Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381

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WBL-20-004

10 CFR 50 Appendix H

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
Facility Operating License No. NPF-96
NRC Docket No. 50-391

Subject: **Watts Bar Nuclear Plant (WBN) Unit 2 – Analysis of Capsule U from Watts Bar Unit 2 Reactor Vessel Radiation Surveillance Program**

In accordance with 10 CFR 50, Appendix H.IV.A, TVA is submitting the Technical Report WCAP-18518-NP, "Analysis of Capsule U from the Watts Bar Unit 2 Reactor Vessel Radiation Surveillance Program," Revision 0, dated March 2020. Westinghouse Electric Company LLC developed this report, which details the test results of Surveillance Capsule U withdrawn in the End of Cycle (EOC) 2 refueling outage after 2.0 effective full power years (EFPY) of operation. This WCAP is provided in the enclosure.

The reactor vessel capsule fluence values for WBN Unit 2 Capsule U were determined using the RAPTOR-M3G computer code as described in WCAP-18124-NP-A, Revision 0 "Fluence Determination with RAPTOR-M3G and FERRET." Although this WCAP has been found acceptable by NRC for referencing in licensing applications, it has not yet been incorporated into the WBN licensing basis. A license amendment to this effect is planned for submission by August 2020.

There are no regulatory commitments contained in this letter. Please direct any questions concerning this matter to Tony Brown, WBN Licensing Manager, at (423) 365-7720.

Respectfully,

Anthony L. Williams IV
Site Vice President
Watts Bar Nuclear Plant

Enclosure: WCAP-18518-NP, "Analysis of Capsule U from the Watts Bar Unit 2 Reactor Vessel Radiation Surveillance Program," Revision 0, dated March 2020.

cc (Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant

WCAP-18518-NP
Revision 0

March 2020

Analysis of Capsule U from the Watts Bar Unit 2 Reactor Vessel Radiation Surveillance Program

WCAP-18518-NP
Revision 0

Analysis of Capsule U from the Watts Bar Unit 2 Reactor Vessel Radiation Surveillance Program

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EXECUTIVE SUMMARY

The purpose of this report is to document the testing results of surveillance Capsule U from Watts Bar Unit 2. Capsule U was removed at 2.0 effective full-power years (EFPY) and post-irradiation mechanical tests of the Charpy V-notch and tensile specimens were performed. A fluence evaluation utilizing the neutron transport and dosimetry cross-section libraries was derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Capsule U received a fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) after irradiation to 2.0 EFPY. The peak clad/base metal interface vessel fluence after 32 EFPY (end-of-license) of plant operation is projected to be 1.94 x 10^{19} n/cm^2 (E > 1.0 MeV).

This evaluation led to the following conclusions: (1) The measured shift in the 30 ft-lb transition temperature of the surveillance forging and weld materials contained in Watts Bar Unit 2 Capsule U are approximately equal to or less than the Regulatory Guide 1.99, Revision 2 [Ref. 1] predictions. (2) The measured percent decreases in upper-shelf energy for the surveillance forging and weld materials contained in Watts Bar Unit 2 Capsule U are typically less than the Regulatory Guide 1.99, Revision 2 [Ref. 1] predictions. The exception is Intermediate Shell Forging 05 in the tangential direction, which experienced a higher than predicted decrease in the upper-shelf energy. (3) The Watts Bar Unit 2 surveillance weld (Heat # 895075) data and sister-plant data are judged to be credible. This credibility evaluation can be found in Appendix D.

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

1 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance Capsule U, the first capsule removed and tested from the Watts Bar Unit 2 reactor pressure vessel, led to the following conclusions:

- Charpy V-notch test data were plotted using a symmetric hyperbolic tangent curve-fitting program. Appendix C presents the CVGRAPH, Version 6.02, Charpy V-notch plots for Capsule U, along with the program baseline data.

- Capsule U received an average fast neutron fluence (E > 1.0 MeV) of 6.04 x 10^{18} n/cm^2 after 2.0 effective full-power years (EFPY) of plant operation.

- Irradiation of the reactor vessel Intermediate Shell Forging 05 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major rolling direction (tangential orientation), resulted in an irradiated 30 ft-lb transition temperature (T_{30}) of -15.6°F. This results in a 30 ft-lb transition temperature increase of 26.7°F for the tangentially oriented specimens.

- Irradiation of the reactor vessel Intermediate Shell Forging 05 Charpy specimens, oriented with the longitudinal axis of the specimen perpendicular to the major rolling direction (axial orientation), resulted in an irradiated 30 ft-lb transition temperature (T_{30}) of -20.5°F. This results in a 30 ft-lb transition temperature increase of 21.3°F for the axially oriented specimens.

- Irradiation of the Surveillance Program Weld Material (Heat # 895075) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature (T_{30}) of -25.1°F. This results in a 30 ft-lb transition temperature increase of 32.6°F.

- Irradiation of the Heat-Affected Zone (HAZ) Material Charpy specimens resulted in an irradiated 30 ft-lb transition temperature (T_{30}) of -103.4°F. This results in a 30 ft-lb transition temperature reduction of 1.6°F. Note that physically a reduction in T_{30} should not occur.

- The average upper-shelf energy of Intermediate Shell Forging 05 (tangential orientation) resulted in an average energy decrease of -45 ft-lb after irradiation. This decrease results in an irradiated average upper-shelf energy of 130 ft-lb for the tangentially oriented specimens.

- The average upper-shelf energy of Intermediate Shell Forging 05 (axial orientation) resulted in an average energy decrease of -5 ft-lb after irradiation. This decrease results in an irradiated average upper-shelf energy of 105 ft-lb for the axially oriented specimens.

- The average upper-shelf energy of the Surveillance Program Weld Material (Heat # 895075) Charpy specimens resulted in an average energy decrease of -9 ft-lb after irradiation. This decrease results in an irradiated average upper-shelf energy of 135 ft-lb for the weld metal specimens.

- The average upper-shelf energy of the HAZ Material Charpy specimens resulted in an average energy decrease of -12 ft-lb after irradiation. This decrease results in an irradiated average upper-shelf energy of 118 ft-lb for the HAZ Material.

- Comparisons of the measured 30 ft-lb shift in transition temperature values and upper-shelf energy decreases to those predicted by Regulatory Guide 1.99, Revision 2 [Ref. 1] for the Watts Bar Unit 2 reactor vessel surveillance materials are presented in Table 5-10.

- Based on the credibility evaluation presented in Appendix D, the Watts Bar Unit 2 surveillance weld material (Heat # 895075) is credible.

- The maximum calculated 32 EFPY (end-of-license) neutron fluence (E > 1.0 MeV) for the Watts Bar Unit 2 reactor vessel beltline using the Regulatory Guide 1.99, Revision 2 [Ref. 1] attenuation formula (i.e., Equation # 3 in the Guide) is as follows:

  Calculated (32 EFPY): Vessel peak clad/base metal interface fluence* = 1.94×10^{19} n/cm^2
  Vessel peak quarter-thickness (1/4T) fluence = 1.17×10^{19} n/cm^2

  *This fluence value is documented in Table 6-5.

2 INTRODUCTION

This report presents the results of the examination of Capsule U, the first capsule removed and tested in the continuing surveillance program, which monitors the effects of neutron irradiation on the Tennessee Valley Authority (TVA) Watts Bar Unit 2 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Watts Bar Unit 2 reactor pressure vessel materials was designed and recommended by Westinghouse Electric Company LLC. A detailed description of the surveillance program is contained in WCAP-9455 [Ref. 2] "Tennessee Valley Authority Watts Bar Unit No. 2 Reactor Vessel Radiation Surveillance Program." The surveillance program covers the 40-year design life of the reactor pressure vessel and is based on ASTM E185-73 [Ref. 3], "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." Capsule U was removed from the reactor after 2.0 EFPY of exposure and shipped to the Westinghouse Churchill Laboratory Services, where the post-irradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

This report summarizes the testing and post-irradiation data obtained from surveillance Capsule U removed from the Watts Bar Unit 2 reactor vessel and presents 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 SA508 Class 2 (base material of the Watts Bar Unit 2 reactor pressure vessel beltline) are well documented in the literature. Generally, low-alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness during high-energy irradiation.

A method for ensuring the integrity of reactor pressure vessels has been presented in "Fracture Toughness Criteria for Protection Against Failure," Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code [Ref. 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 American Society of Testing and Materials (ASTM) E208-06 [Ref. 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 (axial) to the major rolling direction of the forging. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{Ic} curve) which appears in Appendix G to Section XI of the ASME Code [Ref. 4]. The K_{Ic} curve is a lower bound of static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{Ic} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined using these allowable stress intensity factors.

RT_{NDT} and, in turn, the operating limits of nuclear power plants, are 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, are monitored by a reactor vessel surveillance program, such as the Watts Bar Unit 2 reactor vessel radiation surveillance program, in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are 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 (initial RT_{NDT} + M + ΔRT_{NDT}) is used to index the material to the K_{Ic} curve and, in turn, to set operating limits for the nuclear power plant that take into account the effects of irradiation on the reactor vessel materials.

4 DESCRIPTION OF PROGRAM

Six surveillance capsules for monitoring the effects of neutron exposure on the Watts Bar Unit 2 reactor pressure vessel core region (beltline) materials were inserted in the reactor vessel prior to initial plant startup. The six capsules were positioned in the reactor vessel, as shown in Figure 4-1, between the neutron shield pads and the vessel wall, at various azimuthal locations for this Westinghouse four-loop plant. The vertical center of the capsules is opposite the vertical center of the core. The capsules contain specimens made from the following:

- Intermediate Shell Forging 05 (tangential orientation), Heat # 527828
- Intermediate Shell Forging 05 (axial orientation), Heat # 527828
- Weld metal fabricated with weld wire heat no. 895075 with type Grau L.O. (LW320) flux, lot P46 which is equivalent to the heat number, Flux Type, and Flux Lot number used in the actual fabrication of the intermediate shell to lower shell circumferential weld seam
- Weld heat-affected zone (HAZ) material of Intermediate Shell Forging 05

Test material obtained from the Intermediate Shell Forging 05 (after thermal heat treatment and forming of the forging) was taken at least one forging thickness from the quenched edges of the forging. All test specimens were machined from the ¼ thickness location of the forging after performing a simulated post-weld stress-relieving treatment on the test material. Test specimens were also removed from the weld and heat-affected zone metal of stress-relieved weldments joining Intermediate Shell Forging 05 and Lower Shell Forging 04. All heat-affected zone specimens were obtained from the weld heat-affected zone of Intermediate Shell Forging 05.

Charpy V-notch impact specimens from Intermediate Shell Forging 05 were machined in the tangential orientation (longitudinal axis of the specimen parallel to the major rolling direction) and also in the axial orientation (longitudinal axis of the specimen perpendicular to the major rolling direction). The core-region weld Charpy impact specimens were machined from the weldment such that the long dimension of each Charpy specimen was perpendicular to the weld direction. The notch of the weld metal Charpy specimens was machined such that the direction of crack propagation in the specimen was in the welding direction.

Tensile specimens from Intermediate Shell Forging 05 were machined both in the tangential and axial orientations. Tensile specimens from the weld metal were oriented perpendicular to the welding direction.

Compact tension test specimens from Intermediate Shell Forging 05 were machined in both the tangential and axial orientations. Compact tension test specimens from the weld metal were machined perpendicular to the weld direction with the notch oriented in the direction of the weld. All specimens were fatigue precracked according to ASTM E399 [Ref. 6].

All six capsules contain dosimeter wires of pure iron, copper, nickel, and aluminum-0.15 weight percent cobalt (cadmium-shielded and unshielded). In addition, cadmium-shielded dosimeters of Neptunium (^{237}Np) and Uranium (^{238}U) were placed in the capsules to measure the integrated flux at specific neutron energy levels.

The capsules contain thermal monitors made from two low-melting-point eutectic alloys, which were sealed in Pyrex tubes. These thermal monitors were located in three different positions in the capsule. These thermal monitors are used to detect the maximum temperature attained by the test specimens during irradiation. The composition of the two eutectic alloys and their melting points are as follows:

2.5% Ag, 97.5% Pb	Melting Point: 579°F (304°C)
1.75% Ag, 0.75% Sn, 97.5% Pb	Melting Point: 590°F (310°C)

The chemical composition and the heat treatment of the various mechanical specimens in Capsule U are presented in Table 4-1 and Table 4-2, respectively. The data in the tables were obtained from the original surveillance program report, WCAP-9455 [Ref. 2], Appendix A.

Capsule U was removed after 2.0 EFPY of plant operation. This capsule contained Charpy V-notch specimens, pre-cracked bend bar specimens, compact tension specimens, tensile specimens, dosimeters, and thermal monitors.

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

Table 4-1 Chemical Composition (wt. %) of the Watts Bar Unit 2 Reactor Vessel Surveillance Materials (Unirradiated)

Element	Intermediate Shell Forging 05[(a)]		Surveillance Weld Metal[(b)]	
			Westinghouse[(c)]	Rotterdam[(d)]
C	0.21	0.22	0.060	0.069
S	0.012	0.011	0.008	0.010
Co	< 0.01	0.007	0.015	-----
Cu	0.05	0.07	0.016	0.05
Si	0.28	0.26	0.21	0.22
Mo	0.55	0.60	0.52	0.56
Ni	0.78	0.80	0.69	0.70
Mn	0.72	0.68	1.80	1.97
Cr	0.28	0.32	0.020	0.05
V	< 0.01	0.01	< 0.001	-----
P	0.012	0.011	0.015	0.010
Al	< 0.01	0.003	-----	-----

Notes:

(a) All analyses were conducted by Rotterdam Dockyard Company/Krupp ladle analysis.

(b) The surveillance weldment is identical to the closing girth seam weldment between forging 04 and 05. The closing seam used weld wire heat no. 895075 with type Grau L.O. (LW320) flux, lot P46, except for the 1-inch root pass at the I.D. of the vessel. This root pass used weld wire heat no. 899680 with type Grau L.O. (LW320) flux, lot P23, with an as-deposited copper and phosphorous content of 0.03 and 0.009, respectively. The surveillance weldment specimens were not removed from this root area.

(c) All analyses were performed by Westinghouse.

(d) All analyses were performed by Rotterdam Dockyard Company.

Table 4-2 Heat Treatment History of the Watts Bar Unit 2 Reactor Vessel Surveillance Materials

Material	Temperature (°F)	Time (hr)	Cooling
Intermediate shell forging 05[a]	1675-1700	4	Water-quenched
	1230-1240	6	Air-cooled
Weldment	1140 ± 25	22	Furnace-cooled
	1140 ± 25	14	Furnace-cooled

Notes:

(a) The surviellence forging also received the stress relief treatment given to the weld.

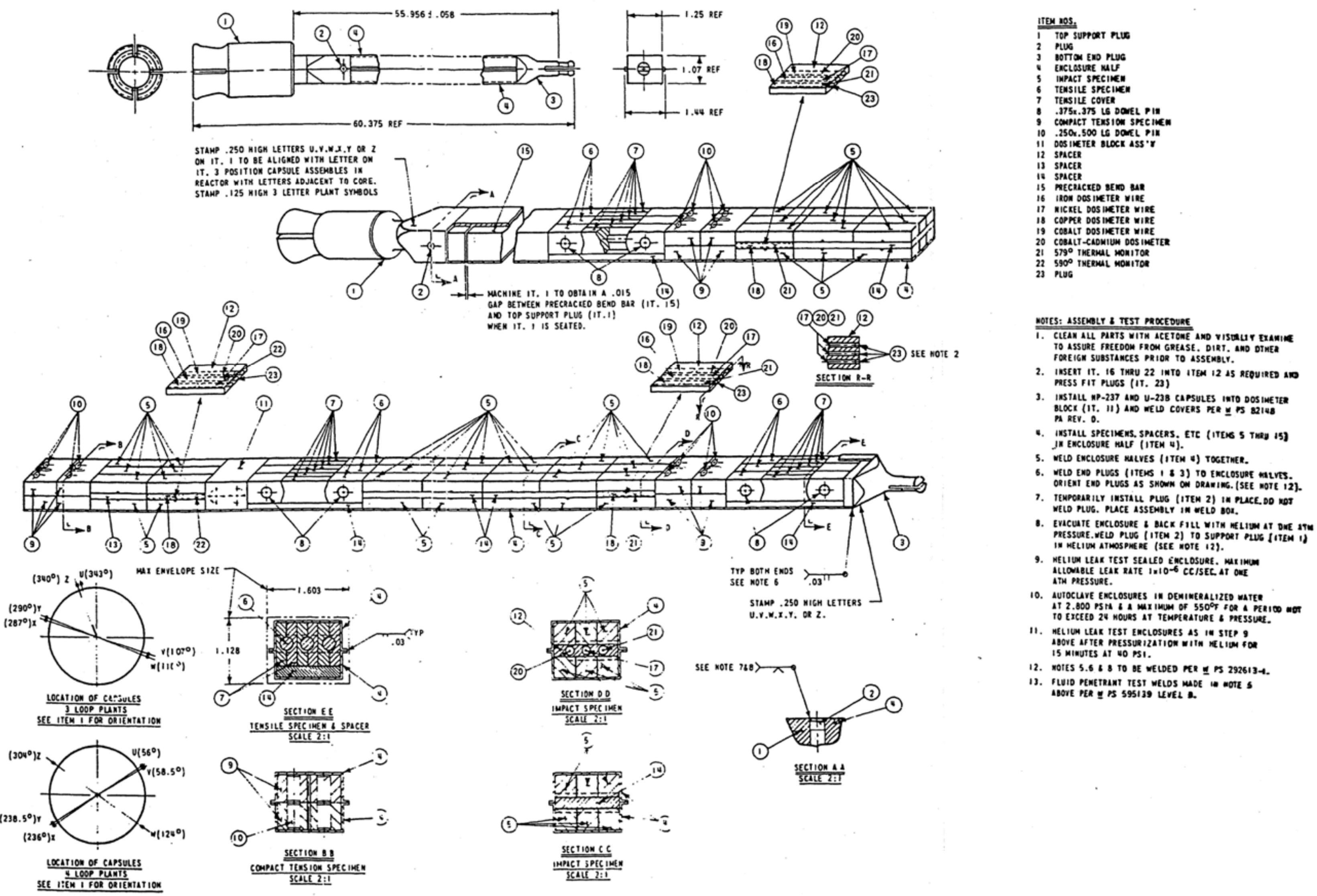

Figure 4-1 Watts Bar Unit 2 Irradiation Capsule Assembly and Reactor Vessel Location

	BEND BAR	TENSILE	COMPACT	COMPACT	CHARPY	CHARPY	CHARPY	COMPACT	COMPACT	CHARPY	CHARPY	DOSIMETER	TENSILE	CHARPY	CHARPY	CHARPY	CHARPY	CHARPY	COMPACT	COMPACT	TENSILE	
		BW18			BW90 BH90	BW87 BH87	BW84 BH84			BW81 BH81	BW78 BH78		BL18	BT90 BL90	BT87 BL87	BT84 BL84	BT81 BL81	BT78 BL78			BT18	
Z	BL6	BW17	BW24 BW23	BW22 BW21	BW89 BH89	BW86 BH86	BW83 BH83	BL24 BL23	BL22 BL21	BW80 BH80	BW77 BH77	359	BL17	BT89 BL89	BT86 BL86	BT83 BL83	BT80 BL80	BT77 BL77	BT24 BT23	BT22 BT21	BT17	
		BW16			BW88 BH88	BW85 BH85	BW82 BH82			BW79 BH79	BW76 BH76		BL16	BT88 BL88	BT85 BL85	BT82 BL82	BT79 BL79	BT76 BL76			BT16	
		BW15			BW75 BH75	BW72 BH72	BW69 BH69			BW66 BH66	BW63 BH63		BL15	BT75 BL75	BT72 BL72	BT69 BL69	BT66 BL66	BT63 BL63			BT15	
Y	BL5	BW14	BW20 BW19	BW18 BW17	BW74 BH74	BW71 BH71	BW68 BH68	BL20 BL19	BL18 BL17	BW65 BH65	BW62 BH62	358	BL14	BT74 BL74	BT71 BL71	BT68 BL68	BT65 BL65	BT62 BL62	BT20 BT19	BT18 BT17	BT14	
		BW13			BW73 BH73	BW70 BH70	BW67 BH67			BW64 BH64	BW61 BH61		BL13	BT73 BL73	BT70 BL70	BT67 BL67	BT64 BL64	BT61 BL61			BT13	
		BW12			BW60 BH60	BW57 BH57	BW54 BH54			BW51 BH51	BW48 BH48		BL12	BT60 BL60	BT57 BL57	BT54 BL54	BT51 BL51	BT48 BL48			BT12	
X	BL4	BW11	BW16 BW15	BW14 BW13	BW59 BH59	BW56 BH56	BW53 BH53	BL16 BL15	BL14 BL13	BW50 BH50	BW47 BH47	357	BL11	BT59 BL59	BT56 BL56	BT53 BL53	BT50 BL50	BT47 BL47	BT16 BT15	BT14 BT13	BT11	
		BW10			BW58 BH58	BW55 BH55	BW52 BH52			BW49 BH49	BW46 BH46		BL10	BT58 BL58	BT55 BL55	BT52 BL52	BT49 BL49	BT46 BL46			BT10	
		BW9			BW45 BH45	BW42 BH42	BW39 BH39			BW36 BH36	BW33 BH33		BL9	BT45 BL45	BT42 BL42	BT39 BL39	BT36 BL36	BT33 BL33			BT9	
W	BL3	BW8	BW12 BW11	BW10 BW9	BW44 BH44	BW41 BH41	BW38 BH38	BL12 BL11	BL10 BL9	BW35 BH35	BW32 BH32	356	BL8	BT44 BL44	BT41 BL41	BT38 BL38	BT35 BL35	BT32 BL32	BT12 BT11	BT10 BT9	BT8	
		BW7			BW43 BH43	BW40 BH40	BW37 BH37			BW34 BH34	BW31 BH31		BL7	BT43 BL43	BT40 BL40	BT37 BL37	BT34 BL34	BT31 BL31			BT7	
		BW6			BW30 BH30	BW27 BH27	BW24 BH24			BW21 BH21	BW18 BH18		BL6	BT30 BL30	BT27 BL27	BT24 BL24	BT21 BL21	BT18 BL18			BT6	
V	BL2	BW5	BW8 BW7	BW6 BW5	BW29 BH29	BW26 BH26	BW23 BH23	BL8 BL7	BL6 BL5	BW20 BH20	BW17 BH17	355	BL5	BT29 BL29	BT26 BL26	BT23 BL23	BT20 BL20	BT17 BL17	BT8 BT7	BT6 BT5	BT5	
		BW4			BW28 BH28	BW25 BH25	BW22 BH22			BW19 BH19	BW16 BH16		BL4	BT28 BL28	BT25 BL25	BT22 BL22	BT19 BL19	BT16 BL16			BT4	
		BW3			BW15 BH15	BW12 BH12	BW9 BH9			BW6 BH6	BW3 BH3		BL3	BT15 BL15	BT12 BL12	BT9 BL9	BT6 BL6	BT3 BL3			BT3	
U	BL1	BW2	BW4 BW3	BW2 BW1	BW14 BH14	BW11 BH11	BW8 BH8	BL4 BL3	BL2 BL1	BW5 BH5	BW2 BH2	354	BL2	BT14 BL14	BT11 BL11	BT8 BL8	BT5 BL5	BT2 BL2	BT4 BT3	BT2 BT1	BT2	SPACER
		BW1			BW13 BH13	BW10 BH10	BW7 BH7			BW4 BH4	BW1 BH1		BL1	BT13 BL13	BT10 BL10	BT7 BL7	BT4 BL4	BT1 BL1			BT1	

LEGEND:

BL– INTERMEDIATE FORGING 05, HEAT NO. 527828 (TANGENTIAL)
BT– INTERMEDIATE FORGING 05, HEAT NO. 527828 (AXIAL)
BW – WELD METAL
BH – HEAT-AFFECTED ZONE MATERIAL

Figure 4-2 Specimen Locations in the Watts Bar Unit No. 2 Surveillance Test Capsules

5 TESTING OF SPECIMENS FROM CAPSULE U

5.1 OVERVIEW

The post-irradiation mechanical testing of the Charpy V-notch impact specimens and tensile specimens was performed at the Westinghouse Churchill Laboratory Services Hot Cell Facility. Testing was performed in accordance with 10 CFR 50, Appendix H [Ref. 7] and ASTM Specification E185-82 [Ref. 8].

Capsule U was opened upon receipt at the hot cell laboratory. The specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in WCAP-9455 [Ref. 2]. All of the items were in their proper locations.

Examination of the thermal monitors indicated that the three temperature monitors had not melted. Based on this examination, the maximum temperature to which the specimens were exposed was less than 579°F (304°C).

The Charpy impact tests were performed per ASTM Specification E185-82 [Ref. 8] and E23-18 [Ref. 9] on a Tinius-Olsen Model 74, 358J machine. The Charpy machine striker was instrumented with an Instron®[1] Impulse system. Instrumented testing and calibration were performed to ASTM E2298-18 [Ref. 10].

The instrumented striker load signal data acquisition rate was 819 kHz with data acquired for 10 ms. From the load-time curve, the load of general yielding (F_{gy}), the maximum load (F_m) and the time to maximum load were 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 brittle fracture initiation/load at initiation of unstable crack propagation (F_{bf}). The termination load after the fast load drop is identified as the arrest load/load at end of unstable crack propagation (F_a). F_{gy}, F_m, F_{bf}, and F_a were determined per the guidance in ASTM Standard E2298-18 [Ref. 10].

The pre-maximum load energy (W_m) was determined by integrating the load-time record to the maximum load point via the instrumented Charpy software. The integrated total impact energy (W_t) is compared to the absorbed energy measured from the dial energy (KV).

Percent shear was determined from post-fracture photographs using the ratio-of-areas method in compliance with ASTM E23-18 [Ref. 9] and A370-18 [Ref. 11]. The lateral expansion was measured using a dial gage rig similar to that shown in the same ASTM Standards.

[1]Instron is a registered trademark of Instron Corporation.

Tensile tests were performed on a 250 kN Instron® screw driven tensile machine (Model 5985) per ASTM E185-82 [Ref. 8]. Testing met ASTM Specifications E8/E8M-16 [Ref. 12] for room temperature or E21-17 [Ref. 13] for elevated temperatures.

The tensile specimens were, nominally, 4.23 inches long with a 1.00 inch gage length and 0.250 inch in diameter, per WCAP-9455 [Ref. 2]. Strain measurements were made using an extensometer, which was attached to the 1.00 inch gage section of the tensile specimen. The strain rate obtained met the requirement of ASTM E8/E8M-16 [Ref. 12] and ASTM E21-17 [Ref. 13].

Elevated test temperatures were obtained with a three-zone electric resistance split-tube Instron SF-16 furnace with an 11-inch hot zone. For the elevated tests, temperature was measured by two Type N thermocouples in contact with the gage section of the specimen per ASTM E21-17 [Ref. 13]. Tensile specimens were soaked at temperature (± 5ºF) for a minimum of 20 minutes before testing. All tests were conducted in air.

The yield load, ultimate load, fracture load, uniform elongation, and elongation at fracture were determined directly from the load-extension curve. The yield strength (0.2% offset method), ultimate tensile strength, and fracture strength were calculated using the original cross-sectional area. Yield point elongation (YPE) was calculated as the difference in strain between the upper yield strength and the onset of uniform strain hardening using the methodology described in ASTM E8/E8M-16 [Ref. 12]. The final diameter and final gage length were determined from post-fracture photographs. This final diameter measurement was used to calculate the fracture stress (fracture true stress) and the percent reduction in area. The reported total elongation is the elongation at fracture.

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 U, which received a fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) in 2.0 EFPY of operation, are presented in Tables 5-1 through 5-8 and are compared with the unirradiated results as shown in Figures 5-1 through 5-12. The unirradiated capsule results were taken from WCAP-9455 [Ref. 2]. The original program unirradiated material input data, were updated using CVGRAPH, Version 6.02.

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

- Irradiation of the reactor vessel Intermediate Shell Forging 05 Charpy specimens, oriented with the tangential axis of the specimen parallel to the major rolling direction (tangential orientation), resulted in an irradiated 30 ft-lb transition temperature (T_{30}) of -15.6°F and an irradiated 50 ft-lb transition temperature (T_{50}) of 12.7°F. This results in a 30 ft-lb transition temperature increase of 26.7°F (ΔT_{30} = 26.7°F) and a 50 ft-lb transition temperature increase of 33.6°F (ΔT_{50} = 33.6°F) for the tangentially oriented specimens.

- Irradiation of the reactor vessel Intermediate Shell Forging 05 Charpy specimens, oriented with the tangential axis of the specimen perpendicular to the major rolling direction (axial orientation), resulted in an irradiated 30 ft-lb transition temperature (T_{30}) of -20.5°F and an irradiated 50 ft-lb transition temperature (T_{50}) of 24.2°F. This results in a 30 ft-lb transition temperature increase of 21.3°F (ΔT_{30} = 21.3°F) and a 50 ft-lb transition temperature increase of 18.1°F (ΔT_{50} = 18.1°F) for the axially oriented specimens.

- Irradiation of the Surveillance Program Weld Material (Heat # 895075) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature (T_{30}) of -25.1°F and an irradiated 50 ft-lb transition temperature (T_{50}) of 14.3°F. This results in a 30 ft-lb transition temperature increase of 32.6°F (ΔT_{30} = 32.6°F) and a 50 ft-lb transition temperature increase of 33.5°F (ΔT_{30} = 33.5°F).

- Irradiation of the HAZ Material Charpy specimens resulted in an irradiated 30 ft-lb transition temperature (T_{30}) of -103.4°F and an irradiated 50 ft-lb transition temperature (T_{50}) of -61.7°F. This results in a 30 ft-lb transition temperature reduction of 1.6°F (ΔT_{30} = -1.6°F) and a 50 ft-lb transition temperature increase of 19.8°F (ΔT_{50} = 19.8°F). Note that physically, a reduction in T_{30} (ΔT_{30} = -1.6°F) should not occur. However, this can be indicated due to the standard error of the measurements and/or variation of material properties within a sample/heat. When this measured reduction is observed, all downstream analyses which use the results should conservatively assume that no shift in T_{30} has occurred, i.e., ΔT_{30} = 0°F.

- The average upper-shelf energy of Intermediate Shell Forging 05 (tangential orientation) resulted in an average energy decrease of -45 ft-lb after irradiation. This decrease results in an irradiated average upper-shelf energy of 130 ft-lb for the tangentially oriented specimens.

- The average upper-shelf energy of Intermediate Shell Forging 05 (axial orientation) resulted in an average energy decrease of -5 ft-lb after irradiation. This decrease results in an irradiated average upper-shelf energy of 105 ft-lb for the axially oriented specimens.

- The average upper-shelf energy of the Surveillance Program Weld Material (Heat # 895075) Charpy specimens resulted in an average energy decrease of -9 ft-lb after irradiation. This decrease results in an irradiated average upper-shelf energy of 135 ft-lb for the weld metal specimens.

- The average upper-shelf energy of the HAZ Material Charpy specimens resulted in an average energy decrease of -12 ft-lb after irradiation. This decrease results in an irradiated average upper-shelf energy of 118 ft-lb for the HAZ Material.

- Comparisons of the measured 30 ft-lb shift in transition temperature values and upper-shelf energy decreases to those predicted by Regulatory Guide 1.99, Revision 2 [Ref. 1] for the Watts Bar Unit 2 reactor vessel surveillance materials are presented in Table 5-10.

The fracture appearance of each irradiated Charpy specimen from the various materials is shown in Figures 5-13 through 5-16. The fractures show an increasingly ductile or tougher appearance with increasing test temperature. Load-time records for the individual instrumented Charpy specimens are contained in Appendix B.

5.3 TENSILE TEST RESULTS

The results of the tensile tests performed on the various materials contained in Capsule U irradiated to 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) are presented in Table 5-11 and are compared with unirradiated results as shown in Figure 5-17 through Figure 5-19.

The results of the tensile tests performed on the Intermediate Shell Forging 05 (tangential orientation) indicated that irradiation to 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) caused increases in the 0.2 percent offset yield strength and the ultimate tensile strength when compared to unirradiated data in WCAP-9455 [Ref. 2]. See Figure 5-17.

The results of the tensile tests performed on the Intermediate Shell Forging 05 (axial orientation) indicated that irradiation to 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) caused increases in the 0.2 percent offset yield strength and the ultimate tensile strength when compared to unirradiated data in WCAP-9455 [Ref. 2]. See Figure 5-18.

The results of the tensile tests performed on the Surveillance Program Weld Material (Heat # 895075) indicated that irradiation to 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) caused increases in the 0.2 percent offset yield strength and the ultimate tensile strength when compared to unirradiated data [Ref. 2]. See Figure 5-19.

The fractured tensile specimens for the Intermediate Shell Forging 05 (tangential orientation) material are shown in Figure 5-20; the fractured tensile specimens for the Intermediate Shell Forging 05 (axial orientation) are shown in Figure 5-21; and the fracture tensile specimens for the Surveillance Program Weld Material (Heat # 895075) are shown in Figure 5-22. The engineering stress-strain curves for the tensile tests are shown in Figure 5-23 through Figure 5-31.

Table 5-1 Charpy V-notch Data for the Watts Bar Unit 2 Intermediate Shell Forging 05 Irradiated to a Fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) (Tangential Orientation)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
BL10	-60	-51	8	11	6	0.2	5
BL4	-50	-46	21	28	14	0.4	10
BL8	-30	-34	32	43	22	0.6	10
BL15	-20	-29	21	28	18	0.5	10
BL6	-15	-26	28	38	21	0.5	15
BL3	-10	-23	26	35	19	0.5	15
BL5	-5	-21	42	57	33	0.8	20
BL11	0	-18	48	65	34	0.9	25
BL1	10	-12	52	71	39	1.0	25
BL9	40	4	57	77	42	1.1	35
BL13	75	24	106	144	75	1.9	70
BL2	120	49	124	168	86	2.2	100
BL7	170	77	130	176	83	2.1	100
BL14	200	93	138	187	90	2.3	100
BL12	220	104	130	176	85	2.2	100

Table 5-2 Charpy V-notch Data for the Watts Bar Unit 2 Intermediate Shell Forging 05 Irradiated to a Fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) (Axial Orientation)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
BT1	-60	-51	14	19	9	0.2	10
BT9	-50	-46	22	30	14	0.4	15
BT2	-35	-37	10	14	10	0.3	10
BT6	-30	-34	31	42	22	0.6	15
BT4	-20	-29	39	53	28	0.7	15
BT15	-15	-26	29	39	21	0.5	15
BT10	-10	-23	27	37	21	0.5	15
BT8	0	-18	41	56	29	0.7	15
BT3	10	-12	49	66	39	1.0	20
BT12	30	-1	61	83	49	1.2	35
BT14	75	24	73	99	54	1.4	55
BT13	120	49	74	100	58	1.5	60
BT5	170	77	102	138	73	1.9	100
BT11	200	93	109	148	76	1.9	100
BT7	220	104	103	140	80	2.0	100

Table 5-3 Charpy V-notch Data for the Watts Bar Unit 2 Surveillance Program Weld Material (Heat # 895075) Irradiated to a Fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
BW13	-60	-51	21	28	17	0.4	25
BW8	-50	-46	20	27	16	0.4	20
BW15	-30	-34	21	28	17	0.4	35
BW7	-25	-32	23	31	21	0.5	30
BW4	-20	-29	26	35	23	0.6	30
BW14	-15	-26	50	68	38	1.0	45
BW9	-10	-23	39	53	28	0.7	40
BW1	0	-18	38	52	29	0.7	35
BW12	10	-12	52	71	42	1.1	40
BW3	60	16	68	92	52	1.3	55
BW6	75	24	110	149	76	1.9	80
BW2	120	49	86	117	71	1.8	60
BW5	170	77	123	167	84	2.1	100
BW10	200	93	138	187	90	2.3	100
BW11	220	104	143	194	86	2.2	100

Table 5-4 Charpy V-notch Data for the Watts Bar Unit 2 Heat-Affected Zone (HAZ) Material Irradiated to a Fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	°F	°C	ft-lbs	Joules	mils	mm	%
BH5	-115	-82	29	39	16	0.4	25
BH3	-110	-79	42	57	23	0.6	25
BH10	-100	-73	14	19	7	0.2	20
BH2	-90	-68	39	53	23	0.6	25
BH12	-80	-62	33	45	22	0.6	35
BH7	-70	-57	63	85	37	0.9	55
BH15	-60	-51	52	71	31	0.8	45
BH14	-35	-37	77	104	48	1.2	60
BH8	-30	-34	33	45	25	0.6	45
BH11	-20	-29	71	96	42	1.1	60
BH1	10	-12	98	133	56	1.4	75
BH4	75	24	117	159	68	1.7	100
BH13	150	66	105	142	74	1.9	100
BH6	200	93	125	169	73	1.9	100
BH9	220	104	127	172	73	1.9	100

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Table 5-5 Instrumented Charpy Impact Test Results for the Watts Bar Unit 2 Intermediate Shell Forging 05 Irradiated to a Fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) (Tangential Orientation)

Sample Number	Test Temp (°F)	Total Dial Energy, KV (ft-lb)	Total Instrumented Energy, W_t (ft-lb)	Difference, (KV-W_t)/KV (%)	Energy to Max Load, W_m (ft-lb)	Maximum Load, F_m (lb)	Time to F_m (msec)	General Yield Load, F_{gy} (lb)	Fracture Load, F_{bf} (lb)	Arrest Load, F_a (lb)
BL10	-60	7.5	7.43	1%	3.53	4000	0.09	3400	3400	0
BL4	-50	20.5	19.69	4%	3.40	4000	0.09	3200	4000	0
BL8	-30	31.5	29.43	7%	28.27	4100	0.51	3200	4100	0
BL15	-20	20.5	19.23	6%	16.24	4000	0.31	3100	3800	0
BL6	-15	28.25	27.00	4%	23.49	4000	0.43	3200	3900	0
BL3	-10	26.25	24.73	6%	23.33	4000	0.44	3100	4000	0
BL5	-5	42.25	39.81	6%	34.04	4200	0.6	3200	3900	0
BL11	0	47.5	43.09	9%	34.71	4200	0.63	3200	3800	0
BL1	10	52	48.59	7%	33.53	4100	0.61	3000	3900	0
BL9	40	56.5	49.42	13%	32.8	4100	0.6	3000	3900	0
BL13	75	105.5	93.06	12%	32.35	4000	0.6	2900	2300	800
BL2	120	124	114.35	8%	31.08	3900	0.6	2700	0	0
BL7	170	129.75	119.56	8%	30.42	3900	0.61	2600	0	0
BL14	200	137.5	127.37	7%	29.79	3800	0.6	2400	0	0
BL12	220	130	119.59	8%	29.73	3800	0.61	2500	0	0

Table 5-6 Instrumented Charpy Impact Test Results for the Watts Bar Unit 2 Intermediate Shell Forging 05 Irradiated to a Fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) (Axial Orientation)

Sample Number	Test Temp (°F)	Total Dial Energy, KV (ft-lb)	Total Instrumented Energy, W_t (ft-lb)	Difference, $(KV-W_t)/KV$ (%)	Energy to Max Load, W_m (ft-lb)	Maximum Load, F_m (lb)	Time to F_m (msec)	General Yield Load, F_{gy} (lb)	Fracture Load, F_{bf} (lb)	Arrest Load, F_a (lb)
BT1	-60	14	13.34	5%	3.23	4000	0.09	3300	3800	0
BT9	-50	22	21.29	3%	19.5	4100	0.36	3200	3700	0
BT2	-35	10	9.69	3%	3.4	4000	0.09	3300	3800	0
BT6	-30	31	29.14	6%	28.34	4100	0.5	3100	4000	0
BT4	-20	38.5	36.24	6%	34.27	4100	0.61	3200	4000	0
BT15	-15	29.25	28.24	3%	19.05	4000	0.36	3100	3900	0
BT10	-10	27	25.02	7%	23.36	4000	0.43	3100	3800	0
BT8	0	41	38.46	6%	33.72	4100	0.6	3100	4000	0
BT3	10	49	45.29	8%	33.36	4100	0.6	3000	3900	0
BT12	30	61.25	56.00	9%	33.63	4200	0.6	3000	3800	300
BT14	75	73	61.86	15%	32.23	4000	0.6	2900	3200	1000
BT13	120	74.25	62.98	15%	30.71	3900	0.61	2700	3000	1400
BT5	170	102	94.57	7%	30.29	3900	0.61	2200	0	0
BT11	200	108.5	100.54	7%	31.38	3800	0.62	2500	0	0
BT7	220	103	95.27	8%	28.95	3800	0.6	2500	0	0

Table 5-7 Instrumented Charpy Impact Test Results for the Watts Bar Unit 2 Surveillance Program Weld Material (Heat # 895075) Irradiated to a Fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Total Dial Energy, KV (ft-lb)	Total Instrumented Energy, W_t (ft-lb)	Difference, (KV-W_t)/KV (%)	Energy to Max Load, W_m (ft-lb)	Maximum Load, F_m (lb)	Time to F_m (msec)	General Yield Load, F_{gy} (lb)	Fracture Load, F_{bf} (lb)	Arrest Load, F_a (lb)
BW13	-60	20.5	17.62	14%	3.5	4100	0.09	3200	3900	800
BW8	-50	19.5	17.77	9%	3.21	3900	0.09	3100	3500	0
BW15	-30	20.5	16.63	19%*	3.24*	3900*	0.09*	3100*	3600*	800*
BW7	-25	22.5	20.27	10%	3.4	3800	0.09	3100	3800	1400
BW4	-20	26	24.63	5%	3.2	3900	0.09	3000	3700	500
BW14	-15	50	45.66	9%	33.33	4000	0.6	3100	3700	2100
BW9	-10	38.5	34.85	9%	28.03	3900	0.51	3200	3900	1600
BW1	0	37.5	35.35	6%	27.91	4000	0.5	3100	3700	1400
BW12	10	52	47.42	9%	32.81	4000	0.61	3000	3800	1600
BW3	60	68	61.59	9%	13.71	4000	0.6	2800	3600	1400
BW6	75	109.5	100.56	8%	32.02	4000	0.6	2800	3000	2800
BW2	120	86.25	79.06	8%	31.19	3900	0.6	2800	2700	1500
BW5	170	123	113.84	7%	51.15	3700	0.99	2600	0	0
BW10	200	138	128.24	7%	48.83	3700	0.95	2500	0	0
BW11	220	143	131.6	8%	47.95	3700	0.95	2200	0	0

* The difference between instrumented Charpy and dial values was greater than 15%, but the values were not adjusted as required by E2298-18 [Ref. 10].

Table 5-8 Instrumented Charpy Impact Test Results for the Watts Bar Unit 2 Heat-Affected Zone (HAZ) Material Irradiated to a Fluence of 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Total Dial Energy, KV (ft-lb)	Total Instrumented Energy, W_t (ft-lb)	Difference, (KV-W_t)/KV (%)	Energy to Max Load, W_m (ft-lb)	Maximum Load, F_m (lb)	Time to F_m (msec)	General Yield Load, F_{gy} (lb)	Fracture Load, F_{bf} (lb)	Arrest Load, F_a (lb)
BH5	-115	28.5	25.53	10%	4.63	4300	0.12	3500	4100	0
BH3	-110	41.75	38.45	8%	37.53	4400	0.62	3700	4400	0
BH10	-100	13.5	12.4	8%	3.57	4400	0.1	3400	3500	0
BH2	-90	38.5	36.1	6%	3.44	4300	0.09	3400	4200	0
BH12	-80	33	28.22	14%	3.57	4300	0.1	3500	4100	900
BH7	-70	62.5	52.99	15%	35.93	4300	0.61	3400	3900	900
BH15	-60	52	44.08	15%	35.91	4300	0.6	3300	4000	600
BH14	-35	76.5	68.28	11%	35.41	4300	0.6	3400	4100	1600
BH8	-30	32.5	26.61*	18%*	3.46*	4000*	0.09*	3200*	3600*	1700*
BH11	-20	71	62.25	12%	34.87	4200	0.6	3200	3800	1600
BH1	10	98	91.49	7%	34.13	4200	0.61	3000	3600	1900
BH4	75	116.5	109.21	6%	53.54	4100	0.95	3000	0	0
BH13	150	104.5	97.83	6%	30.99	3900	0.6	2700	0	0
BH6	200	124.5	116.79	6%	41.01	4000	0.79	2800	0	0
BH9	220	127	117.92	7%	49.78	3800	0.95	2500	0	0

* The difference between instrumented Charpy and dial values was greater than 15%, but the values were not adjusted as required by E2298-18 [Ref. 10].

Table 5-9 Effect of Irradiation to 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV) on the Charpy V-Notch Toughness Properties of the Watts Bar Unit 2 Reactor Vessel Surveillance Capsule U Materials

Material	Average 30 ft-lb Transition Temperature(a) (°F)			Average 35 mil Lateral Expansion Temperature(a) (°F)			Average 50 ft-lb Transition Temperature(a) (°F)			Average Energy Absorption ≥ 95% Shear(b) (ft-lb)		
	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔE
Intermediate Shell Forging 05 (Tangential)	-42.3	-15.6	26.7	-20.7	9.1	29.8	-20.9	12.7	33.6	175	130	-45
Intermediate Shell Forging 05 (Axial)	-41.8	-20.5	21.3	1.3	15.0	13.7	6.1	24.2	18.1	110	105	-5
Surveillance Weld Material (Heat # 895075)	-57.7	-25.1	32.6	-24.7	3.7	28.4	-19.2	14.3	33.5	144	135	-9
Heat-Affected Zone (HAZ) Material	-101.8	-103.4	-1.6(c)	-69.7	-46.1	23.6	-81.5	-61.7	19.8	130	118	-12

Notes:

(a) Average value is determined by CVGRAPH, Version 6.02 (see Appendix C).

(b) Upper-shelf Energy (USE) values are a calculated average from unirradiated and Capsule U Charpy test results for specimens that achieved greater than or equal to 95% shear.

(c) Note that physically a reduction in T_{30} should not occur.

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

Material	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper-Shelf Energy Decrease	
			Predicted[(a)] (°F)	Measured[(b)] (°F)	Predicted[(a)] (%)	Measured[(b)] (%)
Intermediate Shell Forging 05 (Tangential)	U	0.604	26.6	26.7	17	26
Intermediate Shell Forging 05 (Axial)	U	0.604	26.6	21.3	17	5
Surveillance Weld Material (Heat # 895075)	U	0.604	38.6	32.6	17	6
Heat-Affected Zone Material	U	0.604	- - -	-1.6	- - -	9

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2 [Ref. 1], methodology using the capsule fluence and best-estimate weight percent values of copper and nickel of the surveillance material.

(b) Calculated by CVGRAPH, Version 6.02 using measured Charpy data (See Appendix C).

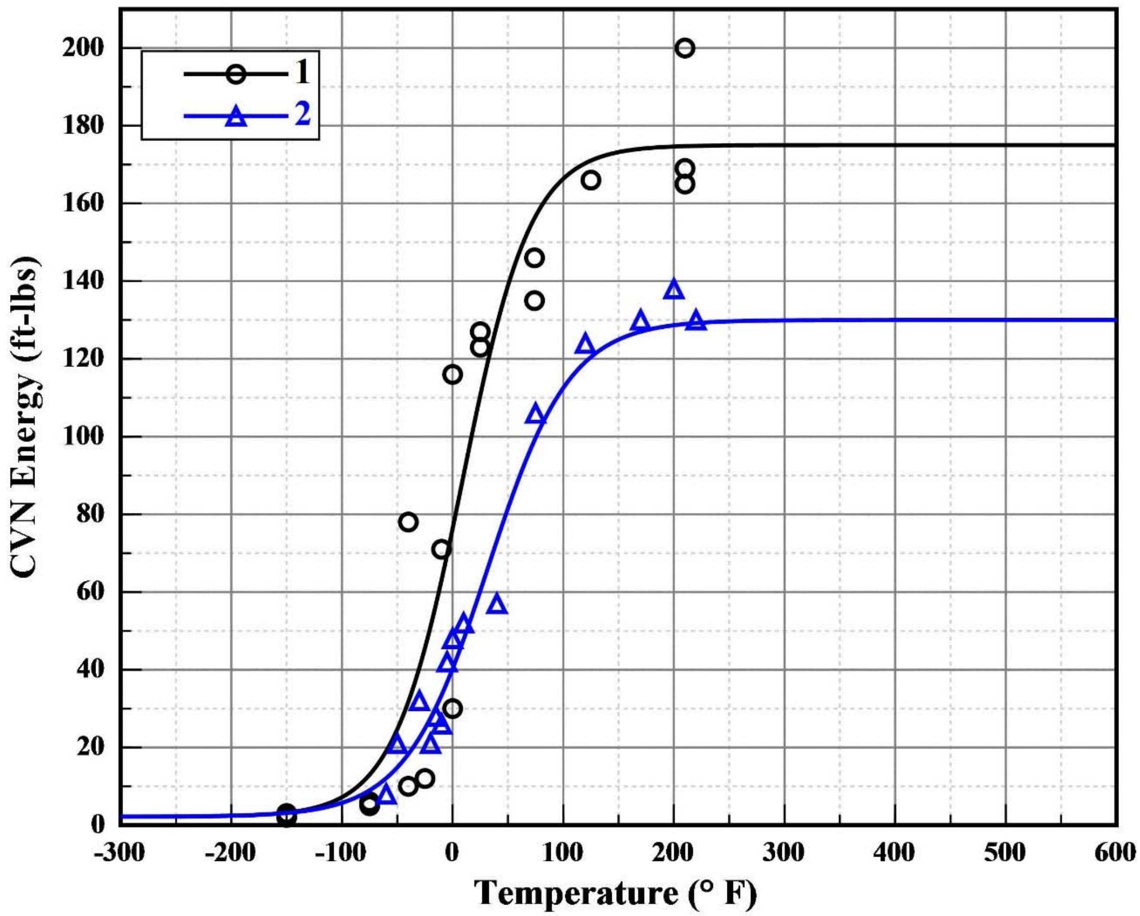
Table 5-11 Tensile Properties of the Watts Bar Unit 2 Capsule U Reactor Vessel Surveillance Materials Irradiated to 6.04×10^{18} n/cm^2 (E > 1.0 MeV)

Material	Sample Number	Test Temp. (°F)	0.2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Strength (ksi)	Fracture True Stress (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Intermediate Shell Forging 05 (Tangential)	BL1	78	63.0	86.5	2.51	51.1	201	12.6	31.9	75
	BL2	300	58.8	80.4	2.54	51.7	217	10.6	24.7	76
	BL3	550	57.3	86.2	2.76	56.3	190	9.1	22.9	70
Intermediate Shell Forging 05 (Axial)	BT1	78	66.8	89.7	2.99	60.8	200	11.3	25.2	70
	BT2	300	61.8	83.0	2.91	59.4	195	10.2	22.2	70
	BT3	550	60.4	87.1	3.08	62.7	179	9.9	21.3	65
Surveillance Weld Material (Heat # 895075)	BW1	78	74.5	88.2	2.57	52.3	227	12.8	30.2	77
	BW2	300	70.2	81.1	2.40	48.9	199	11.6	26.9	75
	BW3	550	64.1	83.4	2.57	52.4	188	6.5	19.7	72

Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 9:53 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508CL2	Tangential	527828
2	Watts Bar 2	U	SA508Cl2	Tangential	527828



Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1	0.00E+000	2.2	175	0	-42.3	0	-20.9	0
2	6.04E+018	2.2	130	-45	-15.6	26.7	12.7	33.6

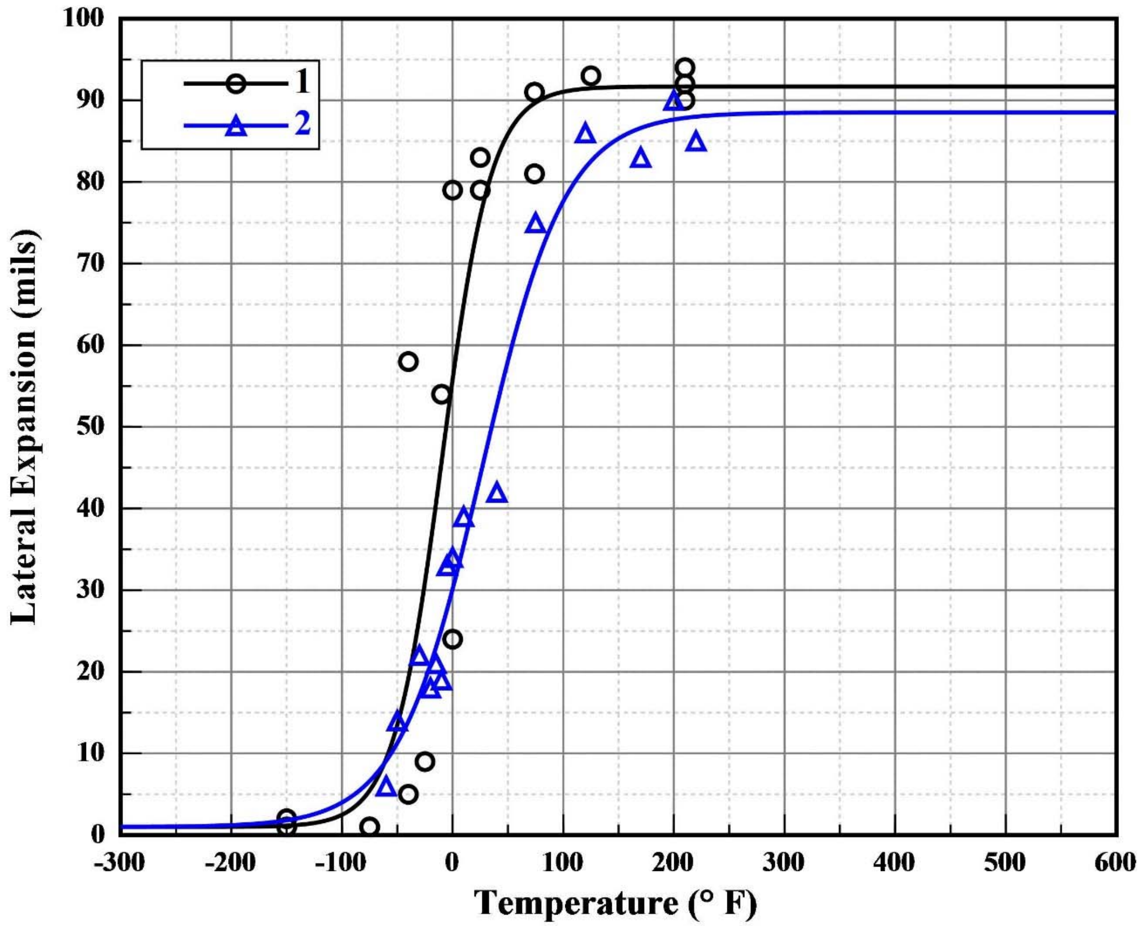
Figure 5-1 Charpy V-Notch Impact Energy vs. Temperature for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)

Note: Data for Capsule U was taken from Table 5-1.

Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 9:59 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508CL2	Tangential	527828
2	Watts Bar 2	U	SA508Cl2	Tangential	527828



Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1	0.00E+000	1	91.68	0	-20.7	0
2	6.04E+018	1	88.51	-3.17	9.1	29.8

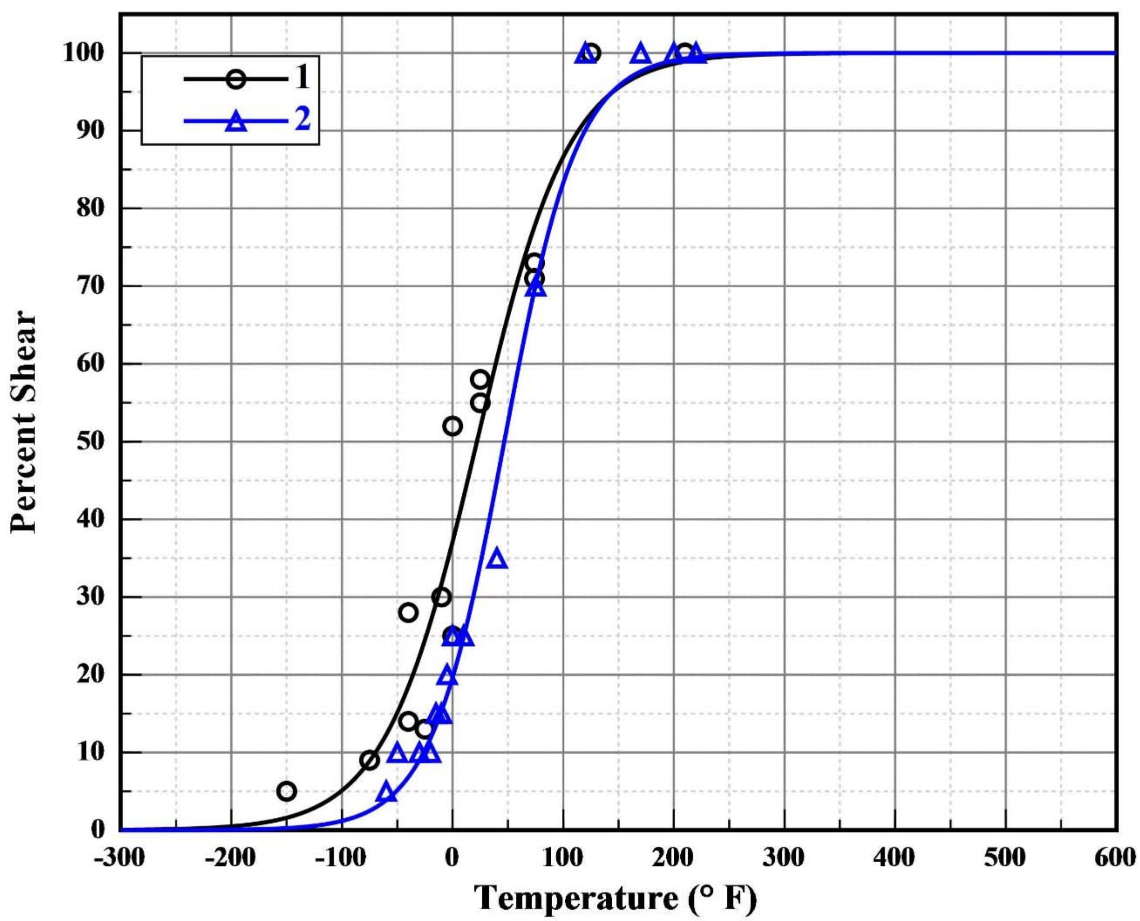
Figure 5-2 Charpy V-Notch Lateral Expansion vs. Temperature for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)

Note: Data for Capsule U was taken from Table 5-1.

Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 10:36 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508CL2	Tangential	527828
2	Watts Bar 2	U	SA508Cl2	Tangential	527828



Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1	0.00E+000	0	100	0	22	0
2	6.04E+018	0	100	0	46.7	24.7

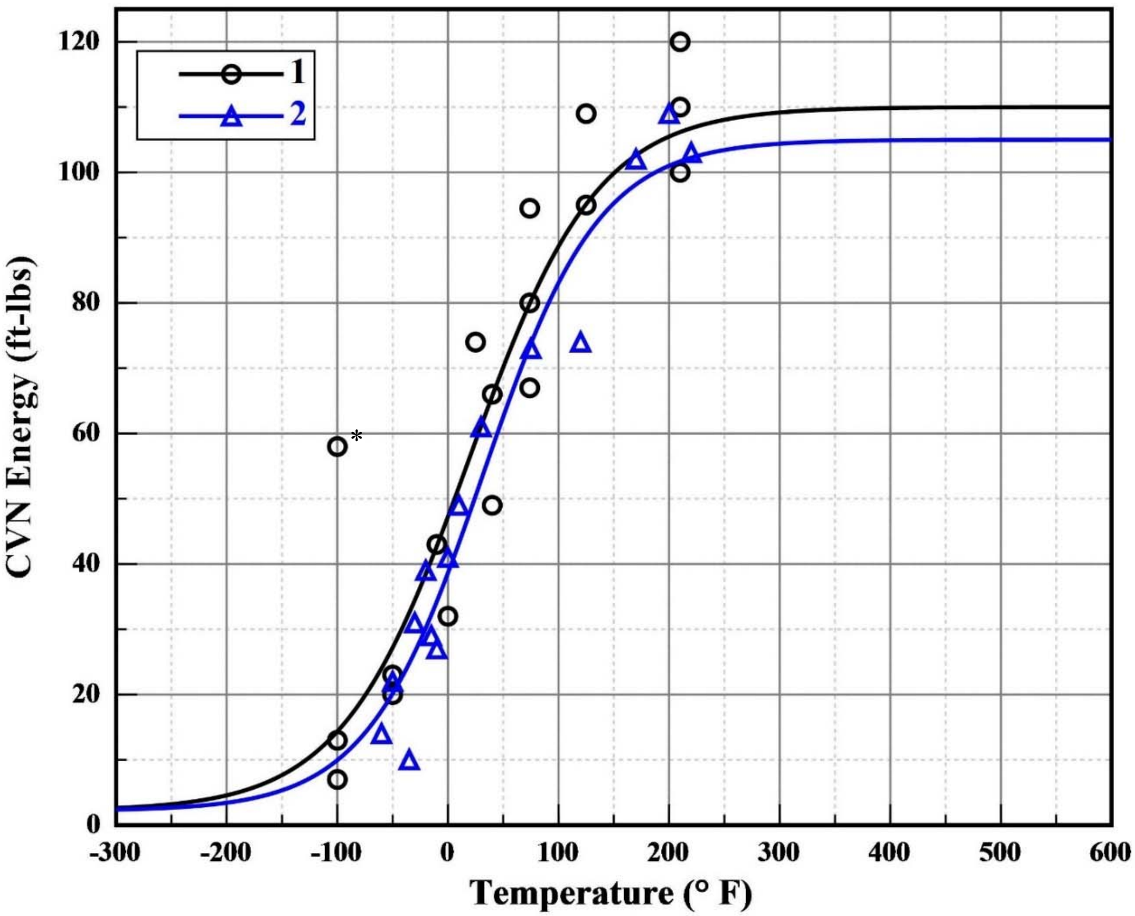
Figure 5-3 Charpy V-Notch Percent Shear vs. Temperature for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)

Note: Data for Capsule U was taken from Table 5-1.

Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 10:35 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508CL2	Axial	527828
2	Watts Bar 2	U	SA508CL2	Axial	527828



Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1	0.00E+000	2.2	110	0	-41.8	0	6.1	0
2	6.04E+018	2.2	105	-5	-20.5	21.3	24.2	18.1

* Legibility for this test temperature is not clear. It could be 0°F rather than the -100°F shown in this plot. Using -100°F is consistent with WCAP-9455, Rev. 4 and conservative, providing a larger T_{30} shift. This may be revisited, if necessary, during a future evaluation, e.g. next tested capsule's credibility evaluation.

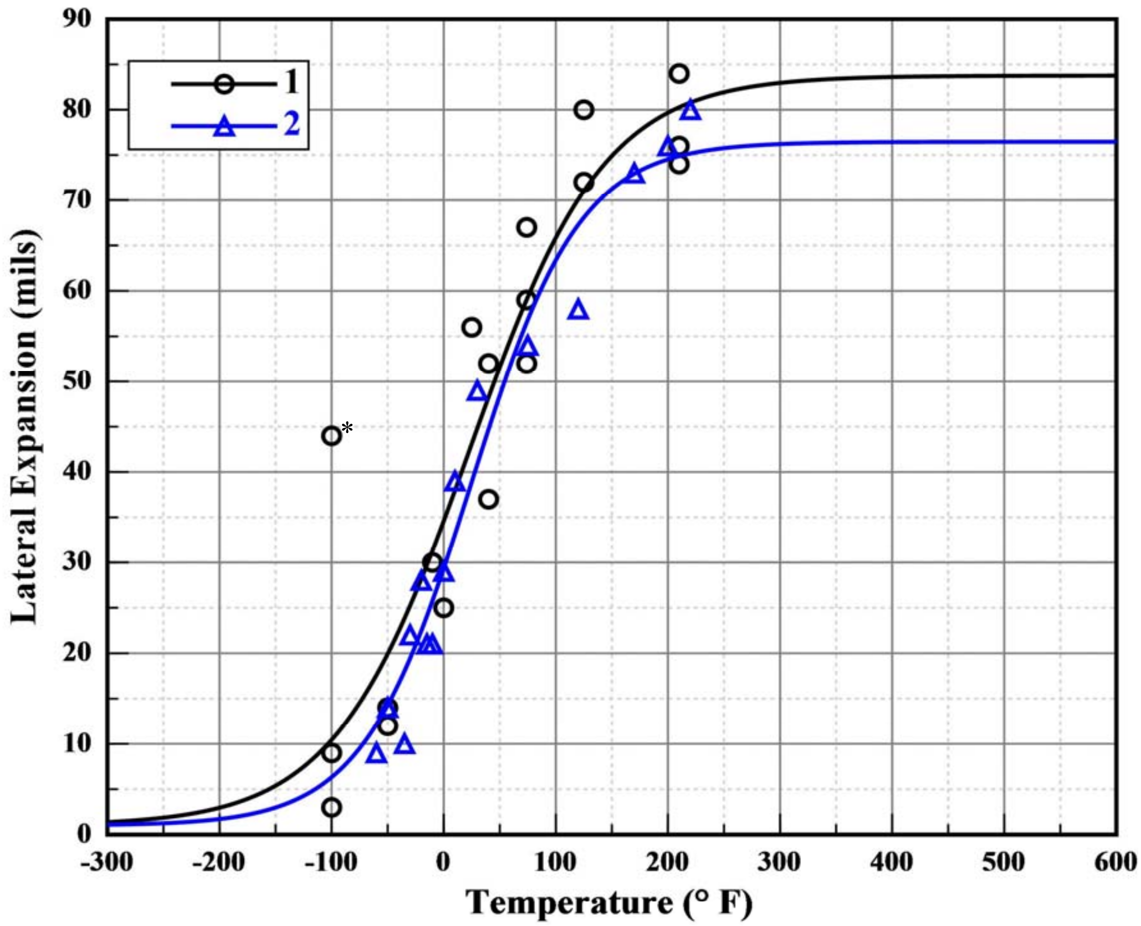
Figure 5-4 Charpy V-Notch Impact Energy vs. Temperature for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)

Note: Data for Capsule U was taken from Table 5-2.

Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 10:30 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508CL2	Axial	527828
2	Watts Bar 2	U	SA508CL2	Axial	527828



Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1	0.00E+000	1	83.77	0	1.3	0
2	6.04E+018	1	76.47	-7.3	15	13.7

* Legibility for this test temperature is not clear. It could be 0°F rather than the -100°F shown in this plot. Using -100°F is consistent with WCAP-9455, Rev. 4 and conservative, providing a larger T_{30} shift. This may be revisited, if necessary, during a future evaluation, e.g. next tested capsule's credibility evaluation.

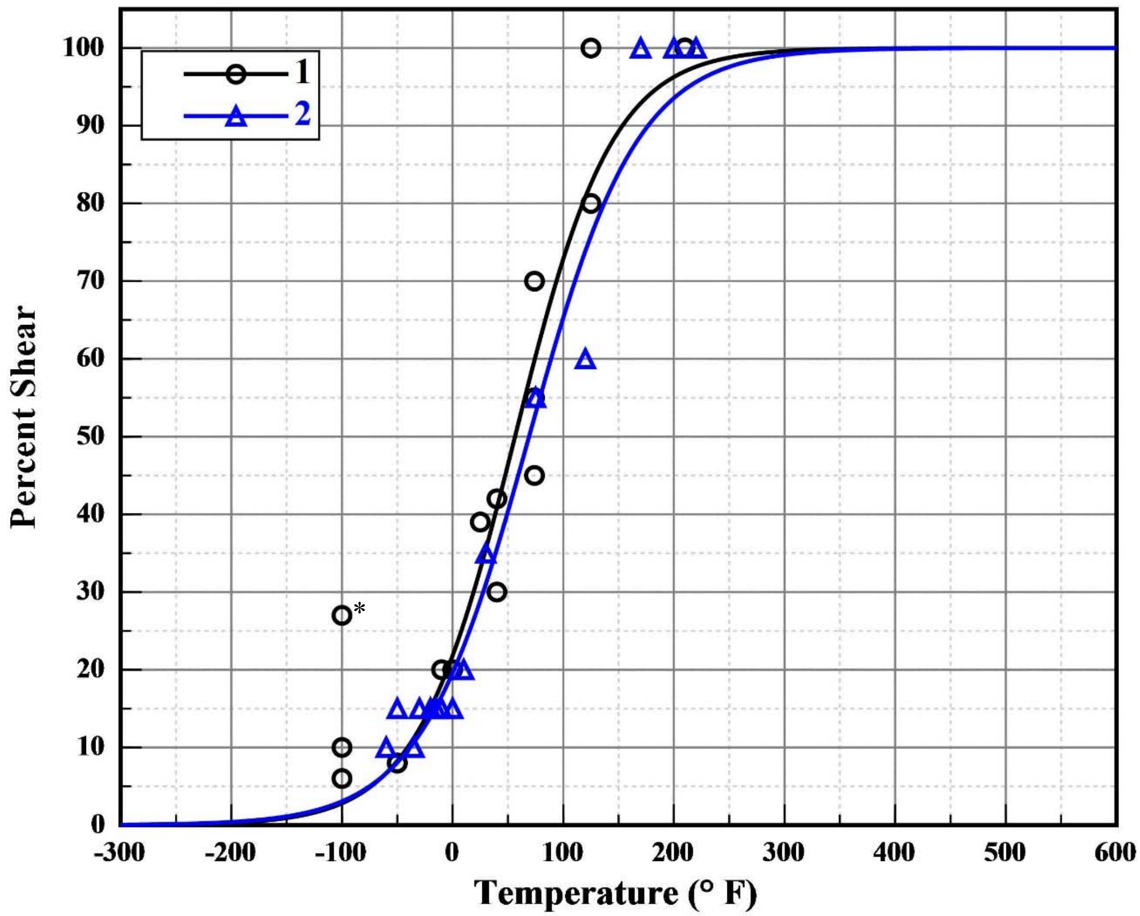
Figure 5-5 Charpy V-Notch Lateral Expansion vs. Temperature for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)

Note: Data for Capsule U was taken from Table 5-2.

Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 10:55 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508CL2	Axial	527828
2	Watts Bar 2	U	SA508CL2	Axial	527828



Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1	0.00E+000	0	100	0	56.4	0
2	6.04E+018	0	100	0	69.1	12.7

* Legibility for this test temperature is not clear. It could be 0°F rather than the -100°F shown in this plot. Using -100°F is consistent with WCAP-9455, Rev. 4 and conservative, providing a larger T_{30} shift. This may be revisited, if necessary, during a future evaluation, e.g. next tested capsule's credibility evaluation.

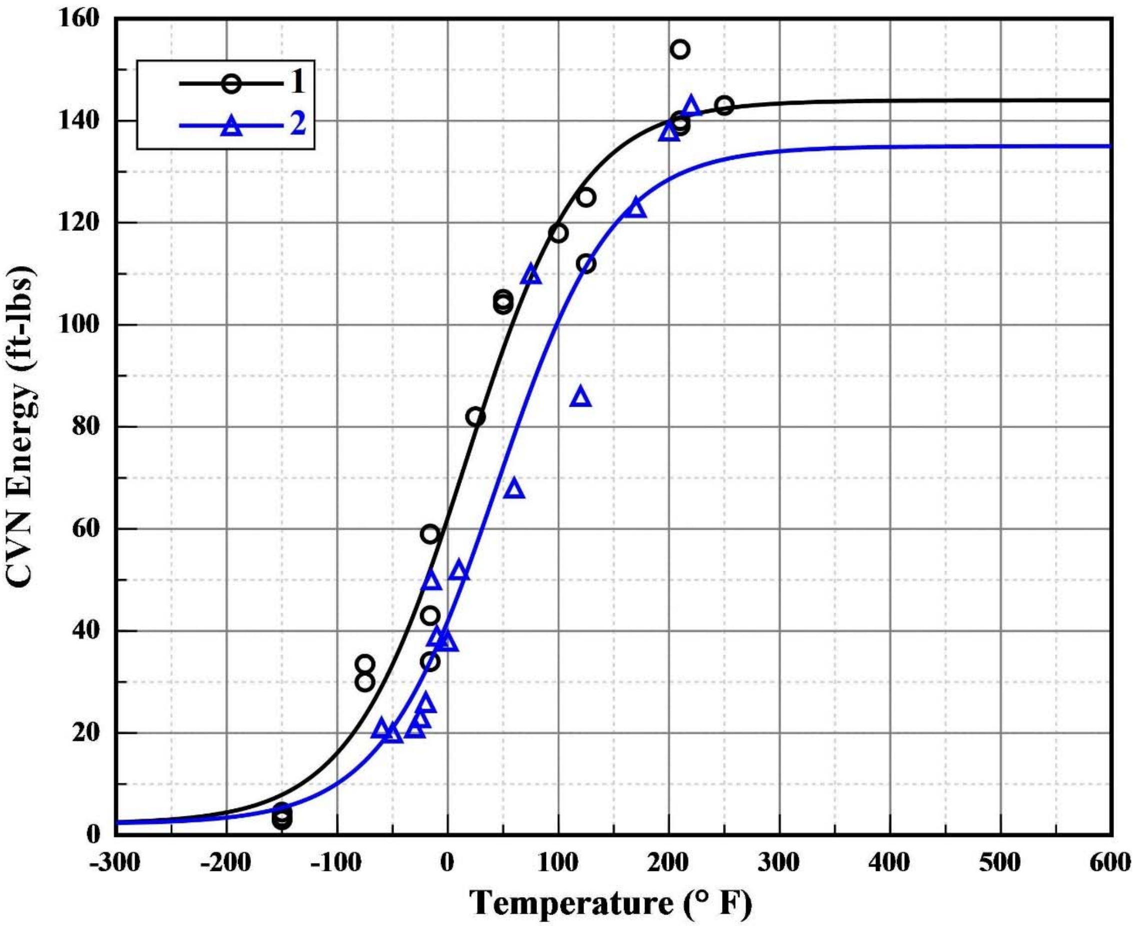
Figure 5-6 Charpy V-Notch Percent Shear vs. Temperature for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)

Note: Data for Capsule U was taken from Table 5-2.

Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:02 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SAW	NA	895075
2	Watts Bar 2	U	SAW	NA	895075



Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1	0.00E+000	2.2	144	0	-57.7	0	-19.2	0
2	6.04E+018	2.2	135	-9	-25.1	32.6	14.3	33.5

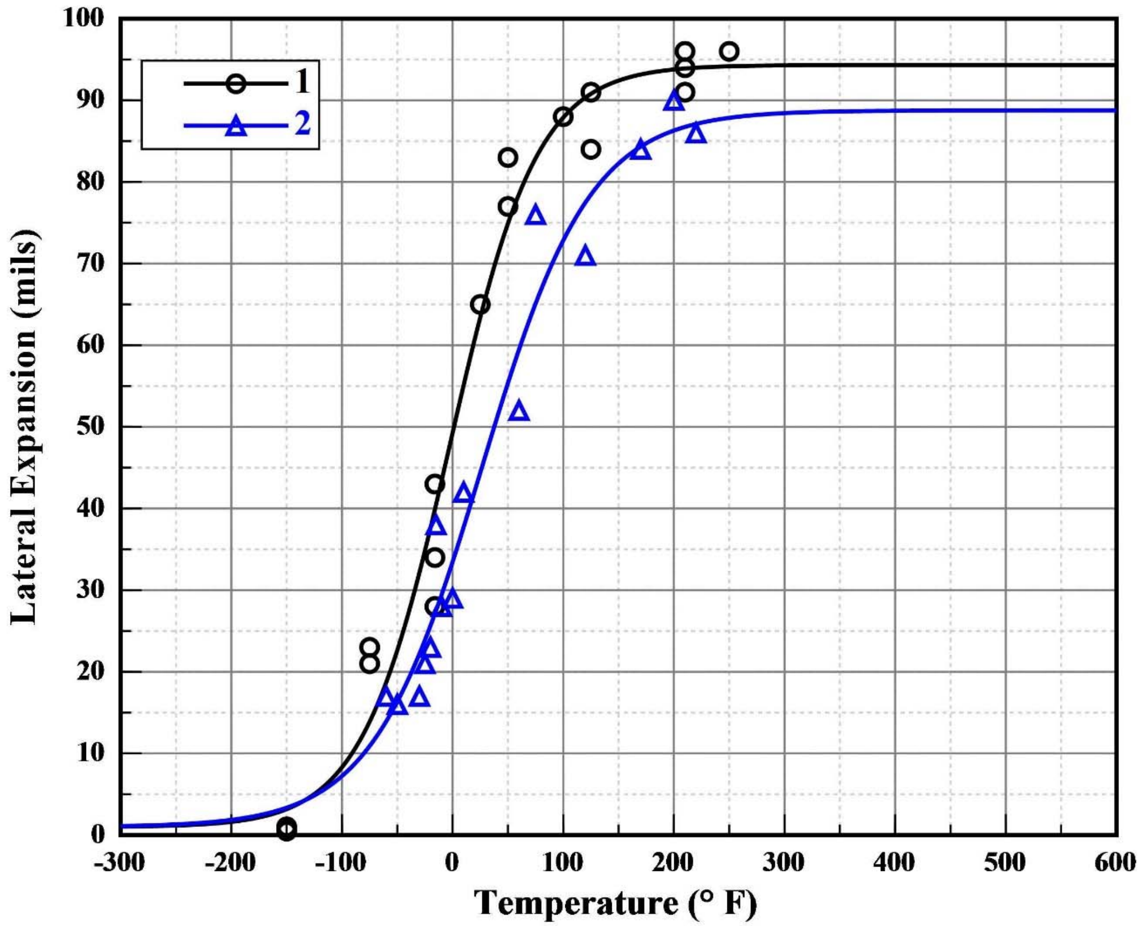
Figure 5-7 Charpy V-Notch Impact Energy vs. Temperature for Watts Bar Unit 2 Reactor Vessel Surveillance Program Weld Material (Heat # 895075)

Note: Data for Capsule U was taken from Table 5-3.

Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:01 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SAW	NA	895075
2	Watts Bar 2	U	SAW	NA	895075



Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1	0.00E+000	1	94.32	0	-24.7	0
2	6.04E+018	1	88.76	-5.56	3.7	28.4

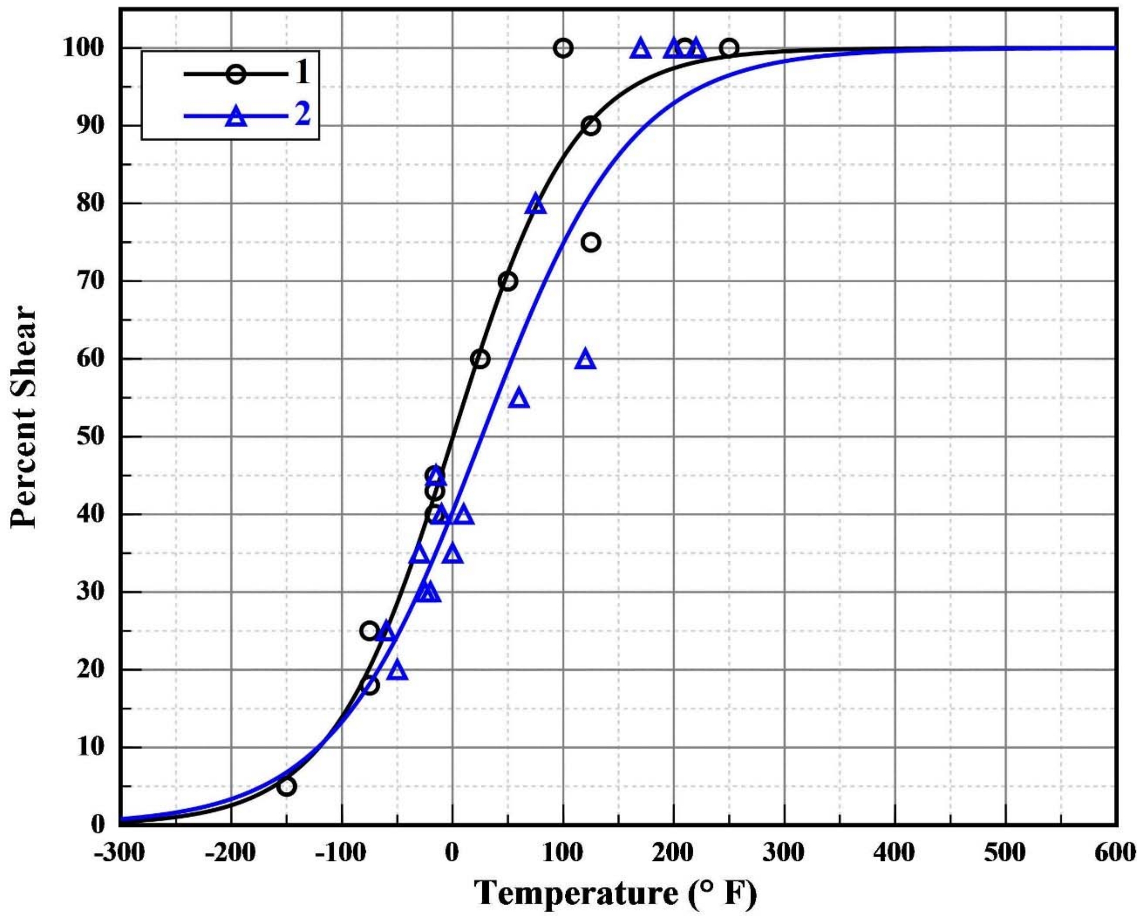
Figure 5-8 Charpy V-Notch Lateral Expansion vs. Temperature for Watts Bar Unit 2 Reactor Vessel Surveillance Program Weld Material (Heat # 895075)

Note: Data for Capsule U was taken from Table 5-3.

Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:03 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SAW	NA	895075
2	Watts Bar 2	U	SAW	NA	895075



Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1	0.00E+000	0	100	0	0.4	0
2	6.04E+018	0	100	0	26.5	26.1

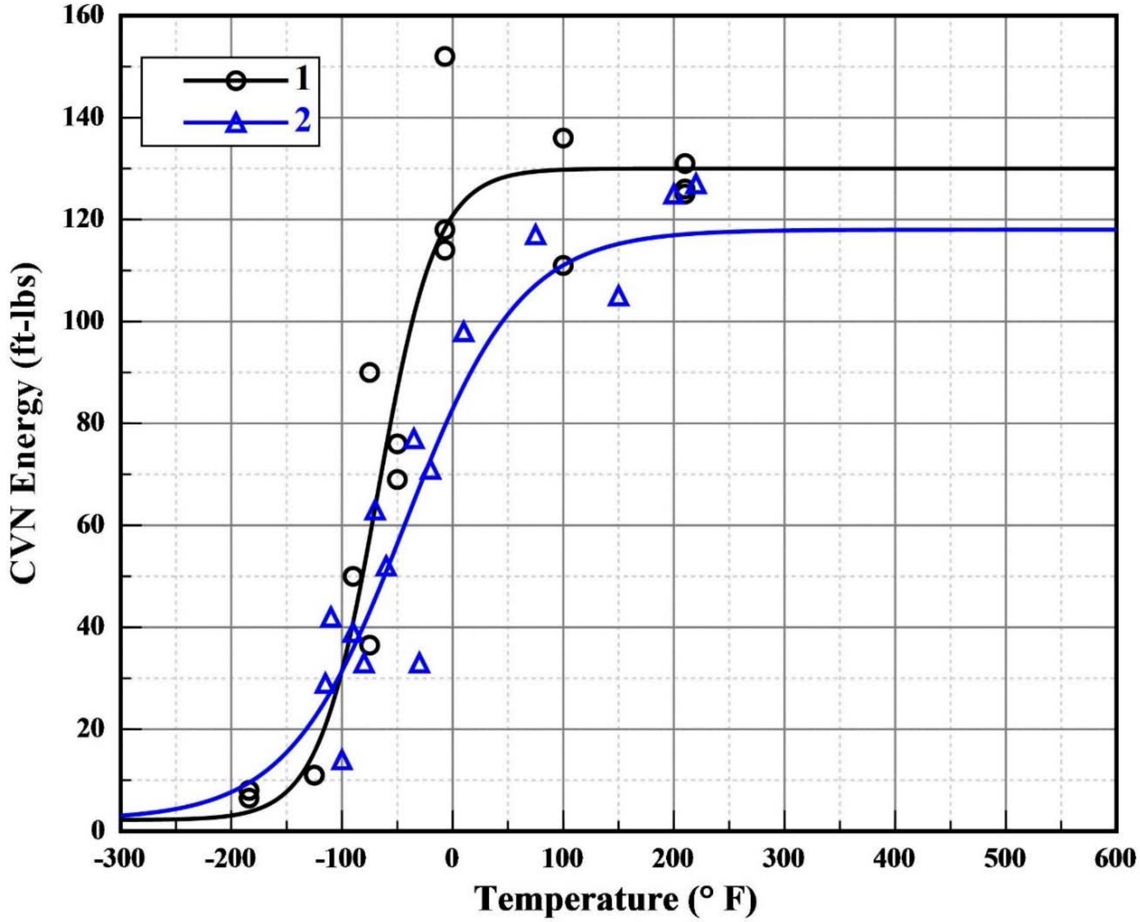
Figure 5-9 Charpy V-Notch Percent Shear vs. Temperature for Watts Bar Unit 2 Reactor Vessel Surveillance Program Weld Material (Heat # 895075)

Note: Data for Capsule U was taken from Table 5-3.

Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:17 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508Cl2	NA	527828
2	Watts Bar 2	U	SA508CL2	NA	527828



Curve	Fluence	LSE	USE	d-USE	T @30	d-T @30	T @50	d-T @50
1	0.00E+000	2.2	130	0	-101.8	0	-81.5	0
2	6.04E+018	2.2	118	-12	-103.4	-1.60	-61.7	19.8

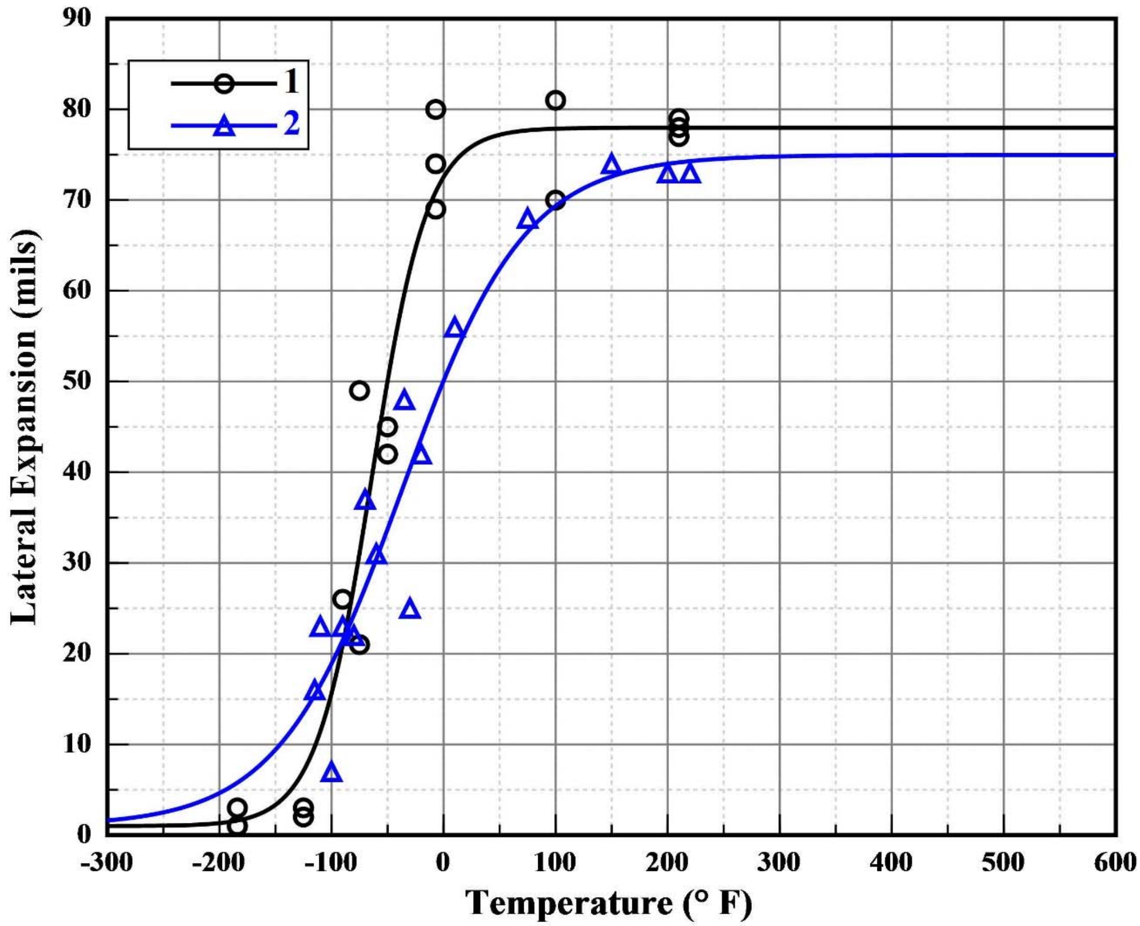
Figure 5-10 Charpy V-Notch Impact Energy vs. Temperature for Watts Bar Unit 2 Reactor Vessel Heat-Affected Zone Material

Note: Data for Capsule U was taken from Table 5-4.

Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:10 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508Cl2	NA	527828
2	Watts Bar 2	U	SA508CL2	NA	527828



Curve	Fluence	LSE	USE	d-USE	T @35	d-T @35
1	0.00E+000	1	77.97	0	-69.7	0
2	6.04E+018	1	74.97	-3	-46.1	23.6

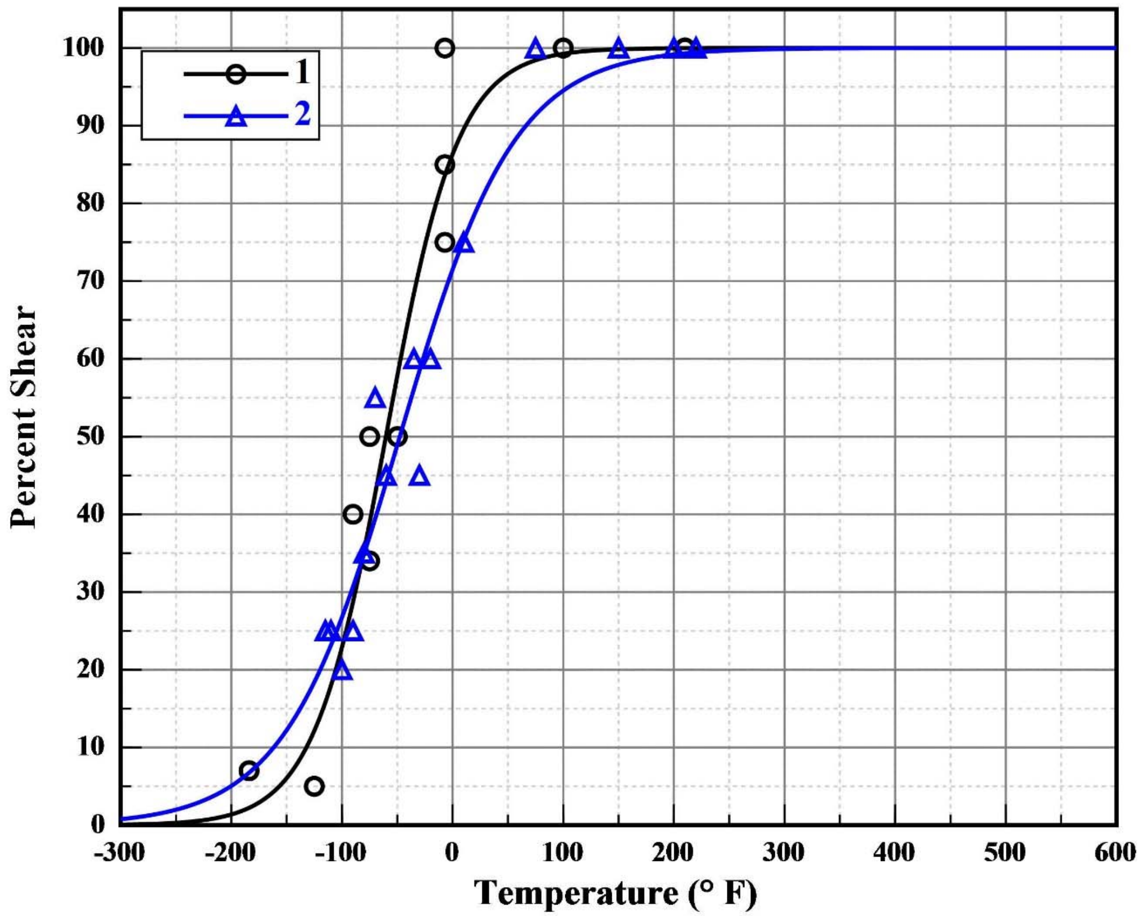
Figure 5-11 Charpy V-Notch Lateral Expansion vs. Temperature for Watts Bar Unit 2 Reactor Vessel Heat-Affected Zone Material

Note: Data for Capsule U was taken from Table 5-4.

Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:16 AM

Curve	Plant	Capsule	Material	Ori.	Heat #
1	Watts Bar 2	Unirrad	SA508Cl2	NA	527828
2	Watts Bar 2	U	SA508CL2	NA	527828



Curve	Fluence	LSE	USE	d-USE	T @50	d-T @50
1	0.00E+000	0	100	0	-60.2	0
2	6.04E+018	0	100	0	-47.7	12.5

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

Note: Data for Capsule U was taken from Table 5-4.

Figure 5-13 Charpy Impact Specimen Fracture Surfaces for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)

Figure 5-14 Charpy Impact Specimen Fracture Surfaces for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)

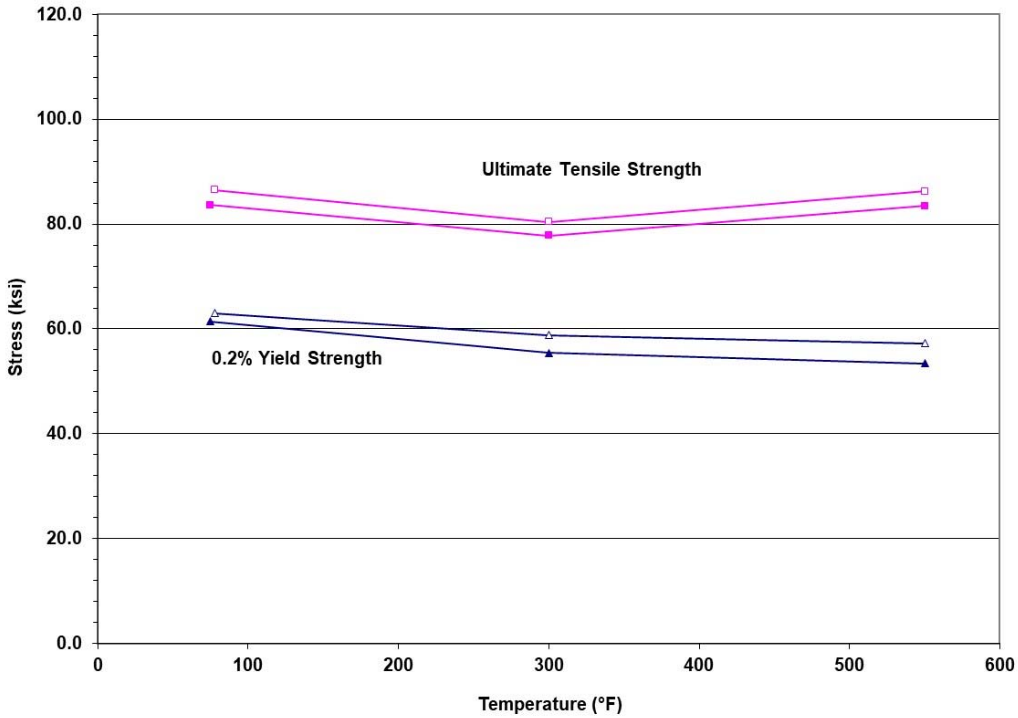
BW13, -60°F BW8, -50°F BW15, -30°F BW7, -25°F BW4, -20°F

BW14, -15°F BW9, -10°F BW1, 0°F BW12, 10°F BW3, 60°F

BW6, 75°F BW2, 120°F BW5, 170°F BW10 200°F BW11, 220°F

Figure 5-15 Charpy Impact Specimen Fracture Surfaces for the Watts Bar Unit 2 Reactor Vessel Surveillance Program Weld Material (Heat # 895075)

Figure 5-16 Charpy Impact Specimen Fracture Surfaces for the Watts Bar Unit 2 Reactor Vessel Heat-Affected Zone Material



Legend: ▲, ●, and ■ are unirradiated
Δ, ○, and □ are irradiated to 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV)

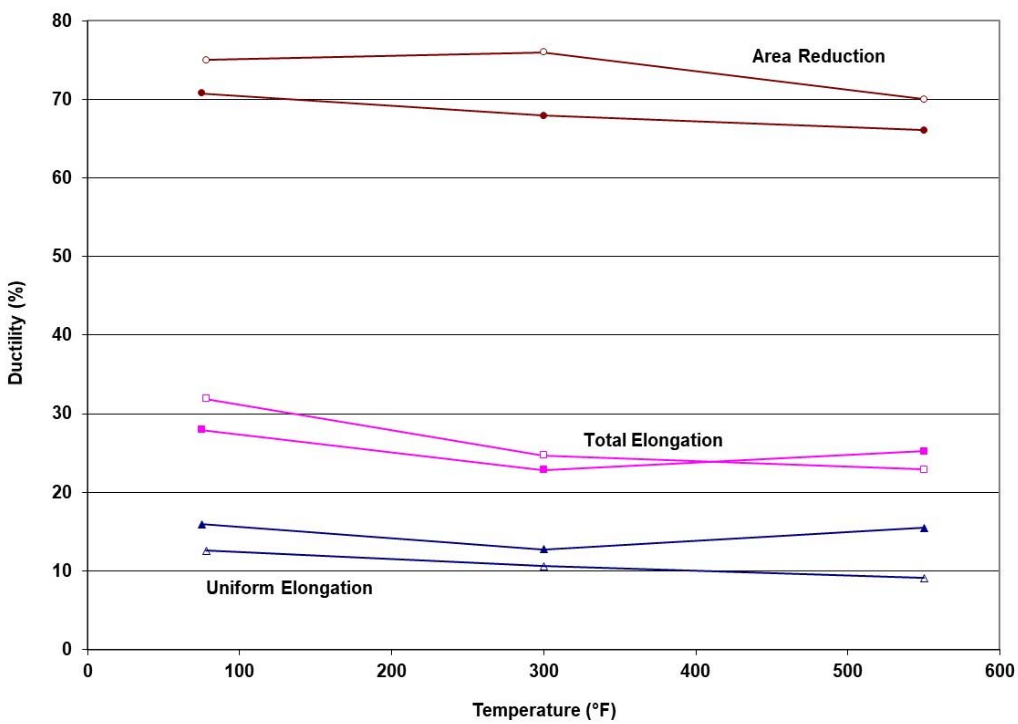
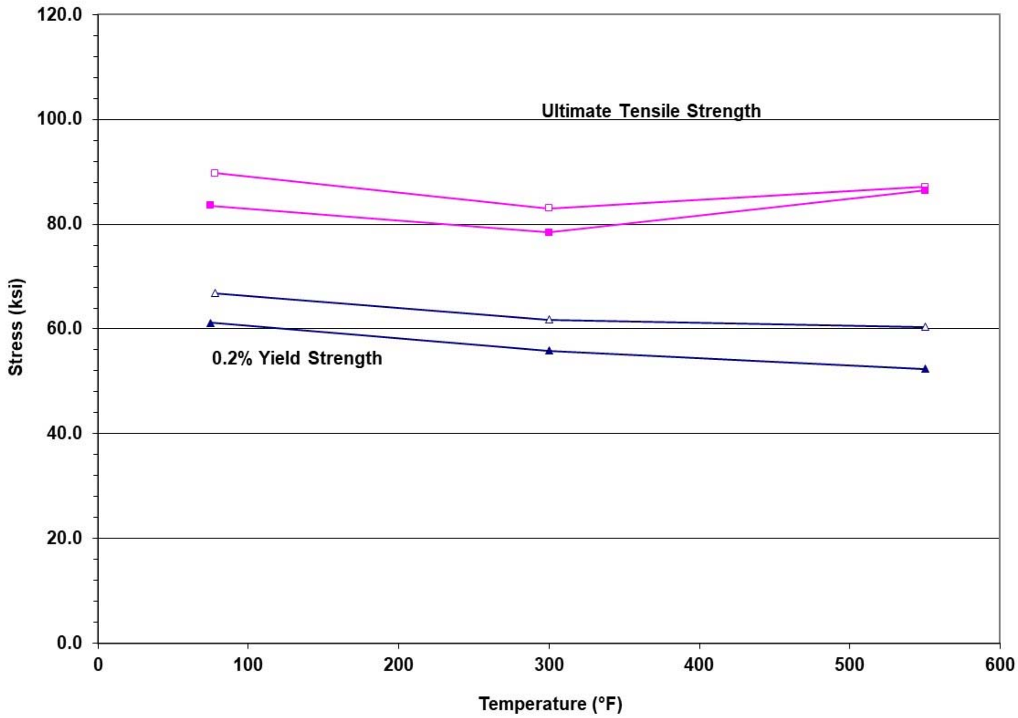


Figure 5-17 Tensile Properties for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation)



Legend: ▲, ●, and ■ are unirradiated
Δ, ○, and □ are irradiated to 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV)

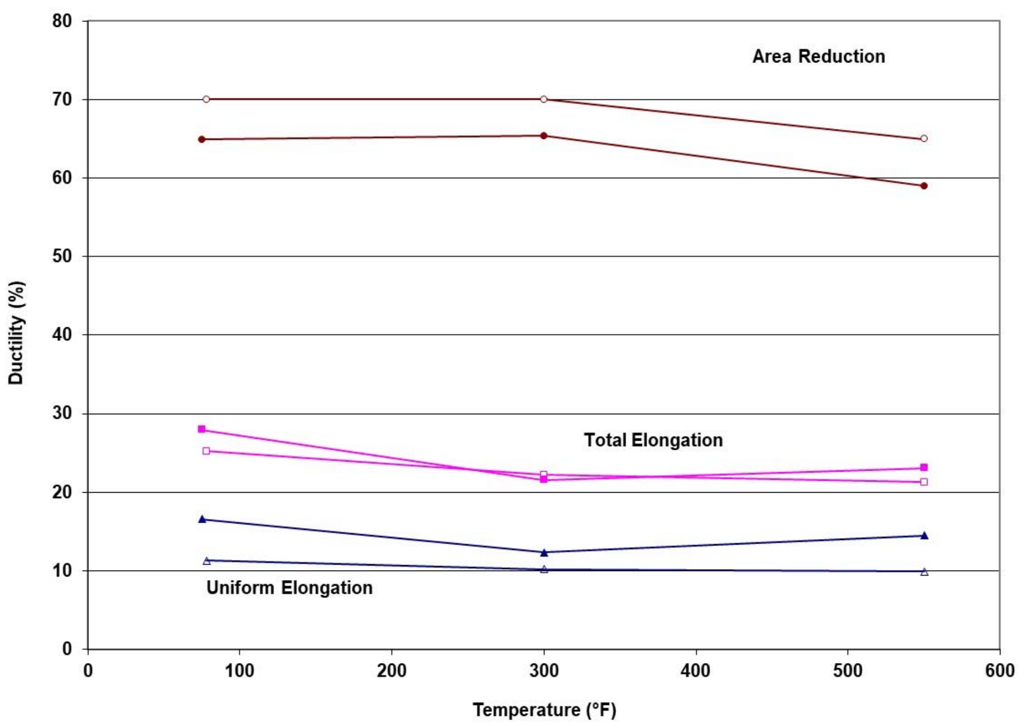
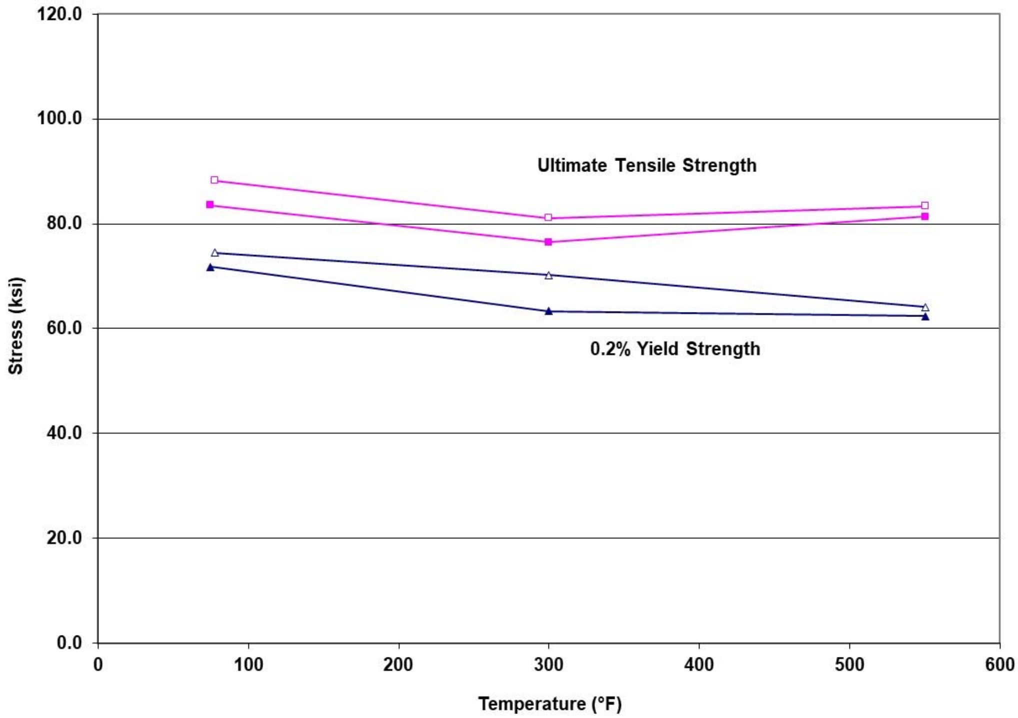


Figure 5-18 Tensile Properties for Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation)



Legend: ▲, ●, and ■ are unirradiated
Δ, ○, and □ are irradiated to 6.04 x 10^{18} n/cm^2 (E > 1.0 MeV)

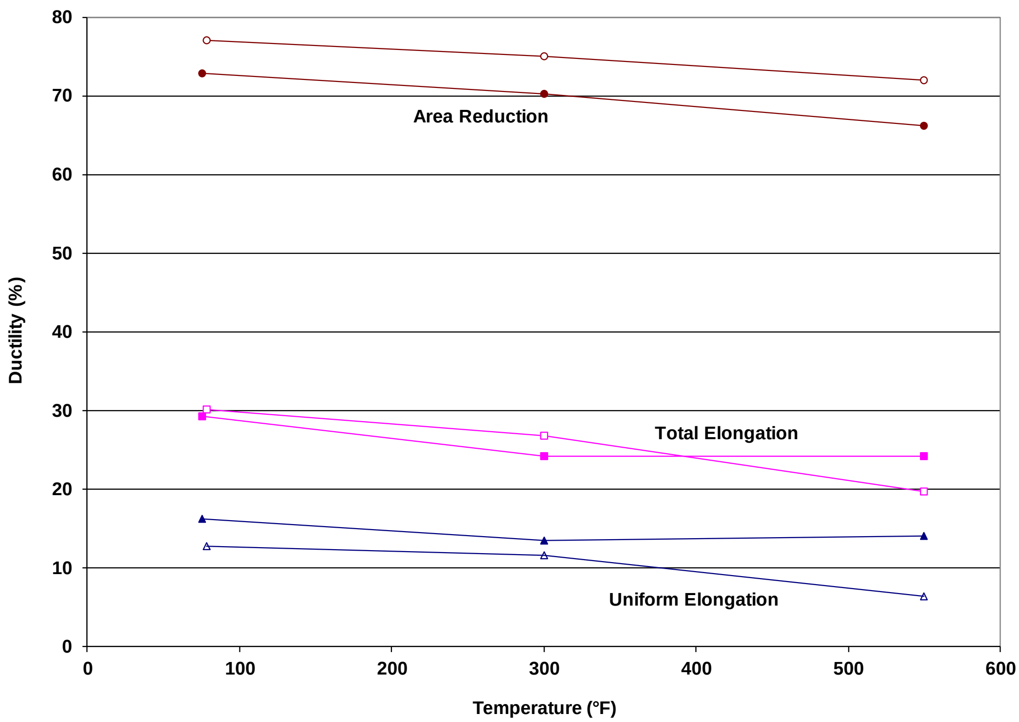


Figure 5-19 Tensile Properties for Watts Bar Unit 2 Reactor Vessel Surveillance Program Weld Material (Heat # 895075)

BL1 tested at 78°F

BL2 tested at 300°F

BL3 tested at 550°F

Figure 5-20 Fractured Tensile Specimens from Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Tangential Orientation) [Scale in 1/10th of inch]

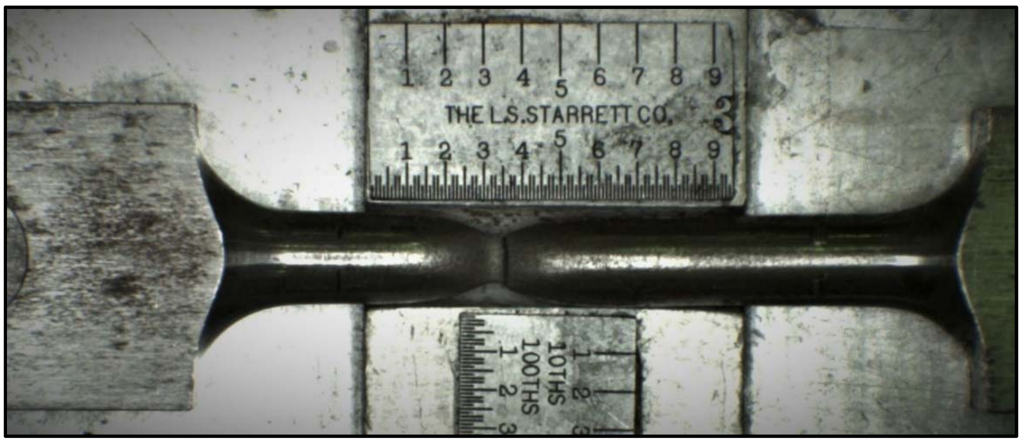
BT1 tested at 78°F

BT2 tested at 300°F

BT3 tested at 550°F

Figure 5-21 Fractured Tensile Specimens from Watts Bar Unit 2 Reactor Vessel Intermediate Shell Forging 05 (Axial Orientation) [Scale in 1/10th of inch]

BW1 tested at 78°F



BW2 tested at 300°F

BW3 tested at 550°F

Figure 5-22 Fractured Tensile Specimens from Watts Bar Unit 2 Reactor Vessel Surveillance Program Weld Material (Heat # 895075) [Scale in 1/10th of inch]

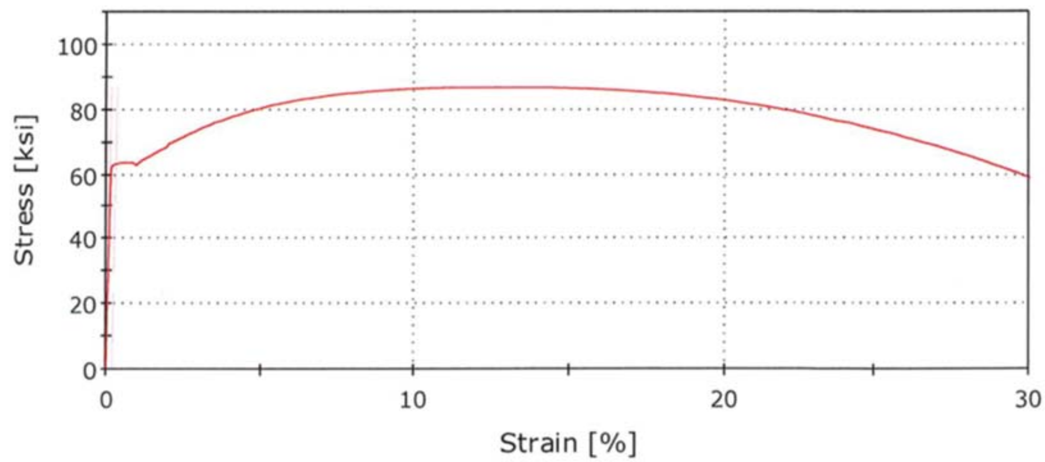


Figure 5-23 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BL1, Tested at 78°F

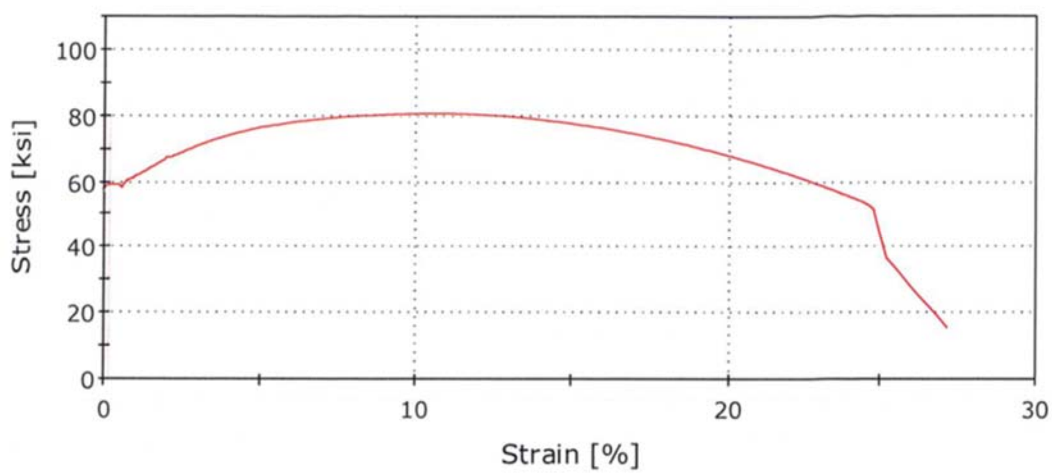


Figure 5-24 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BL2, Tested at 300°F

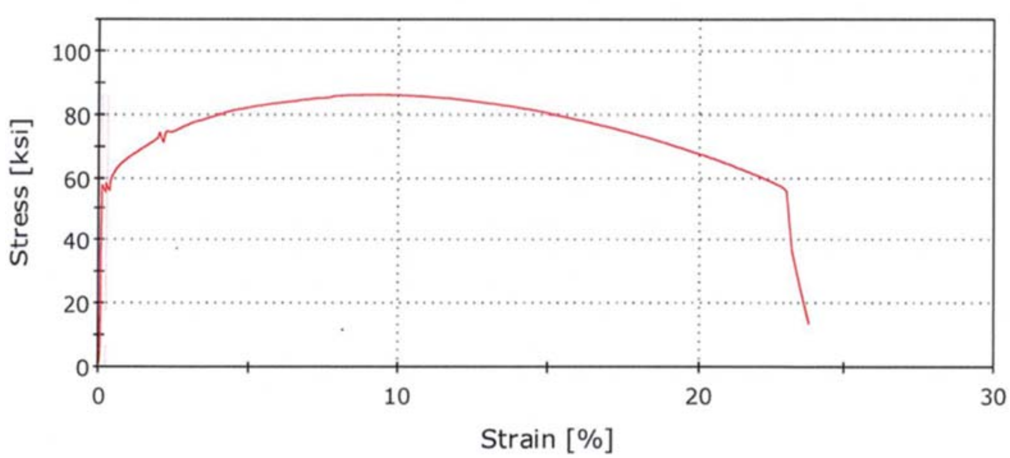


Figure 5-25 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BL3, Tested at 550°F

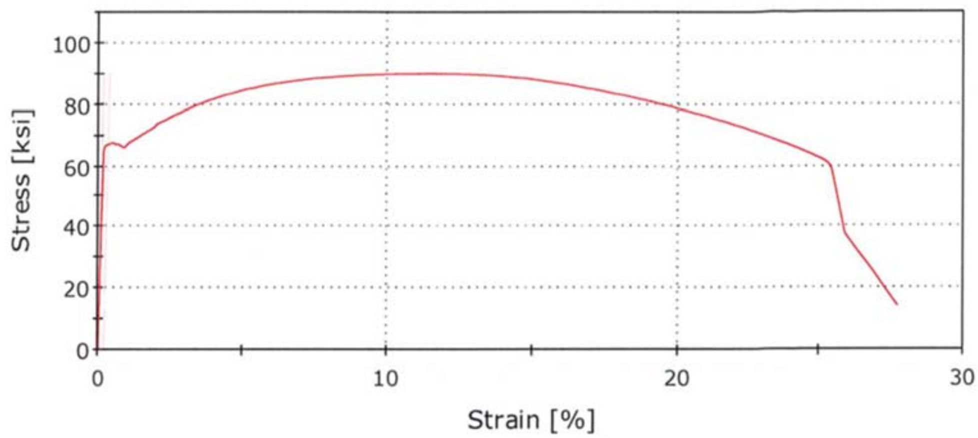


Figure 5-26 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BT1, Tested at 78°F

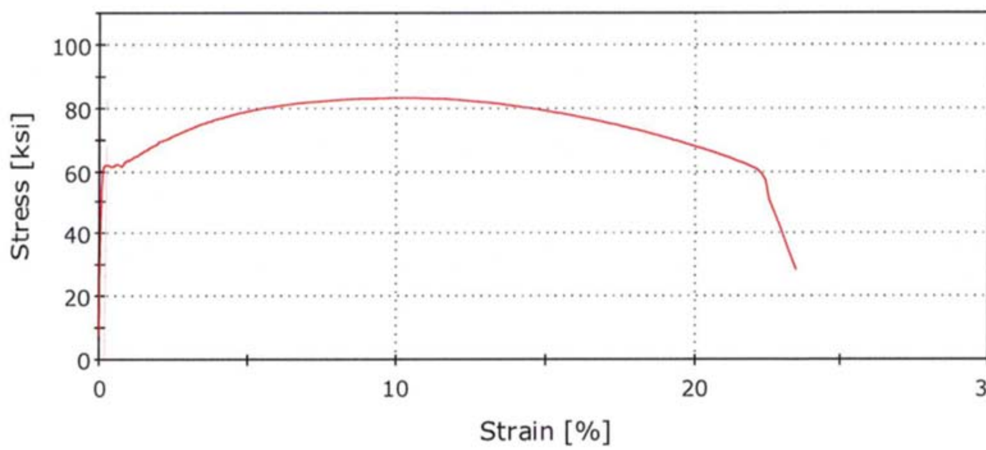


Figure 5-27 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BT2, Tested at 300°F

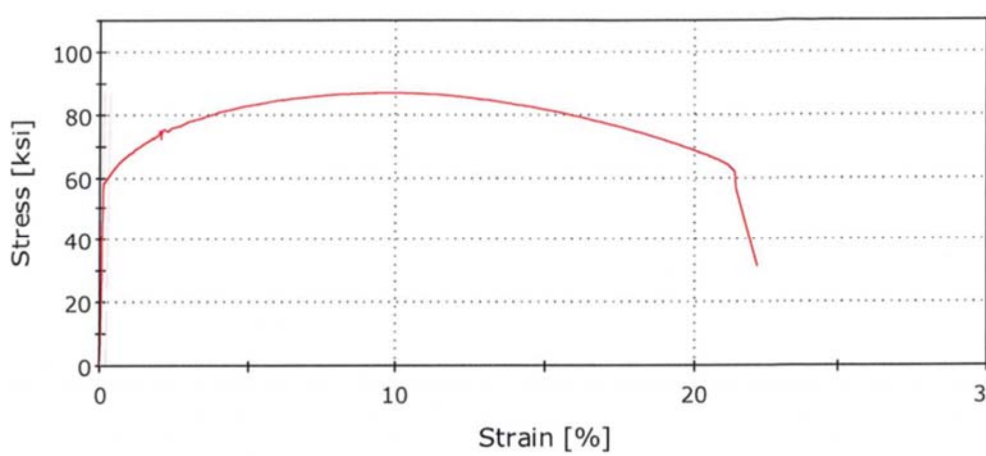


Figure 5-28 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BT3, Tested at 550°F

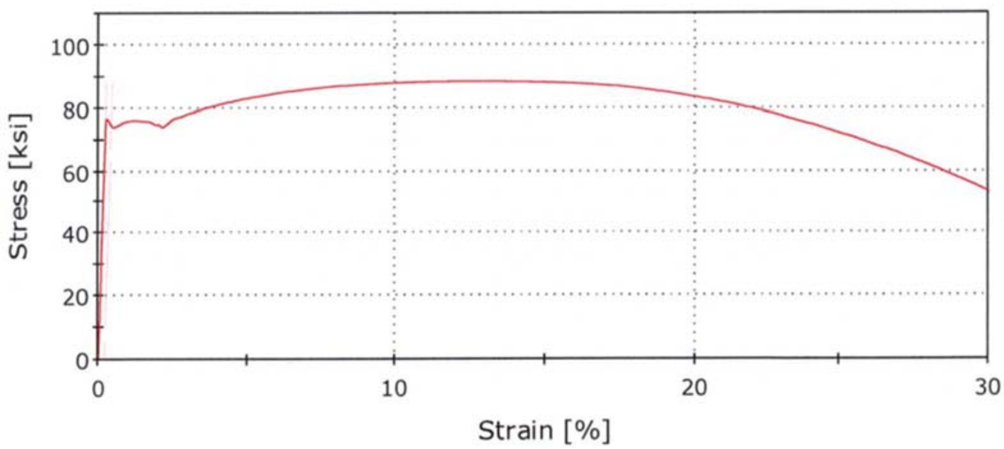


Figure 5-29 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BW1, Tested at 78°F

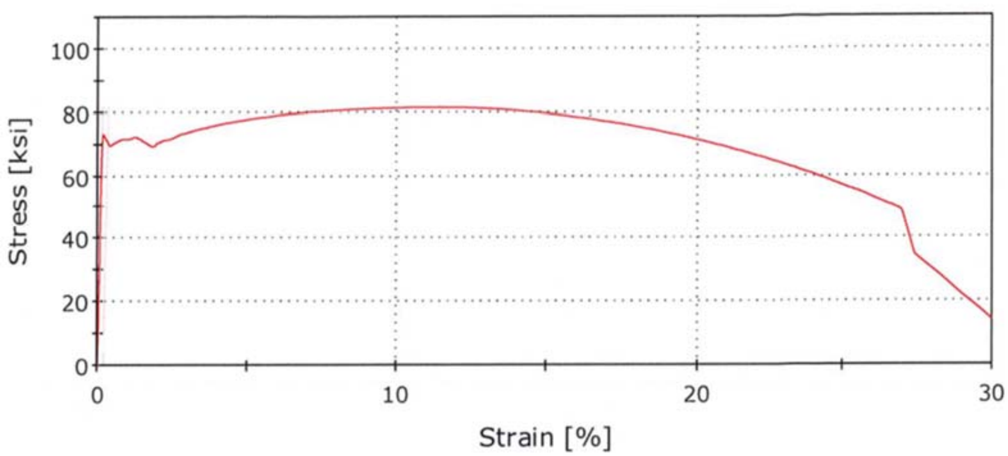


Figure 5-30 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BW2, Tested at 300°F

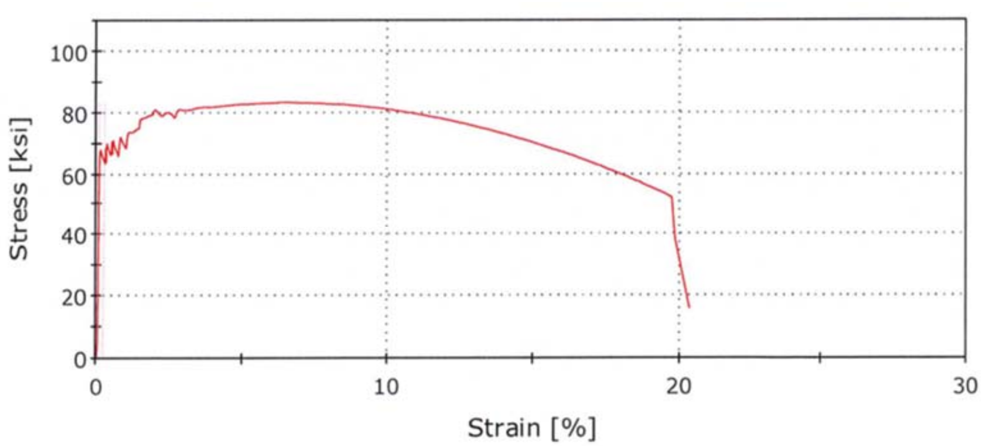


Figure 5-31 Engineering Stress-Strain Curve for Watts Bar Unit 2, Capsule U, Tensile Specimen BW3, Tested at 550°F

6 RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 INTRODUCTION

This section describes a discrete ordinates (S_n) transport analysis performed for the Watts Bar Unit 2 reactor to determine the neutron radiation environment within the reactor pressure vessel and surveillance capsules. In this analysis, fast neutron exposure parameters in terms of fast neutron (E > 1.0 MeV) fluence and iron atom displacements (dpa) were established on a plant- and fuel-cycle-specific basis. An evaluation of the most recent dosimetry sensor set from Capsule U, withdrawn at the end of the 2nd plant operating cycle, is provided. Comparisons of the results from the dosimetry evaluations with the analytical predictions served to validate the plant-specific neutron transport calculations. These validated calculations subsequently form the basis for projections of the neutron exposure of the reactor pressure vessel for operating periods extending to 36 effective full-power years (EFPY).

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

Because of this potential shift away from a threshold fluence toward an energy-dependent damage function for data correlation, ASTM Standard Practice E853-18, "Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results," [Ref. 14] recommends reporting displacements per iron atom along with fluence (E > 1.0 MeV) to provide a database for future reference. The energy-dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693-94, "Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements per Atom" [Ref. 15]. The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the reactor vessel wall has been promulgated in Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [Ref. 1].

All of the calculations and dosimetry evaluations described in this section and in Appendix A were based on nuclear cross-section data derived from ENDF/B-VI. Furthermore, the neutron transport and dosimetry evaluation methodologies follow the guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Ref. 16]. Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET" [Ref. 17].

6.2 DISCRETE ORDINATES ANALYSIS

The arrangement of the surveillance capsules in the Watts Bar Unit 2 reactor vessel is shown in Figure 4-1. Six irradiation capsules attached to the neutron pad are included in the reactor design that constitutes the reactor vessel surveillance program. Capsules U, X, V, Y, W, and Z are located at azimuthal angles of 56.0°, 236.0°, 58.5°, 238.5°, 124.0°, and 304.0°, respectively. These full-core positions correspond to the following octant symmetric locations represented in Figure 6-1 through Figure 6-3: 34.0° from the core cardinal axes (for the 56.0° and 236.0° dual surveillance capsule holder locations found in octants with a 20.0° neutron pad segment); 31.5° from the core cardinal axes (for the 58.5° and 238.5° dual surveillance capsule holder locations found in octants with a 20.0° neutron pad segment); and 34.0° from the core cardinal axes (for the 304.0° and the 124.0° single surveillance capsule holder locations found in octants with a 17.5° neutron pad segment). The stainless steel specimen containers are 1.182-inch by 1-inch and are approximately 56 inches in height. The containers are positioned axially such that the test specimens are centered on the core midplane, thus spanning the central 5 feet of the 12-foot high reactor core.

From a neutronic standpoint, the surveillance capsules and associated support structures are significant. The presence of these materials has a significant effect on both the spatial distribution of neutron exposure rate and the neutron spectrum in the vicinity of the capsules. However, the capsules are far enough apart that they do not interfere with one another. In order to determine the neutron environment at the test specimen location, the capsules themselves must be included in the analytical model.

In performing the fast neutron exposure evaluations for the Watts Bar Unit 2 reactor vessel and surveillance capsules, plant-specific 3D forward transport calculations were carried out to directly solve for the space- and energy-dependent neutron exposure rate, $\phi(r,\theta,z,E)$.

For the Watts Bar Unit 2 transport calculations, the models depicted in Figure 6-1 through Figure 6-3 were utilized. The reactor is octant symmetric with three different neutron pad and surveillance capsule configurations: octants with 20.0° neutron pads and surveillance capules located at 31.5° and 34°, octants with 17.5° neutron pads and a single surveillance capsule at 34.0°, and octants with 15.0° neutron pads and without surveillance capsules.

Each octant model contained a representation of the reactor core, the reactor internals, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological shield wall. These models formed the basis for the calculated results and enabled making comparisons to the surveillance capsule dosimetry evaluations. In developing these analytical models, nominal design dimensions were generally employed for the various structural components. In addition, water temperatures, and hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full-power operating conditions. The coolant densities were treated on a fuel-cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, et cetera.

A section view of the RAPTOR-M3G model of the Watts Bar Unit 2 reactor is shown in Figure 6-4. The model extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation approximately eight feet below the active fuel to five feet above the active fuel.

Each of the three RAPTOR-M3G models consisted of 200 radial mesh, 132 azimuthal mesh, and 347 vertical mesh. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the RAPTOR-M3G calculations was set at a value of 0.001.

The core power distributions used in the plant-specific transport analysis for the first 2 fuel cycles at Watts Bar Unit 2 included cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions. Actual operating characteristics through Cycle 2 have been evaluated; projections of future neutron exposure are based upon expected core loading patterns and operating characteristics for the following five fuel cycles. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of fuel-cycle-averaged neutron exposure rate, which when multiplied by the appropriate fuel cycle length, generated the incremental fast neutron exposure for each fuel cycle. In constructing these core source distributions, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of individual fuel assemblies. From these assembly-dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

All of the transport calculations supporting this analysis were carried out using the RAPTOR-M3G discrete ordinates code and the BUGLE-96 cross-section library, as described in [Ref. 17]. The BUGLE-96 library provides a coupled 47-neutron, 20-gamma-group cross-section data set produced specifically for light-water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion, and angular discretization was modeled with an S_{12} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

Selected results from the neutron transport analyses are provided in Table 6-1 through Table 6-8. In Table 6-1, the calculated exposure rates expressed in terms of fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate are given at the radial and azimuthal center of the surveillance capsule positions. Integrated neutron exposure levels are presented in Table 6-2 in terms of fast neutron (E > 1.0 MeV) fluence and Table 6-3 in terms of iron dpa. These results, representative of the average axial exposure of the material specimens, establish the calculated exposure of the surveillance capsules to date and projected into the future.

Neutron exposure data pertinent to selected pressure vessel materials are given in Table 6-4 and Table 6-5 for fast neutron (E > 1.0 MeV) fluence rate and fluence. Similar data are provided in Table 6-6 and Table 6-7 for dpa/s and dpa. The data presented represent the maximum neutron exposure experienced by the reactor pressure vessel (RPV) materials that will constitute inputs to the reactor vessel integrity analysis. The reported data considers both the inner and outer radius of the RPV base metal, and accounts for the possibility of higher neutron exposure values occurring on the outer surface of the RPV (as compared to the inner surface) for materials that are distant from the active core. In each case, the data are provided for each operating cycle of the Watts Bar Unit 2 reactor. Note that, for any given fuel cycle, the location of the maximum neutron exposure rate may or may not coincide with the location of the maximum neutron exposure.

These data tabulations include both plant- and fuel-cycle-specific calculated neutron exposures at the end of Cycle 2 and projections to 32 and 36 EFPY. Projections of neutron exposure beyond the end of Cycle 7 are based on the expected core loading pattern and operating characteristics of Cycle 7. The projections of future exposure account for an anticipated power uprate from 3411 MWt to 3459 MWt occurring at the beginning of Cycle 4, and the presence of Tritium-Producing Burnable Absorber Rods (TPBARs) in the fuel for Cycle 4 and beyond.

Updated lead factors for the Watts Bar Unit 2 surveillance capsules are provided in Table 6-8. The capsule lead factor is defined as the ratio of the calculated fluence (E > 1.0 MeV) at the geometric radial and azimuthal center of the surveillance capsule to the corresponding maximum calculated fluence at the pressure vessel clad/base metal interface.

6.3 NEUTRON DOSIMETRY

The validity of the calculated neutron exposures previously reported in Section 6.2 is demonstrated by a direct comparison against the measured sensor reaction rates and a least-squares evaluation performed for each of the capsule dosimetry sets. However, since the neutron dosimetry measurement data merely serve to validate the calculated results, only the direct comparison of measured-to-calculated results for surveillance Capsule U is provided in this section of the report. For completeness, the assessment based on both direct and least-squares evaluation comparisons is documented in Appendix A.

The direct comparison of measured versus calculated fast neutron threshold reaction rates for the sensors from Capsule U, which was withdrawn from Watts Bar Unit 2 at the end of the 2nd fuel cycle, is summarized below.

Reaction	Reaction Rate (rps/atom)		M/C
	Measured (M)	Calculated (C)	
Cu-63 (n,α) Co-60	4.11E-17	4.52E-17	0.91
Fe-54 (n,p) Mn-54	5.24E-15	5.29E-15	0.99
Ni-58 (n,p) Co-58	7.21E-15	7.48E-15	0.96
U-238 (n,f) Cs-137	2.44E-14	2.98E-14	0.72
Np-237 (n,f) Cs-137	3.54E-13	3.07E-13	1.01
Average			**0.92**
Standard Deviation (%)			**12.7**

The measured-to-calculated (M/C) reaction rate ratios for the Capsule U threshold reactions range from 0.72 to 1.01, and the average M/C ratio is 0.92 ± 12.7% (1σ). This direct comparison falls within the ± 20% criterion specified in Regulatory Guide 1.190. This comparison validates the current analytical results described in Section 6.2; therefore, the calculations are deemed applicable for Watts Bar Unit 2.

6.4 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the Watts Bar Unit 2 surveillance capsule and reactor pressure vessel is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology was carried out in the following four stages:

1. **Simulator Benchmark Comparisons:** Comparisons of calculations with measurements from simulator benchmarks, including the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL) and the VENUS-1 Experiment.

2. **Operating Reactor and Calculational Benchmarks:** Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment. Also considered are comparisons of calculations performed with RAPTOR-M3G to results published in the NRC fluence calculation benchmark.

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

4. **Plant-Specific Benchmarking:** Comparisons of the plant-specific calculations with all available dosimetry results from the Watts Bar Unit 2 surveillance program.

The first phase of the methods qualification (simulator benchmark comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This phase, however, did not test the accuracy of commercial core neutron source calculations nor did it address uncertainties in operational or geometric variables that impact power reactor calculations. The second phase of the qualification (operating reactor and calculational benchmark comparisons) addressed uncertainties in these additional areas that are primarily methods-related and would tend to apply generically to all fast neutron exposure evaluations. The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations, as well as to a lack of knowledge relative to various plant-specific input parameters. The overall calculational uncertainty applicable to the Watts Bar Unit 2 analysis was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with Watts Bar Unit 2 measurements) was used solely to demonstrate the validity of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used in any way to modify the calculated surveillance capsule and pressure vessel neutron exposures previously described in Section 6.2. As such, the validation of the Watts Bar Unit 2 analytical model based on the measured plant dosimetry is completely described in Appendix A.

The following summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in Westinghouse Report WCAP-18124-NP-A, *Fluence Determination with RAPTOR-M3G and FERRET* [Ref. 17].

Description	**Capsule and Vessel IR**
Simulator Benchmark Comparisons	3%
H.B. Robinson Benchmark Comparisons	5%
Analytical Sensitivity Studies	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%
Net Calculational Uncertainty	**13%**

The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random, and no systematic bias was applied to the analytical results. The plant-specific measurement comparisons described in Appendix A support these uncertainty assessments for Watts Bar Unit 2.

The NRC-issued Safety Evaluation for WCAP-18124-NP appears in Section A of Ref. 17. The NRC identified two "Limitations and Conditions" associated with the application of RAPTOR-M3G and FERRET, which are reproduced here for convenience:

1. Applicability of WCAP-18124-NP, Revision 0 is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to the response parameters of interest (e.g. pressure-temperature limits, material stress/strain), margin assessment, or a combination thereof, for applications of the method to components including, but not limited to, the RPV upper circumferential weld and the reactor coolant system inlet and outlet nozzles and reactor vessel internal components.

2. Least squares adjustment is acceptable if the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing the discrepancy should be disqualified.

The primary purpose of this report is to describe the evaluation of a surveillance capsule. The neutron exposure values applicable to the surveillance capsules and the maximum reactor pressure vessel neutron exposure values used to derive the surveillance capsule lead factors are completely covered by the benchmarking and uncertainty analyses in WCAP-18124-NP. Therefore, Limitation #1 does not strictly apply. Note, however, that this report does contain neutron exposure values for materials that are *outside* the qualification basis of WCAP-18124-NP (i.e. "extended beltline" materials). Should values outside the qualification basis of WCAP-18124-NP be cited in future evaluations of reactor vessel integrity, additional justification should be supplied, as stated in Limitation #1.

Limitation #2 applies in situations where the least squares analysis is used to *adjust* the calculated values of neutron exposure. In this report, the least squares analysis is provided only as a supplemental check on the results of the dosimetry evaluation. The least squares analysis was *not* used to modify the calculated surveillance capsule or reactor pressure vessel neutron exposure. Therefore, Limitation #2 does not apply.

Table 6-1 Calculated Neutron Exposure Rates at the Geometric Center of the Surveillance Capsules

Cycle	Operating Time (EFPY)	Neutron (E > 1.0 MeV) Fluence Rate (n/cm^2-s)			Iron Atom Displacement Rate (dpa/s)		
		34.0° Single	31.5° Dual	34.0° Dual	34.0° Single	31.5° Dual	34.0° Dual
1	0.74	1.09E+11	9.24E+10	1.10E+11	2.23E-10	1.85E-10	2.24E-10
2	2.00	8.70E+10	7.36E+10	8.75E+10	1.77E-10	1.47E-10	1.77E-10

Table 6-2 Calculated Fast Neutron (E > 1.0 MeV) Fluence at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Neutron (E > 1.0 MeV) Fluence (n/cm^2)					
		Capsule U (34.0° Dual)	Capsule V (31.5° Dual)	Capsule W (34.0° Single)	Capsule X (34.0° Dual)	Capsule Y (31.5° Dual)	Capsule Z (34.0° Single)
1	0.74	2.57E+18	2.16E+18	2.56E+18	2.57E+18	2.16E+18	2.56E+18
2	2.00	6.04E+18	5.07E+18	6.00E+18	6.04E+18	5.07E+18	6.00E+18
Future	32.00	-	7.84E+19	9.10E+19	9.16E+19	7.84E+19	9.10E+19
Future	36.00	-	8.83E+19	1.03E+20	1.03E+20	8.83E+19	1.03E+20

Table 6-3 Calculated Iron Atom Displacements (dpa) at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacements (dpa)					
		Capsule U (34.0° Dual)	Capsule V (31.5° Dual)	Capsule W (34.0° Single)	Capsule X (34.0° Dual)	Capsule Y (31.5° Dual)	Capsule Z (34.0° Single)
1	0.74	5.23E-03	4.33E-03	5.22E-03	5.23E-03	4.33E-03	5.22E-03
2	2.00	1.23E-02	1.01E-02	1.22E-02	1.23E-02	1.01E-02	1.22E-02
Future	32.00	-	1.56E-01	1.85E-01	1.85E-01	1.56E-01	1.85E-01
Future	36.00	-	1.76E-01	2.08E-01	2.09E-01	1.76E-01	2.08E-01

Table 6-4 Calculated Neutron Fluence Rate for Selected Pressure Vessel Materials

Cycle	Cumulative Operating Time (EFPY)	Maximum Neutron (E > 1.0 MeV) Fluence Rate (n/cm^2-s)			
		Bottom Head Peel 02 to Bottom Head Ring 03 Circ. Weld	Bottom Head Ring 03 to Lower Shell 04 Circ. Weld[a]	Lower Shell 04	Lower Shell 04 to Int. Shell 05 Circ. Weld
1	0.74	5.68E+06	3.12E+09	2.31E+10	2.19E+10
2	2.00	4.80E+06	2.62E+09	1.88E+10	1.79E+10

Cycle	Cumulative Operating Time (EFPY)	Maximum Neutron (E > 1.0 MeV) Fluence Rate (n/cm^2-s)			
		Int. Shell 05	Int. Shell 05 to Upper Shell 06 Circ. Weld[b]	Inlet Nozzle to Upper Shell 06 Weld (Lowest Extent)	Outlet Nozzle to Upper Shell 06 Weld (Lowest Extent)
1	0.74	2.19E+10	5.47E+08	2.42E+07	1.22E+07
2	2.00	1.82E+10	5.05E+08	2.37E+07	1.12E+07

Notes:

(a) The Bottom Head Ring 03 to Lower Shell 04 Circumferential Weld exposure value is representative of the maximum exposure to the Bottom Head Ring 03.

(b) The Intermediate Shell 05 to Upper Shell 06 Circumferential Weld exposure value is representative of the maximum exposure to Upper Shell 06.

Table 6-5 Calculated Neutron Fluence for Selected Pressure Vessel Materials

Cycle	Cumulative Operating Time (EFPY)	Maximum Neutron (E > 1.0 MeV) Fluence (n/cm^2) Bottom Head Peel 02 to Bottom Head Ring 03 Circ. Weld	Bottom Head Ring 03 to Lower Shell 04 Circ. Weld[(a)]	Lower Shell 04	Lower Shell 04 to Int. Shell 05 Circ. Weld
1	0.74	1.33E+14	7.29E+16	5.40E+17	5.11E+17
2	2.00	3.23E+14	1.77E+17	1.28E+18	1.22E+18
Future	32.00	4.93E+15	2.47E+18	1.94E+19	1.83E+19
Future	36.00	5.55E+15	2.78E+18	2.19E+19	2.06E+19

Cycle	Cumulative Operating Time (EFPY)	Maximum Neutron (E > 1.0 MeV) Fluence (n/cm^2) Int. Shell 05	Int. Shell 05 to Upper Shell 06 Circ. Weld[(b)]	Inlet Nozzle to Upper Shell 06 Weld (Lowest Extent)	Outlet Nozzle to Upper Shell 06 Weld (Lowest Extent)
1	0.74	5.12E+17	1.28E+16	5.64E+14	2.85E+14
2	2.00	1.23E+18	3.28E+16	1.50E+15	7.06E+14
Future	32.00	1.86E+19	5.12E+17	2.40E+16	1.16E+16
Future	36.00	2.10E+19	5.75E+17	2.70E+16	1.31E+16

Notes:

(a) The Bottom Head Ring 03 to Lower Shell 04 Circumferential Weld exposure value is representative of the maximum exposure to the Bottom Head Ring 03.

(b) The Intermediate Shell 05 to Upper Shell 06 Circumferential Weld exposure value is representative of the maximum exposure to Upper Shell 06.

Table 6-6 Calculated dpa/s for Selected Pressure Vessel Materials

Cycle	Cumulative Operating Time (EFPY)	Maximum Iron Atom Displacement Rate (dpa/s)			
		Bottom Head Peel 02 to Bottom Head Ring 03 Circ. Weld	Bottom Head Ring 03 to Lower Shell 04 Circ. Weld[(a)]	Lower Shell 04	Lower Shell 04 to Int. Shell 05 Circ. Weld
1	0.74	3.94E-14	4.97E-12	3.67E-11	3.49E-11
2	2.00	3.29E-14	4.17E-12	2.98E-11	2.86E-11

Cycle	Cumulative Operating Time (EFPY)	Maximum Iron Atom Displacement Rate (dpa/s)			
		Int. Shell 05	Int. Shell 05 to Upper Shell 06 Circ. Weld[(b)]	Inlet Nozzle to Upper Shell 06 Weld (Lowest Extent)	Outlet Nozzle to Upper Shell 06 Weld (Lowest Extent)
1	0.74	3.49E-11	9.10E-13	1.06E-13	7.84E-14
2	2.00	2.89E-11	8.36E-13	9.07E-14	6.73E-14

Notes:

(a) The Bottom Head Ring 03 to Lower Shell 04 Circumferential Weld exposure value is representative of the maximum exposure to the Bottom Head Ring 03.

(b) The Intermediate Shell 05 to Upper Shell 06 Circumferential Weld exposure value is representative of the maximum exposure to Upper Shell 06.

Table 6-7 Calculated dpa for Selected Pressure Vessel Materials

Cycle	Cumulative Operating Time (EFPY)	Maximum Iron Atom Displacements (dpa)			
		Bottom Head Peel 02 to Bottom Head Ring 03 Circ. Weld	Bottom Head Ring 03 to Lower Shell 04 Circ. Weld[(a)]	Lower Shell 04	Lower Shell 04 to Int. Shell 05 Circ. Weld
1	0.74	9.19E-07	1.16E-04	8.57E-04	8.15E-04
2	2.00	2.22E-06	2.81E-04	2.04E-03	1.95E-03
Future	32.00	3.39E-05	3.94E-03	3.04E-02	2.91E-02
Future	36.00	3.81E-05	4.43E-03	3.42E-02	3.28E-02

Cycle	Cumulative Operating Time (EFPY)	Maximum Iron Atom Displacements (dpa)			
		Int. Shell 05	Int. Shell 05 to Upper Shell 06 Circ. Weld[(b)]	Inlet Nozzle to Upper Shell 06 Weld (Lowest Extent)	Outlet Nozzle to Upper Shell 06 Weld (Lowest Extent)
1	0.74	8.16E-04	2.13E-05	2.47E-06	1.83E-06
2	2.00	1.96E-03	5.44E-05	6.06E-06	4.50E-06
Future	32.00	2.94E-02	8.50E-04	9.94E-05	7.38E-05
Future	36.00	3.31E-02	9.55E-04	1.12E-04	8.31E-05

Notes:

(a) The Bottom Head Ring 03 to Lower Shell 04 Circumferential Weld exposure value is representative of the maximum exposure to the Bottom Head Ring 03.

(b) The Intermediate Shell 05 to Upper Shell 06 Circumferential Weld exposure value is representative of the maximum exposure to Upper Shell 06.

Table 6-8 Calculated Surveillance Capsule Lead Factors

Cycle	Cumulative Operating Time (EFPY)	Lead Factor					
		Capsule U (34.0° Dual)	Capsule V (31.5° Dual)	Capsule W (34.0° Single)	Capsule X (34.0° Dual)	Capsule Y (31.5° Dual)	Capsule Z (34.0° Single)
1	0.74	4.76	4.00	4.73	4.76	4.00	4.73
2	2.00	4.70	3.95	4.67	4.70	3.95	4.67
Future	7.34[(a)]	-	3.95	4.66	4.69	3.95	4.66
Future	32.00	-	4.04	4.69	4.72	4.04	4.69
Future	36.00	-	4.03	4.68	4.71	4.03	4.68

Notes:

(a) 7.34 EFPY is the expected reactor operating time following the end of Cycle 6, and represents the interval at which Capsule W should be removed. See Section 7 for more information about the recommended schedule for removing surveillance capsules.

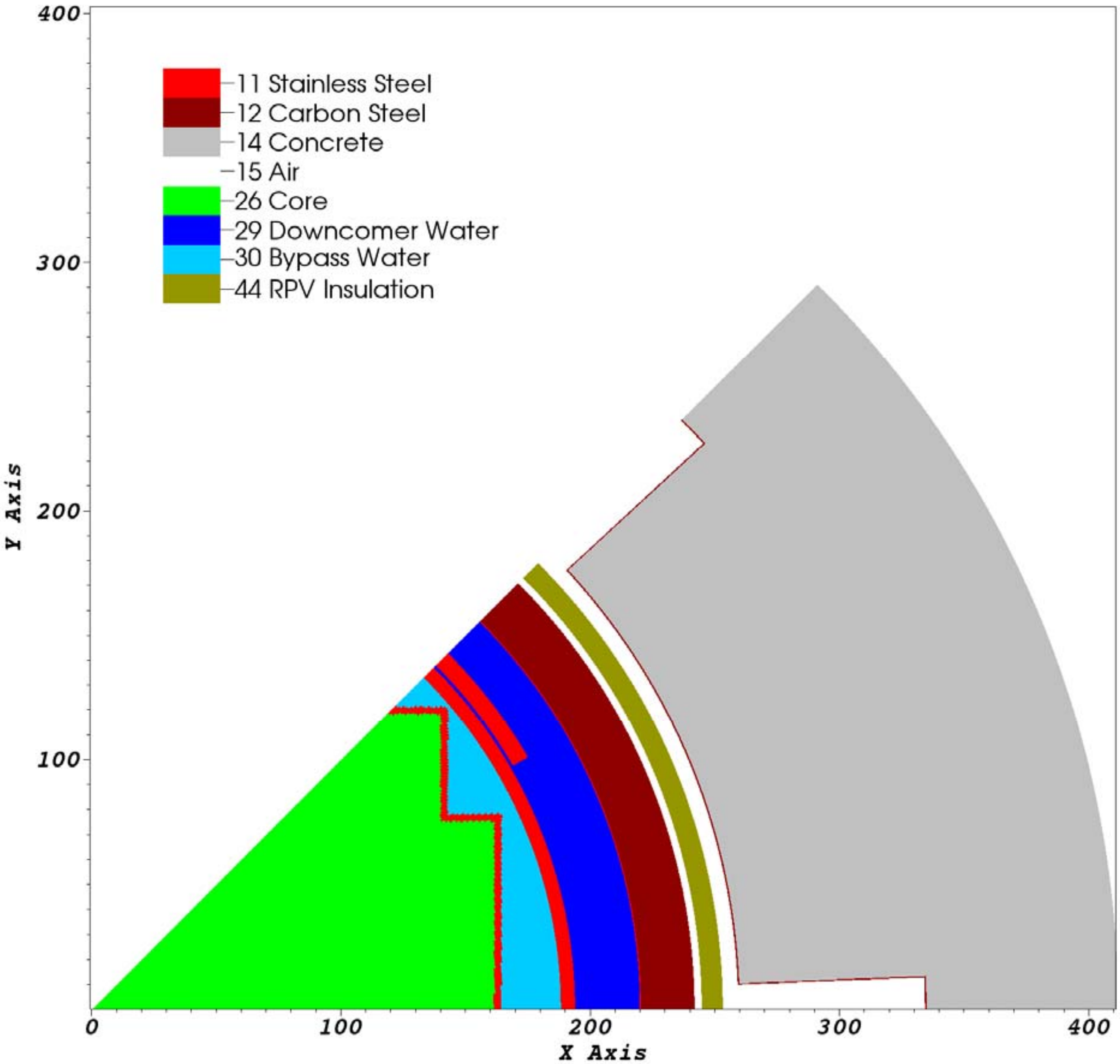


Figure 6-1 Watts Bar Unit 2 Plan View of the Reactor Geometry at the Core Midplane 15.0° Neutron Pad Configuration

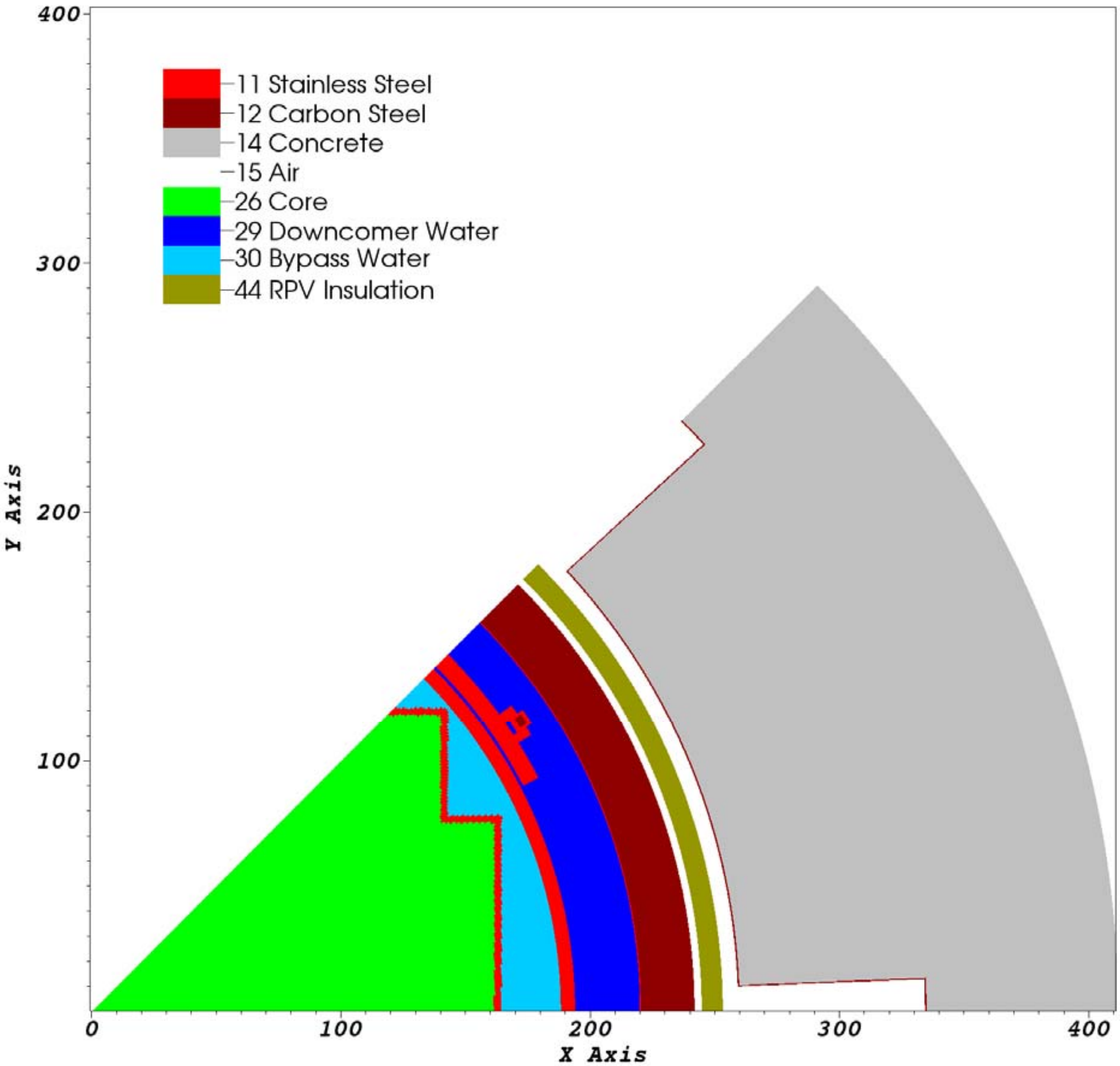


Figure 6-2 Watts Bar Unit 2 Plan View of the Reactor Geometry at the Core Midplane 17.5° Neutron Pad Configuration

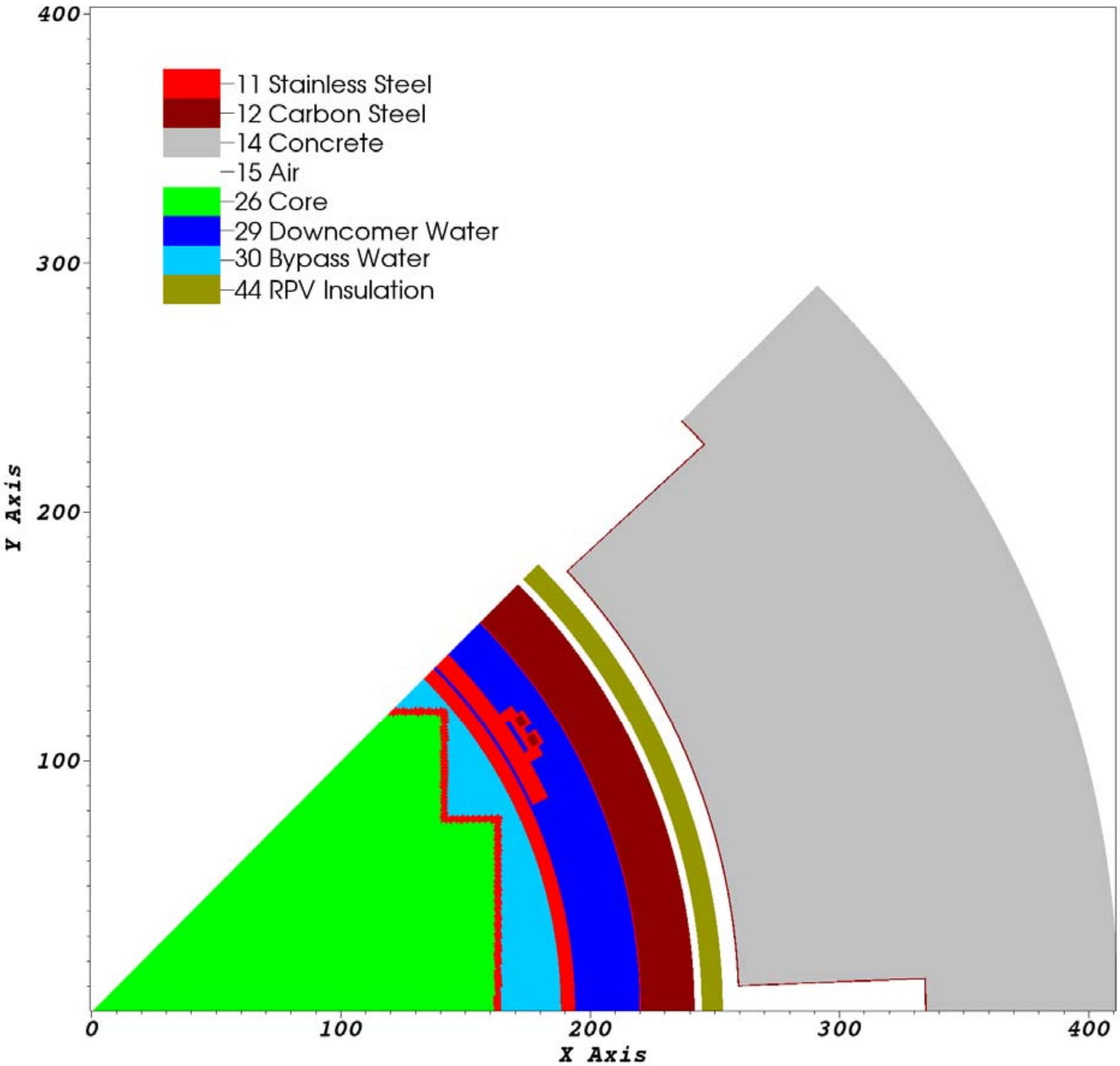


Figure 6-3 Watts Bar Unit 2 Plan View of the Reactor Geometry at the Core Midplane 20.0° Neutron Pad Configuration

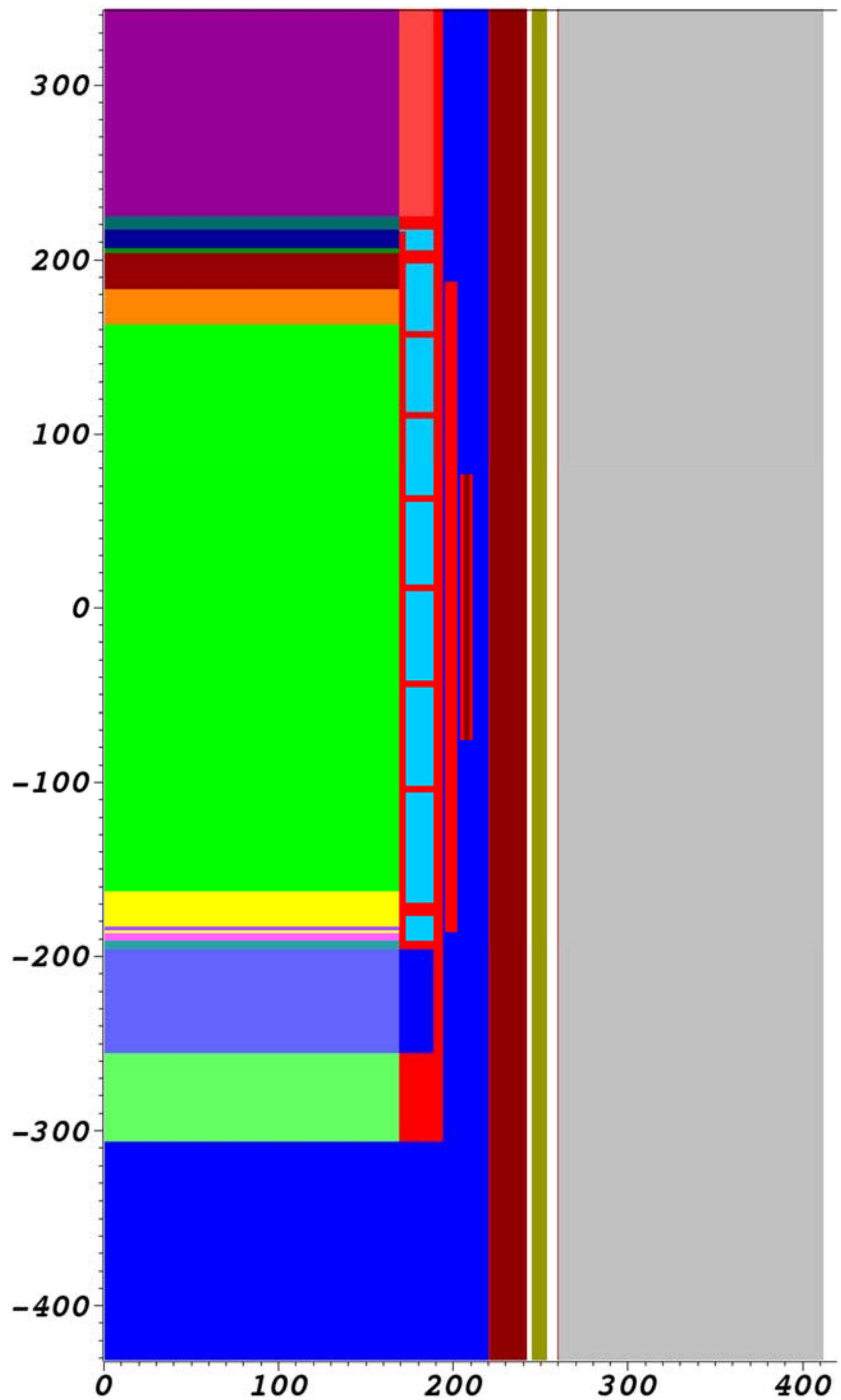


Figure 6-4 Watts Bar Unit 2 Section View of the Reactor Geometry at the 34.0° Azimuthal Angle - 20.0° Neutron Pad Configuration

7 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule (Table 7-1) meets the requirements of ASTM E185-82 [Ref. 8]. Note that it is recommended for future capsule(s) to be removed from the Watts Bar Unit 2 reactor vessel.

Table 7-1 Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Lead Factor	Withdrawal EFPY[(a)]	Fluence (n/cm^2, E > 1.0 MeV)
U	Dual 34°	4.70	2.0 EFPY (EOC 2)	0.604 x 10^{19}
W	Single 34°	4.66	7.0 EFPY[(b)] (EOC 6)	1.94 x 10^{19}
X	Dual 34°	4.69	7.0 EFPY to 13.7 EFPY[(c)]	1.94 x 10^{19} to 3.88 x 10^{19} [(c)]
Z	Single 34°	4.69	Standby[(d)]	- - -[(d)]
V	Dual 31.5°	4.04	Standby[(d)]	- - -[(d)]
Y	Dual 31.5°	4.04	Standby[(d)]	- - -[(d)]

Notes:

(a) Effective full-power years (EFPY) from plant startup. The projected EFPY and end-of-cycle (EOC) values assume the Measurement Uncertainty Recapture (MUR) uprate and TPBARs are implemented at the beginning of Cycle 4.

(b) This capsule should be withdrawn at the outage nearest to but following 7.0 EFPY of operation. This outage is projected to occur at EOC 6. However, the capsule withdrawal should be based on actual plant EFPY as opposed to the estimated cycle.

(c) Capsule X must be withdrawn between 7.0 EFPY and 13.7 EFPY in order satisfy the requirements of the third capsule for EOL per ASTM E185-82 [Ref. 8]. However, if the capsule is removed after 11.7 EFPY (but still before 13.7 EFPY), this capsule will satisfy the requirements of the third capsule for both end of license (EOL, 40 years) and end of a potential license extension (60 years) per ASTM E185-82 [Ref. 8] and NUREG-1801, Revision 2 [Ref. 18]. Thus, if possible, the capsule should be pulled between 11.7 EFPY and 13.7 EFPY, but the capsule must be pulled between 7.0 EFPY and 13.7 EFPY. The removal EFPY of the third capsule should be revisited at a later date, such as after Capsule W is removed.

(d) Capsules Z, V, and Y should remain in the reactor. The potential for future removal and storage of some or all of these standby capsules should be revisited at a later date, such as with the withdrawal and testing of Capsule W. If additional metallurgical data is needed, such as in support of a first (60 years) or second (80 years) license renewal, withdrawal and testing of these capsules should be considered when planning for withdrawal of Capsule X in anticipation of any license renewal effort. Note, ASTM E185-82 and NUREG-1801, Revision 2 recommend that the capsules not experience twice the end of life fluence. Therefore, the potential for future removal and storage of some or all of these standby capsules should be revisited at a later date, such as with the withdrawal and testing of Capsule W.

8 REFERENCES

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2. Westinghouse Report WCAP-9455, Revision 4, "Tennessee Valley Authority Watts Bar Unit No. 2 Reactor Vessel Radiation Surveillance Program," August 2019.

3. ASTM E185-73, *Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,* 1973.

4. Appendix G of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, *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,* ASTM.

6. ASTM E399, *Standard Test Method for Linear-Elastic Plane-Strain Fracture Toughness* K_{Ic} *of Metallic Materials*, ASTM.

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APPENDIX A VALIDATION OF THE RADIATION TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

A.1 NEUTRON DOSIMETRY

Comparisons of measured dosimetry results to both the calculated and least-squares adjusted values for Capsule U are provided in this appendix. The sensor sets have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” [Ref. A-1]. One of the main purposes for providing this material is to demonstrate that the overall measurements agree with the calculated and least-squares adjusted values to within ± 20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in Section 6.2 of this report.

A.1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of Capsule U are presented. The capsule designation, location within the reactor, and time of withdrawal are as follows:

Capsule	Azimuthal Location	Withdrawal Time	Irradiation Time (EFPY)
U	56º	End of Cycle 2	2.00

The passive neutron sensors included in these evaluations are summarized as follows:

Sensor Material	Reaction Of Interest	Capsule U
Copper	Cu-63 (n,α) Co-60	X
Iron	Fe-54 (n,p) Mn-54	X
Nickel	Ni-58 (n,p) Co-58	X
Uranium-238	U-238 (n,f) Cs-137	X
Neptunium-237	Np-237 (n,f) Cs-137	X
Cobalt-Aluminum[(a)]	Co-59 (n,γ) Co-60	X

Notes:

(a) The cobalt-aluminum and uranium sensors include both bare and cadmium-covered sensors.

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

The use of passive monitors does not yield a direct measure of the energy-dependent neutron exposure rate 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 exposure rate has on the target material over the course of the irradiation period. An accurate assessment of the average neutron exposure rate incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

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

The radiometric counting of the sensors from Capsule U was carried out by Pace Analytical Services, Inc. The radiometric counting followed established ASTM procedures.

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

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

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

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/g).
N_0	=	Number of target element atoms per gram of sensor.
F	=	Atom fraction of the target isotope in the target element.
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 ϕ (E > 1.0 MeV) during irradiation period j to the time weighted average ϕ (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).
$t_{d,j}$	=	Decay time following irradiation period j (sec).

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

In the equation describing the reaction rate calculation, the ratio $[P_j]/[P_{ref}]$ accounts for month-by-month variation of reactor core power level within any given fuel cycle as well as over multiple fuel cycles. The ratio C_j, which was calculated for each fuel cycle using the transport methodology discussed in Section 6.2, accounts for the change in sensor reaction rates caused by variations in exposure rate 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, the additional C_j term should be employed. The impact of changing exposure rate levels for constant power operation can be quite significant for sensor sets that have been irradiated for many cycles in a reactor that has transitioned from non-low-leakage to low-leakage fuel management or for sensor sets contained in surveillance capsules that have been moved from one capsule location to another.

The fuel-cycle-specific neutron exposure rates and the computed values for C_j are listed in Table A-3 and Table A-4, respectively. These exposure rates represent the capsule- and cycle-dependent results at the radial and azimuthal center of the respective capsules at core midplane.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets, additional corrections were made to the U-238 measurements to account for the presence of ^{235}U impurities in the sensors, as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. Corrections were also made to the U-238 and Np-237 sensor reaction rates to account for gamma-ray-induced fission reactions that occurred over the course of the surveillance capsule irradiations. The correction factors corresponding to the Watts Bar Unit 2 fission sensor reaction rates are summarized as follows:

Correction	Capsule U
U-235 Impurity/Pu Build-in	0.8609
U-238 (γ,f)	0.9637
Net U-238 Correction	0.8296
Np-237 (γ,f)	0.9898

The correction factors were applied in a multiplicative fashion to the decay-corrected cadmium-covered uranium fission sensor reaction rates.

Results of the sensor reaction rate determinations are given in Table A-5. In Table A-5, the measured specific activities, decay-corrected saturated specific activities, and computed reaction rates for each sensor are listed.

A.1.2 Least-Squares Evaluation of Sensor Sets

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

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

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

For the least-squares evaluation of the Watts Bar Unit 2 dosimetry, the FERRET code [Ref. A-2] was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine the best-estimate values of exposure parameters (fluence rate (E > 1.0 MeV) and dpa) and their associated uncertainties.

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

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy-dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the Watts Bar Unit 2 application, the calculated neutron spectrum was obtained from the results of plant-specific neutron transport calculations described in Section 6.2 of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section A.1.1. The dosimetry reaction cross-sections and uncertainties were obtained from the SNLRML dosimetry cross-section library [Ref. A-3].

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum were input to the least-squares procedure in the form of variances and covariances. The assignment of the input uncertainties followed the guidance provided in ASTM Standard E944, "Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance" [Ref. A-4].

The following provides a summary of the uncertainties associated with the least-squares evaluation of the Watts Bar Unit 2 surveillance capsule sensor sets.

Reaction Rate Uncertainties

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

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

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

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

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

For sensors included in the Watts Bar Unit 2 surveillance program, the following uncertainties in the fission spectrum averaged cross-sections are provided in the SNLRML documentation package.

Reaction	Uncertainty
Cu-63 (n,α) Co-60	4.08–4.16%
Fe-54 (n,p) Mn-54	3.05–3.11%
Ni-58 (n,p) Co-58	4.49–4.56%
Co-59 (n,γ) Co-60	0.79–3.59%
U-238 (n,f)	0.54–0.64%
Np-237 (n,f)	10.32-10.97%

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

Calculated Neutron Spectrum

The neutron spectra inputs to the least-squares adjustment procedure were obtained directly from the results of plant-specific transport calculations for each surveillance capsule irradiation period and location. The spectrum for each capsule was input in an absolute sense (rather than as simply a relative spectral shape).

Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

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

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

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

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

Where:

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

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

The set of parameters defining the input covariance matrix for the Watts Bar Unit 2 calculated spectra was as follows:

Exposure Rate Normalization Uncertainty (R_n)	15%
Exposure Rate Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Exposure Rate Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

A.1.3 Comparisons of Measurements and Calculations

Results of the least-squares evaluations are provided in Table A-6. In these tables, measured, calculated, and best-estimate values for sensor reaction rates are given. Also provided in these tabulations are ratios of the measured reaction rates to both the calculated and least-squares adjusted reaction rates. These ratios of measured-to-calculated (M/C) and measured-to-best estimate (M/BE) illustrate the consistency of the fit of the calculated neutron energy spectra to the measured reaction rates both before and after adjustment. Additionally, comparisons of the calculated and best-estimate values of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate are tabulated along with the best estimate-to-calculated (BE/C) ratios observed for each of the capsules.

The data comparisons provided in Table A-6 show that the adjustments to the calculated spectra are relatively small and within the assigned uncertainties for the calculated spectra, measured sensor reaction rates, and dosimetry reaction cross-sections. Further, these results indicate that the use of the least-squares evaluation results in a reduction in the uncertainties associated with the exposure of the surveillance capsules. From Section 6.4 of this report, the calculational uncertainty is specified as 13% at the 1σ level.

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

In the case of the direct comparison of the measured and calculated sensor reaction rates, for the individual threshold foils considered in the least-squares analysis, the M/C comparisons of the fast neutron threshold reactions range from 0.72 to 1.01. The overall average M/C ratio is 0.92 with an associated standard deviation of 12.7%.

In the case of the comparison of the best-estimate and calculated fast neutron exposure parameters, the BE/C comparisons are 0.92 and 0.93 for fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate, respectively.

Based on these comparisons, it is concluded that the calculated fast neutron exposures provided in Section 6.2 of this report are validated for use in the assessment of the condition of the materials comprising the beltline region of the Watts Bar Unit 2 reactor pressure vessel.

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

Reaction of Interest	Atomic Weight (g/g-atom)	Target Atom Fraction	Product Half-life (days)	Fission Yield (%)	90% Response Range[(a)] (MeV)
Cu-63 (n,α) Co-60	63.546	0.6917	1925.28	-	4.53–11.0
Fe-54 (n,p) Mn-54	55.845	0.05845	312.13	-	2.27–7.54
Ni-58 (n,p) Co-58	58.693	0.68077	70.86	-	1.98–7.51
Co-59 (n,γ) Co-60	58.933	0.0015	1925.28	-	non-threshold
U-238 (n,f) Cs-137	238.051	1.00	10975.76	0.0602	1.44–6.69
Np-237 (n,f) Cs-137	237.048	1.00	10975.76	0.0627	0.68-5.61

Note:

(a) Energies between which 90% of activity is produced (U-235 fission spectrum) [Ref. A-5]

Table A-2 Monthly Thermal Generation for the Watts Bar Unit 2 Reactor

Cycle 1		Cycle 2	
Month	**MWt-h**	**Month**	**MWt-h**
May-16	30290	Dec-17	1491562
Jun-16	347922	Jan-18	2530416
Jul-16	1056046	Feb-18	2287280
Aug-16	664736	Mar-18	2532054
Sep-16	314358	Apr-18	2210328
Oct-16	1745340	May-18	2440366
Nov-16	2369963	Jun-18	1738791
Dec-16	2526323	Jul-18	2432179
Jan-17	2530416	Aug-18	2020404
Feb-17	2284824	Sep-18	2451827
Mar-17	1588980	Oct-18	2534509
Apr-17	0	Nov-18	2455101
May-17	0	Dec-18	2536965
Jun-17	819	Jan-19	2535328
Jul-17	18829	Feb-19	1829660
Aug-17	2011398	Mar-19	2534509
Sep-17	2446915	Apr-19	970907
Oct-17	2184950		
Nov-17	0		

Table A-3 Surveillance Capsule Fluence Rates for C_j Calculation, Core Midplane Elevation

Cycle	Cycle Length (EFPY)	Fluence Rate (E > 1.0 MeV, n/cm^2-s)
		CapsuleU
1	0.74	1.10E+11
2	1.26	8.75E+10
Average	--	9.59E+10

Table A-4 Surveillance Capsule C_j Factors, Core Midplane Elevation

Cycle	Cycle Length (EFPY)	C_j
		Capsule U
1	0.74	1.15
2	1.26	0.91

Table A-5 Measured Sensor Activities and Reaction Rates for Surveillance Capsule U

Sample ID	Target Isotope[a]	Product Isotope	Measured Activity[b] (dps/g)	Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
6	Cu-63	Co-60	5.71E+04	2.63E+05	4.01E-17		
12	Cu-63	Co-60	6.04E+04	2.78E+05	4.25E-17		
18	Cu-63	Co-60	5.80E+04	2.67E+05	4.08E-17	4.11E-17	4.11E-17
8	Fe-54	Mn-54	1.96E+06	3.30E+06	5.24E-15		
14	Fe-54	Mn-54	2.09E+06	3.52E+06	5.58E-15		
20	Fe-54	Mn-54	1.83E+06	3.08E+06	4.89E-15	5.24E-15	5.24E-15
7	Ni-58	Co-58	2.08E+07	4.95E+07	7.08E-15		
13	Ni-58	Co-58	2.14E+07	5.09E+07	7.29E-15		
19	Ni-58	Co-58	2.13E+07	5.07E+07	7.26E-15	7.21E-15	7.21E-15
1	U-238	Cs-137	1.75E+05	3.93E+06	2.58E-14	2.58E-14	2.14E-14
2	Np-237	Cs-137	2.21E+06	4.97E+07	3.12E-13	3.12E-13	3.09E-13
3	Co-59 (B)	Co-60	1.73E+07	7.97E+07	5.20E-12		
4	Co-59 (B)	Co-60	1.59E+07	7.33E+07	4.78E-12		
9	Co-59 (B)	Co-60	1.62E+07	7.46E+07	4.87E-12		
10	Co-59 (B)	Co-60	1.39E+07	6.40E+07	4.18E-12		
15	Co-59 (B)	Co-60	1.70E+07	7.83E+07	5.11E-12		
16	Co-59 (B)	Co-60	1.49E+07	6.86E+07	4.48E-12	4.77E-12	4.77E-12
5	Co-59	Co-60	8.73E+06	4.02E+07	2.62E-12		
11	Co-59	Co-60	9.16E+06	4.22E+07	2.75E-12		
17	Co-59	Co-60	9.11E+06	4.20E+07	2.74E-12	2.71E-12	2.71E-12

Note:

(a) (B) denotes “Bare” for the Co-59 sensors; the other Co-59 sensors are cadmium-shielded.

(b) Measured activity is decay corrected to 6/25/2019.

Table A-6 Least-Squares Evaluation of Dosimetry in Capsule U (34° Dual Position, Core Midplane, Withdrawn at the End of Cycle 2)

Capsule U – 34.0° Dual - Withdrawn EOC 2				X^2/DOF=	0.897	
Reaction	**Measured (rps/atom)**	**Calculated (rps/atom)**	**Best Estimate (rps/atom)**	**M/C**	**M/BE**	**BE/C**
Cu-63(n,a)Co-60	4.11E-17	4.52E-17	4.20E-17	0.91	0.98	0.93
Fe-54(n,p)Mn-54	5.24E-15	5.29E-15	5.02E-15	0.99	1.04	0.95
Ni-58(n,p)Co-58	7.21E-15	7.48E-15	7.07E-15	0.96	1.02	0.94
Co-59(n,g)Co-60	4.77E-12	5.03E-12	4.74E-12	0.95	1.01	0.94
Co-59(n,g)Co-60 Cd	2.70E-12	3.28E-12	2.73E-12	0.82	0.99	0.83
U-238(n,f) Cd	2.14E-14	2.98E-14	2.76E-14	0.72	0.78	0.93
Np-237(n,f) Cd	3.09E-13	3.07E-13	2.95E-13	1.01	1.04	0.96
Threshold Foil Average				**0.92**	**0.97**	**0.94**
% Standard Deviation				**12.7**	**11.3**	**1.4**

Integral Quantity	**Units**	**Calculated**	**% Unc.**	**Best Est.**	**% Unc.**	**BE/C**
Fluence Rate (E > 1.0 MeV)	n/cm^2-s	9.64E+10	13	8.91E+10	6	0.92
Iron Displacement Rate	dpa/s	1.92E-10	13	1.78E-10	8	0.93

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

Capsule	M/C Ratio				
	Cu-63 (n,α)	Fe-54 (n,p)	Ni-58 (n,p)	U-238 (n,f)	Np-237 (n,f)
U	0.91	0.99	0.96	0.72	1.01
Average	**0.91**	**0.99**	**0.96**	**0.72**	**1.01**
% Standard Deviation	-	-	-	-	-

Reaction	Average M/C	% Standard Deviation
Cu-63 (n,α)	0.91	-
Fe-54 (n,p)	0.99	-
Ni-58 (n,p)	0.96	-
U-238 (n,f)	0.72	-
Np-237 (n,f)	1.01	-
Linear Average	**0.92**	**12.7**

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

Capsule	Neutron (E > 1.0 MeV) Fluence Rate		Iron Displacement Rate	
	BE/C	% Standard Deviation	BE/C	% Standard Deviation
U	0.92	6	0.93	8
Average	**0.92**	-	**0.93**	-

A.2 REFERENCES

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

A-2 A. Schmittroth, *FERRET Data Analysis Core*, HEDL-TME 79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

A-3 RSICC Data Library Collection DLC-178, *SNLRML Recommended Dosimetry Cross-Section Compendium*, July 1994.

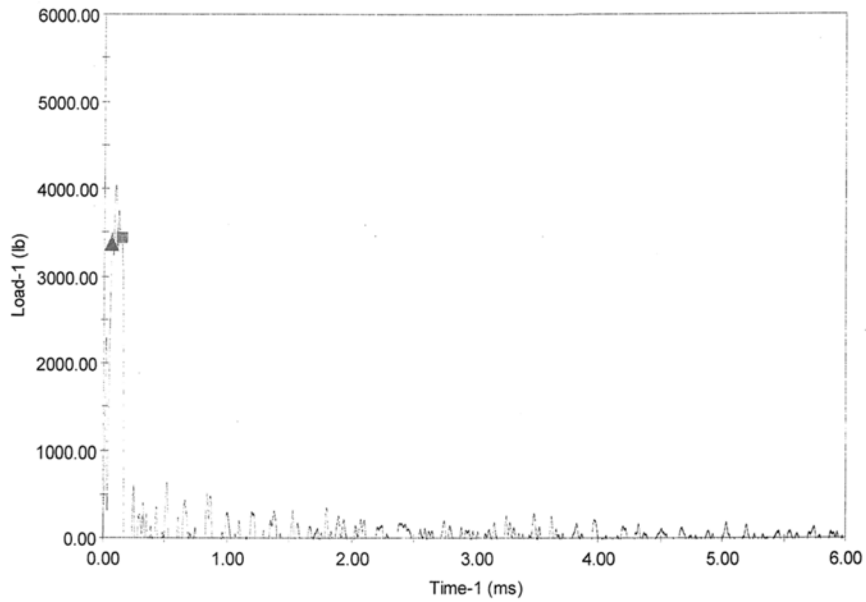
A-4 ASTM Standard E944-19, *Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance*, 2019.

A-5 ASTM Standard E844-18, *Standard Guide for Sensor Set Design and Irradiation for Reactor Surveillance*, 2018.

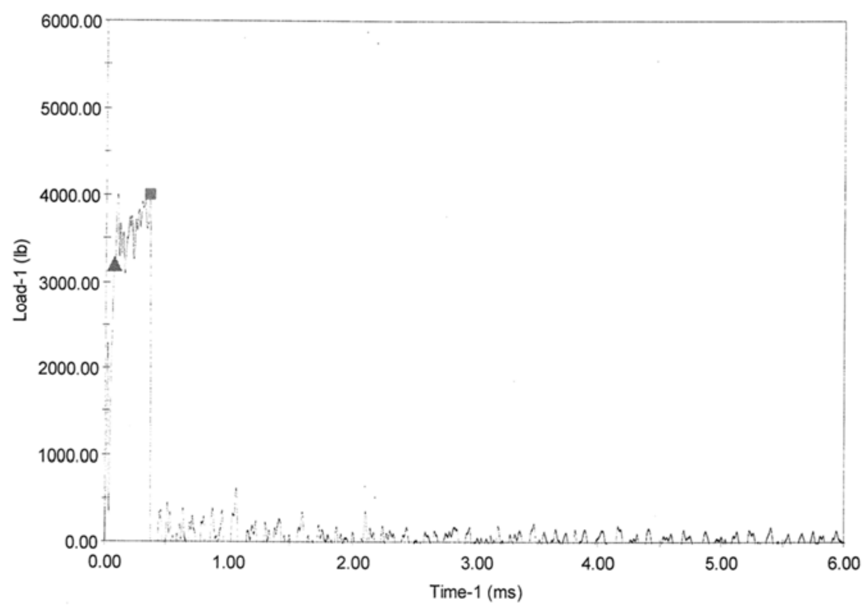
APPENDIX B LOAD-TIME RECORDS FOR CHARPY SPECIMEN TESTS FROM CAPSULE U

- "BLXX" denotes Intermediate Shell Forging 05, tangential orientation
- "BTXX" denotes Intermediate Shell Forging 05, axial orientation
- "BWXX" denotes weld material
- "BHXX" denotes heat-affected zone material

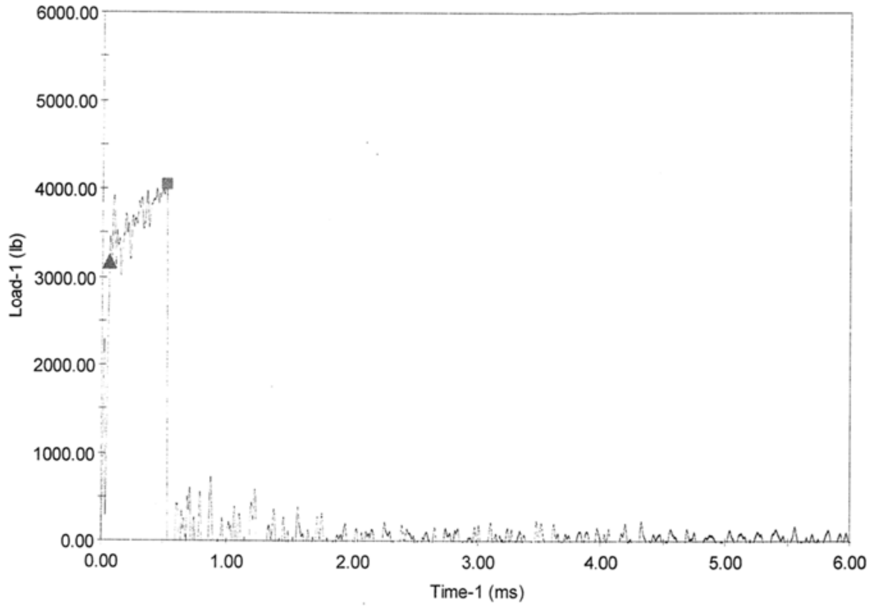
Note that the instrumented Charpy data is not required per ASTM Standards E185-82 or E23-18.



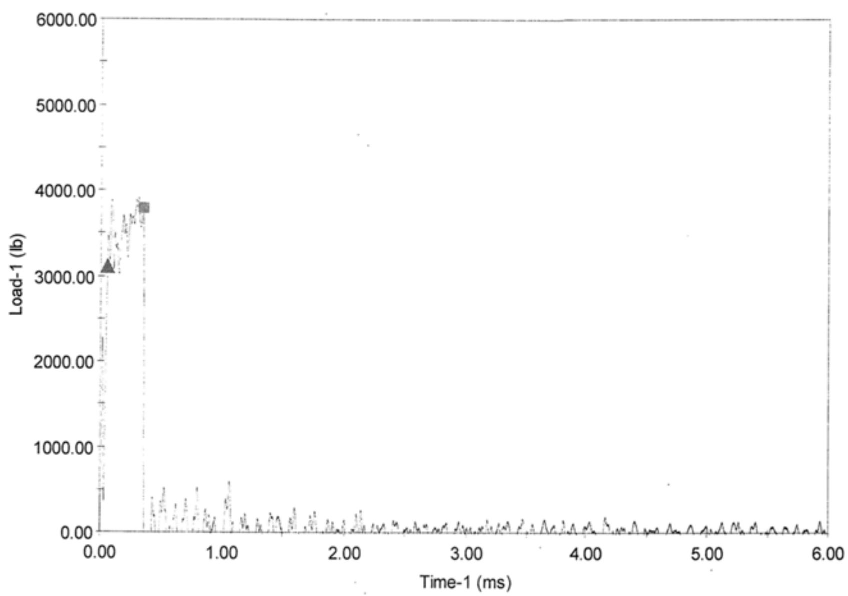
BL10 tested at -60 °F.



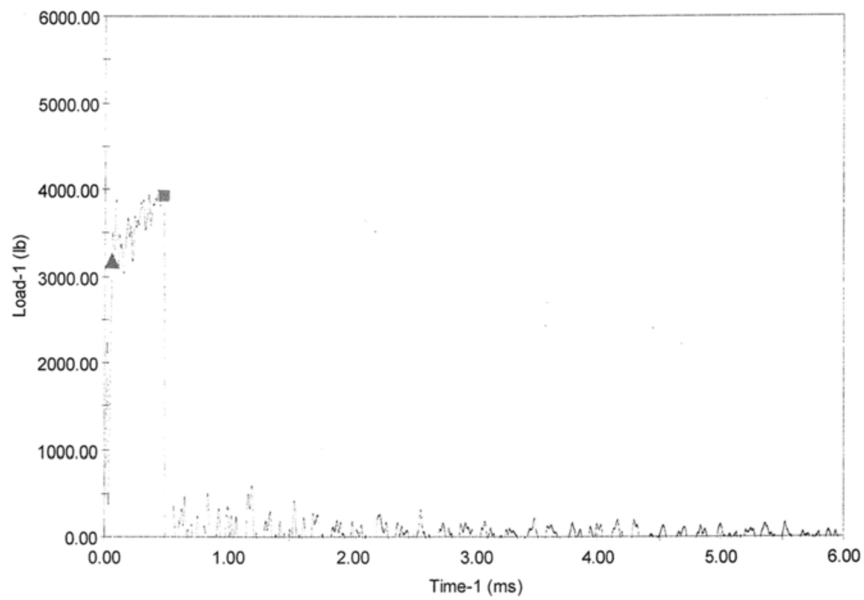
BL4 tested at -50 °F.



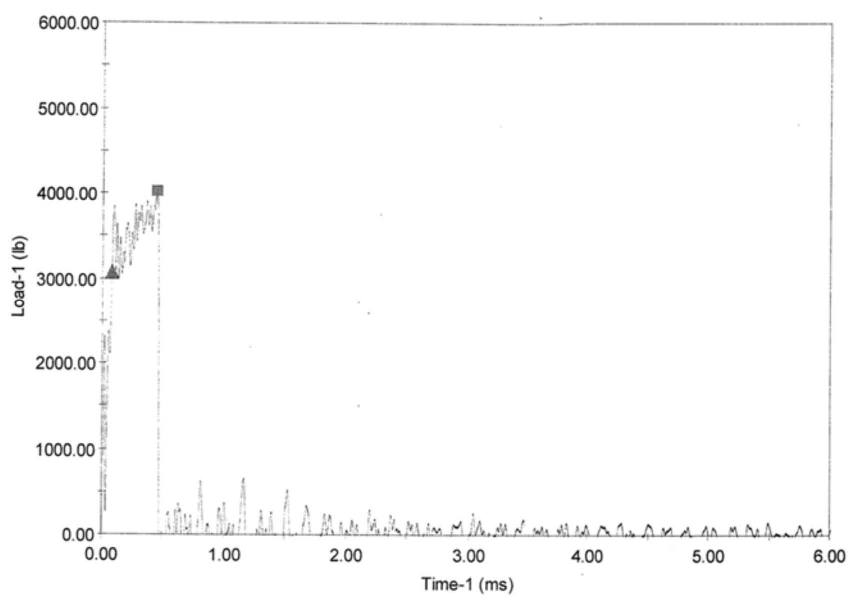
BL8: Tested at -30°F



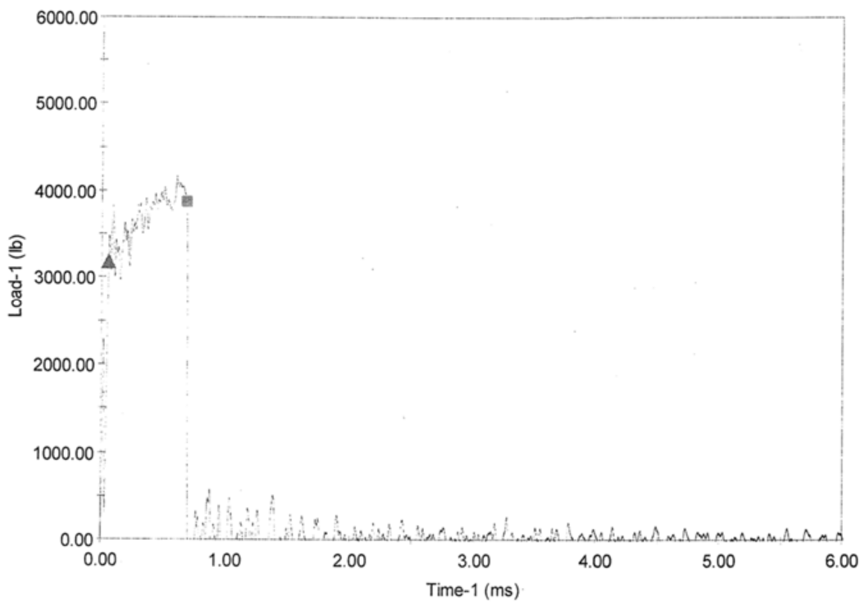
BL15: Tested at -20°F



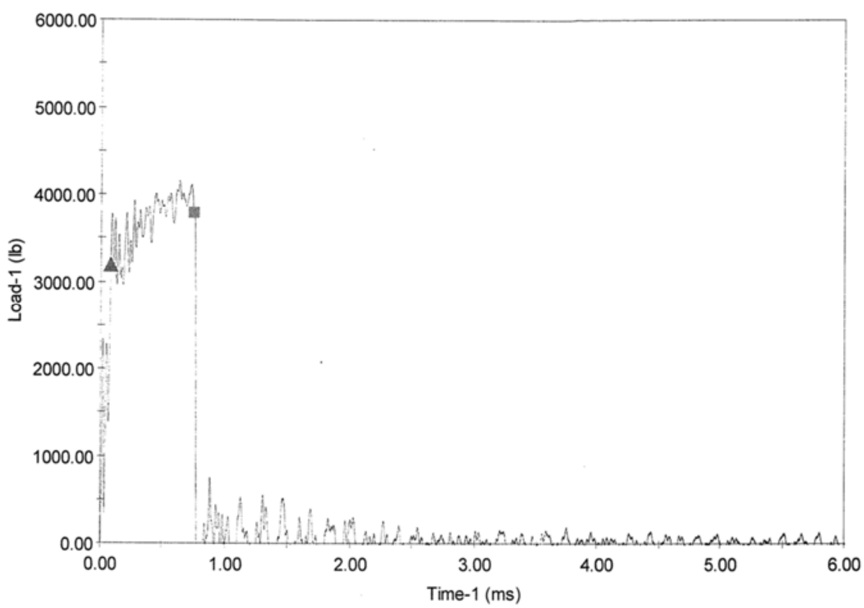
BL6: Tested at -15°F



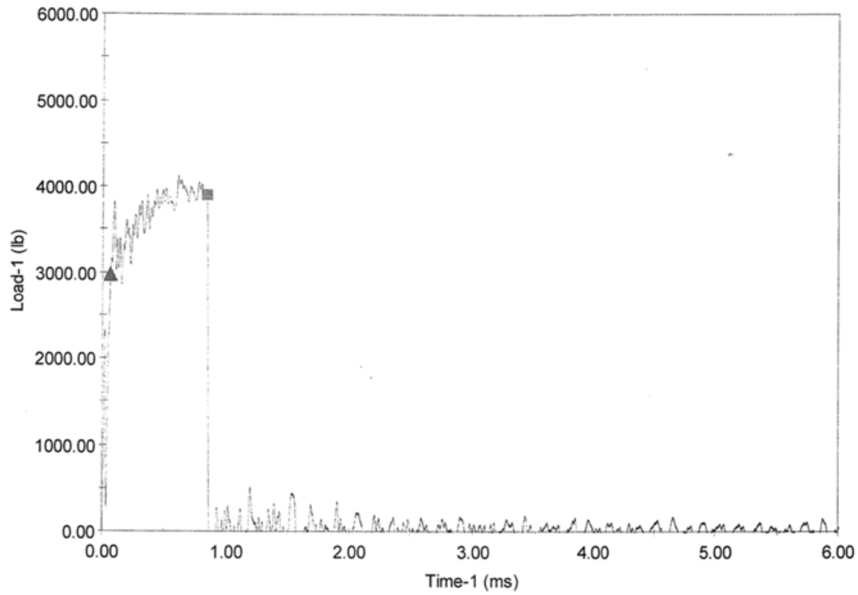
BL3: Tested at -10°F



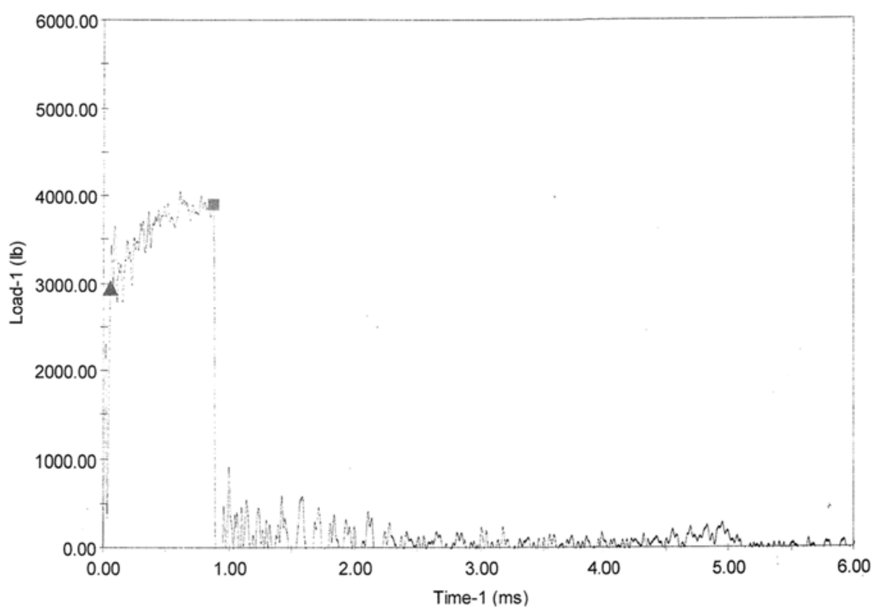
BL5: Tested at -5°F



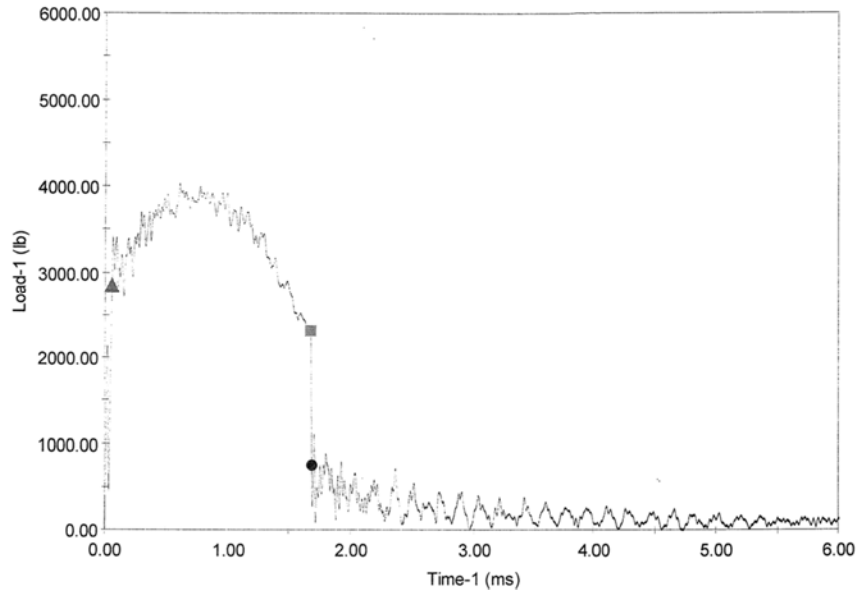
BL11: Tested at 0°F



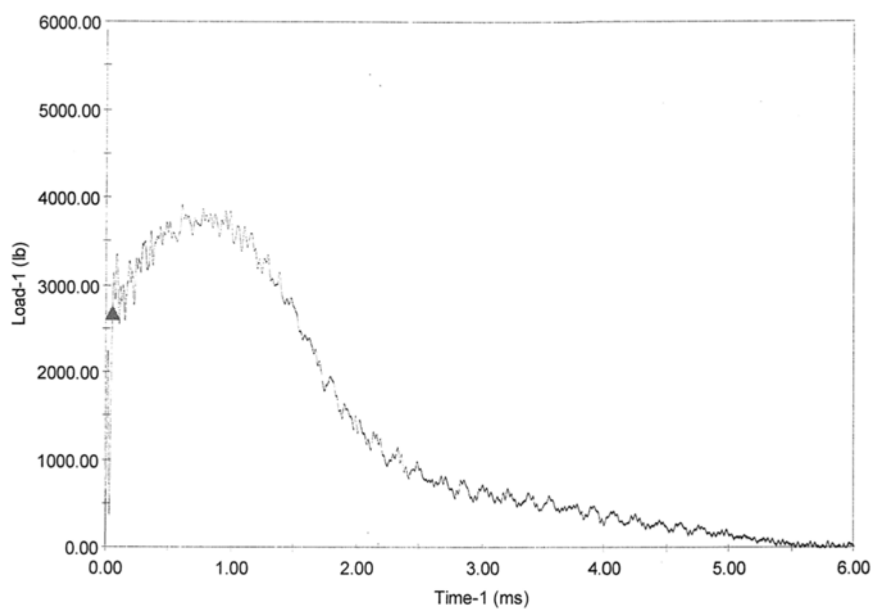
BL1: Tested at 10°F



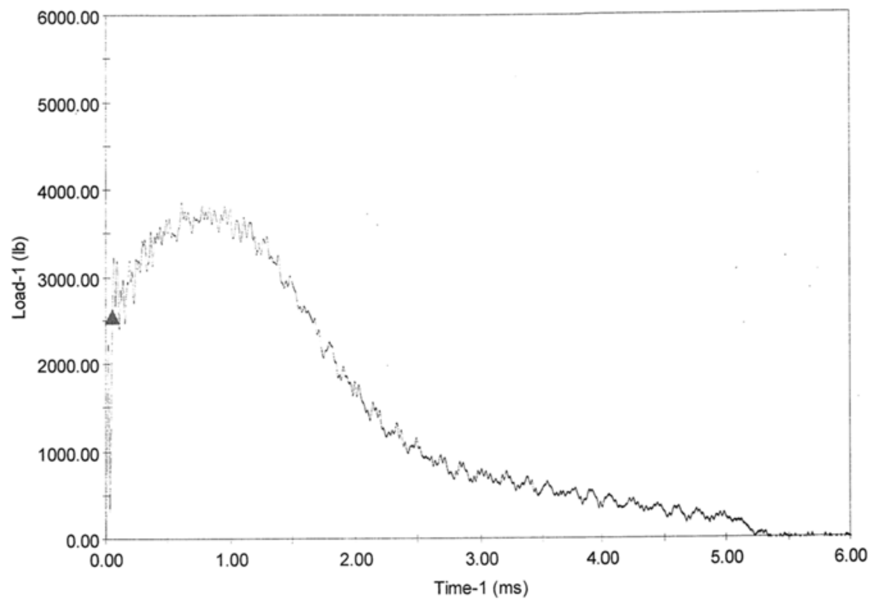
BL9: Tested at 40°F



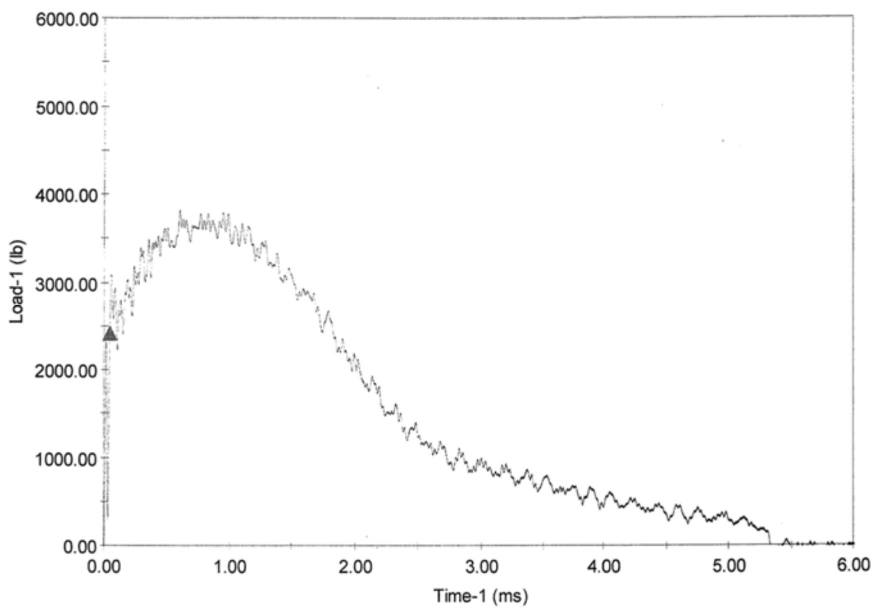
BL13: Tested at 75°F



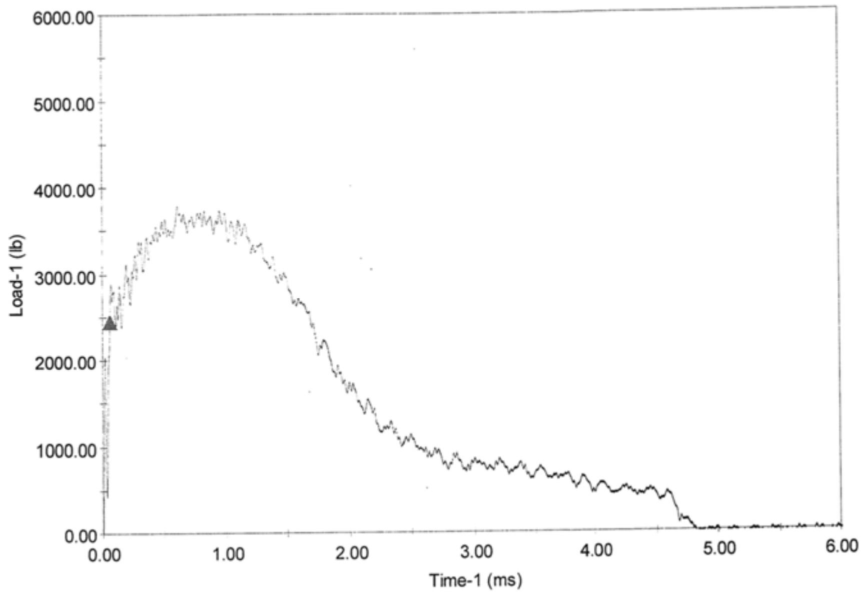
BL2: Tested at 120°F



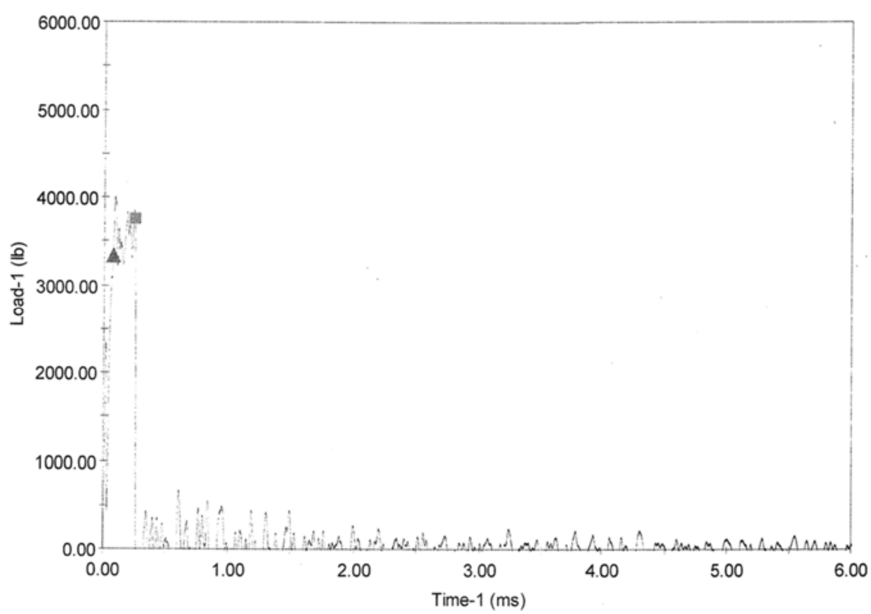
BL7: Tested at 170°F



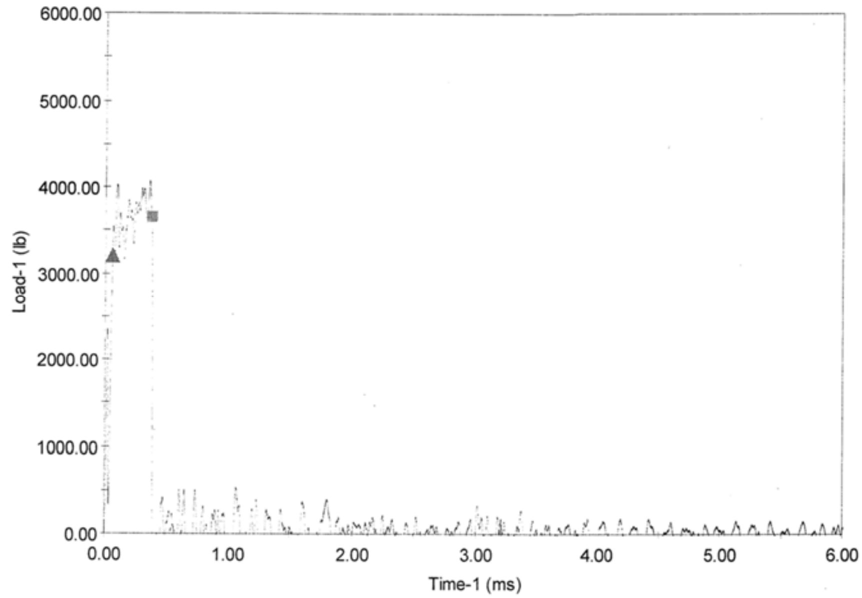
BL14: Tested at 200°F



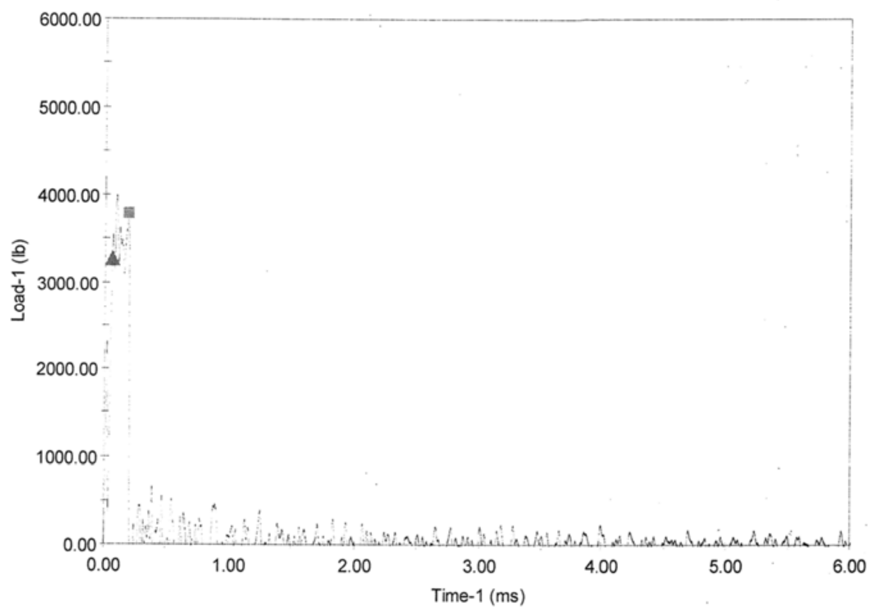
BL12: Tested at 220°F



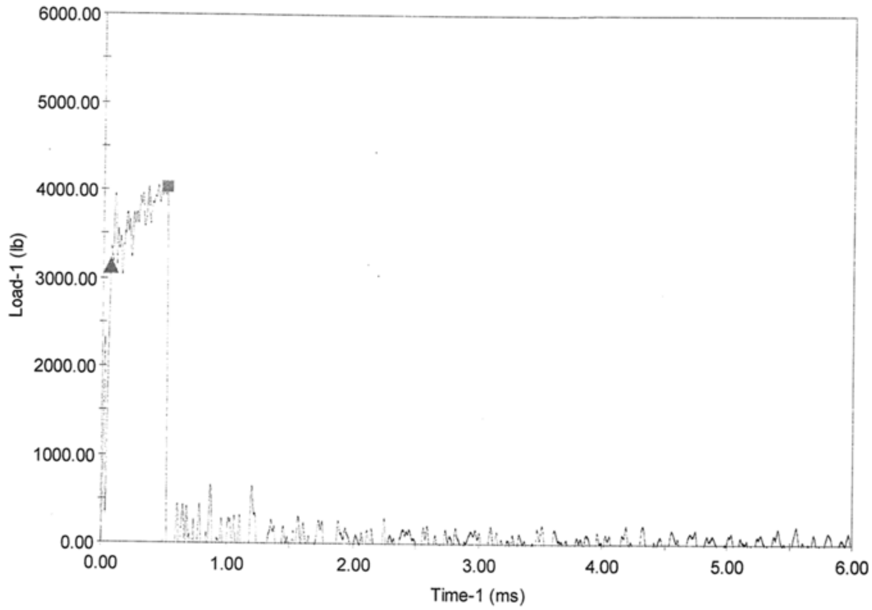
BT1: Tested at -60°F



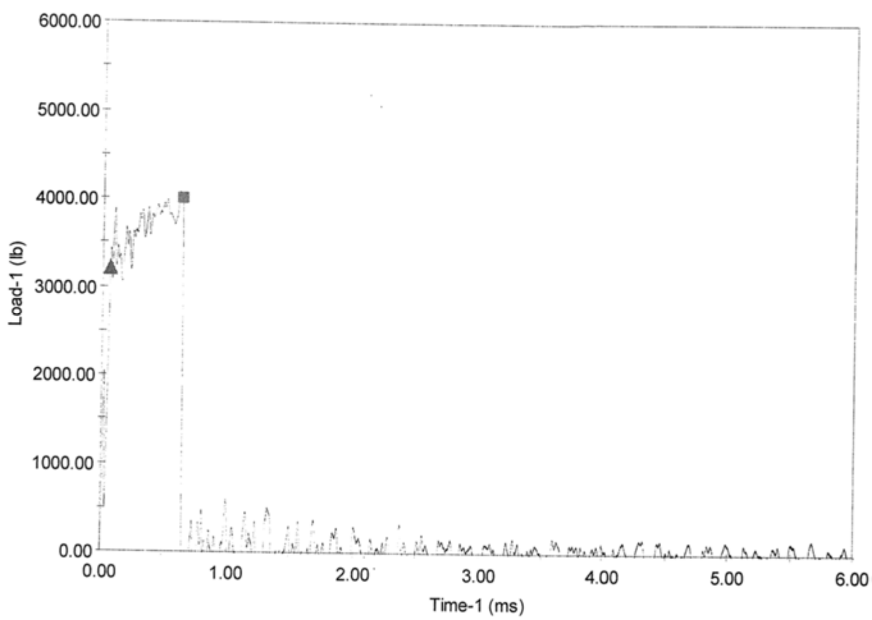
BT9: Tested at -50°F



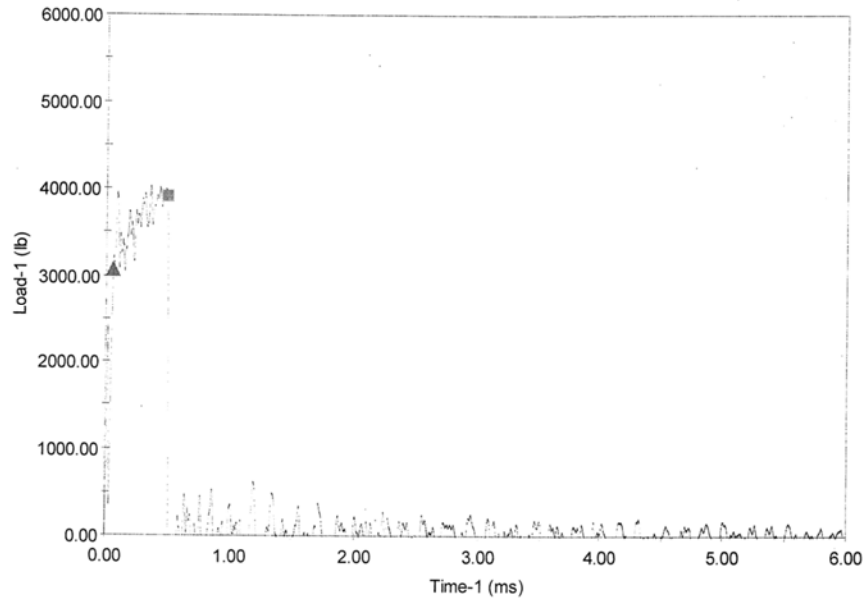
BT2: Tested at -35°F



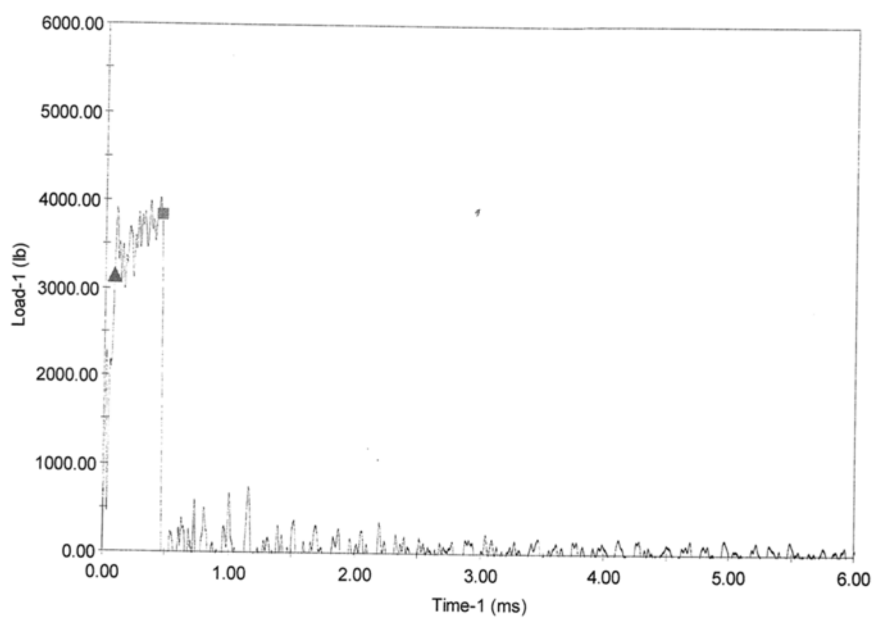
BT6: Tested at -30°F



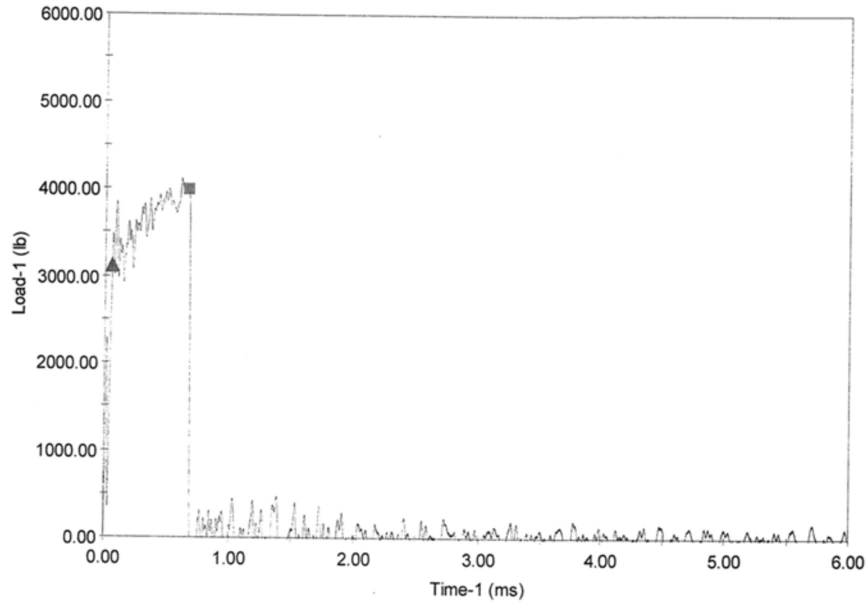
BT4: Tested at -20°F



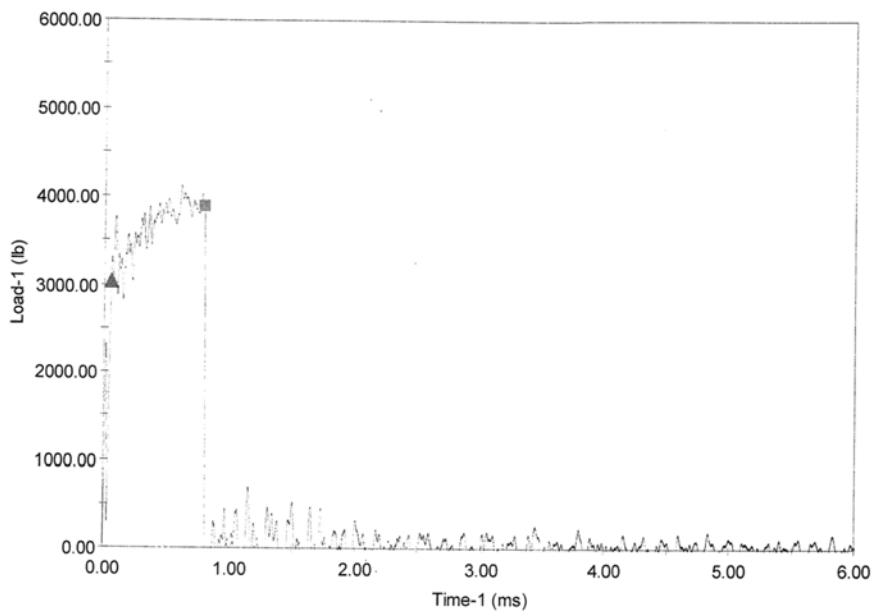
BT15: Tested at -15°F



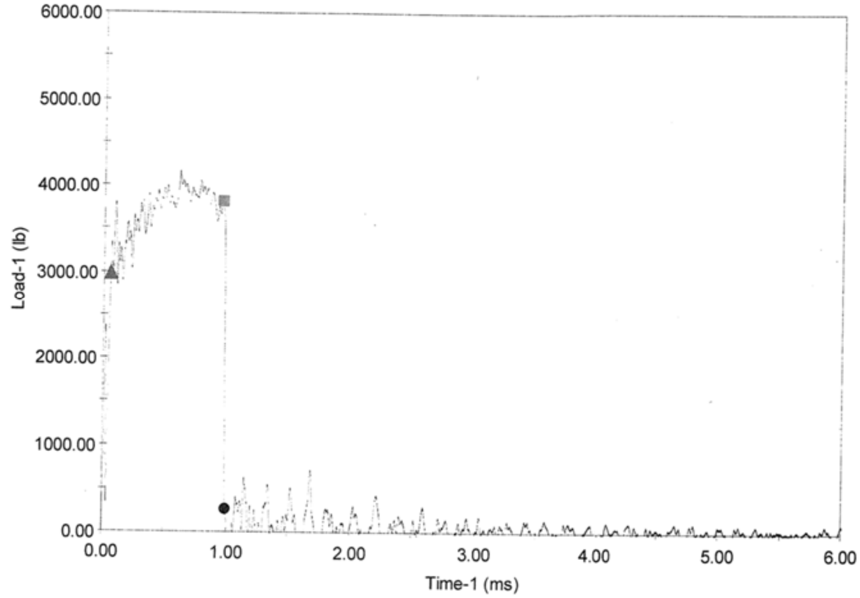
BT10: Tested at -10°F



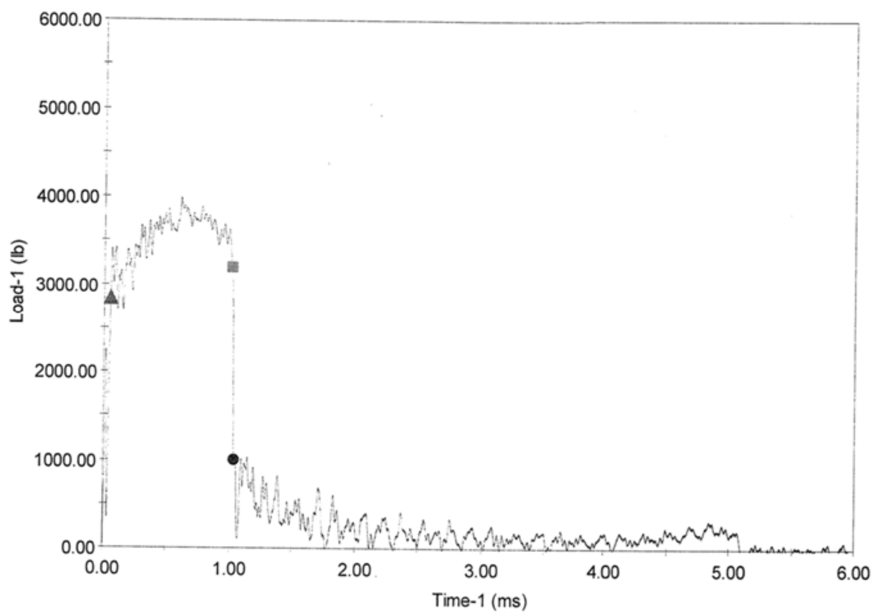
BT8: Tested at 0°F



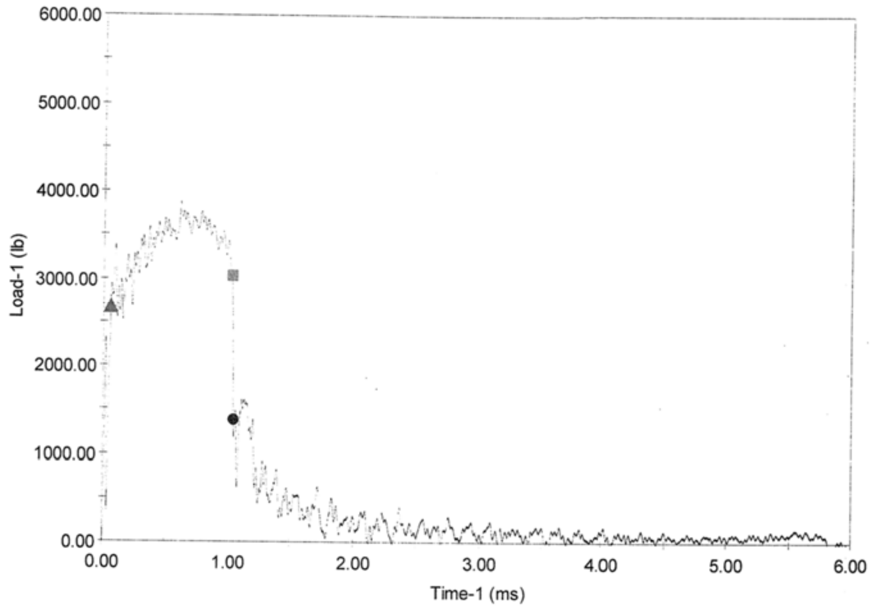
BT3: Tested at 10°F



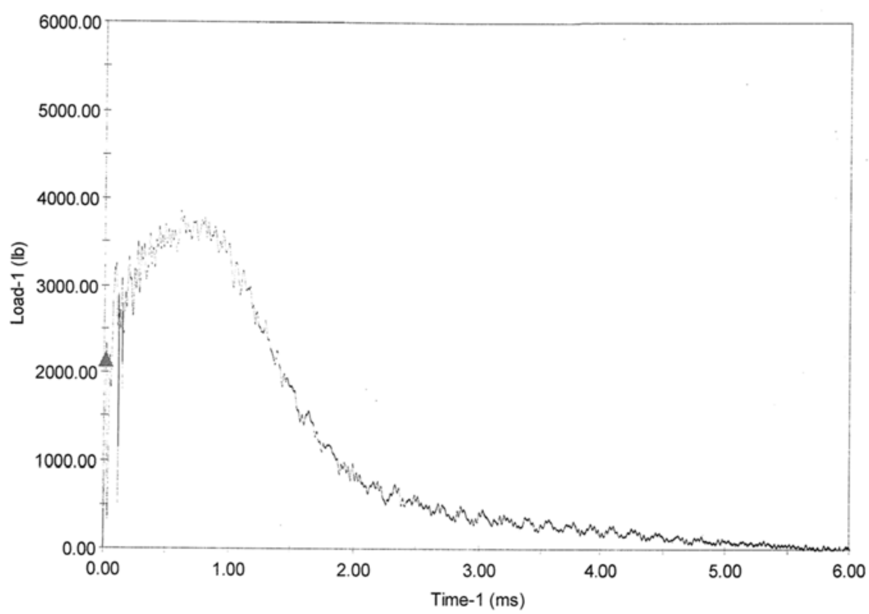
BT12: Tested at 30°F



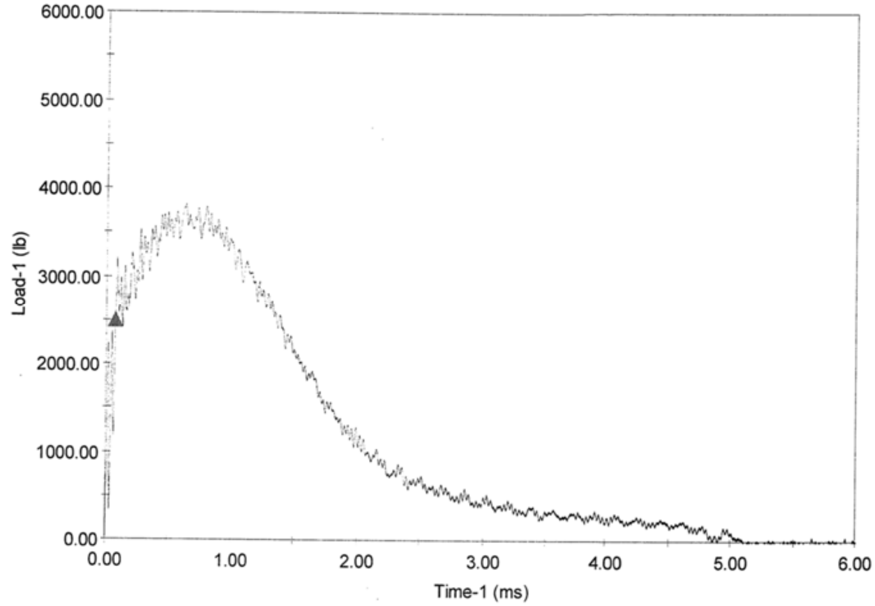
BT14: Tested at 75°F



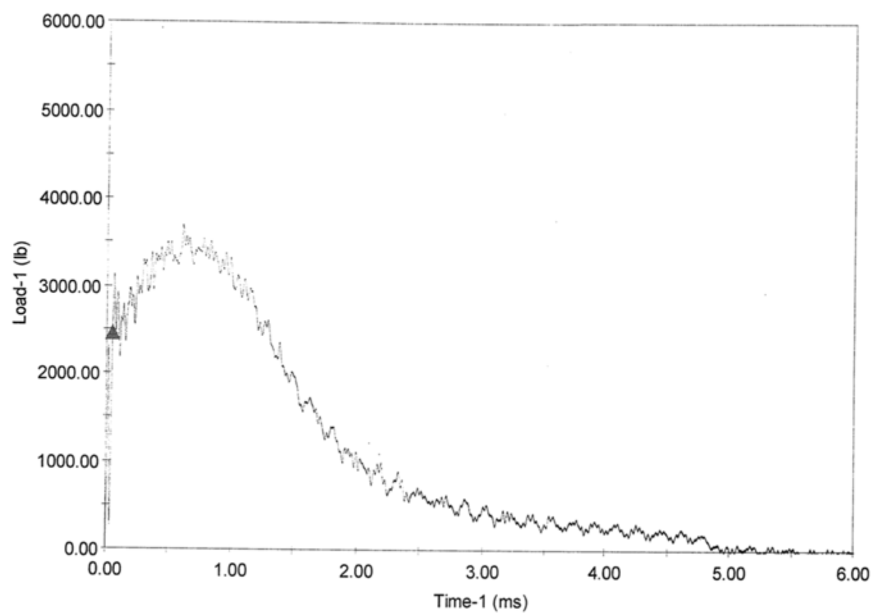
BT13: Tested at 120°F



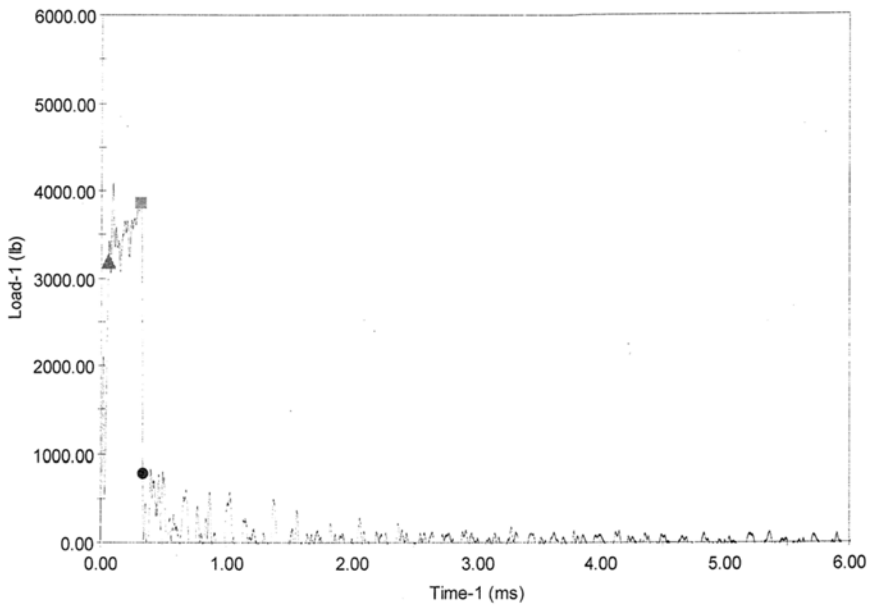
BT5: Tested at 170°F



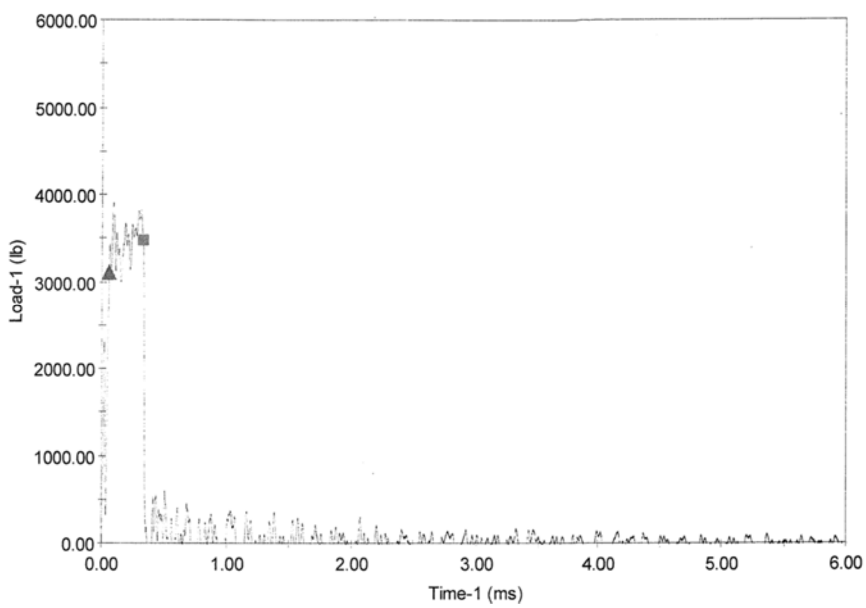
BT11: Tested at 200°F



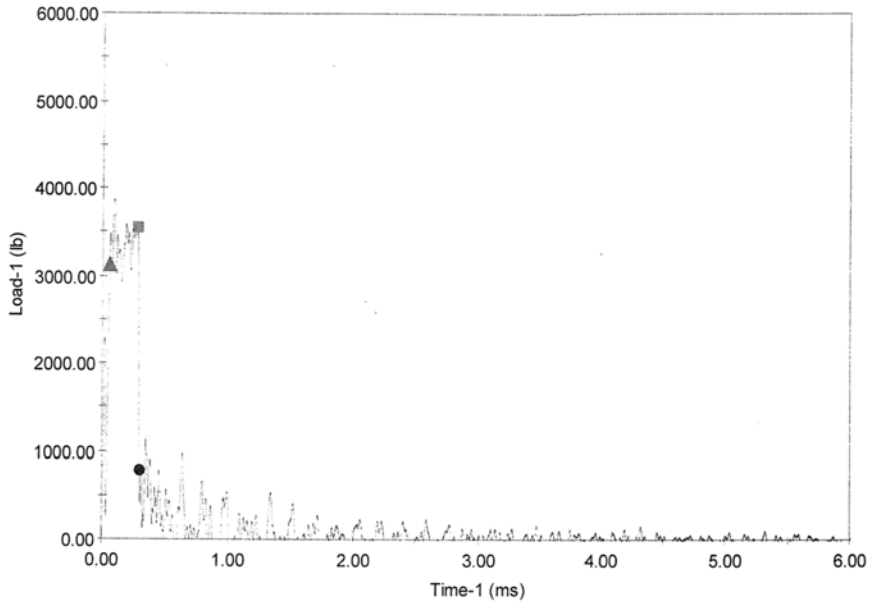
BT7: Tested at 220°F



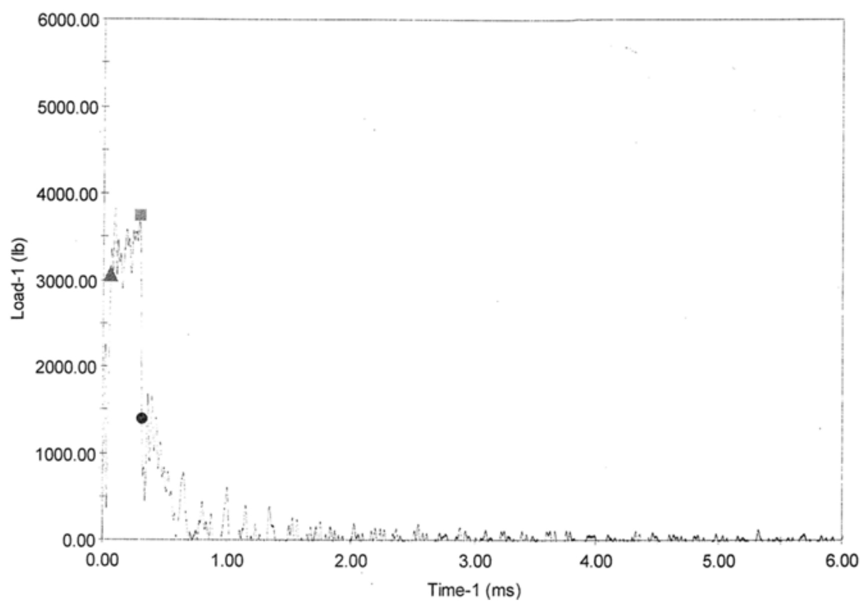
BW13: Tested at -60°F



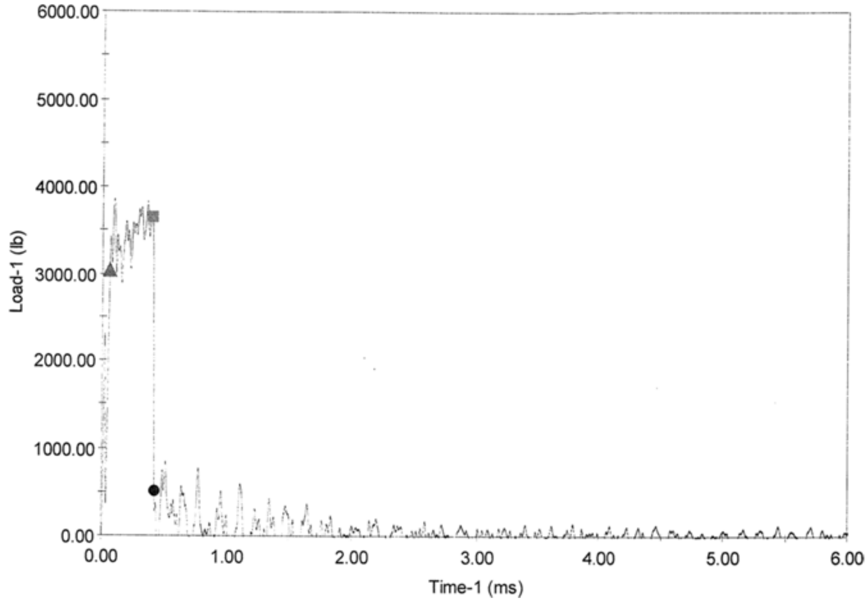
BW8: Tested at -50°F



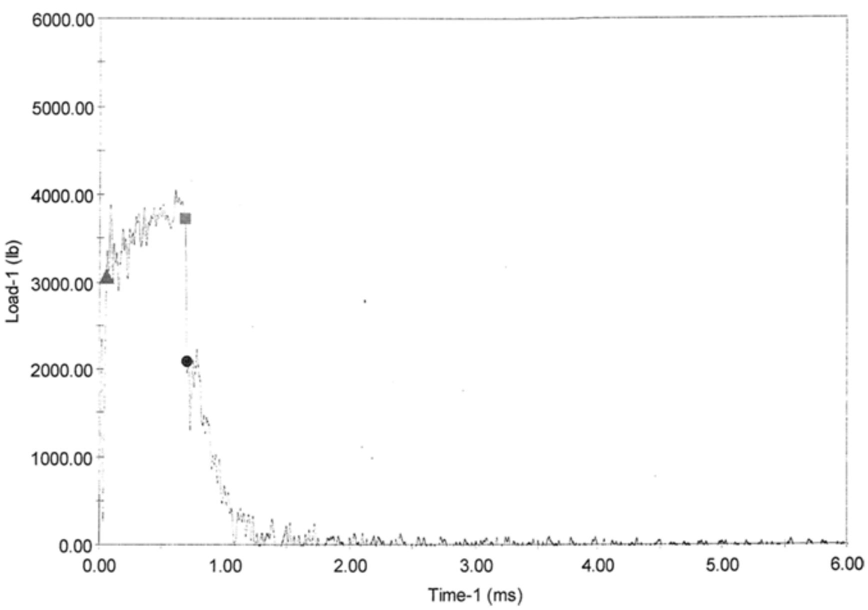
BW15: Tested at -30°F



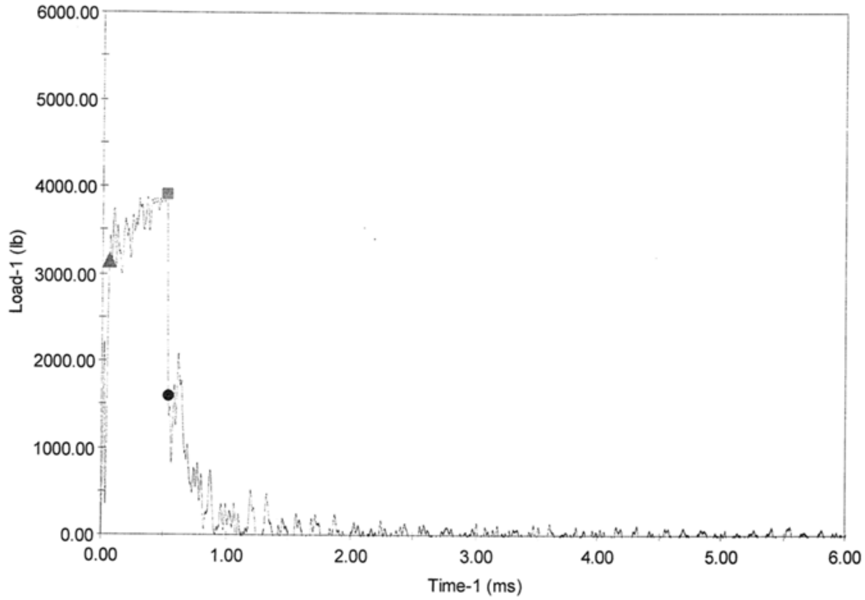
BW7: Tested at -25°F



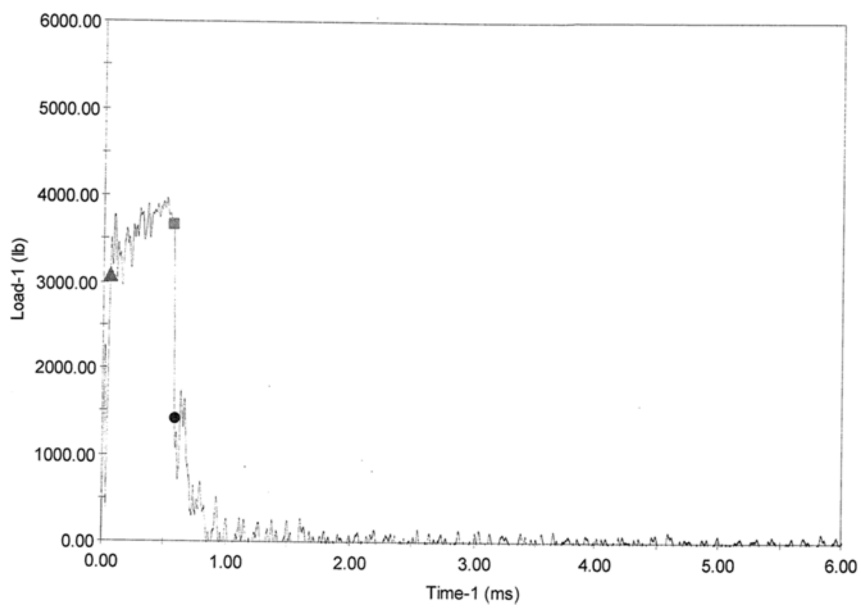
BW4: Tested at -20°F



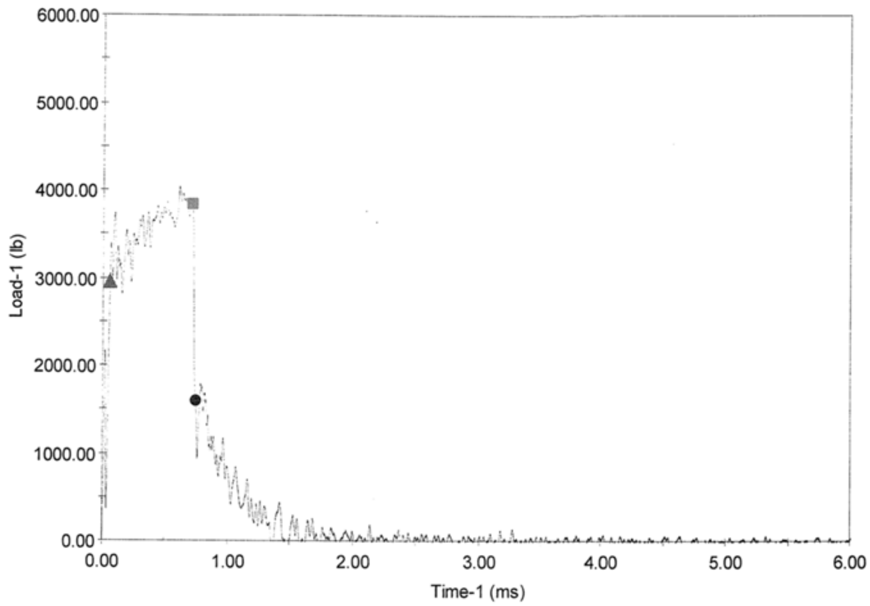
BW14: Tested at -15°F



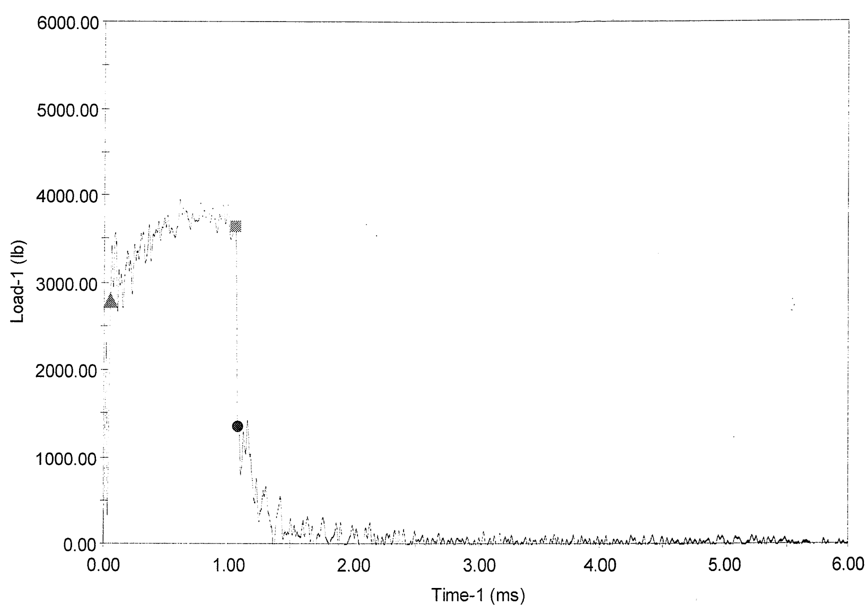
BW9: Tested at -10°F



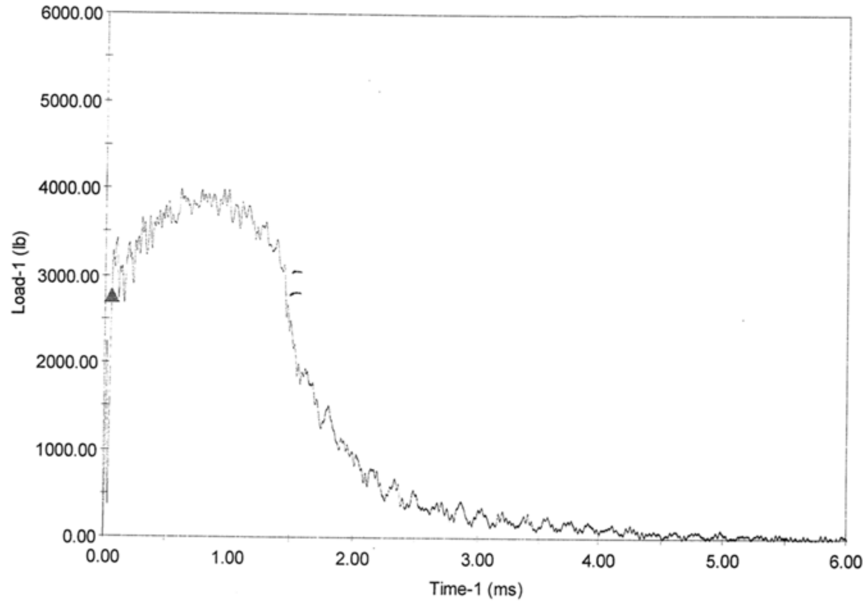
BW1: Tested at 0°F



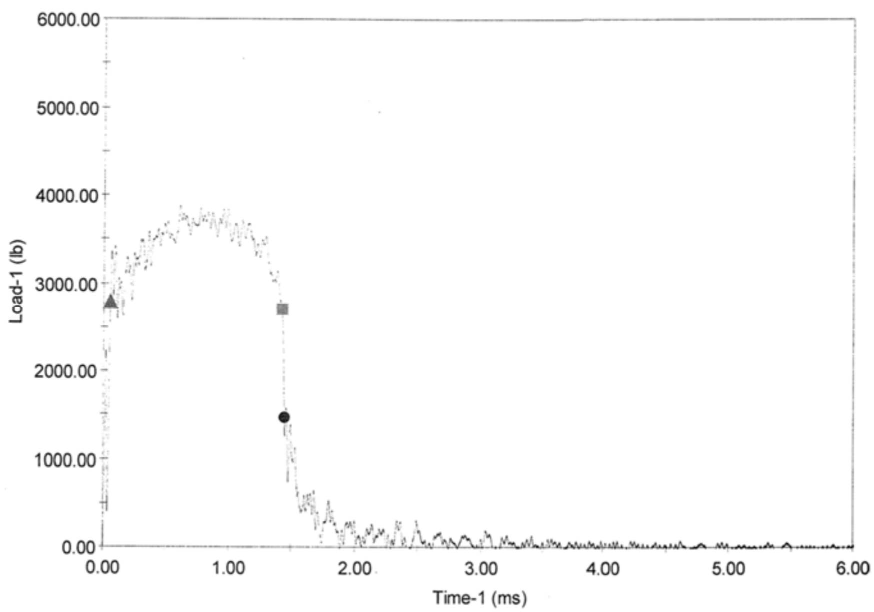
BW12: Tested at 10°F



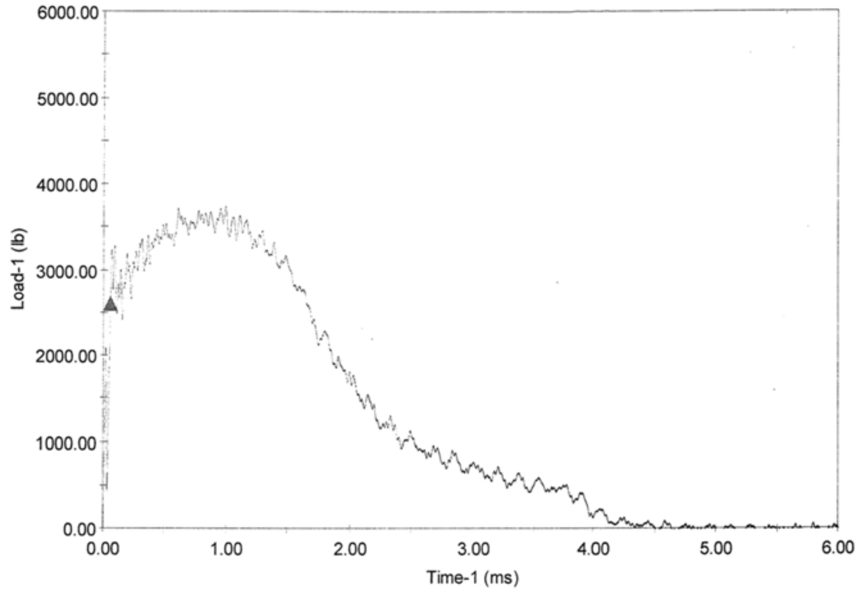
BW3: Tested at 60°F



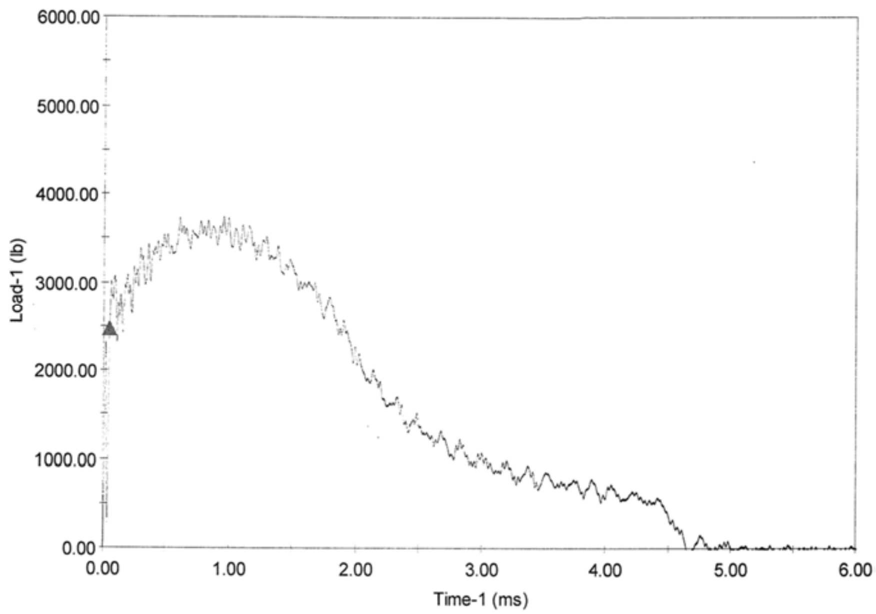
BW6: Tested at 75°F



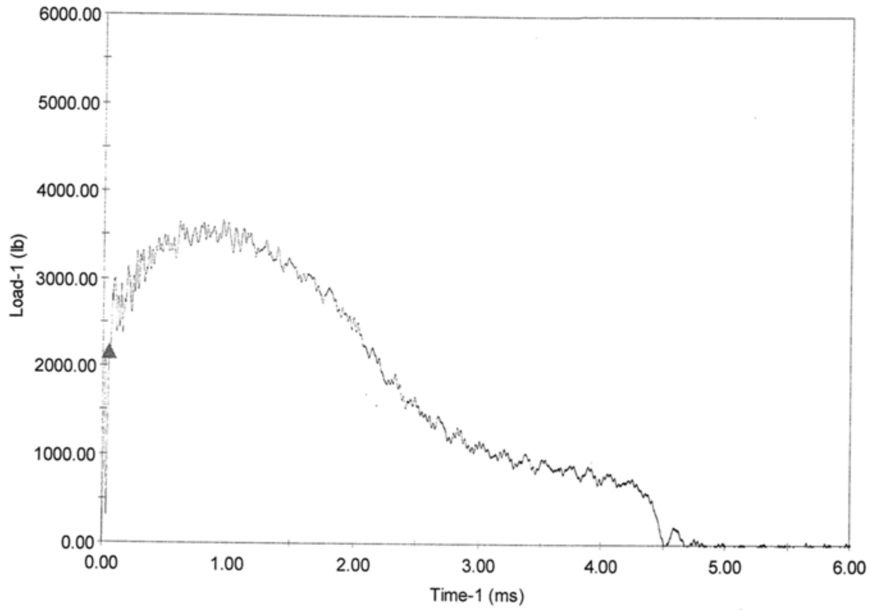
BW2: Tested at 120°F



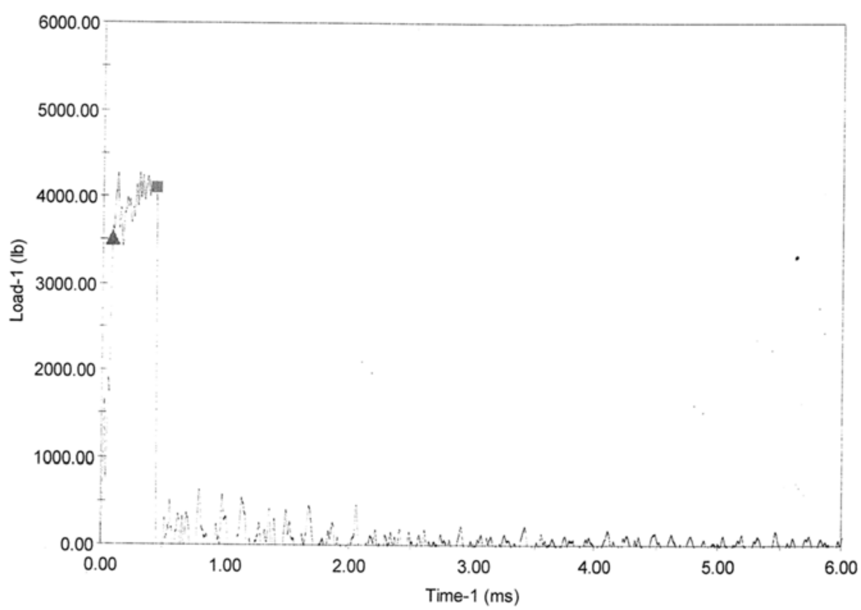
BW5: Tested at 170°F



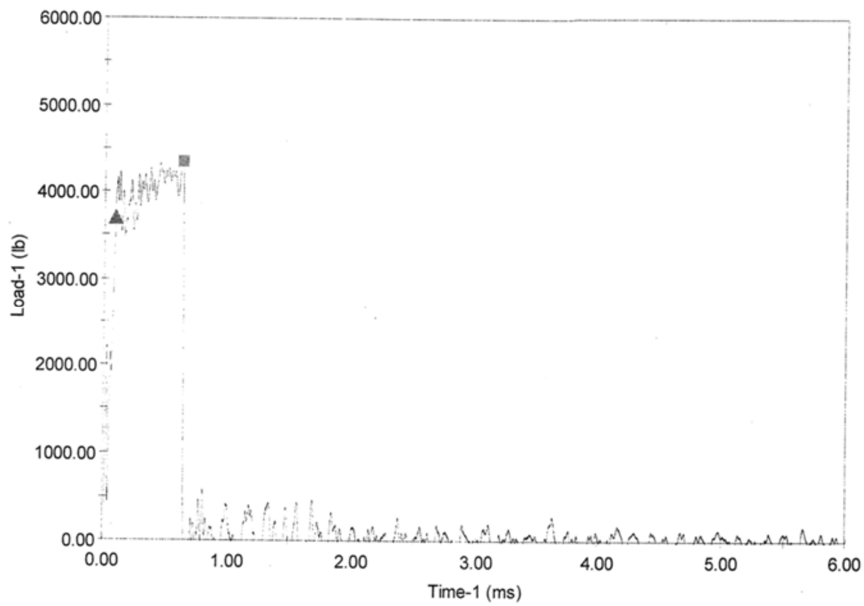
BW10: Tested at 200°F



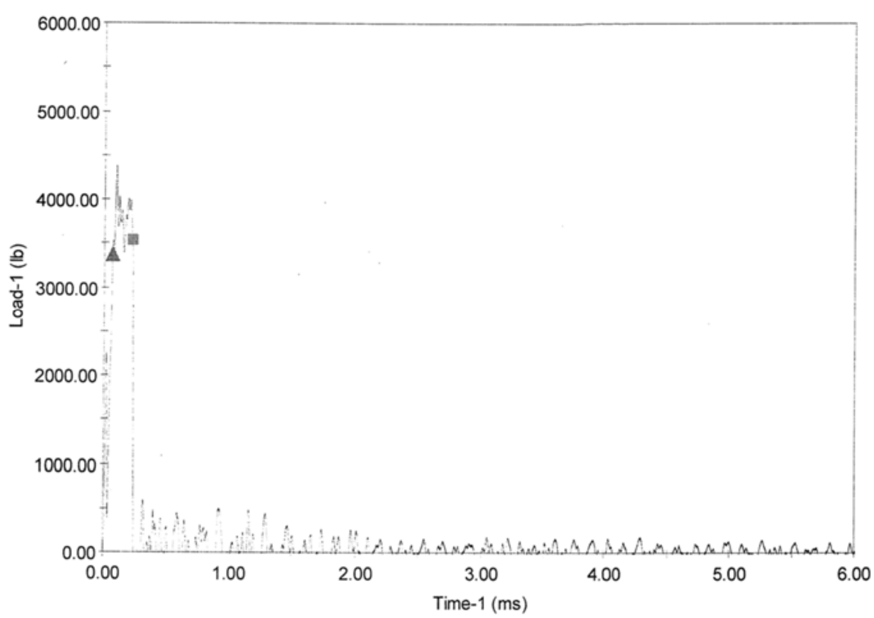
BW11: Tested at 220°F



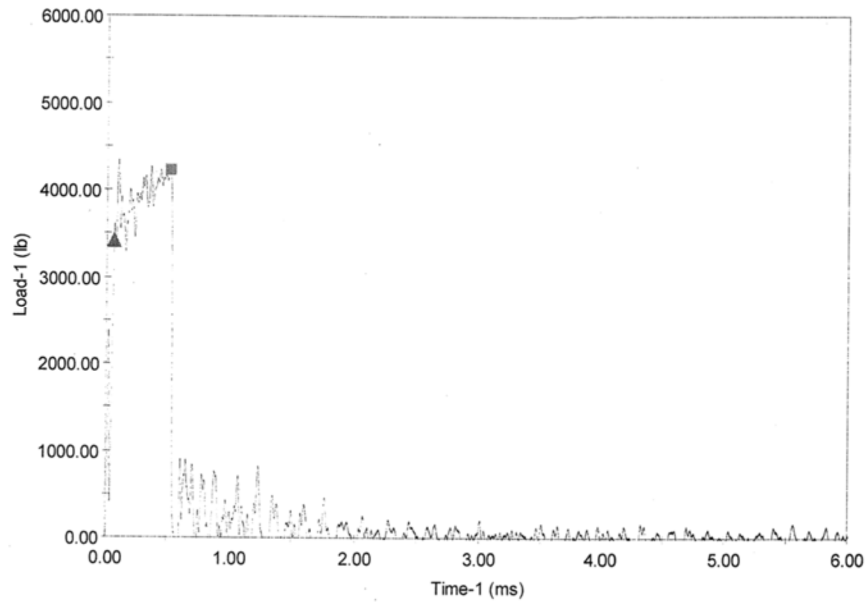
BH5: Tested at -115°F



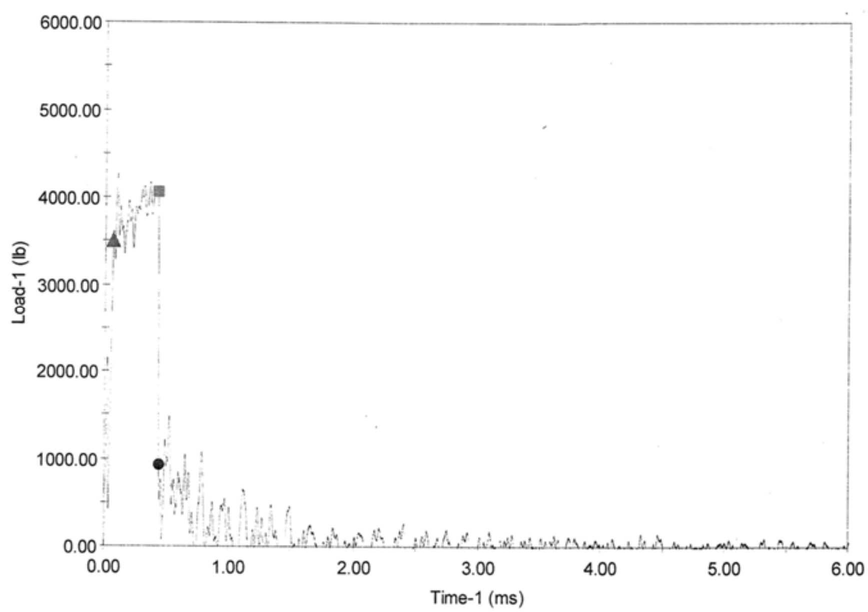
BH3: Tested at -110°F



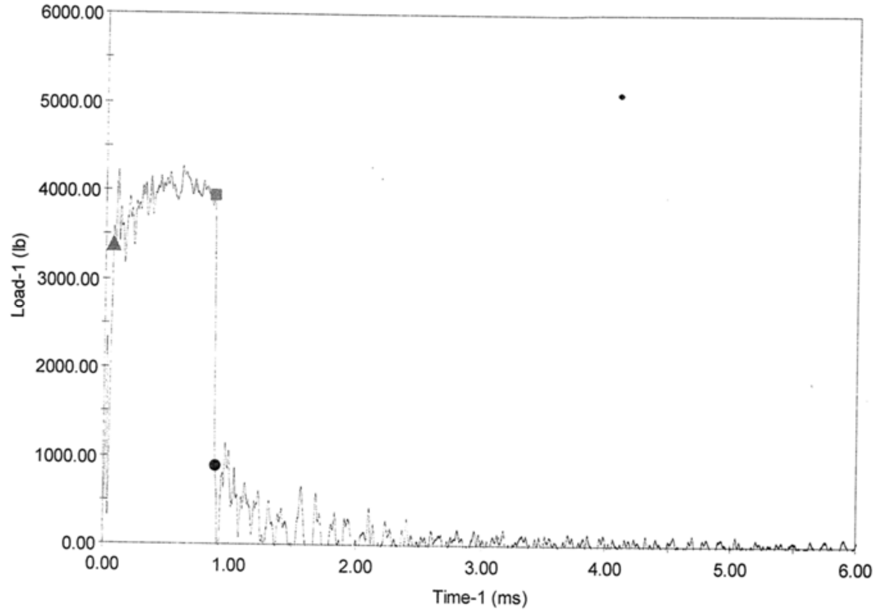
BH10: Tested at -100°F



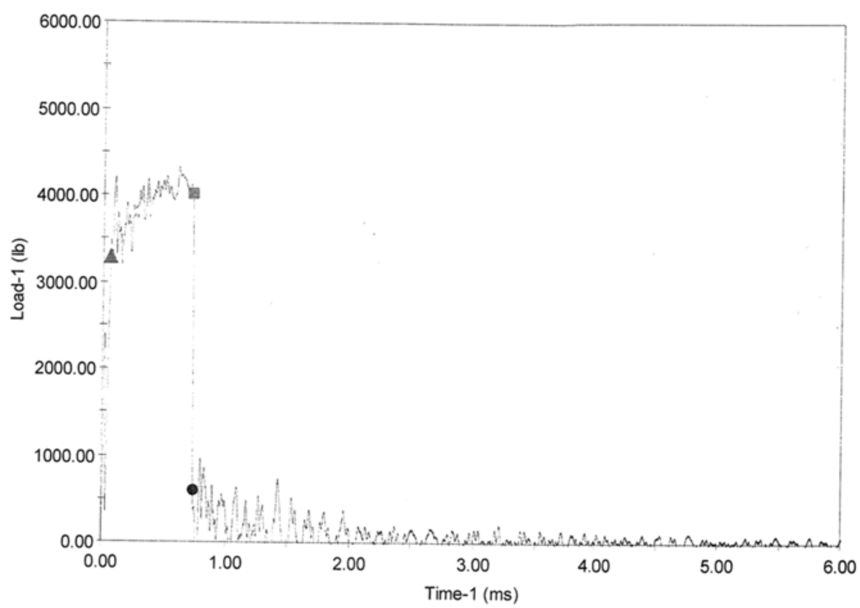
BH2: Tested at -90°F



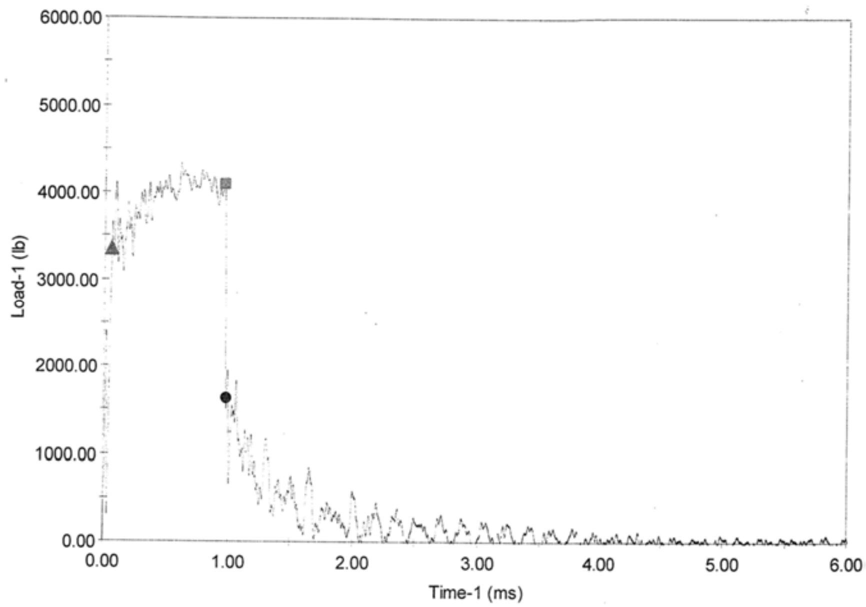
BH12: Tested at -80°F



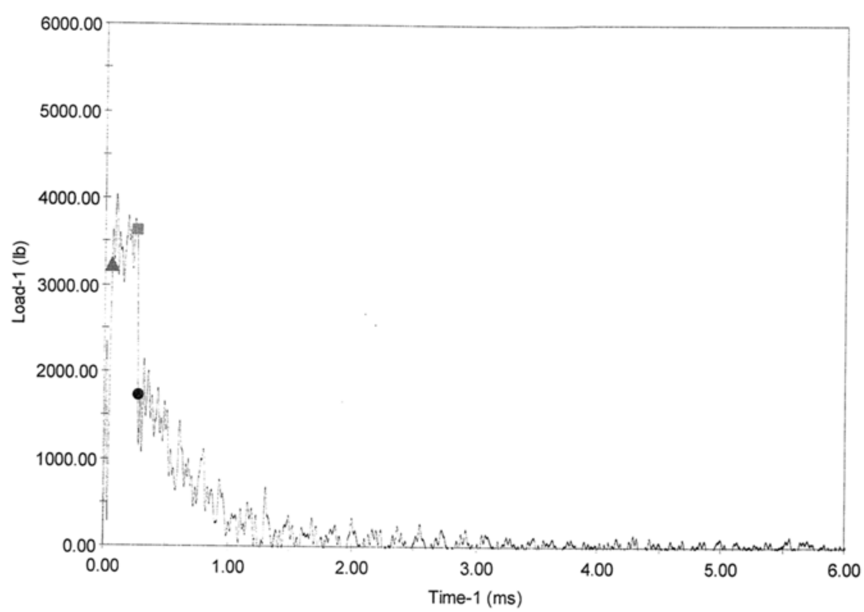
BH7: Tested at -70°F



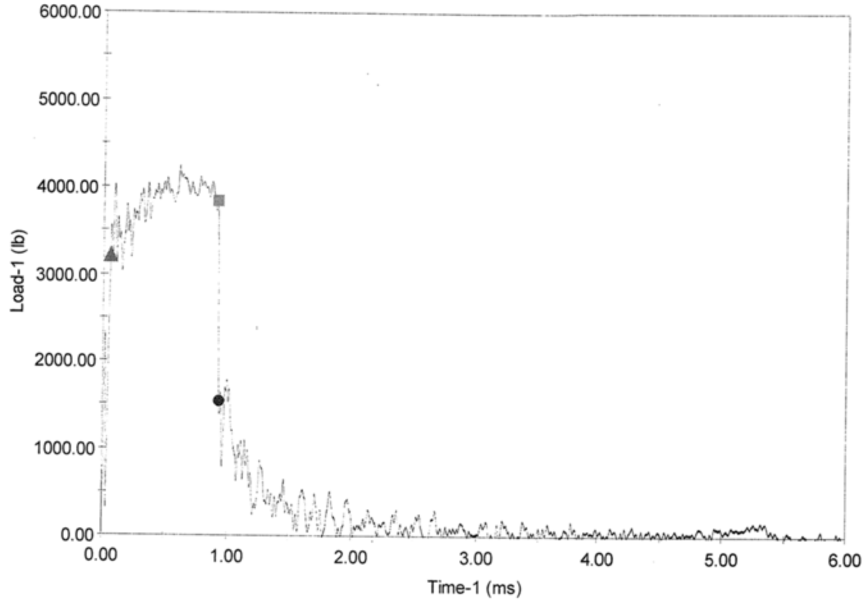
BH15: Tested at -60°F



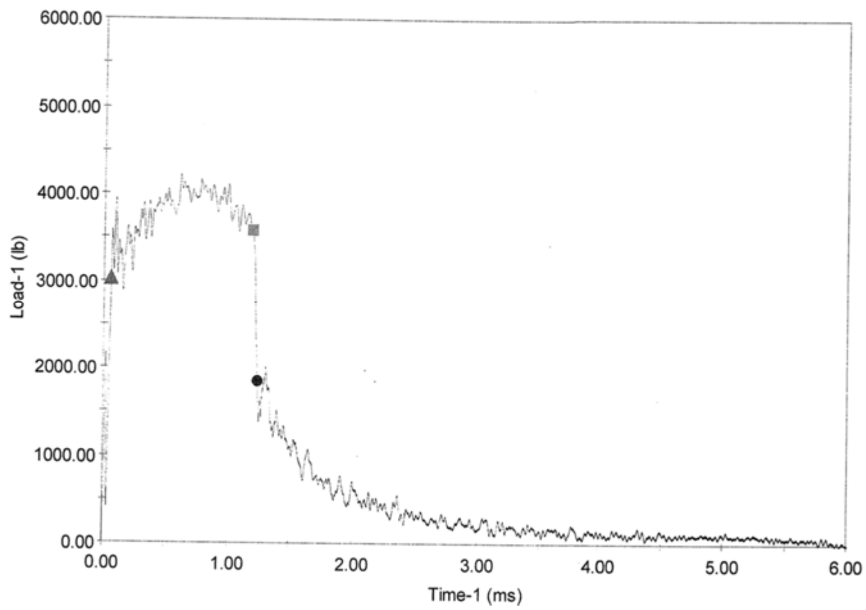
BH14: Tested at -35°F



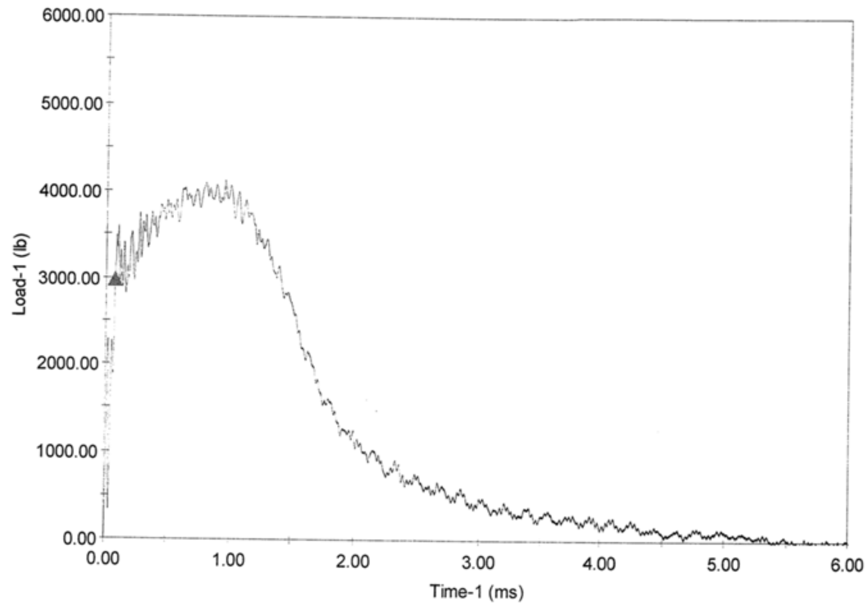
BH8: Tested at -30°F



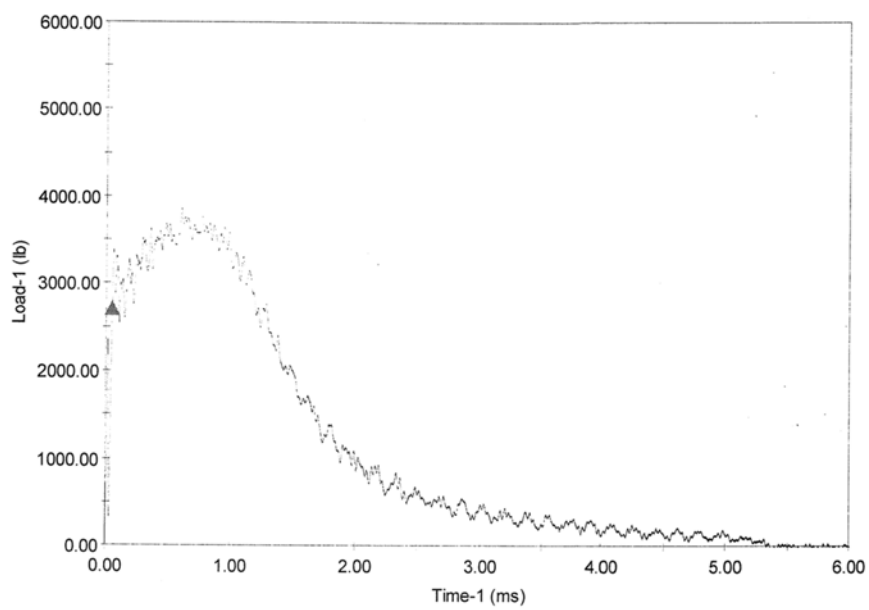
BH11: Tested at -20°F



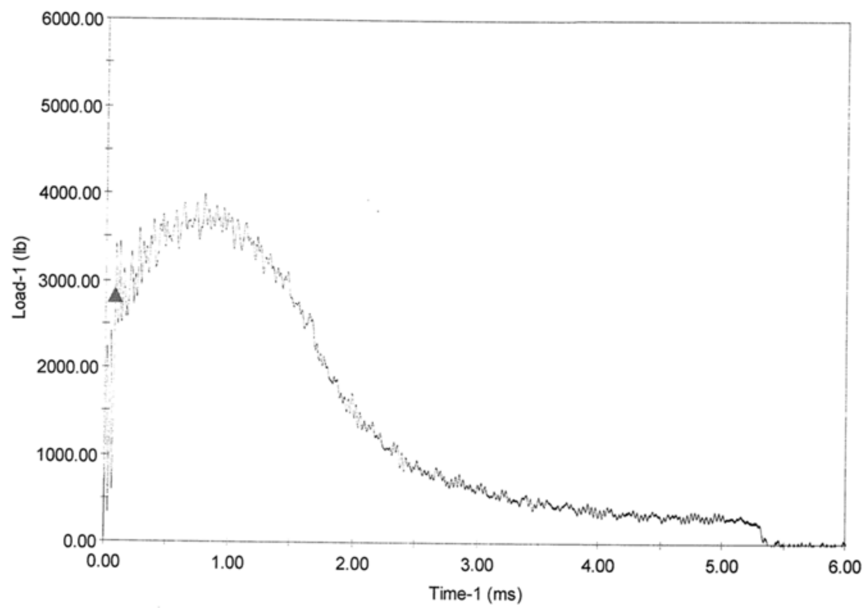
BH1: Tested at 10°F



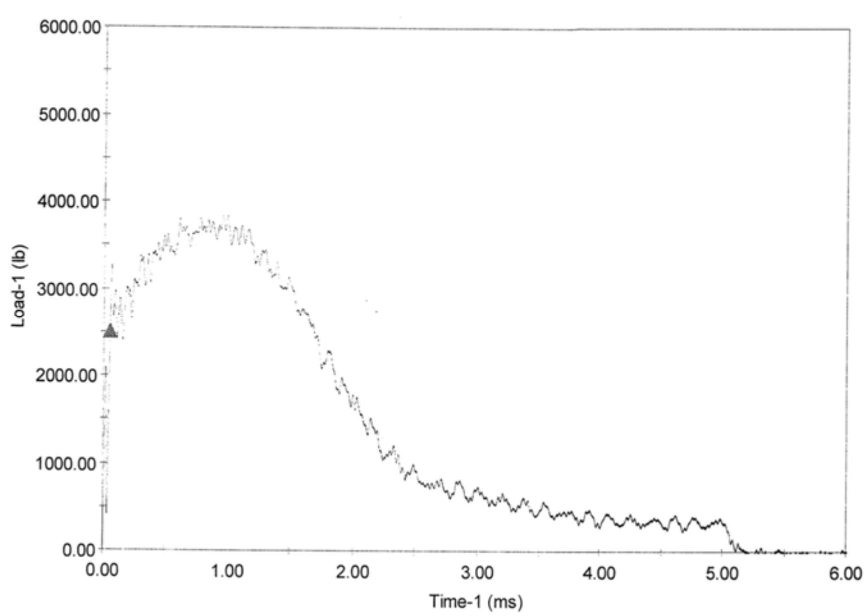
BH4: Tested at 75°F



BH13: Tested at 150°F



BH6: Tested at 200°F



BH9: Tested at 220°F

APPENDIX C CHARPY V-NOTCH PLOTS FOR BASELINE AND CAPSULE U USING SYMMETRIC HYPERBOLIC TANGENT CURVE-FITTING METHOD

C.1 METHODOLOGY

Contained in Table C-1 are the upper-shelf energy (USE) values that are used as input for the generation of the Charpy V-notch plots using CVGRAPH, Version 6.02. The definition for USE is given in ASTM E185-82 [Ref. C-1], Section 4.18, and reads as follows:

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

Westinghouse reports the average of all Charpy data (≥ 95% shear) as the USE, excluding any values that are deemed outliers using engineering judgment. Hence, the Capsule U USE values reported in Table C-1 were determined by applying this methodology to the Charpy data tabulated in Table 5-1 through Table 5-4 of this report. USE values documented in Table C-1 for the unirradiated material were also determined by applying the methodology described above to the Charpy impact data reported in WCAP-9455 [Refs. C-2 and C-3]. The USE values reported in Table C-1 were used in generation of the Charpy V-notch curves.

The lower-shelf energy values were fixed at 2.2 ft-lb for all cases. The lower-shelf lateral expansion values were fixed at 1.0 mil in order to be consistent with the previous capsule analysis. Similarly, the upper-shelf energy must also be fixed for curve-fitting the Charpy V-Notch (CVN) Energy data using the values reported in in Table C-1. However, the upper-shelf lateral expansion is not fixed in CVGRAPH.

USE is expected to decrease as a function of fluence and copper content. As expected, this decrease in USE was exhibited from the unirradiated materials to the Capsule U materials.

Table C-1 Upper-Shelf Energy Values (ft-lb) Fixed in CVGRAPH

Material	Unirradiated (ft-lb)	Capsule U (ft-lb)
Intermediate Shell Forging 05, Heat # 527828 (Tangential)	175	130
Intermediate Shell Forging 05, Heat # 527828 (Axial)	110	105
Weld Metal Heat # 895075	144	135
HAZ	130	118

CVGRAPH, Version 6.02 plots of all surveillance data are provided in this appendix, on the pages following the reference list.

C.2 REFERENCES

C-1 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, 1982.

C-2 Westinghouse Report WCAP-9455, Revision 4, "Tennessee Valley Authority Watts Bar Unit No. 2 Reactor Vessel Radiation Surveillance Program," August 2019.

C-3 Westinghouse Report WCAP-9455, Revision 2, "Tennessee Valley Authority Watts Bar Unit No. 2 Reactor Vessel Radiation Surveillance Program," June 1995.

C.3 CVGRAPH VERSION 6.02 INDIVIDUAL PLOTS OF UNIRRADIATED SPECIMENS

Unirradiated Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:28 AM

A = 88.60 B = 86.40 C = 61.94 T0 = 8.85 D = 0.00

Correlation Coefficient = 0.943

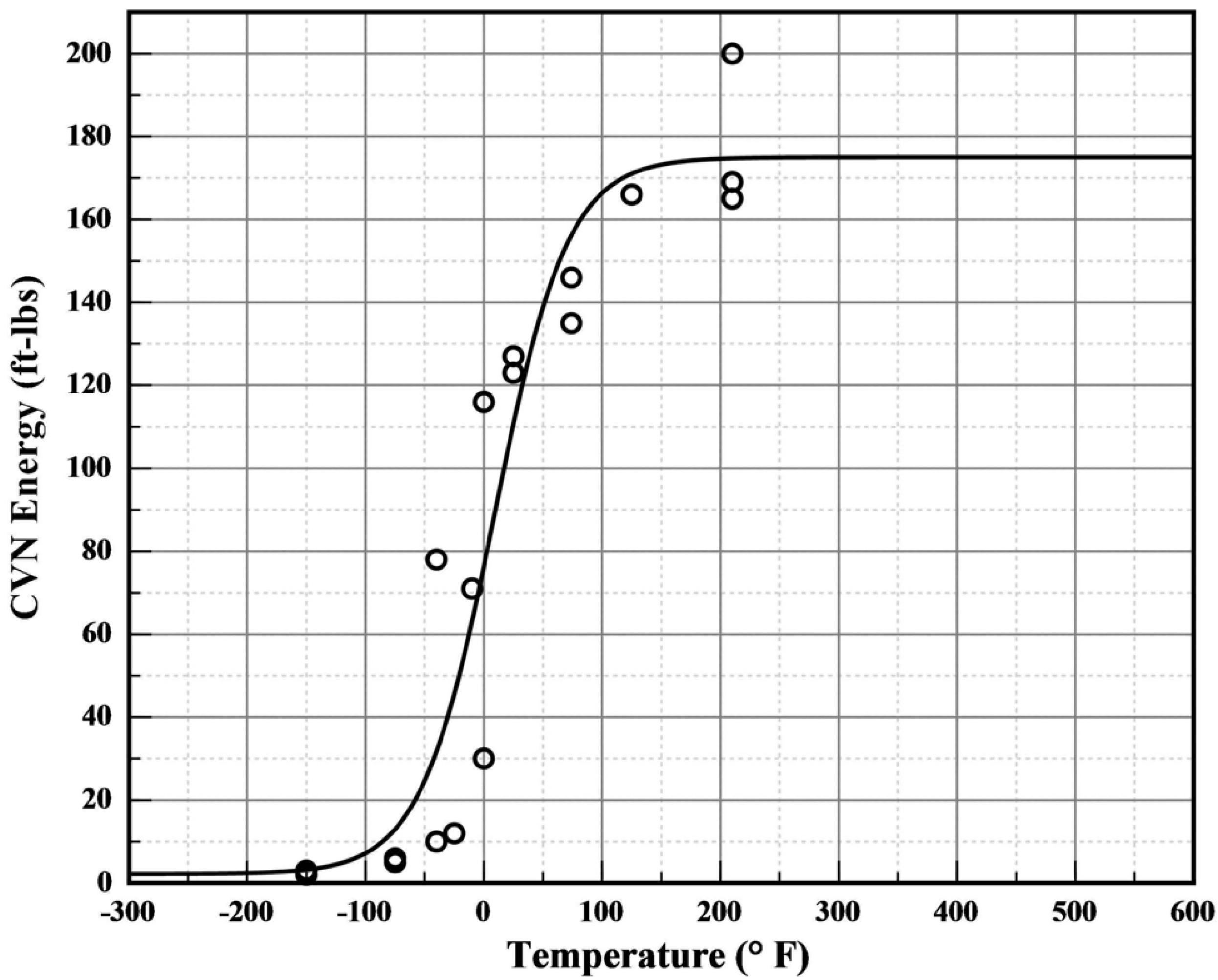
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 175.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-42.30° F Temp@35 ft-lbs=-36.00° F Temp@50 ft-lbs=-20.90° F

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **Tangential** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Tangential** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Intermediate Shell Forging 05 (Tangential)

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-150	2.0	3.2	-1.22
-150	3.0	3.2	-0.22
-75	6.0	13.0	-7.01
-75	5.0	13.0	-8.01
-40	10.0	31.8	-21.78
-40	78.0	31.8	46.22
-25	12.0	45.6	-33.59
-10	71.0	63.1	7.91
0	116.0	76.3	39.66
0	30.0	76.3	-46.34
25	127.0	110.6	16.36
25	123.0	110.6	12.36
74	135.0	156.2	-21.21
74	146.0	156.2	-10.21
125	166.0	171.0	-5.03
210	200.0	174.7	25.26
210	165.0	174.7	-9.74
210	169.0	174.7	-5.74

CVGraph 6.02 01/09/2020 Page 2/2

Unirradiated Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:31 AM

A = 46.34 B = 45.34 C = 44.30 T0 = -9.39 D = 0.00

Correlation Coefficient = 0.921

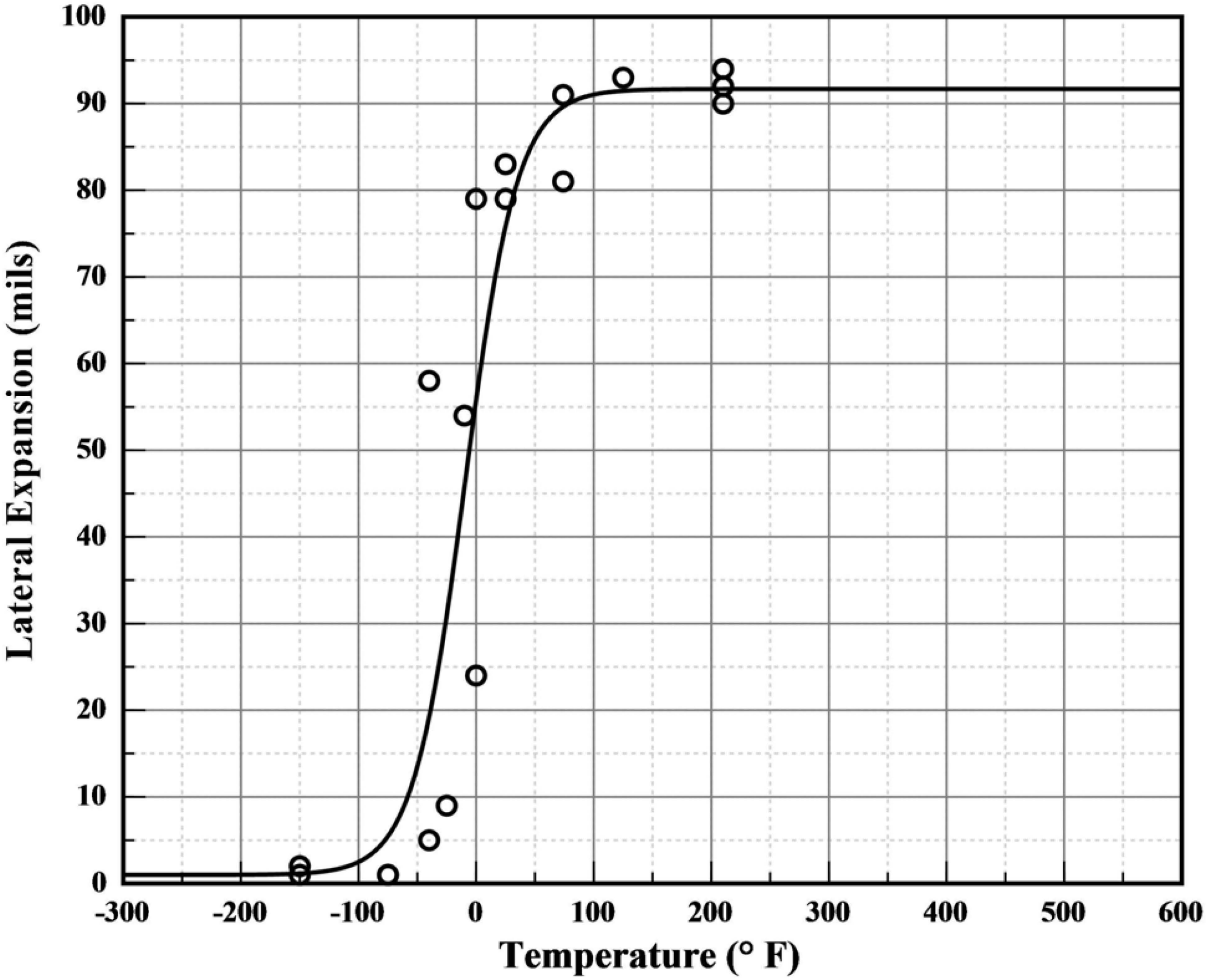
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 91.68 Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils=-20.70° F

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **Tangential** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Tangential** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Intermediate Shell Forging 05 (Tangential)

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-150	2.0	1.2	0.84
-150	1.0	1.2	-0.16
-75	1.0	5.5	-4.46
-75	1.0	5.5	-4.46
-40	5.0	19.2	-14.20
-40	58.0	19.2	38.80
-25	9.0	31.0	-22.00
-10	54.0	45.7	8.28
0	79.0	55.8	23.19
0	24.0	55.8	-31.81
25	83.0	75.8	7.16
25	79.0	75.8	3.16
74	81.0	89.6	-8.63
74	91.0	89.6	1.37
125	93.0	91.5	1.53
210	94.0	91.7	2.32
210	90.0	91.7	-1.68
210	92.0	91.7	0.32

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Unirradiated Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:33 AM

A = 50.00 B = 50.00 C = 83.46 T0 = 21.98 D = 0.00

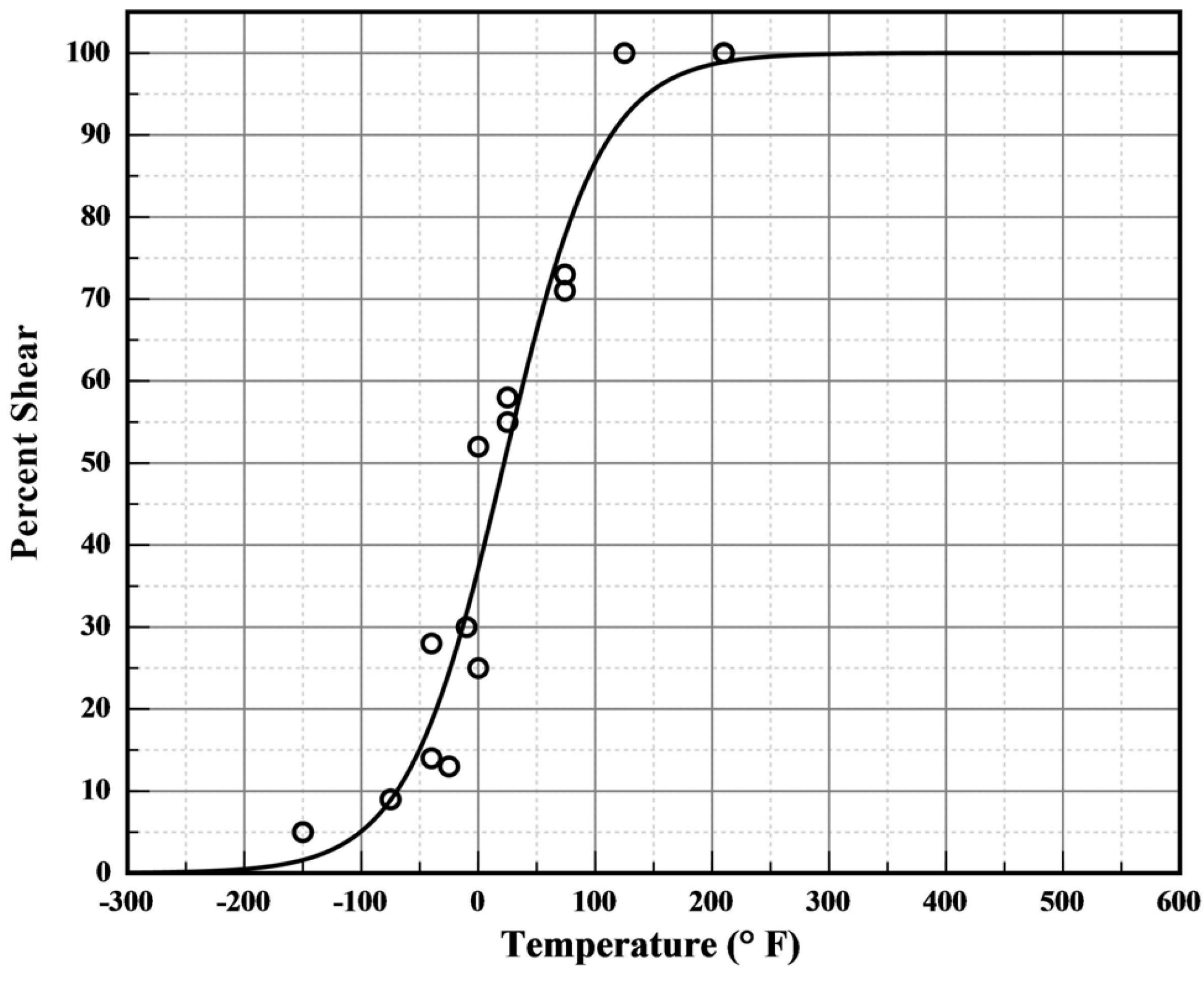
Correlation Coefficient = 0.982

Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf %Shear = 100.00 (Fixed) Lower Shelf %Shear = 0.00 (Fixed)

Temperature at 50% Shear = 22.00

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Tangential** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Tangential** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Intermediate Shell Forging 05 (Tangential)
Charpy V-Notch Data

Temperature (° F)	Input %Shear	Computed %Shear	Differential
-150	5.0	1.6	3.40
-150	5.0	1.6	3.40
-75	9.0	8.9	0.08
-75	9.0	8.9	0.08
-40	14.0	18.5	-4.46
-40	28.0	18.5	9.54
-25	13.0	24.5	-11.49
-10	30.0	31.7	-1.73
0	52.0	37.1	14.87
0	25.0	37.1	-12.13
25	58.0	51.8	6.19
25	55.0	51.8	3.19
74	71.0	77.7	-6.67
74	73.0	77.7	-4.67
125	100.0	92.2	7.81
210	100.0	98.9	1.09
210	100.0	98.9	1.09
210	100.0	98.9	1.09

CVGraph 6.02 01/09/2020 Page 2/2

Unirradiated Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:39 AM

A = 56.10 B = 53.90 C = 115.54 T0 = 19.17 D = 0.00

Correlation Coefficient = 0.917

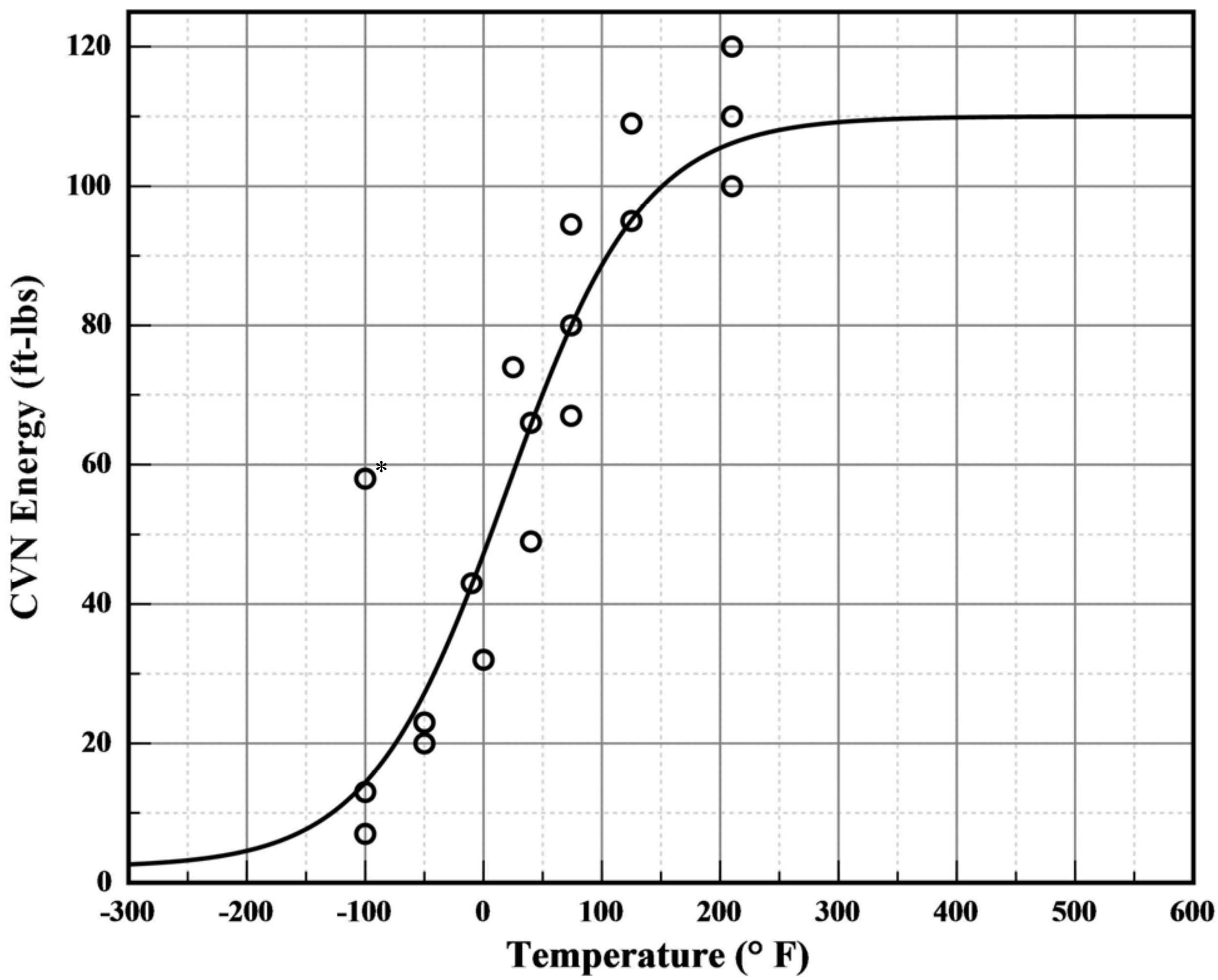
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 110.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-41.80° F Temp@35 ft-lbs=-28.60° F Temp@50 ft-lbs= 6.10° F

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **Axial** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

* See note on next page

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Axial** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Intermediate Shell Forging 05 (Axial)

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-100	13.0	14.4	-1.36
-100	7.0	14.4	-7.36
-100*	58.0	14.4	43.65
-50	23.0	27.2	-4.20
-50	20.0	27.2	-7.20
-10	43.0	42.8	0.23
0	32.0	47.2	-15.24
25	74.0	58.8	15.18
40	49.0	65.7	-16.71
40	66.0	65.7	0.29
74	94.5	79.9	14.58
74	67.0	79.9	-12.92
74	80.0	79.9	0.08
125	95.0	95.1	-0.12
125	109.0	95.1	13.88
210	120.0	106.2	13.82
210	100.0	106.2	-6.18
210	110.0	106.2	3.82

* Legibility for this test temperature is not clear. It could be 0°F rather than the -100°F shown here. Using -100°F is consistent with WCAP-9455, Rev. 4 and conservative, providing a larger T_{30} shift. This may be revisited, if necessary, during a future evaluation, e.g. next tested capsule's credibility evaluation.

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Unirradiated Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:37 AM

A = 42.38 B = 41.38 C = 119.89 T0 = 22.85 D = 0.00

Correlation Coefficient = 0.915

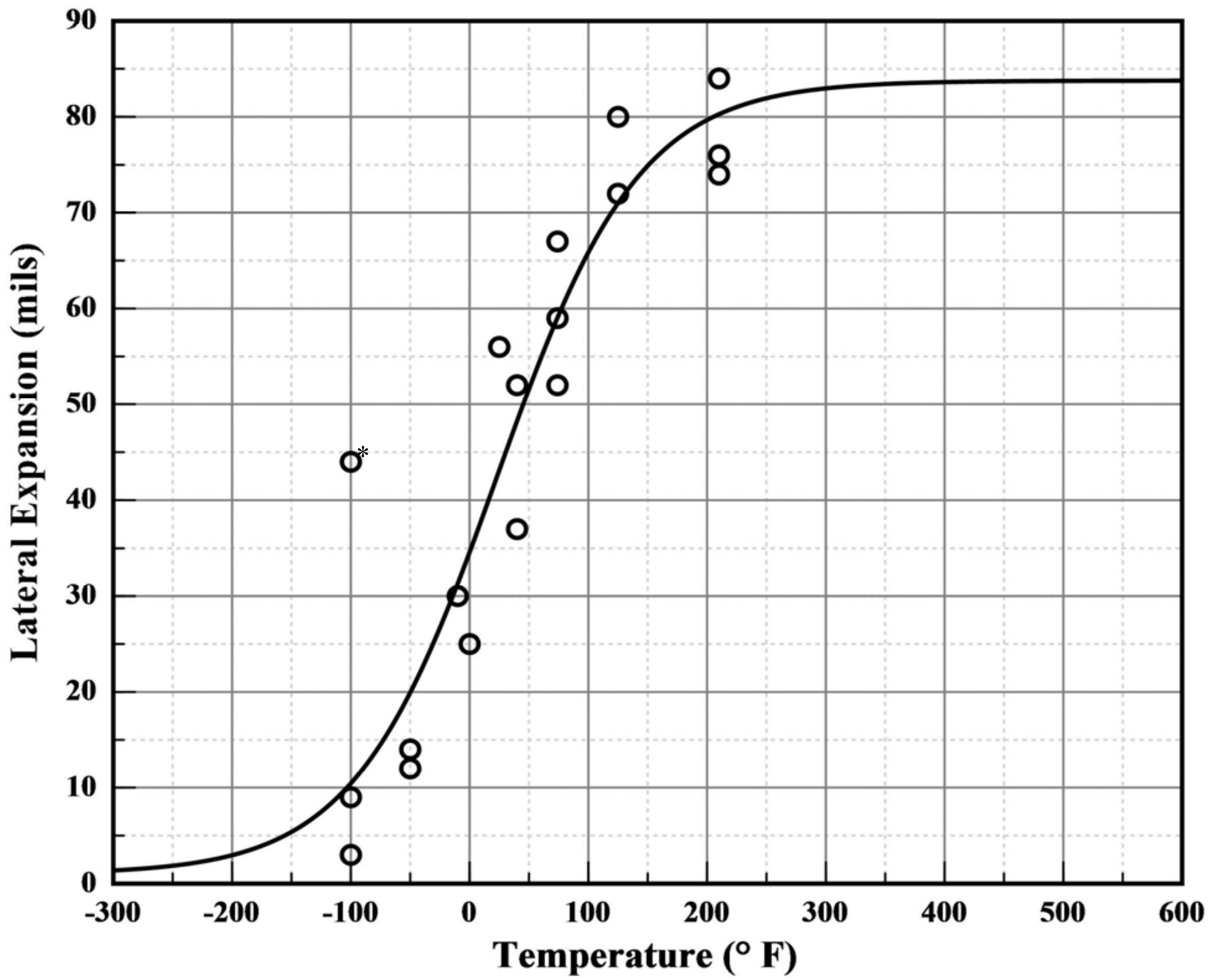
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 83.77 Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 1.30° F

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **Axial** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



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* See note on next page

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Axial** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Intermediate Shell Forging 05 (Axial)

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-100	9.0	10.4	-1.45
-100	3.0	10.4	-7.45
-100*	44.0	10.4	33.55
-50	14.0	19.9	-5.93
-50	12.0	19.9	-7.93
-10	30.0	31.3	-1.32
0	25.0	34.6	-9.59
25	56.0	43.1	12.87
40	37.0	48.3	-11.27
40	52.0	48.3	3.73
74	67.0	59.0	7.96
74	52.0	59.0	-7.04
74	59.0	59.0	-0.04
125	72.0	71.0	0.97
125	80.0	71.0	8.97
210	84.0	80.3	3.73
210	76.0	80.3	-4.27
210	74.0	80.3	-6.27

* Legibility for this test temperature is not clear. It could be 0°F rather than the -100°F shown here. Using -100°F is consistent with WCAP-9455, Rev. 4 and conservative, providing a larger T_{30} shift. This may be revisited, if necessary, during future evaluation, e.g. next tested capsule's credibility evaluation.

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Unirradiated Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:38 AM

A = 50.00 B = 50.00 C = 88.38 T0 = 56.39 D = 0.00

Correlation Coefficient = 0.967

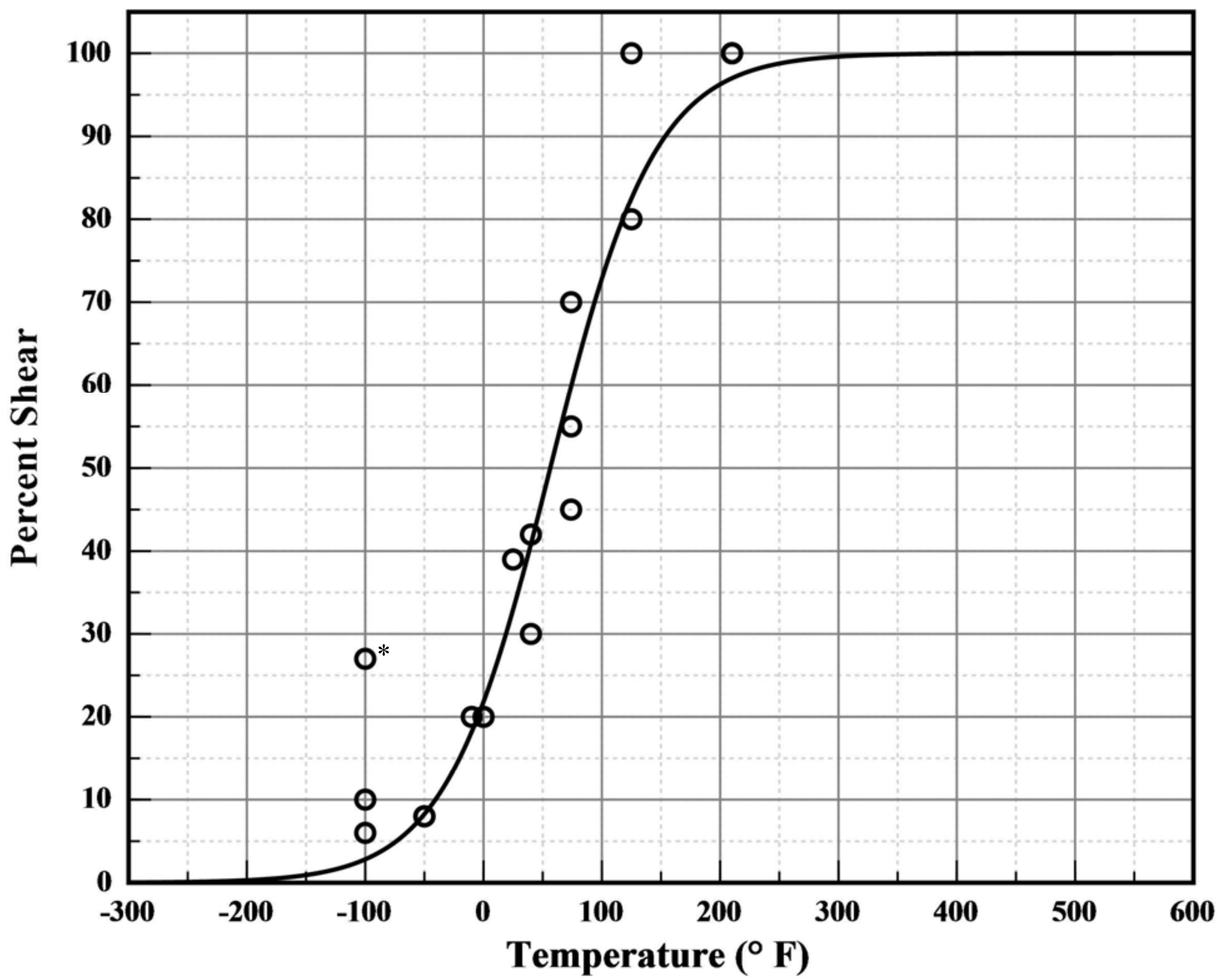
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf %Shear = 100.00 (Fixed) Lower Shelf %Shear = 0.00 (Fixed)

Temperature at 50% Shear = 56.40

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **Axial** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



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* See note on next page

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Axial** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Intermediate Shell Forging 05 (Axial)

Charpy V-Notch Data

Temperature (° F)	Input %Shear	Computed %Shear	Differential
-100	10.0	2.8	7.18
-100	6.0	2.8	3.18
-100*	27.0	2.8	24.18
-50	8.0	8.3	-0.26
-50	8.0	8.3	-0.26
-10	20.0	18.2	1.79
0	20.0	21.8	-1.82
25	39.0	33.0	6.05
40	30.0	40.8	-10.83
40	42.0	40.8	1.17
74	70.0	59.8	10.17
74	45.0	59.8	-14.83
74	55.0	59.8	-4.83
125	80.0	82.5	-2.53
125	100.0	82.5	17.47
210	100.0	97.0	3.00
210	100.0	97.0	3.00
210	100.0	97.0	3.00

* Legibility for this test temperature is not clear. It could be 0°F rather than the -100°F shown here. Using -100°F is consistent with WCAP-9455, Rev. 4 and conservative, providing a larger T_{30} shift. This may be revisited, if necessary, during future evaluation, e.g. next tested capsule's credibility evaluation.

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Unirradiated Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:41 AM

A = 73.10 B = 70.90 C = 104.77 T0 = 16.13 D = 0.00

Correlation Coefficient = 0.987

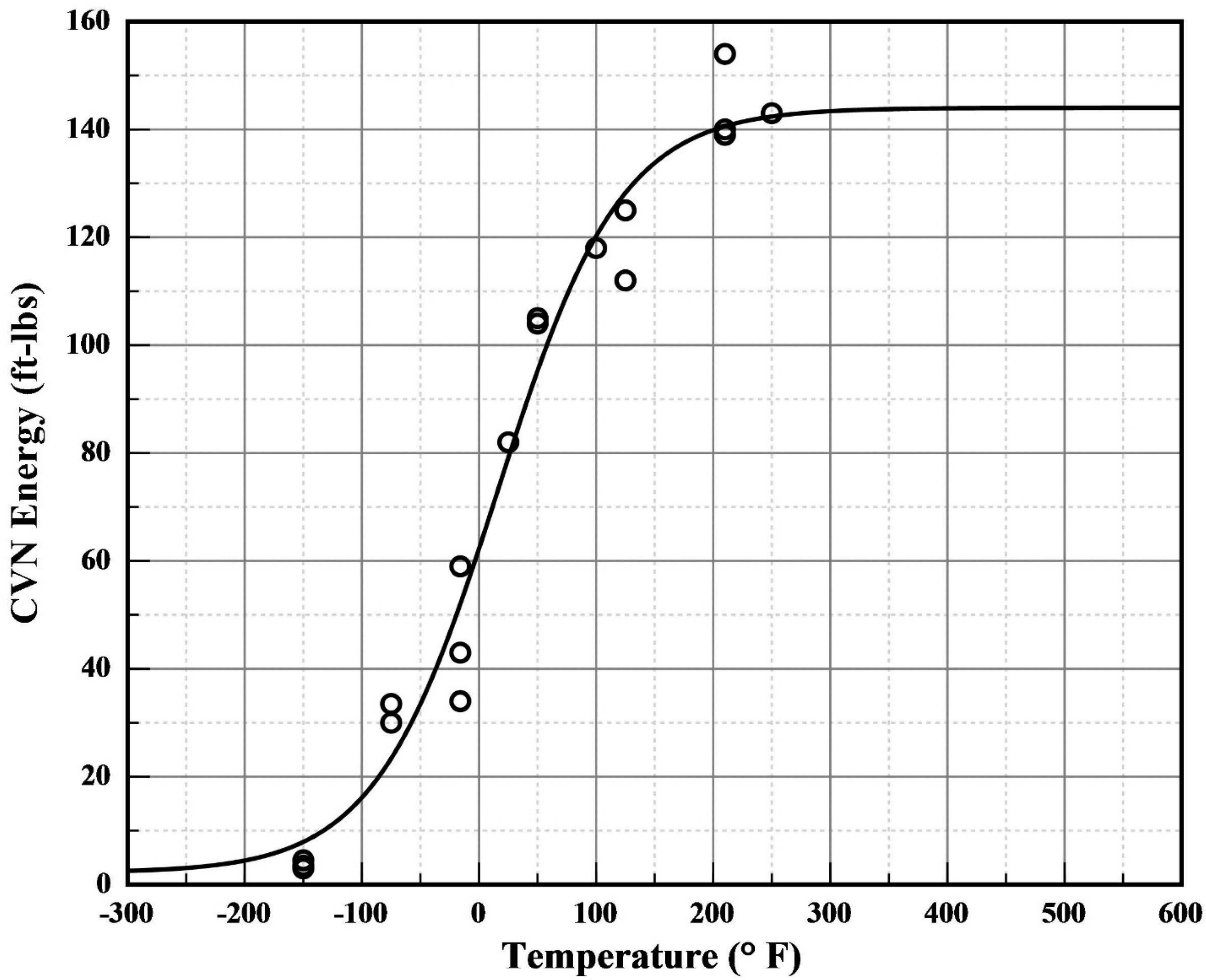
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 144.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-57.70° F Temp@35 ft-lbs=-46.70° F Temp@50 ft-lbs=-19.20° F

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**

Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



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Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Surveillance Program Weld Metal

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-150	4.5	7.9	-3.41
-150	3.5	7.9	-4.41
-150	3.0	7.9	-4.91
-75	30.0	23.4	6.62
-75	33.5	23.4	10.12
-16	43.0	52.0	-9.02
-16	34.0	52.0	-18.02
-16	59.0	52.0	6.98
25	82.0	79.1	2.91
50	105.0	95.3	9.74
50	104.0	95.3	8.74
100	118.0	120.2	-2.20
125	125.0	128.2	-3.23
125	112.0	128.2	-16.23
210	139.0	140.6	-1.58
210	154.0	140.6	13.42
210	140.0	140.6	-0.58
250	143.0	142.4	0.61

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Unirradiated Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:43 AM

A = 47.66 B = 46.66 C = 78.92 T0 = -2.74 D = 0.00

Correlation Coefficient = 0.990

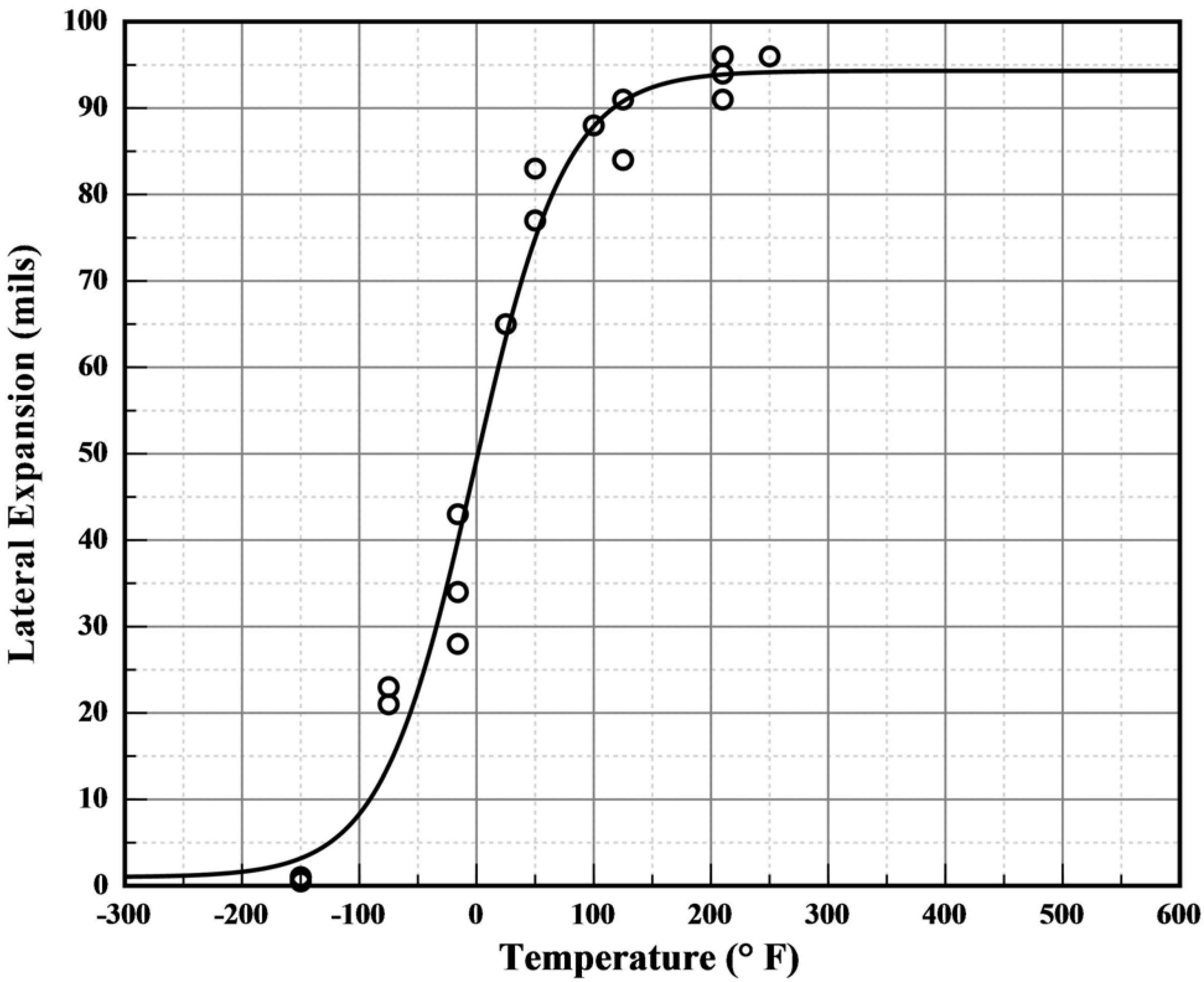
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 94.32 Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils=-24.70° F

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**

Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



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Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Surveillance Program Weld Metal

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-150	1.0	3.2	-2.18
-150	1.0	3.2	-2.18
-150	0.5	3.2	-2.68
-75	21.0	13.9	7.11
-75	23.0	13.9	9.11
-16	34.0	39.9	-5.89
-16	28.0	39.9	-11.89
-16	43.0	39.9	3.11
25	65.0	63.4	1.58
50	77.0	74.9	2.10
50	83.0	74.9	8.10
100	88.0	87.9	0.11
125	91.0	90.8	0.21
125	84.0	90.8	-6.79
210	94.0	93.9	0.10
210	96.0	93.9	2.10
210	91.0	93.9	-2.90
250	96.0	94.2	1.83

CVGraph 6.02 01/09/2020 Page 2/2

Unirradiated Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:44 AM

A = 50.00 B = 50.00 C = 110.15 T0 = 0.36 D = 0.00

Correlation Coefficient = 0.989

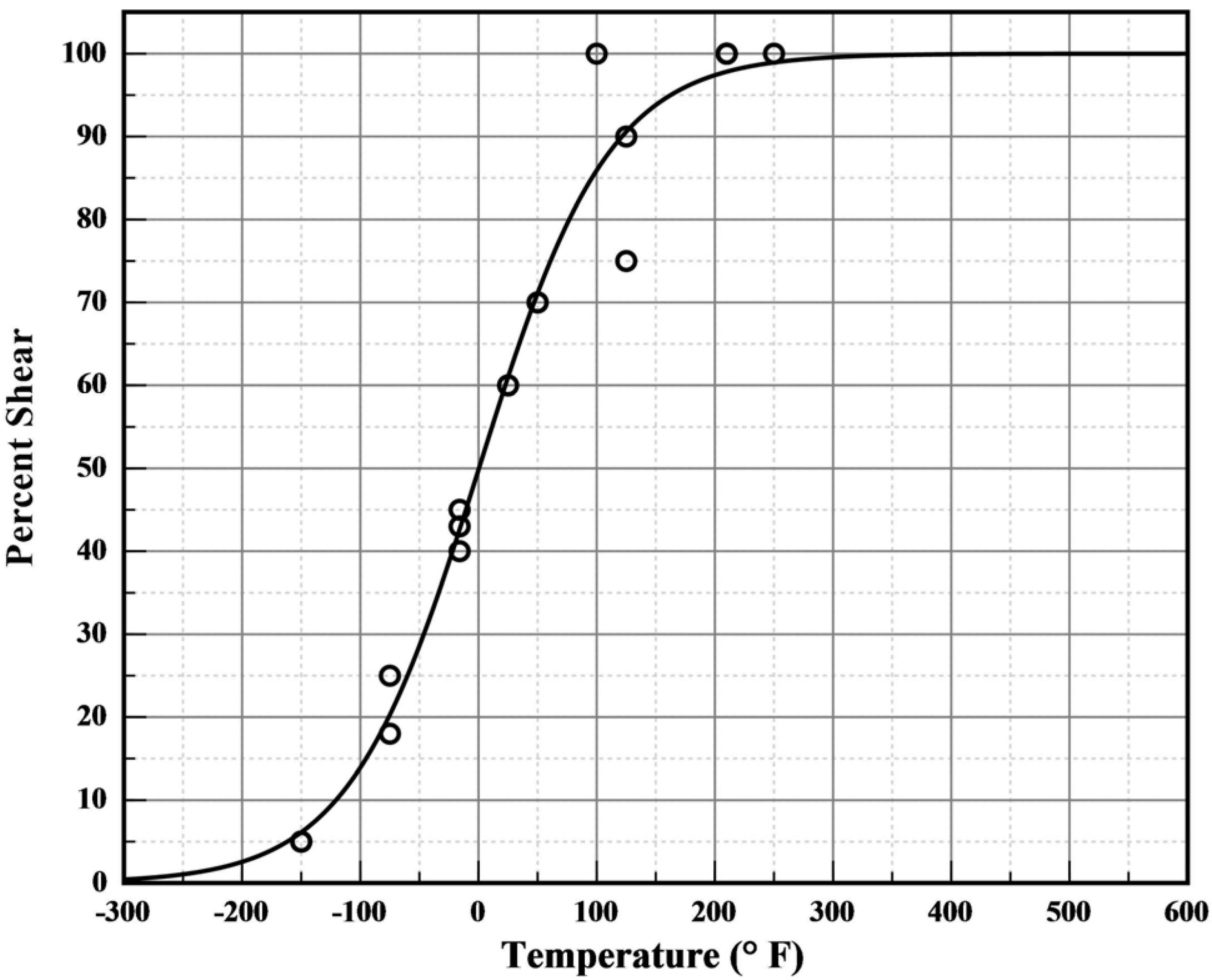
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf %Shear = 100.00 (Fixed) Lower Shelf %Shear = 0.00 (Fixed)

Temperature at 50% Shear = 0.40

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**

Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Surveillance Program Weld Metal

Charpy V-Notch Data

Temperature (° F)	Input %Shear	Computed %Shear	Differential
-150	5.0	6.1	-1.12
-150	5.0	6.1	-1.12
-150	5.0	6.1	-1.12
-75	18.0	20.3	-2.29
-75	25.0	20.3	4.71
-16	43.0	42.6	0.37
-16	40.0	42.6	-2.63
-16	45.0	42.6	2.37
25	60.0	61.0	-1.00
50	70.0	71.1	-1.12
50	70.0	71.1	-1.12
100	100.0	85.9	14.07
125	90.0	90.6	-0.58
125	75.0	90.6	-15.58
210	100.0	97.8	2.17
210	100.0	97.8	2.17
210	100.0	97.8	2.17
250	100.0	98.9	1.06

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Unirradiated Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:45 AM

A = 66.10 B = 63.90 C = 53.13 T0 = -67.87 D = 0.00

Correlation Coefficient = 0.960

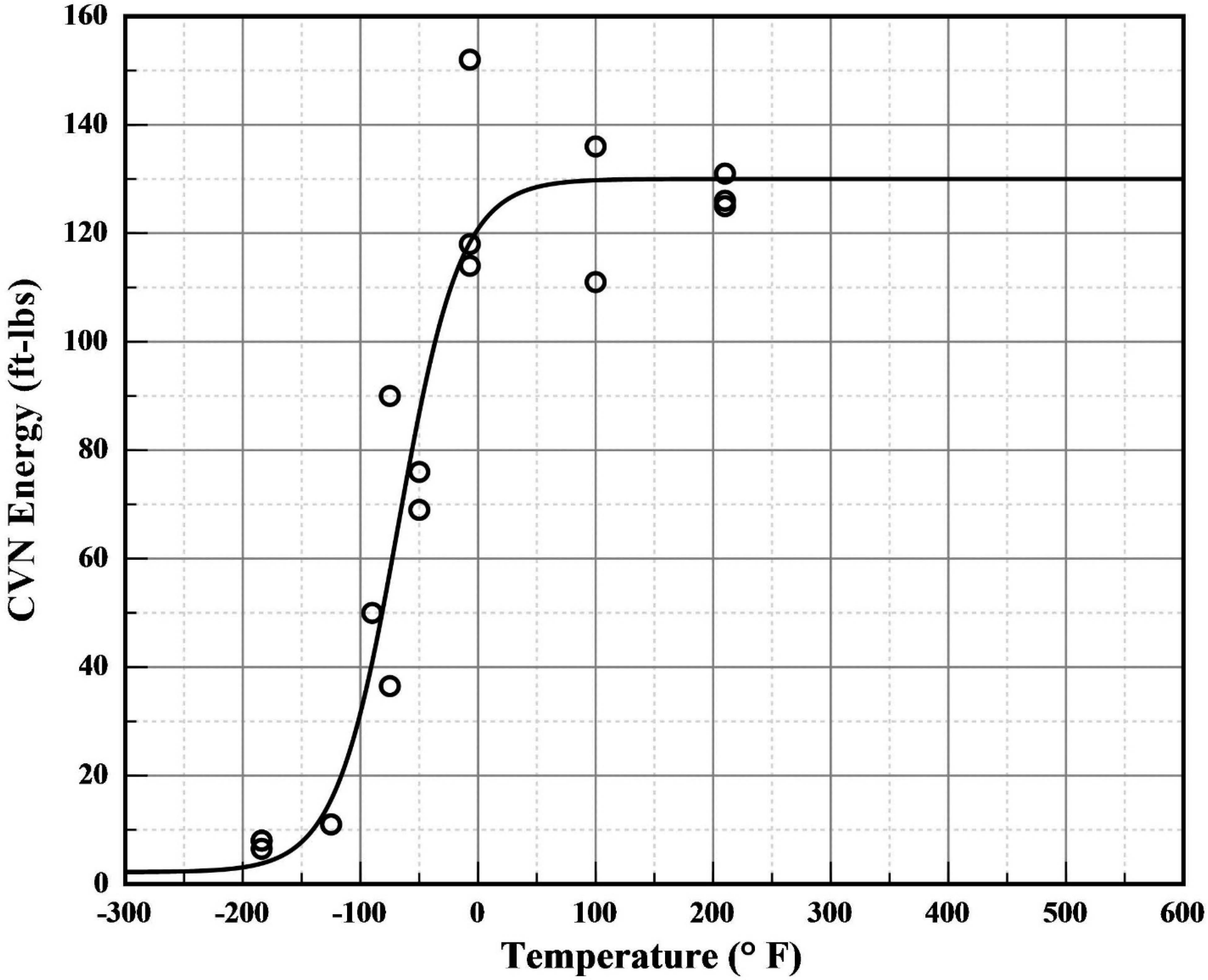
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 130.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-101.80° F Temp@35 ft-lbs=-96.10° F Temp@50 ft-lbs=-81.50° F

Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**

Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



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Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Heat-Affected Zone (HAZ)

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-184	8.0	3.8	4.21
-184	6.5	3.8	2.71
-184	8.0	3.8	4.21
-125	11.0	15.5	-4.53
-125	11.0	15.5	-4.53
-90	50.0	40.9	9.08
-75	36.5	57.6	-21.08
-75	90.0	57.6	32.42
-50	76.0	86.8	-10.82
-50	69.0	86.8	-17.82
-7	152.0	118.3	33.73
-7	114.0	118.3	-4.27
-7	118.0	118.3	-0.27
100	111.0	129.8	-18.77
100	136.0	129.8	6.23
210	126.0	130.0	-4.00
210	131.0	130.0	1.00
210	125.0	130.0	-5.00

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Unirradiated Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:46 AM

A = 39.49 B = 38.49 C = 49.67 T0 = -63.96 D = 0.00

Correlation Coefficient = 0.979

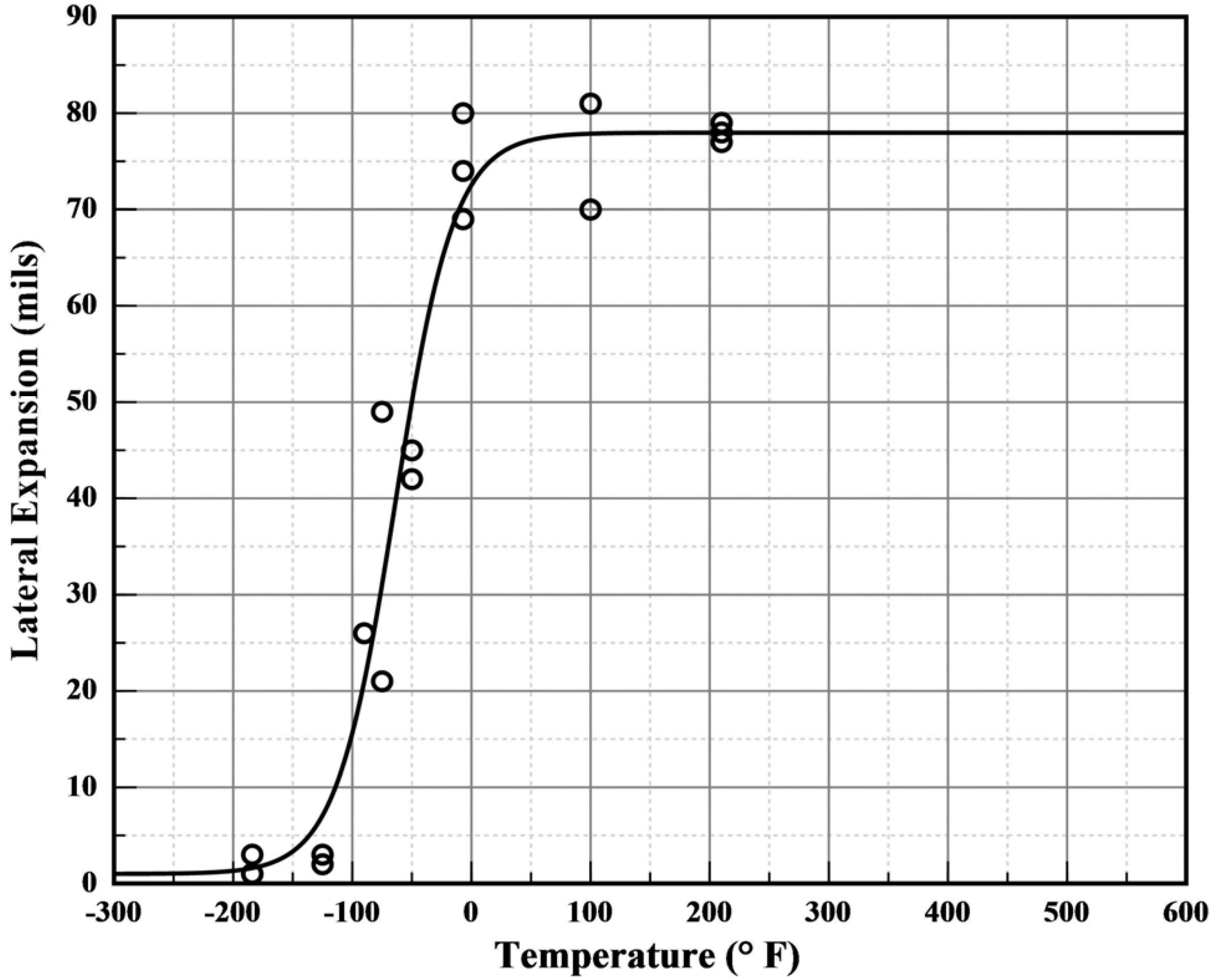
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 77.97 Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils=-69.70° F

Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**

Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



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Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Heat-Affected Zone (HAZ)

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-184	3.0	1.6	1.39
-184	1.0	1.6	-0.61
-184	1.0	1.6	-0.61
-125	2.0	7.1	-5.07
-125	3.0	7.1	-4.07
-90	26.0	21.0	5.02
-75	21.0	31.1	-10.07
-75	49.0	31.1	17.93
-50	45.0	50.0	-5.03
-50	42.0	50.0	-8.03
-7	80.0	70.9	9.08
-7	74.0	70.9	3.08
-7	69.0	70.9	-1.92
100	70.0	77.9	-7.87
100	81.0	77.9	3.13
210	77.0	78.0	-0.97
210	79.0	78.0	1.03
210	78.0	78.0	0.03

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Unirradiated Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:47 AM

A = 50.00 B = 50.00 C = 65.42 T0 = -60.30 D = 0.00

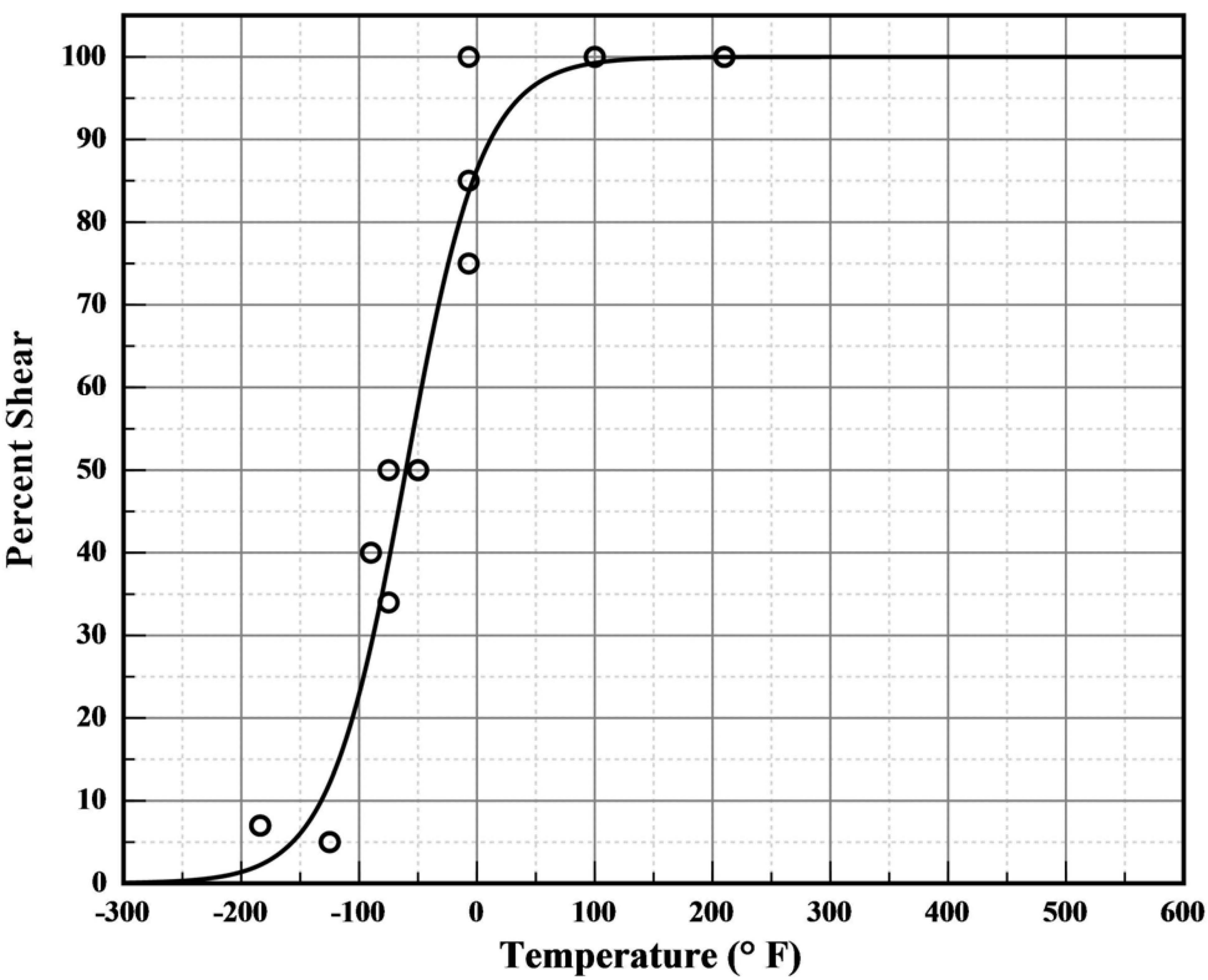
Correlation Coefficient = 0.983

Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf %Shear = 100.00 (Fixed) Lower Shelf %Shear = 0.00 (Fixed)

Temperature at 50% Shear = -60.20

Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



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Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Unirradiated Heat-Affected Zone (HAZ)

Charpy V-Notch Data

Temperature (° F)	Input %Shear	Computed %Shear	Differential
-184	7.0	2.2	4.77
-184	7.0	2.2	4.77
-184	7.0	2.2	4.77
-125	5.0	12.2	-7.15
-125	5.0	12.2	-7.15
-90	40.0	28.7	11.26
-75	34.0	39.0	-4.95
-75	50.0	39.0	11.05
-50	50.0	57.8	-7.81
-50	50.0	57.8	-7.81
-7	100.0	83.6	16.39
-7	75.0	83.6	-8.61
-7	85.0	83.6	1.39
100	100.0	99.3	0.74
100	100.0	99.3	0.74
210	100.0	100.0	0.03
210	100.0	100.0	0.03
210	100.0	100.0	0.03

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C.4 CVGRAPH VERSION 6.02 INDIVIDUAL PLOTS OF CAPSULE U SPECIMENS

Capsule U Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:49 AM

A = 66.10 B = 63.90 C = 74.07 T0 = 31.76 D = 0.00

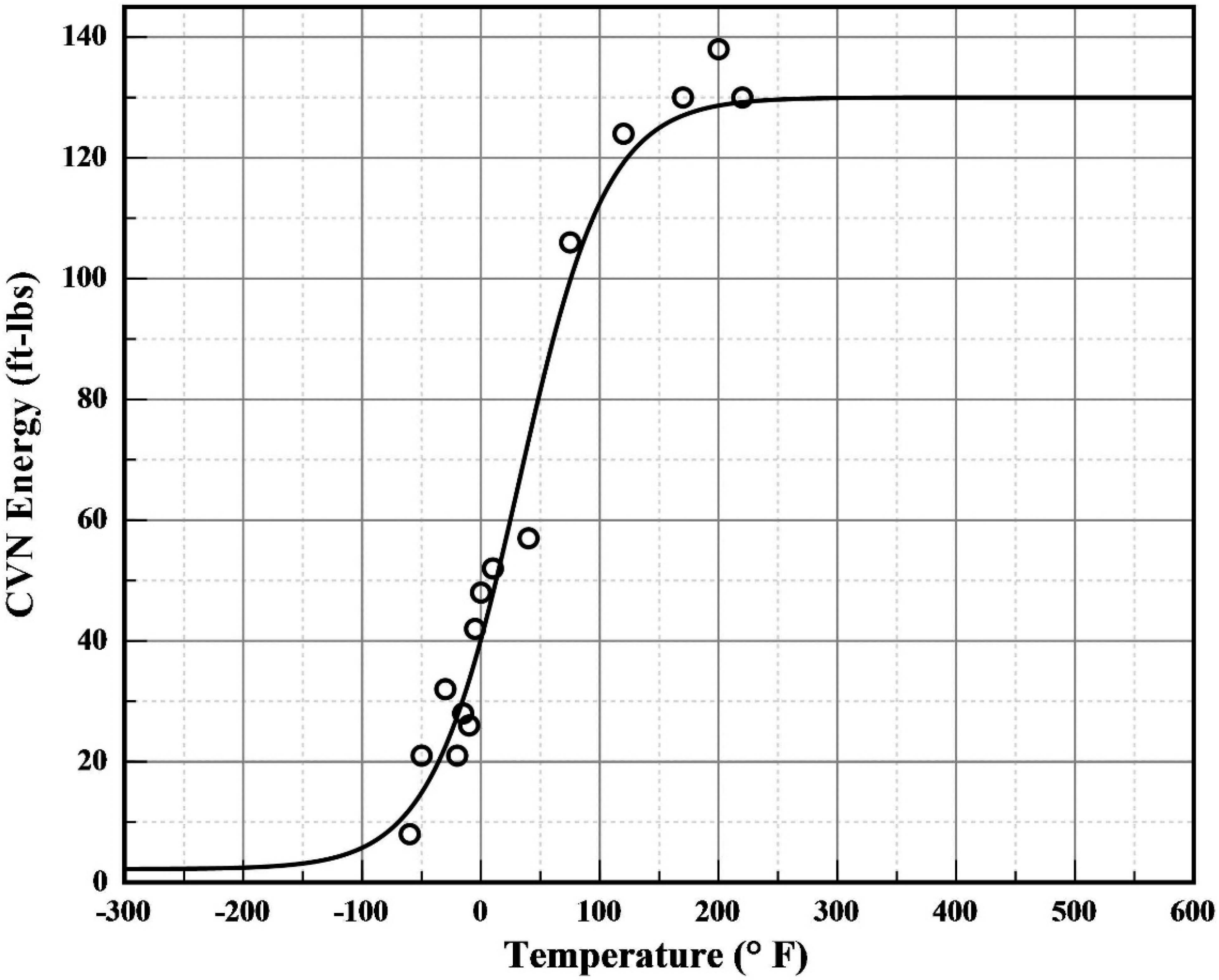
Correlation Coefficient = 0.988

Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 130.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-15.60° F Temp@35 ft-lbs= -7.60° F Temp@50 ft-lbs= 12.70° F

Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**
Orientation: **Tangential** Capsule: **U** Fluence: **6.04E+018 n/cm²**



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Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**
Orientation: **Tangential** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Intermediate Shell Forging 05 (Tangential)

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-60	8.0	12.1	-4.10
-50	21.0	14.9	6.14
-30	32.0	22.5	9.51
-20	21.0	27.5	-6.53
-15	28.0	30.4	-2.38
-10	26.0	33.5	-7.46
-5	42.0	36.8	5.24
0	48.0	40.3	7.74
10	52.0	47.8	4.15
40	57.0	73.2	-16.18
75	106.0	99.7	6.33
120	124.0	119.2	4.80
170	130.0	127.0	2.99
200	138.0	128.7	9.35
220	130.0	129.2	0.79

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:50 AM

A = 44.76 B = 43.76 C = 75.61 T0 = 26.23 D = 0.00

Correlation Coefficient = 0.987

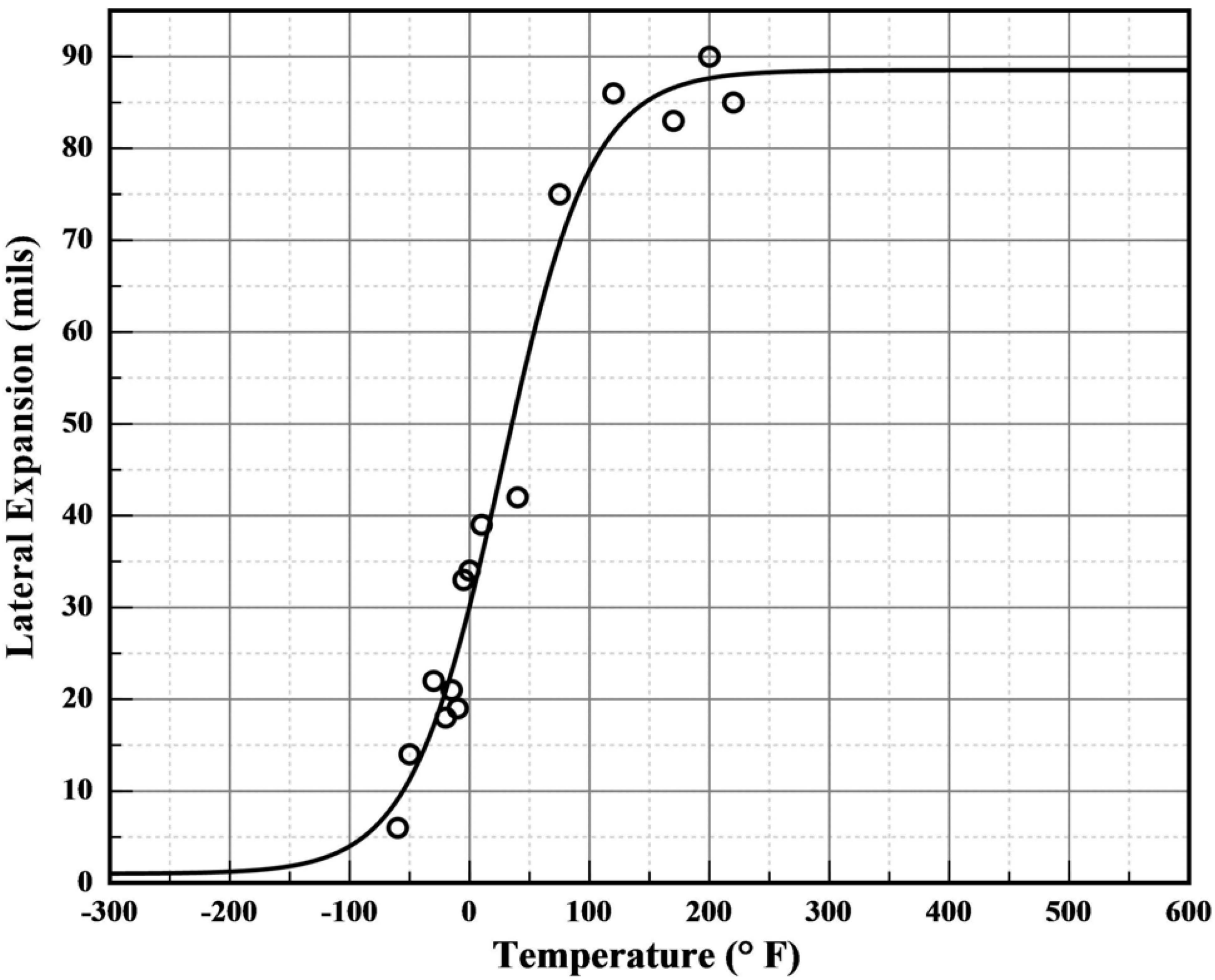
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 88.51 Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 9.10° F

Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**

Orientation: **Tangential** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**
Orientation: **Tangential** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Intermediate Shell Forging 05 (Tangential)

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-60	6.0	9.1	-3.11
-50	14.0	11.3	2.72
-30	22.0	17.1	4.87
-20	18.0	20.9	-2.90
-15	21.0	23.0	-2.01
-10	19.0	25.3	-6.26
-5	33.0	27.6	5.36
0	34.0	30.2	3.84
10	39.0	35.5	3.49
40	42.0	52.6	-10.64
75	75.0	69.6	5.37
120	86.0	81.8	4.24
170	83.0	86.6	-3.61
200	90.0	87.6	2.36
220	85.0	88.0	-3.00

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Intermediate Shell Forging 05 (Tangential)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:51 AM

A = 50.00 B = 50.00 C = 66.36 T0 = 46.69 D = 0.00

Correlation Coefficient = 0.994

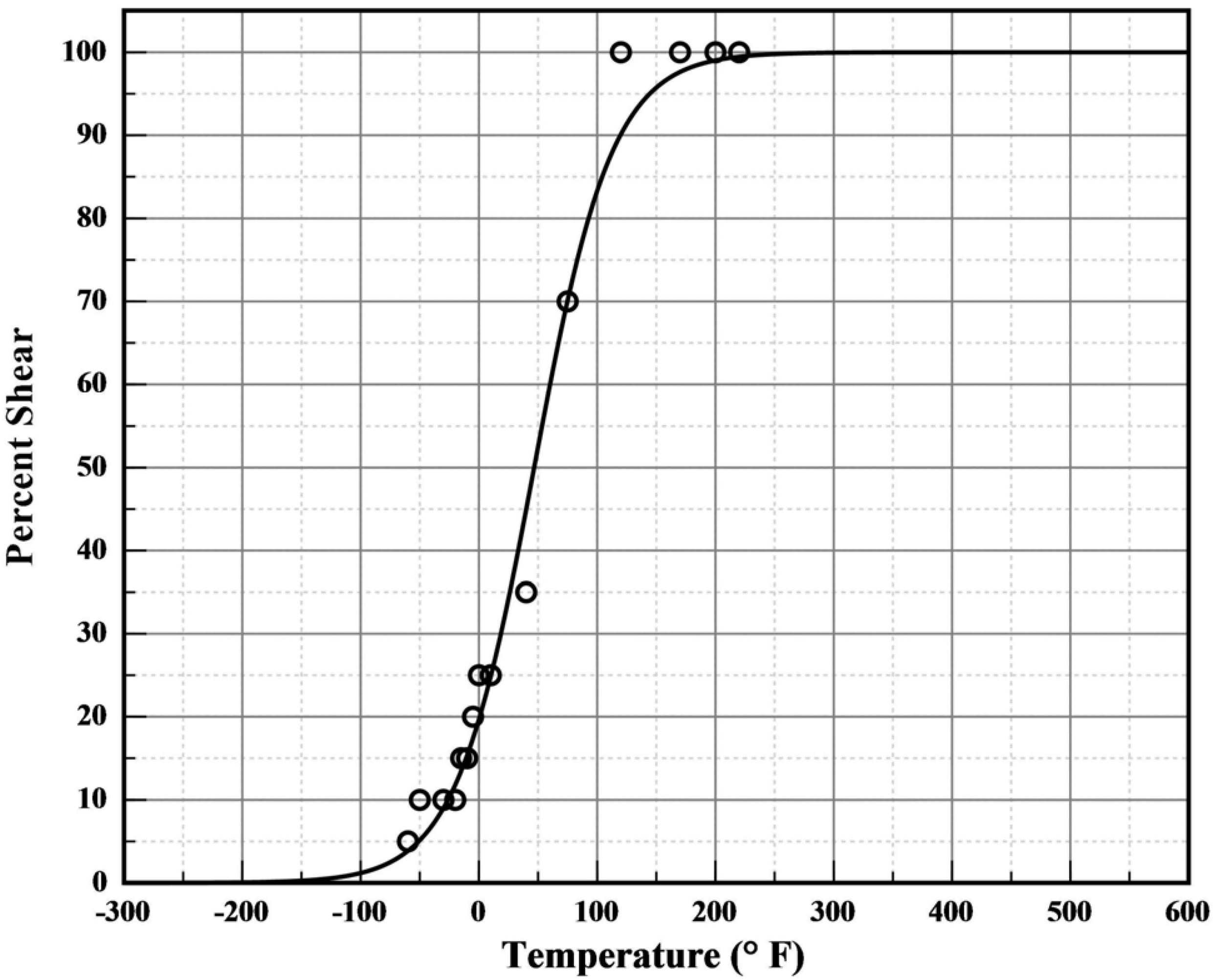
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf %Shear = 100.00 (Fixed) Lower Shelf %Shear = 0.00 (Fixed)

Temperature at 50% Shear = 46.70

Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**

Orientation: **Tangential** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508Cl2** Heat: **527828**
Orientation: **Tangential** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Intermediate Shell Forging 05 (Tangential)

Charpy V-Notch Data

Temperature (° F)	Input %Shear	Computed %Shear	Differential
-60	5.0	3.9	1.14
-50	10.0	5.1	4.85
-30	10.0	9.0	0.98
-20	10.0	11.8	-1.82
-15	15.0	13.5	1.52
-10	15.0	15.3	-0.33
-5	20.0	17.4	2.61
0	25.0	19.7	5.33
10	25.0	24.9	0.13
40	35.0	45.0	-9.98
75	70.0	70.1	-0.12
120	100.0	90.1	9.89
170	100.0	97.6	2.37
200	100.0	99.0	0.98
220	100.0	99.5	0.54

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:52 AM

A = 53.60 B = 51.40 C = 104.98 T0 = 31.50 D = 0.00

Correlation Coefficient = 0.974

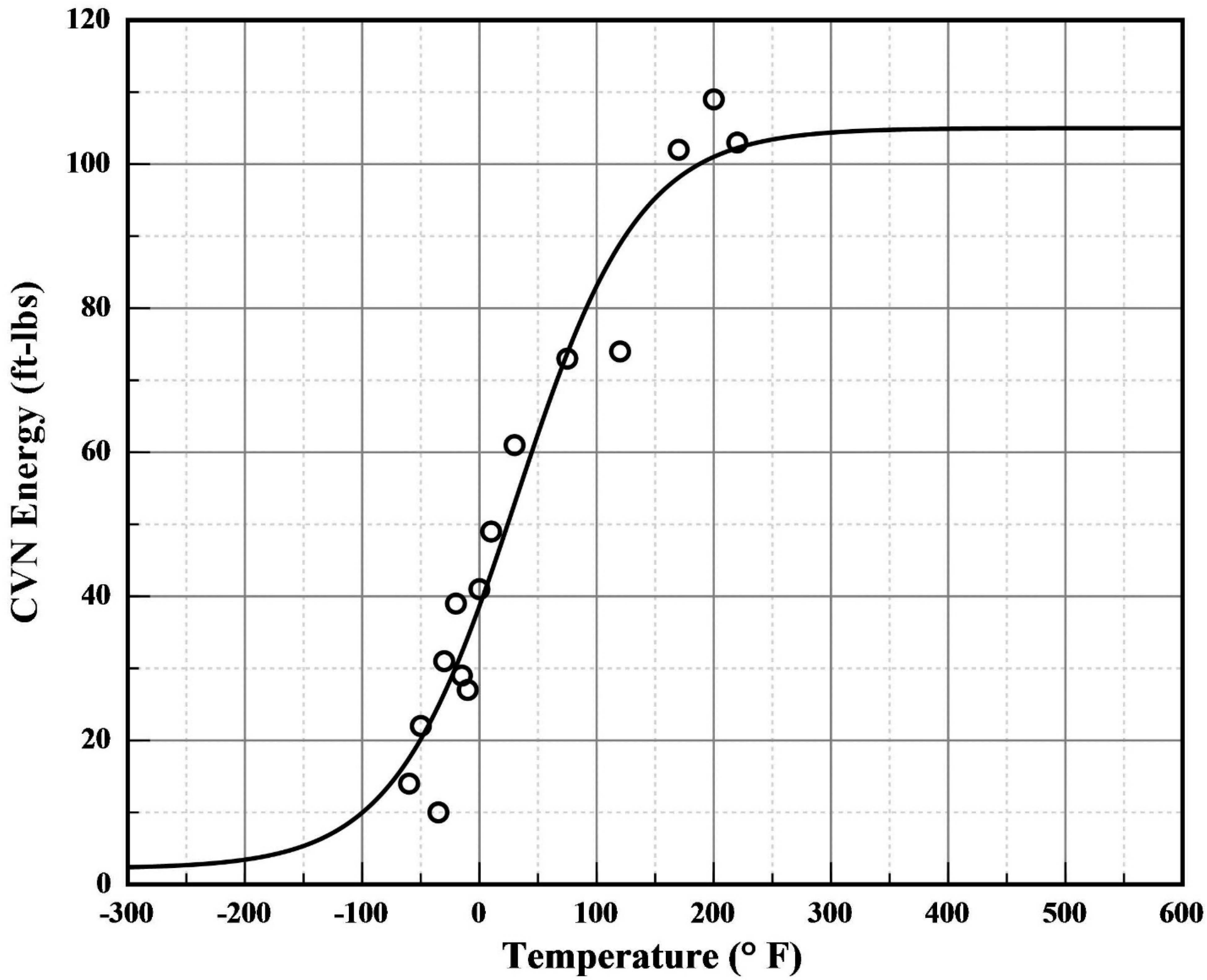
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 105.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-20.50° F Temp@35 ft-lbs= -8.20° F Temp@50 ft-lbs= 24.20° F

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **Axial** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Axial** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Intermediate Shell Forging 05 (Axial)

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-60	14.0	17.5	-3.51
-50	22.0	20.2	1.84
-35	10.0	24.8	-14.79
-30	31.0	26.5	4.48
-20	39.0	30.2	8.77
-15	29.0	32.2	-3.21
-10	27.0	34.3	-7.28
0	41.0	38.6	2.38
10	49.0	43.2	5.78
30	61.0	52.9	8.14
75	73.0	73.8	-0.76
120	74.0	88.9	-14.93
170	102.0	98.1	3.86
200	109.0	101.0	7.99
220	103.0	102.2	0.76

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:54 AM

A = 38.73 B = 37.73 C = 96.58 T0 = 24.51 D = 0.00

Correlation Coefficient = 0.978

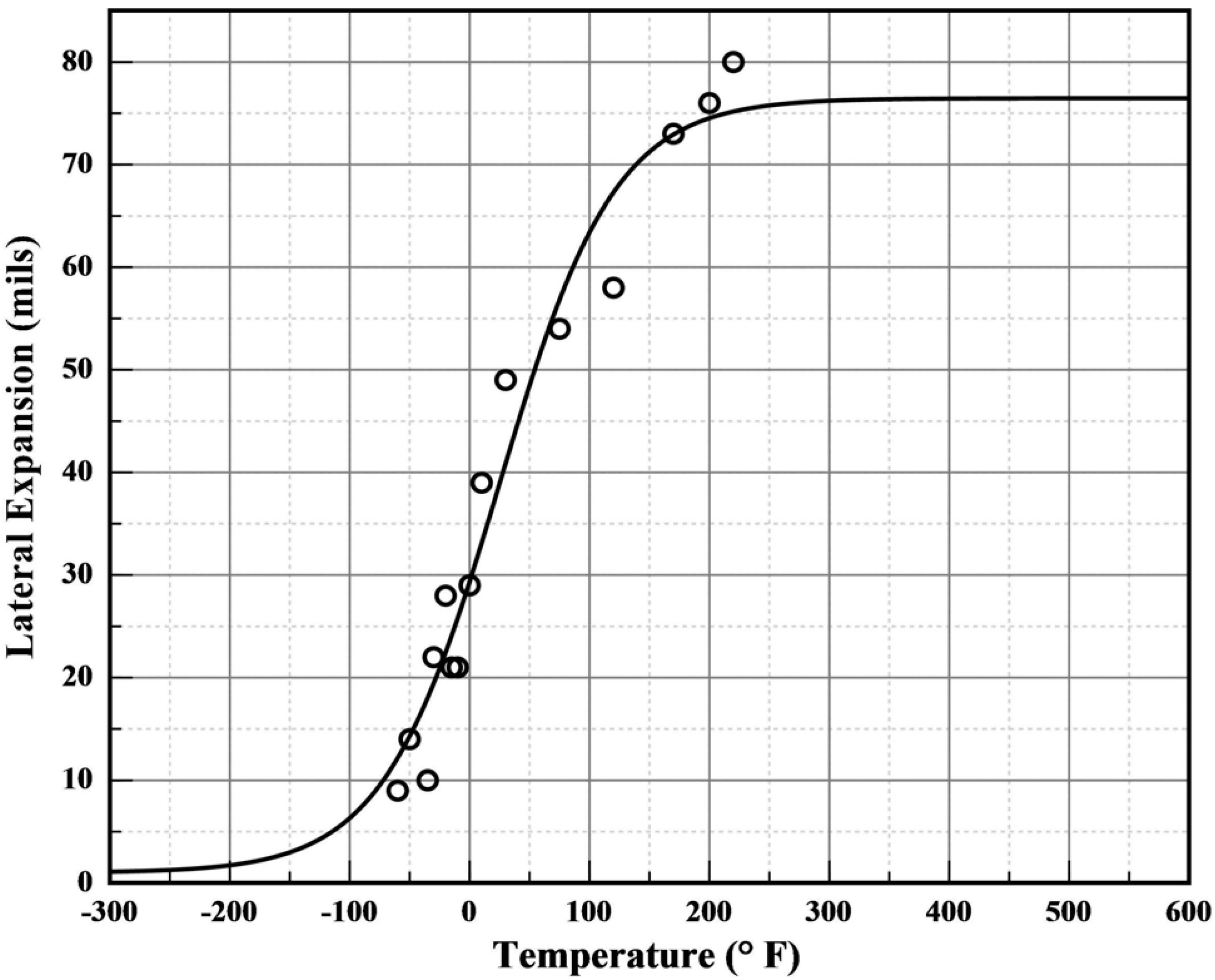
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 76.47 Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 15.00° F

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **Axial** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Axial** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Intermediate Shell Forging 05 (Axial)

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-60	9.0	12.2	-3.17
-50	14.0	14.3	-0.29
-35	10.0	18.0	-8.04
-30	22.0	19.4	2.56
-20	28.0	22.5	5.52
-15	21.0	24.1	-3.10
-10	21.0	25.8	-4.79
0	29.0	29.4	-0.36
10	39.0	33.1	5.90
30	49.0	40.9	8.13
75	54.0	56.8	-2.84
120	58.0	67.3	-9.29
170	73.0	72.9	0.07
200	76.0	74.5	1.48
220	80.0	75.2	4.83

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Intermediate Shell Forging 05 (Axial)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:55 AM

A = 50.00 B = 50.00 C = 97.89 T0 = 69.10 D = 0.00

Correlation Coefficient = 0.986

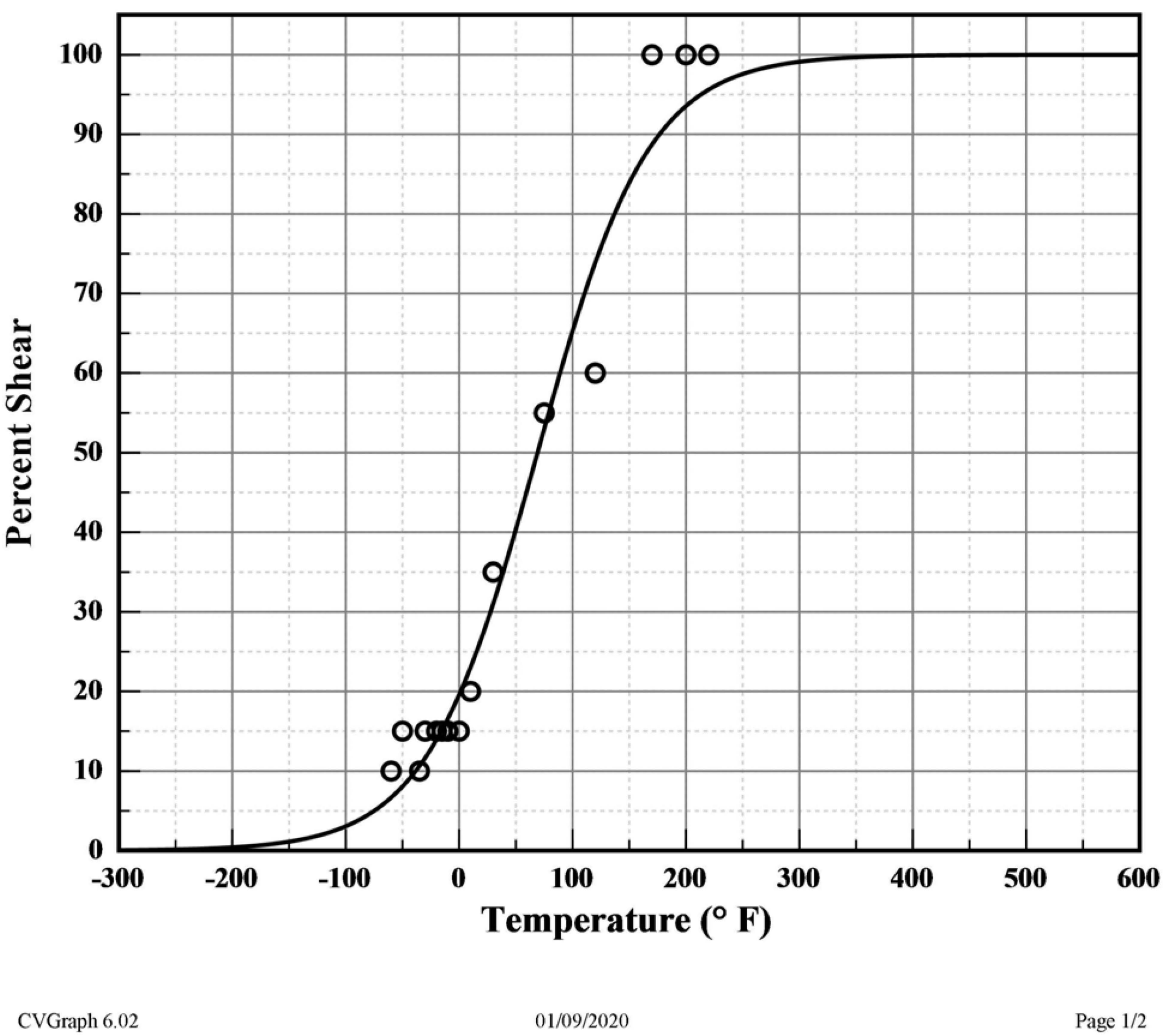
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf %Shear = 100.00 (Fixed) Lower Shelf %Shear = 0.00 (Fixed)

Temperature at 50% Shear = 69.10

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **Axial** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **Axial** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Intermediate Shell Forging 05 (Axial)

Charpy V-Notch Data

Temperature (° F)	Input %Shear	Computed %Shear	Differential
-60	10.0	6.7	3.33
-50	15.0	8.1	6.93
-35	10.0	10.6	-0.65
-30	15.0	11.7	3.34
-20	15.0	13.9	1.06
-15	15.0	15.2	-0.21
-10	15.0	16.6	-1.57
0	15.0	19.6	-4.59
10	20.0	23.0	-3.01
30	35.0	31.0	3.98
75	55.0	53.0	1.99
120	60.0	73.9	-13.88
170	100.0	88.7	11.29
200	100.0	93.5	6.45
220	100.0	95.6	4.38

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:56 AM

A = 68.60 B = 66.40 C = 104.80 T0 = 44.44 D = 0.00

Correlation Coefficient = 0.967

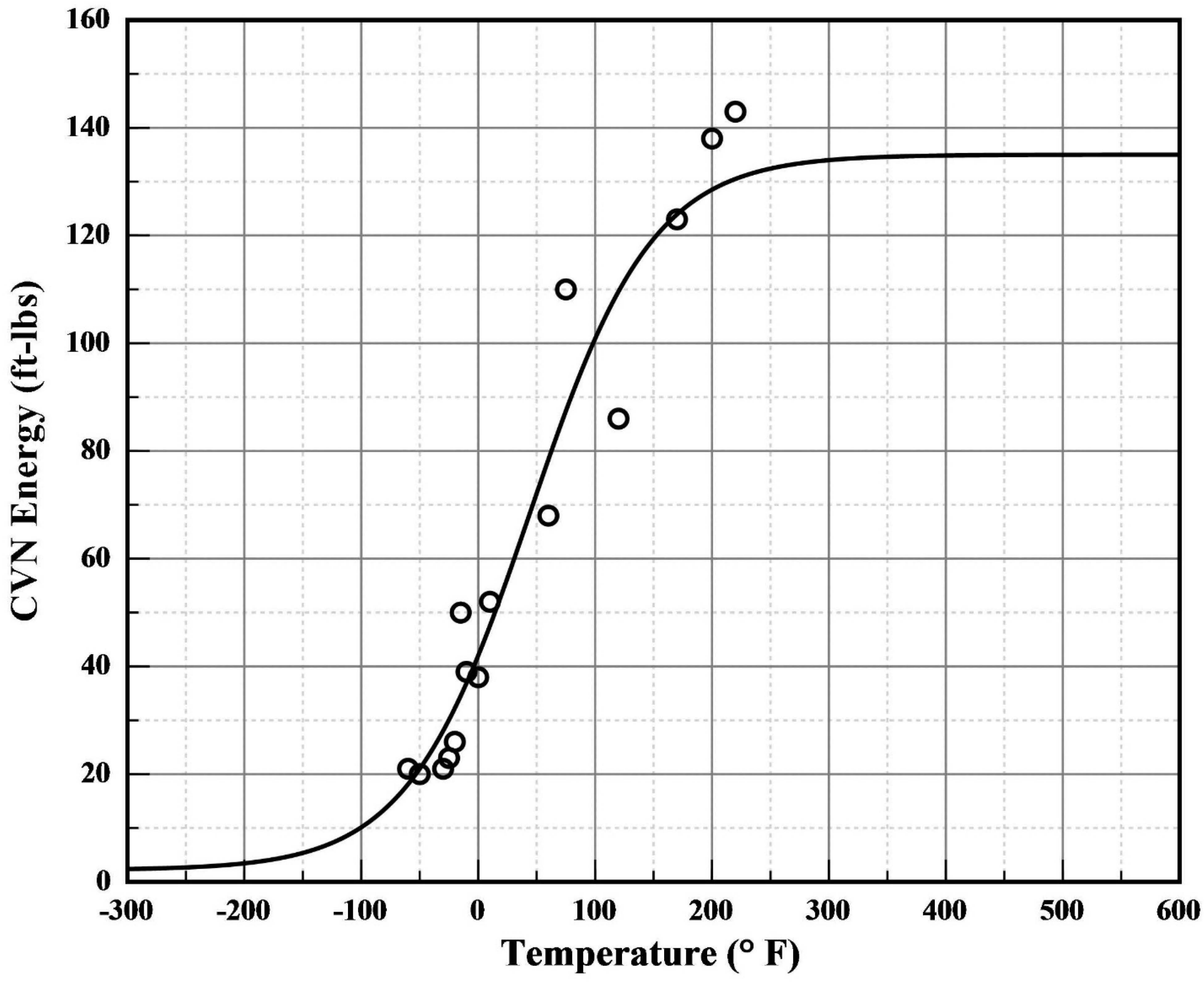
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 135.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-25.10° F Temp@35 ft-lbs=-13.90° F Temp@50 ft-lbs= 14.30° F

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**

Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**
Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Surveillance Program Weld Metal

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-60	21.0	18.1	2.87
-50	20.0	21.0	-1.00
-30	21.0	28.0	-7.04
-25	23.0	30.1	-7.08
-20	26.0	32.2	-6.24
-15	50.0	34.5	15.48
-10	39.0	36.9	2.09
0	38.0	42.0	-4.02
10	52.0	47.5	4.47
60	68.0	78.4	-10.39
75	110.0	87.4	22.57
120	86.0	109.6	-23.61
170	123.0	123.9	-0.92
200	138.0	128.5	9.49
220	143.0	130.5	12.50

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:57 AM

A = 44.88 B = 43.88 C = 98.10 T0 = 26.09 D = 0.00

Correlation Coefficient = 0.980

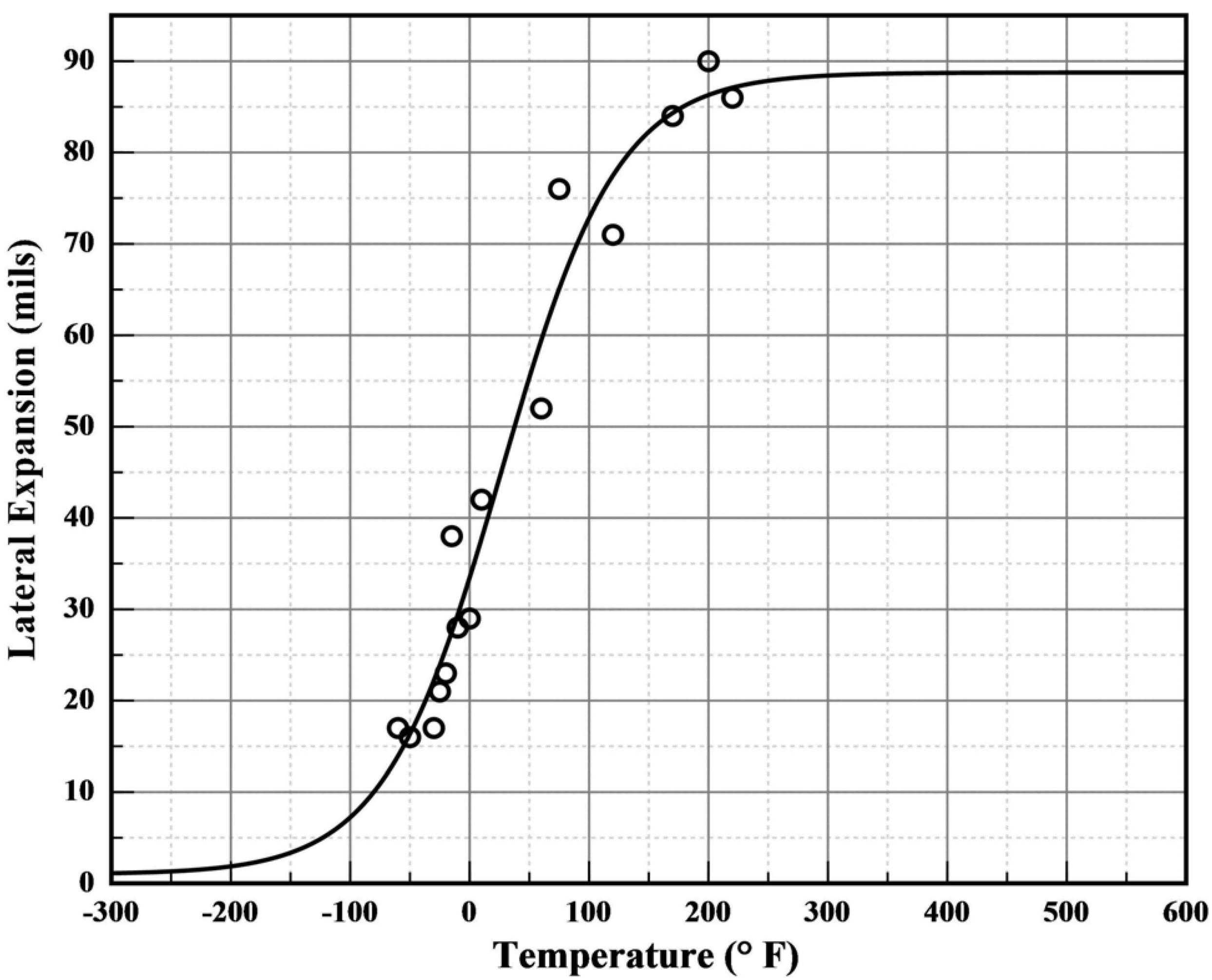
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 88.76 Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 3.70° F

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**

Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**
Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Surveillance Program Weld Metal

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-60	17.0	13.9	3.06
-50	16.0	16.3	-0.35
-30	17.0	22.2	-5.21
-25	21.0	23.9	-2.89
-20	23.0	25.7	-2.66
-15	38.0	27.5	10.50
-10	28.0	29.4	-1.43
0	29.0	33.5	-4.48
10	42.0	37.7	4.25
60	52.0	59.5	-7.47
75	76.0	65.1	10.89
120	71.0	77.5	-6.48
170	84.0	84.3	-0.33
200	90.0	86.3	3.70
220	86.0	87.1	-1.11

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Surveillance Program Weld Metal

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:58 AM

A = 50.00 B = 50.00 C = 134.87 T0 = 26.42 D = 0.00

Correlation Coefficient = 0.955

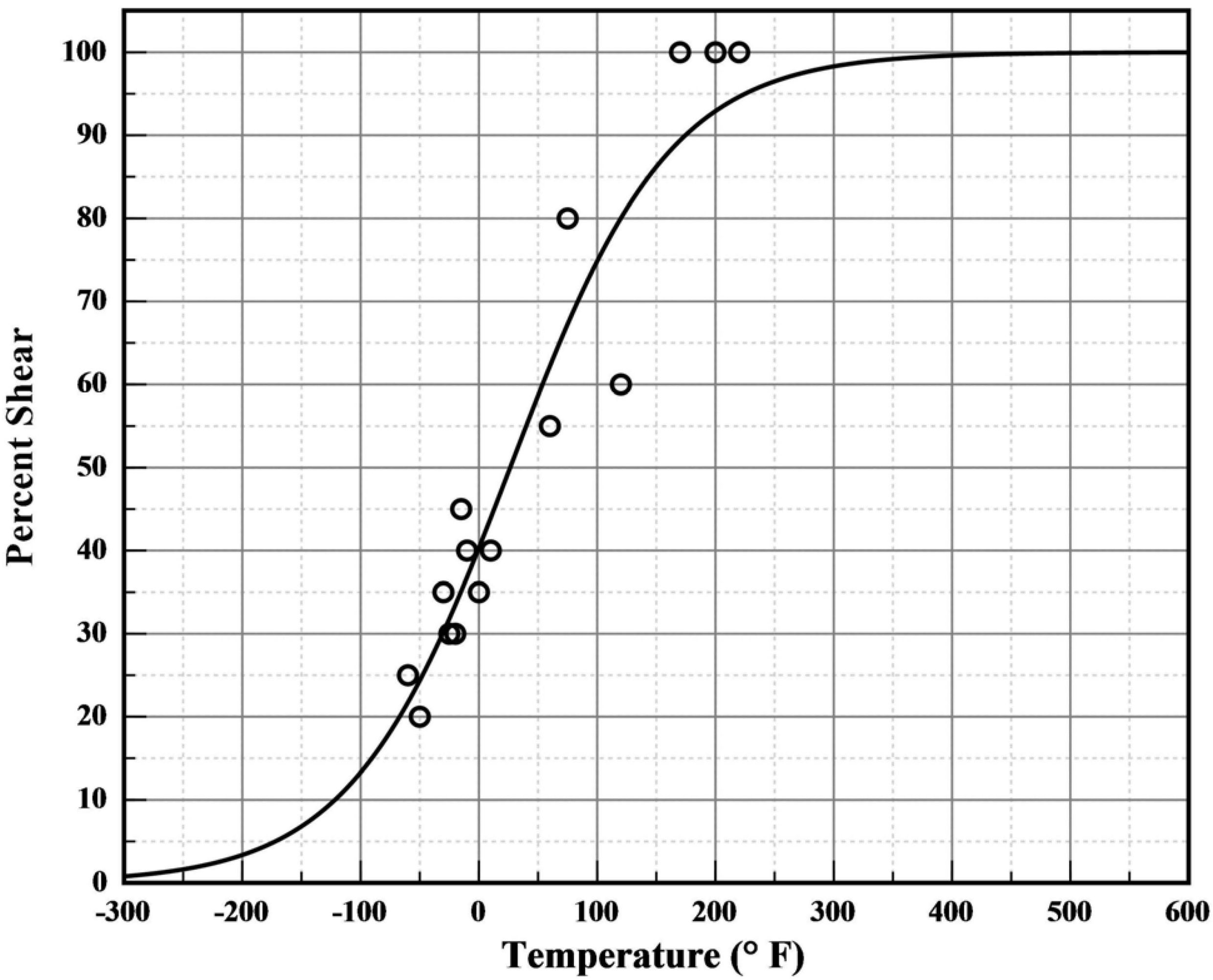
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf %Shear = 100.00 (Fixed) Lower Shelf %Shear = 0.00 (Fixed)

Temperature at 50% Shear = 26.50

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**

Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SAW** Heat: **895075**
Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Surveillance Program Weld Metal

Charpy V-Notch Data

Temperature (° F)	Input %Shear	Computed %Shear	Differential
-60	25.0	21.7	3.27
-50	20.0	24.4	-4.36
-30	35.0	30.2	4.78
-25	30.0	31.8	-1.81
-20	30.0	33.4	-3.44
-15	45.0	35.1	9.89
-10	40.0	36.8	3.18
0	35.0	40.3	-5.33
10	40.0	43.9	-3.94
60	55.0	62.2	-7.20
75	80.0	67.3	12.73
120	60.0	80.0	-20.02
170	100.0	89.4	10.63
200	100.0	92.9	7.08
220	100.0	94.6	5.36

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 11:59 AM

A = 60.10 B = 57.90 C = 104.39 T0 = -43.34 D = 0.00

Correlation Coefficient = 0.931

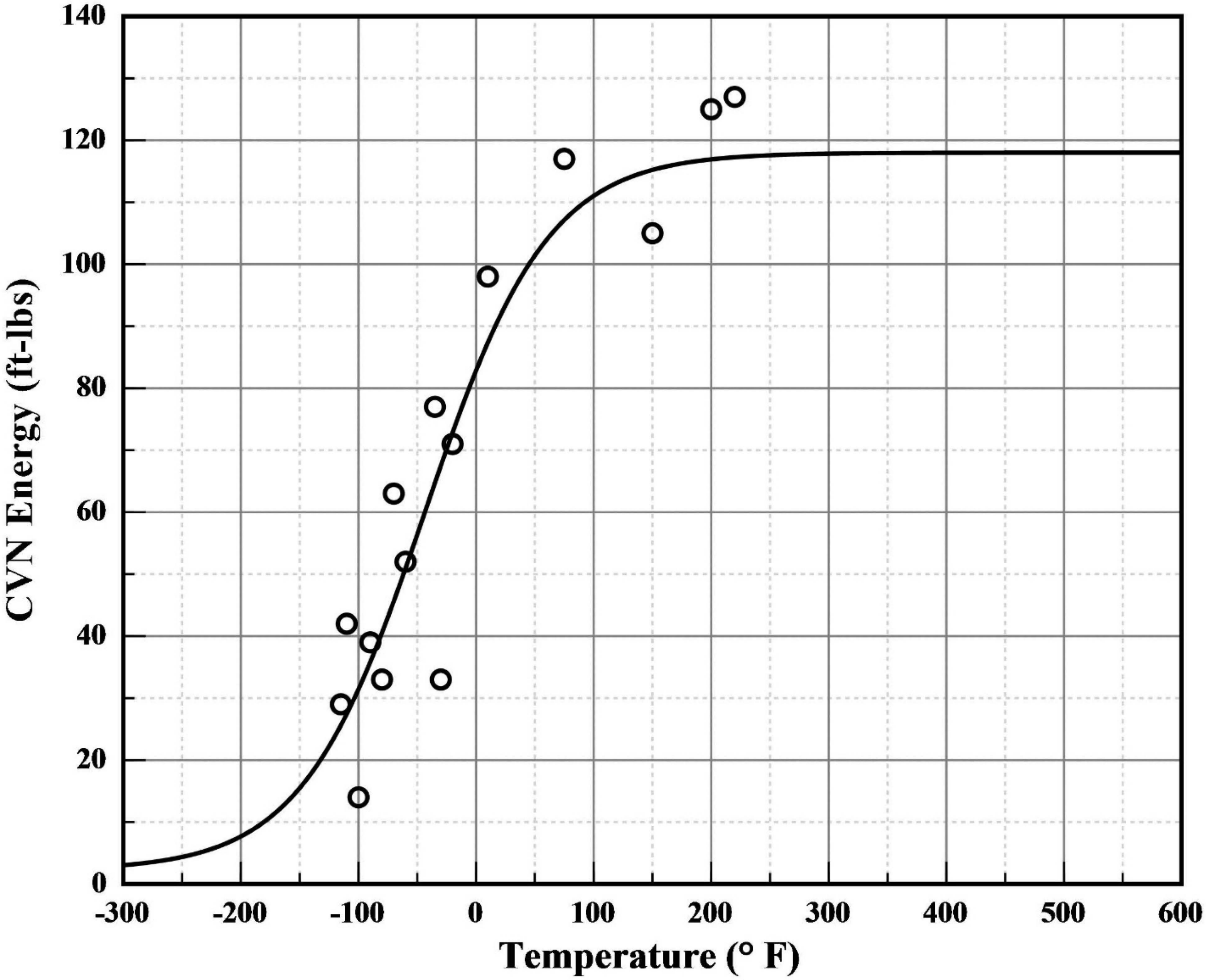
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 118.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-103.40° F Temp@35 ft-lbs=-91.70° F Temp@50 ft-lbs=-61.70° F

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Heat-Affected Zone (HAZ)

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-115	29.0	25.6	3.39
-110	42.0	27.4	14.55
-100	14.0	31.4	-17.43
-90	39.0	35.8	3.18
-80	33.0	40.6	-7.56
-70	63.0	45.6	17.37
-60	52.0	50.9	1.06
-35	77.0	64.7	12.29
-30	33.0	67.5	-34.46
-20	71.0	72.8	-1.83
10	98.0	87.4	10.65
75	117.0	107.1	9.87
150	105.0	115.2	-10.22
200	125.0	116.9	8.08
220	127.0	117.3	9.74

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 12:02 PM

A = 37.98 B = 36.98 C = 109.83 T0 = -37.27 D = 0.00

Correlation Coefficient = 0.956

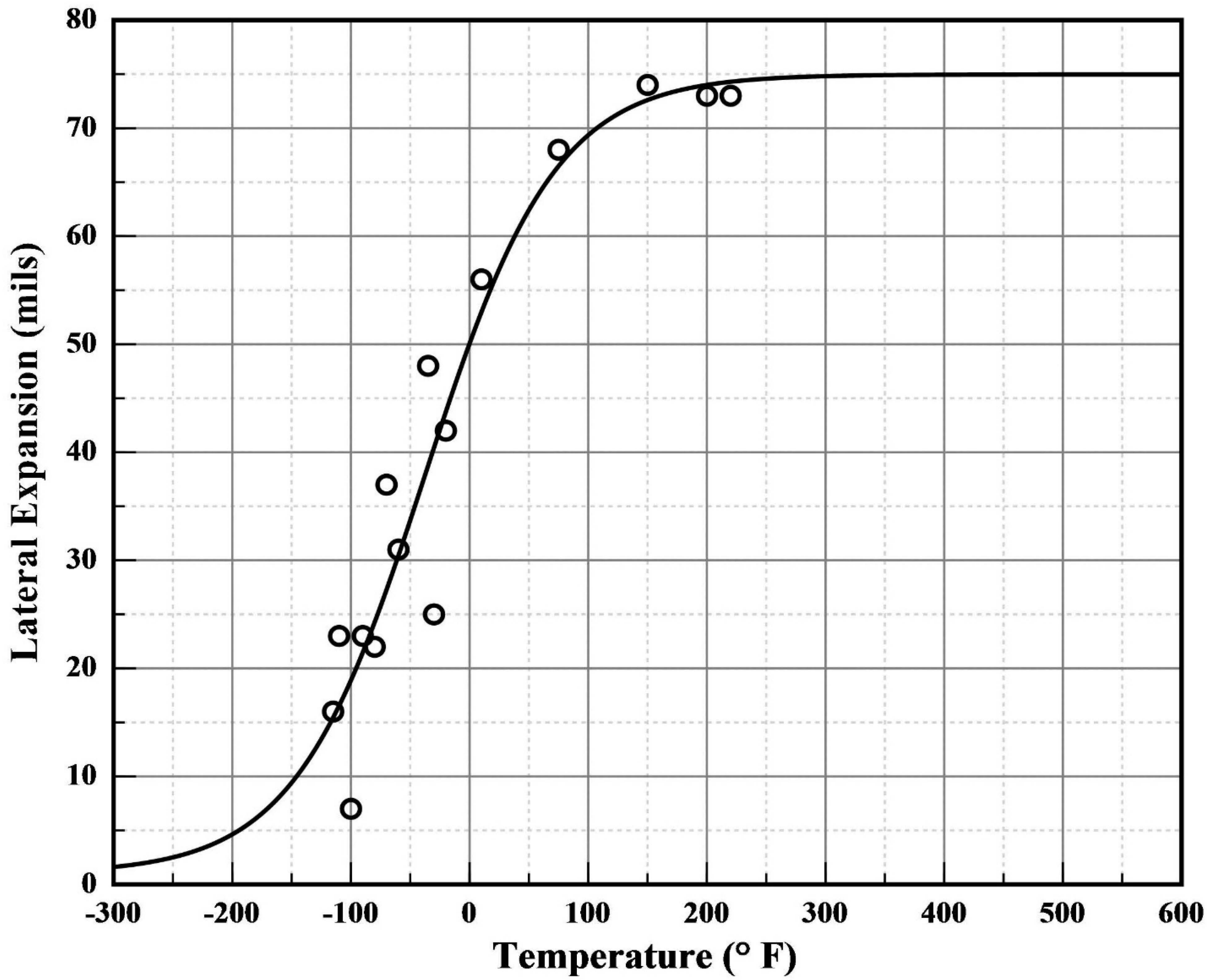
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 74.97 Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils=-46.10° F

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Heat-Affected Zone (HAZ)
Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-115	16.0	15.5	0.55
-110	23.0	16.5	6.46
-100	7.0	18.9	-11.89
-90	23.0	21.5	1.52
-80	22.0	24.3	-2.28
-70	37.0	27.3	9.72
-60	31.0	30.4	0.56
-35	48.0	38.7	9.25
-30	25.0	40.4	-15.43
-20	42.0	43.8	-1.75
10	56.0	53.0	3.01
75	68.0	66.5	1.51
150	74.0	72.6	1.40
200	73.0	74.0	-1.00
220	73.0	74.3	-1.29

CVGraph 6.02 01/09/2020 Page 2/2

Capsule U Heat-Affected Zone (HAZ)

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 1/9/2020 12:03 PM

A = 50.00 B = 50.00 C = 103.78 T0 = -47.73 D = 0.00

Correlation Coefficient = 0.976

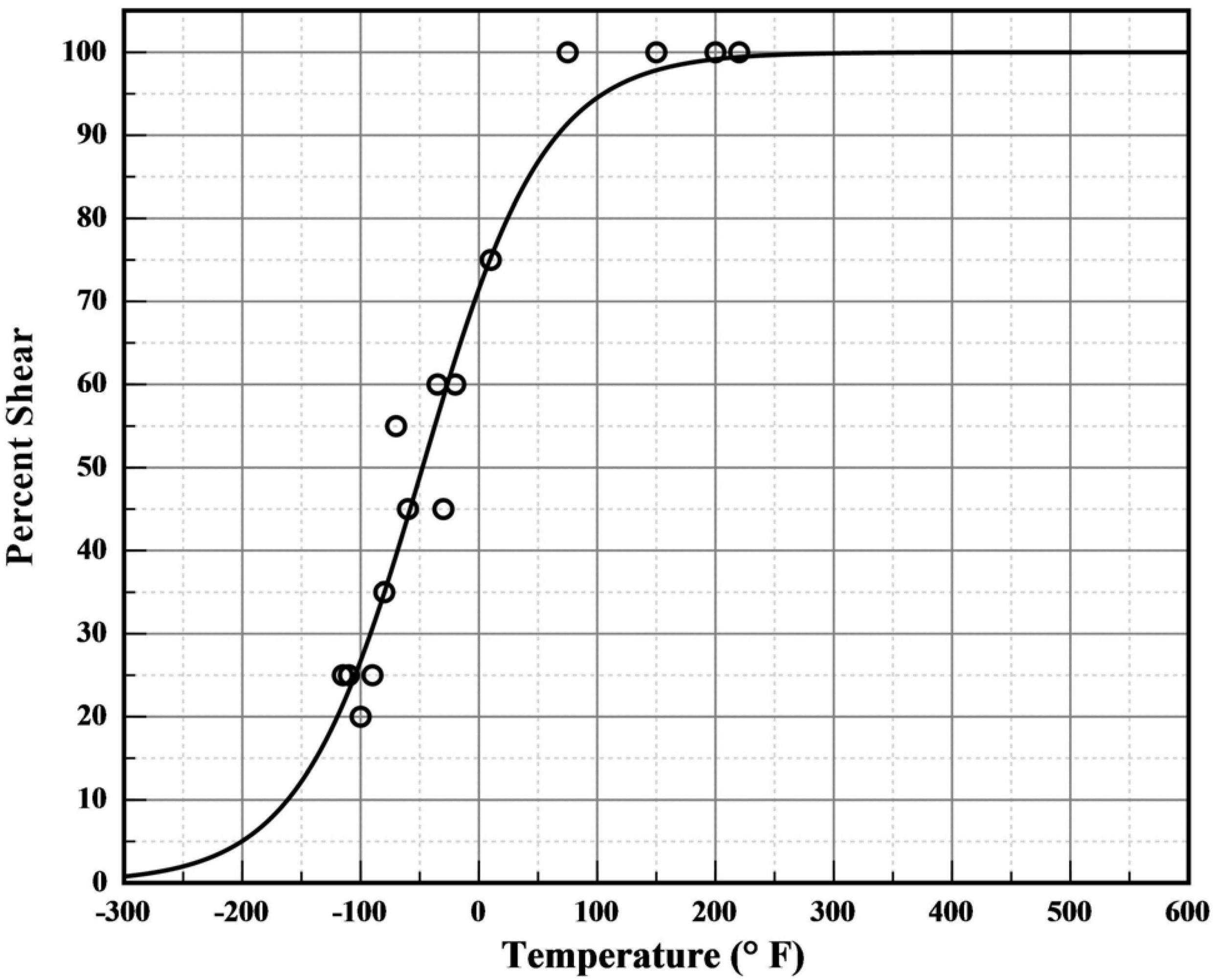
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf %Shear = 100.00 (Fixed) Lower Shelf %Shear = 0.00 (Fixed)

Temperature at 50% Shear = -47.70

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**

Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**



CVGraph 6.02 01/09/2020 Page 1/2

Plant: **Watts Bar 2** Material: **SA508CL2** Heat: **527828**
Orientation: **NA** Capsule: **U** Fluence: **6.04E+018 n/cm²**

Capsule U Heat-Affected Zone (HAZ)

Charpy V-Notch Data

Temperature (° F)	Input %Shear	Computed %Shear	Differential
-115	25.0	21.5	3.52
-110	25.0	23.1	1.85
-100	20.0	26.8	-6.75
-90	25.0	30.7	-5.69
-80	35.0	34.9	0.06
-70	55.0	39.4	15.57
-60	45.0	44.1	0.88
-35	60.0	56.1	3.90
-30	45.0	58.5	-13.46
-20	60.0	63.1	-3.05
10	75.0	75.3	-0.26
75	100.0	91.4	8.59
150	100.0	97.8	2.17
200	100.0	99.2	0.84
220	100.0	99.4	0.57

CVGraph 6.02 01/09/2020 Page 2/2

APPENDIX D WATTS BAR UNIT 2 SURVEILLANCE PROGRAM CREDIBILITY EVALUATION

D.1 INTRODUCTION

Regulatory Guide 1.99, Revision 2 [Ref. D-1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

Capsule U is the first surveillance capsule to be removed and tested from the Watts Bar Unit 2 reactor vessel. In accordance with Regulatory Guide 1.99, Revision 2, the credibility of the surveillance data will be judged based on five criteria. However, criterion 3 requires at least two data sets in order to determine the credibility. Since this is the first capsule withdrawn from Watts Bar Unit 2, this criterion cannot be applied to the surveillance forging. For this reason, the credibility of the surveillance forging cannot be determined due to the limited data available. The surveillance weld Heat # was utilized in the surveillance programs of sister-plants; therefore, criterion 3 can be applied to the surveillance weld with consideration of all available sister-plant data.

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

D.2 EVALUATION

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

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" [Ref. D-2], as follows:

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

The Watts Bar Unit 2 reactor vessel consists of the following beltline region materials:

- Upper Shell Forging 06, Heat # 411572
- Intermediate Shell Forging 05, Heat # 527828
- Lower Shell Forging 04, Heat # 528658
- Bottom Head Ring 03, Heat # 5329

- Upper Shell Forging to Intermediate Shell Forging Circumferential Weld Seam W06 (Weld Wire Heat # 899680 with type Grau L.O. (LW320) flux, lot P23)
- Intermediate Shell Forging 05 to Lower Shell Forging 04 Circumferential Weld Seam W05 (Weld Wire Heat # 895075 with type Grau L.O. (LW320) flux, lot P46)
- Lower Shell Forging 04 to Bottom Head Ring 03 Circumferential Weld Seam W04 (Weld Wire Heat # 899680 with type Grau L.O. (LW320) flux, lot P23)

The Watts Bar Unit 2 surveillance program utilizes tangential and axial test specimens from the Intermediate Shell Forging 05, Heat # 527828. The surveillance weldment is identical to the closing girth seam weldment between forging 04 and 05. The closing seam used weld wire Heat # 895075 with type Grau L.O. (LW320) flux, lot P46, except for the 1-inch root pass at the I.D. of the vessel. This root pass used weld wire Heat # 899680 with type Grau L.O. (LW320) flux, lot P23, with an as-deposited copper and phosphorous content of 0.03 and 0.009, respectively. However, the surveillance weldment specimens were not removed from this root area.

Per WCAP-9455 [Ref. D-3], the Watts Bar Unit 2 surveillance program was developed to the requirements of ASTM E185-73. At the time of the surveillance program development, the Upper Shell Forging 06 and Bottom Head Ring 03 were not considered a "beltline" material. Of the other beltline forgings, Intermediate Shell Forging 05 was foreseen to be the most limiting forging. Intermediate Shell Forging 05 has the highest estimated initial and end of life RT_{NDT} and the lowest initial upper-shelf energy value of the Watts Bar Unit 2 beltline forgings. The chemistry values (Cu and Ni weight percent) for the beltline forgings are relatively consistent and no forging is clearly differentiated from the rest by its high copper or nickel content. Therefore, Intermediate Shell Forging 05 was appropriately selected as the base metal material for the surveillance program.

Intermediate Shell Forging 05 to Lower Shell Forging 04 Circumferential Weld Seam W05 was considered the only weld in the beltline region and therefore, was representative of all the beltline welds. Hence, the surveillance program weld was fabricated with the same weld wire heat (# 895075), the same type flux (LW320), and the same flux lot (# P46) as the Intermediate to Lower Shell Forging Circumferential Weld Seam W05.

Therefore, the materials selected for use in the Watts Bar Unit 2 surveillance program were those judged to be most likely limiting with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed.

<u>Based on the discussion above, Criterion 1 is met for the Watts Bar Unit 2 surveillance program.</u>

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

Based on engineering judgment, the scatter in the data presented in these plots, as documented in Section 5, is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the Watts Bar Unit 2 surveillance materials unambiguously.

Hence, the Watts Bar Unit 2 surveillance program meets Criterion 2.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [Ref. D-4].

This criterion requires at least two data sets in order to determine the credibility. Since this is the first capsule withdrawn from Watts Bar Unit 2, this criterion cannot be applied to the surveillance forging. However, since the surveillance weld Heat # was utilized in the surveillance programs of sister-plants, this criterion can be applied to the surveillance weld.

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

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to the industry at a meeting held by the NRC on February 12 and 13, 1998 [Ref. D-5]. Of the five cases, Case 4 ("Surveillance Data from Plant and Other Sources") most closely represents the situation for the Watts Bar Unit 2 Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075) weld material. Since only one capsule has been tested, an evaluation of the Watts Bar Unit 2 surveillance data alone cannot be completed.

Evaluation of Weld Data from All Sources (Case 4)

In accordance with the NRC Case 4 guidelines, the data from all sources should be adjusted to the mean chemical composition of all the data. Data applicable to the Watts Bar Unit 2 surveillance weld material is also available from the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance programs. Since data are from multiple sources, the data must be adjusted for chemical and irradiation environment differences. The chemistry adjustment ratios are shown below.

Watts Bar Unit 2 Heat # 895075 surveillance weld metal

Cu Wt. % = 0.033, Ni Wt. % = 0.70 (from Table D-1) results in a Regulatory Guide, Position 1.1 Chemistry Factor (CF) = 44.9°F

Catawba Unit 1 Heat # 895075 surveillance weld metal

Cu Wt. % = 0.05, Ni Wt. % = 0.73 (from Table D-1) results in a Regulatory Guide, Position 1.1 CF = 68°F

Watts Bar Unit 1 Heat # 895075 surveillance weld metal

Cu Wt. % = 0.03, Ni Wt. % = 0.75 (from Table D-1) results in a Regulatory Guide, Position 1.1 CF = 41°F

McGuire Unit 2 Heat # 895075 surveillance weld metal

Cu Wt. % = 0.04, Ni Wt. % = 0.74 (from Table D-1) results in a Regulatory Guide, Position 1.1 CF = 54°F

Heat # 895075 surveillance data average composition (considering all available capsules)

The average Cu Wt. % = 0.039 and average Ni Wt. % = 0.74 (from Table D-1) results in a Regulatory Guide, Position 1.1 CF = 52.7°F

The ratio procedure is then applied considering the average chemical composition. The following ratios are applied to the ΔRT_{NDT} in Table D-1:

$$Ratio_{WB2} = CF_{Average} / CF_{WB2\ Surv.\ Weld} = 52.7 / 44.9 = 1.17$$

$$Ratio_{Catawba1} = CF_{Average} / CF_{Catawba1\ Surv.\ Weld} = 52.7 / 68 = 0.78$$

$$Ratio_{WB1} = CF_{Average} / CF_{WB1\ Surv.\ Weld} = 52.7 / 41 = 1.29$$

$$Ratio_{McGuire2} = CF_{Average} / CF_{McGuire2\ Surv.\ Weld} = 52.7 / 54 = 0.98$$

Table D-1 calculates the adjusted ΔRT_{NDT} for weld Heat # 895075 in order to calculate the interim CF for the credibility evaluation.

Table D-1 Mean Chemical Composition and Temperature for Weld Heat # 895075

Material	Capsule	Cu(a) (Wt. %)	Ni(a) (Wt. %)	Chemistry Ratio	Inlet Temp.(b) (°F)	Temp. Adjust.(c) (°F)	Measured ΔRT_{NDT}(d) (°F)	Adjusted ΔRT_{NDT}(e) (°F)
Watts Bar Unit 2 Surveillance Weld (Heat # 895075)	U	0.033	0.70	1.17	559	-0.4	32.6	37.67
Catawba Unit 1 Surveillance Weld (Heat # 895075)	Z	0.05	0.73	0.78	562	2.6	1.91	3.52
	Y	0.05	0.73	0.78	562	2.6	17.79	15.90
	V	0.05	0.73	0.78	562	2.6	26.5	22.70
Watts Bar Unit 1 Surveillance Weld (Heat # 895075)	U	0.03	0.75	1.29	560	0.6	0.0	0.77
	W	0.03	0.75	1.29	560	0.6	30.5	40.12
	X	0.03	0.75	1.29	560	0.6	25.8	34.06
	Z	0.03	0.75	1.29	560	0.6	13.9	18.71
McGuire Unit 2 Surveillance Weld (Heat # 895075)	V	0.04	0.74	0.98	557	-2.4	38.51	35.39
	X	0.04	0.74	0.98	557	-2.4	35.93	32.86
	U	0.04	0.74	0.98	557	-2.4	23.81	20.98
	W	0.04	0.74	0.98	557	-2.4	43.76	40.53
MEAN	---	0.039	0.74	-	559.4		-	-

Notes:

(a) Watts Bar Unit 2 data is the average of the values in Table 4-1. Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 data is taken from WCAP-18191-NP [Ref. D-6].

(b) Watts Bar Unit 2 temperature is determined by averaging (time-weighted) the inlet temperatures for all cycles prior to the capsule being removed. Watts Bar Unit 1 and McGuire Unit 2 data is taken from WCAP-18191-NP. Catawba Unit 1 data is taken from WCAP-17669-NP [Ref. D-7].

(c) Temperature Adjustment = $T_{capsule} - T_{average.}$

(d) Watts Bar Unit 2 data is taken from Section 5. Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 data is taken from WCAP-18191-NP.

(e) Adjusted ΔRT_{NDT} = ($\Delta RT_{NDT, Measured}$ + Temp. Adjustment) x (Chemistry Ratio).

Table D-2 calculates the interim CF for weld Heat # 895075 considering all available data adjusted to account for chemical and irradiation environment differences.

Table D-2 Heat # 895075 Interim Chemistry Factor Using All Available Surveillance Data

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	Adjusted ΔRT_{NDT}[(a)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Watts Bar Unit 2 Surveillance Weld (Heat # 895075)	U	0.604	0.859	37.67	32.35	0.738
Catawba Unit 1 Surveillance Weld (Heat # 895075)	Z	0.286	0.658	3.52	2.31	0.433
	Y	1.290	1.071	15.90	17.03	1.147
	V	2.270	1.222	22.70	27.73	1.493
Watts Bar Unit 1 Surveillance Weld (Heat # 895075)	U	0.447	0.776	0.77	0.60	0.602
	W	1.080	1.022	40.12	40.98	1.044
	X	1.710	1.148	34.06	39.08	1.317
	Z	2.40	1.236	18.71	23.12	1.528
McGuire Unit 2 Surveillance Weld (Heat # 895075)	V	0.302	0.672	35.39	23.78	0.452
	X	1.380	1.089	32.86	35.80	1.187
	U	1.900	1.176	20.98	24.67	1.382
	W	2.82	1.276	40.53	51.71	1.628
				SUM:	319.18	12.949
	$CF_{Surv. Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (319.18) \div (12.949) =$ **24.6°F**					

Notes:

(a) Fluence taken from Section 6.0 for Watts Bar Unit 2 and WCAP-18191-NP [Ref. D-6] for Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2. Adjusted ΔRT_{NDT} taken from Table D-1.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-3.

Table D-3 Heat # 895075 Surveillance Capsule Data Scatter about the Best-Fit Line Using All Available Surveillance Data

Material	Capsule	CF[(a)] ($Slope_{best-fit}$) (°F)	Capsule Fluence[(b)] (x 10^{19} n/cm^2)	FF[(c)]	Adjusted ΔRT_{NDT}[(d)] (°F)	Predicted ΔRT_{NDT}[(e)] (°F)	Scatter ΔRT_{NDT}[(f)] (°F)	<28°F (Weld)
Watts Bar Unit 2 Surveillance Weld (Heat # 895075)	U	24.6	0.604	0.859	37.7	21.1	16.5	Yes
Catawba Unit 1 Surveillance Weld (Heat # 895075)	Z	24.6	0.286	0.658	3.5	16.2	12.7	Yes
	Y	24.6	1.29	1.071	15.9	26.3	10.4	Yes
	V	24.6	2.27	1.222	22.7	30.1	7.4	Yes
Watts Bar Unit 1 Surveillance Weld (Heat # 895075)	U	24.6	0.447	0.776	0.8	19.1	18.3	Yes
	W	24.6	1.08	1.022	40.1	25.1	15.0	Yes
	X	24.6	1.71	1.148	34.1	28.2	5.8	Yes
	Z	24.6	2.40	1.236	18.7	30.4	11.7	Yes
McGuire Unit 2 Surveillance Weld (Heat # 895075)	V	24.6	0.302	0.672	35.4	16.5	18.9	Yes
	X	24.6	1.38	1.089	32.9	26.8	6.1	Yes
	U	24.6	1.90	1.176	21.0	28.9	7.9	Yes
	W	24.6	2.82	1.276	40.5	31.4	9.1	Yes

Notes:

(a) CF calculated in Table D-2.

(b) Fluence taken from Section 6 for Watts Bar Unit 2 and WCAP-18191-NP [Ref. D-6] for Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2.

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(d) Adjusted ΔRT_{NDT} taken from Table D-1.

(e) Predicted ΔRT_{NDT} = CF x FF.

(f) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Adjusted ΔRT_{NDT}].

Table D-3 indicates that 12 of the 12 surveillance data points fall inside the +/- 1σ of 28°F scatter band for surveillance weld materials. 100% of the data are bounded; therefore, the surveillance weld data is deemed "credible" per the third criterion.

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

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

Hence, Criterion 4 is met for the Watts Bar Unit 2 surveillance program.

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

Correlation Monitor Materials (CMM) were included in some plants' reactor vessel surveillance programs in order to provide data to improve the predictive model and confirm Revision 2 of Regulatory Guide 1.99. See NUREG/CR-6413, ORNL/TM-13133 [Ref. D-8]. However, the Watts Bar Unit 2 surveillance program does not contain correlation monitor material. Hence, this criterion is not applicable to the Watts Bar Unit 2 surveillance program.

D.3 CONCLUSION

Based on the preceding responses to the 5 criteria of Regulatory Guide 1.99, Revision 2, Section B, the Watts Bar Unit 2 surveillance weld data for Heat # 895075 are deemed credible. Since only one capsule has been withdrawn and tested containing the Watts Bar Unit 2 surveillance forging material, insufficient data exists to determine the credibility of the surveillance forging material.

D.4 REFERENCES

D-1 U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

D-2 Code of Federal Regulations 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.

D-3 Westinghouse Report WCAP-9455, Revision 4, "Tennessee Valley Authority Watts Bar Unit No. 2 Reactor Vessel Radiation Surveillance Program," August 2019.

D-4 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, 1982.

D-5 K. Wichman, M. Mitchell, and A. Hiser, USNRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, *NRC/Industry Workshop on RPV Integrity Issues*, February 12, 1998. *[ADAMS Accession Number ML110070570]*.

D-6 Westinghouse Report WCAP-18191-NP, Revision 1, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations," February 2020.

D-7 Westinghouse Report WCAP-17669-NP, Revision 1, "Catawba Unit 1 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," October 2015.

D-8 NUREG/CR-6413, ORNL/TM-13133, "Analysis of the Irradiation Data for A302B and A533B Correlation Monitor Materials," April 1996.

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