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ANALYSIS OF CAPSULE U FROM THE GEORGIA POWER COMPANY VOGTLE UNIT 1 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

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PREFACE

This report has been technically reviewed and verified.

	Reviewer
Sections 1 through 5 and 7, 8	N. K. Ray
Section 6	E. P. Lippincott

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SECTION 1
SUMMARY OF RESULTS

The analysis of the reactor vessel material contained in Capsule U, the first surveillance capsule to be removed from the Georgia Power Company Vogtle Unit 1 reactor pressure vessel, resulted in the following conclusions:

- o The capsule received an average fast neutron fluence (E > 1.0 MeV) of 3.41×10^{18} n/cm^2.

- o Irradiation of the reactor vessel intermediate shell Plate B8805-3, to 3.41×10^{18} n/cm, resulted in no 30 and 50 ft-lb transition temperature increase for specimens oriented normal to the major working direction (transverse orientation) and a 15°F increase for specimens oriented parallel to the major working direction (longitudinal orientation).

- o Weld metal irradiated to 3.41×10^{18} n/cm^2 experienced a 15°F increase in the 30 and 50 ft-lb transition temperature.

- o Irradiation to 3.41×10^{18} n/cm^2 resulted in no decrease in the average upper shelf energy of Plate B8805-3 (transverse orientation) or the weld metal. Both materials exhibit a more than adequate upper shelf level for continued safe plant operation.

- o Comparison of the 30 ft-lb transition temperature increases for the Vogtle Unit 1 surveillance material with predicted increases using the methods of NRC Regulatory Guide 1.99, Revision 2, demonstrated that the Plate B8805-3 material and weld metal transition temperature increases were less than predicted.

- o Surveillance capsule test results for reactor vessel intermediate shell Plate B8805-3 and vessel core region weld metal indicate that these materials are not highly sensitive to irradiation at neutron fluences up to 3.41×10^{18} n/cm^2.

SECTION 2
INTRODUCTION

This report presents the results of the examination of Capsule U, the first capsule to be removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Georgia Power Company Vogtle Unit 1 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Georgia Power Company Vogtle Unit 1 reactor pressure vessel materials was designed and recommended by the Westinghouse Electric Corporation. A description of the surveillance program and the preirradiation mechanical properties of the reactor vessel materials are presented by Singer.[1] The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels". Westinghouse Electric Corporation personnel performed the postirradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens.

This report summarizes testing and the postirradiation data obtained from surveillance Capsule U removed from the Georgia Power Company Vogtle Unit 1 reactor vessel and discusses the analysis of the data.

SECTION 3
BACKGROUND

The ability of the large steel pressure vessel, which contains 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 Vogtle Unit 1 reactor pressure vessel beltline) are well documented in industry literature. Generally, low alloy ferritic materials demonstrate an increase in hardness and tensile properties and a decrease in ductility and toughness under certain conditions of irradiation.

A method for performing analyses to guard against fast fracture in reactor pressure vessels has been presented in "Protection Against Non-ductile Failure," Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature (RT_{NDT}).

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) 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 material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

RT_{NDT} and 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 or changes in mechanical properties of a given reactor pressure vessel steel can be monitored by a reactor surveillance program such as the Vogtle Unit 1 Reactor Vessel Radiation Surveillance Program.[1] A surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is used to index the material to the K_{IR} curve and to set operating limits for the nuclear power plant which reflect the effects of irradiation on the reactor vessel materials.

SECTION 4
DESCRIPTION OF PROGRAM

Six surveillance capsules for monitoring the effects of neutron exposure on the Vogtle Unit 1 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 neutron shield pads and the vessel wall at locations shown in Figure 4-1. The vertical center of the capsules is opposite the vertical center of the core.

Capsule U (Figure 4-2) was removed after 1.14 effective full power years of plant operation. This capsule contained Charpy V-notch impact, tensile, and 1/2T - Compact Tension fracture mechanics specimens from the reactor vessel intermediate shell Plate B8805-3, submerged arc weld metal identical to that used for the beltline region girth and longitudinal weld seams of the reactor vessel and Charpy V-notch specimens from weld heat-affected zone (HAZ) material. All heat-affected zone specimens were obtained from within the HAZ of Plate B8805-3 of the representative weld.

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. In addition, a chemical analysis using Inductively Coupled Plasma Spectrometry (ICPS) was performed on irradiated Charpy specimens from the intermediate shell Plate B8805-3 and weld metal and is reported in Table 4-1.

All test specimens were machined from the 1/4 thickness location of the plate. Test specimens represent material taken at least one plate thickness from the quenched end of the plate. All base metal 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. The 1/2T Compact Tension (CT) test specimens in Capsule U were machined such that the simulated crack in the specimen would propagate normal and parallel to the major working direction for the plate specimens and parallel to the weld direction for weld specimens. All specimens were fatigue precracked per ASTM E399-70T.

Capsule U contained dosimeter wires of pure iron, copper, nickel, and unshielded aluminum-cobalt. In addition, cadmium-shielded dosimeters of Neptunium (Np^{237}) and Uranium (U^{238}) were contained in the capsule.

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

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

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

TABLE 4-1

CHEMICAL COMPOSITION OF
THE VOGTLE UNIT 1 REACTOR VESSEL
SURVEILLANCE MATERIALS

Element	Plate B8805-3 (Wt. %)			Weld Metal[c] (Wt. %)			
C	0.25	0.22	-	0.14	0.09	0.13	-
S	0.010	0.011	-	0.009	0.009	0.010	-
N_2	0.008	0.006	-	-	0.007	0.004	-
Co	0.009	0.006	0.013 (a)	-	0.01	0.005	0.006 (b)
Cu	0.06	0.058	0.053 (a)	0.03	0.04	0.037	0.035 (b)
Si	0.26	0.28	-	0.16	0.17	0.19	-
Mo	0.53	0.57	0.431 (a)	0.52	0.63	0.61	0.475 (b)
Ni	0.60	0.61	0.586 (a)	-	0.10	0.10	0.091 (b)
Mn	1.32	1.32	1.262 (a)	1.06	1.17	1.15	1.057 (b)
Cr	0.04	0.057	0.049 (a)	-	0.05	0.052	0.044 (b)
V	0.003	<0.002	<0.002 (a)	0.005	0.007	0.003	0.006 (b)
P	0.003	0.017	0.010 (a)	0.007	0.008	0.017	0.008 (b)
Sn	0.017	0.019	-	-	0.003	<0.002	-
Al	0.029	0.030	-	-	0.009	0.002	-
Ti	<0.01	0.004	-	-	<0.01	0.006	-
Pb	<0.001	<0.001	-	-	<0.001	<0.001	-
W	<0.01	<0.01	-	-	0.02	<0.01	-
Zr	<0.001	<0.002	-	-	0.001	<0.002	-
As	0.001	0.003	-	-	0.006	0.004	-
Cb	<0.01	<0.002	-	-	0.01	<0.002	-
B	<0.001	<0.001		-	<0.001	<0.001	-

(a) Analysis performed on irradiated Charpy plate specimen AT-5

(b) Analysis performed on irradiated Charpy weld specimen AW-12

(c) Surveillance weld specimens were made of the same wire and flux as the Intermediate to lower shell girth weld seam (Wire Heat 83653 and Linde 0091 Flux Lot 3536)

TABLE 4-2

HEAT TREATMENT OF THE VOGTLE UNIT 1
REACTOR VESSEL SURVEILLANCE MATERIALS

Material	Temperature (°F)	Time (hr)	Coolant
Intermediate Shell	1575/1625	4	Water quenched
Plate B8805-3	1200/1250	4	Air cooled
	1100/1200	17.5	Furnace cooled
Weld Metal	1100/1200	12.75	Furnace cooled

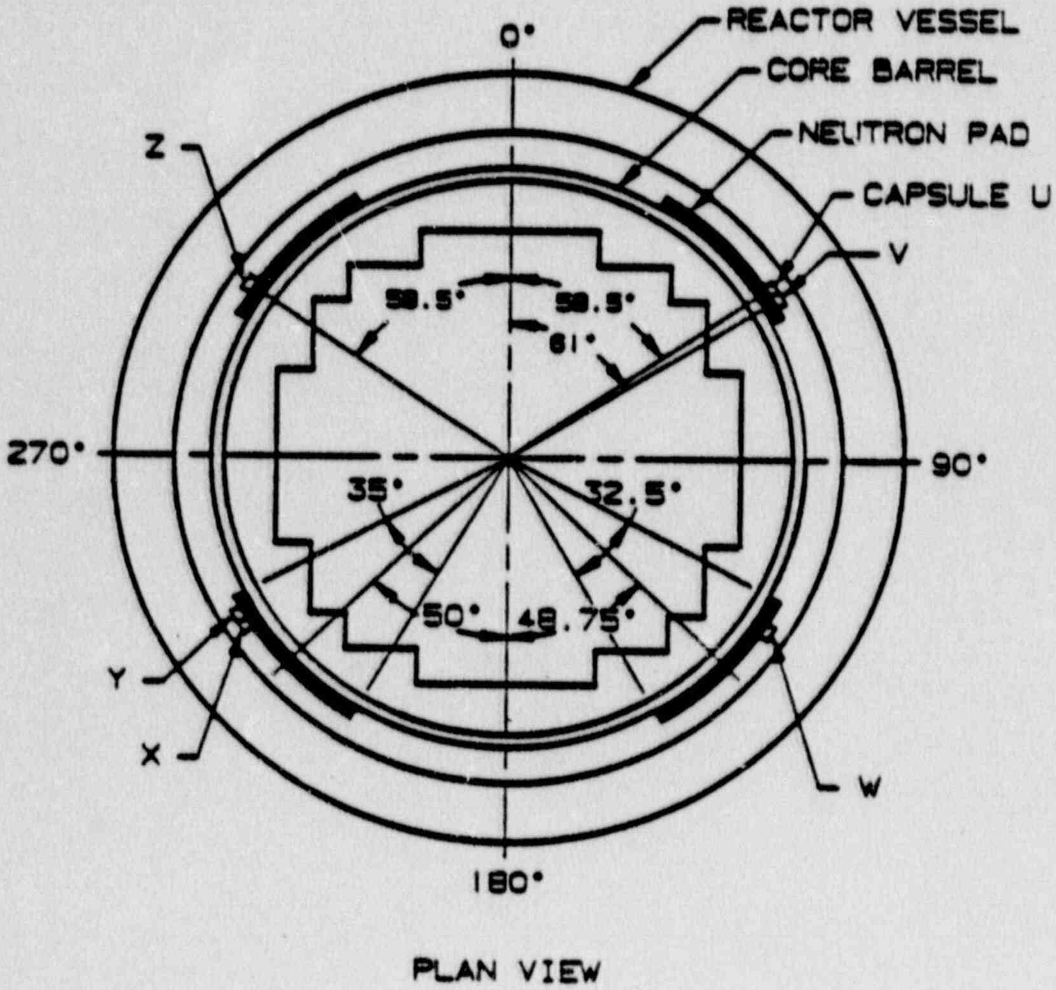
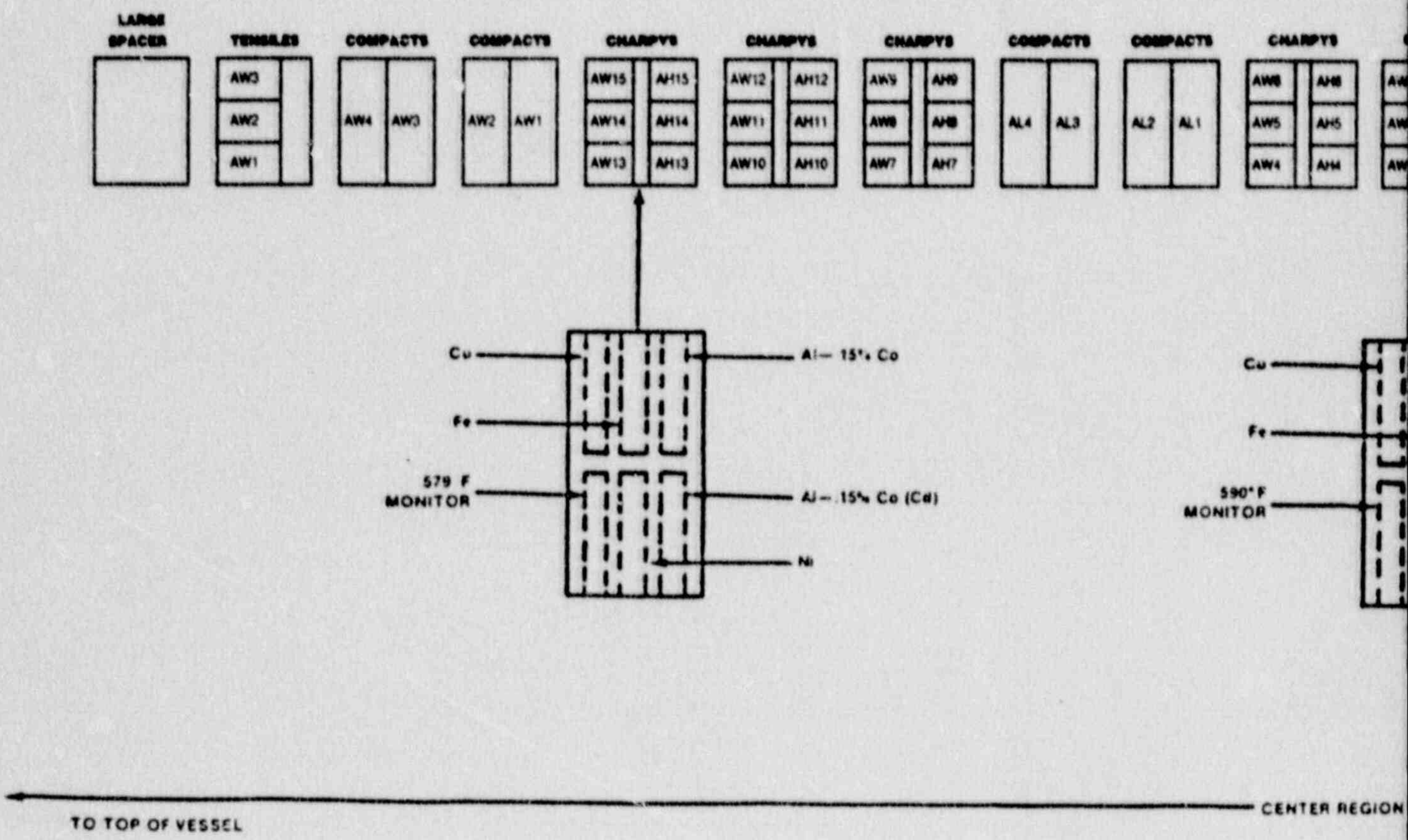


Figure 4-1. Arrangement of Surveillance Capsules in the Vogtle Unit 1 Reactor Vessel

LEGEND: AL - INTERMEDIATE SHELL PLATE B8805-3 (LONGITUDINAL)
AT - INTERMEDIATE SHELL PLATE B8805-3 (TRANSVERSE)
AW - WELD METAL
AH - HEAT-AFFECTED-ZONE MATERIAL



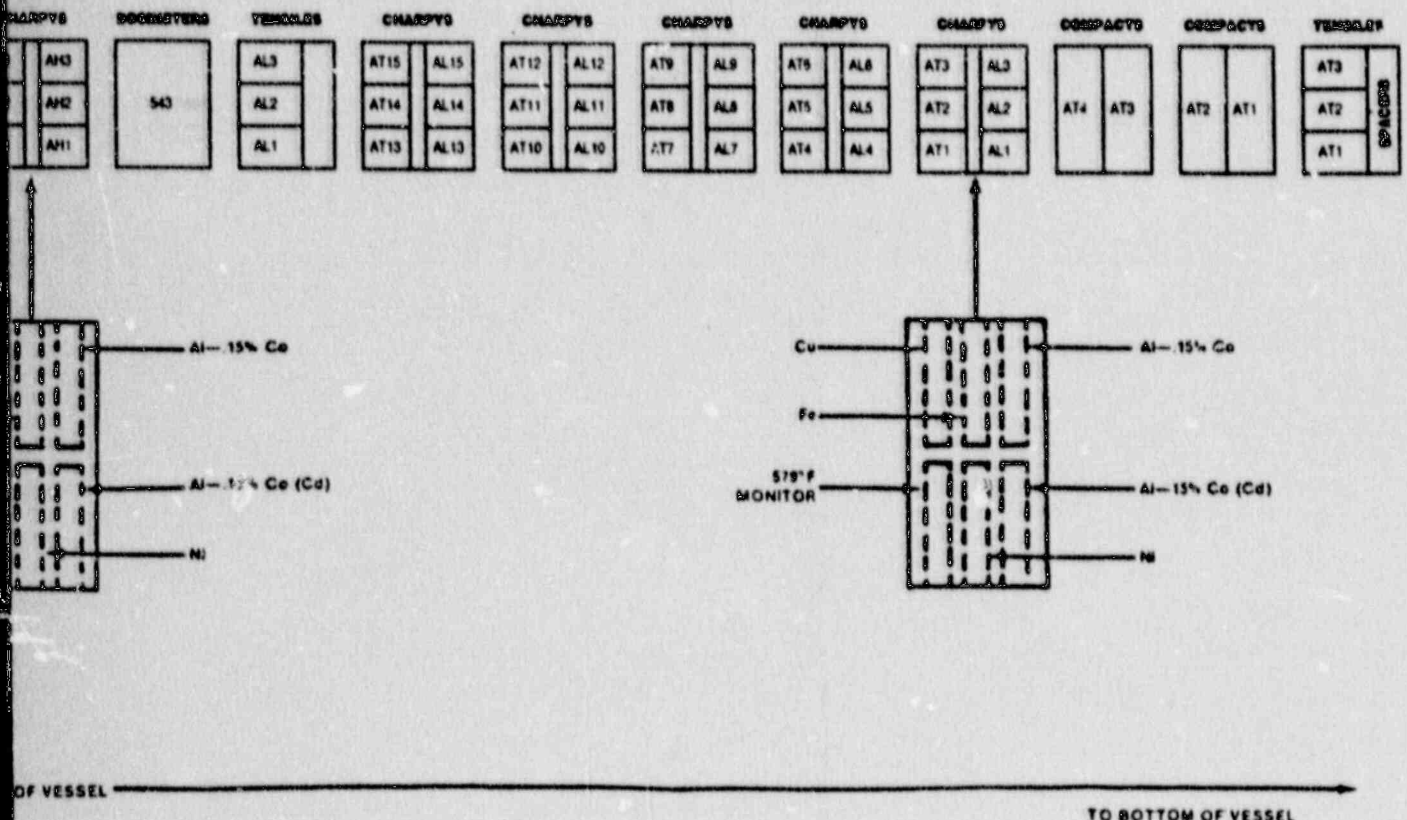


Figure 4-2 Capsule U Diagram Showing Location of Specimens, Thermal Monitors, and Dosimeters

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SECTION 5
TESTING OF SPECIMENS FROM CAPSULE U

5-1. OVERVIEW

The postirradiation mechanical testing of the Charpy V-notch and tensile specimens was performed at the Westinghouse Research and Development Laboratory with consultation by Westinghouse Nuclear Energy Systems personnel. Testing was performed in accordance with 10CFR50, Appendices G and H[2], ASTM Specification E185-82 and Westinghouse Procedure MHL 8402, Revision 1 as modified by Westinghouse RMF Procedures 8102, Revision 1 and 8103, Revision 1.

Upon receipt of the capsule at the laboratory, the specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in WCAP-11011.[1] No discrepancies were found.

Examination of the two low-melting 304°C (579°F) and 310°C (590°F) eutectic alloys indicated no melting of either type of thermal monitor. Based on this examination, the maximum temperature to which the test specimens were exposed was less than 304°C (579°F).

The Charpy impact tests were performed per ASTM Specification E23-82 and RMF Procedure 8103, Revision 1 on a Tinius-Olsen Model 74, 358J machine. The tup (striker) of the Charpy machine is instrumented with an Effects Technology model 500 instrumentation system. 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, 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.

During Charpy impact specimen testing of specimen AL-13 and AH-7 the Effects Technology instrumentation system malfunctioned and therefore no load-time and energy-time signals were recorded. The lack of this instrumented data will have no effect on the final analysis of the Charpy data since instrumented data is only supplied as supplemental test data.

The yield stress (σy) is calculated from the three point bend formula. The flow stress is calculated from the average of the yield and maximum loads, also using the three point bend formula.

Percentage shear was determined from postfracture photographs using the ratio-of-areas methods in compliance with ASTM Specification A370-77. The lateral expansion was measured using a dial gage rig similar to that shown in the same specification.

Tension tests were performed on a 20,000-pound Instron, split-console test machine (Model 1115) per ASTM Specifications E8-83 and E21-79, and RMF Procedure 8102, Revision 1. All pull rods, grips, and pins were made of Inconel 718 hardened to Rc45. 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 inch per minute throughout the test.

Deflection 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 is 1.00 inch. The extensometer is rated as Class B-2 per ASTM E83-67.

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 temperature. 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 indicated that this method is accurate to plus or minus 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 postfracture 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 Charpy V-notch impact tests performed on the various materials contained in Capsule U irradiated to approximately 550°F at 3.41 x 10^{18} n/cm^2 are presented in Tables 5-1 through 5-4 and Figures 5-1 through 5-4. The transition temperature increases and upper shelf energy decreases for the Capsule U material are shown in Table 5-5.

Irradiation of the vessel intermediate shell Plate B8805-3 material (transverse orientation) specimens to 3.41 x 10^{18} n/cm^2 (Figure 5-1) resulted in no 30 and 50 ft-lb transition temperature increase, and an upper shelf energy increase of 2 ft-lb when compared to the unirradiated data.[1]

Irradiation of the vessel intermediate shell Plate B8805-3 material (longitudinal orientation) specimens to 3.41 x 10^{18} m/cm^2 (Figure 5-2) resulted in a 30 and 50 ft-lb transition temperature increase of 15°F and an upper shelf energy decrease of 12 ft-lb when compared to the unirradiated data.

Weld metal irradiated to 3.41×10^{18} n/cm^2 (Figure 5-3) resulted in a 30 and 50 ft-lb transition temperature increase of 15°F, and an upper shelf energy increase of 11 ft-lb.

Weld HAZ metal irradiated to 3.41×10^{18} n/cm^2 (Figure 5-4) resulted in no 30 and 50 ft-lb transition temperature increase and an upper shelf energy decrease of 8 ft-lb.

The fracture appearance of each irradiated Charpy specimen from the various materials is shown in Figures 5-5 through 5-8 and show an increasing ductile or tougher appearance with increasing test temperature.

Table 5-6 shows a comparison of the 30 ft-lb transition temperature (ΔRT_{NDT}) increases for the various Vogtle Unit 1 surveillance materials with predicted increases using the methods of NRC Regulatory Guide 1.99, Revision 2.[3] This comparison shows that the transition temperature increase resulting from irradiation to 3.41×10^{18} n/cm^2 is less than predicted by the Guide for Plate B8805-3. The weld metal transition temperature increase resulting from 3.41×10^{18} n/cm^2 is also less than the Regulatory Guide prediction.

5-3. TENSION TEST RESULTS

The results of tension tests performed on Plate B8805-3 (transverse and longitudinal orientation) and weld metal irradiated to 3.41×10^{18} n/cm^2 are shown in Table 5-7 and Figures 5-9, 5-10 and 5-11, respectively. These results show that irradiation produced a 0.2 percent yield strength increase no greater than 5 ksi for Plate B8805-3 and the weld metal. Fractured tension specimens for each of the materials are shown in Figures 5-12, 5-13 and 5-14. A typical stress-strain curve for the tension specimens is shown in Figure 5-15.

5-4. COMPACT TENSION TESTS

It was decided that the 1/2T - Compact Tension fracture mechanics specimens will not be tested and will be stored at the Hot Cell at the Westinghouse R&D Center, since the U.S. Nuclear Regulatory Commission has recommended that surveillance capsule fracture mechanics specimens not be tested at this time.

TABLE 5-1

CHARPY V-NOTCH IMPACT DATA FOR THE VOGTLE UNIT 1
REACTOR VESSEL SHELL PLATE B8805-3
IRRADIATED AT 550°F, FLUENCE 3.41 x 10^{18} n/cm^2 (E > 1.0 MeV)

Sample No.	Temperature (°F)	Temperature (°C)	Impact Energy (ft-lb)	Impact Energy (J)	Lateral Expansion (mils)	Lateral Expansion (mm)	Shear (%)
			Longitudinal Orientation				
AL1	-75	(-59)	11.0	(15.0)	7.0	(0.18)	5
AL7	-50	(-46)	18.0	(24.5)	12.0	(0.30)	5
AL9	-25	(-32)	14.0	(19.0)	9.0	(0.23)	5
AL6	-10	(-23)	25.0	(34.5)	16.5	(0.42)	15
AL8	0	(-18)	38.0	(51.5)	28.0	(0.71)	20
AL5	0	(-18)	23.0	(31.0)	23.0	(0.58)	15
AL14	10	(-12)	29.0	(39.5)	22.0	(0.56)	20
AL15	25	(- 4)	44.0	(59.5)	32.0	(0.81)	25
AL3	50	(10)	58.0	(78.5)	41.0	(1.04)	35
AL10	72	(22)	88.0	(119.5)	55.0	(1.40)	45
AL11	150	(66)	97.0	(131.5)	70.0	(1.78)	65
AL4	200	(93)	125.0	(169.5)	85.0	(2.16)	100
AL13	250	(121)	136.0	(184.5)	80.0	(2.03)	100
AL12	350	(177)	143.0	(194.0)	81.0	(2.06)	100
AL2	400	(204)	130.0	(176.5)	81.0	(2.06)	100
			Transverse Orientation				
AT4	-75	(-59)	14.0	(19.0)	10.0	(0.25)	5
AT1	-50	(-46)	18.0	(24.5)	12.0	(0.30)	10
AT9	-25	(-32)	15.0	(20.5)	10.0	(0.25)	10
AT11	10	(-12)	32.0	(43.5)	28.0	(0.71)	20
AT5	10	(-12)	33.0	(44.5)	24.0	(0.61)	20
AT3	25	(- 4)	33.0	(44.5)	26.0	(0.66)	20
AT10	25	(- 4)	37.0	(50.0)	30.0	(0.76)	25
AT2	50	(10)	47.0	(63.5)	36.0	(0.91)	30
AT7	50	(10)	49.0	(66.5)	36.0	(0.91)	35
AT13	72	(22)	50.0	(68.0)	36.0	(0.91)	45
AT6	150	(66)	75.0	(101.5)	64.0	(1.63)	95
AT8	200	(93)	93.0	(126.0)	67.0	(1.70)	100
AT15	250	(121)	96.0	(130.0)	68.0	(1.73)	100
AT12	350	(177)	101.0	(137.0)	72.0	(1.83)	100
AT14	400	(204)	103.0	(139.0)	73.0	(1.86)	100

TABLE 5-2

CHARPY V-NOTCH IMPACT DATA FOR THE VOGTLE UNIT 1 REACTOR VESSEL WELD METAL AND HAZ METAL IRRADIATED AT 550°F FLUENCE 3.41 x 10^{18} n/cm^2 (E > 1.0 MeV)

Sample No.	Temperature (°F)	(°C)	Impact Energy (ft-lb)	(J)	Lateral (mils)	Expansion (mm)	Shear (%)
			Weld Metal				
AW15	-100	(-73)	3.0	(4.5)	2.0	(0.05)	2
AW13	- 50	(-46)	12.0	(16.5)	12.0	(0.30)	10
AW12	- 30	(-34)	18.0	(24.5)	15.0	(0.38)	25
AW3	- 30	(-34)	12.0	(16.5)	12.0	(0.30)	20
AW4	- 30	(-34)	17.0	(23.0)	17.0	(0.43)	25
AW9	- 25	(-32)	90.0	(122.0)	66.0	(1.68)	85
AW2	- 25	(-32)	16.0	(21.5)	16.0	(0.41)	25
AW6	- 25	(-32)	70.0	(95.0)	50.0	(1.27)	65
AW1	0	(-18)	52.0	(70.5)	40.0	(1.02)	60
AW7	25	(- 4)	115.0	(156.0)	80.0	(2.03)	95
AW10	72	(22)	142.0	(192.5)	75.0	(1.91)	95
AW8	200	(93)	137.0	(185.5)	85.0	(2.16)	100
AW14	250	(121)	158.0	(214.0)	85.0	(2.16)	100
AW5	350	(177)	147.0	(199.5)	86.0	(2.18)	100
AW11	350	(177)	181.0	(245.5)	80.0	(2.03)	100
			HAZ Metal				
AH6	-125	(-87)	12.0	(16.3)	8.0	(0.20)	10
AH10	- 75	(-59)	27.0	(36.5)	14.0	(0.36)	15
AH15	- 75	(-59)	24.0	(32.5)	16.0	(0.41)	15
AH2	- 75	(-59)	38.0	(51.5)	25.0	(0.64)	30
AH1	- 70	(-57)	43.0	(58.5)	30.0	(0.76)	30
AH4	- 70	(-57)	67.0	(91.0)	41.0	(1.04)	55
AH3	- 60	(-51)	96.0	(130.0)	54.0	(1.37)	70
AH8	- 50	(-46)	59.0	(80.0)	33.0	(0.84)	55
AH14	0	(-18)	62.0	(84.0)	41.0	(1.04)	60
AH5	25	(- 4)	89.0	(120.5)	56.0	(1.42)	85
AH7	72	(22)	128.0	(173.5)	69.0	(1.75)	90
AH11	150	(66)	116.0	(157.5)	58.0	(1.47)	100
AH13	250	(121)	132.0	(179.0)	82.0	(2.08)	100
AH12	250	(121)	195.0	(264.5)	SPECIMEN DID NOT BREAK		
AH9	350	(177)	138.0	(187.0)	76.0	(1.93)	100

TABLE 5-3

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR VOGTLE UNIT 1

REACTOR VESSEL SHELL PLATE B8805-3

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in^2) Charpy Ed/A	Maximum Em/A	Prop Ep/A	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)
Longitudinal Orientation													
AL1	-75	11.0	89	74	14	0.00	85	4.60	330	4.60	-	0	76
AL7	-50	18.0	143	126	19	-7.45	145	4.60	475	4.60	-	-1	76
AL9	-25	14.0	113	92	21	-7.45	125	3.85	385	3.85	-	-1	63
AL6	-10	25.0	201	147	54	0.00	55	4.25	435	4.10	-	0	70
AL5	0	23.0	185	155	31	0.00	70	4.10	475	4.10	-	0	68
AL8	0	38.0	306	228	78	0.00	60	4.40	595	4.35	-	0	72
AL14	10	29.0	234	184	49	0.00	60	4.25	525	4.15	0.15	0	70
AL3	50	58.0	467	270	197	0.00	60	4.25	705	4.15	0.55	0	70
AL10	72	88.0	709	295	413	0.00	50	4.30	725	3.75	1.35	0	71
AL11	150	97.0	781	313	469	-7.45	35	4.05	775	3.05	1.85	-1	67
AL4	200	125.0	1007	274	733	0.00	30	3.95	725	-	-	0	65
AL13	250	136.0	1097	-	-	-	-	-	-	-	-	-	-
AL12	350	143.0	1151	300	851	0.40	5	3.70	770	-	-	13	68
AL2	400	130.0	1047	270	777	2.25	60	3.65	695	-	-	74	97
Transverse Orientation													
AT4	-75	14.0	113	75	38	1.35	20	4.90	285	4.55	-	45	104
AT1	-50	18.0	145	115	30	-7.45	25	4.55	330	4.55	-	-1	75
AT9	-25	15.0	121	69	52	1.70	50	3.85	220	3.80	-	57	92
AT11	10	32.0	285	181	76	1.45	55	4.30	505	4.20	-	47	95
AT5	10	33.0	266	186	79	-7.45	50	4.35	510	4.03	-	-1	72
AT3	25	33.0	266	211	55	1.70	50	4.30	595	4.25	-	56	99
AT10	25	37.0	298	226	72	0.00	40	4.30	580	4.15	-	0	71
AT2	50	47.0	378	279	99	-7.45	75	4.65	700	4.44	0.45	-1	77
AT7	50	49.0	395	222	172	-7.45	70	4.30	610	4.25	0.25	-1	71
AT13	72	50.0	403	228	175	0.00	40	4.40	555	4.35	1.10	0	73
AT6	150	75.0	604	259	345	0.00	70	4.05	695	3.55	1.90	0	67
AT8	200	93.0	749	250	499	0.00	50	3.90	695	-	-	0	64
AT15	250	96.0	773	232	541	0.00	45	3.95	610	-	-	0	65
AT12	350	101.0	813	243	570	0.15	20	3.80	625	-	-	4	65
AT14	400	103.0	829	228	601	1.20	20	3.80	625	-	-	39	82

TABLE 5-4

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR VOGTLE UNIT 1 REACTOR VESSEL WELD METAL AND HAZ METAL

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in^2): Charpy Ed/A	Normalized Energies (ft-lb/in^2): Maximum Em/A	Normalized Energies (ft-lb/in^2): Prop Ep/A	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)
Weld Metal													
AW15	-100	3.0	24	10	14	0.01	15	2.20	75	2.15	-	2	37
AW13	- 50	12.0	97	44	52	-7.45	45	3.90	220	3.80	0.25	-1	64
AW3	- 30	12.0	97	20	76	1.40	50	2.95	160	2.95	1.00	46	71
AW4	- 30	17.0	137	33	104	1.05	15	3.80	145	3.70	0.65	35	81
AW12	- 30	18.0	145	37	108	0.00	40	4.35	195	4.35	1.10	0	71
AW2	- 25	16.0	129	62	66	1.25	15	4.45	300	4.45	1.15	41	94
AW6	- 25	70.0	564	333	231	-7.45	50	4.70	765	4.35	1.20	-1	77
AW9	- 25	90.0	725	311	414	0.00	55	4.30	745	3.45	0.95	0	71
AW1	0	52.0	419	265	154	0.00	30	4.15	650	4.15	1.85	0	69
AW7	25	115.0	926	316	610	3.20	125	4.25	705	2.60	1.10	106	123
AW10	72	142.0	1143	332	811	0.00	45	4.10	800	3.00	2.35	0	68
AW8	200	137.0	1103	325	779	0.00	60	3.70	910	-	-	0	61
AW14	250	158.0	1272	324	949	-7.45	45	3.70	875	-	-	-1	61
AW5	350	147.0	1184	286	897	2.15	60	3.45	765	-	-	71	93
AW11	350	181.0	1457	251	1207	2.45	95	3.50	670	-	-	81	99
HAZ Metal													
AH6	-125	12.0	97	75	22	1.85	45	5.25	255	5.15	0.15	61	117
AH10	-100	27.0	217	176	41	-7.45	55	4.50	430	4.40	-	-1	74
AH15	- 75	24.0	193	158	35	-7.45	35	4.50	410	4.50	0.20	-1	74
AH2	- 75	38.0	306	257	49	1.85	45	5.30	620	5.20	0.80	62	118
AH1	- 70	43.0	346	320	27	-7.45	20	5.35	640	5.30	-	-1	88
AH4	- 70	67.0	540	334	206	2.00	50	5.35	805	4.90	0.25	65	121
AH3	- 60	96.0	773	359	414	-7.45	40	5.15	750	3.85	1.00	-1	85
AH8	- 50	59.0	475	245	230	-7.45	30	4.60	550	4.45	1.20	-1	76
AH14	0	62.0	499	313	186	0.00	55	4.45	750	4.25	1.05	0	74
AH5	25	89.0	717	302	414	1.45	45	4.50	770	3.85	2.7	47	98
AH7	72	128.0	1032	-	-	-	-	-	-	-	-	-	-
AH11	150	116.0	934	260	674	-7.45	10	3.95	645	-	-	-1	65
AH12*	250	195.0	1570	405	1165	3.35	95	4.70	835	-	-	111	133
AH13	250	132.0	1063	315	748	2.50	55	3.95	745	-	-	83	107
AH9	350	138.0	1111	308	803	2.35	75	3.65	790	-	-	77	99

*Specimen AH12 Did Not Break

TABLE 5-5
THE EFFECT OF 550°F IRRADIATION AT 3.41 x 10^{18} n/cm^2 (E > 1.0 MeV) ON THE NOTCH TOUGHNESS PROPERTIES OF THE VOGTLE UNIT 1 REACTOR VESSEL MATERIALS

	Average 30 ft-lb Temp (°F)			Average 35 mil Lateral Expansion Temp (°F)			Average 50 ft-lb Temp (°F)			Average Energy Absorption at Full Shear (ft-lb)		
Material	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	Δ(ft-lb)
Plate B8805-3 (Longitudinal)	-15	0	15	10	25	15	20	35	15	122	134	+12
Plate B8805-3 (Transverse)	-15	15	0	55	55	0	65	65	0	96	98	+2
Weld Metal	-40	-25	15	-35	-20	15	-25	-10	15	145	156	+11
HAZ Metal	-75	-75	0	-45	-45	0	-50	-50	0	136	128	-8

TABLE 5-6

COMPARISON OF VOGTLE UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE CHARPY IMPACT TEST RESULTS
WITH REGULATORY GUIDE 1.99 REVISION 2 PREDICTIONS

Material	Capsule	Fluence $10^{19} n/cm^2$	ΔRT_{NDT} (°F) Meas.	ΔRT_{NDT} (°F) Pred.	USE DECREASE (%) Meas.	USE DECREASE (%) Pred.
Plate B8805-3 (Longitudinal)	U	0.341	15	26	0	15
Plate B8805-3 (Transverse)	U	0.341	0	26	0	15
Weld Metal	U	0.341	15	20	0	15

TABLE 5-7

TENSILE PROPERTIES FOR VOGTLE UNIT 1

REACTOR VESSEL MATERIAL IRRADIATED TO 3.41 x 10^{18} n/cm^2 (E > 1.0 MeV)

Material	Sample Number	Test Temp. (°F)	0.2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Stress (ksi)	Fracture Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Plate	AL3	73	74.9	97.8	3.00	184.7	61.1	9.0	23.0	67
B8805-3	AL2	175	71.3	91.3	2.80	188.6	57.0	10.5	24.0	70
(Long. Orient.)	AL1	550	66.2	92.7	3.28	164.4	66.8	10.5	21.5	59
Plate	AT3	73	76.4	98.8	3.30	268.9	67.2	10.5	24.6	64
B8805-3	AT2	175	70.8	93.7	3.30	186.7	67.2	10.5	21.9	64
(Transv. Orient.)	AT1	550	68.8	94.7	3.55	192.8	72.3	9.8	20.1	51
Weld	AW2	0	76.4	90.7	2.75	213.4	56.0	10.5	23.7	74
	AW3	73	73.8	88.6	2.70	199.4	55.0	10.5	23.7	72
	AW1	550	68.8	90.7	2.85	183.6	58.1	8.3	20.3	68

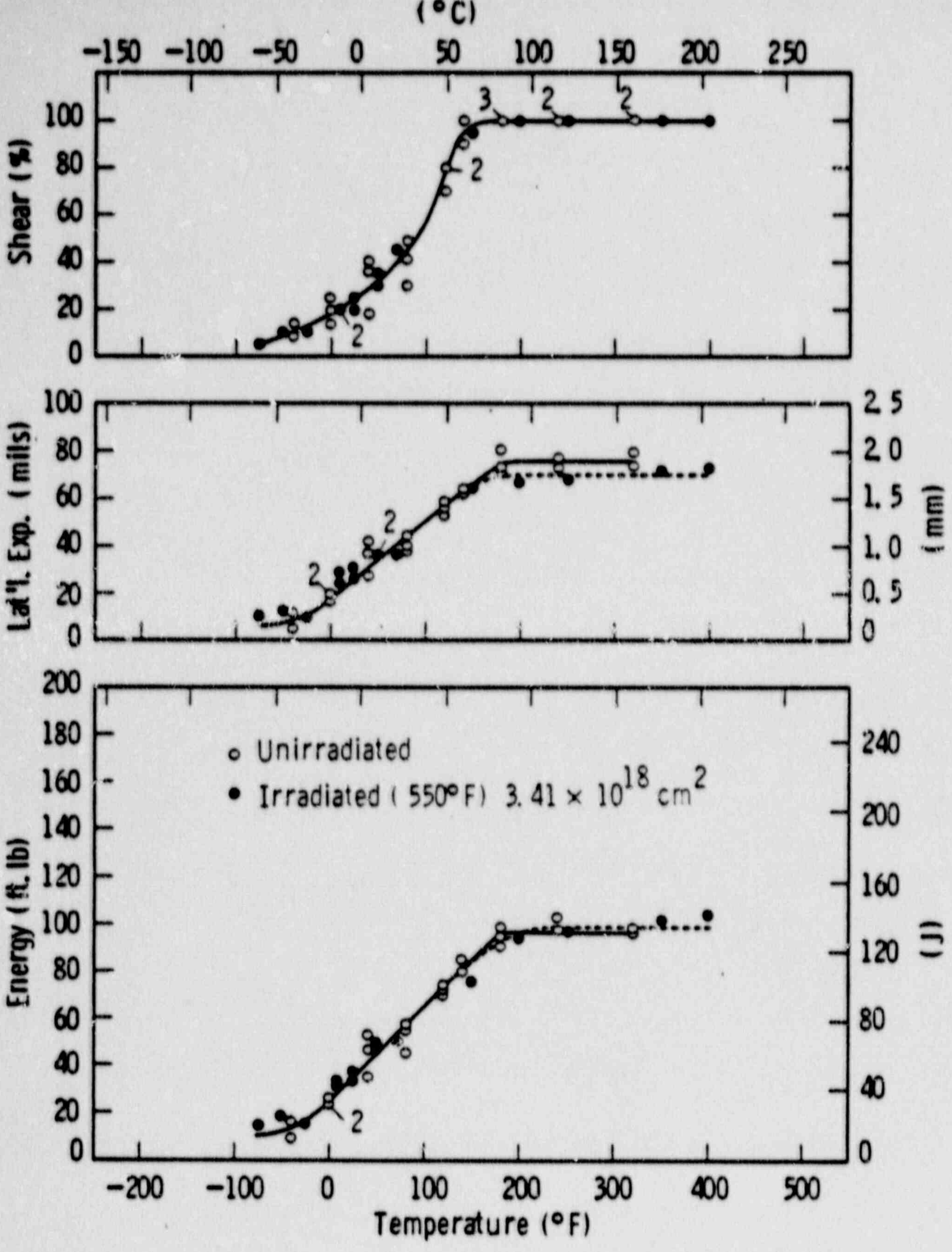

FIGURE 5-1 CHARPY V-NOTCH IMPACT DATA FOR VOGTLE UNIT 1 REACTOR VESSEL SHELL PLATE B8805-3 (TRANSVERSE ORIENTATION)

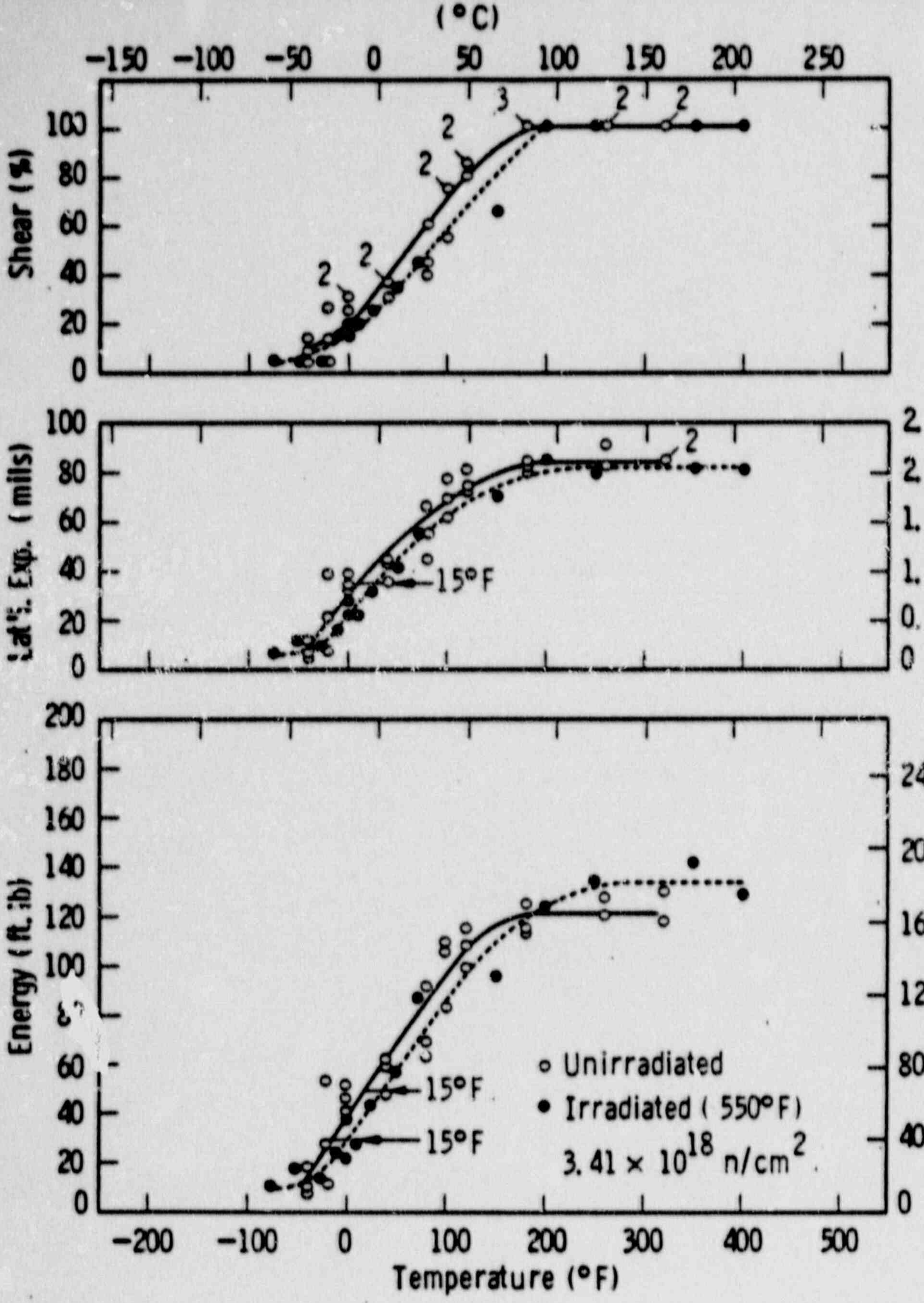

FIGURE 5-2 CHARPY V-NOTCH IMPACT DATA FOR VOGTLE UNIT 1 REACTOR VESSEL SHELL PLATE B8805-3 (LONGITUDINAL ORIENTATION)

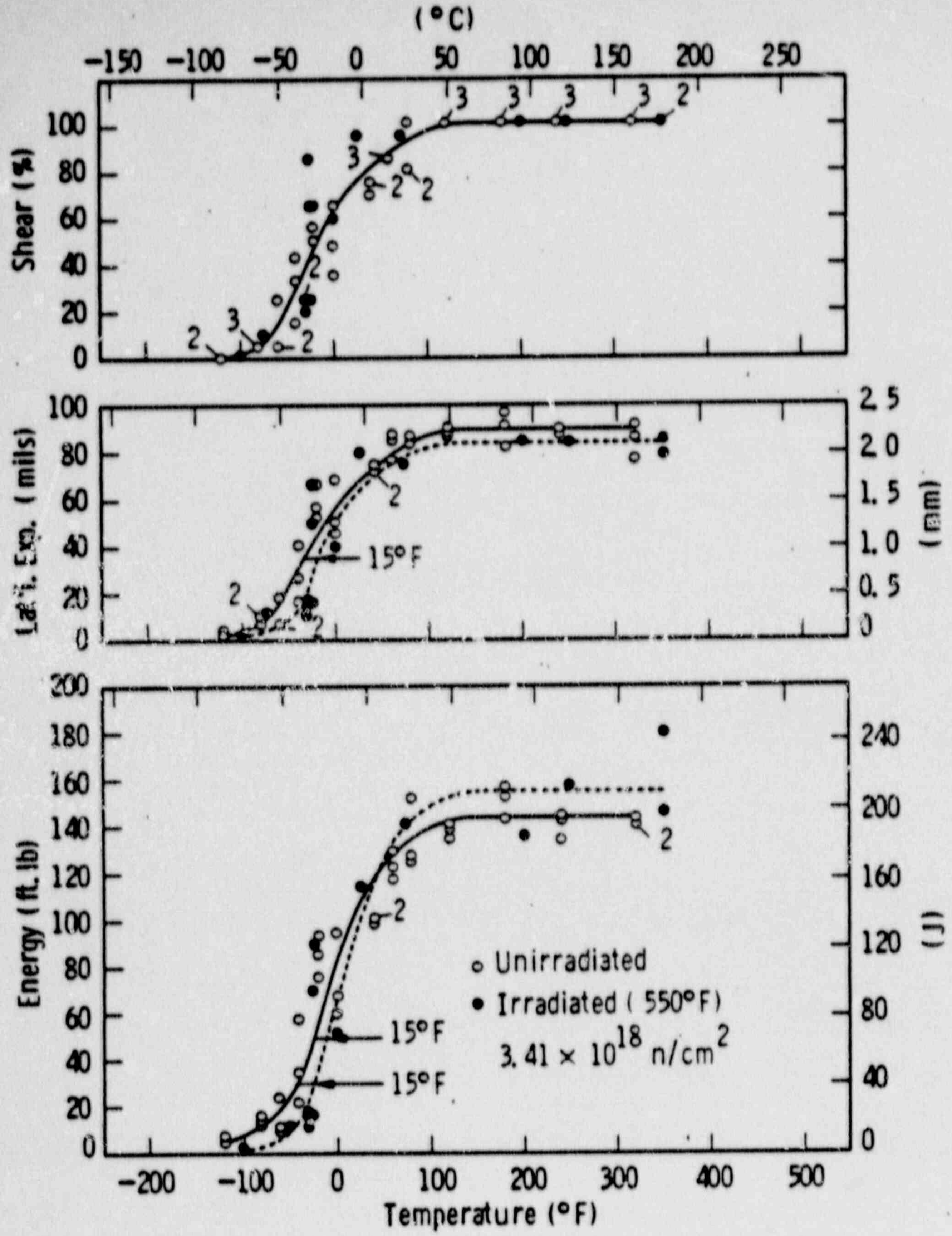

FIGURE 5-3 CHARPY V-NOTCH IMPACT DATA FOR VOGTLE UNIT 1 REACTOR VESSEL WELD METAL

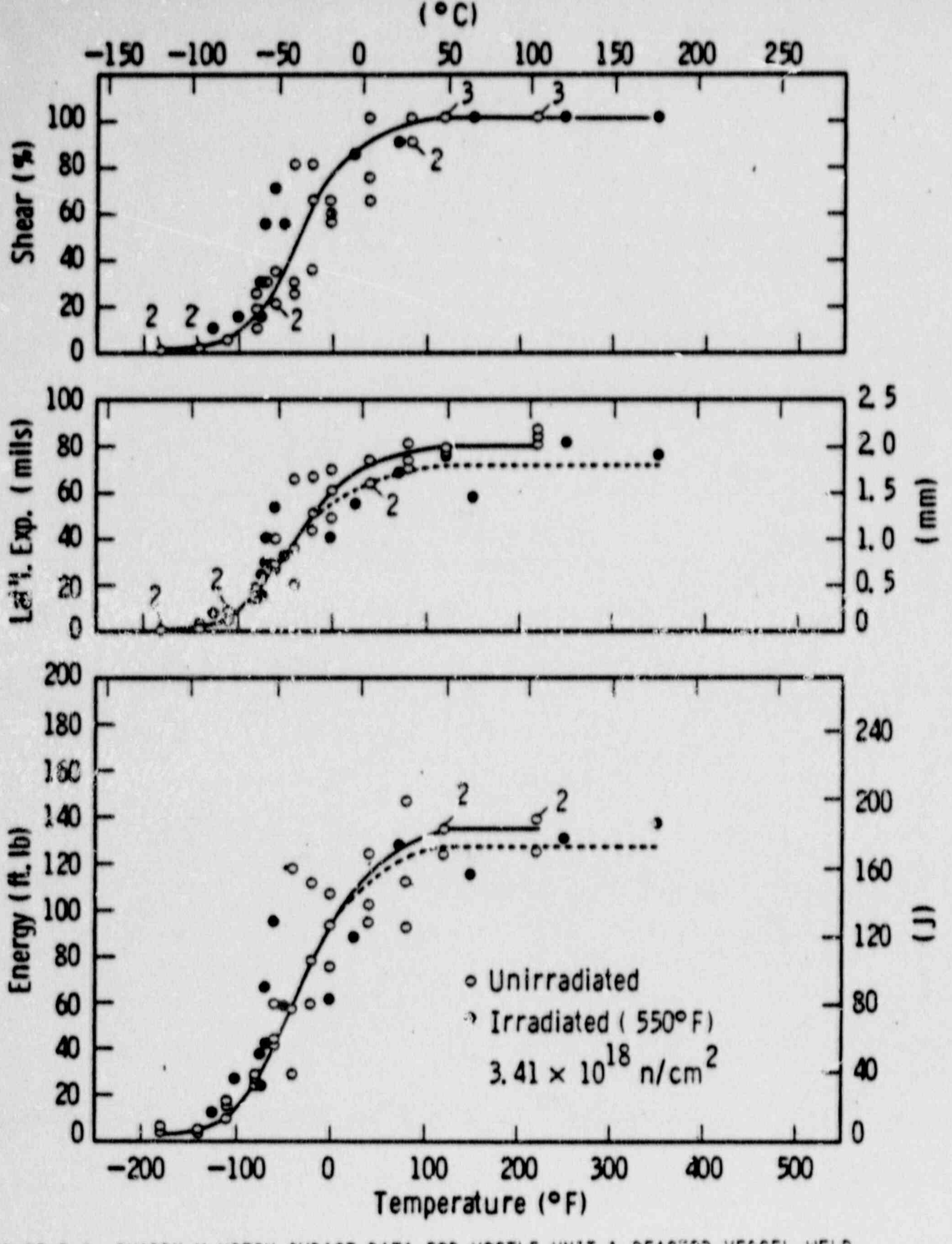

FIGURE 5-4 CHARPY V-NOTCH IMPACT DATA FOR VOGTLE UNIT 1 REACTOR VESSEL WELD HEAT AFFECTED ZONE METAL

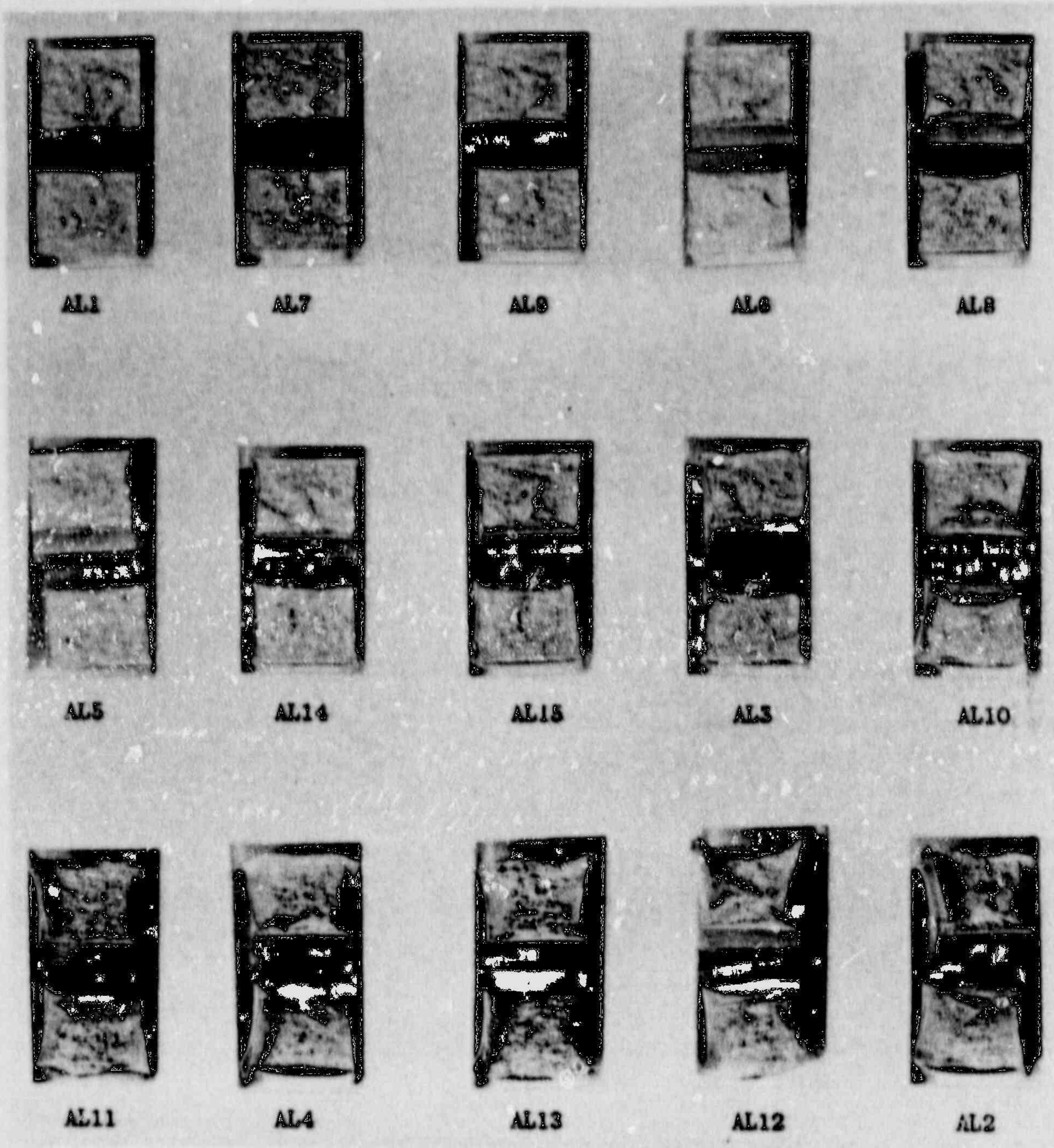

FIGURE 5-5 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR VOGTLE UNIT 1 REACTOR VESSEL SHELL PLATE B8805-3 (LONGITUDINAL ORIENTATION)

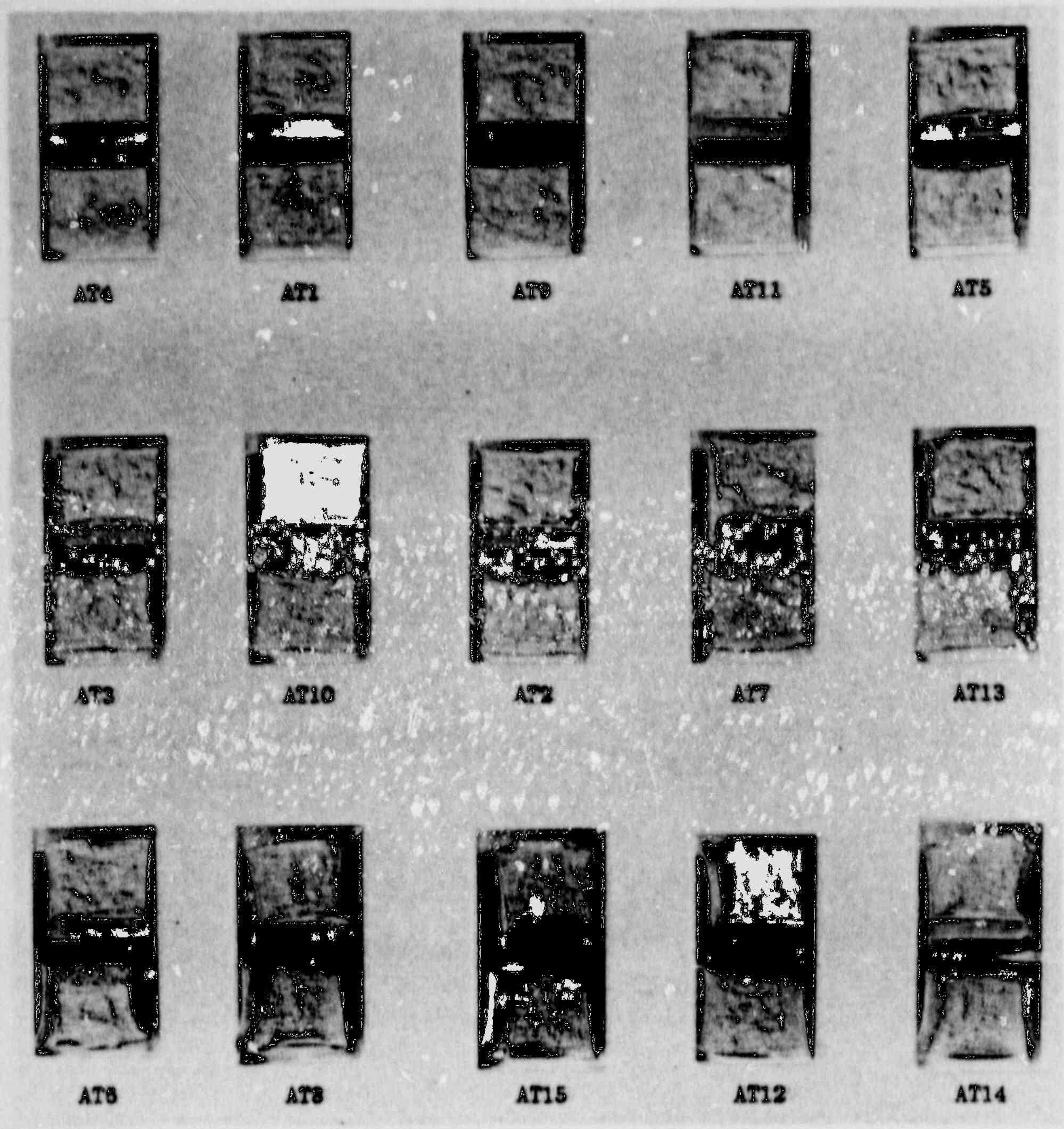

FIGURE 5-6 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR VOGTLE UNIT 1 REACTOR VESSEL SHELL PLATE B8805-3 (TRANSVERSE ORIENTATION)

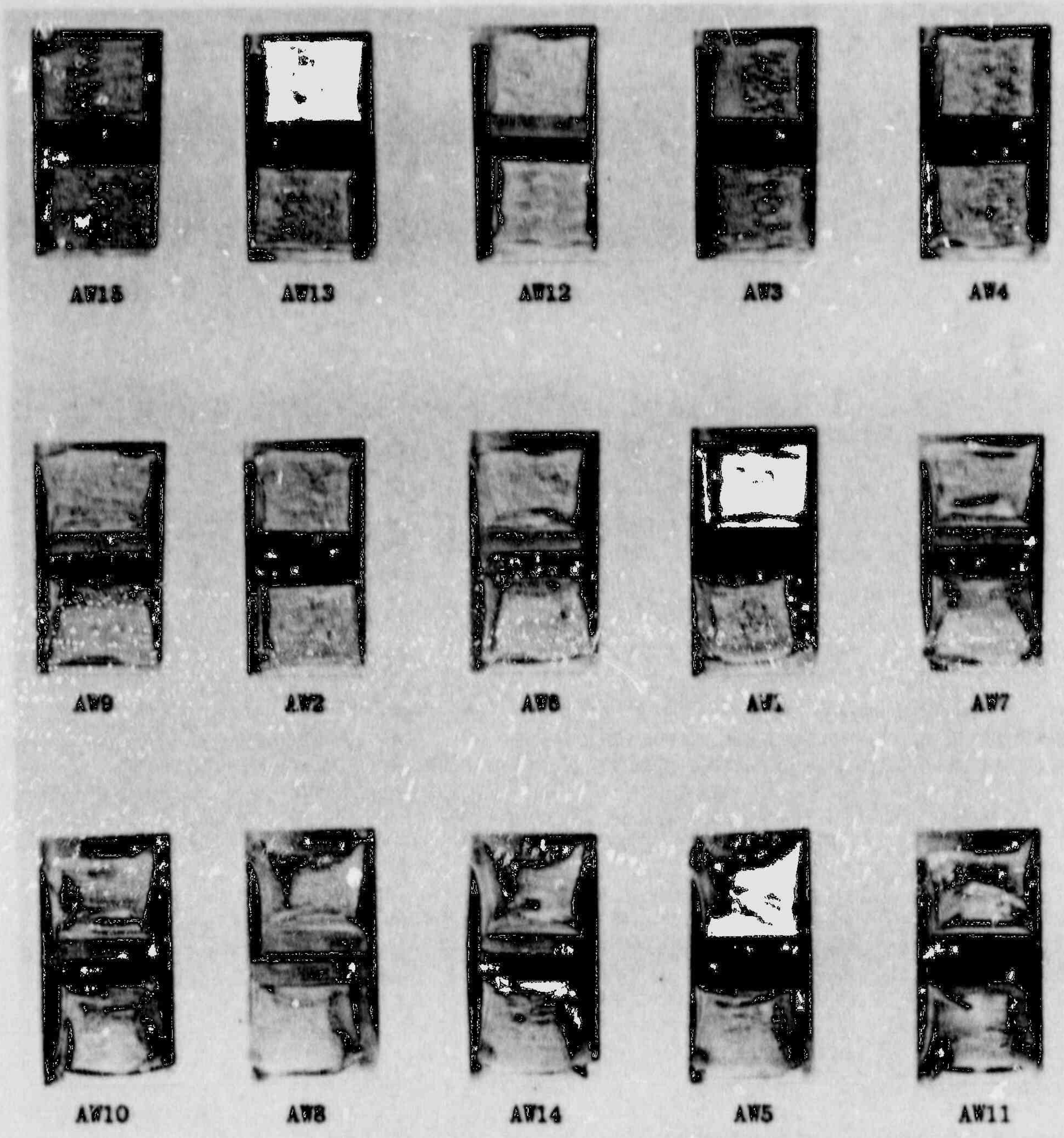

FIGURE 5-7 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR VOGTLE UNIT 1 REACTOR VESSEL WELD METAL

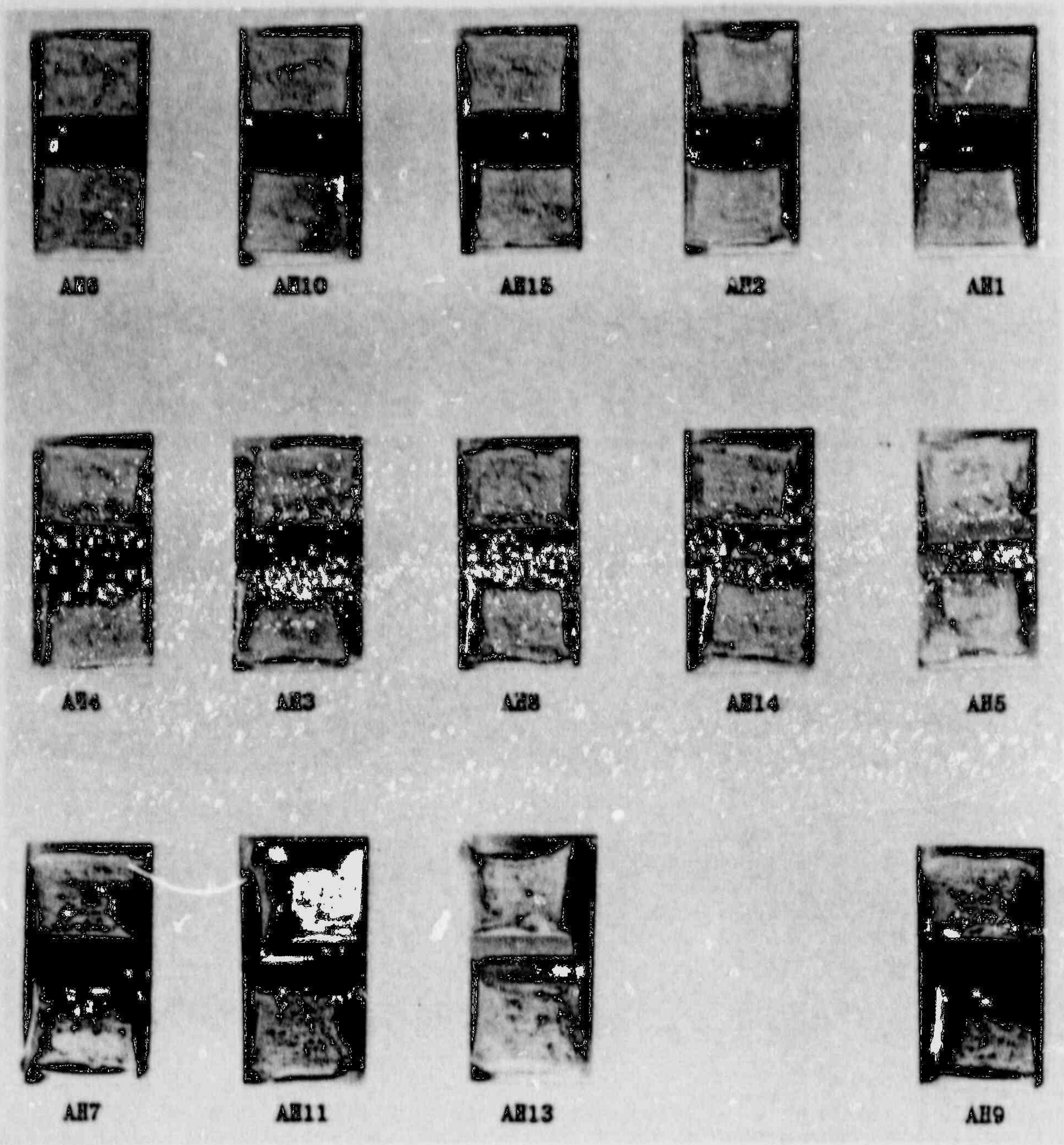

FIGURE 5-8 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR VOGTLE UNIT 1 REACTOR VESSEL WELD HAZ METAL

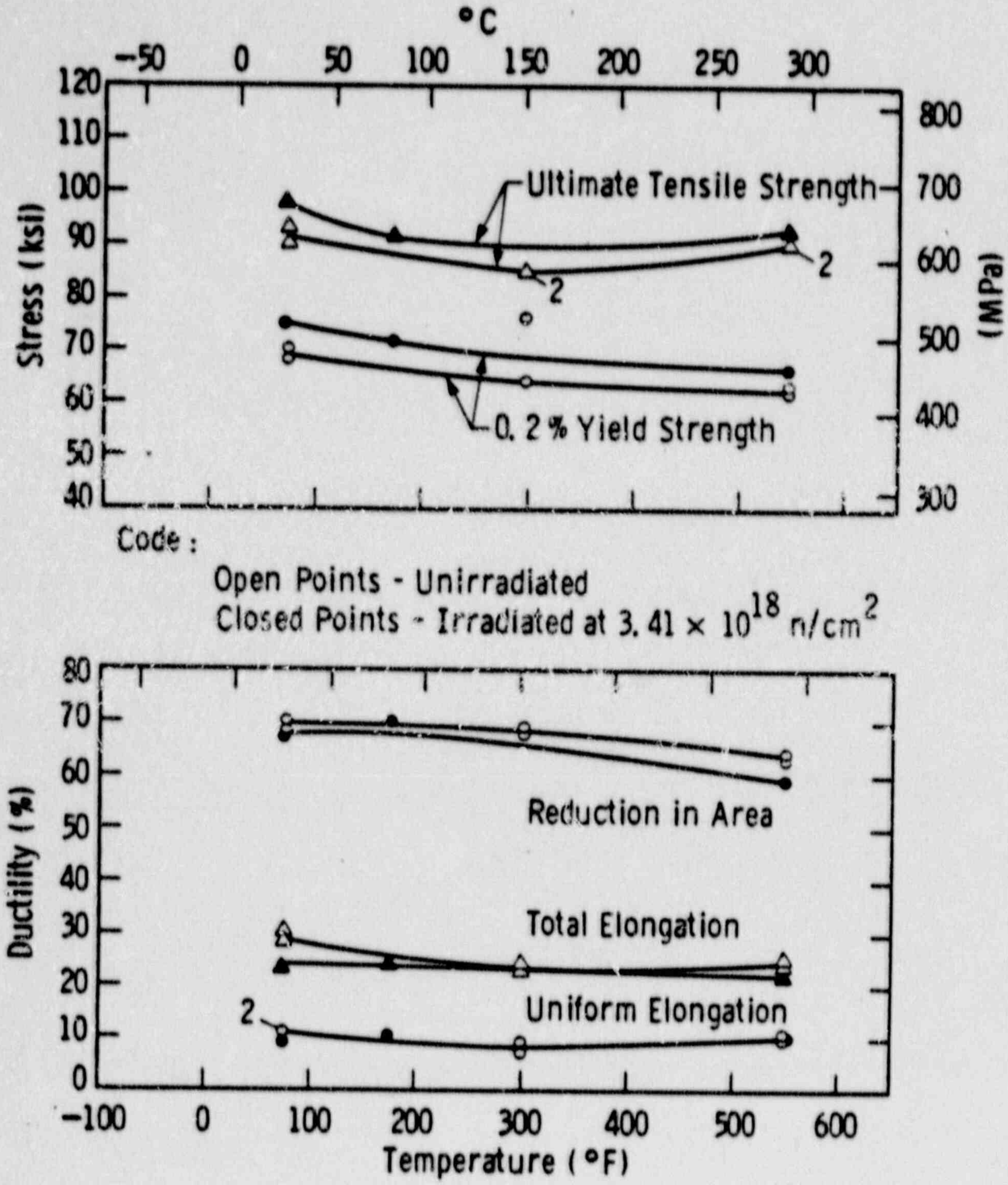

FIGURE 5-9 TENSILE PROPERTIES FOR VOGTLE UNIT 1 REACTOR VESSEL SHELL PLATE B8805-3 (LONGITUDINAL ORIENTATION)

Curve 757141-A

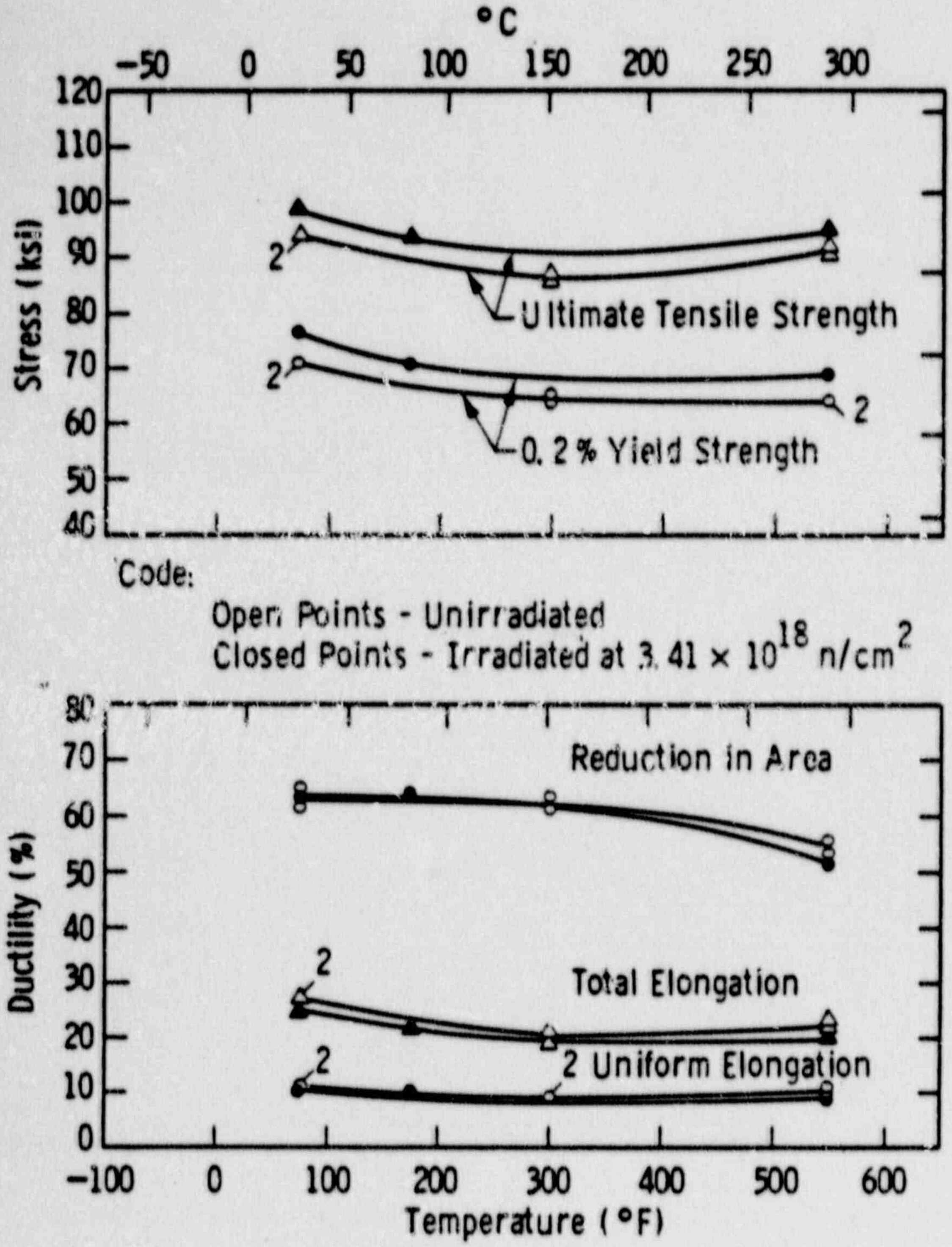

FIGURE 5-10 TENSILE PROPERTIES FOR VOGTLE UNIT 1 REACTOR VESSEL SHELL PLATE B8805-3 (TRANSVERSE ORIENTATION)

Curve 757142-A

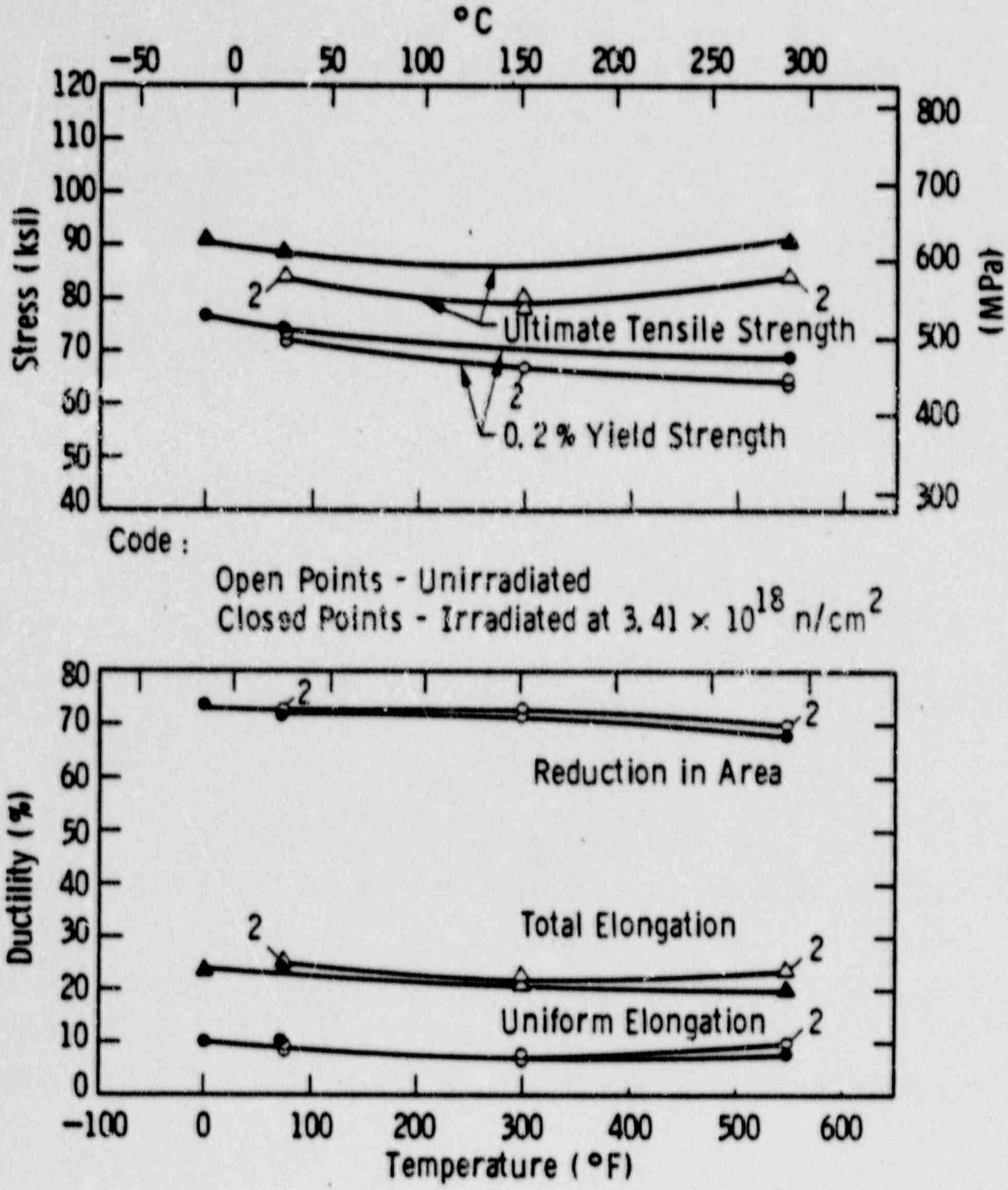

FIGURE 5-11 TENSILE PROPERTIES FOR VOGTLE UNIT 1 REACTOR VESSEL WELD METAL

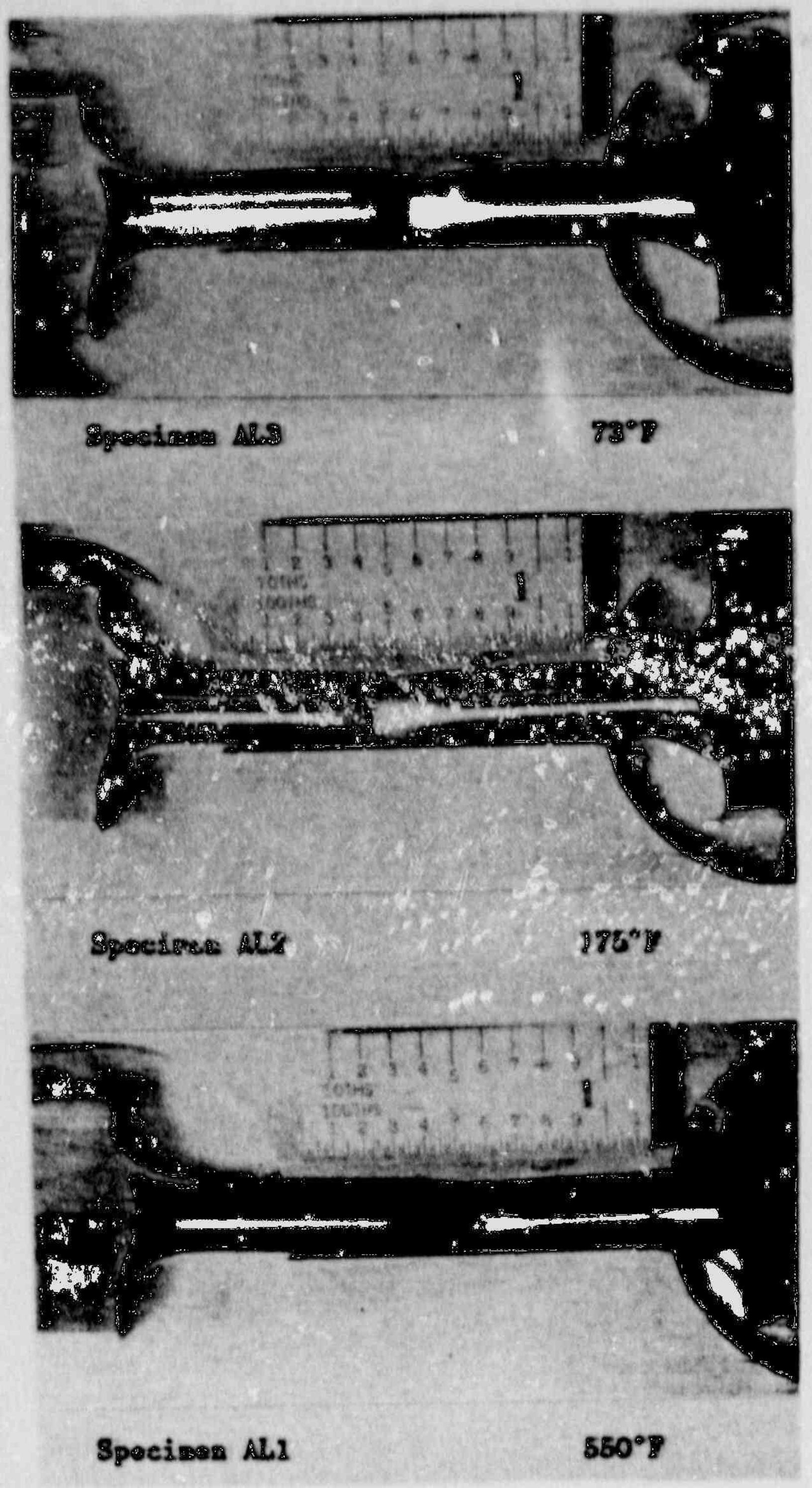

FIGURE 5-12 FRACTURED TENSILE SPECIMENS FOR VOGTLE UNIT 1 REACTOR VESSEL SHELL PLATE B8805-3 (LONGITUDINAL ORIENTATION)

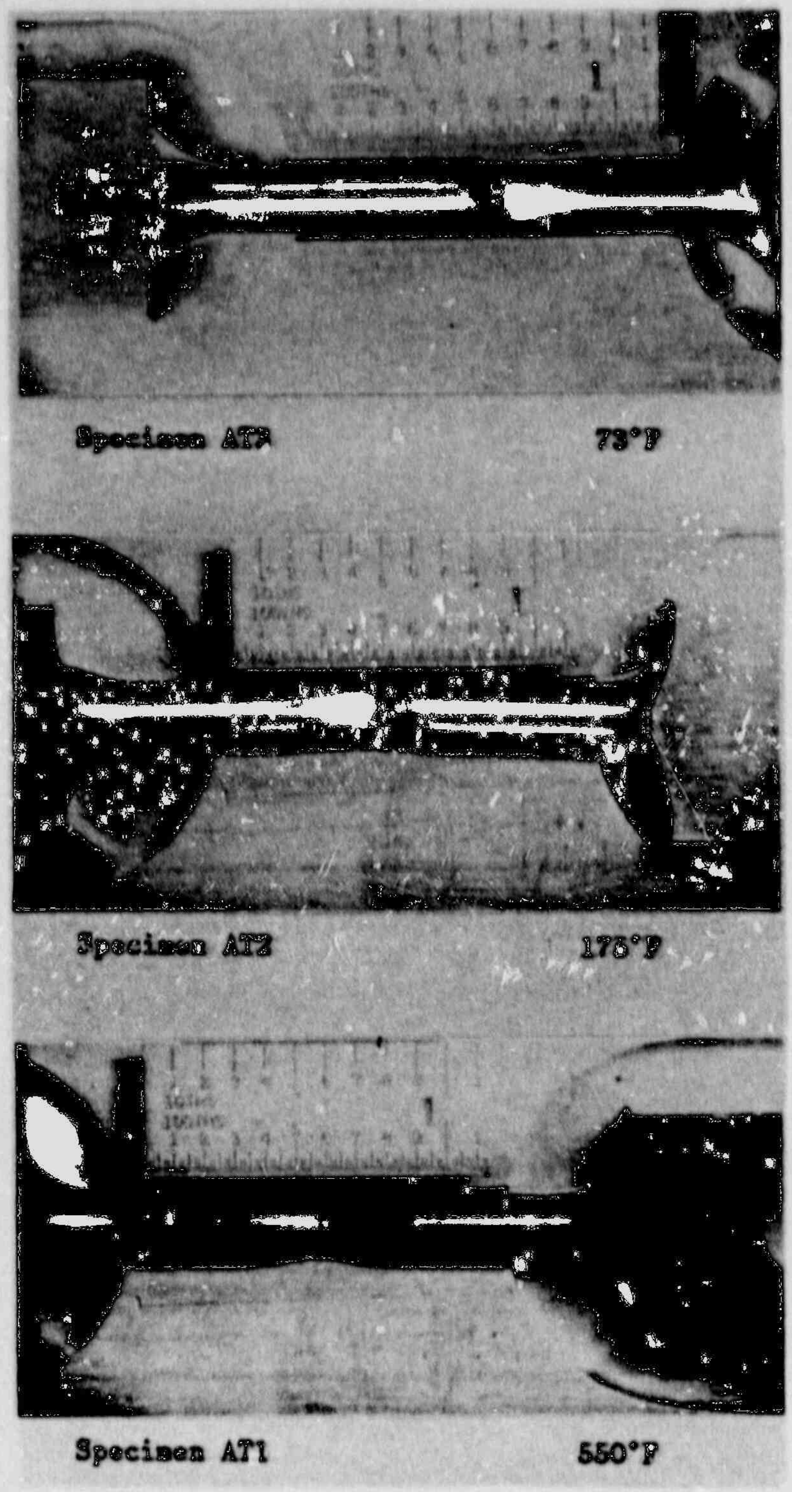

FIGURE 5-13 FRACTURED TENSILE SPECIMENS FOR VOGTLE UNIT 1 REACTOR VESSEL SHELL PLATE B8805-3 (TRANSVERSE ORIENTATION)

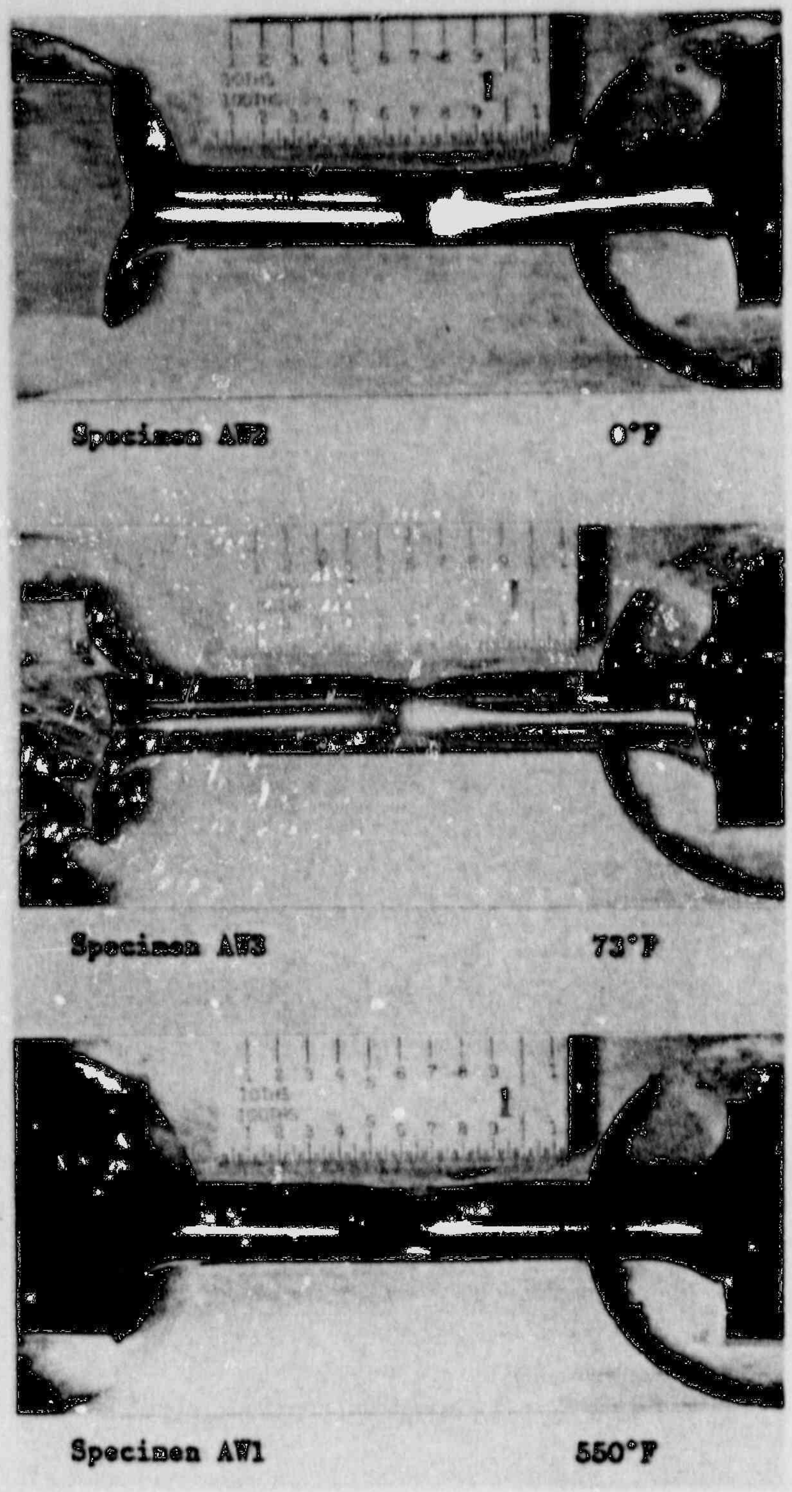

FIGURE 5-14 FRACTURED TENSILE SPECIMENS FOR VOGTLE UNIT 1 REACTOR VESSEL WELD METAL

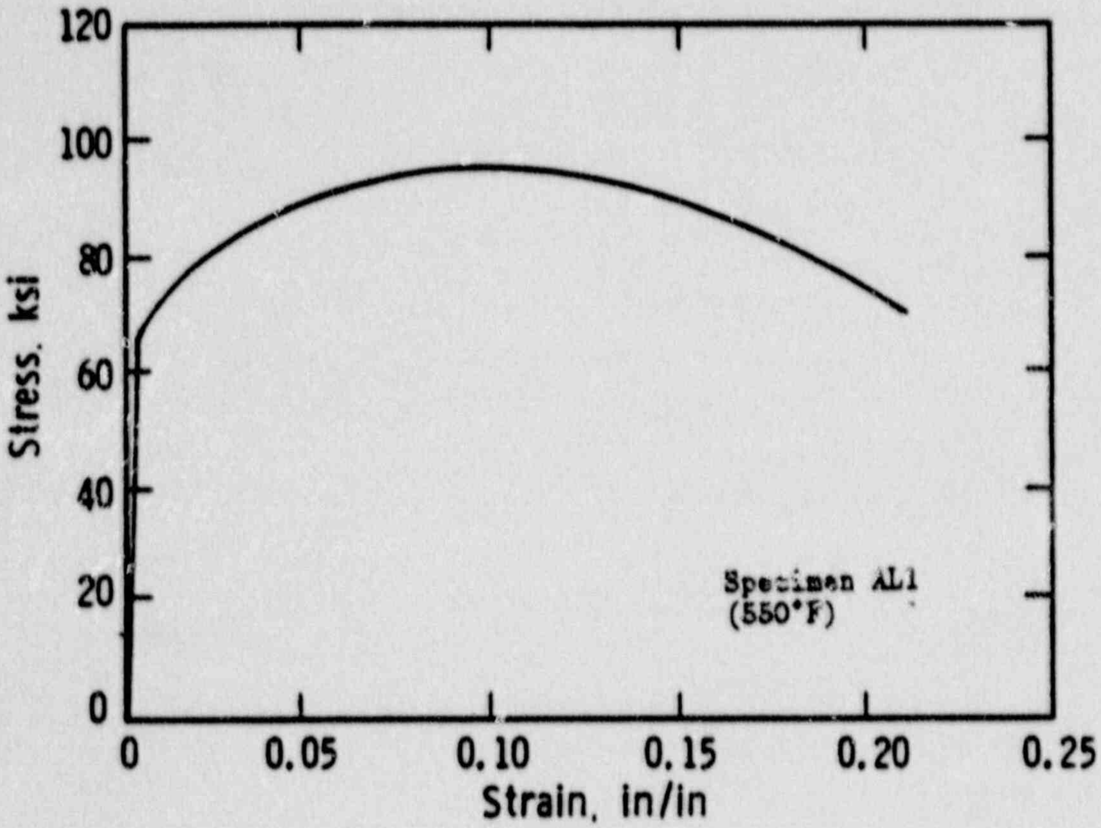

FIGURE 5-15 TYPICAL STRESS-STRAIN CURVE FOR TENSION SPECIMENS

SECTION 6
RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 INTRODUCTION

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

The use of fast neutron fluence (E > 1.0 MeV) to correlate measured materials properties changes to the neutron exposure of the material for light water reactor applications has traditionally been accepted for development of damage trend curves as well as for the implementation of trend curve data to assess vessel condition. In recent years, however, it has been suggested that an exposure model that accounts for differences in neutron energy spectra between surveillance capsule locations and positions within the vessel wall could lead to an improvement in the uncertainties associated with damage trend curves as well as to a more accurate evaluation of damage gradients through the pressure 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 Ferritic Steels in Terms of Displacements per Atom." The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the pressure vessel wall has already been promulgated in Revision 2 to the Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

This section provides the results of the neutron dosimetry evaluations performed in conjunction with the analysis of test specimens contained in surveillance capsule U. Fast neutron exposure parameters in terms of fast neutron fluence (E > 1.0 MeV), fast neutron fluence (E > 0.1 Mev), and iron atom displacements (dpa) are established for the capsule irradiation history. The analytical formalism relating the measured capsule exposure to the exposure of the vessel wall is described and used to project the integrated exposure of the vessel itself. Also uncertainties associated with the derived exposure parameters at the surveillance capsule and with the projected exposure of the pressure vessel are provided.

6.2 DISCRETE ORDINATES ANALYSIS

A plan view of the reactor geometry at the core midplane is shown in Figure 4-1. Six irradiation capsules attached to the neutron pads are included in the reactor design to constitute the reactor vessel surveillance program. The capsules are located at azimuthal angles of 58.5°, 61.0°, 121.5°, 238.5°, 241.0°, and 301.5° relative to the core cardinal area as shown in Figure 4-1.

A plan view of a dual surveillance capsule holder attached to the neutron pad is shown in Figure 6-1. The stainless steel specimen containers are 1.182 by 1-inch and approximately 56 inches in height. The containers are positioned axially such that the specimens are centered on the core midplane, thus spanning the central 5 feet of the 12-foot high reactor core.

From a neutron transport standpoint, the surveillance capsule structures are significant. They have a marked effect on both the distribution of neutron flux and the neutron energy spectrum in the water annulus between the neutron pad and the reactor vessel. In order to properly determine the neutron environment at the test specimen locations, the capsules themselves must be included in the analytical model.

In performing the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, two distinct sets of transport calculations were carried out. The first, a single computation in the conventional forward mode, was used primarily to obtain relative neutron energy distributions throughout the reactor geometry as well as to establish relative radial distributions of exposure parameters (ϕ(E > 1.0 Mev.) ϕ(E > 0.1 Mev), and dpa) through the vessel wall. The neutron spectral information was required for the interpretation of neutron dosimetry withdrawn from the surveillance capsule as well as for the determination of exposure parameter ratios; i.e., dpa/ϕ(E > 1.0 MeV), within the pressure vessel geometry. The relative radial gradient information was required to permit the projection of measured exposure parameters to locations interior to the pressure vessel wall; i.e., the 1/4T, 1/2T, and 3/4T locations.

The second set of calculations consisted of a series of adjoint analyses relating the fast neutron flux (E > 1.0 MeV) at surveillance capsule positions, and several azimuthal locations on the pressure vessel inner radius to neutron source distributions within the reactor core. The importance functions generated from these adjoint analyses provided the basis for all absolute exposure projections and comparison with measurement. These importance functions, when combined with cycle specific neutron source distributions, yielded absolute predictions of neutron exposure at the locations of interest for the Cycle 1 irradiation; and established the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles. It is important to note that the cycle specific neutron source distributions utilized in these analyses included not only spatial variations of fission rates within the reactor core; but, also accounted for the effects of varying neutron yield per fission and fission spectrum introduced by the build-up of plutonium as the burnup of individual fuel assemblies increased.

The absolute cycle specific data from the adjoint evaluations together with relative neutron energy spectra and radial distribution information from the forward calculation provided the means to:

1. Evaluate neutron dosimetry obtained from surveillance capsule locations.

2. Extrapolate dosimetry results to key locations at the inner radius and through the thickness of the pressure vessel wall.

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

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

The forward transport calculation for the reactor model summarized in Figures 4-1 and 6-1 was carried out in R, θ geometry using the DOT two-dimensional discrete ordinates code [4] and the SAILOR cross-section library [5]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In these analyses anisotopic scattering was treated with a P_3 expansion of the cross-sections and the angular discretization was modeled with an S_8 order of angular quadrature.

The reference core power distribution utilized in the forward analysis was derived from statistical studies of long-term operation of Westinghouse 4-loop plants. Inherent in the development of this reference core power distribution is the use of an out-in fuel management strategy; i.e., fresh fuel on the core periphery. Furthermore, for the peripheral fuel assemblies, a 2σ uncertainty derived from the statistical evaluation of plant to plant and cycle to cycle variations in peripheral power was used. Since it is unlikely that a single reactor would have a power distribution at the nominal $+2\sigma$

level for a large number of fuel cycles, the use of this reference distribution is expected to yield somewhat conservative results.

All adjoint analyses were also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the SAILOR library. Adjoint source locations were chosen at several azimuthal locations along the pressure vessel inner radius as well as the geometric cente... each surveillance capsule. Again, these calculations were run in R, θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case, ϕ (E > 1.0 MeV). Having the importance functions and appropriate core source distributions, the response of interest could be calculated as:

$$R(r, \theta) = \int_r \int_\theta \int_E I(r, \theta, E)\, S(r, \theta, E)\, r\, dr\, d\theta\, dE$$

where: $R(r, \theta)$ = ϕ (E > 1.0 MeV) at radius r and azimuthal angle θ

$I(r, \theta, E)$ = Adjoint importance function at radius, r, azimuthal angle θ, and neutron source energy E.

$S(r, \theta, E)$ = Neutron source strength at core location r, θ and energy E.

Although the adjoint importance functions used in the Vogtle Unit 1 analysis were based on a response function defined by the threshold neutron flux (E > 1.0 MeV), prior calculations have shown that, while the implementation of low leakage loading patterns significantly impact the magnitude and the spatial distribution of the neutron field, changes in the relative neutron energy spectrum are of second order. Thus, for a given location the ratio of dpa/ϕ (E > 1.0 MeV) is insensitive to changing core source distributions. In the application of these adjoint important functions to the Vogtle Unit 1 reactor, therefore, calculation of the iron displacement rates (dpa) and the neutron flux (E > 0.1 MeV) were computed on a cycle specific basis by using dpa/ϕ (E > 1.0 MeV) and ϕ (E > 0.1 MeV)/ϕ (E > 1.0 MeV) ratios from the forward analysis in conjunction with the cycle specific ϕ (E > 1.0 MeV) solutions from the individual adjoint evaluations.

The reactor core power distribution used in the plant specific adjoint calculations was taken from the fuel cycle design report for the first operating cycle of Vogtle Unit 1 [6]. The relative power levels in fuel assemblies that are significant contributors to the neutron exposure of the pressure vessel and surveillance capsules are summarized in Figure 6-2. For comparison purposes, the core power distribution (design basis) used in the reference forward calculation is also illustrated in Figure 6-2.

Selected results from the neutron transport analyses performed for the Vogtle Unit 1 reactor are provided in Tables 6-1 through 6-5. The data listed in these tables establish the means for absolute comparisons of analysis and measurement for the capsule irradiation period and provide the means to correlate dosimetry results with the corresponding neutron exposure of the pressure vessel wall.

In Table 6-1, the calculated exposure parameters (ϕ (E > 1.0 MeV), ϕ (E > 0.1 MeV), and dpa) are given at the geometric center of the two surveillance capsule positions for both the design basis and the plant specific core power distributions. The plant specific data, based on the adjoint transport analysis, are meant to establish the absolute comparison of measurement with analysis. The design basis data derived from the forward calculation are provided as a point of reference against which plant specific fluence evaluations can be compared. Similar data is given in Table 6-2 for the pressure vessel inner radius. Again, the three pertinent exposure parameters are listed for both the design basis and the Cycle 1 plant specific power distributions. It is important to note that the data for the vessel inner radius were taken at the clad/base metal interface; and, thus, represent the maximum exposure levels of the vessel wall itself.

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

For example, the neutron flux (E > 1.0 MeV) at the 1/4T position on the 45° azimuth is given by:

$$\phi_{1/4T}(45°) = \phi(220.27,\ 45°)\ F\ (225.75,\ 45°)$$

where $\phi_{1/4T}(45°)$ = Projected neutron flux at the 1/4T position on the 45° azimuth

ϕ (220.27, 45°) = Projected or calculated neutron flux at the vessel inner radius on the 45° azimuth.

F (225.75, 45°) = Relative radial distribution function from Table 6-3.

Similar expressions apply for exposure parameters in terms of ϕ(E > 0.1 MeV) and dpa/sec.

The DOT calculations were carried out for a typical octant of the reactor. However, for the neutron pad arrangement in Vogtle Unit 1, the pad extent for all octants is not the same. For the analysis of the flux to the pressure vessel, an octant was chosen with the neutron pad extending from 32.5° to 45° (12.5°) which produces the maximum vessel flux. Other octants have neutron pads extending 22.5° or 20° which provide more shielding. For the octant with the 12.5° pad, the maximum flux to the vessel occurs near 25° and the values in the tables for the 25° angle are vessel maximum values. Exposure values for 0°, 15°, and 45° can be used for all octants; values in the tables for 25° and 35° are maximum values and only apply to octants with a 12.5° neutron pad extent.

6.3 NEUTRON DOSIMETRY

The passive neutron sensors included in the Vogtle Unit 1 surveillance program are listed in Table 6-6. Also given in Table 6-6 are the primary nuclear reactions and associated nuclear constants that were used in the evaluation of the neutron energy spectrum within the capsule and the subsequent determination of the various exposure parameters of interest (ϕ (E > 1.0 Mev), ϕ (E > 0.1 MeV), dpa).

The relative locations of the neutron sansors within the capsules are shown in Figure 4-2. The iron, nickel, copper, and cobalt-aluminum monitors, in wire form, were placed in holes drilled in spacers at several axial levels within the capsules. The cadmium-shielded neptunium and uranium fission monitors were accommodated within the dosimeter block located near the center of the capsule.

The use of passive monitors such as those listed in Table 6-6 does not yield a direct measure of the energy dependent flux level at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

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

The specific activity of each of the neutron monitors was determined using established ASTM procedures [7 through 20]. Following sample preparation and weighing, the activity of each monitor was determined by means of a lithium-drifted germanium, Ge(Li), gamma spectrometer. The irradiation history of the Vogtle Unit 1 reactor during Cycle 1 was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" for the applicable period.

The irradiation history applicable to capsule U is given in Table 6-7. Measured and saturated reaction product specific activities as well as measured full power reaction rates are listed in Table 6-8. Reaction rate values were derived using the pertinent data from Tables 6-6 and 6-7.

Values of key fast neutron exposure parameters were derived from the measured reaction rates using the FERRET least squares adjustment code [21]. The FERRET approach used the measured reaction rate data and the calculated neutron energy spectrum at the the center of the surveillance capsule as input and proceeded to adjust a priori (calculated) group fluxes to produce a best fit (in a least squares sense) to the reaction rate data. The exposure parameters along with associated uncertainties where then obtained from the adjusted spectra.

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

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

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

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

relates a set of measured reaction rates R_i to a single spectrum ϕ_g by the multigroup cross section σ_{ig}. (In this case, FERRET also adjusts the cross-sections.) The lognormal approach automatically accounts for the physical constraint of positive fluxes, even with the large assigned uncertainties.

In the FERRET analysis of the dosimetry data, the continuous quantities (i.e., fluxes and cross-sections) were approximated in 53 groups. The calculated fluxes from the discrete ordinates analysis were expanded into the FERRET group structure using the SAND-II code [22]. This procedure was carried out by first expanding the a priori spectrum into the SAND-II 620 group structure using a spline interpolation procedure for interpolation in regions where group boundaries do not coincide. The 620-point spectrum was then easily collapsed to the group scheme used in FERRET.

The cross-sections were also collapsed into the 53 energy-group structure using SAND II with calculated spectra (as expanded to 620 groups) as weighting functions. The cross sections were taken from the ENDF/B-V dosimetry file. Uncertainty estimates and 53 x 53 covariance matrices were constructed for each cross section. Correlations between cross sections were neglected due to data and code limitations, but are expected to be unimportant, since these are not a major component of the cross section uncertainty treatment.

For each set of data or a priori values, the inverse of the corresponding relative covariance matrix M is used as a statistical weight. In some cases, as for the cross sections, a multigroup covariance matrix is used. More often, a simple parameterized form is used:

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

where R_N specifies an overall fractional normalization uncertainty (i.e., complete correlation) for the corresponding set of values. The fractional uncertainties R_g specify additional random uncertainties for group g that are correlated with a correlation matrix:

$$P_{gg'} = (1 - \theta) \delta_{gg'} + \theta \exp \left[\frac{-(g-g')^2}{2\gamma^2} \right]$$

The first term specifies purely random uncertainties while the second term describes short-range correlations over a range γ (θ specifies the strength of the latter term.)

For the a priori calculated fluxes, a short-range correlation of $\gamma = 6$ groups was used. This choice implies that neighboring groups are strongly correlated when θ is close to 1. Strong long-range correlations (or anticorrelations) were justified based on information presented by R. E. Maerker [23]. Maerker's results are closely duplicated when $\gamma = 6$. For the integral reaction rate covariances, simple normalization and random uncertainties were combined as deduced from experimental uncertainties.

Results of the FERRET evaluation of the capsule U dosimetry are given in Table 6-9 The data summarized in Table 6-9 indicated that the capsule received an integrated exposure of 3.41×10^{18} n/cm^2 (E > 1.0 MeV) with an associated uncertainty of $\pm$ 8%. Also reported are capsule exposures in terms of fluence (E > 0.1 MeV) and iron atom displacements (dpa). Summaries of the fit of the adjusted spectrum are provided in Table 6-10. In general, excellent results were achieved in the fits of the adjusted spectrum to the individual experimental reaction rates. The adjusted spectrum itself is tabulated in Table 6-11 for the FERRET 53 energy group structure.

A summary of the measured and calculated neutron exposure of capsule U is presented in Table 6-12. The agreement between calculation and measurement falls within $\pm$ 12% for all exposure parameters listed. The calculated fast neutron exposure (Φ (E > 1.0 MeV), Φ (E > 0.1 MeV), dpa) values agreed with the measurements to within 1-3% whereas, the thermal neutron exposure calculated for Cycle 1 exceeded the measured value by 12 percent.

Neutron exposure projections at key locations on the pressure vessel inner radius are given in Table 6-13. Along with the current (1.14 EFPY) exposure derived from the capsule U measurements, projections are also provided for an exposure period of 16 EFPY and to end of vessel design life (32 EFPY). The calculated design basis exposure rates given in Table 6-2 were used to perform projections beyond the end of Cycle 1.

In the calculation of exposure gradients for use in the development of heatup and cooldown curves for the Vogtle Unit 1 reactor coolant system, exposure projections to 16 EFPY and 32 EFPY were employed. Data based on both a fluence (E > 1.0 MeV) slope and a plant specific dpa slope through the vessel wall are provided in Table 6-14. In order to access RT_{NDT} vs. fluence trend curves, dpa equivalent fast neutron fluence levels for the 1/4T and 3/4T positions were defined by the relations

$$\phi'(1/4T) = \phi\,(\text{Surface})\left(\frac{\text{dpa}\ (1/4T)}{\text{dpa}\ (\text{Surface})}\right)$$

$$\phi'(3/4T) = \phi\,(\text{Surface})\left(\frac{\text{dpa}\ (3/4T)}{\text{dpa}\ (\text{Surface})}\right)$$

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

In Table 6-15 updated lead factors are listed for each of the Vogtle Unit 1 surveillance capsules. These data may be used as a guide in establishing future withdrawal schedules for the remaining capsules.

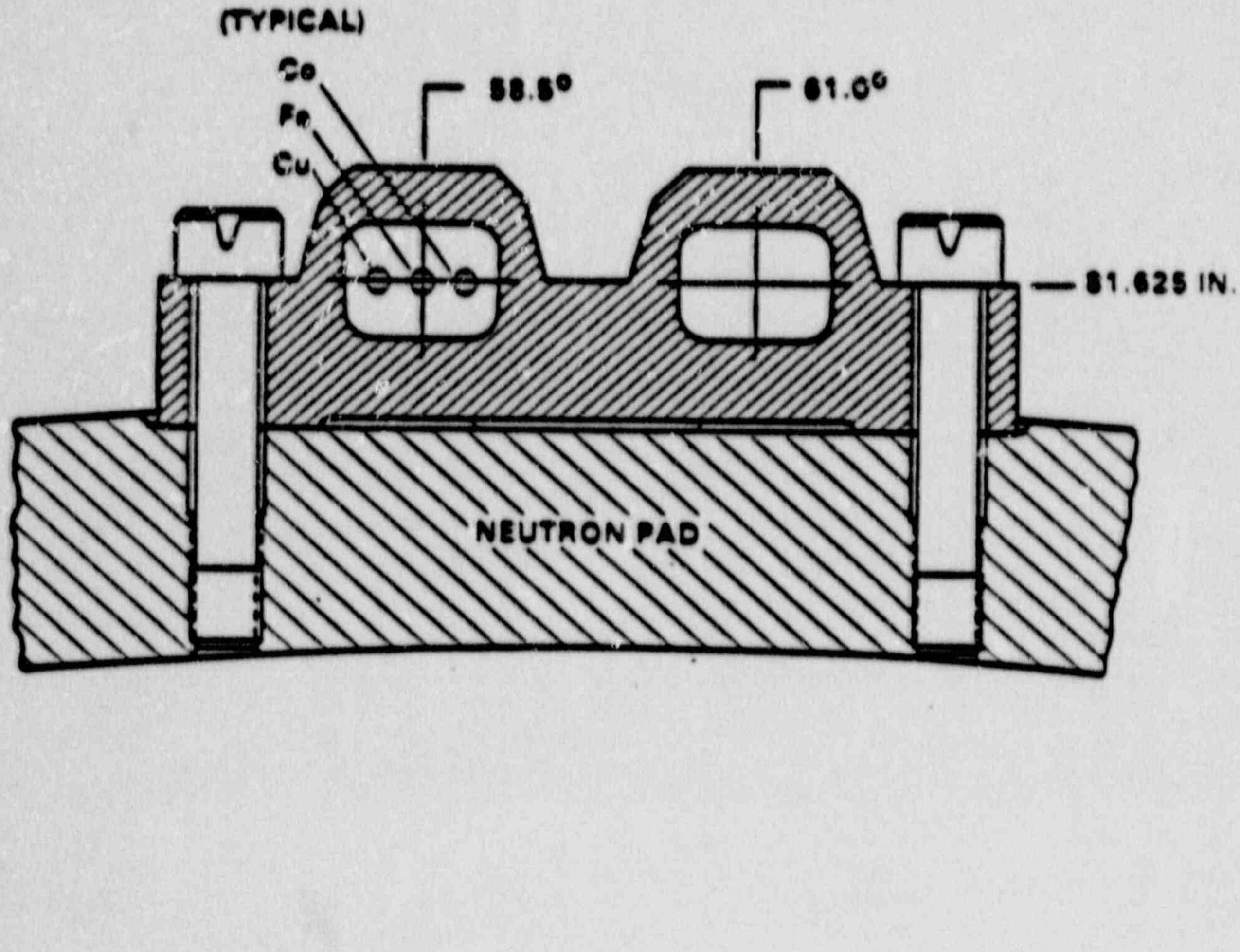

Figure 6-1. Plan View of a Dual Reactor Vessel Surveillance Capsule

0.72 1.01	0.75 1.04	0.67 0.96	0.55 0.77	Cycle 1 Design Basis	
0.99 1.02	1.04 1.10	0.95 1.00	0.97 1.05	0.81 1.10	0.50 0.71
1.14 1.05	1.10 0.87	1.12 0.87	1.05 1.07	0.98 1.00	0.96 1.05
1.15 1.09	1.18 1.06	1.13 0.88	1.13 1.10	1.18 1.04	
1.19 0.90	1.15 1.04	1.19 1.12	1.14 0.92		

Figure 6-2. Core Power Distributions Used in Transport Calculations for Vogtle Unit 1

TABLE 6-1

CALCULATED FAST NEUTRON EXPOSURE PARAMETERS AT THE SURVEILLANCE CAPSULE CENTER

	DESIGN BASIS		CYCLE 1	
	29.0°	31.5°	29.0°	31.5°
ϕ (E> 1.0 MeV) (n/cm^2-sec)	1.13×10^{11}	1.21×10^{11}	8.65×10^{10}	9.23×10^{10}
ϕ (E> 0.1 MeV) (n/cm^2-sec)	5.07×10^{11}	5.44×10^{11}	3.89×10^{11}	4.15×10^{11}
dpa/sec	2.21×10^{-10}	2.37×10^{-10}	1.69×10^{-10}	1.81×10^{-10}

TABLE 6-2

CALCULATED FAST NEUTRON EXPOSURE PARAMETERS AT
THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

DESIGN BASIS

	0°	15°	25°	35°	45°
ϕ(E> 1.0Mev) (n/cm^2-sec)	1.78×10^{10}	2.66×10^{10}	3.01×10^{10}	2.45×10^{10}	2.81×10^{10}
ϕ(E> 0.1Mev) (n/cm^2-sec)	3.70×10^{10}	5.60×10^{10}	8.22×10^{10}	6.96×10^{10}	7.04×10^{10}
dpa/sec	2.77×10^{-11}	4.12×10^{-11}	5.04×10^{-11}	4.15×10^{-11}	4.48×10^{-11}

CYCLE 1 SPECIFIC

	0°	15°	25°	35°	45°
ϕ(E> 1.0Mev) (n/cm^2-sec)	1.36×10^{10}	2.01×10^{10}	2.30×10^{10}	1.87×10^{10}	2.13×10^{10}
ϕ(E> 0.1Mev) (n/cm^2-sec)	2.83×10^{10}	4.23×10^{10}	6.28×10^{10}	5.31×10^{10}	5.34×10^{10}
dpa/sec	2.12×10^{-11}	3.12×10^{-11}	3.85×10^{-11}	3.17×10^{-11}	3.39×10^{-11}

TABLE 6-3

RELATIVE RADIAL DISTRIBUTIONS OF NEUTRON FLUX (E > 1.0 MeV) WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0°	15°	25°	35°	45°
220.27[1]	1.00	1.00	1.00	1.00	1.00
220.64	0.976	0.979	0.980	0.977	0.979
221.66	0.888	0.891	0.893	0.891	0.889
222.99	0.768	0.770	0.772	0.770	0.766
224.31	0.653	0.653	0.657	0.655	0.648
225.63	0.551	0.550	0.554	0.552	0.543
226.95	0.462	0.460	0.465	0.463	0.452
228.28	0.386	0.384	0.388	0.386	0.375
229.60	0.321	0.319	0.324	0.321	0.311
230.92	0.267	0.265	0.271	0.267	0.257
232.25	0.221	0.219	0.223	0.221	0.211
233.57	0.183	0.181	0.185	0.183	0.174
234.89	0.151	0.149	0.153	0.151	0.142
236.22	0.124	0.122	0.126	0.124	0.116
237.54	0.102	0.100	0.104	0.102	0.0945
238.86	0.0828	0.0817	0.0846	0.0835	0.0762
240.19	0.0671	0.0660	0.0689	0.0679	0.0608
241.51	0.0538	0.0522	0.0550	0.0545	0.0471
242.17[2]	0.0506	0.0488	0.0518	0.0521	0.0438

NOTES: 1) Base Metal Inner Radius
2) Base Metal Outer Radius

TABLE 6-4

RELATIVE RADIAL DISTRIBUTIONS OF NEUTRON FLUX (E > 0.1 MeV)
WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0°	15°	25°	35°	45°
220.27(1)	1.00	1.00	1.00	1.00	1.00
220.64	1.00	1.00	1.00	1.00	1.00
221.66	1.00	1.00	1.00	0.999	0.995
222.99	0.974	0.969	0.974	0.959	0.956
224.31	0.927	0.920	0.927	0.907	0.901
225.63	0.874	0.865	0.874	0.850	0.842
226.95	0.818	0.808	0.818	0.792	0.782
228.28	0.761	0.750	0.716	0.734	0.721
229.60	0.705	0.693	0.704	0.677	0.662
230.92	0.649	0.637	0.649	0.621	0.605
232.25	0.594	0.582	0.594	0.567	0.549
233.57	0.540	0.529	0.542	0.515	0.495
234.89	0.487	0.478	0.490	0.465	0.443
236.22	0.436	0.428	0.440	0.416	0.392
237.54	0.386	0.380	0.392	0.369	0.343
238.86	0.337	0.333	0.344	0.324	0.295
240.19	0.289	0.287	0.298	0.279	0.248
241.51	0.244	0.238	0.249	0.233	0.201
242.17(2)	0.233	0.226	0.237	0.223	0.188

NOTES: 1) Base Metal Inner Radius
2) Base Metal Outer Radius

TABLE 6-5

RELATIVE RADIAL DISTRIBUTIONS OF IRON DISPLACEMENT RATE (dpa) WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0°	15°	25°	35°	45°
220.27(1)	1.00	1.00	1.00	1.00	1.00
220.64	0.984	0.981	0.984	0.983	0.984
221.66	0.912	0.909	0.917	0.921	0.915
222.99	0.815	0.812	0.826	0.833	0.821
224.31	0.722	0.719	0.737	0.747	0.730
225.63	0.638	0.634	0.656	0.668	0.647
226.95	0.563	0.559	0.584	0.597	0.572
228.28	0.497	0.493	0.519	0.533	0.506
229.60	0.439	0.435	0.462	0.475	0.447
230.92	0.387	0.383	0.410	0.423	0.394
232.25	0.341	0.338	0.364	0.376	0.347
233.57	0.300	0.297	0.322	0.334	0.305
234.89	0.263	0.261	0.285	0.295	0.266
236.22	0.230	0.228	0.250	0.260	0.231
237.54	0.199	0.198	0.218	0.227	0.199
238.86	0.171	0.170	0.189	0.196	0.169
240.19	0.145	0.144	0.161	0.167	0.140
241.51	0.121	0.119	0.135	0.139	0.113
242.17(2)	0.116	0.113	0.128	0.134	0.106

NOTES: 1) Base Metal Inner Radius
2) Base Metal Outer Radius

TABLE 6-6

NUCLEAR PARAMETERS FOR NEUTRON FLUX MONITORS

Monitor Material	Reaction of Interest	Target Weight Fraction	Response Range	Product Half-Life	Fission Yield (%)
Copper	$Cu^{63}(n,\alpha)Co^{60}$	0.6917	E> 4.7 MeV	5.272 yrs	
Iron	$Fe^{54}(n,p)Mn^{54}$	0.0582	E> 1.0 MeV	312.2 days	
Nickel	$Ni^{58}(n,p)Co^{58}$	0.6830	E> 1.0 MeV	70.90 days	
Uranium-238*	$U^{238}(n,f)Cs^{137}$	1.0	E> 0.4 MeV	30.12 yrs	5.99
Neptunium-237*	$Np^{237}(n,f)Cs^{137}$	1.0	E> 0.08 MeV	30.12 yrs	6.50
Cobalt-Aluminum*	$Co^{59}(n,\gamma)Co^{60}$	0.0015	0.4ev>E> 0.015 MeV	5.272 yrs	
Cobalt-Aluminum	$Co^{59}(n,\gamma)Co^{60}$	0.0015	E> 0.015 MeV	5.272 yrs	

*Denotes that monitor is cadmium shielded.

TABLE 6-7

IRRADIATION HISTORY OF NEUTRON SENSORS CONTAINED IN CAPSULE U

Irradiation Period	P_j (MW_t)	$\frac{P_j}{P_{Ref.}}$	Irradiation Time (days)	Decay Time (days)
3/87	716	.210	4	694
4/87	1108	.325	30	664
5/87	1404	.412	31	633
6/87	1055	.309	30	603
7/87	2467	.723	31	572
8/87	3373	.989	31	541
9/87	3407	.999	30	511
10/87	951	.279	31	480
11/87	2677	.785	30	450
12/87	3317	.972	31	419
1/88	1835	.538	31	388
2/88	1993	.584	29	359
3/88	3302	.968	31	328
4/88	2649	.777	30	298
5/88	3402	.998	31	267
6/88	3396	.996	30	237
7/88	2984	.875	31	206
8/88	3215	.943	31	175
9/88	3260	.956	30	145
10/88	2813	.825	7	138

NOTE: Reference Power = 3411 MW_t

TABLE 6-8

MEASURED SENSOR ACTIVITIES AND REACTION RATES

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Cu-63 (n,α) Co-60			
Top	4.82×10^4	3.68×10^5	
Middle	4.38×10^4	3.35×10^5	
Bottom	4.44×10^4	3.39×10^5	
Average	4.55×10^4	3.48×10^5	5.30×10^{-17}
Fe-54(n,p) Mn-54			
Top	1.49×10^6	3.57×10^6	
Middle	1.34×10^6	3.21×10^6	
Bottom	1.36×10^6	3.25×10^6	
Average	1.40×10^6	3.34×10^6	5.32×10^{-15}
Ni-58 (n,p) Co-58			
Top	1.26×10^7	5.42×10^7	
Middle	1.16×10^7	4.99×10^7	
Bottom	1.17×10^7	5.04×10^7	
Average	1.20×10^7	5.15×10^7	7.35×10^{-15}
U-238 (n,f) Cs-137 (Cd)			
Middle	1.10×10^5	4.29×10^6	2.83×10^{-14}

TABLE 6-8

MEASURED SENSOR ACTIVITIES AND REACTION RATES - cont'd

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Np-237(n,f) Cs-137 (Cd)			
Middle	1.24×10^{6}	4.84×10^{7}	2.93×10^{-13}
Co-59 (n,γ) Co-60			
Top	1.03×10^{7}	7.87×10^{7}	
Middle	1.01×10^{7}	7.72×10^{7}	
Bottom	1.05×10^{7}	8.03×10^{7}	
Average	1.03×10^{7}	7.87×10^{7}	5.14×10^{-12}
Co-59 (n,γ) Co-60 (Cd)			
Top	5.21×10^{6}	3.98×10^{7}	
Middle	5.46×10^{6}	4.17×10^{7}	
Bottom	5.58×10^{6}	4.27×10^{7}	
Average	5.42×10^{6}	4.14×10^{7}	2.70×10^{-12}

TABLE 6-9

SUMMARY OF NEUTRON DOSIMETRY RESULTS

	TIME AVERAGED EXPOSURE RATES	
ϕ (E> 1.0 MeV) (n/cm^2-sec)	9.47×10^{10}	$\pm$ 8%
ϕ (E> 0.1 MeV) (n/cm^2-sec)	4.11×10^{11}	$\pm$ 15%
dpa/sec	1.80×10^{-10}	$\pm$ 11%
ϕ(E> 0.414 eV) (n/cm^2-sec)	3.92×10^{10}	$\pm$ 30%
	INTEGRATED CAPSULE EXPOSURE	
Φ (E> 1.0 MeV) (n/cm^2)	3.41×10^{18}	$\pm$ 8%
Φ (E> 0.1 MeV) (n/cm^2)	1.48×10^{19}	$\pm$ 15%
dpa	6.48×10^{-3}	$\pm$ 11%
Φ (E> 0.414 eV) (n/cm^2)	1.41×10^{18}	$\pm$ 30%

NOTE: Total Irradiation Time = 1.14 EFPY

TABLE 6-10

COMPARISON OF MEASURED AND FERRET CALCULATED
REACTION RATES AT THE SURVEILLANCE CAPSULE CENTER

Reaction	Measured	Adjusted Calculation	C/M
Cu-63 (n,α) Co-60	5.30×10^{-17}	5.41×10^{-17}	1.02
Fe-54 (n,p) Mn-54	5.32×10^{-15}	5.25×10^{-15}	0.99
Ni-58 (n,p) Co-58	7.35×10^{-15}	7.25×10^{-15}	0.99
U-238 (n,f) Cs-137 (Cd)	2.83×10^{-14}	2.90×10^{-14}	1.03
Np-237 (n,f) Cs-137 (Cd)	2.93×10^{-13}	2.96×10^{-13}	1.01
Co-59 (n,γ) Co-60 (Cd)	2.70×10^{-12}	2.70×10^{-12}	1.00
Co-59 (n,γ) Co-60	5.14×10^{-12}	5.14×10^{-12}	1.00

TABLE 6-11

ADJUSTED NEUTRON ENERGY SPECTRUM AT THE SURVEILLANCE CAPSULE CENTER

Group	Energy (Mev)	Adjusted Flux (n/cm^2-sec)	Group	Energy (Mev)	Adjusted Flux (n/cm^2-sec)
1	1.73×10^{1}	7.58×10^{6}	28	9.12×10^{-3}	1.92×10^{10}
2	1.49×10^{1}	1.72×10^{7}	29	5.53×10^{-3}	2.49×10^{10}
3	1.35×10^{1}	6.68×10^{7}	30	3.36×10^{-3}	7.80×10^{9}
4	1.16×10^{1}	1.50×10^{8}	31	2.84×10^{-3}	7.48×10^{9}
5	1.00×10^{1}	3.30×10^{8}	32	2.40×10^{-3}	7.24×10^{9}
6	8.61×10^{0}	5.65×10^{8}	33	2.04×10^{-3}	2.05×10^{10}
7	7.41×10^{0}	1.30×10^{9}	34	1.23×10^{-3}	1.90×10^{10}
8	6.07×10^{0}	1.86×10^{9}	35	7.49×10^{-4}	1.73×10^{10}
9	4.97×10^{0}	3.91×10^{9}	36	4.54×10^{-4}	1.71×10^{10}
10	3.68×10^{0}	5.12×10^{9}	37	2.75×10^{-4}	1.86×10^{10}
11	2.87×10^{0}	1.06×10^{10}	38	1.67×10^{-4}	2.07×10^{10}
12	2.23×10^{0}	1.44×10^{10}	39	1.01×10^{-4}	2.00×10^{10}
13	1.74×10^{0}	2.00×10^{10}	40	6.14×10^{-5}	1.97×10^{10}
14	1.35×10^{0}	2.24×10^{10}	41	3.73×10^{-5}	1.89×10^{10}
15	1.11×10^{0}	4.09×10^{10}	42	2.26×10^{-5}	1.81×10^{10}
16	8.21×10^{-1}	4.66×10^{10}	43	1.37×10^{-5}	1.73×10^{10}
17	6.39×10^{-1}	4.83×10^{10}	44	8.32×10^{-6}	1.62×10^{10}
18	4.98×10^{-1}	3.50×10^{10}	45	5.04×10^{-6}	1.47×10^{10}
19	3.88×10^{-1}	4.93×10^{10}	46	3.06×10^{-6}	1.35×10^{10}
20	3.02×10^{-1}	5.09×10^{10}	47	1.86×10^{-6}	1.23×10^{10}
21	1.83×10^{-1}	5.06×10^{10}	48	1.13×10^{-6}	8.95×10^{9}
22	1.11×10^{-1}	4.06×10^{10}	49	6.83×10^{-7}	8.97×10^{9}
23	6.74×10^{-2}	2.83×10^{10}	50	4.14×10^{-7}	9.63×10^{9}
24	4.09×10^{-2}	1.61×10^{10}	51	2.51×10^{-7}	8.13×10^{9}
25	2.55×10^{-2}	2.12×10^{10}	52	1.52×10^{-7}	6.99×10^{9}
26	1.99×10^{-2}	1.04×10^{10}	53	9.24×10^{-8}	1.45×10^{10}
27	1.50×10^{-2}	1.33×10^{10}			

NOTE: Tabulated energy levels represent the upper energy of each group.

TABLE 6-12

COMPARISON OF CALCULATED AND MEASURED
EXPOSURE LEVELS FOR CAPSULE U

	Calculated	Measured	C/M
Φ(E> 1.0 MeV) (n/cm^2)	3.32×10^{18}	3.41×10^{18}	0.97
Φ(E> 0.1 MeV) (n/cm^2)	1.49×10^{19}	1.48×10^{19}	1.01
dpa	6.51×10^{-3}	6.48×10^{-3}	1.01
Φ(E> 0.414 eV) (n/cm^2)	1.58×10^{18}	1.41×10^{18}	1.12

TABLE 6-13
NEUTRON EXPOSURE PROJECTIONS AT KEY LOCATIONS
ON THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

	AZIMUTHAL ANGLE				
	0°	15°	25°[(a)]	35°	45°
1.14 EFPY					
Φ(E> 1.0 MeV) (n/cm^2)	5.02×10^{17}	7.43×10^{17}	8.50×10^{17}	6.91×10^{17}	7.87×10^{17}
Φ(E> 0.1 MeV) (n/cm^2)	1.01×10^{18}	1.51×10^{18}	2.25×10^{18}	1.90×10^{18}	1.91×10^{18}
dpa	7.59×10^{-4}	1.12×10^{-3}	1.38×10^{-3}	1.14×10^{-3}	1.21×10^{-3}
16.0 EFPY					
Φ(E> 1.0 MeV) (n/cm^2)	8.85×10^{18}	1.32×10^{19}	1.50×10^{19}	1.22×10^{19}	1.40×10^{19}
Φ(E> 0.1 MeV) (n/cm^2)	1.84×10^{19}	2.78×10^{19}	4.08×10^{19}	3.45×10^{19}	3.49×10^{19}
dpa	1.37×10^{-2}	2.04×10^{-2}	2.50×10^{-2}	2.06×10^{-2}	2.22×10^{-2}
32.0 EFPY					
Φ(E> 1.0 MeV) (n/cm^2)	1.78×10^{19}	2.66×10^{19}	3.02×10^{19}	2.46×10^{19}	2.82×10^{18}
Φ(E> 0.1 MeV) (n/cm^2)	3.70×10^{19}	5.60×10^{19}	8.23×10^{19}	6.97×10^{19}	7.05×10^{19}
dpa	2.77×10^{-2}	4.12×10^{-2}	5.05×10^{-2}	4.16×10^{-2}	4.48×10^{-2}

(a) Maximum point on the pressure vessel

TABLE 6-14

NEUTRON EXPOSURE VALUES FOR USE IN THE GENERATION OF HEATUP/COOLDOWN CURVES

16 EFPY

	NEUTRON FLUENCE (E> 1.0 MeV) SLOPE (n/cm^2)			dpa SLOPE (equivalent n/cm^2)		
	Surface	1/4 T	3/4 T	Surface	1/4 T	3/4 T
0°	8.85×10^{18}	4.81×10^{18}	1.03×10^{18}	8.85×10^{18}	5.58×10^{18}	1.94×10^{18}
15°	1.32×10^{19}	7.15×10^{18}	1.50×10^{18}	1.32×10^{19}	8.28×10^{18}	2.86×10^{18}
25°(a)	1.50×10^{19}	8.19×10^{18}	1.77×10^{18}	1.50×10^{19}	9.74×10^{18}	3.57×10^{18}
35°	1.22×10^{19}	6.64×10^{18}	1.42×10^{18}	1.22×01^{19}	8.08×10^{18}	3.03×10^{18}
45°	1.40×10^{19}	7.49×10^{18}	1.51×10^{18}	1.40×10^{19}	8.96×10^{18}	3.07×10^{18}

32 EFPY

	NEUTRON FLUENCE (E> 1.0 MeV) SLOPE (n/cm^2)			dpa SLOPE (equivalent n/cm^2)		
	Surface	1/4 T	3/4 T	Surface	1/4 T	3/4 T
0°	1.78×10^{19}	9.67×10^{18}	2.06×10^{18}	1.78×10^{19}	1.12×10^{19}	3.90×10^{18}
15°	2.66×10^{19}	1.44×10^{19}	3.03×10^{18}	2.66×10^{19}	1.68×10^{19}	5.77×10^{18}
25°(a)	3.02×10^{19}	1.65×10^{19}	3.56×10^{18}	3.02×10^{19}	1.96×10^{19}	7.19×10^{18}
35°	2.46×10^{19}	1.34×10^{19}	2.85×10^{18}	2.46×01^{19}	1.63×10^{19}	6.10×10^{18}
45°	2.82×10^{19}	1.51×10^{19}	3.05×10^{18}	2.82×10^{19}	1.80×10^{19}	6.18×10^{18}

(a) Maximum point on the pressure vessel

3725s/080989 10

TABLE 6-15

UPDATED LEAD FACTORS FOR VOGTLE
UNIT 1 SURVEILLANCE CAPSULES

Capsule	Lead Factor
U	4.01[a]
X	4.02
W	4.02
Z	4.02
V	3.75
Y	3.75

(a) Plant specific evaluation

SECTION 7
SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following removal schedule meets ASTM E185-82 and is recommended for future capsules to be removed from the Vogtle Unit 1 reactor vessel:

Capsule	Vessel Location (deg)	Lead Factor	Removal Time[a]	Estimated Capsule Fluence (n/cm^2)
U	58.5	4.01	1.14	3.41×10^{18}[b]
Y	241	3.75	5	1.77×10^{19}[c]
V	61	3.75	9	3.19×10^{19}[d]
X	238.5	4.02	15	5.69×10^{19}
W	121.5	4.02	Standby	--
Z	301.5	4.02	Standby	--

a) Effective full power years from plant startup

b) Actual fluence

c) Approximate fluence at vessel 1/4 thickness at end of life (32 EFPY)

d) Approximate fluence at vessel inner wall at end of life (32 EFPY)

SECTION 8
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APPENDIX A
HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

1.0 INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature) for the reactor vessel. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material fracture toughness properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Rev. 2 (Radiation Embrittlement of Reactor Vessel Materials)[1]. The value, "f", given in Figuro 1 is the calculated value of the neutron fluence at the location of interest (inner surface, 1/4T, or 3/4T) in the vessel at the location of the postulated defect, n/cm^2 ($E > 1.0$ MeV) divided by 10^{19}. The fluence factor is determined from Figure A-1.

2.0 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan[A-2]. The pre-irradiation fracture-toughness properties of the Vogtle Unit 1 reactor vessel are presented in Table A-2.

3.0 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code[A-3]. The K_{IR} curve is given by the following equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T - RT_{NDT} + 160)] \qquad (1)$$

where

K_{IR} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code[A-3] as follows:

$$C\, K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

where

K_{IM} = stress intensity factor caused by membrane (pressure) stress

K_{IT} = stress intensity factor caused by the thermal gradients

K_{IR} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw.

During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4 T crack during heatup is lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same time coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 1983 Amendment to 10CFR50[A-4] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure.

Table A-2 indicates that the limiting RT_{NDT} of 20°F occurs in the closure head flange of Vogtle Unit 1, so the minimum allowable temperature of this region is 140°F. These limits are less restrictive than the limits shown on Figures A-2 and A-3.

4.0 HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed in Section 3.0.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures A-2 through A-4. This is in addition to other criteria which must be met before the reactor is made critical.

The leak limit curve shown in Figures A-2 and A-3 represents minimum temperature requirements at the leak test pressure specified by applicable codes[A-2,A-3]. The leak test limit curve was determined by methods of References A-2 and A-4.

Figures A-2 through A-4 define limits for ensuring prevention of nonductile failure for the Vogtle Unit 1 primary reactor coolant system.

5.0 ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99 Rev. 2[A-1] the adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (3)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = [CF]f^{(0.28-0.10 \log f)} \qquad (4)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(\text{depth } X)} = f_{\text{surface}}{}^{(e^{-.24x})} \qquad (5)$$

where x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface. The resultant fluence is then put into Equation (4) to calculate ΔRT_{NDT} at the specific depth.

CF (°F) is the chemistry factor, obtained from Reference A-1. Beltline region materials of Vogtle Unit 1 are considered for the limiting material. Limiting material is found to be intermediate shell plate B8805-2. The calculation of ART for the limiting material is shown in Table A-1. This calculation was used to develop the Vogtle Unit 1 heatup and cooldown curves and are shown in Figures A-2 through A-4.

TABLE A-1

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR LIMITING VOGTLE UNIT 1 REACTOR VESSEL MATERIAL - INTERMEDIATE SHELL PLATE B8805-2

Parameter	Regulatory Guide 1.99 - Revision 2 16 EFPY 1/4 T	3/4 T
Chemistry Factor, CF (°F)	51	51
Fluence, f (10^{19} n/cm^2)[(a)]	0.90	0.32
Fluence Factor, ff	0.97	0.69

****

ΔRT_{NDT} = CF x ff (°F)	50	35
Initial RT_{NDT}, I (°F) [(b)]	20	20
Margin, M (°F) [(c)]	34	34

****

Revision 2 to Regulatory Guide 1.99

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	104	89

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 Mev) = 1.5 at 16 EFPY. The Vogtle Unit 1 reactor vessel wall thickness is 8.63 inches at the beltline region.

(b) The initial RT_{NDT} (I) values for the plates and welds are measured values.

(c) Margin is calculated as, $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is assumed to be zero since the initial RT_{NDT} values were obtained from conservative (i.e., "upper bound") test results. However, the standard deviation for ΔRT_{NDT}, σ_Δ, is 28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}.

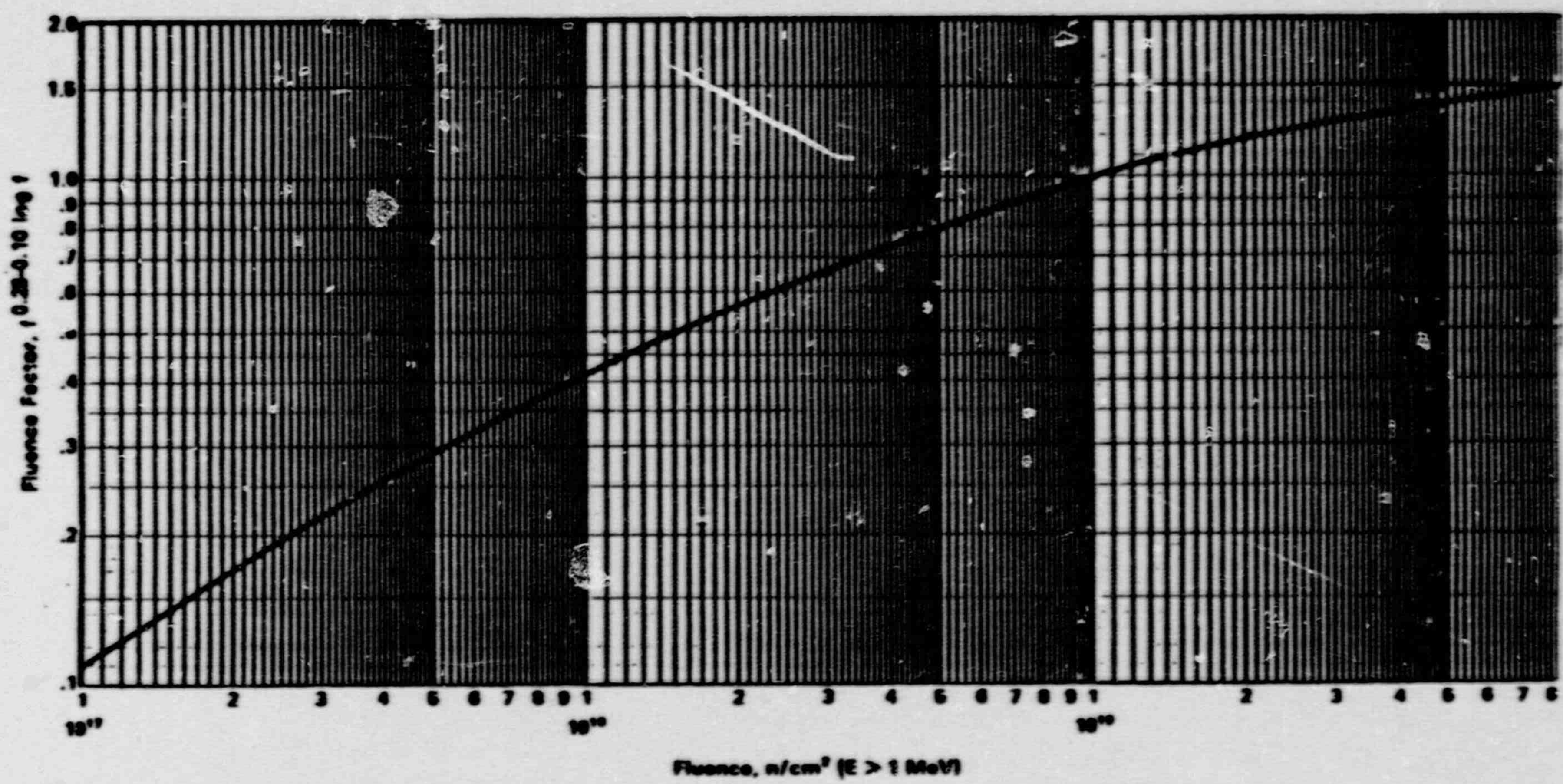

Figure A-1. Fluence Factor for Use in the Expression for ΔRT_{NDT}

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B8805-2

INITIAL RT_{NDT}: 20°F

RT_{NDT} AFTER 16 EFPY: 1/4T, 104°F

3/4T, 89°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

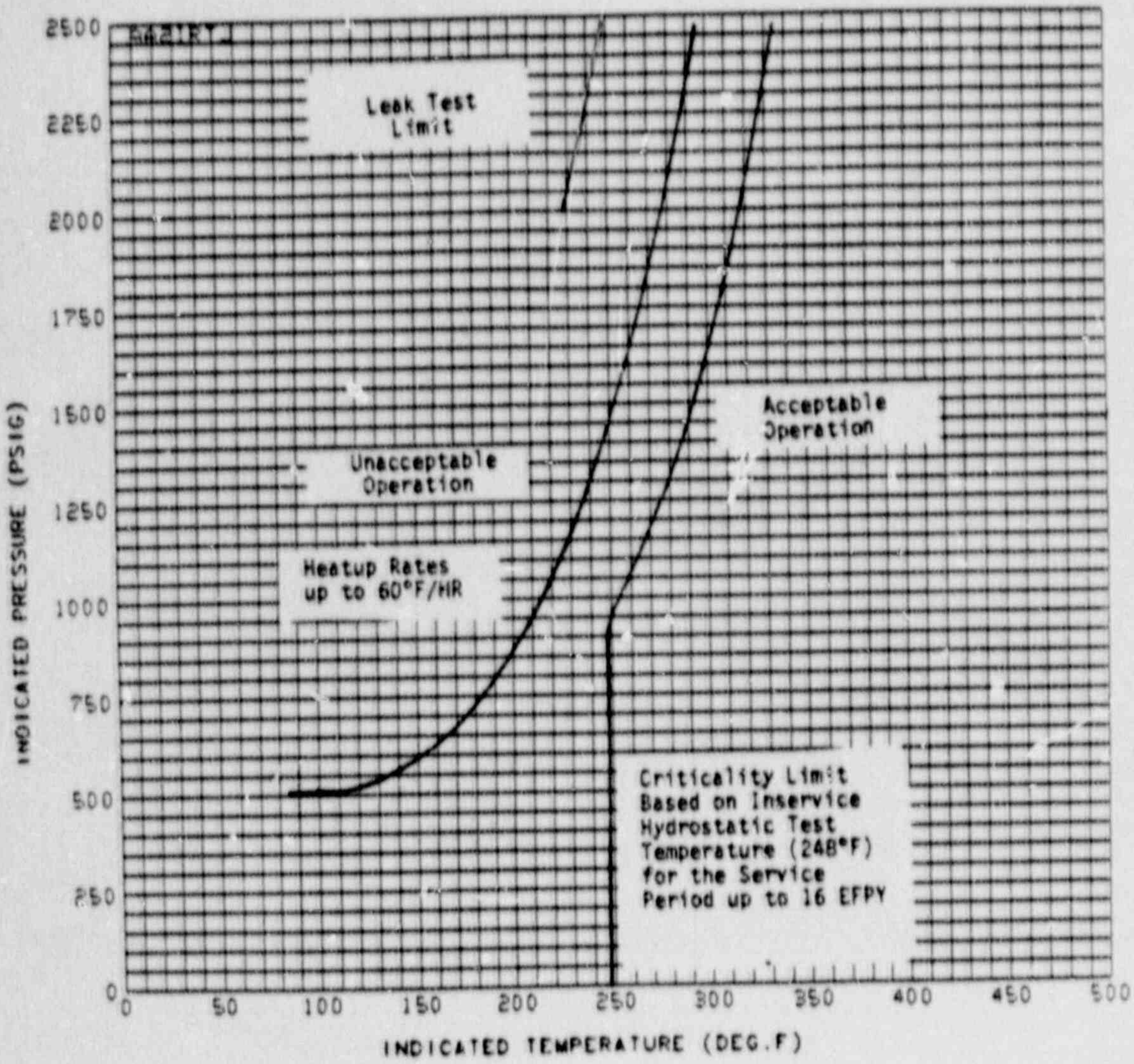

Figure A-2. Vogtle Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B8805-2

INITIAL RT_{NDT}: 20°F

RT_{NDT} AFTER 16 EFPY: 1/4T, 104°F

3/4T, 89°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

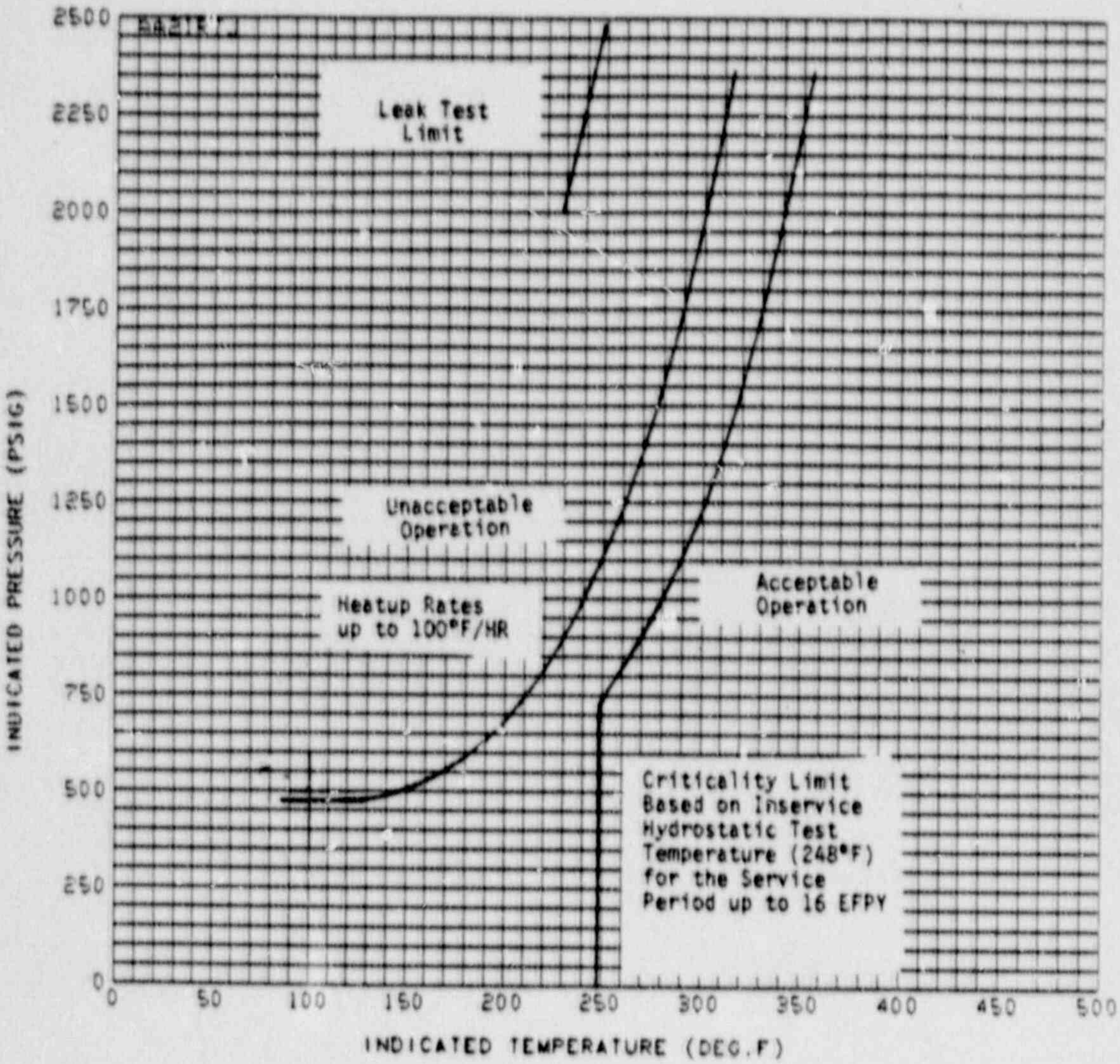

Figure A-3. Vogtle Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B8805-2

INITIAL RT_{NDT}: 20°F

RT_{NDT} AFTER 16 EFPY: 1/4T, 104°F

3/4T, 89°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

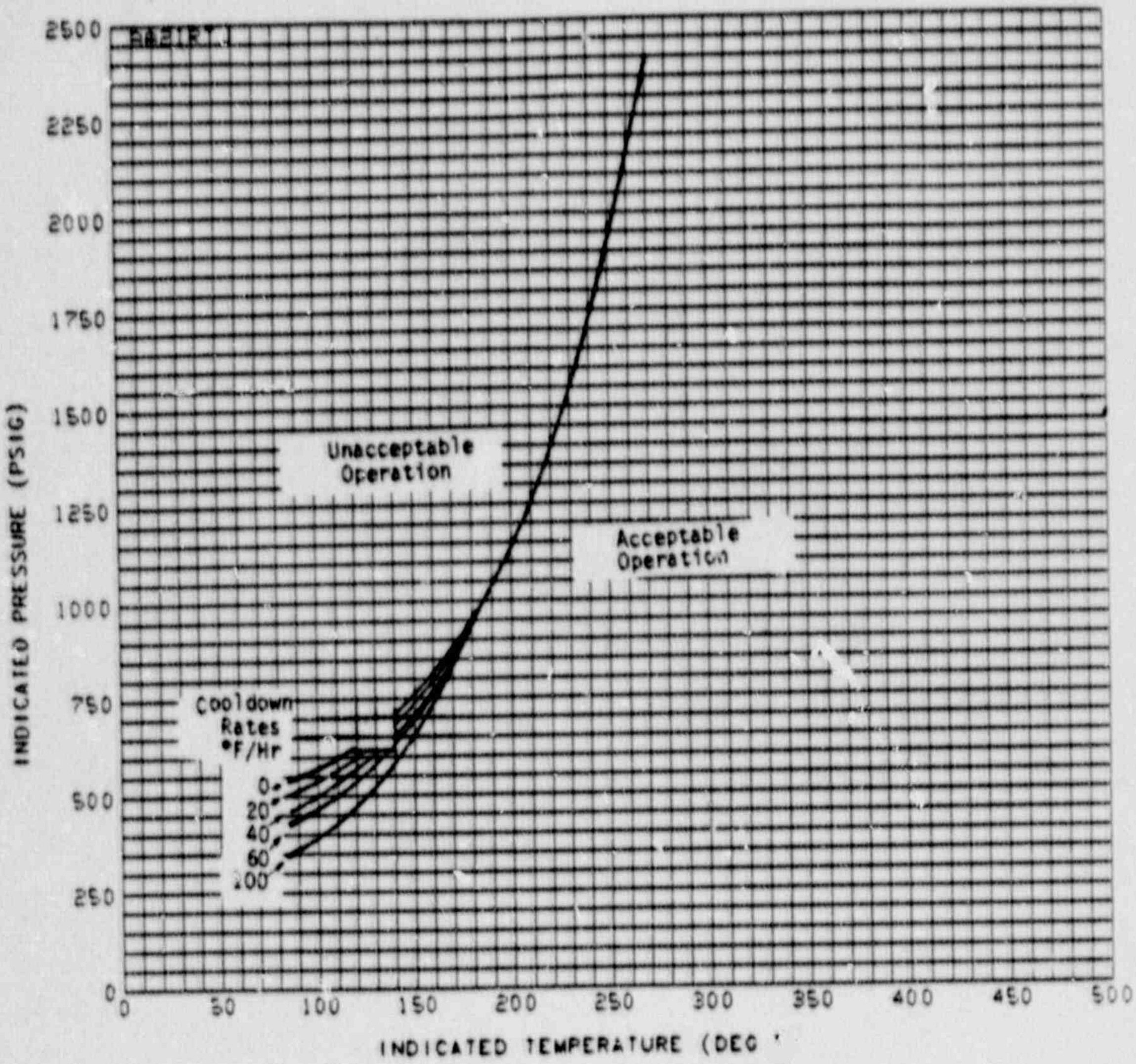

Figure A-4. Vogtle Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 16 EFPY

TABLE 2
VOGTLE UNIT 1 REACTOR VESSEL
MATERIAL PROPERTIES (UNIRRADIATED)

Component	Material Code No.	Cu (%)	Ni (%)	NDTT (°F)	RT_{NDT} (°F)	Shelf Energy NMWD (b) (ft-lb)
Closure Head Dome	B8807-1	0.16	0.67	-50	15	88
Closure Head Torus	B8808-1	0.14	0.56	-30	8	85
Closure Head Flange	B8801-1	-	0.70	20	20	132
Vessel Flange	B8802-1	-	0.71	0	0	119
Inlet Nozzle	B8809-1	-	0.86	-20	-20	107
Inlet Nozzle	B8809-2	-	0.84	-10	-10	95
Inlet Nozzle	B8809-3	-	0.82	-10	-10	117
Inlet Nozzle	B8809-4	-	0.87	-20	-20	105
Outlet Nozzle	B8810-1	-	0.82	-10	-10	>124
Outlet Nozzle	B8810-2	-	0.79	-10	-10	>100
Outlet Nozzle	B8810-3	-	0.77	-10	-10	>102
Outlet Nozzle	B8810-4	-	0.80	-10	-10	> 75
Nozzle Shell	B8804-1	0.14	0.62	-10	28	94
Nozzle Shell	B8804-2	0.10	0.58	-40	15	104
Nozzle Shell	B8804-3	0.14	0.69	-30	40	92
Inter. Shell	B8805-1	0.08	0.59	0	0	90
Inter. Shell	B8805-2	0.08	0.59	-10	20	100
Inter. Shell	B8805-3	0.06	0.60	-20	30	107
Lower Shell	B8606-1	0.05	0.59	-50	20	116
Lower Shell	B8606-2	0.05	0.58	-10	20	113
Lower Shell	B8606-3	0.06	0.64	-20	10	118
Bottom Head Torus	B8813-1	0.13	0.50	-40	-10	88
Bottom Head Dome	B8812-1	0.10	0.53	-40	-28	122
Inter & Lower Shell Vert. Weld Seams and Girth Seam	G1.43 (a)	0.03	0.10	-80	-80	>129

(a) Weld wire heat 83653 and Linde 0091 Flux lot 8536

(b) Normal to major working direction

6.0 REFERENCES

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

A-2. "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

A-3. ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendixes, "Rules for Construction of Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure," pp. 558-563, 1986 Edition, American Society of Mechanical Engineers, New York, 1986.

A-4. Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 48 No. 104, May 27, 1983.

APPENDIX A

HEATUP AND COOLDOWN DATA POINTS

GAE COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE B8805-2 04/13/99

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOVIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)	
1	85.000	537.81	
2	90.000	548.48	
3	95.000	559.88	
4	100.000	572.07	
5	105.000	585.08	
6	110.000	599.18	
7	115.000	614.36	
8	120.000	~~630.65~~	
9	125.000	~~648.06~~	
10	130.000	~~666.09~~	621
11	135.000	~~687.04~~	
12	140.000	~~709.01~~	
13	145.000	732.03	
14	150.000	757.22	
15	155.000	784.06	
16	160.000	813.12	
17	165.000	844.19	
18	170.000	877.82	
19	175.000	913.59	
20	180.000	952.14	
21	185.000	993.57	
22	190.000	1038.09	
23	195.000	1085.90	
24	200.000	1137.21	
25	205.000	1192.29	
26	210.000	1251.40	
27	215.000	1314.83	
28	220.000	1382.95	
29	225.000	1456.83	
30	230.000	1534.29	
31	235.000	1618.21	
32	240.000	1708.16	
33	245.000	1804.84	
34	250.000	1907.63	
35	255.000	2017.68	
36	260.000	2135.77	
37	265.000	2261.80	
38	270.000	2396.03	

GAE COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE B8805-2 04/13/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = ADVIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	499.72
2	90.000	510.87
3	95.000	522.80
4	100.000	535.68
5	105.000	549.61
6	110.000	564.58
7	115.000	580.60
8	120.000	597.91

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)	
9	125.000	616.80	
10	130.000	~~[illegible]~~	621
11	135.000	~~[illegible]~~	
12	140.000	~~[illegible]~~	
13	145.000	706.29	
14	150.000	733.00	
15	155.000	761.97	

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
16	160.000	792.02
17	165.000	826.20
18	170.000	862.20
19	175.000	900.74
20	180.000	942.16
21	185.000	986.69
22	190.000	1034.53

GAE COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE B8805-2 04/13/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH - ADWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	461.11	9	125.000	585.31	16	160.000	774.40
2	90.000	472.90	10	130.000	606.77	17	165.000	810.27
3	95.000	485.53	11	135.000	~~[illegible]~~ 621	18	170.000	848.78
4	100.000	499.23	12	140.000	~~654.64~~	19	175.000	890.24
5	105.000	514.03	13	145.000	681.28	20	180.000	934.61
6	110.000	529.96	14	150.000	710.15	21	185.000	982.73
7	115.000	547.02	15	155.000	741.04	22	190.000	1034.36
8	120.000	565.51						

GAE COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE B8805-2 04/13/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH - ADWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	422.06
2	90.000	434.57
3	95.000	448.01
4	100.000	462.60
5	105.000	478.37
6	110.000	495.25
7	115.000	513.61
8	120.000	533.37
9	125.000	554.80
10	130.000	577.57
11	135.000	602.25
12	140.000	~~[illegible]~~ 621
13	145.000	657.61
14	150.000	688.43
15	155.000	721.84
16	160.000	757.67
17	165.000	796.24
18	170.000	837.73
19	175.000	882.44
20	180.000	930.53
21	185.000	982.34
22	190.000	1038.07

GAE COOLDOWN CURVES REG. GUIDE 1.99,REV.2 FOR PLATE B8805-2 04/13/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH - AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	342.66	8	120.000	470.29	15	155.000	688.04
2	90.000	356.83	9	125.000	494.83	16	160.000	730.81
3	95.000	372.28	10	130.000	521.42	17	165.000	776.03
4	100.000	388.88	11	135.000	550.03	18	170.000	824.42
5	105.000	406.86	12	140.000	580.99	19	175.000	876.78
6	110.000	426.47	13	145.000	614.35	20	180.000	933.16
7	115.000	447.50	14	150.000	650.27			

GAE 60F/HR HEATUP CURVE REG. GUIDE 1.99 REV.2 FOR PLATE B8805-2 04/13/89

THE FOLLOWING DATA WERE CALCULATED FOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (16.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	228
2485	248

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	21776	91846
2485	26803	114484

GAE 60F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 FOR PLATE B8805-2 04/13/89

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)			INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~507.04~~		16	160.000	628.06	30	230.000	1175.03
2	90.000	~~530.54~~		17	165.000	651.42	31	235.000	1239.56
3	95.000	~~534.44~~	510.71	18	170.000	675.89	32	240.000	1308.56
4	100.000	~~544.38~~		19	175.000	702.40	33	245.000	1382.82
5	105.000	~~541.07~~		20	180.000	731.06	34	250.000	1462.25
6	110.000	510.71		21	185.000	762.31	35	255.000	1547.26
7	115.000	513.21		22	190.000	795.84	36	260.000	1638.27
8	120.000	518.04		23	195.000	832.00	37	265.000	1735.81
9	125.000	525.21		24	200.000	871.14	38	270.000	1840.19
10	130.000	534.23		25	205.000	913.14	39	275.000	1951.71
11	135.000	545.42		26	210.000	958.28	40	280.000	2070.58
12	140.000	558.43		27	215.000	1006.84	41	285.000	2197.56
13	145.000	573.37		28	220.000	1058.95	42	290.000	2333.11
14	150.000	589.95		29	225.000	1114.80	43	295.000	2477.26
15	155.000	606.58							

GAE 100F/HR HEATUP CURVE 04/13/89

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 100.0

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = (1-AOVIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~[illegible]~~	17	165.000	536.35	33	245.000	1048.62
2	90.000	~~534.30~~	18	170.000	551.29	34	250.000	1106.54
3	95.000	~~[illegible]~~	19	175.000	568.06	35	255.000	1168.01
4	100.000	~~[illegible]~~ 479.84	20	180.000	586.58	36	260.000	1235.87
5	105.000	~~[illegible]~~	21	185.000	607.21	37	265.000	1307.60
6	110.000	~~[illegible]~~	22	190.000	629.83	38	270.000	1384.74
7	115.000	~~[illegible]~~	23	195.000	654.50	39	275.000	1467.21
8	120.000	~~[illegible]~~	24	200.000	681.35	40	280.000	1555.63
9	125.000	479.84	25	205.000	710.78	41	285.000	1650.20
10	130.000	480.69	26	210.000	742.49	42	290.000	1751.29
11	135.000	483.46	27	215.000	776.84	43	295.000	1858.40
12	140.000	487.86	28	220.000	814.20	44	300.000	1974.80
13	145.000	494.15	29	225.000	854.36	45	305.000	2098.07
14	150.000	502.14	30	230.000	897.68	46	310.000	2229.65
15	155.000	511.87	31	235.000	944.36	47	315.000	2369.60
16	160.000	523.29	32	240.000	994.57			