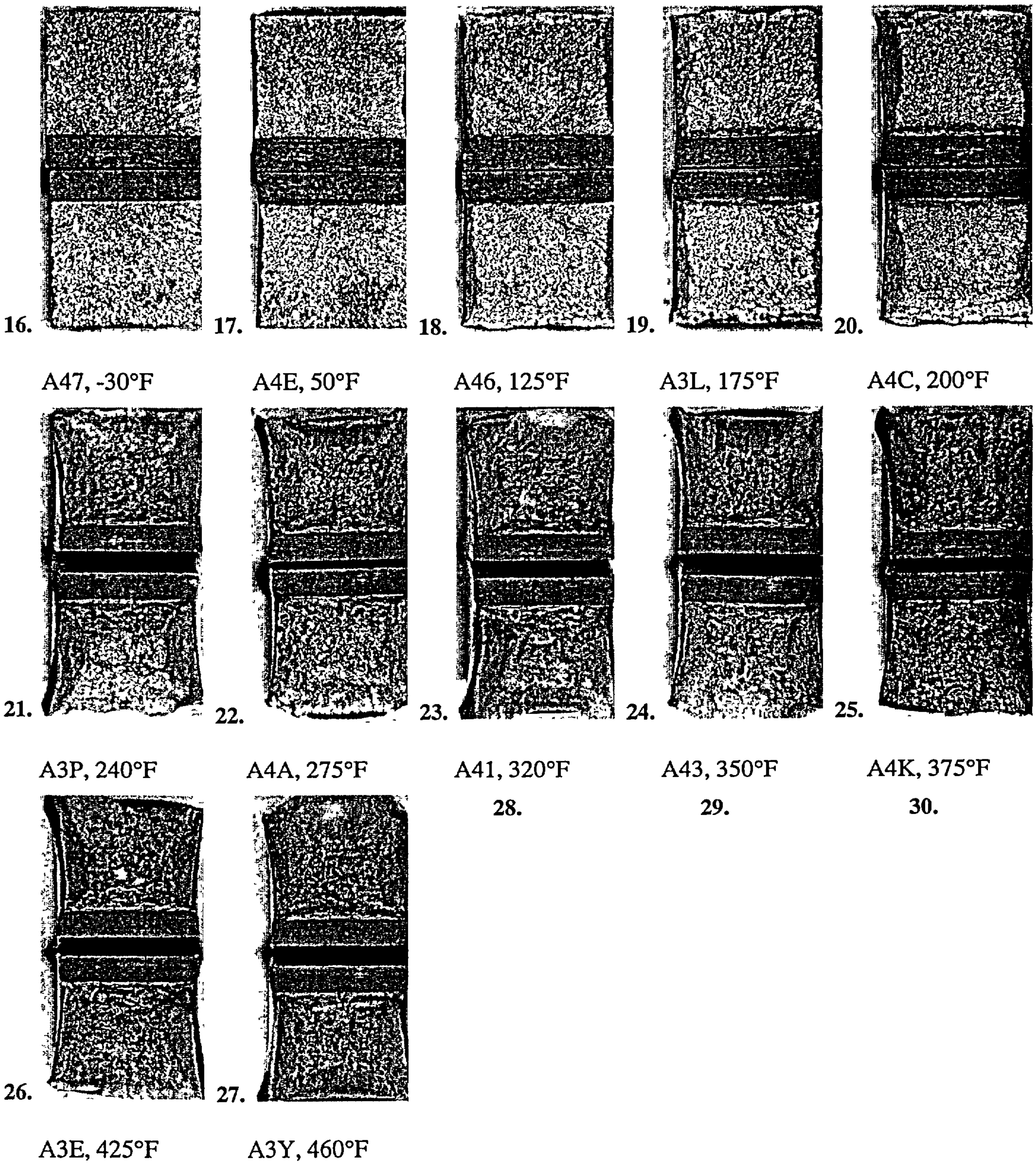


Figure 5-14 Charpy Impact Specimen Fracture Surfaces for HSST Plate O1MY Correlation Monitor Material (Longitudinal Orientation)

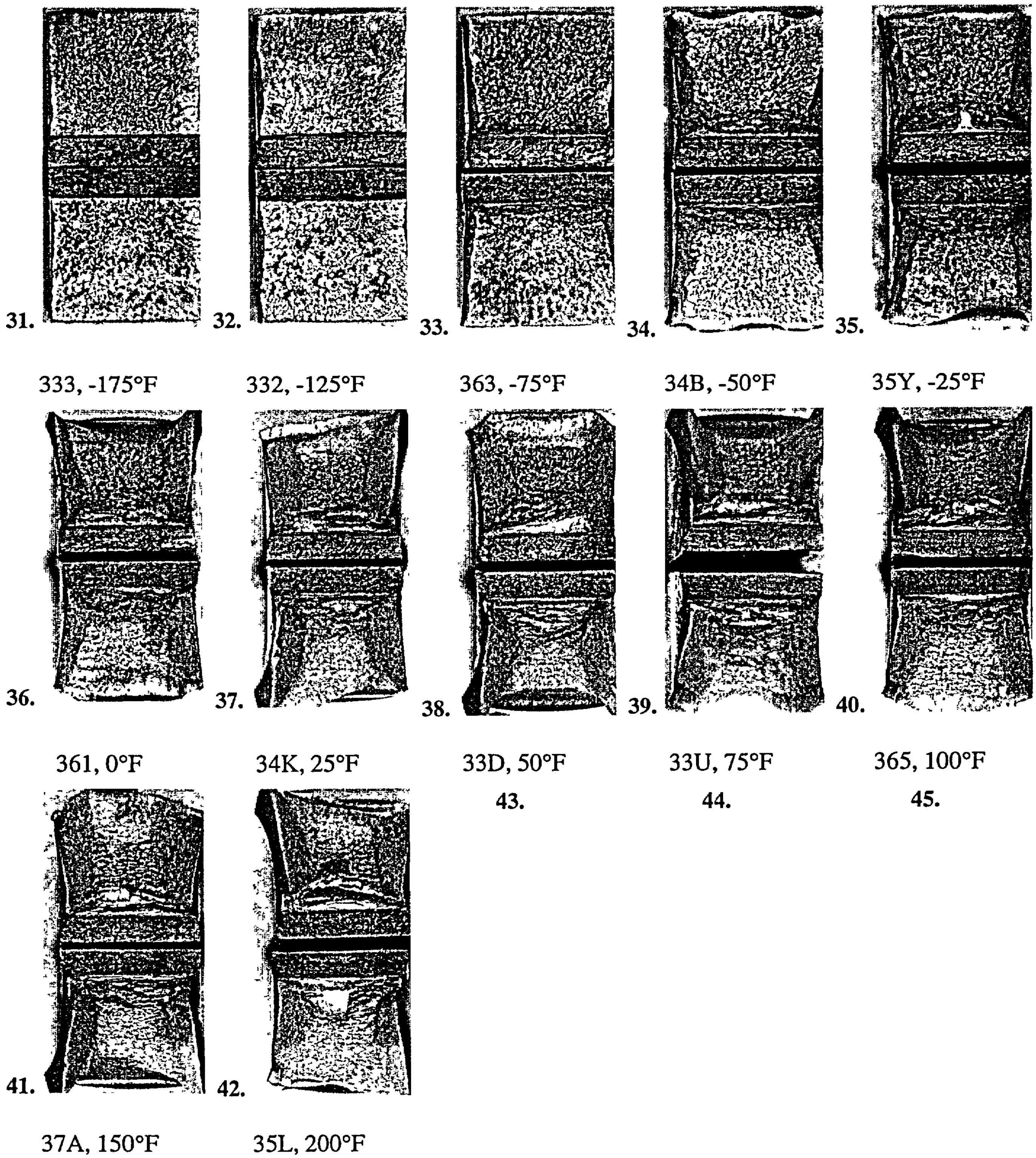


Figure 5-15 Charpy Impact Specimen Fracture Surfaces for Waterford Unit 3 Reactor Vessel Weld Metal Specimens

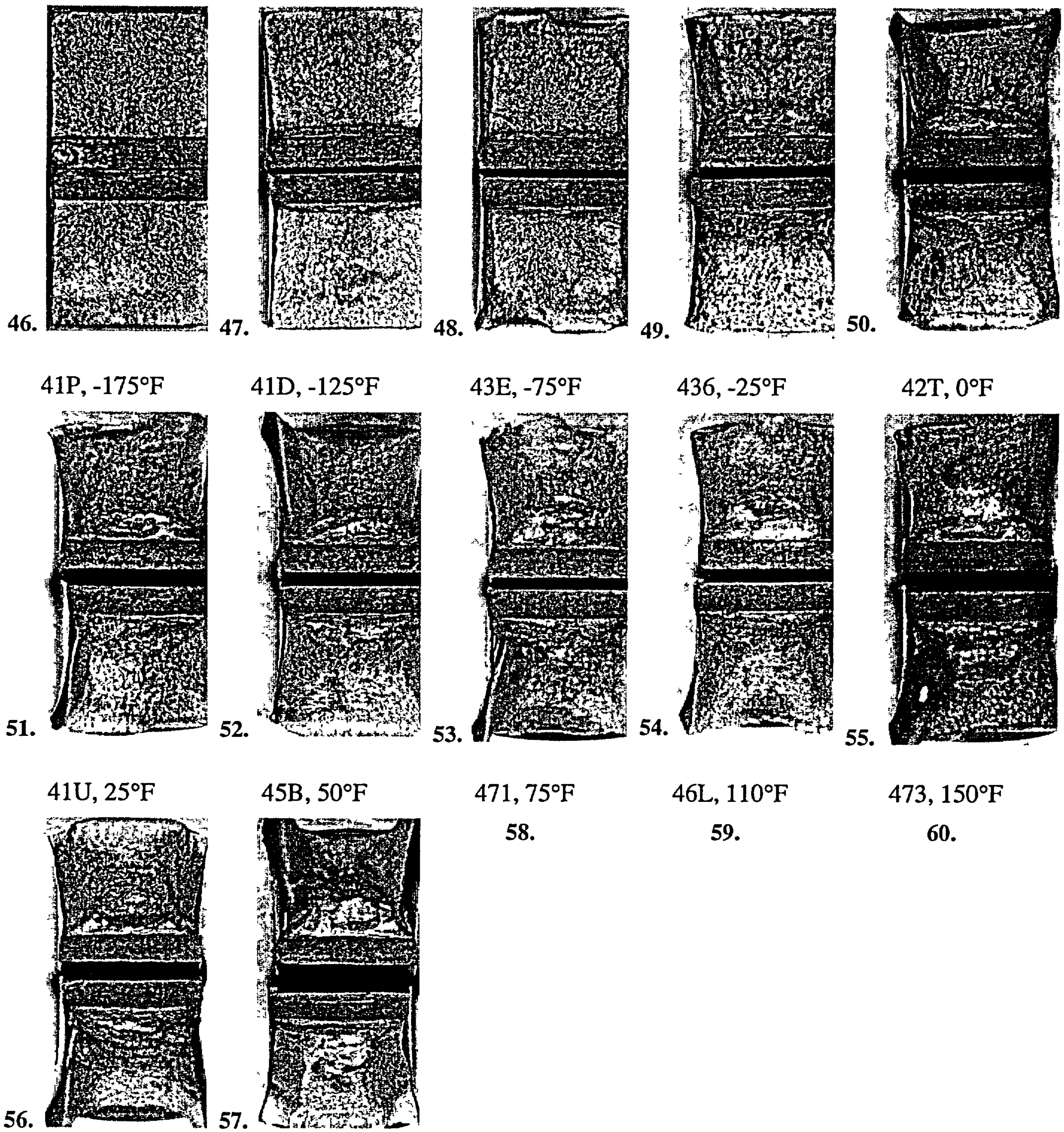
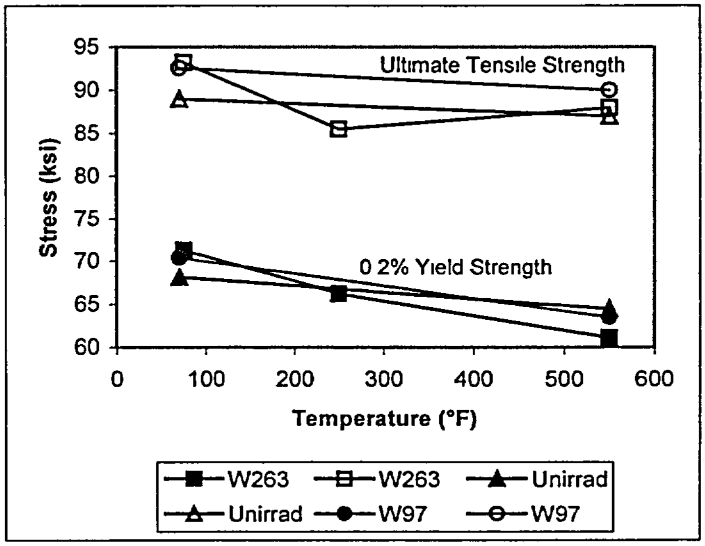


Figure 5-16 Charpy Impact Specimen Fracture Surfaces for Waterford Unit 3 Reactor Vessel Heat Affected Zone (HAZ)



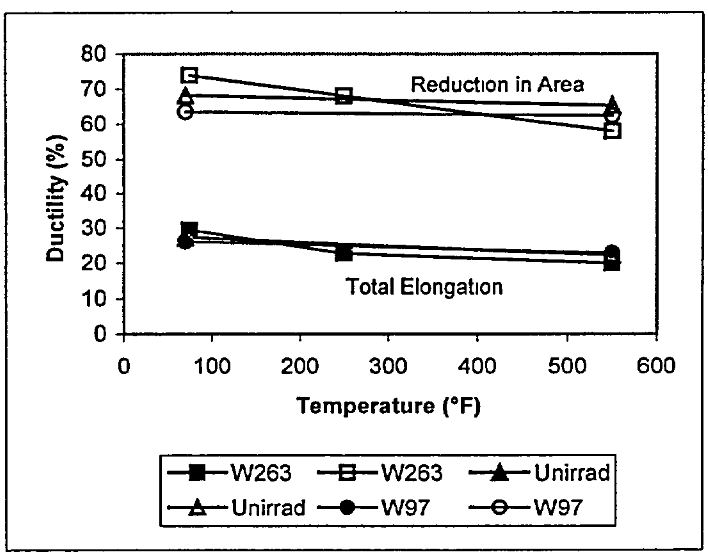
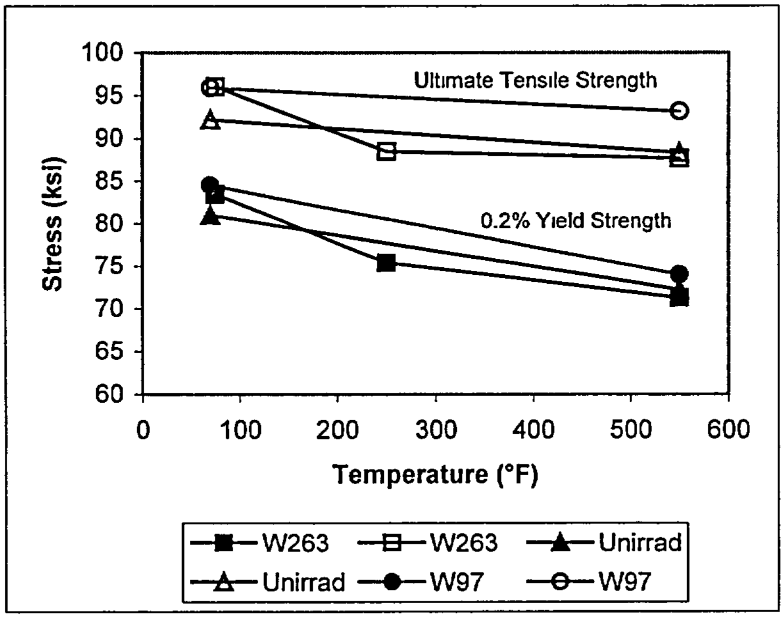


Figure 5-17 Tensile Properties for Waterford Unit 3 Reactor Vessel Lower Shell Plate M-1004-2 (Transverse Orientation)



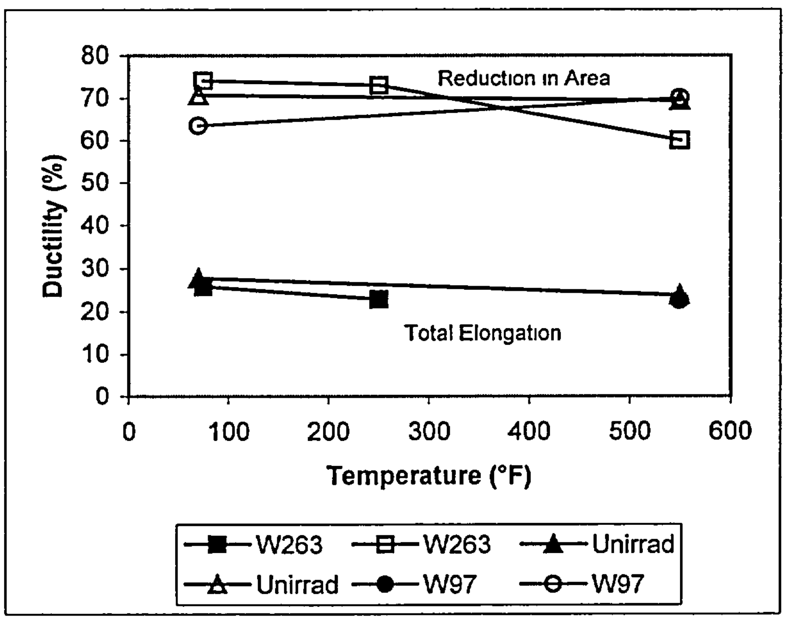


Figure 5-18 Tensile Properties for Waterford Unit 3 Reactor Vessel Weld Metal

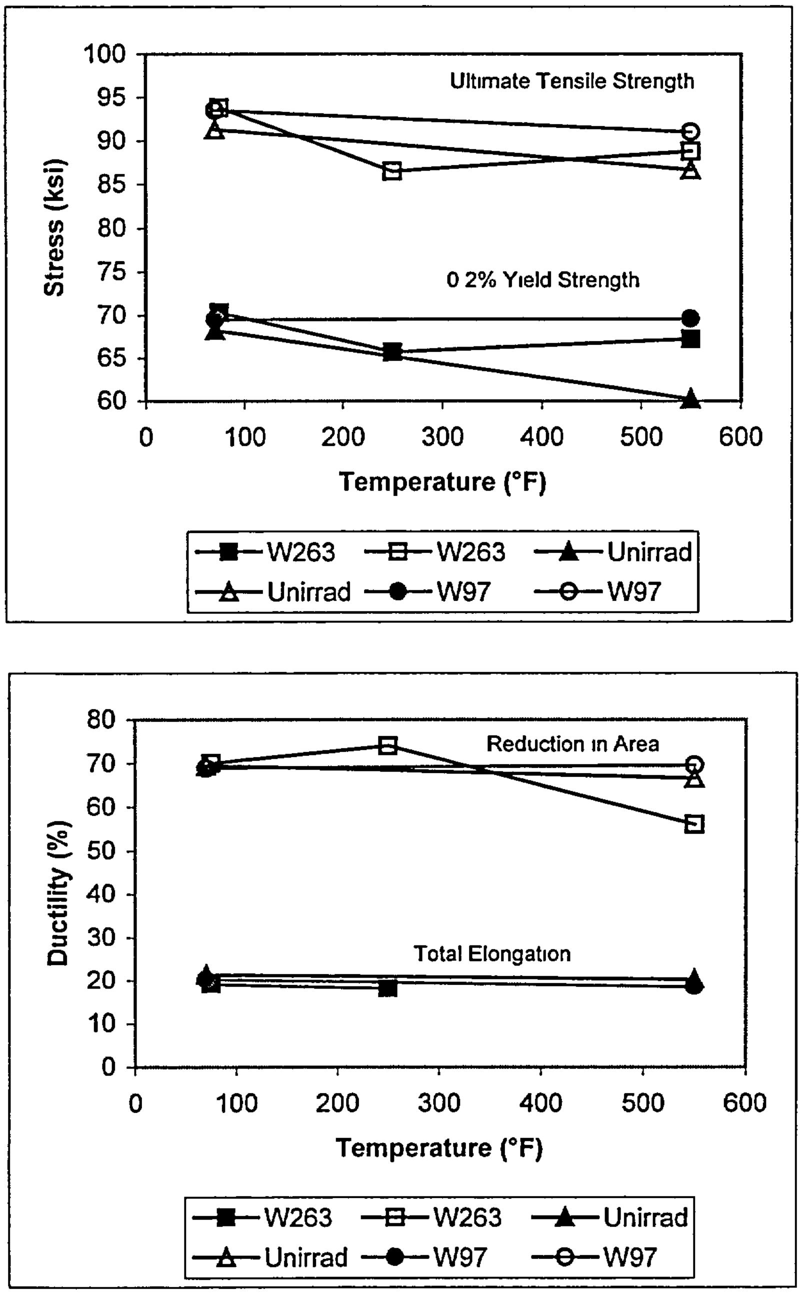


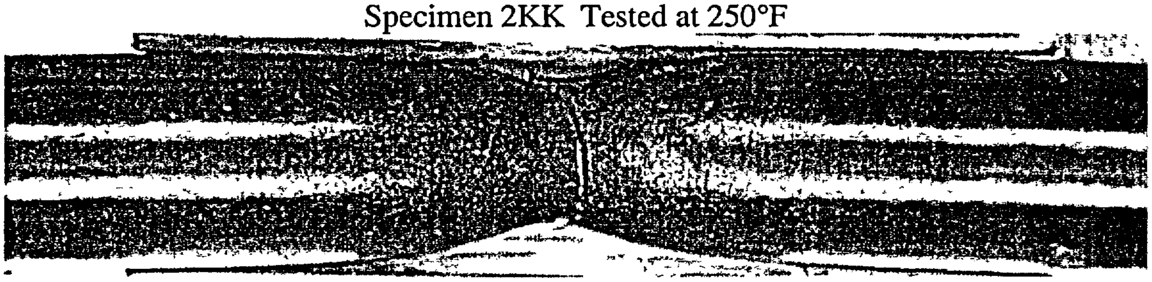
Figure 5-19 Tensile Properties for Waterford Unit 3 Reactor Vessel Heat-Affected-Zone (HAZ)

61. 

Specimen 2J2 Tested at 75°F

62. 

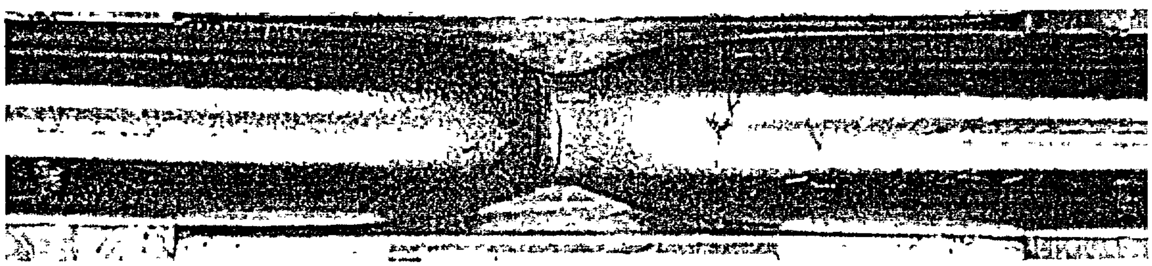
Specimen 2KK Tested at 250°F



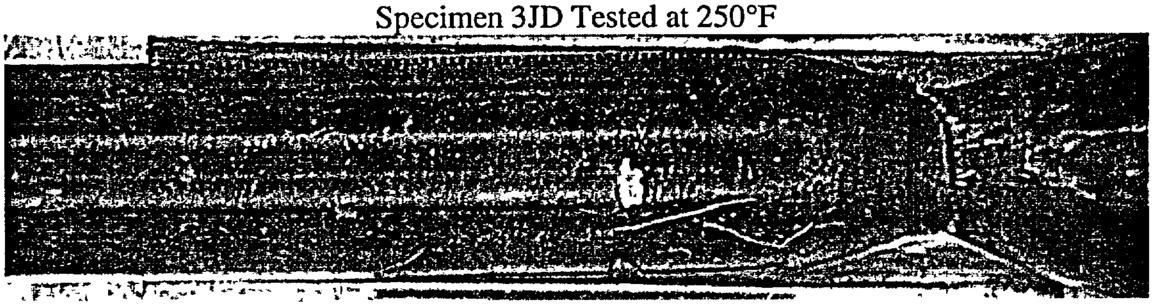
Specimen 2KL Tested at 550°F

Figure 5-20 Fractured Tensile Specimens from Waterford Unit 3 Reactor Vessel Plate M-1004-2 (Transverse Orientation)

Specimen 3K3 Tested at 75 °F

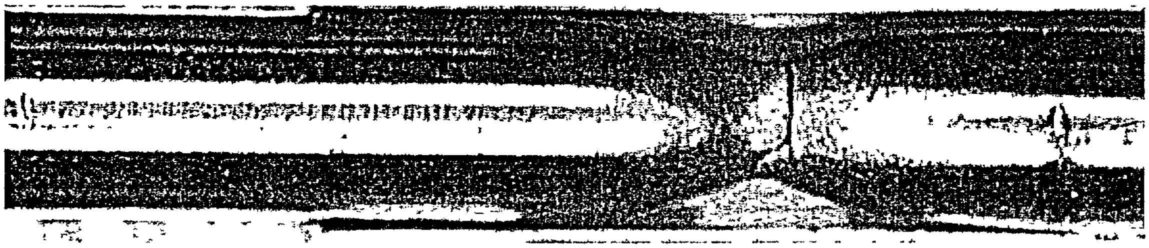


Specimen 3JD Tested at 250°F



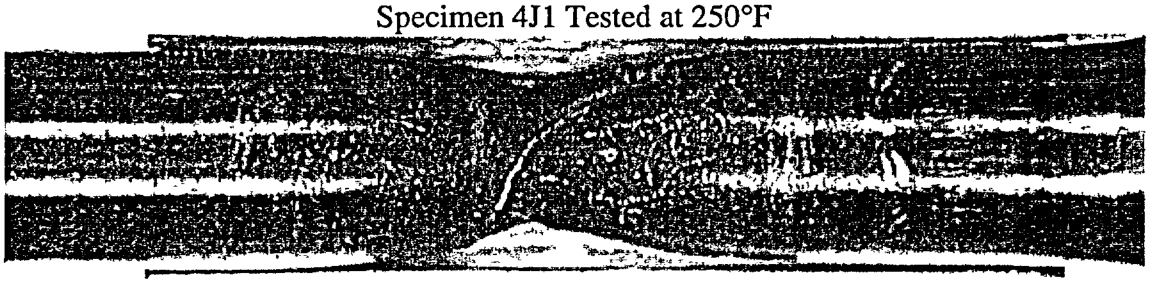
Specimen 3J7 Tested at 550°F

Figure 5-21 Fractured Tensile Specimens from Waterford Unit 3 Reactor Vessel Weld Metal



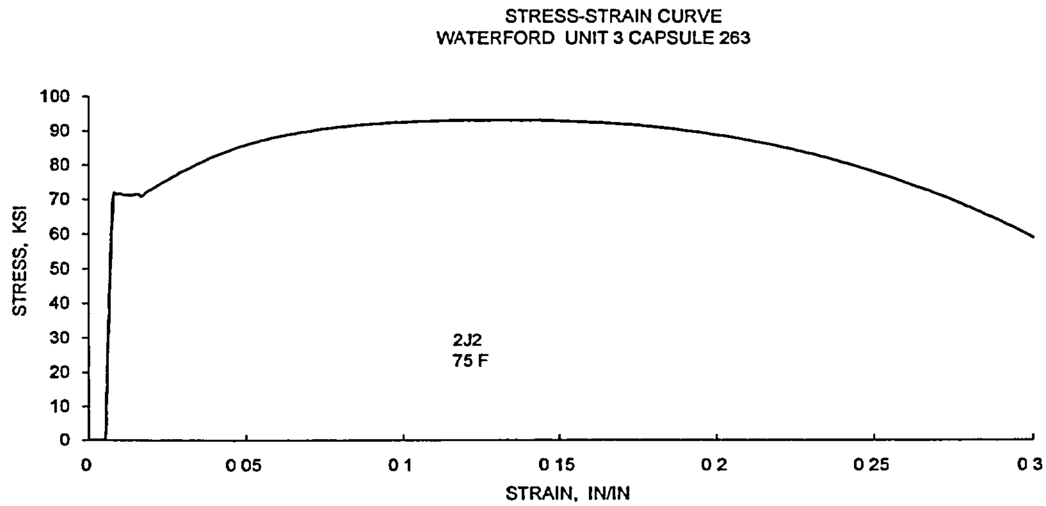
Specimen 4JB Tested at 75°F

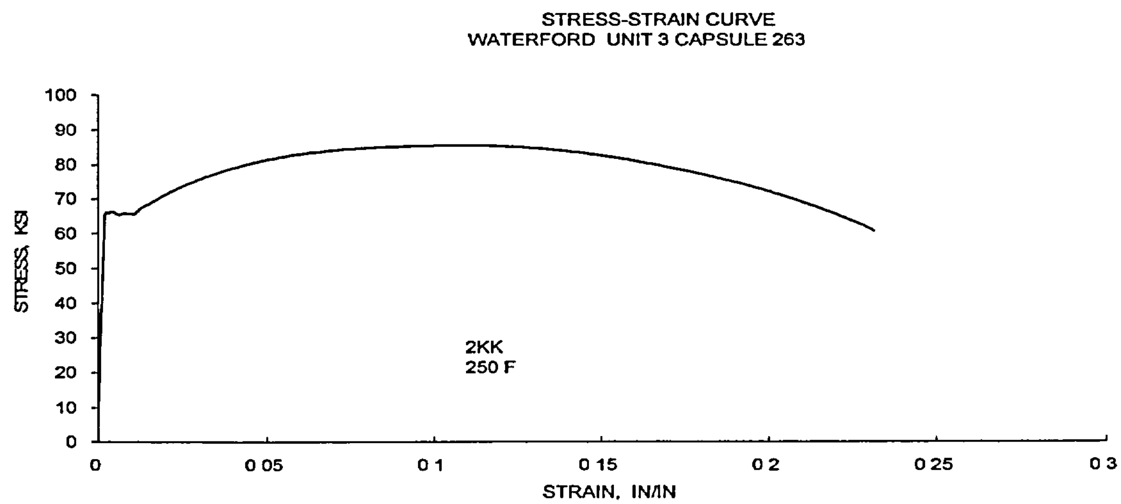
Specimen 4J1 Tested at 250°F



Specimen 4KA Tested at 550°F

Figure 5-22 Fractured Tensile Specimens from Waterford Unit 3 Reactor Vessel Heat-Affected-Zone (HAZ)





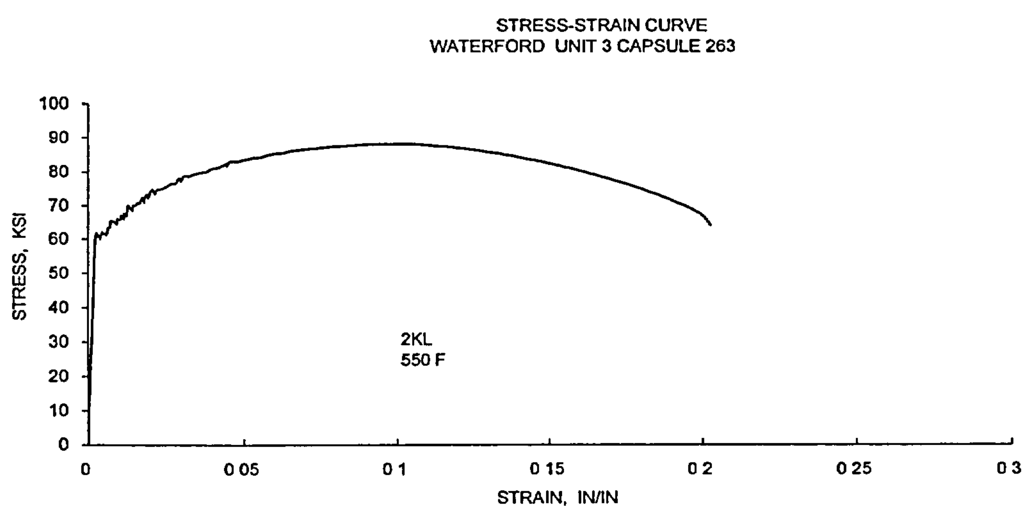
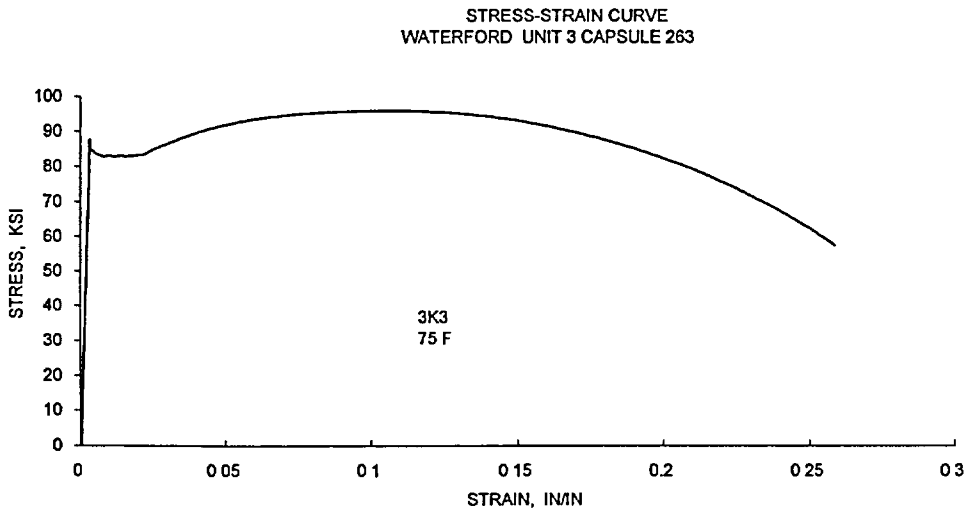
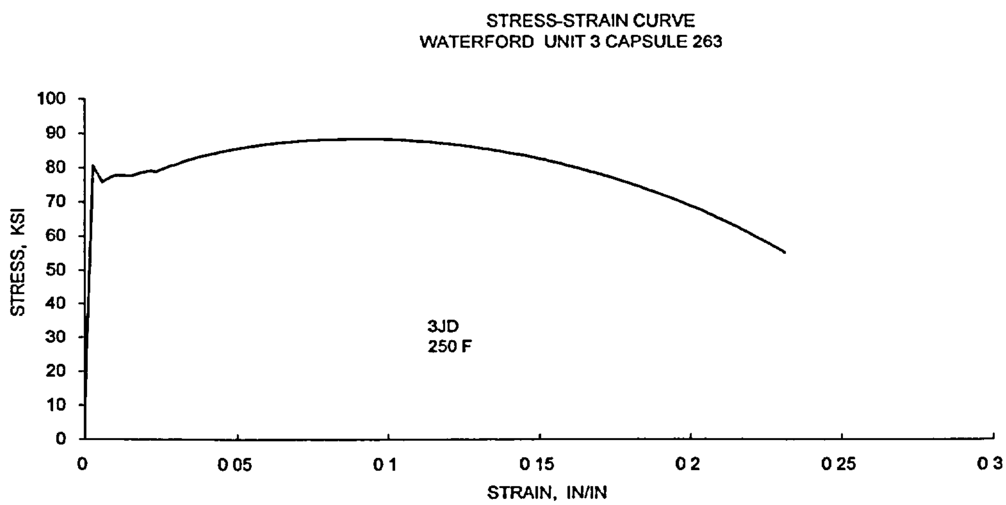


Figure 5-23 Engineering Stress-Strain Curves for Plate M-1004-2 Tensile Specimens 2J2, 2KK and 2KL (Transverse Orientation)





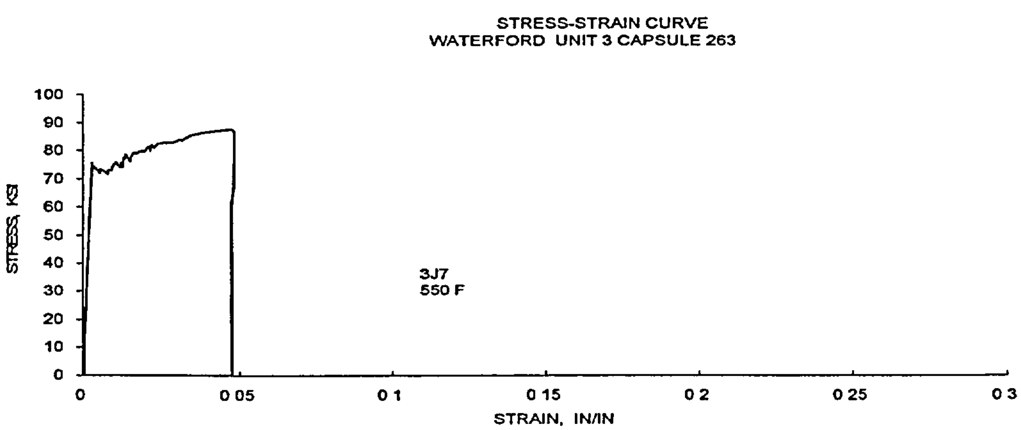
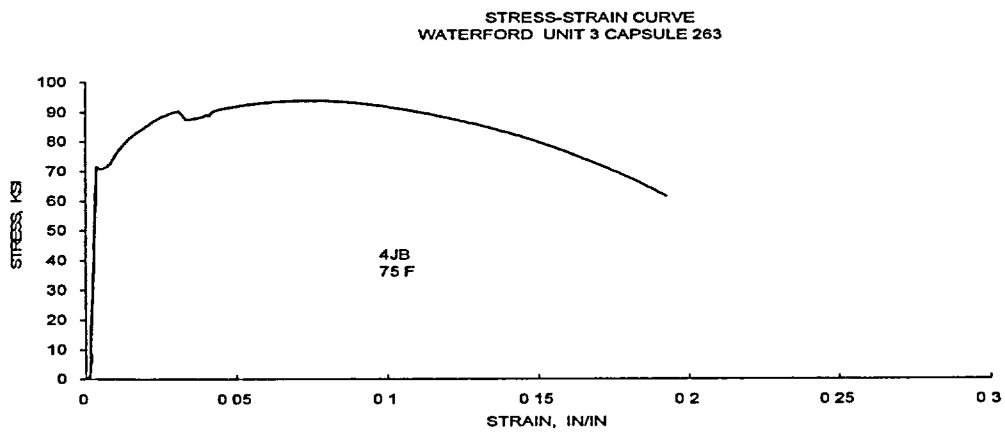
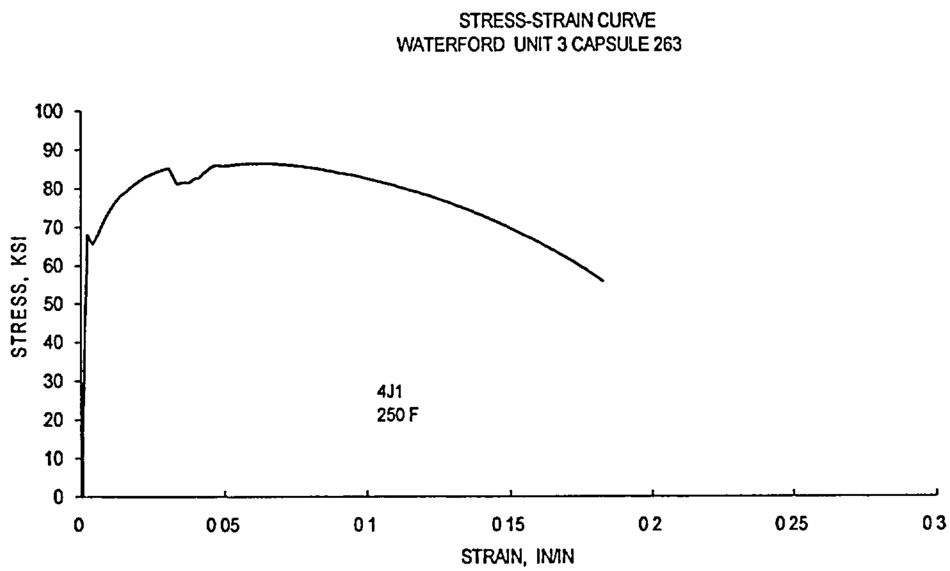


Figure 5-24 Engineering Stress-Strain Curves for Weld Metal Tensile Specimens 3K3, 3JD, and 3J7. [Note: Specimen 3J7 broke outside the gage length.]





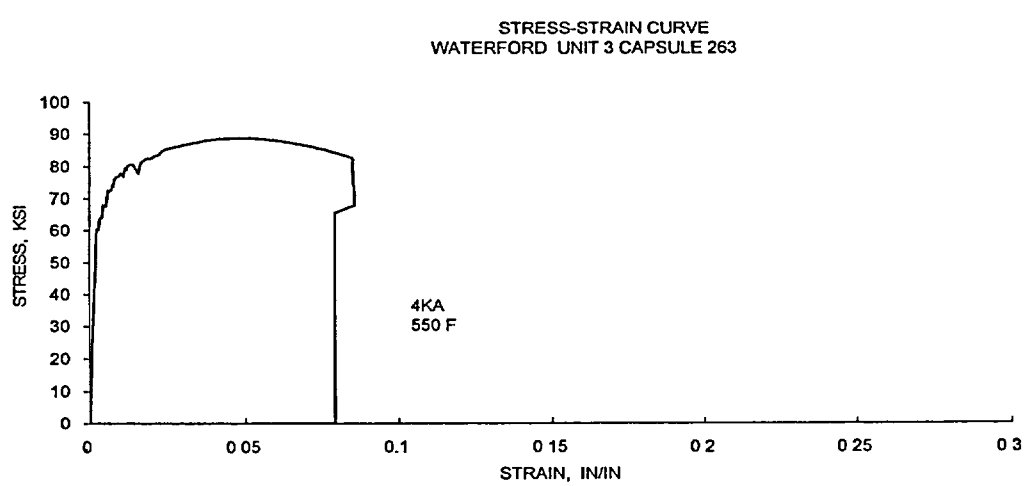


Figure 5-25 Engineering Stress-Strain Curves for Heat-Affected-Zone (HAZ) Material Tensile Specimens 4JB, 4J1 and 4KA. [Note: Specimen 4KA broke at the clip gage knife edge.]

6 RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 INTRODUCTION

Knowledge of the neutron environment within the reactor vessel and surveillance capsule geometry is required as an integral part of LWR reactor vessel surveillance programs for two reasons. First, the neutron environment (energy spectrum, flux, fluence) to which the test specimens were exposed must be known to interpret the neutron radiation induced material property changes observed in the test specimens. Second, a relationship must be established between the neutron environment at various positions within the reactor vessel and that experienced by the test specimens to relate the changes observed in the test specimens to the present and future condition of the reactor vessel. The first requirement is normally met by employing a combination of rigorous analytical techniques and measurements obtained with passive neutron flux monitors contained in each of the surveillance capsules. The second requirement is normally met by the derivation of information solely from analysis.

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 development of damage trend curves as well as for the implementation of trend curve data to assess vessel condition. It has also 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 reduce the uncertainties and increase the accuracy associated with damage trend curves when evaluating damage gradients through the reactor vessel wall. One energy dependent damage function for data correlation is displacements per iron atom (dpa).

In order to provide the dpa values in the data base for future reference, ASTM Standard Practice E853[12], "Analysis and Interpretation of Light-Water Reactor Surveillance Results," recommends reporting both displacements per iron atom (dpa) and neutron fluence (E > 1.0 MeV). 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 is reflected in the through-wall fluence adjustment factor in Regulatory Guide 1.99, Revision 2[1], "Radiation Embrittlement of Reactor Vessel Materials."

This section describes a discrete ordinates S_n transport analysis performed for the Waterford Unit 3 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 W-263, withdrawn at the end of the eleventh plant operating cycle, is provided. In addition, to provide an up-to-date data base applicable to the Waterford Unit 3 reactor, the sensor set from the previously withdrawn capsule[13] (W-97) was re-analyzed using the current dosimetry evaluation methodology. These dosimetry updates are presented in Appendix E of this report. Comparisons of the results from these dosimetry evaluations with the analytical predictions served to validate the plant specific neutron transport calculations. These validated calculations subsequently formed the basis for providing projections of the neutron exposure of the reactor pressure vessel for operating periods extending to 48 Effective Full Power Years (EFPY).

All of the calculations and dosimetry evaluations described in this section and in Appendix E 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."[14] The specific calculational methods applied are also consistent with those described in WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology."[15]

6.2 DISCRETE ORDINATES ANALYSIS

A plan view of the Waterford Unit 3 reactor geometry at the core midplane is shown in Figure 4-1. Six irradiation capsules attached to the reactor pressure vessel are included in the reactor design that constitutes the reactor vessel surveillance program. The capsules are located at azimuthal angles of 83°, 97°, 263°, 277° (7° from the core cardinal axes), and 104°, 284° (14° from the core cardinal axes) as shown in Figure 4-1. The capsule assemblies are centered on the core midplane, thus spanning the central portion of the active fuel zone.

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 location, the capsules themselves must be included in the analytical model.

The fast neutron exposure evaluations for the Waterford Unit 3 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 Waterford Unit 3.

For the Waterford Unit 3 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 7° and 14°, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological shield wall. 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. Note that a variation of this model in which the material composition of the surveillance capsules were redefined as water was utilized to determine the neutron exposure of the pressure vessel wall at the 15° degree azimuth. This accounts for the fact that the peak neutron exposure of the vessel at the 15° azimuth occurs in octants of the core that do not have a 14° surveillance capsule. In developing this analytical model, nominal design dimensions were employed for the various structural components with two exceptions. Specifically, the radius to the center of the surveillance capsule holder as well as the pressure vessel inner radius (PVIR) were taken from the as-built drawings for the Waterford Unit 3 reactor. This was done to account for key differences between the nominal versus as-built dimensions.

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 with a 107% RCS flow rate. 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 153 radial by 82 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 Waterford Unit 3 calculations (see Figure 6-2) extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation 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 (except for the PVIR as-built dimension) 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 girth ribs located between the core shroud and core barrel regions were also explicitly included in the model. The r,z geometric mesh description of the reactor model consisted of 151 radial by 94 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 151 radial mesh intervals included in the r,z model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

The core power distributions used in the plant specific transport analysis were taken from the appropriate Waterford Unit 3 fuel cycle designs. The data extracted from the design calculations represented cycle dependent fuel assembly enrichments, burnups, axial power distributions and pin-by-pin power distributions for assemblies having a face or part of a face on the periphery of the core. 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[16] and the BUGLE-96 cross-section library.[17] 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 6-10. In Table 6-1, the calculated exposure rates and integrated exposures, expressed in terms of both neutron fluence (E > 1.0 MeV) and dpa, are given at the radial and azimuthal center of the two azimuthally symmetric surveillance capsule positions (7° and 14°). These results, representative of the axial midplane of the active core, establish the calculated exposure of the surveillance capsules withdrawn to date as well as projected into the future.

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 for the middle to lower shell circumferential weld located approximately 11.4 inches below the core midplane. Tables 6-3 and 6-4 contain comparable results for the middle shell plates and the lower shell plates, respectively. Due to the symmetry in the reactor geometry, each of the middle and lower shell plates spanning 120° sectors experience neutron exposure levels characteristic of the 0°, 15°, 30°, and 45° azimuths.

In Tables 6-5 and 6-6, 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 the azimuthal locations of longitudinal welds located in the middle and lower shell courses, respectively. All of the data provided in Tables 6-2 through 6-6 were taken at the axial location of the maximum exposure experienced by each material based on the results of the three-dimensional synthesized neutron exposure evaluations.

Both calculated fluence (E > 1.0 MeV) and dpa data are provided in Tables 6-1 through 6-6. These data tabulations include both plant and fuel cycle specific calculated neutron exposures at the end of the eleventh operating fuel cycle (reactor power of 3390 MWt) as well as projections for the current operating fuel cycle, i.e., cycle twelve (reactor power of 3441 MWt), cycle thirteen (reactor power of 3441 MWt), and cycle fourteen (reactor power of 3716 MWt) and beyond to 32 and 48 effective full power years (EFPY). The projections were based on the assumption that the reactor power level and spatial power distribution from fuel cycle twelve was representative of cycle thirteen and the assumed cycle lengths were 524 EFPD and 490 EFPD, respectively. Projections for cycle fourteen and beyond were based on the assumption that future operation would continue to make use of low leakage fuel management and that a representative equilibrium spatial power distribution from the ongoing major uprate program would be typical of future operating cycles. Furthermore, to provide a degree of conservatism in the cycle fourteen and beyond projected fluence, a positive bias of 5% was applied to the neutron source in all fuel assemblies located on the core periphery.

Radial gradient information applicable to fast (E > 1.0 MeV) neutron fluence and dpa are given in Tables 6-7 and 6-8, respectively. The data, based on the cumulative integrated exposures from cycles one through twelve, 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 6-7 and 6-8. The calculated fast neutron exposures for the two surveillance capsules withdrawn from the Waterford Unit 3 reactor are provided in Table 6-9. These assigned neutron exposure levels are based on the plant and fuel cycle specific neutron transport calculations performed for the Waterford Unit 3 reactor.

Updated lead factors for the Waterford Unit 3 surveillance capsules are provided in Table 6-10. 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 6-10, the lead factors for capsules that have been withdrawn from the reactor (W-97 and W-263) were based on the calculated fluence values for the irradiation period corresponding to the time of withdrawal for the individual capsules. For the capsules remaining in the reactor (W-83, W-104, W-277, and W-284), the lead factors correspond to the calculated fluence values at the end of cycle twelve, the current operating fuel cycle for Waterford Unit 3.

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 E.

The direct comparison of measured versus calculated fast neutron threshold reaction rates for the sensors from Capsule W-263, that was withdrawn from Waterford Unit 3 at the end of the eleventh fuel cycle, is summarized below.

Reaction	Reaction Rates (rps/atom)		M/C
	Measured	Calculated	Ratio
$^{63}Cu(n,\alpha)^{60}Co$ (Cd)	4.95E-17	4.86E-17	1.02
$^{54}Fe(n,p)^{54}Mn$	4.80E-15	4.33E-15	1.11
$^{58}Ni(n,p)^{58}Co$ (Cd)	6.65E-15	5.66E-15	1.17
	Average:		1.10
	% Standard Deviation:		7.1

The measured-to-calculated (M/C) reaction rate ratios for the Capsule W-263 threshold reactions range from 1.02 to 1.17, and the average M/C ratio is 1.10 ± 7.1% (1σ). This direct comparison falls well within the ± 20% criterion specified in Regulatory Guide 1.190[14]; furthermore, it is consistent with the full set of comparisons given in Appendix E for all measured dosimetry removed to date from the Waterford Unit 3 reactor. As a result, these comparisons validate the current analytical results described in Section 6.2 and are deemed applicable for Waterford Unit 3.

6.4 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the Waterford Unit 3 surveillance capsule and reactor pressure vessel is based on the recommended approach provided in Regulatory

Guide 1.190[14]. 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 Waterford Unit 3 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 Waterford Unit 3 analysis was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with Waterford Unit 3 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 Waterford Unit 3 analytical model based on the measured plant dosimetry is completely described in Appendix E.

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 15.

	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 E support these uncertainty assessments for Waterford Unit 3.

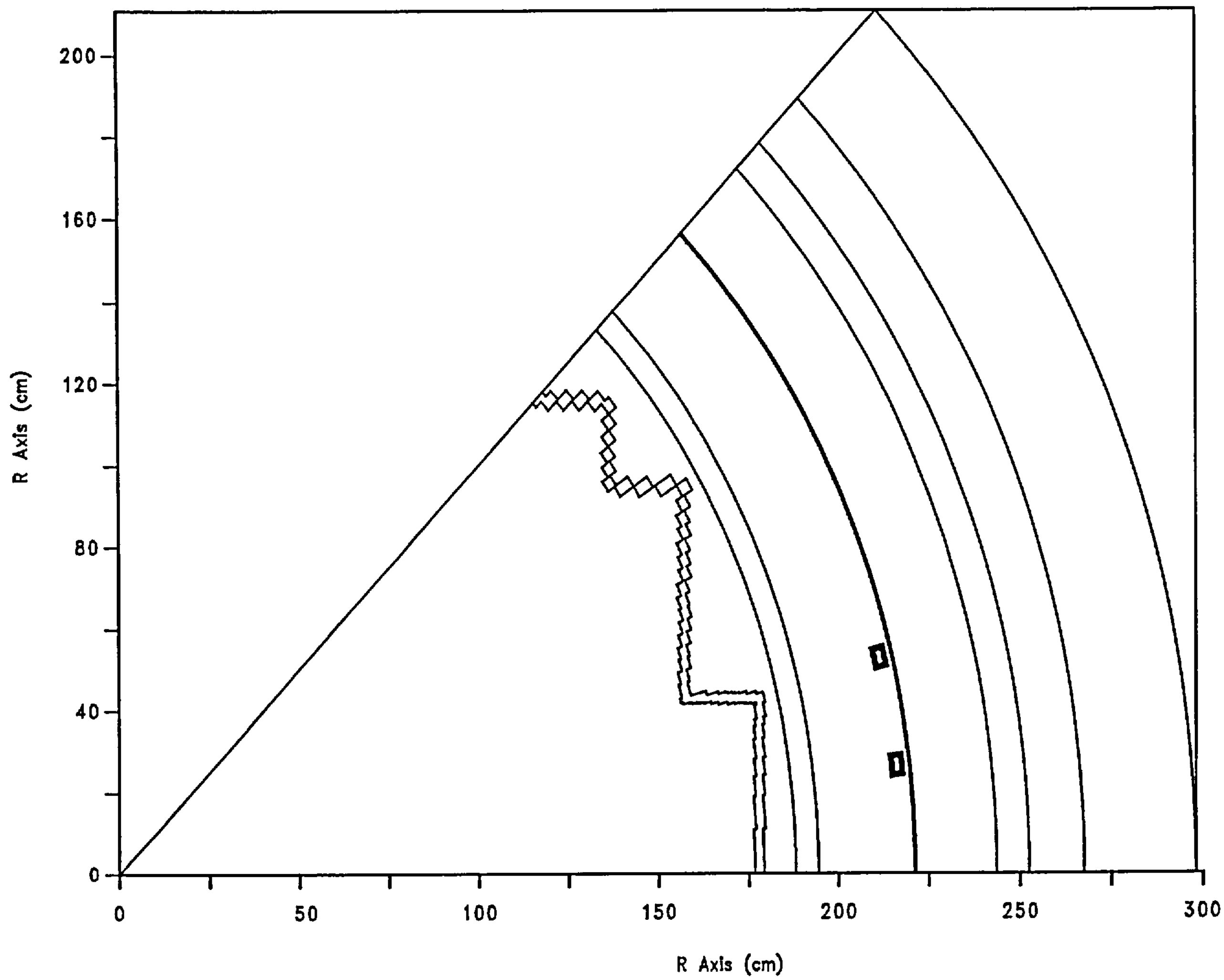


Figure 6-1 Waterford Unit 3 r,θ Reactor Geometry at the Core Midplane

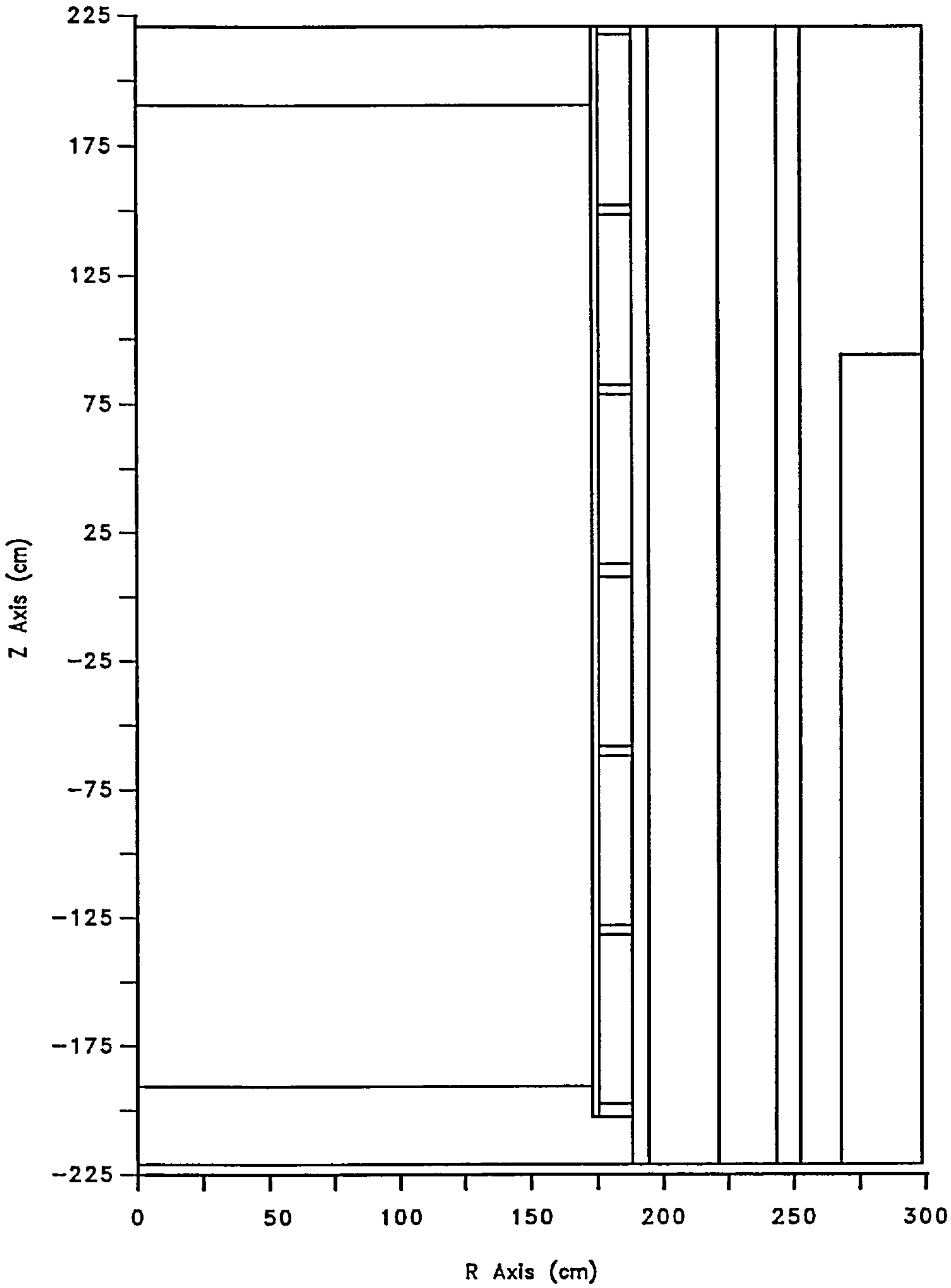


Figure 6-2 Waterford Unit 3 r,z Reactor Geometry

Table 6-1 Calculated Neutron Exposure Rates and Integrated Exposures

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Flux (E > 1.0 MeV) [n/cm²-s]	
				7°	14°
1	3.28E+07	3.28E+07	1.04	5.62E+10	3.97E+10
2	3.18E+07	6.46E+07	2.05	4.37E+10	3.07E+10
3	3.64E+07	1.01E+08	3.20	4.38E+10	2.99E+10
4	3.82E+07	1.39E+08	4.41	3.98E+10	2.79E+10
5	3.93E+07	1.79E+08	5.66	3.98E+10	2.78E+10
6	4.09E+07	2.19E+08	6.95	3.90E+10	2.33E+10
7	4.26E+07	2.62E+08	8.30	2.13E+10	1.70E+10
8	4.27E+07	3.05E+08	9.66	2.60E+10	1.83E+10
9	4.55E+07	3.50E+08	11.10	2.50E+10	1.85E+10
10	4.43E+07	3.95E+08	12.50	2.38E+10	1.80E+10
11	4.19E+07	4.36E+08	13.83	1.88E+10	1.40E+10
12 (Pjt)	4.53E+07	4.82E+08	15.27	2.38E+10	1.72E+10
13 (Pjt)	4.23E+07	5.24E+08	16.61	2.38E+10	1.72E+10
Future	2.21E+08	1.01E+09	32.00	2.73E+10	2.04E+10
Future	3.79E+08	1.51E+09	48.00	2.73E+10	2.04E+10

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm²]	
				7°	14°
1	3.28E+07	3.28E+07	1.04	1.84E+18	1.30E+18
2	3.18E+07	6.46E+07	2.05	3.23E+18	2.28E+18
3	3.64E+07	1.01E+08	3.20	4.83E+18	3.37E+18
4	3.82E+07	1.39E+08	4.41	6.35E+18	4.43E+18
5	3.93E+07	1.79E+08	5.66	7.91E+18	5.52E+18
6	4.09E+07	2.19E+08	6.95	9.51E+18	6.48E+18
7	4.26E+07	2.62E+08	8.30	1.04E+19	7.20E+18
8	4.27E+07	3.05E+08	9.66	1.15E+19	7.98E+18
9	4.55E+07	3.50E+08	11.10	1.27E+19	8.83E+18
10	4.43E+07	3.95E+08	12.50	1.37E+19	9.62E+18
11	4.19E+07	4.36E+08	13.83	1.45E+19	1.02E+19
12 (Pjt)	4.53E+07	4.82E+08	15.27	1.56E+19	1.10E+19
13 (Pjt)	4.23E+07	5.24E+08	16.61	1.66E+19	1.17E+19
Future	2.21E+08	1.01E+09	32.00	2.98E+19	2.16E+19
Future	3.79E+08	1.51E+09	48.00	4.36E+19	3.19E+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]	Displacement Rate [dpa/s]	
				7°	14°
1	3.28E+07	3.28E+07	1.04	8.20E-11	5.82E-11
2	3.18E+07	6.46E+07	2.05	6.38E-11	4.51E-11
3	3.64E+07	1.01E+08	3.20	6.40E-11	4.39E-11
4	3.82E+07	1.39E+08	4.41	5.81E-11	4.10E-11
5	3.93E+07	1.79E+08	5.66	5.82E-11	4.09E-11
6	4.09E+07	2.19E+08	6.95	5.70E-11	3.43E-11
7	4.26E+07	2.62E+08	8.30	3.12E-11	2.50E-11
8	4.27E+07	3.05E+08	9.66	3.81E-11	2.70E-11
9	4.55E+07	3.50E+08	11.10	3.66E-11	2.72E-11
10	4.43E+07	3.95E+08	12.50	3.48E-11	2.65E-11
11	4.19E+07	4.36E+08	13.83	2.77E-11	2.05E-11
12 (Pjt)	4.53E+07	4.82E+08	15.27	3.48E-11	2.54E-11
13 (Pjt)	4.23E+07	5.24E+08	16.61	3.48E-11	2.54E-11
Future	2.21E+08	1.01E+09	32.00	3.99E-11	3.01E-11
Future	3.79E+08	1.51E+09	48.00	3.99E-11	3.01E-11

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Displacements [dpa]	
				7°	14°
1	3.28E+07	3.28E+07	1.04	2.69E-03	1.91E-03
2	3.18E+07	6.46E+07	2.05	4.72E-03	3.34E-03
3	3.64E+07	1.01E+08	3.20	7.05E-03	4.94E-03
4	3.82E+07	1.39E+08	4.41	9.27E-03	6.51E-03
5	3.93E+07	1.79E+08	5.66	1.16E-02	8.12E-03
6	4.09E+07	2.19E+08	6.95	1.39E-02	9.52E-03
7	4.26E+07	2.62E+08	8.30	1.52E-02	1.06E-02
8	4.27E+07	3.05E+08	9.66	1.69E-02	1.17E-02
9	4.55E+07	3.50E+08	11.10	1.85E-02	1.30E-02
10	4.43E+07	3.95E+08	12.50	2.01E-02	1.42E-02
11	4.19E+07	4.36E+08	13.83	2.12E-02	1.50E-02
12 (Pjt)	4.53E+07	4.82E+08	15.27	2.28E-02	1.62E-02
13 (Pjt)	4.23E+07	5.24E+08	16.61	2.43E-02	1.72E-02
Future	2.21E+08	1.01E+09	32.00	4.37E-02	3.18E-02
Future	3.79E+08	1.51E+09	48.00	6.38E-02	4.70E-02

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

Table 6-2 Calculated Neutron Exposure of the Middle Shell to Lower Shell Circumferential Weld (101-171)

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 0°	15°	30°	45°
1	3.28E+07	3.28E+07	1.04	1.48E+18	9.23E+17	8.05E+17	6.35E+17
2	3.18E+07	6.46E+07	2.05	2.67E+18	1.63E+18	1.47E+18	1.12E+18
3	3.64E+07	1.01E+08	3.20	4.05E+18	2.42E+18	2.12E+18	1.61E+18
4	3.82E+07	1.39E+08	4.41	5.36E+18	3.21E+18	2.82E+18	2.15E+18
5	3.93E+07	1.79E+08	5.66	6.72E+18	4.01E+18	3.52E+18	2.69E+18
6	4.09E+07	2.19E+08	6.95	8.14E+18	4.68E+18	4.00E+18	3.14E+18
7	4.26E+07	2.62E+08	8.30	8.87E+18	5.21E+18	4.56E+18	3.58E+18
8	4.27E+07	3.05E+08	9.66	9.80E+18	5.78E+18	5.01E+18	3.99E+18
9	4.55E+07	3.50E+08	11.10	1.07E+19	6.39E+18	5.49E+18	4.36E+18
10	4.43E+07	3.95E+08	12.50	1.16E+19	6.97E+18	6.05E+18	4.84E+18
11	4.19E+07	4.36E+08	13.83	1.22E+19	7.40E+18	6.45E+18	5.23E+18
12 (Pjt)	4.53E+07	4.82E+08	15.27	1.31E+19	7.96E+18	6.93E+18	5.65E+18
13 (Pjt)	4.23E+07	5.24E+08	16.61	1.40E+19	8.49E+18	7.39E+18	6.03E+18
Future	2.21E+08	1.01E+09	32.00	2.47E+19	1.57E+19	1.41E+19	1.17E+19
Future	3.79E+08	1.51E+09	48.00	3.59E+19	2.32E+19	2.12E+19	1.75E+19

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa] 0°	15°	30°	45°
1	3.28E+07	3.28E+07	1.04	2.25E-03	1.42E-03	1.23E-03	9.77E-04
2	3.18E+07	6.46E+07	2.05	4.06E-03	2.51E-03	2.24E-03	1.72E-03
3	3.64E+07	1.01E+08	3.20	6.16E-03	3.73E-03	3.24E-03	2.48E-03
4	3.82E+07	1.39E+08	4.41	8.17E-03	4.94E-03	4.32E-03	3.32E-03
5	3.93E+07	1.79E+08	5.66	1.02E-02	6.17E-03	5.38E-03	4.15E-03
6	4.09E+07	2.19E+08	6.95	1.24E-02	7.20E-03	6.12E-03	4.83E-03
7	4.26E+07	2.62E+08	8.30	1.35E-02	8.03E-03	6.97E-03	5.52E-03
8	4.27E+07	3.05E+08	9.66	1.49E-02	8.90E-03	7.68E-03	6.15E-03
9	4.55E+07	3.50E+08	11.10	1.63E-02	9.84E-03	8.41E-03	6.72E-03
10	4.43E+07	3.95E+08	12.50	1.76E-02	1.07E-02	9.26E-03	7.46E-03
11	4.19E+07	4.36E+08	13.83	1.86E-02	1.14E-02	9.88E-03	8.06E-03
12 (Pjt)	4.53E+07	4.82E+08	15.27	2.00E-02	1.23E-02	1.06E-02	8.70E-03
13 (Pjt)	4.23E+07	5.24E+08	16.61	2.13E-02	1.31E-02	1.13E-02	9.30E-03
Future	2.21E+08	1.01E+09	32.00	3.77E-02	2.42E-02	2.17E-02	1.80E-02
Future	3.79E+08	1.51E+09	48.00	5.48E-02	3.58E-02	3.24E-02	2.70E-02

Table 6-3 Calculated Neutron Exposure of the Middle Shell Plates (M-1003-1, M-1003-2, and M-1003-3)

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
				0°	15°	30°	45°
1	3.28E+07	3.28E+07	1.04	1.48E+18	9.25E+17	8.07E+17	6.37E+17
2	3.18E+07	6.46E+07	2.05	2.68E+18	1.64E+18	1.47E+18	1.12E+18
3	3.64E+07	1.01E+08	3.20	4.06E+18	2.43E+18	2.12E+18	1.61E+18
4	3.82E+07	1.39E+08	4.41	5.38E+18	3.22E+18	2.83E+18	2.16E+18
5	3.93E+07	1.79E+08	5.66	6.75E+18	4.02E+18	3.53E+18	2.70E+18
6	4.09E+07	2.19E+08	6.95	8.19E+18	4.70E+18	4.02E+18	3.15E+18
7	4.26E+07	2.62E+08	8.30	8.92E+18	5.24E+18	4.58E+18	3.60E+18
8	4.27E+07	3.05E+08	9.66	9.85E+18	5.81E+18	5.04E+18	4.01E+18
9	4.55E+07	3.50E+08	11.10	1.08E+19	6.42E+18	5.52E+18	4.38E+18
10	4.43E+07	3.95E+08	12.50	1.16E+19	7.00E+18	6.08E+18	4.86E+18
11	4.19E+07	4.36E+08	13.83	1.23E+19	7.43E+18	6.48E+18	5.25E+18
12 (Pjt)	4.53E+07	4.82E+08	15.27	1.32E+19	7.99E+18	6.96E+18	5.66E+18
13 (Pjt)	4.23E+07	5.24E+08	16.61	1.40E+19	8.52E+18	7.40E+18	6.05E+18
Future	2.21E+08	1.01E+09	32.00	2.48E+19	1.58E+19	1.42E+19	1.17E+19
Future	3.79E+08	1.51E+09	48.00	3.60E+19	2.33E+19	2.12E+19	1.75E+19

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]			
				0°	15°	30°	45°
1	3.28E+07	3.28E+07	1.04	2.26E-03	1.42E-03	1.23E-03	9.79E-04
2	3.18E+07	6.46E+07	2.05	4.07E-03	2.52E-03	2.25E-03	1.73E-03
3	3.64E+07	1.01E+08	3.20	6.18E-03	3.74E-03	3.25E-03	2.49E-03
4	3.82E+07	1.39E+08	4.41	8.20E-03	4.95E-03	4.33E-03	3.33E-03
5	3.93E+07	1.79E+08	5.66	1.03E-02	6.19E-03	5.40E-03	4.17E-03
6	4.09E+07	2.19E+08	6.95	1.25E-02	7.24E-03	6.15E-03	4.86E-03
7	4.26E+07	2.62E+08	8.30	1.36E-02	8.07E-03	7.01E-03	5.54E-03
8	4.27E+07	3.05E+08	9.66	1.50E-02	8.94E-03	7.71E-03	6.18E-03
9	4.55E+07	3.50E+08	11.10	1.64E-02	9.88E-03	8.44E-03	6.75E-03
10	4.43E+07	3.95E+08	12.50	1.77E-02	1.08E-02	9.30E-03	7.49E-03
11	4.19E+07	4.36E+08	13.83	1.87E-02	1.14E-02	9.92E-03	8.09E-03
12 (Pjt)	4.53E+07	4.82E+08	15.27	2.01E-02	1.23E-02	1.07E-02	8.73E-03
13 (Pjt)	4.23E+07	5.24E+08	16.61	2.13E-02	1.31E-02	1.13E-02	9.32E-03
Future	2.21E+08	1.01E+09	32.00	3.78E-02	2.43E-02	2.17E-02	1.80E-02
Future	3.79E+08	1.51E+09	48.00	5.50E-02	3.59E-02	3.25E-02	2.70E-02

Note: The maximum exposure after cycle one occurs an axial elevation of 8.3 inches below the midplane of the active fuel. The maximum exposure for all other times occurs at an axial elevation 16 8 inches above the midplane of the active fuel.

Table 6-4 Calculated Neutron Exposure of the Lower Shell Plates (M-1004-1, M-1004-2, and M-1004-3)

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
				0°	15°	30°	45°
1	3.28E+07	3.28E+07	1.04	1.48E+18	9.23E+17	8.05E+17	6.35E+17
2	3.18E+07	6.46E+07	2.05	2.67E+18	1.63E+18	1.47E+18	1.12E+18
3	3.64E+07	1.01E+08	3.20	4.05E+18	2.42E+18	2.12E+18	1.61E+18
4	3.82E+07	1.39E+08	4.41	5.36E+18	3.21E+18	2.82E+18	2.15E+18
5	3.93E+07	1.79E+08	5.66	6.72E+18	4.01E+18	3.52E+18	2.69E+18
6	4.09E+07	2.19E+08	6.95	8.14E+18	4.68E+18	4.00E+18	3.14E+18
7	4.26E+07	2.62E+08	8.30	8.87E+18	5.21E+18	4.56E+18	3.58E+18
8	4.27E+07	3.05E+08	9.66	9.80E+18	5.78E+18	5.01E+18	3.99E+18
9	4.55E+07	3.50E+08	11.10	1.07E+19	6.39E+18	5.49E+18	4.36E+18
10	4.43E+07	3.95E+08	12.50	1.16E+19	6.97E+18	6.05E+18	4.84E+18
11	4.19E+07	4.36E+08	13.83	1.22E+19	7.40E+18	6.45E+18	5.23E+18
12 (Pjt)	4.53E+07	4.82E+08	15.27	1.31E+19	7.96E+18	6.93E+18	5.65E+18
13 (Pjt)	4.23E+07	5.24E+08	16.61	1.40E+19	8.49E+18	7.39E+18	6.03E+18
Future	2.21E+08	1.01E+09	32.00	2.47E+19	1.57E+19	1.41E+19	1.17E+19
Future	3.79E+08	1.51E+09	48.00	3.59E+19	2.32E+19	2.12E+19	1.75E+19

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]			
				0°	15°	30°	45°
1	3.28E+07	3.28E+07	1.04	2.25E-03	1.42E-03	1.23E-03	9.77E-04
2	3.18E+07	6.46E+07	2.05	4.06E-03	2.51E-03	2.24E-03	1.72E-03
3	3.64E+07	1.01E+08	3.20	6.16E-03	3.73E-03	3.24E-03	2.48E-03
4	3.82E+07	1.39E+08	4.41	8.17E-03	4.94E-03	4.32E-03	3.32E-03
5	3.93E+07	1.79E+08	5.66	1.02E-02	6.17E-03	5.38E-03	4.15E-03
6	4.09E+07	2.19E+08	6.95	1.24E-02	7.20E-03	6.12E-03	4.83E-03
7	4.26E+07	2.62E+08	8.30	1.35E-02	8.03E-03	6.97E-03	5.52E-03
8	4.27E+07	3.05E+08	9.66	1.49E-02	8.90E-03	7.68E-03	6.15E-03
9	4.55E+07	3.50E+08	11.10	1.63E-02	9.84E-03	8.41E-03	6.72E-03
10	4.43E+07	3.95E+08	12.50	1.76E-02	1.07E-02	9.26E-03	7.46E-03
11	4.19E+07	4.36E+08	13.83	1.86E-02	1.14E-02	9.88E-03	8.06E-03
12 (Pjt)	4.53E+07	4.82E+08	15.27	2.00E-02	1.23E-02	1.06E-02	8.70E-03
13 (Pjt)	4.23E+07	5.24E+08	16.61	2.13E-02	1.31E-02	1.13E-02	9.30E-03
Future	2.21E+08	1.01E+09	32.00	3.77E-02	2.42E-02	2.17E-02	1.80E-02
Future	3.79E+08	1.51E+09	48.00	5.48E-02	3.58E-02	3.24E-02	2.70E-02

Note: The maximum exposure occurs at the axial elevation of the circumferential weld, i.e., 11.4 inches below the midplane of the active fuel.

Table 6-5 Calculated Neutron Exposure of the Middle Shell Longitudinal Welds

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm²]		
				Weld 101-124A	Weld 101-124B	Weld 101-124C
1	3.28E+07	3.28E+07	1.04	1.48E+18	8.07E+17	8.07E+17
2	3.18E+07	6.46E+07	2.05	2.68E+18	1.47E+18	1.47E+18
3	3.64E+07	1.01E+08	3.20	4.06E+18	2.12E+18	2.12E+18
4	3.82E+07	1.39E+08	4.41	5.38E+18	2.83E+18	2.83E+18
5	3.93E+07	1.79E+08	5.66	6.75E+18	3.53E+18	3.53E+18
6	4.09E+07	2.19E+08	6.95	8.19E+18	4.02E+18	4.02E+18
7	4.26E+07	2.62E+08	8.30	8.92E+18	4.58E+18	4.58E+18
8	4.27E+07	3.05E+08	9.66	9.85E+18	5.04E+18	5.04E+18
9	4.55E+07	3.50E+08	11.10	1.08E+19	5.52E+18	5.52E+18
10	4.43E+07	3.95E+08	12.50	1.16E+19	6.08E+18	6.08E+18
11	4.19E+07	4.36E+08	13.83	1.23E+19	6.48E+18	6.48E+18
12 (Pjt)	4.53E+07	4.82E+08	15.27	1.32E+19	6.96E+18	6.96E+18
13 (Pjt)	4.23E+07	5.24E+08	16.61	1.40E+19	7.40E+18	7.40E+18
Future	2.21E+08	1.01E+09	32.00	2.48E+19	1.42E+19	1.42E+19
Future	3.79E+08	1.51E+09	48.00	3.60E+19	2.12E+19	2.12E+19

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]		
				Weld 101-124A	Weld 101-124B	Weld 101-124C
1	3.28E+07	3.28E+07	1.04	2.26E-03	1.23E-03	1.23E-03
2	3.18E+07	6.46E+07	2.05	4.07E-03	2.25E-03	2.25E-03
3	3.64E+07	1.01E+08	3.20	6.18E-03	3.25E-03	3.25E-03
4	3.82E+07	1.39E+08	4.41	8.20E-03	4.33E-03	4.33E-03
5	3.93E+07	1.79E+08	5.66	1.03E-02	5.40E-03	5.40E-03
6	4.09E+07	2.19E+08	6.95	1.25E-02	6.15E-03	6.15E-03
7	4.26E+07	2.62E+08	8.30	1.36E-02	7.01E-03	7.01E-03
8	4.27E+07	3.05E+08	9.66	1.50E-02	7.71E-03	7.71E-03
9	4.55E+07	3.50E+08	11.10	1.64E-02	8.44E-03	8.44E-03
10	4.43E+07	3.95E+08	12.50	1.77E-02	9.30E-03	9.30E-03
11	4.19E+07	4.36E+08	13.83	1.87E-02	9.92E-03	9.92E-03
12 (Pjt)	4.53E+07	4.82E+08	15.27	2.01E-02	1.07E-02	1.07E-02
13 (Pjt)	4.23E+07	5.24E+08	16.61	2.13E-02	1.13E-02	1.13E-02
Future	2.21E+08	1.01E+09	32.00	3.78E-02	2.17E-02	2.17E-02
Future	3.79E+08	1.51E+09	48.00	5.50E-02	3.25E-02	3.25E-02

Note: The maximum exposure after cycle one occurs an axial elevation of 8.3 inches below the midplane of the active fuel. The maximum exposure for all other times occurs at an axial elevation 16 8 inches above the midplane of the active fuel.

Table 6-6 Calculated Neutron Exposure of the Lower Shell Longitudinal Welds

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2]		
				Weld 101-142A	Weld 101-142B	Weld 101-142C
1	3.28E+07	3.28E+07	1.04	1.48E+18	8.05E+17	8.05E+17
2	3.18E+07	6.46E+07	2.05	2.67E+18	1.47E+18	1.47E+18
3	3.64E+07	1.01E+08	3.20	4.05E+18	2.12E+18	2.12E+18
4	3.82E+07	1.39E+08	4.41	5.36E+18	2.82E+18	2.82E+18
5	3.93E+07	1.79E+08	5.66	6.72E+18	3.52E+18	3.52E+18
6	4.09E+07	2.19E+08	6.95	8.14E+18	4.00E+18	4.00E+18
7	4.26E+07	2.62E+08	8.30	8.87E+18	4.56E+18	4.56E+18
8	4.27E+07	3.05E+08	9.66	9.80E+18	5.01E+18	5.01E+18
9	4.55E+07	3.50E+08	11.10	1.07E+19	5.49E+18	5.49E+18
10	4.43E+07	3.95E+08	12.50	1.16E+19	6.05E+18	6.05E+18
11	4.19E+07	4.36E+08	13.83	1.22E+19	6.45E+18	6.45E+18
12 (Pjt)	4.53E+07	4.82E+08	15.27	1.31E+19	6.93E+18	6.93E+18
13 (Pjt)	4.23E+07	5.24E+08	16.61	1.40E+19	7.39E+18	7.39E+18
Future	2.21E+08	1.01E+09	32.00	2.47E+19	1.41E+19	1.41E+19
Future	3.79E+08	1.51E+09	48.00	3.59E+19	2.12E+19	2.12E+19

Cycle	Cycle Length [EFPS]	Cumulative Irradiation Time [EFPS]	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]		
				Weld 101-142A	Weld 101-142B	Weld 101-142C
1	3.28E+07	3.28E+07	1.04	2.25E-03	1.23E-03	1.23E-03
2	3.18E+07	6.46E+07	2.05	4.06E-03	2.24E-03	2.24E-03
3	3.64E+07	1.01E+08	3.20	6.16E-03	3.24E-03	3.24E-03
4	3.82E+07	1.39E+08	4.41	8.17E-03	4.32E-03	4.32E-03
5	3.93E+07	1.79E+08	5.66	1.02E-02	5.38E-03	5.38E-03
6	4.09E+07	2.19E+08	6.95	1.24E-02	6.12E-03	6.12E-03
7	4.26E+07	2.62E+08	8.30	1.35E-02	6.97E-03	6.97E-03
8	4.27E+07	3.05E+08	9.66	1.49E-02	7.68E-03	7.68E-03
9	4.55E+07	3.50E+08	11.10	1.63E-02	8.41E-03	8.41E-03
10	4.43E+07	3.95E+08	12.50	1.76E-02	9.26E-03	9.26E-03
11	4.19E+07	4.36E+08	13.83	1.86E-02	9.88E-03	9.88E-03
12 (Pjt)	4.53E+07	4.82E+08	15.27	2.00E-02	1.06E-02	1.06E-02
13 (Pjt)	4.23E+07	5.24E+08	16.61	2.13E-02	1.13E-02	1.13E-02
Future	2.21E+08	1.01E+09	32.00	3.77E-02	2.17E-02	2.17E-02
Future	3.79E+08	1.51E+09	48.00	5.48E-02	3.24E-02	3.24E-02

Note: The maximum exposure occurs at the axial elevation of the circumferential weld, i.e., 11.4 inches below the midplane of the active fuel.

Table 6-7 Relative Radial Distribution of Neutron Fluence (E > 1.0 MeV)

RADIUS (cm)	AZIMUTHAL ANGLE			
	0°	15°	30°	45°
221.36	1.000	1.000	1.000	1.000
226.84	0.557	0.563	0.558	0.562
232.31	0.270	0.280	0.271	0.276
237.79	0.125	0.133	0.126	0.129
243.26	0.052	0.060	0.055	0.058
Note:	Base Metal Inner Radius = 221.36 cm Base Metal 1/4T* = 226.84 cm Base Metal 1/2T* = 232.31 cm Base Metal 3/4T* = 237.79 cm Base Metal Outer Radius = 243.26 cm			

*excludes cladding in thickness dimension

Table 6-8 Relative Radial Distribution of Iron Atom Displacements (dpa)

RADIUS (cm)	AZIMUTHAL ANGLE			
	0°	15°	30°	45°
221.36	1.000	1.000	1.000	1.000
226.84	0.630	0.640	0.631	0.636
232.31	0.374	0.391	0.377	0.384
237.79	0.217	0.235	0.223	0.230
243.26	0.111	0.130	0.122	0.129
Note:	Base Metal Inner Radius = 221.36 cm Base Metal 1/4T* = 226.84 cm Base Metal 1/2T* = 232.31 cm Base Metal 3/4T* = 237.79 cm Base Metal Outer Radius = 243.26 cm			

*excludes cladding in thickness dimension

Table 6-9 Calculated Fast Neutron Exposure of Surveillance Capsules

Capsule	Irradiation Time [EFPY]	Fluence (E > 1.0 MeV) [n/cm^2]	Iron Displacements [dpa]
W-97	4.41	6.35E+18	9.27E-03
W-263	13.83	1.45E+19	2.12E-02

Table 6-10 Calculated Surveillance Capsule Lead Factors

Capsule ID And Location	Status	Lead Factor
W-97 (7°)	Withdrawn EOC 4	1.18
W-263 (7°)	Withdrawn EOC 11	1.18
W-83 (7°)	In Reactor	1.18
W-104 (14°)	In Reactor	0.83
W-277 (7°)	In Reactor	1.18
W-284 (14°)	In Reactor	0.83

Note: (1) Lead factors for capsules remaining in the reactor are based on cycle specific exposure calculations through the current operating fuel reload, i.e., Cycle 12.

7 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule meets the intent of ASTM E185-82 and is recommended for future capsules to be removed from the Waterford Unit 3 reactor vessel. This recommended removal schedule is applicable to 32 EFPY of operation.

Table 7-1 Waterford Unit 3 Reactor Vessel Surveillance Capsule Withdrawal Schedule

Capsule	Location	Lead Factor[(a)]	Removal Time (EFPY)[(b)]	Fluence (n/cm^2, E > 1.0 MeV)
97°	97°	1.18	4.44[(c)]	6.47×10^{18} [(c)]
104°	104°	0.83	Standby	
284°	284°	0.83	Standby	
263°	263°	1.18	13.83	1.45×10^{19}
83°	83°	1.18	26	2.47×10^{19} [(d)]
277°	277°	1.18	Standby	

Notes:

(a) Updated based on Capsule 263° dosimetry analysis.

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

(c) From Capsule 97° capsule evaluation report, Reference 13.

(d) Capsule 83° will reach the EOL (32 EFPY) vessel inside surface fluence of 2.47×10^{19} n/cm^2 (E > 1.0 MeV) at approximately 26 EFPY.

8 REFERENCES

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5. ASTM E208, *Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA

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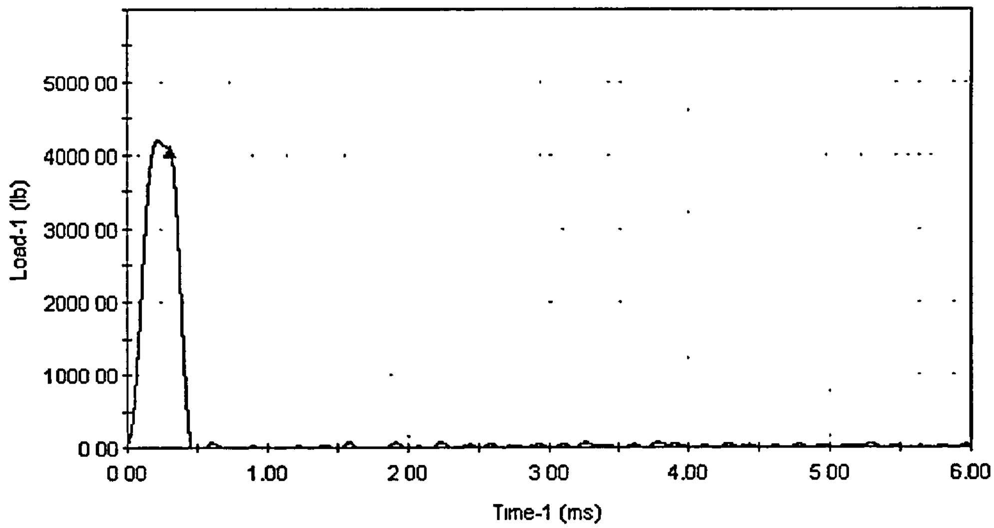
17. RSIC DLC-185, "BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications", March 1996

18. C-PENG-ER-004, Revision 0, "The Reactor Vessel Group Records Evaluation Program Phase II Final Report for the Waterford 3 Reactor Pressure Vessel Plates, Forgings, Welds and Cladding", October 1995.

APPENDIX A

INSTRUMENTED CHARPY IMPACT TEST CURVES

- Specimen prefix “2” denotes Lower Shell Plate M-1004-2, Transverse Orientation
- Specimen prefix “A” denotes Correlation Monitor Material, Longitudinal Orientation
- Specimen prefix “3” denotes Weld Material
- Specimen prefix “4” denotes Heat-Affected Zone material



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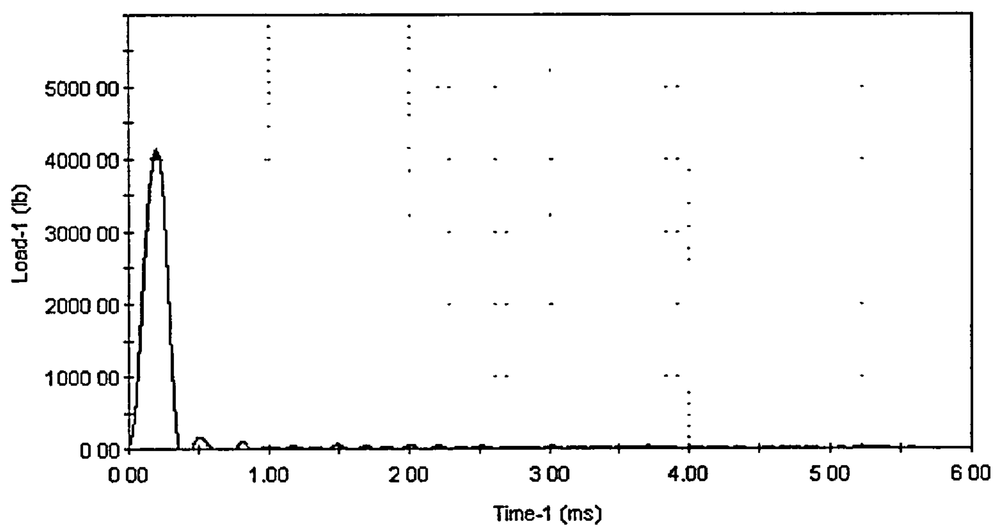

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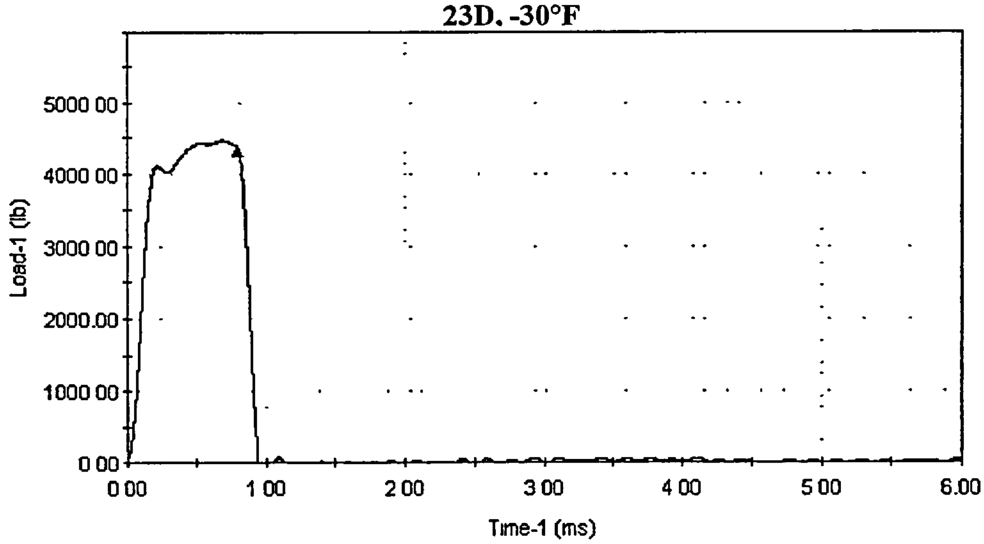

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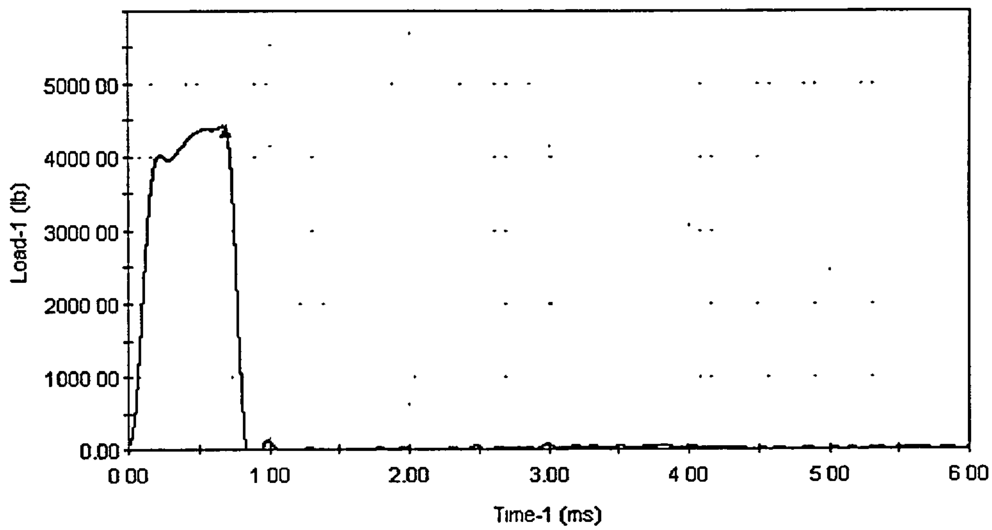

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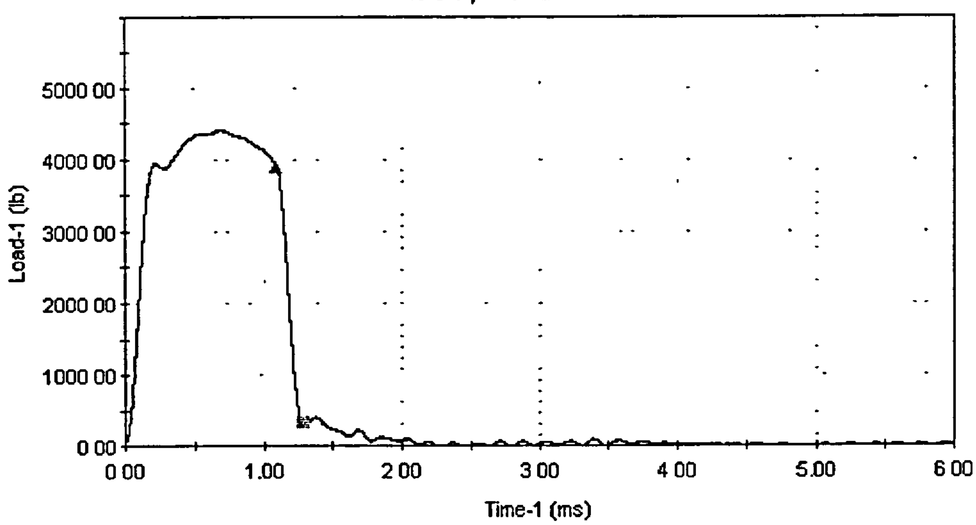

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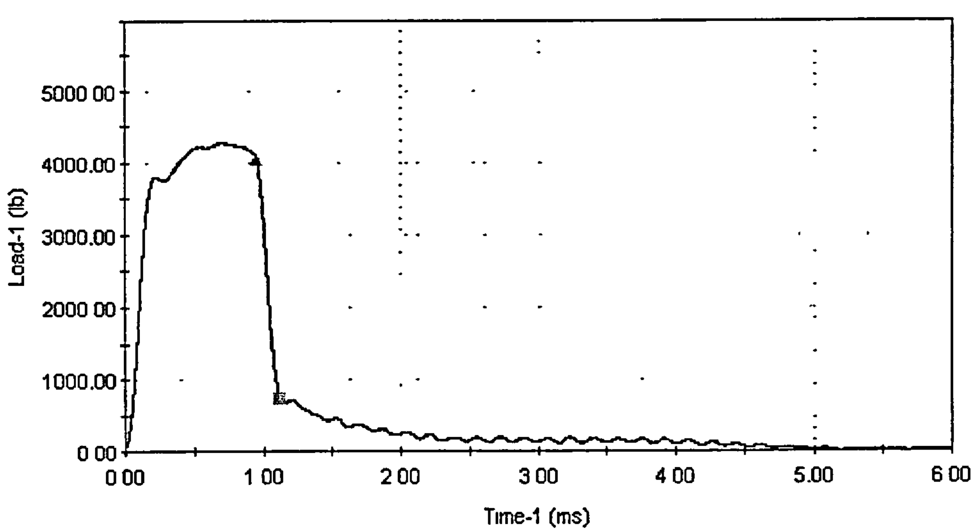

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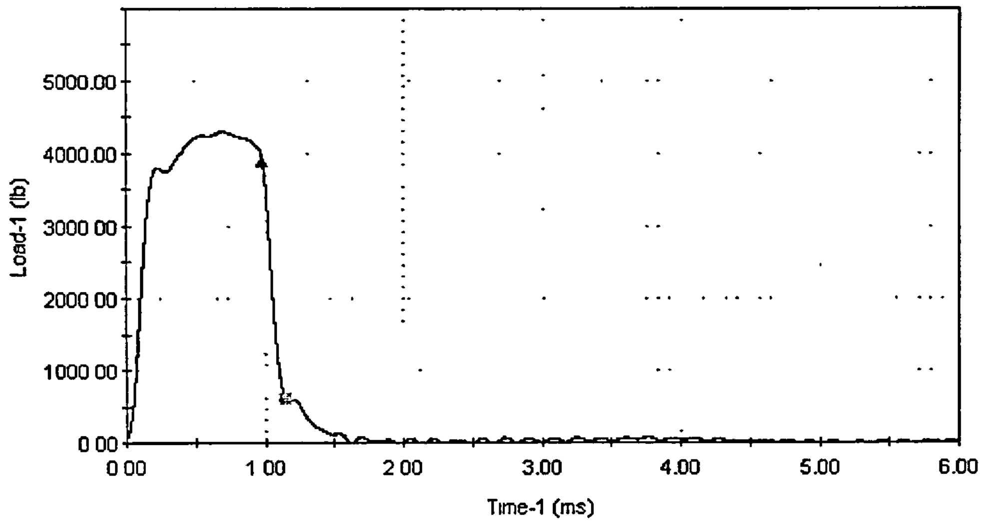

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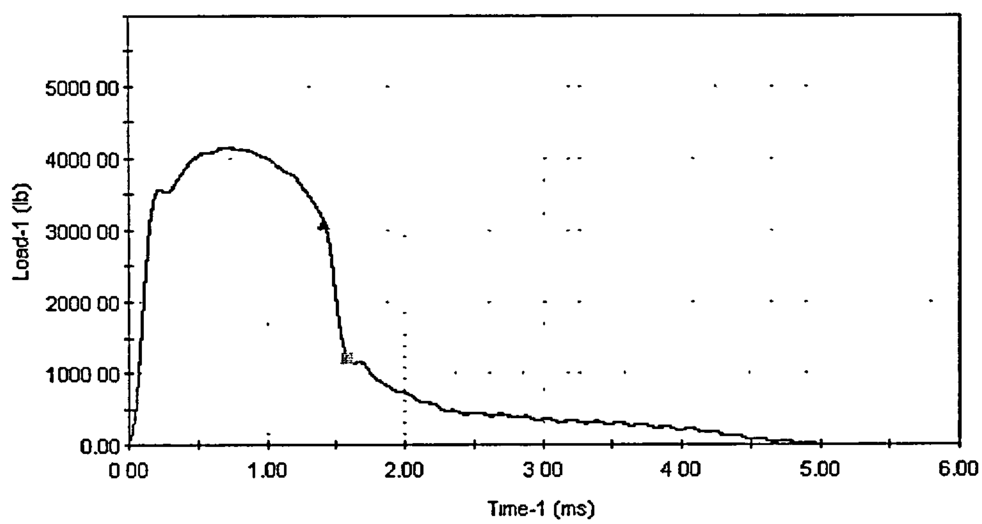

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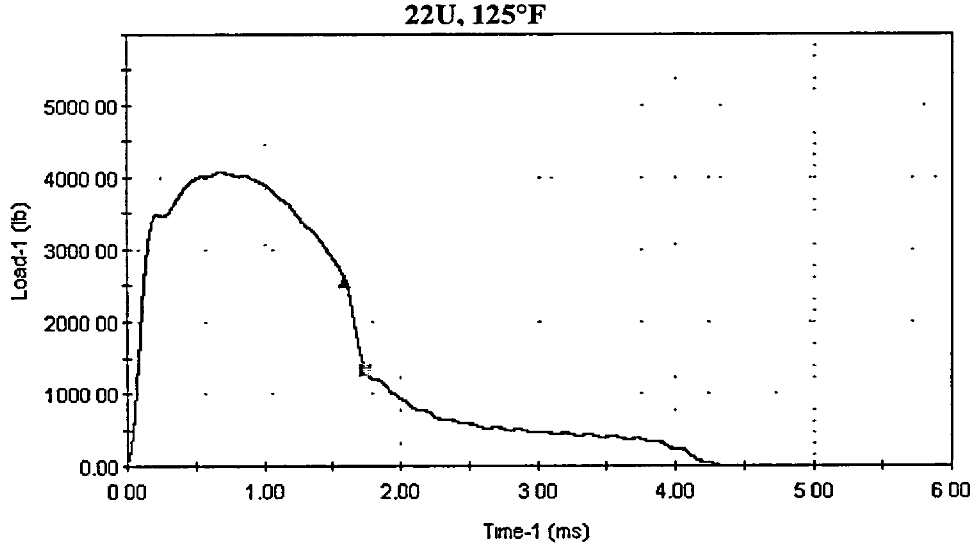

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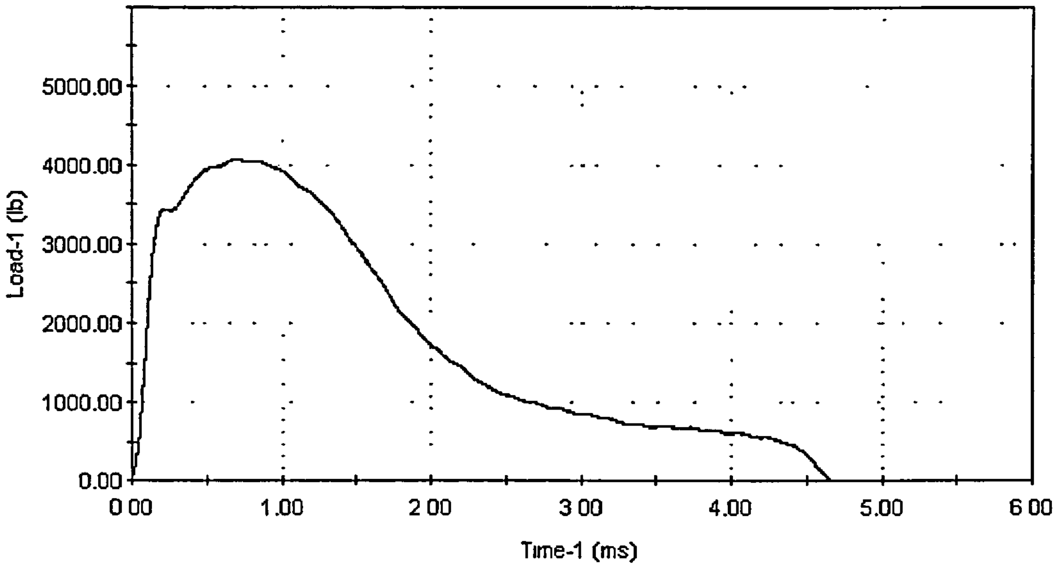
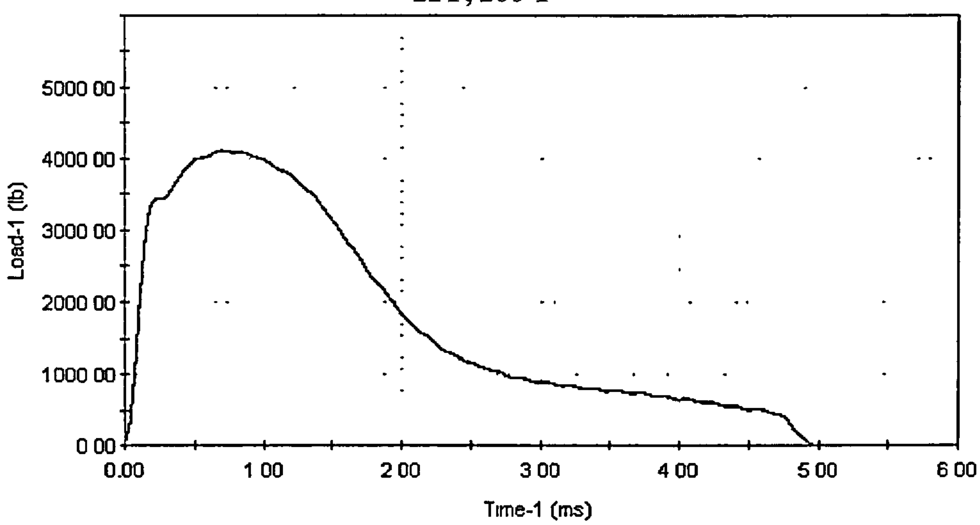


22Y, 200°F



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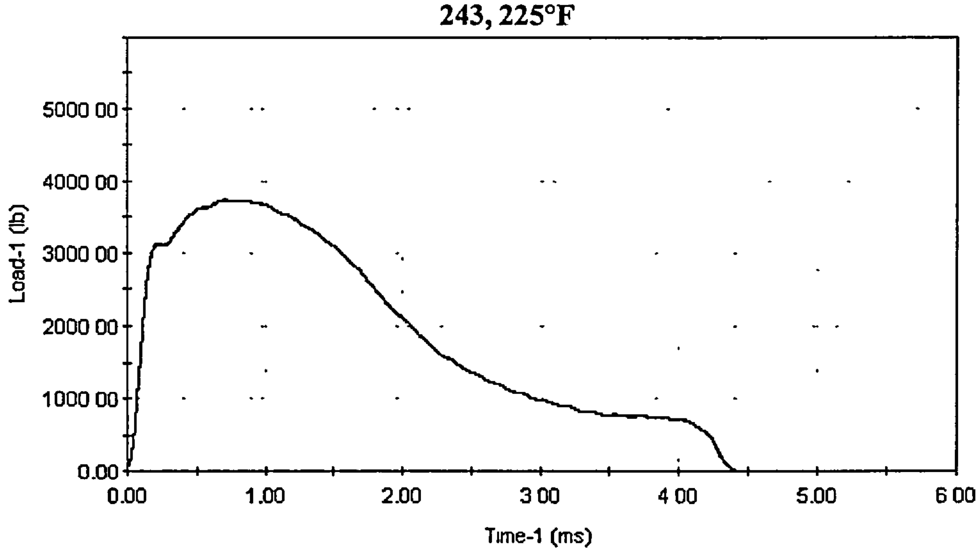
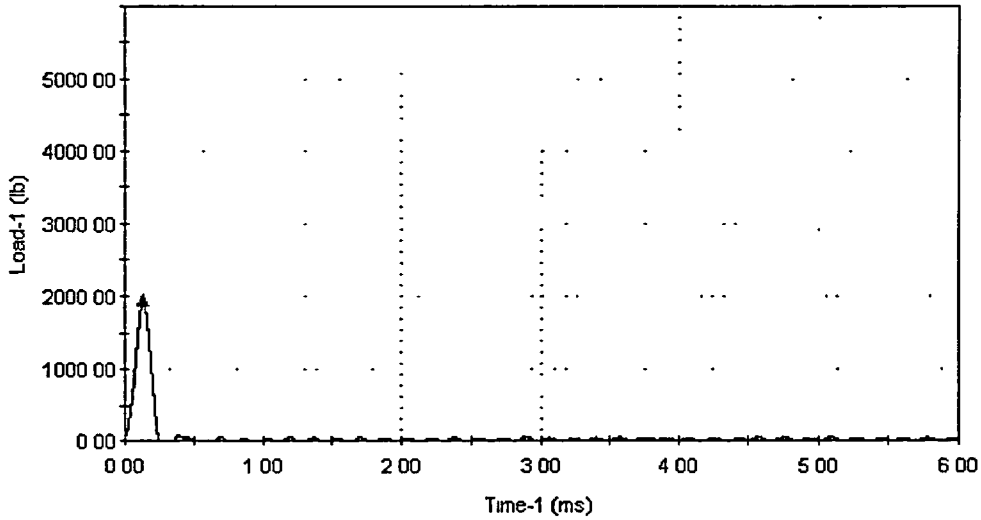
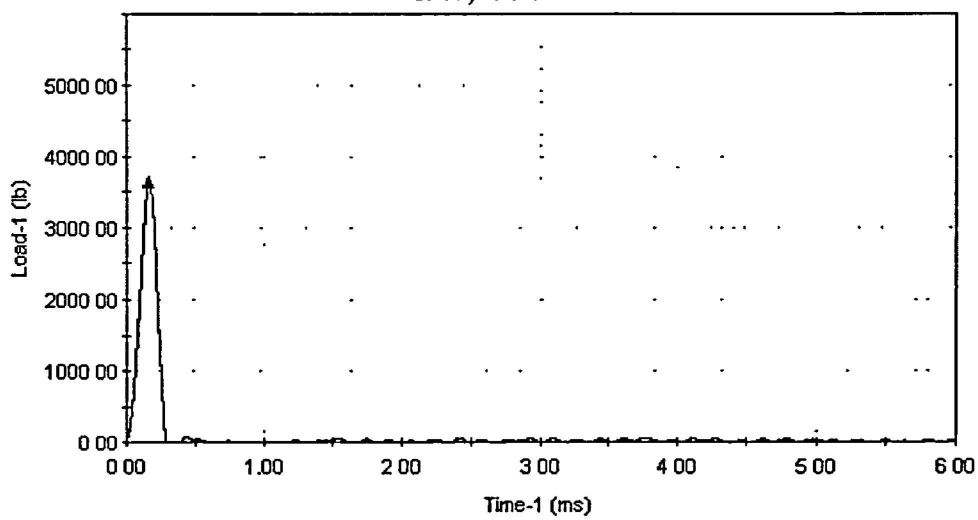


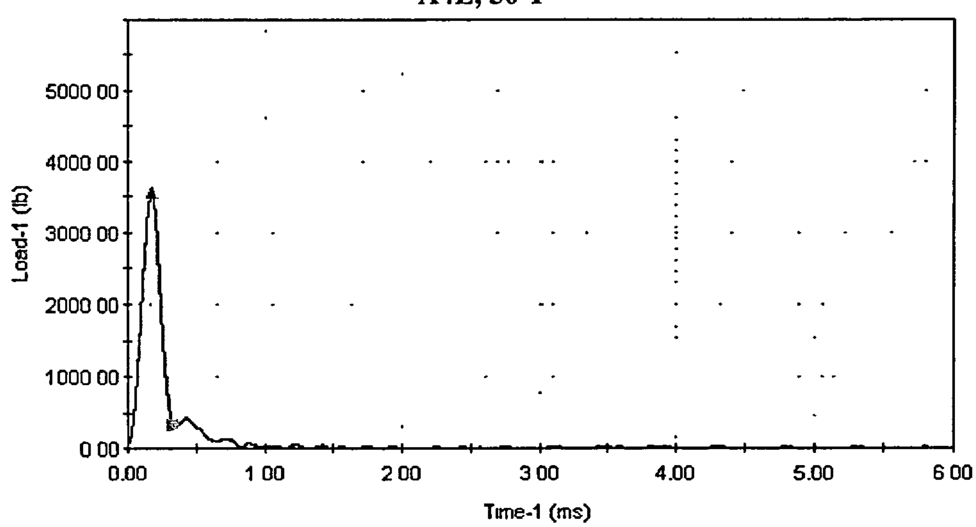
24M, 350°F



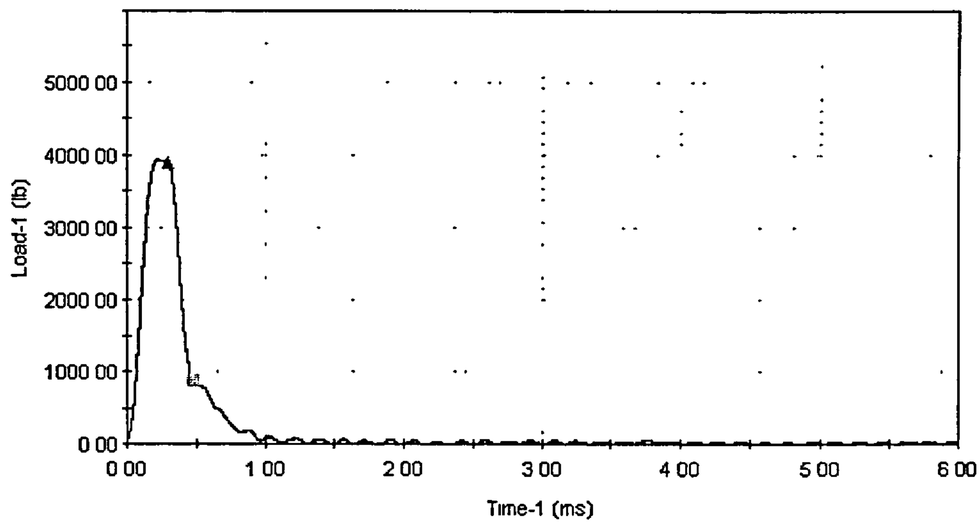
A47, -30°F



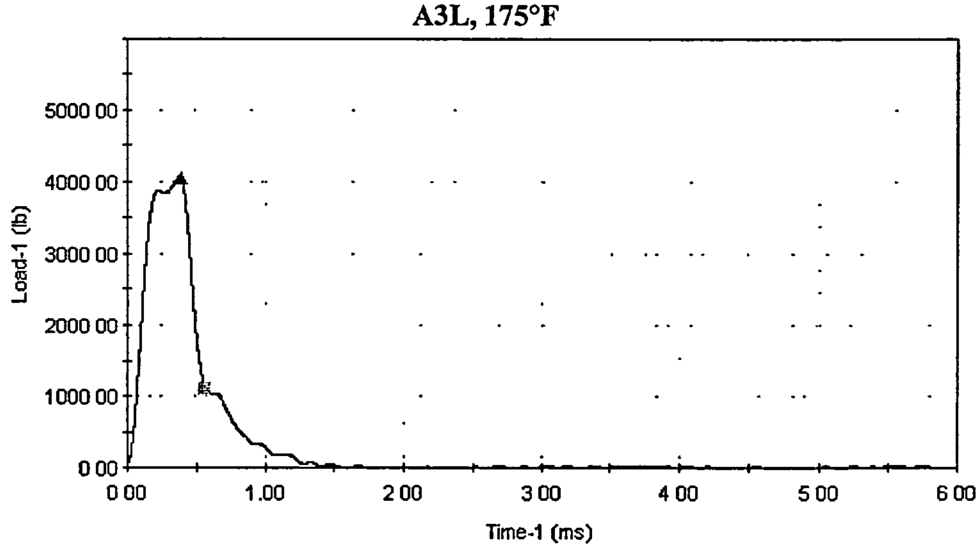
A4E, 50°F



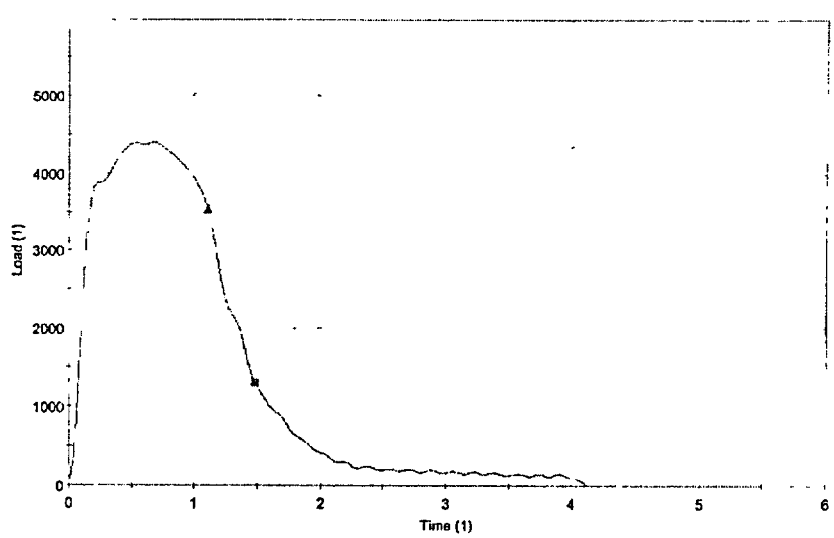
A46, 125°F



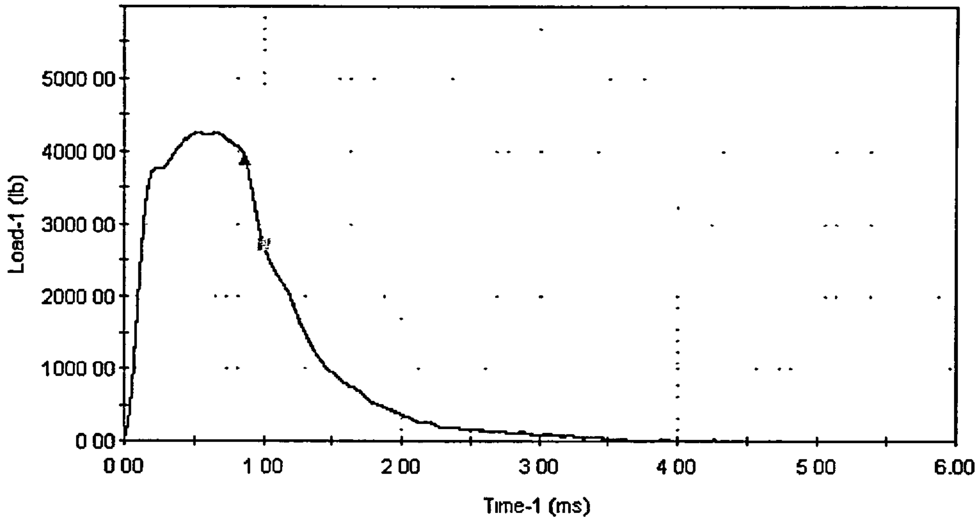
A3L, 175°F



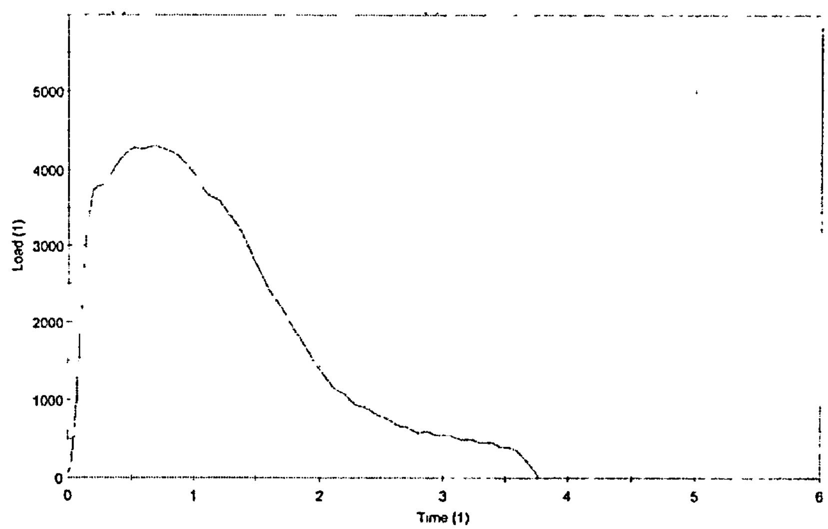
A4C, 200°F



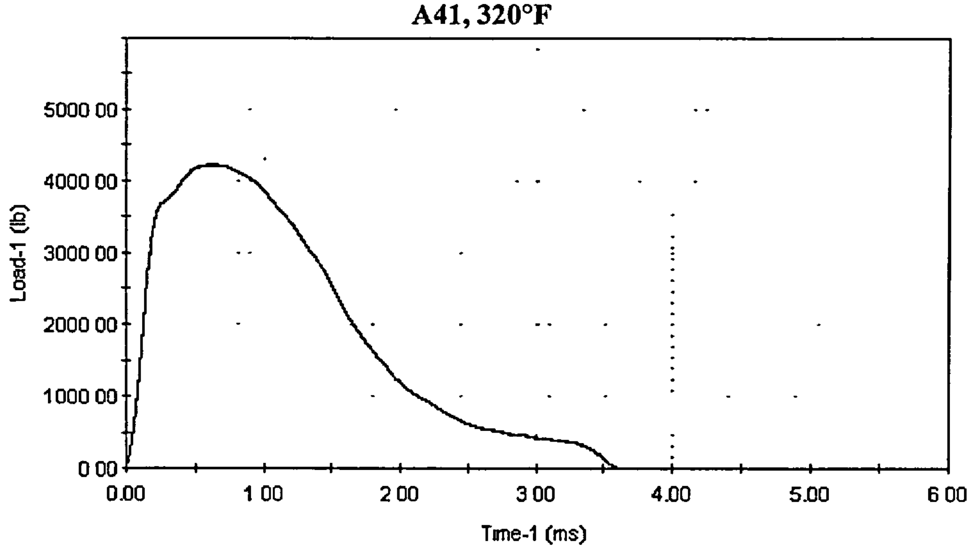
A3P, 240°F



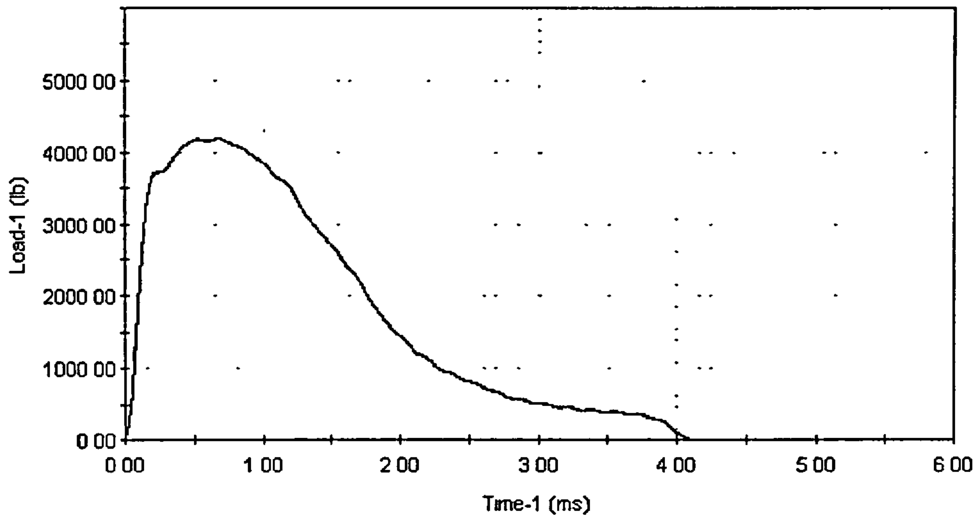
A4A, 275°F



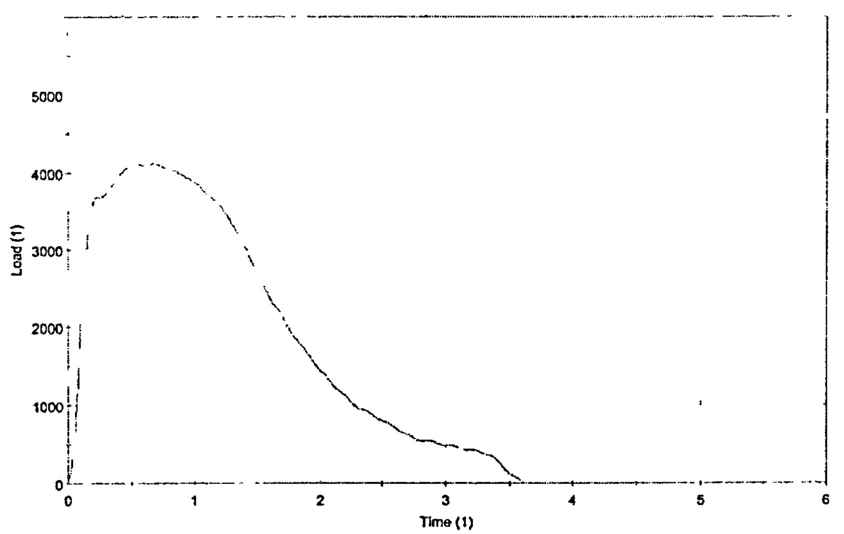
A41, 320°F



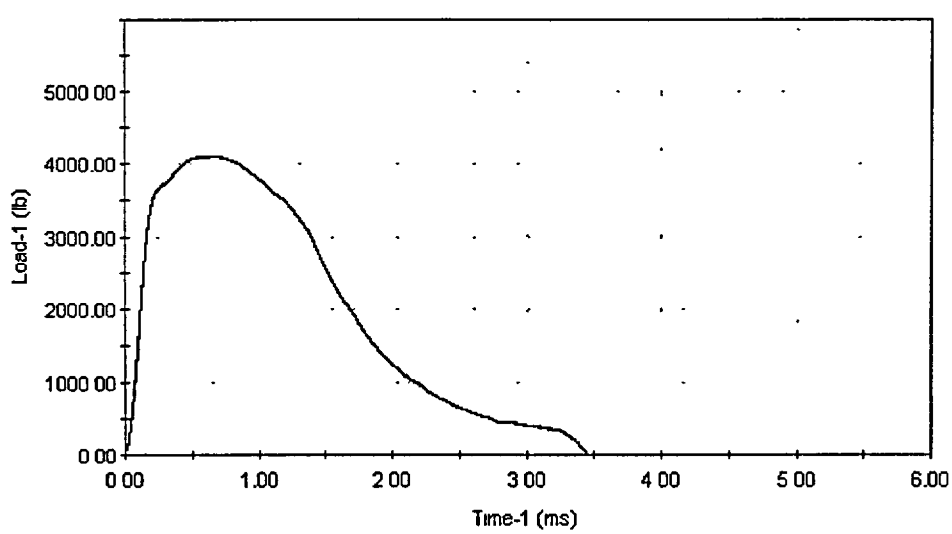
A43, 350°F



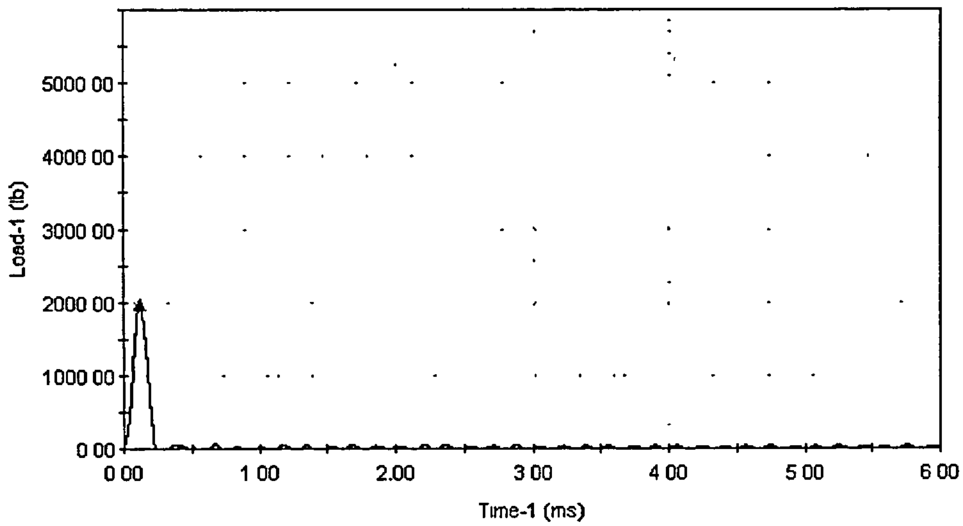
A4K, 375°F



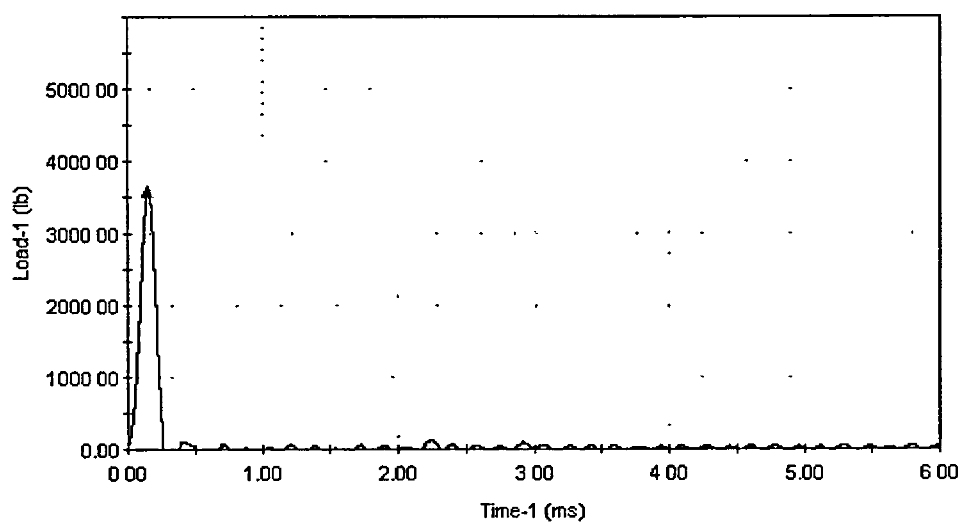
A3E, 425°F



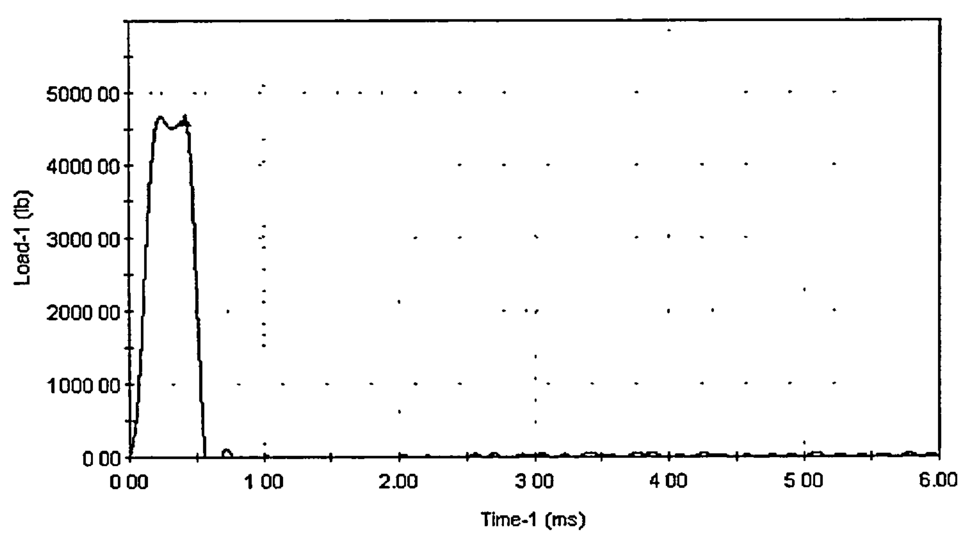
A3Y, 460°F



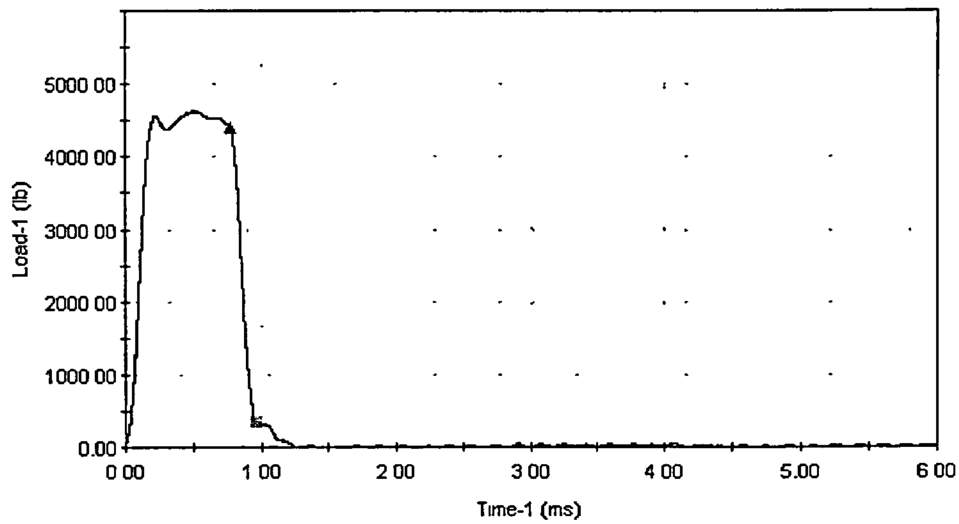
333, -175°F



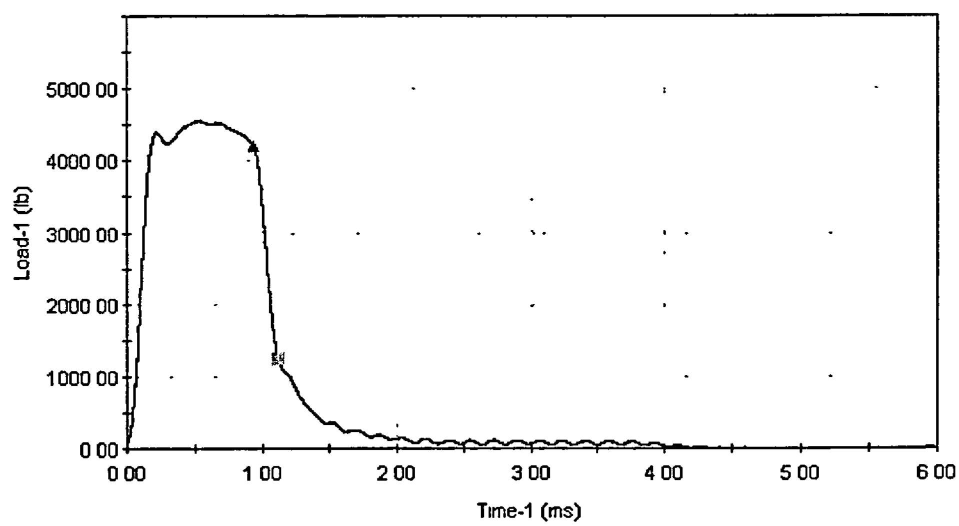
332, -125°F



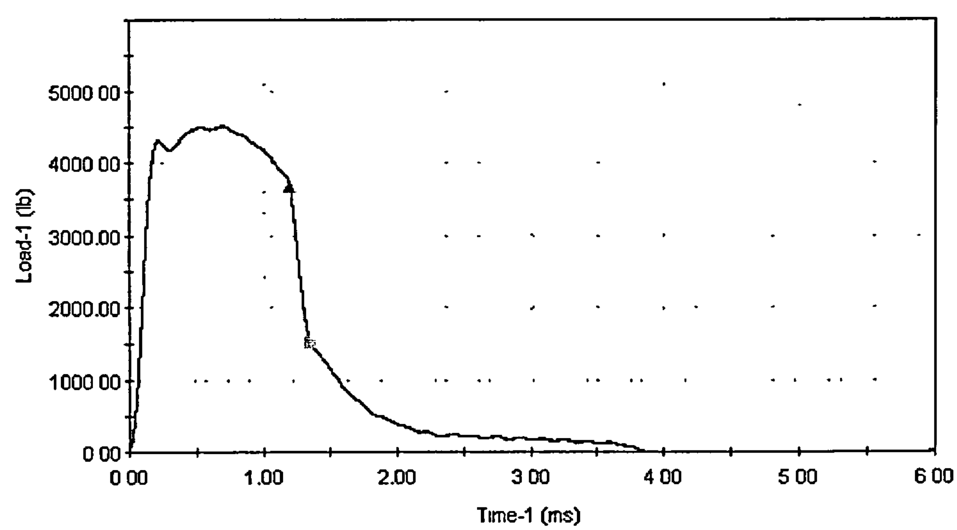
363, -75°F



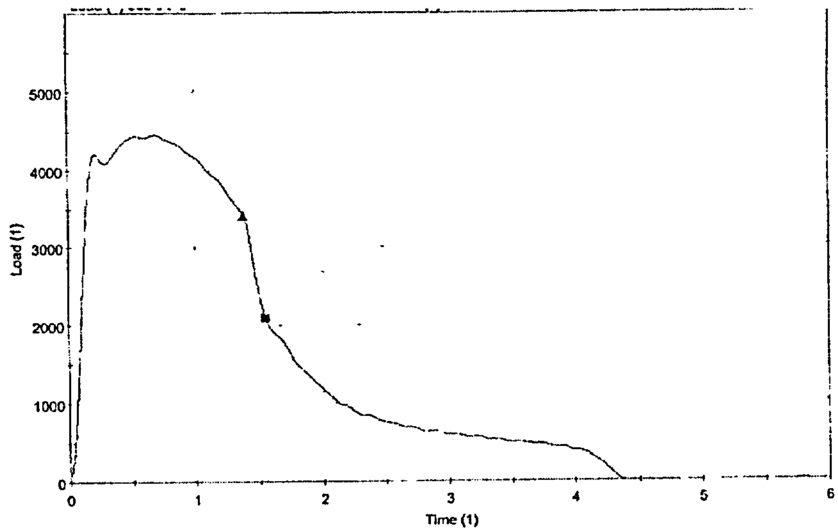
34B, -50°F



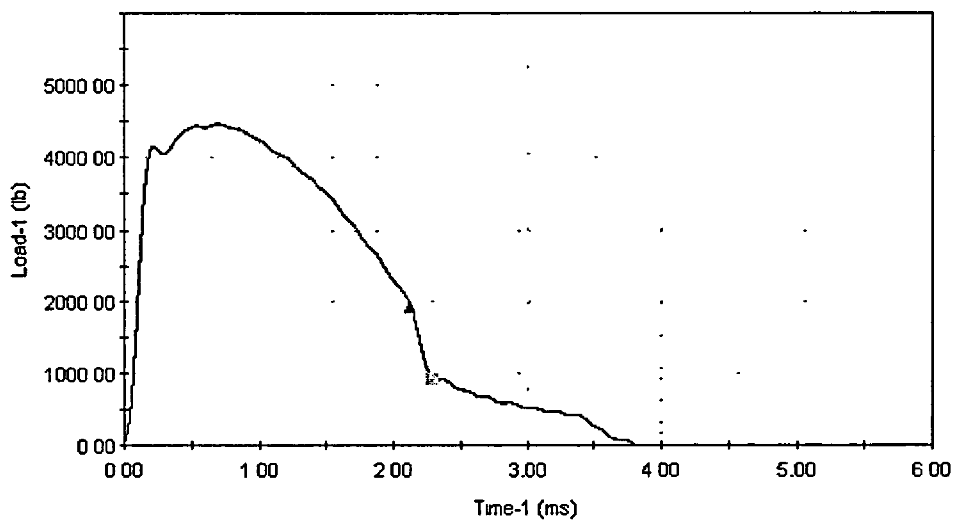
35Y, -25°F



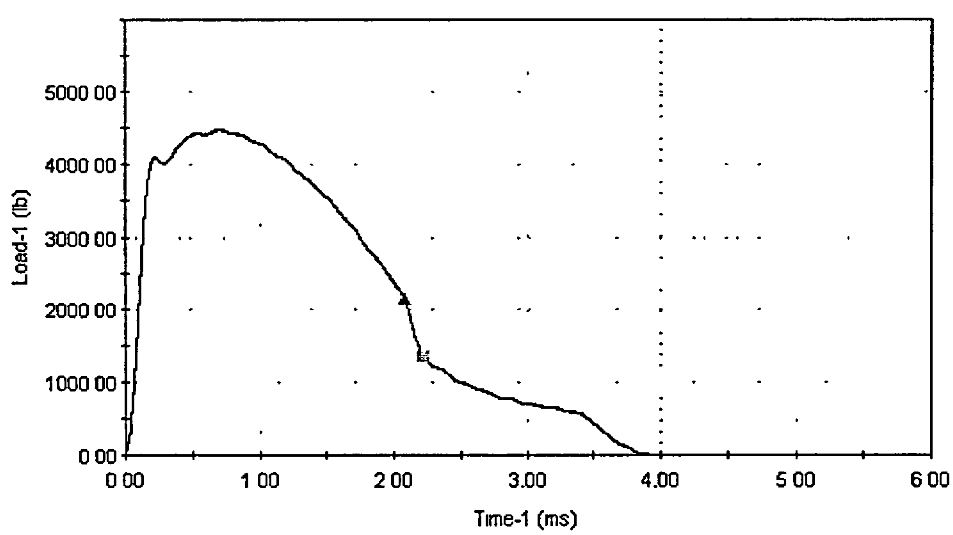
361, 0°F



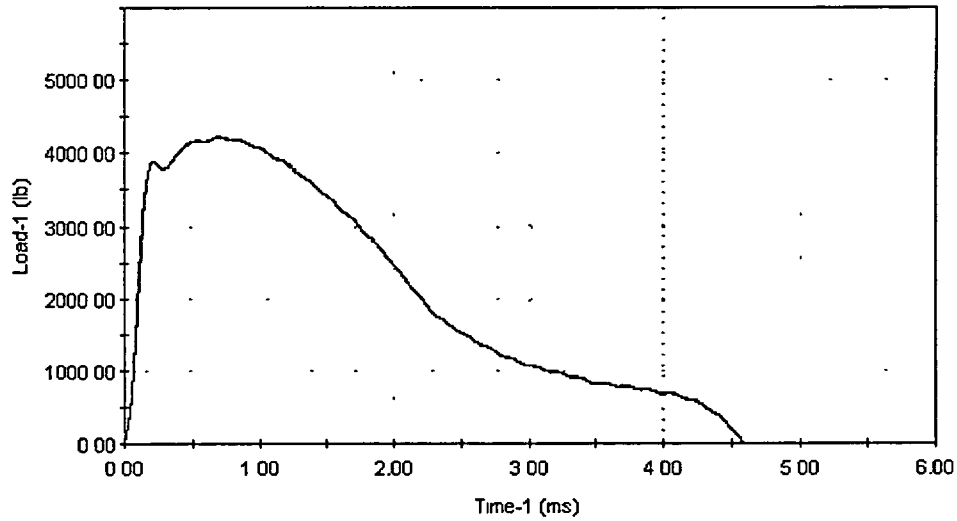
34K, 25°F



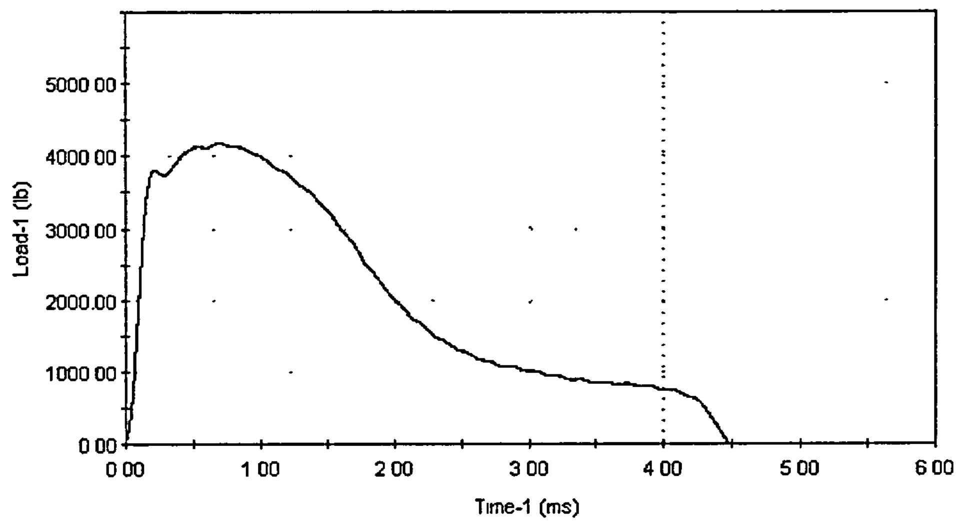
33D, 50°F



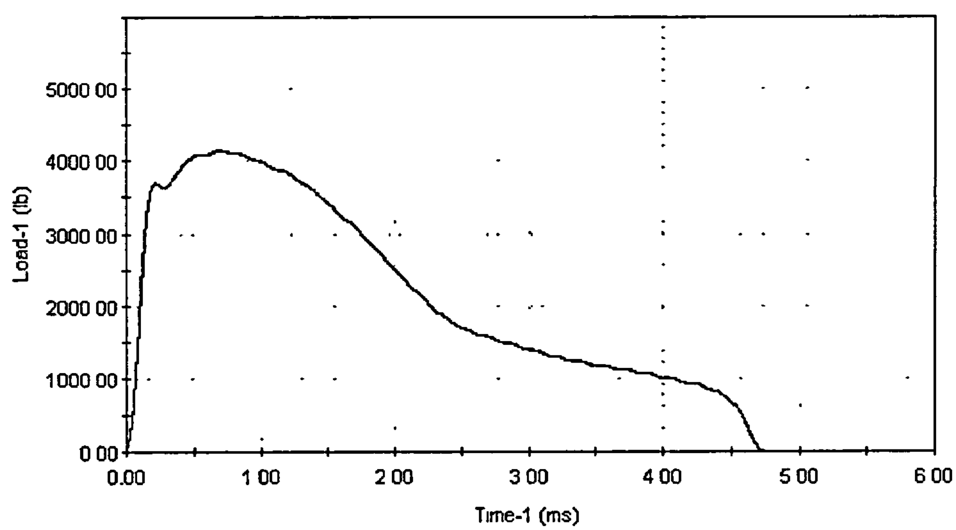
33U, 75°F



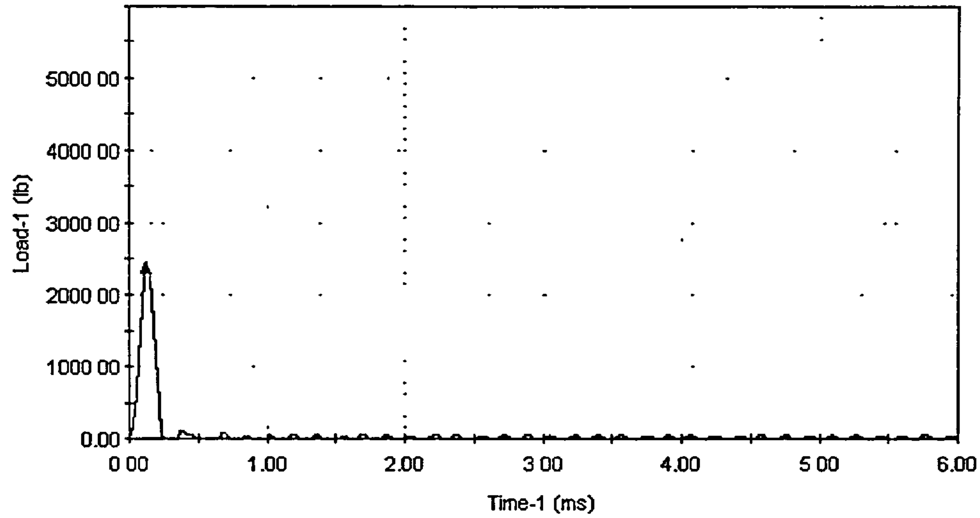
365, 100°F



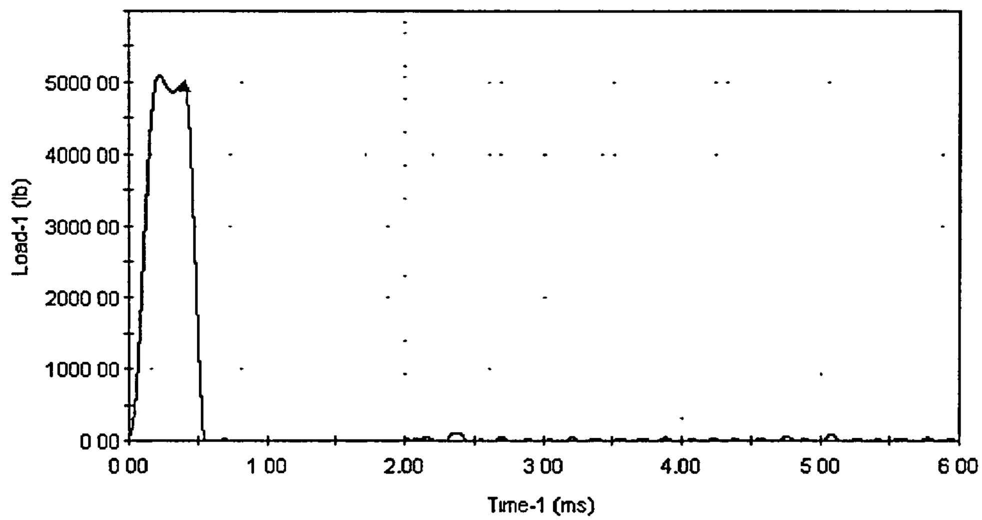
37A, 150°F



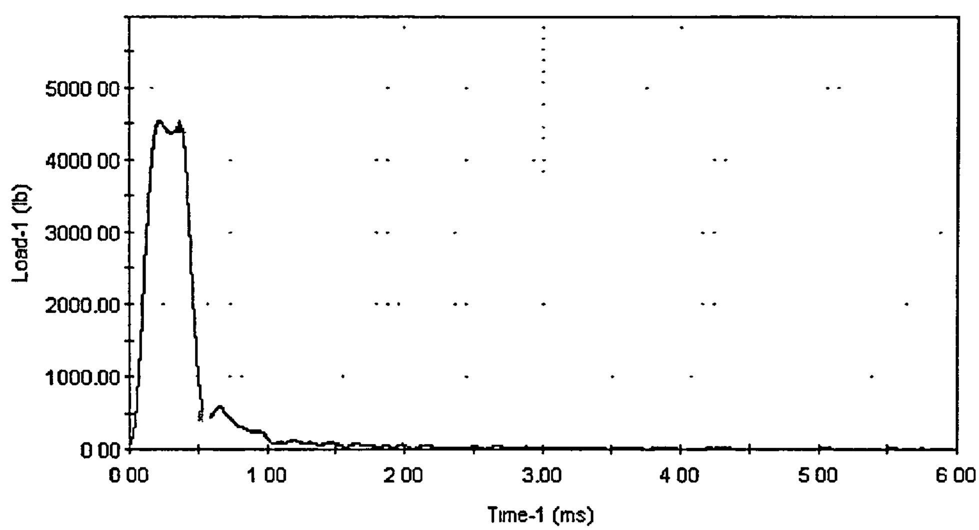
35L, 200°F



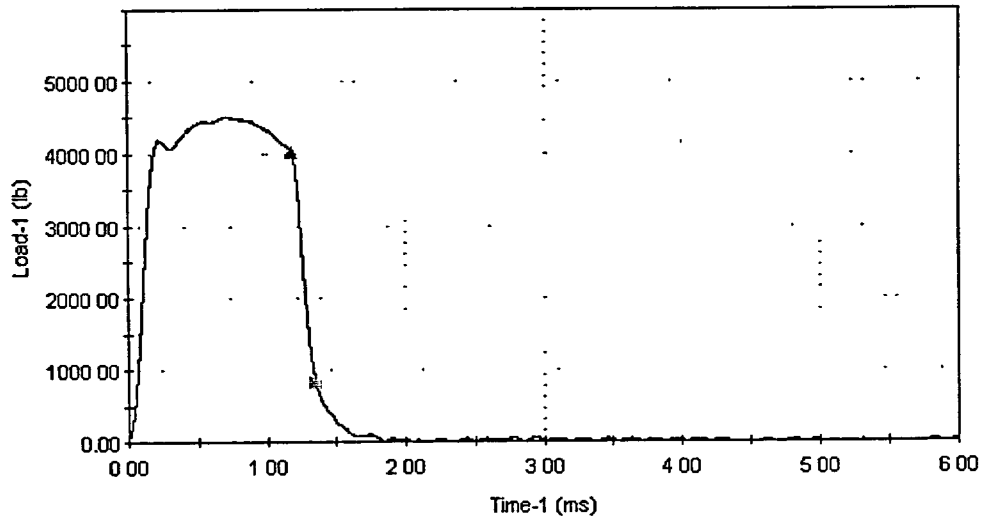
41P, -175°F



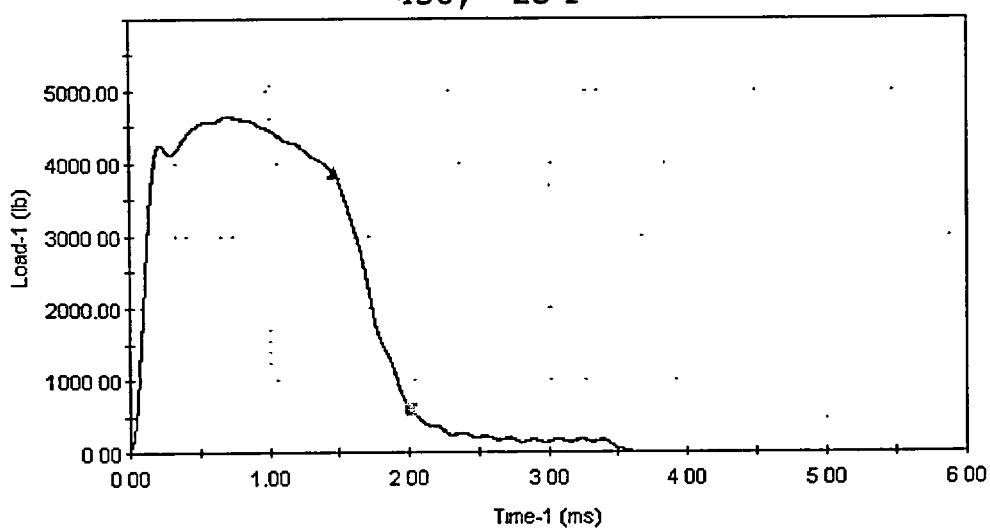
41D, -125°F



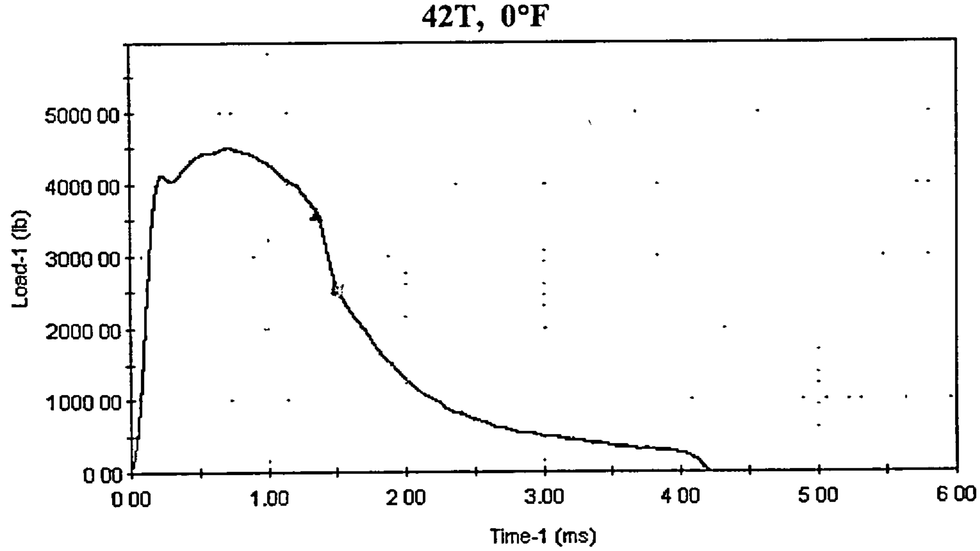
43E, -75°F



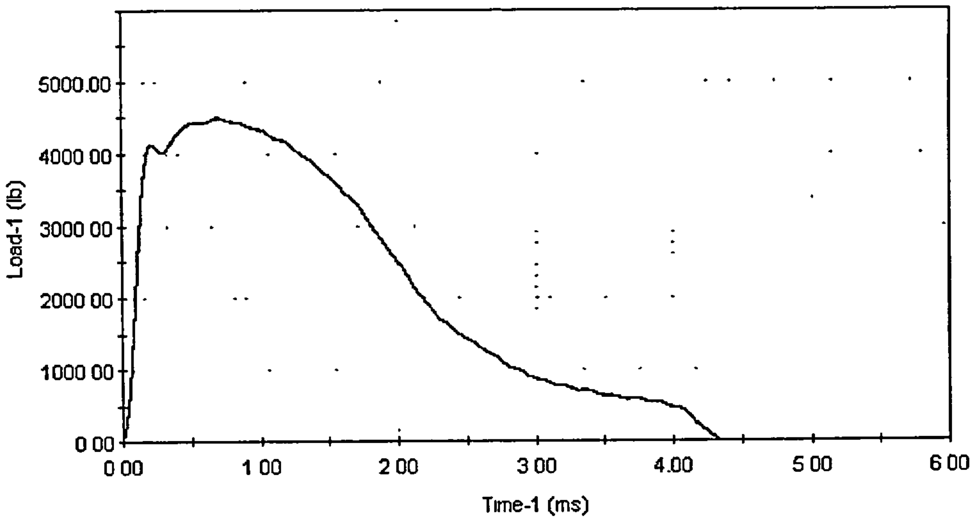
436, -25°F



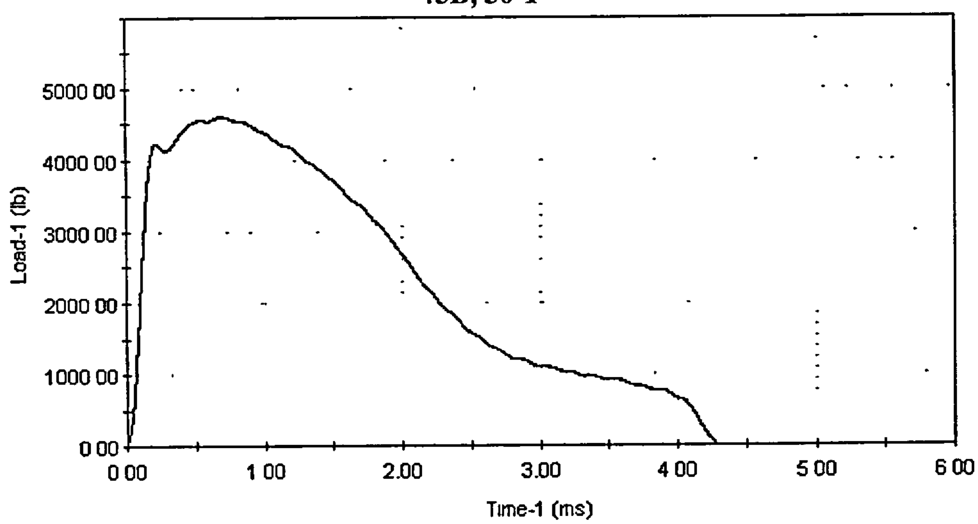
42T, 0°F



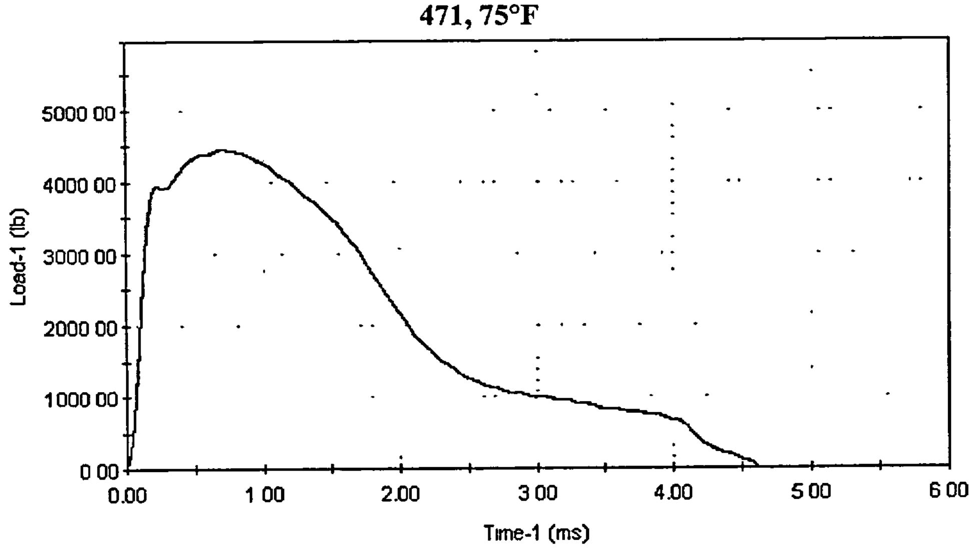
41U, 25°F



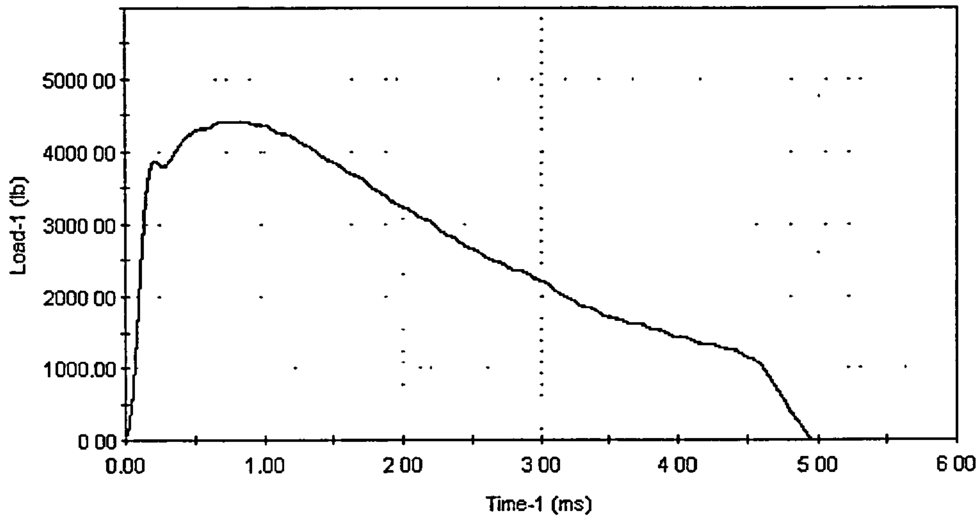
45B, 50°F



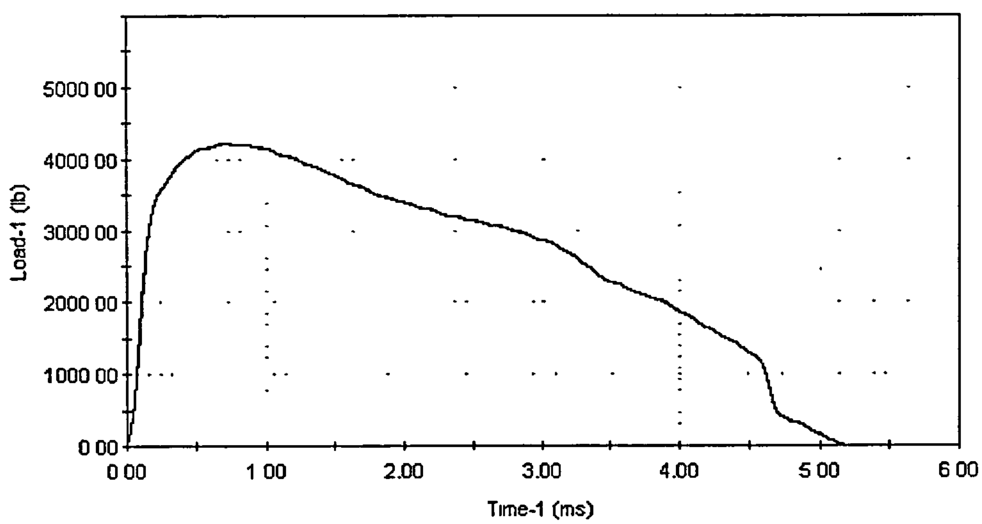
471, 75°F



46L, 110°F



473, 150°F



44A, 325°F

APPENDIX B

Charpy V-Notch Plots for Each Capsule

Charpy V-notch plots for each capsule are given in the following pages. They were determined using the Hyperbolic Tangent Curve-Fitting Method. Contained in Table B-1 are the upper shelf energy values used as input for the generation of the Charpy V-notch plots using CVGRAPH, Version 4.1. Lower shelf energy values were fixed at 2.2 ft-lb. The unirradiated and irradiated upper shelf energy values were calculated per the ASTM E185-82 definition of upper shelf energy.

Table B-1 Upper Shelf Energy Values Fixed in CVGRAPH

Material	Unirradiated	Capsule 97°	Capsule 263°
Lower Shell Plate M-1004-2 (Longitudinal Orientation)	170 ft-lb	155 ft-lb	---
Lower Shell Plate M-1004-2 (Transverse Orientation)	141 ft-lb	124 ft-lb	131 ft-lb
Weld Metal (Heat # 88114)	156 ft-lb	143 ft-lb	145 ft-lb
HAZ Material	170 ft-lb	156 ft-lb	163 ft-lb
Correlation Monitor Material (HSST Plate 01 MY)	133 ft-lb	---	113 ft-lb

UNIRRADIATED PLATE M-1004-2 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:53:15 on 09-30-2002

Page 1

Coefficients of Curve 1

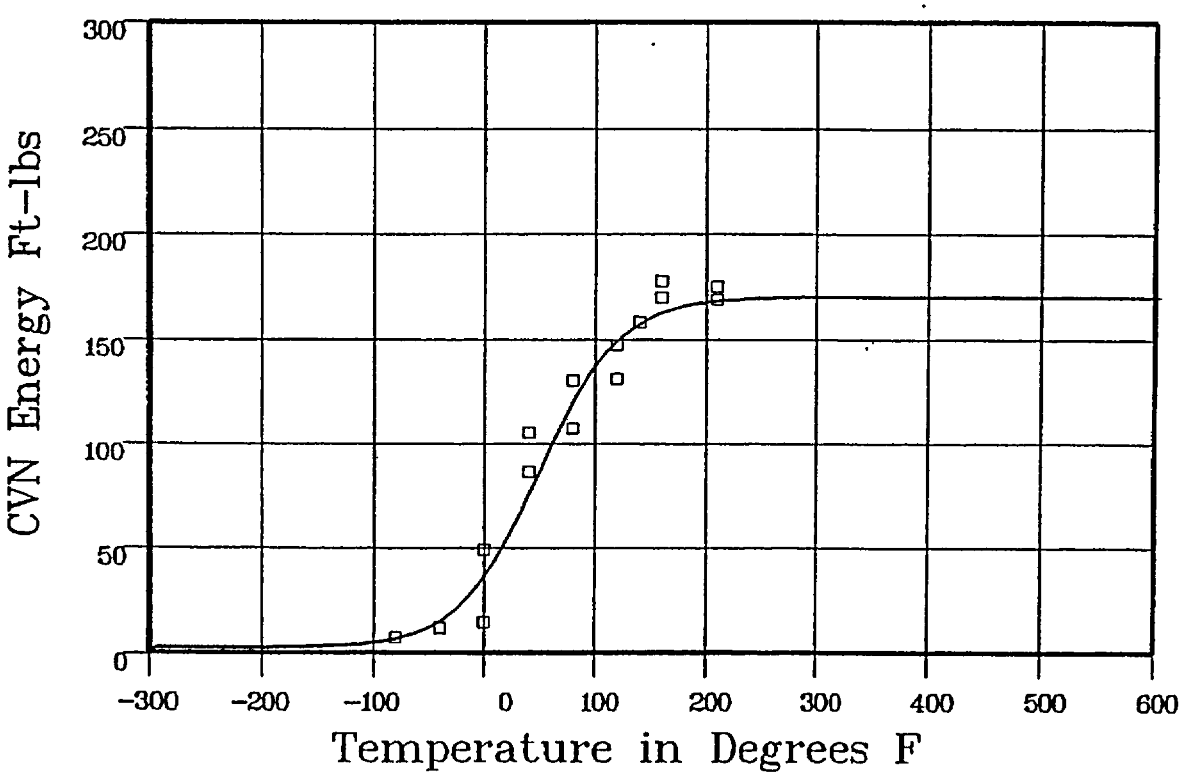
A = 86.09	B = 83.9	C = 72.21	T0 = 44.99

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

Upper Shelf Energy: 170 Fixed Temp. at 30 ft-lbs: -13.3 Temp. at 50 ft-lbs: 11.7 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: UNIRR Material: PLATE SA533B1 Ori.: LT Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-80	7	7.3	-.3
-40	11.5	16.75	-5.25
-40	12	16.75	-4.75
0	48.5	39.68	8.81
0	14.5	39.68	-25.18
40	105	80.3	24.69
40	86	80.3	5.69
80	130	123.85	6.14
80	107	123.85	-16.85

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

UNIRRADIATED PLATE M-1004-2 (LONGITUDINAL)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: UNIRR Total Fluence.

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
120	147	151.31	-4.31
120	131	151.31	-20.31
140	158	158.72	-.72
160	177.5	163.33	14.16
160	169.5	163.33	6.16
210	175	168.27	6.72
210	168.5	168.27	.22

SUM of RESIDUALS = -5.08

UNIRRADIATED PLATE M–1004–2 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:58:42 on 09-30-2002
Page 1
Coefficients of Curve 1

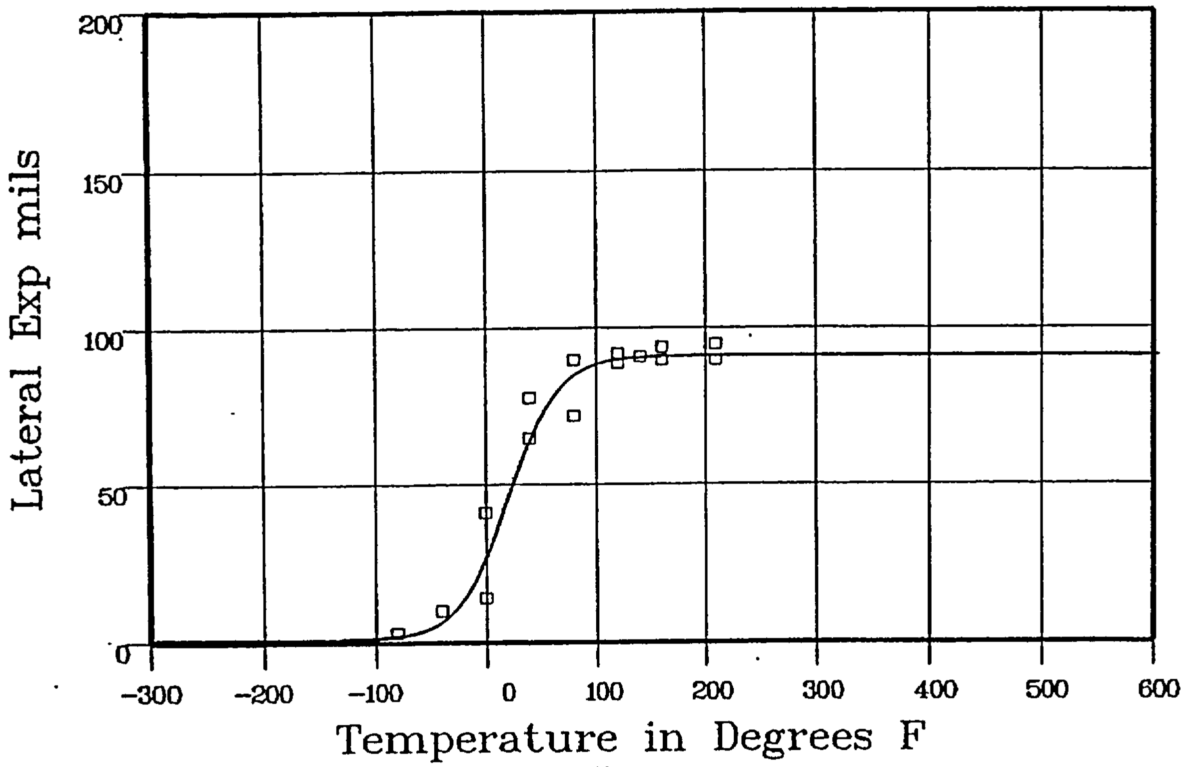
A = 46.13	B = 45.13	C = 44.98	T0 = 16.4

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

Upper Shelf L.E.: 91.27 Temperature at L.E. 35: 5 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap.: UNIRR Material: PLATE SA533B1 Ori.: LT Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-80	3	2.22	.77
-40	10	7.8	2.19
-40	10	7.8	2.19
0	41	30.37	10.62
0	14	30.37	-16.37
40	78	67.85	10.14
40	65	67.85	-2.85
80	90	86.23	3.76
80	72	86.23	-14.23

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

UNIRRADIATED PLATE M-1004-2 (LONGITUDINAL)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed LE	Differential
120	92	90.38	1.61
120	89	90.38	-1.38
140	91	90.91	.08
160	94	91.12	2.87
160	90	91.12	-1.12
210	95	91.26	3.73
210	90	91.26	-1.26

SUM of RESIDUALS = .78

UNIRRADIATED PLATE M-1004-2 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:25:59 on 10-01-2002
Page 1
Coefficients of Curve 1

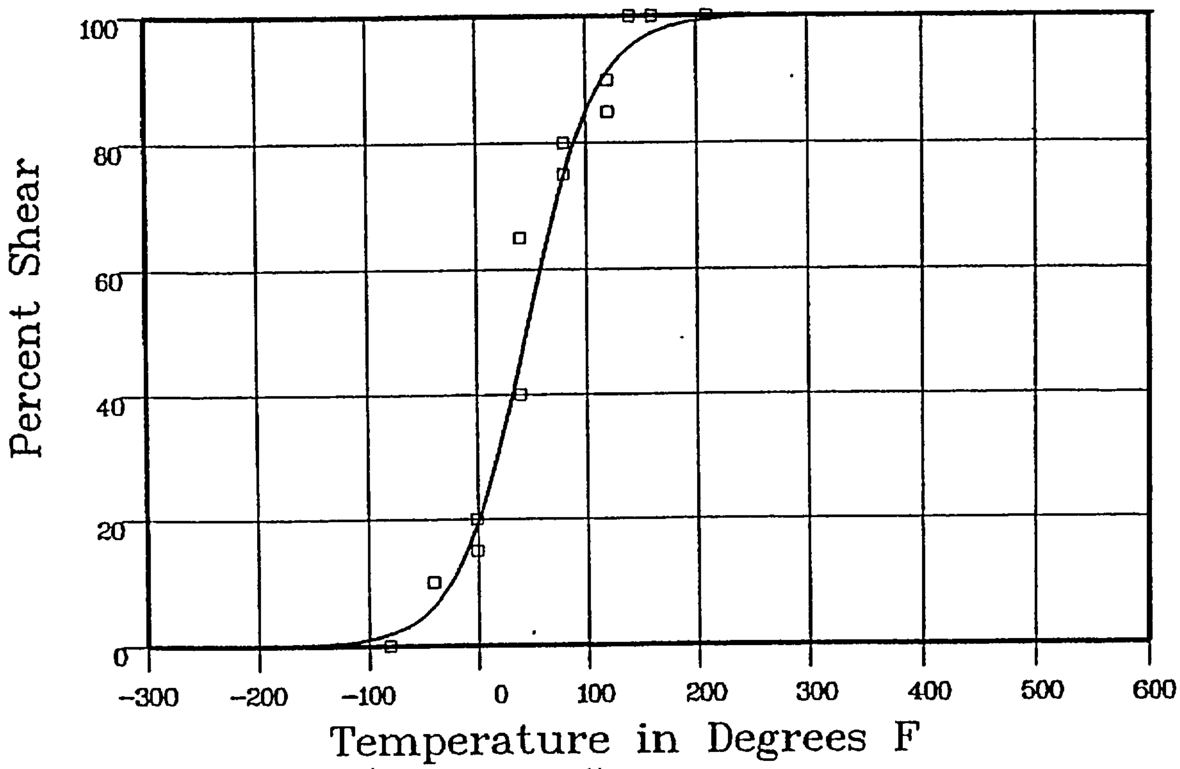
A = 50	B = 50	C = 63.15	T0 = 40.78

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

Temperature at 50% Shear: 40.7

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: UNIRR Material: PLATE SA533B1 Ori: LT Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-80	0	2.13	-2.13
-40	10	7.18	2.81
-40	10	7.18	2.81
0	20	21.56	-1.56
0	15	21.56	-6.56
40	65	49.38	15.61
40	40	49.38	-9.38
80	80	77.58	2.41
80	75	77.58	-2.58

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

UNIRRADIATED PLATE M-1004-2 (LONGITUDINAL)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
120	90	92.47	-2.47
120	85	92.47	-7.47
140	100	95.85	4.14
160	100	97.75	2.24
160	100	97.75	2.24
210	100	99.53	.46
210	100	99.53	.46

SUM of RESIDUALS = 1.03

CAPSULE 97 PLATE M-1004-2 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:30:59 on 09-30-2002

Page 1

Coefficients of Curve 2

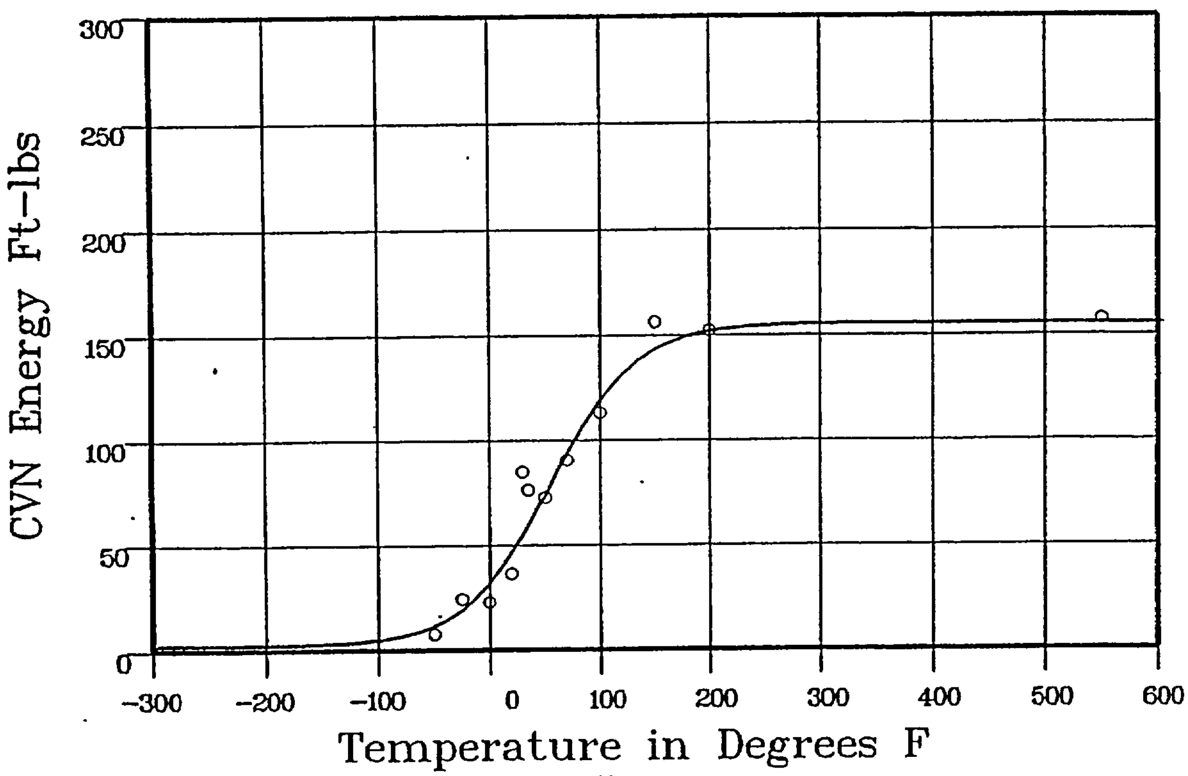
A = 78.59	B = 76.4	C = 76.75	T0 = 50.62

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

Upper Shelf Energy: 155 Fixed Temp. at 30 ft-lbs: -7 Temp. at 50 ft-lbs: 20.4 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: W-97 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: W-97 Material: PLATE SA533B1 Ori.: LT Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-50	7.5	12.54	-5.04
-25	24	20.89	3.1
0	22.5	34.43	-11.93
20	36	49.63	-13.63
30	84.5	58.54	25.95
35	76	63.25	12.74
50	72.5	77.97	-5.47
70	90	97.48	-7.48

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

CAPSULE 97 PLATE M-1004-2 (LONGITUDINAL)

Page 2

Material PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
100	113	121.93	-8.93
150	156	144.33	11.66
200	152	151.94	.05
550	157	154.99	2

SUM of RESIDUALS = 3.01

CAPSULE 97 PLATE M–1004–2 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:58:42 on 09-30-2002
Page 1
Coefficients of Curve 2

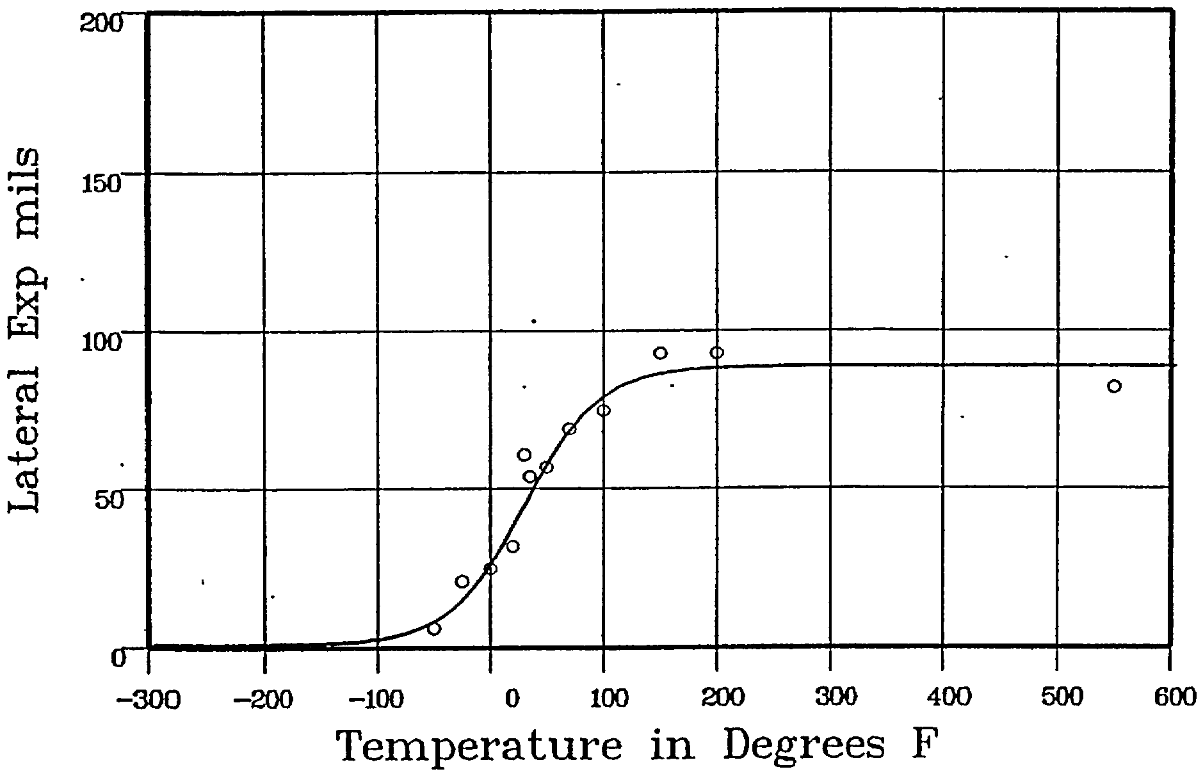
A = 44.9	B = 43.9	C = 66.46	T0 = 25.78

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

Upper Shelf L.E.: 88.81 Temperature at L.E. 35: 10.5 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: W-97 Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: W-97 Material: PLATE SA533B1 Ori.: LT Heat #: M-1004-2

Charpy V–Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-50	6	9.14	-3.14
-25	21	16.65	4.34
0	25	28.68	-3.68
20	32	41.09	-9.09
30	61	47.68	13.31
35	54	50.95	3.04
50	57	60.23	-3.23
70	69	70.45	-1.45

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

CAPSULE 97 PLATE M-1004-2 (LONGITUDINAL)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
100	75	80.31	-5.31
150	93	86.77	6.22
200	93	88.35	4.64
550	82	88.81	-6.81

SUM of RESIDUALS = -1.15

CAPSULE 97 PLATE M-1004-2 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:25:59 on 10-01-2002
Page 1
Coefficients of Curve 2

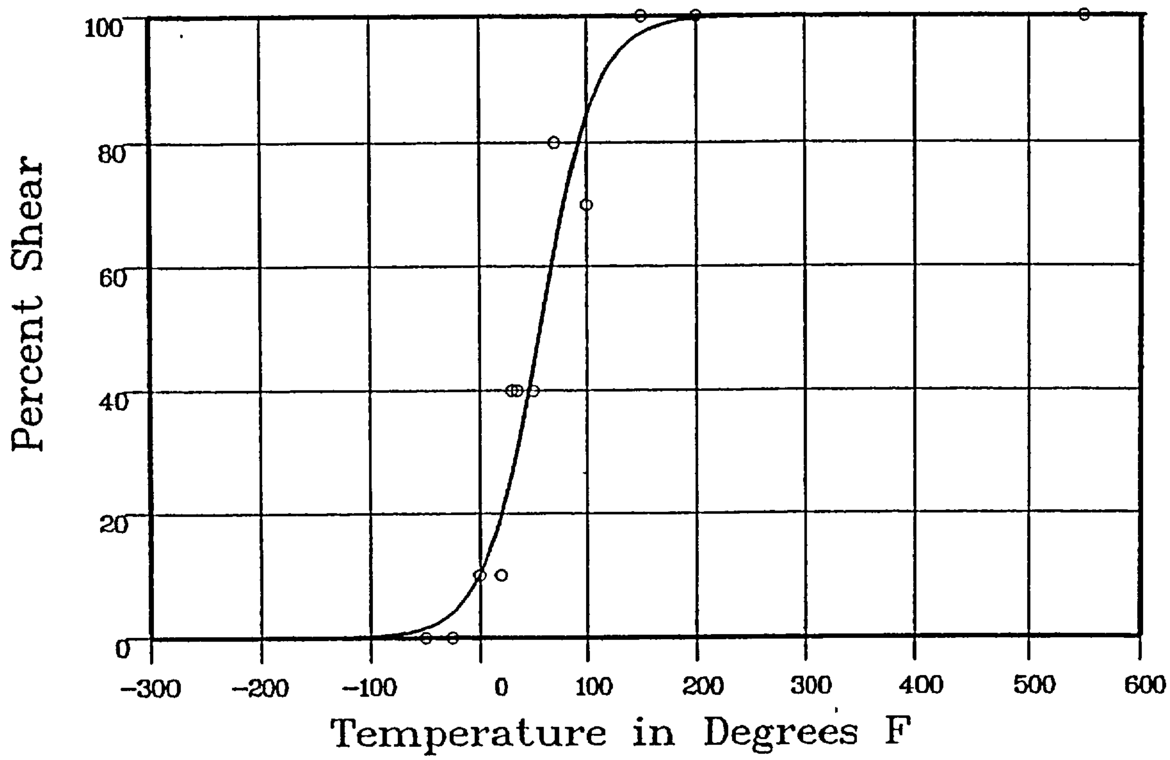
A = 50	B = 50	C = 51.69	T0 = 52.03

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

Temperature at 50% Shear: 52

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: W-97 Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: W-97 Material: PLATE SA533B1 Ori.: LT Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-50	0	1.89	-1.89
-25	0	4.83	-4.83
0	10	11.78	-1.78
20	10	22.45	-12.45
30	40	29.89	10.1
35	40	34.09	5.9
50	40	48.03	-8.03
70	80	66.71	13.28

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

CAPSULE 97 PLATE M-1004-2 (LONGITUDINAL)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: LT

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
100	70	86.48	-16.48
150	100	97.79	2.2
200	100	99.67	.32
550	100	100	0

SUM of RESIDUALS = -13.65

UNIRRADIATED PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:51:35 on 09-30-2002

Page 1

Coefficients of Curve 1

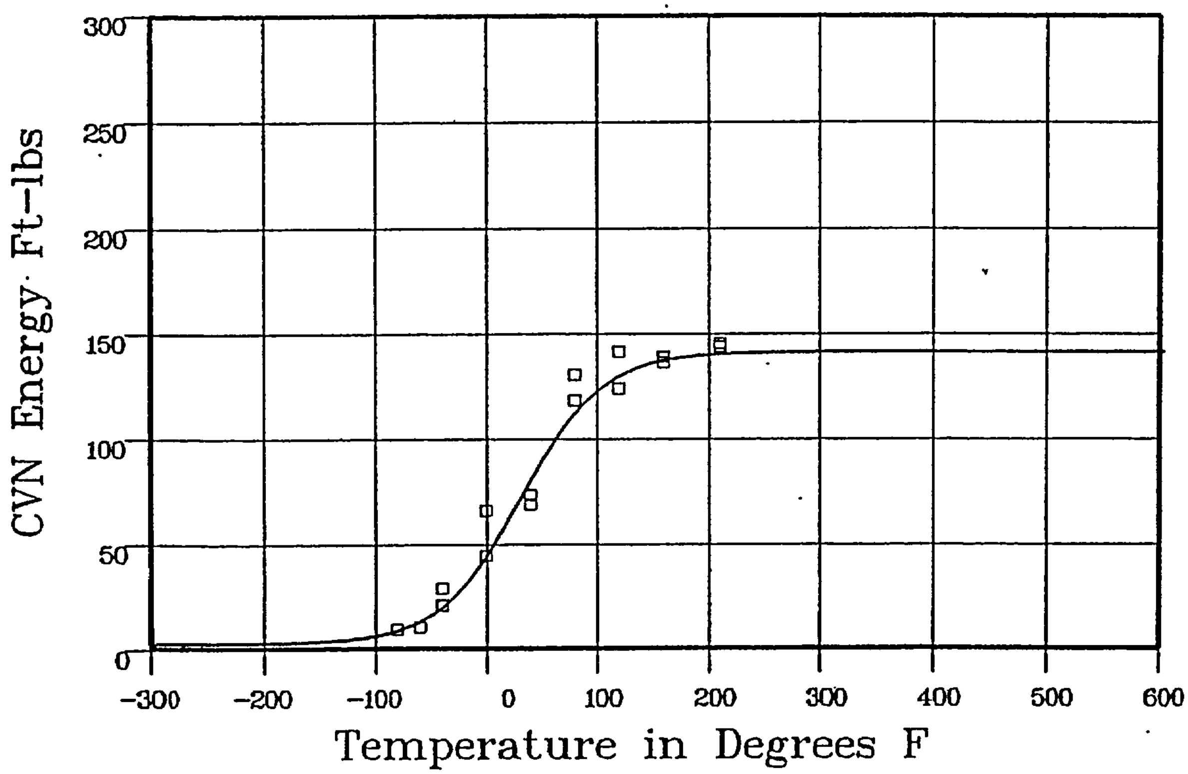
A = 71.59	B = 69.4	C = 73.82	T0 = 26.66

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

Upper Shelf Energy: 141 Fixed Temp. at 30 ft-lbs: -24.4 Temp. at 50 ft-lbs: 2.8 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: UNIRR Material: PLATE SA533B1 Ori: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-80	9	9.51	-.51
-60	10	14.3	-4.3
-40	28.5	21.78	6.71
-40	20.5	21.78	-1.28
0	65.5	47.57	17.92
0	44	47.57	-3.57
40	73	84	-11
40	68.5	84	-15.5
80	130	114.52	15.47

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

UNIRRADIATED PLATE M-1004-2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
80	118	114.52	3.47
120	141	130.74	10.25
120	123.5	130.74	-7.24
160	136	137.35	-1.35
160	138.5	137.35	1.14
210	145	140.04	4.95
210	143.5	140.04	3.45

SUM of RESIDUALS = 18.62

UNIRRADIATED PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:37:03 on 10-01-2002
Page 1
Coefficients of Curve 1

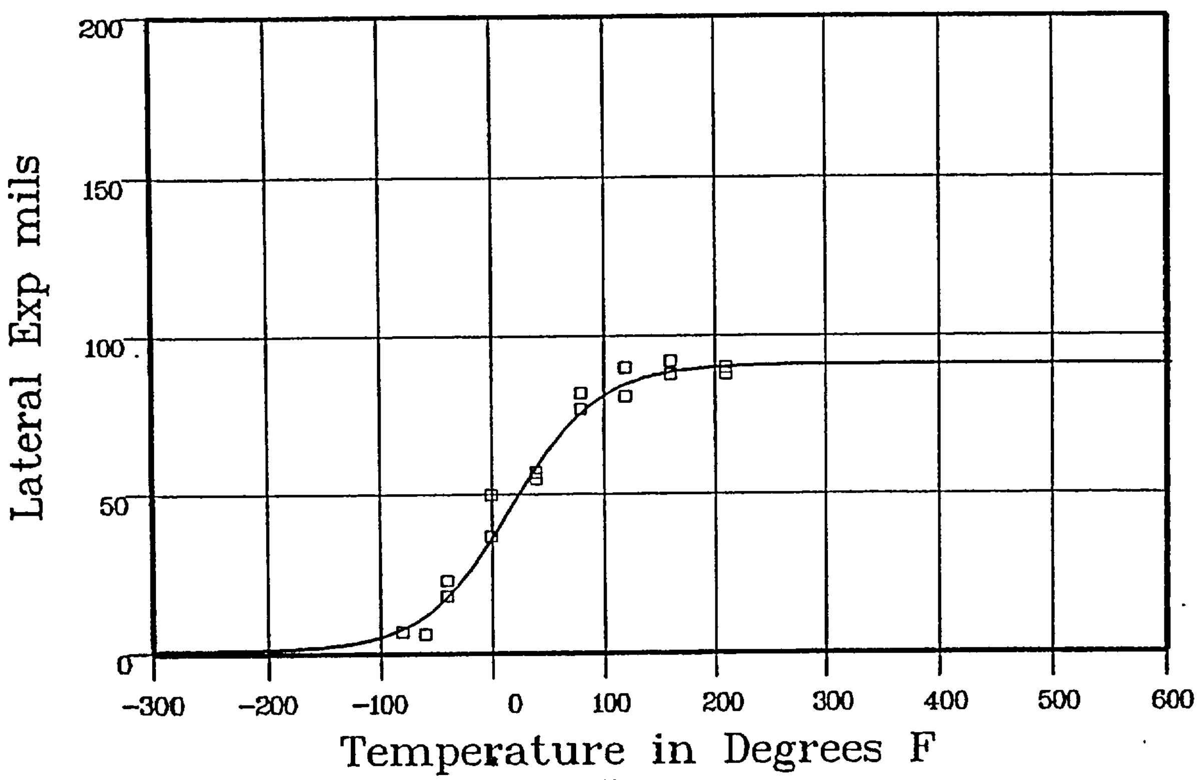
A = 45.95	B = 44.95	C = 78	T0 = 12.65

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

Upper Shelf L.E.: 90.9 Temperature at L.E. 35: -6.7 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: UNIRR Material: PLATE SA533B1 Ori: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-80	7	8.64	-1.64
-60	6	13.07	-7.07
-40	23	19.5	3.49
-40	18	19.5	-1.5
0	50	38.72	11.27
0	37	38.72	-1.72
40	57	61.09	-4.09
40	55	61.09	-6.09
80	77	77.32	-.32

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

UNIRRADIATED PLATE M-1004-2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
80	82	77.32	4.67
120	90	85.51	4.48
120	81	85.51	-4.51
160	88	88.89	-.89
160	92	88.89	3.1
210	90	90.33	-.33
210	88	90.33	-2.33

SUM of RESIDUALS = -3.53

UNIRRADIATED PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:41:58 on 10-01-2002

Page 1

Coefficients of Curve 1

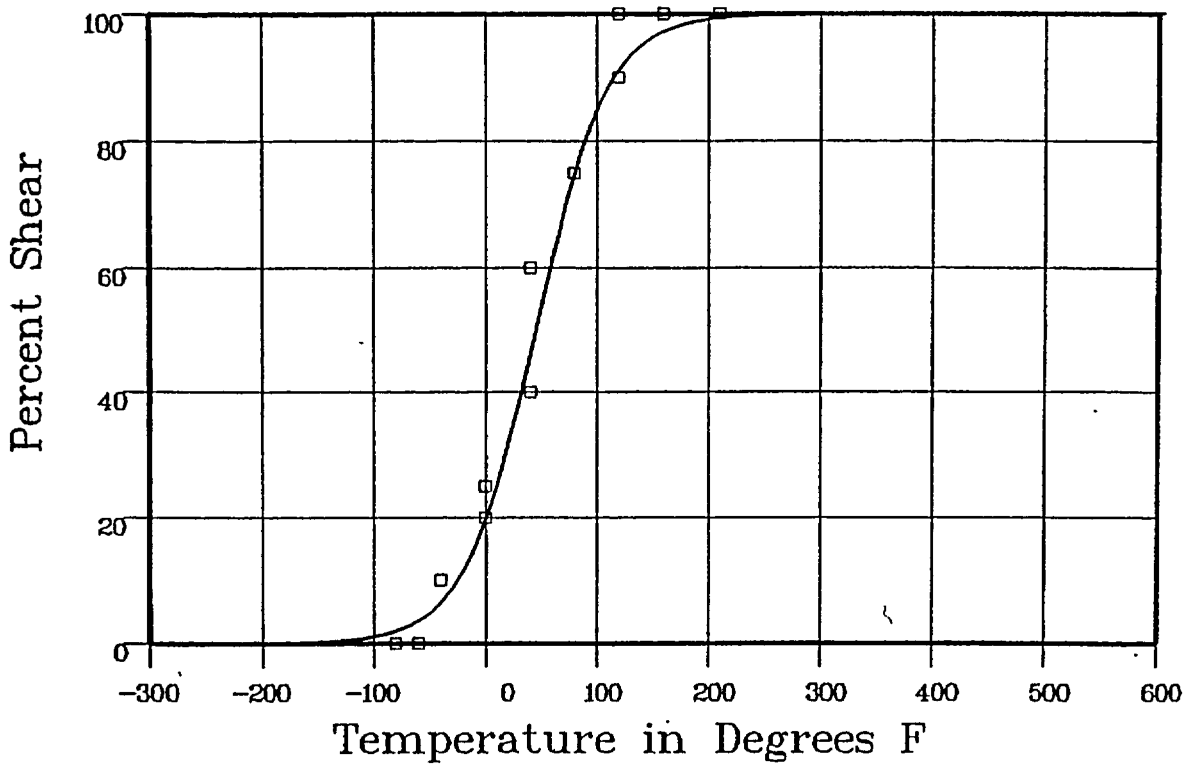
A = 50	B = 50	C = 64.03	T0 = 40.31

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

Temperature at 50% Shear: 40.3

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: UNIRR Material: PLATE SA533B1 Ori: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-80	0	2.28	-2.28
-60	0	4.17	-4.17
-40	10	7.52	2.47
-40	10	7.52	2.47
0	25	22.11	2.88
0	20	22.11	-2.11
40	60	49.75	10.24
40	40	49.75	-9.75
80	75	77.54	-2.54

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

UNIRRADIATED PLATE M-1004-2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
80	75	77.54	-2.54
120	100	92.33	7.66
120	90	92.33	-2.33
160	100	97.67	2.32
160	100	97.67	2.32
210	100	99.5	.49
210	100	99.5	.49

SUM of RESIDUALS = 5.62

CAPSULE 97 PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:51:35 on 09-30-2002
Page 1
Coefficients of Curve 2

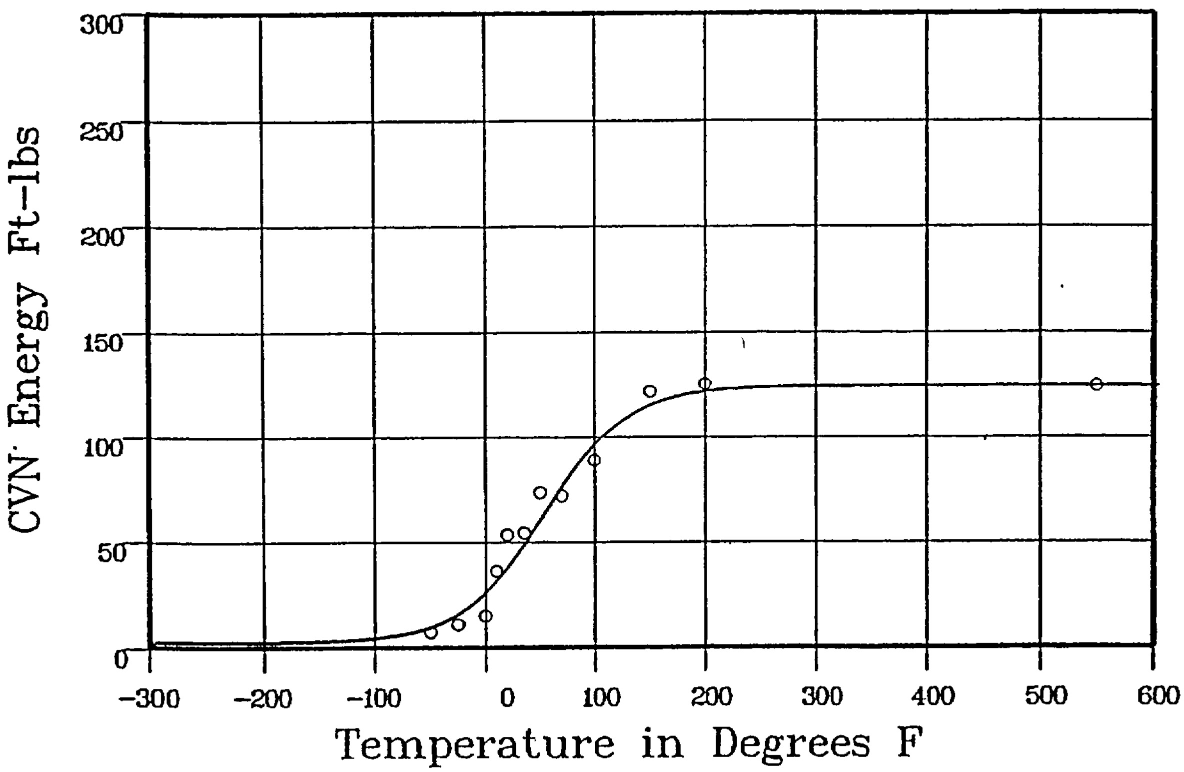
A = 63.09	B = 60.9	C = 75.89	T0 = 49.68

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

Upper Shelf Energy: 124 Fixed Temp at 30 ft-lbs: 3.4 Temp. at 50 ft-lbs: 33.1 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-97 Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: W-97 Material PLATE SA533B1 Ori: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-50	7	10.41	-3.41
-25	11	17.13	-6.13
0	15	28.09	-13.09
10	36	33.87	2.12
20	53	40.42	12.57
35	54	51.45	2.54
50	73	63.35	9.64
70	71.5	79.02	-7.52

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

CAPSULE 97 PLATE M-1004-2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
100	88.5	98.43	-9.93
150	121.5	115.91	5.58
200	125	121.72	3.27
550	124	123.99	0

SUM of RESIDUALS = -4.34

CAPSULE 97 PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:37:03 on 10-01-2002

Page 1

Coefficients of Curve 2

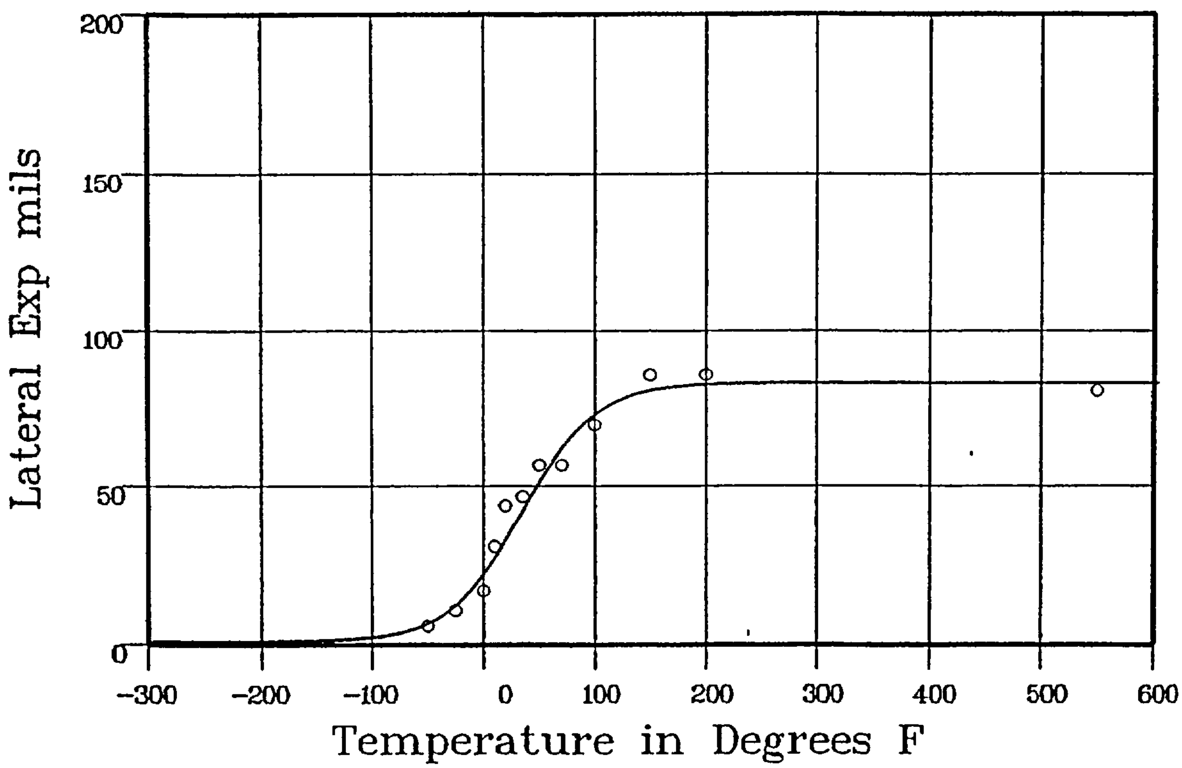
A = 42.2	B = 41.2	C = 65.77	T0 = 30.45

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

Upper Shelf L.E.: 83.4 Temperature at L.E. 35: 18.8 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-97 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: W-97 Material: PLATE SA533B1 Ori.: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-50	6	7.56	-1.56
-25	11	13.87	-2.87
0	17	24.37	-7.37
10	31	29.78	1.21
20	44	35.7	8.29
35	47	45.04	1.95
50	57	54.09	2.9
70	57	64.36	-7.36

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

CAPSULE 97 PLATE M-1004-2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed LE.	Differential
100	70	74.52	-4.52
150	86	81.28	4.71
200	86	82.92	3.07
550	81	83.4	-2.4

SUM of RESIDUALS = -3.96

CAPSULE 97 PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:41:58 on 10-01-2002

Page 1

Coefficients of Curve 2

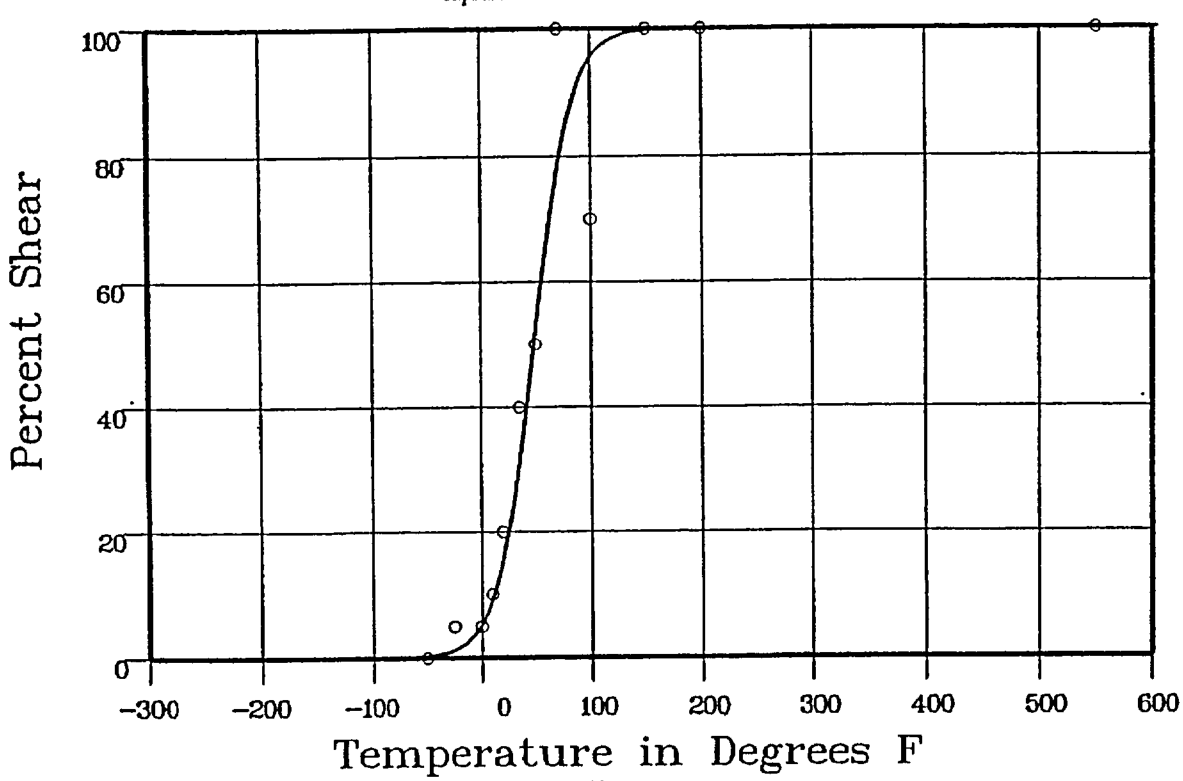
A = 50	B = 50	C = 33.47	T0 = 44.06

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

Temperature at 50% Shear: 44

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-97 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-97 Material: PLATE SA533B1 Ori: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-50	0	.36	-.36
-25	5	1.58	3.41
0	5	6.7	-1.7
10	10	11.55	-1.55
20	20	19.19	.8
35	40	36.78	3.21
50	50	58.77	-8.77
70	100	82.48	17.51

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

CAPSULE 97 PLATE M-1004-2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
100	70	96.58	-26.58
150	100	99.82	.17
200	100	99.99	0
550	100	100	0

SUM of RESIDUALS = -13.85

CAPSULE 263 PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:51:35 on 09-30-2002

Page 1

Coefficients of Curve 3

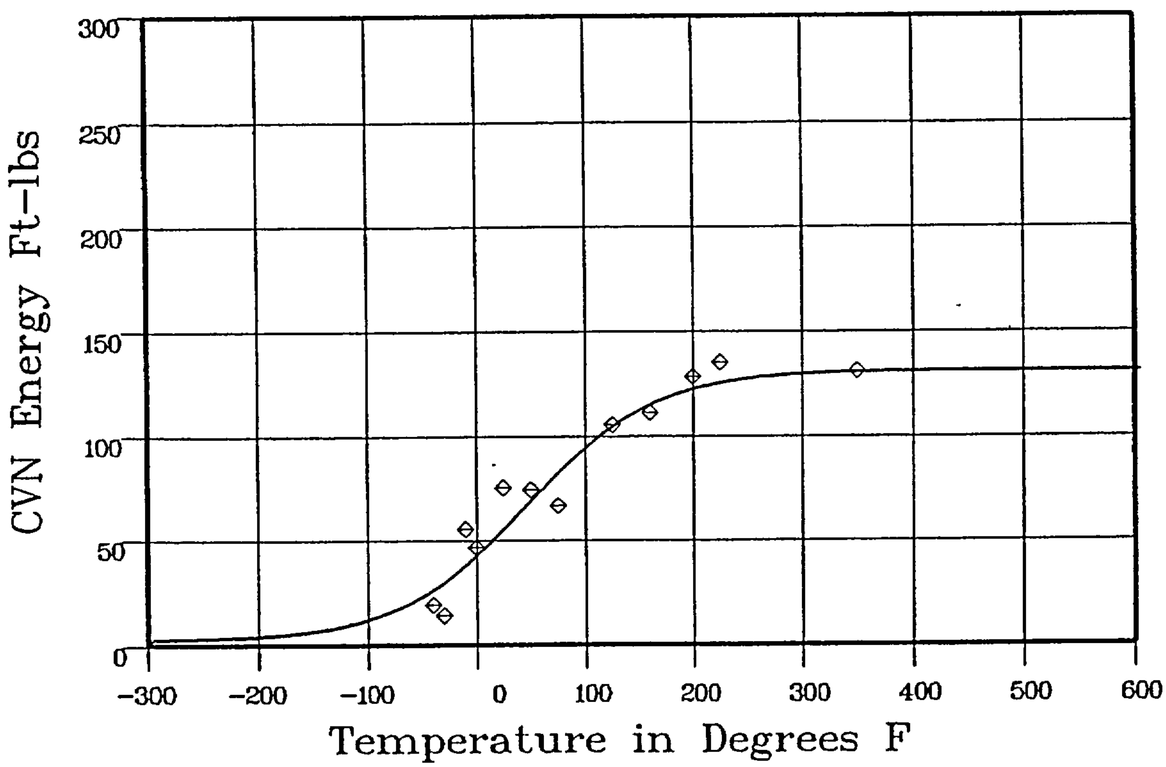
A = 66.59	B = 64.4	C = 116.94	T0 = 41.85

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

Upper Shelf Energy: 131 Fixed Temp. at 30 ft-lbs: -33.5 Temp. at 50 ft-lbs: 11 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-263 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-263 Material: PLATE SA533B1 Ori: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-40	19	27.67	-8.67
-30	14	31.35	-17.35
-10	55	39.77	15.22
0	46	44.48	1.51
25	75	57.37	17.62
50	74	71.07	2.92
75	66	84.37	-18.37
125	105	105.96	-.96

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

CAPSULE 263 PLATE M-1004-2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-263 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
160	111	115.92	-4.92
200	128	122.92	5.07
225	135	125.61	9.38
350	131	130.34	.65

SUM of RESIDUALS = 2.09

CAPSULE 263 PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:37:03 on 10-01-2002

Page 1

Coefficients of Curve 3

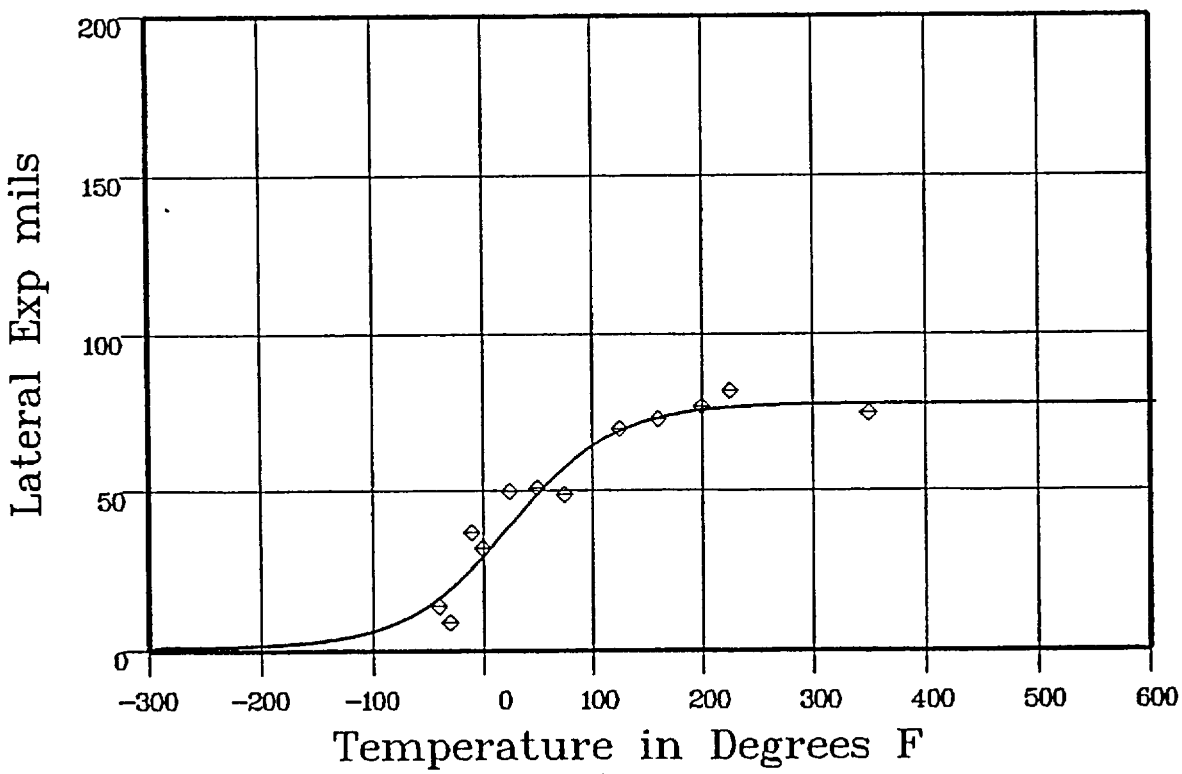
A = 39.43	B = 38.43	C = 95.57	T0 = 20.88

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

Upper Shelf L.E.: 77.86 Temperature at L.E. 35: 9.8 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-263 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: W-263 Material: PLATE SA533B1 Ori.: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-40	14	17.79	-3.79
-30	9	20.7	-11.7
-10	37	27.42	9.57
0	32	31.16	.83
25	50	41.08	8.91
50	51	50.79	.2
75	49	59.13	-10.13
125	70	70.05	-.05

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

CAPSULE 263 PLATE M–1004–2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-263 Total Fluence:

Charpy V–Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed LE.	Differential
160	73	73.9	-.9
200	77	76.09	.9
225	82	76.81	5.18
350	75	77.79	-2.79

SUM of RESIDUALS = -3.76

CAPSULE 263 PLATE M-1004-2 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 14:41:58 on 10-01-2002

Page 1

Coefficients of Curve 3

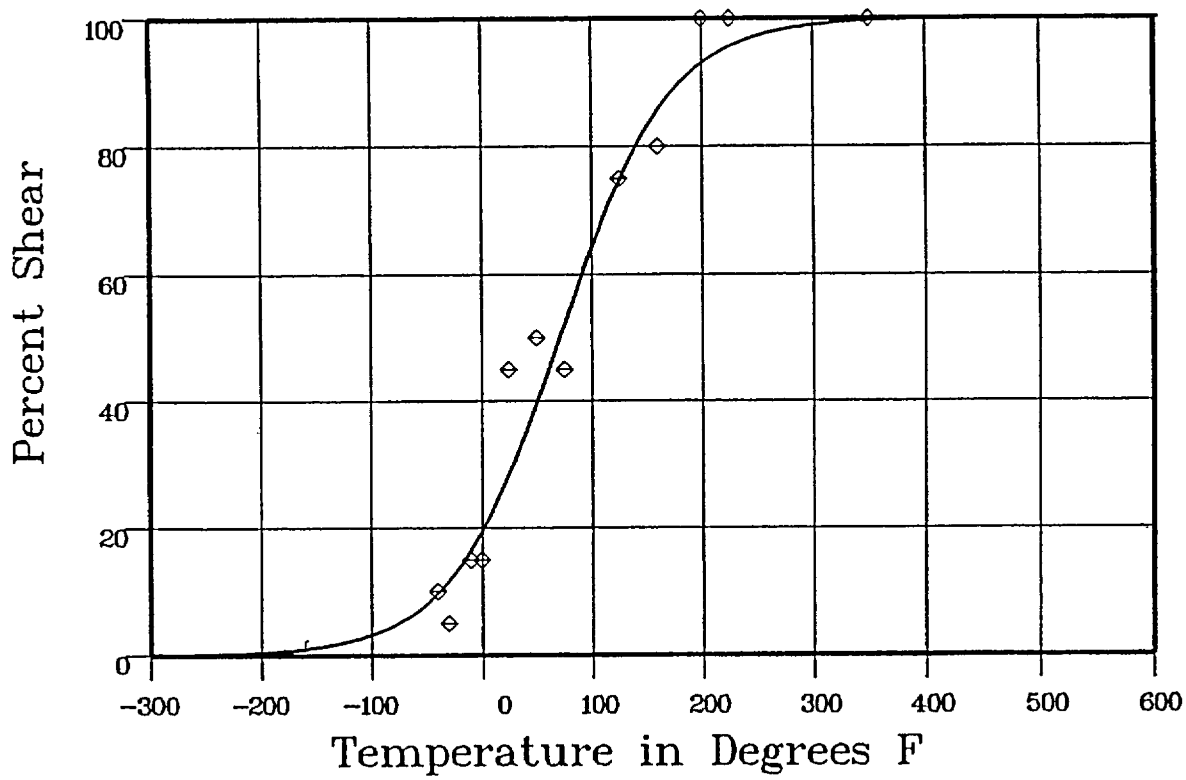
A = 50	B = 50	C = 99.72	T0 = 66.3

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

Temperature at 50% Shear: 66.3

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-263 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: W-263 Material: PLATE SA533B1 Ori.: TL Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-40	10	10.6	-.6
-30	5	12.65	-7.65
-10	15	17.79	-2.79
0	15	20.91	-5.91
25	45	30.39	14.6
50	50	41.89	8.1
75	45	54.34	-9.34
125	75	76.44	-1.44

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

CAPSULE 263 PLATE M–1004–2 (TRANSVERSE)

Page 2

Material: PLATE SA533B1 Heat Number: M-1004-2 Orientation: TL

Capsule: W-263 Total Fluence:

Charpy V–Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
160	80	86.75	-6.75
200	100	93.59	6.4
225	100	96.01	3.98
350	100	99.66	.33

SUM of RESIDUALS = -1.09

WELD METAL (UNIRRADIATED)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:13:21 on 10-01-2002

Page 1

Coefficients of Curve 1

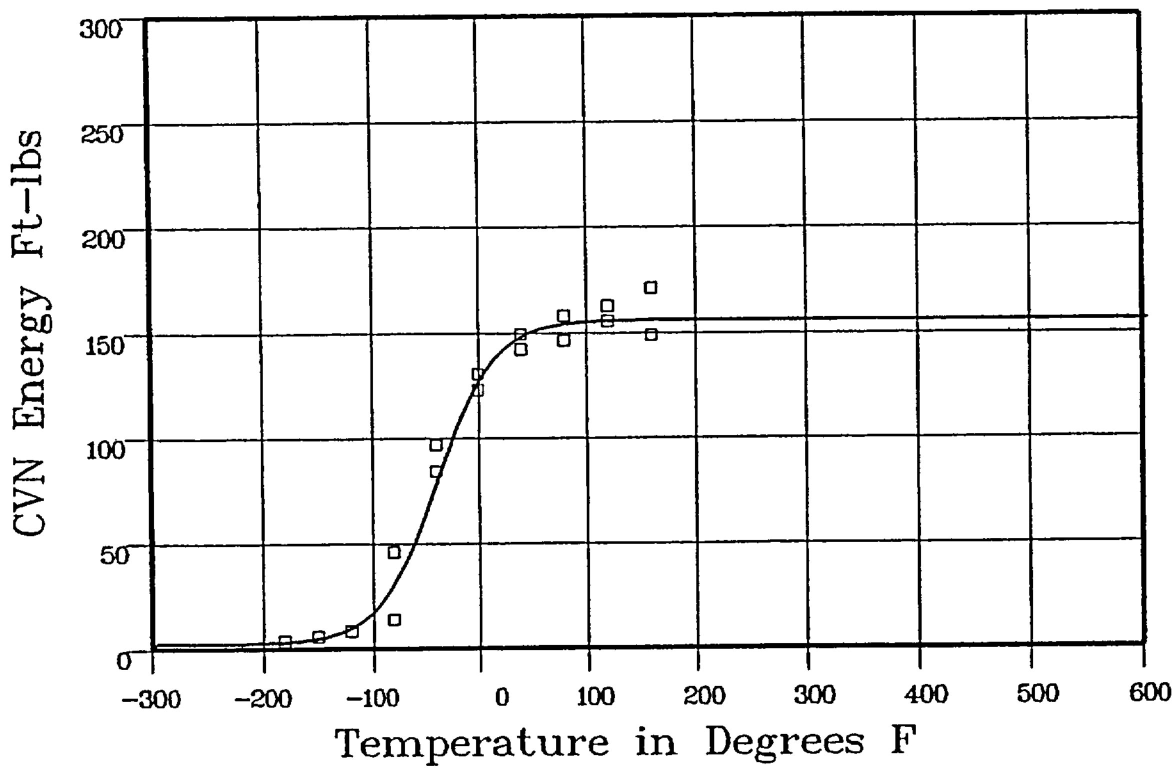
A = 79.09	B = 76.9	C = 54.25	T0 = -43.59

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

Upper Shelf Energy: 156 Fixed Temp. at 30 ft-lbs: -84.5 Temp. at 50 ft-lbs: -65.1 Lower Shelf Energy: 2.19 Fixed

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: UNIRR Material: WELD L 124/0091 Ori.: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-180	3.5	3.2	.29
-150	5.5	5.18	.31
-120	8	10.88	-2.88
-80	13.5	34.06	-20.56
-80	45.5	34.06	11.43
-40	83.5	84.18	-.68
-40	96.5	84.18	12.31
0	130.5	130.31	.18
0	122.5	130.31	-7.81

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

WELD METAL (UNIRRADIATED)

Page 2

Material WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
40	149	149.25	-.25
40	142	149.25	-7.25
80	146	154.4	-8.4
80	158	154.4	3.59
120	155.5	155.63	-.13
120	162.5	155.63	6.86
160	171	155.91	15.08
160	148.5	155.91	-7.41

SUM of RESIDUALS = -5.27

WELD METAL (UNIRRADIATED)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:17:12 on 10-01-2002
Page 1
Coefficients of Curve 1

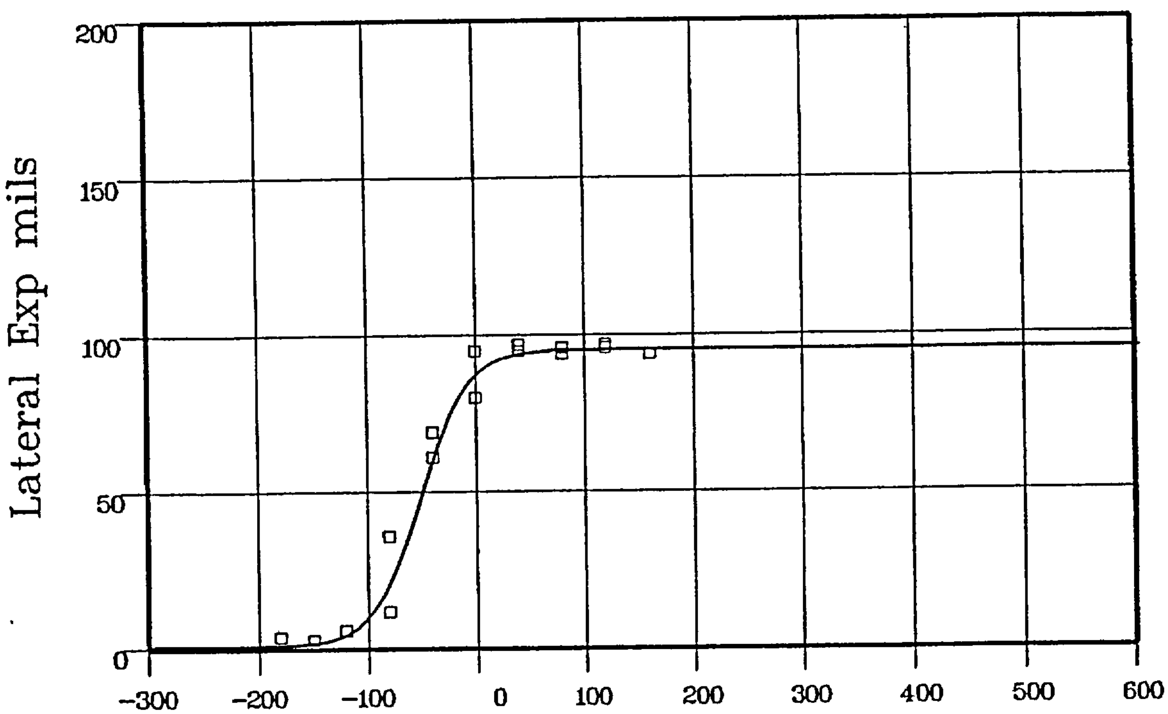
A = 48.24	B = 47.24	C = 43.52	T0 = -55.78

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

Upper Shelf L.E.: 95.49 Temperature at L.E. 35: -68.3 Lower Shelf L.E.: 1 Fixed

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: UNIRR Material: WELD L 124/0091 Ori: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-180	4	1.31	2.68
-150	3	2.22	.77
-120	6	5.69	.3
-80	12	24.37	-12.37
-80	36	24.37	11.62
-40	61	64.66	-3.66
-40	69	64.66	4.33
0	95	88.73	6.26
0	80	88.73	-8.73

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

WELD METAL (UNIRRADIATED)

Page 2

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
40	97	94.34	2.65
40	95	94.34	.65
80	94	95.3	-1.3
80	96	95.3	.69
120	96	95.46	.53
120	97	95.46	1.53
160	94	95.48	-1.48
160	94	95.48	-1.48
			SUM of RESIDUALS = 3.02

WELD METAL (UNIRRADIATED)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:21:23 on 10-01-2002

Page 1

Coefficients of Curve 1

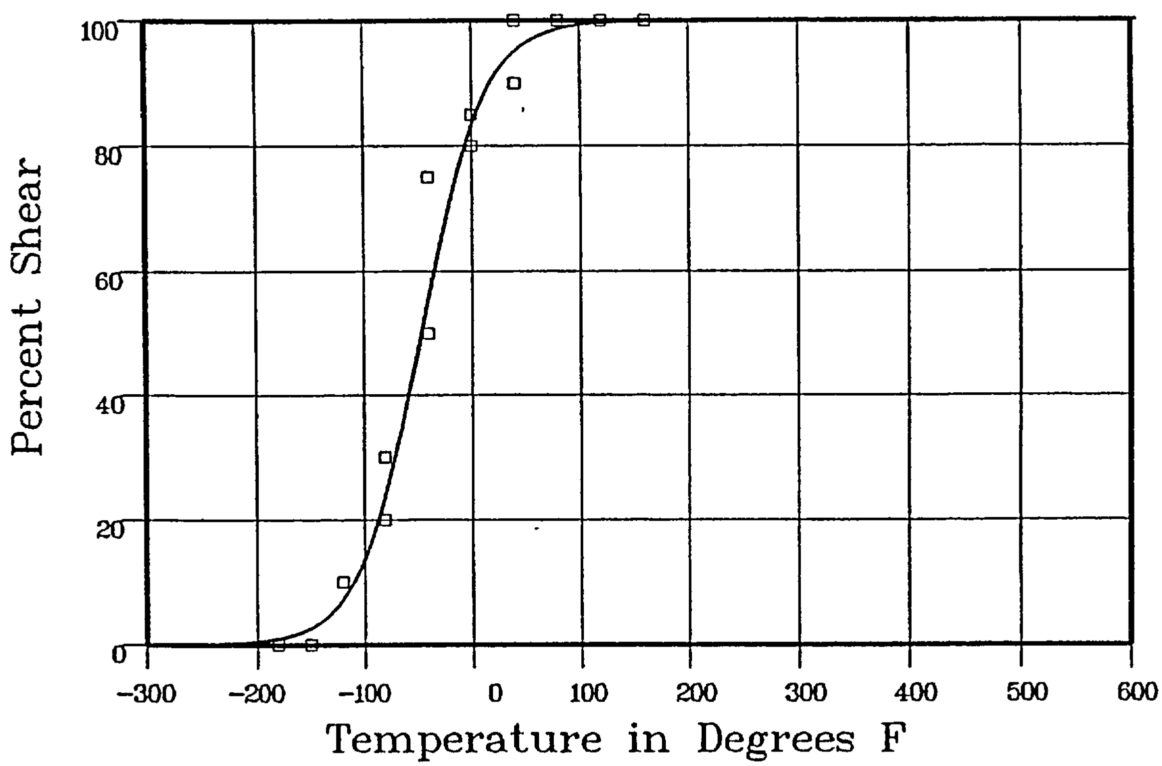
A = 50	B = 50	C = 57.69	T0 = -51.09

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

Temperature at 50% Shear: -51

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: UNIRR Material: WELD L 124/0091 Ori.: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-180	0	1.13	-1.13
-150	0	3.14	-3.14
-120	10	8.4	1.59
-80	20	26.85	-6.85
-80	30	26.85	3.14
-40	50	59.49	-9.49
-40	75	59.49	15.5
0	85	85.46	-.46
0	80	85.46	-5.46

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

WELD METAL (UNIRRADIATED)

Page 2

Material WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
40	100	95.92	4.07
40	90	95.92	-5.92
80	100	98.94	1.05
80	100	98.94	1.05
120	100	99.73	.26
120	100	99.73	.26
160	100	99.93	.06
160	100	99.93	.06

SUM of RESIDUALS = -5.38

WELD METAL (UNIRRADIATED)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:21:23 on 10-01-2002

Page 1

Coefficients of Curve 1

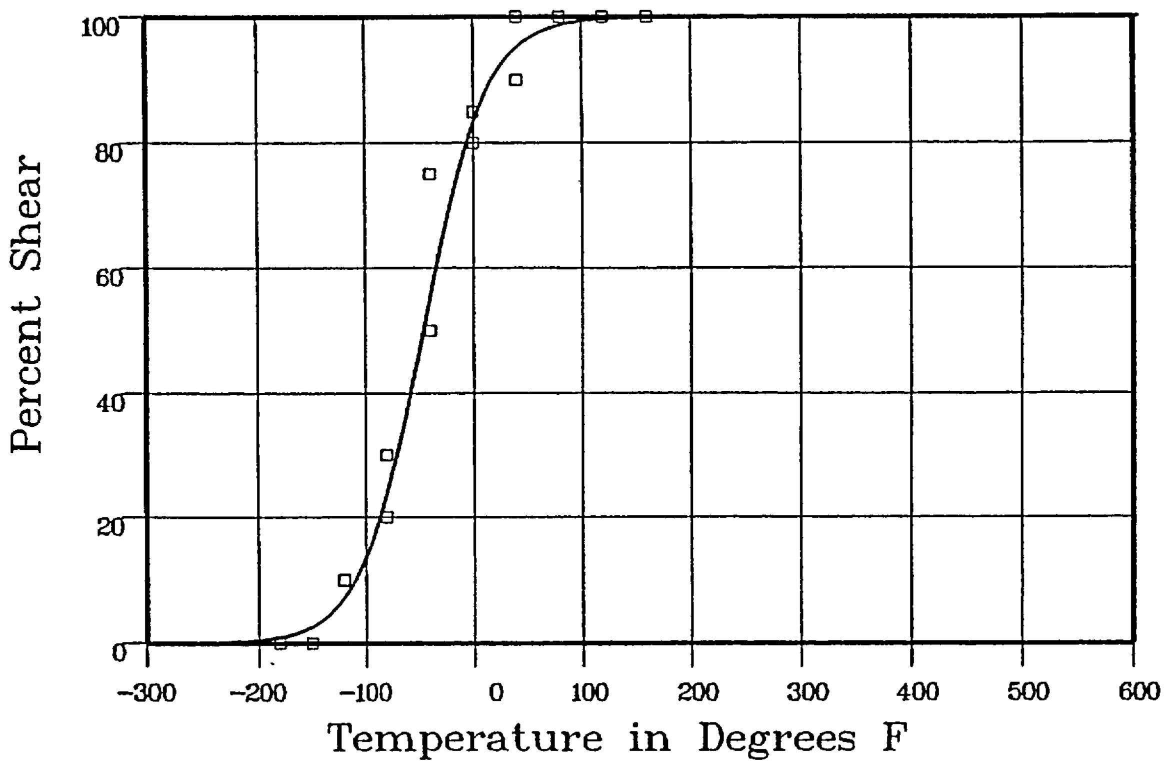
A = 50	B = 50	C = 57.69	T0 = -51.09

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

Temperature at 50% Shear: -51

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: UNIRR Material: WELD L 124/0091 Ori.: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-180	0	1.13	-1.13
-150	0	3.14	-3.14
-120	10	8.4	1.59
-80	20	26.85	-6.85
-80	30	26.85	3.14
-40	50	59.49	-9.49
-40	75	59.49	15.5
0	85	85.46	-.46
0	80	85.46	-5.46

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

WELD METAL (UNIRRADIATED)

Page 2

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
40	100	95.92	4.07
40	90	95.92	-5.92
80	100	98.94	1.05
80	100	98.94	1.05
120	100	99.73	.26
120	100	99.73	.26
160	100	99.93	.06
160	100	99.93	.06

SUM of RESIDUALS = -5.38

WELD METAL (CAPSULE 97)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:13:21 on 10-01-2002

Page 1

Coefficients of Curve 2

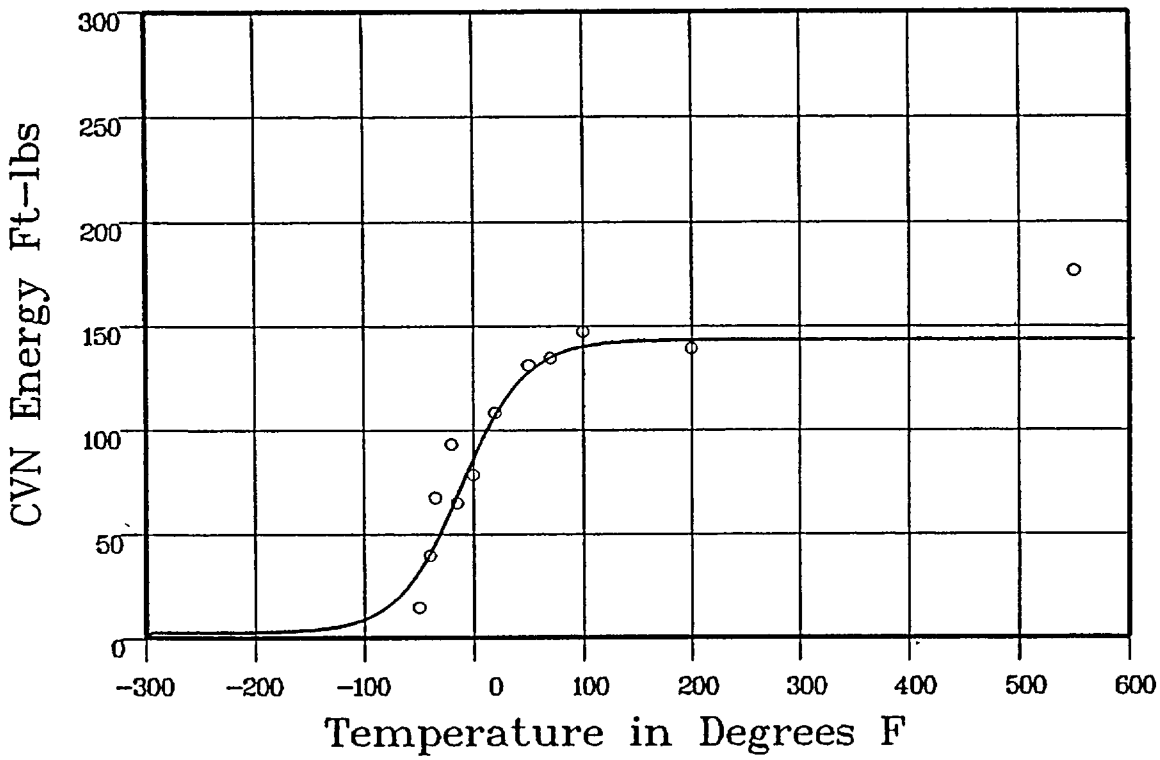
A = 72.59	B = 70.4	C = 58.32	T0 = -15.46

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

Upper Shelf Energy: 143 Fixed Temp. at 30 ft-lbs: -56.3 Temp. at 50 ft-lbs: -34.8 Lower Shelf Energy: 2.19 Fixed

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-97 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-97 Material: WELD L 124/0091 Ori: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-50	14.5	35.19	-20.69
-40	39.5	44.62	-5.12
-35	67	49.86	17.13
-20	93	67.14	25.85
-15	64.5	73.16	-8.66
0	78	90.84	-12.84
20	108	110.81	-2.81
50	131	129.51	1.48

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

WELD METAL (CAPSULE 97)

Page 2

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
70	134.5	135.86	-1.36
100	147	140.36	6.63
200	139	142.91	-3.91
550	176.5	143	33.5

SUM of RESIDUALS = 29.19

WELD METAL (CAPSULE 97)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:17:12 on 10-01-2002
Page 1
Coefficients of Curve 2

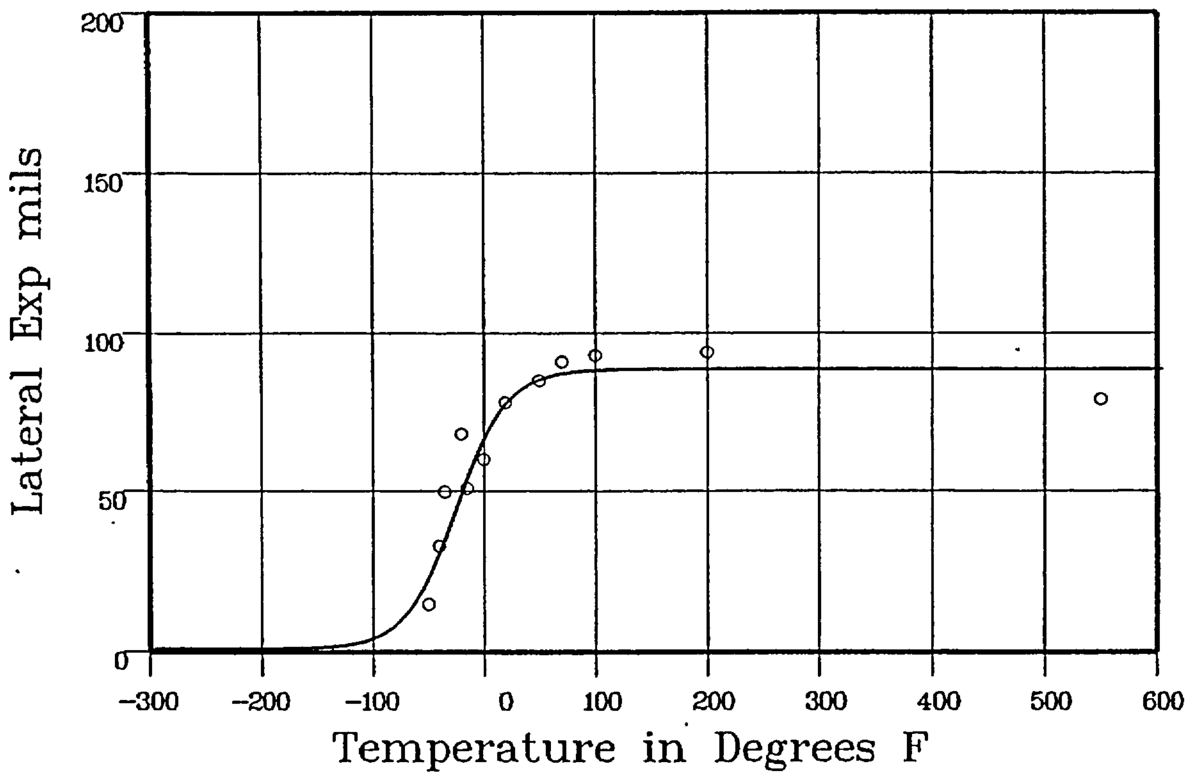
A = 44.83	B = 43.83	C = 46.28	T0 = -29.06

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

Upper Shelf LE: 88.67 Temperature at LE 35: -39.6 Lower Shelf LE: 1 Fixed

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-97 Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: W-97 Material: WELD L 124/0091 Ori: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed LE	Differential
-50	15	26.25	-11.25
-40	33	34.66	-1.66
-35	50	39.24	10.75
-20	68	53.31	14.68
-15	51	57.76	-6.76
0	60	69.24	-9.24
20	78	79.28	-1.28
50	85	85.89	-.89

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

WELD METAL (CAPSULE 97)

Page 2

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
70	91	87.48	3.51
100	93	88.34	4.65
200	94	88.67	5.32
550	79	88.67	-9.67

SUM of RESIDUALS = -1.84

WELD METAL (CAPSULE 97)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:21:23 on 10-01-2002

Page 1

Coefficients of Curve 2

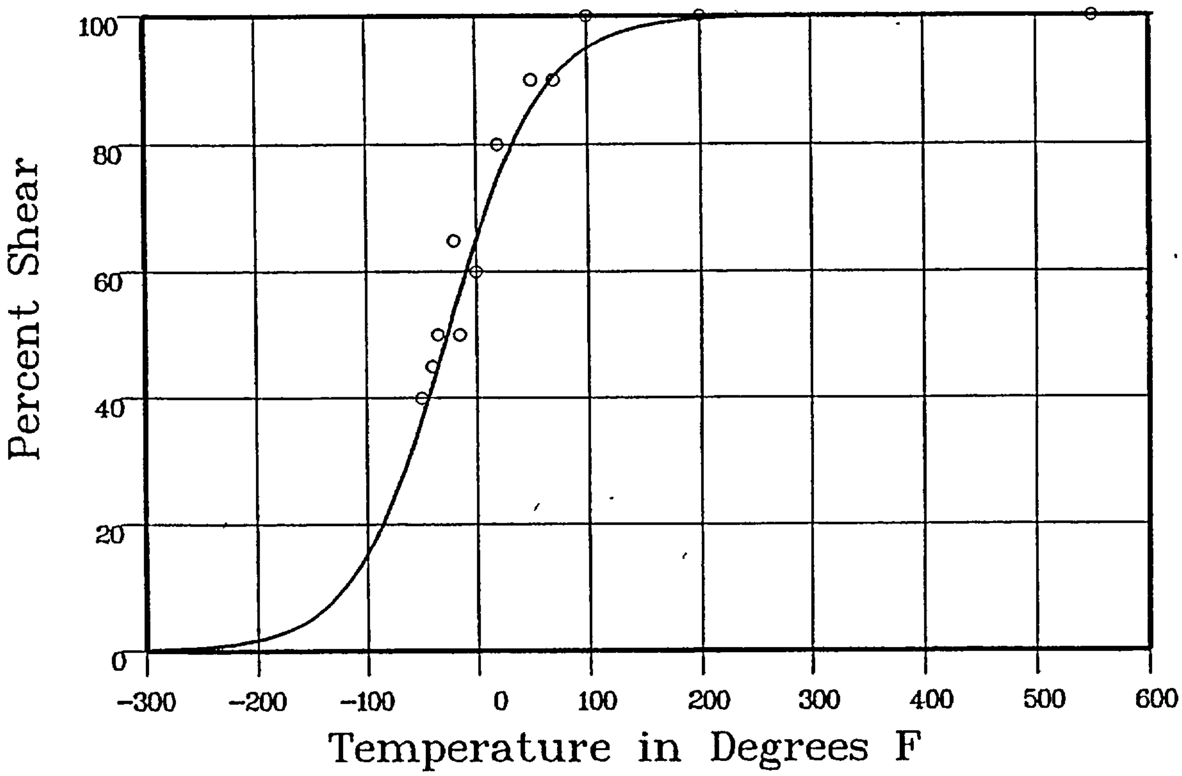
A = 50	B = 50	C = 85.34	T0 = -30.93

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

Temperature at 50% Shear: -30.9

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-97 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-97 Material: WELD L 124/0091 Ori: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-50	40	39.01	.98
-40	45	44.71	.28
-35	50	47.62	2.37
-20	65	56.37	8.62
-15	50	59.22	-9.22
0	60	67.36	-7.36
20	80	76.73	3.26
50	90	86.95	3.04

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

WELD METAL (CAPSULE 97)

Page 2

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
70	90	91.41	-1.41
100	100	95.55	4.44
200	100	99.55	.44
550	100	99.99	0

SUM of RESIDUALS = 5.46

WELD METAL (CAPSULE 263)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:13:21 on 10-01-2002

Page 1

Coefficients of Curve 3

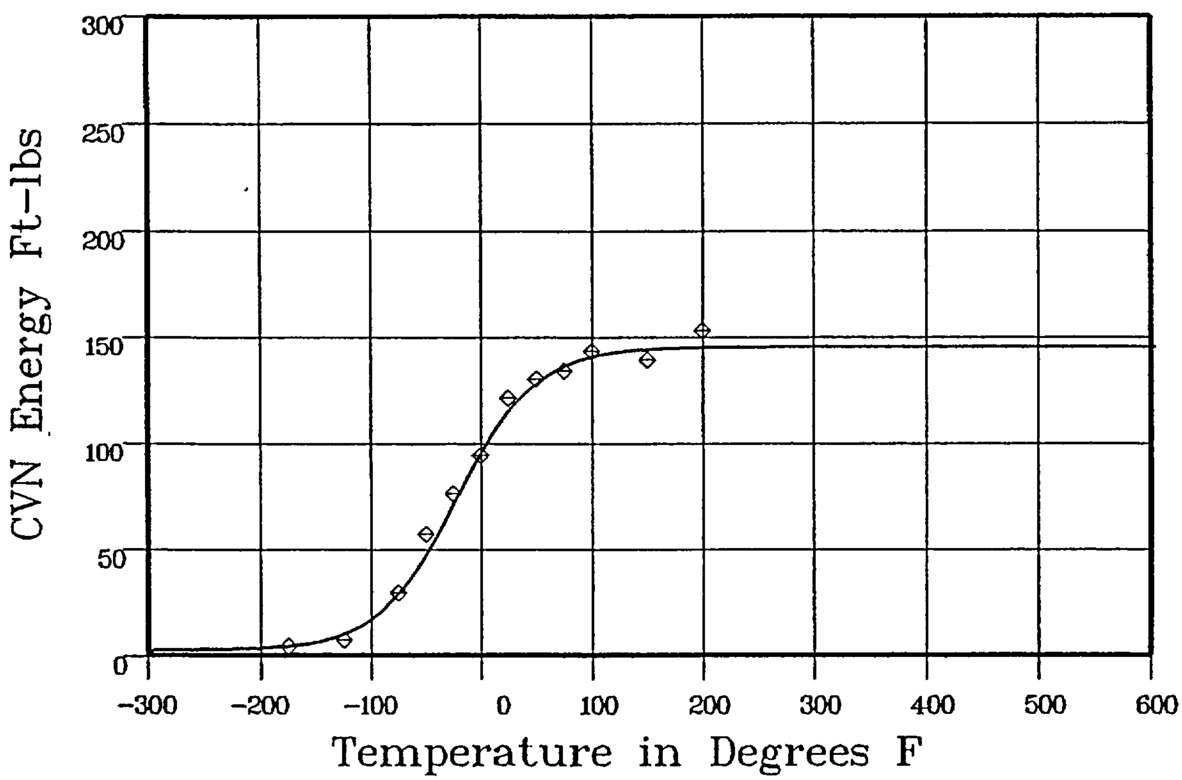
A = 73.59	B = 71.4	C = 71.83	T0 = -26.71

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

Upper Shelf Energy: 145 Fixed Temp. at 30 ft-lbs: -77.7 Temp. at 50 ft-lbs: -51.3 Lower Shelf Energy: 2.19 Fixed

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-263 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-263 Material: WELD L 124/0091 Ori: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-175	4	4.46	-.46
-125	7	10.89	-3.89
-75	29	31.73	-2.73
-50	57	51.23	5.76
-25	76	75.3	.69
0	94	98.99	-4.99
25	121	117.64	3.35
50	130	129.91	.08

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

WELD METAL (CAPSULE 263)

Page 2

Material WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-263 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
75	134	137.05	-3.05
100	143	140.92	2.07
150	139	143.96	-4.96
200	153	144.74	8.25

SUM of RESIDUALS = .11

WELD METAL (CAPSULE 263)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:17:12 on 10-01-2002
Page 1
Coefficients of Curve 3

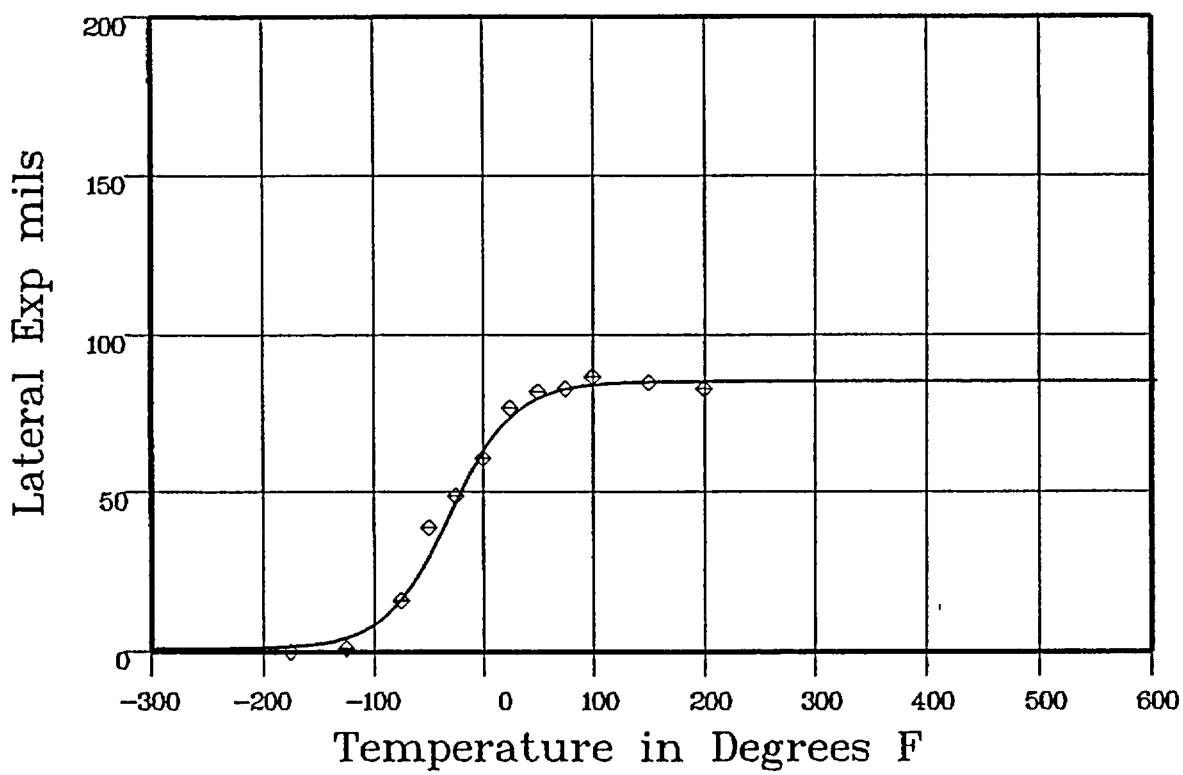
A = 43.17	B = 42.17	C = 59.76	T0 = -35.06

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

Upper Shelf L.E.: 85.35 Temperature at L.E. 35: -46.8 Lower Shelf L.E.: 1 Fixed

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-263 Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: W-263 Material: WELD L 124/0091 Ori: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-175	0	1.77	-1.77
-125	1	4.96	-3.96
-75	16	18.55	-2.55
-50	39	32.85	6.14
-25	49	50.21	-1.21
0	61	65.42	-4.42
25	77	75.39	1.6
50	82	80.72	1.27

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

WELD METAL (CAPSULE 263)

Page 2

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-263 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed LE.	Differential
75	83	83.28	-.28
100	87	84.44	2.55
150	85	85.18	-.18
200	83	85.32	-2.32

SUM of RESIDUALS = -5.15

WELD METAL (CAPSULE 263)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:21:23 on 10-01-2002
Page 1
Coefficients of Curve 3

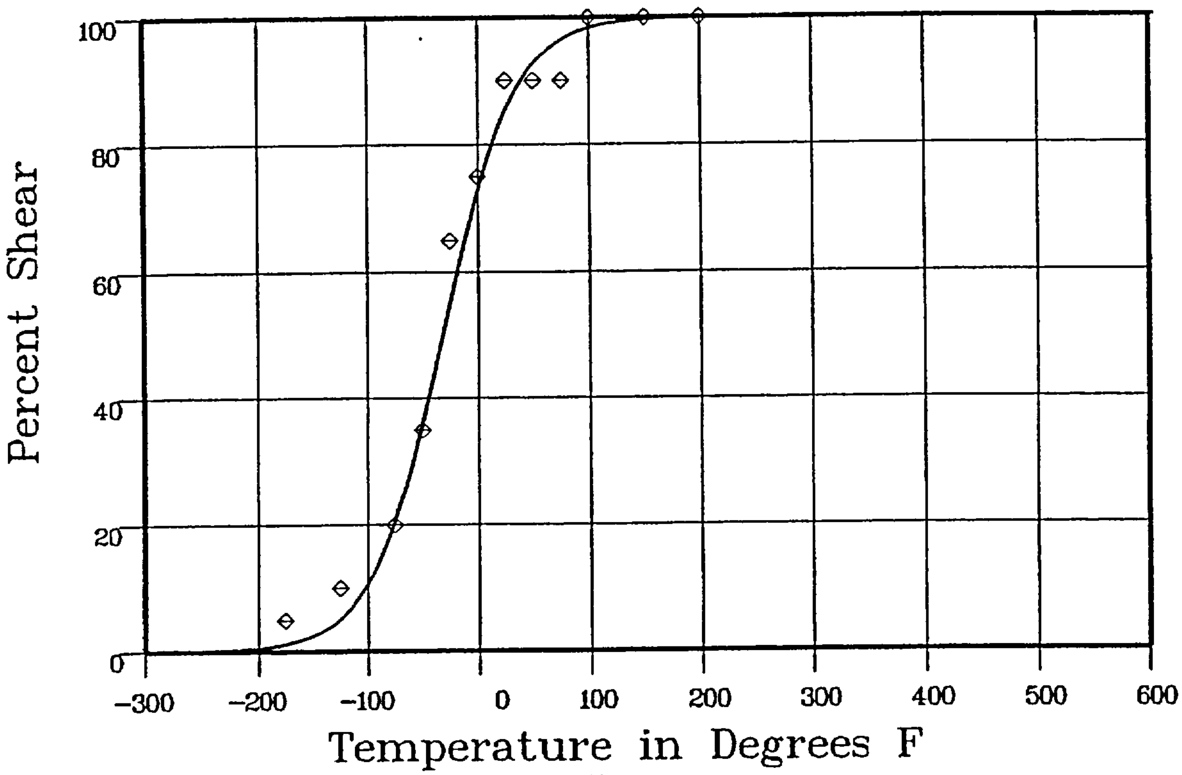
A = 50	B = 50	C = 64.17	T0 = -36.37

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

Temperature at 50% Shear: -36.3

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-263 Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: W-263 Material: WELD L 124/0091 Ori: Heat #: 88114/0145

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-175	5	1.31	3.68
-125	10	5.94	4.05
-75	20	23.08	-3.08
-50	35	39.54	-4.54
-25	65	58.77	6.22
0	75	75.64	-.64
25	90	87.13	2.86
50	90	93.65	-3.65

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

WELD METAL (CAPSULE 263)

Page 2

Material: WELD L 124/0091 Heat Number: 88114/0145 Orientation:

Capsule: W-263 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
75	90	96.98	-6.98
100	100	98.59	1.4
150	100	99.7	.29
200	100	99.93	.06

SUM of RESIDUALS = -.3

HEAT AFFECTED ZONE UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:35:22 on 10-01-2002

Page 1

Coefficients of Curve 1

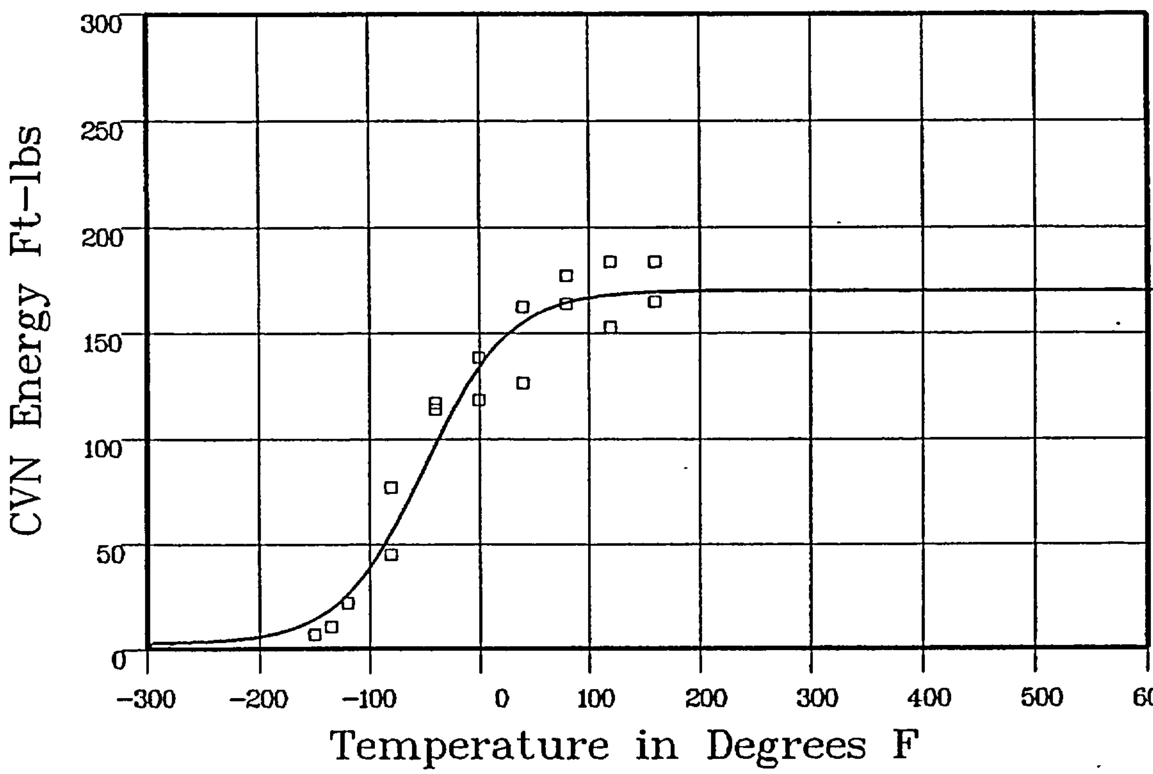
A = 86.09	B = 83.9	C = 77.59	T0 = -54.37

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

Upper Shelf Energy: 170 Fixed Temp. at 30 ft-lbs: -117 Temp. at 50 ft-lbs: -90 Lower Shelf Energy: 2.19 Fixed

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: UNIRR Material: HEAT AFFD ZONE SA533B1 Ori.: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-150	6.5	15.34	-8.84
-135	10.5	20.86	-10.36
-120	21.5	28.3	-6.8
-80	76.5	59.35	17.14
-80	44.5	59.35	-14.85
-40	113.5	101.46	12.03
-40	116.5	101.46	15.03
0	118	136.84	-18.84
0	138	136.84	1.15

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

HEAT AFFECTED ZONE UNIRRADIATED

Page 2

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
40	126	156.45	-30.45
40	162	156.45	5.54
80	177	164.9	12.09
80	163.5	164.9	-1.4
120	152.5	168.14	-15.64
120	183.5	168.14	15.35
160	164.5	169.33	-4.83
160	183.5	169.33	14.16

SUM of RESIDUALS = -19.54

HEAT AFFECTED ZONE UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:42:12 on 10-01-2002

Page 1

Coefficients of Curve 1

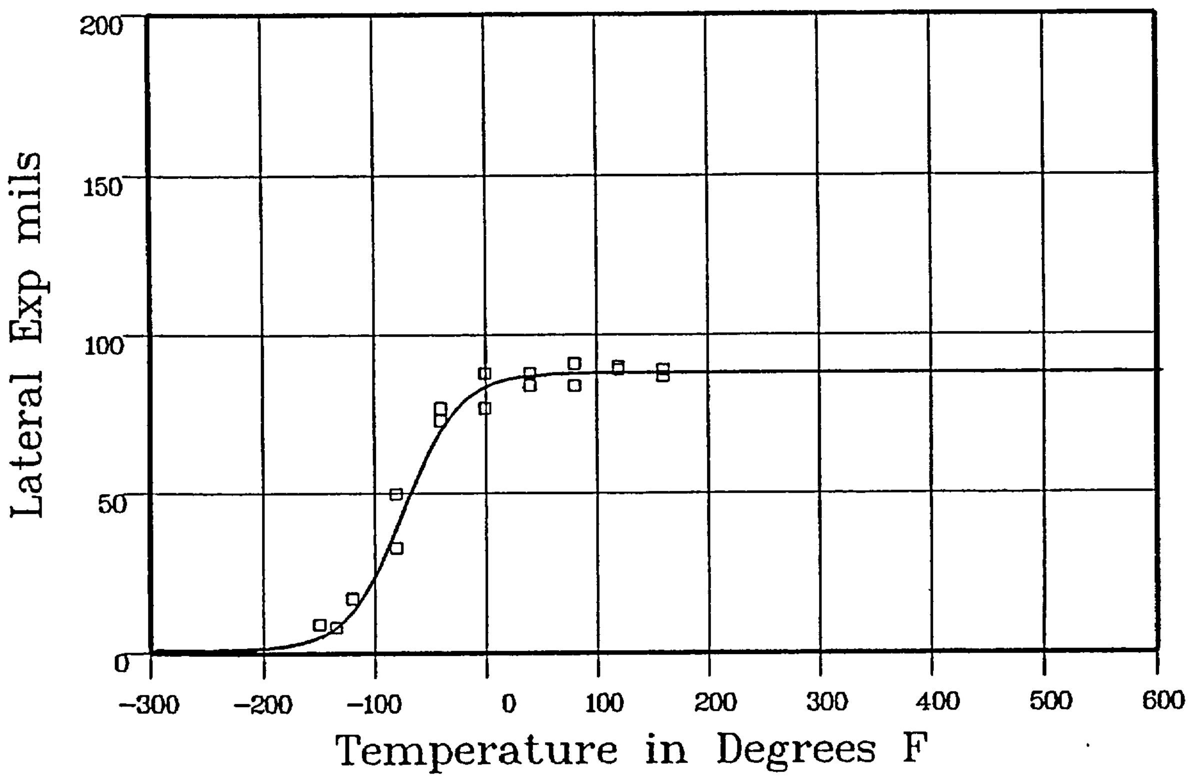
A = 44.56	B = 43.56	C = 50.48	T0 = -78.28

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

Upper Shelf L.E.: 88.13 Temperature at L.E. 35: -89.5 Lower Shelf L.E.: 1 Fixed

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: UNIRR Material: HEAT AFFD ZONE SA533B1 Ori.: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-150	9	5.8	3.19
-135	8	9.33	-1.33
-120	17	15	1.99
-80	50	43.08	6.91
-80	33	43.08	-10.08
-40	73	72.45	.54
-40	77	72.45	4.54
0	77	84.38	-7.38
0	88	84.38	3.61

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

HEAT AFFECTED ZONE UNIRRADIATED

Page 2

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
40	84	87.34	-3.34
40	88	87.34	.65
80	84	87.97	-3.97
80	91	87.97	3.02
120	90	88.1	1.89
120	89	88.1	.89
160	87	88.13	-1.13
160	89	88.13	.86
		SUM of RESIDUALS =	.91

HEAT AFFECTED ZONE UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:47:52 on 10-01-2002

Page 1

Coefficients of Curve 1

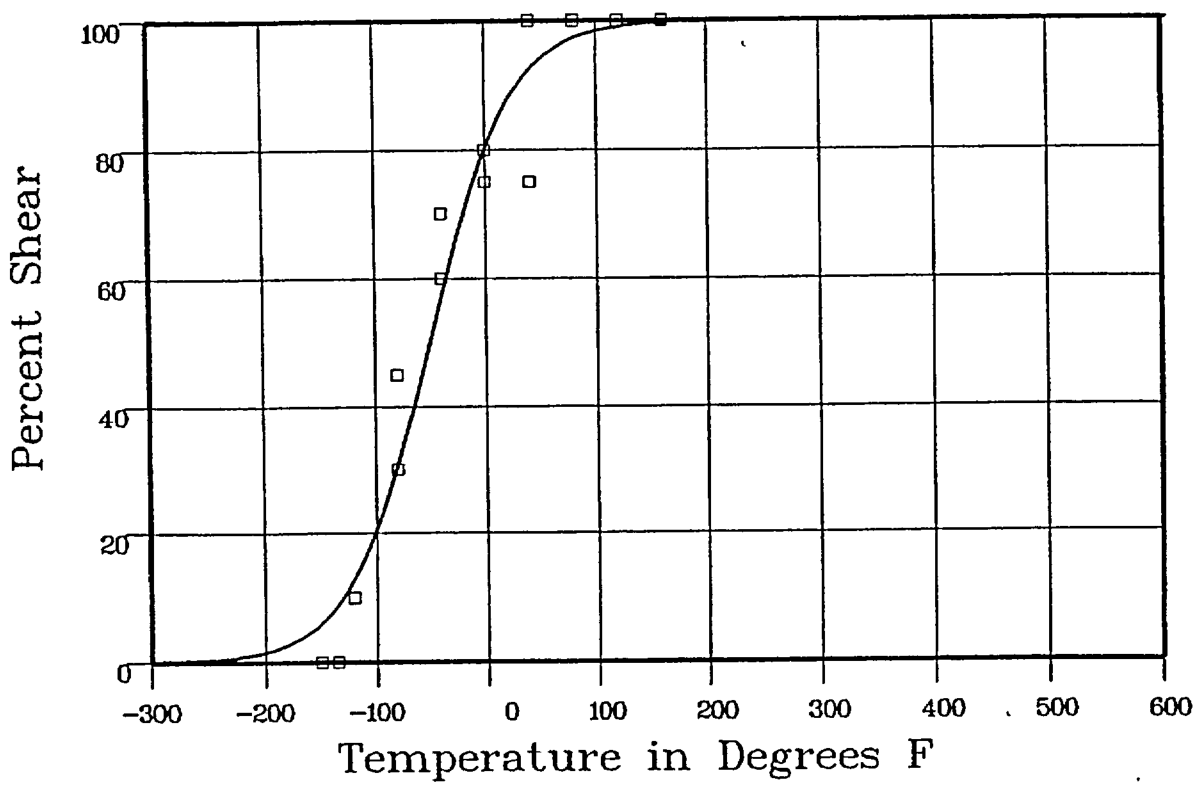
A = 50	B = 50	C = 72.35	T0 = -55.31

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

Temperature at 50% Shear: -55.3

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: UNIRR Material: HEAT AFFD ZONE SA533B1 Ori: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-150	0	6.8	-6.8
-135	0	9.94	-9.94
-120	10	14.32	-4.32
-80	45	33.57	11.42
-80	30	33.57	-3.57
-40	70	60.42	9.57
-40	60	60.42	-.42
0	75	82.18	-7.18
0	80	82.18	-2.18

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

HEAT AFFECTED ZONE UNIRRADIATED

Page 2

Material HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
40	75	93.3	-18.3
40	100	93.3	6.69
80	100	97.68	2.31
80	100	97.68	2.31
120	100	99.22	.77
120	100	99.22	.77
160	100	99.74	.25
160	100	99.74	.25

SUM of RESIDUALS = -18.34

HEAT AFFECTED ZONE CAPSULE 97

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:35:22 on 10-01-2002

Page 1

Coefficients of Curve 2

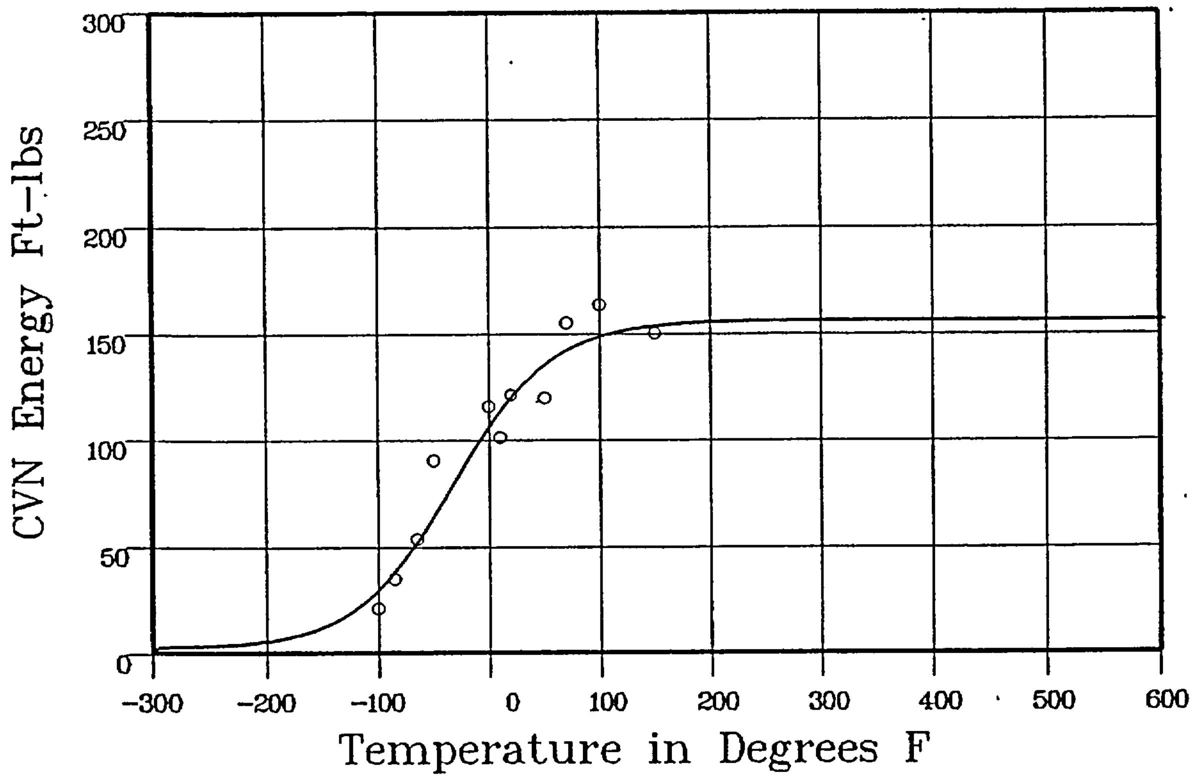
A = 79.09	B = 76.9	C = 88.57	T0 = -36.56

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

Upper Shelf Energy: 156 Fixed Temp at 30 ft-lbs: -103.4 Temp. at 50 ft-lbs: -71.8 Lower Shelf Energy: 2.19 Fixed

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-97 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-97 Material: HEAT AFFD ZONE SA533B1 Ori: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-100	21	31.84	-10.84
-85	35	40.79	-5.79
-65	53.5	55.22	-1.72
-50	90	67.52	22.47
0	115.5	109.15	6.34
10	101	116.17	-15.17
20	121	122.46	-1.46
50	119.5	136.91	-17.41

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

HEAT AFFECTED ZONE CAPSULE 97

Page 2

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
70	155	143.27	11.72
100	163.5	149.26	14.23
150	150	153.75	-3.75

SUM of RESIDUALS = -1.39

HEAT AFFECTED ZONE CAPSULE 97

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:42:12 on 10-01-2002

Page 1

Coefficients of Curve 2

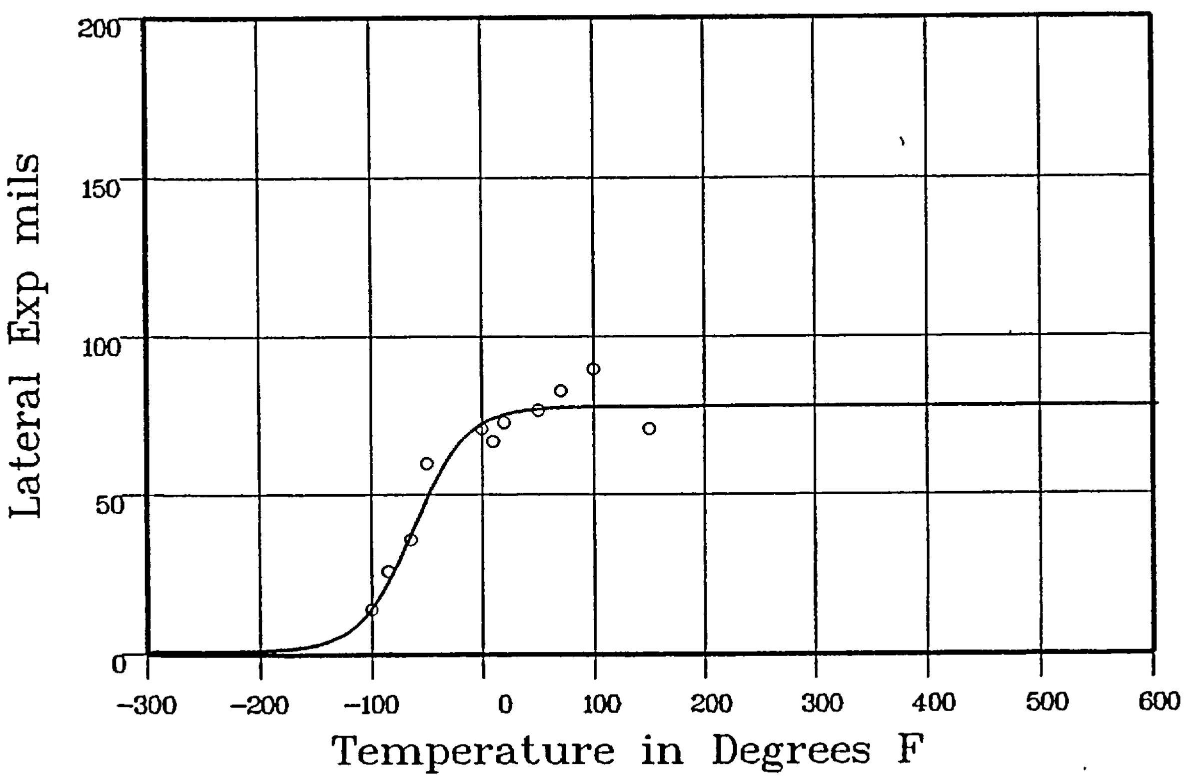
A = 39.53	B = 38.53	C = 48.11	T0 = -66.56

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

Upper Shelf L.E.: 78.06 Temperature at L.E. 35: -72.2 Lower Shelf L.E.: 1 Fixed

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-97 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: W-97 Material: HEAT AFFD ZONE SA533B1 Ori.: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-100	14	16.36	-2.36
-85	26	25.44	.55
-65	36	40.78	-4.78
-50	60	52.29	7.7
0	71	73.5	-2.5
10	67	74.99	-7.99
20	73	76	-3
50	77	77.45	-.45

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

HEAT AFFECTED ZONE CAPSULE 97

Page 2

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
70	83	77.79	5.2
100	90	77.98	12.01
150	71	78.05	-7.05

SUM of RESIDUALS = -2.68

HEAT AFFECTED ZONE CAPSULE 97

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:47:52 on 10-01-2002
Page 1

Coefficients of Curve 2

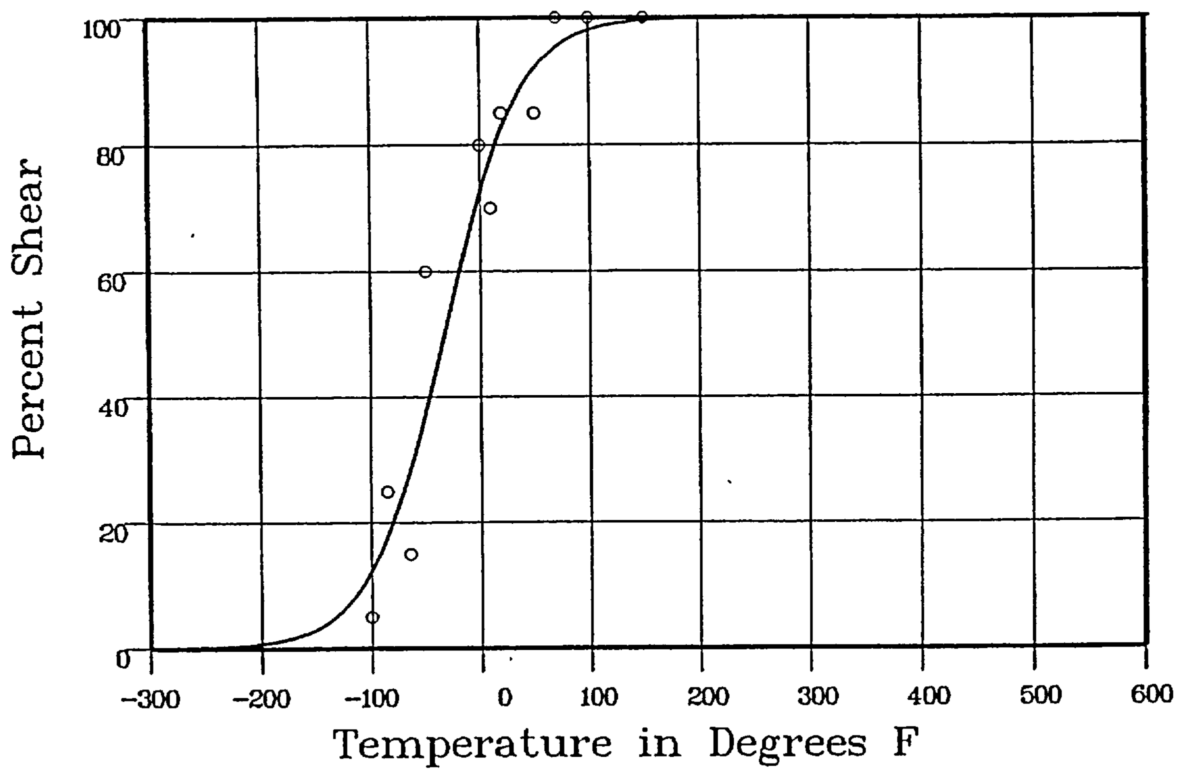
A = 50	B = 50	C = 67.96	T0 = -37.5

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

Temperature at 50% Shear: -37.5

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-97 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-97 Material: HEAT AFFD ZONE SA533B1 Ori: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-100	5	13.71	-8.71
-85	25	19.81	5.18
-65	15	30.8	-15.8
-50	60	40.9	19.09
0	80	75.09	4.9
10	70	80.18	-10.18
20	85	84.45	.54
50	85	92.92	-7.92

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

HEAT AFFECTED ZONE CAPSULE 97

Page 2

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-97 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
70	100	95.94	4.05
100	100	98.28	1.71
150	100	99.6	.39

SUM of RESIDUALS = -6.71

HEAT AFFECTED ZONE CAPSULE 263

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:35:22 on 10-01-2002

Page 1

Coefficients of Curve 3

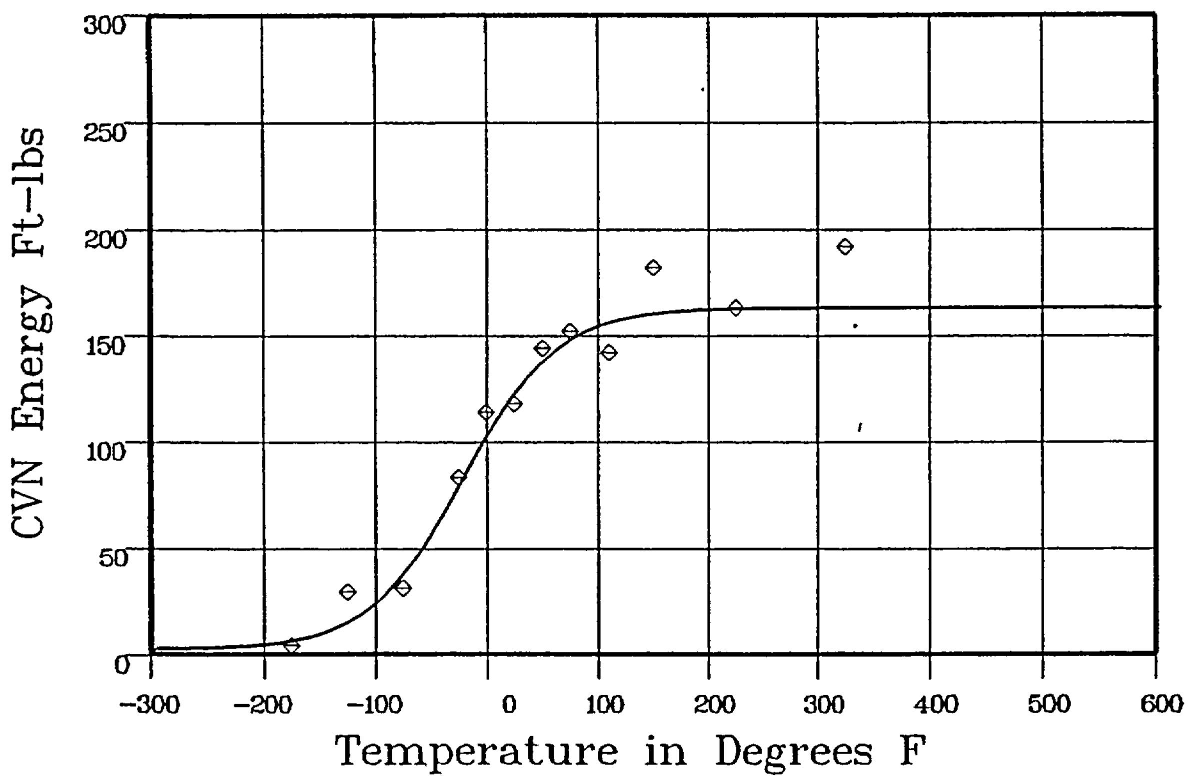
A = 82.59	B = 80.4	C = 84.55	T0 = -25.78

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

Upper Shelf Energy: 163 Fixed Temp. at 30 ft-lbs: -91.9 Temp. at 50 ft-lbs: -62.1 Lower Shelf Energy: 2.19 Fixed

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-263 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-263 Material: HEAT AFFD ZONE SA533B1 Ori: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-175	4	6.78	-2.78
-125	29	16.24	12.75
-75	31	40.45	-9.45
-25	83	83.34	-.34
0	114	106.38	7.61
25	118	125.81	-7.81
50	144	140.04	3.95
75	152	149.42	2.57

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

HEAT AFFECTED ZONE CAPSULE 263

Page 2

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-263 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
110	142	156.77	-14.77
150	182	160.52	21.47
225	163	162.57	.42
325	192	162.95	29.04

SUM of RESIDUALS = 42.69

HEAT AFFECTED ZONE CAPSULE 263

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:42:12 on 10-01-2002

Page 1

Coefficients of Curve 3

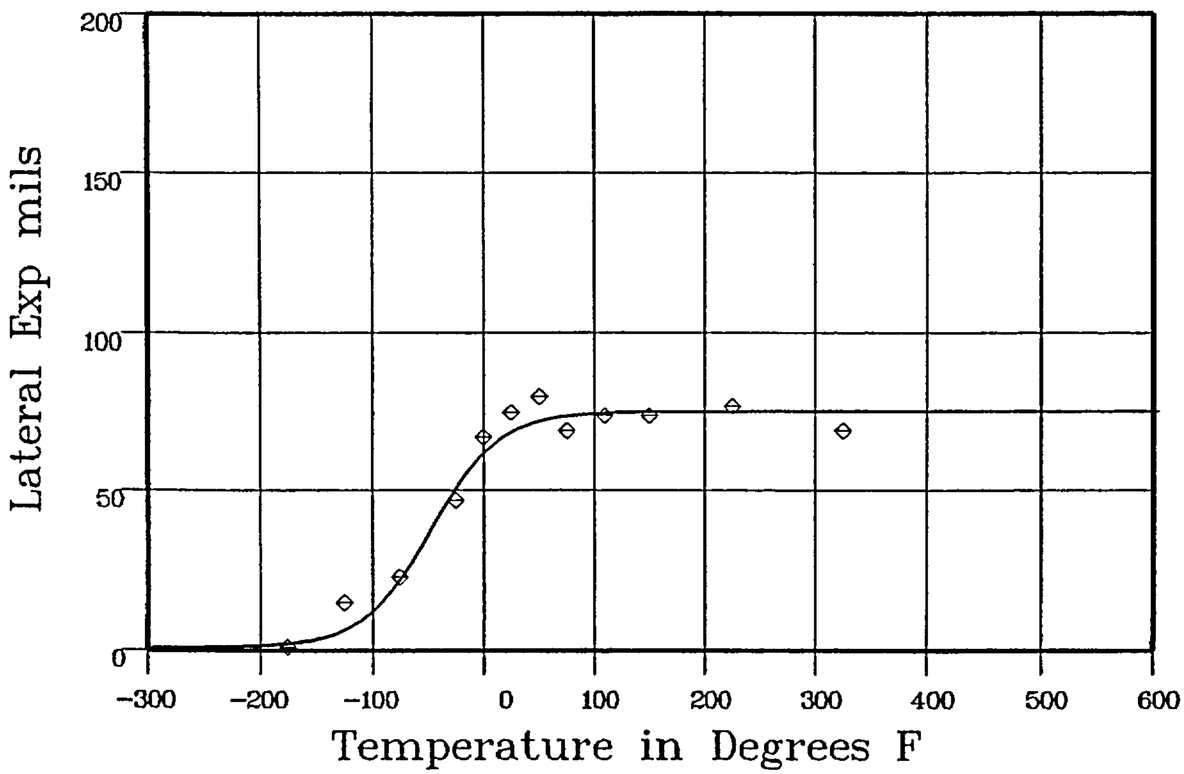
A = 38.09	B = 37.09	C = 61.59	T0 = -51.41

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

Upper Shelf L.E.: 75.19 Temperature at L.E. 35: -56.5 Lower Shelf L.E.: 1 Fixed

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-263 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: W-263 Material: HEAT AFFD ZONE SA533B1 Ori: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-175	1	2.31	-1.31
-125	15	7.23	7.76
-75	23	24.54	-1.54
-25	47	53.1	-6.1
0	67	63.43	3.56
25	75	69.47	5.52
50	80	72.53	7.46
75	69	73.99	-4.99

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

HEAT AFFECTED ZONE CAPSULE 263

Page 2

Material: HEAT AFFD ZONE SA533B1   Heat Number: M-1004-2   Orientation:

Capsule: W-263   Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed LE	Differential
110	74	74.8	-.8
150	74	75.08	-1.08
225	77	75.18	1.81
325	69	75.19	-6.19

SUM of RESIDUALS = 4.08

HEAT AFFECTED ZONE CAPSULE 263

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:47:52 on 10-01-2002
Page 1
Coefficients of Curve 3

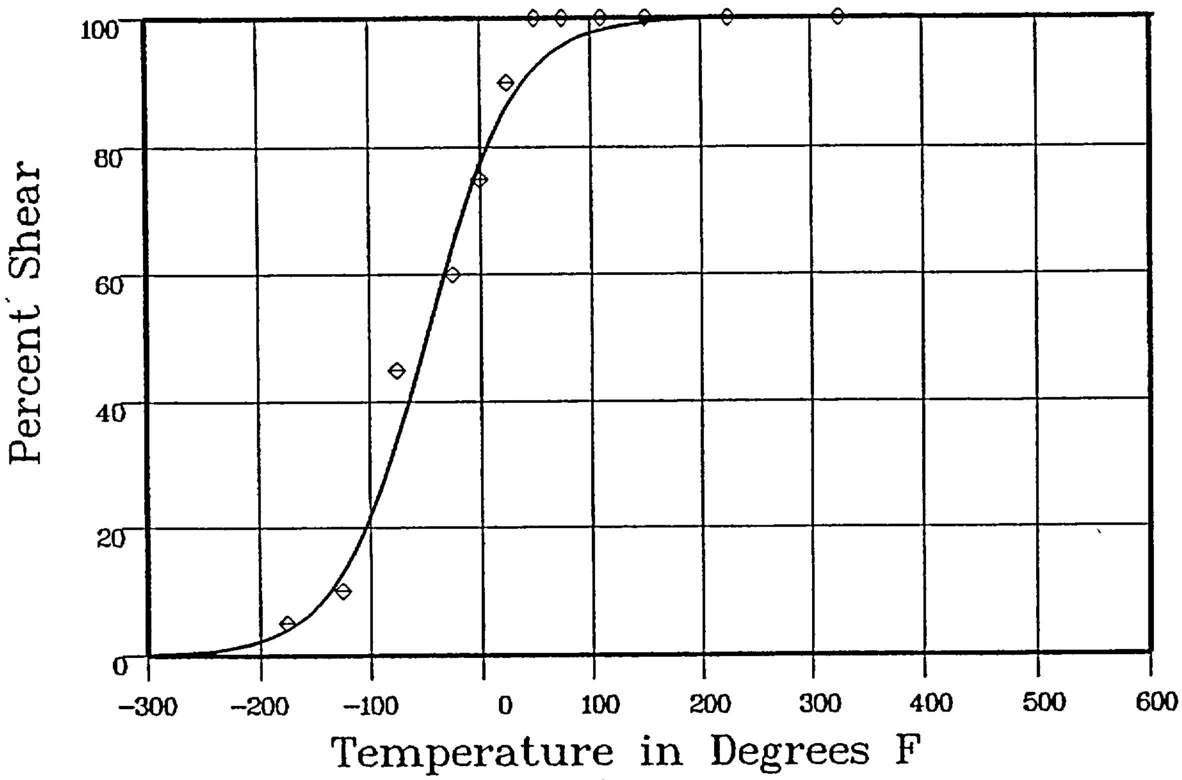
A = 50	B = 50	C = 79.46	T0 = -53.43

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

Temperature at 50% Shear: -53.4

Material: HEAT AFFD ZONE SA533B1 Heat Number: M-1004-2 Orientation:

Capsule: W-263 Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: W-263 Material: HEAT AFFD ZONE SA533B1 Ori: Heat #: M-1004-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-175	5	4.48	.51
-125	10	14.17	-4.17
-75	45	36.75	8.24
-25	60	67.16	-7.16
0	75	79.33	-4.33
25	90	87.8	2.19
50	100	93.1	6.89
75	100	96.2	3.79

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

HEAT AFFECTED ZONE CAPSULE 263

Page 2

Material: HEAT AFFD ZONE SA533B1   Heat Number: M-1004-2   Orientation:

Capsule: W-263   Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
110	100	98.39	1.6
150	100	99.4	.59
225	100	99.9	.09
325	100	99.99	0

SUM of RESIDUALS = 8.27

STANDARD REFERENCE MATERIAL UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:20:51 on 10-01-2002
Page 1
Coefficients of Curve 1

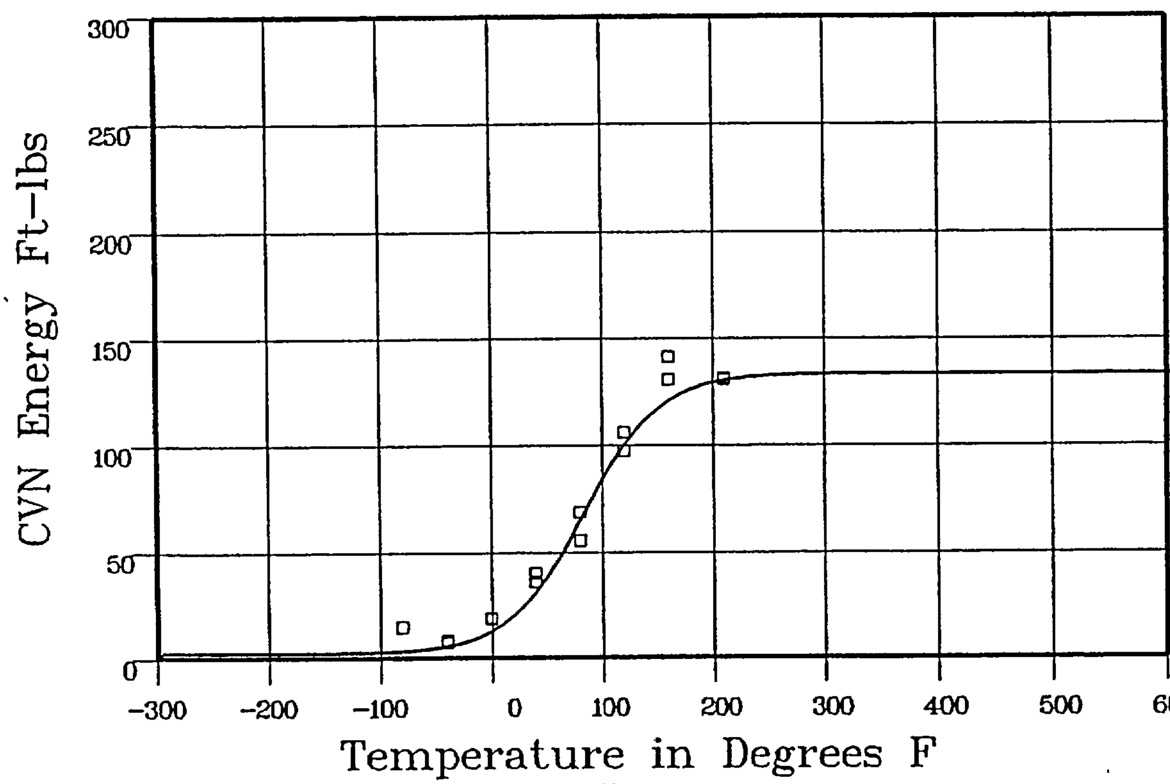
A = 67.59	B = 65.4	C = 67.85	T0 = 78.75

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

Upper Shelf Energy: 133 Fixed Temp at 30 ft-lbs: 34.3 Temp. at 50 ft-lbs: 60 Lower Shelf Energy: 2.19 Fixed

Material: SRM HSST01 Heat Number: A1008-1 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap.: UNIRR Material: SRM HSST01 Ori.: LT Heat #: A1008-1

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-80	14.5	3.4	11.09
-40	7	6.03	.96
-40	8	6.03	1.96
0	18.5	13.89	4.6
0	18	13.89	4.1
40	39.5	33.84	5.65
40	35.5	33.84	1.65
80	68	68.8	-.8
80	55	68.8	-13.8

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

STANDARD REFERENCE MATERIAL UNIRRADIATED

Page 2

Material: SRM HSST01 Heat Number: A1008-1 Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
120	97	103.09	-6.09
120	105.5	103.09	2.4
160	130	122.07	7.92
160	141	122.07	18.92
210	131	130.32	.67
210	130	130.32	-.32

SUM of RESIDUALS = 38.98

STANDARD REFERENCE MATERIAL UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:23:23 on 10-01-2002

Page 1

Coefficients of Curve 1

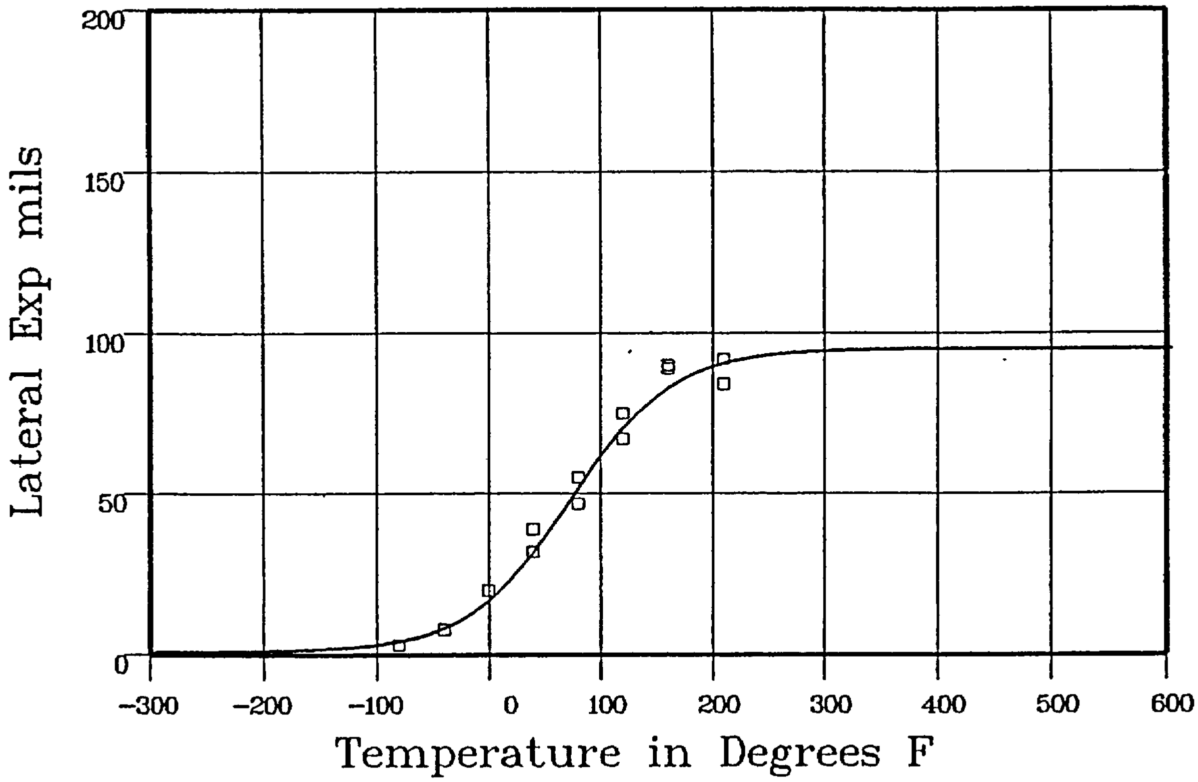
A = 48.12	B = 47.12	C = 92.08	T0 = 68.29

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

Upper Shelf L.E: 95.25 Temperature at L.E. 35. 41.9 Lower Shelf L.E: 1 Fixed

Material: SRM HSST01 Heat Number: A1008-1 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap: UNIRR Material: SRM HSST01 Ori: LT Heat #: A1008-1

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-80	3	4.61	-1.61
-40	8	9.19	-1.19
-40	8	9.19	-1.19
0	20	18.43	1.56
0	20	18.43	1.56
40	39	34.08	4.91
40	32	34.08	-2.08
80	55	54.08	.91
80	47	54.08	-7.08

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

STANDARD REFERENCE MATERIAL UNIRRADIATED

Page 2

Material: SRM HSST01 Heat Number: A1008-1 Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed LE	Differential
120	67	72.12	-5.12
120	75	72.12	2.87
160	90	83.93	6.06
160	89	83.93	5.06
210	84	91.1	-7.1
210	92	91.1	.89

SUM of RESIDUALS = -1.54

STANDARD REFERENCE MATERIAL UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:25:54 on 10-01-2002
Page 1
Coefficients of Curve 1

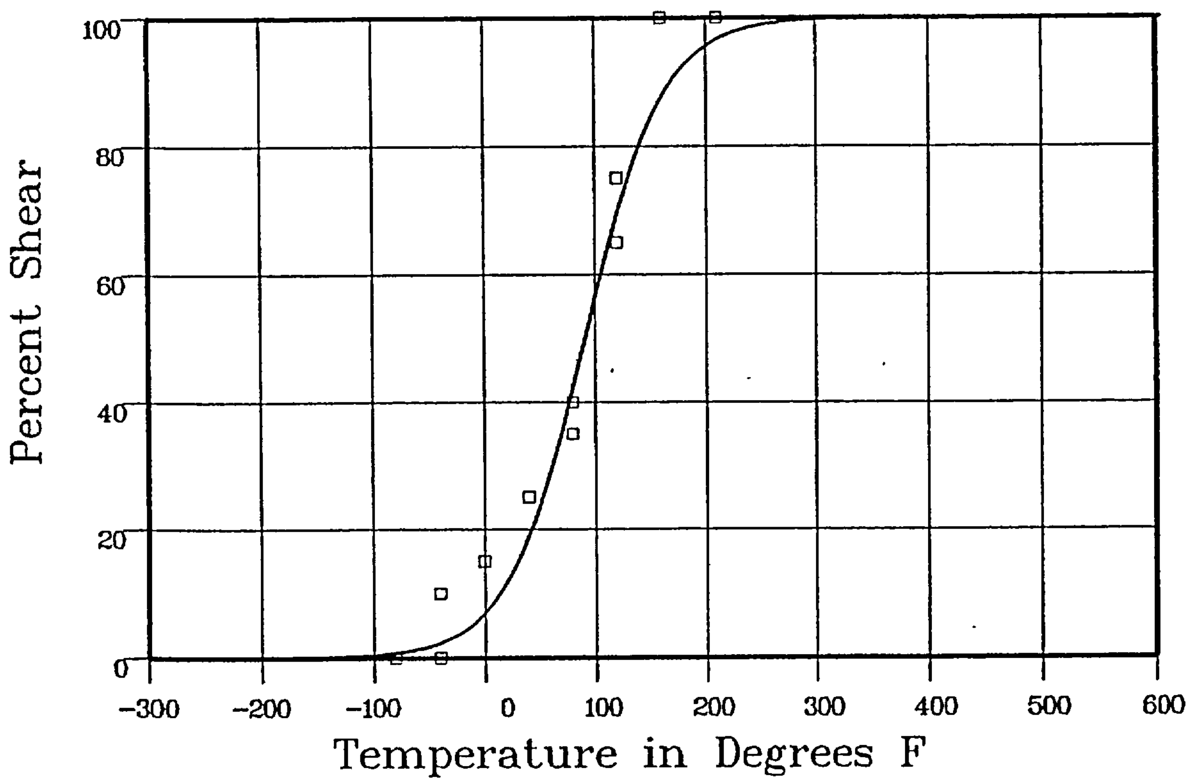
A = 50	B = 50	C = 70.44	T0 = 86.38

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

Temperature at 50% Shear: 86.3

Material: SRM HSST01 Heat Number: A1008-1 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap: UNIRR Material: SRM HSST01 Ori: LT Heat #: A1008-1

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-80	0	.88	-.88
-40	0	2.69	-2.69
-40	10	2.69	7.3
0	15	7.92	7.07
0	15	7.92	7.07
40	25	21.13	3.86
40	25	21.13	3.86
80	40	45.48	-5.48
80	35	45.48	-10.48

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

STANDARD REFERENCE MATERIAL UNIRRADIATED

Page 2

Material: SRM HSST01 Heat Number: A1008-1 Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
120	65	72.2	-7.2
120	75	72.2	2.79
160	100	88.99	11
160	100	88.99	11
210	100	97.09	2.9
210	100	97.09	2.9

SUM of RESIDUALS = 33.06

STANDARD REFERENCE MATERIAL CAPSULE 263

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:20:51 on 10-01-2002

Page 1

Coefficients of Curve 2

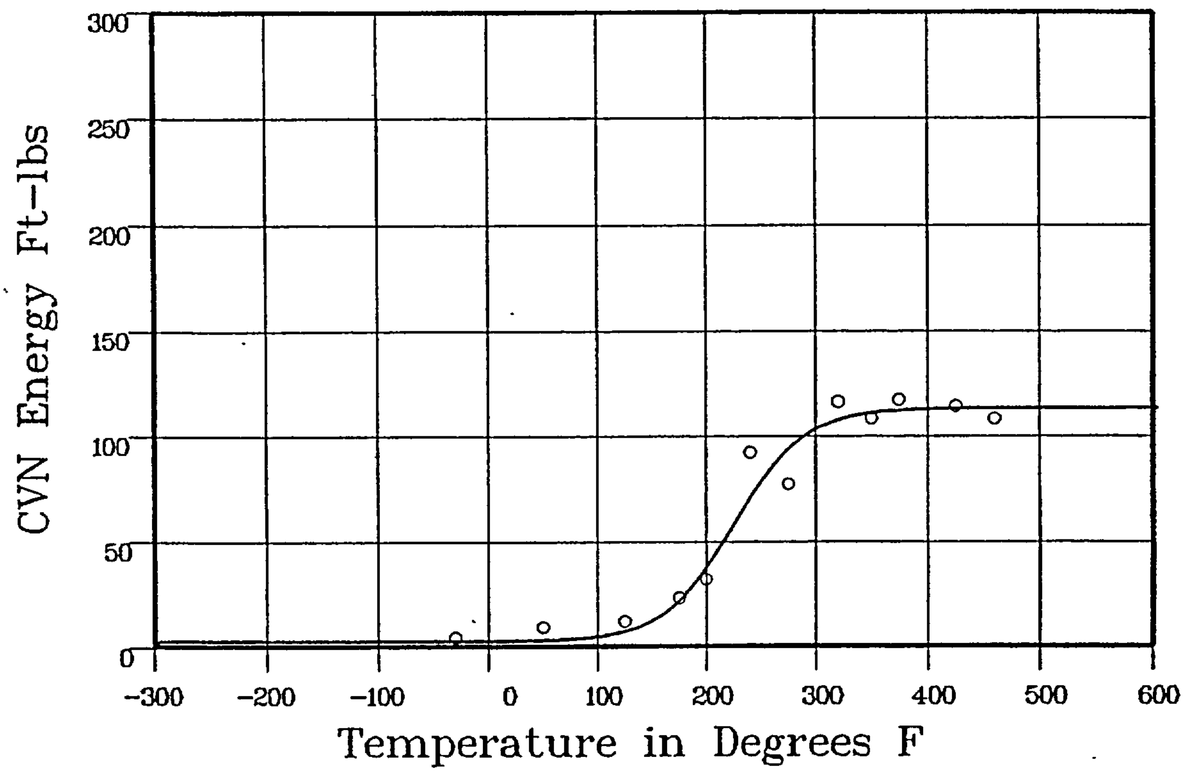
A = 57.59	B = 55.4	C = 64.79	T0 = 220.31

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

Upper Shelf Energy: 113 Fixed   Temp. at 30 ft-lbs: 184.8   Temp. at 50 ft-lbs: 211.3   Lower Shelf Energy: 2.19 Fixed

Material: SRM SA533B1   Heat Number: A1008-1   Orientation: LT

Capsule: W-263   Total Fluence:



Data Set(s) Plotted

Plant: WF3   Cap.: W-263   Material: SRM SA533B1   Ori.: LT   Heat #: A1008-1

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-30	4	2.24	1.75
50	9	2.77	6.22
125	12	7.75	4.24
175	23	24.14	-1.14
200	32	40.78	-8.78
240	92	73.93	18.06
275	77	95.7	-18.7
320	116	108.11	7.88

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

STANDARD REFERENCE MATERIAL CAPSULE 263

Page 2

Material: SRM SA533B1 Heat Number: A1008-1 Orientation: LT

Capsule: W-263 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
350	108	111.01	-3.01
375	117	112.07	4.92
425	114	112.8	1.19
460	108	112.93	-4.93

SUM of RESIDUALS = 7.72

STANDARD REFERENCE MATERIAL CAPSULE 263

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:23:23 on 10-01-2002
Page 1
Coefficients of Curve 2

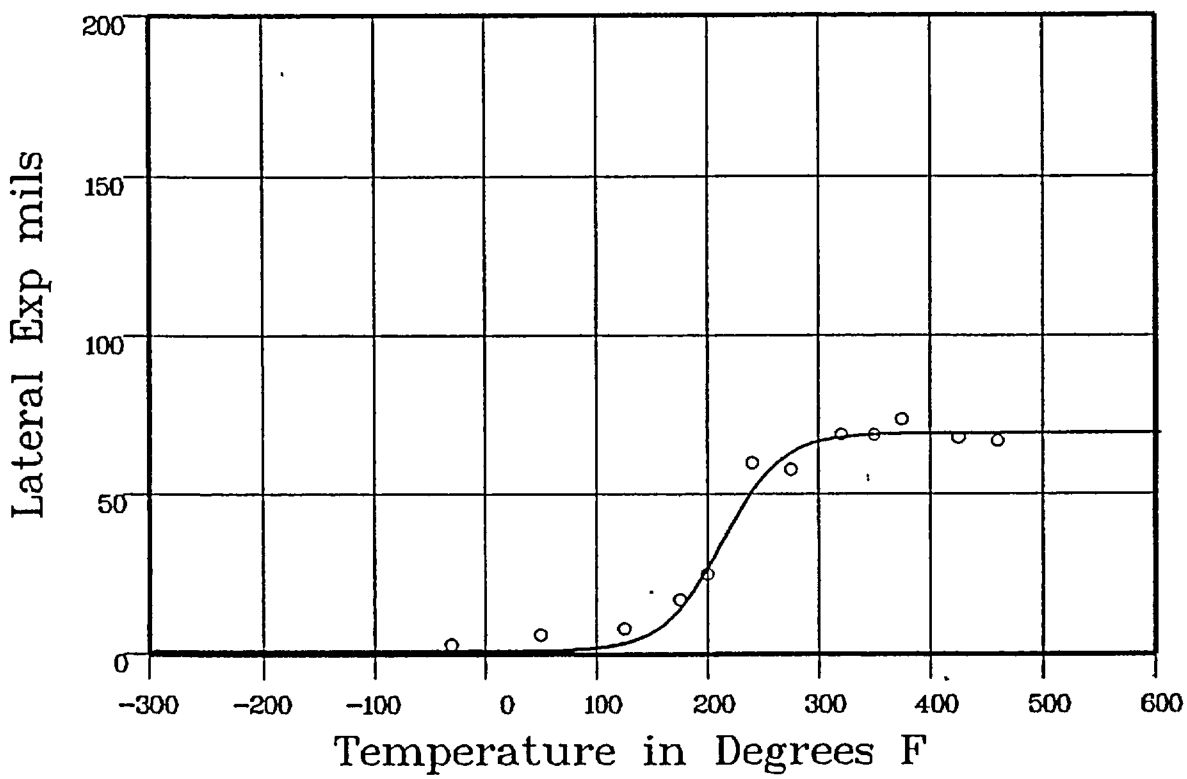
A = 35.22	B = 34.22	C = 53.34	T0 = 208.88

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

Upper Shelf L.E.: 69.45 Temperature at L.E. 35: 208.5 Lower Shelf L.E.: 1 Fixed

Material: SRM SA533B1 Heat Number: A1008-1 Orientation: LT

Capsule: W-263 Total Fluence:



Data Set(s) Plotted
Plant: WF3 Cap.: W-263 Material: SRM SA533B1 Ori.: LT Heat #: A1008-1

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-30	3	1	1.99
50	6	1.17	4.82
125	8	3.82	4.17
175	17	16	.99
200	25	29.57	-4.57
240	60	53.19	6.8
275	58	64.16	-6.16
320	69	68.41	.58

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

STANDARD REFERENCE MATERIAL CAPSULE 263

Page 2

Material: SRM SA533B1 Heat Number: A1008-1 Orientation. LT

Capsule: W-263 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed LE.	Differential
350	69	69.11	-.11
375	74	69.32	4.67
425	68	69.43	-1.43
460	67	69.45	-2.45

SUM of RESIDUALS = 9.31

STANDARD REFERENCE MATERIAL CAPSULE 263

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 16:25:54 on 10-01-2002

Page 1

Coefficients of Curve 2

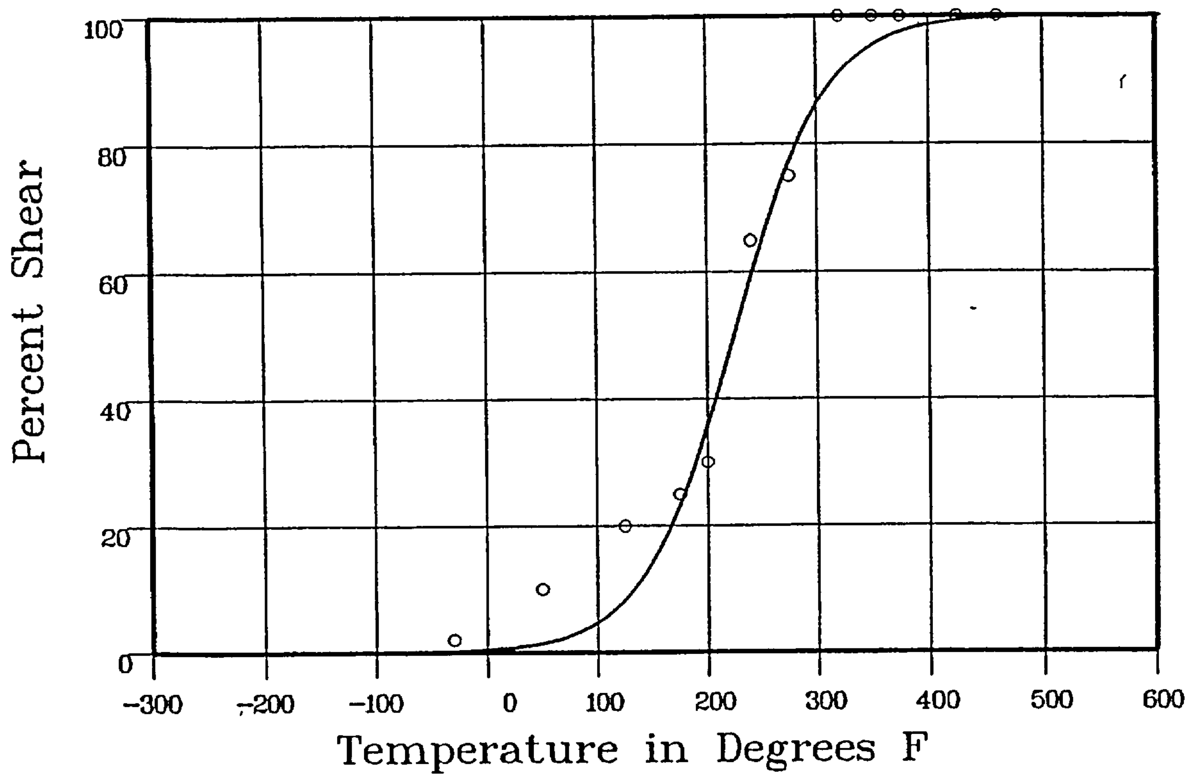
A = 50	B = 50	C = 82.74	T0 = 219.84

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

Temperature at 50% Shear: 219.8

Material: SRM SA533B1 Heat Number: A1008-1 Orientation: LT

Capsule: W-263 Total Fluence:



Data Set(s) Plotted

Plant: WF3 Cap.: W-263 Material: SRM SA533B1 Ori.: LT Heat #: A1008-1

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-30	2	.23	1.76
50	10	1.62	8.37
125	20	9.17	10.82
175	25	25.27	-.27
200	30	38.23	-8.23
240	65	61.94	3.05
275	75	79.13	-4.13
320	100	91.84	8.15

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

STANDARD REFERENCE MATERIAL CAPSULE 263

Page 2

Material: SRM SA533B1 Heat Number: A1008-1 Orientation: LT

Capsule: W-263 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
350	100	95.87	4.12
375	100	97.7	2.29
425	100	99.3	.69
460	100	99.69	.3

SUM of RESIDUALS = 26.95

APPENDIX C

Charpy V-Notch Shift Results for Each Capsule

On the following pages are the Charpy V-notch shift results for each capsule based on using the Hyperbolic Tangent Curve-Fitting Method (CVGRAPH, Version 4.1).

Table C-1 Changes in Average 30 and 50 ft-lb Temperatures for Lower Shell Plate M-1004-2 (Longitudinal Orientation), CVGRAPH 4.1

Capsule	Unirradiated T_{30}	CVGRAPH Fit T_{30}	ΔT_{30}	Unirradiated T_{50}	CVGRAPH Fit T_{50}	ΔT_{50}
97°	-13.37°F	-7.06°F	6.3°F	11.76°F	20.42°F	8.7°F
263°	--	--	--	---	--	--

Table C-2 Changes in Average 35 mil Lateral Expansion Temperatures and Average Energy Absorption at Full Shear for Lower Shell Plate M-1004-2 (Longitudinal Orientation), CVGRAPH 4.1

Capsule	Unirradiated T_{35}	CVGRAPH Fit T_{35}	ΔT_{35}	Unirradiated Upper Shelf Energy	CVGRAPH Fit Upper Shelf Energy	ΔE
97°	5.07°F	10.52°F	5.4°F	170 ft-lb	155 ft-lb	-15 ft-lb
263°	--	--	--	--	--	--

Table C-3 Changes in Average 30 and 50 ft-lb Temperatures for Lower Shell Plate M-1004-2 (Transverse Orientation), CVGRAPH 4.1

Capsule	Unirradiated T_{30}	CVGRAPH Fit T_{30}	ΔT_{30}	Unirradiated T_{50}	CVGRAPH Fit T_{50}	ΔT_{50}
97°	-24.44°F	3.45°F	27.9°F	2.89°F	33.10°F	30.2°F
263°	-24.44°F	-33.57°F	-9.1°F	2.89°F	11.01°F	8.1°F

Table C-4 Changes in Average 35 mil Lateral Expansion Temperatures for Lower Shell Plate M-1004-2 (Transverse Orientation), CVGRAPH 4.1

Capsule	Unirradiated T_{35}	CVGRAPH Fit T_{35}	ΔT_{35}	Unirradiated Upper Shelf Energy	CVGRAPH Fit Upper Shelf Energy	ΔE
97°	-6.73°F	18.83°F	25.6°F	141 ft-lb	124 ft-lb	-17 ft-lb
263°	-6.73°F	9.81°F	16.5°F	141 ft-lb	131 ft-lb	-10 ft-lb

Table C-5 Changes in Average 30 and 50 ft-lb Temperatures for Surveillance Weld Material, CVGRAPH 4.1

Capsule	Unirradiated T_{30}	CVGRAPH Fit T_{30}	ΔT_{30}	Unirradiated T_{50}	CVGRAPH Fit T_{50}	ΔT_{50}
97°	-84.58°F	-56.36°F	28.2°F	-65.19°F	-34.87°F	30.3°F
263°	-84.58°F	-77.71°F	6.9°F	-65.19°F	-51.38°F	13.8°F

Table C-6 Changes in Average 35 mil Lateral Expansion Temperatures and Average Energy Absorption at Full Shear for Surveillance Weld Material, CVGRAPH 4.1

Capsule	Unirradiated T_{35}	CVGRAPH Fit T_{35}	ΔT_{35}	Unirradiated Upper Shelf Energy	CVGRAPH Fit Upper Shelf Energy	ΔE
97°	-68.31°F	-39.63°F	28.7°F	156 ft-lb	143 ft-lb	-13 ft-lb
263°	-68.31°F	-46 80°F	21.5°F	156 ft-lb	145 ft-lb	-11 ft-lb

Table C-7 Changes in Average 30 and 50 ft-lb Temperatures for the Heat-Affected-Zone Material CVGRAPH 4.1

Capsule	Unirradiated T_{30}	CVGRAPH Fit T_{30}	ΔT_{30}	Unirradiated T_{50}	CVGRAPH Fit T_{50}	ΔT_{50}
97°	-117.09°F	-103.49°F	13.6°F	-90.08°F	-71.83°F	18.2°F
263°	-117.09°F	-91.96°F	25.1°F	-90.08°F	-62.15°F	27.9°F

Table C-8 Changes in Average 35 mil Lateral Expansion Temperatures and Average Energy Absorption at Full Shear for the Heat-Affected-Zone Material, CVGRAPH 4.1

Capsule	Unirradiated T_{35}	CVGRAPH Fit T_{35}	ΔT_{35}	Unirradiated Upper Shelf Energy	CVGRAPH Fit Upper Shelf Energy	ΔE
97°	-89.55°F	-72.24°F	17.3°F	170 ft-lb	156 ft-lb	-14 ft-lb
263°	-89.55°F	-56.57°F	33.0°F	170 ft-lb	163 ft-lb	-7 ft-lb

Table C-9 Changes in Average 30 and 50 ft-lb Temperatures for the Correlation Monitor Material HSST Plate 01, CVGRAPH 4.1

Capsule	Unirradiated T_{30}	CVGRAPH Fit T_{30}	ΔT_{30}	Unirradiated T_{50}	CVGRAPH Fit T_{50}	ΔT_{50}
97°	---	---	---	---	---	---
263°	34.31°F	184.87°F	150.5°F	60.02°F	211.36°F	151.3°F

Table C-10 Changes in Average 35 mil Lateral Expansion Temperatures and Average Energy Absorption at Full Shear for the Correlation Monitor Material HSST Plate 01, CVGRAPH 4.1

Capsule	Unirradiated T_{35}	CVGRAPH Fit T_{35}	ΔT_{35}	Unirradiated Upper Shelf Energy	CVGRAPH Fit Upper Shelf Energy	ΔE
97°	---	---	---	---	---	---
263°	41.94°F	208.53°F	166.6°F	133 ft-lb	113 ft-lb	-20 ft-lb

APPENDIX D

Waterford Unit 3 Surveillance Data Credibility Analysis

INTRODUCTION:

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

To date there have been two surveillance capsules removed from the Waterford Unit 3 reactor vessel. To use these surveillance data sets, they must be shown to be credible. In accordance with the discussion of Regulatory Guide 1.99, Revision 2, there are five requirements that must be met for the surveillance data to be judged credible.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the Waterford Unit 3 reactor vessel surveillance data and determine if the Waterford Unit 3 surveillance data are credible.

EVALUATION:

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

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

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

The Waterford Unit 3 reactor vessel consists of the following beltline region materials:

- Intermediate Shell Plates M-1003-1, 2 and 3,
- Lower Shell Plates M-1004-1, 2 and 3,
- Intermediate-to-lower shell circumferential weld seam 101-171 (Heat 88114, Linde 0091)
- Intermediate shell plate longitudinal weld seams 101-124 A, B & C (Heat BOLA and HODA).
- Lower shell longitudinal weld seams 101-142A, B & C (Heat 83653, Linde 0091).

Per TR-C-MCS-001, "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Waterford-Unit 3 Reactor Vessel Materials", the surveillance materials in the surveillance program are those judged most limiting. This is further demonstrated in the Nuclear Regulatory Commission's (NRC) Reactor Vessel Integrity Database (RVID), Version 2.01, in which the surveillance plate and weld are predicted to be the most limiting in terms of having the highest adjusted reference temperature after exposure to a neutron fluence of 3.68×10^{19} n/cm^2.

Hence, Criterion 1 is met for the Waterford Unit 3 reactor vessel surveillance program materials.

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

Plots of Charpy energy versus temperature for the unirradiated and irradiated condition are presented in Appendix B of this report. Based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper shelf energy of the Waterford Unit 3 surveillance materials unambiguously. Hence, the Waterford Unit 3 surveillance data meet this criterion.

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

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

Following is the calculation of the best fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2.

Table D-1 Waterford Unit 3 Surveillance Capsule Data

Material	Capsule	Capsule f(a)	FF(b)	ΔRT_{NDT}(c)	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Plate M-1004-2(a) (Longitudinal)	97°	0.647	0.878	6	5.3	0.771
				SUM:	5.3	0.771
	$CF_{Plate RW} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (5.3) \div (0.771) =$ **6.9 °F**					
Lower Shell Plate M-1004-2 (Transverse)	97°	0.647	0.878	28	24.6	0.771
	263°	1.45	1.103	0 (-9^d)	0	1.217
				SUM:	24.6	1.988
	$CF_{Plate WR} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (24.6) \div (1.988) =$ **12.4°F**					
Surveillance Weld Material	97°	0.647	0.878	28	24.6	.771
	263°	1.45	1.103	7	7.7	1.217
				SUM:	32.3	1.988
	$CF_{Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (32.3\ °F) \div (1.988) =$ **16.2°F**					

Notes:

(a) f = calculated fluence from capsule 97° and 263° analysis results, (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from Table 5-10.

(d) Assume 0° F shift for negative measured value.

(e) No longitudinal base metal plate M-1004-2 specimens are in capsule 263°.

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

Table D-2 Best Fit Evaluation for Waterford Unit 3 Surveillance Materials

Base Material	CF (°F)	FF	Measured ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit[a] ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)	< 17°F (Base Metals) < 28°F (Weld Metal)
Lower Shell Plate M-1004-2 (Longitudinal)	6.9	0.878	6	6	0	Not Applicable (single measurement[b])
Lower Shell Plate M-1004-2 (Transverse)	12.4	0.878	28	11	17	Yes
	12.4	1.103	0[c]	14	14	Yes
Surveillance Weld Metal	16.2	0.878	28	14	14	Yes
	16.2	1.103	7	18	11	Yes

NOTES:

(a) Best Fit Line Per Equation 2 of Reg. Guide 1.99 Rev. 2 Position 1.1.
(b) Only one set of longitudinal orientation Charpy specimens from Capsule 97°.
(c) The measured Charpy 30 ft-lb shift was negative (-9°F) which is a non-physical characteristic. The scatter of the Charpy data in the lower shelf region (around 30 ft-lb) for the data set was about 25% to 30% of the measurement value, which could have contributed to the negative shift. Additional analysis of two other indices (50 ft-lb and 35 mils LE) show the scatter was well within the permitted scatter of 17°F. Therefore, the measured Charpy 30 ft-lb shift of 0°F is used for the analysis.

Table D-2 demonstrates that the measured shift values for the transverse orientation plate and for the weld are within the 1σ scatter band (17°F for the plate and 28°F for the weld). Therefore, the Waterford Unit 3 surveillance plate (transverse orientation) and weld data meet this criterion.

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

The capsule specimens are located in the reactor between the thermal shield and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the reactor vessel.

The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. This is supported by the fact that the 558° temperature monitors in the surveillance capsule melted but the 579° temperature monitors did not. Hence, this criterion is met.

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

The Waterford Unit 3 surveillance program has correlation monitor material from HSST Plate 01MY. NUREG/CR-6413 (ORNL/TM-13133) contains a plot of residual vs. fast neutron fluence for the correlation monitor materials from the HSST Program (Figure 11 in the NUREG report). This figure shows a 2σ uncertainty of 50°F. The data used for this plot is contained in Table 14 (in the NUREG Report). The data from the Waterford Unit 3 Capsule 263° are compared to the NUREG data trend in Table D-3.

Table D-3 Calculation of Residual vs. Fast Fluence

Capsule	Fluence ($x\ 10^{19}\ n/cm^2$)	Fluence Factor (FF)	Measured Shift	RG 1.99 Shift (CF*FF)[a]	Residual (Meas.- RG Shift)
263°	1.45	1.103	150	145	5

(a) Per NUREG/CR-6413, ORNL/TM-13133, the Cu and Ni values for the Correlation Monitor Material is 0.18 Cu and 0.66 Ni. This equates to a Chemistry Factor of 136.1°F from Reg. Guide 1.99 Rev. 2.

Table D-3 shows a difference of only 5°F. That is much less than the 2σ uncertainty of 50°F, the allowable scatter in NUREG/CR-6413, ORNL/TM-13133. Hence, this criterion is met.

CONCLUSION:

Based on the preceding responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B and 10 CFR 50.61, the Waterford Unit 3 surveillance data meet credibility requirements 1,2, 3, 4 and 5 of Regulatory Guide 1.99, Revision 2. Meeting these five criteria permits the use of the derived Chemistry Factors of 12.4°F for the transverse orientation plate and 16.2°F for the weld, and permits the use of half the normal value of σ_Δ for predicting shift.

APPENDIX E

VALIDATION OF THE RADIATION TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

E.1 Neutron Dosimetry

Comparisons of measured dosimetry results to both the calculated and least squares adjusted values for all surveillance capsules withdrawn from service to date at Waterford Unit 3 are described herein. The sensor sets from these capsules have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."[E-1] One of the main purposes for presenting this material is to demonstrate that the overall measurements agree with the calculated and least squares adjusted values to within ± 20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in Section 6.2 of this report. This information may also be useful in the future, in particular, as least squares adjustment techniques become accepted in the regulatory environment.

E.1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of the two neutron sensor sets withdrawn to date as a part of the Waterford Unit 3 Reactor Vessel Materials Surveillance Program are presented. The capsule designation, location within the reactor, and time of withdrawal of each of these dosimetry sets were as follows:

Capsule ID	Equivalent Azimuthal Location	Withdrawal Time	Irradiation Time [EFPY]
W-97	7°	End of Cycle 4	4.41
W-263	7°	End of Cycle 11	13.83

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

The passive neutron sensors included in the evaluations of Surveillance Capsules W-97 and W-263 are summarized as follows:

Sensor Material	Reaction Of Interest	Capsule W-97	Capsule W-263
Copper	$^{63}Cu(n,\alpha)^{60}Co$	X	X
Iron	$^{54}Fe(n,p)^{54}Mn$	X	X
Nickel	$^{58}Ni(n,p)^{58}Co$	X	X
Titanium	$^{46}Ti(n,p)^{46}Sc$	X	X
Uranium-238*	$^{238}U(n,f)^{137}Cs$	X	X
Cobalt-Aluminum*	$^{59}Co(n,\gamma)^{60}Co$	X	X

* These measurements include both bare and cadmium-covered sensors.

With regard to the neutron sensors listed above, it should be recognized that both of these capsules also contained sulfur sensors as well. The reaction of interest in these sensors is $^{32}S(n,p)^{32}P$; however, due to the short half-life of ^{32}P (14.28 days), this reaction was not measured for Capsule W-263 as part of the present evaluation, nor for Capsule W-97 as reported in Reference E-2. Further note that the bare uranium sensor measurements for Capsules W-97 and W-263 were excluded from this assessment. The bare $^{238}U(n,f)$ measurement is dominated by contributions from thermal neutron reactions in ^{235}U impurities. These thermal contributions add significant uncertainty to the determination of the $^{238}U(n,f)$ reaction rate. The cadmium-covered ^{238}U sensor provides greater accuracy for the measurement of this fast neutron reaction.

Pertinent physical and nuclear characteristics of the passive neutron sensors are listed in Table E-1. The use of passive monitors such as those listed above does not yield a direct measure of the energy dependent neutron flux at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

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

The radiometric counting of the neutron sensors from Capsule W-97 was carried out by Babcock & Wilcox (B&W).[E-2] The radiometric counting of the sensors from Capsule W-263 was completed at the Pace Analytical Services Laboratory located at the Westinghouse Waltz Mill Site. In all cases, the radiometric counting followed established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor was determined by means of a high-resolution gamma spectrometer. For the copper, iron, nickel, titanium, and cobalt-aluminum sensors, these analyses were performed by direct counting of each of the individual samples. In the case of the uranium fission sensors, the analyses were carried out by direct counting preceded by dissolution and chemical separation of cesium from the sensor material.

The irradiation history of the reactor over the irradiation periods experienced by Capsules W-97 and W-263 was based on the reported monthly power generation of Waterford Unit 3 from initial reactor criticality through the end of the dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations. The irradiation history applicable to Capsules W-97 and W-263 is given in Table E-2.

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

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

where:

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

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

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

Preliminary calculations for the reactions whose products have short half-lives indicated that C_j factors based on cycle average flux values were not appropriate due to a substantial increase in peripheral power from beginning to end of the fuel cycle. The effect of this power change was accounted for by subdividing the cycles immediately preceding the capsule withdrawal (4 and 11) into thirds. This approach better defines the irradiation conditions for the sensors with short half-life reaction products.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets, corrections were made to the ^{238}U measurements to account for the presence of ^{235}U impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. Corrections were also made to the ^{238}U sensor reaction rates to account for gamma ray induced fission reactions that occurred over the course of the capsule irradiations. The correction factors applied to the Waterford Unit 3 fission sensor reaction rates are summarized as follows:

Correction	Capsule W-97	Capsule W-263
^{235}U Impurity/Pu Build-in	0.860	0.827
^{238}U(γ,f)	0.872	0.875
Net ^{238}U Correction	0.750	0.724

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

Results of the sensor reaction rate determinations for Capsules W-97 and W-263 are given in Table E-4. In Table E-4, the measured specific activities, decay corrected saturated specific activities, and computed reaction rates for each sensor indexed to the radial center of the capsule are listed. The fission sensor reaction rates are listed both with and without the applied corrections for ^{238}U impurities, plutonium build-in, and gamma ray induced fission effects.

Examination of the Table E-4 results revealed that the average cadmium covered uranium fission monitor reaction rate for Capsule W-263 was more than 500% lower than Capsule W-97. Due to the fact that these two capsules were irradiated in symmetrically equivalent locations and the half-life of cesium-137 is 30.07 years, the measured reaction rate for the fission monitors in Capsule W-263 should be greater than Capsule W-97. Based on this observation, the cadmium-covered uranium measurements for Capsule W-263 was rejected; i.e., it was not utilized in the least squares adjustment calculation for these capsules.

E.1.2 Least Squares Evaluation of Sensor Sets

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

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

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

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

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

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

For the Waterford Unit 3 application, the calculated neutron spectrum was obtained from the results of plant specific neutron transport calculations described in Section 6.2 of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section E.1.1. The dosimetry reaction cross-sections and uncertainties were obtained from the Sandia National Laboratory Radiation Metrology Laboratory (SNLRML) dosimeter cross-section library[E-4]. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)".

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

The following provides a summary of the uncertainties associated with the least squares evaluation of the Waterford Unit 3 surveillance capsule sensor sets.

Reaction Rate Uncertainties

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

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

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	5%
$^{54}Fe(n,p)^{54}Mn$	5%
$^{58}Ni(n,p)^{58}Co$	5%
$^{46}Ti(n,p)^{46}Sc$	5%
$^{238}U(n,f)^{137}Cs$	10%
$^{59}Co(n,\gamma)^{60}Co$	5%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

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

For sensors included in the Waterford Unit 3 surveillance program, the following uncertainties in the fission spectrum averaged cross-sections are provided in the SNLRML documentation package.

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	4.08-4.16%
$^{54}Fe(n,p)^{54}Mn$	3.05-3.11%
$^{58}Ni(n,p)^{58}Co$	4.49-4.56%
$^{46}Ti(n,p)^{46}Sc$	4.51-4.87%
$^{238}U(n,f)^{137}Cs$	0.54-0.64%
$^{59}Co(n,\gamma)^{60}Co$	0.79-3.59%

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

Calculated Neutron Spectrum

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

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

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

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

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

where

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

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when $g = g'$, and is 0.0 otherwise.

The set of parameters defining the input covariance matrix for the Waterford Unit 3 calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	29%
(E < 0.68 eV)	52%
Short Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

E.1.3 Comparisons of Measurements and Calculations

Results of the least squares evaluations of the dosimetry from the Waterford Unit 3 surveillance capsules withdrawn to date are provided in Tables E-5 and E-6. In Table E-5, measured, calculated, and best-estimate values for sensor reaction rates are given for each capsule. Also provided in this tabulation are ratios of the measured reaction rates to both the calculated and least squares adjusted reaction rates. These ratios of M/C and M/BE illustrate the consistency of the fit of the calculated neutron energy spectra to the measured reaction rates both before and after adjustment. In Table E-6, comparison of the calculated and best estimate values of neutron flux (E > 1.0 MeV) and iron atom displacement rate are tabulated along with the BE/C ratios observed for each of the capsules.

The data comparisons provided in Tables E-5 and E-6 show that the adjustments to the calculated spectra are relatively small and well within the assigned uncertainties for the calculated spectra, measured sensor reaction rates, and dosimetry reaction cross-sections. Further, these results indicate that the use of the least squares evaluation results in a reduction in the uncertainties associated with the exposure of the surveillance capsules. From Section 6.4 of this report, it may be noted that the uncertainty associated with the unadjusted calculation of neutron fluence (E > 1.0 MeV) and iron atom displacements at the surveillance capsule locations is specified as 12% at the 1σ level. From Table E-6, it is noted that the corresponding uncertainties associated with the least squares adjusted exposure parameters have been reduced to 6-7% for neutron flux (E > 1.0 MeV) and 5-6% for iron atom displacement rate. Again, the uncertainties from the least squares evaluation are at the 1σ level.

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

It should be noted that although comparisons between the measured and calculated values for the ^{46}Ti sensors are included in Table E-7, they were not used in determining the average measurement to calculation (M/C) ratio since a bias exists in the SNLRML cross section for the ^{46}Ti(n,p) reaction. This bias may be observed in the data contained in ASTM Standard Practice E261, "Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques." Specifically, Table 3 of ASTM E261 indicates that the sum in quadrature of the experimental uncertainty and the calculated uncertainty for ^{46}Ti(n,p)^{46}Sc in the ^{235}U thermal fission field is 6.86%. Also indicated in the same table is the ratio of the calculated cross-section to the experimentally measured cross section (C/E) that is given as 0.899. Since the difference between the calculated and measured cross-section is greater than the uncertainties involved supports the hypothesis that the calculated cross-section is biased low.

In the case of the direct comparison of measured and calculated sensor reaction rates, the M/C comparisons for fast neutron reactions range from 0.76–1.23 for the 7 samples included in the data set. The overall average M/C ratio for the entire set of Waterford Unit 3 data is 1.04 with an associated standard deviation of 14.7%.

In the comparisons of best estimate and calculated fast neutron exposure parameters, the corresponding BE/C comparisons for the capsule data sets range from 0.93–1.14 for neutron flux (E > 1.0 MeV) and from 0.95 to 1.12 for iron atom displacement rate. The overall average BE/C ratios for neutron flux (E > 1.0 MeV) and iron atom displacement rate are 1.04 with a standard deviation of 14.4% and 1.04 with a standard deviation of 12.0%, respectively.

Based on these comparisons, it is concluded that the calculated fast neutron exposures provided in Section 6.2 of this report are validated for use in the assessment of the condition of the materials comprising the beltline region of the Waterford Unit 3 reactor pressure vessel.

Appendix E References

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

E-2. BAW-2177, "Analysis of Capsule W-97, Entergy Operations, Inc., Waterford Generating Station, Unit No. 3," A. L. Lowe Jr., et al., November 1992.

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

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

Table E-1 Nuclear Parameters Used In The Evaluation Of Neutron Sensors

Monitor Material	Reaction of Interest	Target Atom Fraction	90% Response Range (MeV)	Product Half-life	Fission Yield (%)
Copper	^{63}Cu (n,α)	0.6917	5.0 – 12.0	5.271 y	
Iron	^{54}Fe (n,p)	0.0585	2.4 – 8.8	312.3 d	
Nickel	^{58}Ni (n,p)	0.6808	2.1 – 8.8	70.82 d	
Titanium	^{46}Ti (n,p)	0.0825	4.1 – 10.5	8.379 d	
Uranium-238	^{238}U (n,f)	1.0000	1.5 – 7.9	30.07 y	6.02
Cobalt-Aluminum	^{59}Co (n,γ)	0.0017	non-threshold	5.271 y	

Notes: The 90% response range is defined such that, in the neutron spectrum characteristic of the Waterford Unit 3 surveillance capsules located at 7° from the core cardinal axes, approximately 90% of the sensor response is due to neutrons in the energy range specified with approximately 5% of the total response due to neutrons with energies below the lower limit and 5% of the total response due to neutrons with energies above the upper limit.

The counting results identified by B&W for the Capsule W-97 reactions were reported in Reference E-2 based on the weight of the target material in the sample rather than the total weight of the dosimeter material. As a result, the target atom fraction used in the analysis of the Capsule W-97 sensors was unity.

Table E-2 Monthly Thermal Generation During the First Eleven Fuel Cycles of The Waterford Unit 3 Reactor (Reactor Power of 3390 MWt)

Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)
1985	3	42830	1988	1	2143917	1991	1	2515171
1985	4	613673	1988	2	2339100	1991	2	2211690
1985	5	809488	1988	3	2298501	1991	3	1196293
1985	6	198642	1988	4	76267	1991	4	0
1985	7	846176	1988	5	3254	1991	5	163566
1985	8	0	1988	6	1934985	1991	6	2275493
1985	9	317589	1988	7	2304929	1991	7	2453834
1985	10	1581557	1988	8	2509305	1991	8	2389958
1985	11	2319818	1988	9	2107712	1991	9	2435284
1985	12	1423475	1988	10	1509724	1991	10	2522290
1986	1	2313146	1988	11	1052050	1991	11	2256967
1986	2	2236098	1988	12	2282880	1991	12	2500136
1986	3	551214	1989	1	2231876	1992	1	2517213
1986	4	2380512	1989	2	2264143	1992	2	1649550
1986	5	2337147	1989	3	2458113	1992	3	2284044
1986	6	2345202	1989	4	2418581	1992	4	2429922
1986	7	1646157	1989	5	2509484	1992	5	2418613
1986	8	2497833	1989	6	2390707	1992	6	2436179
1986	9	2198347	1989	7	2292538	1992	7	2400380
1986	10	2206239	1989	8	2249799	1992	8	2476631
1986	11	2009429	1989	9	1783533	1992	9	1417812
1986	12	0	1989	10	0	1992	10	0
1987	1	0	1989	11	640027	1992	11	1581907
1987	2	1379459	1989	12	2444128	1992	12	2505327
1987	3	2149043	1990	1	1787064	1993	1	2514707
1987	4	2257170	1990	2	1612051	1993	2	2271376
1987	5	2278405	1990	3	2271449	1993	3	2373995
1987	6	2426725	1990	4	2433315	1993	4	2433762
1987	7	2489860	1990	5	2499444	1993	5	2518857
1987	8	2332754	1990	6	2434503	1993	6	2310136
1987	9	1406064	1990	7	2508532	1993	7	2513308
1987	10	1734107	1990	8	2333299	1993	8	2517897
1987	11	2418833	1990	9	2363915	1993	9	2431720
1987	12	2224789	1990	10	1728843	1993	10	2519890
			1990	11	2433754	1993	11	2437790
			1990	12	2516099	1993	12	2517986

Table E-2 (cont'd)
Monthly Thermal Generation During the First Eleven Fuel Cycles of The Waterford Unit 3 Reactor (Reactor Power of 3390 MWt)

Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)
1994	1	2517230	1997	1	2520469	2000	1	2513887
1994	2	2273776	1997	2	2273932	2000	2	2358614
1994	3	318321	1997	3	2520600	2000	3	2346459
1994	4	433275	1997	4	886717	2000	4	2436564
1994	5	2515334	1997	5	0	2000	5	2521322
1994	6	2306353	1997	6	0	2000	6	1873406
1994	7	2518426	1997	7	99594	2000	7	2521097
1994	8	2514659	1997	8	2505653	2000	8	2521192
1994	9	2430858	1997	9	2439155	2000	9	2436861
1994	10	2518011	1997	10	2523657	2000	10	1051538
1994	11	2439425	1997	11	2433511	2000	11	1030289
1994	12	2515464	1997	12	2433511	2000	12	2517757
1995	1	2517937	1998	1	2513765	2001	1	2521026
1995	2	2273361	1998	2	2276816	2001	2	2121310
1995	3	2516977	1998	3	2520985	2001	3	2520656
1995	4	2436285	1998	4	2436390	2001	4	2436332
1995	5	2518637	1998	5	2503708	2001	5	2518333
1995	6	839448	1998	6	2439738	2001	6	2367725
1995	7	2521005	1998	7	2283997	2001	7	2521008
1995	8	2521249	1998	8	2521163	2001	8	2521122
1995	9	1774478	1998	9	1029804	2001	9	2434974
1995	10	0	1998	10	2456652	2001	10	2520999
1995	11	1929688	1998	11	1396144	2001	11	2439796
1995	12	2517767	1998	12	2262325	2001	12	2520997
1996	1	2517086	1999	1	2486638	2002	1	2518017
1996	2	2357937	1999	2	1291559	2002	2	2277045
1996	3	2516925	1999	3	0	2002	3	1704398
1996	4	2436647	1999	4	2183602			
1996	5	2326692	1999	5	2513051			
1996	6	2440018	1999	6	2244073			
1996	7	1247706	1999	7	2521175			
1996	8	2105079	1999	8	1808465			
1996	9	2396700	1999	9	880564			
1996	10	2471709	1999	10	2524076			
1996	11	2295324	1999	11	2080135			
1996	12	2449101	1999	12	2370546			

Table E-2 (cont'd)
Monthly Thermal Generation During the First Eleven Fuel Cycles of The Waterford Unit 3 Reactor (Reactor Power of 3390 MWt)

Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)	Year	Month	Thermal Generation (MWt-hr)
1994	1	2517230	1997	1	2520469	2000	1	2513887
1994	2	2273776	1997	2	2273932	2000	2	2358614
1994	3	318321	1997	3	2520600	2000	3	2346459
1994	4	433275	1997	4	886717	2000	4	2436564
1994	5	2515334	1997	5	0	2000	5	2521322
1994	6	2306353	1997	6	0	2000	6	1873406
1994	7	2518426	1997	7	99594	2000	7	2521097
1994	8	2514659	1997	8	2505653	2000	8	2521192
1994	9	2430858	1997	9	2439155	2000	9	2436861
1994	10	2518011	1997	10	2523657	2000	10	1051538
1994	11	2439425	1997	11	2433511	2000	11	1030289
1994	12	2515464	1997	12	2433511	2000	12	2517757
1995	1	2517937	1998	1	2513765	2001	1	2521026
1995	2	2273361	1998	2	2276816	2001	2	2121310
1995	3	2516977	1998	3	2520985	2001	3	2520656
1995	4	2436285	1998	4	2436390	2001	4	2436332
1995	5	2518637	1998	5	2503708	2001	5	2518333
1995	6	839448	1998	6	2439738	2001	6	2367725
1995	7	2521005	1998	7	2283997	2001	7	2521008
1995	8	2521249	1998	8	2521163	2001	8	2521122
1995	9	1774478	1998	9	1029804	2001	9	2434974
1995	10	0	1998	10	2456652	2001	10	2520999
1995	11	1929688	1998	11	1396144	2001	11	2439796
1995	12	2517767	1998	12	2262325	2001	12	2520997
1996	1	2517086	1999	1	2486638	2002	1	2518017
1996	2	2357937	1999	2	1291559	2002	2	2277045
1996	3	2516925	1999	3	0	2002	3	1704398
1996	4	2436647	1999	4	2183602			
1996	5	2326692	1999	5	2513051			
1996	6	2440018	1999	6	2244073			
1996	7	1247706	1999	7	2521175			
1996	8	2105079	1999	8	1808465			
1996	9	2396700	1999	9	880564			
1996	10	2471709	1999	10	2524076			
1996	11	2295324	1999	11	2080135			
1996	12	2449101	1999	12	2370546			

Table E-3 Calculated Cj Factors at the Surveillance Capsule Center Core Midplane Elevation

Fuel Cycle	$\phi(E > 1.0\ MeV)$ [n/cm²-s]		C_j	
	Capsule W-97	Capsule W-263	Capsule W-97	Capsule W-263
1	5.62E+10	5.62E+10	1.233	1.691
2	4.37E+10	4.37E+10	0.958	1.313
3	4.38E+10	4.38E+10	0.961	1.318
4 BOL	3.80E+10	3.80E+10	0.834	1.144
4 MOL	3.88E+10	3.88E+10	0.852	1.168
4 EOL	4.23E+10	4.23E+10	0.927	1.271
5		3.98E+10		1.198
6		3.90E+10		1.173
7		2.13E+10		0.640
8		2.60E+10		0.783
9		2.50E+10		0.752
10		2.38E+10		0.715
11 BOL		1.74E+10		0.524
11 MOL		1.83E+10		0.549
11 EOL		2.09E+10		0.629
Average	4.56E+10	3.32E+10	1.000	1.000

*Note: Cj factors based on the ratio of the cycle specific fast (E > 1.0 MeV) neutron flux divided by the average flux over the total irradiation period were deemed unsuitable for Capsule W-263 since individual reaction rates did not vary proportionally with the fast flux. As a result of this observation, the Cj terms that were utilized in the final analyses for both Capsules W-97 and W-263 were based on the individual reaction rates determined from the synthesized transport calculations. The final Cj terms, which are based on individual reaction rates, are reported on the following pages of this table.

Table E-3 cont'd
Calculated C_j Factors at the Surveillance Capsule Center
Core Midplane Elevation (Capsule W-97)

Fuel Cycle	Capsule W-97 Reaction Rates [rps/atom]						
	^{63}Cu (n,α)	^{54}Fe (n,p)	^{58}Ni (n,p)	^{46}Ti (n,p)	^{238}U (n,f)	^{59}Co (n,γ)	^{59}Co (n,γ) Cd
1	7.74E-17	7.15E-15	9.34E-15	1.31E-15	2.49E-14	3.10E-12	6.20E-13
2	6.21E-17	5.63E-15	7.35E-15	1.04E-15	1.94E-14	2.36E-12	4.76E-13
3	6.24E-17	5.65E-15	7.38E-15	1.05E-15	1.95E-14	2.37E-12	4.77E-13
4 BOL	5.43E-17	4.91E-15	6.41E-15	9.11E-16	1.69E-14	2.06E-12	4.15E-13
4 MOL	5.56E-17	5.02E-15	6.55E-15	9.33E-16	1.73E-14	2.10E-12	4.23E-13
4 EOL	6.06E-17	5.47E-15	7.13E-15	1.02E-15	1.88E-14	2.28E-12	4.60E-13
Avg	6.43E-17	5.86E-15	7.65E-15	1.08E-15	2.02E-14	2.48E-12	4.99E-13

Fuel Cycle	Capsule W-97 Cj						
	^{63}Cu (n,α)	^{54}Fe (n,p)	^{58}Ni (n,p)	^{46}Ti (n,p)	^{238}U (n,f)	^{59}Co (n,γ)	^{59}Co (n,γ) Cd
1	1.202	1.220	1.221	1.210	1.229	1.251	1.244
2	0.965	0.961	0.961	0.963	0.959	0.953	0.955
3	0.969	0.965	0.964	0.968	0.962	0.955	0.957
4 BOL	0.843	0.837	0.837	0.841	0.835	0.830	0.832
4 MOL	0.864	0.856	0.856	0.860	0.853	0.845	0.848
4 EOL	0.942	0.933	0.932	0.937	0.929	0.919	0.922
Avg.	1.000	1.000	1.000	1.000	1.000	1.000	1.000

Table E-3 (cont'd)
Calculated C_j Factors at the Surveillance Capsule Center
Core Midplane Elevation
(Capsule W-263)

Fuel Cycle	Capsule W-263 Reaction Rates [rps/atom]						
	^{63}Cu (n,α)	^{54}Fe (n,p)	^{58}Ni (n,p)	^{46}Ti (n,p)	^{238}U (n,f)	^{59}Co (n,γ)	^{59}Co (n,γ) Cd
1	7.74E-17	7.15E-15	9.34E-15	1.31E-15	2.49E-14	3.10E-12	6.20E-13
2	6.21E-17	5.63E-15	7.35E-15	1.04E-15	1.94E-14	2.36E-12	4.76E-13
3	6.24E-17	5.65E-15	7.38E-15	1.05E-15	1.95E-14	2.37E-12	4.77E-13
4 BOL	5.43E-17	4.91E-15	6.41E-15	9.11E-16	1.69E-14	2.06E-12	4.15E-13
4 MOL	5.56E-17	5.02E-15	6.55E-15	9.33E-16	1.73E-14	2.10E-12	4.23E-13
4 EOL	6.06E-17	5.47E-15	7.13E-15	1.02E-15	1.88E-14	2.28E-12	4.60E-13
5	5.70E-17	5.15E-15	6.72E-15	9.56E-16	1.77E-14	2.15E-12	4.34E-13
6	5.62E-17	5.07E-15	6.61E-15	9.43E-16	1.74E-14	2.08E-12	4.20E-13
7	3.26E-17	2.83E-15	3.69E-15	5.38E-16	9.54E-15	1.10E-12	2.25E-13
8	3.94E-17	3.45E-15	4.49E-15	6.53E-16	1.17E-14	1.35E-12	2.76E-13
9	3.83E-17	3.33E-15	4.34E-15	6.33E-16	1.12E-14	1.30E-12	2.64E-13
10	3.64E-17	3.16E-15	4.11E-15	6.01E-16	1.07E-14	1.23E-12	2.53E-13
11 BOL	2.71E-17	2.34E-15	3.04E-15	4.46E-16	7.83E-15	8.96E-13	1.83E-13
11 MOL	2.83E-17	2.44E-15	3.18E-15	4.66E-16	8.20E-15	9.42E-13	1.92E-13
11 EOL	3.22E-17	2.79E-15	3.63E-15	5.31E-16	9.39E-15	1.08E-12	2.21E-13
Avg	4.85E-17	4.33E-15	5.65E-15	8.10E-16	1.48E-14	1.77E-12	3.59E-13

Fuel Cycle	Capsule W-263 Cj						
	^{63}Cu (n,α)	^{54}Fe (n,p)	^{58}Ni (n,p)	^{46}Ti (n,p)	^{238}U (n,f)	^{59}Co (n,γ)	^{59}Co (n,γ) Cd
1	1.596	1.651	1.654	1.619	1.679	1.749	1.729
2	1.282	1.300	1.301	1.289	1.309	1.333	1.327
3	1.288	1.305	1.306	1.295	1.314	1.336	1.331
4 BOL	1.120	1.133	1.134	1.125	1.141	1.160	1.157
4 MOL	1.147	1.159	1.159	1.152	1.165	1.182	1.179
4 EOL	1.251	1.262	1.263	1.255	1.268	1.285	1.282
5	1.176	1.189	1.189	1.181	1.195	1.212	1.209
6	1.160	1.169	1.170	1.164	1.172	1.172	1.171
7	0.672	0.653	0.652	0.664	0.644	0.622	0.627
8	0.813	0.796	0.795	0.806	0.787	0.764	0.768
9	0.790	0.768	0.767	0.781	0.757	0.731	0.736
10	0.751	0.729	0.728	0.742	0.719	0.693	0.704
11 BOL	0.559	0.539	0.538	0.551	0.529	0.506	0.510
11 MOL	0.584	0.564	0.563	0.576	0.554	0.531	0.536
11 EOL	0.665	0.644	0.642	0.656	0.633	0.611	0.615
Avg.	1.000	1.000	1.000	1.000	1.000	1.000	1.000

Table E-4 Measured Sensor Activities And Reaction Rates Surveillance Capsule W-97

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co (Cd)	Top	3.15E+05	7.72E+05	8.07E-17
	Middle	2.93E+05	7.18E+05	7.51E-17
	Bottom	3.22E+05	7.89E+05	8.25E-17
	Average			**7.94E-17**
^{54}Fe (n,p) ^{54}Mn	Top	5.41E+07	6.93E+07	6.21E-15
	Middle	5.08E+07	6.50E+07	5.83E-15
	Bottom	5.35E+07	6.85E+07	6.14E-15
	Average			**6.06E-15**
^{58}Ni (n,p) ^{58}Co (Cd)	Top	6.69E+07	7.64E+07	7.35E-15
	Middle	6.54E+07	7.47E+07	7.18E-15
	Bottom	7.14E+07	8.15E+07	7.84E-15
	Average			**7.46E-15**
46Ti (n,p) ^{46}Sc	Top	1.23E+07	1.41E+07	1.08E-15
	Middle	1.31E+07	1.51E+07	1.15E-15
	Bottom	1.52E+07	1.75E+07	1.33E-15
	Average			**1.19E-15**
^{238}U (n,f) ^{137}Cs (Cd)	Top	2.94E+05	3.09E+06	2.03E-14
	Middle	2.94E+05	3.09E+06	2.03E-14
	Bottom	3.12E+05	3.28E+06	2.15E-14
	Average			**2.07E-14**
^{238}U (n,f) ^{137}Cs (Cd)	Including ^{235}U, ^{239}Pu, and γ, fission corrections.			**1.55E-14**
^{59}Co (n,γ) ^{60}Co	Top	1.61E+10	3.97E+10	3.88E-12
	Middle	1.81E+10	4.46E+10	4.36E-12
	Bottom	1.44E+10	3.55E+10	3.47E-12
	Average			**3.90E-12**
^{59}Co (n,γ) ^{60}Co (Cd)	Top	1.91E+09	4.70E+09	4.60E-13
	Middle	1.64E+09	4.04E+09	3.95E-13
	Bottom	1.84E+09	4.53E+09	4.43E-13
	Average			**4.33E-13**

Notes: 1) Measured specific activities are indexed to a counting date of March 15, 1991.
2) The average ^{238}U (n,f) reaction rate of 6.44E-14 includes a correction factor of 0.860 to account for plutonium build-in and an additional factor of 0.872 to account for photo-fission effects in the sensor.

Table E-4 cont'd
Measured Sensor Activities And Reaction Rates
Surveillance Capsule W-263

Reaction	Location	Measured Activity (dps/g)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co (Cd)	Top	2.36E+05	3.81E+05	5.82E-17
	Middle	1.69E+05	2.73E+05	4.17E-17
	Bottom	1.98E+05	3.20E+05	4.88E-17
	Average			**4.96E-17**
^{54}Fe (n,p) ^{54}Mn	Top	1.42E+06	3.14E+06	4.98E-15
	Middle	1.36E+06	3.01E+06	4.77E-15
	Bottom	1.33E+06	2.94E+06	4.67E-15
	Average			**4.81E-15**
^{58}Ni (n,p) ^{58}Co (Cd)	Top	8.83E+06	4.85E+07	6.94E-15
	Middle	8.27E+06	4.54E+07	6.50E-15
	Bottom	8.31E+06	4.56E+07	6.53E-15
	Average			**6.66E-15**
46Ti (n,p) ^{46}Sc	Top	2.22E+05	9.94E+05	9.57E-16
	Middle	2.12E+05	9.49E+05	9.14E-16
	Bottom	2.06E+05	9.22E+05	8.88E-16
	Average			**9.20E-16**
^{238}U (n,f) ^{137}Cs (Cd)	Top	2.45E+05	9.58E+05	6.29E-15
	Middle	6.76E+04	2.64E+05	1.74E-15
	Bottom	1.28E+05	5.01E+05	3.29E-15
	Average			**3.77E-15**
^{238}U (n,f) ^{137}Cs (Cd)	Including ^{235}U, ^{239}Pu, and γ, fission corrections.			**2.73E-15**
^{59}Co (n,γ) ^{60}Co	Top	2.45E+07	4.10E+07	2.36E-12
	Middle	2.41E+07	4.04E+07	2.32E-12
	Bottom	1.96E+07	3.28E+07	1.89E-12
	Average			**2.19E-12**
^{59}Co (n,γ) ^{60}Co (Cd)	Top	3.04E+06	5.07E+06	2.92E-13
	Middle	3.11E+06	5.19E+06	2.99E-13
	Bottom	3.04E+06	5.07E+06	2.92E-13
	Average			**2.94E-13**

Notes: 1) Measured specific activities are indexed to a counting date of July 25, 2002.
2) The average ^{238}U (n,f) reaction rate of 6.44E-14 includes a correction factor of 0.827 to account for plutonium build-in and an additional factor of 0.875 to account for photo-fission effects in the sensor.

Table E-5 Comparison of Measured, Calculated, and Best Estimate Reaction Rates At The Surveillance Capsule Center

Capsule W-97

	Reaction Rate [rps/atom]				
Reaction	Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$ (Cd)	7.94E-17	6.45E-17	7.58E-17	1.23	1.05
$^{54}Fe(n,p)^{54}Mn$	6.06E-15	5.86E-15	6.05E-15	1.03	1.00
$^{58}Ni(n,p)^{58}Co$ (Cd)	7.46E-15	7.66E-15	7.74E-15	0.97	0.96
$^{46}Ti(n,p)^{46}Sc$	1.19E-15	1.02E-15	1.14E-15	1.17	1.04
$^{238}U(n,f)^{137}Cs$ (Cd)	1.55E-14	2.03E-14	1.95E-14	0.76	0.79
$^{59}Co(n,\gamma)^{60}Co$	3.90E-12	2.46E-12	3.88E-12	1.59	1.01
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	4.33E-13	4.75E-13	4.35E-13	0.91	1.00

Notes:
1. The Capsule W-97 calculated results reported above for the individual reaction rates were taken from the synthesized transport calculations at the core midplane after the fourth fuel cycle.

Capsule W-263

	Reaction Rate [rps/atom]				
Reaction	Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$ (Cd)	4.95E-17	4.86E-17	5.22E-17	1.02	0.95
$^{54}Fe(n,p)^{54}Mn$	4.80E-15	4.33E-15	4.90E-15	1.11	0.98
$^{58}Ni(n,p)^{58}Co$ (Cd)	6.65E-15	5.66E-15	6.49E-15	1.17	1.02
$^{46}Ti(n,p)^{46}Sc$	9.20E-16	7.61E-16	8.73E-16	1.21	1.05
$^{238}U(n,f)^{137}Cs$ (Cd)	Rejected	1.48E-14	N/A	N/A	N/A
$^{59}Co(n,\gamma)^{60}Co$	2.19E-12	1.76E-12	2.18E-12	1.24	1.00
$^{59}Co(n,\gamma)^{60}Co$ (Cd)	2.94E-13	3.42E-13	2.97E-13	0.86	0.99

Notes:
1. Measured reaction rate for the cadmium covered uranium fission monitor was rejected since it was significantly lower than that of Capsule W-97. Due to the fact that these two capsules were irradiated in symmetrically equivalent locations and the half-life of cesium-137 is 30.07 years, the measured reaction rate for the fission monitors in Capsule W-263 should have been greater than the Capsule W-97 measurement results.
2. The Capsule W-263 calculated results reported above for the individual reaction rates were taken from the synthesized transport calculations at the core midplane after the eleventh fuel cycle.

Table E-6 Comparison of Calculated and Best Estimate Exposure Rates At The Surveillance Capsule Center

Capsule ID	ϕ(E > 1.0 MeV) [n/cm^2-s]			
	Calculated	Best Estimate	Uncertainty (1σ)	BE/C
W-97	4.56E+10	4.24E+10	6%	0.93
W-263	3.32E+10	3.79E+10	7%	1.14

Notes:

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

Capsule ID	Iron Atom Displacement Rate [dpa/s]			
	Calculated	Best Estimate	Uncertainty (1σ)	BE/C
W-97	6.66E-11	6.28E-11	5%	0.94
W-263	4.86E-11	5.43E-11	6%	1.12

Notes:

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

Table E-7 Comparison of Measured/Calculated (M/C) Sensor Reaction Rate Ratios Including all Fast Neutron Threshold Reactions

Reaction	M/C Ratio	
	Capsule W-97	Capsule W-263
$^{63}Cu(n,\alpha)^{60}Co$ (Cd)	1.23	1.02
$^{54}Fe(n,p)^{54}Mn$	1.03	1.11
$^{58}Ni(n,p)^{58}Co$ (Cd)	0.97	1.17
$^{46}Ti(n,p)^{46}Sc$	1.17[1]	1.21[1]
$^{238}U(n,p)^{137}Cs$ (Cd)	0.76	N/A[2]
Average	1.00[3]	1.10[3]
% Standard Deviation	19.2[3]	7.1[3]

Notes:

1. The M/C values for the ^{46}Ti sensors are listed but not used in the average M/C ratio due to a bias present in the SNLRML cross-section data as discussed in Section E.1.3. For additional information, these calculations were repeated using the ^{46}Ti dosimetry cross-section from the BUGLE-96 data library set. The results of these calculations were M/C ratios of 1.10 and 1.14 for Capsules W-97 and W-263, respectively.
2. The cadmium-covered uranium measurement from Capsule W-263 was rejected.
3. The overall average M/C ratio for the set of 7 sensor measurements is 1.04 with an associated standard deviation of 14.7%.

Table E-8 Comparison of Best Estimate/Calculated (BE/C) Exposure Rate Ratios

Capsule ID	BE/C Ratio	
	$\phi(E > 1.0\ MeV)$	dpa/s
W-97	0.93	0.95
W-263	1.14	1.12
Average	1.04	1.04
% Standard Deviation	14.4	12.0

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