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Analysis of Capsule W-83 from the Dominion Nuclear Connecticut Millstone Unit 2 Reactor Vessel Radiation Surveillance Program

Westinghouse

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Analysis of Capsule W-83 from the Dominion Nuclear Connecticut Millstone Unit 2 Reactor Vessel Radiation Surveillance Program

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PREFACE

This report has been technically reviewed and verified by:

Reviewer:

Sections 1 through 5, 7, 8, Appendices B, C and D	T. J. Laubham	
Section 6, Appendix A	G. K. Roberts	GK Roberts

EXECUTIVE SUMMARY

The purpose of this report is to document the results of the testing of surveillance Capsule W-83 specimens and dosimeters from the Millstone Unit 2 reactor vessel. Capsule W-83 was removed at 15.3 EFPY and post irradiation mechanical testing of the Capsule W-83 Charpy V-notch and tensile specimens was performed along with a fluence evaluation. The surveillance Capsule W-83 fluence was 1.74×10^{19} n/cm^2 after 15.3 EFPY of plant operation. A brief summary of the Charpy V-notch testing results can be found in Section 1 and the updated capsule removal schedule can be found in Section 7. A supplement to this report is a credibility evaluation, which can be found in Appendix D, which shows the Millstone Unit 2 surveillance plate data is not credible, however the weld data is credible.

1 SUMMARY OF RESULTS

The analysis of the reactor vessel materials contained in surveillance Capsule W-83, the third capsule removed from the Millstone Unit 2 reactor pressure vessel, resulted in the following conclusions:

- Capsule W-83 received an average fast neutron fluence (E>1.0 MeV) of 1.74 x10^{19} n/cm^2 after 15.3 effective full power years (EFPY) of plant operation.

- Irradiation of the reactor vessel lower shell plate C-506-1 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation), to 1.74 x10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 119.12°F and a 50 ft-lb transition temperature increase of 140.1°F. This results in an irradiated 30 ft-lb transition temperature of 163.34°F and an irradiated 50 ft-lb transition temperature of 204.45°F for the longitudinally oriented specimens.

- Irradiation of the reactor vessel lower shell plate C-506-1 Charpy specimens, oriented with the longitudinal axis of the specimen normal to the major working direction of the plate (transverse orientation), to 1.74 x10^{19} n/cm^2 (E> 1.0 MeV) resulted in a 30 ft-lb transition temperature increase of 145.78°F and a 50 ft-lb transition temperature increase of 152.93°F. This results in an irradiated 30 ft-lb transition temperature of 164.18°F and an irradiated 50 ft-lb transition temperature of 205.13°F for transversely oriented specimens.

- Irradiation of the weld metal Charpy specimens to 1.74 x10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 56.09°F and a 50 ft-lb transition temperature increase of 76.96°F. This results in an irradiated 30 ft-lb transition temperature of 23.94°F and an irradiated 50 ft-lb transition temperature of 64.7°F.

- Irradiation of the weld Heat-Affected-Zone (HAZ) metal Charpy specimens to 1.74 x10^{19} n/cm^2 (E> 1.0 MeV) resulted in a 50 ft-lb transition temperature increase of 43.18°F and a 50 ft-lb transition temperature increase of 52.58°F. This results in an irradiated 30 ft-lb transition temperature of 31.66°F and an irradiated 50 ft-lb transition temperature of 61.34°F.

- The average upper shelf energy of the lower shell plate C-506-1 (longitudinal orientation) resulted in an average energy decrease of 39 ft-lb after irradiation to 1.74 x10^{19} n/cm^2 (E> 1.0 MeV). This results in an irradiated average upper shelf energy of 92 ft-lb for the longitudinally oriented specimens.

- The average upper shelf energy of the lower shell plate C-506-1 (transverse orientation) resulted in an average energy decrease of 24 ft-lb after irradiation to 1.74 x10^{19} n/cm^2 (E> 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 84 ft-lb for the transversely oriented specimens.

- The average upper shelf energy of the weld metal Charpy specimens resulted an average energy decrease of 23 ft-lb after irradiation to 1.74 x10^{19} n/cm^2 (E> 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 109 ft-lb for the weld metal specimens.

- The average upper shelf energy of the weld HAZ metal Charpy specimens resulted in an average energy decrease of 26 ft-lb after irradiation to 1.74 x10^{19} n/cm^2 (E> 1.0MeV). This results in an irradiated average upper shelf energy of 103 ft-lb for the weld HAZ metal.

- A comparison of the Millstone Unit 2 reactor vessel beltline material test results with the Regulatory Guide 1.99, Revision 2[13] predictions (See Table 5-10) led to the following conclusions:

  - The measured 30 ft-lb shift in transition temperature of the Lower Shell Plate C-506-1 contained in Capsule 83 Longitudinal Orientation is less than the Regulatory Guide 1.99, Rev 2 predictions.

  - The measured 30 ft-lb shift in transition temperature of the Lower Shell Plate C-506-1 contained in Capsule 83 Transverse Orientation is greater than the Regulatory Guide 1.99, Rev 2 predictions. However, the shift value is within two sigma of the predicted value.

  - The measured 30 ft-lb shift in transition temperature of the Weld Metal contained in Capsule 83 is less than the Regulatory Guide 1.99, Rev 2 predictions.

  - The measured percent decrease in upper shelf energy (USE) of the Lower Shell Plate C-506-1 contained in Capsule 83 Longitudinal Orientation are in good agreement with the Regulatory Guide 1.99, Revision 2, predictions.

  - The measured percent decrease in upper shelf energy (USE) of the Lower Shell Plate C-506-1 contained in Capsule 83 Transverse Orientation and Weld Metal are less than the Regulatory Guide 1.99, Revision 2, predictions.

- The surveillance capsule materials exhibit, adequate upper shelf energy level for continued safe plant operation and are expected to maintain an upper shelf energy of greater than 50 ft-lb throughout the life (32 EFPY) and life extension (48 EFPY) of the vessel as required by 10CFR50, Appendix G.

- The peak calculated end-of-license (32 EFPY) and end-of-license renewal (48 EFPY) neutron fluence (E> 1.0 MeV) at the core midplane for the Millstone Unit 2 reactor vessel using the Regulatory Guide 1.99, Revision 2 attenuation formula (ie. Equation # 3 in the guide; $f_{(depth\ x)} = f_{surface} * e^{(-0.24x)}$) is as follows:

Calculated (32 EFPY):	Vessel inner radius* = 2.40×10^{19} n/cm^2
	Vessel 1/4 thickness = 1.43×10^{19} n/cm^2
	Vessel 3/4 thickness = 5.08×10^{18} n/cm^2
Calculated (48 EFPY):	Vessel inner radius* = 3.44×10^{19} n/cm^2
	Vessel 1/4 thickness = 2.05×10^{19} n/cm^2
	Vessel 3/4 thickness = 7.28×10^{18} n/cm^2

*Clad/base metal interface

2 INTRODUCTION

This report presents the results of the examination of Capsule W-83, the third capsule removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Dominion Nuclear Connecticut Millstone Unit 2 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Millstone Unit 2 reactor pressure vessel materials was designed and recommended by Combustion Engineering. A description of the surveillance program and the preirradiation mechanical properties of the reactor vessel materials is presented in CENPD-53, entitled "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Millstone Point – Unit 2 Reactor Vessel Materials"[1]. The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-70, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels"[14]. Capsule W-83 was removed from the reactor after 15.3 EFPY of exposure and was shipped to the Westinghouse Science and Technology Center Hot Cell Facility, where the postirradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

This report summarizes the analysis of the post-irradiation data obtained from surveillance Capsule W-83 removed from the Dominion Nuclear Connecticut Millstone Unit 2 reactor vessel and discusses the re-analysis of the data. The data is also compared to capsules W-97[2] and W-104[3] which were previously removed from the reactor.

3 BACKGROUND

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy, ferritic pressure vessel steels such as SA533 Grade B Class 1 (base material of the Millstone Unit 2 reactor pressure vessel shell plate) 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 under certain conditions of irradiation.

A method for ensuring the integrity of reactor pressure vessels has been presented in "Fracture Toughness Criteria for Protection Against Failure," Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code[4]. The method uses fracture mechanics concepts and is based on the reference nil-ductility transition temperature (RT_{NDT}).

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT per ASTM E-208[5]) or the temperature 60°F less than the 50 ft-lb (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working direction of the plate. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{Ic} curve) which appears in Appendix G to the ASME Code. The K_{Ic} curve is a lower bound of static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{Ic} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined using these allowable stress intensity factors.

RT_{NDT} and, in turn, the operating limits of nuclear power plants can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement changes in mechanical properties of a given reactor pressure vessel steel can be monitored by a reactor surveillance program, such as the Millstone Unit 2 reactor vessel radiation surveillance program[1], in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the initial RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This RT_{NDT} (RT_{NDT} initial + 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 Millstone Unit 2 reactor pressure vessel core region material were inserted in the reactor vessel prior to initial plant startup. The capsules were positioned in the reactor vessel between the thermal shield and the vessel wall at locations shown in Figure 4-1. Since time of installation, the thermal shield has been removed. The vertical center of the capsules is opposite the vertical center of the core. The capsules contain specimens made from Lower Shell Plate C-506-1 (Heat No. C-5667-1), weld metal fabricated with Mil B-4 weld filler wire heat numbers 10137 and 90136 Linde 0091, which is identical to that used in the actual fabrication of the intermediate to lower shell plates circumferential beltline weld.

The surveillance heat-affected-zone (HAZ) material was fabricated by welding together sections of plate C-506-1 and C-506-3 in the same manner as the surveillance weld material with the same post weld heat-treatment. Standard Reference Material (SRM) was included in the Millstone Unit 2 Surveillance Program but not included within Capsule W-83.

The chemistry and heat treatment of the surveillance material are presented in Table 4-1 and Table 4-2, respectively. The chemical analyses reported in table 4-1 were obtained from unirradiated material used in the surveillance program.

All test specimens were machined from the ¼ thickness location. Test specimens represent material taken at least one plate thickness from the quenched end of the plate. All base material Charpy V-notch impact and tensile specimens were oriented with the longitudinal axis of the specimen both normal to (transverse orientation) and parallel to (longitudinal orientation) the principal working direction of the plate. Charpy V-notch specimens from the weld metal were oriented with the longitudinal axis of the specimens transverse to the weld direction. Tensile specimens were oriented with the longitudinal axis of the specimens normal to the welding direction.

Capsule W-83 contained dosimeters of Sulfur, Iron, Copper (shielded), Nickel (shielded) and Cobalt (shielded and unshielded).

Thermal monitors were made from two low-melting eutectic alloys and sealed in Pyrex tubes that were included in the capsule and were located as shown in Figure 4-2. The two eutectic alloys and their melting points are:

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

The arrangement of the various mechanical test specimens, dosimeters and thermal monitors contained in Capsule W-83 are shown in Figure 4-2. A typical Millstone Unit 2 surveillance capsule Charpy impact compartment assembly is shown in Figure 4-3. A typical surveillance capsule tensile and flux-monitor compartment assembly is shown in Figure 4-4.

Table 4-1 Chemical Composition of the Millstone Unit 2 Reactor Vessel Surveillance Materials [2]			
Element	Plate C-506-1 (Heat C-5567-1)	¼ T – ID Weld C-506-2 /C-506-3[a]	¼ T – OD Weld C-506-2/C-506-3[b]
Si	0.12	0.17	0.15
S	0.014	0.013	0.013
P	0.006	0.015	0.016
Mn	1.26	1.13	1.13
C	0.21	0.12	0 12
Cr	0.10	0.04	0.05
Ni	0.61	0.06	0.06
Mo	0.62	0.54	0.53
V	0.004	0.006	0.007
Cb	<0.01	<0 01	<0 01
B	0.0006	0.0003	0.0003
Co	0.011	0.009	0.009
Cu	0 14	0.30	0.21
Al	0.020	<0 001	<0.001
W	<0.01	0 01	<0 01
Ti	<0.01	<0.01	<0.01
As	0.011	0.011	0.012
Sn	0.009	0.004	0.003
Zr	0.002	0.002	0.002
N_2	0.009	0.008	0 009

Notes:

a) Mil B-4 wire heat 90136, Linde 0091 Flux Lot 3998

b) Mil B-4 wire heat 10137, Linde 0091 Flux Lot 3999

Table 4-2 Heat Treatment of the Millstone Unit 2 Reactor Vessel Surveillance Materials

Material	Temperature (°F)	Time (hr)	Coolant
Intermediate Shell Plate C-506-1	Austenitizing: 1600 ± 50	4	Water quenched
	Tempering: 1225 ± 25	4	Furnace Cooled
	Stress Relief : 1150 ± 25	40	Furnace Cooled to 600°F
Weldment	Post Weld Stress Relief: 1150 ± 25	40	Furnace Cooled to 600°F

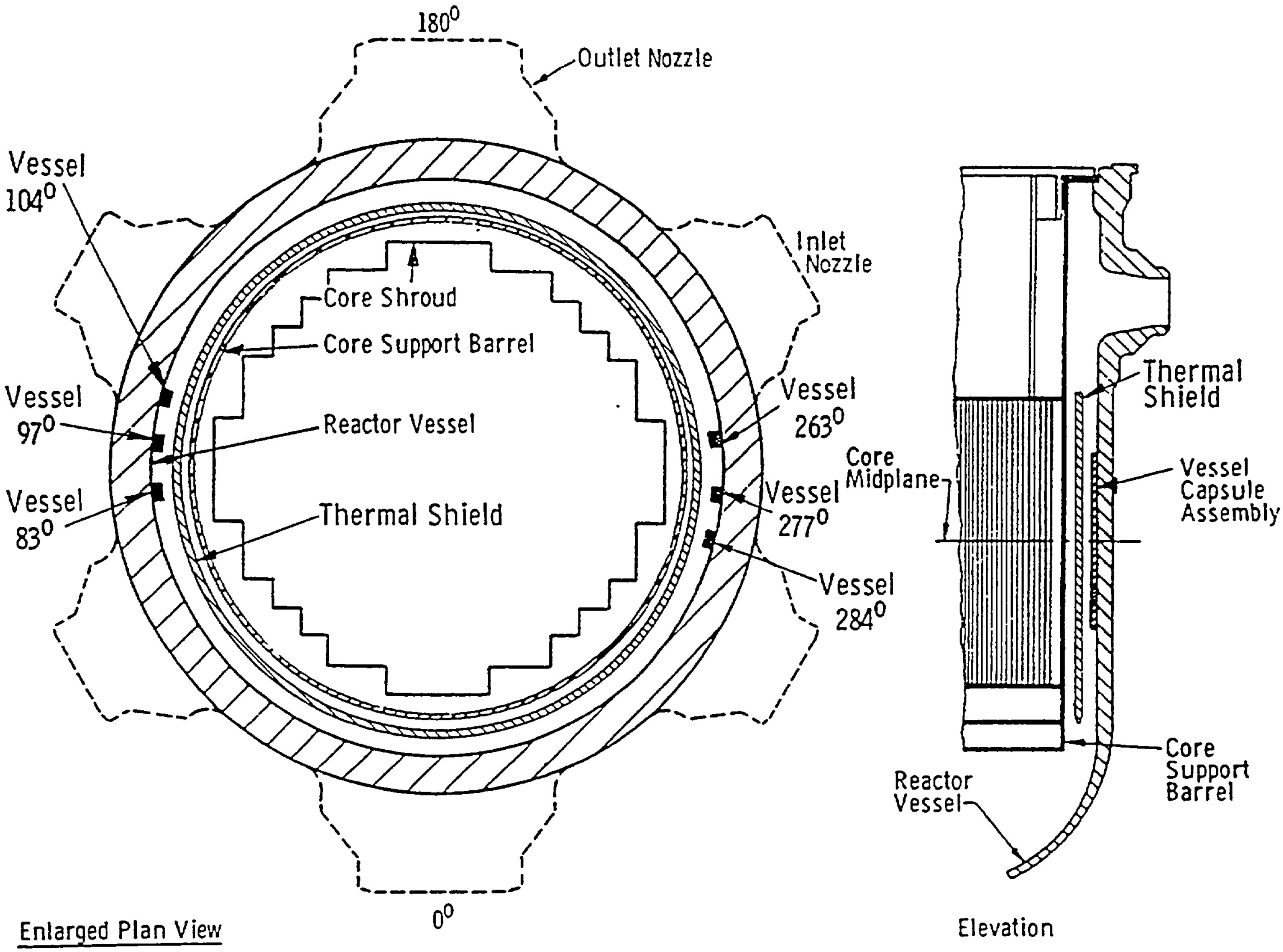

Figure 4-1 Original Arrangement of Surveillance Capsules in the Millstone Unit 2 Reactor Vessel

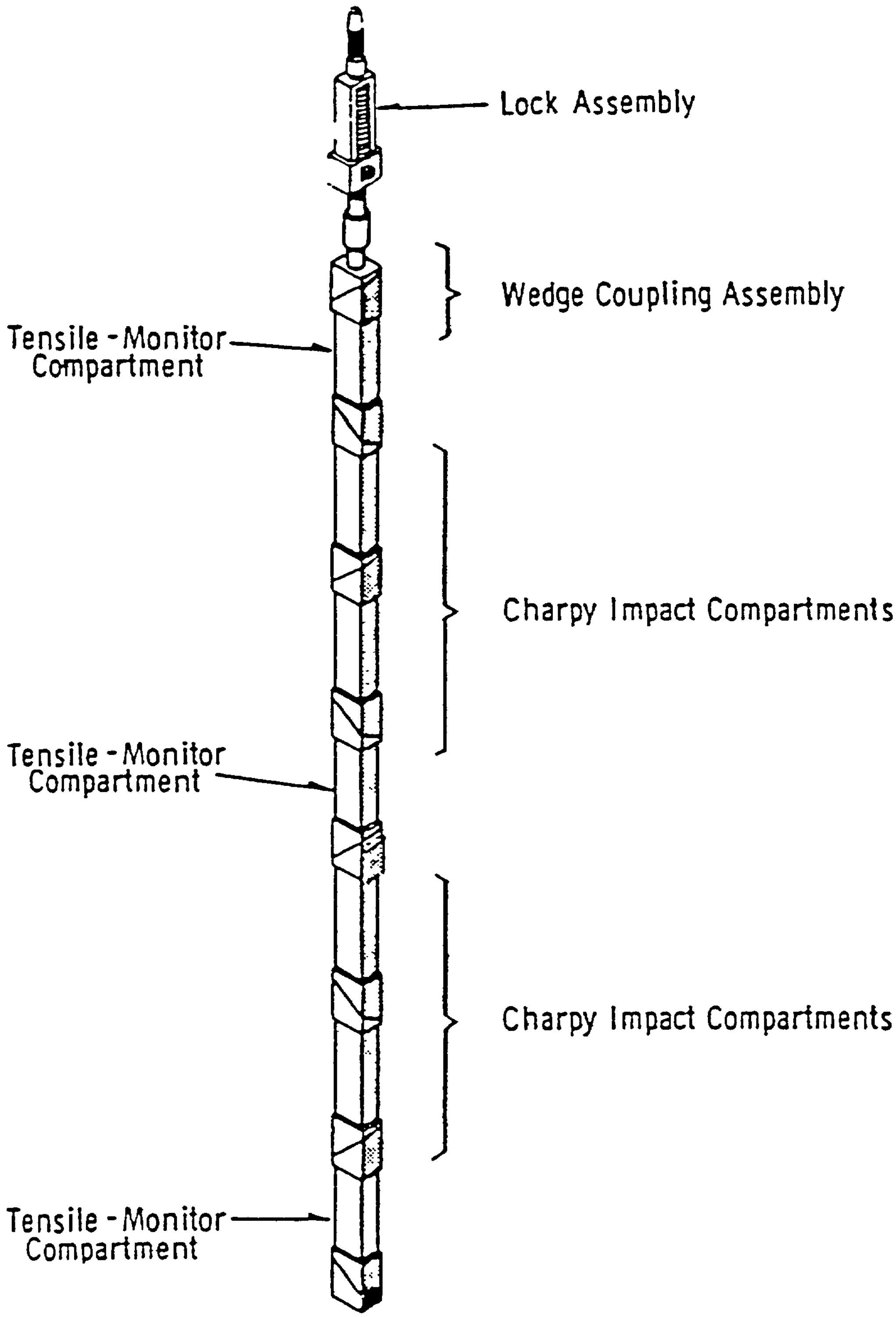

Figure 4-2 Typical Millstone Unit 2 Surveillance Capsule Assembly

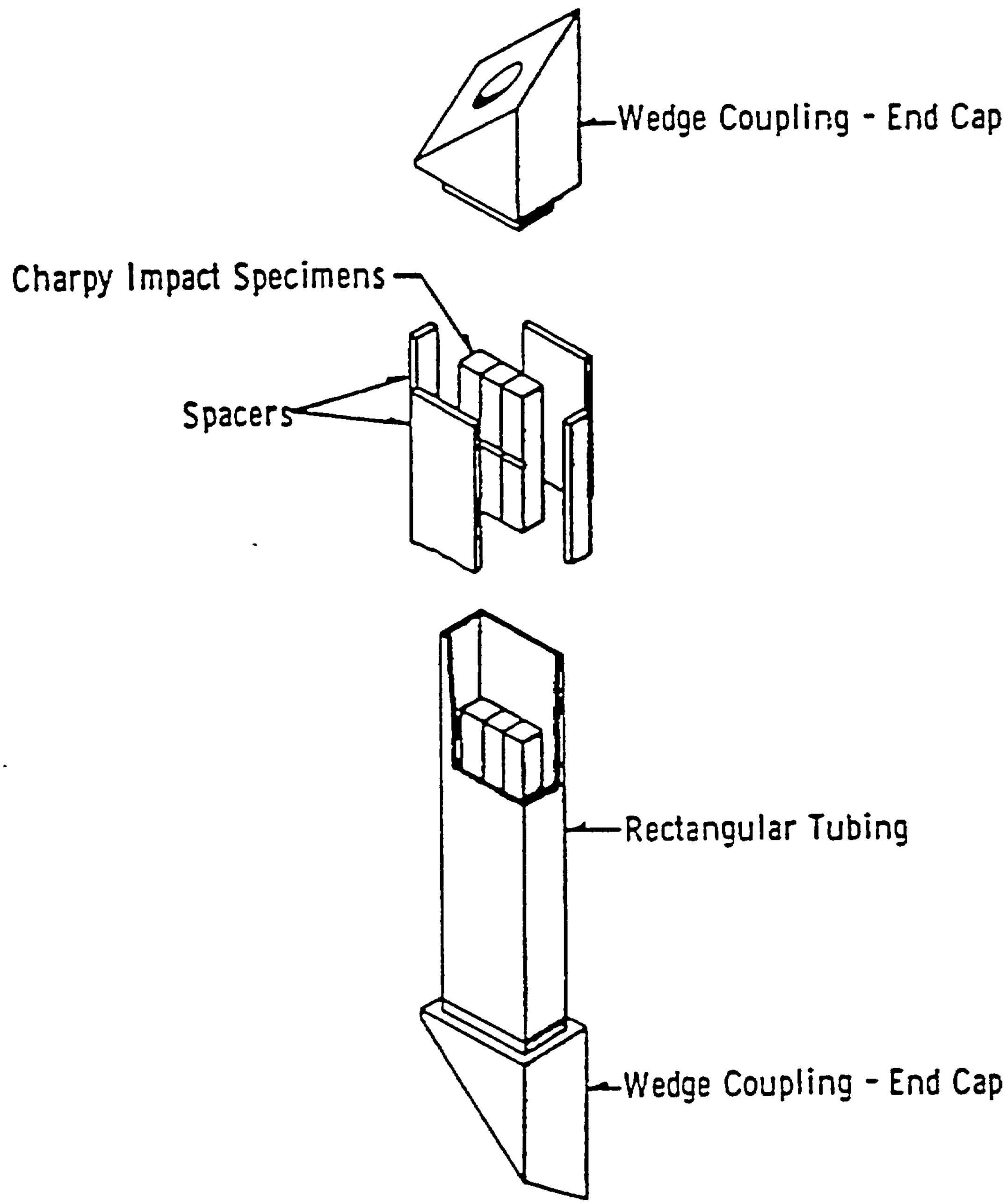

Figure 4-3 Typical Millstone Unit 2 Surveillance Capsule Charpy Impact Compartment Assembly

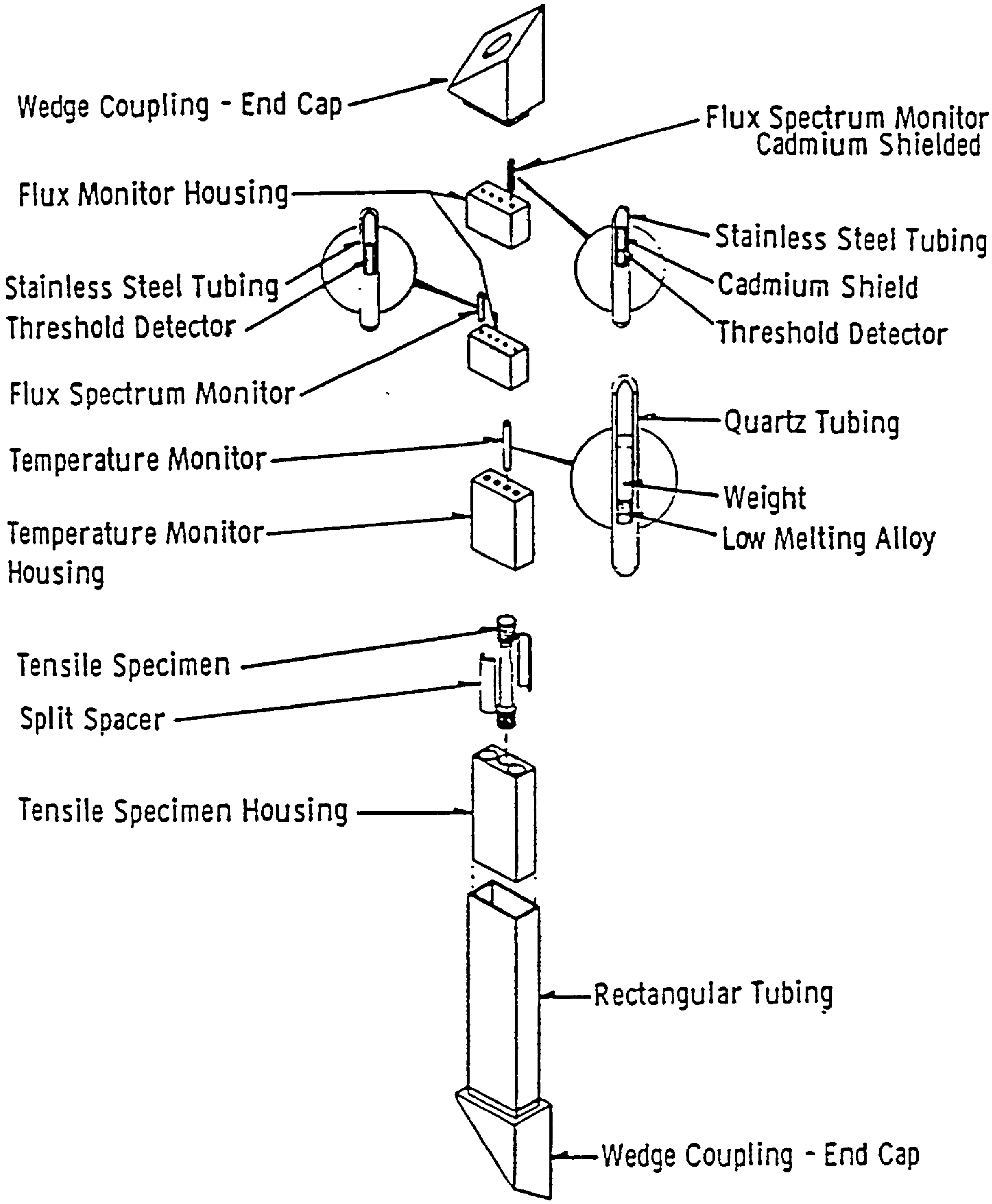

Figure 4-4 Typical Millstone Unit 2 Tensile and Flux-Monitor Compartment Assembly

Table 4-3 Millstone Unit 2 Chemistry and Fluence Values

Beltline Region Location	Fabrication Material Code / Heat No.	Inside Surface Fluence x 10^{19} n/cm^2 EFPY (b)			Chemical Composition(c)		Chemistry Factor (a)	Initial RT_{NDT}
		32	48	54	Cu %	Ni %		
Intermediate Shell Plate SA533, Gr B	C-505-1 / C5843-1	2.40	3.44	3.83	0.13	0.61	91.3	8.1
	C-505-2 / C5843-2	2.40	3.44	3.83	0.13	0.62	91.5	17.5
	C-505-3 / C5843-3	2.40	3.44	3.83	0.13	0.62	91.5	5.0
Lower Shell SA5433, Gr B	C-506-1 / C5667-1	2.38	3.40	3.78	0.15	0.60	110	7.0
	C-506-2 / C5667-2	2.38	3.40	3.78	0.15	0.61	110	-33.7
	C-506-3 / C5667-3	2.38	3.40	3.78	0.14	0.66	101.5	-19.2
Mid. Circumferential Weld Linde 0091	9-203 / 10137 - 3999	2.38	3.40	3.78	0.22	0.04	100	-56.3
	9-203 / 10136 - 3998	2.38	3.40	3.78	0.27	0.07	124.3	-56.3
Intermediate Longitudinal Welds Linde 124	2-203-A / A8746 - 3878	2.40	3.44	3.83	0.15	0.13	77.7	-56.0
Intermediate Longitudinal Welds Linde 124	2-203-B,-C / A8746 - 3878	1.56	2.27	2.53	0.15	0.13	77.7	-56.0
Lower Longitudinal Welds Linde 124	3-203-A / A8746 - 3878	2.38	3.40	3.83	0.15	0.13	77.7	-56.0
Lower Longitudinal Welds Linde 124	3-203-B,-C / A8746 - 3878	1.54	2.24	2.50	0.15	0.13	77.7	-56.0

Notes:

(a) RT_{NDT} calculated per Regulatory Guide 1.99, Revision 2

(b) See Section 6, Table 6-4

(c) See Reference 21

5 TESTING OF SPECIMENS FROM CAPSULE W-83

5.1 OVERVIEW

The post-irradiation mechanical testing of the Charpy V-notch and tensile specimens was performed at the Westinghouse Science and Technology Department Laboratory with consultation by Westinghouse Energy Systems personnel. Testing was performed in accordance with 10CFR50, Appendices G and H[6], ASTM Specification E185-82[7], and Westinghouse Remote Metallographic Facility (RMF) Procedure RMF 8402, Revision 2 as modified by Westinghouse RMF Procedures 8102, Revision 1 and 8103, Revision 1.

Upon receipt of the capsule at the hot cell laboratory, the capsule was visually examined and photographed for identification purposes. The specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in CENDP-53[1]. No discrepancies were found.

Examination of the four low-melting point eutectic alloys indicated that the 1 inch and 1.25 inch monitors melted. Based on this examination, the maximum temperature to which the test specimens were exposed was less than 304°C (580°F).

The Charpy impact tests were performed per ASTM Specification E23-98[8] and RMF Procedure 8103, Revision 1, on a Tinius-Olsen Model 74, 358J machine. The tup (striker) of the Charpy impact test machine is instrumented with an Instron Dynatup Impulse instrumentation system, feeding into an IBM compatible computer. With this system, load-time and energy-time signals can be recorded in addition to the standard measurement of Charpy energy (E_D). From the load-time curve (Appendix A), the load of general yielding (P_{GY}), the time to general yielding (t_{GY}), the maximum load (P_M), and the time to maximum load (t_M) can be determined. Under some test conditions, a sharp drop in load indicative of fast fracture was observed. The load at which fast fracture was initiated is identified as the fast fracture load (P_F), and the load at which fast fracture terminated is identified as the arrest load (P_A).

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

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

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

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

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

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

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

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

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

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

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

Percent shear was determined from post-fracture photographs using the ratio-of-areas methods in compliance with ASTM Specification A370-97a[9] . The lateral expansion was measured using a dial gage rig similar to that shown in the same specification.

Tensile tests were performed on a 20,000-pound Instron, split-console test machine (Model 1115) per ASTM Specification E8-99[10] and E21-92[11], and RMF Procedure 8102, Revision 1 All pull rods, grips, and pins were made of Inconel 718. The upper pull rod was connected through a universal joint to improve axiality of loading. The tests were conducted at a constant crosshead speed of 0 05 inches per minute throughout the test.

Extension measurements were made with a linear variable displacement transducer (LVDT) extensometer. The extensometer knife edges were spring-loaded to the specimen and operated through specimen failure. The extensometer gage length was 1 00 inch. The extensometer is rated Class B-2 per ASTM E-83-93[15].

Elevated test temperatures were obtained with a three-zone electric resistance split-tube furnace with a 9-inch hot zone. All tests were conducted in air.

Because of the difficulty in remotely attaching a thermocouple directly to the specimen, the following procedure was used to monitor specimen temperatures. Chromel-Alumel thermocouples were inserted in shallow holes in the center and each end of the gage section of a dummy specimen and in each grip. In test configuration, with a slight load on the specimen, a plot of specimen temperature versus upper and lower grip and controller temperatures was developed over the range room temperature to 550°F (288°C). The upper grip was used to control the furnace temperature. During the actual testing, the grip temperatures were used to obtain desired specimen temperatures. Experiments have indicated that this method is accurate to $\pm$2°F.

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

5.2 CHARPY V-NOTCH IMPACT TEST RESULTS

The results of the Charpy V-notch impact tests performed on the various materials contained in Capsule W-83 irradiated to approximately 1.74 x10^{19} n/cm^2 after 15.3 EFPY are presented in Tables 5-1 through 5-8 and Figures 5-1 through 5-12. The transition temperature increases and upper shelf energy decreases for the Capsule W-83 material are shown in Table 5-10.

- Irradiation of the reactor vessel lower shell plate C-506-1 Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation), to 1.74 x10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 119.12°F and a 50 ft-lb transition temperature increase of 140.1°F. This results in an irradiated 30 ft-lb transition temperature of 163.34°F and an irradiated 50 ft-lb transition temperature of 204.45°F for the longitudinally oriented specimens.

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- Irradiation of the weld metal Charpy specimens to 1.74 x10^{19} n/cm^2 (E> 1.0MeV) resulted in a 30 ft-lb transition temperature increase of 56.09°F and a 50 ft-lb transition temperature increase of 76.96°F. This results in an irradiated 30 ft-lb transition temperature of 23.94°F and an irradiated 50 ft-lb transition temperature of 64.7°F.

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- The average upper shelf energy of the weld metal Charpy specimens resulted an average energy decrease of 23 ft-lb after irradiation to 1.74 x10^{19} n/cm^2 (E> 1.0 MeV). Hence, this results in an irradiated average upper shelf energy of 109 ft-lb for the weld metal specimens.

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- A comparison of the Millstone Unit 2 reactor vessel beltline material test results with the Regulatory Guide 1.99, Revision 2[13] predictions (See Table 5-10) led to the following conclusions:

  - The measured 30 ft-lb shift in transition temperature of the Lower Shell Plate C-506-1 contained in Capsule 83 Longitudinal Orientation is less than the Regulatory Guide 1.99, Rev 2 predictions.

  - The measured 30 ft-lb shift in transition temperature of the Lower Shell Plate C-506-1 contained in Capsule 83 Transverse Orientation is greater than the Regulatory Guide 1.99, Rev 2 predictions. However, the shift value is less than two sigma of the predicted value.

  - The measured 30 ft-lb shift in transition temperature of the Weld Metal contained in Capsule 83 is less than the Regulatory Guide 1.99, Rev 2 predictions.

  - The measured percent decrease in upper shelf energy (USE) of the Lower Shell Plate C-506-1 contained in Capsule 83 Longitudinal Orientation are in good agreement with the Regulatory Guide 1.99, Revision 2, predictions.

  - The measured percent decrease in upper shelf energy (USE) of the Lower Shell Plate C-506-1 contained in Capsule 83 Transverse Orientation and Weld Metal are less than the Regulatory Guide 1.99, Revision 2, predictions.

- The surveillance capsule materials exhibit a more than adequate upper shelf energy level for continued safe plant operation and are expected to maintain an upper shelf energy of greater than 50 ft-lb throughout the life (32 EFPY) and life extension (48 EFPY) of the vessel as required by 10CFR50, Appendix G.

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

5.3 TENSILE TEST RESULTS

The results of the tensile tests performed on the various materials contained in Capsule W-83 irradiated to 1.74 x 10^{19} n/cm^2 (E>1.0MeV) are presented in Table 5-11 and are compared with unirradiated results as shown in Figures 5-17, 5-18 and 5-19.

The results of the tensile tests performed on the Lower Shell Plate C-506-1 (Longitudinal Orientation) indicate that irradiation to 1.74 x 10^{19} n/cm^2 (E>1.0 MeV) caused an 16 to 18 ksi increase in 0.2 percent yield strength and approximately a 14 to 20 ksi increase in the ultimate tensile strength when compared to unirradiated data (Figure 5-17).

The results of the tensile tests performed on the surveillance weld indicate that irradiation to 1.74 x 10^{19} n/cm^2 (E>1.0 MeV) caused an 4 to 13 ksi increase in 0.2 percent yield strength and approximately a 14 to 16 ksi increase in the ultimate tensile strength when compared to unirradiated data (Figure 5-18).

The results of the tensile tests performed on the Heat Affected Zone Metal indicate that irradiation to 1.74 x 10^{19} n/cm^2 (E>1.0 MeV) caused an 11 to 16 ksi increase in 0.2 percent yield strength and approximately a 12 to 14 ksi increase in the ultimate tensile strength when compared to unirradiated data (Figure 5-19).

The fractured tension specimens for each of the materials are shown in Figures 5-20, 5-21 and 5-22. A typical stress-strain curve for the tension specimens is shown in Figures 5-22 through 5-28.

Table 5-1 Charpy V-notch Impact Data for the Millstone Unit 2 Reactor Vessel Shell Plate C506-1 Irradiated at 550°F, Fluence 1.74 x10^{19} n/cm^2 (E> 1.0 MeV) Longitudinal Orientation							
Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
132	0	-18	3	4	0	0 00	2
146	75	24	12	16	7	0 18	10
117	130	54	32	43	25	0 64	25
13C	175	79	23	31	20	0 51	40
121	175	79	35	47	32	0 81	35
136	200	93	32	43	28	0 71	35
156	215	102	67	91	52	1.32	65
165	225	107	74	100	58	1.47	85
131	250	121	60	81	48	1.22	60
12K	300	149	80	108	68	1.73	100
126	325	163	95	129	72	1.83	100
16D	350	177	102	138	80	2 03	100

Table 5-2 Charpy V-notch Impact Data for the Millstone Unit 2 Reactor Vessel Shell Plate C506-1 Irradiated at 550°F, Fluence 1.74 x10^{19} n/cm^2 (E> 1.0 MeV) Transverse Orientation							
Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
224	0	-18	5	7	1	0.03	2
231	75	24	15	20	9	0.23	10
21L	130	54	17	23	14	0.36	15
253	150	66	38	52	30	0.76	40
213	150	66	24	33	21	0.53	30
245	175	79	27	37	23	0.58	35
212	175	79	32	43	27	0.69	40
211	200	93	37	50	30	0.76	40
23L	225	107	65	88	52	1.32	70
214	275	135	77	104	60	1.52	100
24K	300	149	83	113	65	1.65	100
22A	325	163	92	125	73	1.85	100

Table 5-3 Charpy V-notch Impact Data for the Millstone Unit 2 Reactor Vessel Weld Data Irradiated at 550°F, Fluence 1.74 x10^{19} n/cm^2 (E> 1.0 MeV)

Sample Number	Temperature		Impact Energy		Lateral Expansion		Shear
	F	C	ft-lbs	Joules	mils	mm	%
314	-50	-46	8	11	1	0.03	10
33K	0	-18	9	12	6	0.15	15
34L	30	-1	27	37	23	0.58	30
311	50	10	45	61	36	0.91	40
32A	75	24	68	92	50	1.27	65
36D	100	38	76	103	54	1.37	80
36E	125	52	83	113	64	1.63	90
34C	150	66	76	103	60	1.52	85
337	200	93	84	114	66	1.68	95
323	225	107	109	148	80	2.03	100
336	250	121	101	137	73	1.85	100
312	250	121	117	159	84	2.13	100

Table 5-4 Charpy V-notch Impact Data for the Millstone Unit 2 Reactor Vessel Heat Affected Zone (HAZ) Metal Data Irradiated at 550°F, Fluence 1.74 x10^{19} n/cm^2 (E> 1.0 MeV)							
Sample	Temperature		Impact Energy		Lateral Expansion		Shear
Number	F	C	ft-lbs	Joules	mils	mm	%
42T	-75	-59	7	9	1	0.03	10
46E	-25	-32	18	24	9	0.23	15
42P	0	-18	13	18	6	0.15	20
41E	25	-4	34	46	21	0.53	30
41T	50	10	49	66	31	0.79	45
42U	75	24	24	33	17	0 43	30
427	100	38	102	138	57	1.45	85
43K	150	66	94	127	56	1.42	80
41U	200	93	122	165	71	1.80	90
46B	250	121	95	129	66	1.68	100
45K	300	149	88	119	60	1.52	100
44C	325	163	126	171	76	1.93	100

Table 5-5 Instrumented Charpy Impact Test Results for Millstone Unit 2 Reactor Vessel Shell Plate C506-1 Longitudinal Orientation

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in²)			Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
			Charpy Ed/A	Maximum Em/A	Prop Ep/A								
132	0	3	24	13	11	1633	0.10	1701	0.12	1691	0	54	56
146	75	12	97	53	44	3179	0.14	3995	0.20	3983	0	106	119
117	130	32	258	187	71	3092	0.14	4038	0.47	4002	373	103	119
13C	175	23	185	58	127	2866	0.13	3679	0.21	3584	927	95	109
121	175	35	282	191	91	2998	0.14	4032	0.48	3986	900	100	117
136	200	32	258	165	93	2941	0.14	3873	0.44	3861	878	98	113
156	215	67	540	286	254	3065	0.14	4122	0.66	4071	2338	102	120
165	225	74	596	213	383	2922	0.14	4048	0.53	3963	2694	97	116
131	250	60	483	215	269	2983	0.14	4098	0.52	4035	2396	99	118
12K	300	80	645	205	439	2858	0.14	3909	0.52	n/a	n/a	95	113
126	325	95	765	278	487	2809	0.14	4015	0.67	n/a	n/a	94	114
16D	350	102	822	280	542	2824	0.14	4020	0.67	n/a	n/a	94	114

Table 5-6 Instrumented Charpy Impact Test Results for Millstone Unit 2 Reactor Vessel Shell Plate C506-1 Transverse Orientation													
Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in²)			Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
			Charpy Ed/A	Maximum Em/A	Prop Ep/A								
224	0	5	40	21	19	2605	0.13	2658	0.14	2653	0	87	88
231	75	15	121	68	53	3139	0.14	4094	0.23	4094	0	105	120
21L	130	17	137	58	79	3048	0.14	3815	0.21	3795	650	102	114
253	150	38	306	172	134	3102	0.14	4124	0 43	4102	1163	103	120
213	150	24	193	115	78	2937	0.14	3745	0.33	3721	762	98	111
245	175	27	218	115	103	2902	0.14	3778	0.33	3751	1251	97	111
212	175	32	258	142	116	2980	0.14	3859	0.39	3825	1046	99	114
211	200	37	298	144	154	2887	0.14	3831	0.39	3822	1372	96	112
23L	225	65	524	213	310	3021	0.15	4149	0.52	3926	1188	101	119
214	275	77	620	200	421	2855	0.14	3993	0.51	n/a	n/a	95	114
24K	300	83	669	210	459	2993	0.15	4105	0.51	n/a	n/a	100	118
22A	325	92	741	212	529	2878	0.15	4074	0.53	n/a	n/a	96	116

Table 5-7 Instrumented Charpy Impact Test Results for Millstone Unit 2 Reactor Vessel Weld Metal

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
			Charpy Ed/A	Maximum Em/A	Prop Ep/A								
314	-50	8	64	34	31	3444	0.16	3444	0.16	3439	0	115	115
33K	0	9	73	37	35	3561	0.15	3634	0.17	3624	0	119	120
34L	30	27	218	64	153	3500	0.15	4240	0.21	4014	793	117	129
311	50	45	363	219	144	3354	0.15	4172	0.51	4075	910	112	125
32A	75	68	548	220	328	3393	0.15	4228	0.51	3634	1394	113	127
36D	100	76	612	221	392	3297	0.14	4177	0.51	3789	1881	110	124
36E	125	83	669	216	453	3205	0.14	4109	0.52	3797	2508	107	122
34C	150	76	612	218	394	3171	0.15	4057	0.53	3832	2236	106	120
337	200	84	677	208	469	3039	0.14	3940	0.51	n/a	n/a	101	116
323	225	109	878	289	589	3053	0.14	3993	0.68	n/a	n/a	102	117
336	250	101	814	287	527	2989	0.14	4001	0.67	n/a	n/a	100	116
312	250	117	943	288	655	3016	0.15	4048	0.67	n/a	n/a	100	118

Table 5-8 Instrumented Charpy Impact Test Results for Millstone Unit 2 Reactor Vessel Heat Affected Zone (HAZ) Metal														
Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in²)			Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
			Charpy Ed/A	Maximum Em/A	Prop Ep/A								
42T	-75	7	56	32	25	3489	0.15	3489	0.15	3470	0	116	116
46E	-25	18	145	73	72	3917	0.16	4608	0 22	4444	0	130	142
42P	0	13	105	58	47	3722	0.15	4343	0.2	4341	0	124	134
41E	25	34	274	167	107	3531	0.14	4396	0.39	4381	976	118	132
41T	50	49	395	248	147	3457	0.16	4544	0 54	4456	1072	115	133
42U	75	24	193	69	124	3440	0.14	4270	0 22	3884	1112	115	128
427	100	102	822	330	492	3472	0.15	4500	0.69	3358	380	116	133
43K	150	94	757	309	448	3353	0.15	4339	0.67	3491	697	112	128
41U	200	122	983	322	661	3308	0.15	4412	0.69	1328	376	110	129
46B	250	95	765	300	466	3157	0.15	4216	0.67	n/a	n/a	105	123
45K	300	88	709	227	482	3001	0.15	4183	0.55	n/a	n/a	100	120
44C	325	126	1015	303	712	3081	0.14	4262	0.68	n/a	n/a	103	122

Table 5-9 The Effect of 550°F Irradiation at 1.74 x10^{19} n/cm^2 (E>1.0 MeV) on the Notch Toughness Properties of the Millstone Unit 2 Reactor Vessel Surveillance Capsule Materials

Material	Average 30 (ft-lb) Transition Temperature (°F)			Average 35 mil Lateral Expansion Temperature (°F)			Average 50 ft-lb Transition Temperature (°F)			Average Energy Absorption at Full Shear (ft-lb)		
	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔE
Plate C-506-1 (Longitudinal)	44.22	163.34	119.12	44.98	191.3	146.32	64.35	204.45	140.1	131	92	-39
Plate C-506-1 (Transverse)	18.4	164.18	145.78	24.64	196.84	172.19	52.2	205.13	152.93	108	84	-24
Weld Metal	-32.15	23.94	56.09	-24.52	55.28	79.8	-12.25	64.7	76.96	132	109	-23
HAZ Metal	-11.52	31.66	43.18	0.72	75.4	74.68	8.75	61.34	52.58	129	103	-26

Note: All unirradiated data presented here was taken from CENPD-53[1].

Table 5-10 Comparison of the Millstone Unit 2 Reactor Vessel Surveillance Capsule Charpy Impact Test Results with Regulatory Guide 1.99 Revision 2 Predictions

Material	Capsule	Calculated Fluence (x 10^{19} n/cm^2)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) (a)	Measured (°F)	Predicted (%) (a)	Measured (%) (b)
Lower Shell Plate C-506-1 Longitudinal (Heat # C-5667-1)	W-97	0.324	75.9	65.75	18	28
	W-104	0.949	108.9	87.67	23	20
	W-83	1.74	126.5	119.12	27	30
Lower Shell Plate C-506-1 Transverse (Heat # C-5667-1)	W-97	0.324	75.9	90.83	18	27
	W-83	1.74	126 5	145.78	27	22
Intermediate to Lower Girth Seam 9-203 (Heat # 10137 & 90136)	W-97	0.324	79.2	65.93	30	24
	W-104	0.949	113.6	52.12	38	19
	W-83	1.74	132	56 09	45	17
Heat Affected Zone Metal	W-97	0 324	--	74.26	--	29
	W-83	1.74	--	43.18	--	20
Standard Reference Material	W-104	0.949	--	133.41	--	35

Notes:

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

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

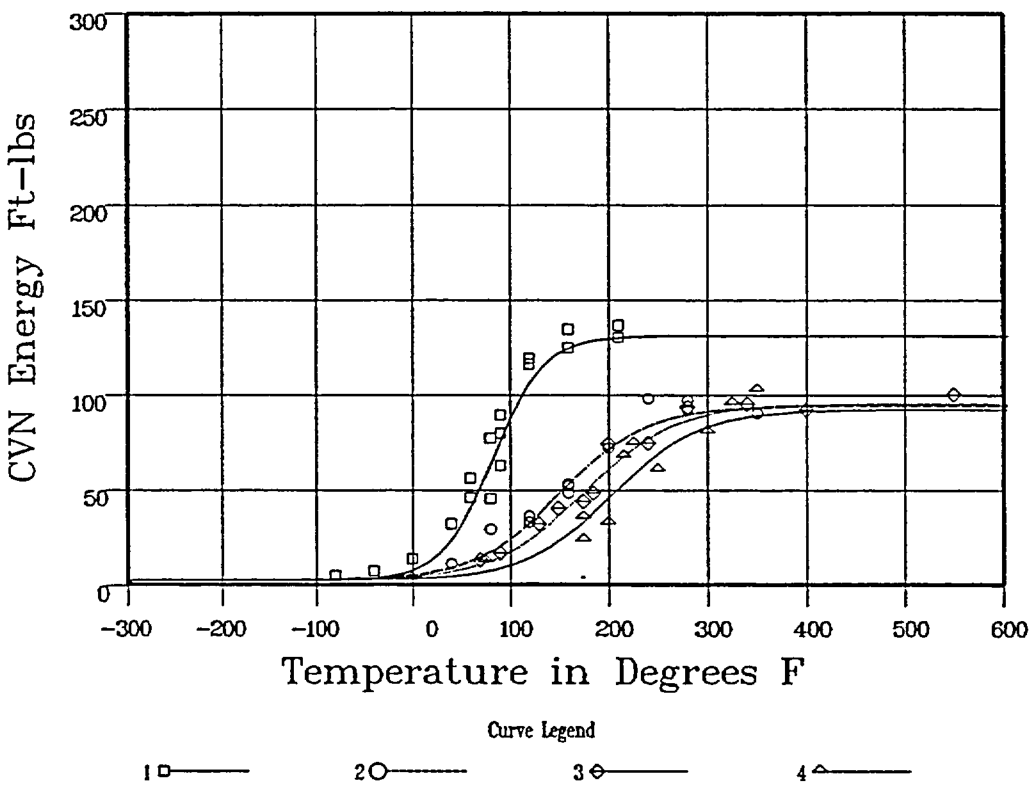
Table 5-11 Tensile Properties for Millstone Unit 2 Reactor Vessel Material Irradiated to 1.74 x10^{19} n/cm^2 (E > 1.0 MeV)										
Material	Sample Number	Test Temp. (°F)	0.2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Stress (ksi)	Fracture Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
1JC	PLATE	75	83.1	103 0	3 38	227.6	68 9	12.5	25 3	70
1J2	PLATE	250	77.3	96 3	3 20	215.5	65 1	11 0	22.8	70
1JL	PLATE	550	70 3	96.3	3 44	149 8	70.1	10.2	19.9	53
3K5	WELD	75	87 5	98.3	3 10	208 8	63.2	11 5	26.0	70
3K3	WELD	250	78 6	90.5	2 95	204 9	60.1	11 0	24.0	71
3K7	WELD	550	75.0	93 3	3 23	176.6	65 8	10.5	21 5	63
4JU	HAZ	75	81 9	101.8	3 35	193 4	68 2	10.8	23 4	65
4JT	HAZ	250	75 8	95.8	3 14	199 3	64 0	10 3	23.0	68
4KK	HAZ	550	72 3	95.3	3 49	176 2	71 1	10 1	20 2	60

LOWER SHELL PLATE C-506-1 (LONGITUDINAL)

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

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	131	0	44.22	0	64.35	0
2	0	2.19	0	94	-37	109.97	65.75	150.11	85.75
3	0	2.19	0	95	-36	131.9	87.67	173.56	109.2
4	0	2.19	0	92	-39	163.34	119.12	204.45	140.1



Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR.	PLATE SA533B1	LT	C-5667-1
2	ML2	W-97	PLATE SA533B1	LT	C-5667-1
3	ML2	W-104	PLATE SA533B1	LT	C-5667-1
4	ML2	W-83	PLATE SA533B1	LT	C-5667-1

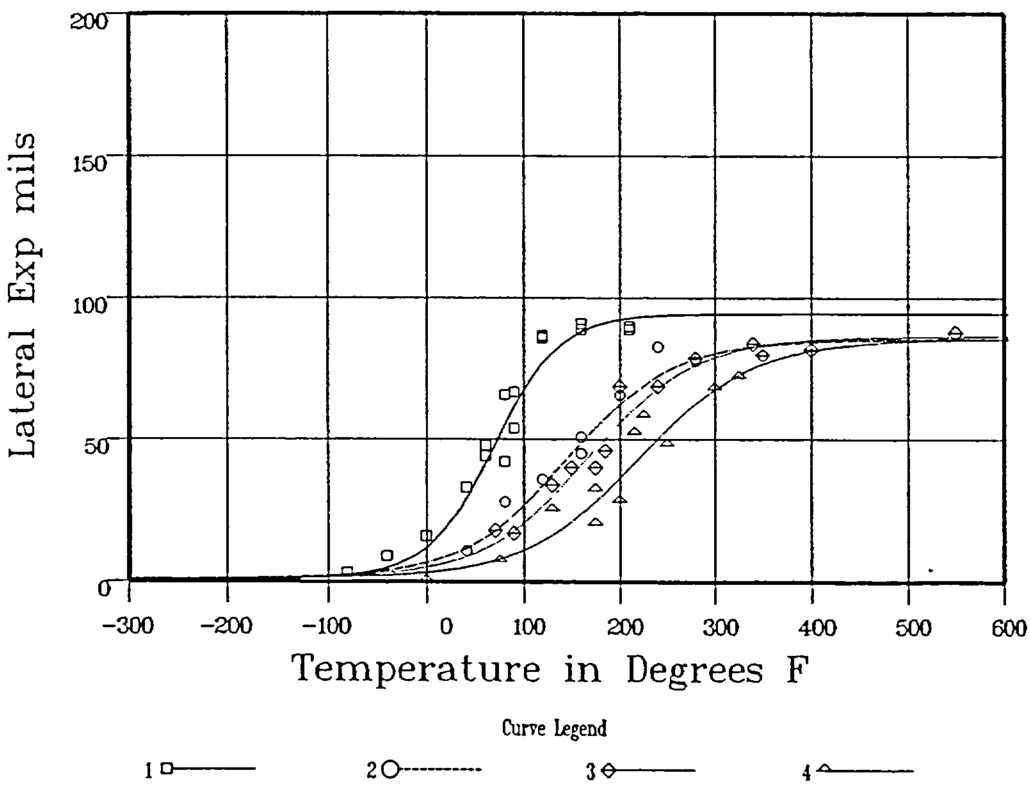
Figure 5-1 Charpy V-Notch Impact Data for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Longitudinal Orientation)

LOWER SHELL PLATE C-506-1 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:30:01 on 10-10-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	94.44	0	44.98	0
2	0	85.56	-8.87	119	74.01
3	0	86.57	-7.86	139.72	94.73
4	0	85.68	-8.75	191.3	146.32



Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	PLATE SA533B1	LT	C-5667-1
2	ML2	W-97	PLATE SA533B1	LT	C-5667-1
3	ML2	W-104	PLATE SA533B1	LT	C-5667-1
4	ML2	W-83	PLATE SA533B1	LT	C-5667-1

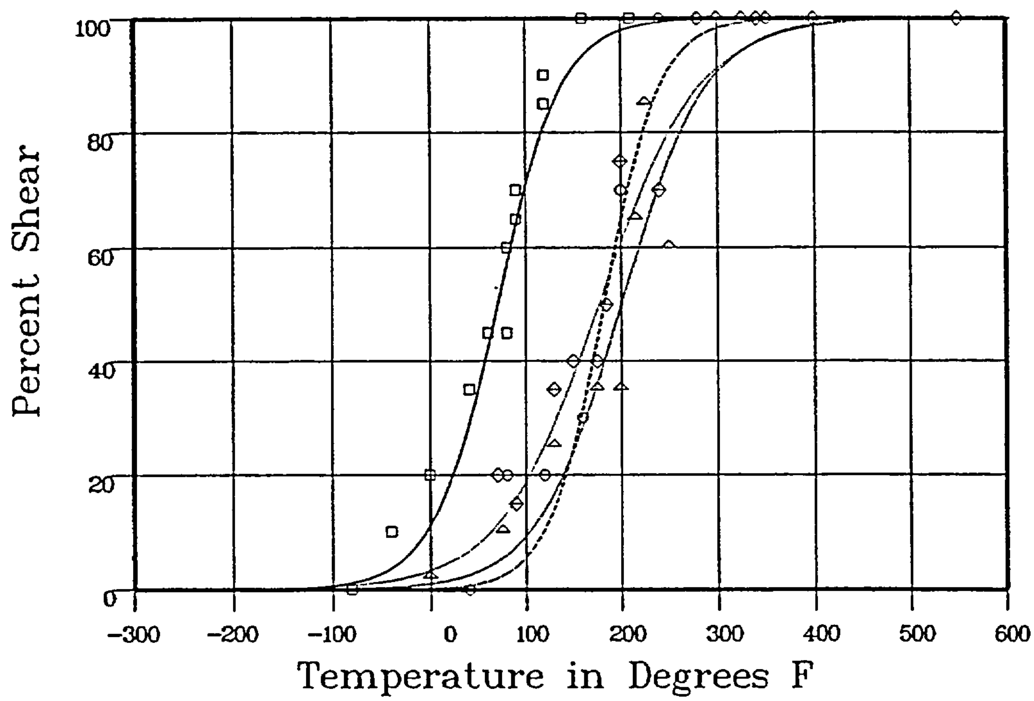
Figure 5-2 Charpy V-Notch Lateral Expansion vs. Temperature for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Longitudinal Orientation)

LOWER SHELL PLATE C-506-1 (LONGITUDINAL)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 15:36:10 on 10-10-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	65.62	0
2	0	177.65	112.03
3	0	172.55	106.93
4	0	196.02	130.39



Curve Legend

1 □——— 2 ○-------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	PLATE SA533B1	LT	C-5667-1
2	ML2	W-97	PLATE SA533B1	LT	C-5667-1
3	ML2	W-104	PLATE SA533B1	LT	C-5667-1
4	ML2	W-83	PLATE SA533B1	LT	C-5667-1

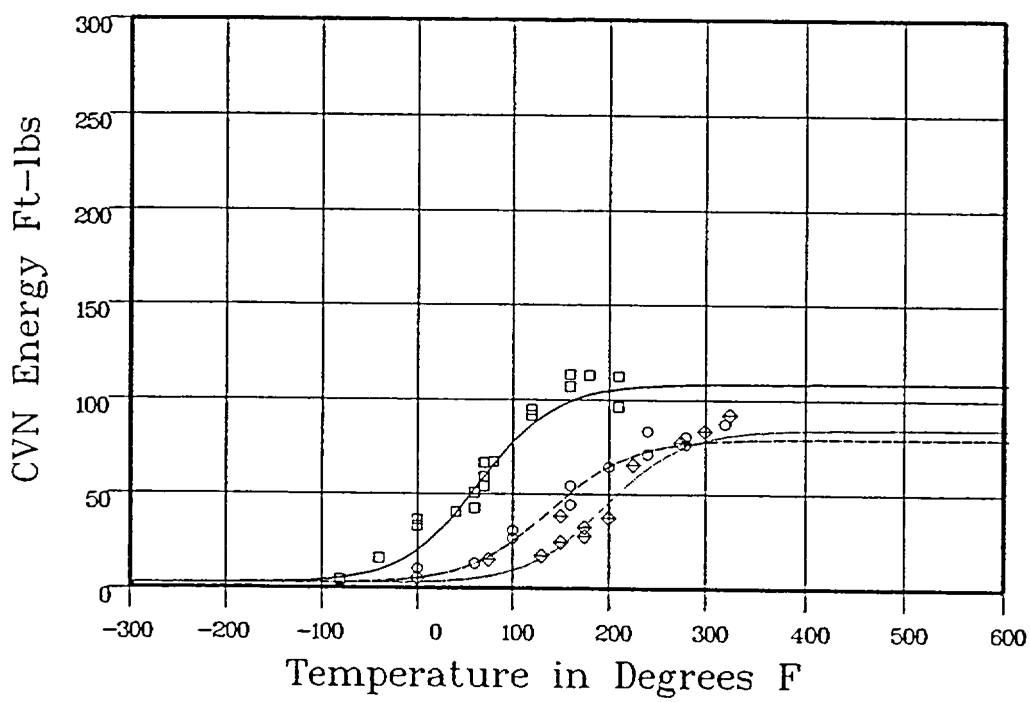
Figure 5-3 Charpy V-Notch Percent Shear vs. Temperature for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Longitudinal Orientation)

LOWER SHELL PLATE C-506-1 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:43:52 on 10-14-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	108	0	18.4	0	52.2	0
2	0	2.19	0	79	-29	109.24	90.83	155.06	102.85
3	0	2.19	0	84	-24	164.18	145.78	205.13	152.93



Curve Legend

1 ——— 2 - - - - 3 ———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	PLATE SA533B1	TL	C-5667-1
2	ML2	W-97	PLATE SA533B1	TL	C-5667-1
3	ML2	W-83	PLATE SA533B1	TL	C-5667-1

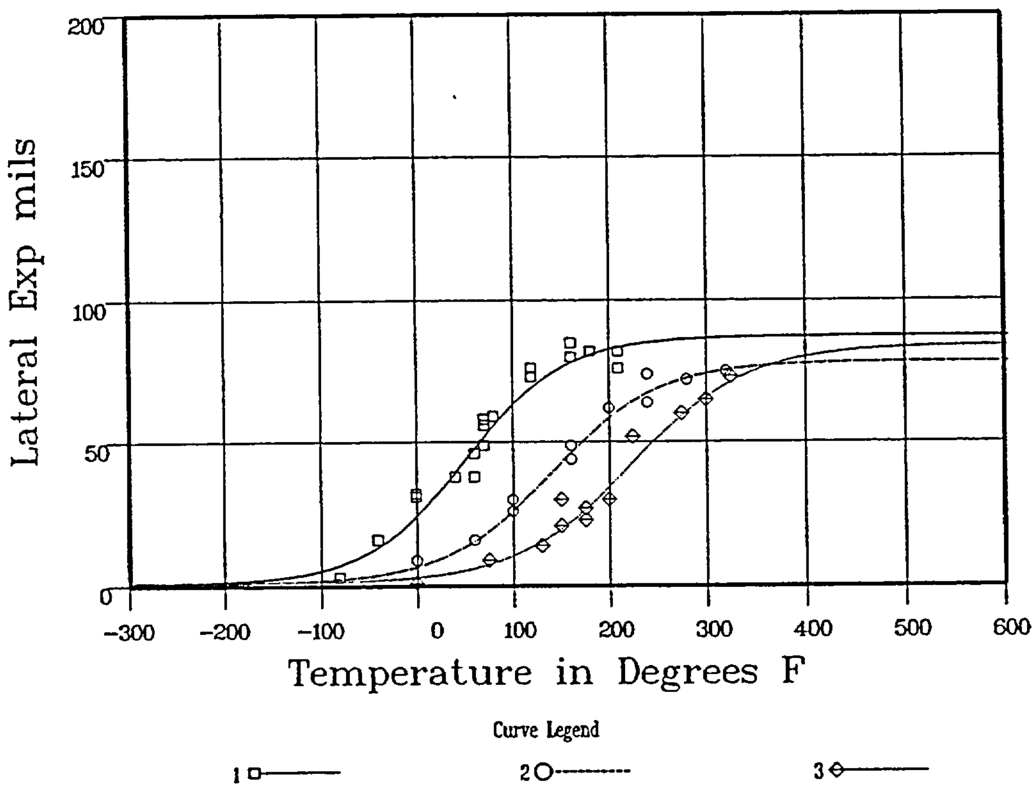
Figure 5-4 Charpy V-Notch Impact Data for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Transverse Orientation)

LOWER SHELL PLATE C-506-1 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 12:04:01 on 10-14-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	87.4	0	24.64	0
2	0	78.24	-9.16	122.63	97.98
3	0	84.22	-3.17	196.84	172.19



Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	PLATE SA533B1	TL	C-5667-1
2	ML2	W-97	PLATE SA533B1	TL	C-5667-1
3	ML2	W-83	PLATE SA533B1	TL	C-5667-1

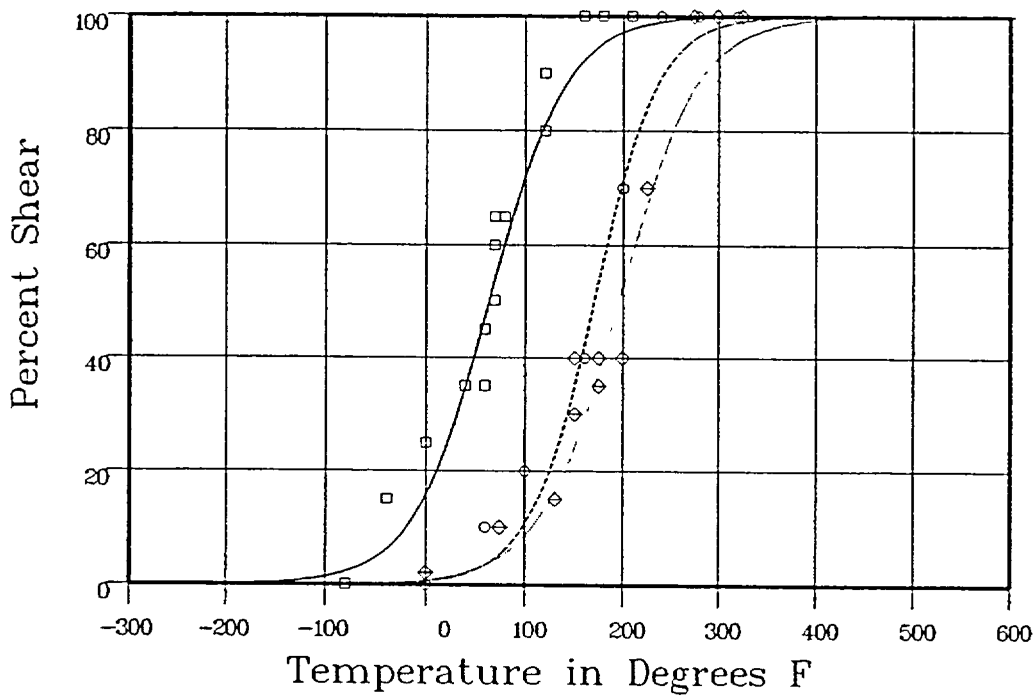
Figure 5-5 Charpy V-Notch Lateral Expansion vs. Temperature for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Transverse Orientation)

LOWER SHELL PLATE C-506-1 (TRANSVERSE)

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 12:13:13 on 10-14-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	59.53	0
2	0	165.52	105.99
3	0	191.71	132.18



Curve Legend

1 □——— 2 ○--------- 3 ◇———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat #
1	ML2	UNIRR	PLATE SA533B1	TL	C-5667-1
2	ML2	W-97	PLATE SA533B1	TL	C-5667-1
3	ML2	W-83	PLATE SA533B1	TL	C-5667-1

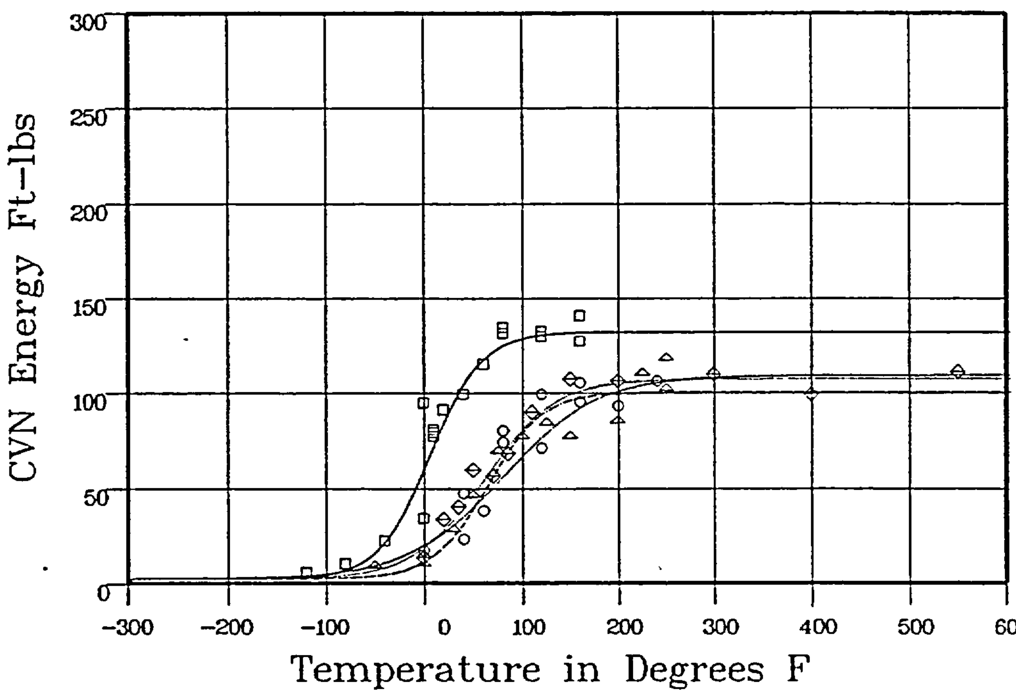
Figure 5-6 Charpy V-Notch Percent Shear vs. Temperature for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Transverse Orientation)

SURVEILLANCE PROGRAM WELD MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:11:31 on 11-21-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	132	0	-32.15	0	-12.25	0
2	0	2.19	0	100	-32	33.78	65.93	58.27	70.52
3	0	2.19	0	107	-25	19.97	52.12	50.37	62.62
4	0	2.19	0	109	-23	23.94	56.09	64.7	76.96



Curve Legend

1 □——— 2 ○--------- 3 ◇———— 4 △————

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	MI2	UNIRR	WELD L 124/0091		90136/10137
2	MI2	W-97	WELD L 124/0091		90136/10137
3	MI2	W-104	WELD L 124/0091		90136/10137
4	MI2	W-83	WELD L 124/0091		90136/10137

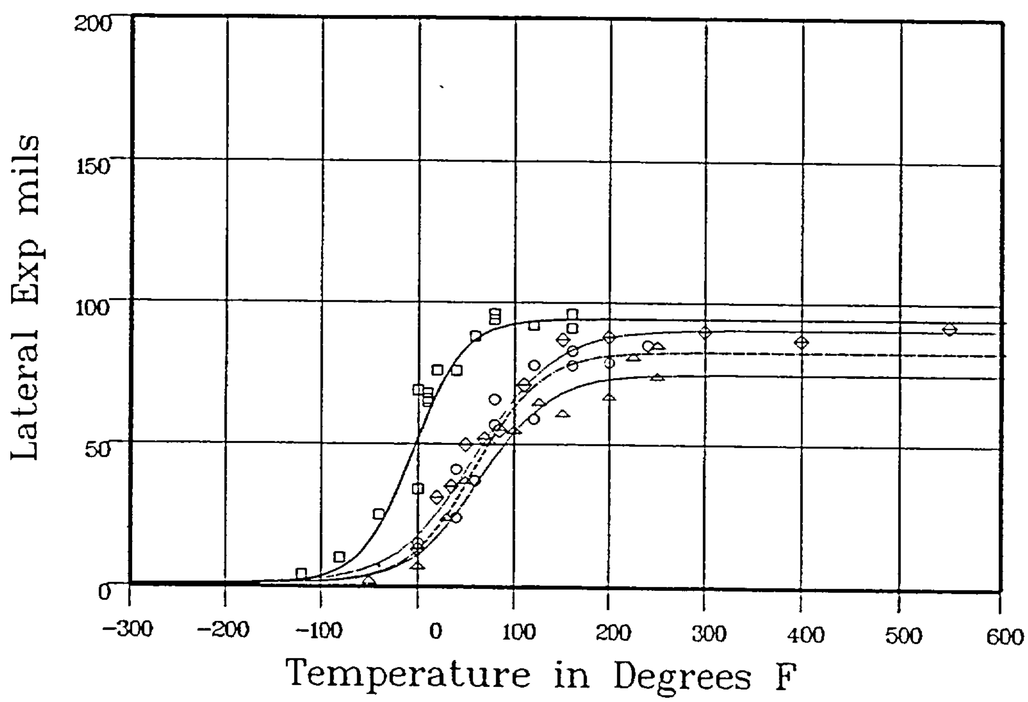
Figure 5-7 Charpy V-Notch Impact Data for Millstone Unit 2 Reactor Vessel Weld Metal

SURVEILLANCE PROGRAM WELD MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 12:20:58 on 10-14-2002

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	94.33	0	-24.52	0
2	0	82.72	-11.6	44.25	68.77
3	0	90.54	-3.79	35.2	59.73
4	0	74.78	-19.55	55.28	79.8



Curve Legend

1 □——— 2 ○-------- 3 ◇——— 4 △———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	WELD	L 124/0091	90136/10137
2	ML2	W-97	WELD	L 124/0091	90136/10137
3	ML2	W-104	WELD	L 124/0091	90136/10137
4	ML2	W-83	WELD	L 124/0091	90136/10137

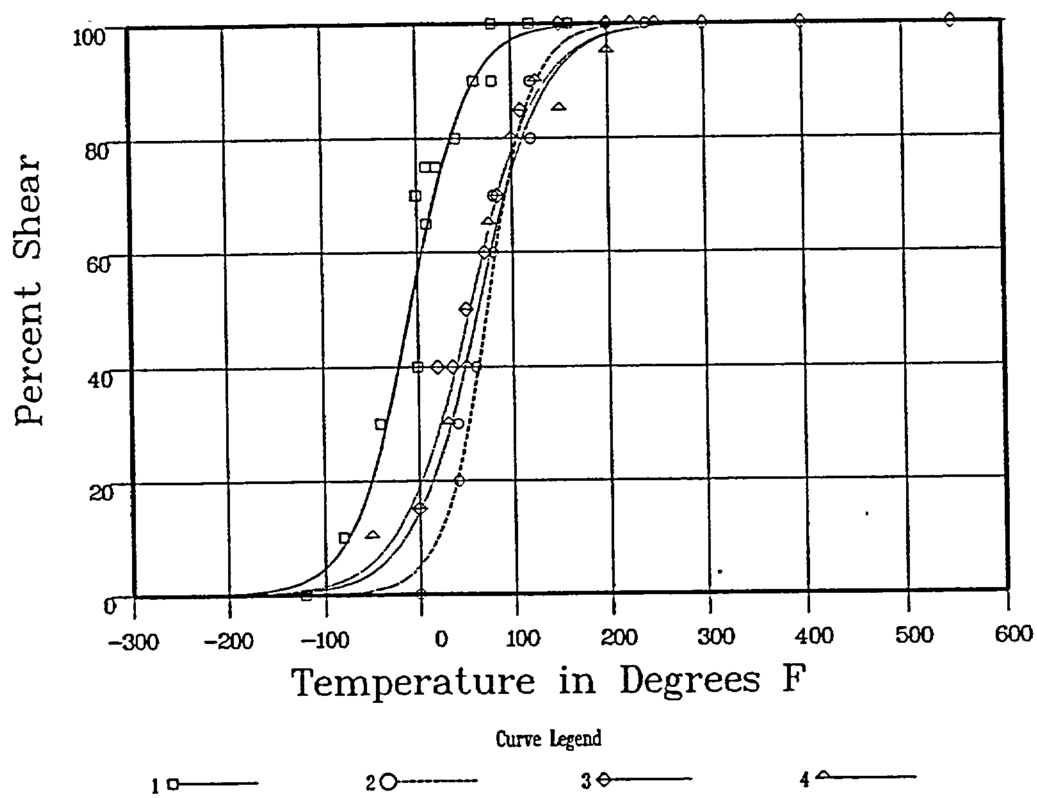
Figure 5-8 Charpy V-Notch Lateral Expansion vs. Temperature for Millstone Unit 2 Reactor Vessel Weld Metal

SURVEILLANCE PROGRAM WELD MATERIAL

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 12:28:54 on 10-14-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	-12.18	0
2	0	67.7	79.89
3	0	49.21	61.4
4	0	58.12	70.31



Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	WELD L 124/0091		90136/10137
2	ML2	W-97	WELD L 124/0091		90136/10137
3	ML2	W-104	WELD L 124/0091		90136/10137
4	ML2	W-83	WELD L 124/0091		90136/10137

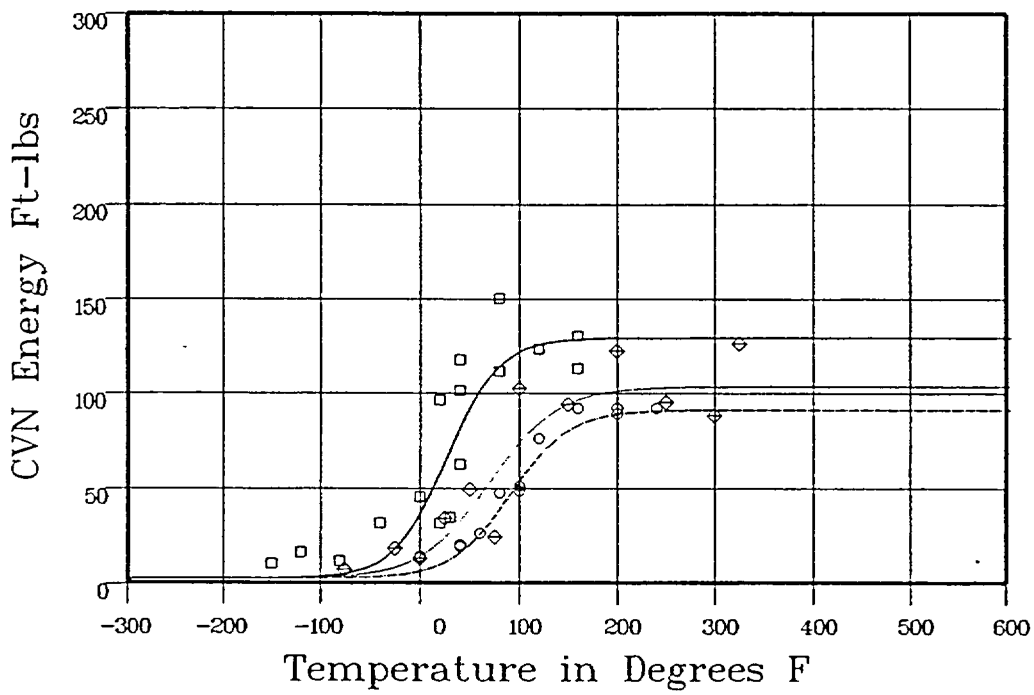
Figure 5-9 Charpy V-Notch Percent Shear vs. Temperature for Millstone Unit 2 Reactor Vessel Weld Metal

HEAT AFFECTED ZONE

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 09:57:39 on 10-15-2002

Results

Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0	2.19	0	129	0	-115.2	0	8.75	0
2	0	2.19	0	91	-38	62.74	74.26	90.83	82.08
3	0	2.19	0	103	-26	31.66	43.18	61.34	52.58



Curve Legend

1 □——— 2 ○--------- 3 ◇———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	HEAT AFFD ZONE SA533B1		C-506-1
2	ML2	W-97	HEAT AFFD ZONE SA533B1		C-506-1
3	ML2	W-83	HEAT AFFD ZONE SA533B1		C-506-1

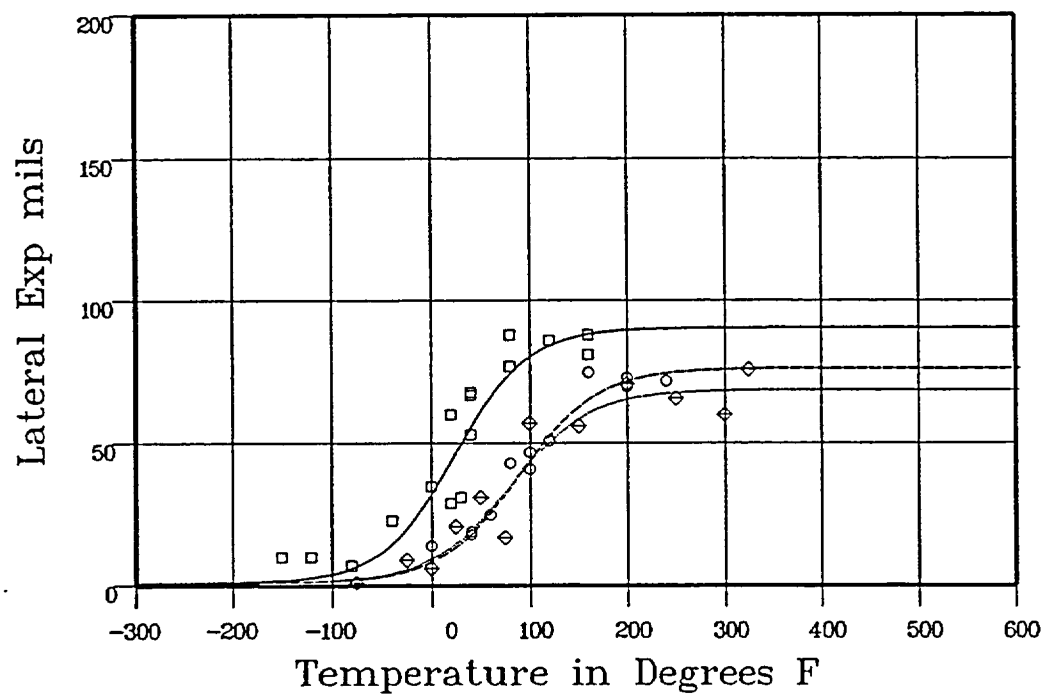
Figure 5-10 Charpy V-Notch Impact Data for Millstone Unit 2 Reactor Vessel Shell Heat Affected Zone Material

HEAT AFFECTED ZONE

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

Results

Curve	Fluence	USE	d-USE	T @ LE35	d-T @ LE35
1	0	90.39	0	.72	0
2	0	76.34	-14.05	76.37	75.65
3	0	68.9	-21.49	75.4	74.68



Curve Legend

1 □——— 2 ○------ 3 ◇———

Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	HEAT AFFD ZONE SA533B1		C-506-1
2	ML2	W-97	HEAT AFFD ZONE SA533B1		C-506-1
3	ML2	W-83	HEAT AFFD ZONE SA533B1		C-506-1

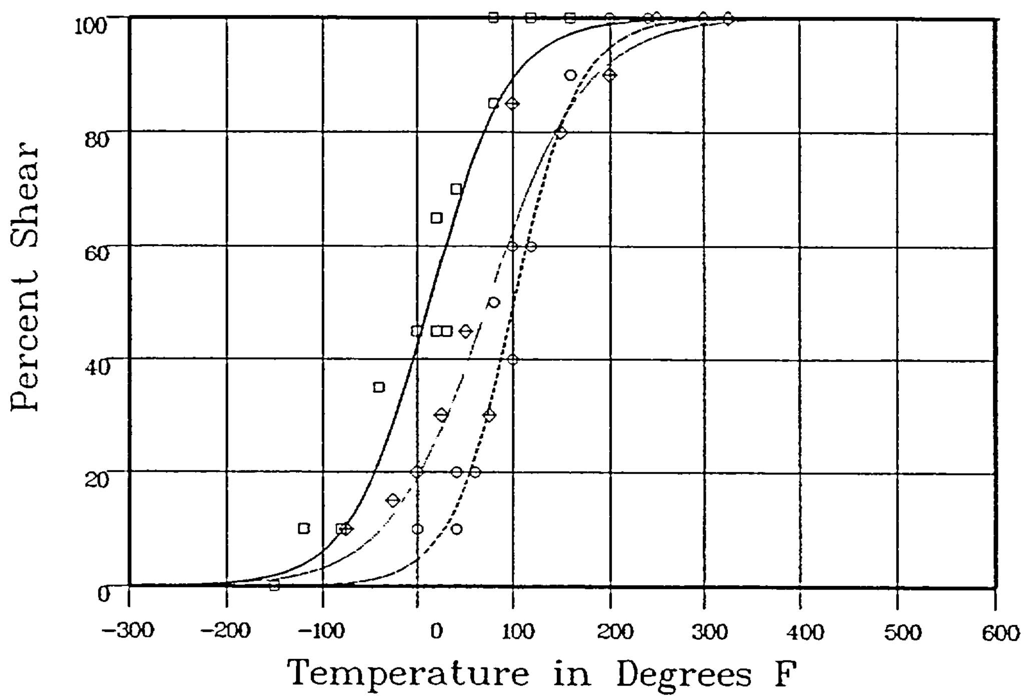
Figure 5-11 Charpy V-Notch Lateral Expansion vs. Temperature for Millstone Unit 2 Reactor Vessel Shell Heat Affected Zone Material

HEAT AFFECTED ZONE

CVGRAPH 4.1 Hyperbolic Tangent Curve Printed at 10:04:13 on 10-15-2002

Results

Curve	Fluence	T @ 50% Shear	d-T @ 50% Shear
1	0	7.96	0
2	0	96.67	88.71
3	0	69.37	61.4



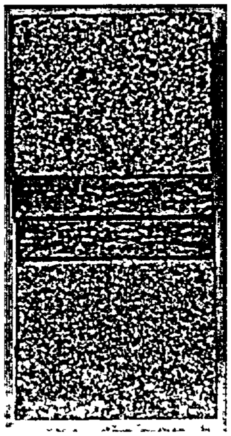
Curve Legend

1 ☐——— 2 ○--------- 3 ◇———

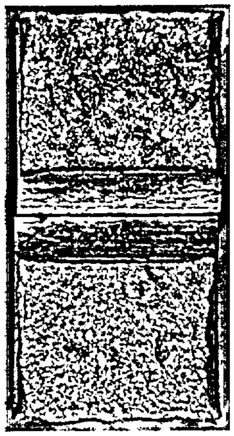
Data Set(s) Plotted

Curve	Plant	Capsule	Material	Ori.	Heat#
1	ML2	UNIRR	HEAT AFFD ZONE SA533B1		C-506-1
2	ML2	W-97	HEAT AFFD ZONE SA533B1		C-506-1
3	ML2	W-83	HEAT AFFD ZONE SA533B1		C-506-1

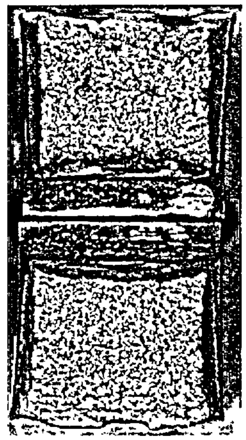
Figure 5-12 Charpy V-Notch Percent Shear vs. Temperature for Millstone Unit 2 Reactor Vessel Heat Affected Zone Material



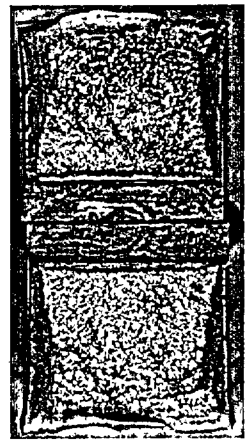
132, 0°F



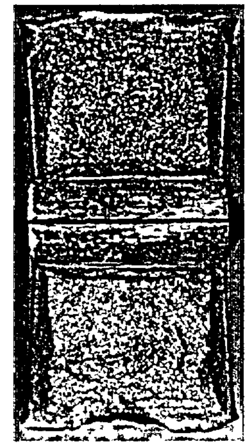
146, 75°F



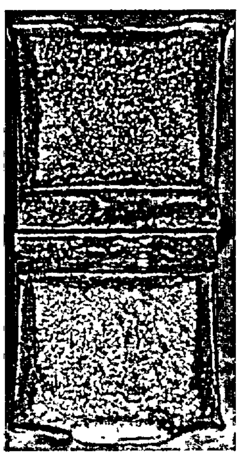
117, 130°F



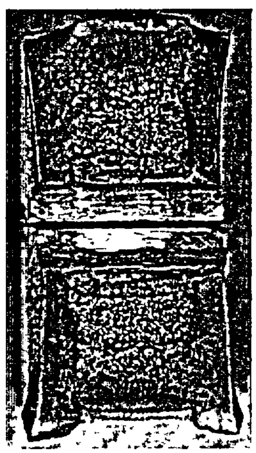
13C, 175°F



121, 175°F

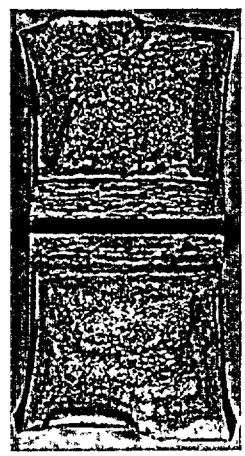


136, 200°F



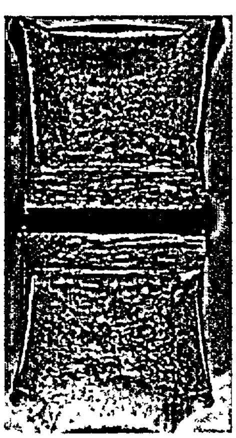
156, 215°F

165, 225°F

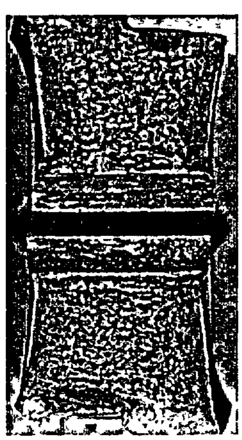


131, 250°F

12K, 300°F



126, 325°F



16D, 350°F

Figure 5-13 Charpy Impact Specimen Fracture Surfaces for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Longitudinal Orientation)

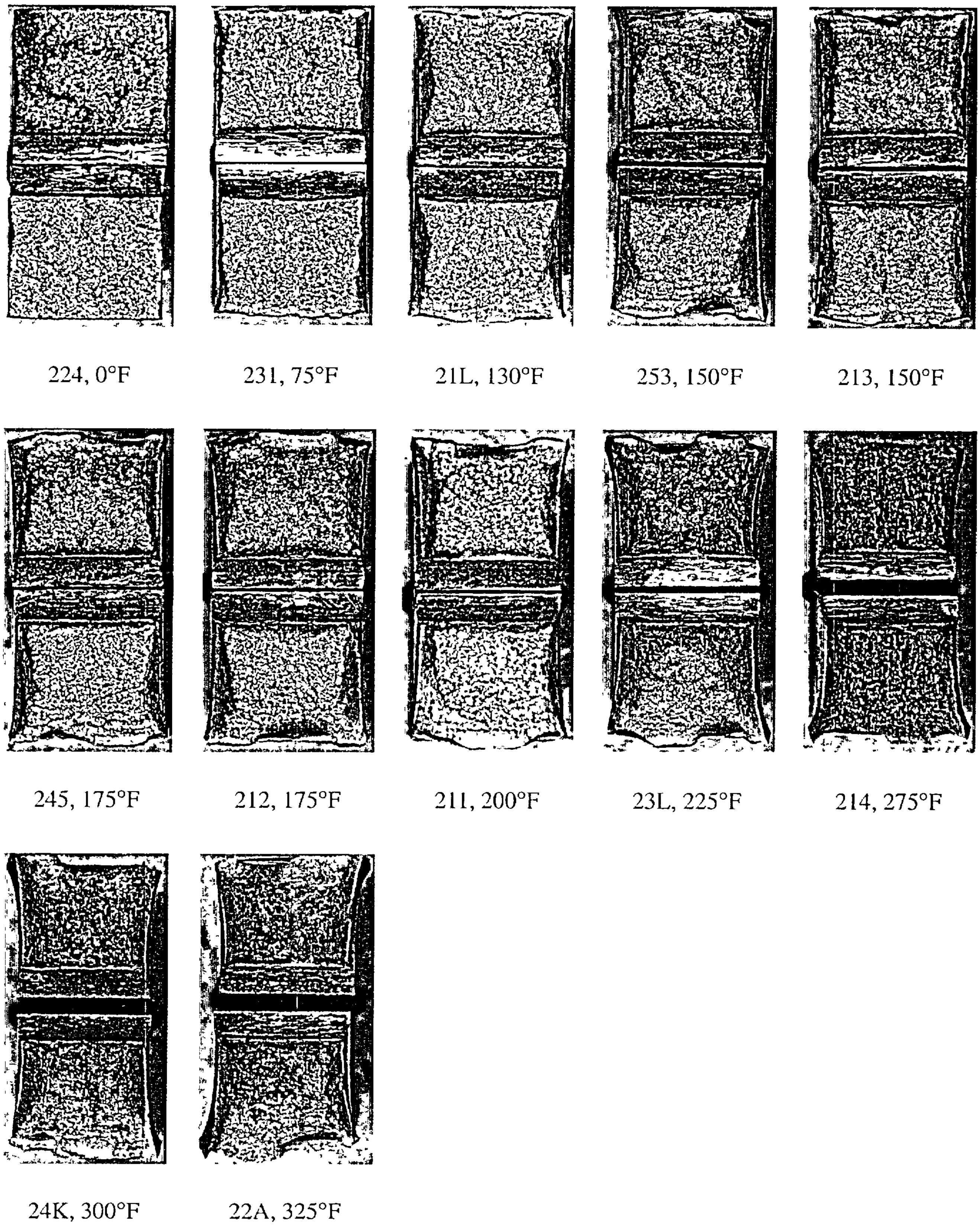
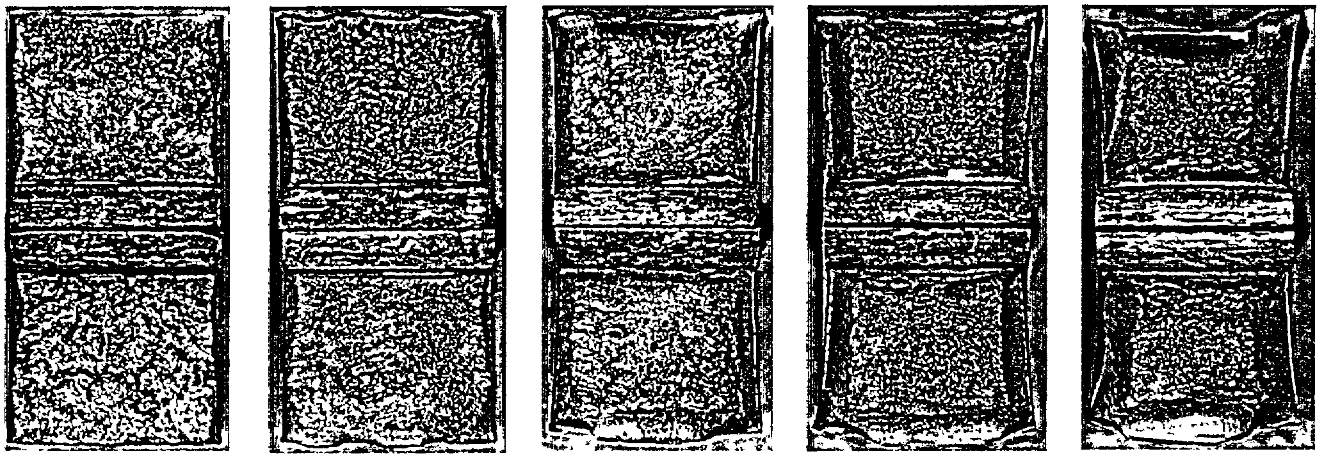
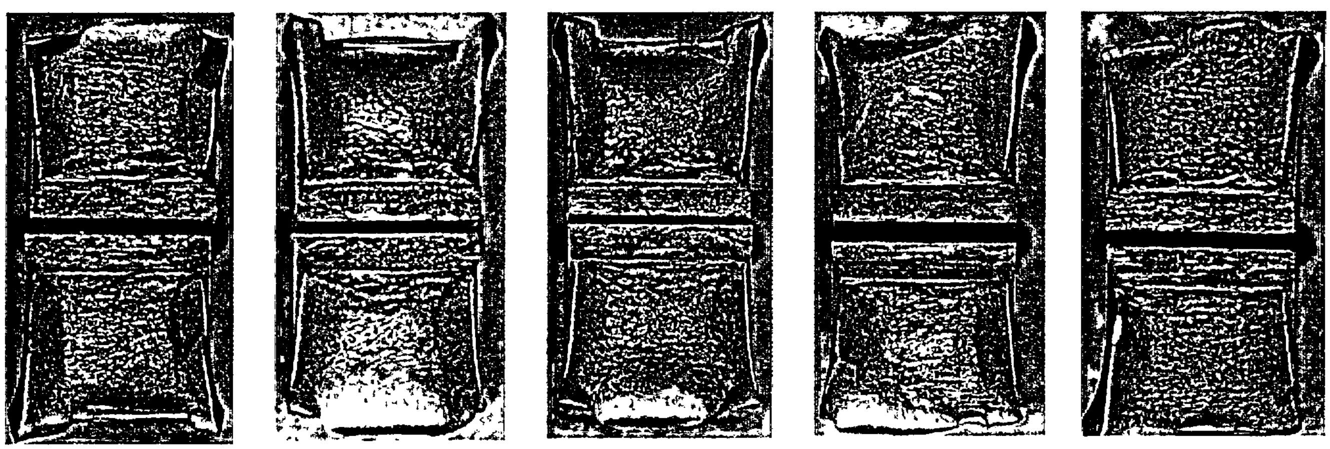


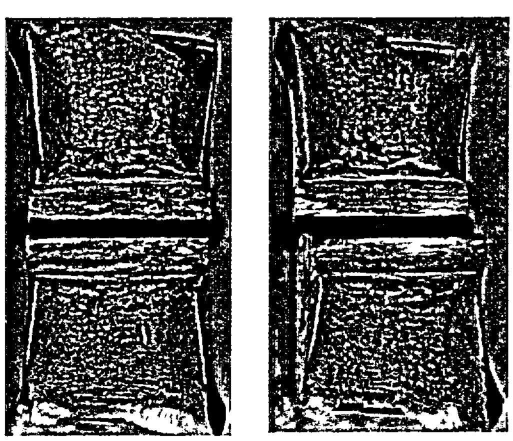
Figure 5-14 Charpy Impact Specimen Fracture Surfaces for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Transverse Orientation)



314, -50°F | 33K, 0°F | 34L, 30°F | 311, 50°F | 32A, 75°F



36D, 100°F | 36E, 125°F | 34C, 150°F | 337, 200°F | 323, 225°F



336, 250°F | 312, 250°F

Figure 5-15 Charpy Impact Specimen Fracture Surfaces for Millstone Unit 2 Reactor Vessel Weld Metal

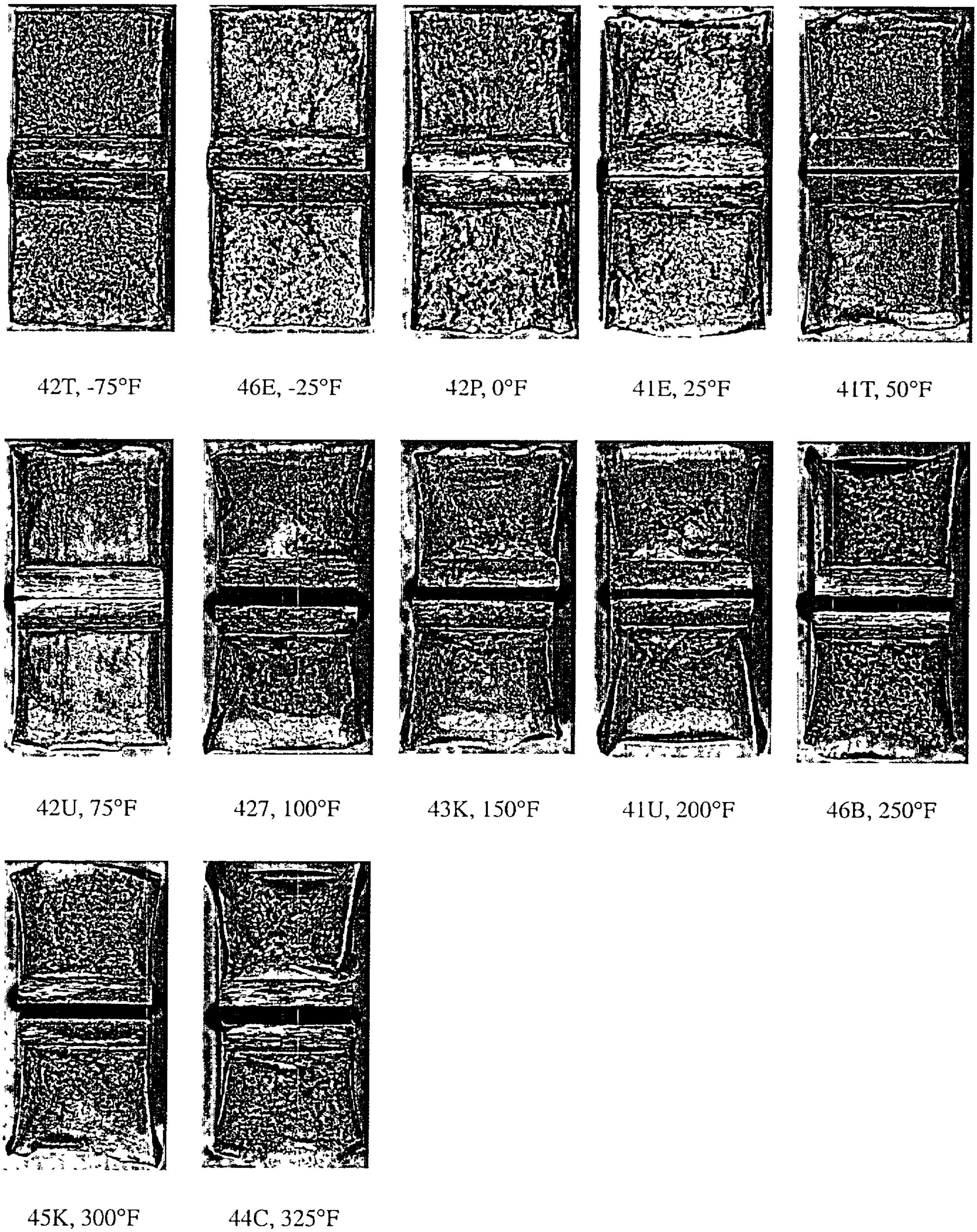


Figure 5-16 Charpy Impact Specimen Fracture Surfaces for Millstone Unit 2 Reactor Vessel Weld HAZ Metal

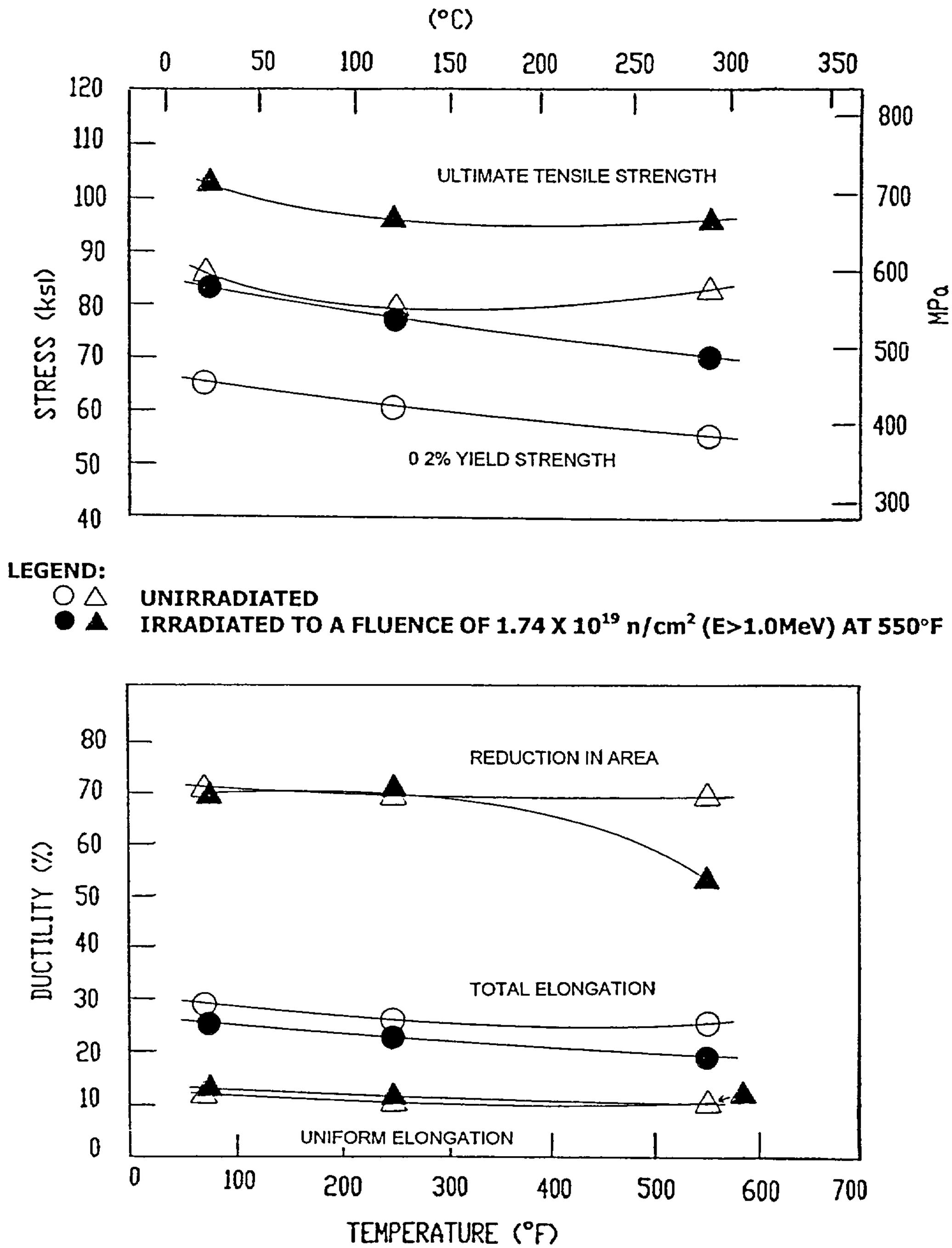


Figure 5-17 Tensile Properties for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Longitudinal Orientation)

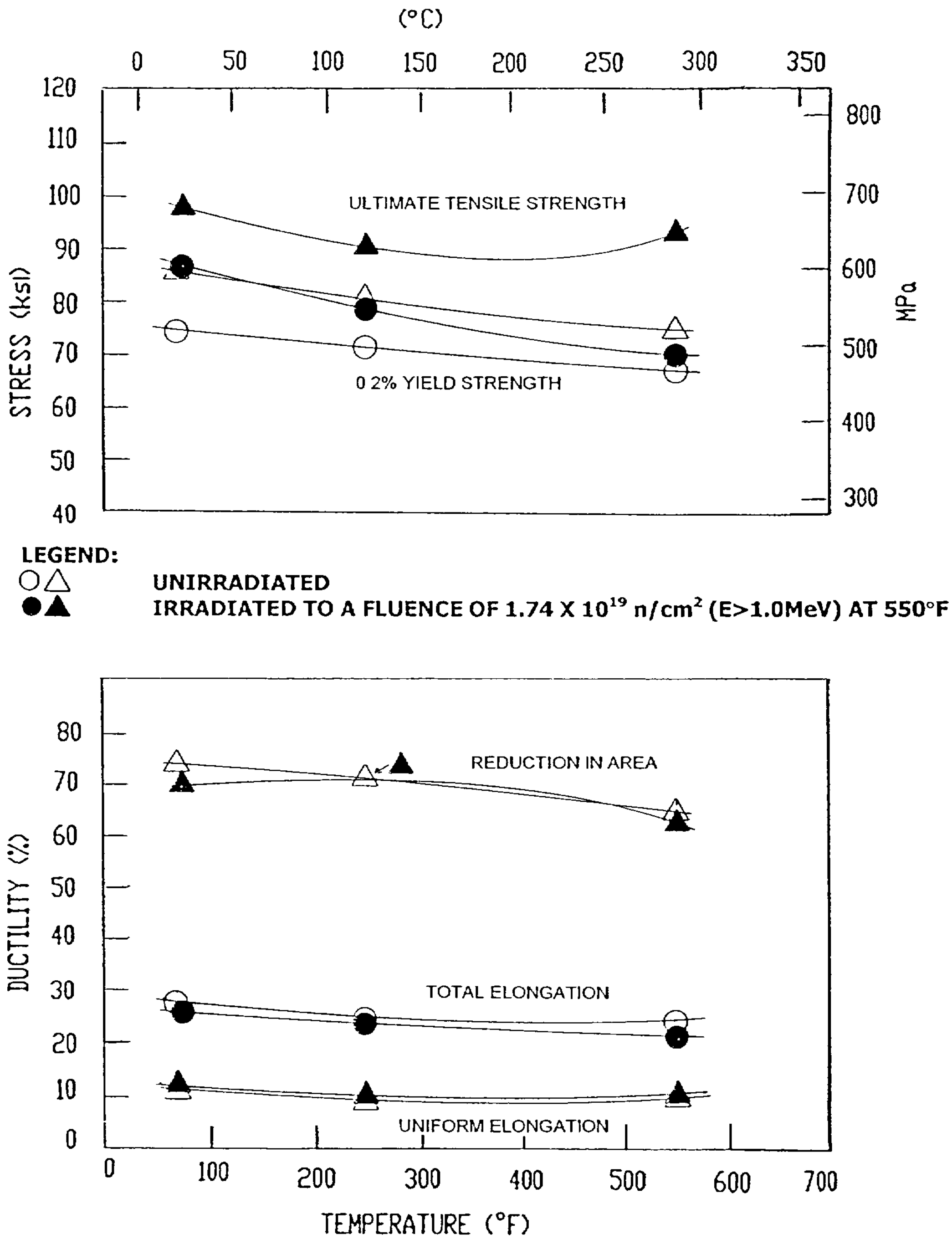


Figure 5-18 Tensile Properties for Millstone Unit 2 Reactor Vessel Weld Metal

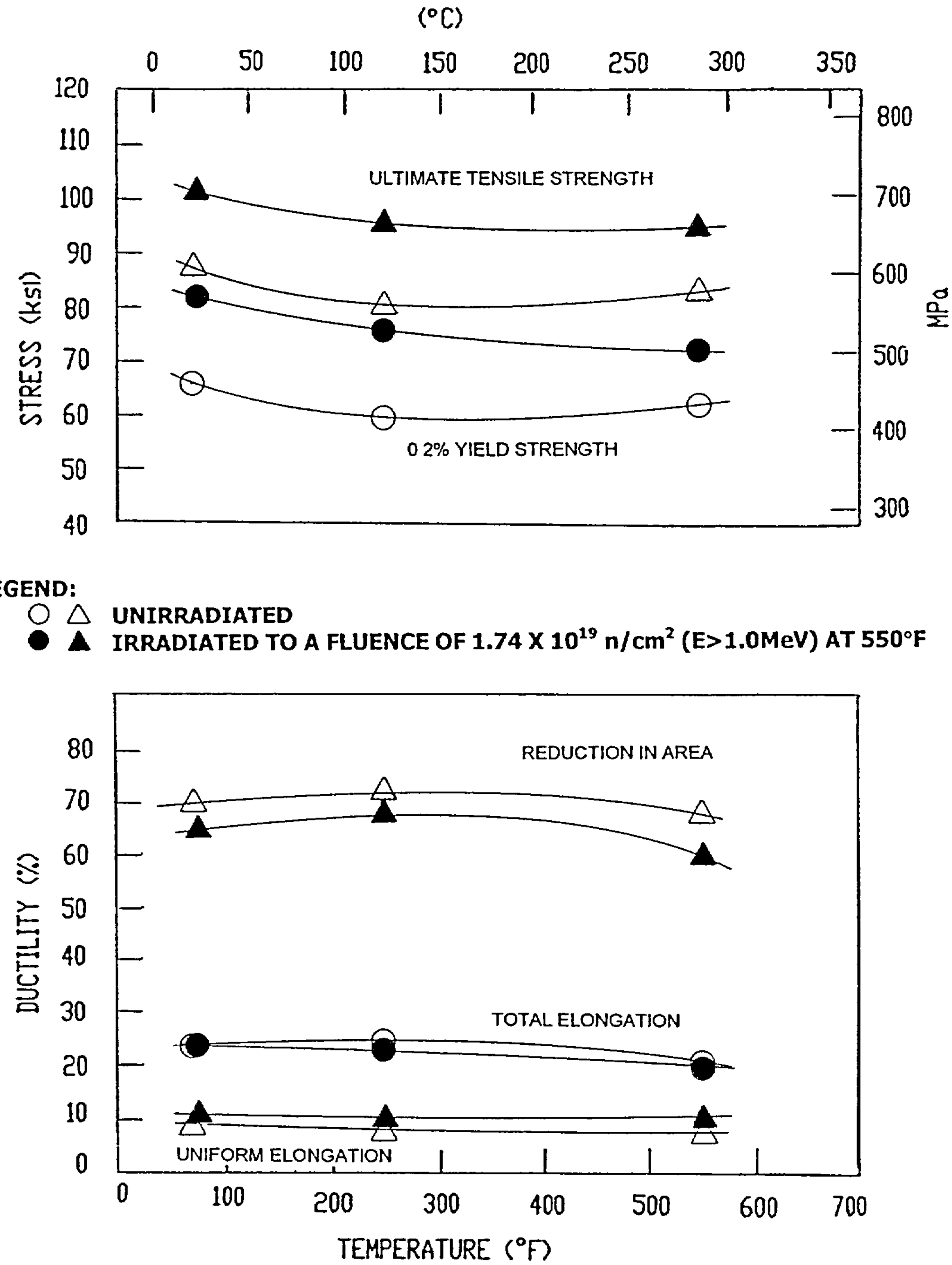
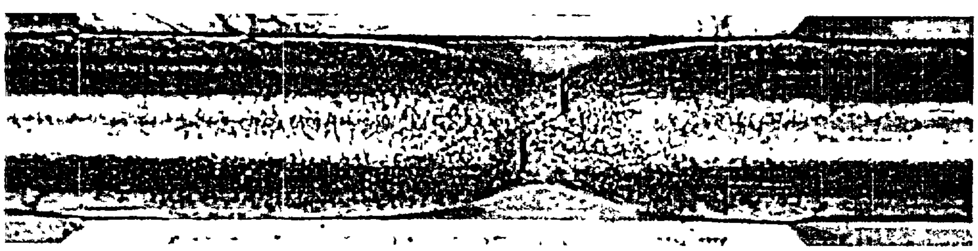
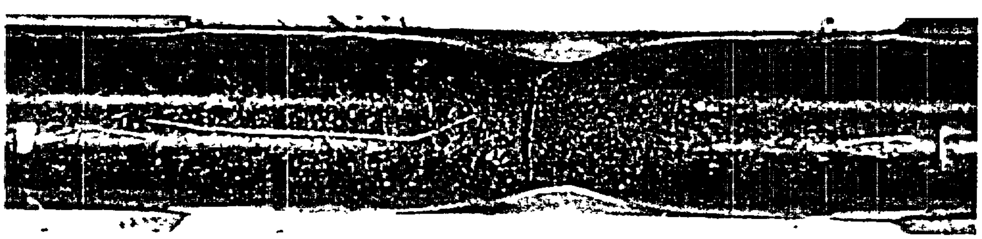


Figure 5-19 Tensile Properties for Millstone Unit 2 Reactor Vessel Heat Affected Zone Material

Specimen 1JC Tested at 75°F

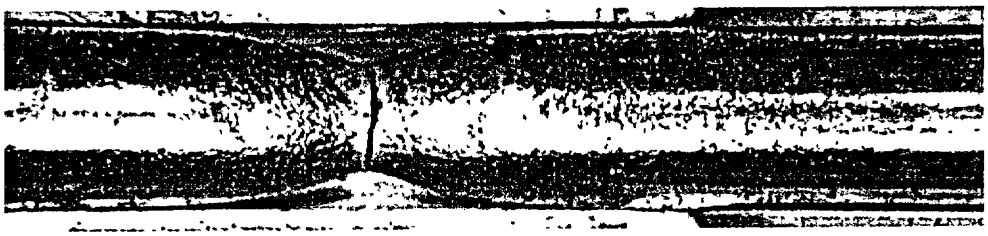


Specimen 1J2 Tested at 250°F



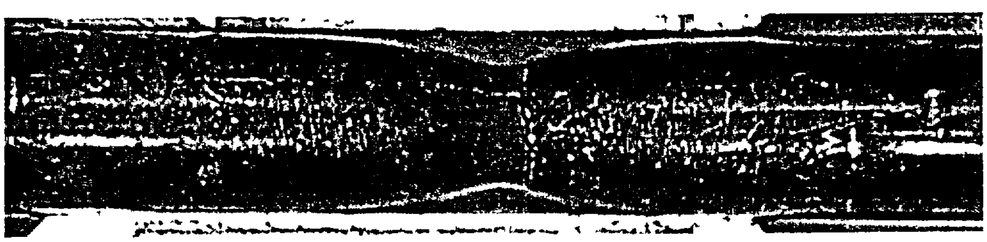
Specimen 1JL Tested at 550°F

Figure 5-20 Fractured Tensile Specimens for Millstone Unit 2 Reactor Vessel Shell Plate C-506-1 (Longitudinal Orientation)



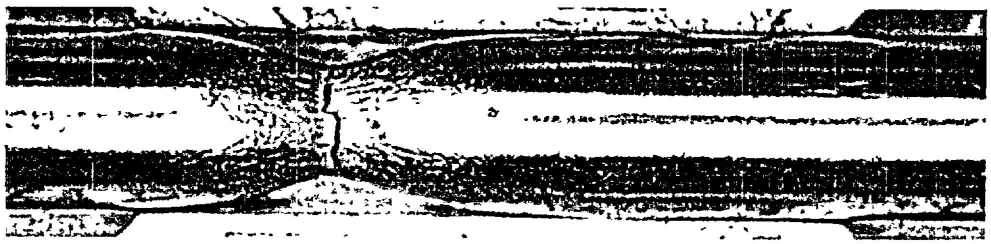
Specimen 3K5 Tested at 75°F

Specimen 3K3 Tested at 250°F

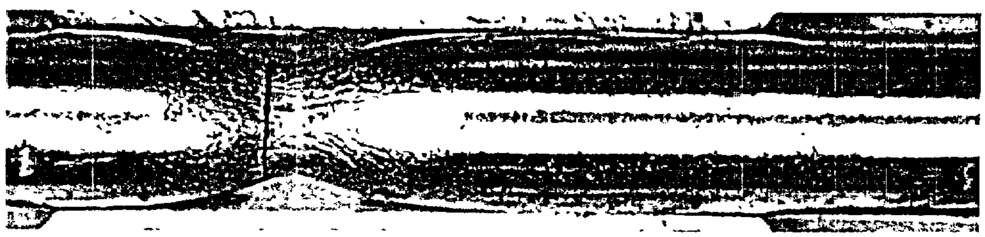


Specimen 3K7 Tested at 550°F

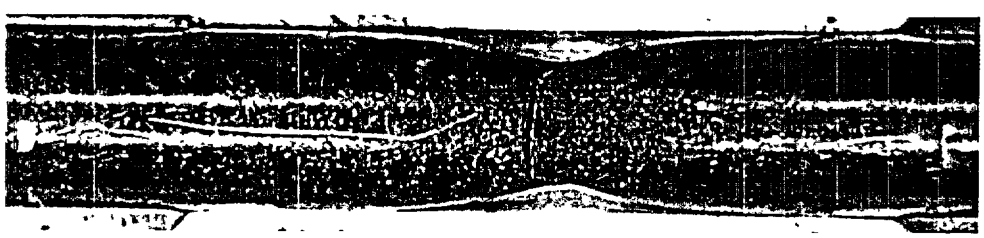
Figure 5-21 Fractured Tensile Specimens for Millstone Unit 2 Reactor Vessel Surveillance Weld Metal



Specimen 4JU Tested at 75°F

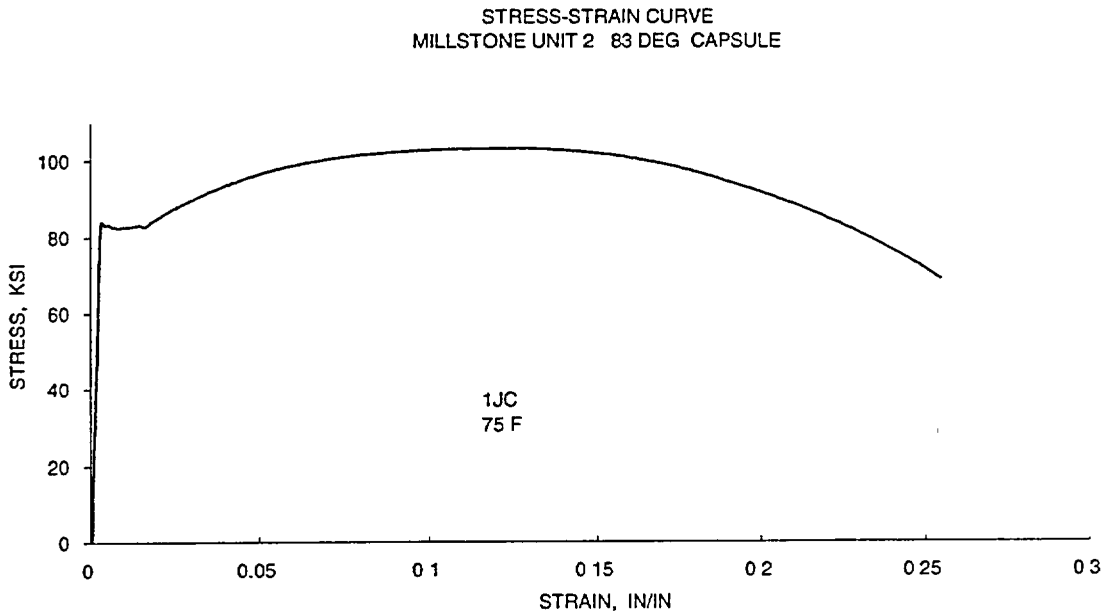


Specimen 4JT Tested at 250°F



Specimen 4KK Tested at 550°F

Figure 5-22 Fractured Tensile Specimens for Millstone Unit 2 Reactor Vessel Surveillance HAZ Metal



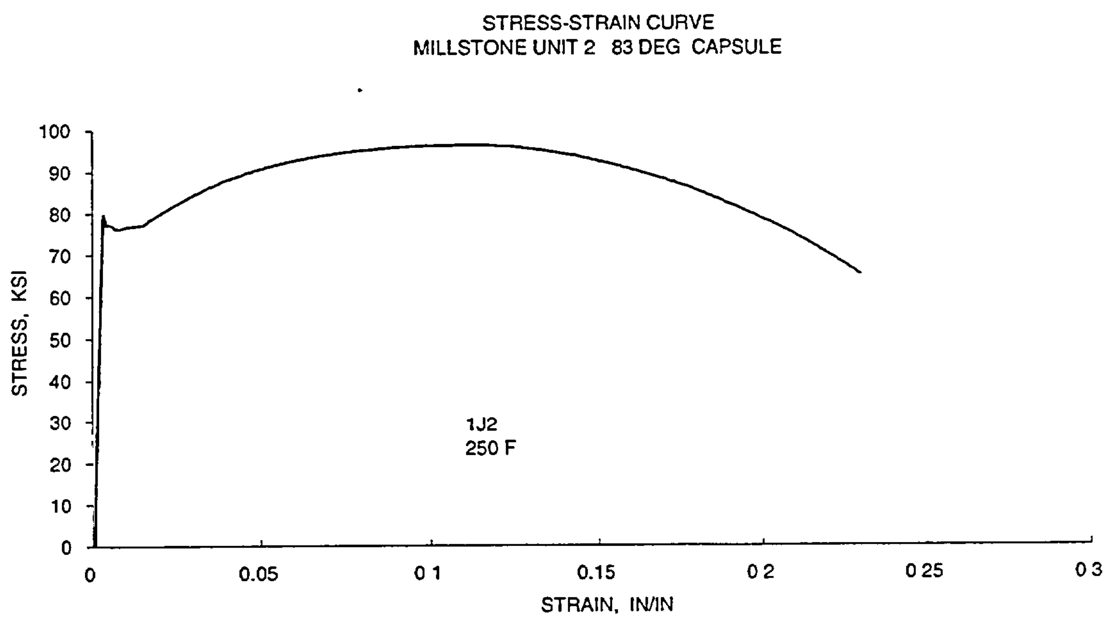


Figure 5-23 Engineering Stress-Strain curves for Millstone Unit 2 Reactor Vessel Intermediate Shell Plate C-506-1, 83° Capsule, Longitudinal Tensile Specimens 1JC and 1J2.

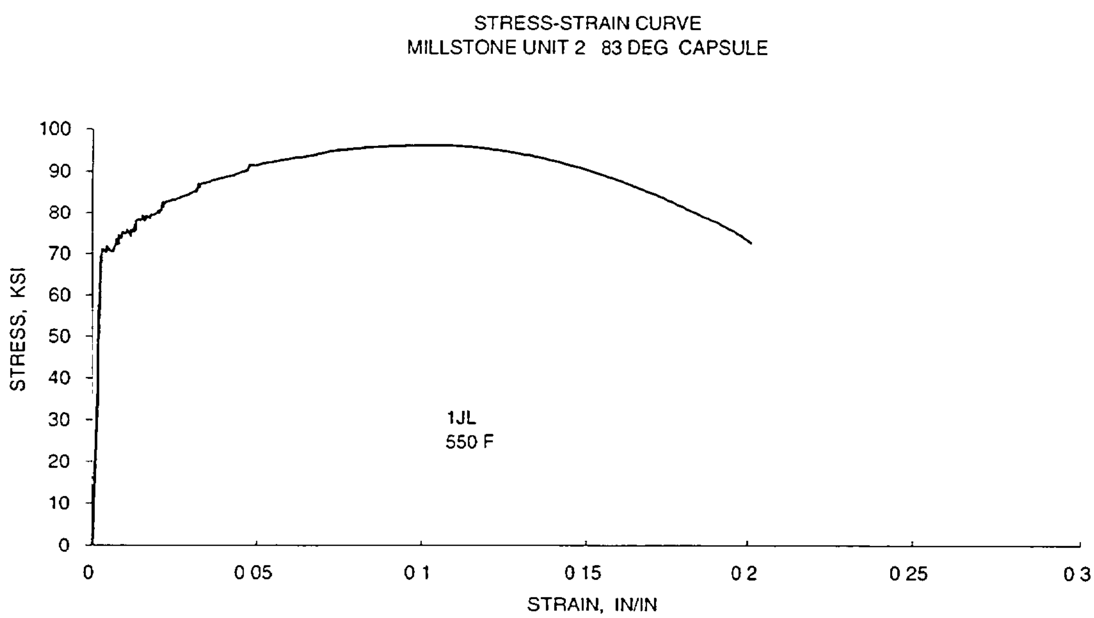
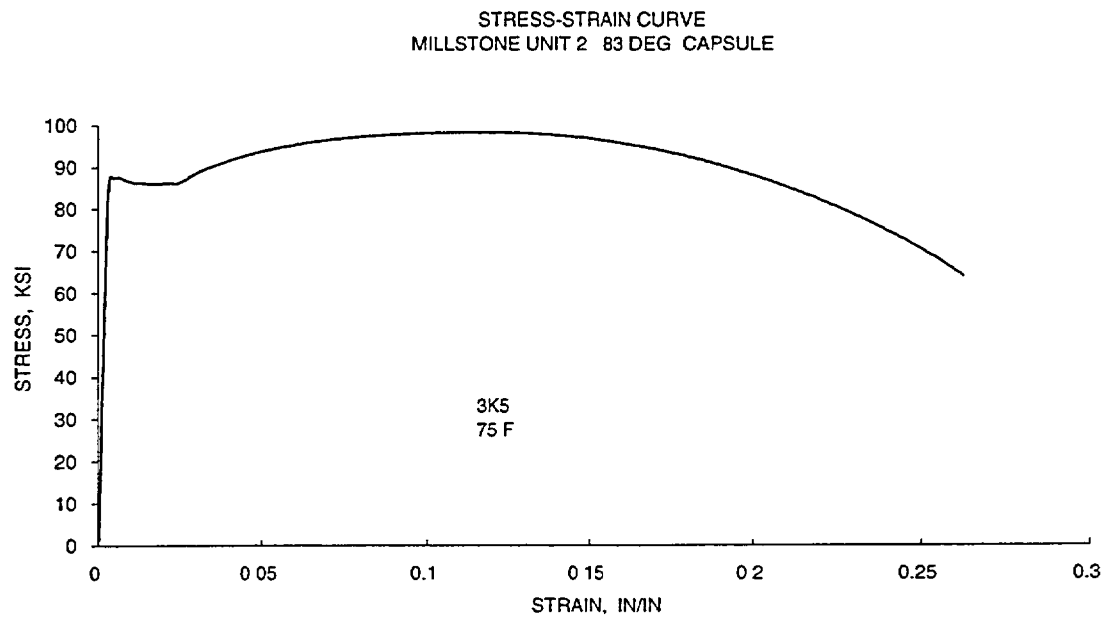


Figure 5-24 Engineering Stress-Strain Curves for Millstone Unit 2 Reactor Vessel Intermediate Shell Plate C-506-1, 83° Capsule, Longitudinal Tensile Specimens 1JL.



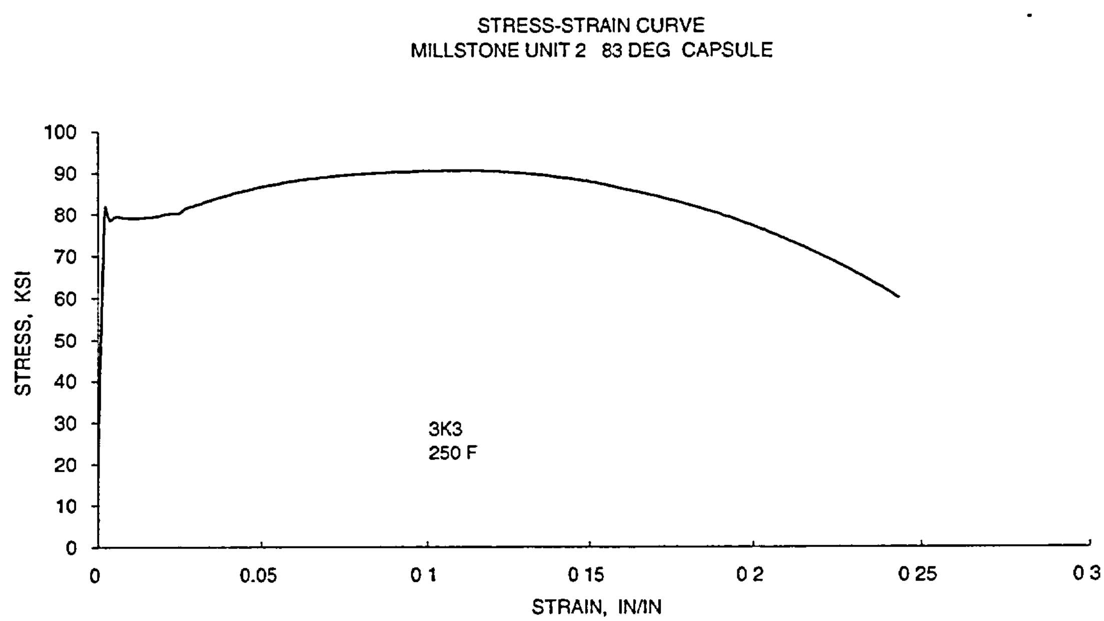


Figure 5-25 Engineering Stress-Strain Curves for Millstone Unit 2 Reactor Vessel Weld Metal 83° Capsule, Tensile Specimens 3K5 and 3K3.

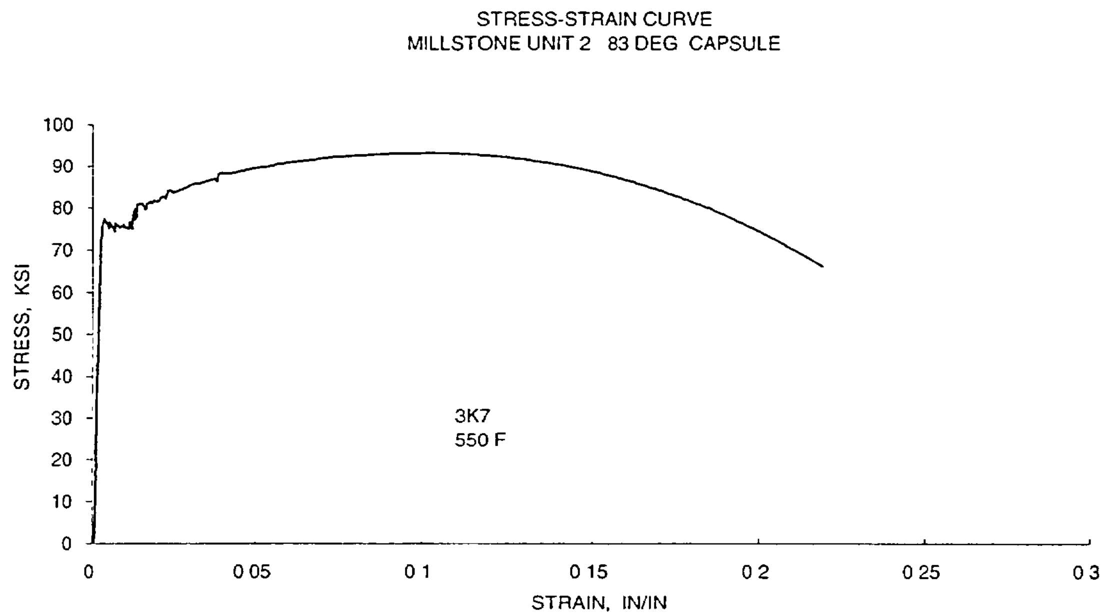
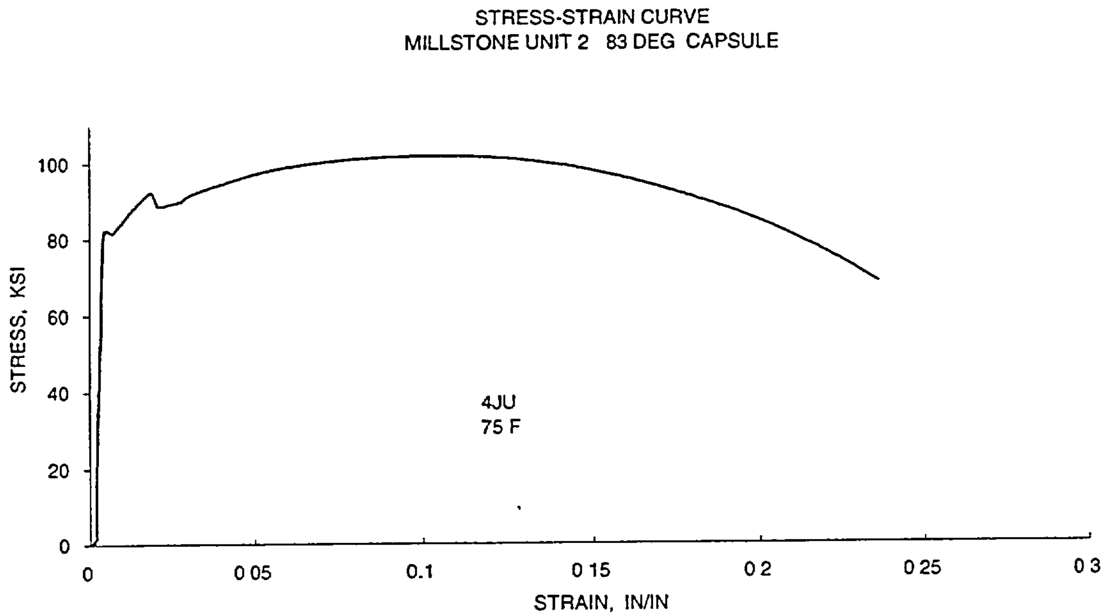


Figure 5-26 Engineering Stress-Strain Curves for Millstone Unit 2 Reactor Vessel Weld Metal 83° Capsule, Tensile Specimens 3K7.



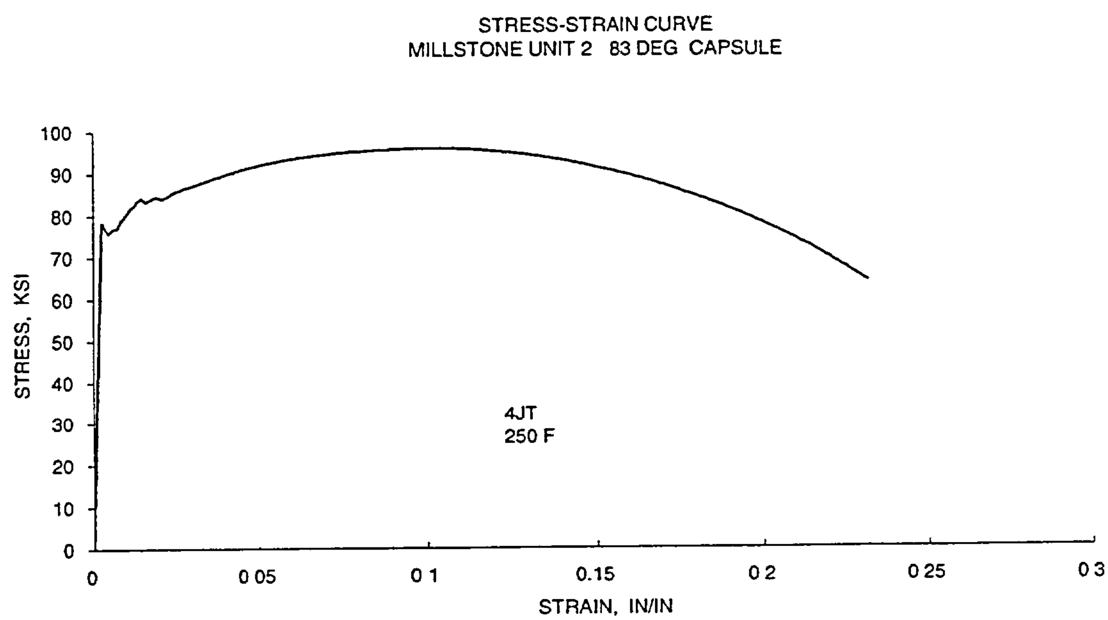


Figure 5-27 Engineering Stress-Strain Curves for Millstone Unit 2 Reactor Vessel HAZ, 83° Capsule, Tensile Specimens 4JU and 4JT.

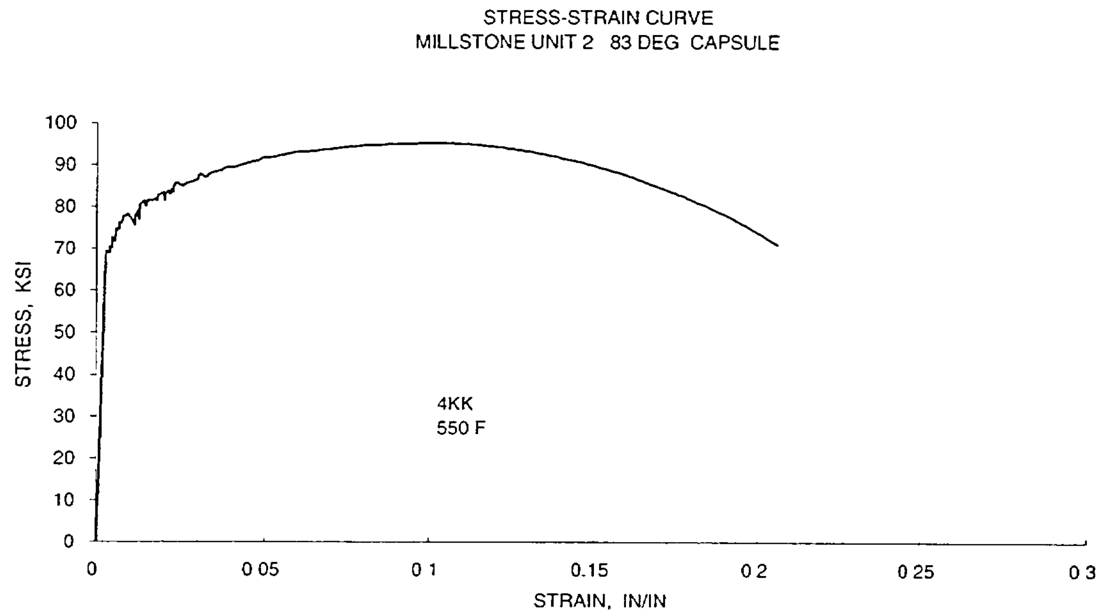


Figure 5-28 Engineering Stress-Strain Curves for Millstone Unit 2 Reactor Vessel HAZ, 83° Capsule, Tensile Specimen 4KK.

6 RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 INTRODUCTION

This section describes a discrete ordinates S_n transport analysis performed for the Millstone Unit 2 reactor to determine the neutron radiation environment within the reactor pressure vessel and surveillance capsules. In this analysis, fast neutron exposure parameters in terms of fast neutron fluence (E > 1.0 MeV) and iron atom displacements (dpa) were established on a plant and fuel cycle specific basis. An evaluation of the most recent dosimetry sensor set from Capsule W-83, withdrawn at the end of the fourteenth plant operating cycle, is provided. In addition, to provide an up-to-date database applicable to the Millstone Unit 2 reactor, the sensor set from previously withdrawn capsules were also re-analyzed using the current dosimetry evaluation methodology. The dosimetry updates are presented in Appendix A of this report. Comparisons of the results from these dosimetry evaluations with the analytical predictions served to validate the plant specific neutron transport calculations. These validated calculations subsequently formed the basis for providing projections of the neutron exposure of the reactor pressure vessel for operating periods extending to 54 Effective Full Power Years (EFPY).

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

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

All of the calculations and dosimetry evaluations described in this section and in Appendix A were based on the latest available nuclear cross-section data derived from ENDF/B-VI and made use of the latest available calculational tools. Furthermore, the neutron transport and dosimetry evaluation methodologies follow the guidance and meet the requirements of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."[16] Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996.[17] The specific calculational methods applied are also consistent with those described in WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology."[18]

6.2 Discrete Ordinates Analysis

A plan view of the Millstone Unit 2 reactor geometry at the core midplane is shown in Figure 4-1. Six irradiation capsules attached to the reactor pressure vessel are included in the reactor design that constitutes the reactor vessel surveillance program. The capsules are located at azimuthal angles of 83°, 97°, 263°, 277° (7° from the core cardinal axes), and 104° and 284° (14° from the core cardinal axes) as shown in Figure 4-1. The surveillance capsules reside within surveillance capsule holders that are attached to the pressure vessel cladding. The center radius of the surveillance capsule holder is 85-7/16 inches (nominal). The 98-3/4 inch high surveillance capsule holder contain an 80.72 inch stacked length of surveillance capsule compartments that are positioned axially such that the test specimens are centered on the core midplane.

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

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

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

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation. This synthesis procedure was carried out for each operating cycle at Millstone Unit 2.

For the Millstone Unit 2 transport calculations, the r,θ model depicted in Figure 6-1 was utilized since the reactor is octant symmetric (with the exception of the surveillance capsules). This r,θ model includes the core, the reactor internals, core barrel, thermal shield (through cycle 5), explicit representations of the surveillance capsules at 7° and 14°, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological cladding and shield wall. This r,θ model was utilized in the synthesis procedure to perform the surveillance capsule dosimetry evaluations and subsequent comparisons with calculated results, in addition to calculating the maximum neutron exposure levels at the pressure vessel wall. Note that a variation of this model in which the material composition of the surveillance capsules was redefined as water was utilized to determine the maximum neutron exposure at the pressure vessel wall in octants of the core that do not contain surveillance capsules. In developing this analytical model, nominal design dimensions were employed for the various structural components with two exceptions. Specifically, the radius to the center of the surveillance capsule holder as well as the pressure vessel inner radius (PVIR) were taken from the as-built drawings for the Millstone Unit 2 reactor This was done to account for key differences between the nominal versus as-built dimensions.

Water temperatures, and hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full power operating conditions. The coolant densities were treated on a fuel cycle specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, et cetera. The geometric mesh description of the r,θ reactor model consisted of 134 radial by 68 azimuthal intervals. The mesh size was chosen to assure that proper convergence of the inner iterations was achieved on a point wise basis. The pointwise inner iteration flux convergence criterion utilized in the r,θ calculations was set at a value of 0.001.

The r,z model used for the Millstone Unit 2 calculations (see Figure 6-2) extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation 1-foot below the lower support assembly plate (inlet plenum region) to 1-foot above the fuel alignment plate (outlet plenum region). As in the case of the r,θ model, nominal design dimensions (except for the PVIR as-built dimension) and full power coolant densities were employed in the calculations. In this case, the homogenous core region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel girth ribs located between the core shroud and core barrel regions were also explicitly included in the model. The r,z geometric mesh description of the reactor model consisted of 140 radial by 104 axial intervals. As in the case of the r,θ calculations, mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,z calculations was also set at a value of 0.001.

The one-dimensional radial model used in the synthesis procedure consisted of the same 140 radial mesh intervals included in the r,z model. Thus, radial synthesis factors could be determined on a mesh wise basis throughout the entire geometry.

The core power distributions used in the plant specific transport analysis were taken from the appropriate Millstone Unit 2 fuel cycle designs. The data extracted from the design calculations represented cycle dependent fuel assembly enrichments, burnups, and axial power distributions. Peripheral fuel assembly pin power distributions for cycles 1-9 (non-low leakage) were utilized from cycle 3. Peripheral fuel assembly pin power distributions for cycles 10-15 (low leakage) were obtained from cycle 15, i.e., the current operating fuel cycle. Peripheral fuel assembly pin power distributions for future operating cycles (also low leakage) are based on the cycle 17 anticipated core reload design. This information was used to develop spatial and energy dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of fuel cycle averaged neutron flux, which when multiplied by the appropriate fuel cycle length, generated the incremental fast neutron exposure for each fuel cycle. In constructing these core source distributions, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of individual fuel assemblies. From these assembly dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

All of the transport calculations supporting this analysis were carried out using the DORT discrete ordinates code Version 3.1[19] and the BUGLE-96 cross-section library.[20] The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and angular discretization was modeled with an S_{16} order of angular quadrature.

Energy and space dependent core power distributions, as well as system operating temperatures, were treated on a fuel cycle specific basis.

Selected results from the neutron transport analyses are provided in Tables 6-1 through 6-9. In Table 6-1, the calculated cycle specific exposure rates and integrated exposures, expressed in terms of both neutron fluence (E > 1.0 MeV) and dpa, are given at the radial and azimuthal center of the two surveillance capsule positions (7° and 14°). These results, representative of the axial midplane of the active core, establish the calculated exposure of the surveillance capsules withdrawn to date as well as projected into the future. In Table 6-2, cycle specific maximum integrated neutron exposures, expressed in terms of both neutron fluence (E > 1 0 MeV) and dpa, are given at the pressure vessel clad base metal interface at azimuthal angles of 0°, 15°, 30°, and 45° relative to the core major axis. These values are applicable to the intermediate shell plates and intermediate shell longitudinal welds. Table 6-3 contains comparable results for the intermediate shell to lower shell circumferential weld located approximately 15-15/16 inches below the core midplane. These values are applicable as maximum values for the lower shell plate and lower shell longitudinal welds. Table 6-4 contains cycle specific integrated neutron exposures, expressed in terms of both neutron fluence (E > 1.0 MeV) and dpa, for the upper shell to intermediate shell circumferential weld located approximately 82-1/16 inches above the core midplane where the maximum value, representative of the 0° azimuth, is reported. Table 6-5 lists the fast fluence (E >1.0 MeV) and dpa for the intermediate shell adjacent to the core barrel holes at 270° (View "V") and 230° (View "T"). The View "V" holes are located at an elevation of approximately 56-11/16 inches above the core midplane. The View "T" holes are located at an elevation of approximately 52-11/16 inches above the core midplane. Neutron exposure streaming factors that were applied in cycle 6 through the future projections are 1.01 (fluence) and 1.01 (dpa) for the View "V" holes, and 1.03 (fluence) and 1 02 (dpa) for the View "T" holes. Due to the symmetry in the reactor geometry, each of the intermediate and lower shell plates and their longitudinal welds spanning 120° sectors experience neutron exposure levels characteristic of the 0°, 15°, 30°, and 45° azimuths.

Calculated fluence (E > 1.0 MeV) and dpa data are provided in Tables 6-1 through 6-5. All of the data provided in Tables 6-2 through 6-4 was taken at the axial location of the maximum exposure experienced by each material based on the results of the three-dimensional synthesized neutron exposure evaluations. The data tabulations include plant fuel cycle specific calculated neutron exposures at the end of the fourteenth operating fuel cycle as well as projections for the current operating fuel cycle, i.e., cycle fifteen, and beyond to 32, 48, and 54 effective full power years (EFPY). The reactor power level for fuel cycles one and two was 2560 MWt. The reactor power level for all subsequent fuel cycles, i.e., fuel cycle three through 54 EFPY, was 2700 MWt (even though cycle three operated at 2560 MWt for approximately one month). The projection for fuel cycle fifteen was based on the reactor power level and spatial power distributions for fuel cycle fifteen and an assumed fuel cycle length of 532 EFPD Projections beyond the current operating fuel cycle were based on the assumption that future operation would continue to make use of low leakage fuel management and that the cycle seventeen spatial power distributions would be typical of future operating cycles.

Radial gradient information applicable to fast (E > 1.0 MeV) neutron fluence and dpa are given in Tables 6-6 and 6-7, respectively. The data, based on the maximum cumulative integrated exposures from cycles one through fifteen, are presented on a relative basis for each exposure parameter at several azimuthal locations. Exposure distributions through the vessel wall may be obtained by multiplying the calculated exposure at the vessel inner radius by the gradient data listed in Tables 6-6 and 6-7.

The calculated fast neutron exposures for the in-vessel surveillance capsules withdrawn from the Millstone Unit 2 reactor are provided in Table 6-8. These assigned neutron exposure levels are based on the plant and fuel cycle specific neutron transport calculations performed for the Millstone Unit 2 reactor.

Updated lead factors for the Millstone Unit 2 surveillance capsules are provided in Table 6-9. The capsule lead factor is defined as the ratio of the calculated fluence (E > 1.0 MeV) at the geometric center of the surveillance capsule to the corresponding maximum calculated fluence at the pressure vessel clad/base metal interface. In Table 6-9, the lead factors for the capsules that have been removed from service are based on the calculated fluence values for the irradiation period corresponding to the time of withdrawal for the individual capsules. For the capsules that remain in service, the lead factors correspond to the calculated fluence values at the end of cycle fifteen, the current operating fuel cycle for Millstone Unit 2.

6.3 Neutron Dosimetry

The validity of the calculated neutron exposures previously reported in Section 6.2 is demonstrated by a direct comparison against the measured sensor reaction rates and via a least squares evaluation performed for each of the capsule dosimetry sets. However, since the neutron dosimetry measurement data merely serves to validate the calculated results, only the direct comparison of measured-to-calculated results for the most recent surveillance capsule removed from service is provided in this section of the report. For completeness, the assessment of other measured dosimetry removed to date, 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 W-83, that was withdrawn from Millstone Unit 2 at the end of the fourteenth fuel cycle, is summarized below.

Reaction	Reaction Rates (rps/atom)		M/C
	Measured	Calculated	Ratio
$^{54}Fe(n,p)^{54}Mn$	4.50E-15	4.54E-15	0.99
$^{58}Ni(n,p)^{58}Co$ (Cd)	5.75E-15	5.95E-15	0.97
		Average:	0.98
		% Standard Deviation:	1.4

The measured-to-calculated (M/C) reaction rate ratios for the Capsule W-83 threshold reactions range from 0.97 to 0.99, and the average M/C ratio is 0.98 ± 1.4% (1σ). This direct comparison falls well within the ± 20% criterion specified in Regulatory Guide 1.190; furthermore, it is consistent with the full set of comparisons given in Appendix A for other measured dosimetry removed to date from the Millstone Unit 2 reactor. As a result, these comparisons validate the current analytical results described in Section 6.2 and are deemed applicable for Millstone Unit 2.

6.4 Calculational Uncertainties

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

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

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

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

4 - Comparisons of the plant specific calculations with available capsule dosimetry results from the Millstone Unit 2 surveillance program.

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

The fourth phase of the uncertainty assessment (comparisons with Millstone 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 Millstone 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 Reference 3.

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

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

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

Figure 6-1

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

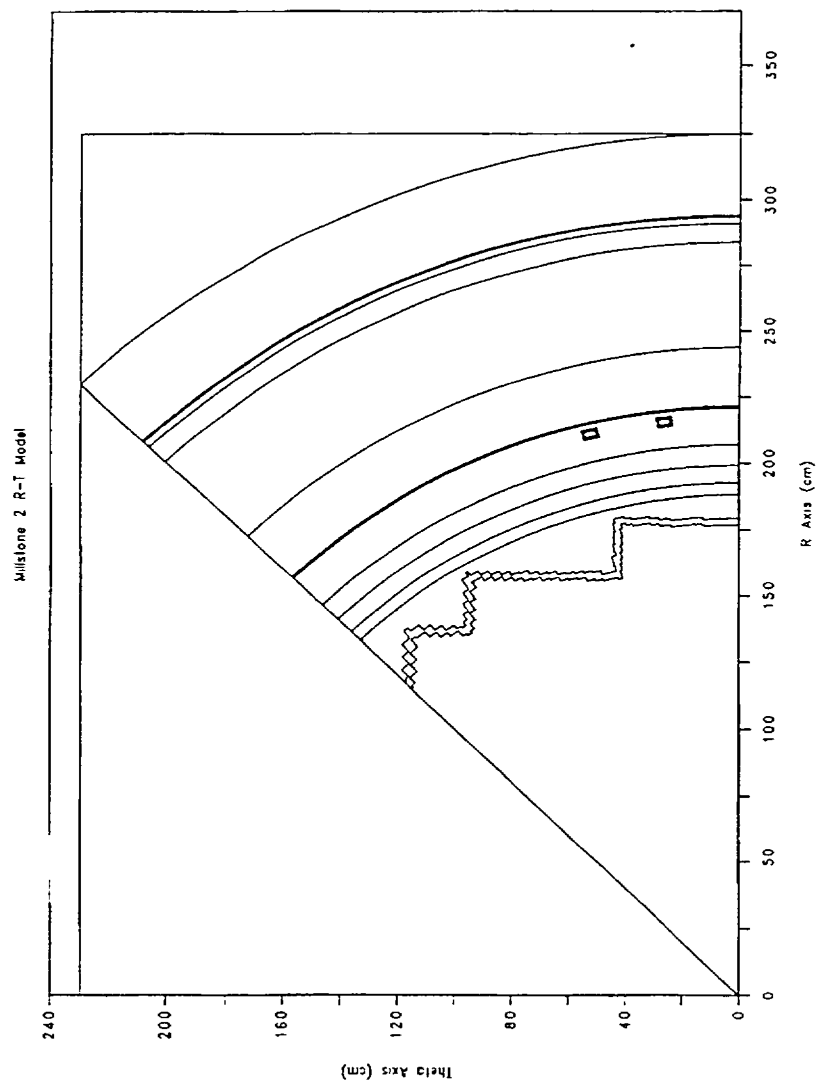


Figure 6-2

Millstone Unit 2 r,z Reactor Geometry

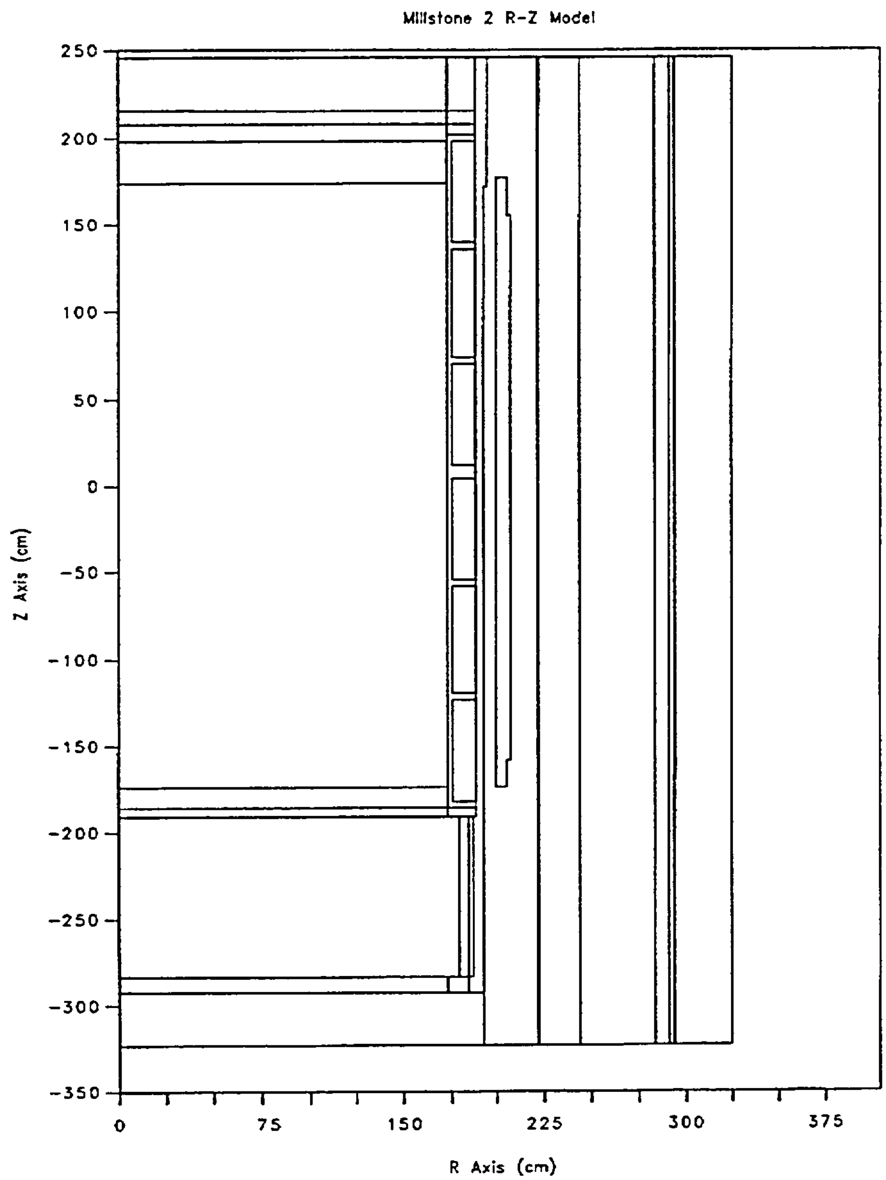


Table 6-1
Calculated Neutron Exposure Rates and Integrated Exposures
At the Surveillance Capsule Center [1]

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Flux (E > 1.0 MeV) [n/cm^2-s] 7°	14°
1	1.3	3.00E+10	2.10E+10
2	2.2	3.35E+10	2.36E+10
3	3.0	3.97E+10	2.80E+10
4	4.0	3.84E+10	2.68E+10
5	5.0	3.79E+10	2.66E+10
6	6.1	5.45E+10	3.90E+10
7	7.1	5.58E+10	4.00E+10
8	8.0	5.59E+10	4.02E+10
9	8.9	5.48E+10	3.94E+10
10	10.0	2.46E+10	1.96E+10
11	11.1	2.81E+10	2.28E+10
12	12.4	2.33E+10	1.88E+10
13	13.7	2.47E+10	2.06E+10
14	15.3	2.46E+10	1.90E+10
Current	16.8	2.65E+10	2.03E+10
Future	32.0	2.54E+10	1.89E+10
Future	48.0	2.54E+10	1.89E+10
Future	54.0	2.54E+10	1.89E+10

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 7°	14°
1	1.3	1.27E+18	8.86E+17
2	2.2	2.13E+18	1.49E+18
3	3.0	3.24E+18	2.28E+18
4	4.0	4.43E+18	3.11E+18
5	5.0	5.63E+18	3.96E+18
6	6.1	7.40E+18	5.22E+18
7	7.1	9.22E+18	6.52E+18
8	8.0	1.09E+19	7.72E+18
9	8.9	1.24E+19	8.78E+18
10	10.0	1.33E+19	9.49E+18
11	11.1	1.42E+19	1.02E+19
12	12.4	1.51E+19	1.10E+19
13	13.7	1.62E+19	1.19E+19
14	15.3	1.74E+19	1.29E+19
Current	16.8	1.87E+19	1.38E+19
Future	32.0	3.08E+19	2.29E+19
Future	48.0	4.37E+19	3.24E+19
Future	54.0	4.85E+19	3.60E+19

[1] Neutron exposure values reported for the surveillance capsules are centered at the core midplane

Table 6-1 cont'd
Calculated Neutron Exposure Rates and Integrated Exposures
At the Surveillance Capsule Center [1]

Cycle	Cumulative Irradiation Time [EFPY]	Iron Atom Displacement Rate [dpa/s]	
		7°	14°
1	1.3	4.59E-11	3.23E-11
2	2.2	5.13E-11	3.63E-11
3	3.0	6.09E-11	4.31E-11
4	4.0	5.90E-11	4.13E-11
5	5.0	5.81E-11	4.10E-11
6	6.1	7.89E-11	5.67E-11
7	7.1	8.07E-11	5.82E-11
8	8.0	8.09E-11	5.85E-11
9	8.9	7.93E-11	5.72E-11
10	10.0	3.57E-11	2.86E-11
11	11.1	4 08E-11	3.32E-11
12	12.4	3.38E-11	2.75E-11
13	13.7	3.59E-11	3.01E-11
14	15.3	3.58E-11	2.78E-11
Current	16.8	3.85E-11	2.97E-11
Future	32.0	3.69E-11	2.76E-11
Future	48.0	3.69E-11	2.76E-11
Future	54.0	3.69E-11	2.76E-11

Cycle	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]	
		7°	14°
1	1.3	1.94E-03	1.36E-03
2	2.2	3.26E-03	2.29E-03
3	3.0	4.96E-03	3.50E-03
4	4.0	6.79E-03	4.78E-03
5	5.0	8.64E-03	6.08E-03
6	6.1	1.12E-02	7.92E-03
7	7.1	1.38E-02	9.81E-03
8	8.0	1.62E-02	1.16E-02
9	8.9	1.84E-02	1.31E-02
10	10.0	1.97E-02	1.41E-02
11	11.1	2.10E-02	1.52E-02
12	12.4	2.24E-02	1.63E-02
13	13.7	2.39E-02	1.76E-02
14	15.3	2.58E-02	1.91E-02
Current	16.8	2.75E-02	2.04E-02
Future	32.0	4.52E-02	3.37E-02
Future	48.0	6.39E-02	4.76E-02
Future	54.0	7.09E-02	5.29E-02

[1] Neutron exposure values reported for the surveillance capsules are centered at the core midplane.

Table 6-2
Calculated Maximum Neutron Exposure at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2] 0°	15°	30°	45°
1	1.3	9.09E+17	5 71E+17	5.13E+17	3.94E+17
2	2.2	1.53E+18	9.69E+17	8.86E+17	6.68E+17
3	3.0	2.32E+18	1.47E+18	1.33E+18	9.70E+17
4	4.0	3.15E+18	1.99E+18	1.78E+18	1.30E+18
5	5.0	4.00E+18	2.52E+18	2.25E+18	1.63E+18
6	6 1	5.39E+18	3.40E+18	3.01E+18	2.23E+18
7	7.1	6.81E+18	4.30E+18	3.77E+18	2 81E+18
8	8.0	8.10E+18	5.13E+18	4 46E+18	3.34E+18
9	8.9	9 26E+18	5 87E+18	5.09E+18	3.83E+18
10	10.0	9.97E+18	6 39E+18	5.63E+18	4.25E+18
11	11.1	1.07E+19	6.92E+18	6.15E+18	4.66E+18
12	12.4	1.14E+19	7.50E+18	6.77E+18	5.15E+18
13	13.7	1.22E+19	8.14E+18	7.45E+18	5.64E+18
14	15 3	1.33E+19	8.88E+18	8.26E+18	6.27E+18
Current	16.8	1.42E+19	9.54E+18	8.87E+18	6.82E+18
Future	32.0	2.40E+19	1.61E+19	1.56E+19	1 29E+19
Future	48.0	3.44E+19	2.30E+19	2.27E+19	1 93E+19
Future	54.0	3 83E+19	2.56E+19	2 53E+19	2.17E+19

Table 6-2 cont'd
Calculated Maximum Neutron Exposure at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]			
		0°	15°	30°	45°
1	1.3	1.44E-03	9.14E-04	8.12E-04	6.28E-04
2	2.2	2.43E-03	1.55E-03	1.40E-03	1.06E-03
3	3.0	3.66E-03	2.34E-03	2.10E-03	1.54E-03
4	4.0	4.99E-03	3.17E-03	2.82E-03	2.08E-03
5	5.0	6.34E-03	4.04E-03	3.56E-03	2.60E-03
6	6.1	8.45E-03	5.38E-03	4.73E-03	3.52E-03
7	7.1	1.06E-02	6.76E-03	5.88E-03	4.41E-03
8	8.0	1.26E-02	8.02E-03	6.94E-03	5.23E-03
9	8.9	1.44E-02	9.16E-03	7.91E-03	5.99E-03
10	10.0	1.54E-02	9.96E-03	8.72E-03	6.63E-03
11	11.1	1.65E-02	1.08E-02	9.53E-03	7.28E-03
12	12.4	1.77E-02	1.17E-02	1.05E-02	8.03E-03
13	13.7	1.89E-02	1.26E-02	1.15E-02	8.79E-03
14	15.3	2.05E-02	1.38E-02	1.28E-02	9.77E-03
Current	16.8	2.19E-02	1.48E-02	1.37E-02	1.06E-02
Future	32.0	3.69E-02	2.49E-02	2.40E-02	2.00E-02
Future	48.0	5.28E-02	3.54E-02	3.48E-02	2.99E-02
Future	54.0	5.87E-02	3.94E-02	3.89E-02	3.36E-02

Table 6-3
Calculated Neutron Exposure of the Intermediate Shell to Lower Shell Circumferential Weld at the Clad/Base Metal Interface

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
		0°	15°	30°	45°
1	1.3	8.83E+17	5.55E+17	4.98E+17	3.83E+17
2	2.2	1.50E+18	9.49E+17	8.67E+17	6.54E+17
3	3.0	2.29E+18	1.45E+18	1.32E+18	9.60E+17
4	4.0	3.12E+18	1.97E+18	1.77E+18	1.29E+18
5	5.0	3.96E+18	2.50E+18	2.23E+18	1.62E+18
6	6.1	5.34E+18	3.37E+18	2.99E+18	2.21E+18
7	7.1	6.74E+18	4.26E+18	3.73E+18	2.78E+18
8	8.0	8.02E+18	5.08E+18	4.42E+18	3.31E+18
9	8.9	9.17E+18	5.81E+18	5.04E+18	3.79E+18
10	10.0	9.88E+18	6.33E+18	5.57E+18	4.21E+18
11	11.1	1.06E+19	6.86E+18	6.09E+18	4.62E+18
12	12.4	1.13E+19	7.43E+18	6.71E+18	5.10E+18
13	13.7	1.21E+19	8.07E+18	7.38E+18	5.59E+18
14	15.3	1.31E+19	8.76E+18	8.15E+18	6.19E+18
Current	16.8	1.41E+19	9.43E+18	8.76E+18	6.73E+18
Future	32.0	2.38E+19	1.59E+19	1.54E+19	1.28E+19
Future	48.0	3.40E+19	2.27E+19	2.24E+19	1.91E+19
Future	54.0	3.78E+19	2.53E+19	2.50E+19	2.15E+19

Table 6-3 cont'd
Calculated Neutron Exposure of the Intermediate Shell to Lower Shell Circumferential Weld at the Clad/Base Metal Interface

Cycle	Cumulative Irradiation Time [EFPY]	Iron Atom Displacements [dpa]			
		0°	15°	30°	45°
1	1.3	1.40E-03	8.88E-04	7.90E-04	6.10E-04
2	2.2	2.38E-03	1.52E-03	1.37E-03	1.04E-03
3	3.0	3.64E-03	2.32E-03	2.09E-03	1.53E-03
4	4.0	4.95E-03	3.15E-03	2.80E-03	2.06E-03
5	5.0	6.29E-03	4.00E-03	3.53E-03	2.58E-03
6	6.1	8.39E-03	5.34E-03	4.70E-03	3.49E-03
7	7.1	1.05E-02	6.70E-03	5.84E-03	4.37E-03
8	8.0	1.25E-02	7.96E-03	6.88E-03	5.19E-03
9	8.9	1.42E-02	9.08E-03	7.84E-03	5.94E-03
10	10.0	1.53E-02	9.87E-03	8.65E-03	6.58E-03
11	11.1	1.64E-02	1.07E-02	9.45E-03	7.21E-03
12	12.4	1.75E-02	1.16E-02	1.04E-02	7.96E-03
13	13.7	1.87E-02	1.25E-02	1.14E-02	8.72E-03
14	15.3	2.03E-02	1.36E-02	1.26E-02	9.64E-03
Current	16.8	2.17E-02	1.46E-02	1.35E-02	1.05E-02
Future	32.0	3.66E-02	2.46E-02	2.37E-02	1.98E-02
Future	48.0	5.22E-02	3.51E-02	3.45E-02	2.95E-02
Future	54.0	5.80E-02	3.90E-02	3.85E-02	3.32E-02

Table 6-4
Calculated Maximum Neutron Exposure of the Upper Shell to Intermediate Shell Circumferential Weld at the Clad/Base Metal Interface

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Exposure	
		Neutron Fluence (E > 1 0 MeV) [n/cm^2]	Iron Atom Displacements [dpa]
1	1.3	9.44E+16	1.52E-04
2	2 2	1.66E+17	2.67E-04
3	3 0	2.53E+17	4.08E-04
4	4.0	3.45E+17	5 55E-04
5	5.0	4.36E+17	7.02E-04
6	6.1	5 30E+17	8.54E-04
7	7.1	6.27E+17	1.01E-03
8	8.0	7.20E+17	1.16E-03
9	8.9	7.99E+17	1.29E-03
10	10.0	8.42E+17	1 36E-03
11	11.1	8 85E+17	1.43E-03
12	12.4	9 36E+17	1.51E-03
13	13.7	9.90E+17	1.60E-03
14	15.3	1.06E+18	1.72E-03
Current	16.8	1.12E+18	1.81E-03
Future	32 0	1.66E+18	2.69E-03
Future	48.0	2.22E+18	3 61E-03
Future	54.0	2 43E+18	3 96E-03

Table 6-5
Calculated Neutron Exposure at the Pressure Vessel Clad/Base Metal Interface
Adjacent to the Core Barrel Holes

Cycle	Cumulative Irradiation Time [EFPY]	Neutron Exposure			
		Adjacent to CB Holes View "V"[1]		Adjacent to CB Holes View "T"[2]	
		Fluence (E > 1.0 MeV) [n/cm^2]	Iron Atom Displacement [dpa]	Fluence (E > 1.0 MeV) [n/cm^2]	Iron Atom Displacement [dpa]
1	1.3	6.78E+17	1.07E-03	3.43E+17	5.46E-04
2	2.2	1.18E+18	1.87E-03	5.98E+17	9.52E-04
3	3.0	1.79E+18	2.84E-03	8.77E+17	1.40E-03
4	4.0	2.44E+18	3.86E-03	1.18E+18	1.87E-03
5	5.0	3.08E+18	4.87E-03	1.47E+18	2.34E-03
6	6.1	4.12E+18	6.45E-03	1.99E+18	3.13E-03
7	7.1	5.20E+18	8.10E-03	2.50E+18	3.91E-03
8	8.0	6.21E+18	9.63E-03	2.98E+18	4.65E-03
9	8.9	7.08E+18	1.10E-02	3.41E+18	5.30E-03
10	10.0	7.56E+18	1.17E-02	3.75E+18	5.82E-03
11	11.1	8.05E+18	1.24E-02	4.10E+18	6.36E-03
12	12.4	8.61E+18	1.33E-02	4.54E+18	7.02E-03
13	13.7	9.21E+18	1.42E-02	4.98E+18	7.69E-03
14	15.3	1.00E+19	1.54E-02	5.55E+18	8.57E-03
Current	16.8	1.07E+19	1.65E-02	6.02E+18	9.28E-03
Future	32.0	1.75E+19	2.68E-02	1.11E+19	1.70E-02
Future	48.0	2.46E+19	3.77E-02	1.64E+19	2.50E-02
Future	54.0	2.72E+19	4.17E-02	1.84E+19	2.81E-02

[1] CB Holes View "V" are located at an azimuth of ~ 270° and an elevation of ~ 56-11/16 inches above the core midplane. Neutron fluence (E > 1.0 MeV) streaming factor = 1.01. Iron atom displacement streaming factor = 1.01.

[2] CB Holes View "T" are located at an azimuth of ~ 230° and an elevation of ~ 52-11/16 inches above the core midplane. Neutron fluence (E > 1.0 MeV) streaming factor = 1.03. Iron atom displacement streaming factor = 1.02.

Table 6-6
Relative Radial Distribution of Neutron Fluence (E > 1.0 MeV)
Within The Reactor Vessel Wall

RADIUS (cm)	AZIMUTHAL ANGLE			
	0°	15°	30°	45°
221 59	1 000	1.000	1.000	1.000
227.07	0.551	0.551	0.551	0.554
232 55	0.266	0 272	0.264	0.271
238.02	0.122	0.128	0.123	0 126
243.50	0 050	0.057	0.053	0 057

Note: Base Metal Inner Radius = 221.59 cm
Base Metal 1/4T = 227.07 cm
Base Metal 1/2T = 232.55 cm
Base Metal 3/4T = 238.02 cm
Base Metal Outer Radius = 243.50 cm

Table 6-7
Relative Radial Distribution of Iron Atom Displacements (dpa)
Within The Reactor Vessel Wall

RADIUS (cm)	AZIMUTHAL ANGLE			
	0°	15°	30°	45°
221.59	1.000	1.000	1 000	1.000
227.07	0.629	0.636	0 628	0.635
232 55	0.373	0.390	0.373	0.386
238.02	0.214	0.233	0.219	0.231
243.50	0.104	0.124	0 115	0.129

Note: Base Metal Inner Radius = 221.59 cm
Base Metal 1/4T = 227.07 cm
Base Metal 1/2T = 232.55 cm
Base Metal 3/4T = 238.02 cm
Base Metal Outer Radius = 243.50 cm

Table 6-8
Calculated Fast Neutron Exposure of Surveillance Capsules
Withdrawn from Millstone Unit 2

Capsule	Cumulative Irradiation Time [EFPY]	Neutron Fluence (E > 1.0 MeV) [n/cm^2]	Iron Atom Displacements [dpa]
W-97	3.0	3.24E+18	4.96E-03
W-97 Supplemental	5.0	7.62E+18	1.10E-02
W-104	10.0	9.49E+18	1.41E-02
W-83	15.3	1.74E+19	2.58E-02

Table 6-9
Calculated Surveillance Capsule Lead Factors

Capsule ID And Location	Status	Lead Factor [1]
W-97 (7°)	Withdrawn EOC3	1.40
W-97 Supplemental (7°)	Inserted BOC6 – Withdrawn EOC10	1.28
W-104 (14°)	Withdrawn EOC10	0.95
W-83 (7°)	Withdrawn EOC14	1.31
W-263 (7°)	In Reactor	1.31
W-277 (7°)	In Reactor	1.31
W-284 (14°)	In Reactor	0.97

[1] Lead factors for capsules remaining in the reactor are based on cycle specific exposure calculations through the current operating fuel reload. i e . Cycle 15.

7 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedule meets the requirements of ASTM E185-82 and is recommended for future capsules to be removed from the Millstone Unit 2 reactor vessel.

Table 7-1

Millstone Unit 2 Reactor Vessel Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Lead Factor	Withdrawal EFPY[(a)]	Fluence (n/cm^2)
W-97	7	1.40	3.0	3.24×10^{18}
W-104	14	0.95	10.0	9.49×10^{18}
W-83	7	1.31	15.3	1.74×10^{19}
W-263	7	1.31[(c)]	EOL[(b)]	See Note (b)
W-277	7	1 31[(c)]	Standby	- -
W-284	14	0.97	Standby	- -

Notes

(a) Effective Full Power Years (EFPY) from Plant Startup.

(b) Capsule W-263 should be removed before it receives a fluence of 4.80×10^{19} n/cm^2 (E>1.0 Mev) (i e. twice the peak vessel EOL surface fluence of 2.40×10^{19} n/cm^2 (E>1.0 MeV)). Capsule W-263 will reach a fluence of approximately 2.40×10^{19} n/cm^2 (E>1.0 MeV) at 23.2 EFPY. This is equal to the reactor vessel peak surface fluence of 2.40×10^{19} (E>1.0 MeV) at 32 EFPY.

(c) The schedule above has Capsule W-263 as the EOL Capsule, however, Capsule W-277 could be withdrawn in place of W-283 since they have the same Lead Factor.

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