WCAP-15589

ANALYSIS OF CAPSULE 38° FROM THE ARIZONA PUBLIC SERVICE COMPANY PALO VERDE UNIT 1 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

Westinghouse Electric Company, LLC

WCAP-15589

Analysis of Capsule 38° from the Arizona Public Service Company Palo Verde Unit 1 Reactor Vessel Radiation Surveillance Program

T. J. Laubham
R.K. Disney
J. Conermann

OCTOBER 2000

Prepared by the Westinghouse Electric Company
for the Arizona Public Service Company

Approved: 
C. H . Boyd, Manager
Equipment & Materials Technology

Approved: 
D. M . Trombola, Manager
Mechanical Systems Integration

WESTINGHOUSE ELECTRIC COMPANY, LLC
Nuclear Services Division
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES (CONTINUED)

LIST OF TABLES (CONTINUED)

LIST OF FIGURES

LIST OF FIGURES (CONTINUED)

LIST OF FIGURES (CONTINUED)

PREFACE

This report has been technically reviewed and verified by:

Reviewer:

Sections 1 through 5, 7, 8, Appendices A, B and C — Ed Terek

Section 6 — G.N Wrights

EXECUTIVE SUMMARY

The purpose of this report is to document the results of the testing of surveillance capsule 38° from Palo Verde Unit 1. Capsule 38° was removed at 9.81 EFPY and post irradiation mechanical tests of the Charpy V-notch and tensile specimens was performed, along with a fluence evaluation based methodology and nuclear data including recently released neutron transport and dosimetry cross-section libraries derived from the ENDF/B-VI database. The calculated peak clad base/metal vessel fluence after 9.81 EFPY of plant operation was 4.65×10^{18} n/cm^2 and the surveillance Capsule 38° calculated fluence was 7.85×10^{18} 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.

1 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance capsule 38° the second capsule to be removed from the Palo Verde Unit 1 reactor pressure vessel, led to the following conclusions: (General Note: Temperatures are reported to two significant digits only to match CVGraph output.)

- The capsule received an average fast neutron calculated fluence (E > 1.0 MeV) of 7.85 x 10^{18} n/cm^2 after 9.81 effective full power years (EFPY) of plant operation.

- Irradiation of the reactor vessel intermediate shell plate M-6701-2 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation), to 7.85 x 10^{18} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 0.57°F and a 50 ft-lb transition temperature increase of 5.9°F. This results in an irradiated 30
ft-lb transition temperature of 8.72°F and an irradiated 50 ft-lb transition temperature of 42.52°F for the longitudinally oriented specimens

- Irradiation of the reactor vessel intermediate shell plate M-6701-2 Charpy specimens, oriented with the longitudinal axis of the specimen normal to the major working direction of the plate (transverse orientation), to 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV) resulted in a 30 ft-lb transition temperature decrease of -19.71°F and a 50 ft-lb transition temperature decrease of -15.98°F. This results in an irradiated 30 ft-lb transition temperature of 10.14°F and an irradiated 50 ft-lb transition temperature of 55.25°F for transversely oriented specimens.

- Irradiation of the weld metal Charpy specimens to 7.85 x 10^{18} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 6.27°F and a 50 ft-lb transition temperature increase of 8.12°F. This results in an irradiated 30 ft-lb transition temperature of –47.0°F and an irradiated 50 ft-lb transition temperature of -25.3°F.

- Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature decrease of -26.59°F and a 50 ft-lb transition temperature decrease of -13.43°F. This results in an irradiated 30 ft-lb transition temperature of -89.79°F and an irradiated 50 ft-lb transition temperature of -40.01°F.

- Irradiation of the standard reference material Charpy specimens to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 114.25°F and a 50 ft-lb transition temperature increase of 128.6°F. This results in an irradiated 30 ft-lb transition temperature of 136.16°F and an irradiated 50 ft-lb transition temperature of 176.21°F.

- The average upper shelf energy of the intermediate shell plate M-6701-2 (longitudinal orientation) resulted in an average energy decrease of 10 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV). This results in an irradiated average upper shelf energy of 141 ft-lb for the longitudinally oriented specimens.

- The average upper shelf energy of the Intermediate shell plate M-6701-2 (transverse orientation) resulted in no average energy decrease of after irradiation to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV).

This results in an irradiated average upper shelf energy of 98 ft-lb for the transversely oriented specimens.

- The average upper shelf energy of the weld metal Charpy specimens resulted an average energy decrease of 6 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 158 ft-lb for the weld metal specimens.

- The average upper shelf energy of the weld HAZ metal Charpy specimens resulted an average energy decrease of 16 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 119 ft-lb for the weld HAZ metal.

- The average upper shelf energy of the standard reference material Charpy specimens resulted an average energy decrease of 24 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 105 ft-lb for the standard reference material.

- A comparison of the Palo Verde Unit 1 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 materials (Weld and Plate) for capsule 38° are less than the Regulatory Guide 1.99, Revision 2, predictions.

  - The measured percent decrease in upper shelf energy of the Capsule 38° surveillance material is less than the Regulatory Guide 1.99, Revision 2, predictions.

- The peak calculated and best estimate end-of-license (32 EFPY) neutron fluence (E> 1.0 MeV) at the core midplane for the Palo Verde Unit 1 reactor vessel using the Regulatory Guide 1.99, Revision 2 attenuation formula (ie. Equation # 3 in the guide; $f_{(depth\ x)} = f_{surface} * e^{(-0.24x)}$) is as follows:

Calculated:	Vessel inner radius*	=	1.64 x 10^{19} n/cm^2
	Vessel 1/4 thickness	=	9.52 x 10^{18} n/cm^2
	Vessel 3/4 thickness	=	3.21 x 10^{18} n/cm^2
Best Estimate:	Vessel inner radius*	=	1.36 x 10^{19} n/cm^2
	Vessel 1/4 thickness	=	7.90 x 10^{18} n/cm^2
	Vessel 3/4 thickness	=	2.66 x 10^{18} n/cm^2

2 INTRODUCTION

This report presents the results of the examination of the Capsule located at 38°, the second capsule to be removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Palo Verde Unit 1 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Arizona Public Service Company Palo Verde Unit 1 reactor pressure vessel materials was designed and recommended by Combustion Engineering. A description of the surveillance program and the preirradiation 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 E185-82, "Standard Practice for conducting Surveillance Tests for light-water cooled Nuclear Power Reactor Vessels". Capsule 38° was removed from the reactor after 9.81 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 38°, removed from the Palo Verde Unit 1 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 Arizona Public Service Company Palo Verde Unit 1 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_{Ia} curve) which appears in Appendix G to the ASME Code[4]. The K_{Ia} curve is a Intermediate bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{Ia} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors. Note that Code Case N-640 now allows the use of the K_{Ic} curve as an alternative to the K_{Ia} 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 Palo Verde Unit 1 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 Palo Verde Unit 1 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 support barrel and the vessel wall at locations shown in Figure 4-1. The vertical center of the capsules is opposite the vertical center of the core.

Capsule 38° was removed after 9.81 effective full power years (EFPY) of plant operation. This capsule contained Charpy V-notch impact and tensile specimens made from reactor vessel intermediate shell course plate M-6701-2, submerged arc weld metal representative of the beltline region welds, heat-affected-zone (HAZ) metal and standard reference material from HSST-01MY plate. All HAZ specimens are obtained within the heat-affected-zone of lower shell plate M-4311-1 and 4311-2.

Charpy V-notch impact specimens from Plate M-6701-2 were with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation). Charpy V-notch impact specimens from Plate M-6701-2 were 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-6701-2 were machined in with the longitudinal axis of the specimen perpendicular 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 38° contained dosimeter wires of sulfur, iron, titanium, nickel (cadmium-shielded), 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% Ag	Melting Point 590°F (310°C)

The chemical Composition and heat treatment of the surveillance material is presented in Tables 4-1 and 4-2. The chemical analysis reported in Table 4-1 was obtained from TR-V-MCM-012. The arrangement of the various mechanical test specimens, dosimeters and thermal monitors contained in capsule 38° is shown in Figure 4-2. A typical Palo Verde Unit 1 surveillance capsule Charpy impact compartment assembly is shown in Figure 4-3, while Figure 4-4 and 4-5 show the Tensile-Monitor Compartment and Charpy Flux & Compact Tension Compartment, respectively.

Table 4-1 Chemical Composition (wt %) of the Palo Verde Unit 1 Reactor Vessel Surveillance Materials

Element	Plate M-6701-2	Weld Metal M-4311-2/M-4311-3
C	0.24	0.16
Mn	1.33	1.08
P	0.004	0.005
S	0.010	0.005
Si	0.28	0.24
Ni	0.60	0.06
Cr	0.04	0.06
Mo	0.53	0.58
V	0.003	0.006
Cb	<0.01	0.01
Ti	<0.01	0.01
Co	0.013	0.018
Cu	0.06	0.04
Al	0.029	0.005
B	<0.001	0.001
W	<0.01	0.02
Sb	0.0015	0.0014
As	<0.001	0.006
Sn	0.003	0.004
Zr	<0.001	0.001
Pb	0.001	0.001
N	0.009	0.007

Table 4-2 Heat Treatment of the Palo Verde Unit 1 Reactor Vessel Surveillance Material			
Material	**Temperature (°F)**	**Time (hrs.)**	**Coolant**
Surveillance Program Test Plate M-6701-2	Austenitizing: 1600 ± 25	4	Water-quenched
	Tempered: 1225 ± 25	4	Air Cooled
	Stress Relief: 1150 ± 25	40	Furnace Cooled to 600°F
Weld Metal	Stress Relief: 1125 ± 25	41 hr & 45 min.	Furnace Cooled

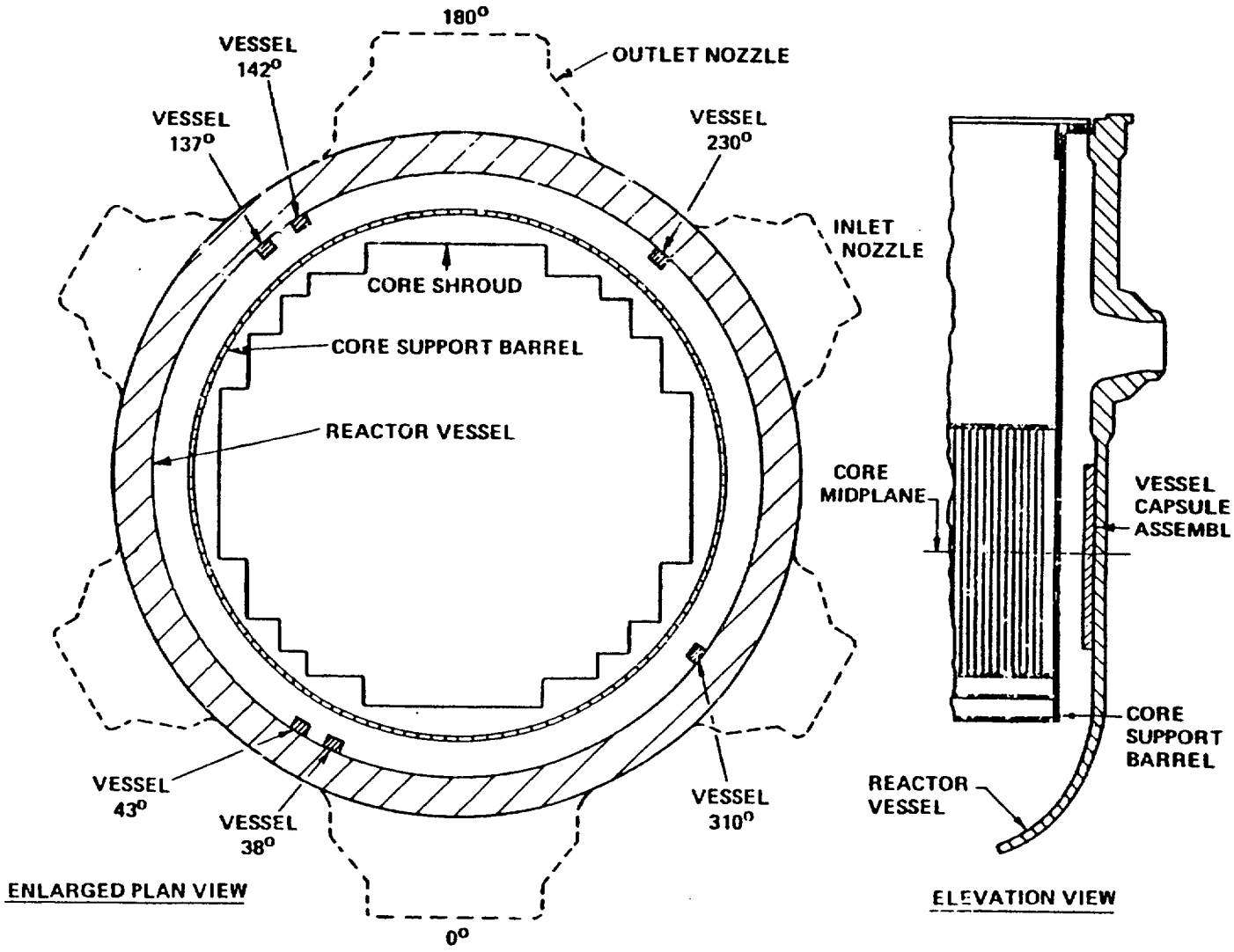


Figure 4-1. Arrangement of Surveillance Capsules in the Palo Verde Unit 1 Reactor Vessel

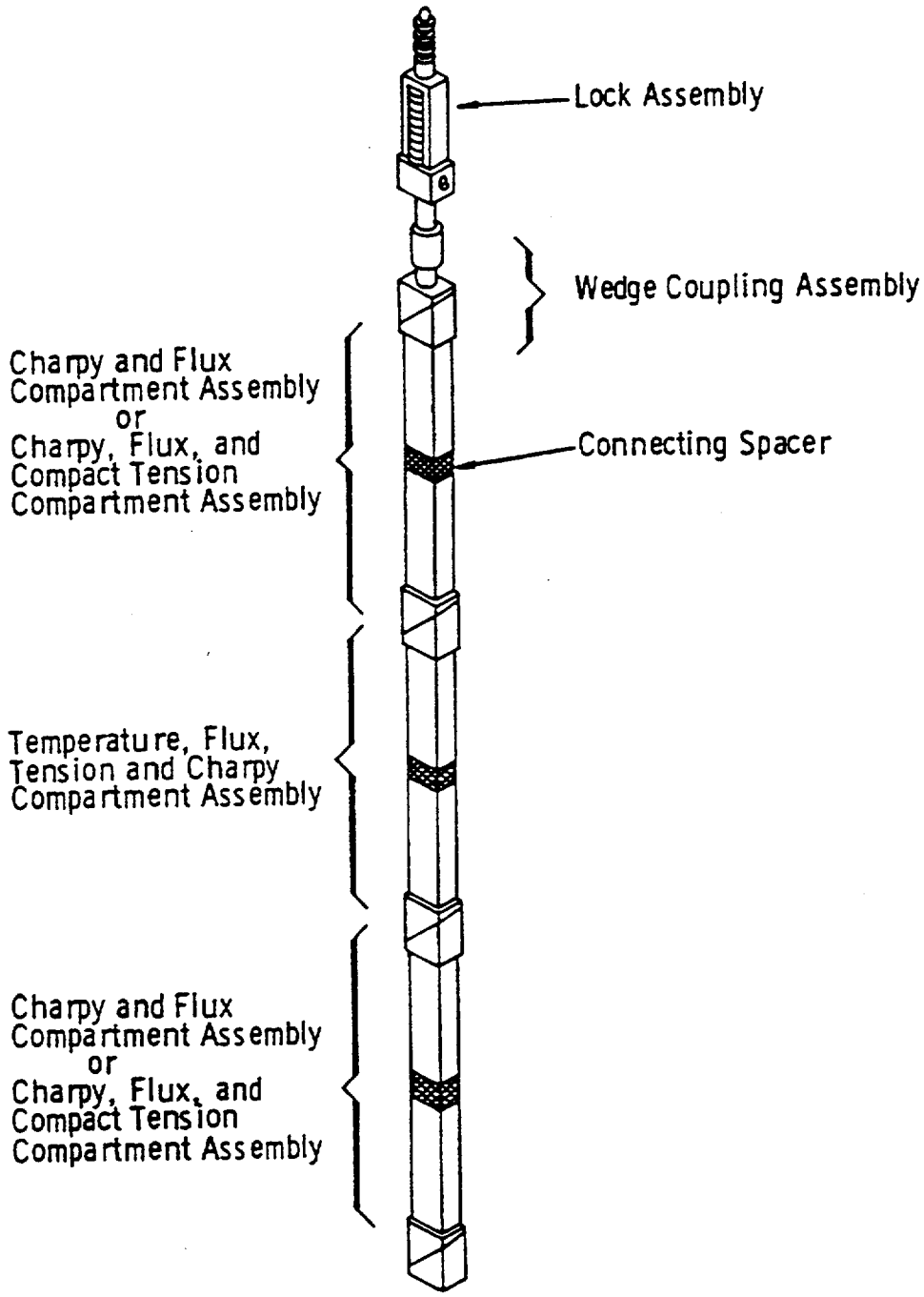


Figure 4-2 Typical Palo Verde Unit 1 Surveillance Capsule Assembly

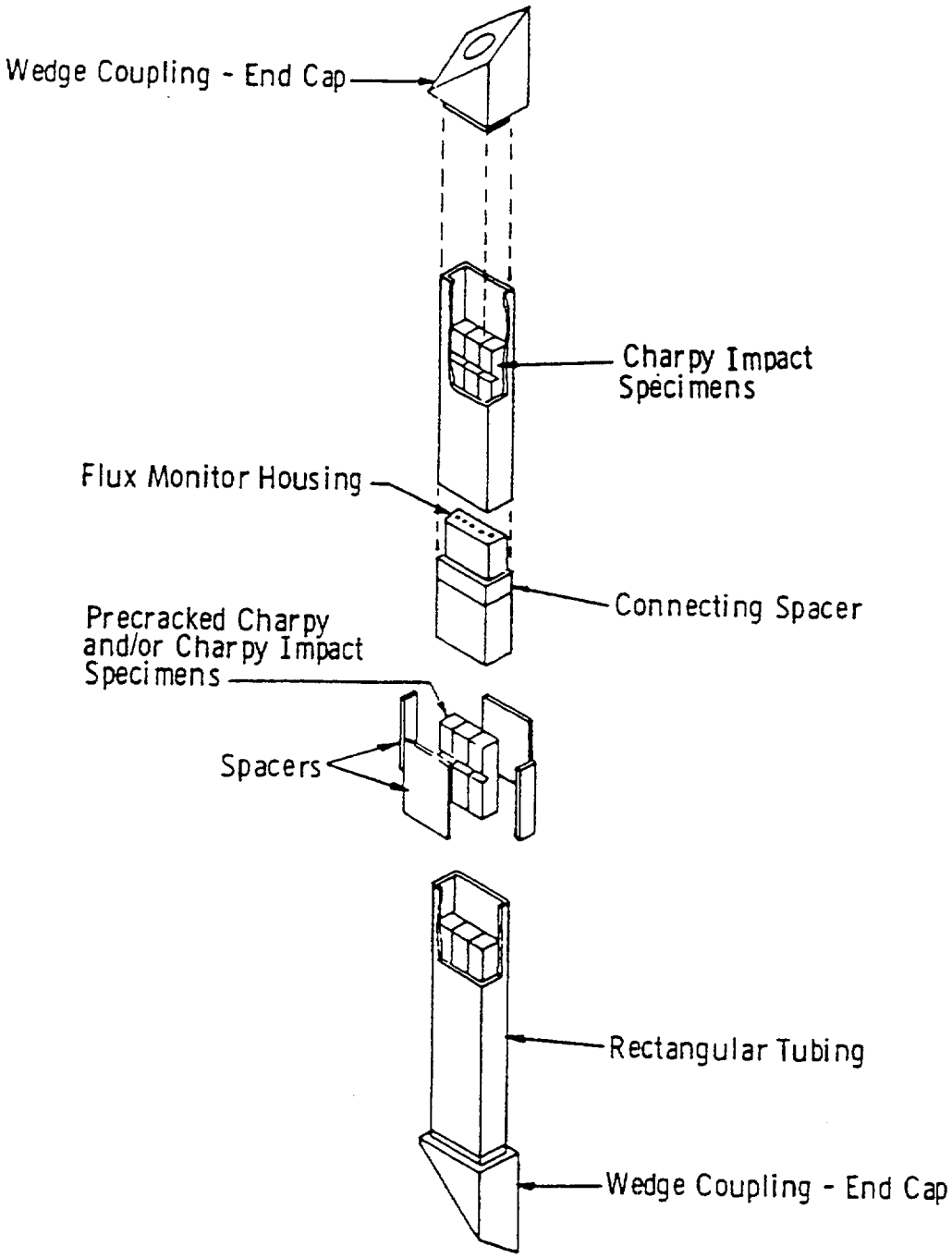


Figure 4-3 Typical Palo Verde Unit 1 Surveillance Capsule Charpy Impact Compartment Assembly

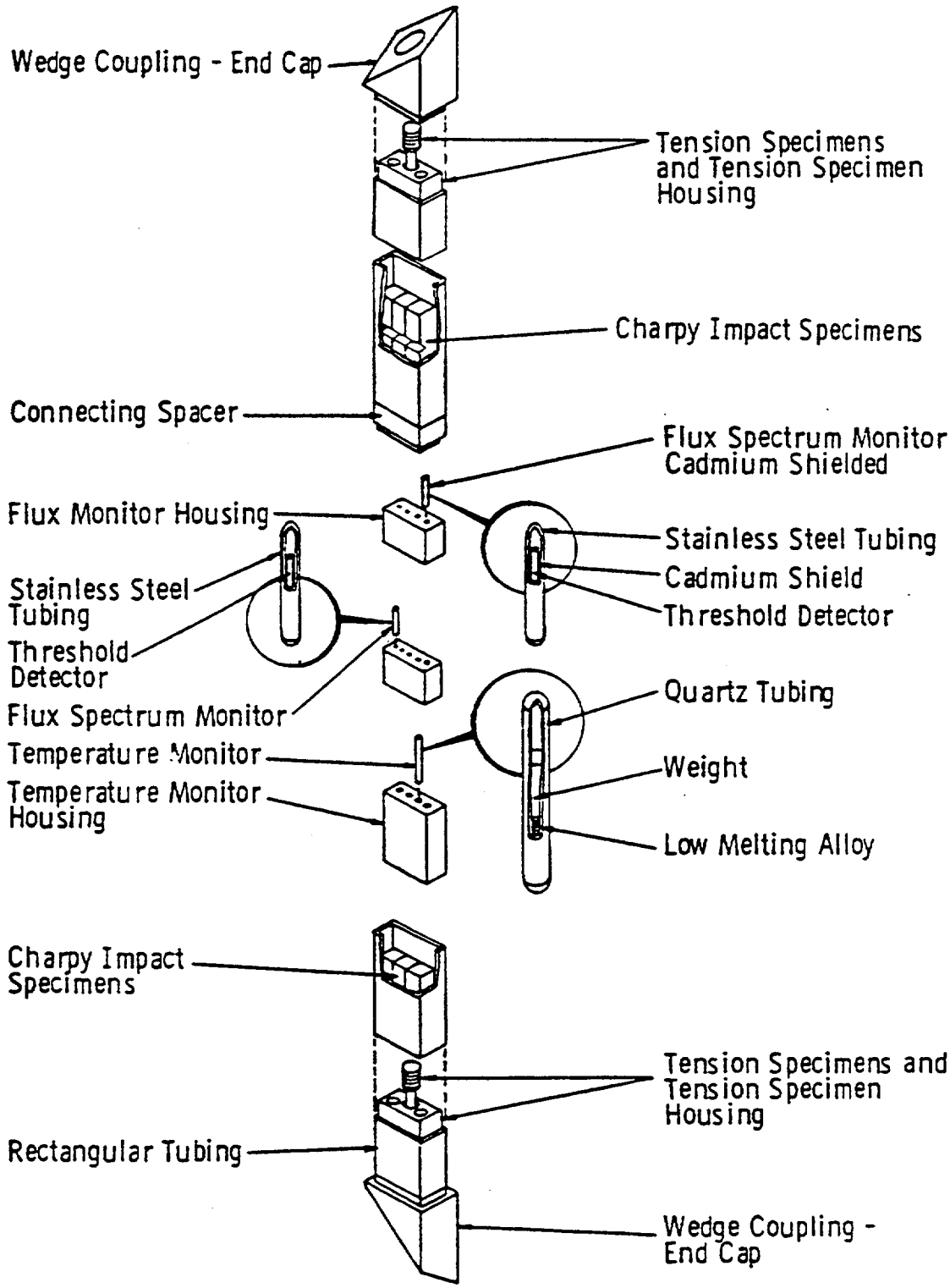


Figure 4-4 Typical Palo Verde Unit 1 Surveillance Capsule Tensile and Flux-Monitor Compartment Assembly

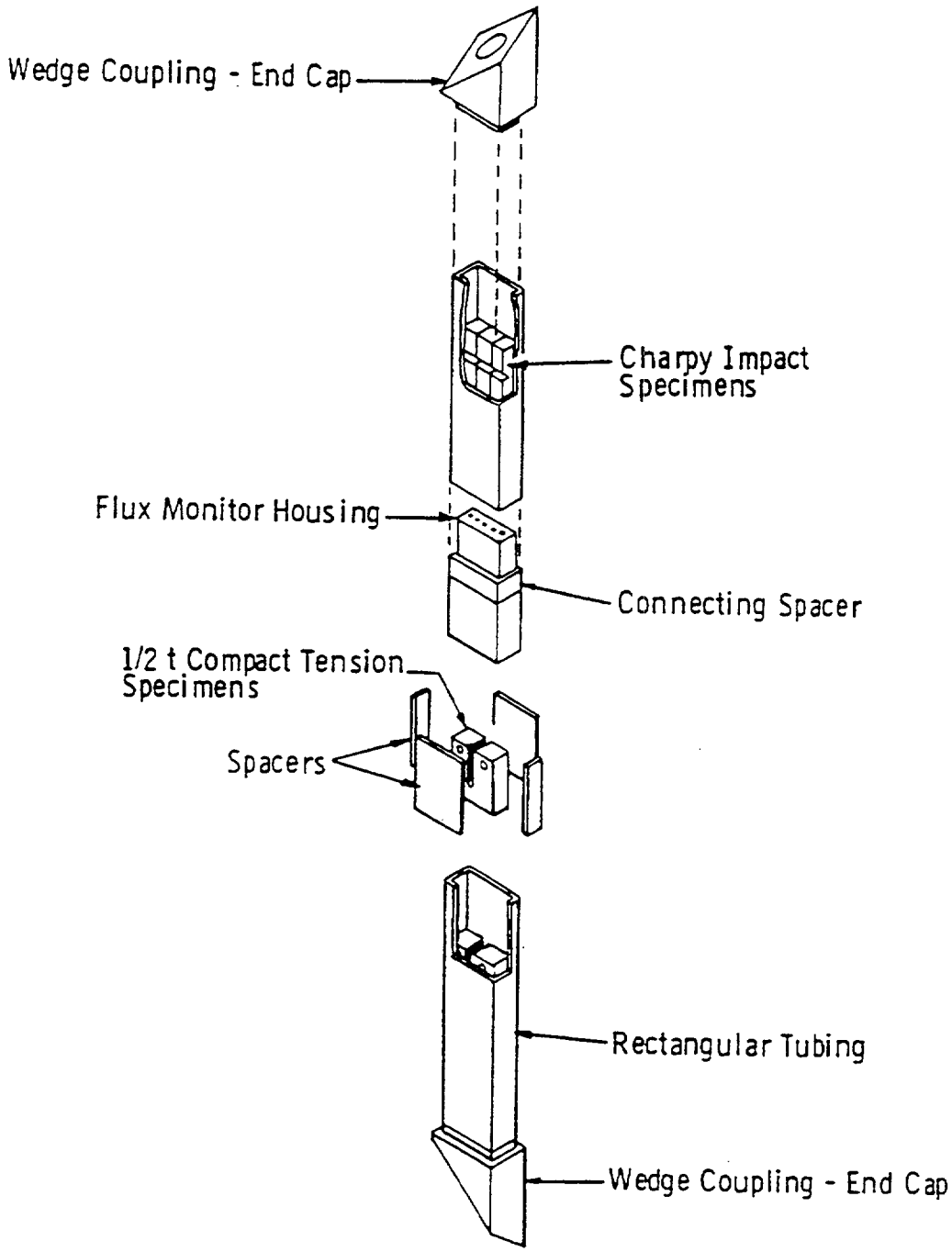


Figure 4-5 Typical Palo Verde Unit 1 Surveillance Capsule Charpy Flux and Compact Tension Compartment Assembly

5 TESTING OF SPECIMENS FROM CAPSULE 38°

5.1 OVERVIEW

The post-irradiation mechanical testing of the Charpy V-notch impact specimens and tensile specimens was performed in the Remote Metallographic Facility (RMF) at the Westinghouse Science and Technology Center. Testing was performed in accordance with 10CFR50, Appendices G and H[2], ASTM Specification E185-82[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 WCAP-14066[3]. No discrepancies were found.

Examination of the four low-melting, eutectic alloy thermal monitors indicated that the two lowest melting point monitors melted, and that the 580°F monitor had signs that some melting had occurred. Based on this examination, the maximum temperature to which the test specimens were exposed to was 580°F.

The Charpy impact tests were performed per ASTM Specification 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 a GRC 930-I instrumentation system, feeding information into an IBM compatible computer. With this system, load-time and energy-time signals can be recorded in addition to the standard measurement of Charpy energy (E_D). From the load-time curve (Appendix A), the load of general yielding (P_{GY}), the time to general yielding (t_{GY}), the maximum load (P_M), and the time to maximum load (t_M) can be determined. Under some test conditions, a sharp drop in load indicative of fast fracture was observed. The load at which fast fracture was initiated is identified as the fast fracture load (P_F), and the load at which fast fracture terminated is identified as the arrest load (P_A). The energy at maximum load (E_M) was determined by comparing the energy-time record and the load-time record. The energy at maximum load is approximately equivalent to the energy required to initiate a crack in the specimen. Therefore, the propagation energy for the crack (E_p) is the difference between the total energy to fracture (E_D) and the energy at maximum load (E_M).

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

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

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

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

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

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

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

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

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

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

Percent shear was determined from post-fracture photographs using the ratio-of-areas methods in compliance with ASTM Specification 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 Specification E8-99[10] and E21-92[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 crosshead speed of 0.05 inches per minute throughout the test.

Extension measurements were made with a linear variable displacement transducer extensometer. The extensometer knife edges were spring-loaded to the specimen and operated through specimen failure. The extensometer gage length was 1.00 inch. The extensometer is rated as Class B-2 per ASTM E83-93[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 griper. In the test configuration, with a slight load on the specimen, a plot of specimen temperature versus upper and lower tensile machine griper and controller temperatures was developed over the range from room temperature to 550°F. During the actual testing, the grip temperatures were used to obtain desired specimen temperatures. Experiments have indicated that this method is accurate to $\pm$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 38°, which received a fluence of 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV) in 9.81 EFPY of operation, are presented in Tables 5-1 through 5-8 and are compared with unirradiated results as shown in Figures 5-1 through 5-12.

The transition temperature increases and upper shelf energy decreases for the capsule 38° materials are summarized in Table 5-9. These results led to the following conclusions:

Irradiation of the reactor vessel intermediate shell plate M-6701-2 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation), to 7.85 x 10^{18} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 0.57°F and a 50 ft-lb transition temperature increase of 5.9°F. This results in an irradiated 30 ft-lb transition temperature of 8.72°F and an irradiated 50 ft-lb transition temperature of 42.52°F for the longitudinally oriented specimens

Irradiation of the reactor vessel intermediate shell plate M-6701-2 Charpy specimens, oriented with the longitudinal axis of the specimen normal to the major working direction of the plate (transverse orientation), to 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV) resulted in a 30 ft-lb transition temperature decrease of -19.71°F and a 50 ft-lb transition temperature decrease of -15.98°F. This results in an irradiated 30 ft-lb transition temperature of 10.14°F and an irradiated 50 ft-lb transition temperature of 55.25°F for transversely oriented specimens.

Irradiation of the weld metal Charpy specimens to 7.85 x 10^{18} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 6.27°F and a 50 ft-lb transition temperature increase of 8.12°F. This results in an irradiated 30 ft-lb transition temperature of –47.0°F and an irradiated 50 ft-lb transition temperature of -25.3°F.

Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature decrease of -26.59°F and a 50 ft-lb transition temperature decrease of -13.43°F. This results in an irradiated 30 ft-lb transition temperature of -89.79°F and an irradiated 50 ft-lb transition temperature of -40.01°F.

Irradiation of the standard reference material Charpy specimens to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 114.25°F and a 50 ft-lb transition temperature increase of 128.6°F. This results in an irradiated 30 ft-lb transition temperature of 136.16°F and an irradiated 50 ft-lb transition temperature of 176.21°F.

The average upper shelf energy of the intermediate shell plate M-6701-2 (longitudinal orientation) resulted in an average energy decrease of 10 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV). This results in an irradiated average upper shelf energy of 141 ft-lb for the longitudinally oriented specimens.

The average upper shelf energy of the Intermediate shell plate M-6701-2 (transverse orientation) resulted in no average energy decrease of after irradiation to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV). This results in an irradiated average upper shelf energy of 98 ft-lb for the transversely oriented specimens.

The average upper shelf energy of the weld metal Charpy specimens resulted an average energy decrease of 6 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 158 ft-lb for the weld metal specimens.

The average upper shelf energy of the weld metal Charpy specimens resulted an average energy decrease of 6 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 158 ft-lb for the weld metal specimens.

The average upper shelf energy of the weld HAZ metal Charpy specimens resulted an average energy decrease of 16 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 119 ft-lb for the weld HAZ metal.

The average upper shelf energy of the standard reference material Charpy specimens resulted an average energy decrease of 24 ft-lb after irradiation to 7.85 x 10^{18} n/cm^2 (E > 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 105 ft-lb for the standard reference material.

A comparison of the Palo Verde Unit 1 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 materials (Weld and Plate) for capsule 38° are less than the Regulatory Guide 1.99, Revision 2, predictions.

- The measured percent decrease in upper shelf energy of the Capsule 38° surveillance material is less than the Regulatory Guide 1.99, Revision 2, predictions.

The fracture appearance of each irradiated Charpy specimen from the various surveillance capsule 38° materials is shown in Figures 5-13 through 5-16 and show an increasingly ductile or tougher appearance with increasing test temperature.

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

The Charpy V-notch data presented in this report is based on a re-plot of all capsule data using CVGRAPH, Version 4.1, which is a hyperbolic tangent curve-fitting program. Hence, Appendix C contains a comparison of the Charpy V-notch shift results for each surveillance material (hand-fitting versus hyperbolic tangent curve-fitting). Additionally, Appendix B presents the CVGRAPH, Version 4.1, Charpy V-notch plots and the program input data.

5.3 TENSILE TEST RESULTS

The results of the tensile tests performed on the various materials contained in capsule 38° irradiated to 7.85×10^{18} n/cm^2 (E > 1.0 MeV) are presented in Table 5-13 and are compared with unirradiated results as shown in Figures 5-21 and 5-22.

The results of the tensile tests performed on the intermediate shell plate M-6701-2 (transverse orientation) indicated that irradiation to 7.85×10^{18} n/cm^2 (E> 1.0 MeV) caused an approximate increase of 0 to 5 ksi in the 0.2 percent offset yield strength and approximately a 0 to 2 ksi increase in the ultimate tensile strength when compared to unirradiated data[1] (Figure 5-21).

The results of the tensile tests performed on the surveillance weld metal indicated that irradiation to 7.85×10^{18} n/cm^2 (E > 1.0 MeV) caused a 5 ksi increase in the 0.2 percent offset yield strength and a 4 to 10 ksi increase in the ultimate tensile strength when compared to unirradiated data (Figure 5-22).

The fractured tensile specimens for the Intermediate shell plate M-6701-2 material are shown in Figure 5-23, while the fractured tensile specimens for the surveillance weld metal and heat-affected-zone material are shown in Figure 5-24.

The engineering stress-strain curves for the tensile tests are shown in Figures 5-25 and 5-26.

Table 5-1 Charpy V-notch Data for the Palo Verde Unit 1 Intermediate Shell Plate M-6701-2 Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV) (Longitudinal Orientation)

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
1A11T	-75	-59	5	7	1	0.03	5
1A127	0	-18	19	26	12	0.30	15
1A112	20	-7	38	52	31	0.79	25
1A125	30	-1	43	58	39	0.99	25
1A13U	50	10	60	81	44	1.12	35
1A111	100	38	91	123	66	1.68	65
1A122	150	66	116	157	74	1.88	85
1A144	225	107	138	187	60	1.52	100
1A13K	275	135	144	195	106	2.69	100

Table 5-2 Charpy V-notch Data for the Palo Verde Unit 1 Intermediate Shell Plate M-6701-2 Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV) (Transverse Orientation)

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
1A255	-75	-59	4	5	5	0.13	2
1A21E	-40	-40	8	11	4	0.10	5
1A25P	0	-18	24	33	21	0.53	10
1A232	5	-15	39	53	38	0.97	15
1A21J	10	-12	39	53	30	0.76	15
1A23A	25	-4	37	50	30	0.76	20
1A25E	50	10	38	52	28	0.71	25
1A25U	50	10	47	64	38	0.97	30
1A261	70	21	55	75	42	1.07	45
1A222	80	27	65	88	51	1.30	50
1A247	125	52	89	121	71	1.80	60
1A21M	150	66	110	149	79	2.01	90
1A256	150	66	62	84	51	1.30	60
1A235	200	93	112	152	84	2.13	100
1A263	250	121	118	160	78	1.98	100

Table 5-3 Charpy V-notch Data for the Palo Verde Unit 1 Surveillance Weld Metal Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV)

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
1A31Y	-96	-71	7	9	7	0.18	10
1A354	-70	-57	19	26	17	0.43	20
1A324	-55	-48	11	15	14	0.36	20
1A3A2	-50	-46	34	46	26	0.66	25
1A3B3	-45	-43	25	34	22	0.56	25
1A372	-25	-32	48	65	37	0.94	40
1A33K	-10	-23	95	129	69	1.75	70
1A32U	0	-18	65	88	52	1.32	60
1A342	15	-9	96	130	69	1.75	70
1A35E	25	-4	114	155	71	1.80	85
1A32M	50	10	129	175	85	2.16	90
1A323	100	38	149	202	90	2.29	100
1A35U	150	66	163	221	92	2.34	100
1A331	200	93	151	205	90	2.29	100
1A33B	250	121	170	231	97	2.46	100

Table 5-4 Charpy V-notch Data for the Palo Verde Unit 1 Heat Affected Zone Metal Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV)

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
1A441	-175	-115	12	16	4	0.10	5
1A442	-120	-84	23	31	12	0.30	10
1A43D	-90	-68	54	73	39	0.99	45
1A41U	-75	-59	53	72	33	0.84	40
1A44Y	-50	-46	18	24	16	0.41	30
1A453	-30	-34	30	41	24	0.61	40
1A416	-25	-32	47	64	46	1.17	50
1A443	0	-18	103	140	64	1.63	85
1A44D	25	-4	62	84	48	1.22	50
1A41M	70	21	108	146	75	1.91	100
1A42B	130	54	100	136	70	1.78	100
1A43T	200	93	148	201	90	2.29	100

Table 5-5 Charpy V-notch Data for the Palo Verde Unit 1 Standard Reference Material Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E> 1.0 MeV)

Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
1AB6M	0	-18	4	5	1	0.03	5
1AB56	100	38	28	38	14	0.36	10
1AB4A	125	52	30	41	28	0.71	15
1AB53	175	79	42	57	42	1.07	25
1AB6P	200	93	45	61	40	1.02	45
1AB5K	225	107	91	123	90	2.29	95
1AB4B	250	121	88	119	72	1.83	90
1AB44	325	163	110	149	71	1.80	100
1AB45	375	191	100	136	82	2.08	100

Table 5-6 Instrumented Charpy Impact Test Results for the Palo Verde Unit 1 Intermediate Shell Plate M-6701-2 Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E>1.0 MeV) (Longitudinal Orientation)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. T_m (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_p/A								
1A11T	-75	5	40	19	21	2216	0.15	2228	0.14	2216	0	74	74
1A127	0	19	153	65	88	3565	0.17	4077	0.23	4077	0	119	127
1A112	20	38	306	212	94	3925	0.18	4503	0.49	4497	447	131	140
1A125	30	43	346	273	74	3857	0.18	4454	0.60	4452	0	128	138
1A13U	50	60	483	293	191	3202	0.17	4040	0.71	3996	518	107	121
1A111	100	91	733	320	414	3750	0.18	4566	0.69	3913	1091	125	138
1A122	150	116	935	308	626	3582	0.18	4401	0.69	3578	2280	119	133
1A144	225	138	1112	314	798	2686	0.17	3625	0.84	n/a	n/a	89	105
1A13K	275	144	1160	353	808	3166	0.18	4109	0.83	n/a	n/a	105	121

Table 5-7 Instrumented Charpy Impact Test Results for the Palo Verde Unit 1 Intermediate Shell Plate M-6701-2 Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E>1.0 MeV) (Transverse Orientation)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. T_m (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_p/A								
1A255	-75	4	32	16	16	2076	0.12	2088	0.13	2076	0	69	69
1A21E	-40	8	64	36	29	3390	0.17	3390	0.17	3390	0	113	113
1A25P	0	24	193	138	55	3412	0.18	3683	0.4	3681	0	114	118
1A232	5	39	314	224	90	3739	0.18	4221	0.53	4215	0	125	133
1A21J	10	39	314	240	75	3748	0.17	4475	0.54	4416	0	125	137
1A23A	25	37	298	198	100	3206	0.17	3742	0.53	3658	0	107	116
1A25E	50	38	306	200	106	3369	0.17	3997	0.51	3963	580	112	123
1A25U	50	47	379	266	113	3101	0.17	3754	0.68	3574	450	103	114
1A261	70	55	443	290	153	3379	0.18	4089	0.7	4072	1096	113	124
1A222	80	65	524	273	250	3219	0.17	3905	0.68	3821	1304	107	119
1A247	125	89	717	297	420	3452	0.18	4237	0.69	3848	1597	115	128
1A21M	150	110	886	306	580	3448	0.17	4326	0.69	3043	2071	115	129
1A256	150	62	500	259	240	2944	0.17	3684	0.69	3582	1313	98	110
1A235	200	112	902	252	651	2797	0.17	3585	0.69	n/a	n/a	93	106
1A263	250	118	951	302	648	2534	0.17	3483	0.84	n/a	n/a	84	100

Table 5-8 Instrumented Charpy Impact Test Results for the Palo Verde Unit 1 Surveillance Weld Metal Irradiated to a Fluence of 7.85×10^{18} n/cm^2 (E>1.0 MeV)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. T_m (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_p/A								
1A31Y	-96	7	56	32	25	3362	0.16	3362	0.16	3362	0	112	112
1A354	-70	19	153	71	82	4128	0.18	4478	0.23	4362	0	137	143
1A324	-55	11	89	40	48	3838	0.17	3845	0.17	3838	0	128	128
1A3A2	-50	34	274	192	82	3870	0.18	4168	0.47	4164	0	129	134
1A3B3	-45	25	201	66	135	4119	0.17	4427	0.22	4231	0	137	142
1A372	-25	48	387	210	176	3686	0.17	4058	0.51	3979	924	123	129
1A33K	-10	95	765	308	457	3790	0.17	4217	0.69	3585	1218	126	133
1A32U	0	65	524	216	308	3555	0.17	4087	0.53	4071	1422	118	127
1A342	15	96	774	320	453	3926	0.18	4323	0.70	3566	1239	131	137
1A35E	25	114	919	329	589	3925	0.17	4482	0.70	3480	1822	131	140
1A32M	50	129	1039	332	707	3147	0.17	3682	0.83	2798	1728	105	114
1A323	100	149	1201	274	927	3407	0.2	3833	0.68	n/a	n/a	113	121
1A35U	150	163	1313	275	1039	3170	0.17	3799	0.69	n/a	n/a	106	116
1A331	200	151	1217	269	948	3132	0.17	3807	0.68	n/a	n/a	104	116
1A33B	250	170	1370	343	1026	3173	0.18	3816	0.84	n/a	n/a	106	116

Table 5-9 Instrumented Charpy Impact Test Results for the Palo Verde Unit 1 Heat Affected Zone Material Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E>1.0 MeV)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. T_m (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_p/A								
1A441	-175	12	97	55	42	4205	0.18	4372	0.2	4366	0	140	143
1A442	-120	23	185	70	115	4217	0.17	4751	0.22	4474	0	140	149
1A43D	-90	54	435	251	184	4413	0.17	4914	0.51	4862	753	147	155
1A41U	-75	53	427	228	199	3862	0.17	4336	0.52	4167	0	129	136
1A44Y	-50	18	145	62	83	4015	0.17	4220	0.21	3976	0	134	137
1A453	-30	30	242	58	184	3478	0.17	3814	0.22	3677	458	116	121
1A416	-25	47	379	223	156	3938	0.17	4347	0.51	4230	1046	131	138
1A443	0	103	830	375	455	4244	0.18	5340	0.70	4541	1555	141	160
1A44D	25	62	500	197	302	3479	0.18	3705	0.52	3570	1888	116	120
1A41M	70	108	870	266	604	3364	0.17	3746	0.67	n/a	n/a	112	118
1A42B	130	100	806	259	547	3228	0.17	3659	0.67	n/a	n/a	107	115
1A43T	200	148	1192	320	872	3704	0.17	4442	0.7	n/a	n/a	123	136

Table 5-10 Instrumented Charpy Impact Test Results for the Palo Verde Unit 1 Standard Reference Material Irradiated to a Fluence of 7.85 x 10^{18} n/cm^2 (E>1.0 MeV)

Sample No.	Test Temp. (°F)	Charpy Energy E_D (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load P_{GY} (lb)	Time to Yield t_{GY} (msec)	Max. Load P_M (lb)	Time to Max. T_m (msec)	Fast Fract. Load P_F (lb)	Arrest Load P_A (lb)	Yield Stress S_Y (ksi)	Flow Stress (ksi)
			Charpy E_D/A	Max. E_M/A	Prop. E_p/A								
1AB6M	0	4	32	8	24	1207	0.11	1207	0.11	1207	0	40	40
1AB56	100	28	226	166	59	3648	0.18	4317	0.43	4315	0	121	133
1AB4A	125	30	242	170	72	3807	0.18	4257	0.42	4228	503	127	134
1AB53	175	42	338	207	132	3582	0.17	4319	0.50	4183	1327	119	132
1AB6P	200	45	363	222	141	3736	0.17	4515	0.51	4513	792	124	137
1AB5K	225	91	733	291	443	3466	0.18	4297	0.67	3190	2469	115	129
1AB4B	250	88	709	267	442	3056	0.17	3862	0.67	1261	575	102	115
1AB44	325	110	886	292	595	3470	0.17	4317	0.66	n/a	n/a	116	130
1AB45	375	100	806	283	523	3328	0.18	4113	0.67	n/a	n/a	111	124

Table 5-11 Effect of Irradiation to 7.85 x 10^{19} n/cm^2 (E>1.0 MeV) on the Notch Toughness Properties of the Palo Verde Unit 1 Reactor Vessel Surveillance Materials

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
Intermediate Shell Plate M-6701-2 (Longitudinal)	8.15	8.72	0.57	35.82	33.19	-2.63	36.62	42.52	5.9	151	141	-10
Intermediate Shell Plate M-6701-2 (Transverse)	29.86	10.14	-19.71	66.22	40.65	-25.56	71.24	5.25	-15.98	98	115	17
Weld Metal	-53.28	-47	6.27	-30.94	-31.79	-0.84	-33.43	-25.3	8.12	164	158	-6
HAZ Metal	-63.2	-89.79	-26.59	-29.74	-42.71	-12.97	-26.57	-40.01	-13.43	135	119	-16
SRM	21.9	136.16	114.25	47.48	153.84	106.35	47.61	176.21	128.6	129	105	-24

a. "Average" is defined as the value read from the curve fit through the data points of the Charpy tests (see Figures 5-1, 5-4, 5-7, 5-10 and 5-13).

b. "Average" is defined as the value read from the curve fit through the data points of the Charpy tests (see Figures 5-2, 5-5, 5-8, 5-11 and 5-14)

Table 5-12 Comparison of the Palo Verde Unit 1 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decrease with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence ($\times 10^{19}$ n/cm^2)[a]	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F)	Measured (°F)	Predicted (%)[b]	Measured (%)
Intermediate Shell Plate M-6701-2 (Longitudinal)	137°	0.433	16.02	33.93	16	15
	38°	0.785	29.05	0.57	18	7
Intermediate Shell Plate M-6701-2 (Transverse)	137°	0.433	16.02	13.55	16	11
	38°	0.785	29.05	-19.71	18	0
Surveillance Program Weld Metal	137°	0.433	22.8	-2.94	16	1
	38°	0.785	23.3	6.27	18	4
Heat Affected Zone Material	137°	0.433	--	-10.98	---	8
	38°	0.785	--	-26.59	---	12

Notes:

(a) Calculated Fluences from capsule 38° dosimetry analysis results (E > 1.0 MeV)

(b) From Figure 2 of Regulatory Guide 1.99, Revision 2, using the Cu wt. Percent and capsule fluence values.

Table 5-13 Tensile Specimens From Intermediate Shell Course Plate M-6701-2, Weld, and Heat Affected Zone Material

Sample Number	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 (%)
1A2JC	PLATE	50	61.1	85.7	3.01	165.8	61.3	15.1	29.6	63
1A2K2	PLATE	175	60.1	79.8	2.87	162.4	58.4	13.4	25.5	64
*1A2J5**	PLATE	*550*	*	*77.7*	*2.71*	*155.6*	*55.3*	*	*20.2*	*64*
1A3JC	WELD	-15	75.4	85.9	2.70	194.4	55.0	13.6	27.3	72
1A3J4	WELD	75	72.8	86.7	2.63	201.2	53.6	11.5	26.1	73
1A3J7	WELD	550	62.1	82.5	2.47	169.8	50.2	10.6	23.8	70

* NOTE: Testing difficulties make the yield strength and uniform elongation of specimen 1A2J5 invalid.

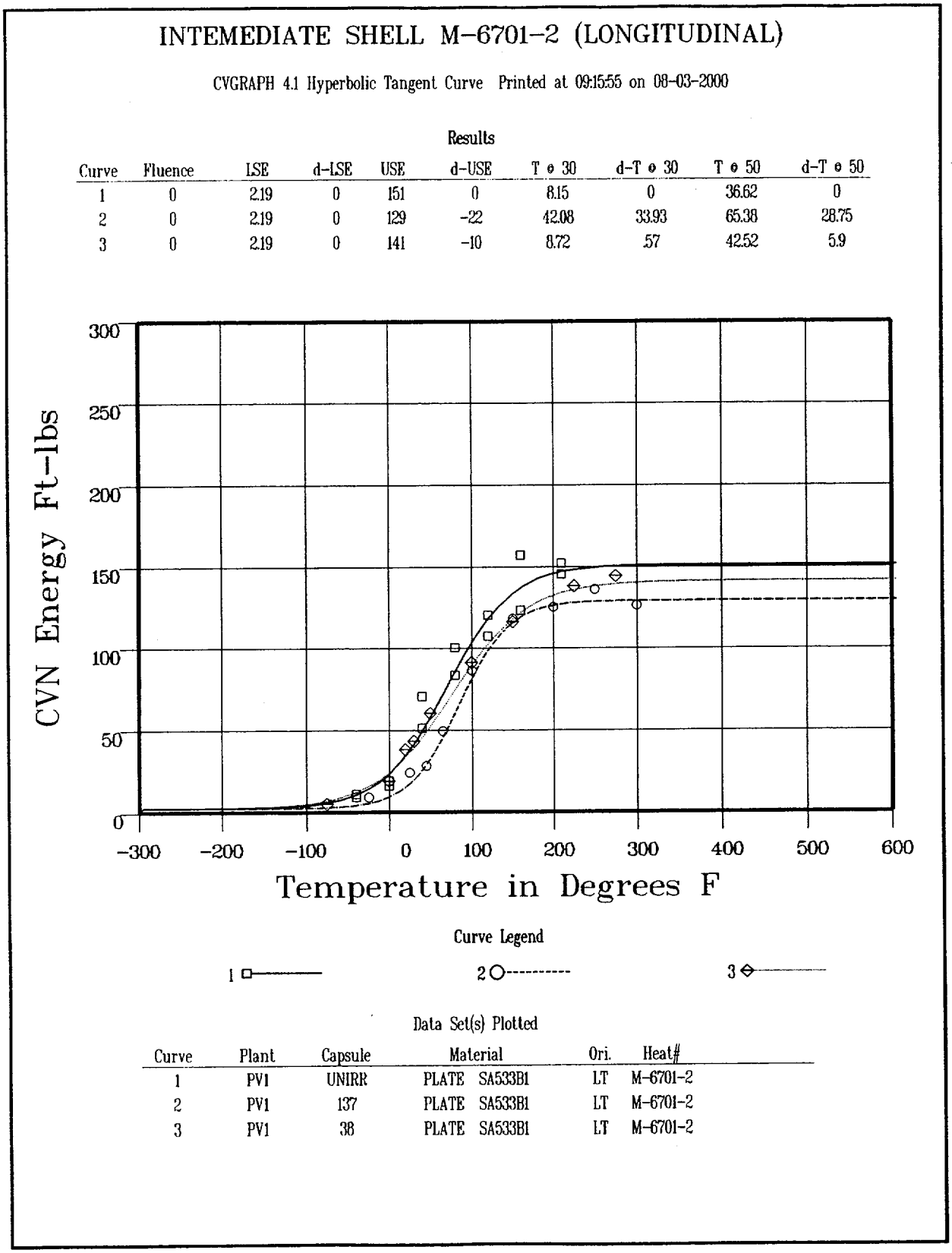


Figure 5-1 Charpy V-Notch Impact Energy vs. Temperature for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate M-6701-2 (Longitudinal Orientation)

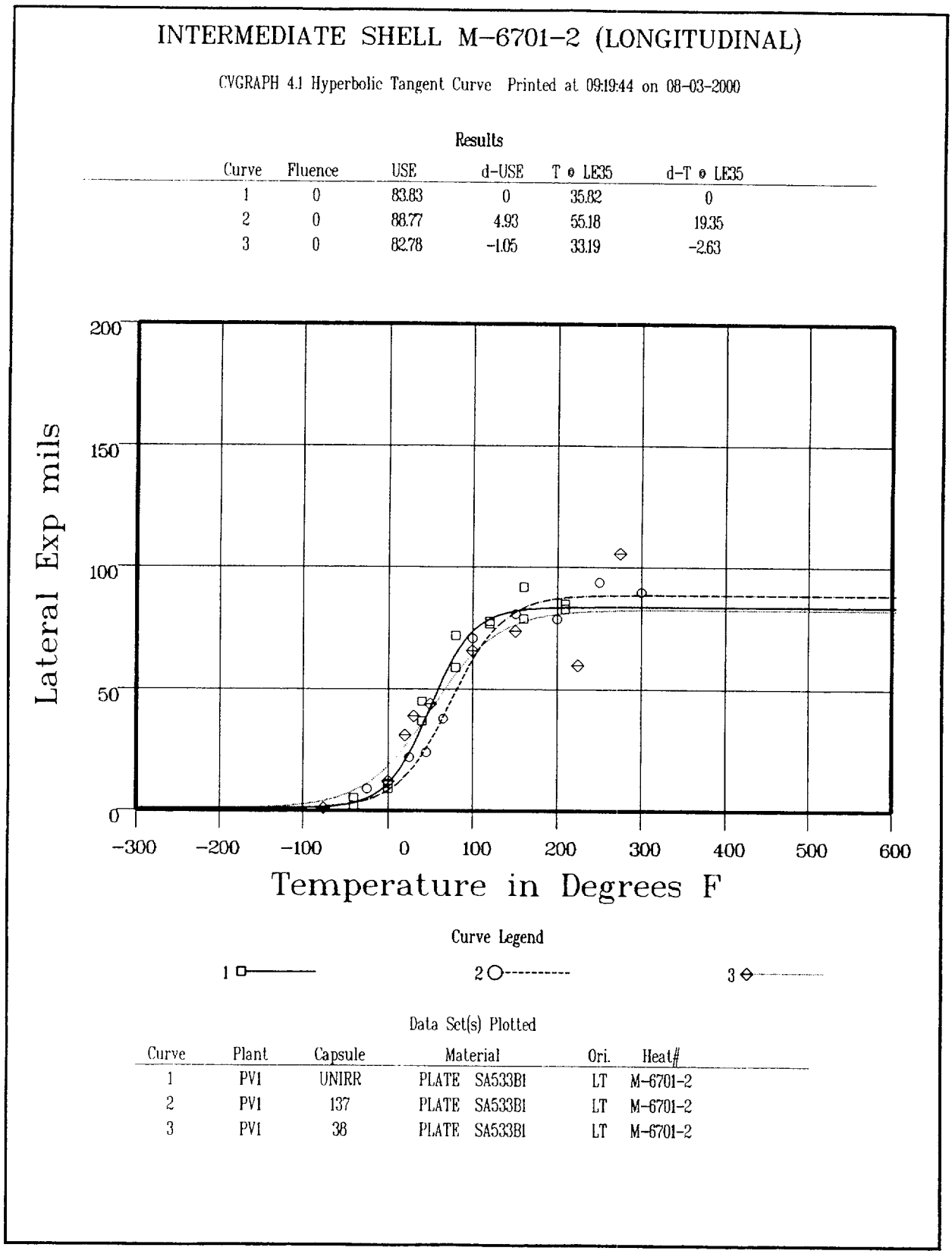


Figure 5-2 Charpy V-Notch Lateral Expansion vs. Temperature for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate C-8-2 (Longitudinal Orientation)

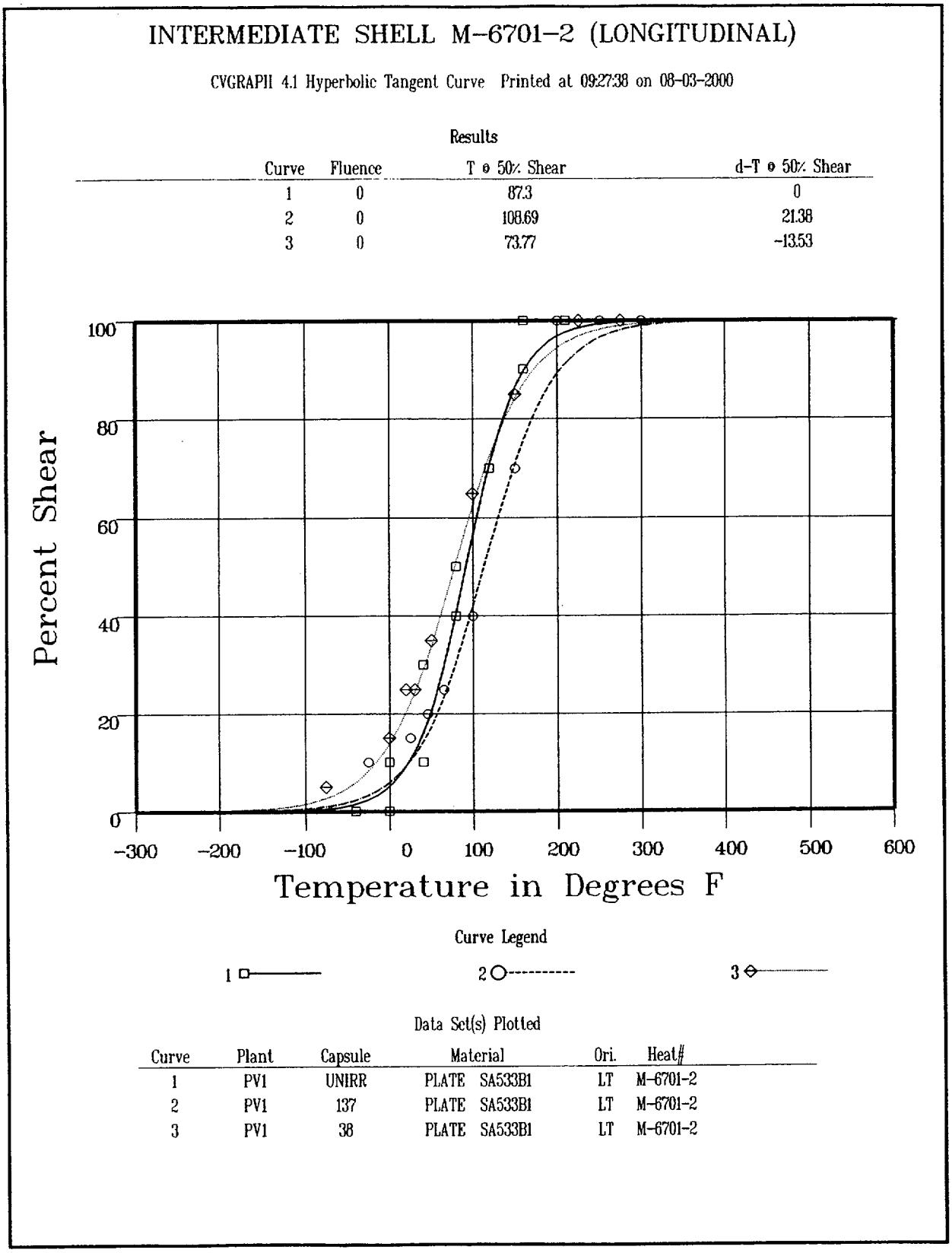


Figure 5-3 Charpy V-Notch Percent Shear vs. Temperature for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate M-6701-2 (Longitudinal Orientation)

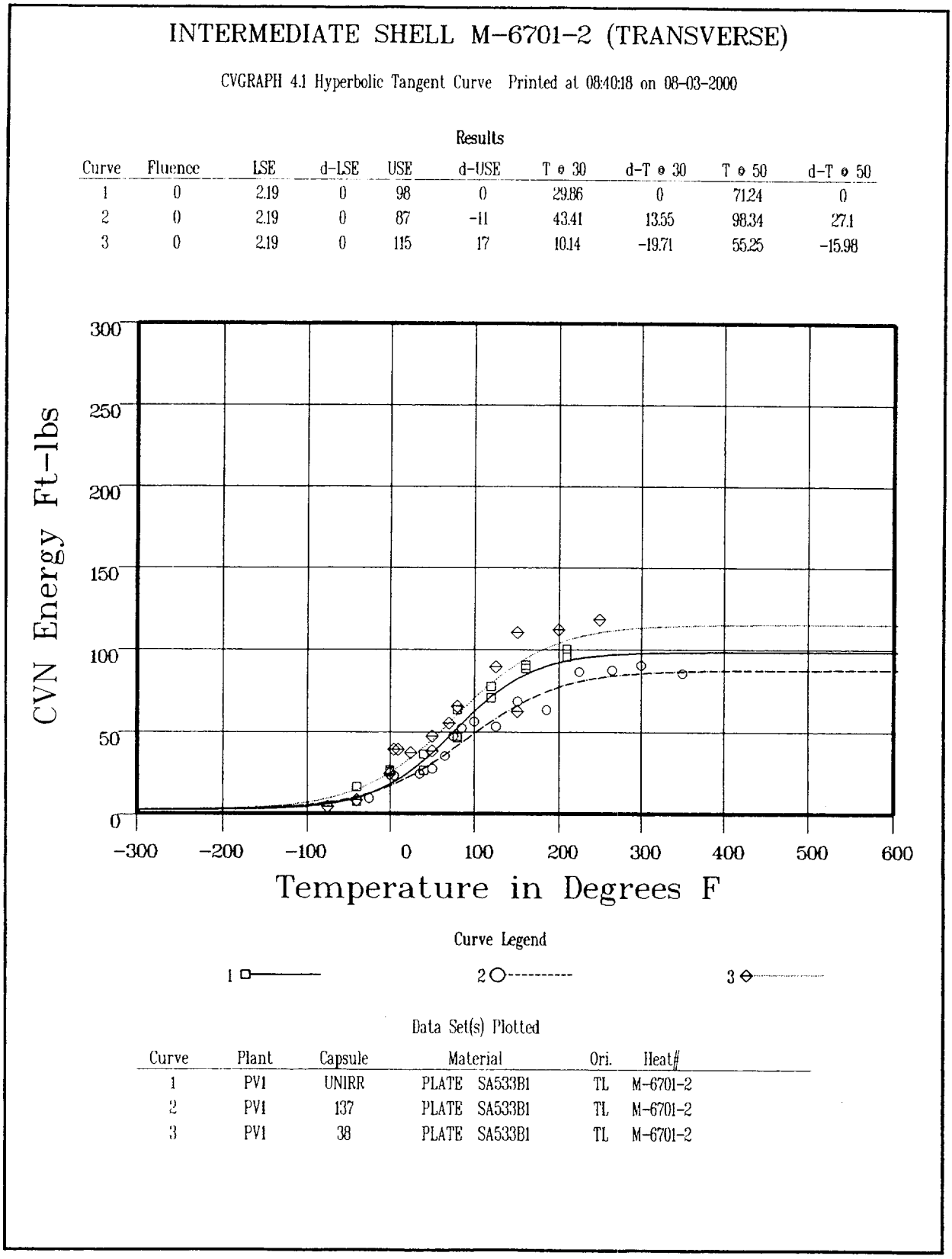


Figure 5-4 Charpy V-Notch Impact Energy vs. Temperature for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate M-6701-2 (Transverse Orientation)

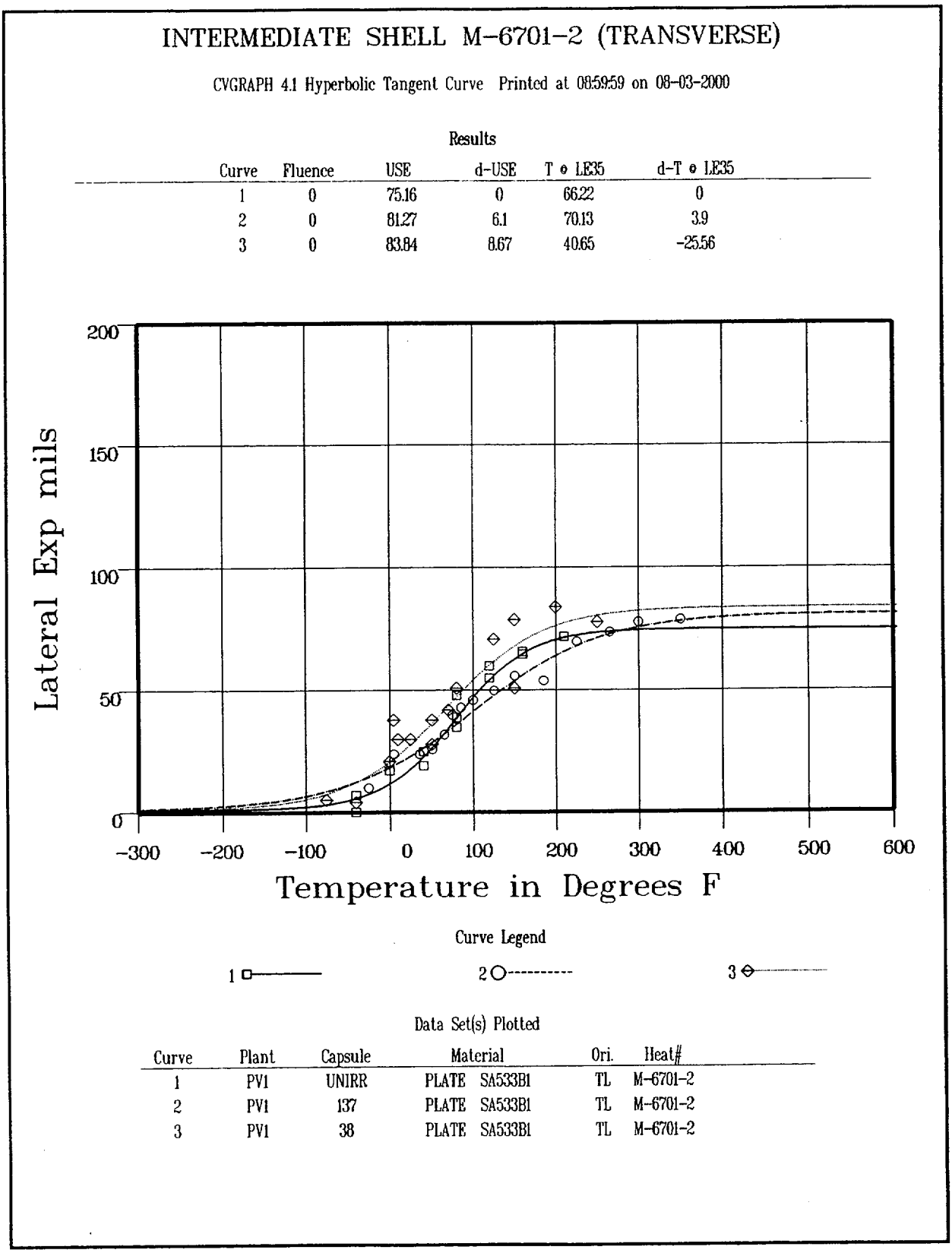


Figure 5-5 Charpy V-Notch Lateral Expansion vs. Temperature for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate C-8-2 (Transverse Orientation)

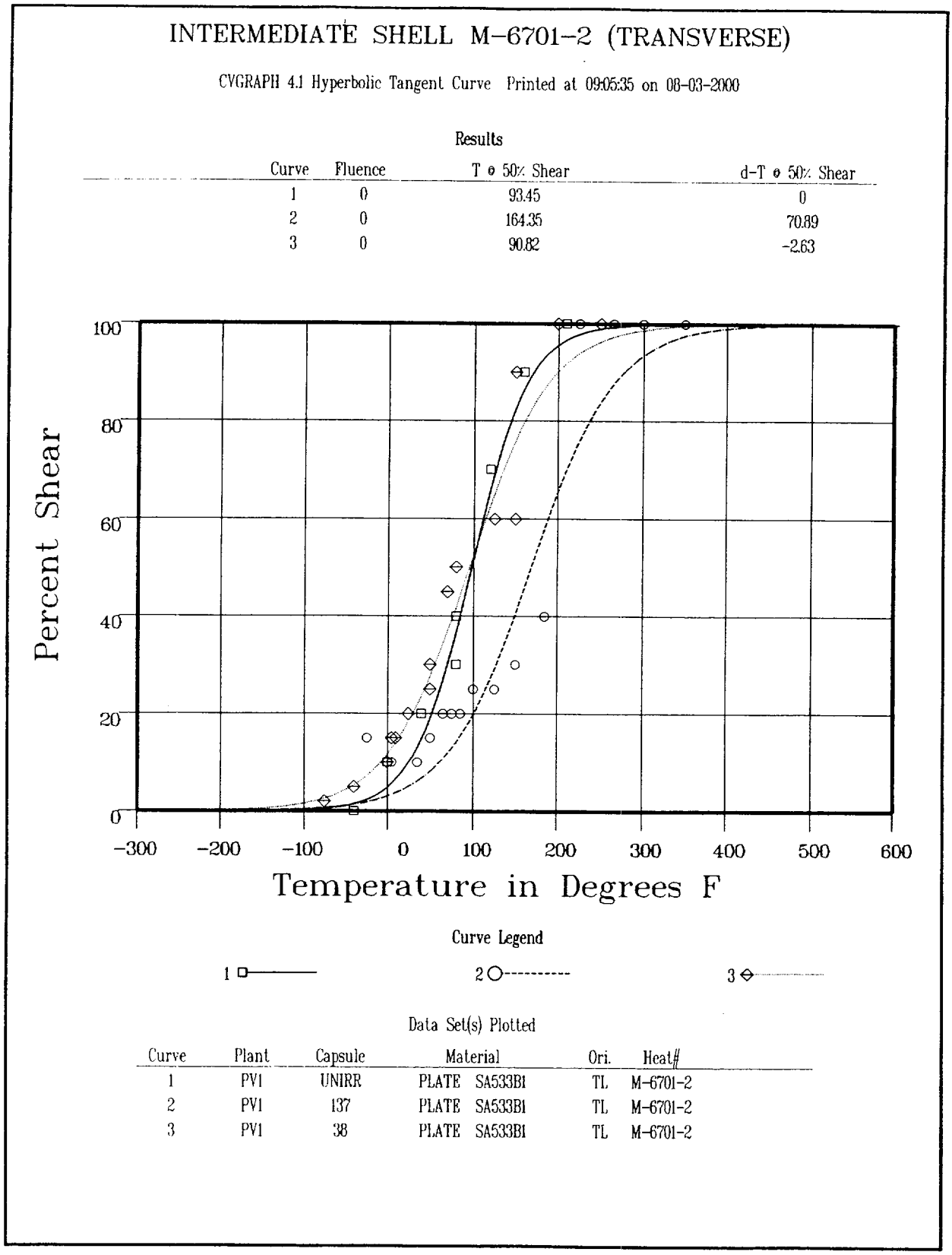


Figure 5-6 Charpy V-Notch Percent Shear vs. Temperature for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate M-6701-2 (Transverse Orientation)

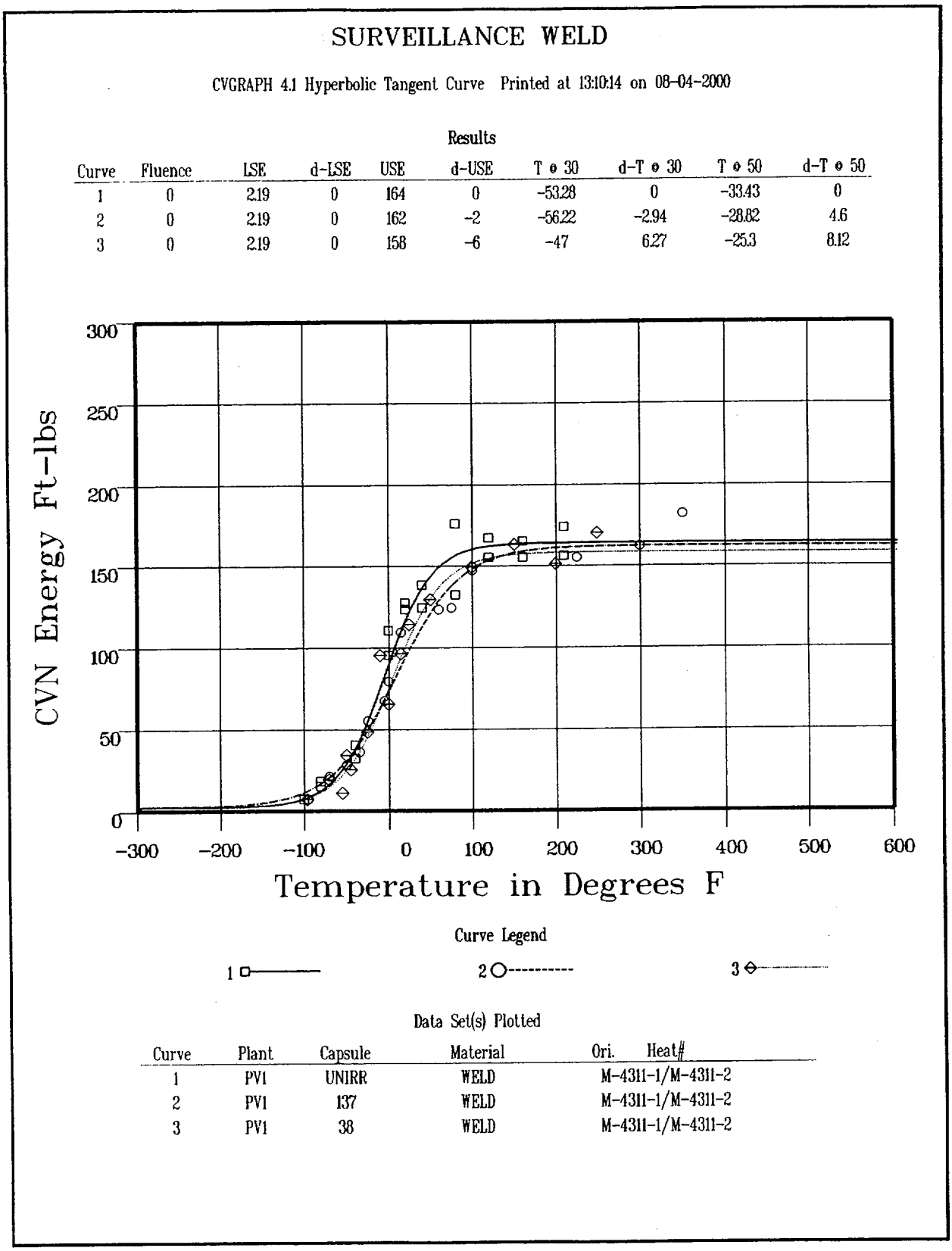


Figure 5-7 Charpy V-Notch Impact Energy vs. Temperature for Palo Verde Unit 1 Reactor Vessel Surveillance Weld Material

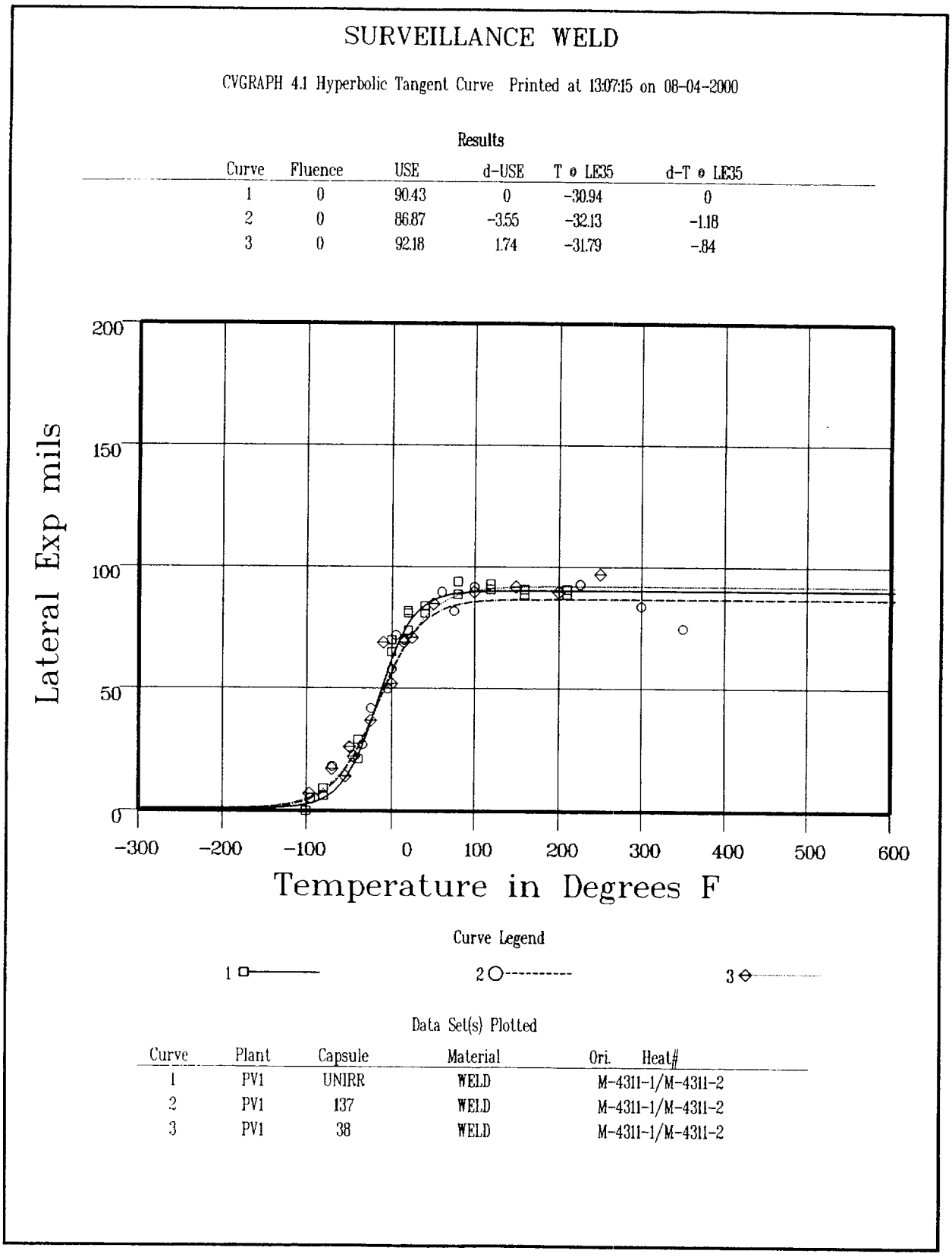


Figure 5-8 Charpy V-Notch Lateral Expansion vs. Temperature for Palo Verde Unit 1 Reactor Vessel Surveillance Weld Metal

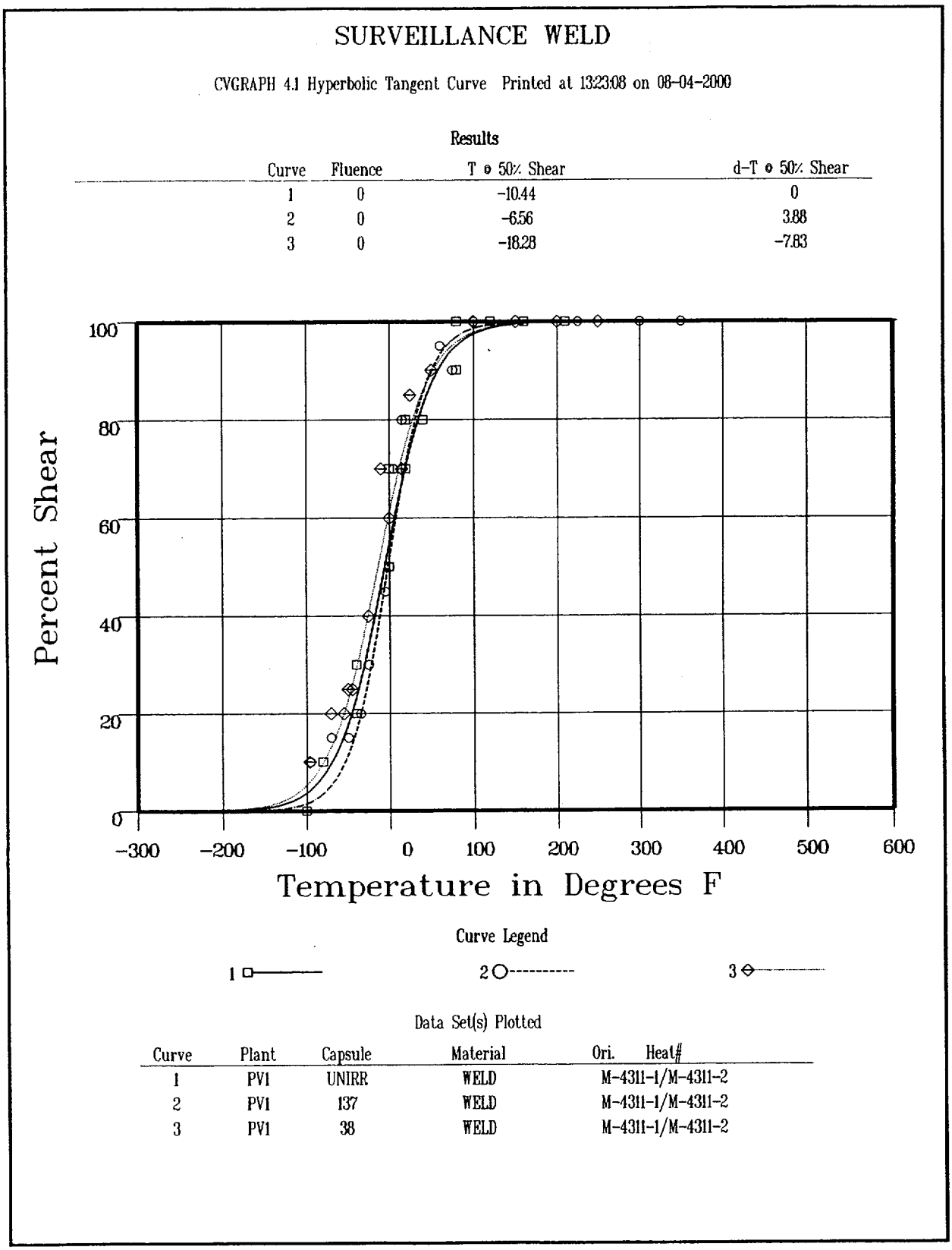


Figure 5-9 Charpy V-Notch Percent Shear vs. Temperature for Palo Verde Unit 1 Reactor Vessel Surveillance Weld Metal

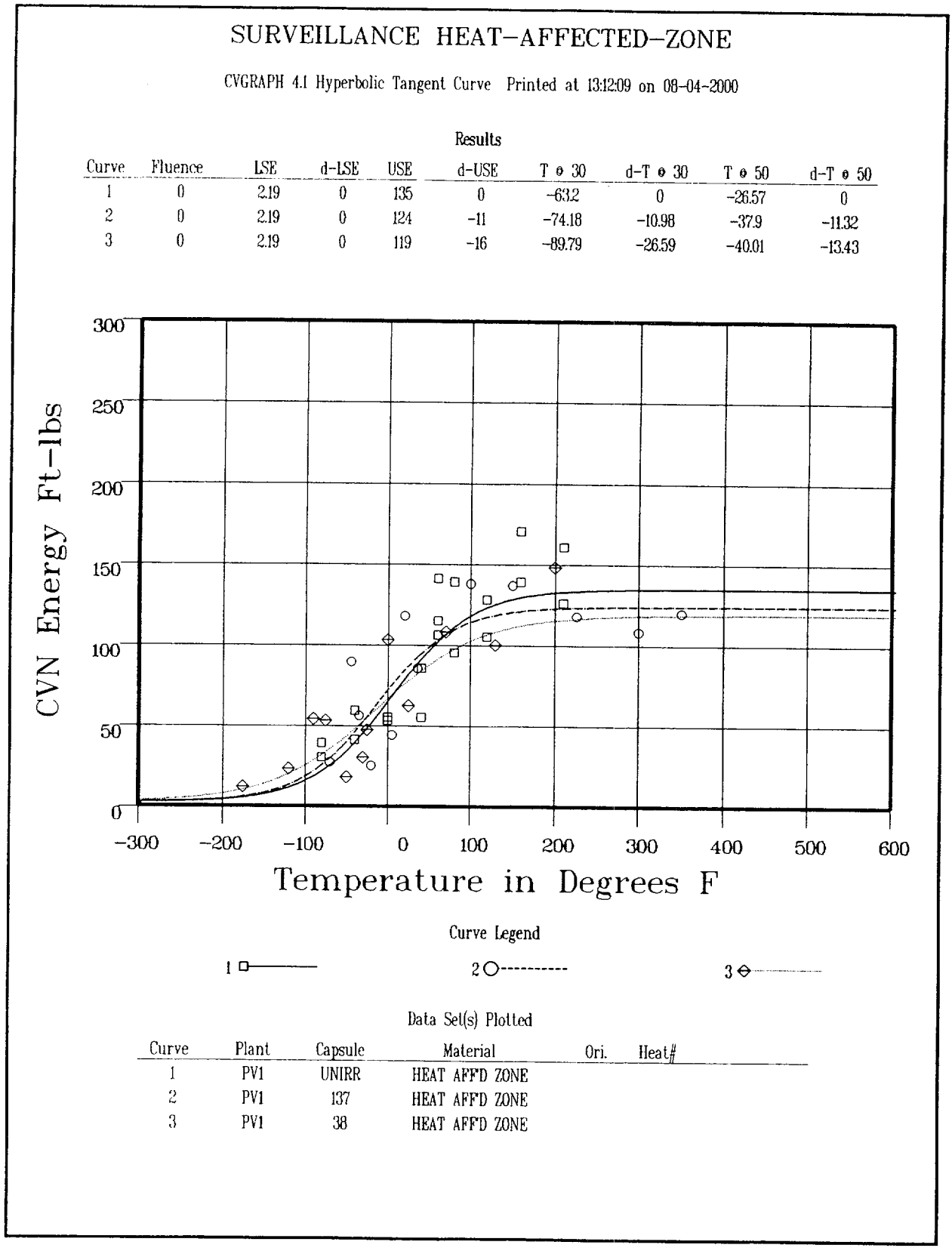


Figure 5-10 Charpy V-Notch Impact Energy vs. Temperature for Palo Verde Unit 1 Reactor Vessel Heat Affected Zone Material

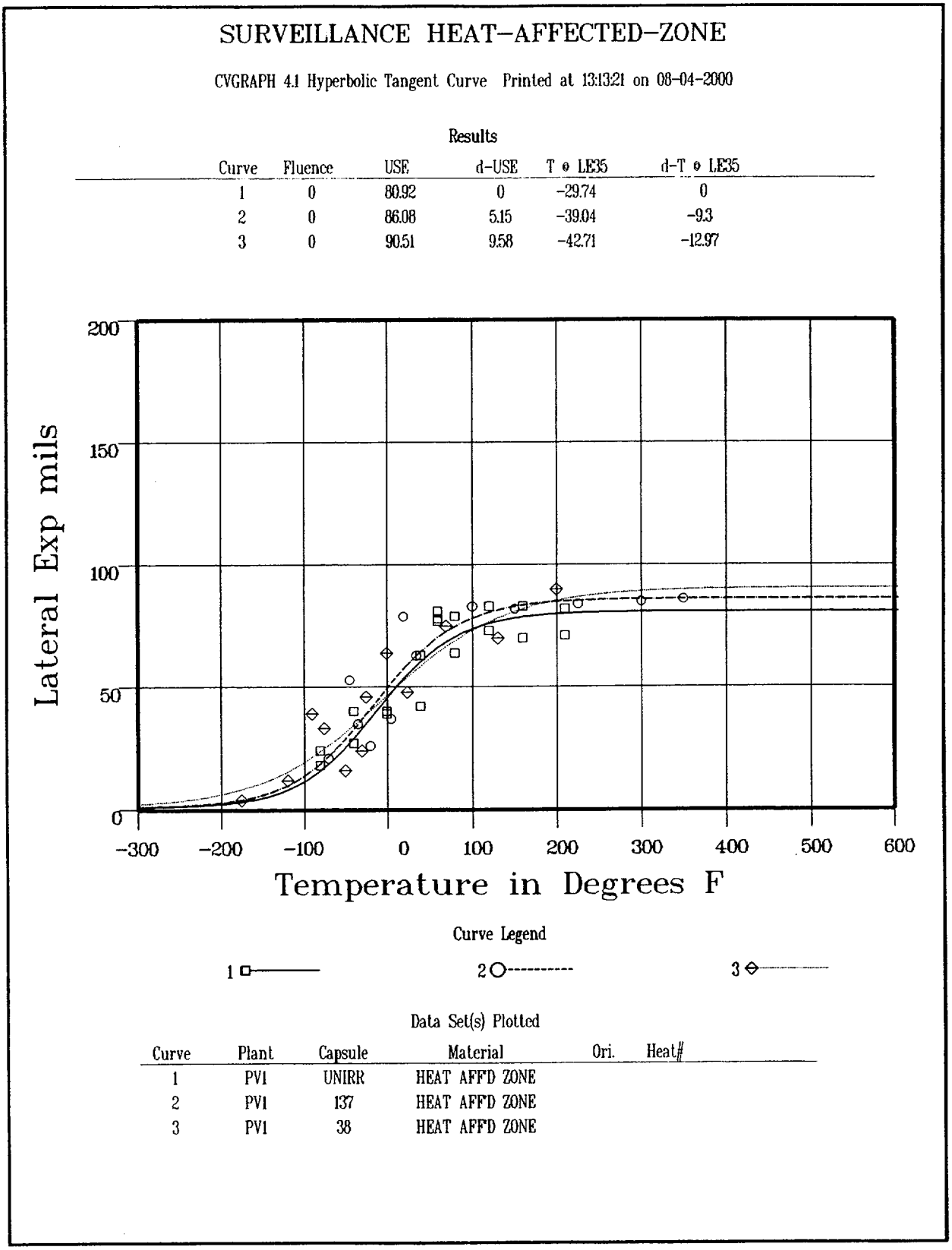


Figure 5-11 Charpy V-Notch Lateral Expansion vs. Temperature for Palo Verde Unit 1 Reactor Vessel Heat Affected Zone Material

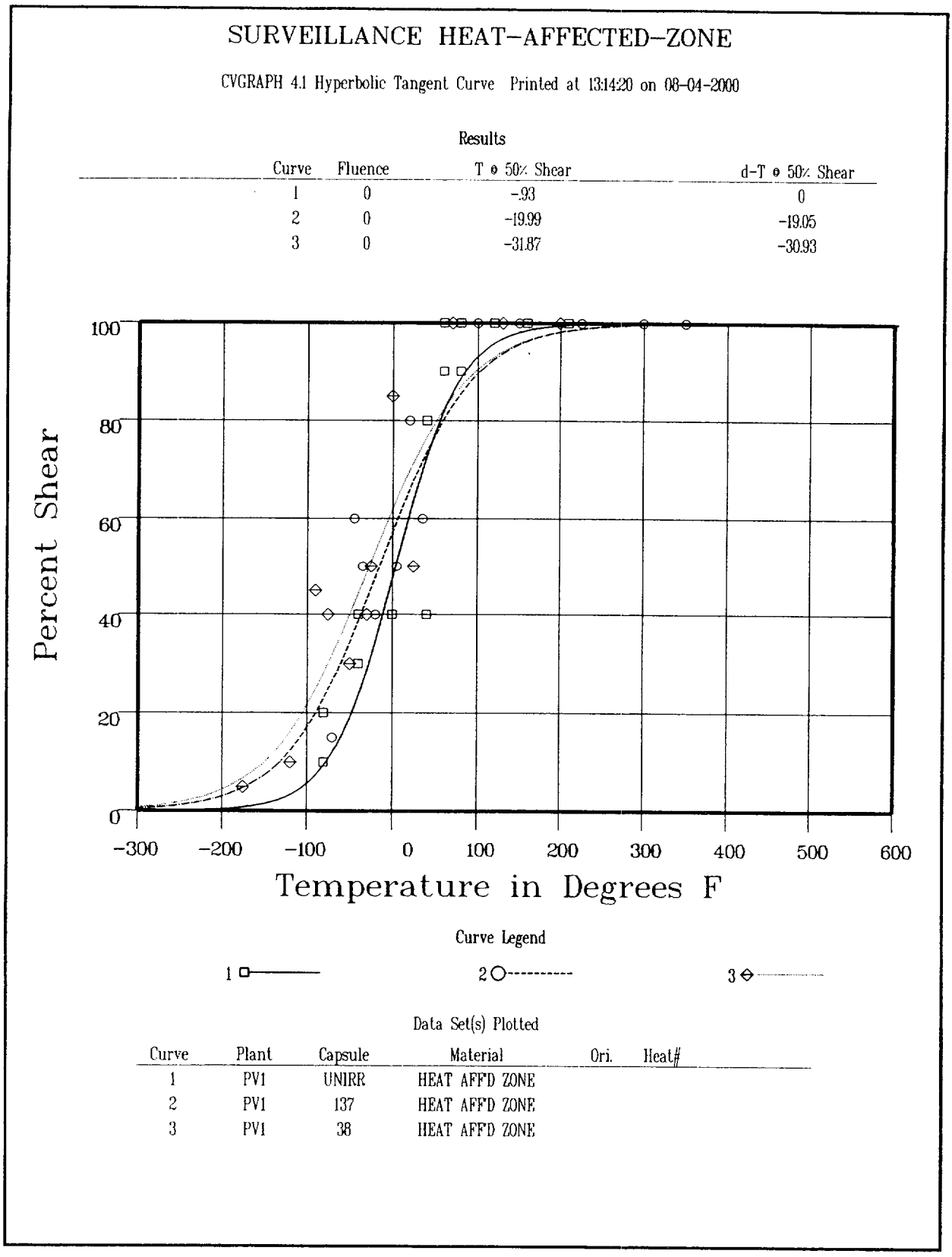


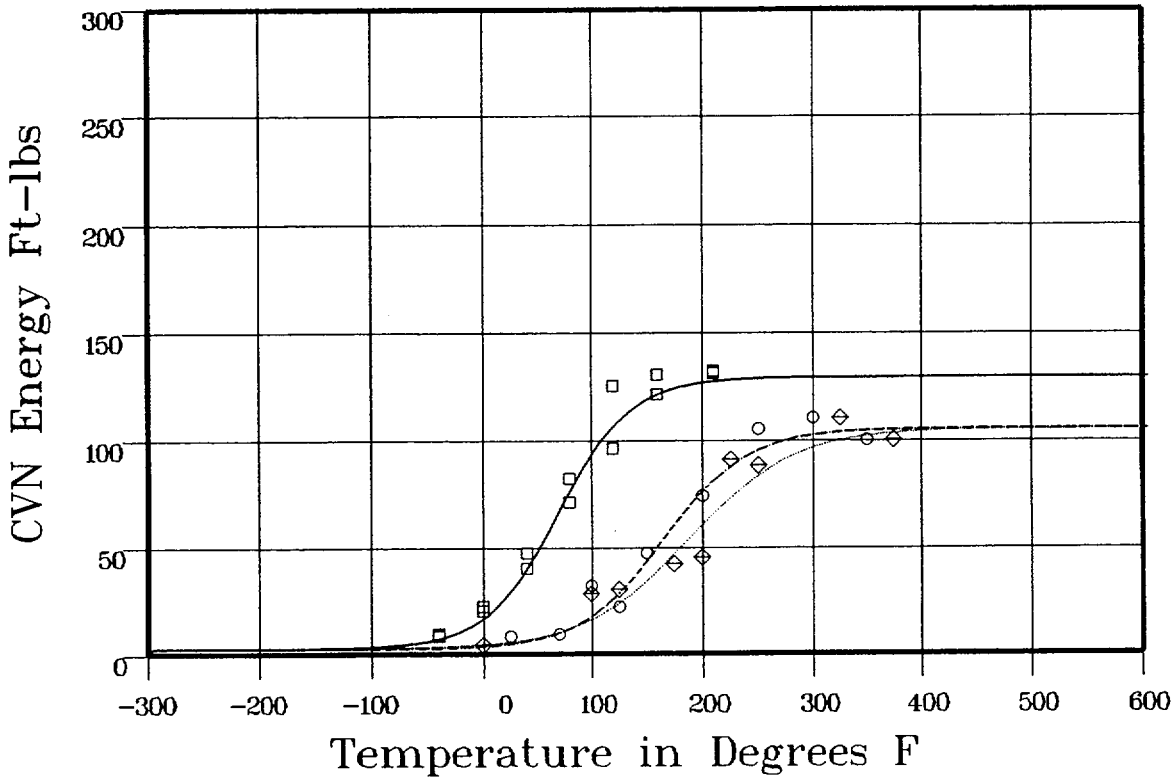
Figure 5-12 Charpy V-Notch Percent Shear vs. Temperature for Palo Verde Unit 1 Reactor Vessel Heat Affected Zone Material

STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 13:16:57 on 08-04-2000

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	129	0	21.9	0	47.61	0
2	0	2.19	0	105	-24	123.29	101.39	155.28	107.66
3	0	2.19	0	105	-24	136.16	114.25	176.21	128.6



Curve Legend

1 □——— 2 ○---------- 3 ◇·····

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	PV1	UNIRR	SRM SA533B1	LT	
2	PV1	137	SRM SA533B1	LT	
3	PV1	38	SRM SA533B1		

Figure 5-13 Charpy V-Notch Impact Energy vs. Temperature for Palo Verde Unit 1 Reactor Vessel Standard Reference Material

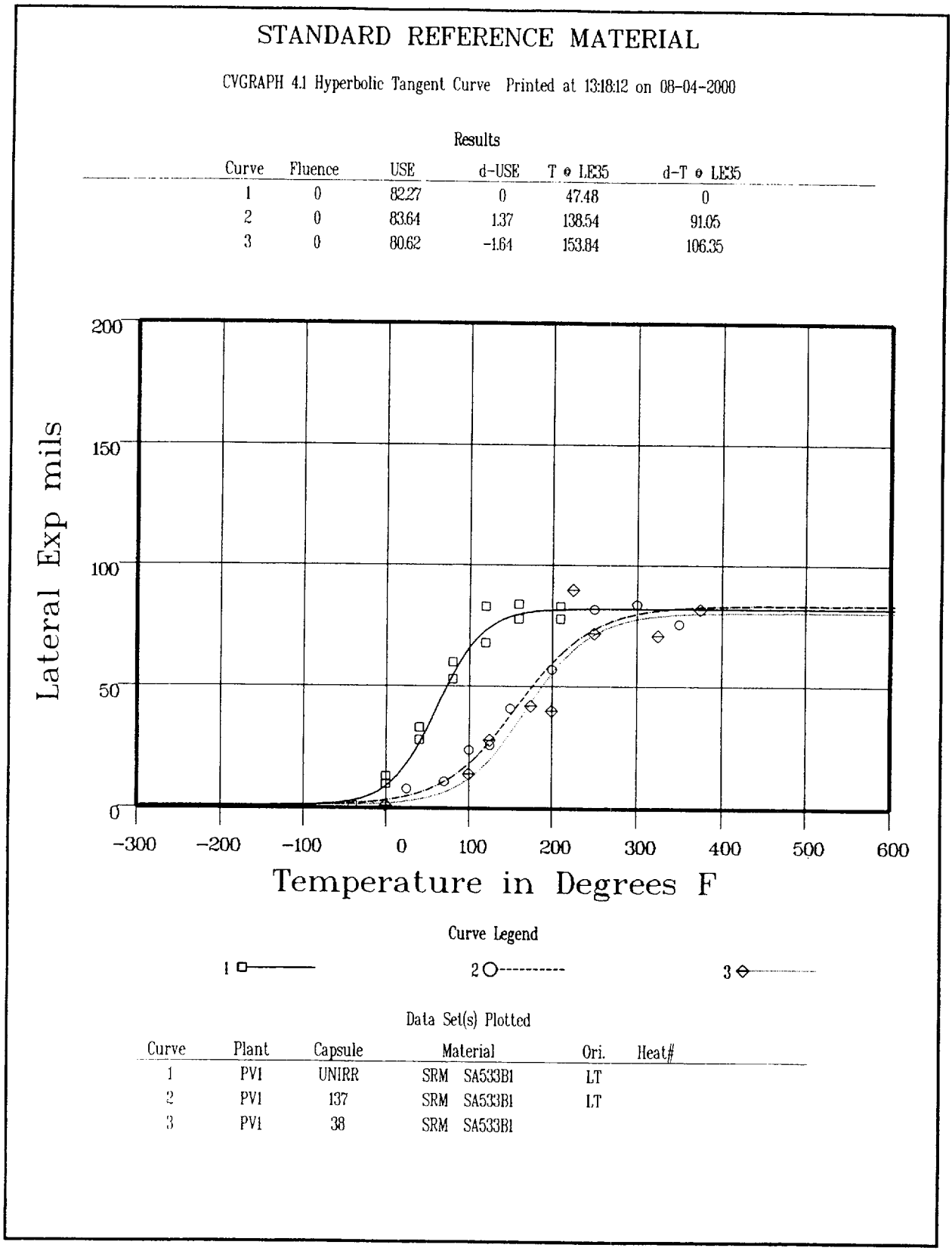


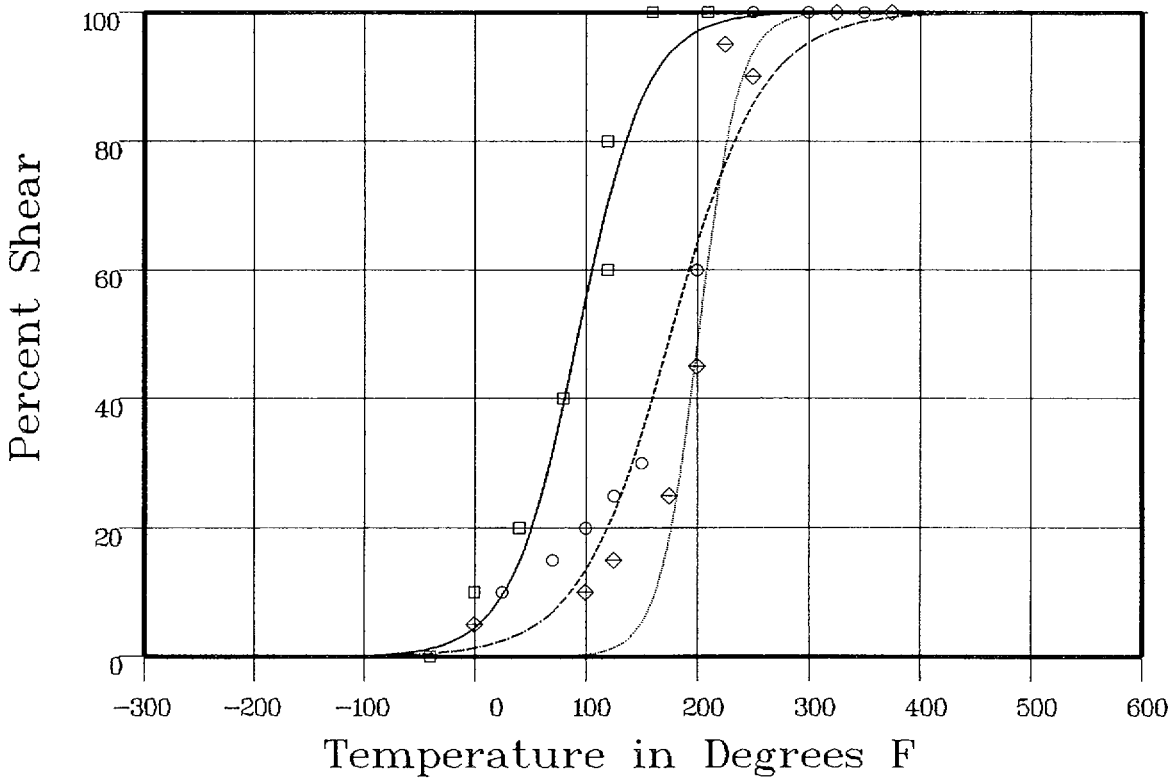
Figure 5-14 Charpy V-Notch Lateral Expansion vs. Temperature for Palo Verde Unit 1 Reactor Vessel Standard Reference Material

STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 13:19:05 on 08-04-2000

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	88.76	0
2	0	171.67	82.91
3	0	196.95	108.18



Curve Legend

1 2 3

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	PV1	UNIRR	SRM SA533B1	LT	
2	PV1	137	SRM SA533B1	LT	
3	PV1	38	SRM SA533B1		

Figure 5-15 Charpy V-Notch Percent Shear vs. Temperature for Palo Verde Unit 1 Reactor Vessel Standard Reference Material

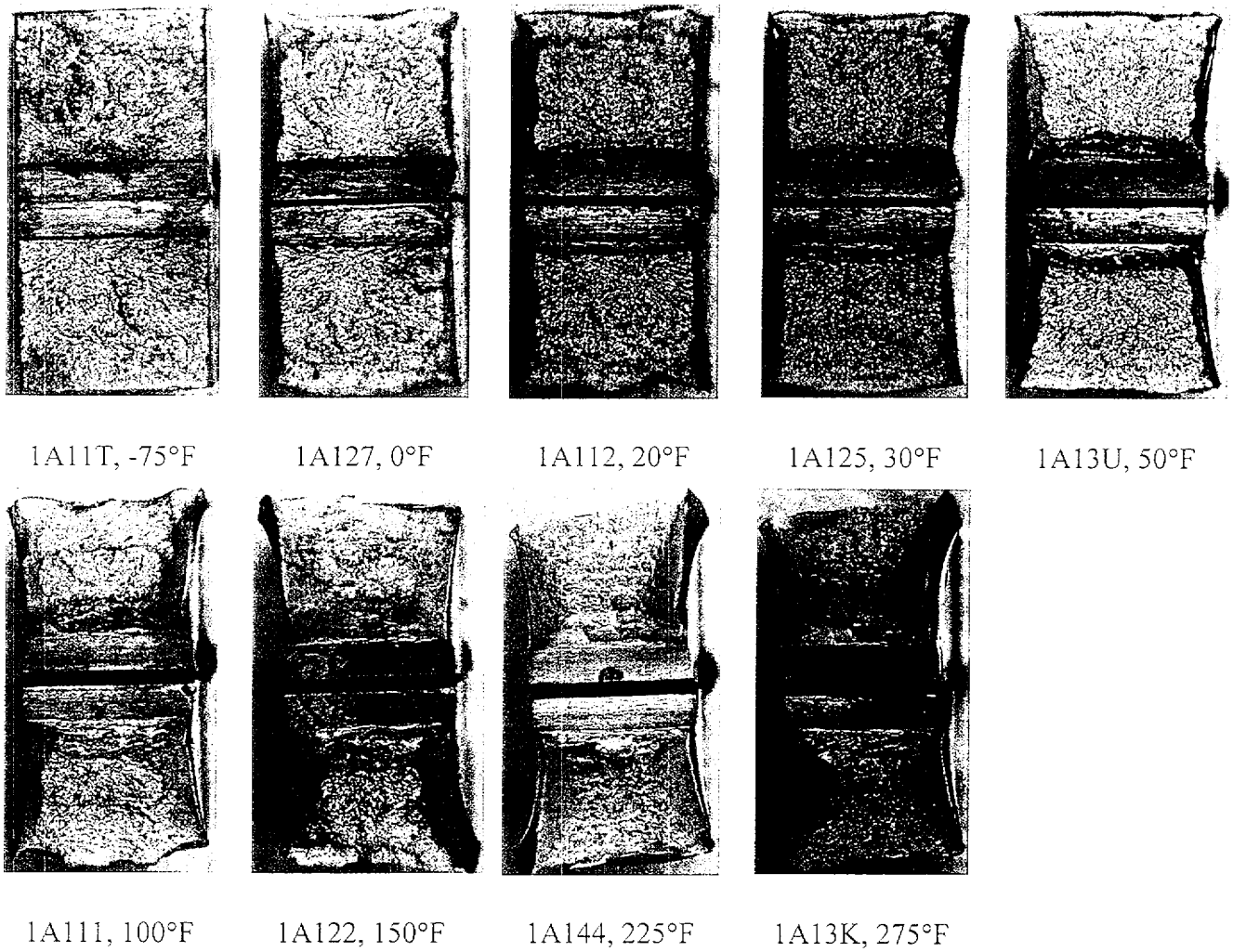


Figure 5-16 Charpy Impact Specimen Fracture Surfaces for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate M-6701-2 (Transverse Orientation)

Figure 5-17 Charpy Impact Specimen Fracture Surfaces for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate M-6701-2 (Longitudinal Orientation)

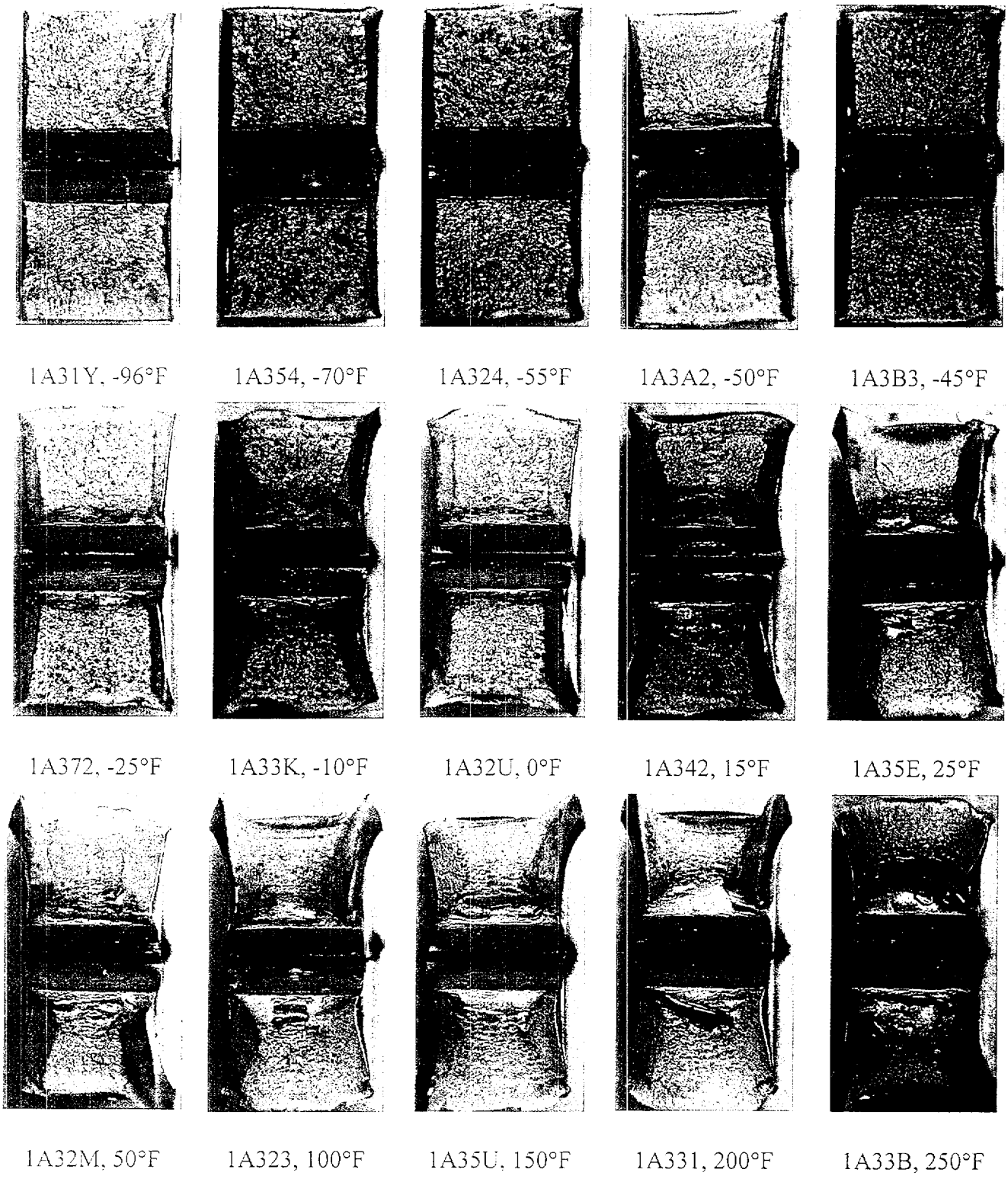


Figure 5-18 Charpy Impact Specimen Fracture Surfaces for Palo Verde Unit 2 Reactor Vessel Weld Metal Specimens

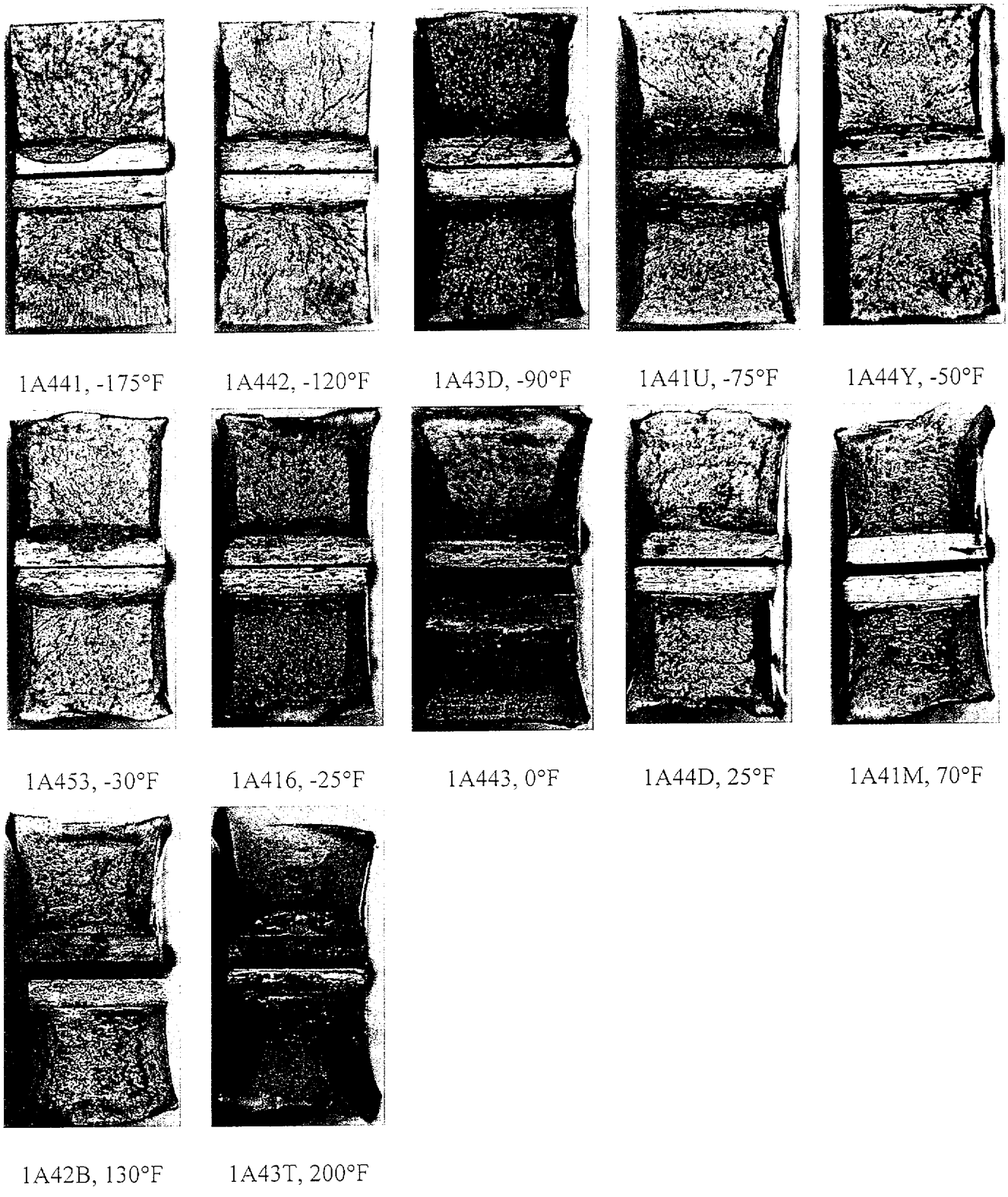


Figure 5-19 **Charpy Impact Specimen Fracture Surfaces for Palo Verde Unit 1 Reactor Vessel Heat Affected Zone (HAZ)**

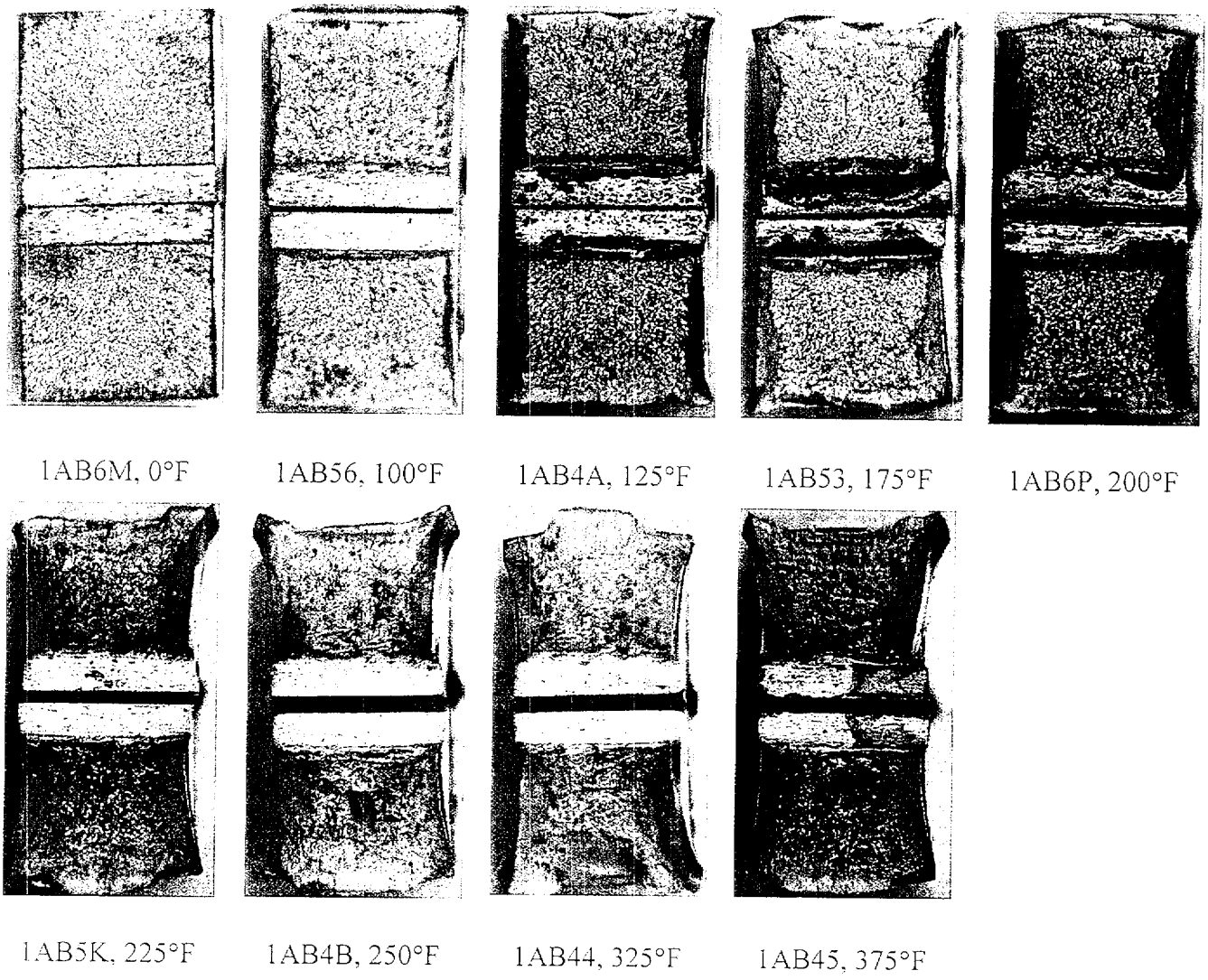
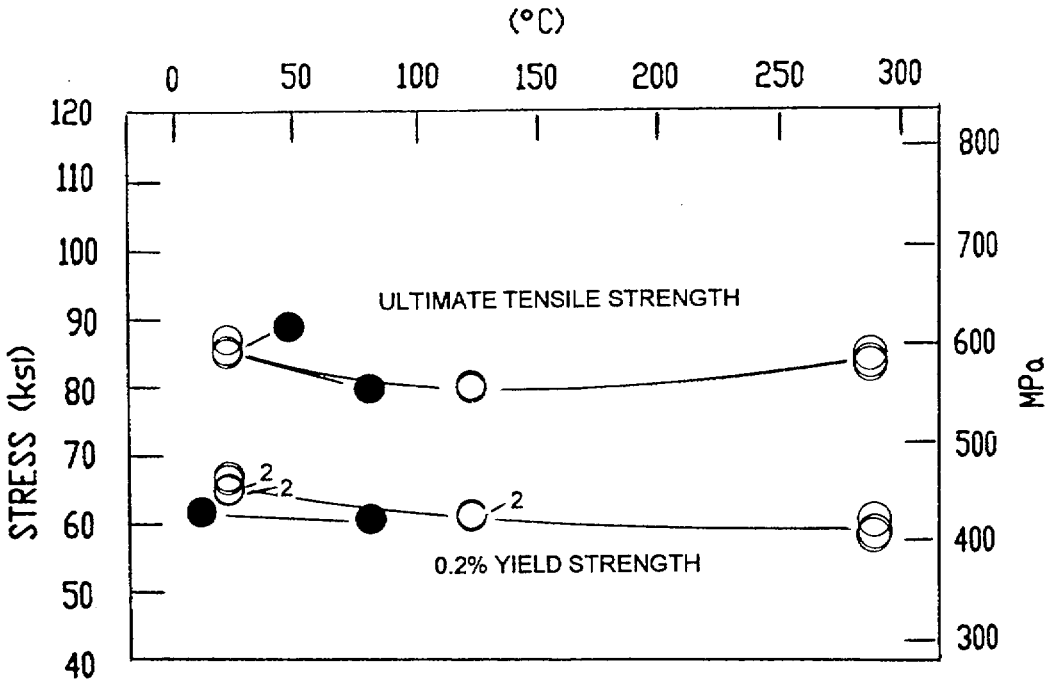


Figure 5-20 **Charpy Impact Specimen Fracture Surfaces for Palo Verde Unit 1 Reactor Vessel Standard Reference Material**



LEGEND:

▲ △ UNIRRADIATED

● ○ IRRADIATED TO A FLUENCE OF 7.85 X 10^{18} n/cm^2 (E>1.0MeV) AT 550°F

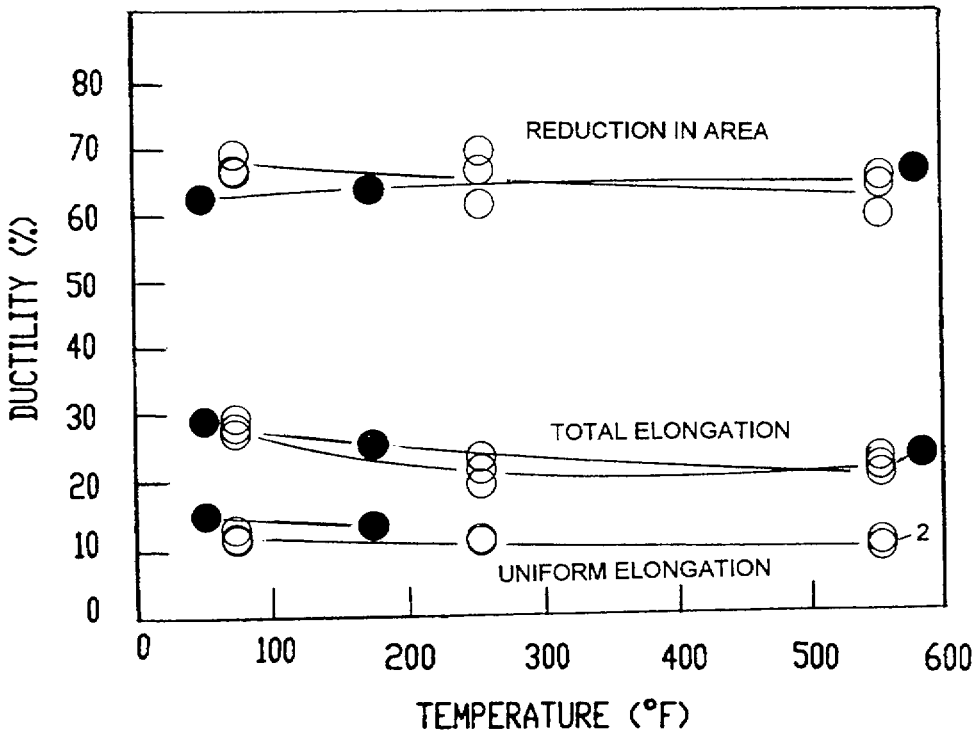


Figure 5-21 **Tensile Properties for Palo Verde Unit 1 Reactor Vessel Intermediate Shell Plate M-6701-2 (Transverse Orientation)**

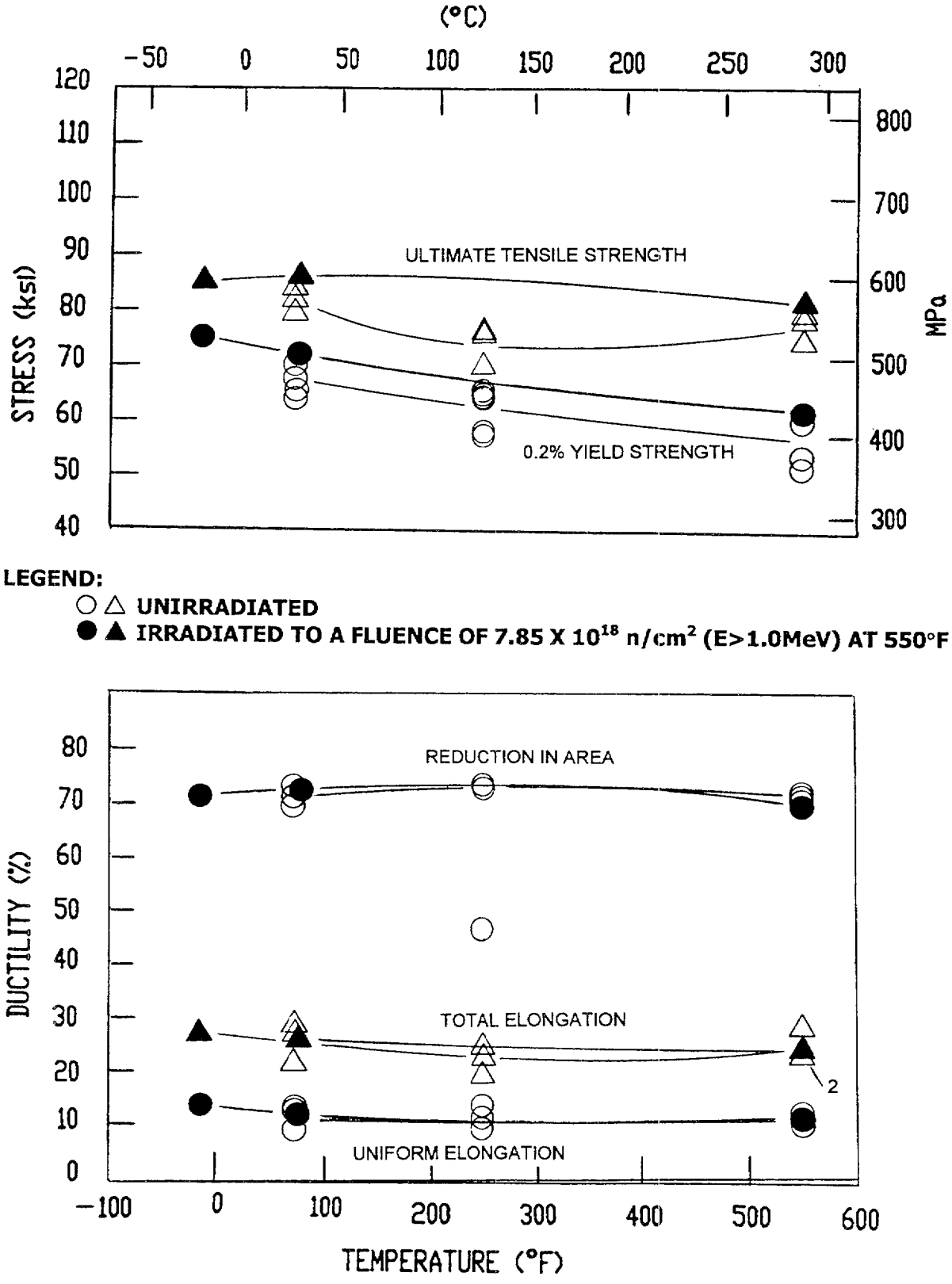
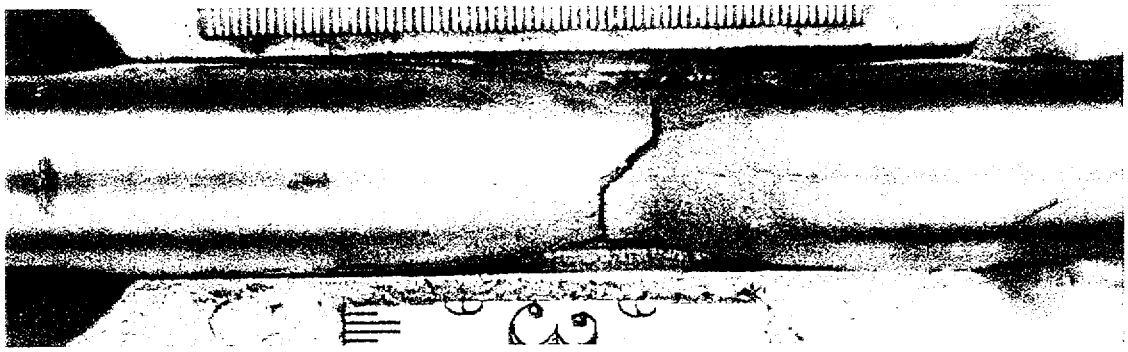
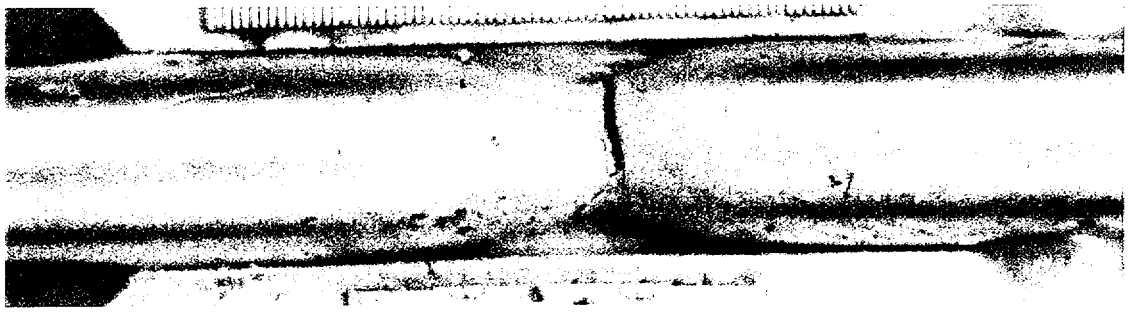


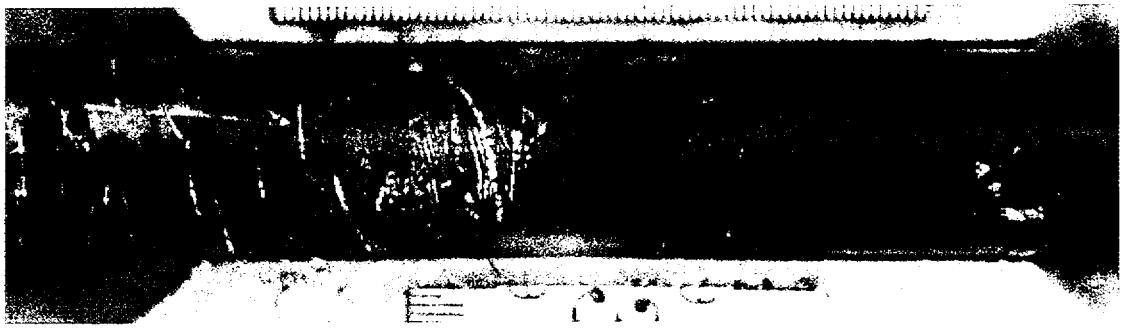
Figure 5-22 Tensile Properties for Palo Verde Unit 1 Reactor Vessel Weld Metal



Specimen 1A2JC Tested at 50°F

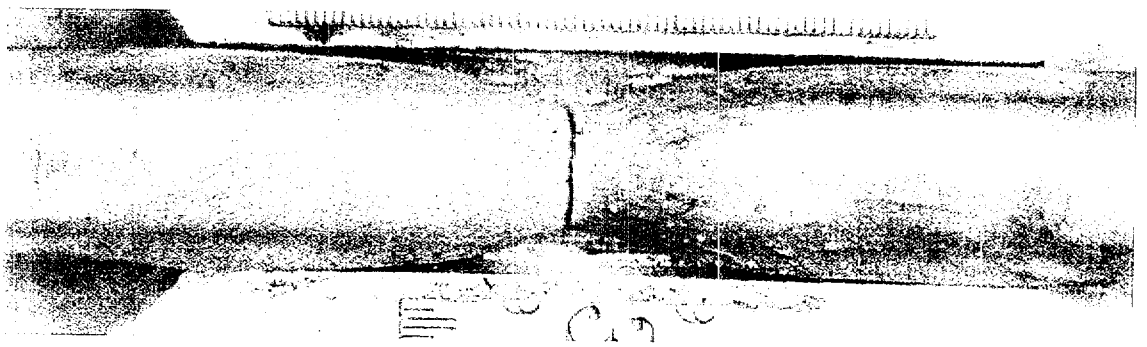


Specimen 1A2K2 Tested at 175°F

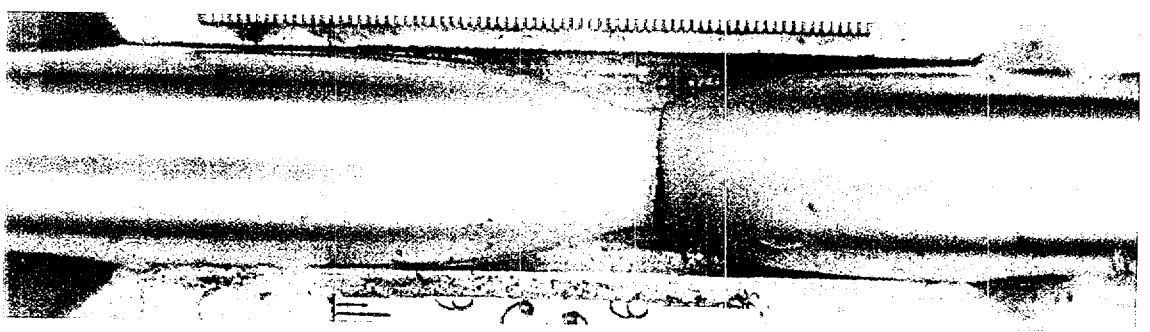


Specimen 1A2J5 Tested at 550°F

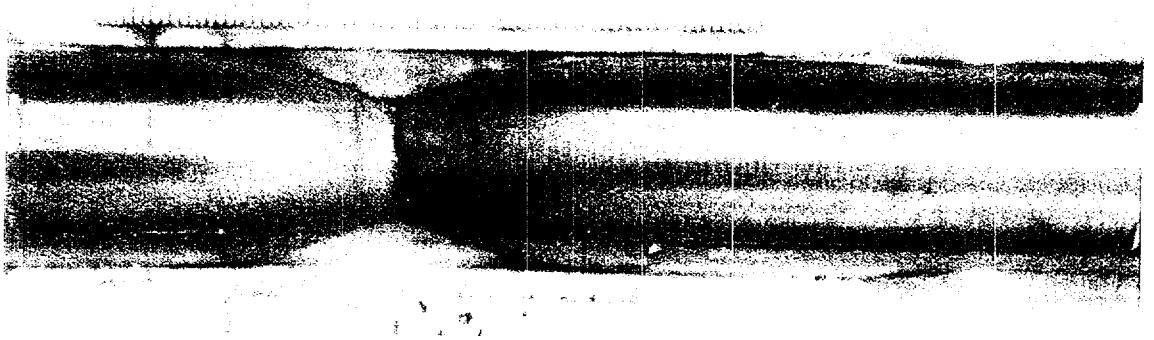
Figure 5-23 Fractured Tensile Specimens from Palo Verde Unit 1 Reactor Vessel Intermediate Shell M-6701-2 (Transverse Orientation)



Specimen 1A3JC Tested at -15°F



Specimen 1A3J4 Tested at 75°F



Specimen 1A3J7 Tested at 550°F

Figure 5-24 Fractured Tensile Specimens from Palo Verde Unit 1 Reactor Vessel Weld Metal

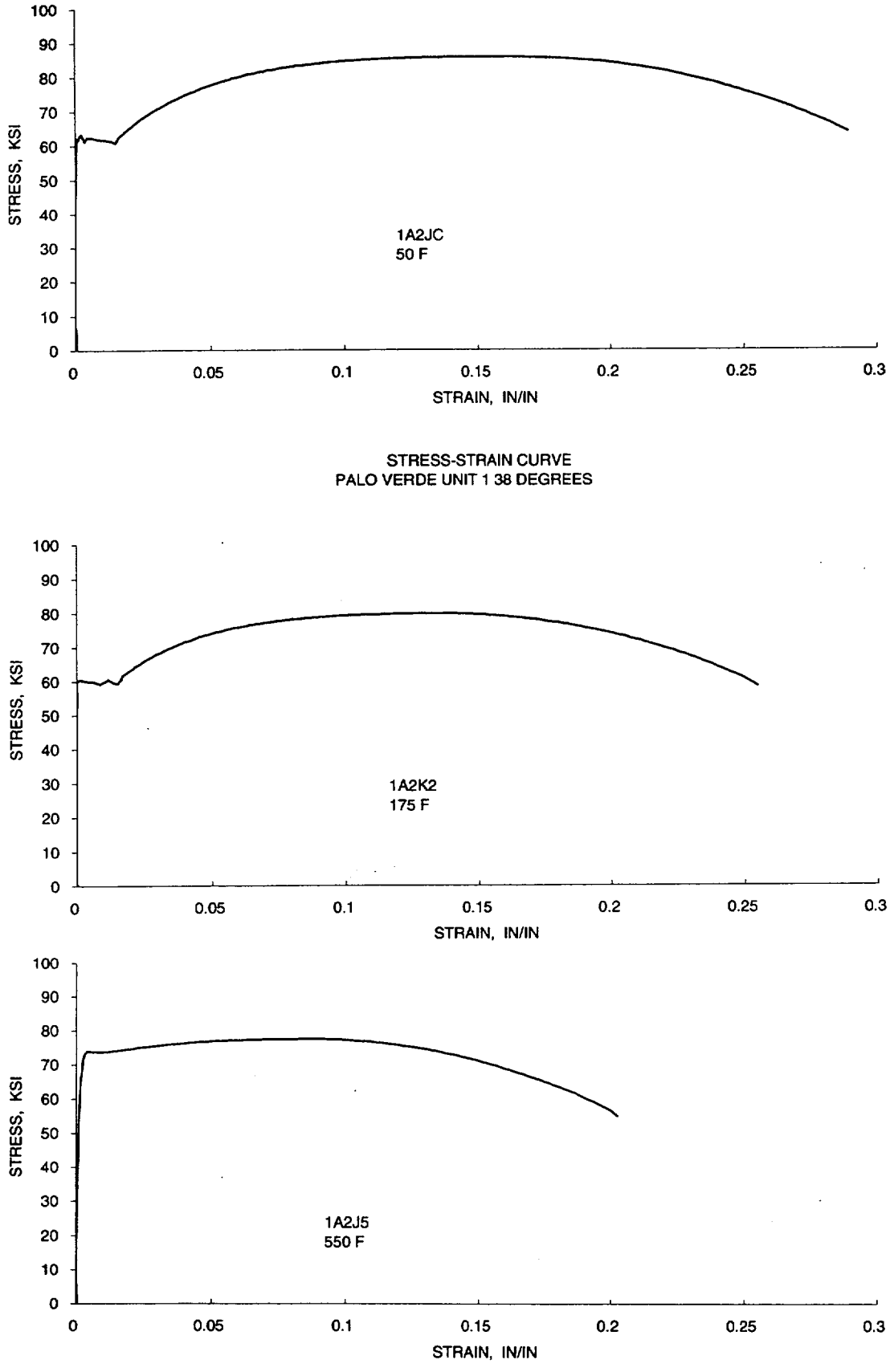


Figure 5-25 Engineering Stress-Strain Curves for Intermediate Shell Plate M-6701-2 Tensile Specimens 1A2JC, 1A2K2 and 1A2J5 (Transverse Orientation)

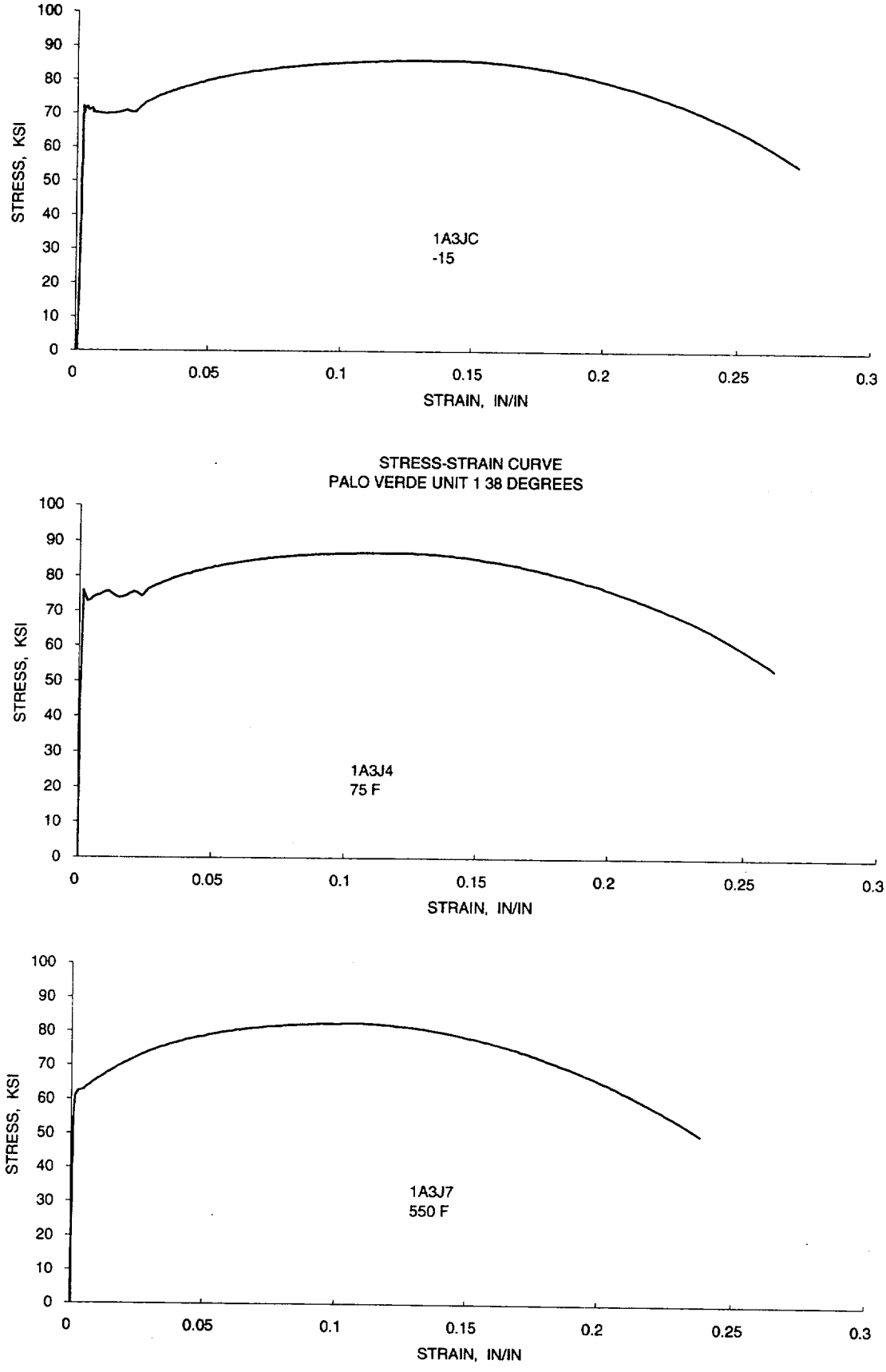


Figure 5-26 Engineering Stress-Strain Curves Weld Metal Tensile Specimens 1A3JC, 1A3J4, and 1A3J7

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, in order to interpret the neutron radiation induced material property changes observed in the test specimens, the neutron environment (energy spectrum, flux, fluence) to which the test specimens were exposed must be known. Second, in order to relate the changes observed in the test specimens to the present and future condition of the reactor vessel, a relationship must be established between the neutron environment at various positions within the reactor vessel and that experienced by the test specimens. The former 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 latter information is generally derived 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. In recent years, however, it has been suggested that an exposure model that accounts for differences in neutron energy spectra between surveillance capsule locations and positions within the vessel wall could lead to an improvement in the uncertainties associated with damage trend curves as well as to a more accurate evaluation of damage gradients through the reactor vessel wall.

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

This section provides the results of the neutron dosimetry evaluations performed in conjunction with the analysis of test specimens contained in surveillance Capsules W137 and W38 which were withdrawn after the fourth and eighth fuel cycles, respectively. This evaluation is based on current state-of-the-art methodology and nuclear data including neutron transport and dosimetry cross-section libraries derived from the
ENDF/B-VI data base. This report provides a consistent up-to-date neutron exposure database for use in evaluating the material properties of the Palo Verde Unit 1 reactor vessel. Included in the neutron exposure database is information related to the standby surveillance capsules W43, W142, W230, and W310.

In each capsule dosimetry evaluation, fast neutron exposure parameters in terms of neutron fluence (E > 1.0 MeV), neutron fluence (E > 0.1 MeV), and iron atom displacements (dpa) are established for the capsule irradiation history. The analytical formalism relating the measured capsule exposure to the exposure of the vessel wall is described and used to project the integrated exposure of the vessel wall. Also, uncertainties associated with the derived exposure parameters at the surveillance capsules and with the projected exposure of the reactor vessel are provided.

All of the calculations and dosimetry evaluations presented in this section have been based on the latest available nuclear cross-section data derived from ENDF/B-VI and the latest available calculational tools and are consistent with the requirements of Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Additionally, the methods used to develop the best estimate pressure vessel fluence are consistent with the NRC approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996.

6.2 DISCRETE ORDINATES ANALYSIS

A plan view of the reactor geometry at the core midplane is shown in Figure 4-1. Six irradiation capsules attached to the reactor vessel are included in the reactor design to constitute the reactor vessel surveillance program. The capsules are located at azimuthal angles of 38°, 43°, 137°, 142°, 230°, and 330° relative to the core cardinal axis as shown in Figure 4-1.

A plan view of the 45 degree R-θ sector model of the reactor including the surveillance capsule holder modeling attached to the reactor vessel is shown in Figure 6-1. The 45-degree model assumes azimuthal symmetry conditions in the reactor and the three capsules modeled at 38°, 40°, and 43° represent the locations of all six surveillance capsules. The stainless steel surveillance capsule holder containers are a 1.968-inch by 1.293-inch inner dimension with a 0.138-inch wall thickness. The stainless steel specimen containers are 1.5 inch by 0.75-inch and approximately 96 inches in height. The containers are positioned axially such that the test specimens are centered on the core midplane, thus spanning the central 8 feet of the 12.5-foot high reactor core.

From a neutronic standpoint, the surveillance capsules and associated support structures are significant. The presence of these materials has a marked effect on both the spatial distributions 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 effect of the surveillance capsules on the neutron environment at the vessel clad base metal interface is shown in Figure 6-2.

In performing the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, two sets of transport calculations were carried out. The first set of two-dimensional R-θ model calculations for each of the eight cycles use a model containing no surveillance capsules. The second set of R-θ computations were for each of the eight cycles with the surveillance capsule modeling shown in Figure 6-1 included at the 38°, 40°, and 43° locations in the 45 degree model. The two sets of calculations were used to obtain relative neutron energy distributions throughout the reactor geometry as well as to establish relative radial distributions of exposure parameters {ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), and dpa/sec} through the vessel wall. The neutron spectral information was required for the interpretation of neutron dosimetry withdrawn from the surveillance capsule as well as for the determination of exposure parameter ratios, i.e., [dpa/sec]/[ϕ(E > 1.0 MeV)], within the reactor vessel geometry. The relative radial gradient information was required to permit the projection of measured exposure parameters to locations interior to the reactor vessel wall, i.e., the ¼T and ¾T locations.

The absolute cycle-specific results from the forward transport calculations included the neutron energy spectra and radial distribution information in the two-dimensional r,θ model and provided the information required to:

1. Evaluate neutron dosimetry obtained from surveillance capsules,

2. Relate dosimetry results to key locations at the inner radius and through the thickness of the reactor vessel wall,

3. Enable a direct comparison of analytical prediction with measurement, and

4. Establish a mechanism for projection of reactor vessel exposure as the design of each new fuel cycle evolves.

The two-dimensional r,θ transport calculation model for the reactor configuration shown in Figure 4-1 is plotted in Figure 6-1. The transport calculations were carried out using the DORT two-dimensional discrete ordinates code Version 3.1[13] and the BUGLE-96 cross-section library [14]. The BUGLE-96 library is a 47 energy group ENDF/B-VI based data set produced specifically for light water reactor applications. In the transport analyses, a forward solution mode is used with anisotropic scattering treated with a P_5 Legendre polynomial expansion of the scattering cross-sections and angular discretization modeled as an S_{16} order of angular quadrature.

The core power distribution utilized in the forward transport calculation for each cycle were derived from assembly power and pin-by-pin power data provided by APS. The cycle averaged axial power distribution derived from APS data is shown in Figure 6-3. The axial power distribution data was used to define the maximum exposure parameter value.

Selected results from the neutron transport analyses are provided in Tables 6-1 through 6-5. The data listed in these tables establish the means for absolute comparisons of analysis and measurement for the Capsules W137 and W38 irradiation periods and provide the means to correlate dosimetry results with the corresponding exposure of the reactor vessel wall. The tabulations also provide the data for the 40° surveillance capsule location.

In Table 6-1, the calculated exposure parameters [$\phi(E > 1.0$ MeV), $\phi(E > 0.1$ MeV), and dpa/sec] are given at the geometric center of the three azimuthally symmetric surveillance capsule positions (38°, 40°, and 43°). All results are based on the Palo Verde Unit 1 core power distributions for the eight cycles of operation. The DORT forward solution transport analyses for each cycle are used to establish the absolute comparison of measurement values with analysis results. Similar neutron exposure rate data are given in Table 6-2 for the reactor vessel inner radius. Again, the three pertinent exposure parameters are listed for the Cycles 1 through 8 based on the cycle-by-cycle core power distributions. Also listed in Table 6-2 are the average exposure values for the both the first 4 cycles of operation, for the 8 cycles of operation, and for cycles 5 through 8. The average values for cycles 5 through 8 are used for exposure projections.

It is important to note that the data for the vessel inner radius were taken at the clad/base metal interface, and, thus, represent the maximum predicted exposure levels of the vessel plates and welds.

Radial gradient information applicable to $\phi(E > 1.0$ MeV), $\phi(E > 0.1$ MeV), and dpa/sec is given in Tables 6-3, 6-4, and 6-5, respectively. The data, obtained from the reference forward neutron transport calculation, are presented on a relative basis for each exposure parameter at several azimuthal locations. Exposure distributions through the vessel wall may be obtained by normalizing the calculated or projected exposure at the vessel inner radius to the gradient data listed in Tables 6-3 through 6-5.

For example, the neutron flux f(E > 1.0 MeV) at the ¼T depth in the reactor vessel wall along the 0° azimuth is given by:

$$\phi_{1/4T}(0°) = \phi(233.756, 0°)\, F(239.409, 0°)$$

where:

$\phi_{1/4T}(0°)$ = Projected neutron flux at the ¼T position on the 0° azimuth.

$\phi(233.756, 0°)$ = Projected or calculated neutron flux at the vessel inner radius on the 0° azimuth.

$F(239.409, 0°)$ = Ratio of the neutron flux at the ¼T position to the flux at the vessel inner radius for the 0° azimuth. This data is obtained from Table 6-3.

Similar expressions apply for exposure parameters expressed in terms of ϕ(E > 0.1 MeV) and dpa/sec where the attenuation function F is obtained from Tables 6-4 and 6-5, respectively.

6.3 NEUTRON DOSIMETRY

The passive neutron sensors included in the Palo Verde Unit 1 surveillance program are listed in Table 6-6. Also given in Table 6-6 are the primary nuclear reactions and associated nuclear constants that were used in the evaluation of the neutron energy spectrum within the surveillance capsules and in the subsequent determination of the various exposure parameters of interest [ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), dpa/sec]. The relative locations of the neutron sensors within the capsules are shown in Figure 4-2. The iron, nickel, copper, titanium, and cobalt-aluminum monitors, in wire form, were placed in holes drilled in spacers at several axial levels within the capsules. The cadmium shielded uranium fission monitors were accommodated within the dosimeter block located near the center of the capsule.

The use of passive monitors such as those listed in Table 6-6 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.

Specific activities for each of the monitors contained in Capsule W137 were determined using established ASTM procedures as documented in prior analysis [15]. The specific activities for each of the monitors contained in Capsule W38 were determined using established ASTM procedures [16 through 29]. Following sample preparation and weighing, the activity of each monitor was determined by means of a high-resolution gamma spectrometer. The irradiation history for the first four operating cycles of the Palo Verde Unit 1 reactor were from NUREG-0020, "Licensed Operating Reactors Status Summary Report". The irradiation history for the Cycles 5 to 8 operating periods of the Palo Verde Unit 1 reactor was obtained from plant personnel [30]. The irradiation history applicable to the exposure of Capsules W137 and W38 is given in Table 6-7.

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 can be calculated for each fuel cycle using the transport technology 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.

Measured and saturated reaction product specific activities as well as the derived full power reaction rates are listed in Table 6-8. All the measurements of fission monitors were updated with the following corrections. The reaction rates of the ^{238}U sensors provided in Table 6-8 includes corrections for ^{235}U impurities, plutonium build-in, and gamma ray induced fission.

Values of key fast neutron exposure parameters were derived from the measured reaction rates using the FERRET least squares adjustment code [31]. The FERRET approach used the measured reaction rate data, sensor reaction cross-sections, and a calculated trial spectrum as input and proceeded to adjust the group fluxes from the trial spectrum to produce a best fit (in a least squares sense) within the constraints of the parameter uncertainties. The best estimate exposure parameters, along with the associated uncertainties, were then obtained from the best-estimate spectrum.

In the FERRET evaluations, a log-normal least squares algorithm weights both the a priori values and the measured data in accordance with the assigned uncertainties and correlations. In general, the measured values, f, are linearly related to the flux, ϕ, by some response matrix, A:

$$f_i^{(s,\alpha)} = \sum_g A_{ig}^{(s)} \phi_g^{(\alpha)}$$

where i indexes the measured values belonging to a single data set s, g designates the energy group, and α delineates spectra that may be simultaneously adjusted. For example,

$$R_i = \sum_g \sigma_{ig} \phi_g$$

relates a set of measured reaction rates, R_i, to a single spectrum, ϕ_g, by the multi-group reaction cross-section, σ_{ig}. The log-normal approach automatically accounts for the physical constraint of positive fluxes, even with large assigned uncertainties.

In the least squares adjustment, the continuous quantities (i.e., neutron spectra and cross-sections) were approximated in a multi-group format consisting of 53 energy groups. The trial input spectrum was converted to the FERRET 53-group structure using the SAND-II code [32]. This procedure was carried out by first expanding the 47 group calculated spectrum into the SAND-II 620 group structure using a SPLINE interpolation procedure in regions where group boundaries do not coincide. The 620 point spectrum was then re-collapsed into the group structure used in FERRET.

The sensor set reaction cross-sections, obtained from the ENDF/B-VI dosimetry file [33], were also collapsed into the 53-energy group structure using the SAND-II code. In this instance, the trial spectrum, as expanded to 620 groups, was employed as a weighting function in the cross-section collapsing procedure. Reaction cross-section uncertainties in the form of a 53 × 53 covariance matrix for each sensor reaction were also constructed from the information contained on the ENDF/B-VI data files. These matrices included energy group to energy group uncertainty correlations for each of the individual reactions. However, correlations between cross-sections for different sensor reactions were not included. The omission of this additional uncertainty information does not significantly impact the results of the adjustment.

Due to the importance of providing a trial spectrum that exhibits a relative energy distribution close to the actual spectrum at the sensor set locations, the neutron spectrum input to the FERRET evaluation was taken from the center of the surveillance capsule modeled in the reference forward transport calculation. While the 53 x 53 group covariance matrices applicable to the sensor reaction cross-sections were developed from the ENDF/B-VI data files, the covariance matrix for the input trial spectrum was constructed from the following relation:

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

where R_n specifies an overall fractional normalization uncertainty (i.e., complete correlation) for the set of values. The fractional uncertainties, R_g, specify additional random uncertainties for group g 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 short range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1 when g = g' and 0 otherwise. For the trial spectrum used in the current evaluations, a short range correlation of $\gamma = 6$ groups was used. This choice implies that neighboring groups are strongly correlated when θ is close to 1. Strong long-range correlations (or anti-correlations) were justified based on information presented by R. E. Maerker [34]. The uncertainties associated with the measured reaction rates included both statistical (counting) and systematic components. The systematic component of the overall uncertainty accounts for counter efficiency, counter calibrations, irradiation history corrections, and corrections for competing reactions in the individual sensors.

Results of the FERRET evaluation of the Capsule W137 and W38 dosimetry are given in Table 6-9. The data summarized in this table include fast neutron exposure evaluations in terms of $\Phi(E > 1.0$ MeV), $\Phi(E > 0.1$ MeV), and dpa. In general, excellent results were achieved in the fits of the best estimate spectra to the individual measured reaction rates. The measured, calculated and best estimate reaction rates for each reaction are given in Table 6-10. An examination of Table 6-10 shows that, in all cases, reaction rates calculated with the best estimate spectra match the measured reaction rates to better than 6%. The best estimate and measured reaction rates compared to calculated reaction rates for the Co monitors show unusually high values. Although the reason has not been identified, a higher Co content in the monitor than that documented and used in the analysis would result in high values. In any event, Co reaction is monitored for an energy range much lower than the fast flux of primary interest; thus the Co data has insignificant effect on the best estimate fast flux from the analysis. The best estimate spectra from the least squares evaluation is given in Table 6-11 in the FERRET 53 energy group structure.

In Table 6-12, absolute comparisons of the best estimate and calculated fluence at the center of Capsules W137 and W38 are presented. The results for the Capsules W137 and W38 dosimetry evaluation (BE/C ratio of 0.832 for $\Phi(E > 1.0$ MeV)) are within expected tolerances compared with results obtained from similar evaluations of dosimetry from other reactors using methodologies based on ENDF/B-VI cross-sections.

6.4 PROJECTIONS OF REACTOR VESSEL EXPOSURE

The best estimate exposure of the Palo Verde Unit 1 reactor vessel was developed using a combination of absolute plant specific transport calculations and all available plant specific measurement data. In the case of Palo Verde Unit 1, the measurement database contains measurements from the five surveillance capsules discussed in this report.

Combining this measurement data base with the plant-specific calculations, the best estimate vessel exposure is obtained from the following relationship:

$$\Phi_{Best\ Est.} = K\ \Phi_{Calc.}$$

where:

$\Phi_{Best\ Est.}$ = The best estimate fast neutron exposure at the location of interest.

K = The plant specific best estimate/calculation (BE/C) bias factor derived from the surveillance capsule dosimetry data.

$\Phi_{Calc.}$ = The absolute calculated fast neutron exposure at the location of interest.

The approach defined in the above equation is based on the premise that the measurement data represent the most accurate plant-specific information available at the locations of the dosimetry; and further, that the use of the measurement data on a plant-specific basis essentially removes biases present in the analytical approach and mitigates the uncertainties that would result from the use of analysis alone.

That is, at the measurement points the uncertainty in the best estimate exposure is dominated by the uncertainties in the measurement process. At locations within the reactor vessel wall, additional uncertainty is incurred due to the analytically determined relative ratios among the various measurement points and locations within the reactor vessel wall.

For Palo Verde Unit 1, the derived plant specific bias factors were 0.832, 0.894, 0.902 for $\Phi(E > 1.0\ MeV)$, $\Phi(E > 0.1\ MeV)$, and dpa, respectively. Bias factors of this magnitude developed with BUGLE-96 are within expected tolerances for fluence calculated using the ENDF/B-VI based cross-section library.

The use of the bias factors derived from the measurement data base acts to remove plant-specific biases associated with the definition of the core source, actual versus assumed reactor dimensions, and operational variations in water density within the reactor. As a result, the overall uncertainty in the best estimate exposure projections within the vessel wall depends on the individual uncertainties in the measurement process, the uncertainty in the dosimetry location, and, in the uncertainty in the calculated ratio of the neutron exposure at the point of interest to that at the measurement location.

The uncertainty in the derived neutron flux for an individual measurement is obtained directly from the results of a least squares evaluation of dosimetry data. The least squares approach combines individual uncertainty in the calculated neutron energy spectrum, the uncertainties in dosimetry cross-sections, and the uncertainties in measured foil specific activities to produce a net uncertainty in the derived neutron flux at the measurement point. The associated uncertainty in the plant specific bias factor, K, derived from the BE/C data base, in turn, depends on the total number of available measurements as well as on the uncertainty of each measurement.

In developing the overall uncertainty associated with the reactor vessel exposure, the positioning uncertainties for dosimetry are taken from parametric studies of sensor position performed as part a series of analytical sensitivity studies included in the qualification of the methodology. The uncertainties in the exposure ratios relating dosimetry results to positions within the vessel wall are again based on the analytical sensitivity studies of the vessel thickness tolerance, downcomer water density variations, and vessel inner radius tolerance. Thus, this portion of the overall uncertainty is controlled entirely by dimensional tolerances associated with the reactor design and by the operational characteristics of the reactor.

The net uncertainty in the bias factor, K, is combined with the uncertainty from the analytical sensitivity study to define the overall fluence uncertainty at the reactor vessel wall. In the case of Palo Verde Unit 1, the derived uncertainties in the bias factor, K, and the additional uncertainty from the analytical sensitivity studies combine to yield a net uncertainty of $\pm$ 7.6%.

Based on this best estimate approach, neutron exposure projections at key locations on the reactor vessel inner radius are given in Table 6-13; furthermore, calculated neutron exposure projections are also provided for comparison purposes. Along with the current (9.81 EFPY) exposure, projections are also provided for exposure periods of 15, 32, 40, 45, and 54 EFPY. Projections for future operation were based on the assumption that the Cycles 5 through 8 exposure rates would continue to be applicable throughout plant life.

In the derivation of best estimate and calculated exposure gradients within the reactor vessel wall for the Palo Verde Unit 1 reactor vessel, exposure projections to 15, 32, 40, 45, and 54 EFPY were also employed. Data based on both a $\Phi(E > 1.0\ MeV)$ slope and a plant-specific dpa slope through the vessel wall are provided in Table 6-14.

In order to assess RT_{NDT} versus fluence curves, dpa equivalent fast neutron fluence levels for the ¼T and ¾T positions were defined by the relations:

$$\phi(\tfrac{1}{4}T) = \phi(0T)\,\frac{dpa(\tfrac{1}{4}T)}{dpa(0T)} \qquad \text{and} \qquad \phi(\tfrac{3}{4}T) = \phi(0T)\,\frac{dpa(\tfrac{3}{4}T)}{dpa(0T)}$$

Using this approach results in the dpa equivalent fluence values listed in Table 6-14.

In Table 6-15, updated lead factors are listed for all of the Palo Verde Unit 1 surveillance capsules.

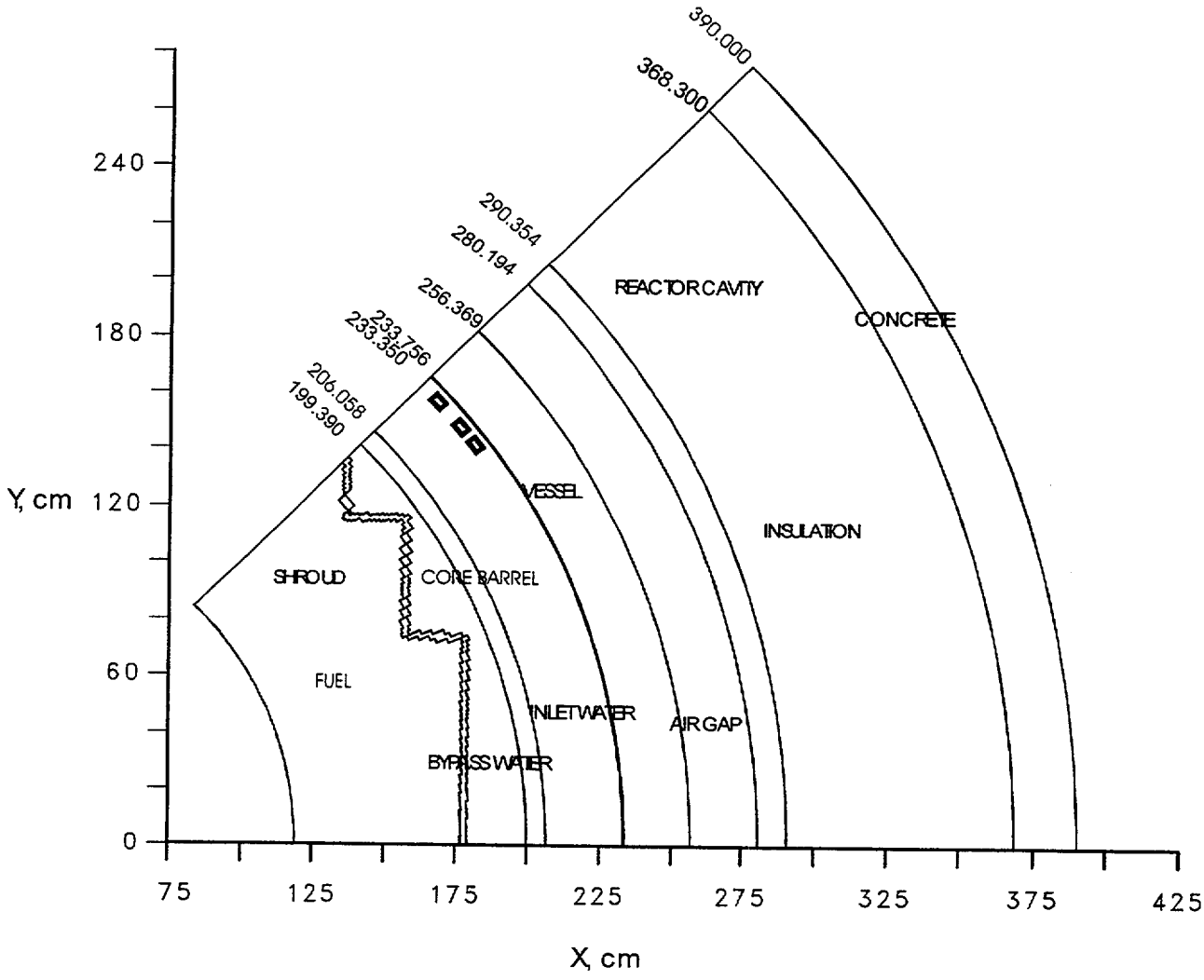


Figure 6-1. Palo Verde Reactor Model (45 Degree R-Θ Sector) Including Vessel Surveillance Capsules

Figure 6-2

Azimuthal Variation of Neutron Flux (E > 1.0 Mev)
At The Reactor Vessel Inner Radius

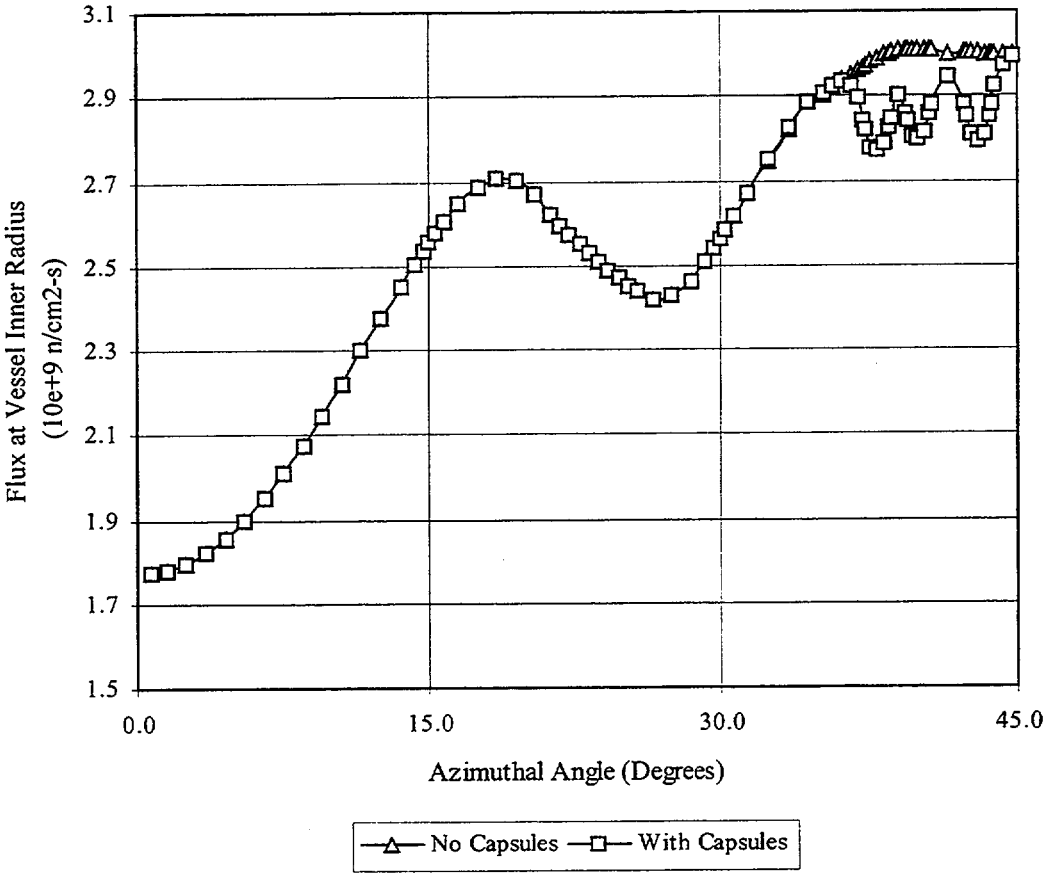


Figure 6-3

Axial Distribution of Reactor Power

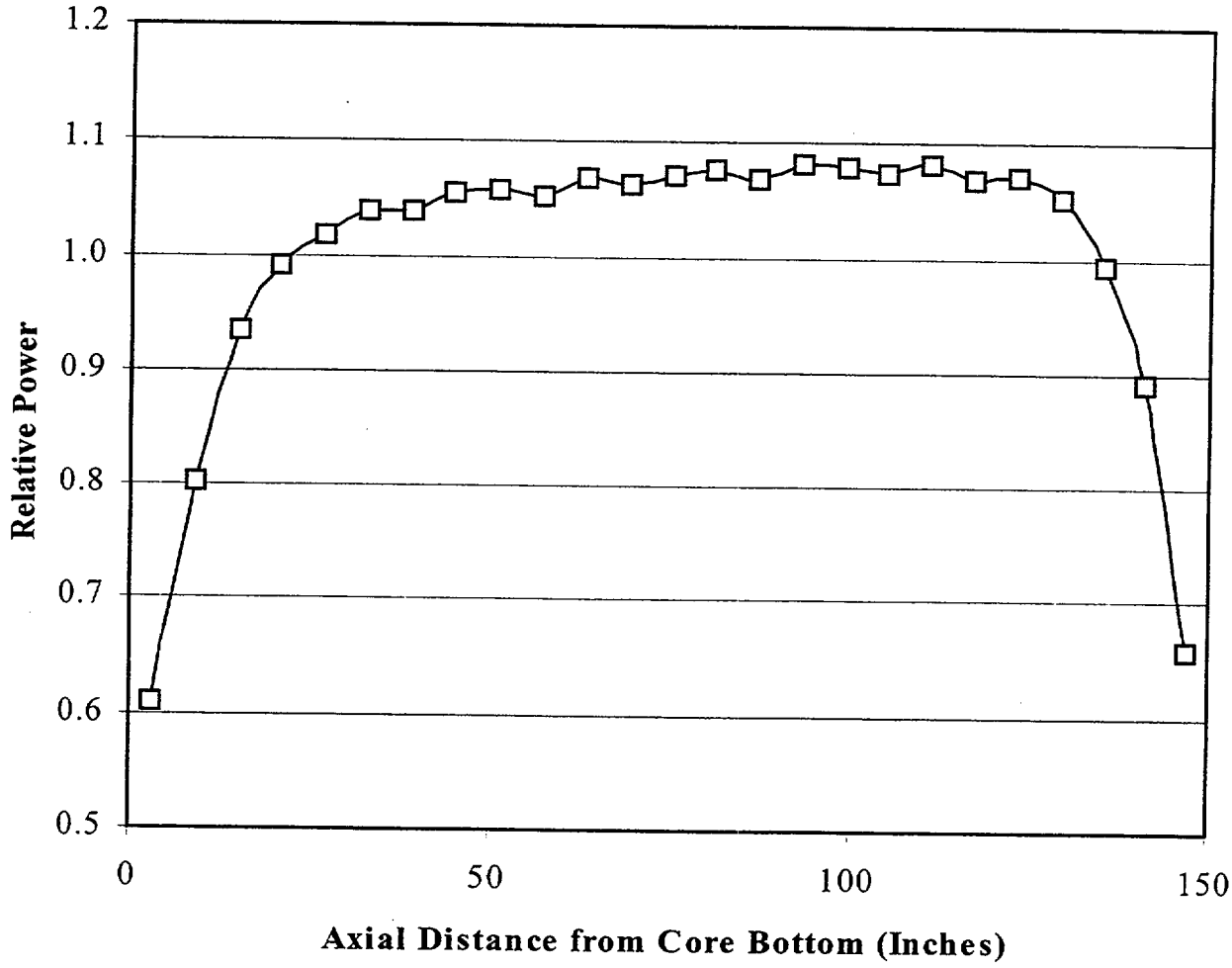


Table 6-1

Calculated Fast Neutron Exposure Rates
at the Center of the Surveillance Capsules
Core Midplane Elevation

Operating Cycle	Capsule Location 38°	40°	43°
	Flux(E>1.0 Mev) [n/cm^2-sec]		
Cycle 1	4.201E+10	4.224E+10	4.167E+10
Cycle 2	2.804E+10	2.803E+10	2.751E+10
Cycle 3	2.493E+10	2.511E+10	2.487E+10
Cycle 4	2.655E+10	2.657E+10	2.606E+10
Cycle 5	2.518E+10	2.621E+10	2.658E+10
Cycle 6	2.647E+10	2.748E+10	2.779E+10
Cycle 7	1.718E+10	1.751E+10	1.745E+10
Cycle 8	1.669E+10	1.708E+10	1.709E+10
Average(1-4)	3.046E+10	3.058E+10	3.013E+10
Average(1-8)	2.542E+10	2.583E+10	2.570E+10
Average(5-8)	2.104E+10	2.170E+10	2.184E+10
	Flux(E>0.1 Mev) [n/cm^2-sec]		
Cycle 1	7.783E+10	7.808E+10	7.655E+10
Cycle 2	5.156E+10	5.143E+10	5.017E+10
Cycle 3	4.581E+10	4.605E+10	4.535E+10
Cycle 4	4.890E+10	4.882E+10	4.761E+10
Cycle 5	4.623E+10	4.803E+10	4.845E+10
Cycle 6	4.865E+10	5.038E+10	5.068E+10
Cycle 7	3.145E+10	3.199E+10	3.171E+10
Cycle 8	3.057E+10	3.122E+10	3.108E+10
Average(1-4)	5.617E+10	5.626E+10	5.512E+10
Average(1-8)	4.677E+10	4.742E+10	4.692E+10
Average(5-8)	3.859E+10	3.972E+10	3.978E+10
	Iron Atom Displacement Rate [dpa]		
Cycle 1	6.094E-11	6.130E-11	6.047E-11
Cycle 2	4.079E-11	4.079E-11	4.004E-11
Cycle 3	3.629E-11	3.654E-11	3.619E-11
Cycle 4	3.860E-11	3.864E-11	3.791E-11
Cycle 5	3.663E-11	3.813E-11	3.866E-11
Cycle 6	3.849E-11	3.996E-11	4.041E-11
Cycle 7	2.503E-11	2.552E-11	2.543E-11
Cycle 8	2.432E-11	2.488E-11	2.489E-11
Average(1-4)	4.426E-11	4.444E-11	4.379E-11
Average(1-8)	3.697E-11	3.757E-11	3.738E-11
Average(5-8)	3.062E-11	3.158E-11	3.179E-11

Table 6-2

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

			Flux (E > 1.0 Mev) [n/cm2-sec]		
Operating Cycle	0 Deg	15 Deg	30 Deg	42.3 Deg	45 Deg
Cycle 1	1.77E+10	2.56E+10	2.57E+10	3.01E+10	3.00E+10
Cycle 2	1.53E+10	1.57E+10	1.75E+10	1.86E+10	1.85E+10
Cycle 3	1.71E+10	1.89E+10	1.62E+10	1.70E+10	1.70E+10
Cycle 4	1.36E+10	1.89E+10	1.75E+10	1.85E+10	1.83E+10
Cycle 5	9.48E+09	1.16E+10	1.34E+10	1.85E+10	1.86E+10
Cycle 6	8.26E+09	1.09E+10	1.39E+10	1.94E+10	1.95E+10
Cycle 7	7.66E+09	1.01E+10	1.07E+10	1.24E+10	1.24E+10
Cycle 8	8.39E+09	9.71E+09	1.02E+10	1.21E+10	1.21E+10
Average (1-4)	1.60E+10	2.01E+10	1.93E+10	2.12E+10	2.11E+10
Average (1-8)	1.20E+10	1.50E+10	1.54E+10	1.80E+10	1.80E+10
Average (5-8)	8.42E+09	1.05E+10	1.19E+10	1.53E+10	1.54E+10

			Flux (E > 0.1 Mev) [n/cm2-sec]		
Operating Cycle	0 Deg	15 Deg	30 Deg	42.3 Deg	45 Deg
Cycle 1	3.74E+10	5.41E+10	5.49E+10	6.43E+10	6.42E+10
Cycle 2	3.20E+10	3.32E+10	3.72E+10	3.97E+10	3.95E+10
Cycle 3	3.57E+10	3.98E+10	3.45E+10	3.63E+10	3.62E+10
Cycle 4	2.85E+10	3.99E+10	3.72E+10	3.94E+10	3.92E+10
Cycle 5	1.98E+10	2.45E+10	2.85E+10	3.91E+10	3.94E+10
Cycle 6	1.73E+10	2.29E+10	2.95E+10	4.11E+10	4.14E+10
Cycle 7	1.60E+10	2.11E+10	2.25E+10	2.62E+10	2.62E+10
Cycle 8	1.75E+10	2.04E+10	2.16E+10	2.56E+10	2.56E+10
Average (1-4)	3.36E+10	4.25E+10	4.11E+10	4.52E+10	4.50E+10
Average (1-8)	2.51E+10	3.16E+10	3.26E+10	3.84E+10	3.84E+10
Average (5-8)	1.76E+10	2.21E+10	2.52E+10	3.25E+10	3.26E+10

			dpa/sec		
Operating Cycle	0 Deg	15 Deg	30 Deg	42.3 Deg	45 Deg
Cycle 1	2.74E-11	3.91E-11	3.94E-11	4.60E-11	4.58E-11
Cycle 2	2.37E-11	2.41E-11	2.69E-11	2.86E-11	2.84E-11
Cycle 3	2.63E-11	2.90E-11	2.50E-11	2.61E-11	2.60E-11
Cycle 4	2.10E-11	2.90E-11	2.69E-11	2.83E-11	2.81E-11
Cycle 5	1.47E-11	1.79E-11	2.07E-11	2.82E-11	2.84E-11
Cycle 6	1.28E-11	1.67E-11	2.13E-11	2.97E-11	2.99E-11
Cycle 7	1.19E-11	1.55E-11	1.64E-11	1.90E-11	1.90E-11
Cycle 8	1.30E-11	1.49E-11	1.57E-11	1.85E-11	1.85E-11
Average (1-4)	2.48E-11	3.09E-11	2.97E-11	3.24E-11	3.23E-11
Average (1-8)	1.85E-11	2.30E-11	2.36E-11	2.76E-11	2.76E-11
Average (5-8)	1.30E-11	1.62E-11	1.83E-11	2.35E-11	2.35E-11

Table 6-3

Relative Radial Distribution Of $\phi(E > 1.0$ MeV)
Within The Reactor Vessel Wall

RADIUS	AZIMUTHAL ANGLE				
(cm)	0°	15°	30°	40°	45°
233.756	1.000	1.000	1.000	1.000	1.000
234.006	0.989	0.989	0.989	0.990	0.989
234.631	0.946	0.945	0.945	0.944	0.945
235.506	0.872	0.870	0.871	0.869	0.871
236.631	0.774	0.772	0.773	0.770	0.771
237.924	0.668	0.665	0.666	0.662	0.664
239.410	0.558	0.554	0.555	0.551	0.553
241.197	0.446	0.442	0.444	0.440	0.441
243.205	0.344	0.341	0.343	0.339	0.340
245.063	0.269	0.267	0.269	0.265	0.265
246.478	0.222	0.220	0.221	0.219	0.218
247.780	0.185	0.183	0.184	0.183	0.182
249.192	0.152	0.150	0.151	0.150	0.150
250.716	0.123	0.121	0.121	0.121	0.120
252.056	0.101	0.099	0.100	0.099	0.099
253.098	0.086	0.085	0.085	0.084	0.084
254.182	0.073	0.071	0.072	0.071	0.071
255.182	0.062	0.060	0.060	0.059	0.059
255.994	0.053	0.051	0.051	0.050	0.050
256.369	0.051	0.049	0.049	0.048	0.047

Note:
Base Metal Inner Radius = 233.756 cm
Base Metal 1/4T = 239.409 cm
Base Metal 1/2T = 245.063 cm
Base Metal 3/4T = 250.716 cm
Base Metal Outer Radius = 256.369 cm

Table 6-4

Relative Radial Distribution Of $\phi(E > 0.1$ MeV)
Within The Reactor Vessel Wall

RADIUS (cm)	AZIMUTHAL ANGLE				
	0°	15°	30°	40°	45°
233.756	1.000	1.000	1.000	1.000	1.000
234.006	1.010	1.010	1.010	1.010	1.009
234.631	1.014	1.011	1.013	1.010	1.011
235.506	1.001	0.996	0.998	0.994	0.996
236.631	0.969	0.963	0.966	0.959	0.961
237.924	0.924	0.915	0.920	0.911	0.913
239.410	0.866	0.855	0.861	0.850	0.852
241.197	0.794	0.781	0.788	0.775	0.777
243.205	0.713	0.700	0.707	0.693	0.694
245.063	0.641	0.627	0.634	0.619	0.619
246.478	0.586	0.572	0.578	0.564	0.564
247.780	0.536	0.521	0.527	0.514	0.514
249.192	0.485	0.469	0.475	0.461	0.461
250.716	0.431	0.415	0.420	0.406	0.406
252.056	0.385	0.370	0.374	0.360	0.359
253.098	0.349	0.333	0.337	0.323	0.322
254.182	0.312	0.296	0.299	0.285	0.284
255.182	0.278	0.262	0.263	0.249	0.248
255.994	0.247	0.231	0.231	0.217	0.216
256.369	0.239	0.222	0.222	0.207	0.206

Note:
Base Metal Inner Radius = 233.756 cm
Base Metal 1/4T = 239.409 cm
Base Metal 1/2T = 245.063 cm
Base Metal 3/4T = 250.716 cm
Base Metal Outer Radius = 256.369 cm

Table 6-5

Relative Radial Distribution Of dpa/sec
Within The Reactor Vessel Wall

RADIUS (cm)	AZIMUTHAL ANGLE				
	0°	15°	30°	40°	45°
233.756	1.000	1.000	1.000	1.000	1.000
234.006	0.990	0.990	0.991	0.991	0.990
234.631	0.953	0.952	0.953	0.952	0.953
235.506	0.891	0.890	0.891	0.889	0.890
236.631	0.810	0.808	0.810	0.807	0.808
237.924	0.721	0.719	0.722	0.717	0.719
239.410	0.629	0.626	0.629	0.624	0.626
241.197	0.534	0.530	0.534	0.528	0.530
243.205	0.444	0.440	0.444	0.438	0.439
245.063	0.374	0.371	0.375	0.369	0.369
246.478	0.327	0.324	0.328	0.322	0.322
247.780	0.289	0.285	0.288	0.284	0.284
249.192	0.253	0.248	0.251	0.247	0.247
250.716	0.218	0.213	0.216	0.211	0.211
252.056	0.190	0.185	0.188	0.183	0.182
253.098	0.169	0.164	0.167	0.162	0.161
254.182	0.150	0.144	0.146	0.141	0.141
255.182	0.132	0.126	0.128	0.122	0.122
255.994	0.117	0.111	0.112	0.106	0.106
256.369	0.113	0.107	0.107	0.102	0.101

Note:
Base Metal Inner Radius = 233.756 cm
Base Metal 1/4T = 239.409 cm
Base Metal 1/2T = 245.063 cm
Base Metal 3/4T = 250.716 cm
Base Metal Outer Radius = 256.369 cm

Table 6-6

Nuclear Parameters Used In The Evaluation Of Neutron Sensors

Monitor Material	Atomic Weight	Reaction of Interest	Target Atom Fraction	Response Range	Product Half-life	Fission Yield (%)
Copper	63.546	$Cu^{63}(n,\alpha)Co^{60}$	0.6917	E > 5 Mev	1925.5d	
Iron	55.845	$Fe^{54}(n,p)Mn^{54}$	0.0585	E > 2 Mev	312.3d	
Nickel	58.693	$Ni^{58}(n,p)Co^{58}$	0.6808	E > 2 Mev	70.82d	
Titanium	45.953	$Ti^{46}(n,p)Sc^{46}$	0.0825	E > 2 Mev	83.79d	
Uranium-238	238.051	$U^{238}(n,f)Cs^{137}$	0.9996	E > 1 Mev	10983.3d	6.02
Uranium-238	238.051	$U^{238}(n,f)Zr^{95}$	0.9996	E > 1 Mev	64.02d	5.15
Uranium-238	238.051	$U^{238}(n,f)Ru^{103}$	0.9996	E > 1 Mev	39.27d	6.26
Cobalt-Al	58.933	$Co^{59}(n,\gamma)Co^{60}$	0.0017	Non-threshold	1925.5d	

Notes:

1. Atomic weight data taken from the Chart of the Nuclides, 15th Edition, Dated 1996.
2. Half-life data and target fraction data for the $Cu^{63}(n,\alpha)$, $Fe^{54}(n,p)$, $Ni^{58}(n,p)$, $Ti^{46}(n,p)$, and $Co^{59}(n,\gamma)$ reactions were taken from ASTM Standard E 1005-97.
3. Half-life and fission yield data for the $U^{238}(n,f)$ reaction taken from ASTM Standard E 1005-97.
4. Target atom fraction for the U^{238} assumed as 350 ppm of U^{235}.

Table 6-7

Monthly Thermal Generation During The First Eight Fuel Cycles
Of The Palo Verde Unit 1 Reactor
(Reactor Power of 3800 MWt)

Cycle 1		Cycle 2		Cycle 3		Cycle 4	
Mo-Year	Thermal Generation (MWt-hr)	Mo-Year	Thermal Generation (MWt-hr)	Mo-Year	Thermal Generation (MWt-hr)	Mo-Year	Thermal Generation (MWt-hr)
Jun-85	488193	Mar-88	1667501	Jun-90	10825	May-92	389734
Jul-85	488786	Apr-88	2300328	Jul-90	2146547	Jun-92	2726926
Aug-85	83645	May-88	2371300	Aug-90	2337547	Jul-92	2825923
Sep-85	1069383	Jun-88	2698106	Sep-90	1652836	Aug-92	2825394
Oct-85	962630	Jul-88	503771	Oct-90	2797086	Sep-92	2545283
Nov-85	194	Aug-88	69686	Nov-90	2704935	Oct-92	2610331
Dec-85	1460729	Sep-88	2565055	Dec-90	2813104	Nov-92	2734833
Jan-86	1265571	Oct-88	2659128	Jan-91	996506	Dec-92	2576546
Feb-86	2011169	Nov-88	2708768	Feb-91	1084003	Jan-93	2645967
Mar-86	566400	Dec-88	2789443	Mar-91	2822719	Feb-93	2461534
Apr-86	0	Jan-89	2775471	Apr-91	2731759	Mar-93	2806238
May-86	597004	Feb-89	2385008	May-91	2824227	Apr-93	2735243
Jun-86	2473534	Mar-89	404819	Jun-91	2725393	May-93	2606624
Jul-86	1814285			Jul-91	2825868	Jun-93	2730282
Aug-86	1746682			Aug-91	2824884	Jul-93	2487672
Sep-86	1929290			Sep-91	1725887	Aug-93	2019059
Oct-86	2413152			Oct-91	2386029	Sep-93	174976
Nov-86	2440676			Nov-91	2700441		
Dec-86	2768558			Dec-91	2822439		
Jan-87	1462574			Jan-92	1430983		
Feb-87	0			Feb-92	1248802		
Mar-87	1861757						
Apr-87	2705174						
May-87	2615160						
Jun-87	2387981						
Jul-87	29941						
Aug-87	2431392						
Sep-87	2502710						
Oct-87	144005						

Table 6-7 (Continued)

Monthly Thermal Generation During The First Eight Fuel Cycles
Of The Palo Verde Unit 1 Reactor
(Reactor Power of 3800 MWt)

Cycle 5		Cycle 6		Cycle 7		Cycle 8	
Mo-Year	Thermal Generation (MWt-hr)	Mo-Year	Thermal Generation (MWt-hr)	Mo-Year	Thermal Generation (MWt-hr)	Mo-Year	Thermal Generation (MWt-hr)
Nov-93	175241	May-95	124734	Oct-96	19736	Apr-98	856293
Dec-93	2408701	Jun-95	2522346	Nov-96	2575629	May-98	2882404
Jan-94	2402290	Jul-95	2818244	Dec-96	2882996	Jun-98	2790255
Feb-94	2170013	Aug-95	2672598	Jan-97	2882444	Jul-98	2879986
Mar-94	2397621	Sep-95	2732753	Feb-97	2602247	Aug-98	2874823
Apr-94	2342627	Oct-95	2826498	Mar-97	2852026	Sep-98	2790115
May-94	2430079	Nov-95	2422865	Apr-97	2790215	Oct-98	2883307
Jun-94	2361925	Dec-95	2334446	May-97	2441952	Nov-98	2790301
Jul-94	2768631	Jan-96	2826311	Jun-97	2774045	Dec-98	2883381
Aug-94	2763506	Feb-96	2267706	Jul-97	2882408	Jan-99	2883251
Sep-94	2669901	Mar-96	2519048	Aug-97	2876879	Feb-99	2603835
Oct-94	2769680	Apr-96	1373041	Sep-97	2789493	Mar-99	2581165
Nov-94	2256552	May-96	2742983	Oct-97	2842908	Apr-99	2790255
Dec-94	2718645	Jun-96	2735425	Nov-97	2782268	May-99	2883223
Jan-95	2808750	Jul-96	2821260	Dec-97	2883231	Jun-99	2786766
Feb-95	2552287	Aug-96	2630990	Jan-98	2883225	Jul-99	2882991
Mar-95	2638589	Sep-96	1682394	Feb-98	2272117	Aug-99	2882823
Apr-95	76			Mar-98	1202072	Sep-99	2778552
						Oct-99	85396

Table 6-8

Measured Sensor Activities And Reaction Rates

Surveillance Capsule W137

Reaction	Location	Measured Activity (dps/gm)	Saturated Activity (dps/gm)	Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.04E+05	2.985E+05	4.553E-17
	Middle	9.81E+04	2.815E+05	4.295E-17
	Bottom	9.89E+04	2.838E+05	4.330E-17
^{54}Fe (n,p) ^{54}Mn	Top	9.79E+05	2.218E+06	3.516E-15
	Middle	9.09E+05	2.059E+06	3.265E-15
	Bottom	9.21E+05	2.087E+06	3.308E-15
^{58}Ni (n,p) ^{58}Co	Middle	3.36E+06	2.979E+07	4.265E-15
^{46}Ti (n,p) ^{46}Sc	Top	1.09E+05	7.165E+05	6.627E-16
	Middle	1.03E+05	6.770E+05	6.262E-16
	Bottom	1.03E+05	6.770E+05	6.262E-16
^{59}Co (n,γ) ^{60}Co	Middle	5.24E+07	1.504E+08	8.657E-12
^{59}Co (n,γ) ^{60}Co (Cd)	Middle	6.35E+06	1.822E+07	1.049E-12
^{238}U (n,f) ^{137}Cs (Cd)	Top	4.12E+04	4.341E+05	2.852E-15
	Middle	1.00E+05	1.054E+06	6.922E-15
	Bottom	7.56E+04	7.966E+05	5.233E-15
^{238}U (n,f) ^{95}Zr (Cd)	Top	4.08E+04	4.449E+05	3.416E-15
	Middle	8.72E+04	9.509E+05	7.300E-15
	Bottom	6.54E+04	7.132E+05	5.475E-15
^{238}U (n,f) ^{103}Ru (Cd)	Top	1.37E+04	5.853E+05	3.697E-15
	Middle	2.70E+05	1.154E+07	7.286E-14
	Bottom	1.94E+04	8.288E+05	5.235E-15
^{238}U (n,f) ^{137}Cs	Top	2.06E+05	2.171E+06	1.426E-14
	Middle	3.68E+05	3.878E+06	2.547E-14
	Bottom	2.87E+05	3.024E+06	1.987E-14
^{238}U (n,f) ^{95}Zr	Top	2.34E+05	2.552E+06	1.959E-14
	Middle	4.18E+05	4.558E+06	3.499E-14
	Bottom	3.03E+05	3.304E+06	2.537E-14
^{238}U (n,f) ^{103}Ru	Top	6.27E+04	2.679E+06	1.692E-14
	Middle	1.08E+05	4.614E+06	2.914E-14
	Bottom	7.09E+04	3.029E+06	1.913E-14

Table 6-8 cont'd

Measured Sensor Activities And Reaction Rates

Surveillance Capsule W38

Reaction	Location	Measured Activity (dps/gm)	Saturated Activity (dps/gm)	Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	Top	1.320E+05	2.494E+05	3.805E-17
	Middle	1.510E+05	2.853E+05	4.352E-17
	Bottom	1.140E+05	2.154E+05	3.286E-17
^{54}Fe (n,p) ^{54}Mn	Top	5.940E+05	1.735E+06	2.751E-15
	Middle	5.560E+05	1.624E+06	2.575E-15
	Bottom	5.470E+05	1.598E+06	2.533E-15
^{58}Ni (n,p) ^{58}Co	Middle	9.210E+05	2.538E+07	3.634E-15
^{46}Ti (n,p) ^{46}Sc	Top	2.550E+04	4.491E+05	4.154E-16
	Middle	3.150E+04	5.548E+05	5.131E-16
	Bottom	3.170E+04	5.583E+05	5.164E-16
^{59}Co (n,γ) ^{60}Co	Middle	6.880E+07	1.300E+08	7.483E-12
^{238}U (n,f) ^{137}Cs	Top	6.790E+05	3.581E+06	2.352E-14
	Middle	4.110E+05	2.167E+06	1.424E-14
	Bottom	1.080E+06	5.695E+06	3.741E-14
^{238}U (n,f) ^{95}Zr	Top	1.000E+05	3.750E+06	2.879E-14
	Middle	6.840E+05	2.565E+07	1.969E-13
	Bottom	1.740E+05	6.524E+07	5.009E-14
^{238}U (n,f) ^{103}Ru	Top	1.360E+04	3.860E+06	2.438E-14
	Middle	8.700E+03	2.470E+06	1.560E-14
	Bottom	2.410E+04	6.481E+06	4.321E-14

Table 6-9

Summary Of Neutron Dosimetry Results
Surveillance Capsule W137

Best Estimate Flux and Fluence for Capsule W137

Quantity	Flux [n/cm^2-sec]	Quantity	Fluence [n/cm^2]	Uncertainty
ϕ (E > 1.0 MeV)	2.584E+10	Φ (E > 1.0 MeV)	3.724E+18	7%
ϕ (E > 0.1 MeV)	5.165E+10	Φ (E > 0.1 MeV)	7.444E+18	10%
ϕ (E < 0.414 eV)	2.790E+11	Φ (E < 0.414 eV)	4.021E+19	7%
dpa/sec	4.018E-11	dpa	5.791E-03	6%

Best Estimate Flux and Fluence for Capsule W38

Quantity	Flux [n/cm^2-sec]	Quantity	Fluence [n/cm^2]	Uncertainty
ϕ (E > 1.0 MeV)	2.041E+10	Φ (E > 1.0 MeV)	6.320E+18	7%
ϕ (E > 0.1 MeV)	3.974E+10	Φ (E > 0.1 MeV)	1.231E+19	10%
ϕ (E < 0.414 eV)	2.583E+11	Φ (E < 0.414 eV)	7.998E+19	6%
dpa/sec	3.187E-11	dpa	9.868E-03	6%

Table 6-10

Comparison Of Measured, Calculated, And Best Estimate
Reaction Rates At The Surveillance Capsule Center

Surveillance Capsule W137

Reaction	Measured	Calculated	Best Estimate	BE / Meas	BE/ Calc	Meas/Calc
$^{63}Cu\ (n,\alpha)^{60}Co$	4.39E-17	4.79E-17	4.30E-17	0.98	0.90	0.92
$^{54}Fe\ (n,p)^{54}Mn$	3.36E-15	4.10E-15	3.44E-15	1.02	0.84	0.82
$^{58}Ni\ (n,p)^{58}Co$	4.26E-15	5.33E-15	4.45E-15	1.04	0.83	0.80
$^{46}Ti\ (n,p)^{46}Sc$	6.38E-16	7.39E-16	6.38E-16	1.00	0.86	0.86
$^{59}Co\ (n,\gamma)^{60}Co$	8.66E-12	1.38E-12	8.47E-12	0.98	6.14	6.28
$^{59}Co\ (n,\gamma)^{60}Co$ (Cd)	1.05E-12	2.89E-13	1.01E-12	0.96	3.49	3.63

Surveillance Capsule W38

Reaction	Measured	Calculated	Best Estimate	BE / Meas	BE/ Calc	Meas/Calc
$^{63}Cu\ (n,\alpha)^{60}Co$	3.81E-17	4.02E-17	3.57E-17	0.94	0.89	0.95
$^{54}Fe\ (n,p)^{54}Mn$	2.62E-15	3.44E-15	2.75E-15	1.05	0.80	0.76
$^{58}Ni\ (n,p)^{58}Co$	3.63E-15	4.48E-15	3.63E-15	1.00	0.81	0.81
$^{46}Ti\ (n,p)^{46}Sc$	4.82E-16	6.20E-16	5.05E-16	1.05	0.81	0.78
$^{59}Co\ (n,\gamma)^{60}Co$	7.48E-12	1.12E-12	7.31E-12	0.98	6.53	6.68

Table 6-11

Best Estimate Neutron Energy Spectrum At The
Center Of Surveillance Capsules

Capsule W137

Group Number	Energy (MeV)	Flux (n/cm^2-sec)	Group #	Energy (MeV)	Flux (n/cm^2-sec)
1	1.73E+01	7.464E+06	28	9.12E-03	2.551E+09
2	1.49E+01	1.539E+07	29	5.53E-03	2.529E+09
3	1.35E+01	5.215E+07	30	3.36E-03	8.815E+08
4	1.16E+01	1.314E+08	31	2.84E-03	9.226E+08
5	1.00E+01	2.861E+08	32	2.40E-03	9.861E+08
6	8.61E+00	4.696E+08	33	2.04E-03	3.216E+09
7	7.41E+00	1.162E+09	34	1.23E-03	3.520E+09
8	6.07E+00	1.626E+09	35	7.49E-04	3.813E+09
9	4.97E+00	2.799E+09	36	4.54E-04	4.112E+09
10	3.68E+00	2.580E+09	37	2.75E-04	4.759E+09
11	2.87E+00	4.124E+09	38	1.67E-04	8.040E+09
12	2.23E+00	3.933E+09	39	1.01E-04	4.976E+09
13	1.74E+00	4.046E+09	40	6.14E-05	4.629E+09
14	1.35E+00	3.114E+09	41	3.73E-05	4.221E+09
15	1.11E+00	4.378E+09	42	2.26E-05	3.821E+09
16	8.21E-01	4.012E+09	43	1.37E-05	3.462E+09
17	6.39E-01	3.740E+09	44	8.32E-06	3.226E+09
18	4.98E-01	2.635E+09	45	5.04E-06	3.144E+09
19	3.88E-01	3.108E+09	46	3.06E-06	3.078E+09
20	3.02E-01	4.676E+09	47	1.86E-06	2.996E+09
21	1.83E-01	4.083E+09	48	1.13E-06	2.817E+09
22	1.11E-01	3.177E+09	49	6.83E-07	3.058E+09
23	6.74E-02	2.971E+09	50	4.14E-07	4.465E+09
24	4.09E-02	2.094E+09	51	2.51E-07	1.949E+10
25	2.55E-02	1.528E+09	52	1.52E-07	4.378E+10
26	1.99E-02	1.203E+09	53	9.24E-08	2.112E+11
27	1.50E-02	2.504E+09			

Note: Tabulated energy levels represent the upper energy in each group.

Table 6-11 cont'd

Best Estimate Neutron Energy Spectrum At The
Center Of Surveillance Capsules

Capsule W38

Group Number	Energy (MeV)	Flux (n/cm^2-sec)	Group #	Energy (MeV)	Flux (n/cm^2-sec)
1	1.73E+01	5.948E+06	28	1.73E+09	1.728E+09
2	1.49E+01	1.237E+07	29	1.67E+09	1.671E+09
3	1.35E+01	4.210E+07	30	5.63E+08	5.633E+08
4	1.16E+01	1.065E+08	31	5.63E+08	5.639E+08
5	1.00E+01	2.329E+08	32	5.66E+08	5.673E+08
6	8.61E+00	3.820E+08	33	1.70E+09	1.708E+09
7	7.41E+00	9.447E+08	34	1.68E+09	1.692E+09
8	6.07E+00	1.309E+09	35	1.62E+09	1.639E+09
9	4.97E+00	2.239E+09	36	1.57E+09	1.589E+09
10	3.68E+00	2.060E+09	37	1.66E+09	1.692E+09
11	2.87E+00	3.270E+09	38	1.70E+09	1.752E+09
12	2.23E+00	3.094E+09	39	1.69E+09	1.738E+09
13	1.74E+00	3.158E+09	40	1.69E+09	1.742E+09
14	1.35E+00	2.410E+09	41	1.69E+09	1.753E+09
15	1.11E+00	3.361E+09	42	1.68E+09	1.762E+09
16	8.21E-01	3.055E+09	43	1.66E+09	1.755E+09
17	6.39E-01	2.826E+09	44	1.65E+09	1.767E+09
18	4.98E-01	1.976E+09	45	1.68E+09	1.827E+09
19	3.88E-01	2.315E+09	46	1.69E+09	1.872E+09
20	3.02E-01	3.456E+09	47	1.68E+09	1.890E+09
21	1.83E-01	3.001E+09	48	1.60E+09	1.830E+09
22	1.11E-01	2.318E+09	49	1.43E+09	2.029E+09
23	6.74E-02	2.154E+09	50	1.70E+09	3.084E+09
24	4.09E-02	1.504E+09	51	5.49E+09	1.411E+10
25	2.55E-02	1.088E+09	52	9.28E+09	3.409E+10
26	1.99E-02	8.444E+08	53	2.01E+10	2.070E+11
27	1.50E-02	1.732E+09			

Note: Tabulated energy levels represent the upper energy in each group.

Table 6-12

Comparison Of Calculated And Best Estimate Integrated Neutron Exposure Of Palo Verde Unit 1 Surveillance Capsules W137 and W38

CAPSULE W137

	Calculated	Best Estimate	BE/C
$\Phi(E > 1.0$ MeV) [n/cm^2]	4.33E+18	3.72E+18	0.86
$\Phi(E > 0.1$ MeV) [n/cm^2]	7.94E+18	7.44E+18	0.94
dpa	6.23E-03	5.79E-03	0.93

CAPSULE W38

	Calculated	Best Estimate	BE/C
$\Phi(E > 1.0$ MeV) [n/cm^2]	7.85E+18	6.32E+18	0.80
$\Phi(E > 0.1$ MeV) [n/cm^2]	1.45E+19	1.23E+19	0.85
dpa	1.13E-02	9.87E-03	0.87

AVERAGE BE/C RATIOS

	BE/C
$\Phi(E > 1.0$ MeV) [n/cm^2]	0.832
$\Phi(E > 0.1$ MeV) [n/cm^2]	0.894
dpa	0.902

Table 6-13

Azimuthal Variations Of The Neutron Exposure Projections
On The Reactor Vessel Clad/Base Metal Interface
At Maximum Fluence Elevation

Best Estimate

	0°	15°	30°	42.3°	45°
9.81 EFPY					
E>1.0 MeV	3.08E+18	3.87E+18	3.96E+18	4.65E+18	4.65E+18
E>0.1 MeV	6.93E+18	8.75E+18	9.03E+18	1.06E+19	1.06E+19
dpa	5.17E-03	6.43E-03	6.59E-03	7.72E-03	7.71E-03
15 EFPY					
E>1.0 MeV	4.23E+18	5.30E+18	5.58E+18	6.74E+18	6.74E+18
E>0.1 MeV	9.51E+18	1.20E+19	1.27E+19	1.54E+19	1.54E+19
dpa	7.09E-03	8.81E-03	9.30E-03	1.12E-02	1.12E-02
32 EFPY					
E>1.0 MeV	7.99E+18	1.00E+19	1.09E+19	1.36E+19	1.36E+19
E>0.1 MeV	1.79E+19	2.26E+19	2.48E+19	3.09E+19	3.10E+19
dpa	1.34E-02	1.66E-02	1.82E-02	2.25E-02	2.26E-02
40 EFPY					
E>1.0 MeV	9.76E+18	1.22E+19	1.34E+19	1.68E+19	1.68E+19
E>0.1 MeV	2.19E+19	2.76E+19	3.05E+19	3.83E+19	3.84E+19
dpa	1.64E-02	2.03E-02	2.23E-02	2.79E-02	2.79E-02
45 EFPY					
E>1.0 MeV	1.09E+19	1.36E+19	1.50E+19	1.88E+19	1.89E+19
E>0.1 MeV	2.44E+19	3.07E+19	3.41E+19	4.28E+19	4.30E+19
dpa	1.82E-02	2.26E-02	2.49E-02	3.12E-02	3.13E-02
54 EFPY					
E>1.0 MeV	1.29E+19	1.61E+19	1.78E+19	2.24E+19	2.25E+19
E>0.1 MeV	2.88E+19	3.63E+19	4.05E+19	5.11E+19	5.12E+19
dpa	2.16E-02	2.68E-02	2.96E-02	3.72E-02	3.73E-02

Note: Maximum neutron exposure projection is at either 42.3° or 45°

Table 6-13, cont'd

Azimuthal Variations Of The Neutron Exposure Projections
On The Reactor Vessel Clad/Base Metal Interface At Maximum Fluence Elevation

Calculated

	0°	15°	30°	42.3°	45°
9.81 EFPY					
E>1.0 MeV	3.71E+18	4.64E+18	4.75E+18	5.59E+18	5.58E+18
E>0.1 MeV	7.76E+18	9.79E+18	1.01E+19	1.19E+19	1.19E+19
dpa	5.73E-03	7.12E-03	7.31E-03	8.55E-03	8.55E-03
15 EFPY					
E>1.0 MeV	5.09E+18	6.37E+18	6.70E+18	8.10E+18	8.10E+18
E>0.1 MeV	1.06E+19	1.34E+19	1.42E+19	1.72E+19	1.72E+19
dpa	7.87E-03	9.77E-03	1.03E-02	1.24E-02	1.24E-02
32 EFPY					
E>1.0 MeV	9.60E+18	1.20E+19	1.31E+19	1.63E+19	1.64E+19
E>0.1 MeV	2.01E+19	2.53E+19	2.78E+19	3.46E+19	3.47E+19
dpa	1.49E-02	1.84E-02	2.01E-02	2.50E-02	2.50E-02
40 EFPY					
E>1.0 MeV	1.17E+19	1.47E+19	1.61E+19	2.02E+19	2.02E+19
E>0.1 MeV	2.45E+19	3.09E+19	3.42E+19	4.28E+19	4.29E+19
dpa	1.82E-02	2.25E-02	2.48E-02	3.09E-02	3.10E-02
45 EFPY					
E>1.0 MeV	1.31E+19	1.63E+19	1.80E+19	2.26E+19	2.27E+19
E>0.1 MeV	2.73E+19	3.44E+19	3.81E+19	4.79E+19	4.81E+19
dpa	2.02E-02	2.51E-02	2.77E-02	3.46E-02	3.47E-02
54 EFPY					
E>1.0 MeV	1.55E+19	1.93E+19	2.14E+19	2.70E+19	2.70E+19
E>0.1 MeV	3.23E+19	4.06E+19	4.53E+19	5.72E+19	5.74E+19
dpa	2.39E-02	2.97E-02	3.29E-02	4.13E-02	4.14E-02

Note: Maximum neutron exposure projection is at either 42.3° or 45°

Table 6-14

Neutron Exposure Values Within The
Palo Verde Unit 1 Reactor Vessel

Best Estimate Fluence (n/cm^2) Based on E > 1.0 MeV Slope

	0°	15°	30°	42.3°	45°
9.81 EFPY					
Surface	3.08E+18	3.87E+18	3.96E+18	4.65E+18	4.65E+18
¼ T	1.72E+18	2.14E+18	2.20E+18	2.58E+18	2.57E+18
¾ T	3.83E+17	4.68E+17	4.82E+17	5.65E+17	5.63E+17
15 EFPY					
Surface	4.23E+18	5.30E+18	5.58E+18	6.74E+18	6.74E+18
¼ T	2.36E+18	2.94E+18	3.10E+18	3.74E+18	3.74E+18
¾ T	5.25E+17	6.42E+17	6.79E+17	8.18E+17	8.17E+17
32 EFPY					
Surface	7.99E+18	1.00E+19	1.09E+19	1.36E+19	1.36E+19
¼ T	4.47E+18	5.55E+18	6.05E+18	7.54E+18	7.54E+18
¾ T	9.92E+17	1.21E+18	1.33E+18	1.65E+18	1.65E+18
40 EFPY					
Surface	9.76E+18	1.22E+19	1.34E+19	1.68E+19	1.68E+19
¼ T	5.46E+18	6.77E+18	7.44E+18	9.33E+18	9.33E+18
¾ T	1.21E+18	1.48E+18	1.63E+18	2.04E+18	2.04E+18
45 EFPY					
Surface	1.09E+19	1.36E+19	1.50E+19	1.88E+19	1.89E+19
¼ T	6.07E+18	7.54E+18	8.31E+18	1.04E+19	1.05E+19
¾ T	1.35E+18	1.65E+18	1.82E+18	2.28E+18	2.28E+18
54 EFPY					
Surface	1.29E+19	1.61E+19	1.78E+19	2.24E+19	2.25E+19
¼ T	7.19E+18	8.92E+18	9.88E+18	1.25E+19	1.25E+19
¾ T	1.60E+18	1.95E+18	2.16E+18	2.72E+18	2.73E+18

Notes:

- Maximum neutron exposure projection is at either 42.3° or 45°
- The ¼T and ¾T values were determined using the calculational methods described in Section 6.2 and not by the empirical relation described in Regulatory Guide 1.99, Rev. 2.

Table 6-14, cont'd

Neutron Exposure Values Within The
Palo Verde Unit 1 Reactor Vessel

Best Estimate Fluence (n/cm^2) Based on dpa Slope

	0°	15°	30°	42.3°	45°
9.81 EFPY					
Surface	3.08E+18	3.87E+18	3.96E+18	4.65E+18	4.65E+18
¼ T	1.96E+18	2.43E+18	2.50E+18	2.93E+18	2.92E+18
¾ T	6.96E+17	8.34E+17	8.64E+17	1.01E+18	1.00E+18
15 EFPY					
Surface	4.23E+18	5.30E+18	5.58E+18	6.74E+18	6.74E+18
¼ T	2.68E+18	3.33E+18	3.53E+18	4.25E+18	4.24E+18
¾ T	9.55E+17	1.14E+18	1.22E+18	1.47E+18	1.45E+18
32 EFPY					
Surface	7.99E+18	1.00E+19	1.09E+19	1.36E+19	1.36E+19
¼ T	5.07E+18	6.28E+18	6.89E+18	8.57E+18	8.56E+18
¾ T	1.80E+18	2.16E+18	2.38E+18	2.95E+18	2.94E+18
40 EFPY					
Surface	9.76E+18	1.22E+19	1.34E+19	1.68E+19	1.68E+19
¼ T	6.19E+18	7.67E+18	8.47E+18	1.06E+19	1.06E+19
¾ T	2.20E+18	2.63E+18	2.93E+18	3.65E+18	3.63E+18
45 EFPY					
Surface	1.09E+19	1.36E+19	1.50E+19	1.88E+19	1.89E+19
¼ T	6.89E+18	8.54E+18	9.46E+18	1.19E+19	1.19E+19
¾ T	2.45E+18	2.93E+18	3.27E+18	4.09E+18	4.07E+18
54 EFPY					
Surface	1.29E+19	1.61E+19	1.78E+19	2.24E+19	2.25E+19
¼ T	8.16E+18	1.01E+19	1.12E+19	1.42E+19	1.41E+19
¾ T	2.90E+18	3.47E+18	3.88E+18	4.88E+18	4.85E+18

Notes:

- Maximum neutron exposure projection is at either 42.3° or 45°
- The ¼T and ¾T values were determined using the calculational methods described in Section 6.2 and not by the empirical relation described in Regulatory Guide 1.99, Rev. 2.

Table 6-14, cont'd

Neutron Exposure Values Within The
Palo Verde Unit 1 Reactor Vessel

Calculated Fluence (n/cm^2) Based on E > 1.0 MeV Slope

	0°	15°	30°	42.3°	45°
9.81 EFPY					
Surface	3.71E+18	4.64E+18	4.75E+18	5.59E+18	5.58E+18
¼ T	2.07E+18	2.58E+18	2.64E+18	3.10E+18	3.09E+18
¾ T	4.60E+17	5.62E+17	5.79E+17	6.78E+17	6.76E+17
15 EFPY					
Surface	5.09E+18	6.37E+18	6.70E+18	8.10E+18	8.10E+18
¼ T	2.84E+18	3.53E+18	3.72E+18	4.50E+18	4.49E+18
¾ T	6.31E+17	7.71E+17	8.16E+17	9.83E+17	9.81E+17
32 EFPY					
Surface	9.60E+18	1.20E+19	1.31E+19	1.63E+19	1.64E+19
¼ T	5.37E+18	6.66E+18	7.27E+18	9.06E+18	9.06E+18
¾ T	1.19E+18	1.45E+18	1.59E+18	1.98E+18	1.98E+18
40 EFPY					
Surface	1.17E+19	1.47E+19	1.61E+19	2.02E+19	2.02E+19
¼ T	6.55E+18	8.14E+18	8.94E+18	1.12E+19	1.12E+19
¾ T	1.46E+18	1.78E+18	1.96E+18	2.45E+18	2.45E+18
45 EFPY					
Surface	1.31E+19	1.63E+19	1.80E+19	2.26E+19	2.27E+19
¼ T	7.30E+18	9.06E+18	9.99E+18	1.26E+19	1.26E+19
¾ T	1.62E+18	1.98E+18	2.19E+18	2.74E+18	2.75E+18
54 EFPY					
Surface	1.55E+19	1.93E+19	2.14E+19	2.70E+19	2.70E+19
¼ T	8.63E+18	1.07E+19	1.19E+19	1.50E+19	1.50E+19
¾ T	1.92E+18	2.34E+18	2.60E+18	3.27E+18	3.27E+18

Notes:

- Maximum neutron exposure projection is at either 42.3° or 45°
- The ¼T and ¾T values were determined using the calculational methods described in Section 6.2 and not by the empirical relation described in Regulatory Guide 1.99, Rev. 2.

Table 6-14, cont'd

Neutron Exposure Values Within The
Palo Verde Unit 1 Reactor Vessel

Calculated Fluence (n/cm^2) Based on dpa Slope

	0°	15°	30°	42.3°	45°
9.81 EFPY					
Surface	3.71E+18	4.64E+18	4.75E+18	5.59E+18	5.58E+18
¼ T	2.35E+18	2.92E+18	3.00E+18	3.53E+18	3.51E+18
¾ T	8.36E+17	1.00E+18	1.04E+18	1.21E+18	1.20E+18
15 EFPY					
Surface	5.09E+18	6.37E+18	6.70E+18	8.10E+18	8.10E+18
¼ T	3.23E+18	4.00E+18	4.24E+18	5.11E+18	5.09E+18
¾ T	1.15E+18	1.37E+18	1.46E+18	1.76E+18	1.75E+18
32 EFPY					
Surface	9.60E+18	1.20E+19	1.31E+19	1.63E+19	1.64E+19
¼ T	6.09E+18	7.55E+18	8.27E+18	1.03E+19	1.03E+19
¾ T	2.17E+18	2.59E+18	2.86E+18	3.55E+18	3.53E+18
40 EFPY					
Surface	1.17E+19	1.47E+19	1.61E+19	2.02E+19	2.02E+19
¼ T	7.44E+18	9.22E+18	1.02E+19	1.27E+19	1.27E+19
¾ T	2.65E+18	3.16E+18	3.51E+18	4.39E+18	4.36E+18
45 EFPY					
Surface	1.31E+19	1.63E+19	1.80E+19	2.26E+19	2.27E+19
¼ T	8.28E+18	1.03E+19	1.14E+19	1.43E+19	1.42E+19
¾ T	2.95E+18	3.52E+18	3.93E+18	4.92E+18	4.89E+18
54 EFPY					
Surface	1.55E+19	1.93E+19	2.14E+19	2.70E+19	2.70E+19
¼ T	9.80E+18	1.21E+19	1.35E+19	1.70E+19	1.70E+19
¾ T	3.48E+18	4.17E+18	4.66E+18	5.86E+18	5.83E+18

Notes:

- Maximum neutron exposure projection is at either 42.3° or 45°
- The ¼T and ¾T values were determined using the calculational methods described in Section 6.2 and not by the empirical relation described in Regulatory Guide 1.99, Rev. 2.

Table 6-15

Updated Lead Factors For Palo Verde Unit 1
Surveillance Capsules

Capsule	Location	Capsule Fluence	Midplane Max. Wall Fluence	Lead Factor
W137[a]	43°	4.33E+18	3.05E+18	1.42
W38[b]	38°	7.85E+18	5.59E+18	1.41
W43	43°	7.95E+18	5.59E+18	1.42
W142	38°	7.86E+18	5.59E+18	1.41
W230	40°	7.99E+18	5.59E+18	1.43
W310	40°	7.99E+18	5.59E+18	1.43

Notes:
[a] - Withdrawn at the end of Cycle 4.
[b] - Withdrawn at the end of Cycle 8.

The surveillance capsule lead factor is defined by:

$$\frac{\Phi^{Surveillance\ Capsule}_{Calculated}}{\Phi^{Clad\ /\ Base\ Metal\ Interface\ Axial\ Peak}_{Calculated}}$$

where Φ is the neutron fluence ($E > 1.0$ MeV) at the time of the capsule withdrawal. In the case of the standby capsules, the neutron fluence is at the time of the latest withdrawn capsule.

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 Palo Verde Unit 1 reactor vessel. This recommended removal schedule is applicable to 32 EFPY of operation.

TABLE 7-1

Palo Verde Unit 1 Reactor Vessel Surveillance Capsule Withdrawal Schedule

Capsule	Location	Lead Factor[a]	Removal Time (EFPY)[b]	Fluence (n/cm^2, $E > 1.0$ MeV)[a]
137°	137°	1.42	4.533	4.33×10^{18} [c]
38°	38°	1.41	9.81	7.85×10^{18} [c]
230°	230°	1.43	15	1.16×10^{19}
310°	310°	1.43	EOL	2.35×10^{19} [d]
43°	43°	1.42	Standby	(e)
142°	142°	1.41	Standby	(e)

Notes:

(a) Updated in Capsule 38° dosimetry analysis.

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

(c) Plant specific evaluation.

(d) The 310° Capsule should be removed at 32 EFPY or at 37.1 EFPY if a License Renewal is obtained from the NRC.

(e) Capsules 43° and 142° will reach an EOL license renewal (54 EFPY) fluence of 2.70×10^{19} n/cm^2 ($E > 1.0$ MeV) at 38 EFPY. Thus, it is recommended that these Capsules be removed at this time and placed in storage.

8 REFERENCES

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

2. Code of Federal Regulations, 10CFR50, Appendix G, *Fracture Toughness Requirements*, and Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, U.S. Nuclear Regulatory Commission, Washington, D.C.

3. TR-F-MCM-012, "Arizona Public Service Company Palo Verde Unit 1 – Evaluation of Baseline Specimens Reactor Vessel Materials Irradiation Surveillance Program", B.C. Chang, January 31, 1987.

4. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, *Fracture Toughness Criteria for Protection Against Failure*

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

6. TR-V-MCM-002, "Summary Report on Manufacture of Test Specimens and Assembly of Capsules For Irradiation Surveillance of Palo Verde Unit 1 Reactor Vessel Materials", A.D. Emery, July 14, 1982.

7. ASTM E185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706 (IF)*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1993.

8. ASTM E23-98, *Standard Test Methods for Notched Bar Impact Testing of Metallic Materials*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1998.

9. ASTM A370-97, *Standard Test Methods and Definitions for Mechanical Testing of Steel Products*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1997.

10. ASTM E8-99, *Standard Test Methods for Tension Testing of Metallic Materials*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1999.

11. ASTM E21-92 (1998), *Standard Test Methods for Elevated Temperature Tension Tests of Metallic Materials*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1998.

12. ASTM E83-93, *Standard Practice for Verification and Classification of Extensometers*, in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1993.

13. RSIC Computer Code Collection CCC-650, "DOORS 3.1 One, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System, ", August 1996.

14. 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

15. WCAP-14066, "Analysis of the 137° Capsule from the Arizona Public Service Company Palo Verde Unit 1 Reactor Vessel Surveillance Program", J.M. Chicots, et. al., May 1994.

16. ASTM Designation E482-89 (Re-approved 1996), *Standard Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

17. ASTM Designation E560-84 (Re-approved 1996), *Standard Recommended Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

18. ASTM Designation E693-94, *Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements per Atom (dpa)*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

19. ASTM Designation E706-87 (Re-approved 1994), *Standard Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standard*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

20. ASTM Designation E853-87 (Re-approved 1995), *Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

21. ASTM Designation E261-98, *Standard Practice for Determining Neutron Fluence Rate, Fluence, and Spectra by Radioactivation Techniques*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

22. ASTM Designation E262-97, *Standard Method for Determining Thermal Neutron Reaction and Fluence Rates by Radioactivation Techniques*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

23. ASTM Designation E263-00, *Standard Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

24. ASTM Designation E264-92 (Re-approved 1996), *Standard Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

25. ASTM Designation E481-97, *Standard Method for Measuring Neutron-Fluence Rate by Radioactivation of Cobalt and Silver*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

26. ASTM Designation E523-92 (Re-approved 1996), *Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

27. ASTM Designation E704-96, *Standard Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

28. ASTM Designation E705-96, *Standard Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

29. ASTM Designation E1005-97, *Standard Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance*, in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 2000.

30. Electronic Mail Correspondence, "Monthly Thermal Power Histoty and Reactor Physics Data", from Messrs. Fernandez/Neville (APS) to Mr. Bencini (Westinghouse), July – August 2000.

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

32. W. N. McElroy, S. Berg and T. Crocket, *A Computer-Automated Iterative Method of Neutron Flux Spectra Determined by Foil Activation*, AFWL-TR-7-41, Vol. I-IV, Air Force Weapons Laboratory, Kirkland AFB, NM, July 1967

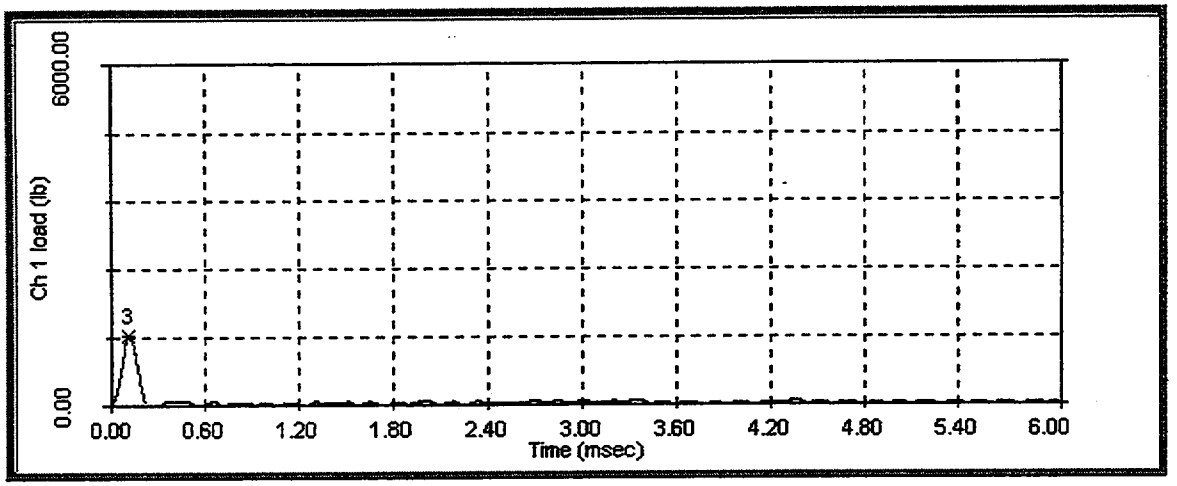
33. RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium", July 1994.

34. EPRI-NP-2188, *Development and Demonstration of an Advanced Methodology for LWR Dosimetry Applications*, R. E. Maerker, et al., 1981.

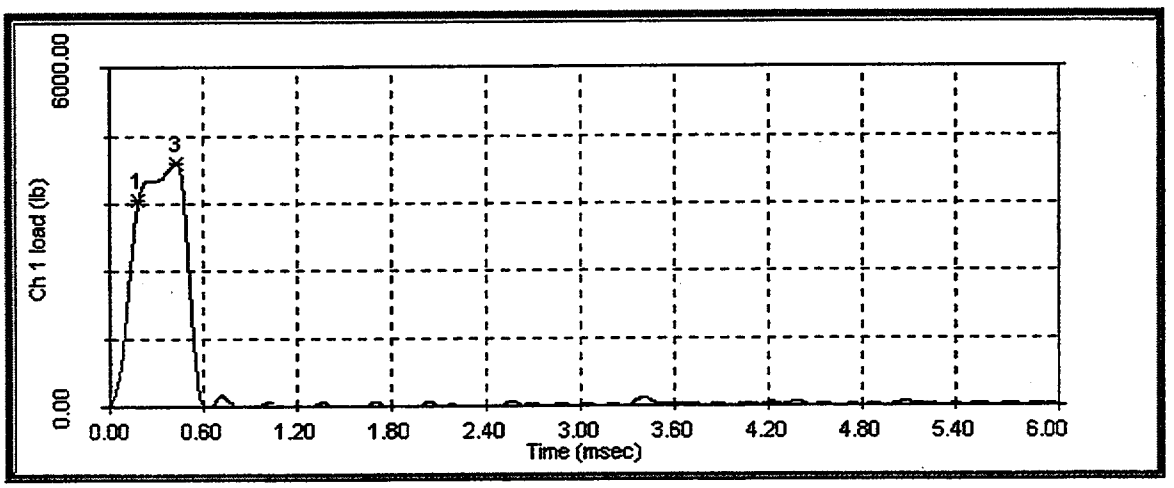
APPENDIX A

INSTRUMENTED CHARPY IMPACT TEST CURVES

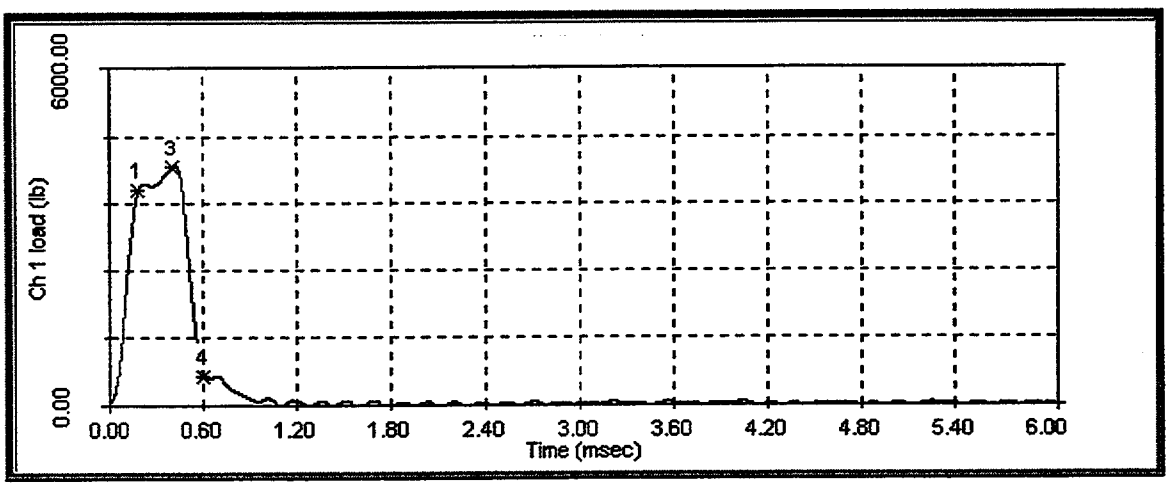
- Specimen prefix "1A1" denotes Intermediate Plate, Longitudinal Orientation
- Specimen prefix "1A2" denotes Intermediate Plate, Transverse Orientation
- Specimen prefix "1A3" denotes weld material
- Specimen prefix "1A4" denotes Heat-Affected Zone material
- Specimen prefix "1AB" denotes Standard Reference Material Plate, Longitudinal Orientation



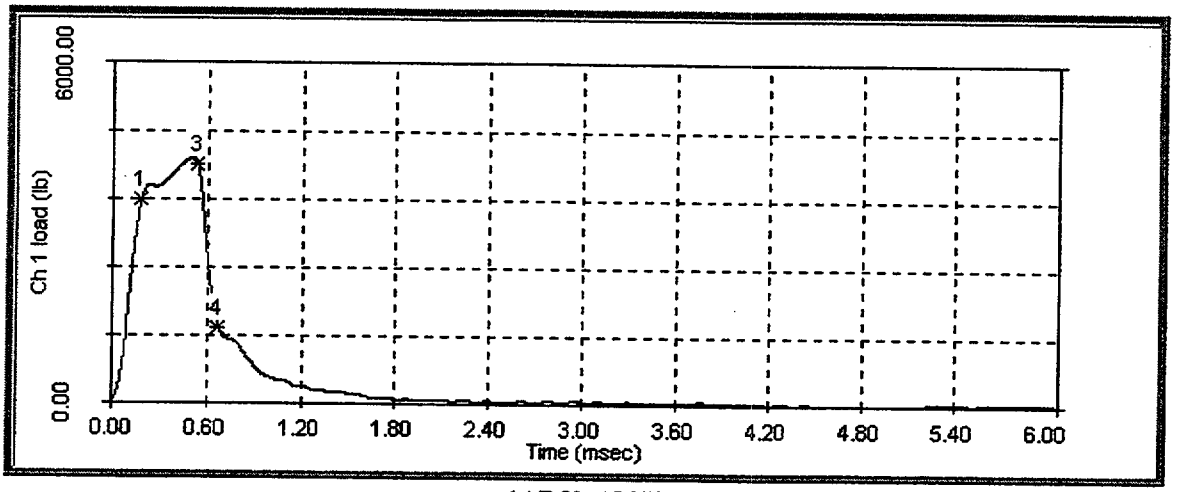
1AB6M, 0°F



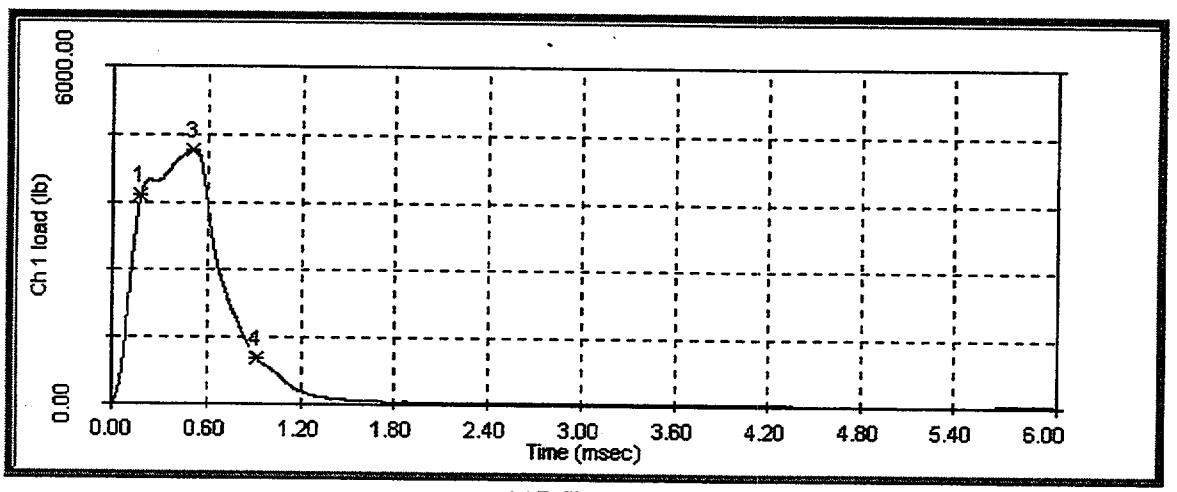
1AB56, 100°F



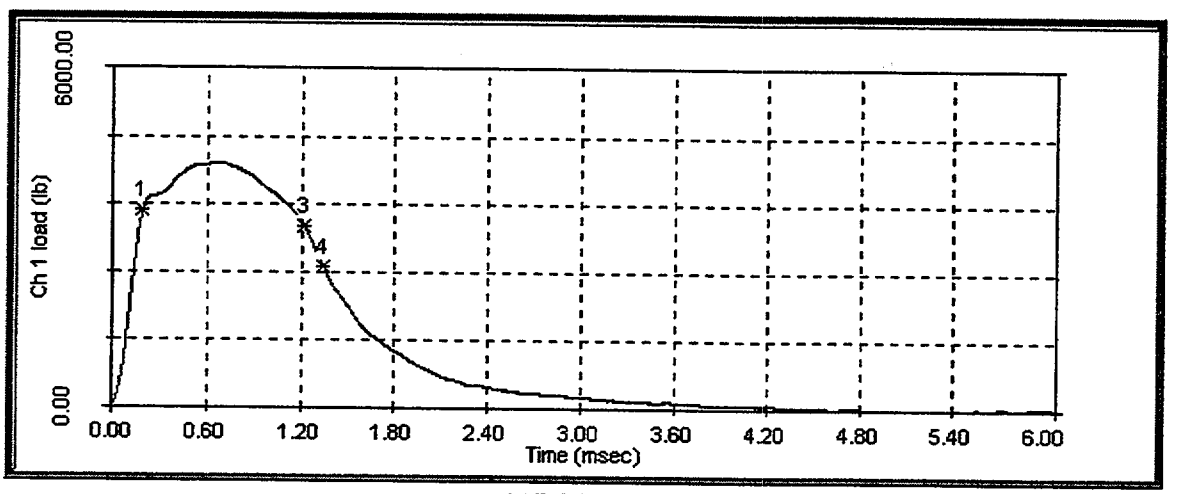
1AB4A, 125°F



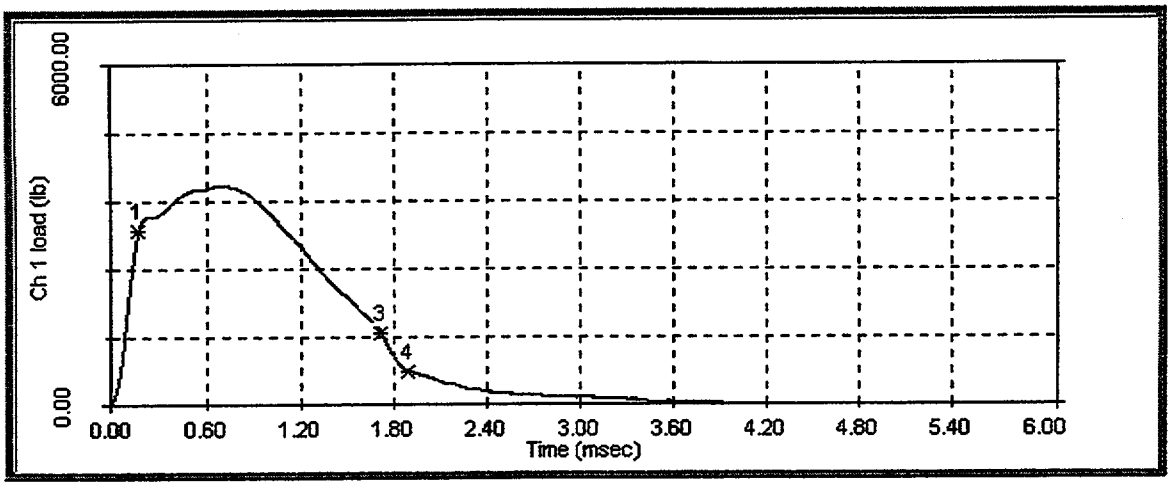
1AB53, 175°F



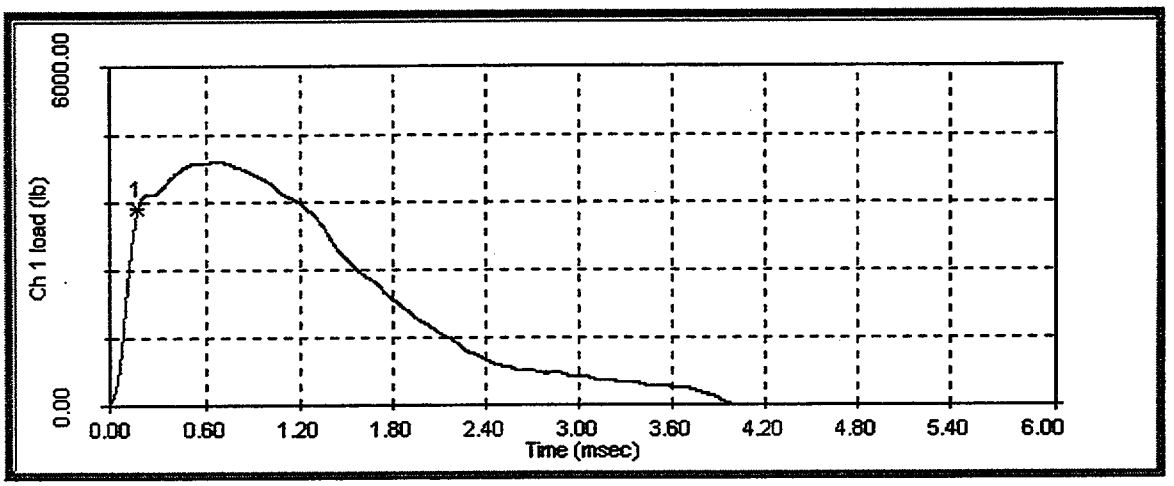
1AB6P, 200°F



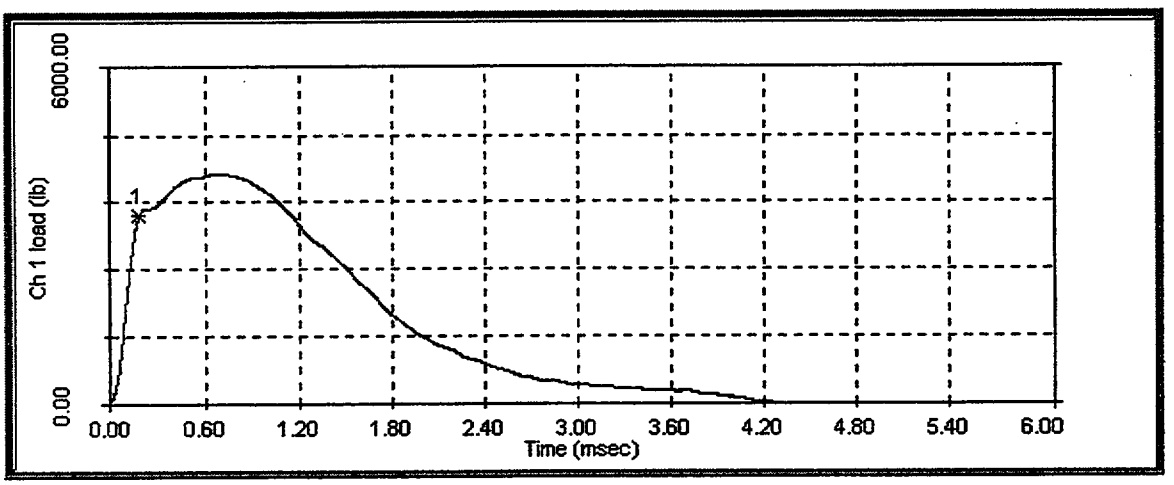
1AB5K, 225°F



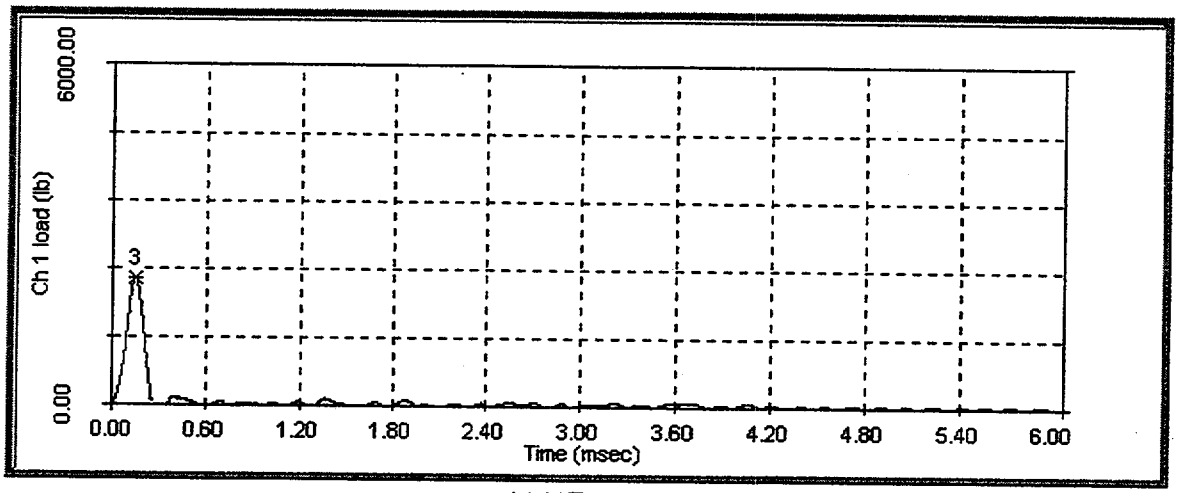
1AB4B, 250°F



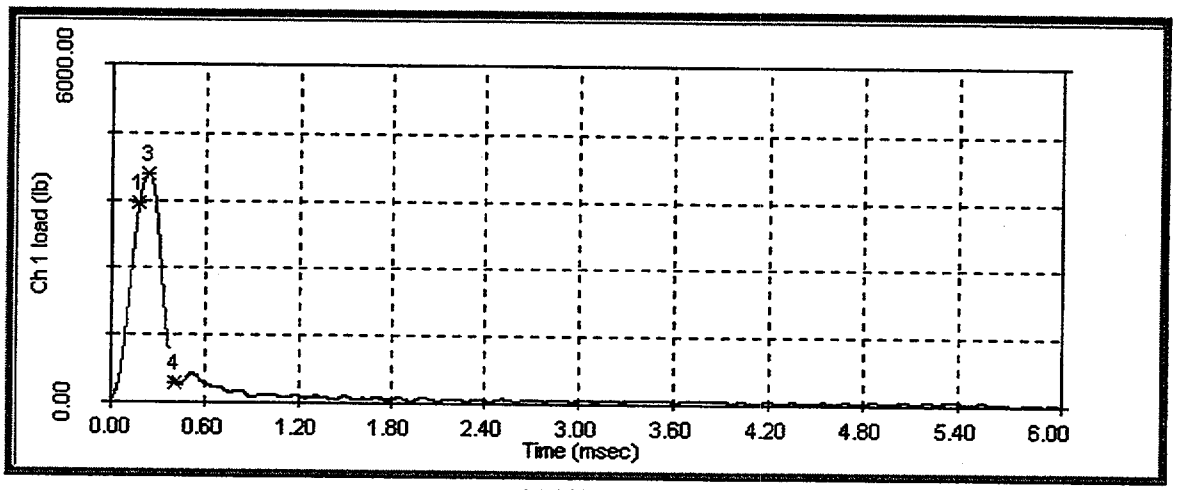
1AB44, 325°F



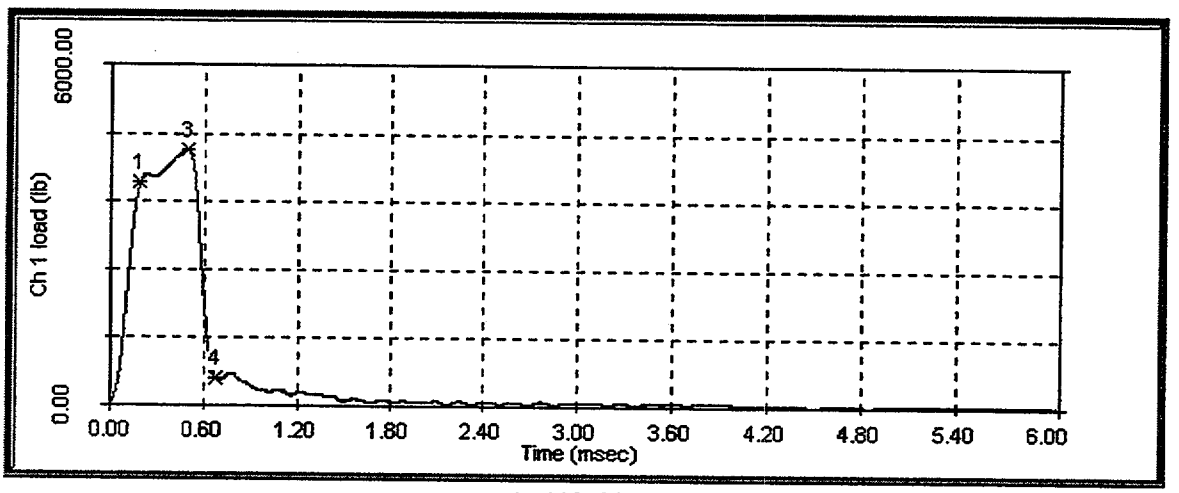
1AB45, 375°F



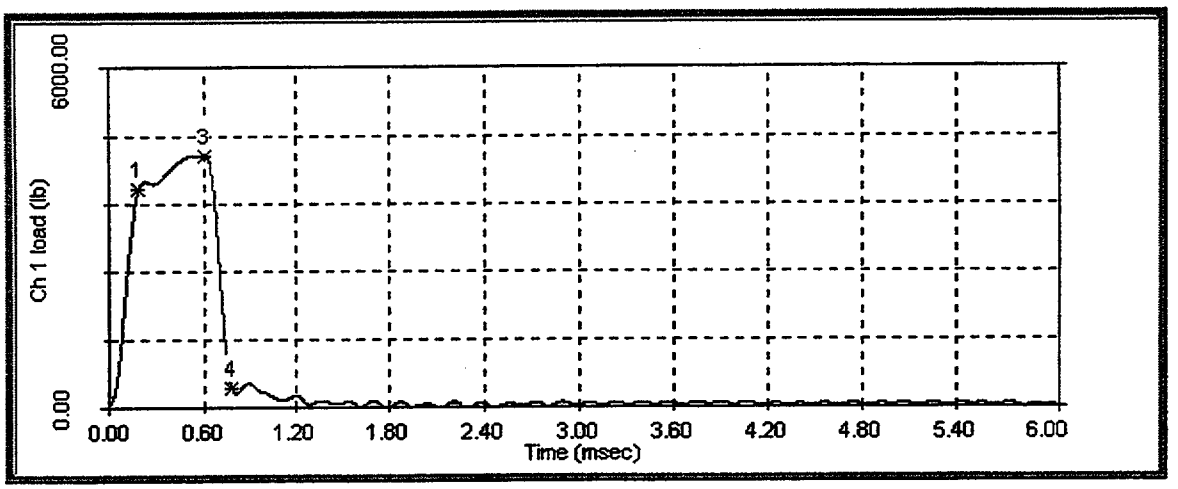
1A11T, -75°F



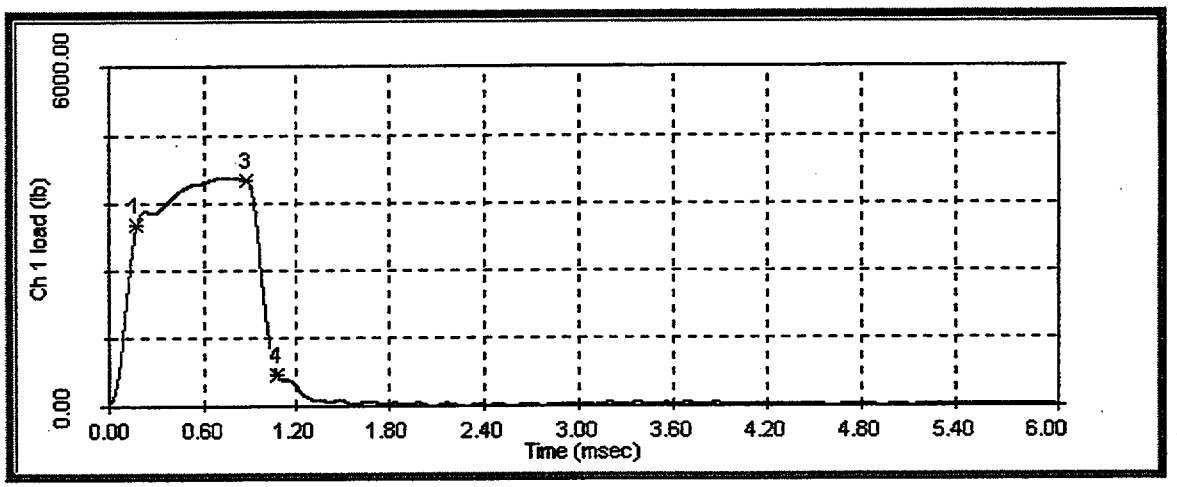
1A127, 0°F



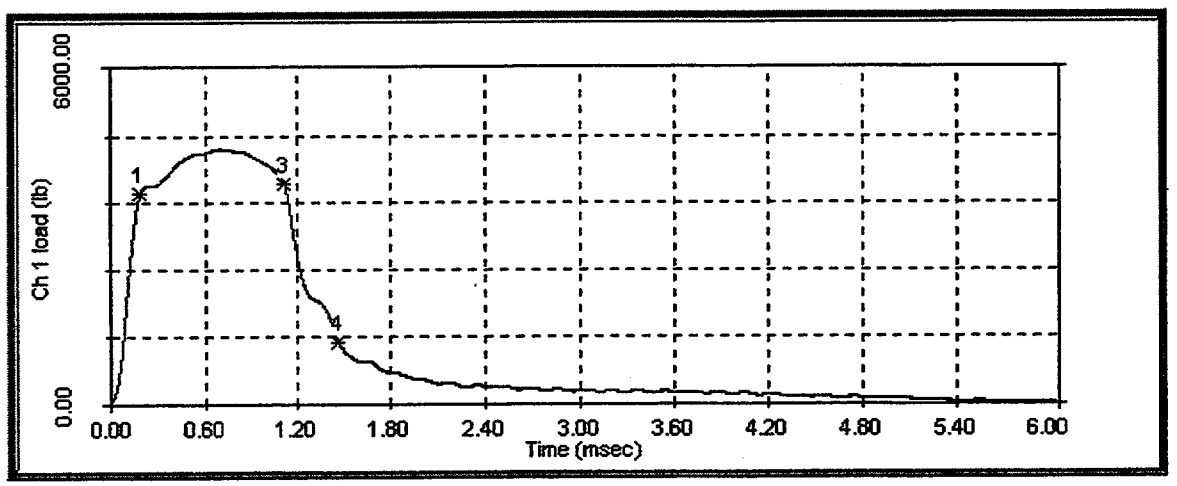
1A112, 20°F



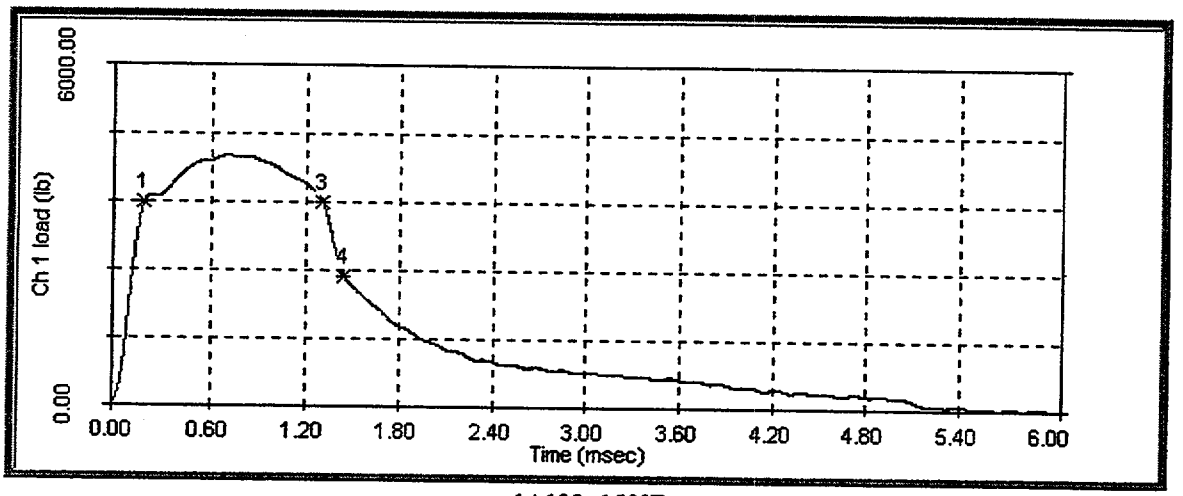
1A125, 30°F



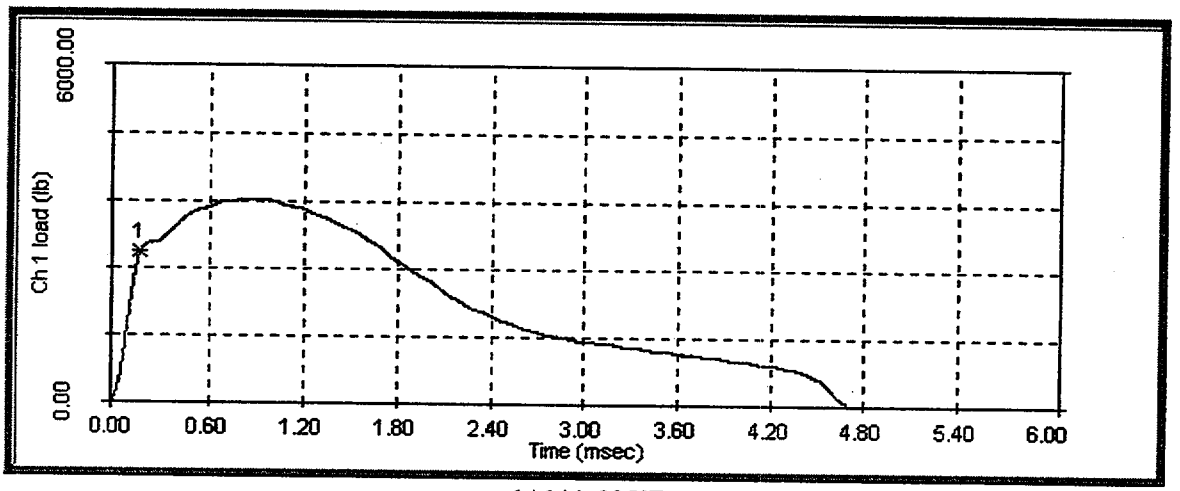
1A13U, 50°F



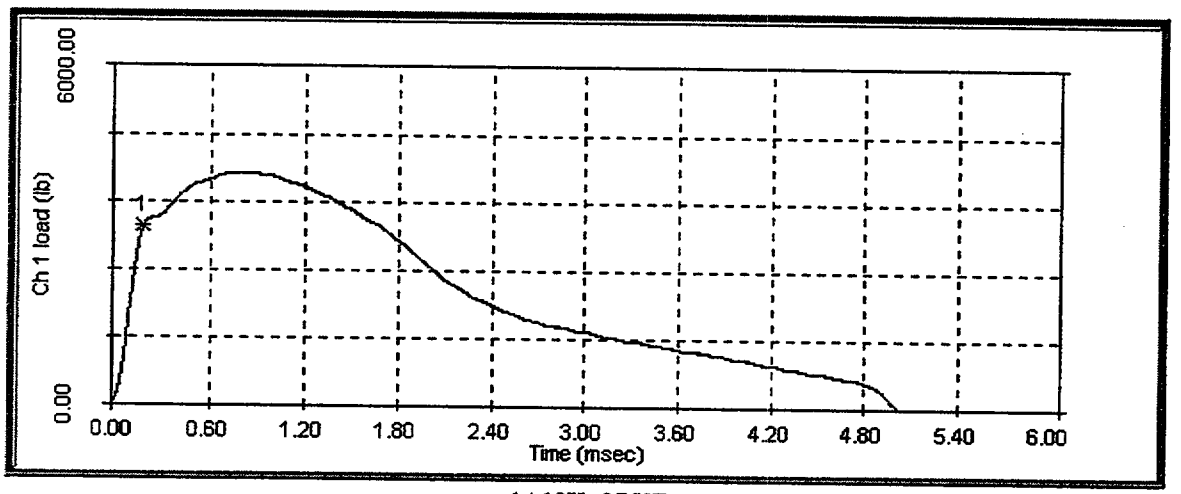
1A111, 100°F



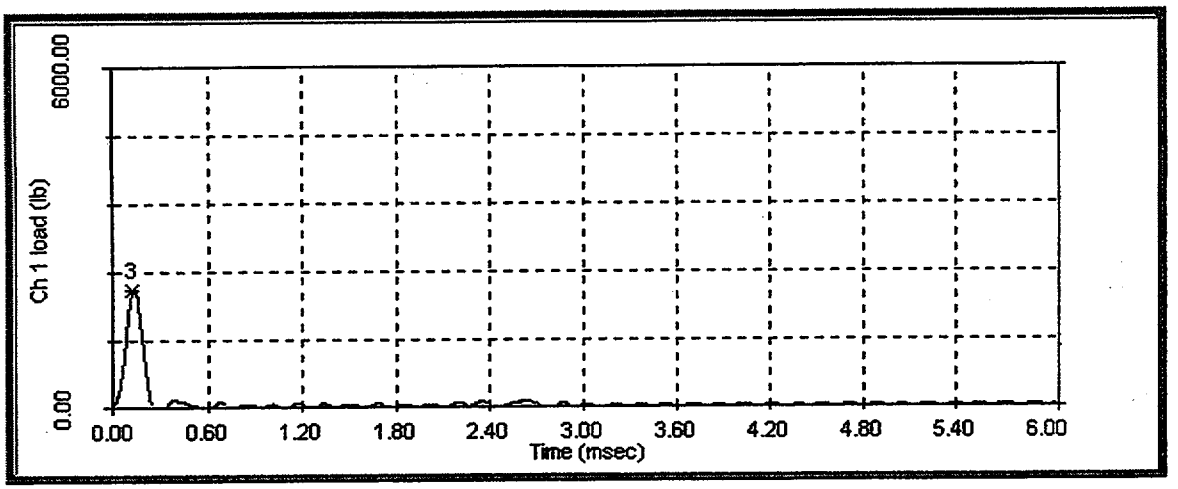
1A122, 150°F



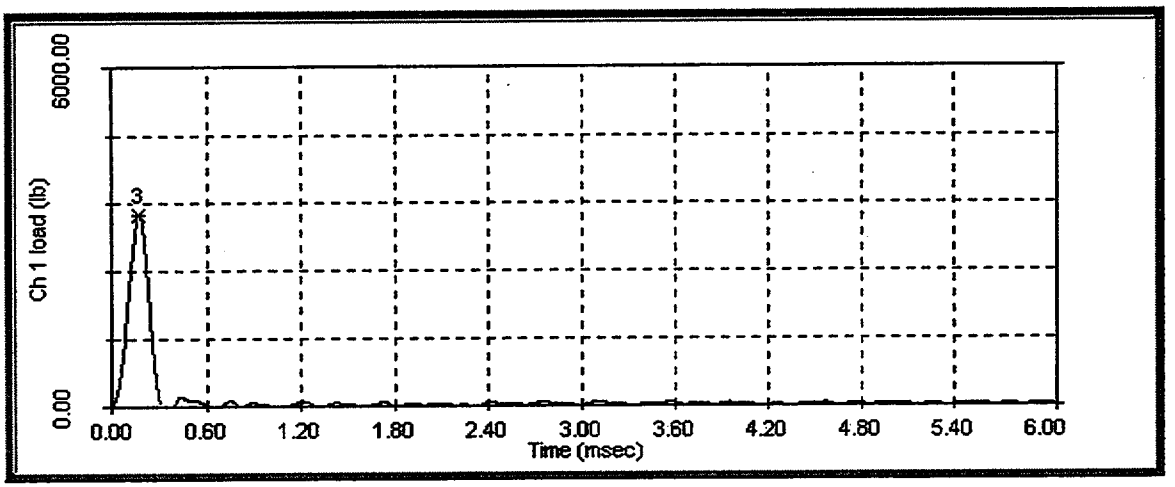
1A144, 225°F



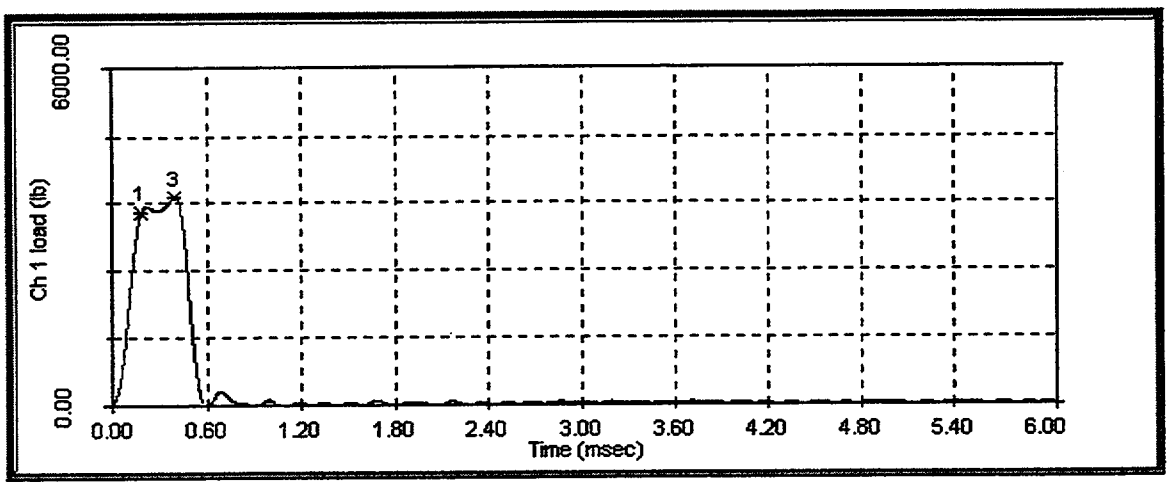
1A13K, 275°F



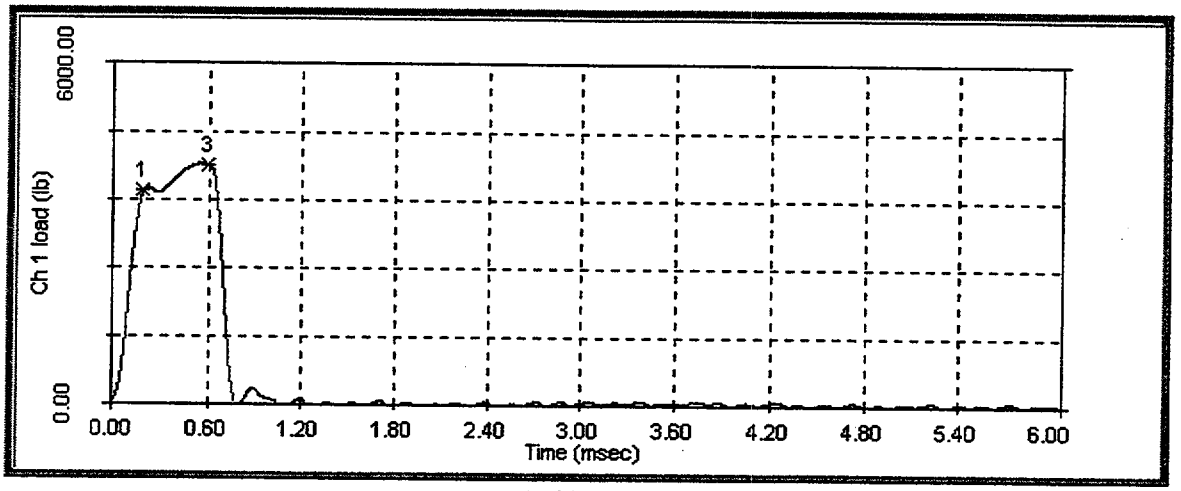
1A255, -75°F



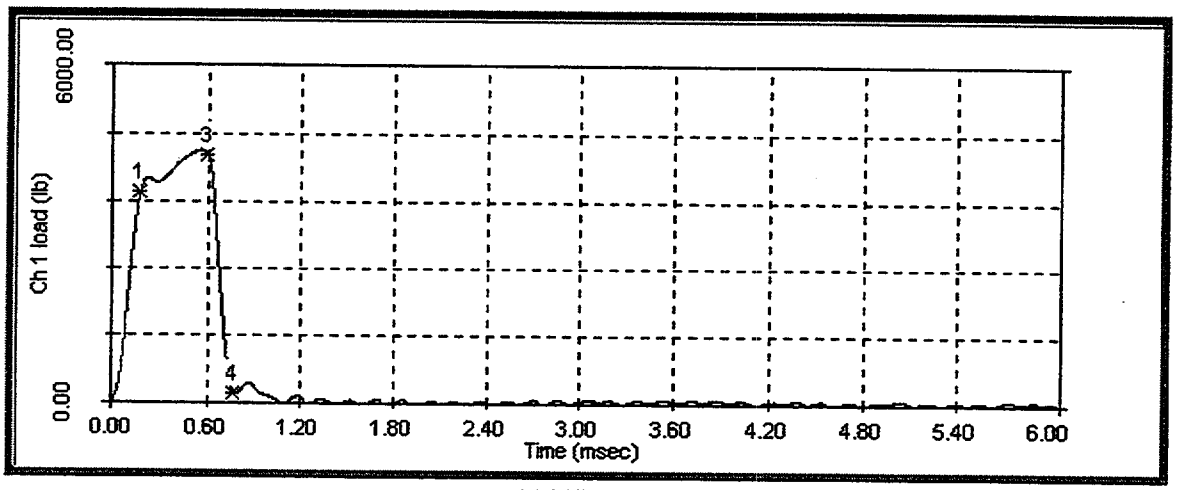
1A21E, -40°F



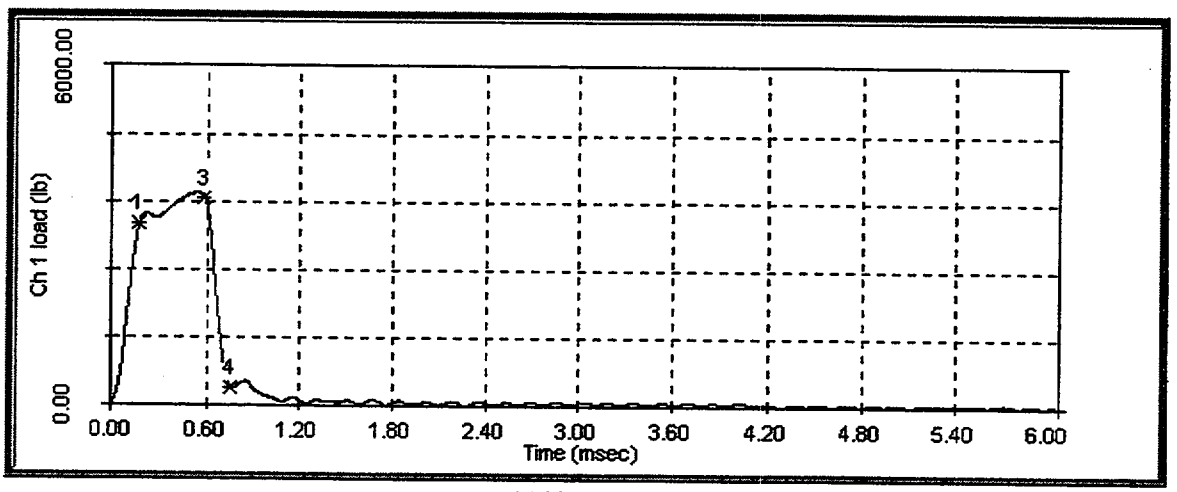
1A25P, 0°F



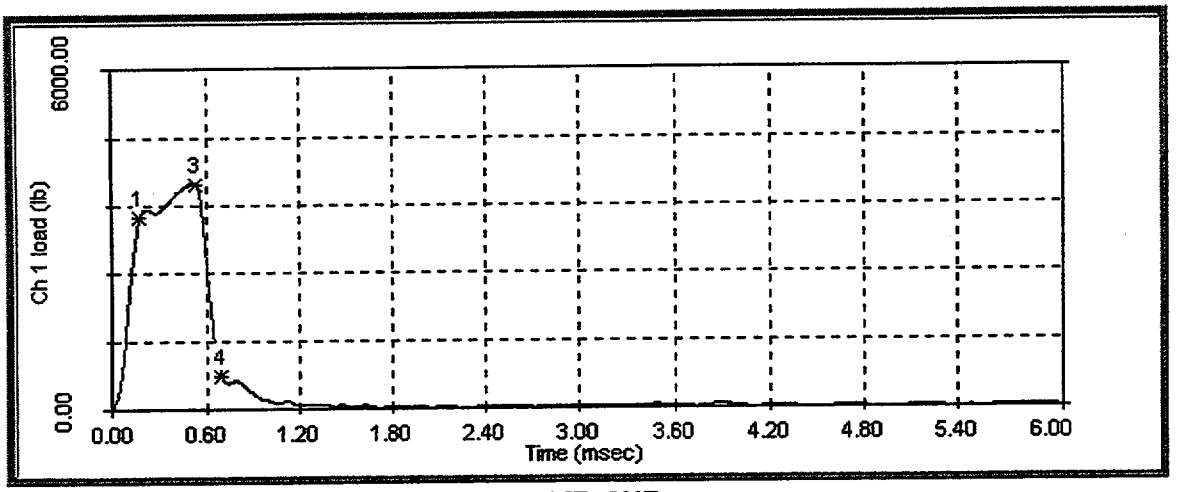
1A232, 5°F



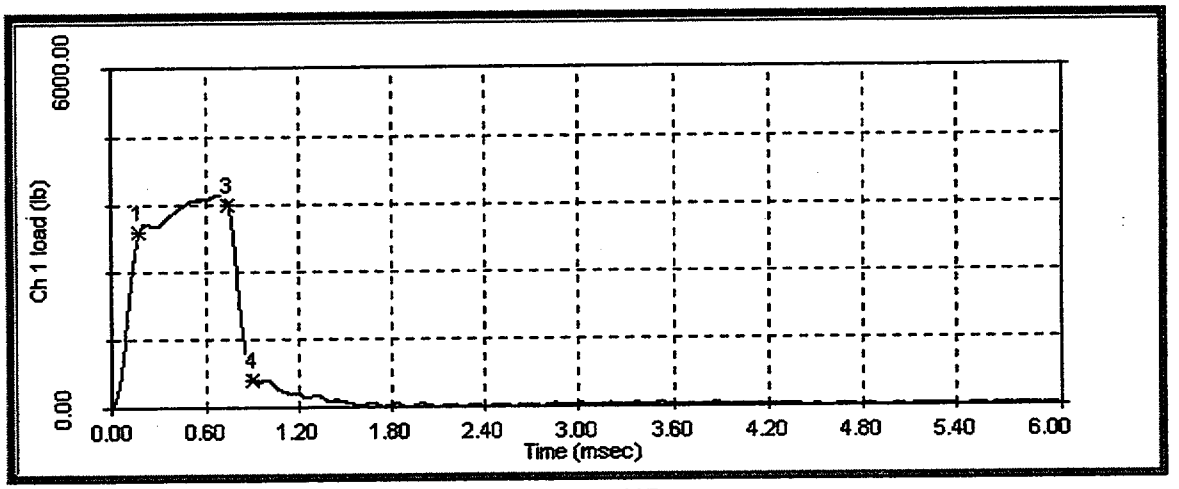
1A21J, 10°F



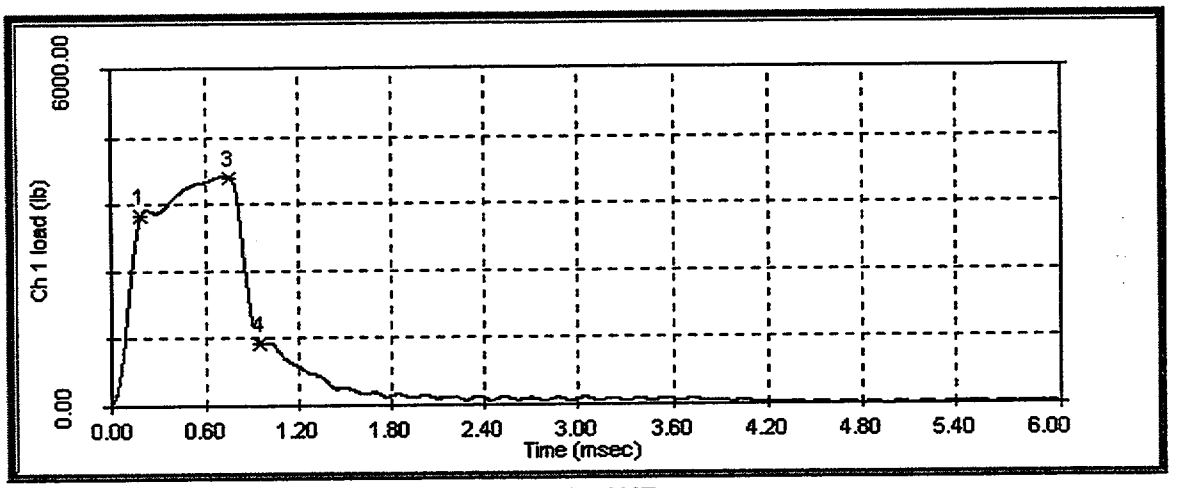
1A23A, 25°F



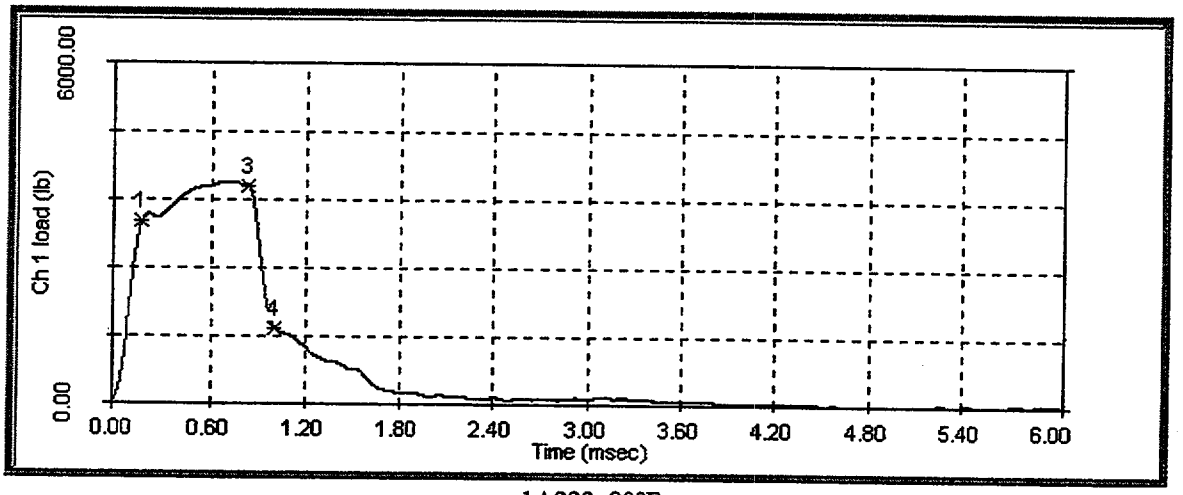
1A25E, 50°F



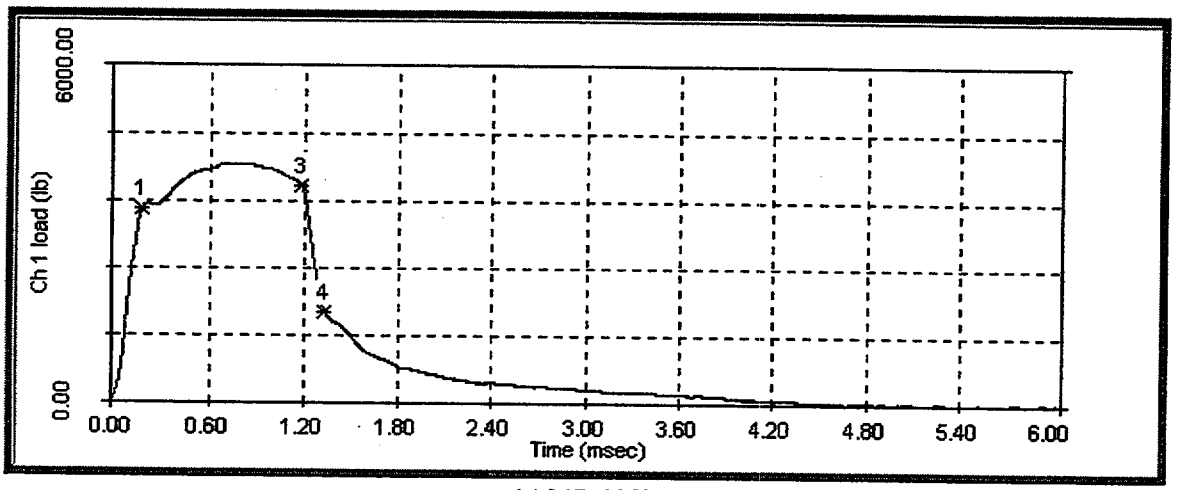
1A25U, 50°F



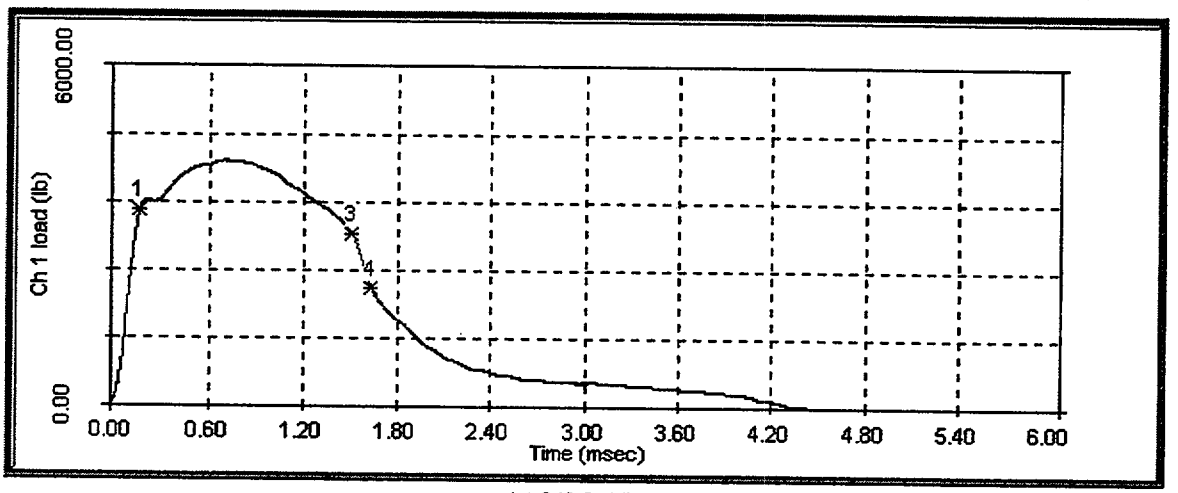
1A261, 70°F



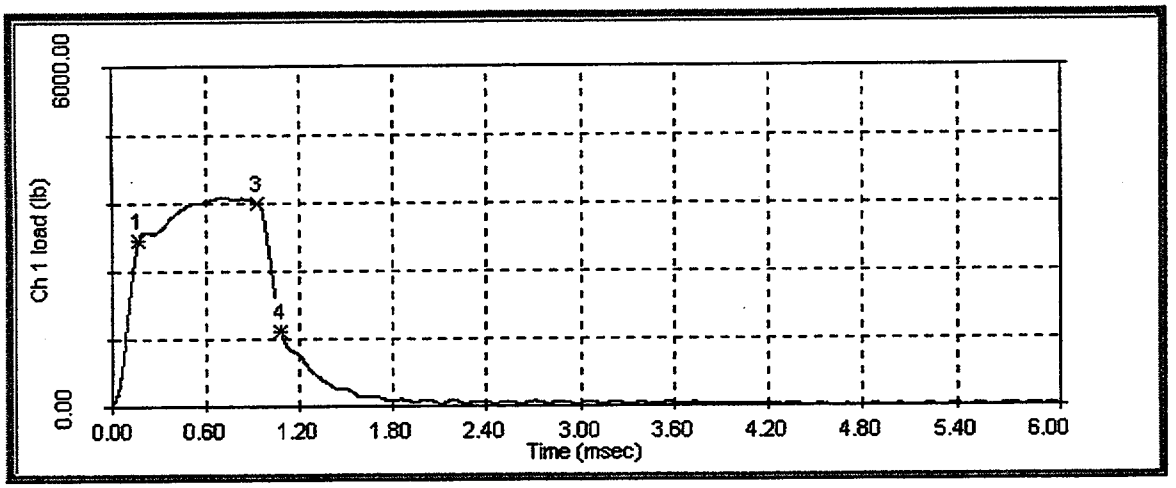
1A222, 80°F



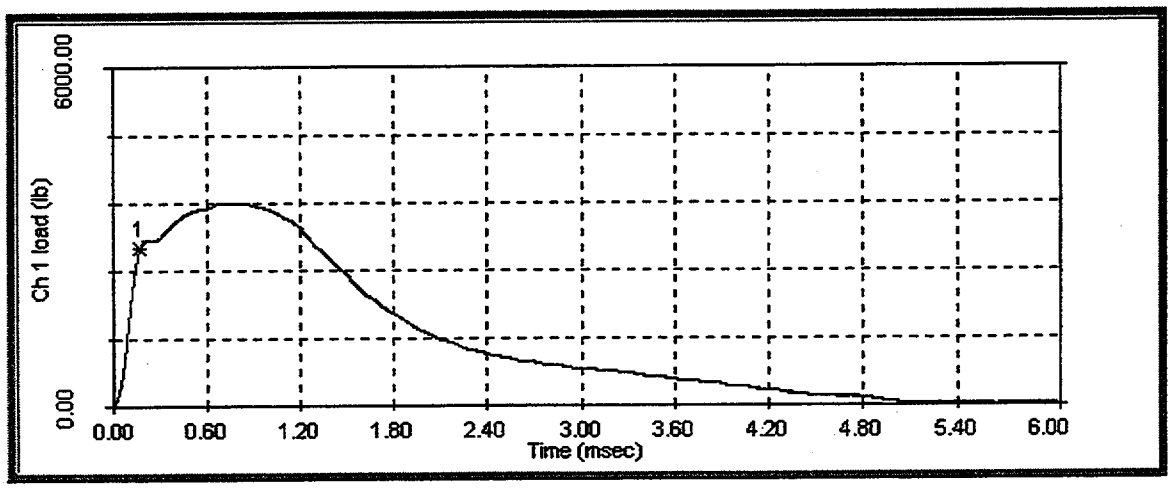
1A247, 125°F



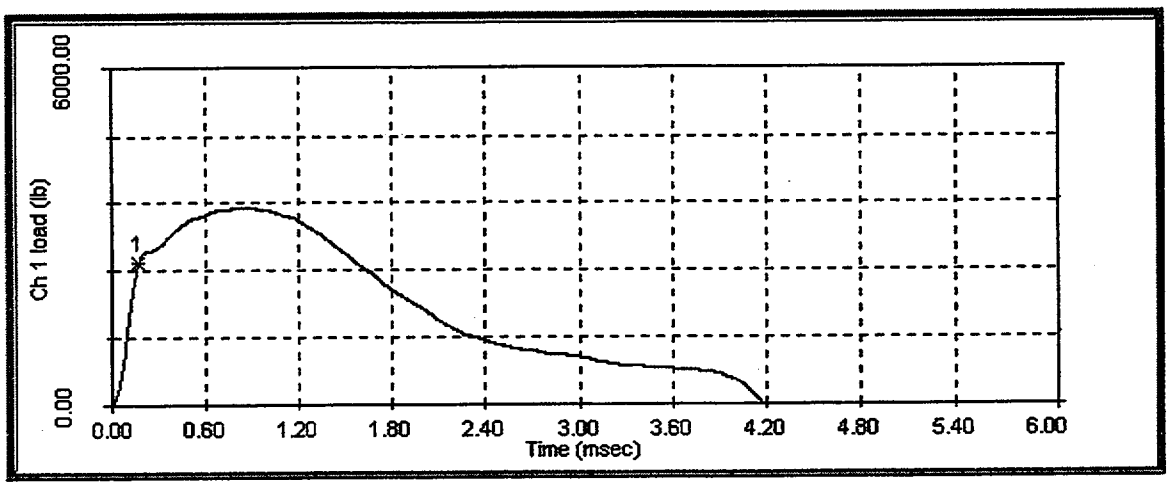
1A21M, 150°F



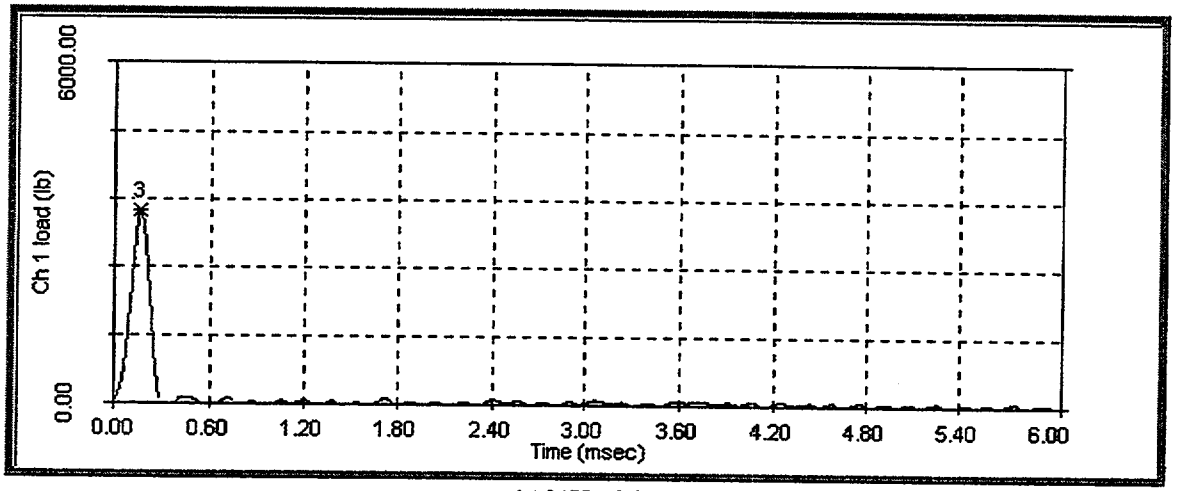
1A256, 150°F



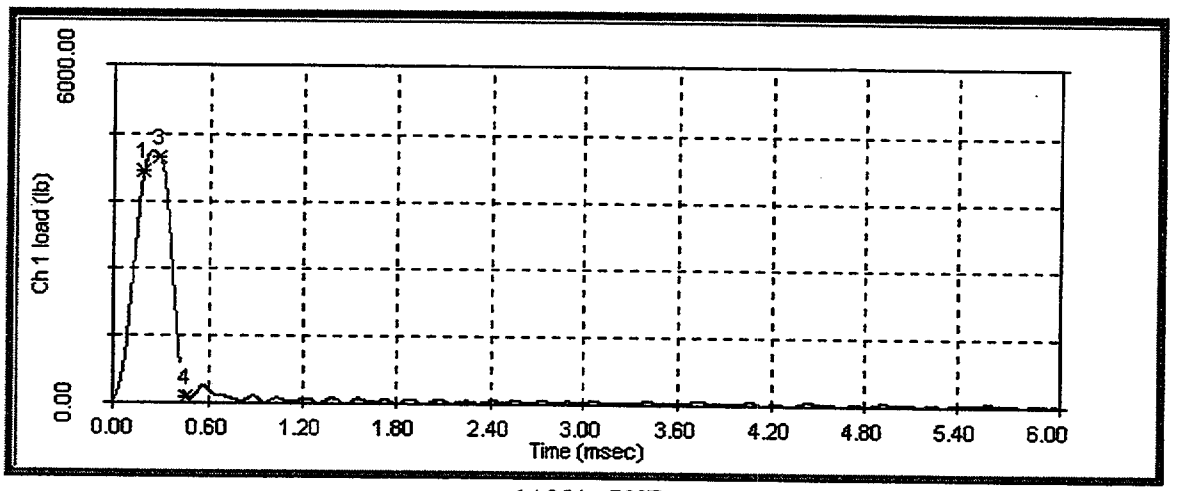
1A235, 200°F



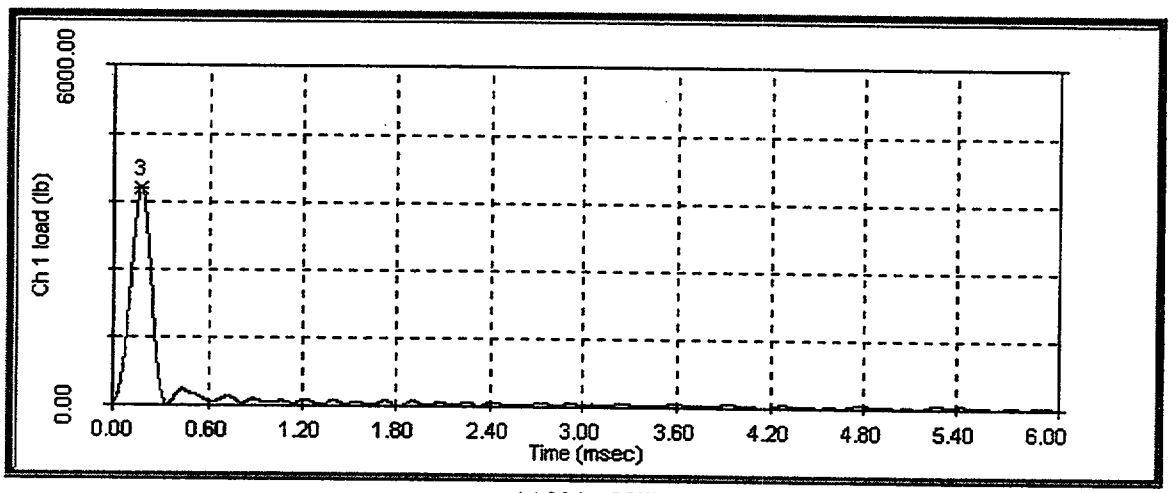
1A263, 250°F



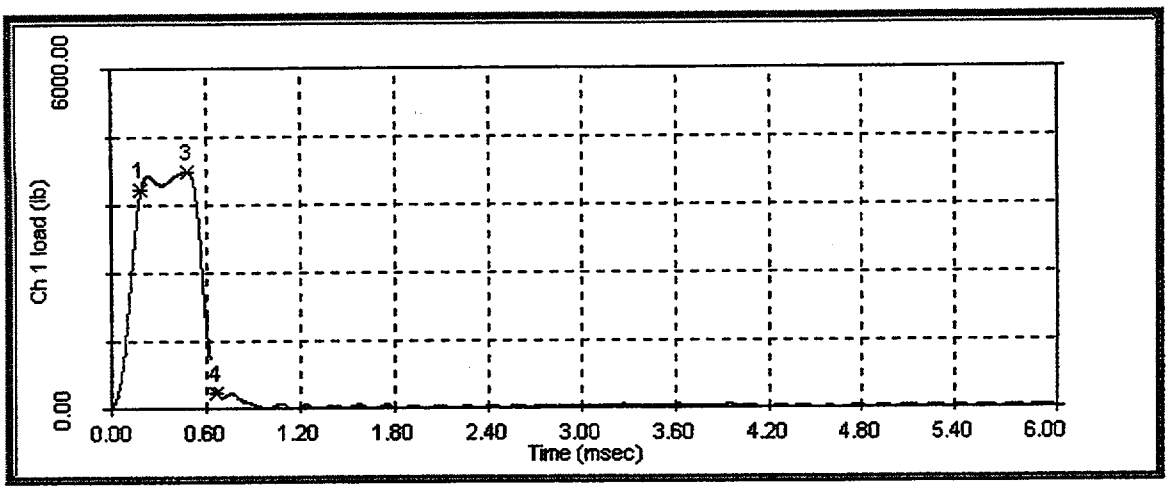
1A31Y, -96°F



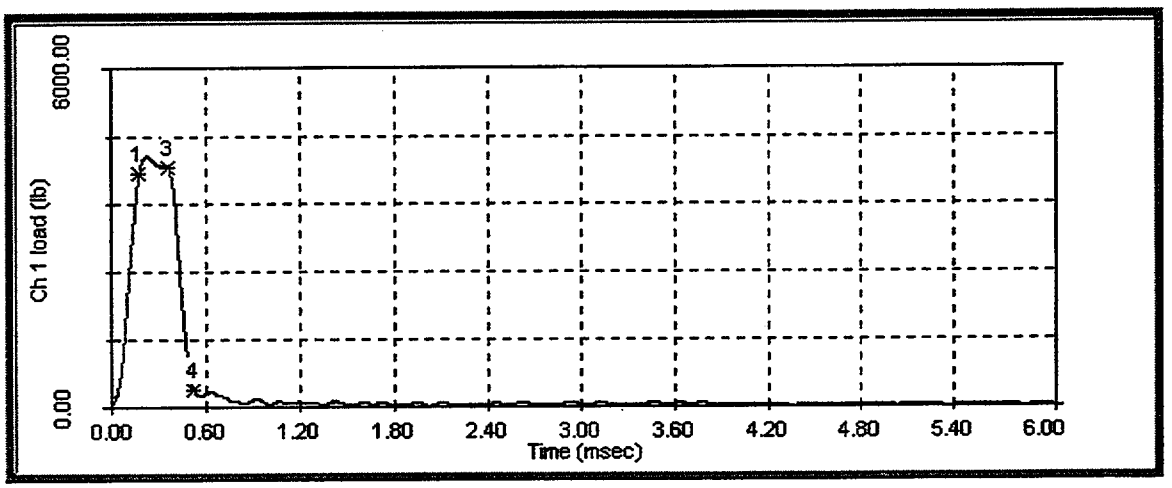
1A354, -70°F



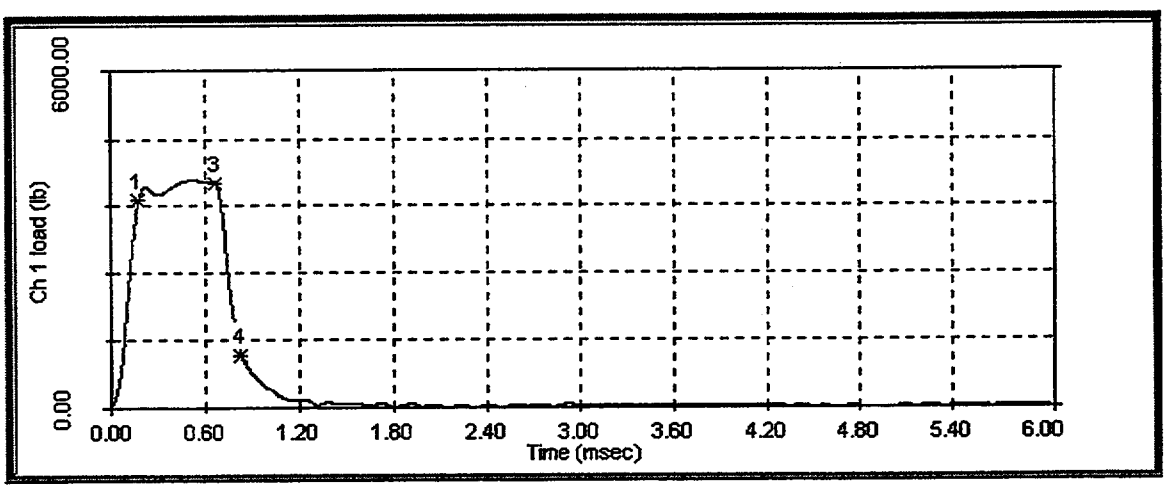
1A324, -55°F



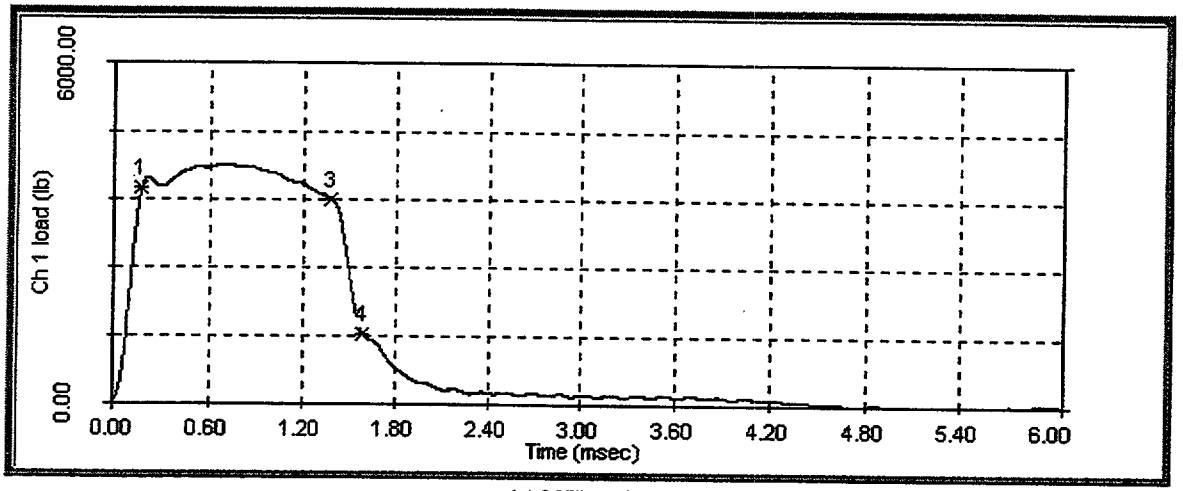
1A3A2, -50°F



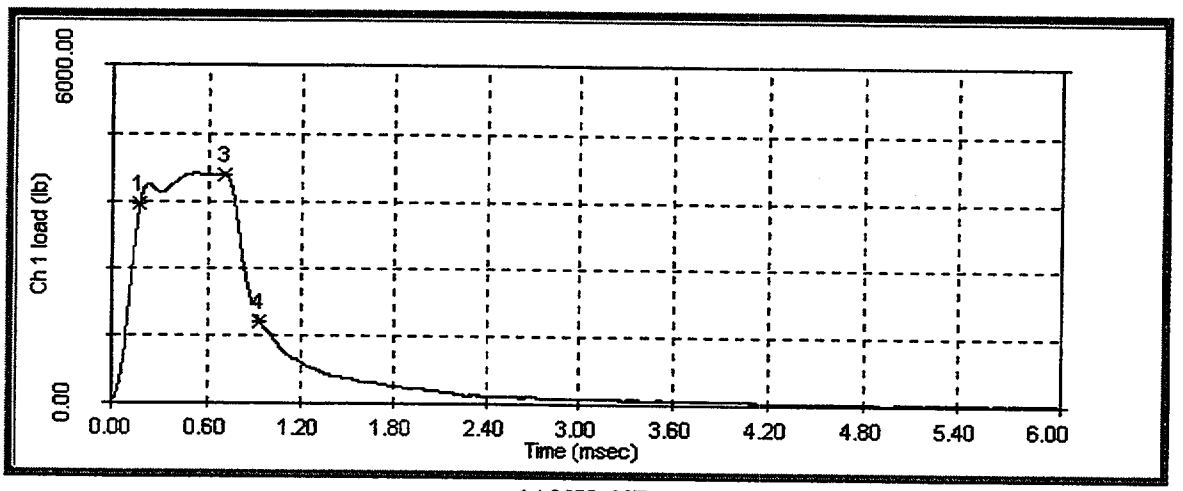
1A3B3, -45°F



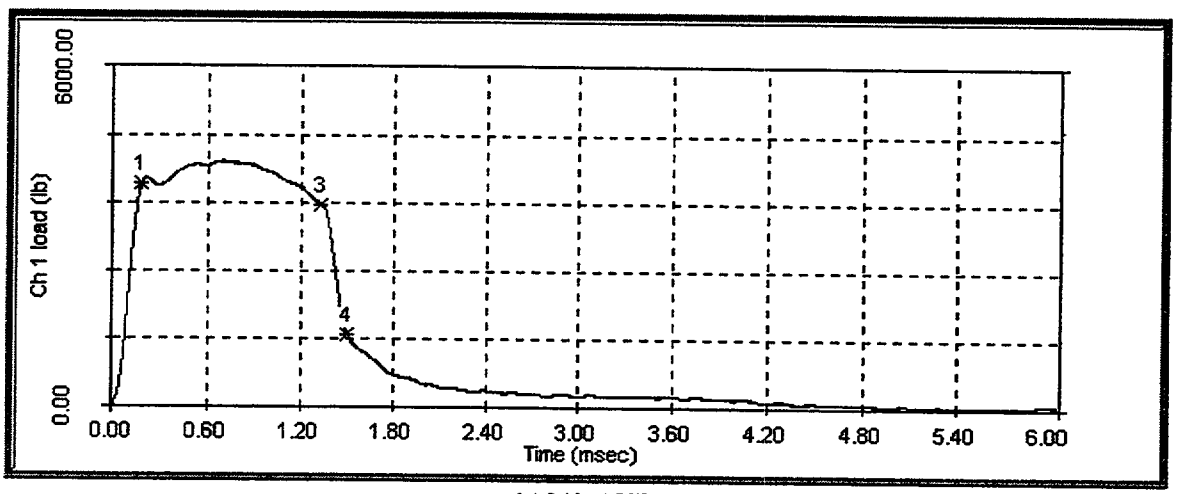
1A372, -25°F



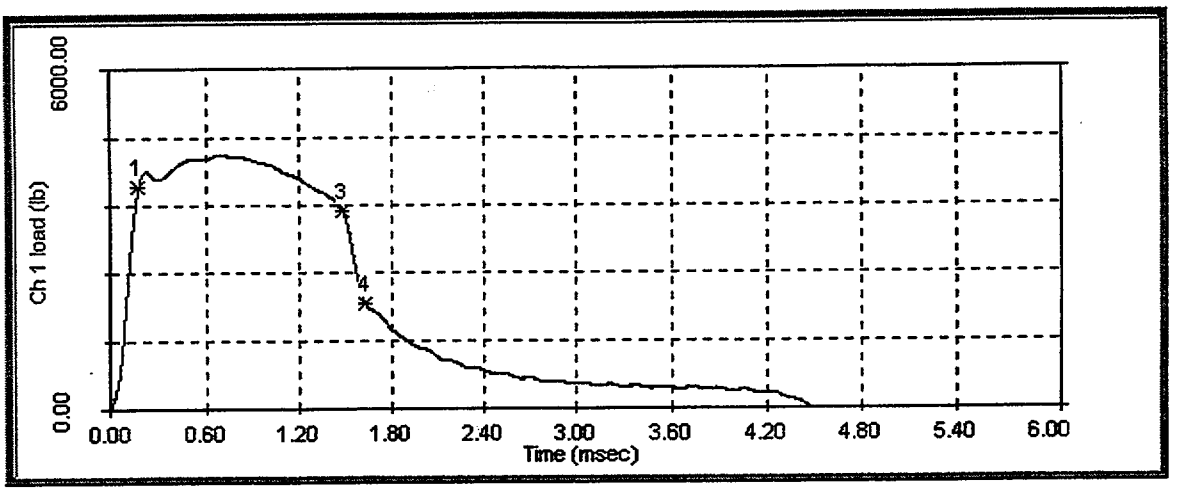
1A33K, -10°F



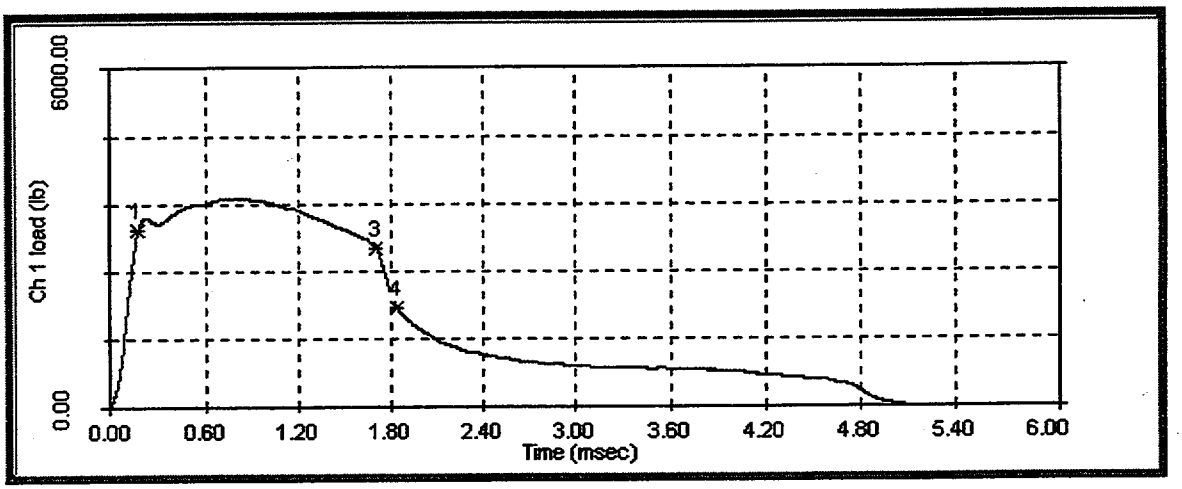
1A32U, 0°F



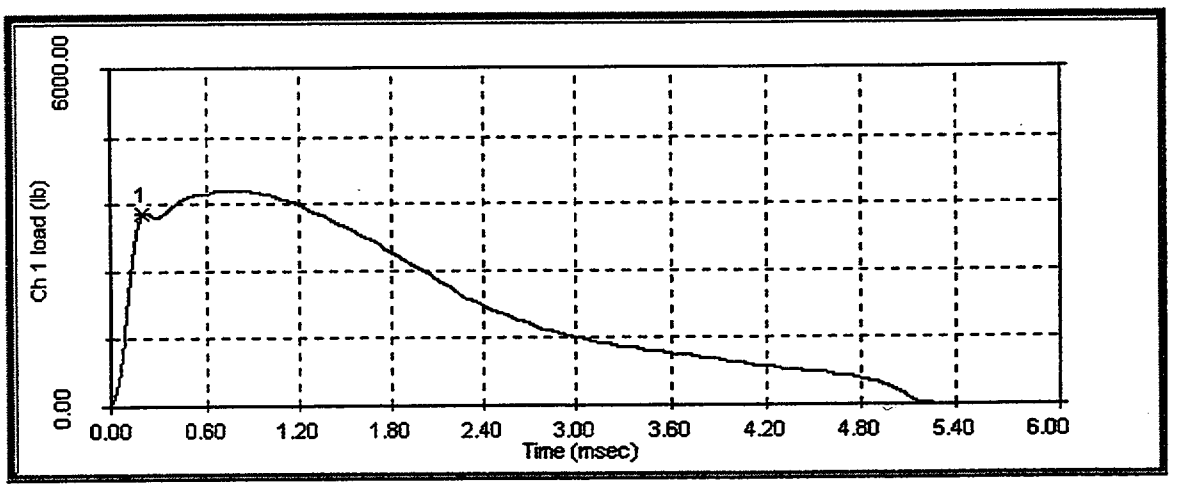
1A342, 15°F



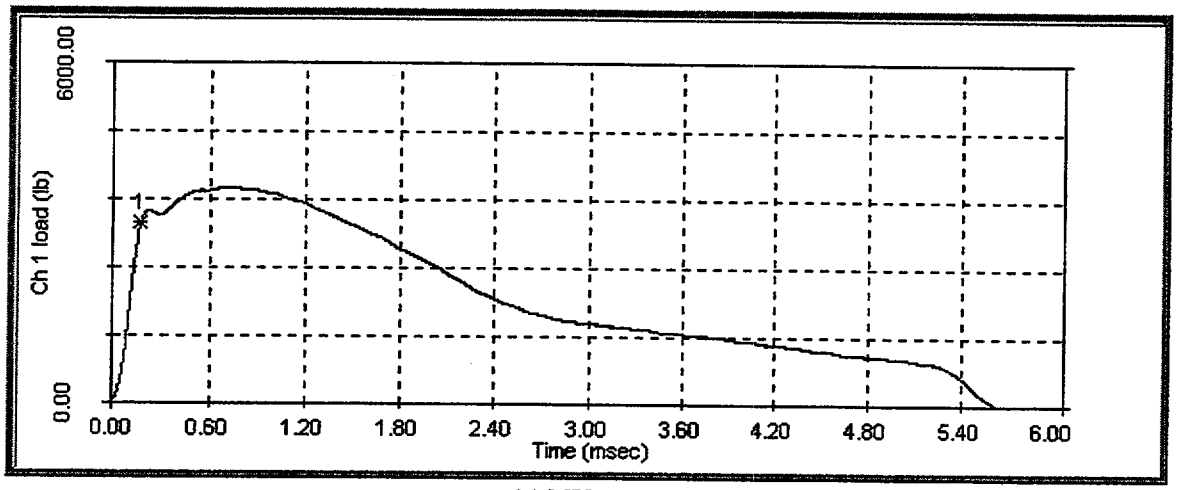
1A35E, 25°F



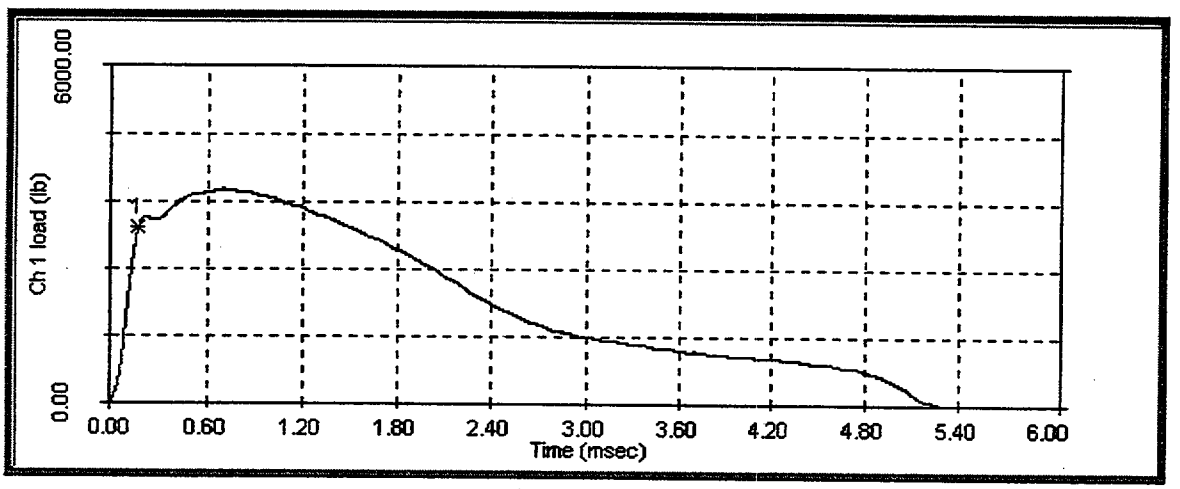
1A32M, 50°F



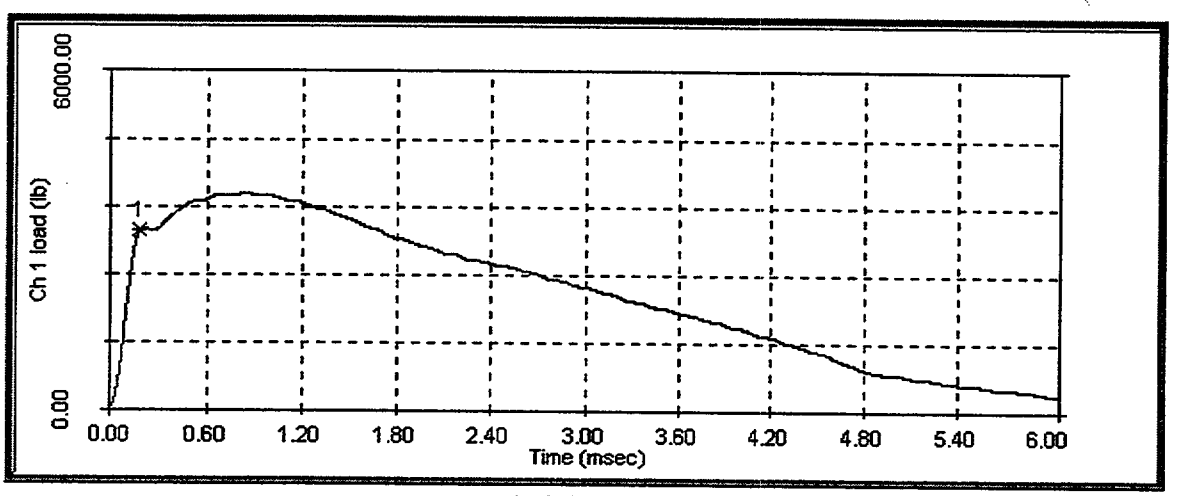
1A323, 100°F



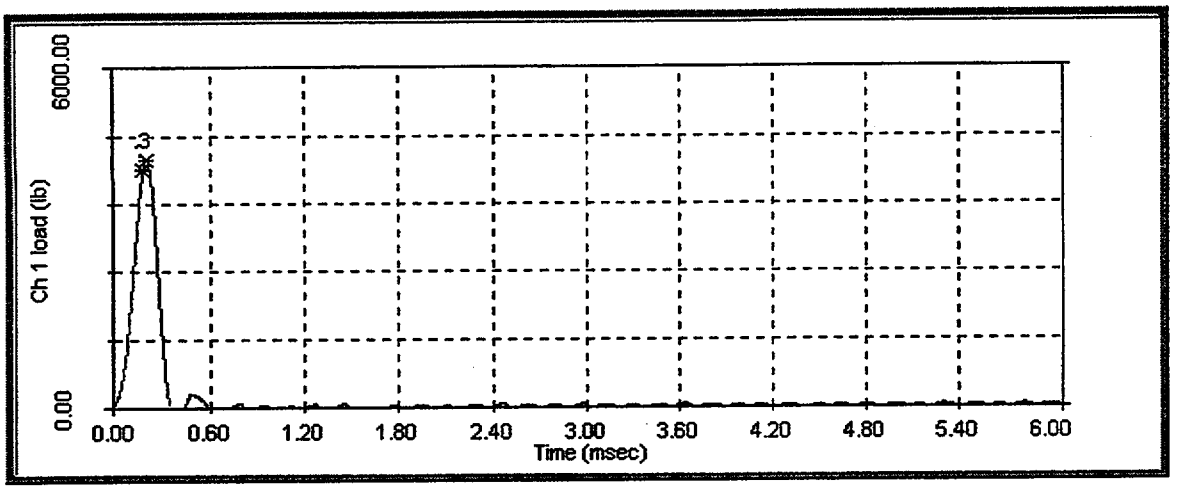
1A35U, 150°F



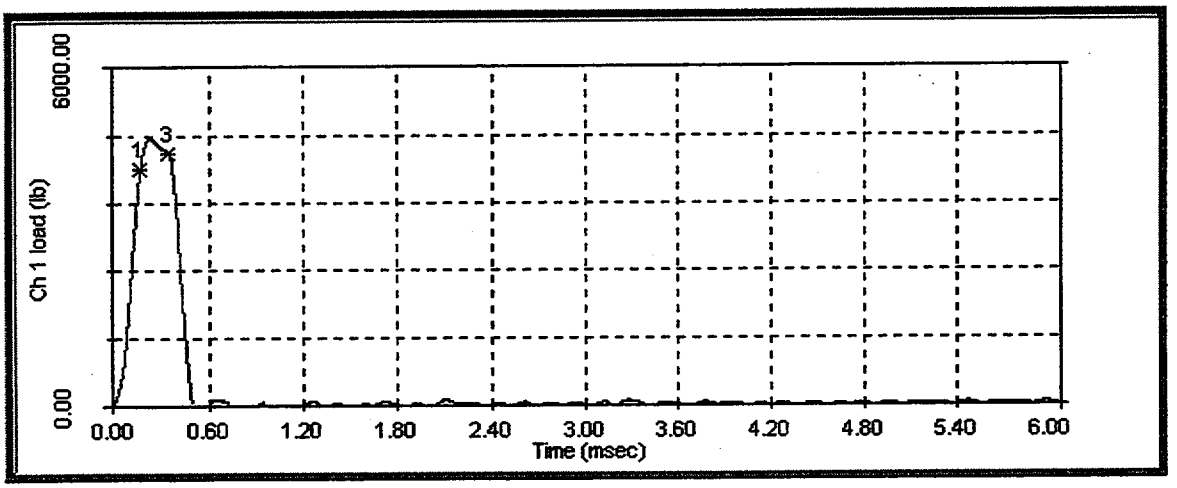
1A331, 200°F



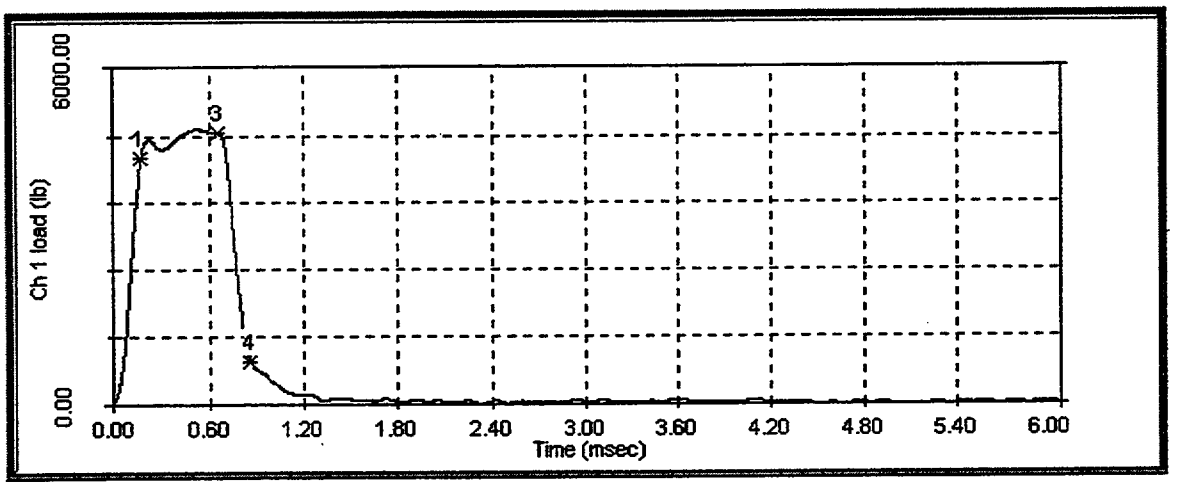
1A33B, 250°F



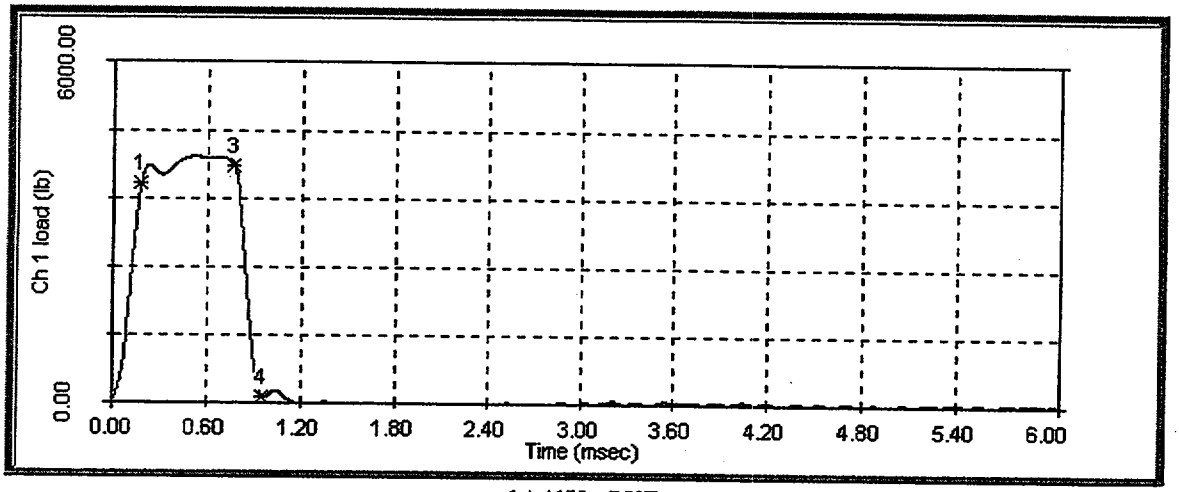
1A441, -175°F



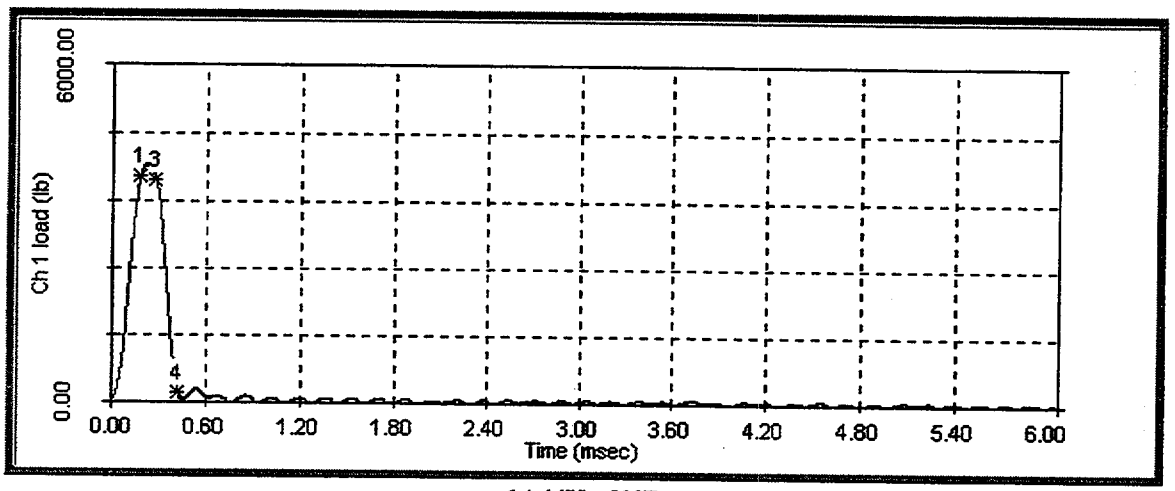
1A442, -120°F



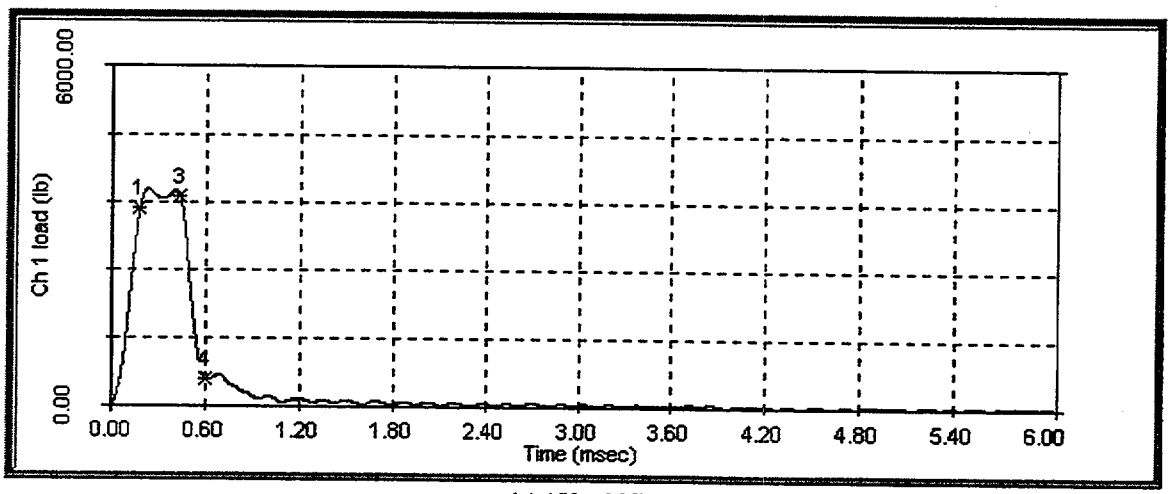
1A43D, -90°F



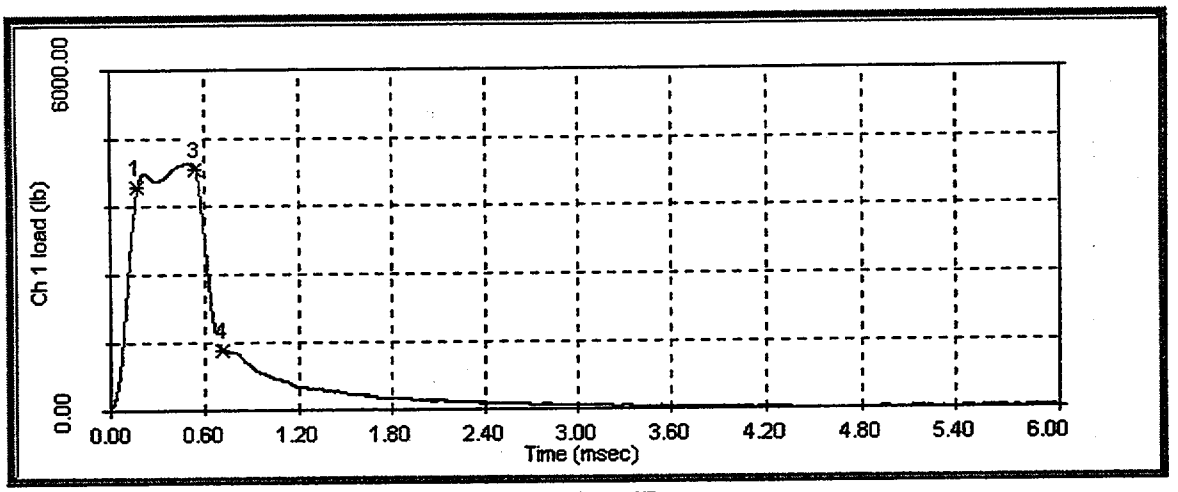
1A41U, -75°F



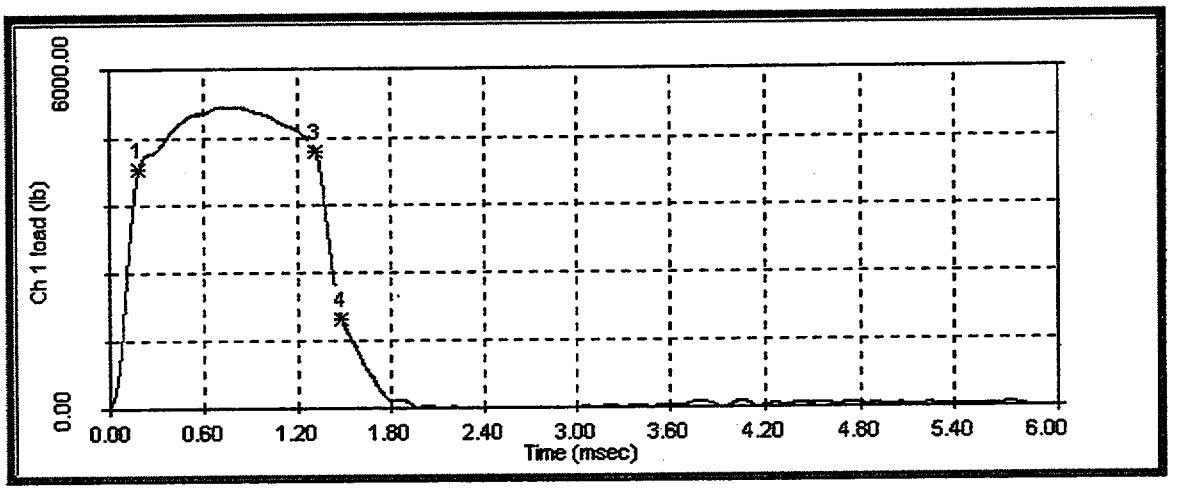
1A44Y, -50°F



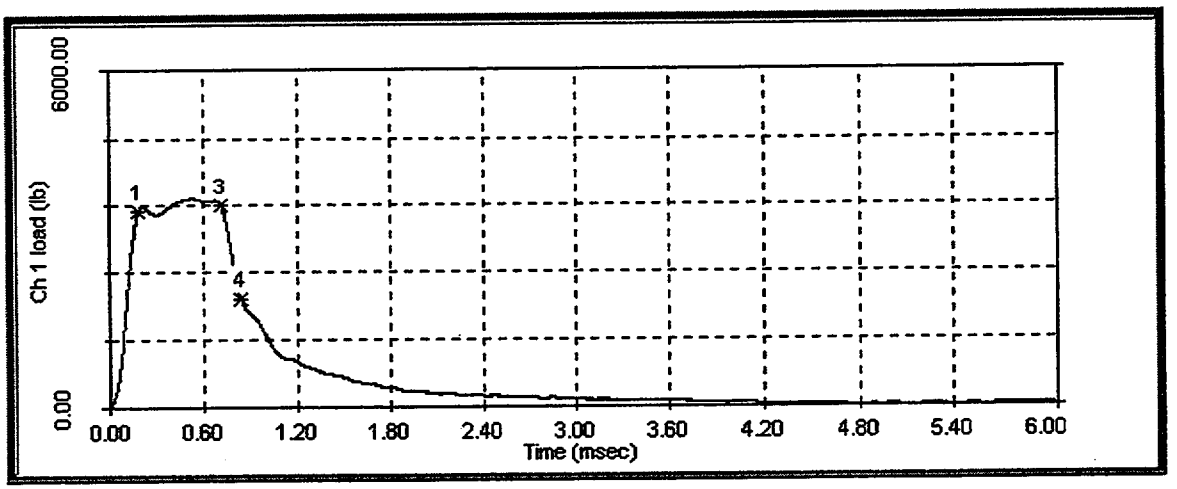
1A453, -30°F



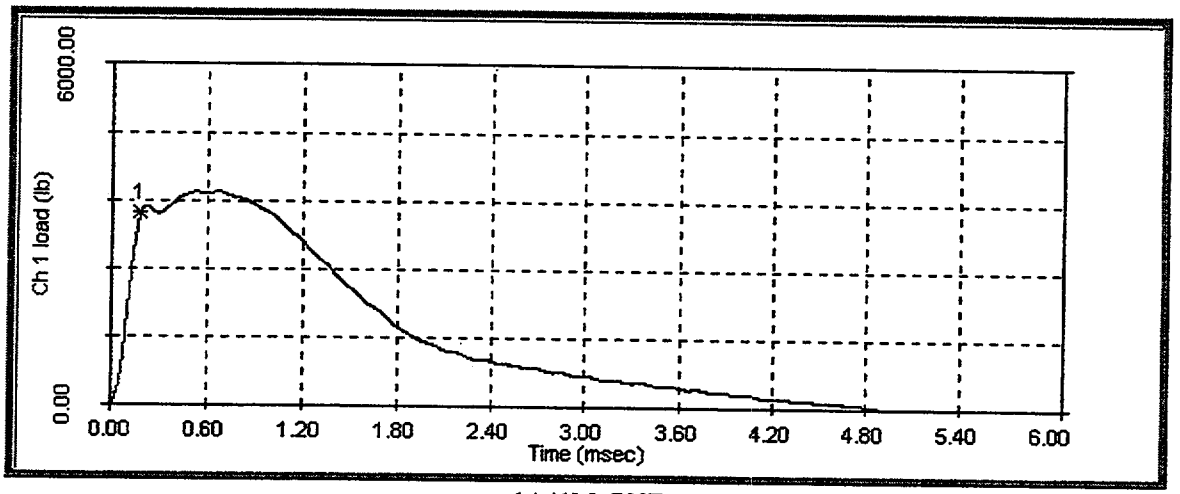
1A416, -25°F



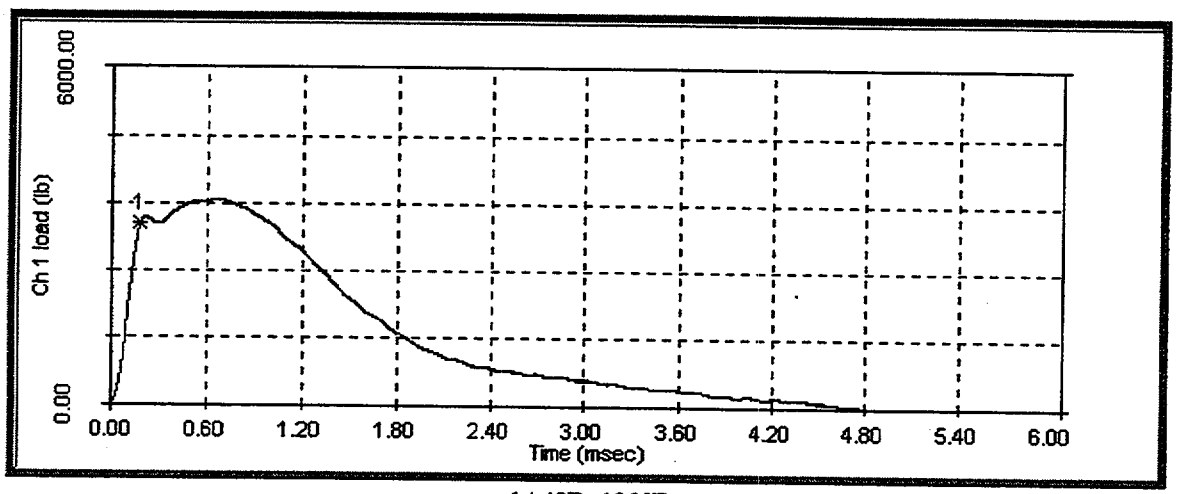
1A443, 0°F



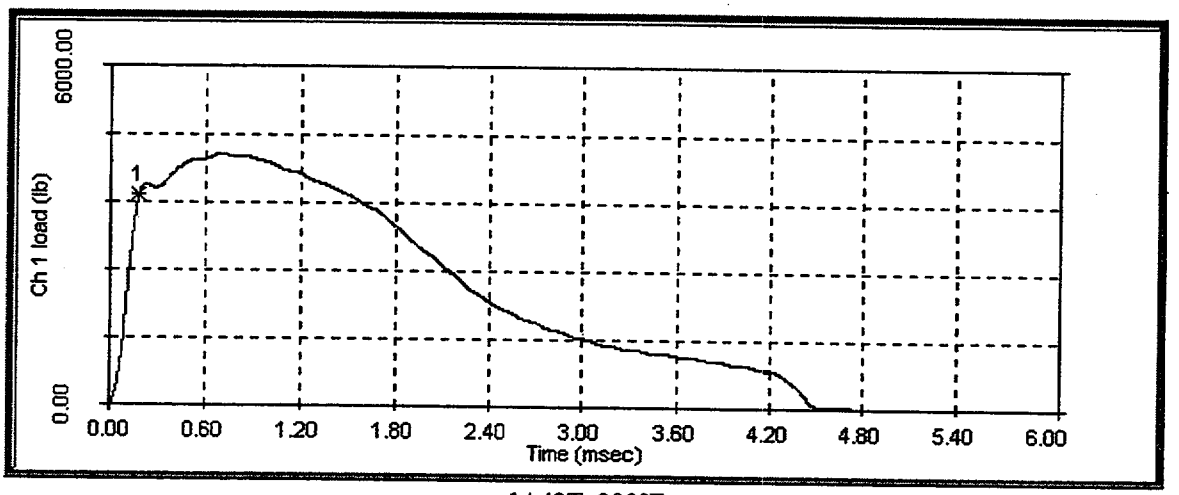
1A44D, 25°F



1A41M, 70°F



1A42B, 130°F



1A43T, 200°F

APPENDIX B

Charpy V-Notch Plots for Each Capsule Using 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. Intermediate 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 137°	Capsule 38°
Lower Shell Plate M-6701-2 (Longitudinal)	151 ft-lb	129 ft-lb	141 ft-lb
Lower Shell Plate M-6701-2 (Transverse)	98 ft-lb	87 ft-lb	115
Surveillance Weld (Heat # 90071)	164 ft-lb	162 ft-lb	158 ft-lb
HAZ Material	135 ft-lb	124 ft-lb	119 ft-lb
Standard Reference Material	129 ft-lb	105 ft-lb	105 ft-lb

UNIRRADIATED (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 08:40:18 on 08-03-2000

Page 1

Coefficients of Curve 1

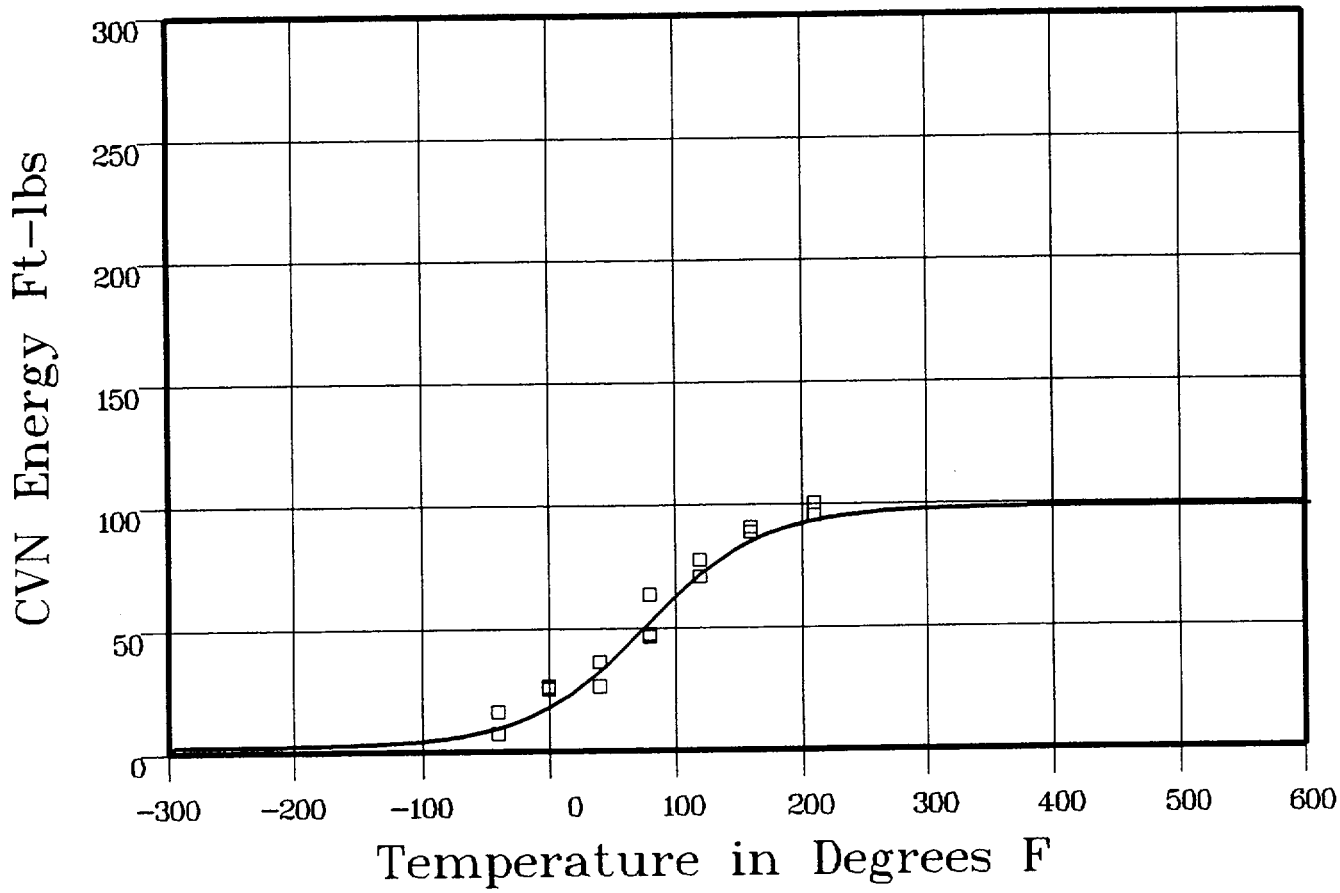
A = 50.09	B = 47.9	C = 92.95	T0 = 71.43

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

Upper Shelf Energy: 98 Fixed Temp. at 30 ft-lbs: 29.8 Temp. at 50 ft-lbs: 71.2 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: PLATE SA533B1 Ori.: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-40	16	10.18	5.81
-40	7	10.18	-3.18
0	26	19.15	6.84
0	25	19.15	5.84
40	26	34.49	-8.49
40	36	34.49	1.5
80	47	54.49	-7.49
80	63	54.49	8.5
80	46	54.49	-8.49

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

UNIRRADIATED (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
120	70	73.07	-3.07
120	77	73.07	3.92
160	88	85.59	2.4
160	90	85.59	4.4
210	100	93.37	6.62
210	95	93.37	1.62

SUM of RESIDUALS = 16.76

UNIRRADIATED (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 08:59:59 on 08-03-2000

Page 1

Coefficients of Curve 1

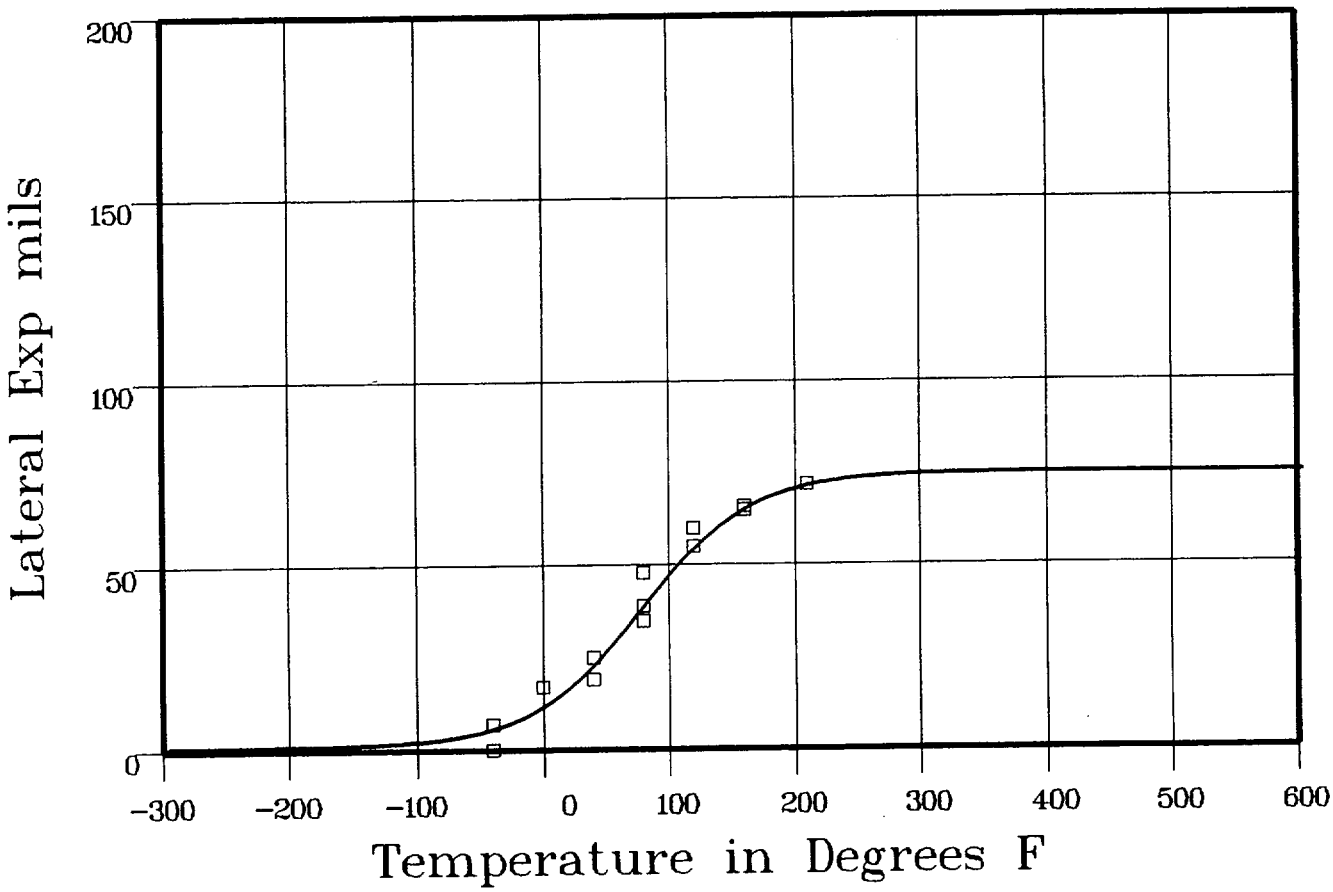
A = 38.08	B = 37.08	C = 87.68	T0 = 73.53

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

Upper Shelf L.E: 75.16 Temperature at L.E. 35: 66.2 Lower Shelf L.E: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: PLATE SA533B1 Ori.: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-40	7	6.17	.82
-40	0	6.17	-6.17
0	17	12.67	4.32
0	17	12.67	4.32
40	19	24.55	-5.55
40	25	24.55	.44
80	39	40.81	-1.81
80	48	40.81	7.18
80	35	40.81	-5.81

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

UNIRRADIATED (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
120	55	56.08	-1.08
120	60	56.08	3.91
160	66	66.11	-.11
160	65	66.11	-1.11
210	72	72.01	-.01
210	72	72.01	-.01
		SUM of RESIDUALS =	-.67

UNIRRADIATED (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:05:35 on 08-03-2000

Page 1

Coefficients of Curve 1

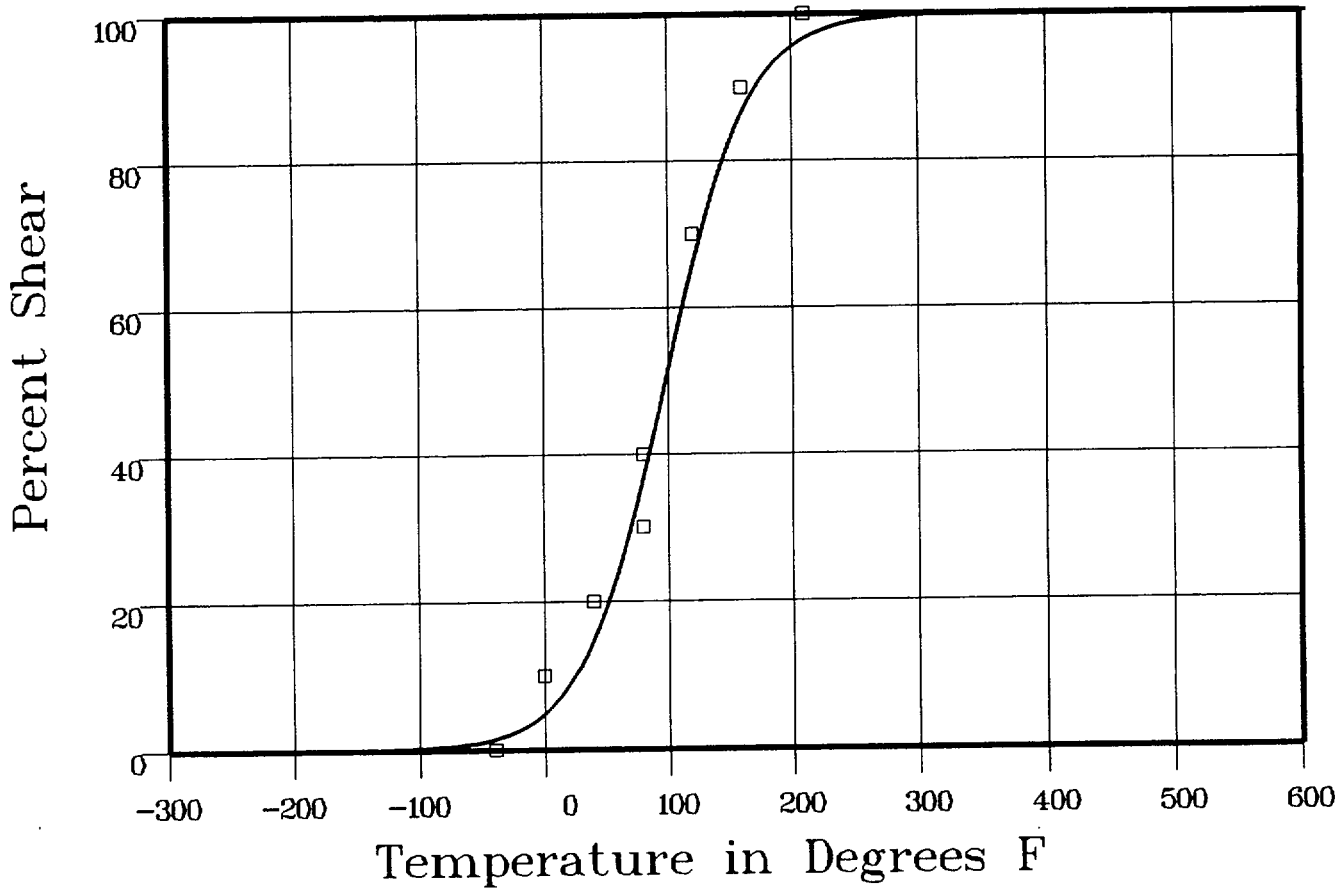
A = 50	B = 50	C = 66.31	T0 = 93.45

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

Temperature at 50% Shear: 93.4

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: PLATE SA533B1 Ori.: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-40	0	1.75	-1.75
-40	0	1.75	-1.75
0	10	5.63	4.36
0	10	5.63	4.36
40	20	16.62	3.37
40	20	16.62	3.37
80	30	39.99	-9.99
80	40	39.99	0
80	40	39.99	0

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

UNIRRADIATED (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
120	70	69	.99
120	70	69	.99
160	90	88.15	1.84
160	90	88.15	1.84
210	100	97.11	2.88
210	100	97.11	2.88

SUM of RESIDUALS = 13.45

CAPSULE 137 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 08:40:18 on 08-03-2000

Page 1

Coefficients of Curve 2

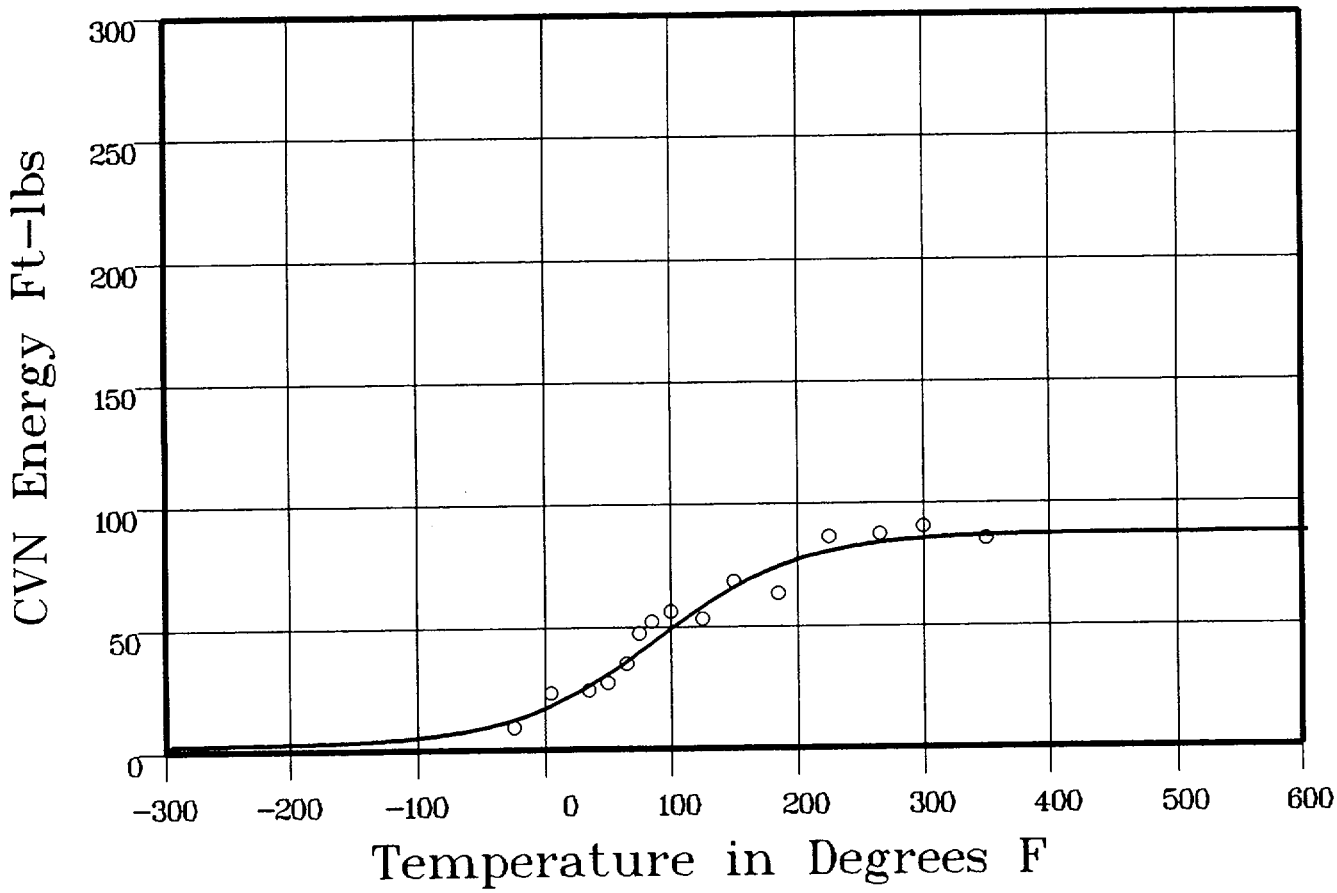
A = 44.59	B = 42.4	C = 112.77	T0 = 83.9

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

Upper Shelf Energy: 87 Fixed Temp. at 30 ft-lbs: 43.4 Temp. at 50 ft-lbs: 98.3 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: PLATE SA533B1 Ori.: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-25	9	12.93	-3.93
5	23	18.98	4.01
35	24	27.28	-3.28
50	27	32.22	-5.22
65	35	37.55	-2.55
75	47	41.25	5.74
85	52	45.01	6.98
100	56	50.61	5.38

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

CAPSULE 137 (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
125	53	59.4	-6.4
150	68	66.94	1.05
185	63	74.89	-11.89
225	86	80.58	5.41
265	87	83.71	3.28
300	90	85.2	4.79
350	85	86.25	-1.25

SUM of RESIDUALS = 2.13

CAPSULE 137 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 08:59:59 on 08-03-2000

Page 1

Coefficients of Curve 2

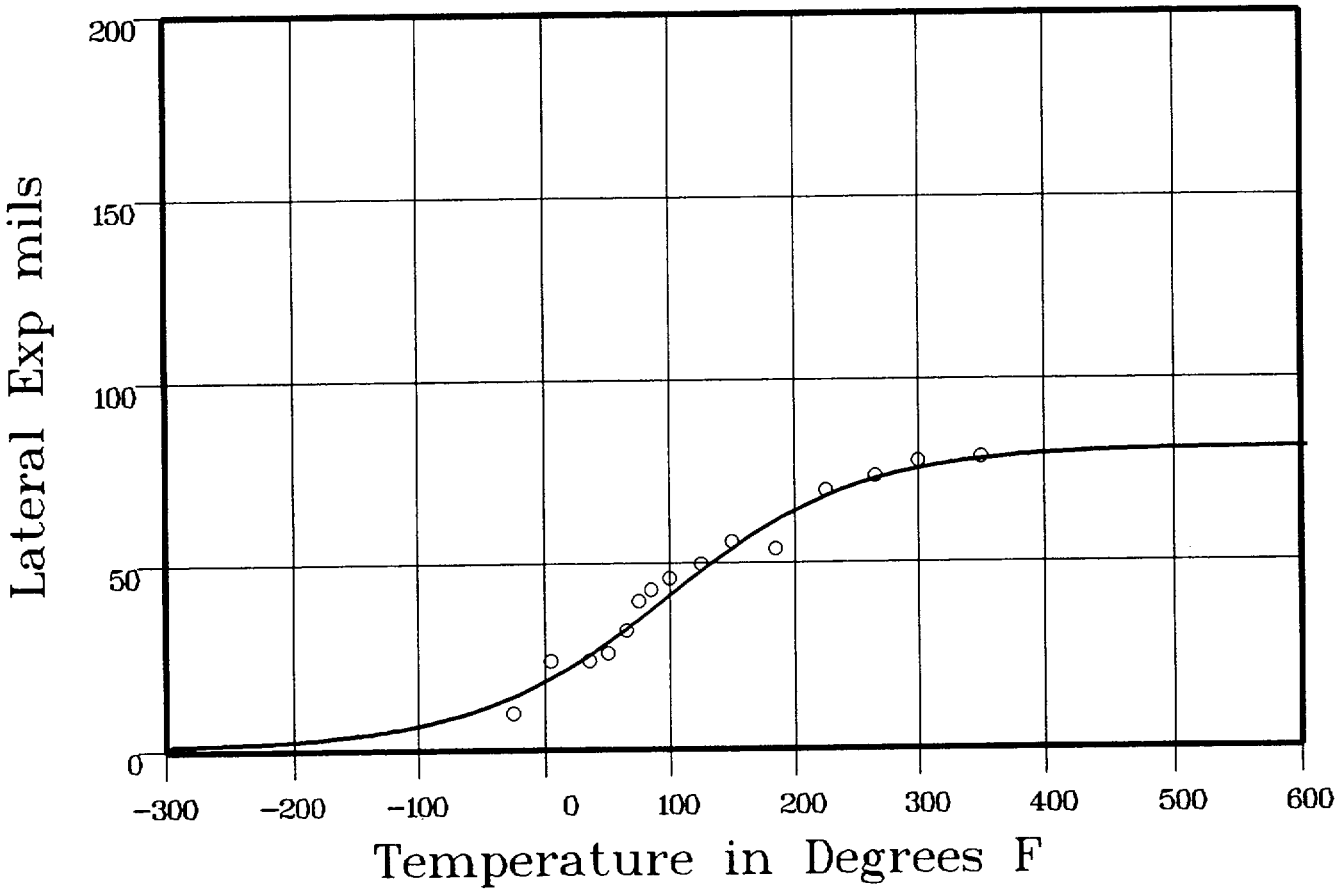
A = 41.13	B = 40.13	C = 153.26	T0 = 93.75

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

Upper Shelf L.E.: 81.27 Temperature at L.E. 35: 70.1 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: PLATE SA533B1 Ori.: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-25	10	15.05	-5.05
5	24	20.18	3.81
35	24	26.46	-2.46
50	26	29.98	-3.98
65	32	33.69	-1.69
75	40	36.25	3.74
85	43	38.84	4.15
100	46	42.77	3.22

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

CAPSULE 137 (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
125	50	49.2	.79
150	56	55.23	.76
185	54	62.55	-8.55
225	70	69	.99
265	74	73.51	.48
300	78	76.17	1.82
350	79	78.53	.46
			SUM of RESIDUALS = -1.48

CAPSULE 137 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:05:35 on 08-03-2000

Page 1

Coefficients of Curve 2

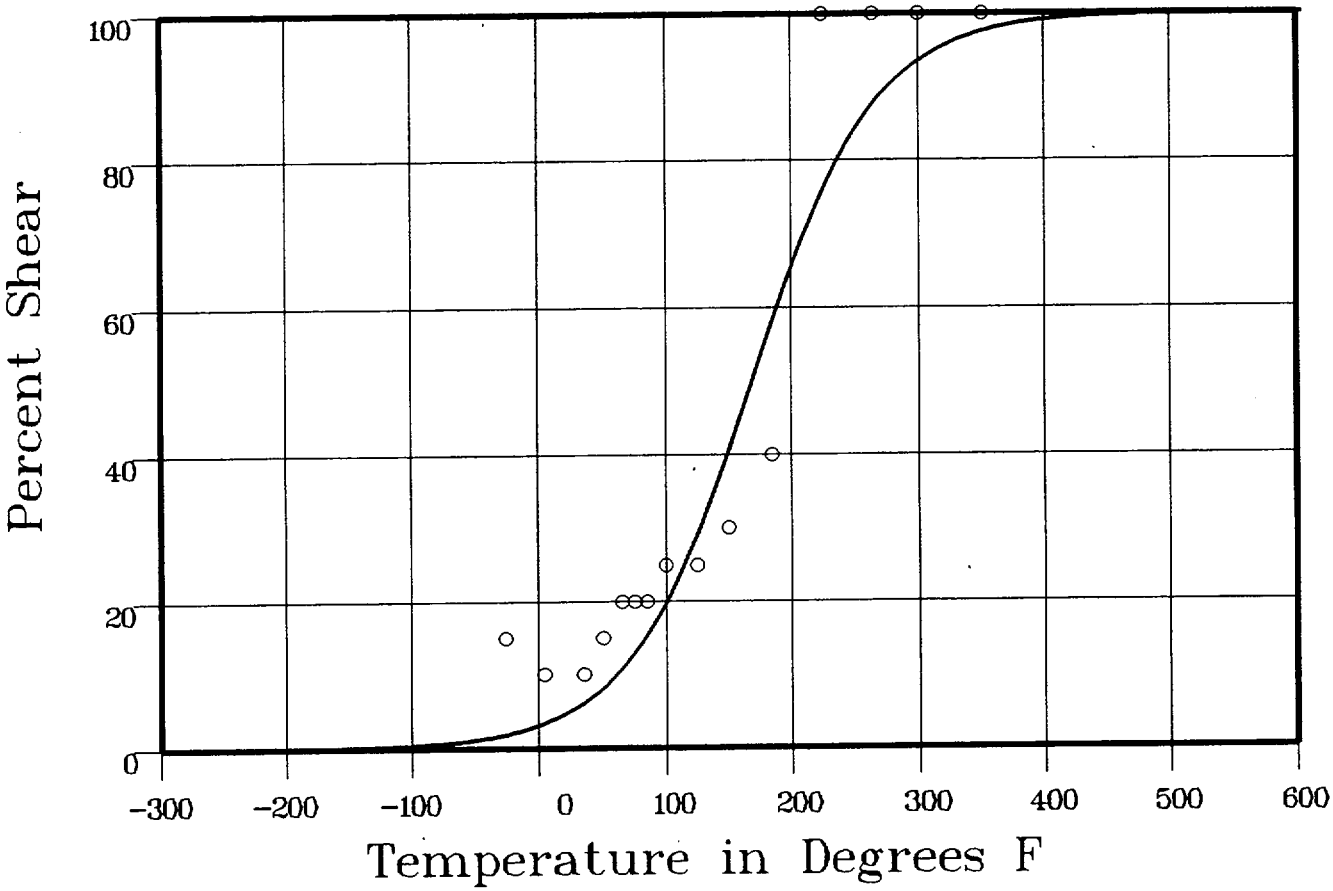
A = 50	B = 50	C = 98.32	T0 = 164.35

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

Temperature at 50% Shear: 164.3

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap: 137 Material: PLATE SA533B1 Ori: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-25	15	2.08	12.91
5	10	3.76	6.23
35	10	6.71	3.28
50	15	8.89	6.1
65	20	11.7	8.29
75	20	13.97	6.02
85	20	16.6	3.39
100	25	21.26	3.73

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

CAPSULE 137 (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
125	25	30.99	-5.99
150	30	42.75	-12.75
185	40	60.34	-20.34
225	100	77.44	22.55
265	100	88.56	11.43
300	100	94.04	5.95
350	100	97.76	2.23
		SUM of RESIDUALS =	53.09

CAPSULE 38 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 08:40:18 on 08-03-2000

Page 1

Coefficients of Curve 3

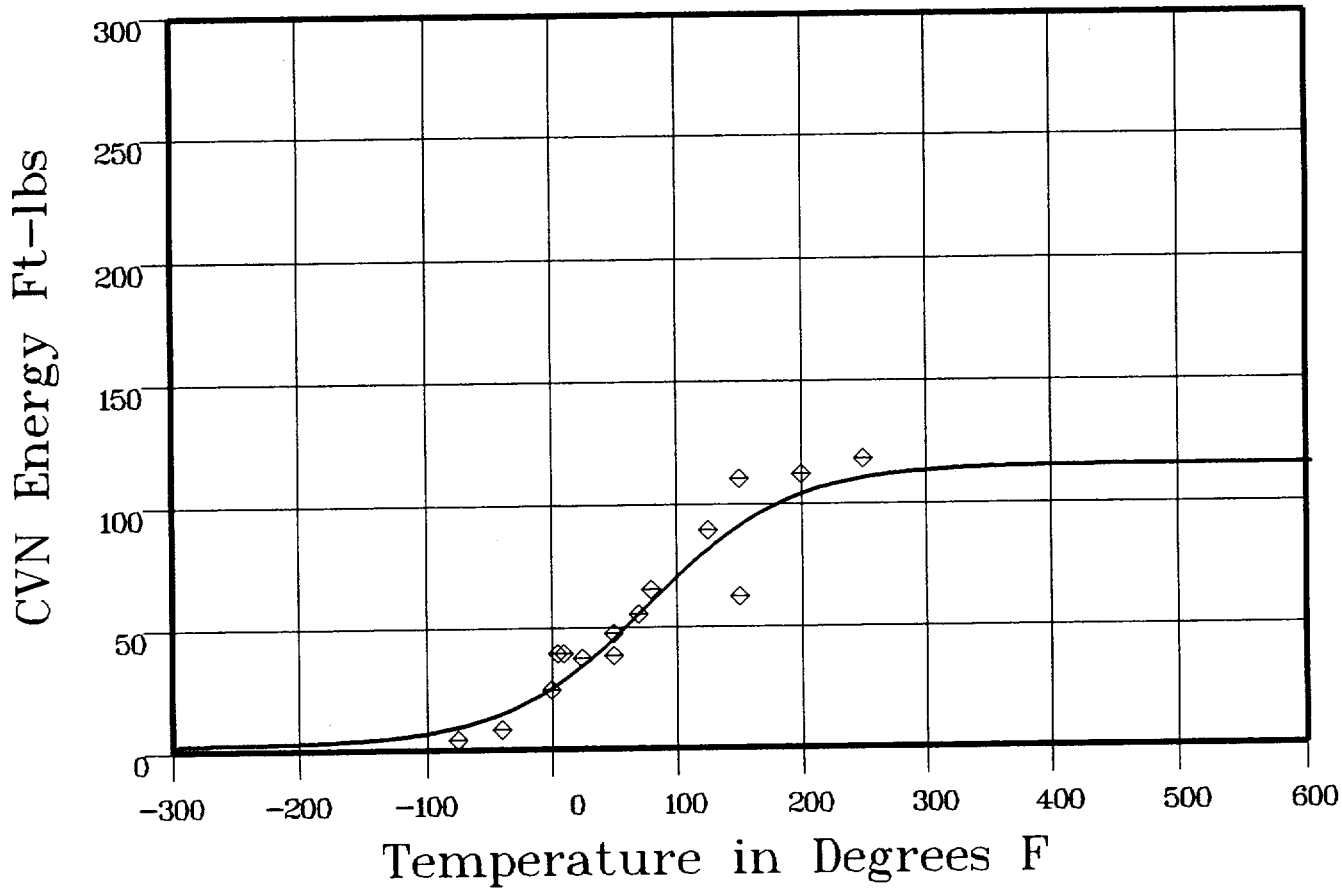
A = 58.59	B = 56.4	C = 111.35	T0 = 72.37

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

Upper Shelf Energy: 115 Fixed Temp. at 30 ft-lbs: 10.1 Temp. at 50 ft-lbs: 55.2 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: PLATE SA533B1 Ori.: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-75	4	9.66	-5.66
-40	8	15.43	-7.43
0	24	26.36	-2.36
5	39	28.1	10.89
10	39	29.94	9.05
25	37	35.95	1.04
50	38	47.41	-9.41
50	47	47.41	-.41

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

CAPSULE 38 (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
70	55	57.39	-2.39
80	65	62.45	2.54
125	89	83.43	5.56
150	62	92.58	-30.58
150	110	92.58	17.41
200	112	104.64	7.35
250	118	110.54	7.45

SUM of RESIDUALS = 3.05

CAPSULE 38 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 08:59:59 on 08-03-2000

Page 1

Coefficients of Curve 3

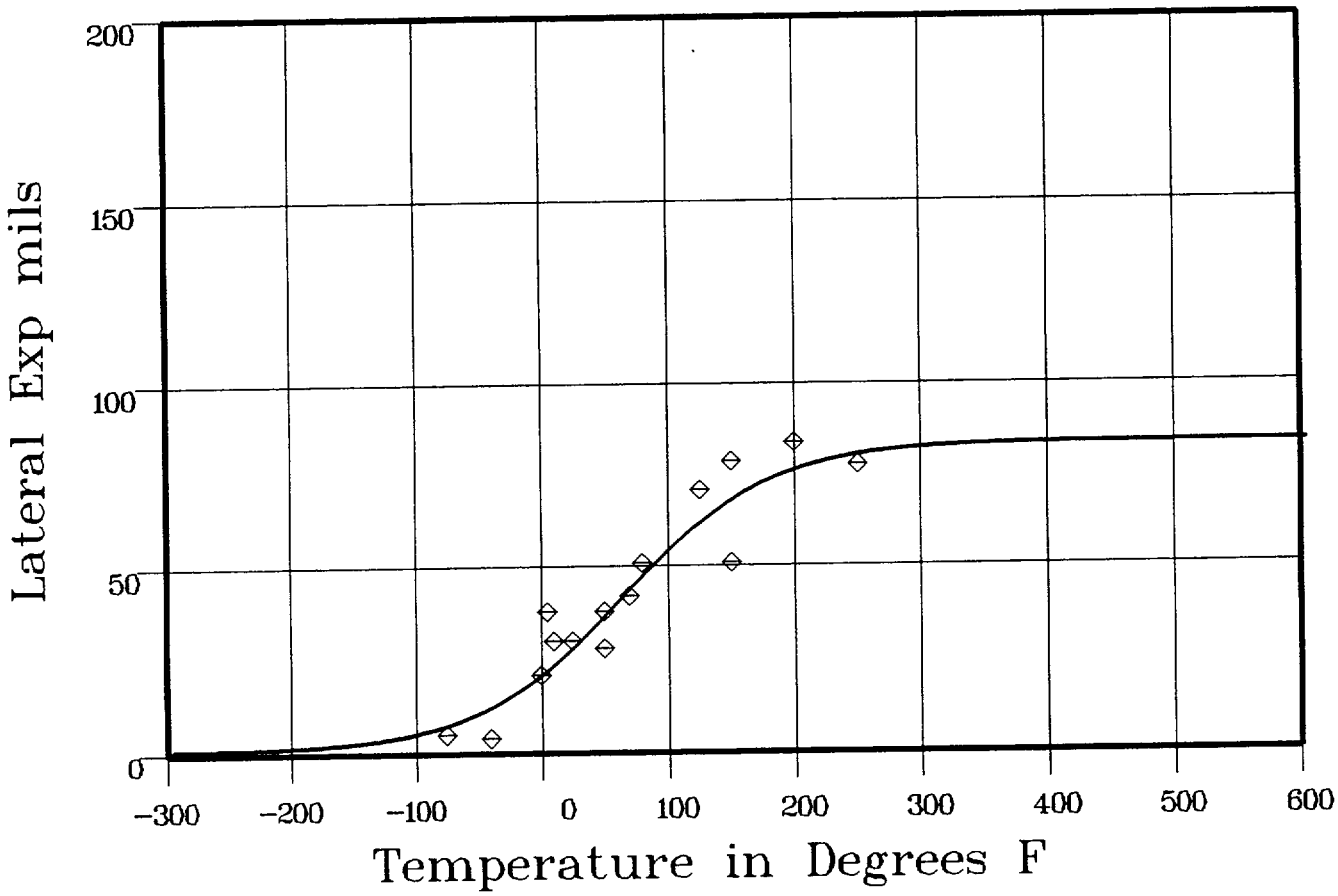
A = 42.42	B = 41.42	C = 114.49	T0 = 61.39

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

Upper Shelf L.E.: 83.84 Temperature at L.E. 35: 40.6 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: PLATE SA533B1 Ori.: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-75	5	8	-3
-40	4	13.04	-9.04
0	21	22.12	-1.12
5	38	23.52	14.47
10	30	24.98	5.01
25	30	29.68	.31
50	28	38.31	-10.31
50	38	38.31	-.31

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

CAPSULE 38 (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
70	42	45.52	-3.52
80	51	49.09	1.9
125	71	63.32	7.67
150	51	69.31	-18.31
150	79	69.31	9.68
200	84	77.08	6.91
250	78	80.87	-2.87

SUM of RESIDUALS = -2.51

CAPSULE 38 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:05:35 on 08-03-2000

Page 1

Coefficients of Curve 3

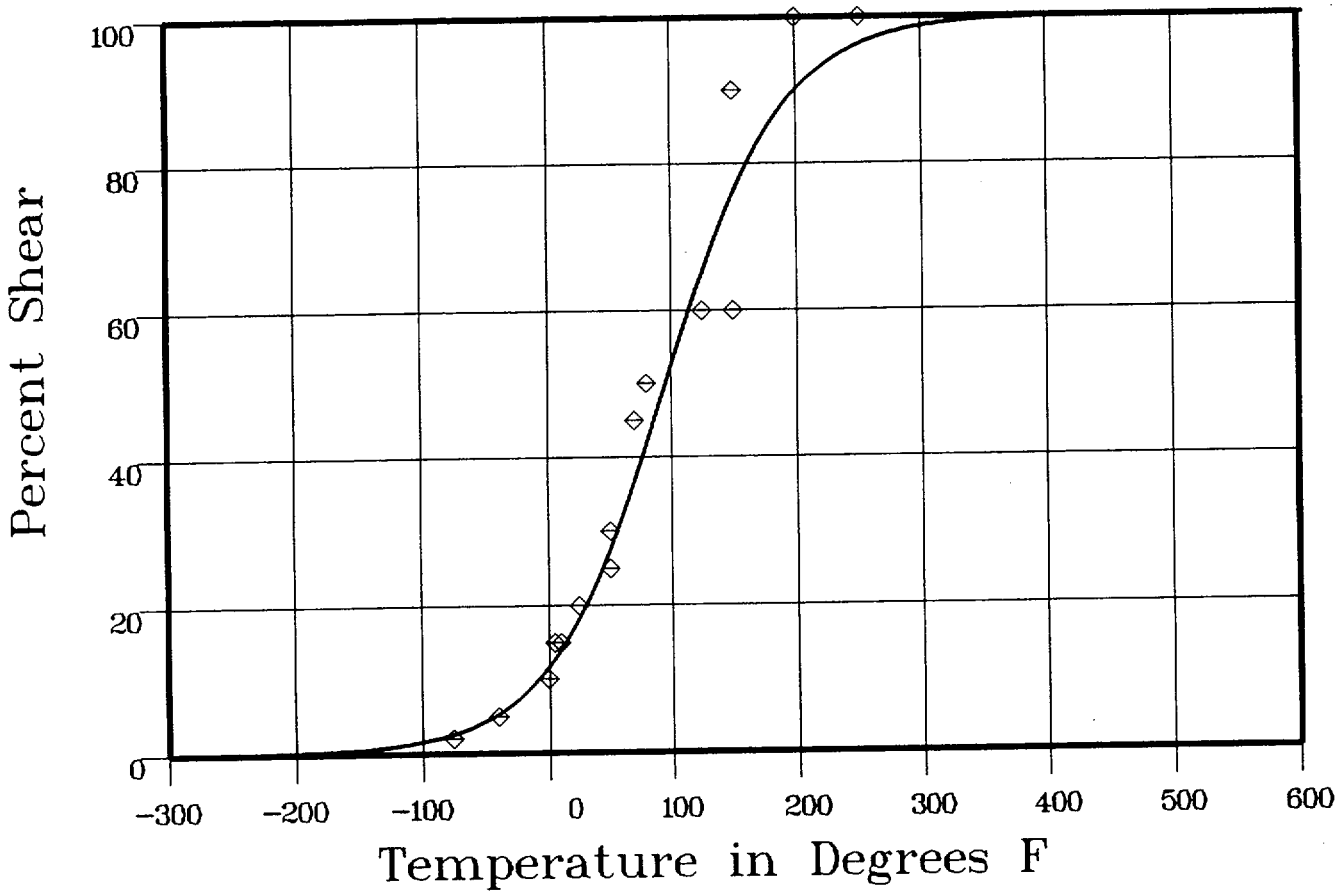
A = 50	B = 50	C = 94.51	T0 = 90.82

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

Temperature at 50% Shear: 90.8

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: PLATE SA533B1 Ori.: TL Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-75	2	2.9	-.9
-40	5	5.9	-.9
0	10	12.76	-2.76
5	15	13.99	1
10	15	15.31	-.31
25	20	19.89	.1
50	25	29.65	-4.65
50	30	29.65	.34

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

CAPSULE 38 (TRANSVERSE)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: TL

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
70	45	39.16	5.83
80	50	44.3	5.69
125	60	67.33	-7.33
150	60	77.76	-17.76
150	90	77.76	12.23
200	100	90.97	9.02
250	100	96.66	3.33
		SUM of RESIDUALS =	2.93

UNIRRADIATED (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:11:17 on 08-03-2000

Page 1

Coefficients of Curve 1

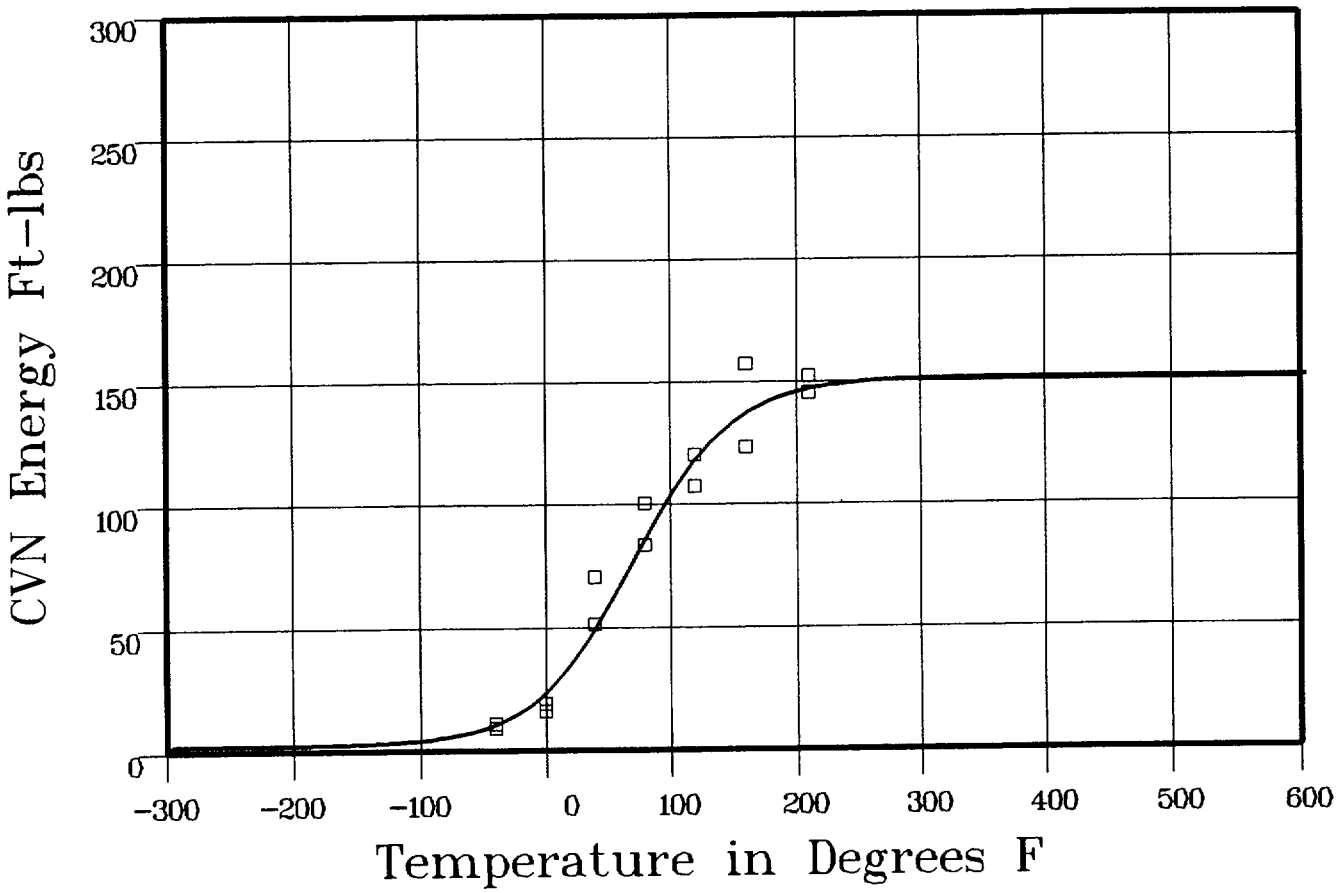
A = 76.59	B = 74.4	C = 78.78	T0 = 66.09

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

Upper Shelf Energy: 151 Fixed Temp. at 30 ft-lbs: 8.1 Temp. at 50 ft-lbs: 36.6 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-40	9	11.62	-2.62
-40	11	11.62	-.62
0	19	25.61	-6.61
0	16	25.61	-9.61
40	70	52.82	17.17
40	51	52.82	-1.82
80	83	89.59	-6.59
80	100	89.59	10.4
120	120	120.81	-.81

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

UNIRRADIATED (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
120	107	120.81	-13.81
160	123	138.44	-15.44
160	157	138.44	18.55
210	145	147.24	-2.24
210	152	147.24	4.75
		SUM of RESIDUALS =	-9.33

UNIRRADIATED (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:19:44 on 08-03-2000

Page 1

Coefficients of Curve 1

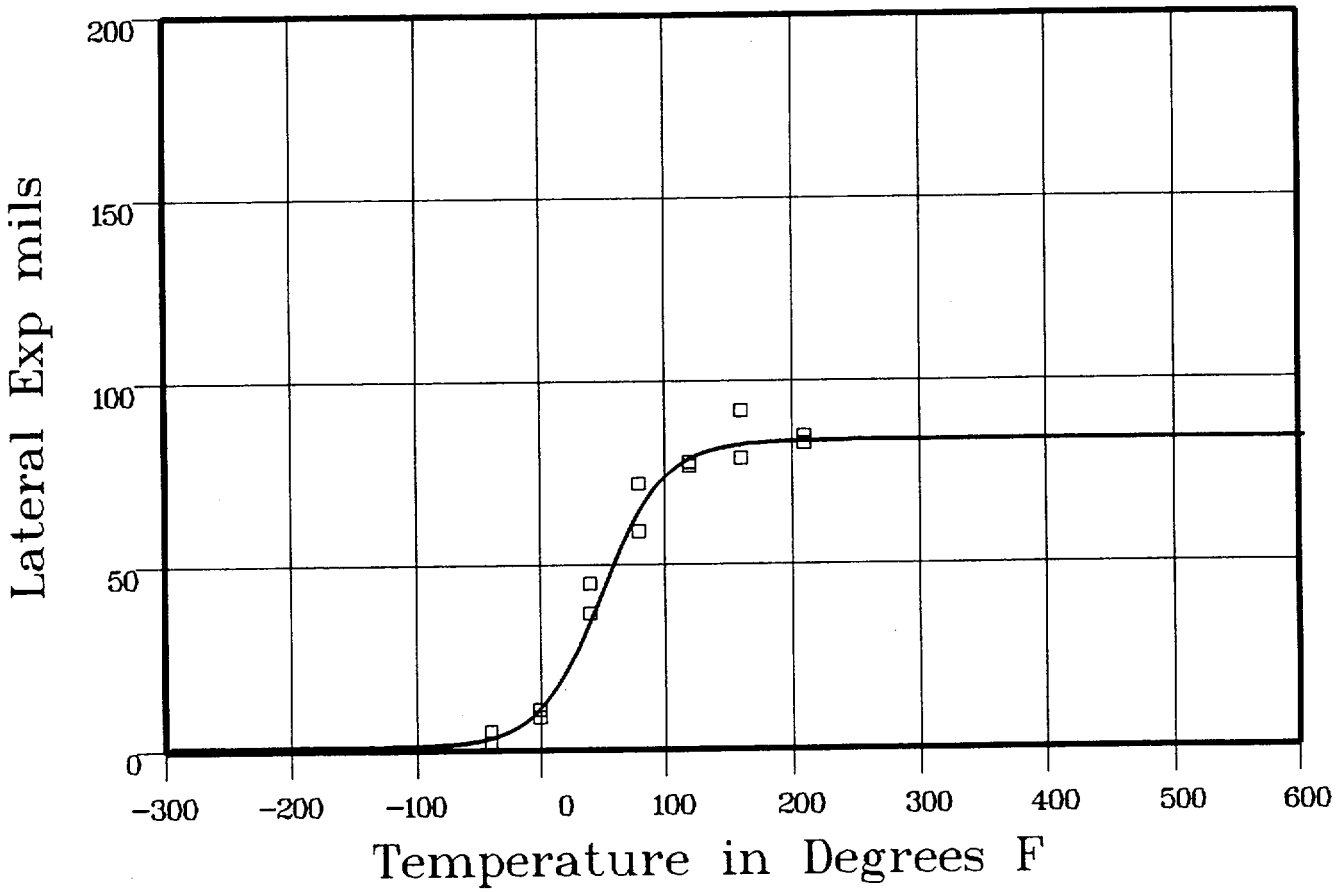
A = 42.41	B = 41.41	C = 50.66	T0 = 45

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

Upper Shelf L.E.: 83.83 Temperature at L.E. 35: 35.8 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-40	2	3.79	-1.79
-40	5	3.79	1.2
0	11	12.98	-1.98
0	9	12.98	-3.98
40	45	38.34	6.65
40	37	38.34	-1.34
80	59	67.21	-8.21
80	72	67.21	4.78
120	78	79.76	-1.76

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

UNIRRADIATED (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
120	77	79.76	-2.76
160	79	82.96	-3.96
160	92	82.96	9.03
210	85	83.71	1.28
210	83	83.71	-.71

SUM of RESIDUALS = -3.56

UNIRRADIATED (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:27:38 on 08-03-2000

Page 1

Coefficients of Curve 1

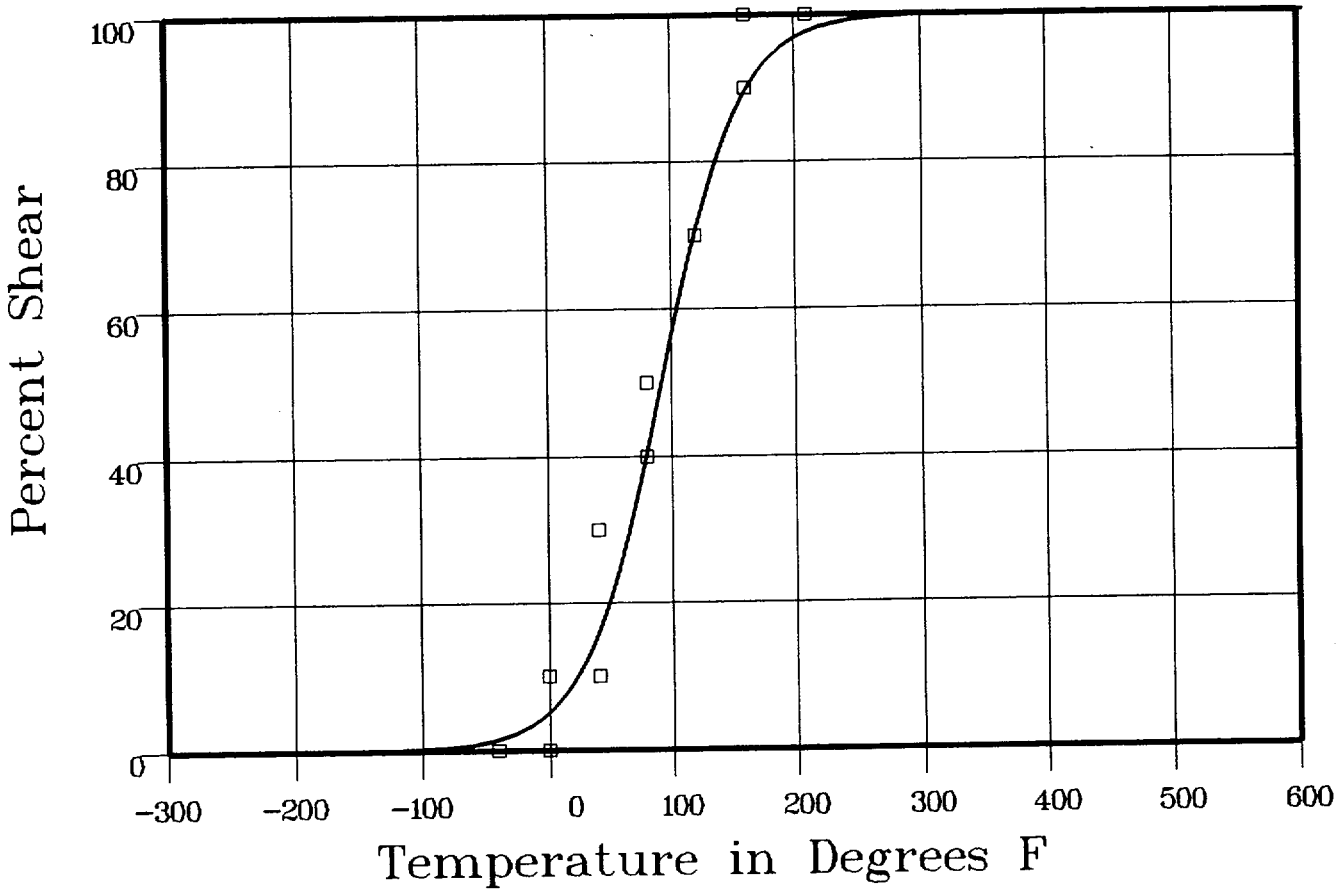
A = 50	B = 50	C = 63.19	T0 = 87.3

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

Temperature at 50% Shear: 87.3

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-40	0	1.74	-1.74
-40	0	1.74	-1.74
0	10	5.93	4.06
0	0	5.93	-5.93
40	30	18.28	11.71
40	10	18.28	-8.28
80	50	44.24	5.75
80	40	44.24	-4.24
120	70	73.78	-3.78

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

UNIRRADIATED (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
120	70	73.78	-3.78
160	90	90.89	-.89
160	100	90.89	9.1
210	100	97.98	2.01
210	100	97.98	2.01

SUM of RESIDUALS = 4.24

CAPSULE 137 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:11:17 on 08-03-2000

Page 1

Coefficients of Curve 2

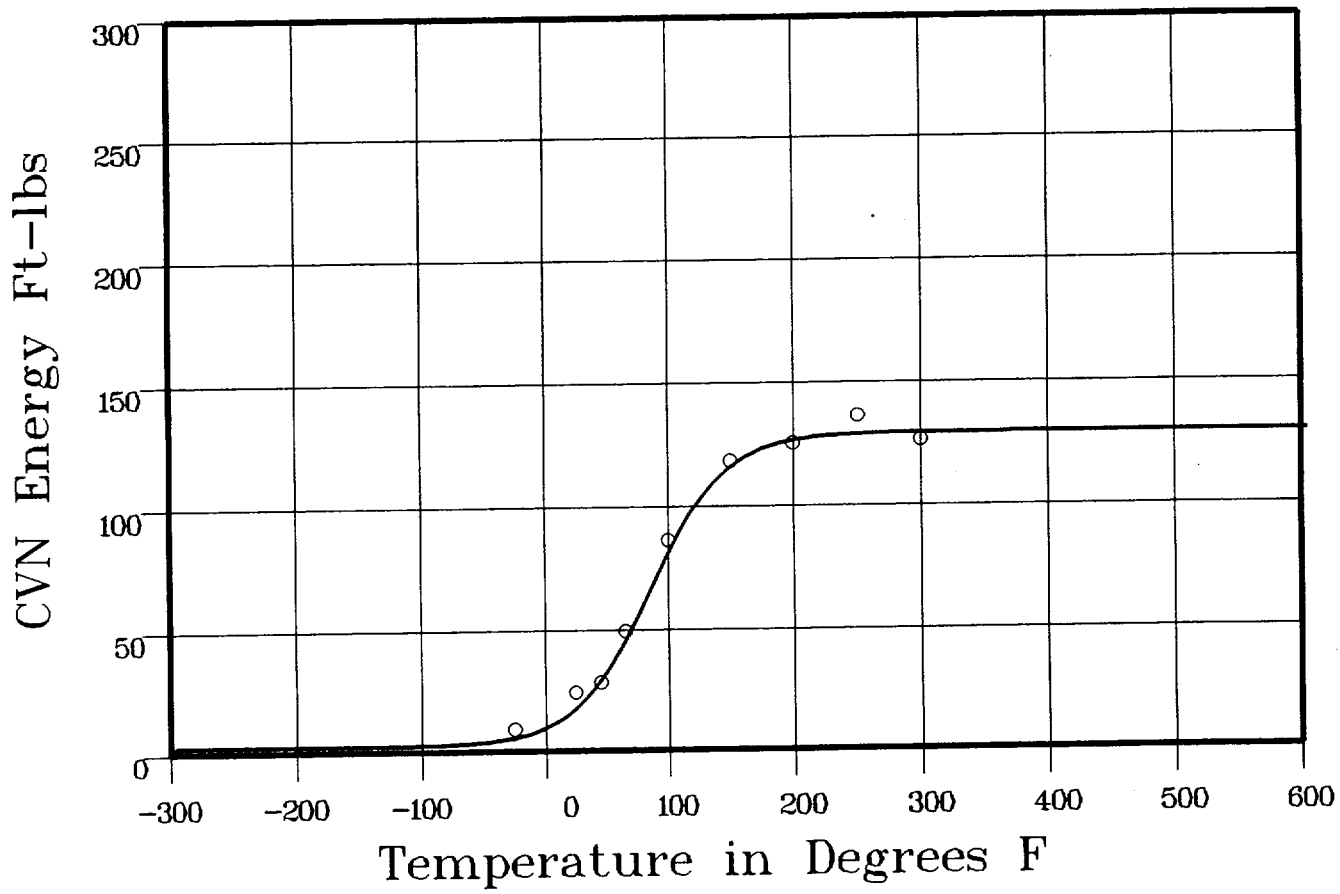
A = 65.59	B = 63.4	C = 60.68	T0 = 80.62

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

Upper Shelf Energy: 129 Fixed Temp. at 30 ft-lbs: 42 Temp. at 50 ft-lbs: 65.3 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap: 137 Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-25	9	5.98	3.01
25	24	19.67	4.32
45	28	32.13	-4.13
65	49	49.62	-.62
100	86	85.18	.81
150	118	117.3	.69
200	125	126.56	-1.56
250	136	128.52	7.47

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

CAPSULE 137 (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
300	126	128.9	-2.9

SUM of RESIDUALS = 7.08

CAPSULE 137 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:19:44 on 08-03-2000

Page 1

Coefficients of Curve 2

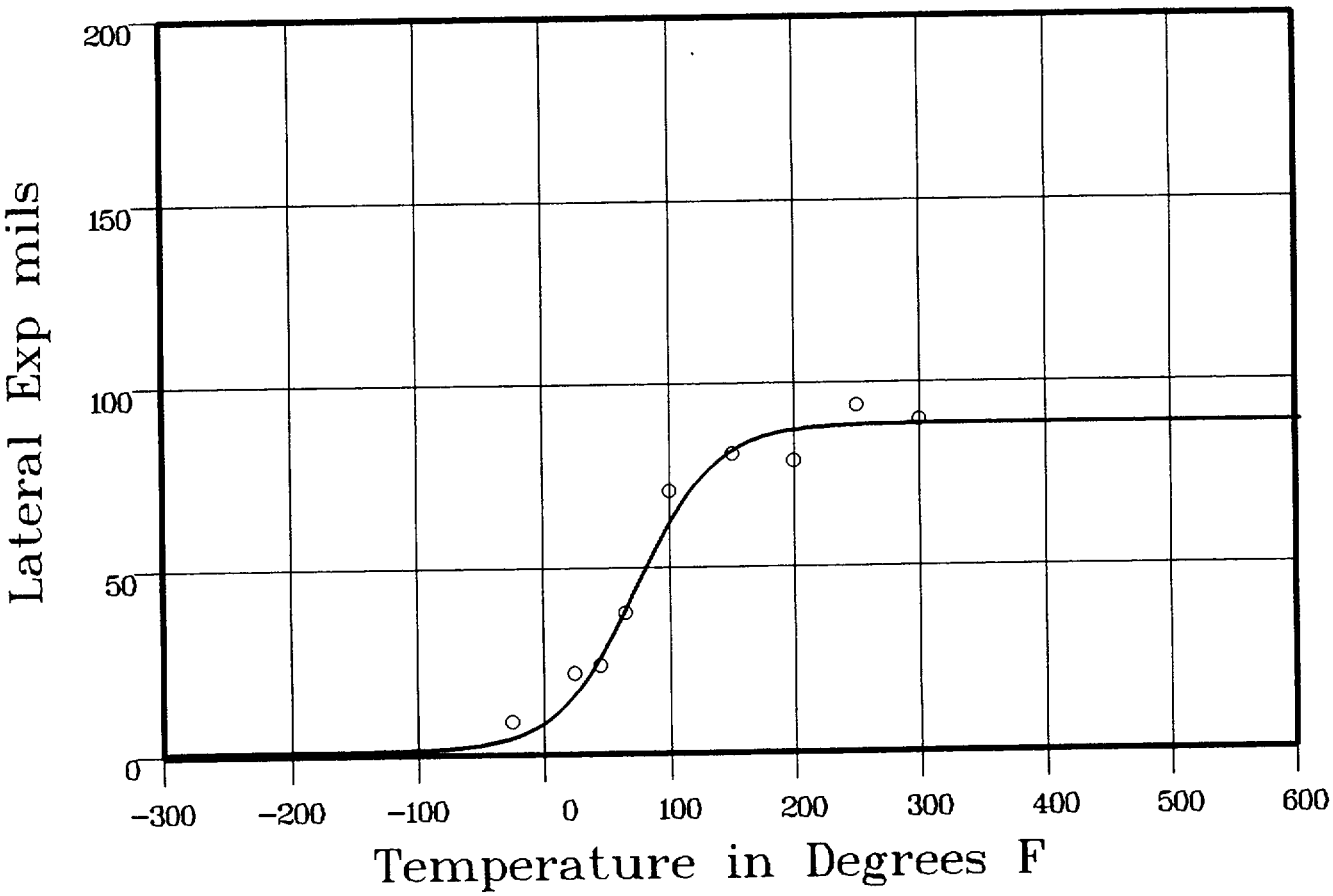
A = 44.88	B = 43.88	C = 61.91	T0 = 69.37

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

Upper Shelf L.E.: 88.77 Temperature at L.E. 35: 55.1 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-25	9	4.97	4.02
25	22	17.9	4.09
45	24	28.45	-4.45
65	38	41.79	-3.79
100	71	64.98	6.01
150	81	82.73	-1.73
200	79	87.5	-8.5
250	94	88.52	5.47

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

CAPSULE 137 (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
300	90	88.72	1.27

SUM of RESIDUALS = 2.4

CAPSULE 137 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:27:38 on 08-03-2000

Page 1

Coefficients of Curve 2

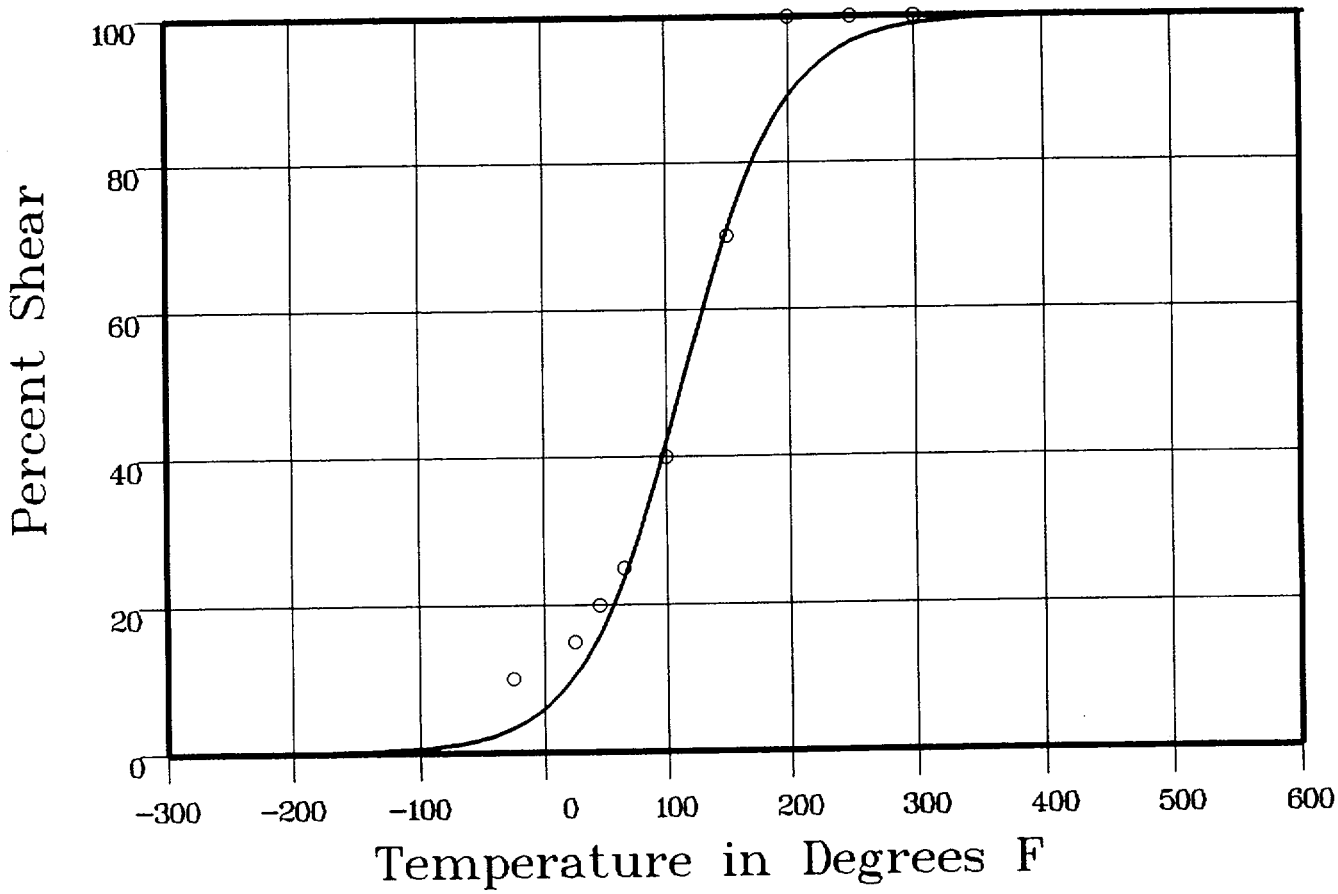
A = 50	B = 50	C = 82.14	T0 = 108.69

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

Temperature at 50% Shear: 108.6

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-25	10	3.71	6.28
25	15	11.53	3.46
45	20	17.49	2.5
65	25	25.65	-.65
100	40	44.72	-4.72
150	70	73.21	-3.21
200	100	90.23	9.76
250	100	96.89	3.1

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

CAPSULE 137 (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
300	100	99.06	.93

SUM of RESIDUALS = 17.46

CAPSULE 38 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:11:17 on 08-03-2000

Page 1

Coefficients of Curve 3

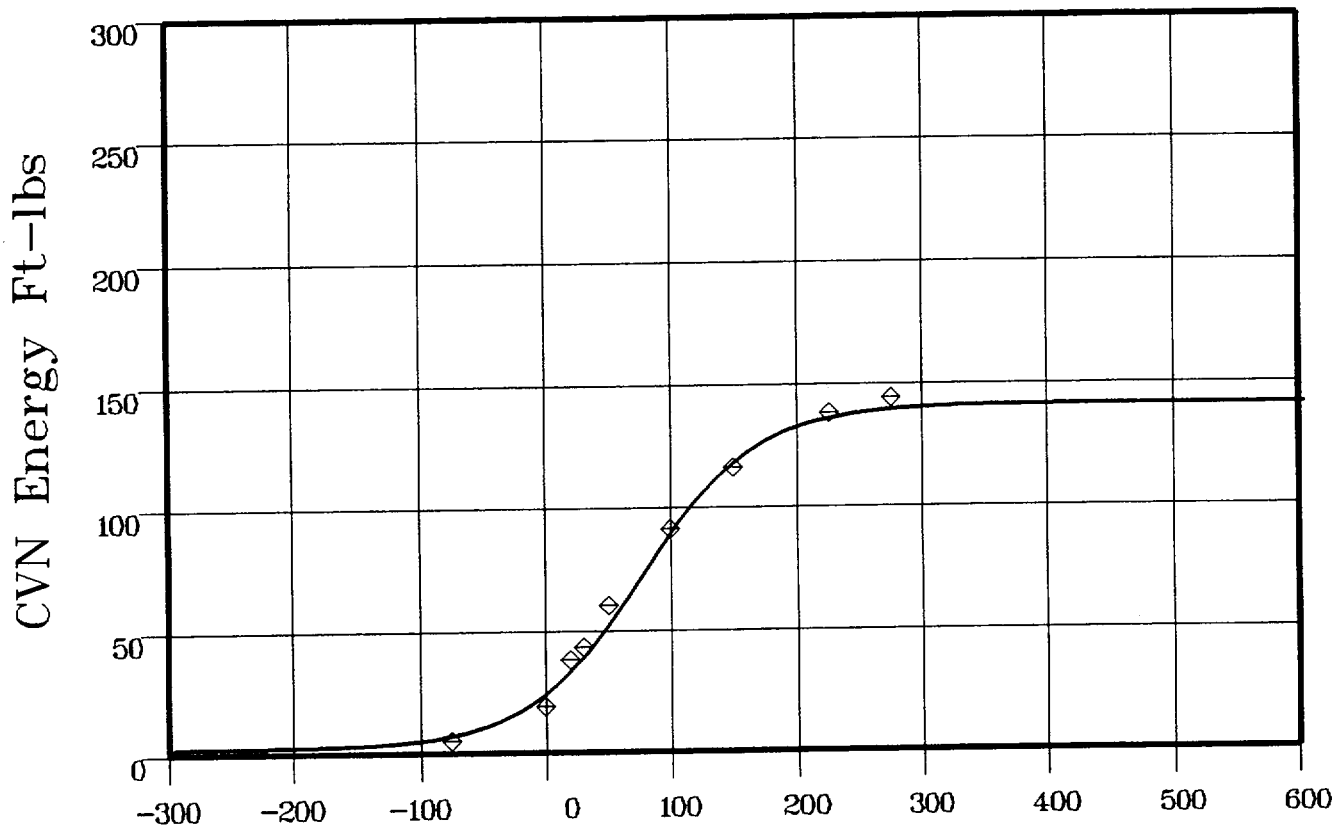
A = 71.59	B = 69.4	C = 91.26	T0 = 71.9

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

Upper Shelf Energy: 141 Fixed Temp. at 30 ft-lbs: 8.7 Temp. at 50 ft-lbs: 42.5 Lower Shelf Energy: 2.19 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PVI Cap.: 38 Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-75	5	7.53	-2.53
0	19	25.98	-6.98
20	38	35.89	2.1
30	43	41.79	1.2
50	60	55.25	4.74
100	91	92.31	-1.31
150	116	119.76	-3.76
225	138	136.31	1.68

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

CAPSULE 38 (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
275	144	139.39	4.6

SUM of RESIDUALS = -.27

CAPSULE 38 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:19:44 on 08-03-2000

Page 1

Coefficients of Curve 3

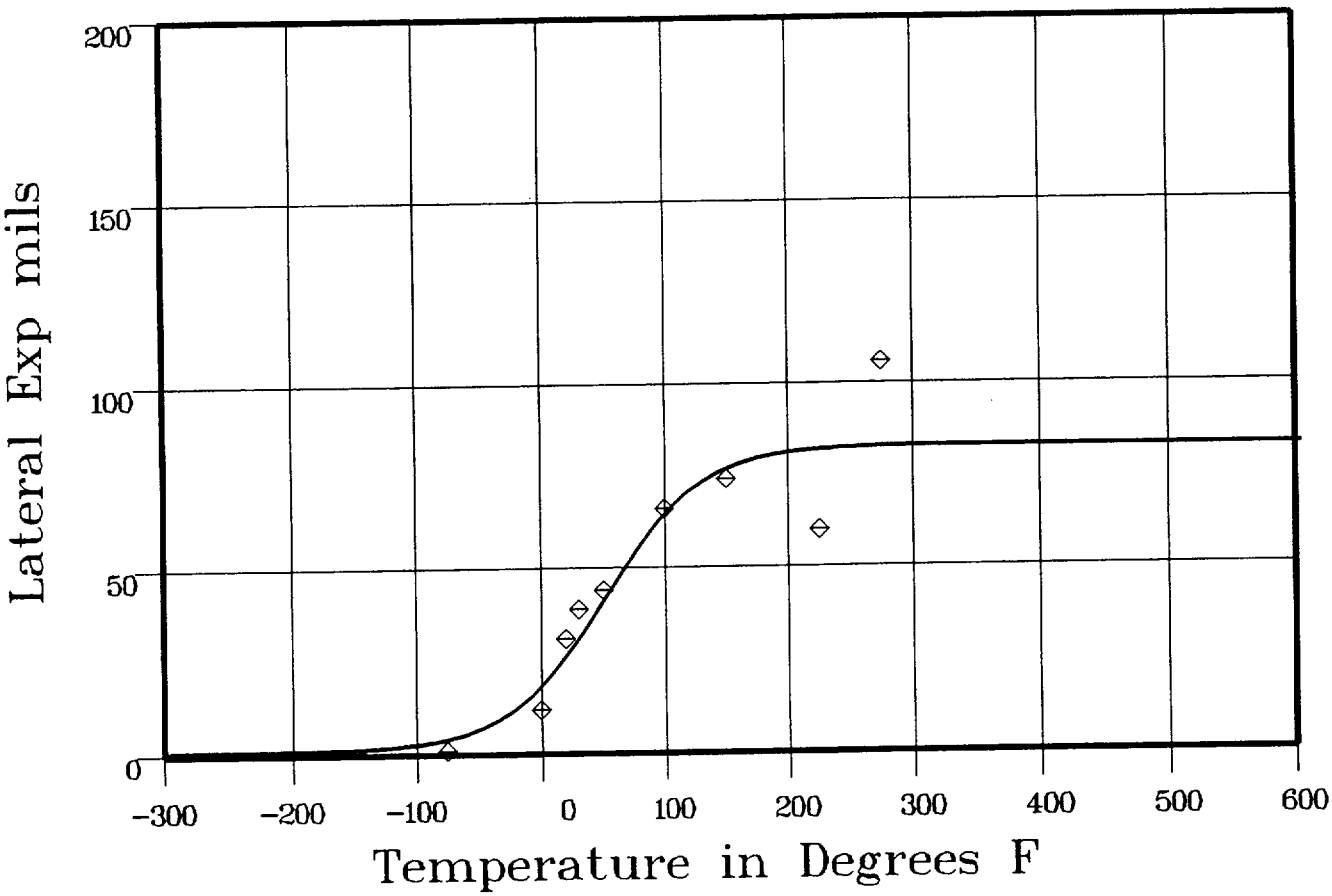
A = 41.89	B = 40.89	C = 78.24	T0 = 46.5

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

Upper Shelf L.E.: 82.78 Temperature at L.E. 35: 33.1 Lower Shelf L.E.: 1 Fixed

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-75	1	4.5	-3.5
0	12	20.09	-8.09
20	31	28.54	2.45
30	39	33.38	5.61
50	44	43.71	.28
100	66	66.17	-.17
150	74	77.36	-3.36
225	60	81.93	-21.93

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

CAPSULE 38 (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
275	106	82.54	23.45

SUM of RESIDUALS = -5.26

CAPSULE 38 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:27:38 on 08-03-2000

Page 1

Coefficients of Curve 3

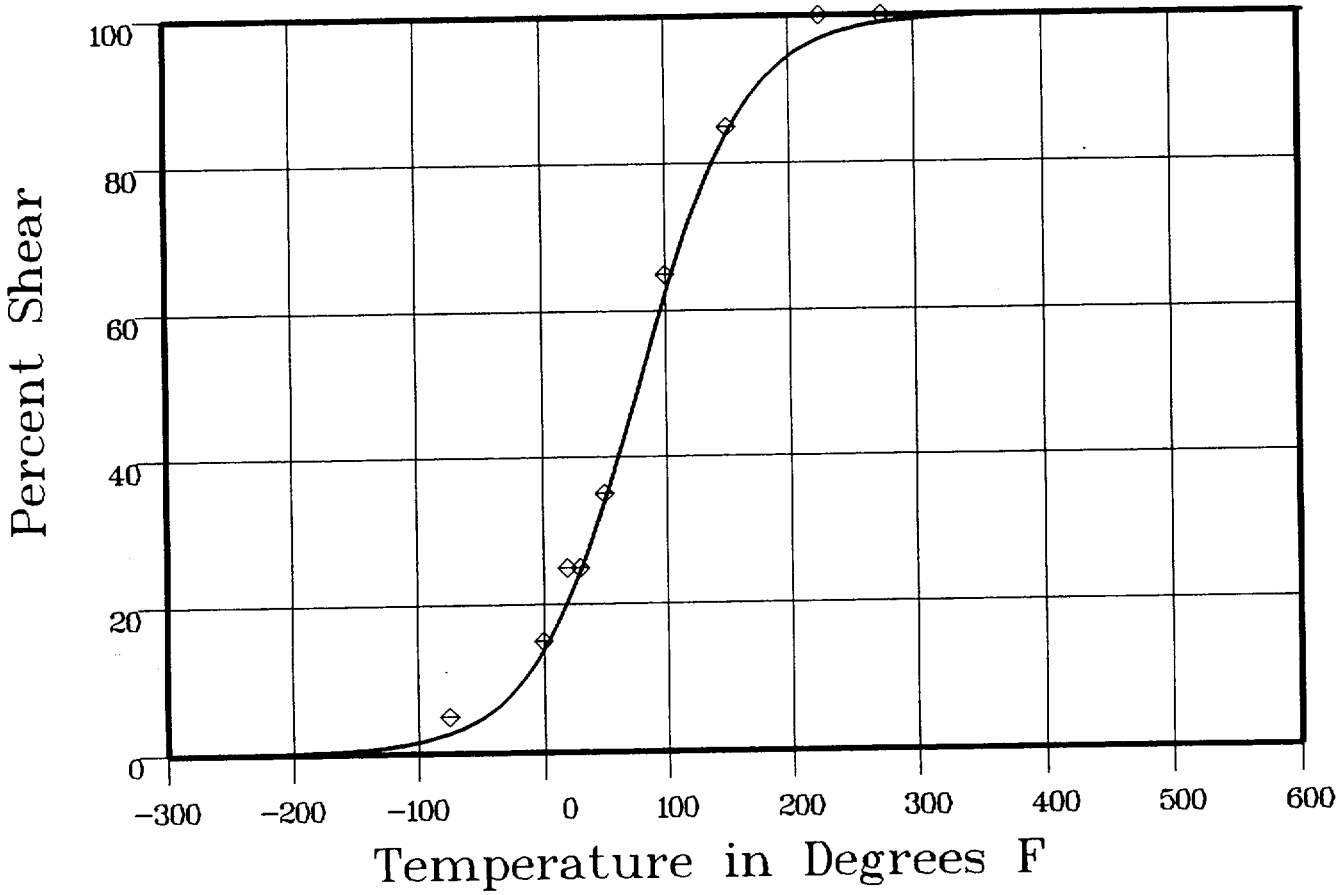
A = 50	B = 50	C = 85.82	T0 = 73.77

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

Temperature at 50% Shear: 73.7

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: PLATE SA533B1 Ori.: LT Heat #: M-6701-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-75	5	3.02	1.97
0	15	15.19	-.19
20	25	22.21	2.78
30	25	26.5	-1.5
50	35	36.49	-1.49
100	65	64.82	.17
150	85	85.52	-.52
225	100	97.13	2.86

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

CAPSULE 38 (LONGITUDINAL)

Material: PLATE SA533B1 Heat Number: M-6701-2 Orientation: LT

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
275	100	99.08	.91

SUM of RESIDUALS = 4.99

UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:42:36 on 08-03-2000

Page 1

Coefficients of Curve 1

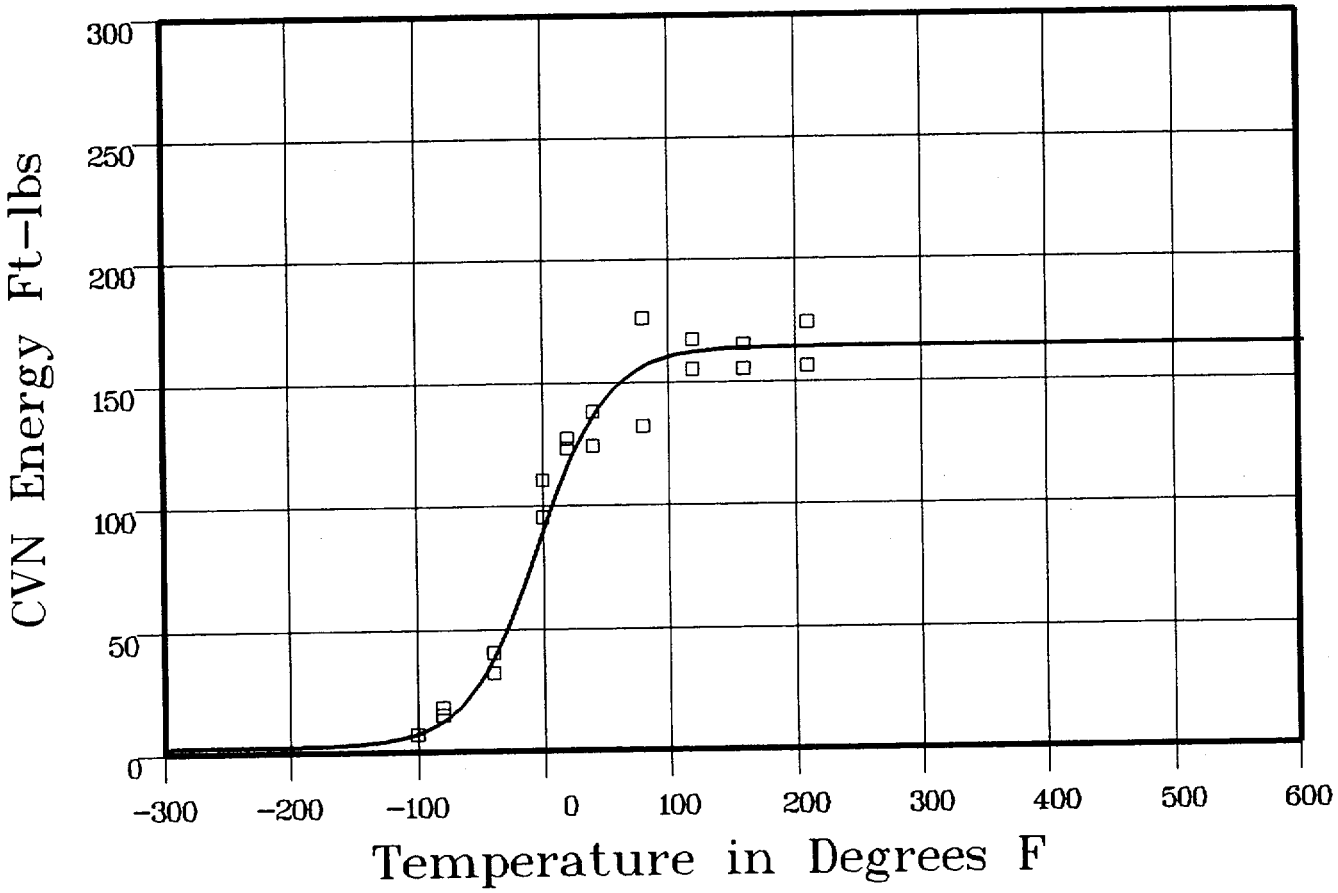
A = 83.09	B = 80.9	C = 56.43	T0 = -8.9

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

Upper Shelf Energy: 164 Fixed Temp. at 30 ft-lbs: -53.2 Temp. at 50 ft-lbs: -33.4 Lower Shelf Energy: 2.19 Fixed

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: WELD Ori.: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-100	7	8.36	-1.36
-80	15	14.25	.74
-80	18	14.25	3.74
-40	32	42.54	-10.54
-40	40	42.54	-2.54
0	110	95.76	14.23
0	95	95.76	-.76
20	123	121.25	1.74
20	127	121.25	5.74

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

UNIRRADIATED

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
20	123	121.25	1.74
40	138	139.7	-1.7
40	124	139.7	-15.7
80	132	157.35	-25.35
80	176	157.35	18.64
120	155	162.33	-7.33
120	167	162.33	4.66
160	165	163.59	1.4
160	155	163.59	-8.59
210	174	163.93	10.06
210	156	163.93	-7.93
		SUM of RESIDUALS =	-19.12

UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:51:31 on 08-03-2000

Page 1

Coefficients of Curve 1

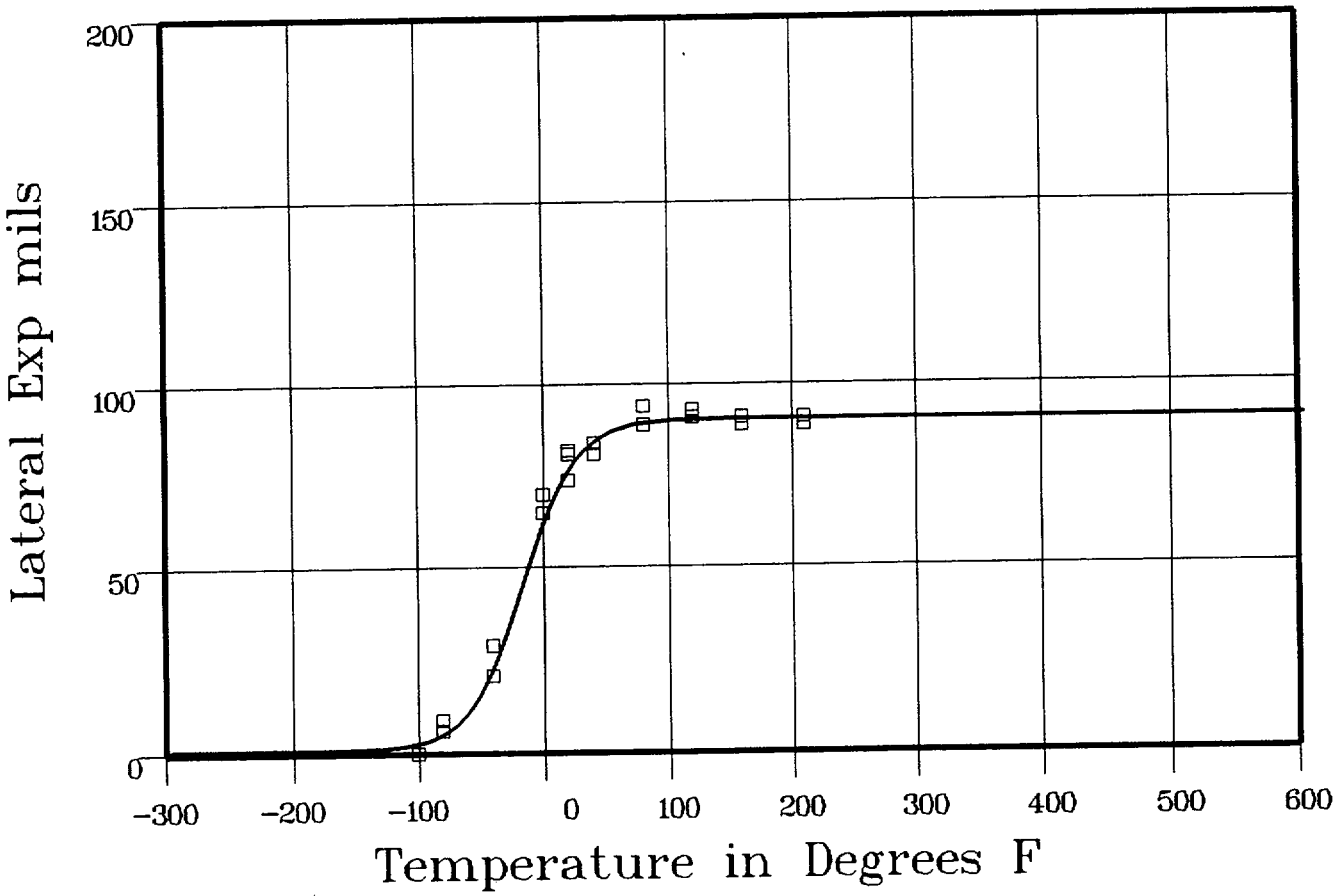
A = 45.71	B = 44.71	C = 42.24	T0 = -20.62

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

Upper Shelf L.E: 90.43 Temperature at L.E. 35: -30.9 Lower Shelf L.E: 1 Fixed

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: WELD Ori.: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-100	0	3.03	-3.03
-80	6	6.07	-.07
-80	9	6.07	2.92
-40	21	26.53	-5.53
-40	29	26.53	2.46
0	70	65.96	4.03
0	65	65.96	-.96
20	74	79.02	-5.02
20	82	79.02	2.97

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

UNIRRADIATED

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
20	81	79.02	1.97
40	81	85.63	-4.63
40	84	85.63	-1.63
80	89	89.67	-.67
80	94	89.67	4.32
120	93	90.31	2.68
120	91	90.31	.68
160	89	90.41	-1.41
160	91	90.41	.58
210	89	90.43	-1.43
210	91	90.43	.56
			SUM of RESIDUALS = -1.21

UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:59:59 on 08-03-2000

Page 1

Coefficients of Curve 1

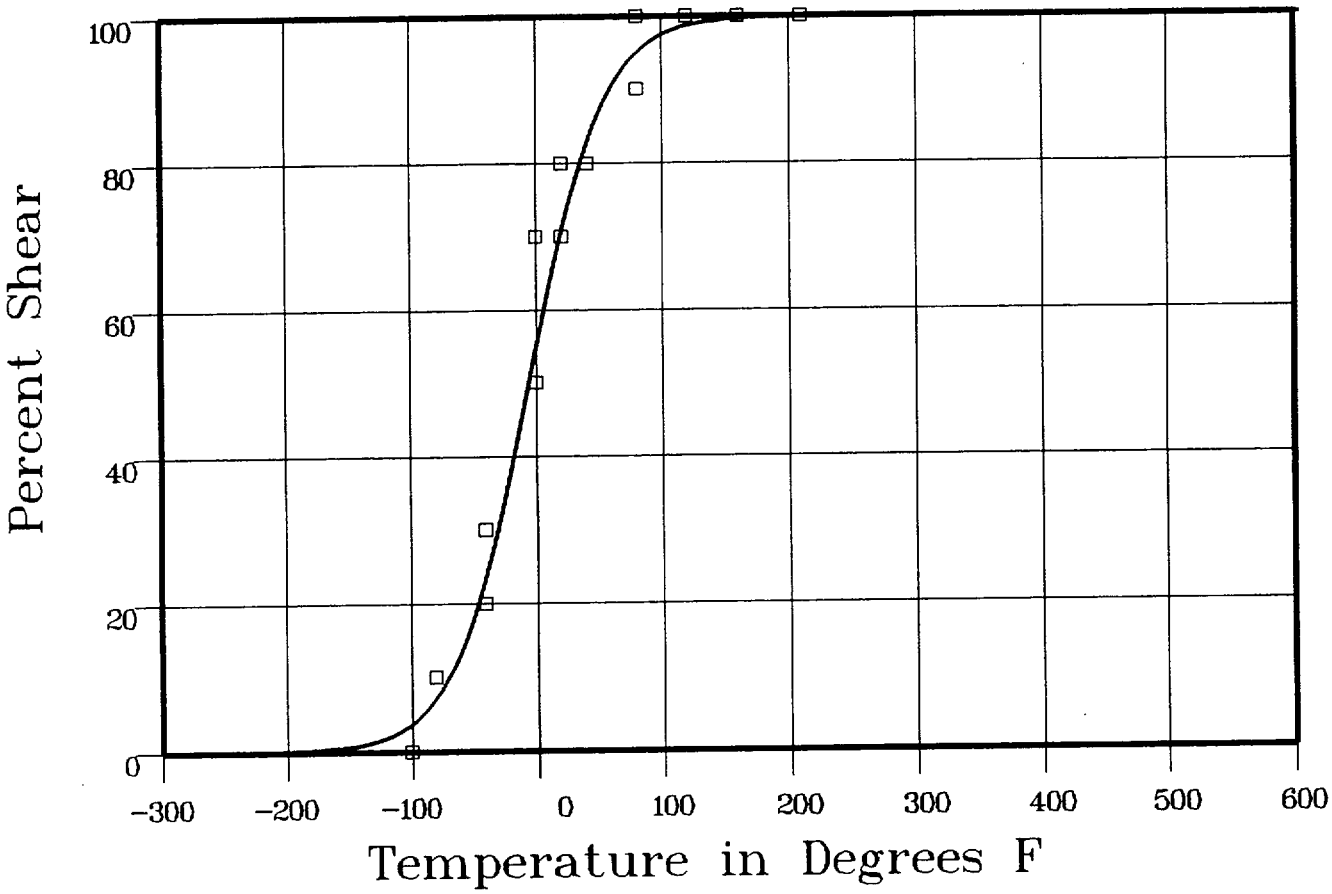
A = 50	B = 50	C = 57.87	T0 = -10.44

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

Temperature at 50% Shear: -10.4

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: WELD Ori.: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-100	0	4.33	-4.33
-80	10	8.29	1.7
-80	10	8.29	1.7
-40	20	26.47	-6.47
-40	30	26.47	3.52
0	50	58.92	-8.92
0	70	58.92	11.07
20	70	74.11	-4.11
20	80	74.11	5.88

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

UNIRRADIATED

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
20	80	74.11	5.88
40	80	85.1	-5.1
40	80	85.1	-5.1
80	90	95.79	-5.79
80	100	95.79	4.2
120	100	98.9	1.09
120	100	98.9	1.09
160	100	99.72	.27
160	100	99.72	.27
210	100	99.95	.04
210	100	99.95	.04

SUM of RESIDUALS = -3.03

CAPSULE 137

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:48:57 on 08-03-2000

Page 1

Coefficients of Curve 2

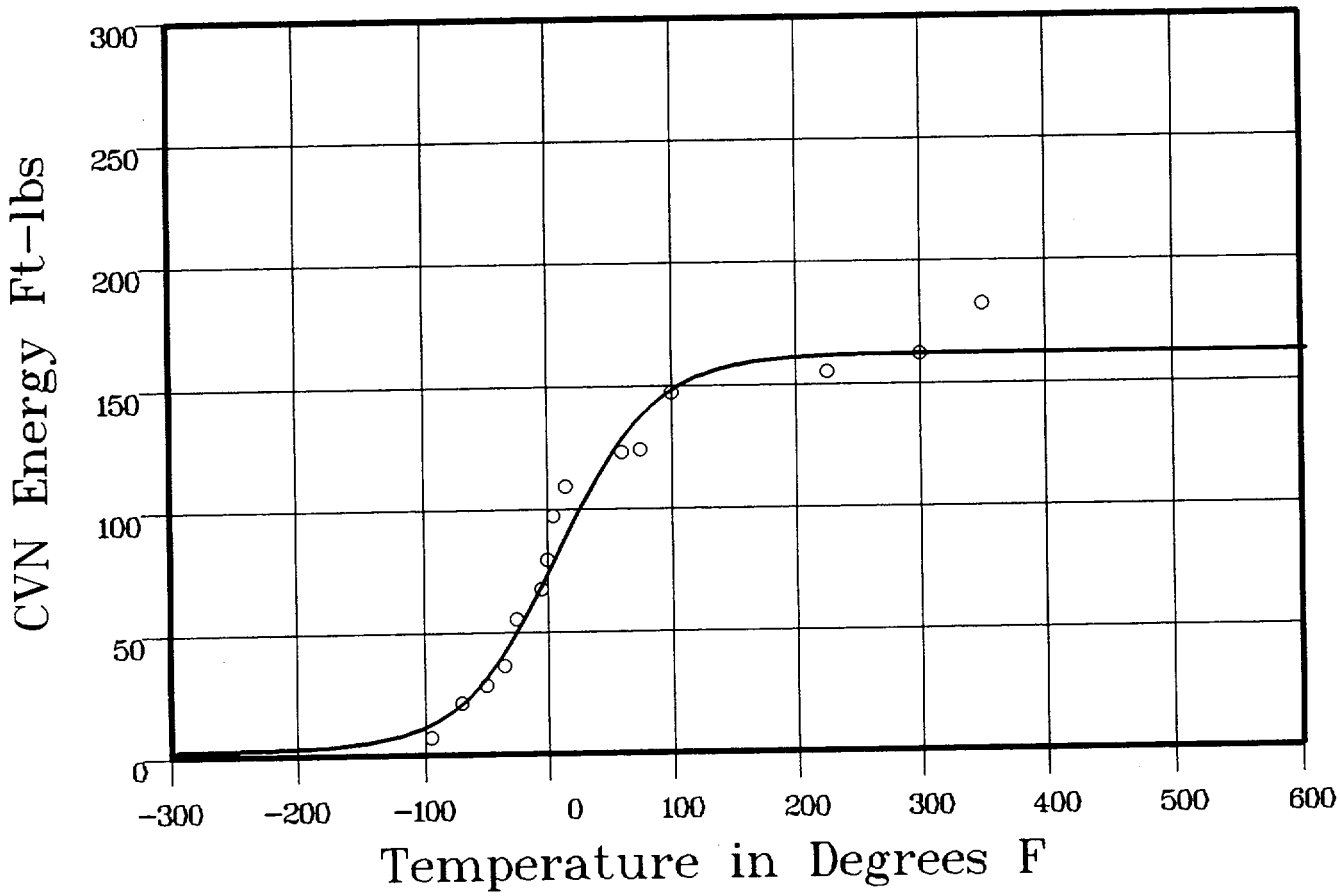
A = 82.09	B = 79.9	C = 77.6	T0 = 4.21

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

Upper Shelf Energy: 162 Fixed Temp. at 30 ft-lbs: -56.2 Temp. at 50 ft-lbs: -28.8 Lower Shelf Energy: 2.19 Fixed

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: WELD Ori.: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-95	7	13.69	-6.69
-70	21	22.76	-1.76
-50	28	33.87	-5.87
-35	36	44.84	-8.84
-25	55	53.36	1.63
-5	67	72.65	-5.65
0	79	77.76	1.23
5	97	82.9	14.09

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

CAPSULE 137

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
15	109	93.12	15.87
60	123	131.33	-8.33
75	124	139.79	-15.79
100	147	149.51	-2.51
225	155	161.46	-6.46
300	162	161.92	.07
350	182	161.97	20.02

SUM of RESIDUALS = -9

CAPSULE 137

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:51:31 on 08-03-2000

Page 1

Coefficients of Curve 2

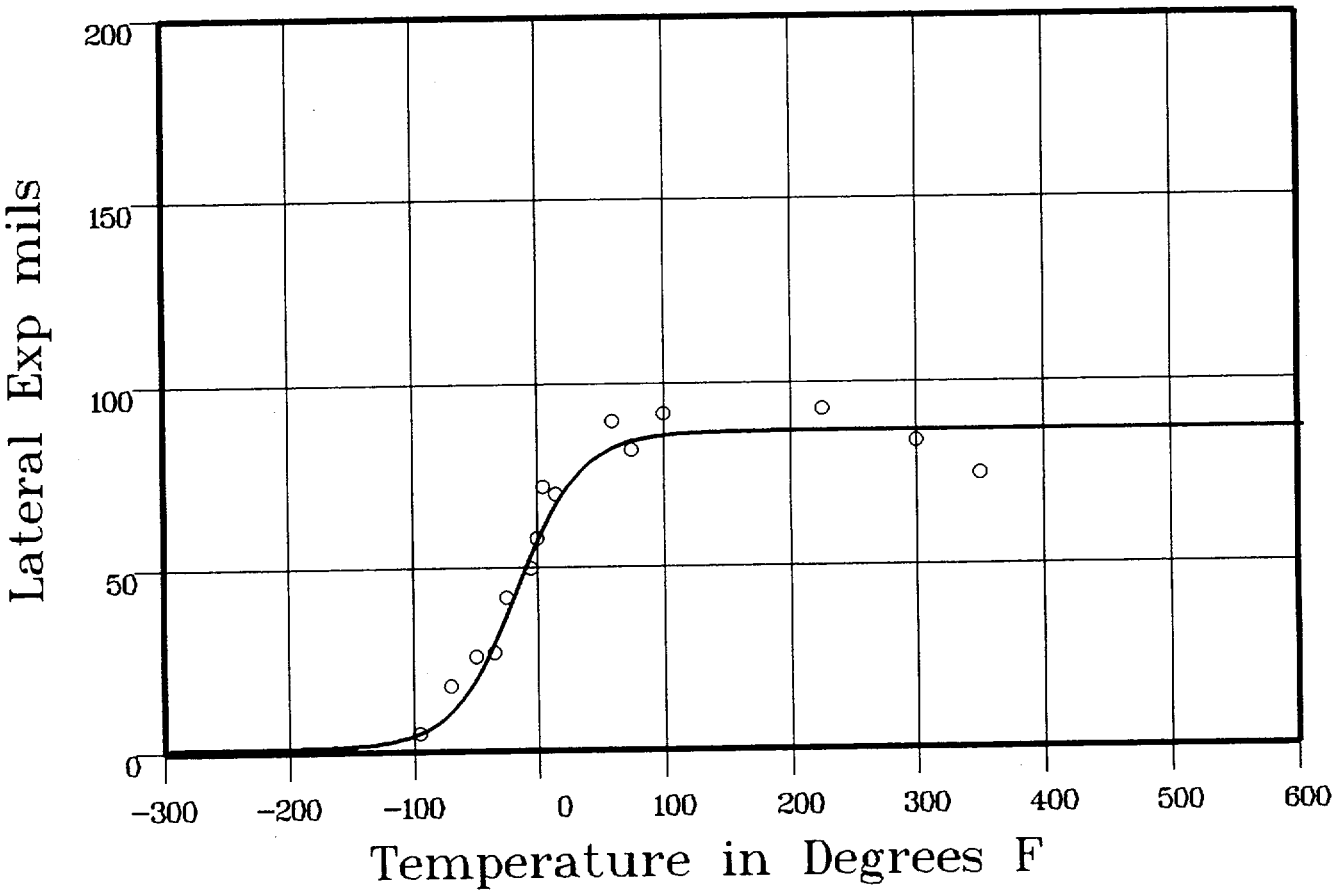
A = 43.93	B = 42.93	C = 52.25	T0 = -21.09

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

Upper Shelf L.E.: 86.87 Temperature at L.E. 35: -32.1 Lower Shelf L.E.: 1 Fixed

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: WELD Ori.: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-95	5	5.79	-.79
-70	18	12.45	5.54
-50	26	22.34	3.65
-35	27	32.77	-5.77
-25	42	40.73	1.26
-5	50	56.76	-6.76
0	58	60.38	-2.38
5	72	63.76	8.23

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

CAPSULE 137

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
15	70	69.63	.36
60	90	83.19	6.8
75	82	84.76	-2.76
100	92	86.05	5.94
225	93	86.87	6.12
300	84	86.87	-2.87
350	75	86.87	-11.87

SUM of RESIDUALS = 4.71

CAPSULE 137

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:59:59 on 08-03-2000

Page 1

Coefficients of Curve 2

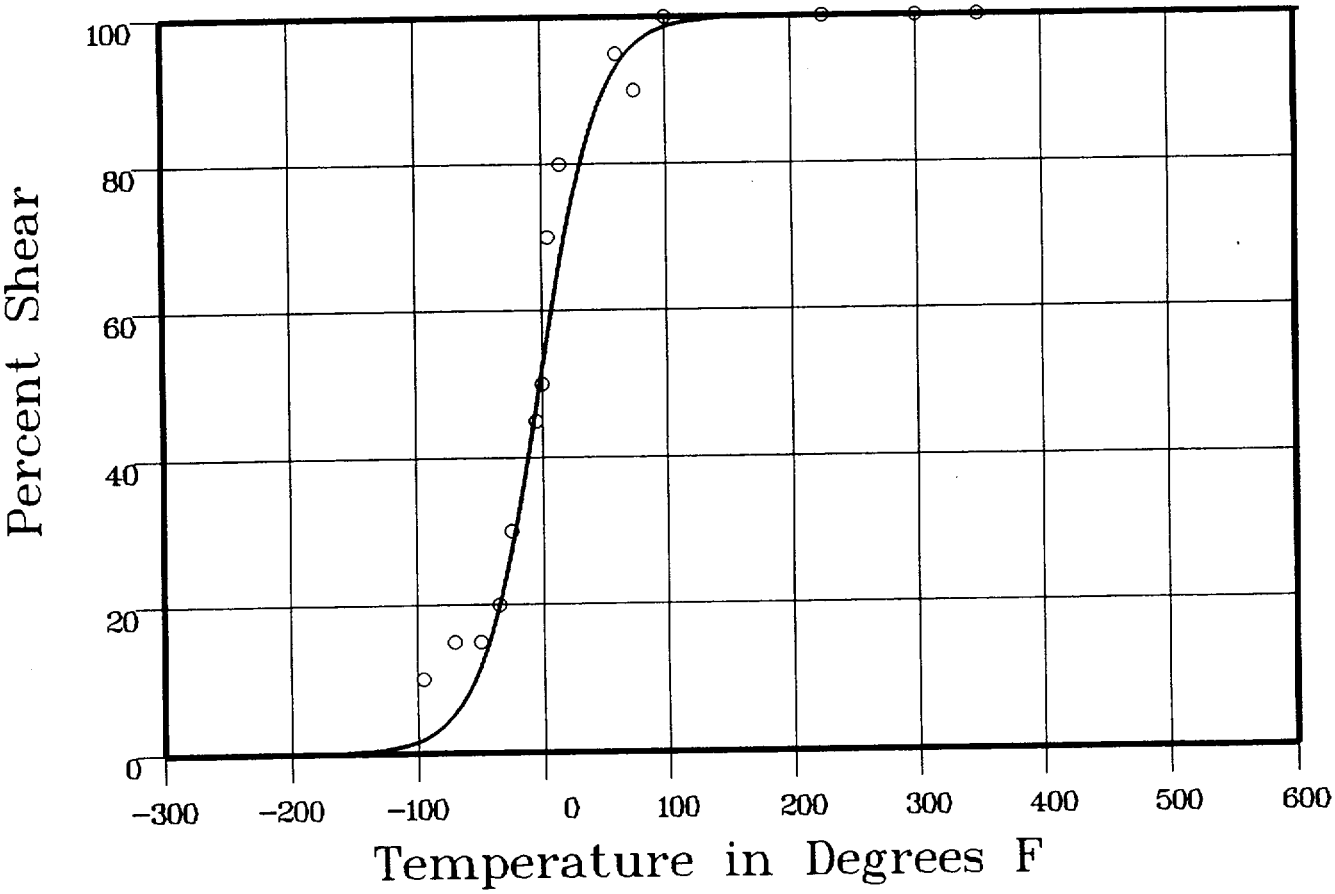
A = 50	B = 50	C = 47.34	T0 = -6.56

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

Temperature at 50% Shear: -6.5

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap: 137 Material: WELD Ori: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-95	10	2.32	7.67
-70	15	6.41	8.58
-50	15	13.76	1.23
-35	20	23.12	-3.12
-25	30	31.45	-1.45
-5	45	51.64	-6.64
0	50	56.88	-6.88
5	70	61.97	8.02

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

CAPSULE 137

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
15	80	71.31	8.68
60	95	94.33	.66
75	90	96.9	-6.9
100	100	98.9	1.09
225	100	99.99	0
300	100	99.99	0
350	100	99.99	0

SUM of RESIDUALS = 10.94

CAPSULE 38

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:42:36 on 08-03-2000

Page 1

Coefficients of Curve 3

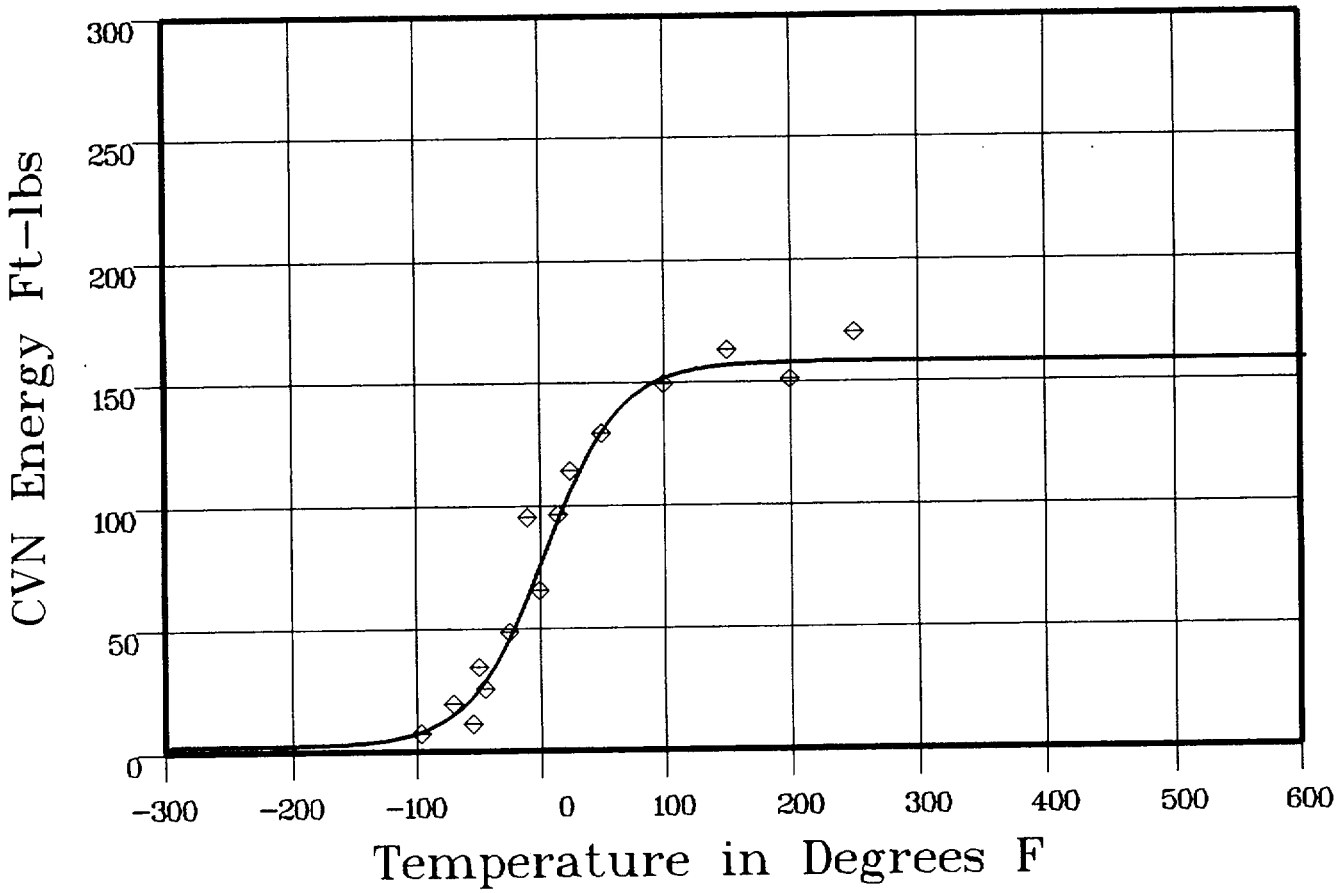
A = 80.09	B = 77.9	C = 60.95	T0 = -.46

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

Upper Shelf Energy: 158 Fixed Temp. at 30 ft-lbs: -47 Temp. at 50 ft-lbs: -25.3 Lower Shelf Energy: 2.19 Fixed

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: WELD Ori.: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-96	7	8.69	-1.69
-70	19	16.63	2.36
-55	11	24.5	-13.5
-50	34	27.82	6.17
-45	25	31.53	-6.53
-25	48	50.33	-2.33
-10	95	68.01	26.98
0	65	80.69	-15.69

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

CAPSULE 38

Page 2

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
15	96	99.45	-3.45
25	114	110.87	3.12
50	129	133.02	-4.02
100	149	152.44	-3.44
150	163	156.89	6.1
200	151	157.78	-6.78
250	170	157.95	12.04
		SUM of RESIDUALS =	-.68

CAPSULE 38

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:51:31 on 08-03-2000

Page 1

Coefficients of Curve 3

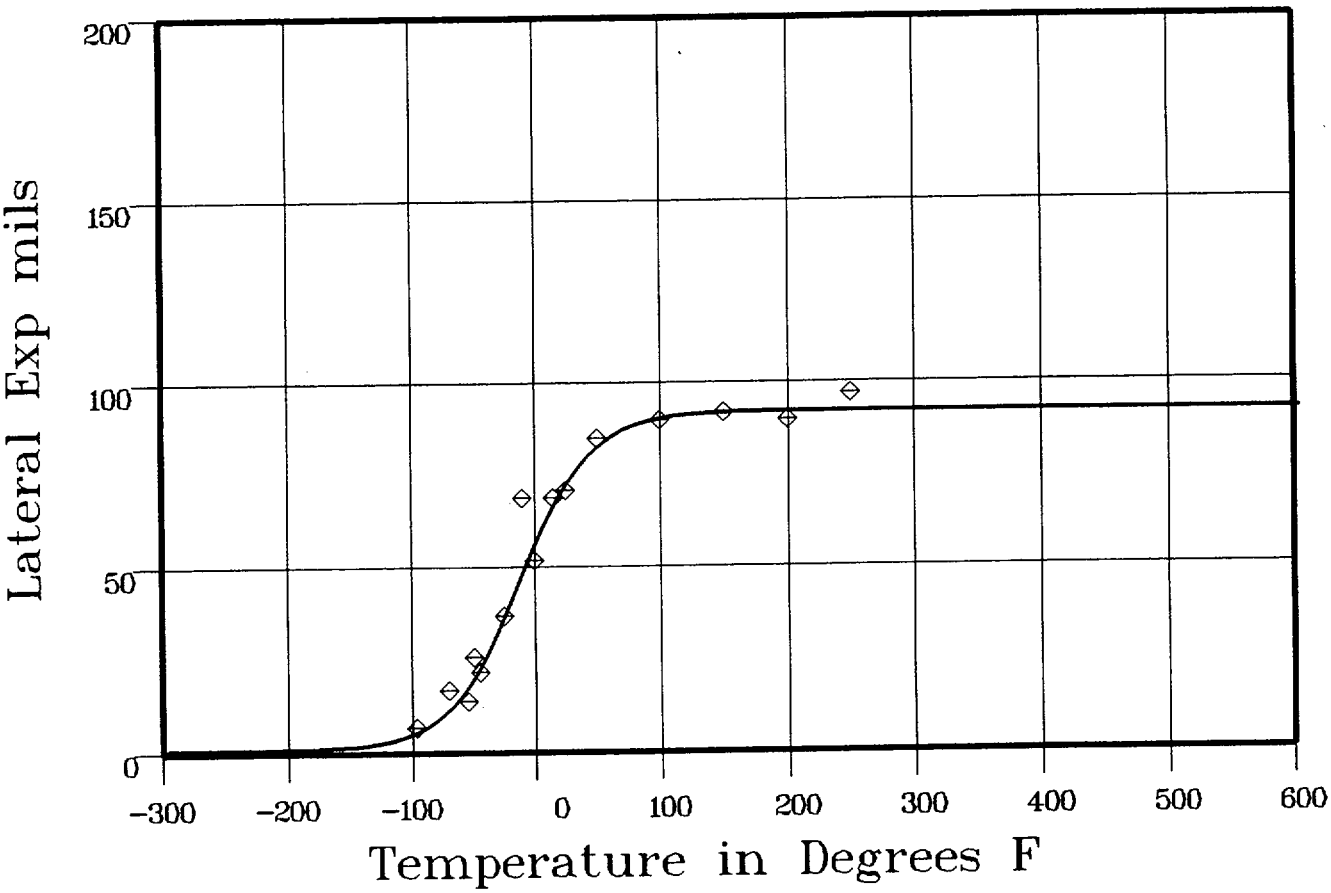
A = 46.59	B = 45.59	C = 57.39	T0 = -16.87

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

Upper Shelf L.E.: 92.18 Temperature at L.E. 35: -31.7 Lower Shelf L.E.: 1 Fixed

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: WELD Ori.: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-96	7	6.44	.55
-70	17	13.37	3.62
-55	14	20.09	-6.09
-50	26	22.85	3.14
-45	22	25.88	-3.88
-25	37	40.17	-3.17
-10	69	52.02	16.97
0	52	59.62	-7.62

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

CAPSULE 38

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
15	69	69.59	-.59
25	71	74.98	-3.98
50	85	84.09	.9
100	90	90.65	-.65
150	92	91.91	.08
200	90	92.13	-2.13
250	97	92.17	4.82
		SUM of RESIDUALS =	1.97

CAPSULE 38

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:59:59 on 08-03-2000

Page 1

Coefficients of Curve 3

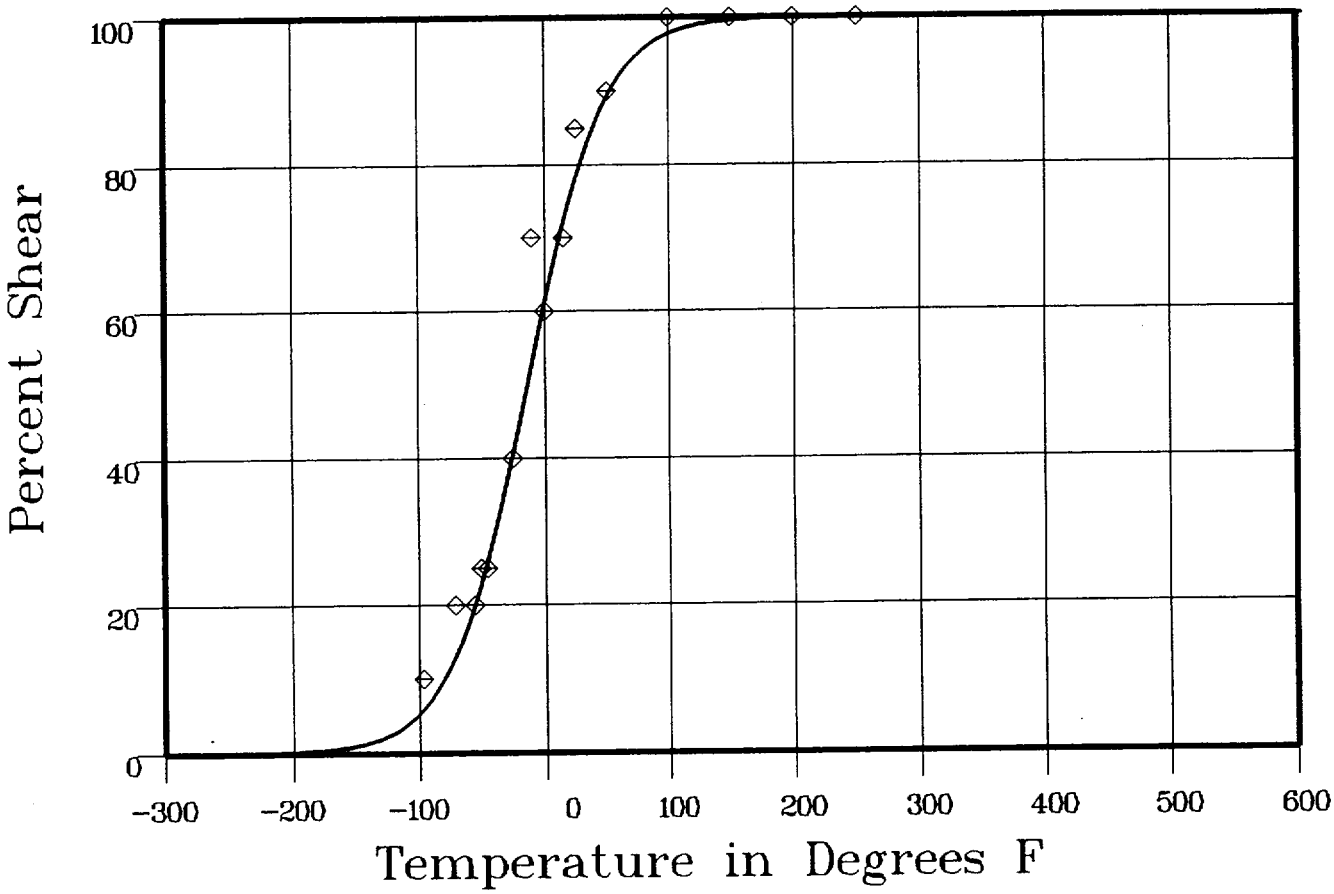
A = 50	B = 50	C = 59.75	T0 = -18.28

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

Temperature at 50% Shear: -18.2

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: WELD Ori.: Heat #: M-4311-1/M-4311-2

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-96	10	6.9	3.09
-70	20	15.04	4.95
-55	20	22.63	-2.63
-50	25	25.7	-.7
-45	25	29.02	-4.02
-25	40	44.4	-4.4
-10	70	56.88	13.11
0	60	64.83	-4.83

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

CAPSULE 38

Material: WELD Heat Number: M-4311-1/M-4311-2 Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
15	70	75.28	-5.28
25	85	80.97	4.02
50	90	90.76	-.76
100	100	98.12	1.87
150	100	99.64	.35
200	100	99.93	.06
250	100	99.98	.01
		SUM of RESIDUALS =	4.84

UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 11:18:01 on 08-03-2000

Page 1

Coefficients of Curve 1

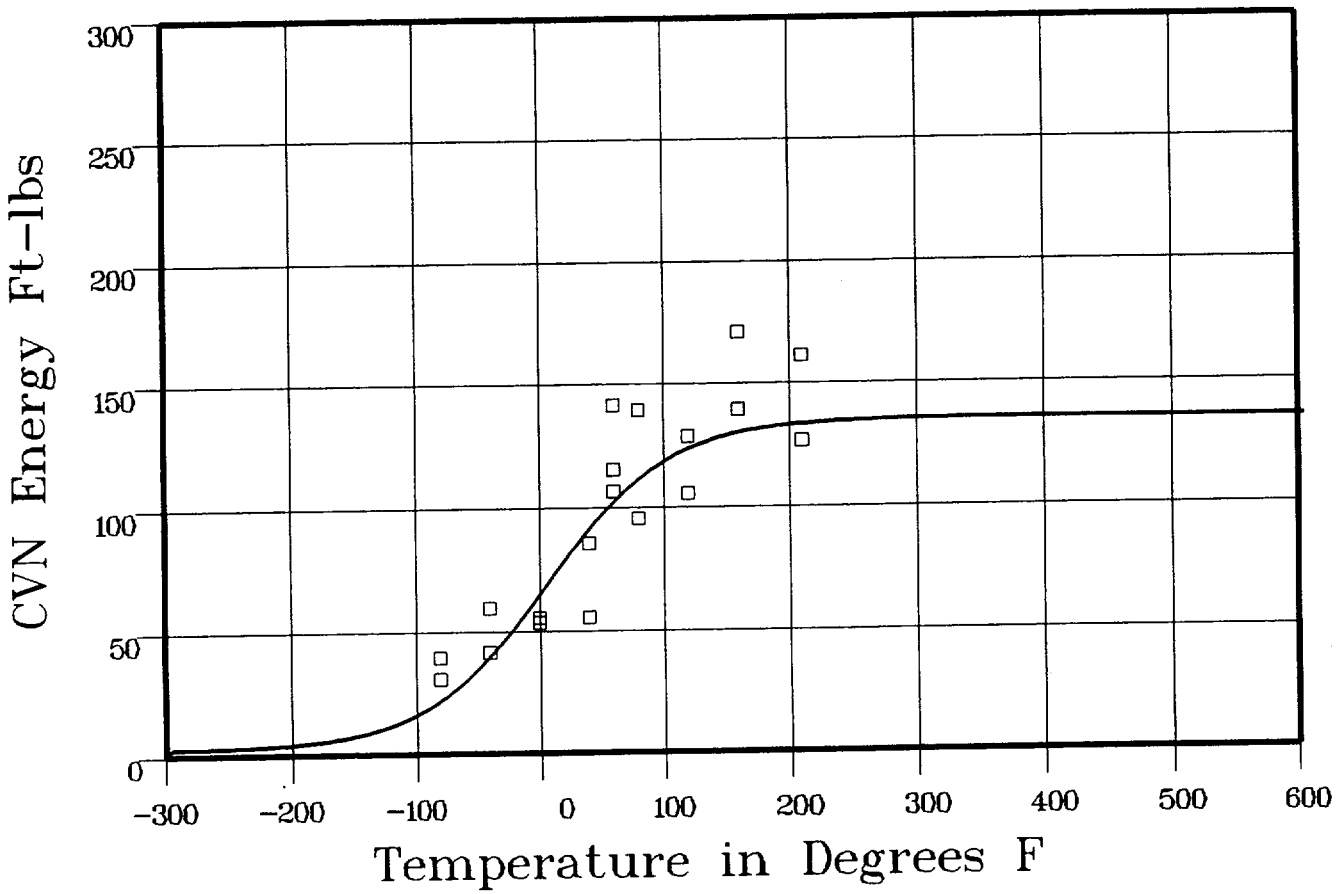
A = 68.59	B = 66.4	C = 97.23	T0 = 1.4

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

Upper Shelf Energy: 135 Fixed Temp. at 30 ft-lbs: -63.2 Temp. at 50 ft-lbs: -26.5 Lower Shelf Energy: 2.19 Fixed

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-80	39	23.16	15.83
-80	30	23.16	6.83
-40	41	41.91	-.91
-40	59	41.91	17.08
0	55	67.63	-12.63
0	53	67.63	-14.63
40	85	93.65	-8.65
40	55	93.65	-38.65
60	106	104.38	1.61

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

UNIRRADIATED

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
60	115	104.38	10.61
60	141	104.38	36.61
80	95	112.99	-17.99
80	139	112.99	26
120	128	124.34	3.65
120	105	124.34	-19.34
160	139	130.1	8.89
160	171	130.1	40.89
210	126	133.2	-7.2
210	161	133.2	27.79

SUM of RESIDUALS = 75.8

UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 11:21:56 on 08-03-2000

Page 1

Coefficients of Curve 1

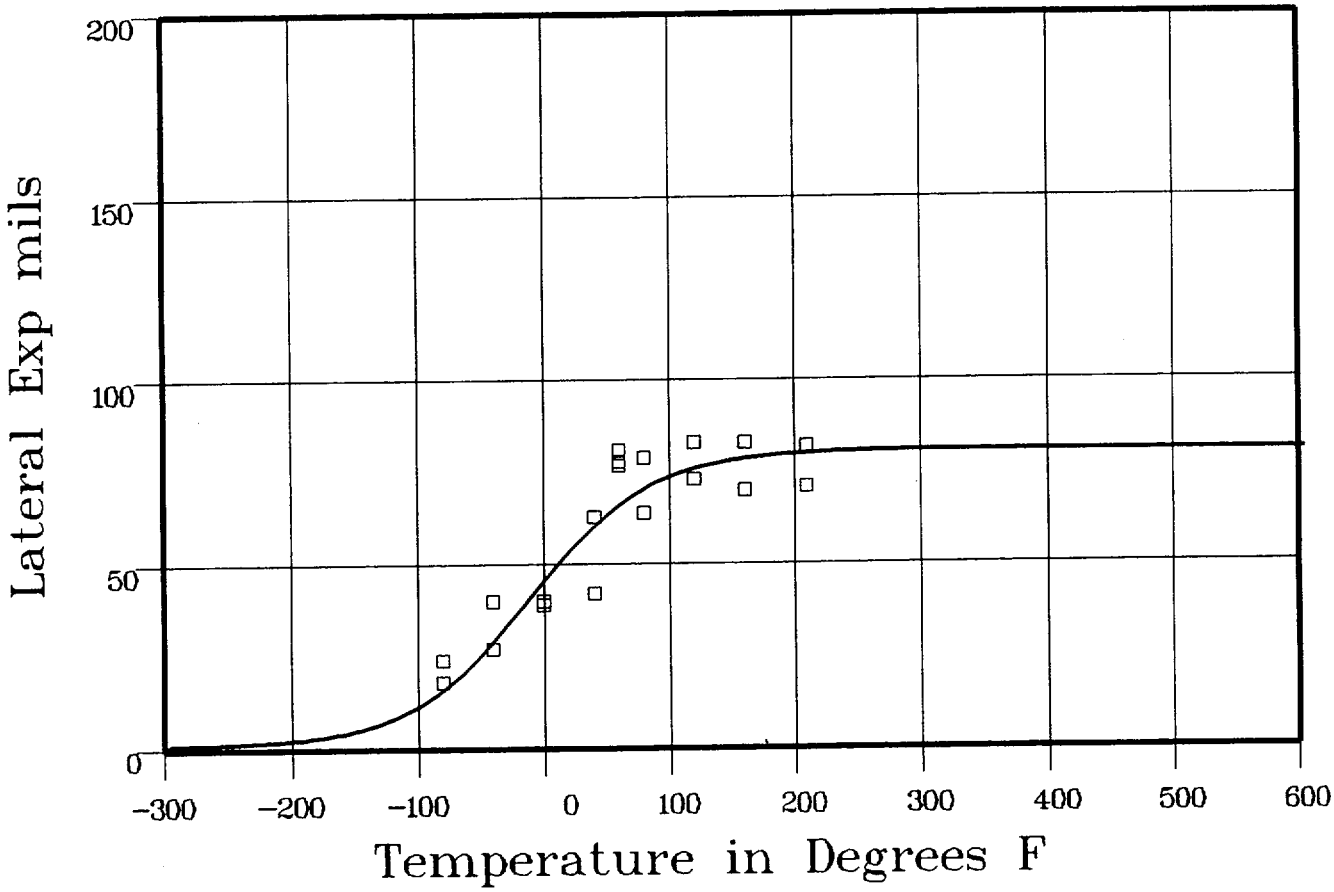
A = 40.96	B = 39.96	C = 94.95	T0 = -15.46

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

Upper Shelf L.E.: 80.92 Temperature at L.E. 35: -29.7 Lower Shelf L.E.: 1 Fixed

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-80	18	17.33	.66
-80	24	17.33	6.66
-40	27	30.86	-3.86
-40	40	30.86	9.13
0	39	47.41	-8.41
0	40	47.41	-7.41
40	63	61.97	1.02
40	42	61.97	-19.97
60	77	67.38	9.61

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

UNIRRADIATED

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
60	81	67.38	13.61
60	78	67.38	10.61
80	64	71.49	-7.49
80	79	71.49	7.5
120	73	76.57	-3.57
120	83	76.57	6.42
160	70	78.99	-8.99
160	83	78.99	4
210	82	80.24	1.75
210	71	80.24	-9.24

SUM of RESIDUALS = 2.08

UNIRRADIATED

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 11:29:47 on 08-03-2000

Page 1

Coefficients of Curve 1

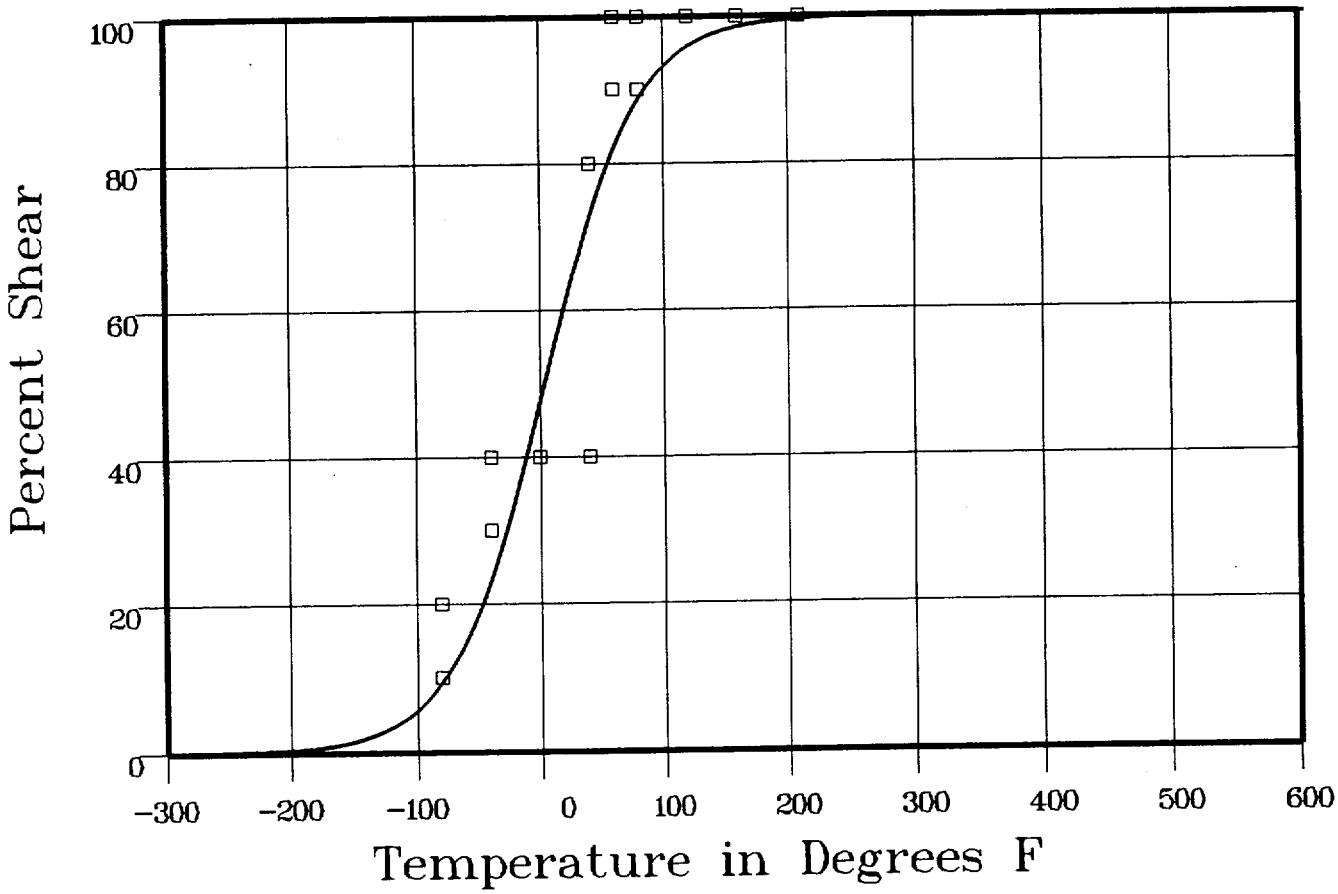
A = 50	B = 50	C = 73.95	T0 = -.93

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

Temperature at 50% Shear: -.9

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-80	10	10.54	-.54
-80	20	10.54	9.45
-40	30	25.79	4.2
-40	40	25.79	14.2
0	40	50.63	-10.63
0	40	50.63	-10.63
40	80	75.15	4.84
40	40	75.15	-35.15
60	100	83.86	16.13

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

UNIRRADIATED

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
60	90	83.86	6.13
60	100	83.86	16.13
80	90	89.92	.07
80	100	89.92	10.07
120	100	96.34	3.65
120	100	96.34	3.65
160	100	98.72	1.27
160	100	98.72	1.27
210	100	99.66	.33
210	100	99.66	.33

SUM of RESIDUALS = 34.81

CAPSULE 137

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:48:11 on 08-03-2000

Page 1

Coefficients of Curve 2

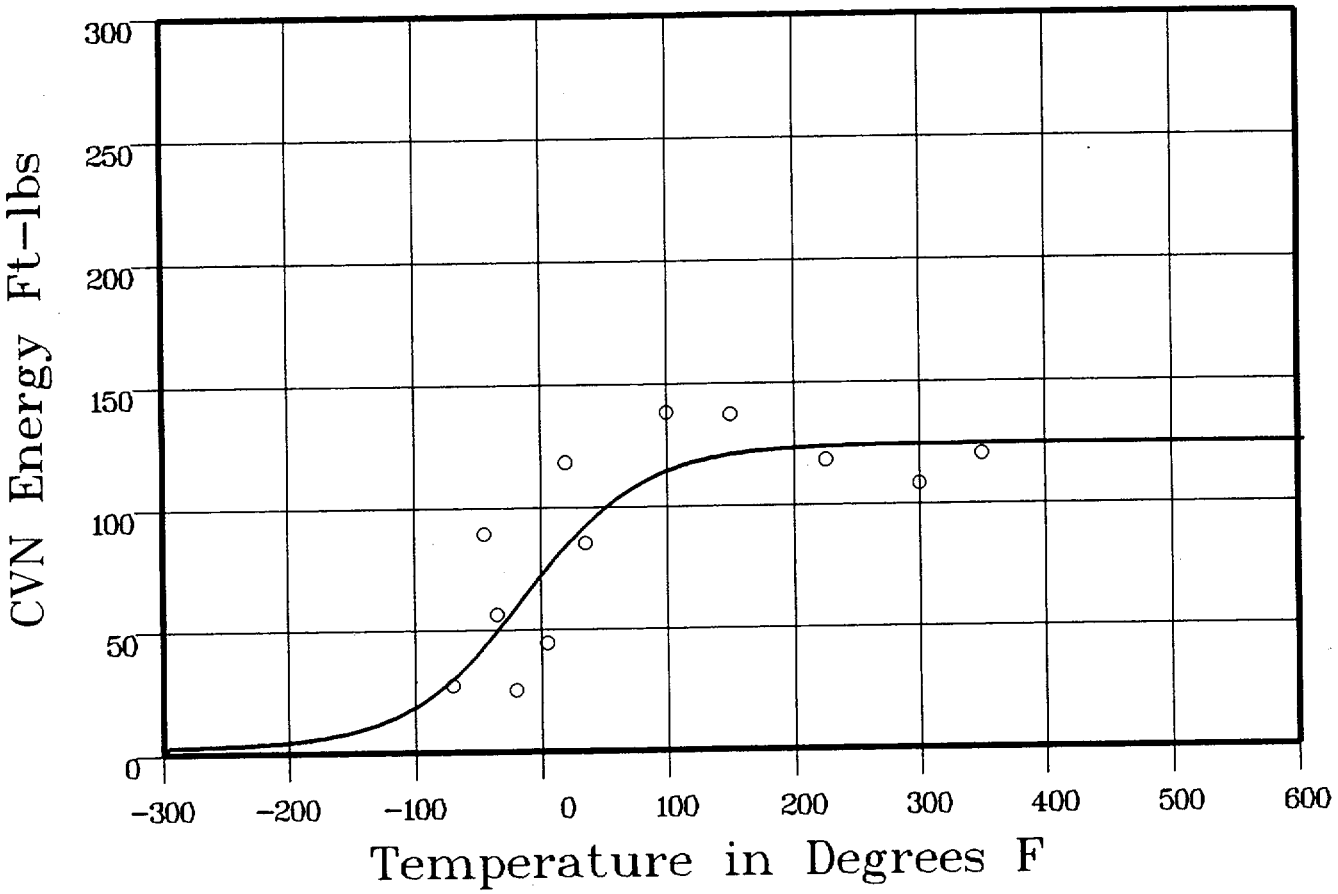
A = 63.09	B = 60.9	C = 92.86	T0 = -17.61

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

Upper Shelf Energy: 124 Fixed Temp. at 30 ft-lbs: -74.1 Temp. at 50 ft-lbs: -37.9 Lower Shelf Energy: 2.19 Fixed

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-70	27	31.98	-4.98
-45	89	45.64	43.35
-35	56	51.83	4.16
-20	25	61.53	-36.53
5	44	77.64	-33.64
20	118	86.5	31.49
35	85	94.33	-9.33
100	138	115.03	22.96

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

CAPSULE 137

Material: HEAT AFFD ZONE Heat Number: Orientation:

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
150	137	120.79	16.2
225	118	123.34	-5.34
300	108	123.86	-15.86
350	120	123.95	-3.95

SUM of RESIDUALS = 8.53

CAPSULE 137

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 11:21:56 on 08-03-2000

Page 1

Coefficients of Curve 2

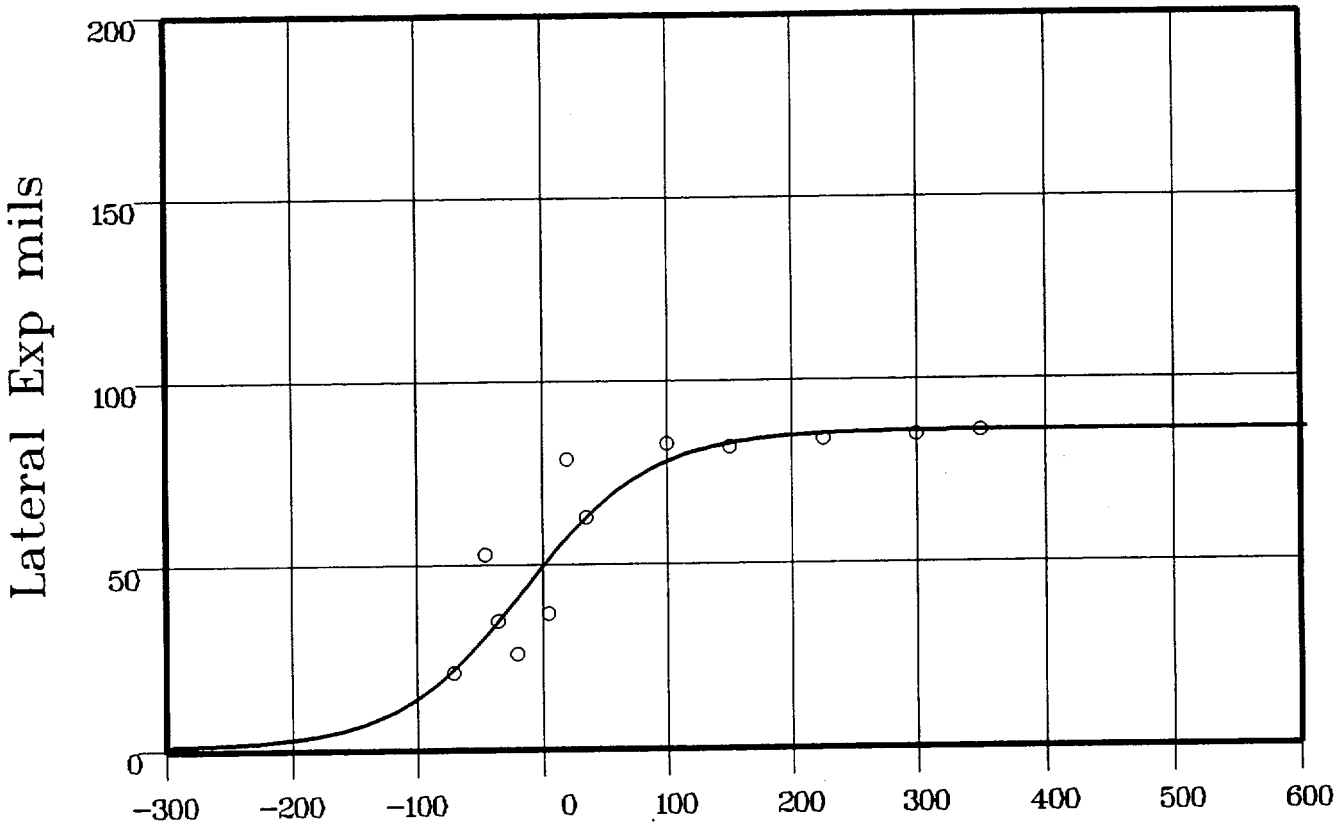
A = 43.54	B = 42.54	C = 99.7	T0 = -18.75

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

Upper Shelf L.E.: 86.08 Temperature at L.E. 35: -39 Lower Shelf L.E.: 1 Fixed

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-70	21	23.41	-2.41
-45	53	32.59	20.4
-35	35	36.67	-1.67
-20	26	43.01	-17.01
5	37	53.49	-16.49
20	79	59.29	19.7
35	63	64.48	-1.48
100	83	78.89	4.1

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

CAPSULE 137

Material: HEAT AFFD ZONE Heat Number: Orientation:

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
150	82	83.29	-1.29
225	84	85.45	-1.45
300	85	85.94	-.94
350	86	86.03	-.03

SUM of RESIDUALS = 1.41

CAPSULE 137

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 11:29:47 on 08-03-2000

Page 1

Coefficients of Curve 2

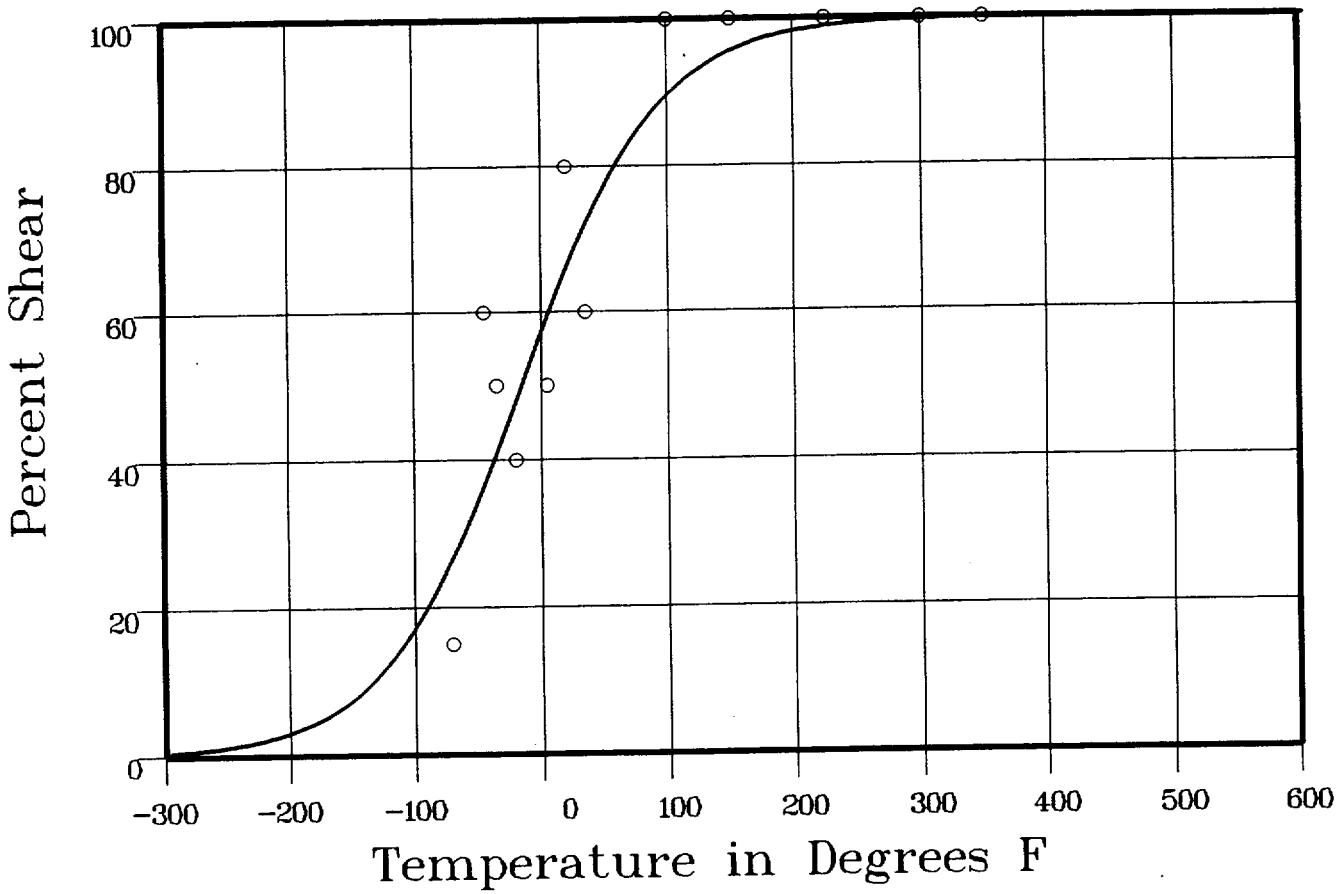
A = 50	B = 50	C = 107.14	T0 = -19.99

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

Temperature at 50% Shear: -19.9

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-70	15	28.22	-13.22
-45	60	38.53	21.46
-35	50	43.04	6.95
-20	40	49.99	-9.99
5	50	61.45	-11.45
20	80	67.84	12.15
35	60	73.62	-13.62
100	100	90.37	9.62

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

CAPSULE 137

Material: HEAT AFFD ZONE Heat Number: Orientation:

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
150	100	95.98	4.01
225	100	98.97	1.02
300	100	99.74	.25
350	100	99.89	.1

SUM of RESIDUALS = 7.29

CAPSULE 38

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:48:11 on 08-03-2000

Page 1

Coefficients of Curve 3

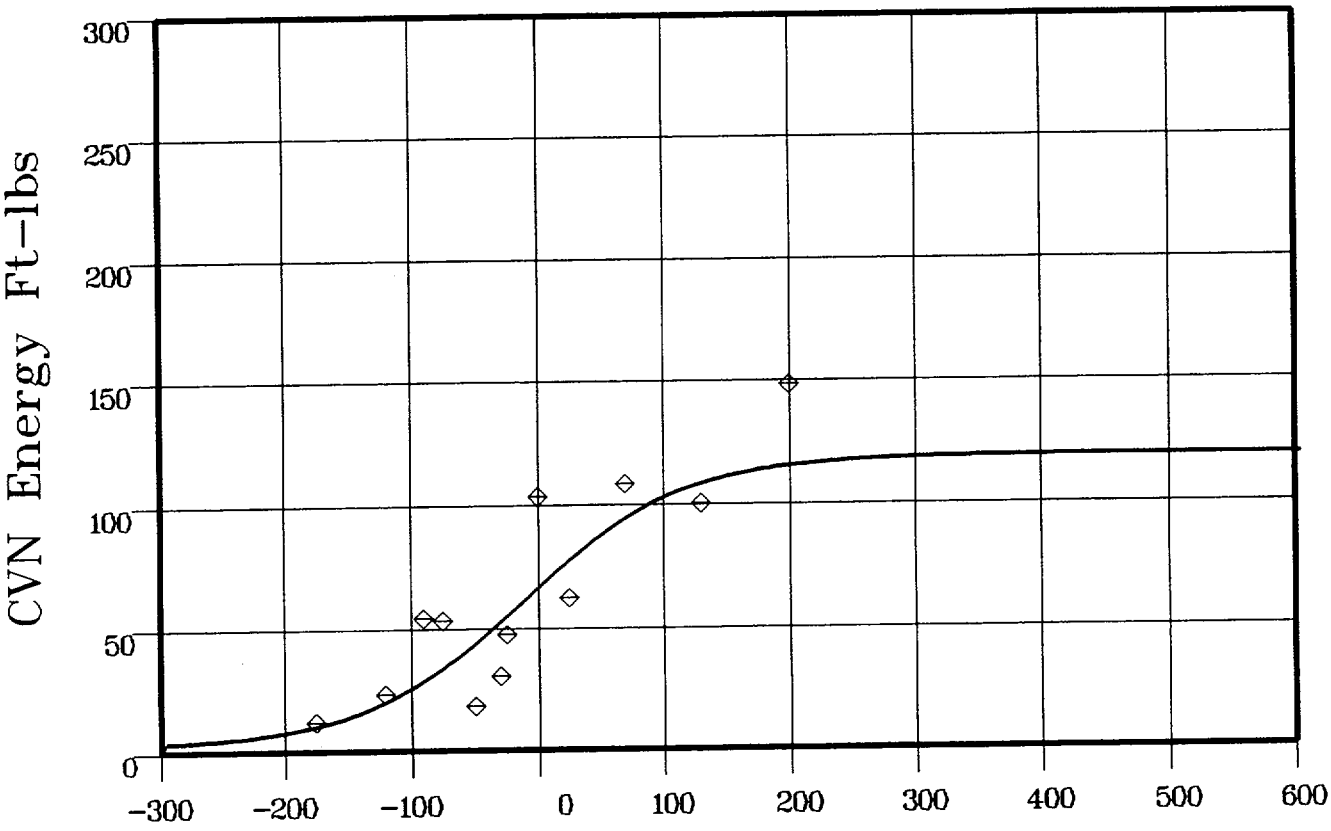
A = 60.59	B = 58.4	C = 125	T0 = -17.07

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

Upper Shelf Energy: 119 Fixed Temp. at 30 ft-lbs: -89.7 Temp. at 50 ft-lbs: -40 Lower Shelf Energy: 2.19 Fixed

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-175	12	10.84	1.15
-120	23	21.06	1.93
-90	54	29.93	24.06
-75	53	35.32	17.67
-50	18	45.56	-27.56
-30	30	54.58	-24.58
-25	47	56.9	-9.9
0	103	68.52	34.47

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

CAPSULE 38

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
25	62	79.54	-17.54
70	108	95.76	12.23
130	100	108.85	-8.85
200	148	115.48	32.51

SUM of RESIDUALS = 35.6

CAPSULE 38

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 11:21:56 on 08-03-2000

Page 1

Coefficients of Curve 3

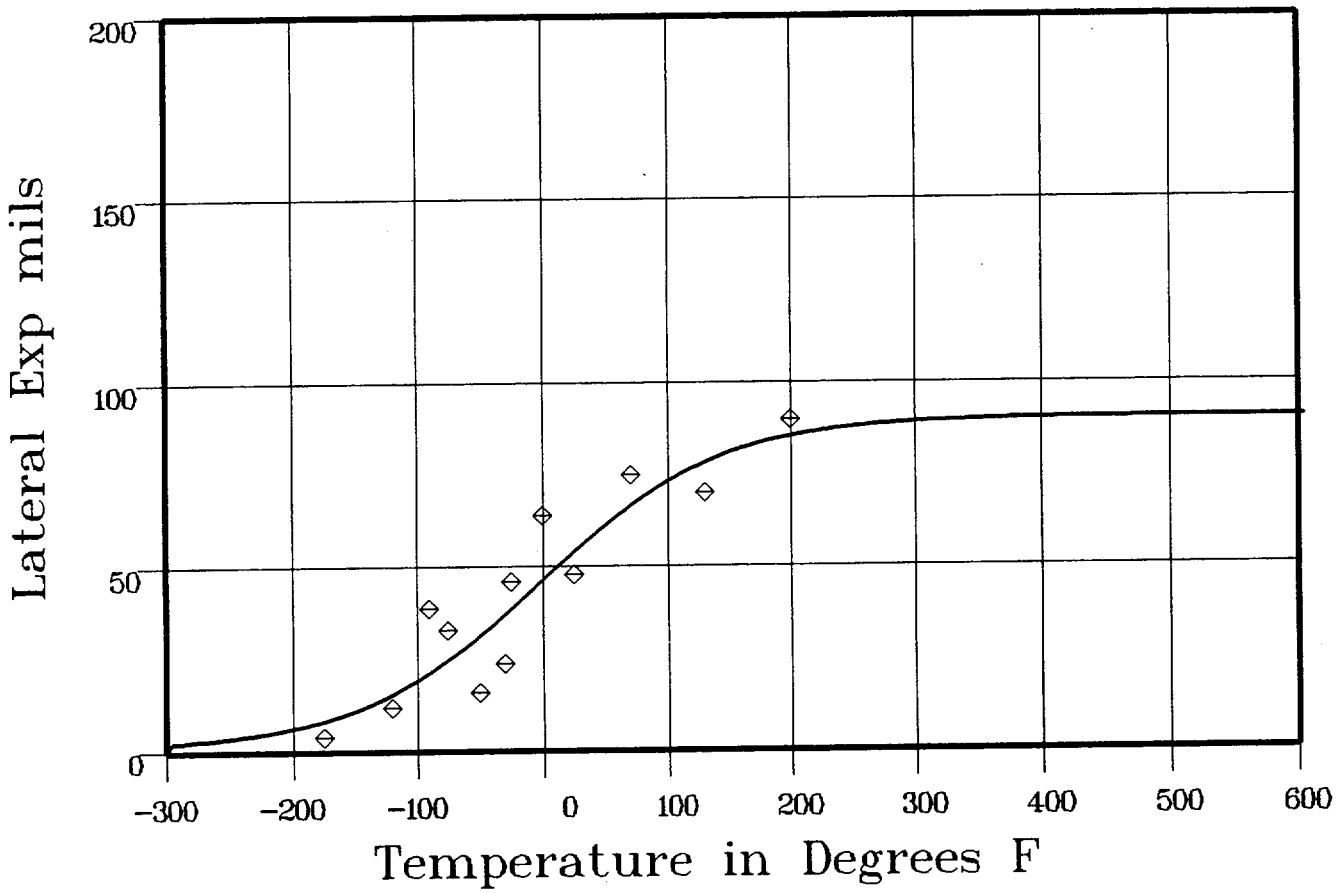
A = 45.75	B = 44.75	C = 143.69	T0 = -7.5

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

Upper Shelf L.E.: 90.51 Temperature at L.E. 35: -42.7 Lower Shelf L.E.: 1 Fixed

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-175	4	8.92	-4.92
-120	12	16.46	-4.46
-90	39	22.55	16.44
-75	33	26.15	6.84
-50	16	32.89	-16.89
-30	24	38.8	-14.8
-25	46	40.33	5.66
0	64	48.08	15.91

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

CAPSULE 38

Material: HEAT AFFD ZONE Heat Number: Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
25	48	55.7	-7.7
70	75	67.79	7.2
130	70	79	-9
200	90	85.78	4.21

SUM of RESIDUALS = -1.51

CAPSULE 38

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 11:29:47 on 08-03-2000

Page 1

Coefficients of Curve 3

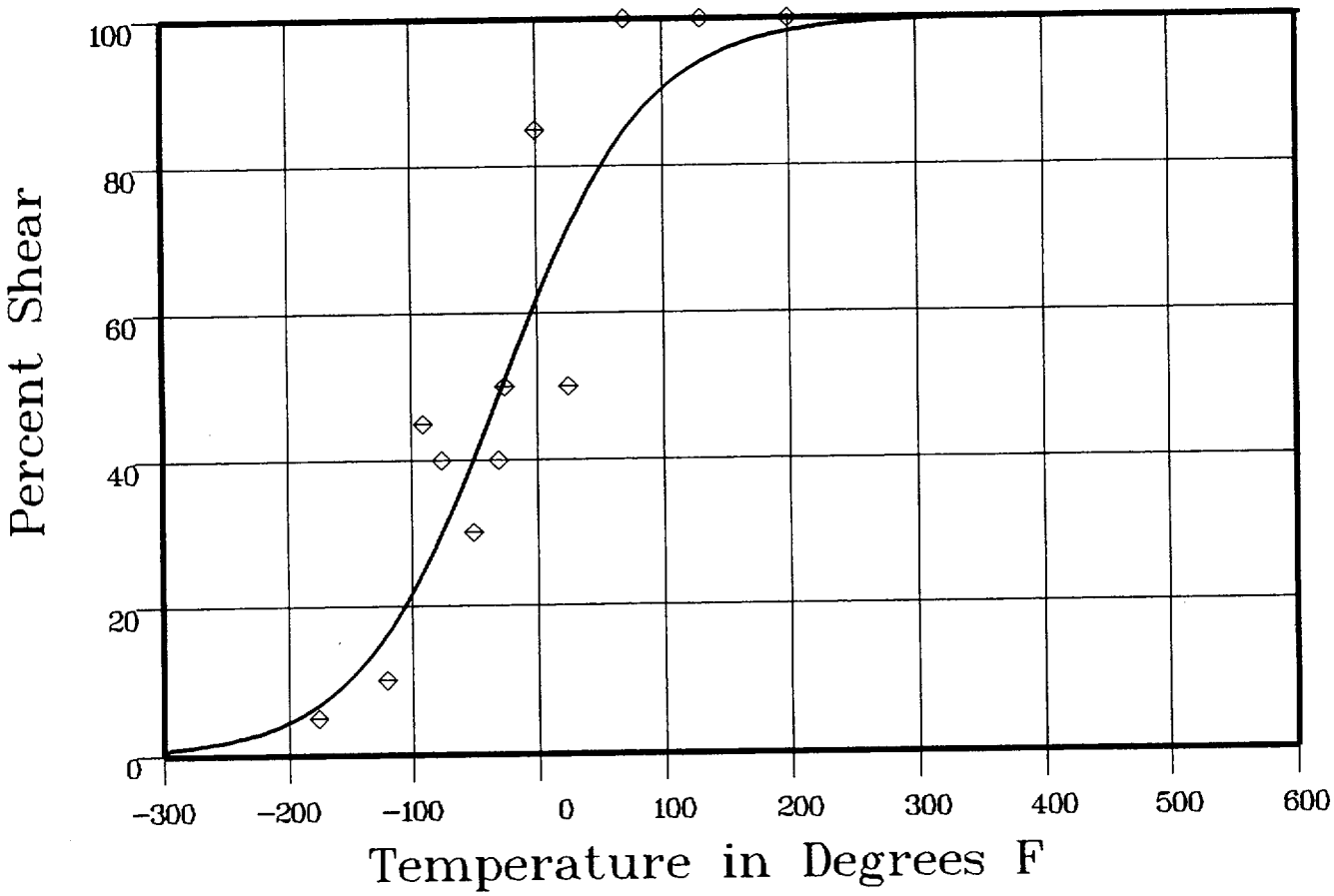
A = 50	B = 50	C = 112.84	T0 = -31.87

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

Temperature at 50% Shear: -31.8

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: HEAT AFF'D ZONE Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-175	5	7.33	-2.33
-120	10	17.33	-7.33
-90	45	26.3	18.69
-75	40	31.77	8.22
-50	30	42.03	-12.03
-30	40	50.83	-10.83
-25	50	53.04	-3.04
0	85	63.75	21.24

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

CAPSULE 38

Material: HEAT AFF'D ZONE Heat Number: Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
25	50	73.26	-23.26
70	100	85.88	14.11
130	100	94.62	5.37
200	100	98.38	1.61

SUM of RESIDUALS = 10.42

UNIRRADIATED STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:46:27 on 08-04-2000

Page 1

Coefficients of Curve 1

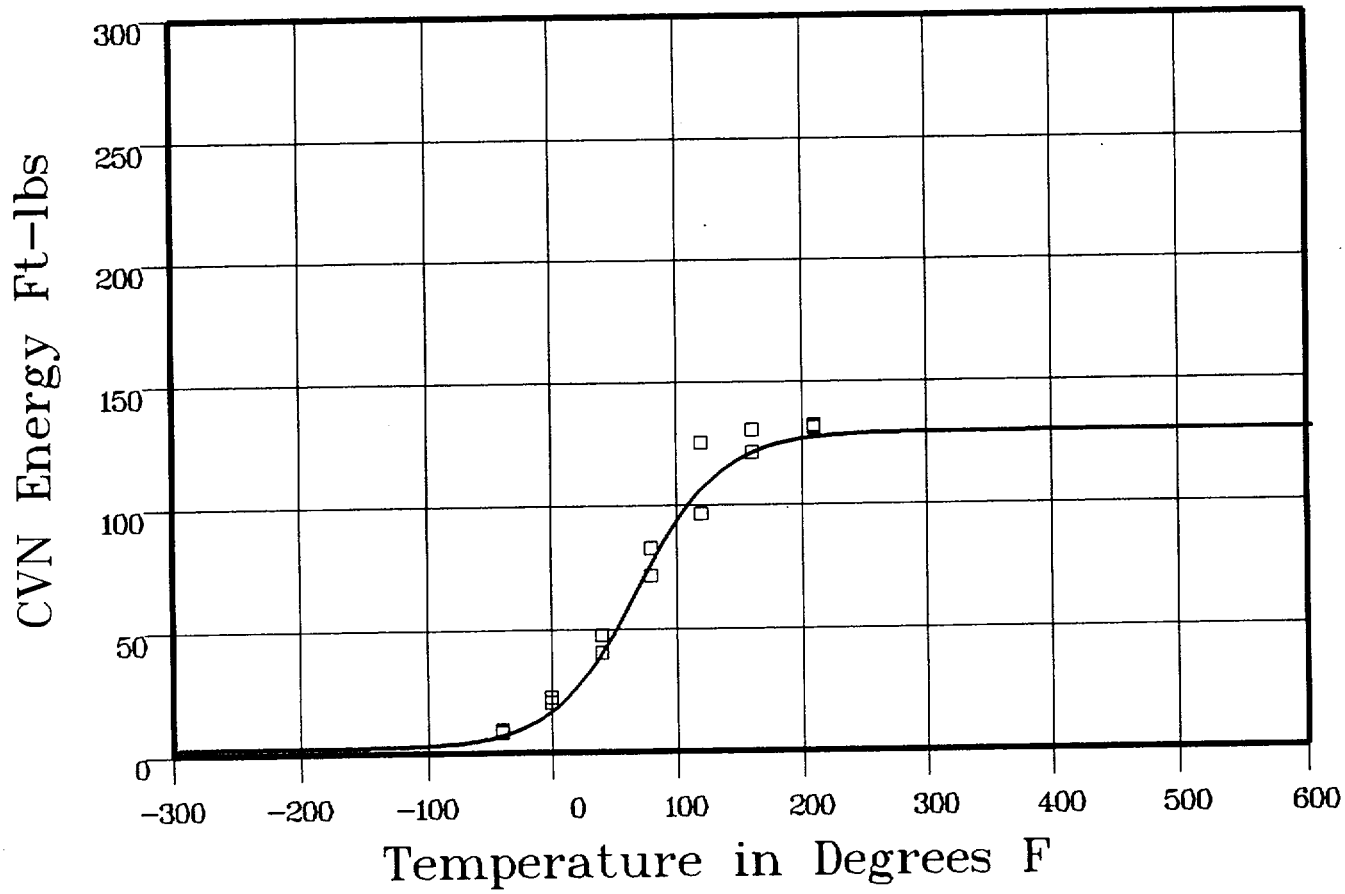
A = 65.59	B = 63.4	C = 66.96	T0 = 64.43

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

Upper Shelf Energy: 129 Fixed Temp. at 30 ft-lbs: 21.9 Temp. at 50 ft-lbs: 47.6 Lower Shelf Energy: 2.19 Fixed

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: SRM SA533B1 Ori.: LT Heat #:

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
-40	9	7.56	1.43
-40	8	7.56	.43
0	20	18.35	1.64
0	22	18.35	3.64
40	47	43.44	3.55
40	40	43.44	-3.44
80	82	80.07	1.92
80	71	80.07	-9.07
120	125	108.73	16.26

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

UNIRRADIATED STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
120	96	108.73	-12.73
160	121	122.09	-1.09
160	130	122.09	7.9
210	131	127.38	3.61
210	132	127.38	4.61

SUM of RESIDUALS = 18.71

UNIRRADIATED STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:49:12 on 08-04-2000

Page 1

Coefficients of Curve 1

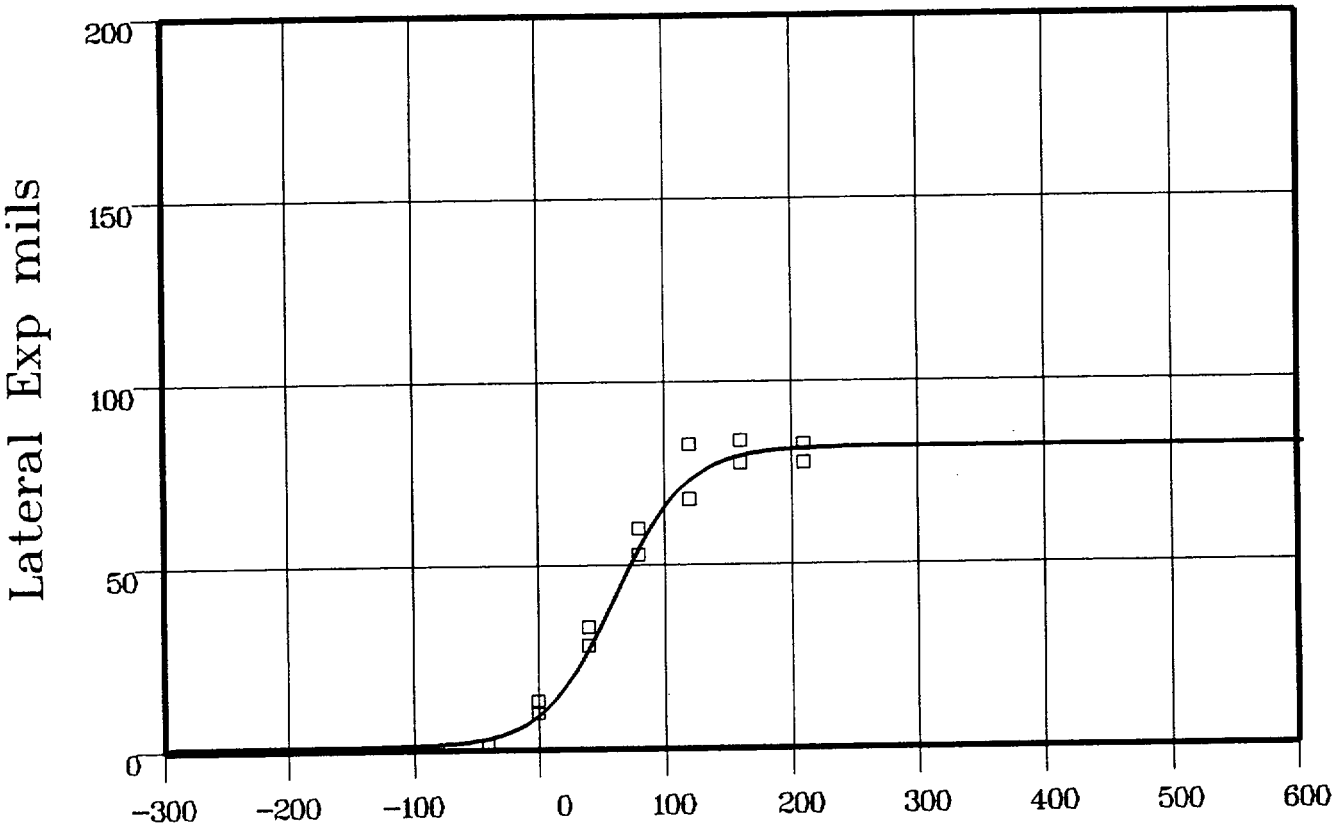
A = 41.63	B = 40.63	C = 56.01	T0 = 56.71

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

Upper Shelf L.E.: 82.27 Temperature at L.E. 35: 47.4 Lower Shelf L.E.: 1 Fixed

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: SRM SA533B1 Ori.: LT Heat #:

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
-40	1	3.49	-2.49
-40	1	3.49	-2.49
0	10	10.47	-.47
0	13	10.47	2.52
40	33	29.85	3.14
40	28	29.85	-1.85
80	60	57.61	2.38
80	53	57.61	-4.61
120	83	74.58	8.41

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

UNIRRADIATED STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
120	68	74.58	-6.58
160	78	80.28	-2.28
160	84	80.28	3.71
210	83	81.93	1.06
210	78	81.93	-3.93

SUM of RESIDUALS = -3.48

UNIRRADIATED STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:51:56 on 08-04-2000

Page 1

Coefficients of Curve 1

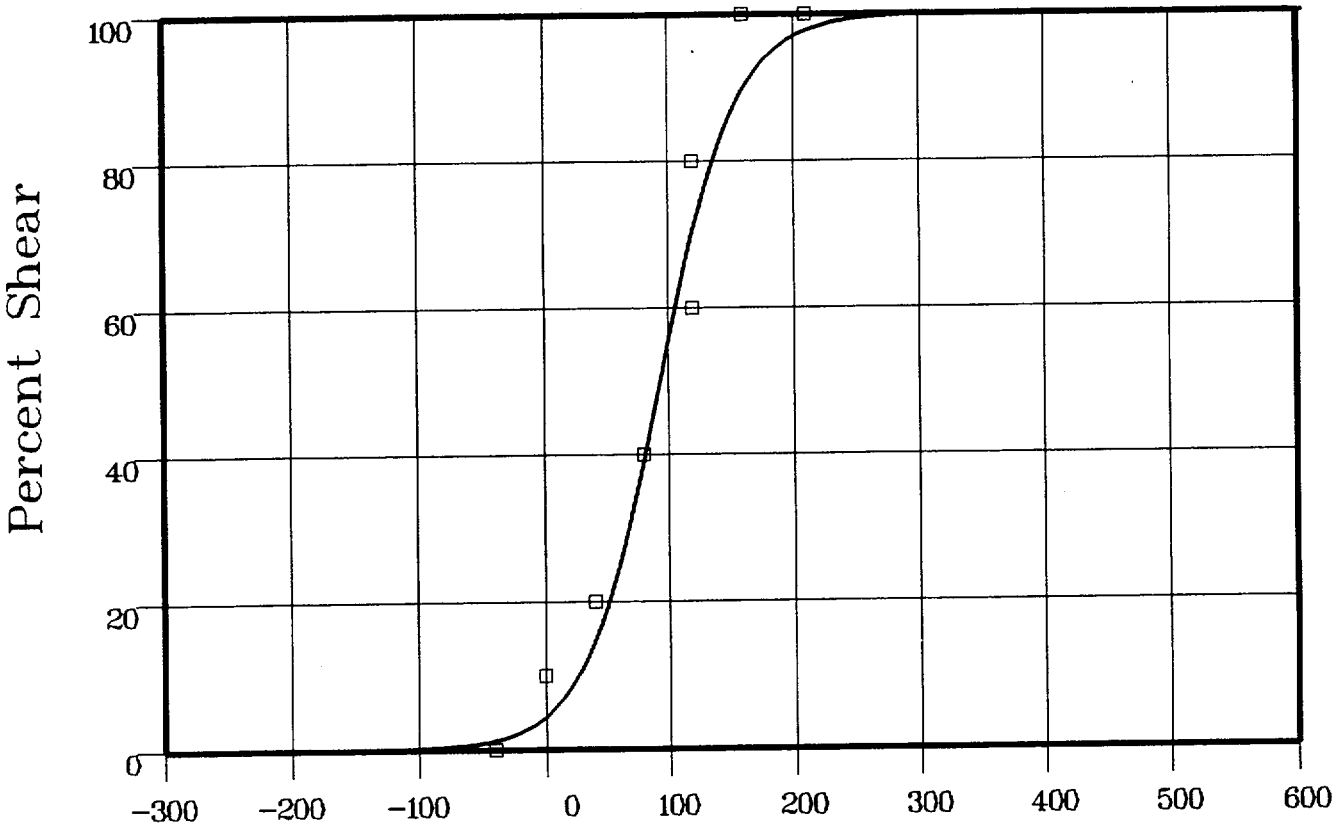
A = 50	B = 50	C = 61.02	T0 = 88.76

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

Temperature at 50% Shear: 88.7

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: UNIRR Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: UNIRR Material: SRM SA533B1 Ori.: LT Heat #:

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
-40	0	1.44	-1.44
-40	0	1.44	-1.44
0	10	5.16	4.83
0	10	5.16	4.83
40	20	16.82	3.17
40	20	16.82	3.17
80	40	42.86	-2.86
80	40	42.86	-2.86
120	80	73.56	6.43

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

UNIRRADIATED STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: UNIRR Total Fluence:

Charpy V–Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
120	60	73.56	-13.56
160	100	91.16	8.83
160	100	91.16	8.83
210	100	98.15	1.84
210	100	98.15	1.84

SUM of RESIDUALS = 21.61

CAPSULE 137 STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:46:27 on 08-04-2000

Page 1

Coefficients of Curve 2

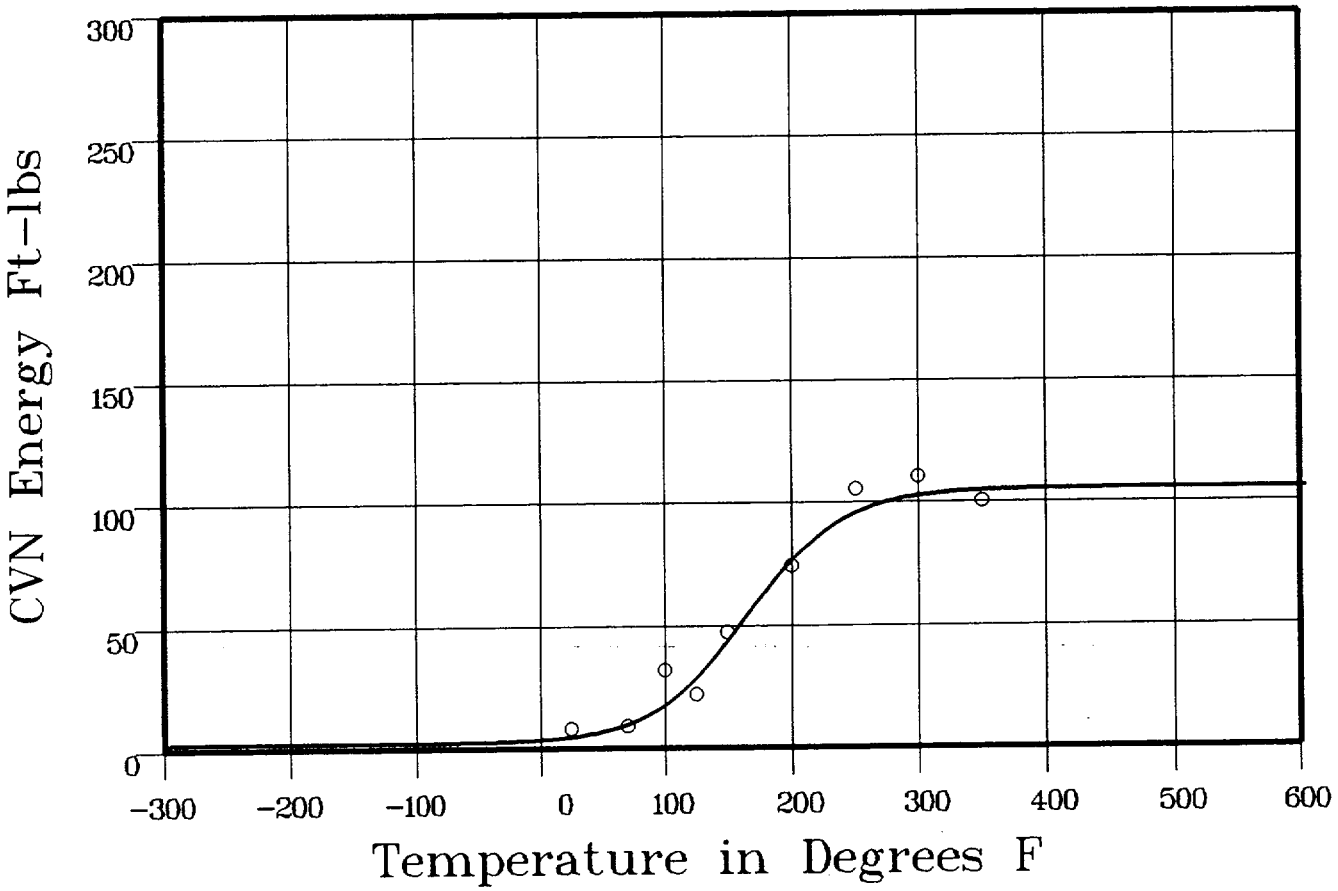
A = 53.59	B = 51.4	C = 75.06	T0 = 160.54

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

Upper Shelf Energy: 105 Fixed Temp. at 30 ft-lbs: 123.2 Temp. at 50 ft-lbs: 155.2 Lower Shelf Energy: 2.19 Fixed

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: SRM SA533B1 Ori.: LT Heat #:

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
25	8	4.9	3.09
70	9	10.65	-1.65
100	32	19.27	12.72
125	22	30.92	-8.92
150	47	46.42	.57
200	74	78.37	-4.37
250	105	96.31	8.68
300	110	102.55	7.44

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

CAPSULE 137 STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
350	100	104.34	-4.34

SUM of RESIDUALS = 13.21

CAPSULE 137 STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:49:12 on 08-04-2000

Page 1

Coefficients of Curve 2

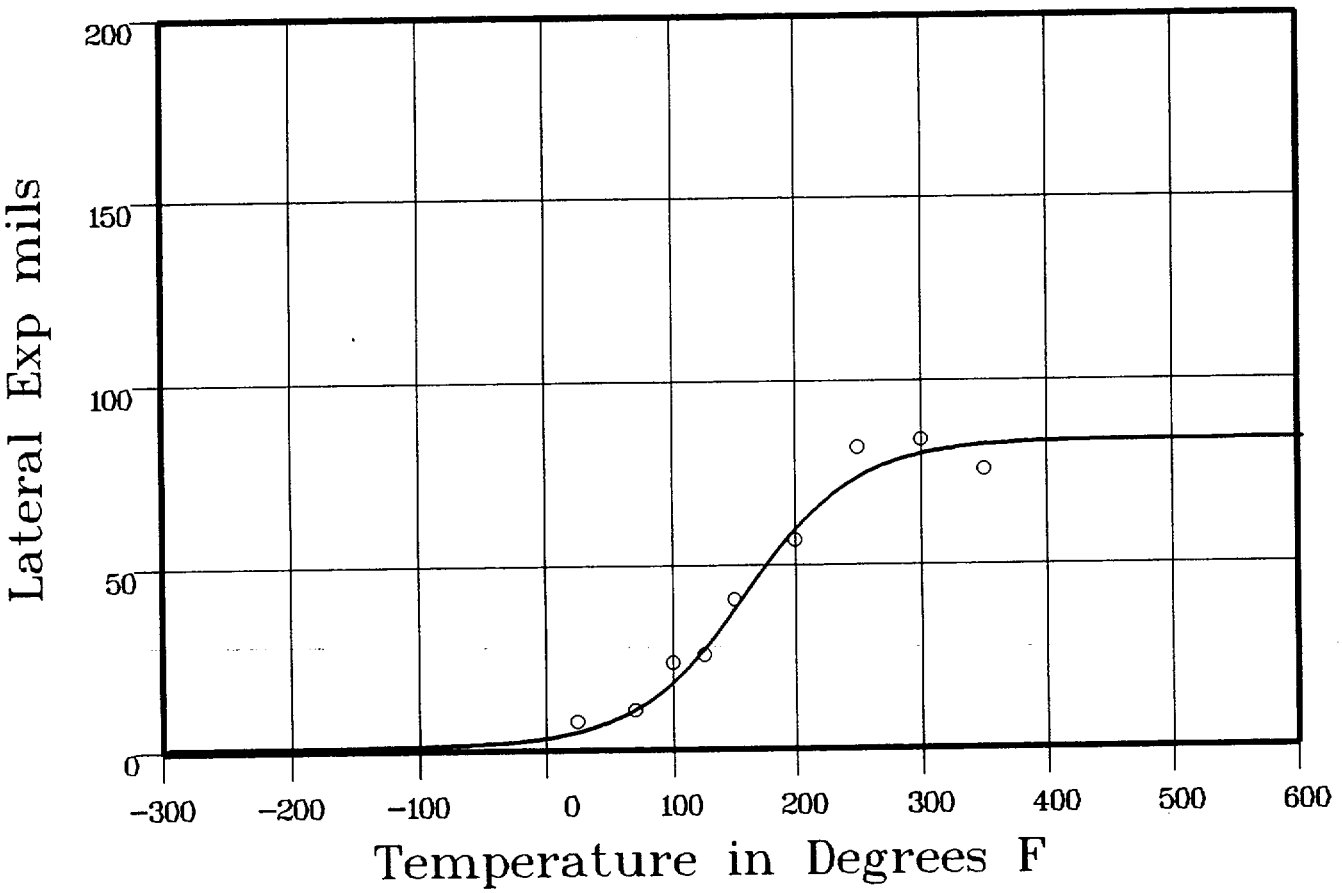
A = 42.32	B = 41.32	C = 90.15	T0 = 154.68

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

Upper Shelf L.E.: 83.64 Temperature at L.E. 35: 138.5 Lower Shelf L.E.: 1 Fixed

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: SRM SA533B1 Ori.: LT Heat #:

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
25	8	5.4	2.59
70	11	11.95	-.95
100	24	19.93	4.06
125	26	29.18	-3.18
150	41	40.17	.82
200	57	61.5	-4.5
250	82	74.74	7.25
300	84	80.47	3.52

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

CAPSULE 137 STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
350	76	82.57	-6.57

SUM of RESIDUALS = 3.04

CAPSULE 137 STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:51:56 on 08-04-2000

Page 1

Coefficients of Curve 2

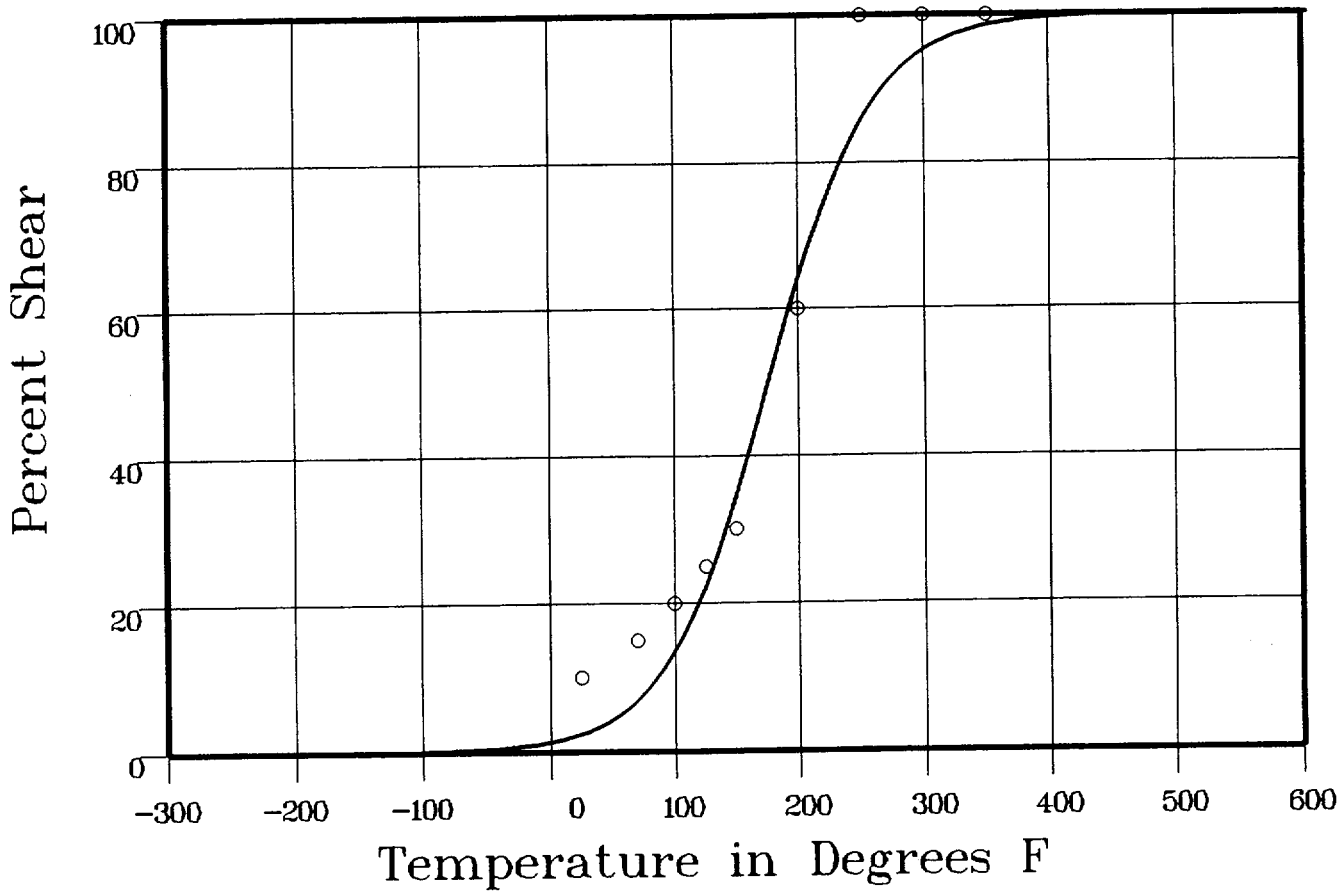
A = 50	B = 50	C = 82.17	T0 = 171.67

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

Temperature at 50% Shear: 171.6

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: 137 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 137 Material: SRM SA533B1 Ori.: LT Heat #:

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
25	10	2.73	7.26
70	15	7.76	7.23
100	20	14.87	5.12
125	25	24.3	.69
150	30	37.1	-7.1
200	60	66.58	-6.58
250	100	87.05	12.94
300	100	95.78	4.21

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

CAPSULE 137 STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation: LT

Capsule: 137 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
350	100	98.71	1.28

SUM of RESIDUALS = 25.07

CAPSULE 38 STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:46:27 on 08-04-2000

Page 1

Coefficients of Curve 3

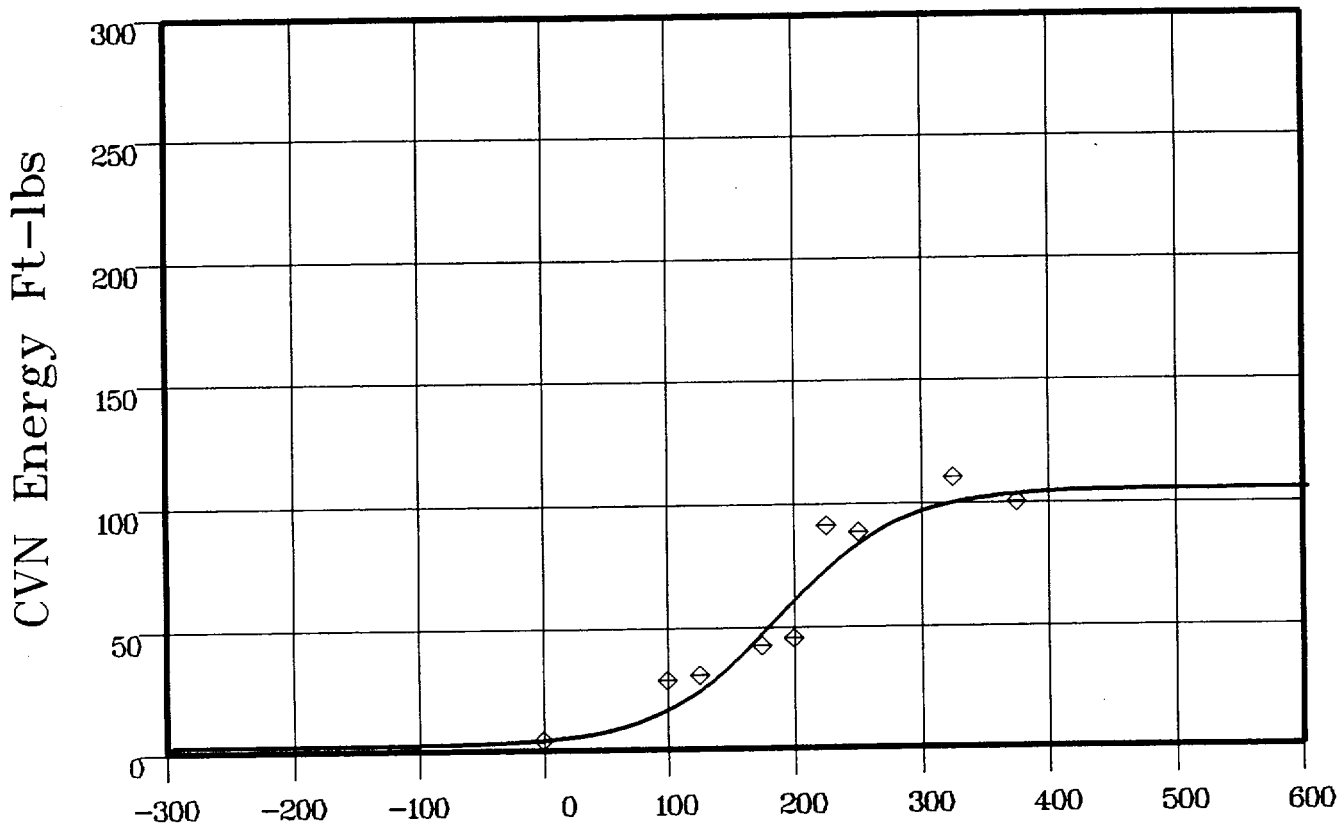
A = 53.59	B = 51.4	C = 94.01	T0 = 182.81

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

Upper Shelf Energy: 105 Fixed Temp. at 30 ft-lbs: 136.1 Temp. at 50 ft-lbs: 176.2 Lower Shelf Energy: 2.19 Fixed

Material: SRM SA533B1 Heat Number: Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: SRM SA533B1 Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input CVN Energy	Computed CVN Energy	Differential
0	4	4.26	-.26
100	28	17.26	10.73
125	30	25.45	4.54
175	42	49.33	-7.33
200	45	62.89	-17.89
225	91	75.23	15.76
250	88	85.13	2.86
325	110	100.23	9.76

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

CAPSULE 38 STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input CVN Energy	Computed CVN Energy	Differential
375	100	103.3	-3.3

SUM of RESIDUALS = 14.87

CAPSULE 38 STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:49:12 on 08-04-2000

Page 1

Coefficients of Curve 3

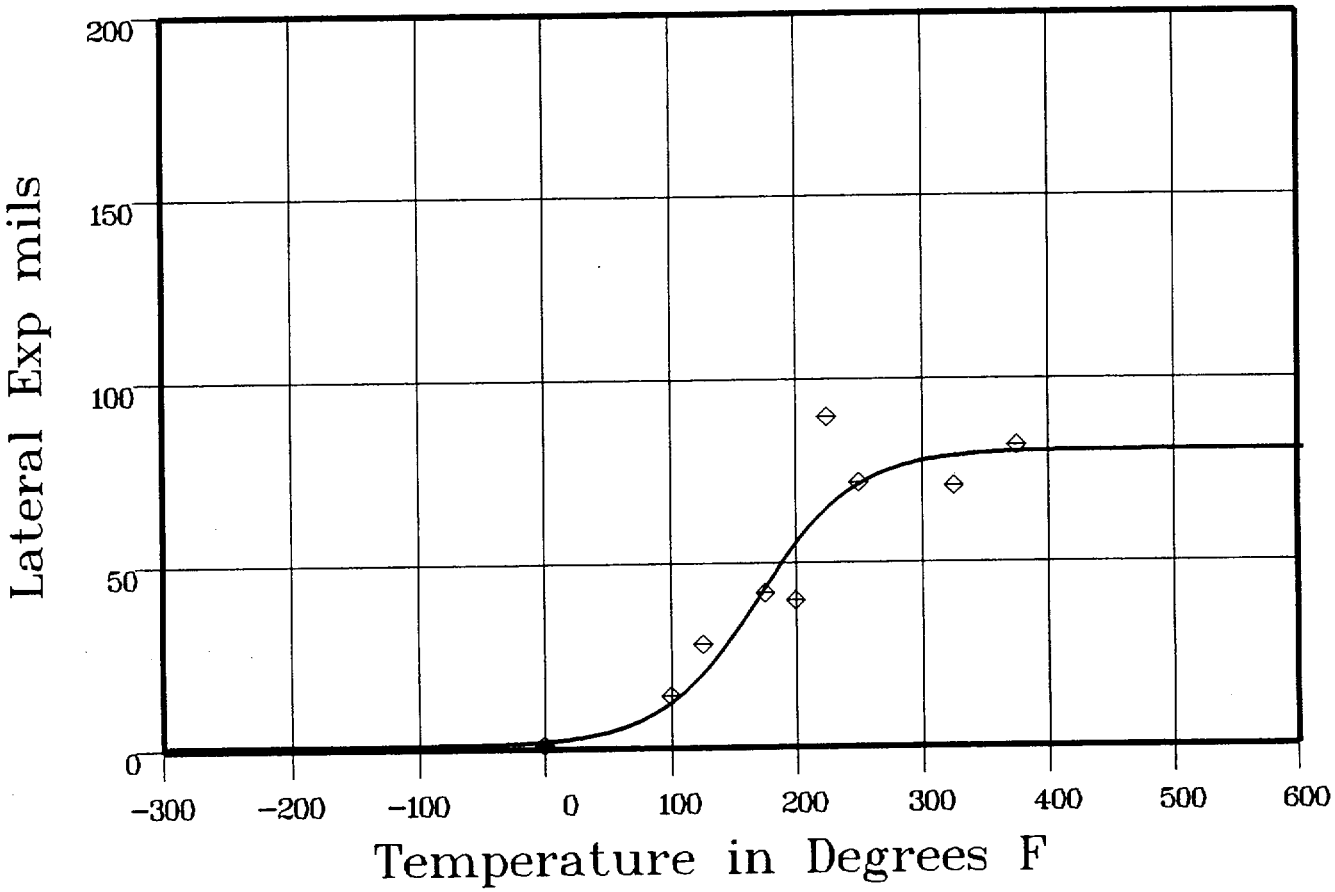
A = 40.81	B = 39.81	C = 77.47	T0 = 165.23

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

Upper Shelf L.E.: 80.62 Temperature at L.E. 35: 153.8 Lower Shelf L.E.: 1 Fixed

Material: SRM SA533B1 Heat Number: Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: SRM SA533B1 Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input Lateral Expansion	Computed L.E.	Differential
0	1	2.1	-1.1
100	14	13.46	.53
125	28	21.81	6.18
175	42	45.8	-3.8
200	40	57.56	-17.56
225	90	66.59	23.4
250	72	72.59	-.59
325	71	79.35	-8.35

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

CAPSULE 38 STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Lateral Expansion	Computed L.E.	Differential
375	82	80.26	1.73

SUM of RESIDUALS = .43

CAPSULE 38 STANDARD REFERENCE MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:51:56 on 08-04-2000

Page 1

Coefficients of Curve 3

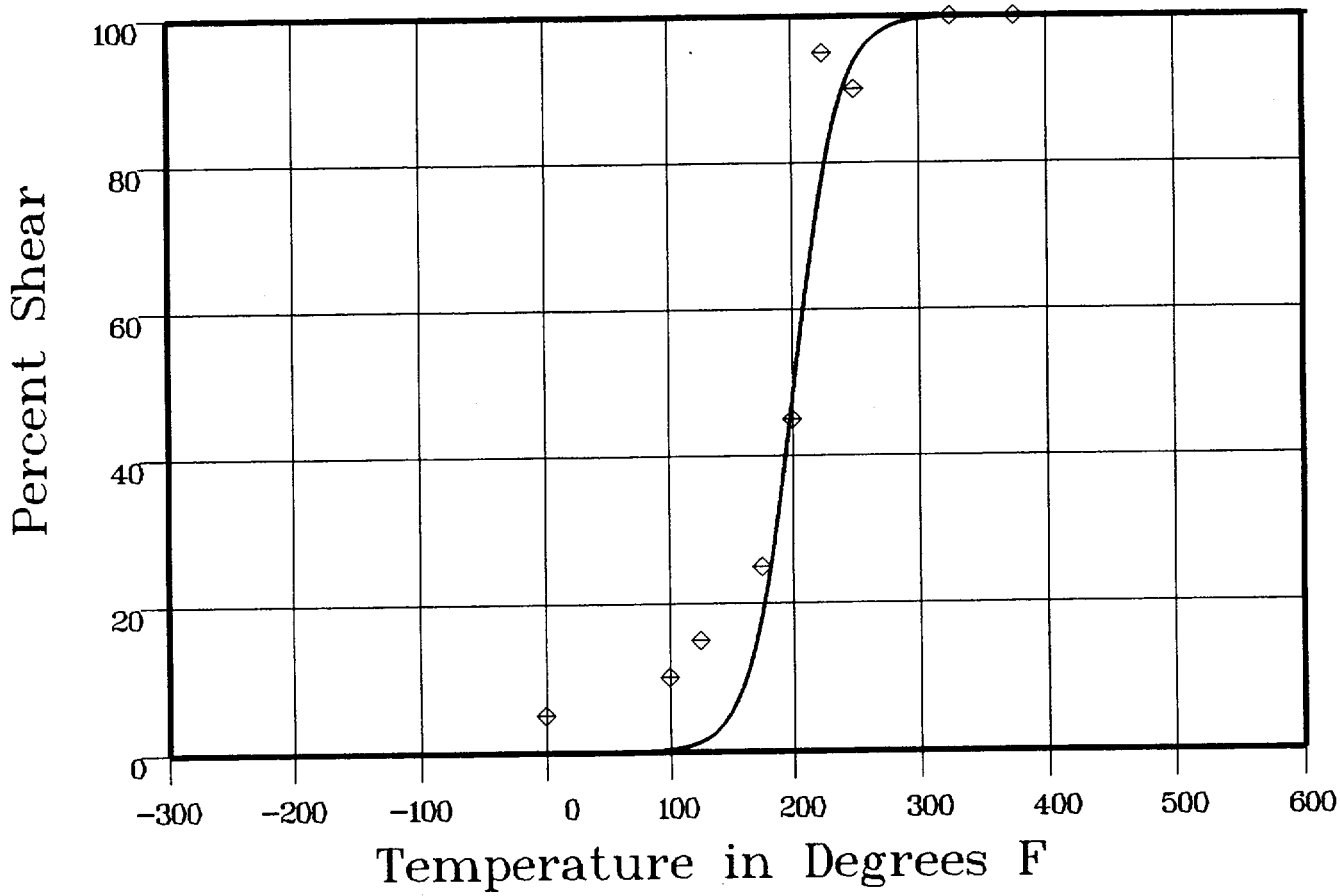
A = 50	B = 50	C = 35.72	T0 = 196.95

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

Temperature at 50% Shear: 196.9

Material: SRM SA533B1 Heat Number: Orientation:

Capsule: 38 Total Fluence:



Data Set(s) Plotted

Plant: PV1 Cap.: 38 Material: SRM SA533B1 Ori.: Heat #:

Charpy V-Notch Data

Temperature	Input Percent Shear	Computed Percent Shear	Differential
0	5	0	4.99
100	10	.43	9.56
125	15	1.74	13.25
175	25	22.62	2.37
200	45	54.24	-9.24
225	95	82.77	12.22
250	90	95.11	-5.11
325	100	99.92	.07

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

CAPSULE 38 STANDARD REFERENCE MATERIAL

Material: SRM SA533B1 Heat Number: Orientation:

Capsule: 38 Total Fluence:

Charpy V-Notch Data (Continued)

Temperature	Input Percent Shear	Computed Percent Shear	Differential
375	100	99.99	0

SUM of RESIDUALS = 28.12

APPENDIX C

Charpy V-Notch Shift Results for Each Capsule Hand-Fit vs. Hyperbolic Tangent Curve-Fitting Method (CVGRAPH, Version 4.1)

TABLE C-1
Changes in Average 30 ft-lb Temperatures for Intermediate Shell Plate M-6701-2
(Longitudinal Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
137°	7°F	40°F	33°F	8.15°F	42.08°F	33.93°F
38°	7°F	---	---	8.15°F	8.72°F	0.57°F

TABLE C-2
Changes in Average 50 ft-lb Temperatures for Intermediate Shell Plate M-6701-2
(Longitudinal Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
137°	37°F	70°F	33°F	36.62°F	65.38°F	28.75°F
38°	37°F	---	---	36.62°F	42.52°F	5.9°F

TABLE C-3
Changes in Average 35 mil Lateral Expansion Temperatures for Intermediate Shell Plate M-6701-2
(Longitudinal Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
137°	35°F	60°F	25°F	35.82°F	55.18°F	19.35°F
38°	35°F	---	---	35.82°F	33.19°F	-2.63°F

TABLE C-4
Changes in Average Energy Absorption at Full Shear for Intermediate Shell Plate M-6701-2
(Longitudinal Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
137°	151 ft-lb	129 ft-lb	-22 ft-lb	151 ft-lb	129 ft-lb	-22 ft-lb
38°	151 ft-lb	---	---	151 ft-lb	141 ft-lb	-10 ft-lb

TABLE C-5
Changes in Average 30 ft-lb Temperatures for Intermediate Shell Plate M-6701-2
(Transverse Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
137°	32°F	40°F	8°F	29.86°F	43.41°F	13.55°F
38°	32°F	---	---	29.86°F	10.14°F	-19.71°F

TABLE C-6
Changes in Average 50 ft-lb Temperatures for Intermediate Shell Plate M-6701-2
(Transverse Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
137°	72°F	100°F	28°F	71.24°F	98.34°F	27.10°F
38°	72°F	---	---	71.24°F	55.25°F	-15.98°F

TABLE C-7
Changes in Average 35 mil Lateral Expansion Temperatures for Intermediate Shell Plate M-6701-2
(Transverse Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
137°	70°F	75°F	5°F	66.22°F	70.13°F	3.90°F
38°	70°F	---	---	66.22°F	40.65°F	-25.56°F

TABLE C-8
Changes in Average Energy Absorption at Full Shear for Intermediate Shell Plate M-6701-2
(Transverse Orientation)
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
137°	98 ft-lb	87 ft-lb	-11 ft-lb	98 ft-lb	87 ft-lb	-11 ft-lb
38°	98 ft-lb	---	---	98 ft-lb	115 ft-lb	17 ft-lb

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

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
137°	-54°F	-45°F	9°F	-53.28°F	-56.22°F	-2.94°F
38°	-54°F	---	---	-53.28°F	-47.00°F	6.27°F

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

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
137°	-33°F	-20°F	13°F	-33.43°F	-28.82°F	4.60°F
38°	-33°F	---	---	-33.43°F	-25.30°F	8.12°F

TABLE C-11
Changes in Average 35 mil Lateral Expansion Temperatures for Surveillance Weld Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
137°	-30°F	-30°F	0°F	-30.94°F	-32.13°F	-1.18°F
38°	-30°F	- -	- -	-30.94°F	-31.79°F	-0.84°F

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

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
137°	164 ft-lb	162 ft-lb	-2 ft-lb	164 ft-lb	162 ft-lb	-2 ft-lb
38°	164 ft-lb	- -	- -	164 ft-lb	158 ft-lb	-6 ft-lb

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

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
137°	-62°F	-62°F	0°F	-63.2°F	-74.18°F	-10.98°F
38°	-62°F	---	---	-63.2°F	-89.79°F	-26.59°F

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

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
137°	-25°F	-25°F	0°F	-26.57°F	-37.9°F	-11.32°F
38°	-25°F	---	---	-26.57°F	-40.01°F	-13.43°F

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

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
137°	-30°F	-30°F	0°F	-29.74°F	-39.04°F	-9.3°F
38°	-30°F	---	---	-29.74°F	-42.71°F	-12.97°F

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

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
137°	135 ft-lb	124 ft-lb	-11 ft-lb	135 ft-lb	124 ft-lb	-11 ft-lb
38°	135 ft-lb	---	---	135 ft-lb	119 ft-lb	-16 ft-lb

TABLE C-17
Changes in Average 30 ft-lb Temperatures for the Standard Reference Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{30}	Unirradiated	CVGRAPH Fit	ΔT_{30}
137°	23°F	130°F	107°F	21.9°F	123.29°F	101.39°F
38°	23°F	---	---	21.9°F	136.16°F	114.25°F

TABLE C-18
Changes in Average 50 ft-lb Temperatures for the Standard Reference Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{50}	Unirradiated	CVGRAPH Fit	ΔT_{50}
137°	50°F	160°F	110°F	47.61°F	155.28°F	107.66°F
38°	50°F	---	---	47.61°F	176.21°F	128.6°F

TABLE C-19
Changes in Average 35 mil Lateral Expansion Temperatures for the Standard Reference Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔT_{35}	Unirradiated	CVGRAPH Fit	ΔT_{35}
137°	45°F	150°F	105°F	47.48°F	138.54°F	91.05°F
38°	45°F	---	---	47.48°F	153.84°F	106.35°F

TABLE C-20
Changes in Average Energy Absorption at Full Shear for the Standard Reference Material
Hand Fit vs. CVGRAPH 4.1

Capsule	Unirradiated	Hand Fit	ΔE	Unirradiated	CVGRAPH Fit	ΔE
137°	129 ft-lb	105 ft-lb	-24 ft-lb	129 ft-lb	105 ft-lb	-24 ft-lb
38°	129 ft-lb	---	---	129 ft-lb	105 ft-lb	-24 ft-lb

ENVIROCARE OF UTAH, INC.

THE SAFE ALTERNATIVE

40-8989

FAX COVER SHEET

DATE:	October 26, 2000		
TO:	Harold LeFevre	FROM:	Jon Carter
TO:	John Lusher	DEPARTMENT:	
PHONE:		PHONE:	(208) 336-1776
FAX:	(301) 415-5398	FAX:	(208) 336-0003

RE: _____

Number of pages including cover sheet: 29

Message: Harold and John -- Attached is a copy of a BNA article that appeared today. I have also attached copies of the materials that were filed with the Commission relative to our 2.206 petition. Thank you.

46 West Broadway, Suite 116, Salt Lake City, Utah 84101
Clive Facility Fax: (435) 884-3549 Mixed Waste (435)-884-6689

Daily Environment Report
No. 208
Thursday, October 26, 2000

NRC Policy Change, Delayed Decision Creates Untenable Situation, Envirocare Says

A change in the Nuclear Regulatory Commission's interpretation of a law, coupled with repeated delays in issuing that interpretation as a final decision, has put a Utah company in an "untenable" situation, according to an attorney representing the firm.

Jon Carter, an attorney representing Envirocare of Utah Inc., told NRC officials that the commission has put the company in an "untenable" situation by changing its interpretation of a law that describes low-activity wastes that must be regulated and by delaying formal issuance of that interpretation.

Envirocare is licensed by NRC to dispose of low-activity radioactive waste. Carter and other Envirocare representatives met with NRC officials Oct. 25 to discuss a wide range of issues.

Envirocare and the Snake River Alliance, an Idaho-based environmental organization, have petitioned the commission to regulate all wastes that are byproducts of uranium or thorium ore processing conducted for the nation's nuclear weapons program.

NRC Acknowledges Inconsistency. Stuart Treby, assistant general counsel at NRC, said the commission has regulated byproduct wastes inconsistently.

For many years, the commission focused on the process by which the wastes were generated, he said. All of these mill tailing wastes were considered 11e.(2) wastes, he told BNA. The designation is based on the section of the Atomic Energy Act (AEA) that describes these wastes.

In 1999, the commission reviewed the act and determined that Congress made a distinction, he said. Uranium and mill ore processing wastes are regulated by the commission only if they were generated at an NRC-licensed facility on or after the law went into effect Nov. 8, 1978, according to the commission's current interpretation of the law.

Byproduct waste generated at a facility that was not licensed by the NRC as of that date is not regulated by the commission, under its current interpretation.

Decision Expected by November. Such waste is being generated by the Formerly Utilized Sites Remedial Action Program (FUSRAP), which was operated by the Department of Energy from its inception in 1974 until 1997, when Congress transferred the program to the U.S. Army Corps of Engineers.

NRC changed its interpretation of the law after the corps took over the program and was trying to identify less expensive ways to dispose of FUSRAP waste, Carter told BNA after the meeting. It is typically cheaper to dispose of byproduct waste at a state-licensed facility than at Envirocare.

Although the commission has stated its current interpretation of the law at public hearings and in letters, it has not made a formal statement. That formal "final action" will result from the Envirocare's and Snake River's petitions, Treby said.

Senior NRC officials are reviewing staff recommendations for that final decision, Treby said. NRC expects to issue its decision by the end of November or earlier, he said. This is the third time NRC has delayed this decision.

NRC's current interpretation of the law and its delayed decision have created several problems, Carter said.

Under the AEA, when a waste disposal facility closes a "cell" containing 11e.(2) wastes, either an authorized state agency or DOE is required to take ownership of the waste. Cells are the term for the area in which the waste is buried. If a state declines to take ownership, the law makes DOE the long-term "custodian" of the waste, Treby said.

Ownership Key Question. If NRC issues its current interpretation of the law as a final decision, DOE would not be required to take ownership of that portion of waste shipped to Envirocare, but generated before the 1978 cutoff, Carter said.

About half of the 1 million cubic yards of waste shipped to Envirocare for disposal in its 11e.(2) cell was generated before 1978, Carter said. If DOE declines to take ownership of that portion, Envirocare could be deemed its custodian, he said.

At the meeting, Treby said Envirocare could argue that DOE must be responsible for the waste since the department identified it as 11e.(2) waste when it shipped the material for disposal.

"From a liability perspective this puts us in a difficult position," Carter said.

Carter and Ken Alkewa, who also represented Envirocare, described another problem.

NRC Reviewing FUSRAP Classifications. The corps is shipping FUSRAP waste, which it has certified as complying with Envirocare's license, i.e., being 11e.(2) wastes, from two New Jersey sites. Yet, the corps also has asked NRC to review the corps' conclusion as to whether this waste qualifies for the 11e.(2) designation.

FUSRAP waste from one site, the Maywood Interim Storage Site, is 11e.(2) waste that can be buried in the federally licensed cell, said Philip Ting of NRC's Office of Nuclear Materials Safety and Safeguards. The Maywood site, located northwest of Hackensack, N.J., is included on EPA's National Priorities List.

A corps official familiar with Maywood could not be reached to say whether FUSRAP waste from that site has been disposed at facilities other than Envirocare, which are regulated by states, not NRC, and hence should not accept 11e.(2) waste.

Carter asked how the NRC would enforce its jurisdiction if the agency decides it does not have authority over waste generated before 1978 in facilities not licensed by NRC.

None of the NRC officials described a specific enforcement plan. But, Ting replied that if NRC learned that a shipment of 11e.(2) waste had been disposed of at a facility not licensed by the agency, that facility "would be in trouble."

NRC has not made a decision about the FUSRAP generated at the other site, called the Wayne Interim Storage Site, Ting said. The 6.8-acre Wayne site is located near Maywood, east of Lincoln Park, N.J.

Carter asked Treby whether Envirocare should continue to dispose of the waste generated at the Wayne site in its NRC-licensed cell. Envirocare has one cell licensed for 11e.(2) waste, and another licensed by the state for naturally occurring radioactive materials.

Treby did not respond directly. However, Envirocare is only supposed to dispose of 11e.(2) wastes in the NRC-licensed cell, he said.

Utah Decision. Utah could allow Envirocare to dispose of the Wayne waste in the cell it operates, Treby said. However, he continued, if he were a state regulator he would not allow the disposal until NRC has made its decision about how it defines 11e.(2) wastes.

After the meeting, Carter told BNA the uncertainty the company is facing is "disconcerting to say the least."

Rail cars with waste from Wayne are headed to Envirocare, he said. The corps has certified the waste as being 11e.(2) material, Carter said. That certification, however, was done in 1995, he said.

Treby's comments during the meeting appeared to suggest that Envirocare should not bury this waste in the NRC-licensed cell, Carter continued. "Should we tell the army to turn the trains around?"

Carter said he intended to seek another meeting with NRC to clarify what Envirocare should be doing with the Wayne waste and additional FUSRAP waste the corps is shipping from St. Louis.

The corps, waste disposal companies, and the Environmental Technology Council, which represents commercial hazardous waste disposal facilities, have filed responses to Envirocare's petition with the NRC.

Legislative history shows that Congress never intended for NRC to regulate FUSRAP waste, the corps, ETC, and companies argued. Congress was aware of other sites that contained mill tailing waste, but the uranium mill tailings act that established FUSRAP listed only 22 specific inactive sites that should be cleaned up by the federal government, the corps said.

Much of the waste cleaned up in FUSRAP can be safely disposed at hazardous waste facilities, the groups argued.

William M. Beckner, executive director of the National Council on Radiation Protection and Measurements, a nonprofit scientific group that develops recommendations on radiation protection, previously told BNA his organization is developing a report calling for risk-based radiation regulation.

By Pat Phibbs

ENVIROCARE OF UTAH, INC.
THE SAFE ALTERNATIVE

October 25, 2000

VIA FACSIMILE – (301) 415-3725

Jack Goldberg
Deputy Assistant General Counsel for Enforcement
Nuclear Regulatory Commission
One White Flint North Building
11555 Rockville Pike
Rockville, MD 20852-2738

Dear Mr. Goldberg:

Representatives of Envirocare of Utah, Inc. (Envirocare) met this morning with Nuclear Regulatory Commission staff including Stuart Treby from the Office of General Counsel. I had a couple of questions about Envirocare's 2.206 petition and Mr. Treby referred me to you.

I am curious as to whether the NRC intends to provide the petitioners an opportunity to review the Commission's decision under the NRC's new 2.206 petition process. We certainly would like to take advantage of this opportunity if it exists.

In addition, I clarified for Mr. Treby that the attached letter and responses relate to issues before the NRC in our 2.206 petition, and we would appreciate a careful consideration by the Commission of these matters.

Please telephone me at (801) 557-4350 if you have any questions. Thank you.

Very truly yours,

Jonathan P. Carter

Jonathan P. Carter
General Counsel

Attachment

cc: Stuart Treby, Via Facsimile, w/o attachment
Karen Cyr, Via Facsimile, w/o attachment

MILLER & CHEVALIER
CHARTERED
655 FIFTEENTH STREET, N.W. SUITE 900
WASHINGTON, D.C. 20005-5701
(202) 626-5800 FAX: (202) 626-0858

October 18, 2000

Mr. Dennis K. Rathbun
Director, Office of Congressional Affairs
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Rathbun:

Envirocare of Utah has reviewed the NRC's responses to questions of the Senate Committee on Environment and Public Works regarding uranium mill tailings regulation that were attached to your letter of September 12, 2000, to Senator Bob Smith. The attached comments on those responses are submitted on behalf of Envirocare. The comments are directed to four of those responses, since, in Envirocare's view, it is those responses that bear most directly on the subject matter of Envirocare's pending section 2.206 petition on mill tailings regulation.

By copy of this letter, we request that the attached comments be considered in connection with that petition.

Yours sincerely,

Leonard Bickwit, Jr.

Attachment

cc: Dr. William D. Travers
NRC, Executive Director for Operations

Stuart A. Treby, Esquire
NRC, Office of General Counsel

Douglas E. Roberts, Vice President, Regulatory and External Affairs
Envirosource Technologies

Robert M. Andersen
Chief Counsel, U.S. Army Corps of Engineers

Gary Richardson, Executive Director
Snake River Alliance, Petitioner

SENATOR BENNETT'S QUESTION 6

Is NRC reversing the position stated in 57 Fed. Reg. 20,527 (May 13, 1992) that materials that satisfy the 11e.(2) definition generated by MED/AEC "qualify as 11e.(2) byproduct material"? And if so, why?

This question and the NRC's response both address a 1992 Request for Public Comment ("the Request") on proposed Commission guidance regarding disposal of "non 11e.(2) byproduct material" in uranium mill tailings piles. The response suggests that the Request's discussion of section 11e.(2) byproduct material is consistent with the Commission's current position that pre-1978 FUSRAP mill tailings[1] are not covered by section 11e.(2) of Atomic Energy Act ("AEA").[2] To the contrary, however, the Request clearly indicates that FUSRAP mill tailings are section 11e.(2) material. The response reaches the opposite conclusion only because it does not focus on critical portions of the Request. Thus, the NRC's current position is in fact a reversal of the position taken in the Request.

The response correctly points out that in the Request, "the term 'non-11e.(2) byproduct material'" refers to waste that is "similar" to section 11e.(2) material but "is not legally considered to be 11e.(2) byproduct material." The response also correctly observes that certain FUSRAP wastes are described by the Request as falling into this category of "non-11e.(2) byproduct material." What the response omits, however, is the reason why these wastes are viewed as not qualifying as 11e.(2) material. That reason is that the particular FUSRAP wastes identified are wastes that are not produced from the processing of source material. It is only those wastes that are referred to in the request as "non-11e.(2) byproduct material," while FUSRAP wastes that were produced from such processing are clearly viewed by the Request as within the coverage of section 11e.(2).

The Request's General Principle

A review of the Request's relevant language makes this clear. At the beginning of the Request, it is stated:

> In the guidance documents and associated staff analyses [included in the Request], the term "non-11e.(2) byproduct material" is used to refer to radioactive waste that is similar in physical and radiological characteristics (for example, low specific activity) to byproduct material, as defined in Section 11e.(2) of the AEA

---

[1] These comments will use the term "pre-1978" material to refer to material over which the NRC asserts that it lacks jurisdiction.

[2] Envirocare acknowledges that it does not know the exact position that the Commission is or will be taking on this matter. For purposes of these comments, it will assume that the position is that mill tailings do not meet the definition of section 11e.(2) of the AEA unless such tailings were produced at a site licensed by the NRC as of the effective date of section 83 of that act or thereafter.

> but does not meet the definition in that section because it is not derived from ore processed primarily for its source material content.

(Emphasis added.) It is then stated, in a reference to such material, that:

> Some licensees have proposed to directly dispose of radioactive wastes in existing uranium mill tailings sites. The materials vary from tailings from extraction processes for metals and rare-earth metals (such as copper, tantalum, columbium, zirconium) to spent resins from water-treatment processes. However, because these materials did not result from the extraction or concentration of uranium or thorium from ore, they are not 11e.(2) byproduct material.

(Emphasis added.) The general principle is thereby established that tailings resulting from the processing of metals and rare-earth metals, as well as other wastes unrelated to AEA source material, are not section 11e.(2) byproduct material and are to be distinguished in that regard from tailings resulting from the processing of uranium and thorium, which are to be considered 11e.(2) material.

Application of the Principle

The Request then applies this general principle to the "Types of Wastes Being Proposed for Disposal Into Tailings Piles." At the beginning of that discussion is the language cited in response to Senator Bennett's question. That language reads as follows:

> The NRC and the Agreement States continue to receive requests for the direct disposal of non-11e.(2) byproduct material into uranium mill tailings piles. The following general categories of non-11e.(2) byproduct material illustrate the requests submitted to NRC and the Agreement States for disposal into uranium mill tailings piles licensed under authority established by title II of UMTRCA.

The first category, mine wastes, are found not to "satisfy the definition of 11e.(2) byproduct material, because they do not result from the extraction or concentration of uranium or thorium from ore." (Emphasis added.) The second, secondary process wastes, are described as tailings created when "natural ores . . . are processed for rare-earth or other metals." (Emphasis added.) These tailings are not viewed as 11e.(2) byproduct material, since "the ore was not processed primarily for its source material content, but for the rare-earth or other metal."

(Emphasis added.)

It is against this background that the discussion of wastes at FUSRAP sites takes place. The full discussion of those wastes is as follows:

> These sites primarily processed material, such as monazite sands, to extract thorium for commercial applications. Government contracts were issued for thorium source material used in the Manhattan Engineering District and early Atomic Energy Commission programs. Wastes resulting from that processing and

> disposed of at these sites would qualify as 11e.(2) byproduct material. However, it is not clear that all the contaminated material at these sites result from processing of ore for thorium. At some sites there was also processing for rare earths and other metals. The DOE, which accepts responsibility for the FUSRAP materials, is investigating options for disposal and control of these materials. DOE estimates that a total of 1.7 million cubic yards of material is located at sites in 13 States. Recent proposals have considered the transportation of FUSRAP materials from New Jersey to tailing piles at uranium mills in other States, such as Utah, Washington, and Wyoming.

(Emphasis added.) There can be no doubt as to the point the Request is making by these observations. While the FUSRAP tailings not resulting from the processing of source material are not section 11e.(2) material, tailings that do result from such processing do in fact constitute 11e.(2) material.

The Meaning of "Would"

The response also argues that when the Request states that FUSRAP "would qualify as 11e.(2) byproduct material" (emphasis added), it means only that such tailings would qualify as such material if they "fell under NRC jurisdiction in the first place." This argument is plainly without merit. The response places great weight on the Request's use of the word "would" in the above-quoted language. The phrasing used, however, is merely a natural way to provide a generalized explanation. In fact, in the paragraph just preceding the one in which the quoted language appears, the discussion of "secondary process wastes" contains the same phrasing: "If the tails contain greater than 0.05 percent uranium and thorium, they would be source material and would thus be licensable and have to be disposed of in compliance with NRC regulations." (Emphasis added.) This is not a reference to what would happen if some other unnamed condition were met.

In sum, one would conclude from the response that the Request means just the opposite of what it says. The only defensible answer to Senator Bennett's question is that the Commission's current position is in fact a reversal of the position it took in the Request.

A marked-up copy of the relevant portions of the Request is attached.

Attachment

SENATOR BENNETT'S QUESTION 4

Please explain why 10 C.F.R. 40.2(b) [sic] makes no reference to such materials having to be licensed by NRC but rather appears to suggest that NRC can regulate such materials whether licensed or not as long as they are not at a DOE controlled Title I site.

The response to this question takes a similarly forced approach to 10 C.F.R. § 40.2a. A fair reading of the regulation again demonstrates that the Commission is reversing previously held positions.

The regulation in question reads as follows:

> **Section 40.2a Coverage of inactive tailings sites.**
>
> (a) Prior to the completion of the remedial action, the Commission will not require a license pursuant to 10 CFR chapter I for possession of residual radioactive materials as defined in this part that are located at a site where milling operations are no longer active, if the site is covered by the remedial action program of Title I of the Uranium Mill Tailings Radiation Control Act of 1978, as amended. The Commission will exert its regulatory role in remedial actions primarily through concurrence and consultation in the execution of the remedial action pursuant to Title I of the Uranium Mill Tailings Radiation Control Act of 1978, as amended. After remedial actions are completed, the Commission will license the long-term care of sites, where residual radioactive materials are disposed, under the requirements set out in § 40.27.
>
> (b) The Commission will regulate byproduct material as defined in this part that is located at a site where milling operations are no longer active, if such site is not covered by the remedial action program of Title I of the Uranium Mill Tailings Radiation Control Act of 1978. The criteria in Appendix A of this Part will be applied to such sites.

The structure of this regulation and the dividing line it draws are stated clearly. The title indicates that the regulation as a whole deals with "inactive tailings sites." The category of inactive sites is divided into two components. Subsection (a) addresses "site[s] where milling operations are no longer active, if the site is covered by the remedial action program of Title I of the UMTRCA." (Emphasis added.) Subsection (b) covers the rest, that is, "site[s] where milling operations are no longer active, if such site is not covered by the remedial action program of Title I of the UMTRCA." (Emphasis added.) There is no suggestion that anything less than all inactive sites are intended to be covered by the section's provisions. In this respect, the regulation reflects the broad statutory language of sections 11e.(2), 81 and 84 of the AEA. From the title of the regulation on down, the clear indication is that the section deals with the entirety of the category of inactive sites.

The Text of the Rule

The response, accordingly, is at odds with the rule's text. It reads an additional restriction into subsection (b), claiming that the subsection applies only to an inactive site that was under active license "as of the effective date of UMTRCA." The response appears to acknowledge the tension between the NRC's current position on section 11e.(2) and the language of the regulation. It argues, however, that "[t]he inconsistency disappears if the intent of the regulation is understood." This leads to some necessary questions. If an essential feature of subsection (b) is that only licensed sites are covered by the subsection, why does the subsection make absolutely no reference to that limitation? Why is there no indication of the Commission's current interpretation either in section 40.2a or in any other section of UMTRCA's implementing regulations? Given that exemptions from licensing and regulation are clearly stated all throughout those regulations and elsewhere in Part 40,[1] why is there no mention of an exemption from regulation for tailings from sites not under license as of November 8, 1981 (i.e., the effective date of Section 83)? How could the Commission consistently fail to include references to an exemption that goes to the heart of the Commission's jurisdiction over mill tailings? The analysis of the Commission's regulations put forward in this response parallels the analysis that has been offered with respect to the legislation itself. In each case, it is asserted that language that on its face unquestionably covers all mill tailings at inactive sites should be read not to do so. In the absence of some compelling explanation as to why this purported exemption was left unexpressed, it is not possible to believe that any such exemption was actually intended.

Other Contemporaneous Commission Actions

Not only are indications of the Commission's current interpretation absent from the proposed and final versions of this regulation; they are also missing from other relevant contemporaneous Commission documents. No mention of the Commission's current interpretation can be found, for instance, in the Commission's final rule of August 24, 1979, relating to mill tailings licensing;[2] in Commission meeting transcripts in 1979 regarding the need for such licensing and for proposed changes in UMTRCA;[3] or in the Executive Legal Director's discussion papers on which the Commission meetings were held.[4] If the Commission in 1979 and 1980 had in fact adopted the interpretation now held by the NRC, there would have been good reason for it to note that interpretation at that time. The meeting transcripts indicate that

---

[1] See, e.g., 10 C.F.R. §§ 40.1, 40.2, 40.11, 40.12, 40.13, 40.14, 40.32 and 150.31.

[2] 44 Fed. Reg. 50,012 (Aug. 24, 1979).

[3] Discussion of SECY-79-88 – Timing of Certain Requirements of the Uranium Mill Tailings Radiation Control Act of 1978 before the Nuclear Regulatory Commission (March 7, 1979); Discussion of SECY-79-88 – Uranium Mill Tailings before the Nuclear Regulatory Commission (May 9, 1979 and May 17, 1979).

[4] SECY-79-88 (Feb. 2, 1979); Staff Response to the Commission Request for Further Information Regarding SECY-79-88 (May 7, 1979).

the Commission was eager to avoid licensing tailings in the immediate aftermath of UMTRCA's enactment at sites licensed by Agreement States.[5] It undertook such licensing only because the Executive Legal Director advised that UMTRCA required such action.[6] Had the Commission actually believed that its licensing responsibilities related only to tailings produced at sites licensed on the effective date of section 83, it could have argued that those responsibilities could not be determined until that effective date had arrived and should accordingly be delayed. Its failure to make that argument is further evidence that the Commission's current position is one that was not held by the Commission at the time.

As a related matter, it should be noted that if the Commission had actually held that position, it could not have justified the licenses it issued in 1979 and 1980. In those years, the Commission issued general licenses to its licensees for the tailings possessed by those licensees. One may legitimately ask under what authority it was functioning. Under the Commission's current position, the NRC at that time could not have known what tailings constituted "byproduct material," since that fact would have been unknowable until November 8, 1981, the effective date of section 83 and the date on which jurisdiction would have been determined. The Commission would essentially have been in limbo prior to that date, since although it had been told to regulate "any byproduct material" immediately upon enactment,[7] it would not have known which tailings were "byproduct material" until three years after enactment. The Commission issued tailings licenses during this period because such an anomalous construction almost certainly never occurred to anyone at the Commission at the time. The total confusion this construction would have caused during these early years strongly suggests that the construction never occurred to anyone in the Congress either.

NRC Actions Since 1998

Envirocare's comments on the NRC's responses to Senator Bennett's Questions 4 and 6 thus support the view that prior to 1998 the Commission interpreted its authority to apply to all mill tailings without exception. The responses themselves, on the other hand, attempt to convey the impression that the position taken in the so-called "Fonner letter" of March 1998 was consistently maintained during the twenty years prior to that letter. For the reasons stated, the responses are not persuasive in that regard.

---

[5] The Commission wished to delay its licensing authority until three years after enactment, as it had proposed in the legislation it originally submitted to the Congress. See H.R. 13382, 95th Cong. § 2 (1978). This bill, introduced by Congressman Udall, was based on the Commission's submission.

[6] It is significant that the Executive Legal Director's advice that licenses were required to be issued immediately was based entirely on sections 81, 84 and 11e.(2) of the AEA. There is no indication in that advice that section 83, which had not yet become effective at the time, in any way limited the reach of those sections with respect to the Commission's obligations under UMTRCA or that section 11e.(2) was otherwise limited in its coverage.

[7] See UMTRCA § 208.

It is worth noting, moreover, that even since 1998 the Commission has taken positions at odds with its current view. First, the Commission has maintained that it can regulate pre-1978 mill tailings on NRC-licensed 11e.(2) sites to the same extent as it regulates post-1978 tailings. In a letter to the Utah Department of Environmental Quality, it stated, "[i]f pre-1978 11e.(2) byproduct material is presented as such to the NRC-licensed Envirocare facility for disposal, Envirocare must comply with all the requirements applicable to disposal of 11e.(2) byproduct material."[8] This statement cannot be reconciled with the Commission's position that pre-1978 material is beyond the Commissions jurisdiction. The Commission cannot regulate non-licensable material to the same extent that it regulates licensable material, even when the non-licensable material is sent to an NRC-licensed site.

Second, the NRC initially indicated that an Envirocare request to dispose of pre-1978 mill tailings in its Utah Agreement State-regulated low-level waste cell should be denied based on 10 CFR § 61.1(b) of the Commission's regulations, which the NRC has asserted restricts the disposal of mill tailings in facilities regulated under Part 61.[9] It took this position notwithstanding that the restrictions in section 61.1(b) apply only to mill tailings defined in Part 40, which the Commission contends do not include pre-1978 mill tailings.

Third, notwithstanding that under Envirocare's section 11e.(2) license, non-11e.(2) material may not be disposed of in its licensed 11e.(2) cell, the Commission has routinely permitted the disposal in that cell of FUSRAP mill tailings, which it contends are non-11e.(2) material.[10] Finally, the Commission has consistently permitted such disposal in the face of its current guidance document on the disposal of non-11e.(2) material, which states, "[R]adioactive material not regulated under the AEA shall not be authorized for disposal in an 11e.(2) byproduct material impoundment."[11]

---

[8] Letter to William J. Sinclair, Director of Utah Dep't of Environmental Quality, Div. of Radiation Control (Sept. 24, 1999).

[9] Id.

[10] The disposal of these materials has occurred pursuant to certification procedures specifically required and approved by the NRC to be included in Envirocare's Standard Operating Procedures. At the time of the adoption of these procedures in April 1994, the NRC stated, "NRC staff has reviewed the information in the procedure and concludes that the procedure will ensure that wastes other than 11e.(2) byproduct material are precluded from disposal in the NRC licensed disposal site. The procedure also will ensure documentation of the constituents in the waste." See Safety Evaluation of the "Procedures for Certification of 11e.(2) Material."

[11] 60 Fed. Reg. 49,296 (Sept. 22, 1995). Since 1994, Envirocare has disposed of approximately one-half million cubic yards of FUSRAP material in its 11e.(2) disposal cell without NRC objection.

What the Commission has been doing, it appears, is regulating this material when it chooses to and not regulating it when chooses not to, without regard to the clear requirements of the law governing this subject matter and the applicable Commission regulations. The Commission should reject this practice and revert to the position it consistently maintained during the twenty years following UMTRCA's enactment.

SENATOR BENNETT'S QUESTION 3

Exactly where in § 83 or in the related legislative history does it say that NRC has no authority over wastes that satisfy the definition of 11e.(2) byproduct material MED or AEC generated by processing for uranium or thorium if generated prior to 1978? (Please assume that such materials are under the control of a private entity and not DOE or are going to be removed from DOE control).

The Commission's response is reducible to several arguments, which are considered here in order.

The Language of Section 83

The Commission argues that the language of section 83 "clearly indicates that NRC's regulatory authority and responsibility for . . . [mill tailings] material are prospective. That is, Congress intended NRC to regulate only those mill tailings at existing licensees' sites and those newly licensed after UMTRCA was enacted." But the language does not so indicate. What it does indicate is that the provisions of that section are to apply only to licenses in effect on the effective date of the section and all future licenses. The language does not address what other sections of UMTRCA are intended to do.

Moreover, in restricting its application to licenses in effect on the effective date of the section and all future licenses, the section does no more than state the obvious. The only category of licenses excluded are licenses not in effect on the section's effective date, i.e., licenses that existed at one time but that terminated prior to November 8, 1981. That is because it would not have made sense for a section requiring that licenses contain certain specified provisions to be applied to licenses that had terminated before the section even came into effect. There is thus nothing meaningful about the selection of the words "[a]ny license issued or renewed after the effective date of this section." The fact is that no other words could reasonably fit in the place in which those words appear.

What is meaningful is that Congress decided to regulate mill tailings primarily through the licenses of the Commission's source material licensees. Congress made section 83 the centerpiece of Title II, and no doubt anticipated that most of that title's requirements would be imposed through the provisions of that section. It also made unmistakably clear, however, that other authorities and obligations would be assigned to the Commission through other sections of Title II. Sections 81 and 84, in conjunction with the definition of section 11e.(2), extend the reach of Title II beyond the licenses referenced in section 83 to all uranium and thorium mill tailings.

There is nothing ambiguous about these statutory instructions. Section 83's requirements clearly apply only to source material licenses in effect on or after November 8, 1981. The requirements of sections 81 and 84 just as clearly apply to all material meeting the definition of section 11e.(2), i.e., all uranium and thorium mill tailings. Moreover, none of the sections in any way conflicts with the others. The requirements for section 83 licenses are more extensive and specific than the requirements of sections 81 and 84, but the requirements are not in conflict.

What sections 81 and 84 unambiguously indicate is that Congress intended for the NRC's authority to encompass all mill tailings, whenever and however produced. Thus, while the regulation of tailings was to be conducted primarily through the licenses of source material licensees, the Commission was to have authority to deal with any and all safety concerns posed by mill tailings. No other reason has been offered, and none suggests itself, as to why these sections read the way they do.

It thus cannot be said that the language of section 83 "clearly indicates . . . that the Congress intended NRC to regulate only those mill tailings materials at existing licensees' sites and those newly licensed after UMTRCA was enacted." On the contrary, sections 83, 81, 84 and 11e.(2), when read in conjunction with each other, unambiguously indicate just the opposite.

Legislative History: Exemption of FUSRAP

The response also states: "It is clear from the legislative history that Congress was aware of the FUSRAP sites and concluded that those sites would not be handled under UMTRCA." It then cites in support of that contention certain portions of the legislative history, which are discussed below. The first and most obvious answer to the contention, however, is that if it were in fact the case, i.e., if Congress "concluded that [FUSRAP] sites would not be handled under UMTRCA," why did Congress not simply say so in the statute? As just noted, the language of Title II is unambiguously comprehensive. The sequence of events in the legislative history confirms that this comprehensiveness was intentional. NRC draft legislation, which combined the ultimately-adopted definition of section 11e.(2) with the then-existing all-inclusive language of section 81, plainly applied the new licensing authority of the Act to all mill tailings. All versions of Title II considered throughout the legislative process were similarly comprehensive. These versions no doubt were reviewed routinely by NRC lawyers, by counsel for the House committees where the legislation was developed, and by the House Legislative Counsel's Office. The absence of any grandfathering provision in sections 81, 84 or 11e.(2) could not have failed to be noticed. It is apparent that both the Congress and the Commission wished the Commission's authority over tailings to be as comprehensive as its authority over any other licensable material under the AEA. In sum, it is not credible that a Congress that truly wished to exclude material associated with FUSRAP from NRC regulation under UMTRCA would have drafted, in a carefully worded statute where other exclusions are clearly stated, language in Title II that unambiguously covers all mill tailings.

Beyond that, the legislative history strongly indicates that such tailings were intended to be included not only under Title II's provisions, but also under Title I. While the record is often murky and confusing on this matter, it shows that (i) the Congress was indeed aware of some sites that ultimately became "FUSRAP sites," even though they were never referred to by that name and were typically referred to as "other sites"; (ii) at the time UMTRCA was considered, all that was in progress was a survey of these other sites — none had actually been selected for remediation;[1] (iii) the text of Title I, which listed the 22 specific sites that were initially selected

[1] See Uranium Mill Tailings Control Act of 1978: Hearings Before the Interstate and Foreign Commerce H. Rep., 95th Cong. 185 (1978) (statement by James Liverman that DOE was

for the Title I remedial program, also provided for a one-year time frame, or "window," during which DOE was permitted to add to that list of sites; (iv) the principal reason for that window was to allow the survey of the "other sites" to be completed so that those sites could be included in Title I if they were found to have mill tailings and otherwise met the title's criteria; and (v) Congress likely believed that if the surveyed sites had mill tailings on them, and did not qualify for Title I's government-owned site exemption, they would in fact be included in Title I.

It is thus not the case that Congress decided, as the NRC response suggests, to approve a two-track system, with Title I operating on one track and FUSRAP on the other. In fact, it is impossible to imagine that Congress could have sanctioned such a system, given the other contemporaneous decisions it was making regarding UMTRCA. In its development of Title I, Congress insisted on significant and unusual regulatory controls, the most important of which were federal or state acquisition of tailings sites and disposal sites, the ultimate transfer to the federal government of the tailings and sites once remedial action was complete, and NRC licensing of DOE or such other federal agency as the President determined should be the ultimate custodian of the land and the tailings. Notwithstanding that the NRC objected to the licensing of DOE, the Congress insisted that such licensing be required. Against this background, it is not conceivable that Congress, to the extent it understood that the "other sites" might have mill tailings on them, would have accepted a separate remediation system for those sites free of the protections Congress had laboriously developed for the Title I program, especially NRC regulation.[2]

Mr. Liverman's Statement

The response bases its arguments to the contrary on three portions of the legislative history: (i) a statement of James Liverman, DOE Acting Assistant Secretary for the Environment, (ii) language in a House committee report with respect to certain reporting requirements under the statute, and (iii) certain statements that the response asserts indicate that Title II was to apply only to existing and future licensed sites. None of this legislative history provides a basis for the NRC's current position.

To begin with, the response's quote from Mr. Liverman can by no means be viewed as the Congress's final word on whether the sites DOE was still surveying would be covered by

---

"currently in the process of evaluating" these sites for radioactivity hazards, that some of these sites would "probably" require remediation, but that "the need for remedial action [had not yet been] determined") ("Commerce Hearings").

[2] Of course, notwithstanding these intentions at the time of enactment, the FUSRAP sites ultimately were never incorporated into the program. They were, however, covered by Title II nonetheless, since that title was drafted comprehensively. As stated in its pending section 2.206 petition, Envirocare is not arguing that the reason Title II was so drafted was to cover such sites. That may or may not have been the case. What is clear is that the title was drafted comprehensively to cover all eventualities, i.e., to cover everything that the Congress thought of, or might not have thought of, with regard to mill tailings.

Title I. For one thing, Mr. Liverman appears to have been of more than one mind on the matter. In another passage, he indicated that after the current survey was complete, "DOE will be in a position to determine which, if any, of these properties could be included in this legislation."[3] Far more important, from all indications Congress disagreed with the statement quoted in the response. As indicated, what Congress enacted in Title I was a provision that designated a one-year window for the post-enactment designation of sites other than the sites listed in the title. In discussing an early draft of this "window" provision, the EPA explained that "DOE has been conducting environmental surveys of old sites that were formerly used for research and development work in the early days of the Nation's atomic energy program. Some of these sites may be found to have similar conditions and would be covered under this bill." (Emphasis added.) In fact, before UMTRCA's enactment, the ongoing survey had already identified one site that involved a serious mill tailings problem — Canonsburg, PA — and this site was immediately added to Title I.[4] As for other sites in the survey, Congress specifically asked whether any of these sites were known to have mill tailings, and DOE indicated in response that it could not yet identify any such sites with mill tailings.[5]

In short, Congress from all indications believed it had successfully provided for the remediation of all inactive mill tailings sites not covered by a specific exemption. After listening to Mr. Liverman, the EPA and the other relevant hearing witnesses, the House Interior Committee explained that the 22 named Title I sites "consist of tailings resulting from operations under Federal contracts. None are now under active license by the Nuclear Regulatory Commission. While it is believed that these sites are the only ones which possess all such characteristics, the bill permits the inclusion [through the window provided] of any other sites meeting those characteristics."[6]

Report Language On Reporting Requirements

The response relies on House Commerce Committee report language that requires reports with regard to remedial activities concerning certain sites that were ultimately included in FUSRAP. The response notes that the sites are identified separately from Title I sites and concludes that Congress "viewed the FUSRAP sites as separate and distinct from the Title I sites."

---

[3] Commerce Hearings at 185.

[4] See UMTRCA § 102(a)(1); Uranium Mill Tailings Control Act of 1978: Hearings Before the Subcomm. on Energy and the Environment of the Comm. on Interior and Insular Affairs, 95th Cong. 49 (1978) ("Interior Hearings"); Commerce Hearings at 298.

[5] See, e.g., Commerce Hearings at 328-32 (giving a list of "all known mill tailings sites located in the United States" that did not include any sites in the survey except for Canonsburg).

[6] H. R. Rep. No. 95-1480, part 2, at 13 (1978) ("Interior Committee Report").

The reporting requirement in question, however, relates to a category of sites specifically exempted from Title I: government-owned or controlled sites.[7] There is no doubt that such sites were excluded from Title I coverage under section 101(6)(A)(i) of UMTRCA. Thus, while it is true that some sites that later became FUSRAP sites are mentioned in the report language, that is only because they qualified for this specific exemption for federally owned sites. The language makes this clear:

> The Committee understands there that [sic] are a number of federally owned or controlled sites with such materials or tailings, such as the TVA site mentioned earlier and a DOE site in Lewiston, N.Y., and some in New Jersey.[8]

(Emphasis added.) As the Commission is aware, FUSRAP deals primarily with privately owned sites. This passage thus in no way implies an exemption from UMTRCA's Title I, much less from Title II, for such sites or for FUSRAP generally.

Moreover, the response's claim that each of the sites mentioned in this passage "was a FUSRAP site at the time Congress enacted UMTRCA" is not correct. First of all, there were no "FUSRAP sites" at the time of enactment. As indicated, no sites had yet been selected for remediation at that time.[9] Moreover, "the TVA site" is "the Tennessee Valley Authority site at Edgemont, South Dakota," which has never been in the FUSRAP program and which was referred to repeatedly in hearings as an example of an excluded federal site.[10]

References to Existing Sites

The response also notes references in the legislative history that it claims suggest that the new authority conferred by Title II was to apply only to milling operations that were active at the time of UMTRCA's enactment. For three reasons, however, those references cannot be relied on to justify the Commission's restrictive interpretation of Title II. First, to the extent that the references can be read as the Commission characterizes them, the references are undeniably imprecise. For example, Title II unquestionably provided the NRC with authority to perform its Title I responsibilities, notwithstanding that those responsibilities do not relate to active mill operations.[11] The Commission also has specifically acknowledged that it was provided

---

[7] This is one of the two principal exemptions from Title I, the other relating to licensed sites. It is significant that there is no independent exemption for FUSRAP sites that do not fall into these two categories.

[8] H. R. Rep. No. 95-1480, part 1, at 41 (1978) ("Commerce Report").

[9] See supra note 1.

[10] Commerce Hearings at 260; see also, e.g., id. at 197, 328.

[11] Both the House Interior Committee report and the House Commerce Committee report on the legislation specify that the new section 84 of the AEA was to be used in part for the

authority under Title II to regulate the Edgemont site, an inactive but licensed site. Just as the legislative history references quoted in the response should not be read to preclude the exercise of these authorities, they should not be read to foreclose the regulation of FUSRAP waste. Rather, they should be read to indicate that Title II is primarily, not exclusively, about active mill operations.

Second, the references can be further explained by the fact, discussed earlier, that the Congress in 1978 assumed that the sites that ultimately were remediated under FUSRAP, if they were found to have mill tailings, were to be included in Title I as a result of the one-year window provision provided by that title. Consistent with that assumption, the Congress probably viewed the universe of mill tailings sites as essentially consisting of Title I sites and active mill operations. It is not surprising, therefore, that casual statements of the sort cited by the Commission appear in the record.

Third, the flavor of those statements does not suggest an intent to restrict the Commission's authority. For example, the full paragraph from the House Commerce Committee report from which one of the references cited in the Commission's response was taken reads as follows:

> The lack of any control over these inactive sites under the 1954 act and other laws to require clean up of these sites is the principal basis for committee action to authorize this remedial program. This situation does not exist at active mill tailings sites. Those sites, even those with tailings derived from Federal contracts, are subject to NRC regulation as a result of the enactment of NEPA in 1970. The NRC can require these operators, as a condition to the granting of a license, to take steps to stabilize these piles, although the control is not adequate. Indeed, the NRC testified that it has obtained commitments from some licensees to cope with the problem to some degree. This bill will provide additional authority to effectively control tailings at these active and all future sites.

(Emphasis indicates the statement that was quoted in the NRC's response.) There is no indication here of an intent to limit the Commission, or to insist that the "additional authority" should never be used at sites that are not active. Such statements should be contrasted with the clear statutory mandates of sections 81, 84 and 11e.(2), which unambiguously provide that the Commission is not to be limited in its jurisdiction over mill tailings, as well as with the substantial legislative history indicating that UMTRCA's coverage was to be comprehensive.[12]

---

performance of these Title I responsibilities. Interior Committee Report at 21; Commerce Report at 45.

[12] See e.g. Commerce Report at 29, ("The committee is convinced that all tailings pose a potential and significant radiation health hazard to the public."); Interior Committee Report at 15, ("The Commission . . . is the lead agency in regulation, oversight and management of uranium mill tailings-related activities. It is one of the major purposes of [UMTRCA] to clarify and reinforce these Commission responsibilities, with respect to uranium mill tailings at both

Conclusion

In sum, neither the statutory language of section 83 nor UMTRCA's legislative history is in conflict with the view that the NRC's authority under UMTRCA relates to all mill tailings. Some further observations in support of this view are relevant here. The first is that Envirocare is not aware of any statutes that are drafted in the way the NRC now reads UMTRCA. It seems fair to ask whether any other statute exists where the controlling definition is drafted in unambiguously broad terms and where the reader is asked to import major limitations on that definition from other sections of the act that do not purport to modify the definition. It may further be asked whether the UMTRCA Congress, notwithstanding the statute's comprehensive objectives, would have denied the Commission authority (i) over all mill tailings on sites whose licenses terminated between 1978 and 1981, (ii) over all pre-1978 mill tailings on sites rejected by FUSRAP, (iii) over all imported mill tailings, and (iv) over all pre-1978 tailings on government sites, whether or not such sites were acquired by private parties prior to remediation. One must further ask whether the Congress would have endorsed legislation that would have left the NRC in regulatory limbo for three years in the manner referred to in the comment on the previous response. Finally one must ask whether the Congress, in a statute designed to curtail dual regulation, EPA regulation, and state regulation of mill tailings would have endorsed a statute where these objectives were essentially thwarted.[13] The Commission's interpretation requires one to accept that all of these unlikely and unfortunate circumstances came together in UMTRCA, notwithstanding that the actual language of the statute and the predominant themes of its legislative history clearly indicate just the opposite. Any such interpretation should be firmly rejected.

---

active and inactive sites.") (Emphasis added.) For a more extensive discussion of the legislative history relating to UMTRCA's comprehensiveness, see Envirocare's Reply to the Supplemental Response of Envirosafe Services of Idaho, Inc. and the Environmental Technology Council and to the U.S. Army Corps of Engineers Letter Response at 23-38 (filed Sept. 13, 2000 with NRC) ("Reply Brief").

[13] See discussion in Reply Brief at 45-49 and comments on subsequent response.

SENATOR BAUCUS'S AND SENATOR GRAHAM'S QUESTION 2

You have taken the position that the NRC does not have authority over the disposal of FUSRAP mill tailings. Does that mean that you cannot regulate the disposal of such material even at a site that is otherwise regulated by the NRC? Please explain your reasoning on this matter.

The response to this question raises important safety concerns. Under the NRC's current position, the Commission's authority over FUSRAP mill tailings disposed of at NRC-licensed sites is necessarily subject to significant limitations. That is because if pre-1978 FUSRAP tailings are not licensable material, they cannot be regulated as such, whether or not they are sent to an NRC-licensed site. If we understand the response correctly, it is consistent with this view. The response observes that in the circumstances identified the NRC could impose its Part 20 dose limits against the licensee. It does not claim, however, that all other regulations that are significant for safety purposes could be imposed with respect to the FUSRAP material.

In Envirocare's view, the imposition and enforcement of a number of such regulations would be beyond the NRC's authority. For example, if pre-1978 material is brought on-site and, as is often the case, occupies a portion of the site separate from the site's post-1978 material, the radon flux standard of Criterion 6 of Part 40's Appendix A could not be imposed by NRC with respect to the pre-1978 material. The same can be said of other standards of safety significance, such as the ground water protection requirements of Appendix A's Table 5, the ALARA requirements of 10 CFR § 20.1101(b), the storage and control requirements of 10 CFR §§ 20.1801 and 20.1802, the posting requirements of 10 CFR § 20.1902(a), and the long-term surveillance plan requirements of the general license issued under 10 CFR § 40.28. While these are all requirements that the NRC has determined are necessary for the protection of public health and safety where post-1978 material is concerned, the pre-1978 material would be free of such requirements. This would be the case notwithstanding that the pre-1978 material in question would be likely to have radiation levels that are on the higher end of the spectrum for such material. As the Commission is aware, the policy of the Army Corps of Engineers has been to send material with higher than normal radiation levels to NRC-licensed sites.[1]

This does not mean, of course, that the material would not be subject to any alternative regulatory regime. From all indications, however, no federal regulation would be available. The Environmental Protection Agency has made clear that it does not regulate pre-1978 mill tailings, since, whatever the NRC's position may be, the EPA views this material as Atomic Energy Act byproduct material.[2] The NRC's position, accordingly, will leave the regulation of this material to state authorities, without regard to the level of competence and experience such authorities

---

[1] Needless to say, none of these same safety standards would apply at sites that are wholly unlicensed by the NRC, such as the RCRA disposal sites to which the Corps is now sending FUSRAP tailings with lower levels of radiation.

[2] Attachment to letter from EPA to Hon. Clint Stennett, Minority Leader, Idaho State Senate, at 3 (June 26, 2000).

may have demonstrated with respect to the regulation of nuclear materials. Some of these states may have no Agreement State relationship of any sort with the NRC.[3]

The NRC's position also will result in a related undesirable consequence: that of dual regulation of disposal sites. In the scenario discussed — where pre-1978 and post-1978 material exist on the same site in separate identifiable locations — the site owner typically will be subjected to two different sets of regulations and requirements. The Commission has recently considered dual regulation scenarios of this sort in other decision-making contexts and has generally regarded them as undesirable.[4] Moreover, it is clear that it was a principal objective of UMTRCA to avoid both dual regulation and state regulation of mill tailings.[5] That the NRC's position will produce just the sort of regulation that the statute was designed to avoid is one of the many anomalous consequences associated with the Commission's position.

An additional safety concern also warrants the Commission's attention. That concern relates to sites that were rejected by DOE for FUSRAP because of "hold harmless" clauses in the contracts under which the relevant waste was produced. These clauses, which arguably freed DOE from responsibility for the clean-up of such waste, have led to the denial of a significant number of FUSRAP applications. It is not clear that the sites involved will ever be remediated. A series of articles in USA Today recently discussed these developments.[6] While we are not in a position to evaluate the dimensions or severity of the risks involved, what is clear is that these sites will not be regulated by the NRC under its current position. That prospect provides an additional safety-related reason for reexamining the NRC's interpretation.

---

[3] The Corps, of course, could evaluate a state's radiation protection program and its competence to administer that program before sending this material to any given NRC-licensed site. There is serious question, however, whether the Corps institutionally is the appropriate agency to make these judgments. There can be no doubt that the Congress that enacted UMTRCA would not have thought so.

[4] See Commission vote sheets for SECY-99-0277 and SECY-99-0012.

[5] See Reply Brief at 48-49.

[6] Peter Eisler, Little Time For Safety As Arms Race Runs At Full Speed, USA Today, Sept. 6, 2000, at 16A; Peter Eisler, Official Sites Got Attention; Private Sites Stayed Private, USA Today, Sept. 6, 2000, at 16A.

57 FR 20525 printed in FULL format.

NUCLEAR REGULATORY COMMISSION
AGENCY: Nuclear Regulatory Commission.

57 FR 20525

May 13, 1992

Uranium Mill Facilities, Request for Public Comments on Revised Guidance on Disposal of Non-Atomic Energy Act of 1954, Section 11e.(2) Byproduct Material in Tailings Impoundments and Position and Guidance on the Use of Uranium Mill Feed Materials Other Than Natural Ores
ACTION: Request for public comment.

SUMMARY: The Nuclear Regulatory Commission (NRC) is soliciting public comment on two guidance documents: "Revised Guidance on Disposal of Non-Atomic Energy Act of 1954, section 11e.(2) Byproduct Material in Tailings Impoundments" and "Position ad Guidance on the Use of Uranium Mill Feed Materials Other Than Natural Ores;" along with the associated staff analyses.

DATES: The comment period expires June 12, 1992.

ADDRESSES: Send written comments to Chief, Rules and Directives Review Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555, or hand deliver to 7920 Norfolk Avenue, Bethesda, MD, between 7:45 a.m. and 4:15 p.m. on Federal workdays.
FOR FURTHER INFORMATION CONTACT: Myron Fliegel, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555; telephone (301) 504-2555.

TEXT: SUPPLEMENTARY INFORMATION:

Discussion

NRC staff has prepared a revision to its licensing guidance, issued July 27, 1988, on the disposal of material other than that defined in section 11e.(2) of the Atomic Energy Act of 1954 (AEA), as amended, in uranium mill tailings impoundments (Part A of the Supplementary Information). The staff has also prepared new licensing guidance on the processing of feed materials other than natural ores in uranium mills (Part B of the Supplementary Information). In developing the guidance, staff analyzed the policy and legal issues involved for each guidance document. In order to solicit input all interested parties on the issues associated with these guidance documents, the NRC is soliciting comments from the public, the Environmental Protection Agency, NRC Agreement States, and regional low-level waste compacts. Comments received will be considered in deciding whether the guidance documents should be revised.

In the guidance documents and associated staff analyses, the term "non-11e.(2) byproduct material" is used to refer to radioactive waste that is similar in physical and radiological characteristics (for example, low specific activity) to byproduct material, as defined in Section 11e.(2) of the AEA but

does not meet the definition in that section because it is not derived from ore processed primarily for its source material content.

The staff analyses in Parts A and B contain additional definitions and extensive background information necessary to understand the summary guidance documents. The reader should consult the analyses for the terms and issues presented in context.

Part A -- Revised Guidance on Disposal of Non-Atomic Energy Act of 1954, Section 11e.(2) Byproduct Material in Tailings Impoundments

1. In reviewing licensee requests for the disposal of source material wastes that have radiological characteristics comparable to those of Atomic Energy Act (AEA) of 1954, section 11e.(2) byproduct material (hereafter designed as "11e(2) byproduct material") in tailings impoundments, staff will follow the guidance set forth below. Licensing of the receipt and disposal of such non-AEA, section 11e.(2) byproduct material (hereafter designated as "non-11e.(2) byproduct material") should be done under 10 CFR Part 40.

2. Naturally occurring and accelerator produced material waste shall not be authorized for disposal in an 11e.(2) byproduct material impoundment.

3. Special nuclear material and Section 11e.(1) product material waste should not be considered as candidates for disposal in a tailings impoundment, without compelling reasons to the contrary. If staff believes that such material should be disposed of in a tailings impoundment in a specific instance, a request for approval by the Commission should be prepared.

4. The 11e.(2) licensee must demonstrate that the material is not subject to applicable Resource Conservation and Recovery Act regulations or other U.S. Environmental Protection Agency standards for hazardous or toxic wastes prior to disposal.

5. The 11e(2) licensee must demonstrate that there are no Comprehensive Environmental Response, Compensation and Liability Act issues related to the disposal of the non-11e(2) byproduct material.

6. The 11e.(2) licensee must demonstrate that there will be no significant environmental impact from disposing of this material.

7. The 11e.(2) license must demonstrate that the proposed disposal will not compromise the reclamation of the tailings impoundment by demonstrating compliance with the reclamation and closure criteria of appendix A of 10 CFR part 40.

8. The 11e.(2) licensee must provide documentation showing approval by the Regional Low-Level Waste Compact in whose jurisdiction the waste originates as well as approval by the Compact in whose jurisdiction the disposal site is located.

9. The Department of Energy should be informed of the Nuclear Regulatory Commission findings and proposed action, with an opportunity to provide comments within 30 days, before granting the license amendment to the 11e.(2) licensee.

10. The mechanism to authorize the disposal of non-11e.(2) byproduct material in a tailings impoundment is an amendment to the mill license under 10 CFR Part 40, authorizing the receipt of the material and its disposal. Additionally, an exemption to the requirements of 10 CFR Part 61, under the authority of @ 61.6, must be granted. The license amendment and the @ 61.6 exemption should be supported with a staff analysis paper addressing the issues discussed in this guidance.

NRC Staff Analysis of Disposal of Non-Atomic Energy Act of 1954, Section 11e.(2) Byproduct Material in Tailings Impoundments

1. Introduction

Recently, the Nuclear Regulatory Commission (NRC) received several requests to allow activities other than the normal processing of native uranium ore at licensed uranium milling facilities. We have, in the past, received, and, in some cases, approved, similar requests. These requests have fallen into two categories. The first category of requests is to allow the processing of feedstock material that is not usually thought of as ore, for the extraction of uranium, and then dispose of the resulting wastes and tailings in the facility's tailings pile. The second category of requests is to allow the direct disposal of non-Atomic Energy Act (AEA) of 1954, section 11e.(2) byproduct material n1 [hereafter designated as "non-11e.(2) byproduct material"], that was not generated onsite, into tailings piles.

n 1 For the purposes of this analysis, the term "non-11e.(2) byproduct material" will be used to refer to radioactive waste that is similar to byproduct material, as defined in the AEA in section 11e.(2), but is not legally considered to be 11e.(2) byproduct material.

In assessing these requests, the staff has raised two policy concerns related to tailings piles. The first concern is that the requested activity might result in complicated, dual, or even multiple regulation of the tailings pile, and the second concern is that the requested activity might jeopardize the ultimate transfer to the United States Government, for perpetual custody and maintenance, of the reclaimed tailings pile.

This analysis addresses the second category of requests, that is, requests to dispose of non-11e.(2) byproduct material in tailings piles. Issues relating to such proposals requesting regulatory consideration of commingling of tailings with other radioactive wastes are discussed. This analysis is limited to options involving commingling with existing tailings impoundments.

2. Background

The Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978 amended the AEA to specifically include uranium and thorium mill tailings and other wastes from the process as radioactive material to be licensed by NRC. Specifically, the definition of byproduct material was revised in Section 11e.(2) of the AEA, to include ". . . the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content."

The definition of byproduct material n2 in Section 11e.(2) of the AEA includes all the wastes resulting from the milling process, not just the radioactive components. In addition, Title II of UMTRCA amended the AEA to explicitly exclude the requirement for the Environmental Protection Agency (EPA) to permit 11e.(2) byproduct material under the Resource Conservation and Recovery Act (RCRA). The designation of 11e.(2) byproduct material contrasts significantly with the situation for source material n3 and other radioactive materials controlled under the authority of the AEA. This possibility for dual regulation by both NRC and EPA can become an issue when dealing with mixed hazardous wastes. As a result of UMTRCA, NRC amended 10 CFR Part 40 to regulate the uranium and thorium tailings and wastes from the milling process. Thus, under normal operation, all the tailings and wastes in an NRC or Agreement State licensed mill producing uranium or thorium are classified as "11e.(2) byproduct material," and are disposed of in tailings piles regulated under Part 40. They are not subject to EPA regulation, under RCRA. However, the EPA Clean Air Act regulations still result in direct EPA permit authority over the mill tailings, whether or not they are commingled with non-11e.(2) byproduct material waste.

n 2 Henceforth, byproduct material as defined in Section 11e.(2) of the AEA will be referred to as "11e.(2) byproduct material."

n 3 Except in the case of source material ore, source material consists only of the radioactive components of the waste, that is; uranium, thorium, or any combination of the two [10 CFR 40.4(h)].

The UMTRCA also required and provided for long-term custody and surveillance of the byproduct material and the land use for its disposal. The Department of Energy (DOE) is the Federal agency currently designated as the "custodial agency" by the AEA. However, the UMTRCA specifically referred only to 11e.(2) byproduct material. UMTRCA contains no provision allowing for the transfer of custody or title, and hence for eventual long-term custody and surveillance of other material, even if the material were no more radioactive or toxic than the uranium or thorium tailings themselves.

3. The Category of Requests for Commingled Disposal To Be Addressed

Some licensees have proposed to directly dispose of radioactive wastes in existing uranium mill tailings sites. The materials vary from tailings from extraction processes for metals and rare-earth metals (such as copper, tantalum, columbium, zirconium) to spent resins from water-treatment processes. However, because these materials did not result from the extraction or concentration of uranium or thorium from ore, they are not 11e.(2) byproduct material. Many of these "orphaned" wastes have elevated concentrations of source material, and unless otherwise exempted, require licensed control, if the materials exceed the 0.05-percent licensable (content of source material by weight) criterion in 10 CFR Part 40. Some of the wastes proposed for commingling contain radioactive material, not regulated by NRC, that classify as naturally-occurring and accelerator-produced radioactive material (NARM) and as such cannot be easily disposed of. In most of the proposals the staff has seen, disposal of these materials in tailings impoundments would not significantly increase the effect on the public health, safety, and environment. Because of the relatively large volumes of these wastes, low-level waste disposal options are limited. These wastes are similar to tailings in volume, radioactivity, and toxicity. Therefore, some waste producers see the mill tailings disposal sites as

providing an economical option for such disposal.

4. Types of Wastes Being Proposed for Disposal Into Tailings Piles

The NRC and the Agreement States continue to receive requests for the direct disposal of non-11e.(2) byproduct material into uranium mill tailings piles. The following general categories of non-11e.(2) byproduct material illustrate the requests submitted to NRC and the Agreement States for disposal into uranium mill tailings piles licensed under authority established by title II of UMTRCA:

4.1 Mine Wastes

To mine uranium or other source material ore from underground or open-pit mines, operators frequently need to dewater the mine cavities. This results in quantities of mine water with suspended or dissolved constituents, some of which are source material. After processing the mine water to satisfy National Pollution Discharge Elimination System or other release requirements, the resultant clean mine water is then discharged offsite. In some cases, the resulting water-treatment filter-cake or sludge residues exceed the 0.05-percent licensable limit for source material. These residues do not satisfy the definition of 11e.(2) byproduct material, because they do not result from the extraction or concentration of uranium or thorium from ore.

NRC and the Agreement States have been contacted by licensees and waste generators that desire to dispose of such filter-cake or sludge residue directly into the tailings piles at licensed uranium mill tailings sites. NRC has indicated that such material does not constitute 11e.(2) byproduct material.

4.2 Secondary Process Wastes

Frequently, natural ores that are processed for rare-earth or other metals have significant concentrations of radioactive elements. Examples include copper, zirconium, and vanadium ores. Sometimes the uranium is captured in a side-stream recovery operation, in which uranium is precipitated out of the pregnant solution, before or after the rare earth or other metal. Although this side-stream recovery operation is licensed by NRC, the tailings (which consist of the crushed depleted ore and the depleted solution after recovery of metals and rare earths) are not 11e.(2) byproduct material. This is because the ore was not processed primarily for its source material content, but for the rare earth or other metal. If the tails contain greater than 0.05 percent uranium and thorium, they would be source material and would thus be licensable and have to be disposed of in compliance with NRC regulations. NRC has received requests from NRC and Agreement State licensees to dispose of such tailings (resulting from processes to extract other metals) into licensed uranium mill tailings piles.

4.3 Formerly Utilized Sites Remedial Action Program (FUSRAP)

These sites primarily processed material, such as monazite sands, to extract thorium for commercial applications. Government contracts were issued for thorium source material used in the Manhattan Engineering District and early Atomic Energy Commission programs. Wastes resulting from that processing and disposed of at these sites would qualify as 11e.(2) byproduct material. However, it is not clear that all the contaminated material at these sites result from

processing of ore for thorium. At some sites there was also processing for rare earths and other metals. The DOE, which accepts responsibility for the FUSRAP materials, is investigating options for disposal and control of these materials. DOE estimates that a total of 1.7 million cubic yards of material is located at sites in 13 States. Recent proposals have considered the transportation of FUSRAP materials from New Jersey to tailing piles at uranium mills in other States, such as Utah, Washington, and Wyoming.

4.4 NARM

These wastes result from a wide range of operations, but are not generally regulated by the AEA. Past requests for disposal in uranium mill tailing ponds have included contaminated resins from ion-exchange well-water purifying operations. NRC has also received inquiries regarding the disposal of construction scrap and radium-contaminated soil from old commercial operations. The individual States usually administer the regulatory responsibility over NARM, but many other Federal agencies have jurisdictional responsibilities related to NARM. These include EPA, the Consumer Product Safety Commission, the Department of Health and Human Services, and the Department of Labor. There is a State-licensed NARM disposal facility in Clive, Utah, licensed to Envirocare of Utah, Inc.

Two common elements run through most of the requests we have received for direct disposal of non-11e.(2) byproduct material in tailings piles: the material is of low specific-activity, and the material is physically similar to 11e.(2) byproduct material. Most of the requests are for bulk material like soil, crushed rock, or sludges, contaminated with source material in relatively low concentrations.

5. Previous Staff Guidance

In response to a request from Region IV, the Director of the Office of Nuclear Material Safety and Safeguards (NMSS) provided guidance for addressing requests to allow the disposal of non-11e.(2) byproduct material in licensed mill tailings impoundments. The staff considered that the types of material proposed for such disposal could be separated into two categories: (1) NARM wastes; and (2) wastes generated by operations regulated under the AEA.

In the guidance, the staff concluded that it would not approve a policy of allowing disposal of NARM wastes in tailings impoundments. A major concern was that NRC did not have authority to regulate NARM. If States or EPA became involved in regulation of NARM, a situation with duplicative jurisdiction with respect to the commingled radioactive materials could be created. Furthermore, the Commission's authority, under section 84c of the AEA, to approve alternatives to requirements, if the NARM wastes were to violate standards, would be impaired.

The staff viewed the other category, wastes generated by operations regulated under the AEA, as potentially acceptable in a mill tailings impoundment. Each such proposal should be considered on a case-specific basis. The guidance identified four findings that would have to be made before NRC would authorize such disposal.