WCAP 11415

WESTINGHOUSE CLASS 3
CUSTOMER DESIGNATED DISTRIBUTION

ANALYSIS OF CAPSULE V FROM THE VIRGINIA ELECTRIC AND POWER COMPANY SURRY UNIT 1 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

S. E. Yanichko
V. A. Perone

February 1987

Work performed under Shop Order No. VCKJ-106

APPROVED: K. R. Balkey for
T. A. Meyer, Manager
Structural Materials and Reliability Technology

Prepared by Westinghouse for the Virginia Electric and Power Company

WESTINGHOUSE ELECTRIC CORPORATION
Power Systems Division
P.O. Box 2728
Pittsburgh, Pennsylvania 15230-2728

PREFACE

This report has been technically reviewed and verified.

	Reviewer	
Sections 1 through 5 and 7	C. C. Heinecke	C C Heinecke
Section 6	S. L. Anderson	S. L. Anderson

TABLE OF CONTENTS

Section	Title	Page
1	SUMMARY OF RESULTS	1-1
2	INTRODUCTION	2-1
3	BACKGROUND	3-1
4	DESCRIPTION OF PROGRAM	4-1
5	TESTING OF SPECIMENS FROM CAPSULE V	5-1
	5-1. Overview	5-1
	5-2. Charpy V-Notch Impact Test Results	5-3
	5-3. Tension Test Results	5-4
	5-4. Wedge Opening Loading Tests	5-4
6	RADIATION ANALYSIS AND NEUTRON DOSIMETRY	6-1
	6-1. Introduction	6-1
	6-2. Discrete Ordinates Analysis	6-1
	6-3. Neutron Dosimetry	6-5
	6-4. Transport Analysis Results	6-10
	6-5. Dosimetry Results	6-12
	6-6. Surveillance Capsule Withdrawal Schedules	6-14
	6-7. Influence of an Energy Dependent Damage Model	6-14
7	SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE	7-1
8	REFERENCES	8-1

LIST OF ILLUSTRATIONS

Figure	Title	Page
4-1	Arrangement of Surveillance Capsules in the Surry Unit 1	4-5
4-2	Capsule V Diagram Showing Location of Specimens, Thermal Monitors, and Dosimeters	4-6
5-1	Irradiated Charpy V-Notch Impact Properties for Surry Unit 1 Reactor Vessel Lower Shell Plate C4415-1	5-17
5-2	Irradiated Charpy V-Notch Impact Properties for Surry Unit 1 Reactor Pressure Vessel Weld Metal	5-18
5-3	Irradiated Charpy V-Notch Impact Properties for Surry Unit 1 Reactor Pressure Vessel Weld Heat Affected Zone Metal	5-19
5-4	Irradiated Charpy V-Notch Impact Properties for Surry Unit 1 A533 Grade B Class 1 Correlation Monitor Material (HSST Plate 02)	5-20
5-5	Tensile Properties for Surry Unit 1 Reactor Vessel Lower Shell Plate C4415-1	5-21
5-6	Tensile Properties for Surry Unit 1 Reactor Vessel Weld Metal	5-22
6-1	Surry Unit 1 Reactor Geometry	6-39
6-2	Reactor Vessel Surveillance Capsule	6-40
6-3	Surry Unit 1 Maximum Fast Neutron (E > 1 MeV) Fluence at the Beltline Weld Locations as a Function of Full Power Operating Time	6-41
6-4	Surry Unit 1 Maximum Fast Neutron (E > 1 MeV) Fluence at the Center of the Surveillance Capsules as a Function of Full Power Operation Time	6-42
6-5	Surry Unit 1 Maximum Fast Neutron (E > 1.0 MeV) Fluence at the Pressure Vessel Inner Radius as a Function of Azimuthal Angle	6-43
6-6	Surry Unit 1 Relative Radial Distribution of Fast Neutron (E > 1.0 MeV) Flux and Fluence Within the Pressure Vessel Wall	6-44
6-7	Surry Unit 1 Relative Axial Variation of Fast Neutron (E > 1.0 MeV) Flux and Fluence Within the Pressure Vessel Wall	6-45

LIST OF TABLES

Table	Title	Page
4-1	Chemical Composition of the Surry Unit 1 Reactor Vessel Surveillance Materials	4-3
4-2	Heat Treatment of the Surry Unit 1 Reactor Vessel Surveillance Materials	4-4
5-1	Charpy V-Notch Impact Data for the Surry Unit 1 Lower Shell Plate C4415-1 Irradiated at 550°F, Fluence 1.94 x 10^{19} n/cm^2 (E > 1 MeV)	5-5
5-2	Charpy V-Notch Impact Data for the Surry Unit 1 Pressure Vessel Weld Metal Irradiated at 550°F, Fluence 1.94 x 10^{19} n/cm^2 (E > 1 MeV)	5-6
5-3	Charpy V-Notch Impact Data for the Surry Unit 1 Pressure Vessel Weld Heat Affected Zone Metal Irradiated at 550°F, Fluence 1.94 x 10^{19} n/cm^2 (E > 1 MeV)	5-7
5-4	Charpy V-Notch Impact Data for the Surry Unit 1 A533 Grade B Class 1 Correlation Monitor Material (HSST Plate 02) at 550°F, Fluence 1.94 x 10^{19} n/cm^2 (E > 1 MeV)	5-8
5-5	Instrumented Charpy Impact Test Results for Surry Unit 1 Lower Shell Plate C4415-1	5-9
5-6	Instrumented Charpy Impact Test Results for Surry Unit 1 Weld Metal	5-10
5-7	Instrumented Charpy Impact Test Results for Surry Unit 1 Weld Heat Affected Zone Metal	5-11
5-8	Instrumented Charpy Impact Test Results for Surry Unit 1 A533 Grade B Class 1 Correlation Monitor Material (HSST Plate 02)	5-12
5-9	The Effect of 550°F Irradiation at 1.94 x 10^{19} (E > 1 MeV) on the Notch Toughness Properties of The Surry Unit 1 Reactor Vessel Materials	5-13
5-10	Summary of Surry Unit 1 Reactor Vessel Surveillance Capsule Charpy Impact Test Results	5-14
5-11	Comparison of Measured ΔRT_{NDT} Versus Regulatory Guide 1.99 Revision 2 Predicted RT_{NDT}	5-15
5-12	Tensile Properties for Surry Unit 1 Reactor Vessel Material Irradiated to 1.94 x 10^{19} n/cm^2	5-16

LIST OF TABLES (Cont)

Table	Title	Page
6-1	47 Energy Group Structure	6-16
6-2	Nuclear Parameters for Neutron Flux Monitors	6-17
6-3	Surry Unit 1 Calculated Fast Neutron (E > 1.0 MeV) Exposure at the Pressure Vessel Inner Radius - 0° Azimuthal Angle	6-18
6-4	Surry Unit 1 Calculated Fast Neutron (E > 1.0 MeV) Exposure at the Pressure Vessel Inner Radius - 15° Azimuthal Angle	6-19
6-5	Surry Unit 1 Calculated Fast Neutron (E > 1.0 MeV) Exposure at the Pressure Vessel Inner Radius - 30° Azimuthal Angle	6-20
6-6	Surry Unit 1 Calculated Fast Neutron (E > 1.0 MeV) Exposure at the Pressure Vessel Inner Radius - 45° Azimuthal Angle	6-21
6-7	Surry Unit 1 Calculated Fast Neutron (E > 1.0 MeV) Exposure at the 15° Surveillance Capsule Center	6-22
6-8	Surry Unit 1 Calculated Fast Neutron (E > 1.0 MeV) Exposure at the 25° Surveillance Capsule Center	6-23
6-9	Surry Unit 1 Calculated Fast Neutron (E > 1.0 MeV) Exposure at the 35° Surveillance Capsule Center	6-24
6-10	Surry Unit 1 Calculated Fast Neutron (E > 1.0 MeV) Exposure at the 45° Surveillance Capsule Center	6-25
6-11	Irradiation History of Surry Unit 1 Reactor Vessel Surveillance Capsule V	6-26
6-12	Measured Flux Monitor Activities from Surry Unit 1, Capsule T	6-30
6-13	Measured Flux Monitor Activities from Surry Unit 1, Capsule W	6-31
6-14	Measured Flux Monitor Activities from Surry Unit 1, Capsule V	6-32
6-15	Calculated Neutron Energy Spectra at the Center of Surry Unit 1 Surveillance Capsules	6-33
6-16	Spectrum Averaged Reaction Cross-Sections at the Center of Surry Unit 1 Surveillance Capsules	6-35

LIST OF TABLES (Cont)

Table	Title	Page
6-17	Thermal Neutron Flux Data from Capsules T, W, and V	6-36
6-18	Comparison of Measured and Calculated Fast Neutron Fluence for Capsules T, W, and V	6-37
6-19	dPa/ϕ (E > 1.0 MeV) Ratios for Surry Unit 1	6-38

SECTION 1
SUMMARY OF RESULTS

The analysis of the reactor vessel material contained in Capsule V, the third surveillance capsule to be removed from the Surry Unit 1 reactor pressure vessel, led to the following conclusions:

- The capsule received an average fast neutron fluence (E > 1.0 MeV) of 1.94×10^{19} n/cm^2.

- Irradiation of the reactor vessel lower shell plate C4415-1, to 1.94×10^{19} n/cm, resulted in 30 and 50 ft-lb transition temperature increases of 110°F and 130°F, respectively for specimens oriented parallel to the major working direction (longitudinal orientation). The upper shelf energy decreased from 125 to 116 ft-lb as a result of the irradiation.

- Weld metal irradiated to 1.94×10^{19} n/cm^2 resulted in a 30 ft-lb transition temperature increase of 240°F. The upper shelf energy decreased to ~ 50 ft-lb as a result of the irradiation.

- Comparison of the 30 ft-lb transition temperature increases (ΔRT_{NDT}) for the Surry Unit 1 surveillance material with predicted increases using the methods of NRC Regulatory Guide 1.99, Revision 2, shows that the plate material and weld metal transition temperature increase were in relatively good agreement with predicted increases.

SECTION 2
INTRODUCTION

This report presents the results of the examination of Capsule V, the third capsule to be removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the Virginia Electric and Power Company Surry Unit 1 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Surry 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 Yanichko.[1] The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E-185-73, "Recommended Practice for Surveillance Tests for Nuclear Reactors".[2] Westinghouse Nuclear Energy Systems personnel were contracted for the preparation of procedures for removing the capsule from the reactor and its shipment to the Westinghouse Research and Development Laboratory, where the postirradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

This report summarizes testing and the postirradiation data obtained from surveillance Capsule V removed from the Surry Unit 1 reactor vessel and discusses the analysis of the data. The data are also compared to capsule T[3] which was removed from the reactor in 1974 and capsule W[4] which was removed in 1978. It sould be noted that only dosimetry was measured for the capsule W. A new reactor vessel surveillance capsule withdrawal schedule was developed to meet the requirements of ASTM E-185-82, as proposed by Babcock and Wilcox[5].

SECTION 3
BACKGROUND

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as SA533 Grade B Class 1 plate (base material of the Surry Unit 1 reactor pressure vessel beltline) are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness 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, in turn, the operating limits of nuclear power plants can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel steel can be monitored by a reactor surveillance program such as the Surry Unit 1 Reactor Vessel Radiation Surveillance Program,[1] in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft lb temperature (ΔRT_{NDT}) due to irradiation is added to the 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, in turn, to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials.

SECTION 4
DESCRIPTION OF PROGRAM

Eight surveillance capsules for monitoring the effects of neutron exposure on the Surry 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 thermal shield 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 V was removed after 8.02 effective full power years of plant operation. The capsule contained Charpy V-notch impact and tensile specimens from the lower shell plate C4415-1 and submerged arc weld metal representative of the beltline weld seams of the reactor vessel, WOL specimens from the weld metal and Charpy V-notch specimens from weld heat-affected zone (HAZ) material (Figure 4-2). All heat-affected zone specimens were obtained from within the HAZ of plate C4415-1 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 was performed on an irradiated Charpy specimen from the weld metal and plate C4415-1 and is reported in table 4-1.

All test specimens were machined from the 1/4 thickness location of the plate after stress relieving. Test specimens represent material taken at least one plate thickness from the quenched edges of the plate. Base metal Charpy V-notch impact specimens were oriented with the longitudinal axis of the specimen parallel to the major working direction of the plate (longitudinal orientation). Charpy V-notch and tensile specimens from the weld metal were oriented with the longitudinal axis of the specimens transverse to the welding direction. The WOL specimens in Capsule V were machined such that the simulated crack in the specimen would propagate parallel to the weld direction.

Capsule V contained dosimeter wires of pure copper, nickel, and aluminum-cobalt (cadmium-shielded and unshielded). 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 V are shown in Figure 4-2.

TABLE 4-1

CHEMICAL COMPOSITION OF
THE SURRY UNIT 1 REACTOR VESSEL
SURVEILLANCE MATERIALS

Element	Intermediate Shell Plate C4326-1	Lower Shell Plate C4415-1		Weld Metal(d)		Correlation Monitor
C	0.23	0.22	0.245(b)	0.10	0.185(c)	0.22
Mn	1.35	1.33	1.46 (b)	1.49	1.47 (c)	1.48
P	0.008	0.014	0.012(b)	0.011	0.011(c)	0.012
S	0.015	0.014	0.017(b)	0.010	0.017(c)	0.018
Si	0.23	0.23	0.42 (b)	0.37	0.43 (c)	0.25
Ni	0.55	0.50	0.569(b)	0.68	0.643(c)	0.68
Cr	0.069	0.078	0.105(b)	0.076	0.074(c)	-
V	0.001(a)	0.001(a)	0.004(b)	0.001	<0.002(c)	-
Mo	0.55	0.55	0.618(b)	0.46	0.405(c)	0.52
Co	0.014	0.015	0.006(b)	0.001	0.011(c)	-
Cu	0.11	0.11	0.115(b)	0.25	0.243(c)	0.14
Sn	0.008	0.008	-	-	-	-
Zn	0.001(a)	0.001(a)	-	-	-	-
Al	0.037	0.036	-	0.013	-	-
N_2	0.007	0.007	-	0.008	-	-
Ti	0.001(a)	0.001(a)	-	-	-	-
Zr	0.002	0.002	-	-	-	-
As	0.007	0.007	-	-	-	-
B	0.003(a)	0.003(a)	-	-	-	-

[a] Not detected. The number indicates the minimum limit of detection.
[b] Analysis performed on irradiated Charpy plate specimen V-25.
[c] Analysis performed on irradiated Charpy weld specimen W-10.
[d] Surveillance weld fabricated from same heat of weld wire (299L44) and Linde 80 flux lot (8596) as used in the vessel lower shell vertical weld seam (L2).

TABLE 4-2

HEAT TREATMENT OF THE SURRY UNIT 1
REACTOR VESSEL SURVEILLANCE MATERIALS

Material	Heat Treatment	
Intermediate shell (Plate C4326-1)	1650°-1700° - 9 hours	- Water-quenched
	1210°F - 9 hours	- Air-cooled
	1125°F - 15-1/2 hours	- Furnace cooled to 600°F
Lower shell (Plate C4415-1)	1650-1700°F - 9 hours	- Water-quenched
	1200°F - 9 hours	- Air-cooled
	1125°F - 15-1/2 hours	- Furnace-cooled to 600°F
Weldment	1125°F - 15-1/2 hours	- Furnace-cooled to 600°F
Correlation Monitor	1675 $\pm$ 25°F - 4 hours	- Air-cooled
	1600 $\pm$ 25°F - 4 hours	- Water-quenched
	1225 $\pm$ 25°F - 4 hours	- Furnace-cooled
	1150 $\pm$ 25°F - 40 hours	- Furnace-cooled to 600°F

14817.1

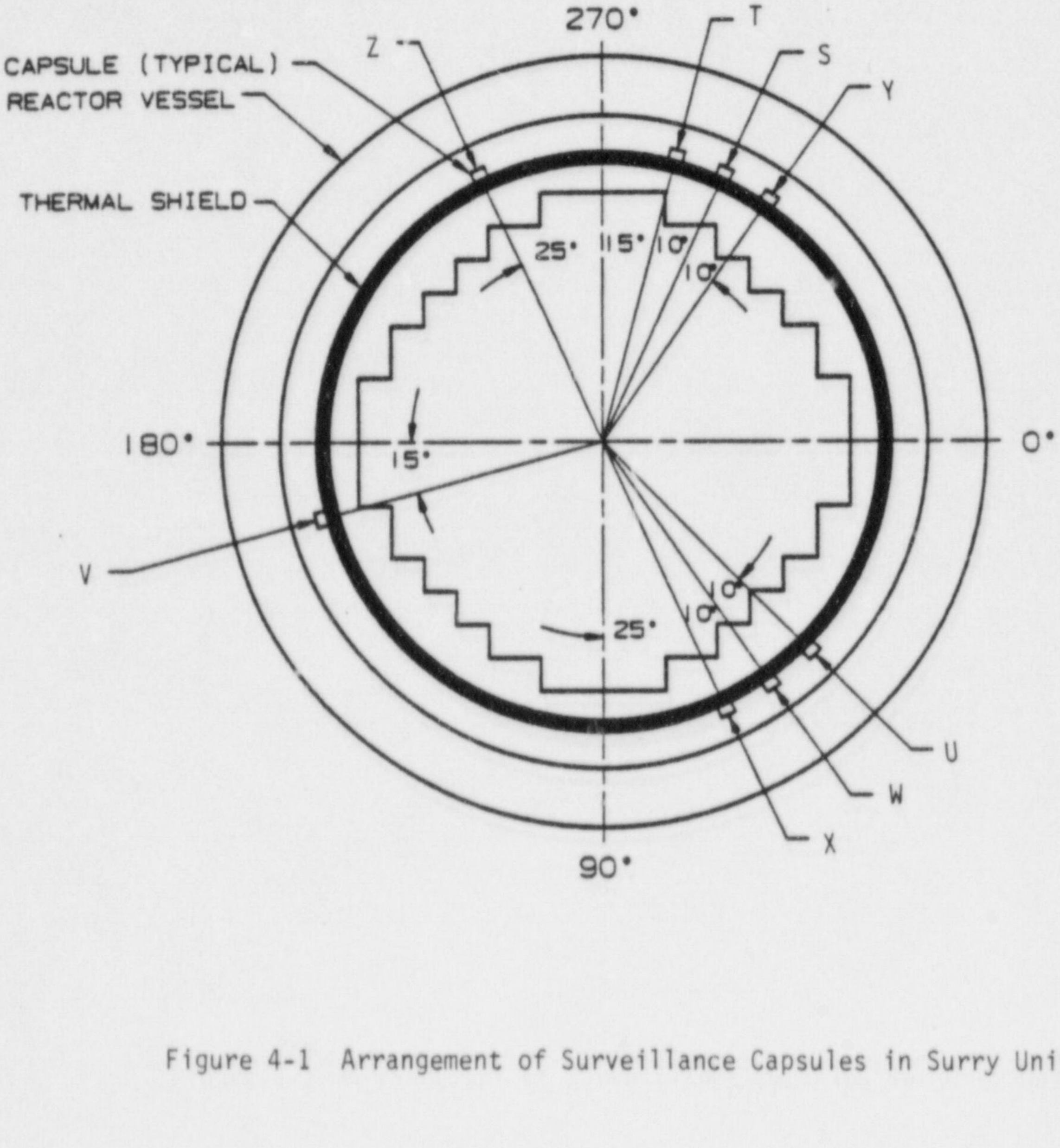

Figure 4-1 Arrangement of Surveillance Capsules in Surry Unit 1

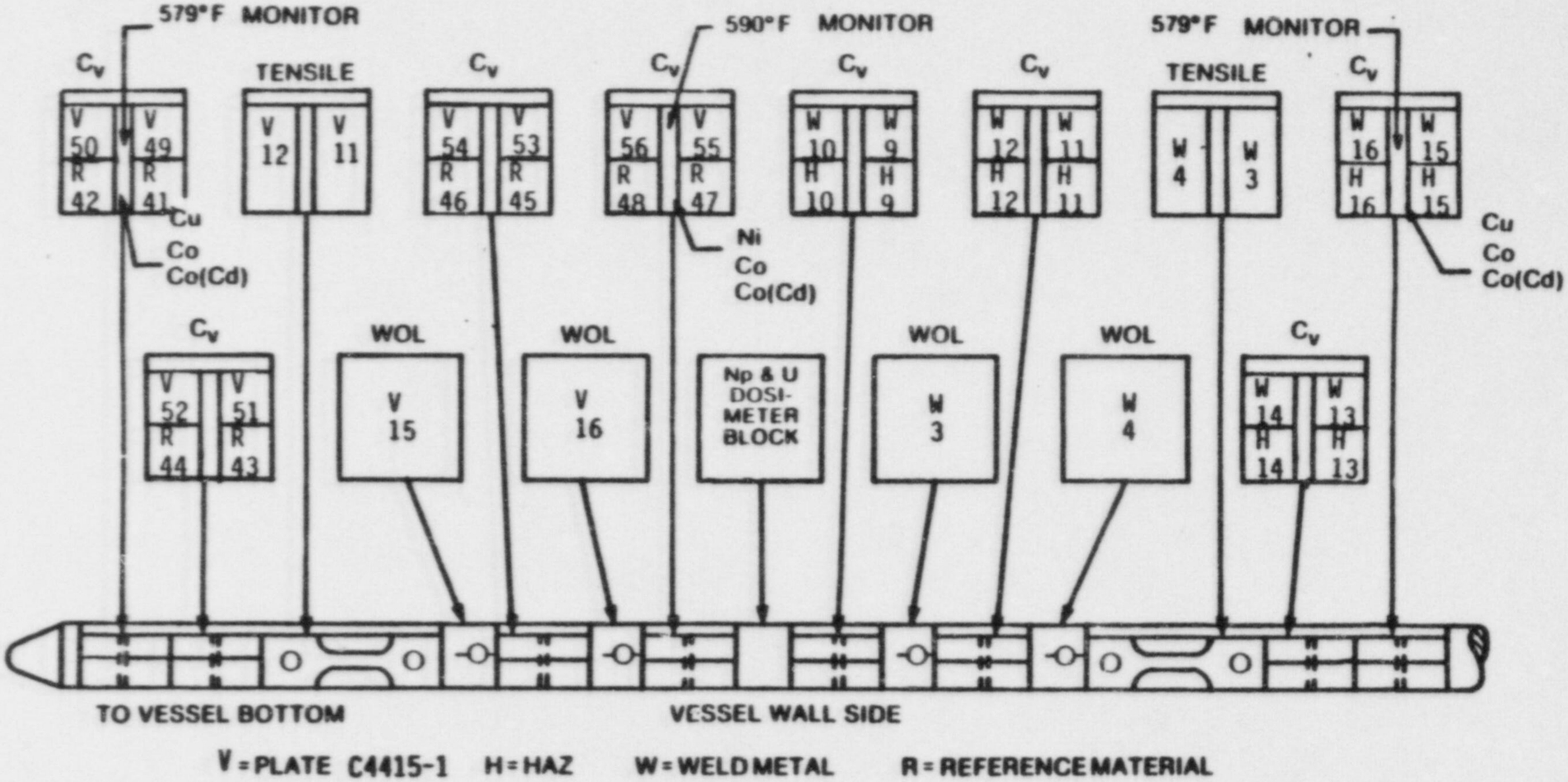

Figure 4-2 Capsule V Diagram Showing Location of Specimens, Thermal Monitors and Dosimeters

SECTION 5
TESTING OF SPECIMENS FROM CAPSULE V

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, ASTM Specification E185-82 and Westinghouse Procedure MHL 8402, Revision 0 as modified by RMF Procedures 8102 and 8103.

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-7723.[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 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, in the Charpy transition region, 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.

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. All pull rods, grips, and pins were made of Inconel 718 hardened to R_C45. 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 V irradiated at 1.94×10^{19} n/cm^2 are presented in Tables 5-1 through 5-8 and Figures 5-1 through 5-4. The transition temperature increases and upper shelf energy decreases for the Capsule V materials are shown in Table 5-9. Table 5-10 summarizes the Charpy impact test results from Capsule V with the previous Capsule T.

Irradiation of vessel lower shell plate C4415-1 material (longitudinal orientation) specimens to 1.94×10^{19} m/cm^2 (Figure 5-1) resulted in 30 and 50 ft-lb transition temperature increases of 110°F and 130°F respectively, and an upper shelf energy decrease of 9 ft lbs.

Weld metal irradiated to 1.94×10^{19} n/cm^2 (Figure 5-2) resulted in a 30 ft-lb transition temperature increase of 240°F and an upper shelf energy decrease of ~ 20 ft-lb which resulted in an upper shelf energy of approximately 50 ft-lb.

Weld HAZ metal irradiated to 1.94×10^{19} n/cm^2 (Figure 5-3) resulted in a 30 and 50 ft-lb transition temperature increases of 80°F and 85°F, respectively, and an upper shelf energy decrease of 8 ft-lb. However because of the large data scatter these values are considered to be highly questionable.

Correlation monitor material (HSST Plate 02) irradiated to 1.94×10^{19} n/cm^2 (Figure 5-4) resulted in 30 and 50 ft-lb transition temperature increases of 145 and 150°F respectively. These increases are in good agreement with other irradiation program tests.

Table 5-11 shows a comparison of the 30 ft-lb transition temperature (ΔRT_{NDT}) increases for the various Surry Unit 1 surveillance materials with predicted increases using the methods of NRC Regulatory Guide 1.99, Revision 2.[6] This comparison shows that the transition temperature increase resulting from irradiation to 0.281 and 1.94×10^{19} n/cm^2 is in relatively good agreement with the increase predicted by the Guide.

5-3. TENSION TEST RESULTS

The results of tension tests performed on plate C4415-1 (longitudinal orientation) and weld metal irradiated to 1.94×10^{19} n/cm^2 are shown in Table 5-12 and Figures 5-5 and 5-6, respectively. These results shown that irradiation produced an increase of 14 to 17 ksi in 0.2 percent yield strength for plate C4415-1 and approximately a 23 to 27 ksi increase for the weld metal.

5-4. WEDGE OPENING LOADING TESTS

At the request of the Virginia Electric and Power Company, Wedge Open Loading (WOL) specimen will not be tested. The specimens will be stored at the Hot Cell at the Westinghouse R&D Center.

TABLE 5-1

CHARPY V-NOTCH IMPACT DATA FOR THE SURRY UNIT 1
LOWER SHELL PLATE C4415-1
IRRADIATED AT 550°F, FLUENCE 1.94 x 10^{19} n/cm^2 (E > 1 MeV)

Sample No.	Temperature °F (°C)	Impact Energy ft-lbs (Joules)	Lateral Expansion mils (mm)	% Shear
V50	50 (10)	11.0 (15.0)	10.0 (0.25)	3
V52	100 (38)	37.0 (50.0)	30.5 (0.77)	[illegible]
V49	150 (66)	50.0 (68.0)	43.0 (1.09)	41
V53	200 (93)	72.0 (97.5)	56.5 (1.44)	66
V54	250 (121)	117.0 (158.5)	79.5 (2.02)	100
V55	300 (149)	116.0 (157.5)	78.5 (1.99)	100
V51	400 (204)	115.0 (156.0)	77.5 (1.97)	100

TABLE 5-2

CHARPY V-NOTCH IMPACT DATA FOR THE SURRY UNIT 1
PRESSURE VESSEL WELD METAL IRRADIATED AT 550°F,
FLUENCE 1.94 x 10^{19} n/cm^2 (E > 1 MeV)

Sample No.	Temperature °F (°C)	Impact Energy ft-lbs (Joules)	Lateral Expansion mils (mm)	% Shear
W12	50 (10)	4.0 (5.5)	3.5 (0.09)	0
W14	150 (66)	17.0 (23.0)	19.5 (0.50)	12
W13	200 (93)	22.0 (30.0)	17.5 (0.44)	28
W16	250 (121)	39.0 (53.0)	28.5 (0.72)	73
W10	250 (121)	33.0 (44.5)	32.0 (0.81)	52
W15	300 (149)	41.0 (55.5)	31.0 (0.79)	96
W11	400 (204)	47.0 (63.5)	41.0 (1.04)	100
W9	450 (232)	52.0 (70.5)	45.5 (1.16)	100

TABLE 5-3

CHARPY V-NOTCH IMPACT DATA FOR THE SURRY UNIT 1
PRESSURE VESSEL WELD HEAT AFFECTED ZONE METAL
IRRADIATED AT 550°F, FLUENCE 1.94 x 10^{19} n/cm^2 (E > 1 MeV)

Sample No.	Temperature °F (°C)	Impact Energy ft-lbs (Joules)	Lateral Expansion mils (mm)	% Shear
H10	-25 (-32)	12.0 (16.5)	10.0 (0.25)	12
H9	25 (-4)	28.0 (38.0)	20.5 (0.52)	31
H13	50 (10)	53.0 (72.0)	31.5 (0.80)	51
H11	100 (38)	87.0 (118.0)	59.0 (1.50)	87
H15	150 (66)	22.0 (30.0)	20.0 (0.51)	58
H14	200 (93)	81.0 (110.0)	61.5 (1.56)	100
H12	300 (149)	52.0 (70.5)	39.0 (0.99)	100
H16	400 (204)	110.0 (149.0)	70.0 (1.78)	100

TABLE 5-4

CHARPY V-NOTCH IMPACT DATA FOR THE SURRY UNIT 1
A533 GRADE B CLASS 1 CORRELATION MONITOR MATERIAL (HSST PLATE 02)
IRRADIATED AT 550°F, FLUENCE 1.94 x 10^{19} n/cm^2 (E > 1 MeV)

Sample No.	Temperature °F (°C)	Impact Energy ft-lbs (Joules)	Lateral Expansion mils (mm)	% Shear
R41	100 (38)	10.0 (13.5)	10.0 (0.25)	4
R43	150 (66)	21.0 (28.5)	18.5 (0.47)	10
R47	200 (93)	33.0 (44.5)	27.5 (0.70)	33
R48	200 (93)	33.0 (44.5)	23.0 (0.58)	26
R46	250 (121)	73.0 (99.0)	45.0 (1.14)	43
R44	300 (149)	92.0 (124.5)	69.0 (1.75)	92
R45	400 (204)	101.0 (137.0)	69.5 (1.77)	100
R42	450 (232)	98.0 (133.0)	72.5 (1.84)	100

TABLE 5-5

INSTRUMENTED CHARPY IMPACT TEST RESULTS

FOR SURRY UNIT 1

LOWER SHELL PLATE C4415-1

Sample No.	Test Temp. (°F)	Charpy Energy (FT LB)	Normalized Energies Charpy Ed/A (FT-LB/in^2)	Maximum Em/A (FT-LB/in^2)	Prop Ep/A (FT-LB/in^2)	Yield Load (KIPS)	Time to Yield (μSec)	Maximum Load (KIPS)	Time to Maximum (μSec)	Fracture Load (KIPS)	Arrest Load (KIPS)	Yield Stress (KSI)	Flow Stress (KSI)
V50	50	11.0	89	36	52	3.05	135	3.10	155	3.15	.25	102	102
V52	100	37.0	298	210	88	3.10	125	4.40	495	4.35	.65	103	124
V49	150	50.0	403	258	145	3.15	135	4.50	585	4.45	1.55	103	126
V53	200	72.0	580	282	297	2.90	135	4.25	670	3.85	2.4	95	118
V54	250	117.0	942	283	659	2.75	135	4.10	690	-	-	91	113
V55	300	116.0	934	270	664	2.90	140	4.15	645	-	-	96	117
V51	400	115.0	926	290	636	2.70	130	4.10	715	-	-	89	112

TABLE 5-6
INSTRUMENTED CHARPY IMPACT TEST RESULTS
FOR SURRY UNIT 1 WELD METAL

Sample No.	Test Temp. (°F)	Charpy Energy (FT LB)	Normalized Energies Charpy Ed/A (FT-LB/in²)	Maximum Em/A (FT-LB/in²)	Prop Ep/A (FT-LB/in²)	Yield Load (KIPS)	Time to Yield (μSec)	Maximum Load (KIPS)	Time to Maximum (μSec)	Fracture Load (KIPS)	Arrest Load (KIPS)	Yield Stress (KSI)	Flow Stress (KSI)
W12	50	4.0	32	16	16	-	-	2.80	85	2.80	.15	-	-
W14	150	17.0	137	89	48	3.10	125	3.80	270	3.75	.50	102	113
W13	200	22.0	177	87	90	2.55	105	3.55	275	3.55	.95	85	101
W10	250	33.0	266	144	122	3.20	115	4.10	350	4.00	2.75	106	121
W16	250	39.0	314	165	149	3.55	130	4.25	380	4.20	3.25	117	129
W15	300	41.0	330	140	190	3.05	125	3.80	370	-	-	101	113
W11	400	47.0	378	157	222	2.65	105	3.80	410	-	-	88	107
W9	450	52.0	419	184	235	3.40	140	4.20	435	-	-	113	126

TABLE 5-7
INSTRUMENTED CHARPY IMPACT TEST RESULTS
FOR SURRY UNIT 1 WELD HEAT AFFECTED ZONE METAL

Sample No.	Test Temp. (°F)	Charpy Energy (FT LB)	Normalized Energies Charpy Ed/A (FT-LB/in^2)	Normalized Energies Maximum Em/A (FT-LB/in^2)	Normalized Energies Prop Ep/A (FT-LB/in^2)	Yield Load (KIPS)	Time to Yield (μSec)	Maximum Load (KIPS)	Time to Maximum (μSec)	Fracture Load (KIPS)	Arrest Load (KIPS)	Yield Stress (KSI)	Flow Stress (KSI)
H10	-25	12.0	97	53	44	3.70	135	4.00	180	4.00	.30	122	127
H9	25	28.0	225	113	113	3.35	125	4.10	295	4.15	1.45	110	123
H13	50	53.0	427	225	202	.85	55	5.25	530	5.3	3.80	27	100
H11	100	87.0	701	290	410	3.85	135	4.95	580	3.70	1.50	128	146
H15	150	22.0	177	107	71	3.10	105	4.05	280	4.05	.80	102	118
H14	200	81.0	652	232	420	3.75	130	4.70	485	-	-	124	140
H12	300	52.0	419	155	264	3.15	135	3.95	400	-	-	105	118
H16	400	110.0	886	255	631	3.30	145	4.50	575	-	-	110	129

2200s-041387:10

TABLE 5-8

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR SURRY UNIT 1

A533 GRADE B CLASS 1 CORRELATION MONITOR MATERIAL (HSST PLATE 02)

Sample No.	Test Temp. (°F)	Charpy Energy (FT LB)	Normalized Energies Charpy Ed/A (FT-LB/in^2)	Normalized Energies Maximum Em/A (FT-LB/in^2)	Normalized Energies Prop Ep/A (FT-LB/in^2)	Yield Load (KIPS)	Time to Yield (μSec)	Maximum Load (KIPS)	Time to Maximum (μSec)	Fracture Load (KIPS)	Arrest Load (KIPS)	Yield Stress (KSI)	Flow Stress (KSI)
R41	100	10.0	81	50	31	3.30	110	3.60	160	3.75	.65	109	114
R43	150	21.0	169	125	44	3.20	100	3.90	325	3.90	.25	106	117
R48	200	33.0	266	141	125	2.75	135	3.75	405	3.75	1.40	91	107
R47	200	33.0	266	164	102	2.90	135	3.90	445	3.90	1.35	95	112
R46	250	73.0	588	263	325	2.70	120	4.05	640	4.00	2.75	89	112
R44	300	92.0	741	265	476	2.65	135	4.05	665	3.30	2.65	88	111
R45	400	101.0	813	234	579	2.60	135	3.90	610	-	-	87	108
R42	450	98.0	789	263	526	2.80	115	4.15	625	-	-	92	115

TABLE 5-9

EFFECT OF 550°F IRRADIATION AT 1.94×10^{19} n/cm^2 (E > MeV) ON THE NOTCH TOUGHNESS PROPERTIES OF THE SURRY UNIT 1 REACTOR VESSEL MATERIALS

	Average 50 ft-lb Temp (°F)			Average 35 mil Lateral Expansion Temp (°F)			Average 30 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 C4415-1	20	150	130	10	130	120	-10	100	110	125	116	9
Weld Metal	50	-	-	0	245	245	-15	225	240	70	49.5	20.5
HAZ Metal	-15	70	85	-20	70	90	-50	30	80	89	81	8
Correlation Material	75	225	150	65	225	160	45	190	145	123	100	23

TABLE 5-10

SUMMARY OF SURRY UNIT NO. 1

REACTOR VESSEL SURVEILLANCE CAPSULE CHARPY IMPACT TEST RESULTS

Material	Fluence 10^{19} n/cm^2	68 Joule 50 ft-lb Trans. Temp. Increase (°F)	41 Joule 30 ft-lb Trans. Temp. Increase (°F)	Decrease in Upper Shelf Energy (ft-lb)
Plate C4415-1 (Long)	0.281 (a)	60	50	5
	1.940 (b)	130	110	9
Weld Metal	0.281	250	165	17
	1.94	-	240	20.5
HAZ Metal	0.281	-	-	-
	1.94	85	80	8
Correlation	0.281	80	70	18
Monitor	1.94	150	145	23

(a) Capsule T

(b) Capsule V

TABLE 5-11

COMPARISON OF

MEASURED Δ RT_{NDT} VERSUS REGULATORY GUIDE 1.99 REVISION 2

PREDICTED ΔRT_{NDT} (a)

Material	Capsule	Fluence 10^{19} n/cm2	Δ RTNDT (30 ft-lb increase) RG. 1.99 Rev. 2 (°F)	Δ RTNDT (30 ft-lb increase) Measured (°F)
Plate C4415-1	T	0.281	47	50
	V	1.94	87	110
Weld Metal	T	0.281	121	165
	V	1.94	222	240
Correlation	T	0.281	65	70
Monitor	V	1.94	120	145

(a) Based on copper and nickel contents reported in WCAP 7723 [1].

TABLE 5-12

TENSILE PROPERTIES FOR SURRY UNIT 1

REACTOR VESSEL MATERIAL IRRADIATED TO 1.94 x 10^{19} n/cm^2

Sample No.	Material	Test Temp. (°F)	2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Stress (ksi)	Fracture Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
V12	Plate C4415-1	250	82.5	99.8	3.30	183.5	67.2	9.8	20.4	63
V11	Plate C4415-1	550	77.4	100.8	3.40	159.4	69.3	9.0	19.7	57
W4	WELD	250	90.7	102.3	3.90	180.9	79.5	[illegible]	19.7	56
W3	WELD	550	82.5	101.9	4.00	165.2	81.5	9.0	17.1	51

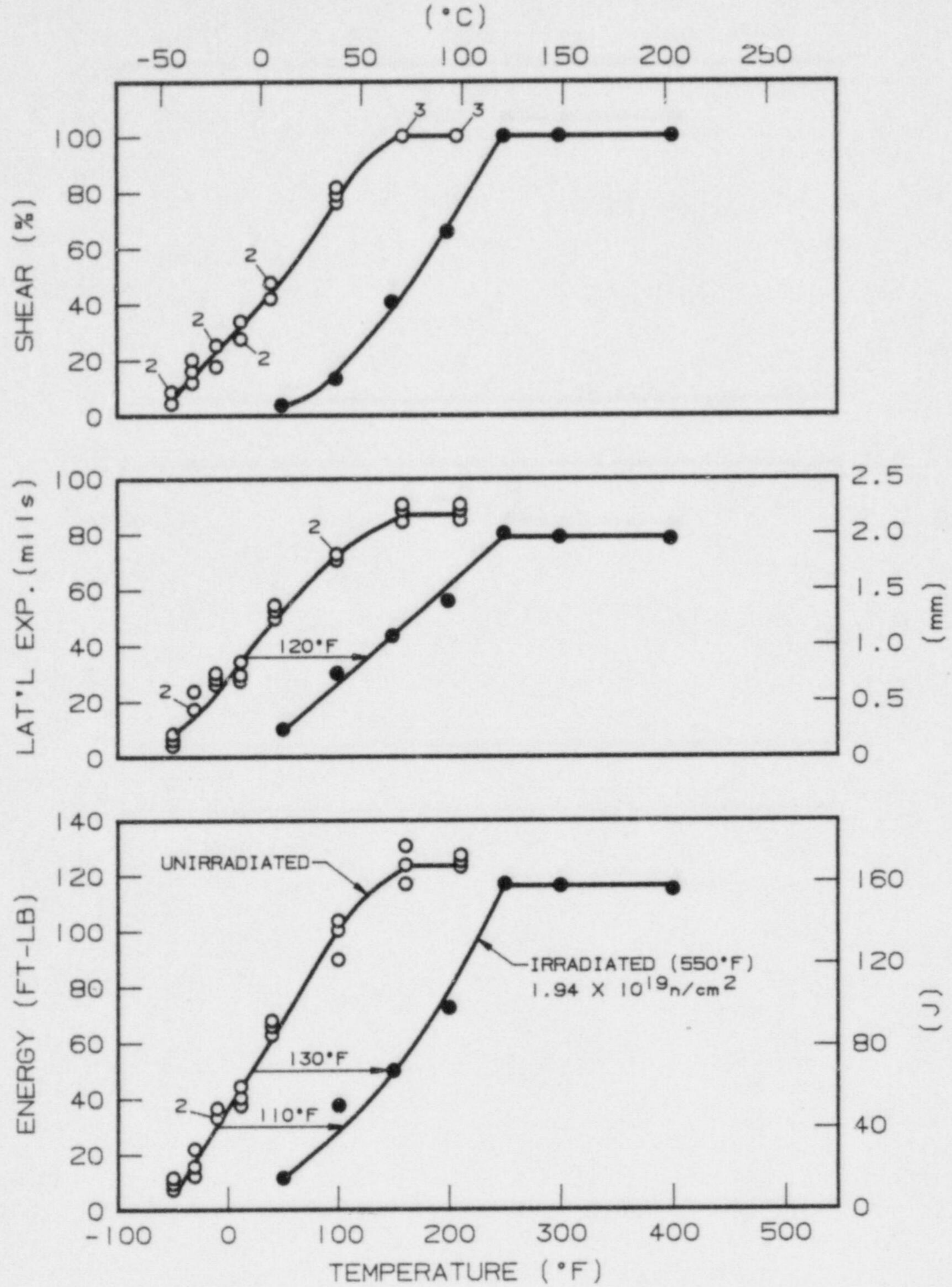

Figure 5-1. Irradiated Charpy V-Notch Impact Properties for Surry Unit 1 Reactor Vessel Lower Shell Plate C4415-1

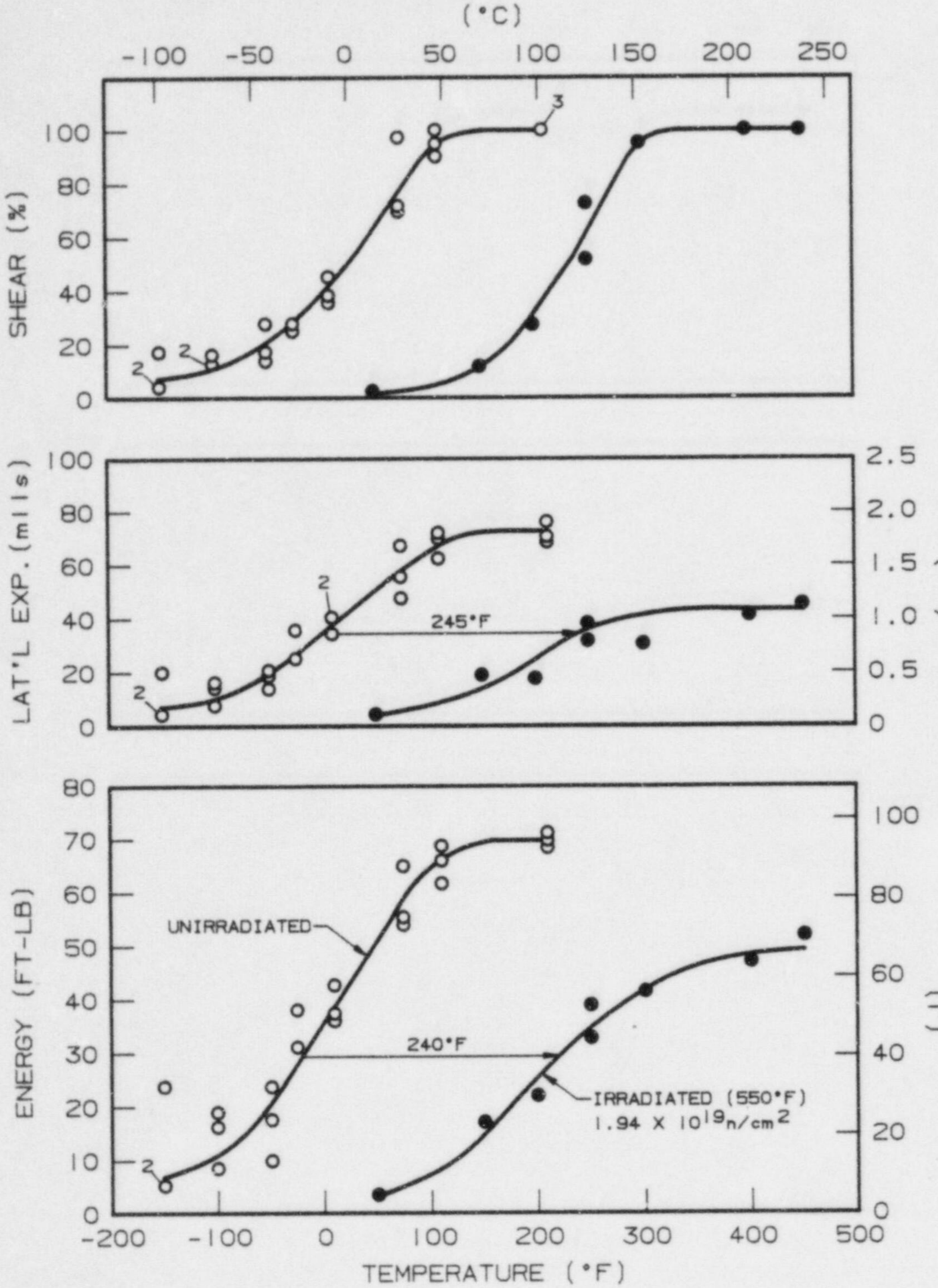

Figure 5-2. Irradiated Charpy V-Notch Impact Properties for Surry Unit 1 Reactor Pressure Vessel Weld Metal

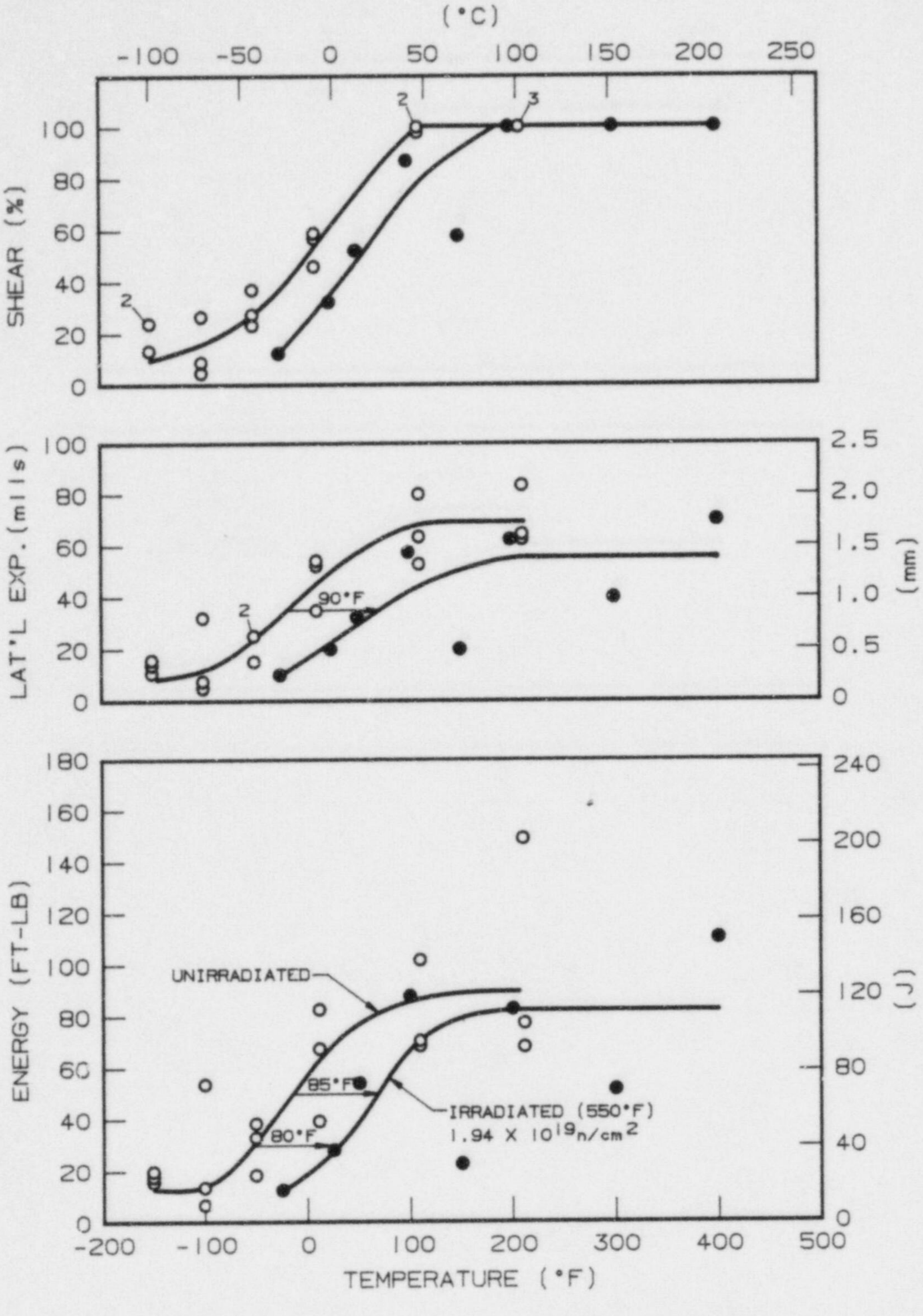

Figure 5-3. Irradiated Charpy V-Notch Impact Properties for Surry Unit 1 Reactor Pressure Vessel Weld Heat Affected Zone Metal

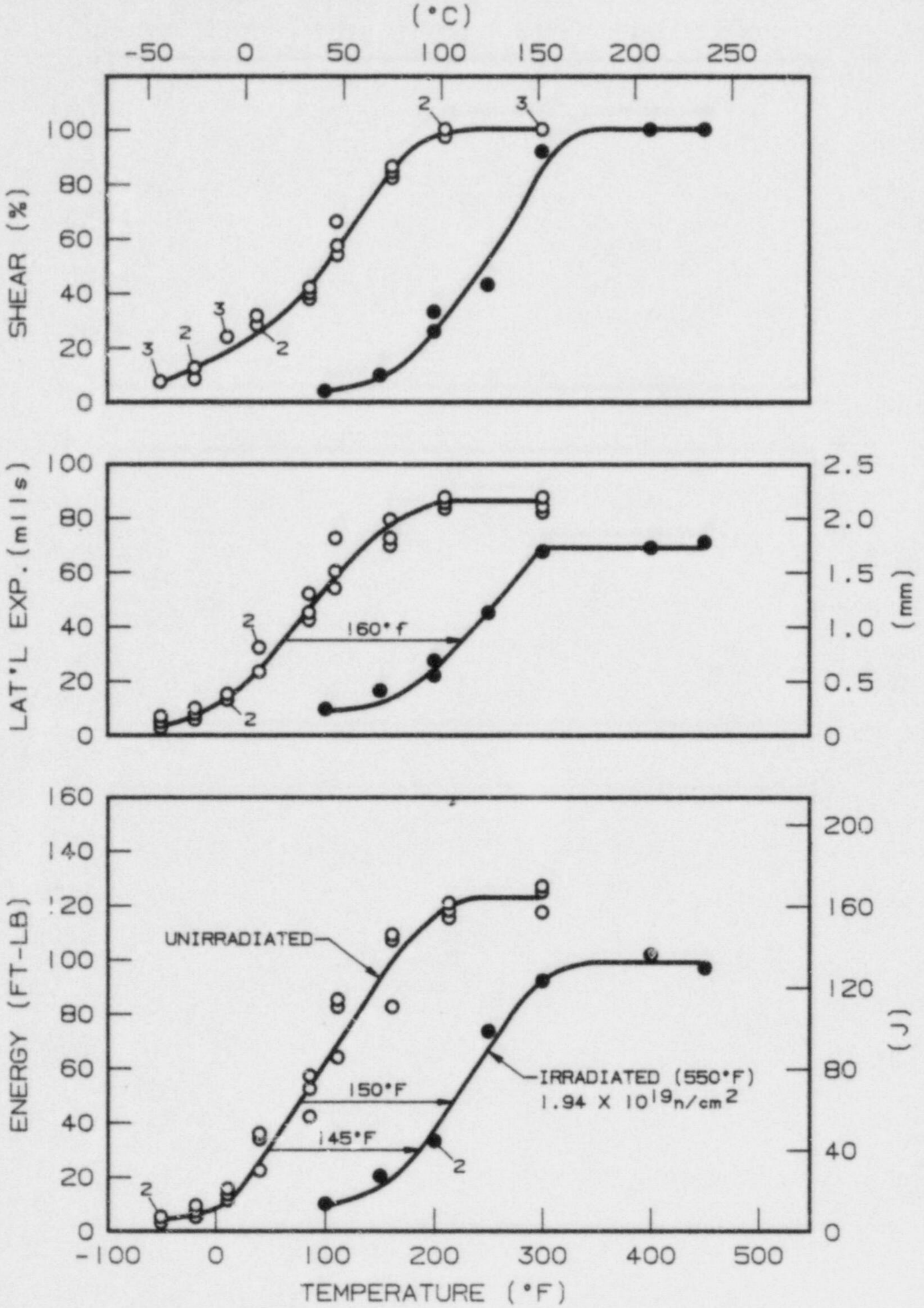

Figure 5-4. Irradiated Charpy V-Notch Impact Properties for Surry Unit 1 A533 Grade B Class 1 Correlation Monitor Material (HSST Plate 02)

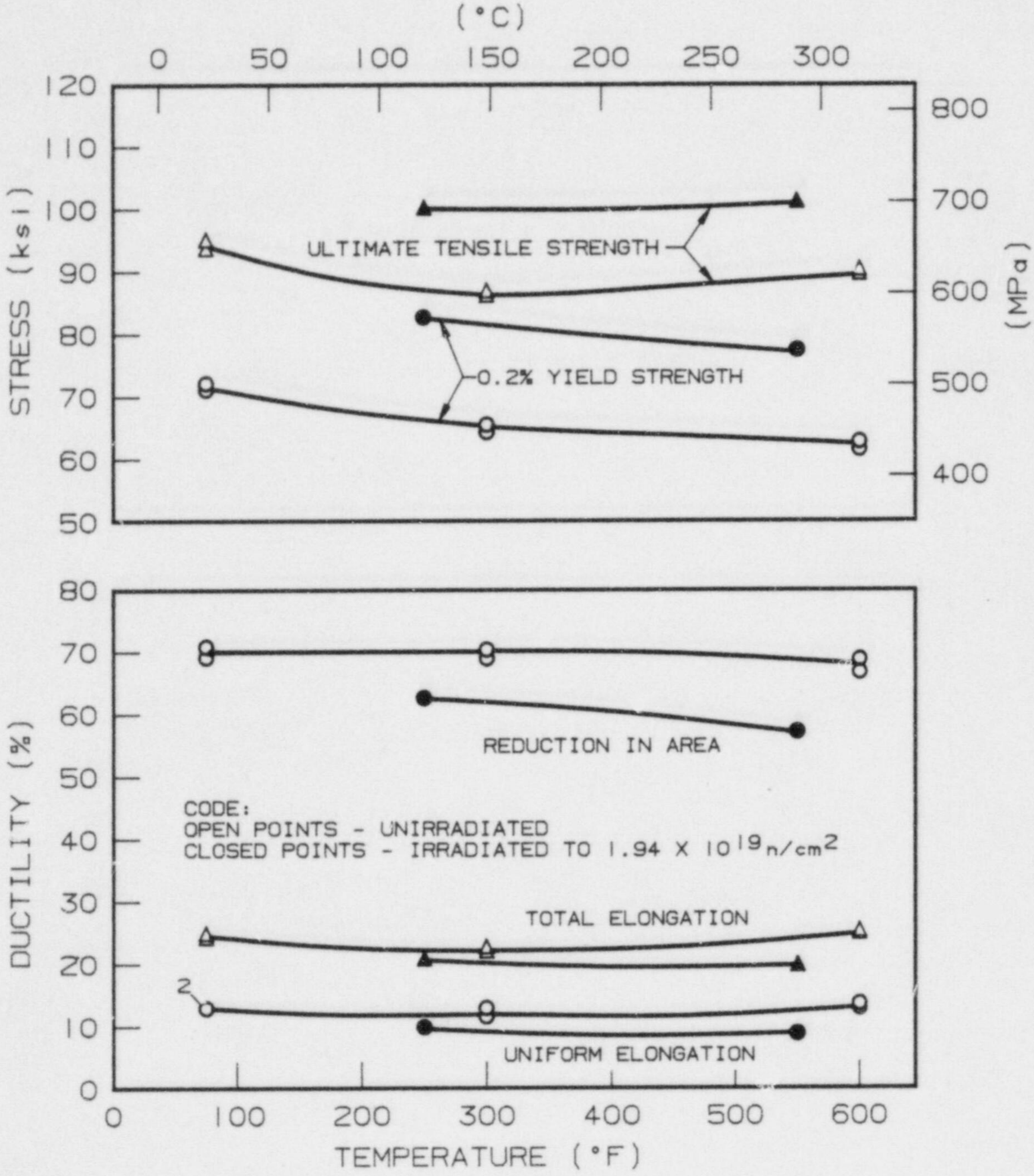

Figure 5-5. Tensile Properties for Surry Unit 1 Reactor Vessel Lower Shell Plate C4415-1

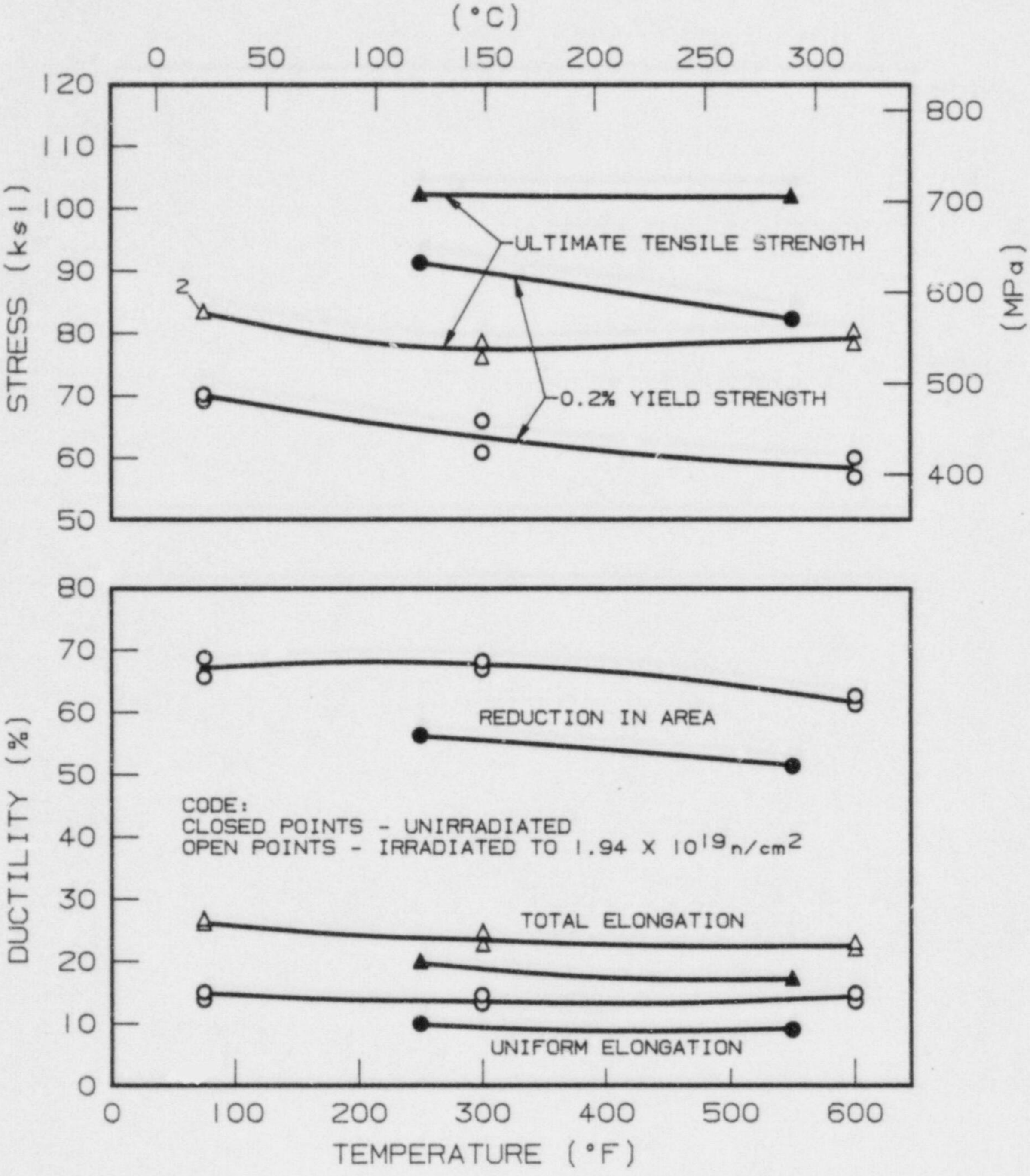

Figure 5-6. Tensile Properties for Surry Unit 1 Reactor Vessel Weld Metal

SECTION 6

RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6-1 INTRODUCTION

Knowledge of the neutron environment within the pressure vessel-surveillance capsule geometry is required as an integral part of LWR pressure vessel surveillance programs for two reasons. First, in the interpretation of radiation-induced property changes observed in materials test specimens, the neutron environment (fluence, flux) to which the test specimens were exposed must be known. Second, in relating the changes observed in the test specimens to the present and future condition of the reactor pressure vessel, a relationship between the environment at various positions within the reactor vessel and that experienced by the test specimens must be established. 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, on the other hand, is derived solely from analysis.

This section describes a discrete ordinates S_n transport analysis performed for the Surry Unit 1 reactor to determine the fast neutron (E > 1.0 Mev) flux and fluence as well as the neutron energy spectra within the reactor vessel and surveillance capsules; and, in turn, to develop data for use in relating neutron exposure of the pressure vessel to that of the surveillance capsules. Based on spectrum-averaged reaction cross sections derived from this calculation, the analysis of the neutron dosimetry contained in Capsule V is discussed and updated evaluations of dosimetry from Capsules T and W are presented.

6-2 DISCRETE ORDINATES ANALYSIS

A plan view of the Surry reactor geometry at the core midplane is shown in Figure 6-1. Since the reactor exhibits 1/8th core symmetry, only a 0°-45°

sector is depicted. Eight irradiation capsules attached to the thermal shield are included in the design to constitute the reactor vessel surveillance program. The capsules are located at 45°, 55°, 65°, 165°, 245°, 285°, 295°, and 305° relative to the major axis at 0°. (Refer to Figure 4-1.)

A plan view of a single surveillance capsule attached to the thermal shield is shown in Figure 6-2. The stainless steel specimen container is 1-inch square and approximately 3 feet in height. The containers are positioned axially such that the specimens are centered on the core midplane, thus spanning the central 3 feet of the 12-foot high reactor core.

From a neutronic standpoint, the surveillance capsule structures are significant. In fact, they have a marked impact on the distributions of neutron flux and energy spectra in the water annulus between the thermal shield and the reactor vessel. Thus, in order to properly ascertain the neutron environment at the test specimen locations, the capsules themselves must be included in the analytical model. Use of at least a two-dimensional computation is, therefore, mandatory.

In the analysis of the neutron environment within the Surry Unit 1 reactor geometry, two sets of transport calculations were carried out. The first, a single computation in the conventional forward mode, was utilized primarily to obtain spectrum-averaged reaction cross sections and gradient corrections for dosimetry reactions. The second set of calculations consisted of a series of adjoint analyses relating the fast neutron (E > 1.0 Mev) flux at the surveillance capsule locations and selected locations on the reactor vessel inner wall to the power distributions in the reactor core. These adjoint importance functions, when combined with cycle-specific core power distributions, yield the plant-specific fast neutron exposure at the surveillance capsule and pressure vessel locations for each operating fuel cycle. Both the forward and adjoint calculations utilized an S_6 angular quadrature.

The forward transport calculation was carried out in R,θ geometry using the DOT two dimensional discrete ordinates code[7] and the SAILOR cross-section library[8]. The SAILOR library is a 47 group, ENDF-BIV based data set produced specifically for light water reactor applications. Anisotropic scattering is treated with a P_3 expansion of the cross-sections. The energy group structure used in the analysis is listed in Table 6-1.

The design basis core power distribution utilized in the forward analysis was derived from statistical studies of long-term operation of Westinghouse 3-loop plants. Inherent in the development of this design basis 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 design basis distribution is expected to yield somewhat conservative results. This is especially true in cases where low leakage fuel management has been employed.

The adjoint analyses were also carried out using the P_3 cross section approximation from the SAILOR library. Adjoint source locations were chosen at the center of each of the surveillance capsules as well as at positions along the inner diameter of the pressure vessel. Again, these calculations were run in R,θ geometry to provide power distribution importance functions for the exposure parameters of interest. Having the adjoint importance functions and appropriate core power distributions, the response of interest is calculated as

$$R_{R,\theta} = \int_R \int_\theta \int_E I(R,\theta,E) F(R,\theta,E) dE R dR d\theta$$

where:

$R_{R,\theta}$ = Response of interest (e.g., ϕ (E > 1.0 MeV)) at radius R and azimuthal angle θ.

$I(R,\theta,E)$ = Adjoint importance function at radius R and azimuthal angle θ for neutron energy group E

$F(R,\theta,E)$ = Full power fission density at radius R and azimuthal angle θ for neutron energy group E

The fission density distributions used reflect the burnup-dependent inventory of fissioning actinides, including U-235, U-238, Pu-239, and Pu-241.

Core power distributions for use in the plant specific fluence evaluations for Surry Unit 1 are derived from measured assembly and cycle burnups for each operating cycle to date. The specific power distribution data used in the analysis is provided in Appendix A of WCAP 11015[9]. The data listed in Appendix A represents cycle averaged relative assembly powers. Therefore, the adjoint results are in terms of fuel cycle averaged neutron flux which when multiplied by the fuel cycle length yields the incremental fast neutron fluence.

Reactor vessel and surveillance capsule neutron fluence projections are made to several future dates. Current neutron fluences, based on past core loadings, are defined as of the end of Cycle 8. Fluence projections are made to the expiration date of the operating license. The expiration date of the operating license for Surry Unit 1 is May 25, 2012 (forty years after the operating license was issued). In addition, projections are made to 60 calendar years beyond issuance of the operating license to illustrate the effect of a 20-year life extension.

A few key assumptions are required to make the fluence projections. In particular, the cycle-averaged core power distribution for Cycle 8 and an 80% capacity factor are assumed to be representative of all future operation. Thus, all fluence projections reflect the low leakage fuel management strategies exemplified by the Cycle 8 core loading. Finally, it is assumed that the Surry Unit 1 core will be uprated from 2441 MWth to 2546 MWth at the beginning of Cycle 11.

The transport methodology, both forward and adjoint, using the SAILOR cross-section library has been benchmarked against the Oakridge National Laboratory (ORNL) Poolside Critical Assembly (PCA) facility as well as against the Westinghouse power reactor surveillance capsule data base [10]. The benchmarking studies indicate that the use of SAILOR cross-sections and generic design basis power distributions produces flux levels that tend to be conservative by 7-22%. When plant specific power distributions are used with the adjoint importance functions, the benchmarking studies show that fluence predictions are within + 15% of measured values at surveillance capsule locations.

6-3 NEUTRON DOSIMETRY

The passive neutron flux monitors included in the Surry Unit 1 surveillance program are listed in Table 6-2. The first five reactions in Table 6-2 are used as fast neutron monitors to relate neutron fluence (E > 1.0 Mev) to measured materials properties changes. To properly account for burnout of the product isotope generated by fast neutron reactions, it is necessary to also determine the magnitude of the thermal neutron flux at the monitor location. Therefore, bare and cadmium-covered cobalt-aluminum monitors are also included.

The relative locations of the various monitors within the surveillance capsules are shown in Figure 4-2. The nickel, copper, and cobalt-aluminum monitors, in wire form, are placed in holes drilled in spacers at several axial levels within the capsules. The iron monitors are obtained by drilling samples from selected Charpy test specimens. In "type-II" capsules such as T, V, X and Z, cadmium-shielded neptunium and uranium fission monitors are accommodated within a dosimeter block located near the center of the capsule. The "type-I" capsules, including S, U, W and Y, do not contain the neptunium and uranium fission monitors.

The use of passive monitors such as those listed in Table 6-2 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 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 analysis of the passive monitors and subsequent derivation of the average neutron flux requires completion of two procedures. First, the disintegration rate of product isotope per unit mass of monitor must be determined. Second, in order to define a suitable spectrum averaged reaction cross section, the neutron energy spectrum at the monitor location must be calculated.

The specific activity of each of the monitors is determined using established ASTM procedures.[11,12,13,14,15] Following sample preparation, the activity of each monitor is determined by means of a lithium-drifted germanium, Ge(Li), gamma spectrometer. The overall standard deviation of the measured data is a function of the precision of sample weighing, the uncertainty in counting, and the acceptable error in detector calibration. For the samples removed from Surry Unit 1, the overall 2σ deviation in the measured data is determined to be ± 10 percent. The neutron energy spectra are determined analytically using the method described in Section 6-1.

Having the measured activity of the monitors and the neutron energy spectra at the locations of interest, the calculation of the neutron flux proceeds as follows.

The reaction product activity in the monitor is expressed as:

$$R = \frac{N_o}{A} f_i \quad Y \int_E \sigma(E)\phi(E) \sum_{j=1}^{N} \frac{P_j}{P_{max}} \; C_j \quad (1 - e^{-\lambda t_j}) \, e^{-\lambda t_d} \qquad (6\text{-}1)$$

where:

R = induced product activity

N_o = Avagadro's number

A = atomic weight of the target isotope

f_i = weight fraction of the target isotope in the target material

Y = number of product atoms produced per reaction

$\sigma(E)$ = energy-dependent reaction cross section

$\phi(E)$ = time-averaged energy-dependent neutron flux at the monitor location with the reactor at full power

P_j = average core power level during irradiation period j

P_{max} = maximum or reference core power level

λ = decay constant of the product isotope

t_j = length of irradiation period j

t_d = decay time following irradiation period j

C_j = $\phi(E>1.0\ MeV)$ during irradiation period j divided by the average $\phi(E>1.0\ MeV)$ over the total irradiation period. C_j is calculated with the adjoint neutron transport method and accounts for the change in neutron monitor response caused by core power distribution variations from cycle to cycle. P_j/P_{max}, which accounts for the month-by-month variation of power level within a cycle, is applied to the full-power-based flux ratio, C_j.

Since neutron flux distributions are calculated using multigroup transport methods and, further, since the prime interest is in the fast neutron flux above 1.0 Mev, spectrum-averaged reaction cross sections are defined such that the integral term in equation (6-1) is replaced by the following relation.

$$\int_E \sigma(E)\ \phi(E)dE = \bar{\sigma}\ \phi(E > 1.0\ \text{Mev})$$

where:

$$\bar{\sigma} = \frac{\int_0^\infty \sigma(E)\ \phi\ (E)dE}{\int_{1.0\ \text{Mev}}^\infty \phi(E)dE} = \frac{\sum_{G=1}^{N} \sigma_g \phi_g}{\sum_{G=G_{1.0\ \text{Mev}}}^{N} \phi_g}$$

Thus, equation (6-1) is rewritten

$$R = \frac{N_o}{A} f_i \; Y \; \bar{\sigma} \; \phi \; (E > 1.0 \text{ Mev}) \sum_{j=1}^{N} \frac{P_j}{P_{max}} C_j \; (1-e^{-\lambda t_j}) \; e^{-\lambda t_d}$$

or, solving for the neutron flux,

$$\phi \; (E > 1.0 \text{ Mev}) = \frac{R}{\frac{N_o}{A} f_i \; Y \; \bar{\sigma} \sum_{j=1}^{N} \frac{P_j}{P_{max}} C_j \; (1-e^{-\lambda t_j}) \; e^{-\lambda t_d}} \qquad (6\text{-}2)$$

The total fluence above 1.0 Mev is then given by

$$\Phi \; (E > 1.0 \text{ Mev}) = \phi \; (E > 1.0 \text{ Mev}) \sum_{j=1}^{N} \frac{P_j}{P_{max}} \; t_j \qquad (6\text{-}3)$$

where:

$\sum_{j=1}^{N} \frac{P_j}{P_{max}} \; t_j$ = total effective full power seconds of operation up to the time of capsule removal

An assessment of the thermal neutron flux levels within the surveillance capsules is obtained from the bare and cadmium-covered Co^{59} (n,γ) Co^{60} data by means of cadmium ratios and the use of a 37-barn 2200 m/sec cross section. Thus,

$$\phi_{Th} = \frac{R_{bare} \quad \frac{D-1}{D}}{\frac{N_o}{A} f_i Y \sigma \sum_{j=1}^{N} \frac{P_j}{P_{max}} \left(1-e^{-\lambda t_j}\right) e^{-\lambda t_d}} \qquad (6\text{-}4)$$

where:

D is defined as $\frac{R_{bare}}{R_{Cd\ covered}}$

6-4 TRANSPORT ANALYSIS RESULTS

Calculated fast neutron (E > 1.0 Mev) exposure results for Surry Unit 1 are presented in Tables 6-3 through 6-10 and in Figures 6-3 through 6-7. Data is presented at several azimuthal locations on the inner radius of the pressure vessel as well as the center of each surveillance capsule.

In Tables 6-3 through 6-6 cycle-specific maximum neutron flux and fluence levels at 0°, 15°, 30°, and 45° on the pressure vessel inner radius are listed the first eight fuel cycles as well as projected to 60 years beyond issuance of the operating license. Similar data for the center of the surveillance capsules located at 15°, 25°, 35°, and 45° are given in Tables 6-7 through 6-10, respectively.

Graphical presentations of the plant specific fast neutron fluence at key locations on the pressure vessel are shown in Figure 6-3 as a function of full power operating time. The pressure vessel data is presented for the 0° location on the circumferential weld as well as for the 45° longitudinal welds. Fast neutron fluence at the surveillance capsule locations is shown as a function of full power operating time in Figure 6-4.

In regard to Figure 6-3 and 6-4, the solid portions of the fluence curves are based directly on the cycle-specific core loadings as of the end of cycle 8. The dashed portions of these curves, however, involve a projection into the future. As mentioned in Section 6-3, the neutron flux average over Cycle 8 was used to project future fluence levels.

It should be noted that implementation of a more severe low leakage pattern would act to reduce the projections of fluence at key locations. On the other hand, relaxation of the current low leakage patterns or a return to out-in fuel management would increase those projections.

In Figure 6-5, the azimuthal variation of maximum fast neutron ($E > 1.0$ MeV) fluence at the inner radius of the pressure vessel is presented as a function of azimuthal angle. Data are presented for both current and projected end-of-life conditions. In Figure 6-6, the relative radial variation of fast neutron flux and fluence within the pressure vessel wall is presented. Similar data showing the relative axial variation of fast neutron flux and fluence over the beltline region of the pressure vessel is shown in Figure 6-7. A three-dimensional description of the fast neutron exposure of the pressure vessel wall can be constructed using the data given in Figures 6-5 through 6-7 along with the relation

$$\phi(R,\theta,Z) = \phi(\theta)\, F(R)\, G(Z)$$

where: $\phi\,(R,\theta,Z)$ = Fast neutron fluence at location R, θ, Z within the pressure vessel wall

$\phi\,(\theta)$ = Fast neutron fluence at azimuthal location θ on the pressure vessel inner radius from Figure 6-5

$F\,(R)$ = Relative fast neutron flux at depth R into the pressure vessel from Figure 6-6

G (Z) = Relative fast neutron flux at axial position Z from Figure 6-7

Analysis has shown that the radial and axial variations within the vessel wall are relatively insensitive to the implementation of low leakage fuel management schemes. Thus, the above relationship provides a vehicle for a reasonable evaluation of fluence gradients within the vessel wall.

6-5 DOSIMETRY RESULTS

The irradiation history of the Surry Unit 1 reactor is given in Table 6-11. The data were obtained from several sources including Nucleonics Week[16], a Surry semi-annual operating report[17], NUREG C020[18] and Virginia Power[19]. Measured saturated activities of the flux monitors contained in Capsules T, W, and V are listed in Tables 6-12 through 6-14, respectively. The measured results for Capsules T and W were derived from Battelle reports[3,4], whereas those for Capsule V were obtained by Westinghouse. The data are presented as measured at the actual monitor radial locations as well as adjusted to the capsule radial (191.15 cm) and azimuthal center where possible. Adjustment factors for each monitor were obtained from reaction rate gradients through each capsule as calculated from the forward transport calculation described in Section 6-2.

In order to derive neutron flux and fluence levels from the measured disintegration rates, suitable spectrum-averaged reaction cross sections are required. The neutron energy spectrum at the radial and azimuthal center of each surveillance capsule, shown in Table 6-15, was taken from the forward calculation. The resulting spectrum-averaged cross sections for each of the five fast neutron reactions are given in Table 6-16.

The fast neutron ($E > 1.0$ Mev) flux levels derived for Capsules T, W, and V are presented in Tables 6-12 through 6-14, respectively. The thermal neutron flux obtained from the cobalt-aluminum monitors is summarized in Table 6-17. Due to the relatively low thermal neutron flux at the capsule locations, no burnout correction was made to any of the measured activities. The maximum error introduced by this assumption is estimated to be less than one percent for the $Ni^{58}(n,p)Co^{58}$ reaction and even less significant for all of the other fast reactions.

A comparison of the measured and calculated fast neutron fluence for each flux monitor of Capsules T, W, and V is shown in Table 6-18. Examination of the data in Table 6-18 shows that neutron fluences corresponding to the average of the monitors at each location agree within 5% of the calculated fluences based on the plant-specific power distributions.

It should be mentioned that, in the case of Capsule V, the excellent agreement between the measured fluences derived from the individual flux monitors and the calculated fluence is due largely to the use of the flux ratio, C_j, mentioned in Section 6-3. Recall that C_j accounts for the impact of power distribution changes on neutron monitor response. For example, low leakage core power distributions in cycles 6 and 8 caused the fast neutron (E>1.0 MeV) flux at the 15° surveillance capsule location to be 15-16 percent lower than the lifetime-average flux at that position. The Co^{58} product from the Ni^{58} n-p reaction has a half-life of only 71 days, which implies that the nickel monitor is probably not sensitive to the irradiation history dating back more than one fuel cycle. Without the power distribution correction factor, the derived flux from the nickel monitor would have been low compared to the other monitors having significantly longer-lived reaction products. Under these circumstances, the results from the nickel monitor would have been ignored.

A similar, but not quite as severe, under-prediction would have occurred if the flux was derived from the iron monitor without use of the C_j factor (Mn^{54} half-life is 312 days). Thus, one can see that meaningful results can be obtained from the entire set of neutron dosimetry when the power distribution correction is made. This correction may be even more important or necessary in future dosimetry analyses where Type I capsules, lacking the uranium and neptunium monitors, are examined.

6-6 SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULES

As discussed in Section 6-4, plant specific fluence evaluations for the center of surveillance capsules located at 15°, 25°, 35°, and 45° were presented in Figure 6-4 for Surry Unit 1. The data presented on those curves represent the best available information upon which to base the future withdrawal schedules for capsules remaining in the Surry Unit 1 reactor.

In the past, withdrawal schedules have been based on the assumption of a constant exposure rate at the surveillance capsule center and a constant lead factor relating capsule exposure to maximum vessel exposure. With the widespread implementation of low leakage fuel management neither of these assumptions can be assumed to be universally valid. It becomes prudent, therefore, to utilize the actual anticipated capsule exposure in conjunction with appropriate materials properties data to establish capsule withdrawal dates that will provide experimental information that is of most benefit.

In evaluating future withdrawal schedules, it must be remembered that the fluence projections shown in Figure 6-4 assume continued operation with the low leakage fuel management scheme currently in place. The validity of this assumption should be verified as each new fuel cycle evolves and if significant changes occur withdrawal schedules should be adjusted accordingly.

6-7 INFLUENCE OF AN ENERGY DEPENDENT DAMAGE MODEL

The use of fast neutron fluence ($E > 1.0$ MeV) to correlate measured materials property 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 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 a reduction 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 calculated 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 (dPa)."

For the Surry Unit 1 pressure vessel, iron atom displacement rates at each surveillance capsule location and at positions within the vessel wall have been calculated. The analysis has indicated that for a given location the ratio of $dPa/\phi(E > 1.0\ MeV)$ is insensitive to changing core power distributions. That is, while implementation of low leakage loading patterns significantly impacts the magnitude and spatial distribution of the neutron field, changes in the relative neutron energy spectrum at a given location are of second order. The $dPa/\phi(E > 1.0\ MeV)$ ratios calculated for key locations in the Surry reactor geometry are given in Table 6-19. The data in Table 6-19 may be used in conjunction with the fast neutron fluence data provided in Section 6-4 to develop distributions of dPa within the surveillance capsules and the reactor pressure vessel.

TABLE 6-1

47 GROUP ENERGY STRUCTURE

Group	Lower Energy (Mev)	Group	Lower Energy (Mev)
1	14.19*	25	0.183
2	12.21	26	0.111
3	10.00	27	0.0674
4	8.61	28	0.0409
5	7.41	29	0.0318
6	6.07	30	0.0261
7	4.97	31	0.0242
8	3.68	32	0.0219
9	3.01	33	0.0150
10	2.73	34	7.10×10^{-3}
11	2.47	35	3.36×10^{-3}
12	2.37	36	1.59×10^{-3}
13	2.35	37	4.54×10^{-4}
14	2.23	38	2.14×10^{-4}
15	1.92	39	1.01×10^{-4}
16	1.65	40	3.73×10^{-5}
17	1.35	41	1.07×10^{-5}
18	1.00	42	5.04×10^{-6}
19	0.821	43	1.86×10^{-6}
20	0.743	44	8.76×10^{-7}
21	0.608	45	4.14×10^{-7}
22	0.498	46	1.00×10^{-7}
23	0.369	47	0.00
24	0.298		

*The upper energy of group 1 is 17.33 Mev.

TABLE 6-2

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 years	
Iron	$Fe^{54}(n,p)Mn^{54}$	0.058	E>1.0 Mev	312.2 days	
Nickel	$Ni^{58}(n,p)Co^{58}$	0.6827	E>1.0 Mev	70.91 days	
Uranium-238*	$U^{238}(n,f)Cs^{137}$	1.0	E>0.4 Mev	30.17 years	6.0
Neptunium-237*	$Np^{237}(n,f)Cs^{137}$	1.0	E>0.08 Mev	30.17 years	6.5
Cobalt-Aluminum*	$Co^{59}(n,\gamma)Co^{60}$	0.0015	0.4eV< E<0.015 Mev	5.272 years	
Cobalt-Aluminum	$Co^{59}(n,\gamma)Co^{60}$	0.0015	E<0.015 Mev	5.272 years	

*Denotes that monitor is cadmium shielded.

TABLE 6-3

SURRY UNIT 1

CALCULATED FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 0° AZIMUTHAL ANGLE[(a)]

Irradiation Interval	Elapsed Irradiation Time (EFPY)	Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2)
CY-1	1.1	5.03×10^{10}	1.70×10^{18}
CY-2	1.6	5.73×10^{10}	2.70×10^{18}
CY-3	2.3	5.22×10^{10}	3.87×10^{18}
CY-4	3.4	4.86×10^{10}	5.49×10^{18}
CY-5	4.6	4.40×10^{10}	7.10×10^{18}
CY-6	5.9	3.96×10^{10}	8.75×10^{18}
CY-7	6.8	5.91×10^{10}	1.05×10^{19}
CY-8[(b)]	8.0	4.05×10^{10}	1.20×10^{19}
CY-9 + CY-10[(c)]	10.3	4.05×10^{10}	1.49×10^{19}
CY-11 → 6/25/2008[(d)]	25.6	4.22×10^{10}	3.54×10^{19}
6/25/2008 → 5/25/2012[(e)]	28.8	4.22×10^{10}	3.96×10^{19}
5/25/2012 → 5/25/2032[(f)]	44.8	4.22×10^{10}	6.09×10^{19}

(a) Applicable to the peak locations (0°, 90°, 180°, 270°) on the intermediate and lower shell plates and the intermediate to lower shell circumferential weld.

(b) Current neutron fluences are defined at the end of CY-8.

(c) At the beginning of CY-11, the core thermal power will be uprated to 2546 MWth. Beyond the end of CY-8 a 80% capacity factor is assumed.

(d) Exposure period from the onset of the uprating to the original license expiration date.

(e) 5/25/2012 corresponds to 40 calendar years beyond issuance of the operating license and is the license expiration date.

(f) 5/25/2032 corresponds to 60 calendar years beyond issuance of the operating license, illustrating the effect of a 20 year life extension.

TABLE 6-4

SURRY UNIT 1

CALCULATED FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 15° AZIMUTHAL ANGLE

Irradiation Interval	Elapsed Irradiation Time (EFPY)	Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2)
CY-1	1.1	2.40×10^{10}	8.12×10^{17}
CY-2	1.6	2.72×10^{10}	1.29×10^{18}
CY-3	2.3	2.49×10^{10}	1.84×10^{18}
CY-4	3.4	2.34×10^{10}	2.62×10^{18}
CY-5	4.6	2.06×10^{10}	3.38×10^{18}
CY-6	5.9	1.88×10^{10}	4.16×10^{18}
CY-7	6.8	2.50×10^{10}	4.92×10^{18}
CY-8[a]	8.0	1.88×10^{10}	5.62×10^{18}
CY-9 + CY-10[b]	10.3	1.88×10^{10}	6.96×10^{18}
CY-11 → 6/25/2008[c]	25.6	1.97×10^{10}	1.65×10^{19}
6/25/2008 → 5/25/2012[d]	28.8	1.97×10^{10}	1.84×10^{19}
5/25/2012 → 5/25/2032[e]	44.8	1.97×10^{10}	2.84×10^{19}

(a) Current neutron fluences are defined at the end of CY-8.

(b) At the beginning of CY-11, the core thermal power will be uprated to 2546 MWth. Beyond the end of CY-8 a 80% capacity factor is assumed.

(c) Exposure period from the onset of the uprating to the original license expiration date.

(d) 5/25/2012 corresponds to 40 calendar years beyond issuance of the operating license and is the license expiration date.

(e) 5/25/2032 corresponds to 60 calendar years beyond issuance of the operating license, illustrating the effect of a 20 year life extension.

TABLE 6-5

SURRY UNIT 1

CALCULATED FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 30° AZIMUTHAL ANGLE

Irradiation Interval	Elapsed Irradiation Time (EFPY)	Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2)
CY-1	1.1	1.30×10^{10}	4.40×10^{17}
CY-2	1.6	1.54×10^{10}	7.09×10^{17}
CY-3	2.3	1.34×10^{10}	1.01×10^{18}
CY-4	3.4	1.30×10^{10}	1.44×10^{18}
CY-5	4.6	1.09×10^{10}	1.84×10^{18}
CY-6	5.9	1.02×10^{10}	2.27×10^{18}
CY-7	6.8	9.80×10^{9}	2.56×10^{18}
CY-8[(a)]	8.0	9.86×10^{9}	2.93×10^{18}
CY-9 + CY-10[(b)]	10.3	9.86×10^{9}	3.63×10^{18}
CY-11 → 6/25/2008[(c)]	25.6	1.03×10^{10}	8.62×10^{18}
6/25/2008 → 5/25/2012[(d)]	28.8	1.03×10^{10}	9.63×10^{18}
5/25/2012 → 5/25/2032[(e)]	44.8	1.03×10^{10}	1.48×10^{19}

(a) Current neutron fluences are defined at the end of CY-8.

(b) At the beginning of CY-11, the core thermal power will be uprated to 2546 MWth. Beyond the end of CY-8 a 80% capacity factor is assumed.

(c) Exposure period from the onset of the uprating to the original license expiration date.

(d) 5/25/2012 corresponds to 40 calendar years beyond issuance of the operating license and is the license expiration date.

(e) 5/25/2032 corresponds to 60 calendar years beyond issuance of the operating license, illustrating the effect of a 20 year life extension.

TABLE 6-6

SURRY UNIT 1

CALCULATED FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 45° AZIMUTHAL ANGLE[a]

Irradiation Interval	Elapsed Irradiation Time (EFPY)	Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2)
CY-1	1.1	8.59×10^{9}	2.91×10^{17}
CY-2	1.6	1.05×10^{10}	4.75×10^{17}
CY-3	2.3	9.08×10^{9}	6.78×10^{17}
CY-4	3.4	8.71×10^{9}	9.68×10^{17}
CY-5	4.6	7.11×10^{9}	1.23×10^{18}
CY-6	5.9	6.86×10^{9}	1.51×10^{18}
CY-7	6.8	6.14×10^{9}	1.70×10^{18}
CY-8[b]	8.0	6.54×10^{9}	1.94×10^{18}
CY-9 + CY-10[c]	10.3	6.54×10^{9}	2.41×10^{18}
CY-11 → 6/25/2008[d]	25.6	6.82×10^{9}	5.71×10^{18}
6/25/2008 → 5/25/2012[e]	28.8	6.82×10^{9}	6.39×10^{18}
5/25/2012 → 5/25/2032[f]	44.8	6.82×10^{9}	9.83×10^{18}

(a) Applicable to the longitudinal welds at 45°, 135°, 225°, 315° in the peak axial flux.

(b) Current neutron fluences are defined at the end of CY-8.

(c) At the beginning of CY-11, the core thermal power will be uprated to 2546 MWth. Beyond the end of CY-8 a 80% capacity factor is assumed.

(d) Exposure period from the onset of the uprating to the original license expiration date.

(e) 5/25/2012 corresponds to 40 calendar years beyond issuance of the operating license and is the license expiration date.

(f) 5/25/2032 corresponds to 60 calendar years beyond issuance of the operating license, illustrating the effect of a 20 year life extension.

TABLE 6-7

SURRY UNIT 1

CALCULATED FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 15° SURVEILLANCE CAPSULE CENTER

Irradiation Interval	Elapsed Irradiation Time (EFPY)	Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2)
CY-1	1.1	8.31×10^{10}	2.81×10^{18}
CY-2	1.6	9.42×10^{10}	4.46×10^{18}
CY-3	2.3	8.61×10^{10}	6.39×10^{18}
CY-4	3.4	8.11×10^{10}	9.08×10^{18}
CY-5	4.6	7.08×10^{10}	1.17×10^{19}
CY-6	5.9	6.47×10^{10}	1.44×10^{19}
CY-7	6.8	8.76×10^{10}	1.70×10^{19}
CY-8 (a)	8.0	6.46×10^{10}	1.94×10^{19}
CY-9 + CY-10 (b)	10.3	6.46×10^{10}	2.40×10^{19}
CY-11 → 6/25/2008 (c)	25.6	6.74×10^{10}	5.67×10^{19}
6/25/2008 → 5/25/2012 (d)	28.8	6.74×10^{10}	6.34×10^{19}
5/25/2012 → 5/25/2032 (e)	44.8	6.74×10^{10}	9.74×10^{19}

(a) Current neutron fluences are defined at the end of CY-8.

(b) At the beginning of CY-11, the core thermal power will be uprated to 2546 MWth. Beyond the end of CY-8 a 80% capacity factor is assumed.

(c) Exposure period from the onset of the uprating to the original license expiration date.

(d) 5/25/2012 corresponds to 40 calendar years beyond issuance of the operating license and is the license expiration date.

(e) 5/25/2032 corresponds to 60 calendar years beyond issuance of the operating license, illustrating the effect of a 20 year life extension.

TABLE 6-8

SURRY UNIT 1

CALCULATED FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 25° SURVEILLANCE CAPSULE CENTER

Irradiation Interval	Elapsed Irradiation Time (EFPY)	Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2)
CY-1	1.1	5.26×10^{10}	1.78×10^{18}
CY-2	1.6	6.14×10^{10}	2.85×10^{18}
CY-3	2.3	5.40×10^{10}	4.06×10^{18}
CY-4	3.4	5.24×10^{10}	5.81×10^{18}
CY-5	4.6	4.48×10^{10}	7.44×10^{18}
CY-6	5.9	4.15×10^{10}	9.17×10^{18}
CY-7	6.8	4.17×10^{10}	1.04×10^{19}
CY-8 [(a)]	8.0	4.02×10^{10}	1.19×10^{19}
CY-9 + CY-10 [(b)]	10.3	4.02×10^{10}	1.48×10^{19}
CY-11 → 6/25/2008 [(c)]	25.6	4.20×10^{10}	3.52×10^{19}
6/25/2008 → 5/25/2012 [(d)]	28.8	4.20×10^{10}	3.93×10^{19}
5/25/2012 → 5/25/2032 [(e)]	44.8	4.20×10^{10}	6.05×10^{19}

(a) Current neutron fluences are defined at the end of CY-8.

(b) At the beginning of CY-11, the core thermal power will be uprated to 2546 MWth. Beyond the end of CY-8 a 80% capacity factor is assumed.

(c) Exposure period from the onset of the uprating to the original license expiration date.

(d) 5/25/2012 corresponds to 40 calendar years beyond issuance of the operating license and is the license expiration date.

(e) 5/25/2032 corresponds to 60 calendar years beyond issuance of the operating license, illustrating the effect of a 20 year life extension.

TABLE 6-9

SURRY UNIT 1

CALCULATED FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 35° SURVEILLANCE CAPSULE CENTER

Irradiation Interval	Elapsed Irradiation Time (EFPY)	Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2)
CY-1	1.1	3.56×10^{10}	1.21×10^{18}
CY-2	1.6	4.28×10^{10}	1.95×10^{18}
CY-3	2.3	3.71×10^{10}	2.78×10^{18}
CY-4	3.4	3.58×10^{10}	3.97×10^{18}
CY-5	4.6	2.93×10^{10}	5.05×10^{18}
CY-6	5.9	2.78×10^{10}	6.21×10^{18}
CY-7	6.8	2.58×10^{10}	6.99×10^{18}
CY-8[a]	8.0	2.68×10^{10}	7.99×10^{18}
CY-9 + CY-10[b]	10.3	2.68×10^{10}	9.90×10^{18}
CY-11 → 6/25/2008[c]	25.6	2.80×10^{10}	2.35×10^{19}
6/25/2008 → 5/25/2012[d]	28.8	2.80×10^{10}	2.62×10^{19}
5/25/2012 → 5/25/2032[e]	44.8	2.80×10^{10}	4.04×10^{19}

(a) Current neutron fluences are defined at the end of CY-8.

(b) At the beginning of CY-11, the core thermal power will be uprated to 2546 MWth. Beyond the end of CY-8 a 80% capacity factor is assumed.

(c) Exposure period from the onset of the uprating to the original license expiration date.

(d) 5/25/2012 corresponds to 40 calendar years beyond issuance of the operating license and is the license expiration date.

(e) 5/25/2032 corresponds to 60 calendar years beyond issuance of the operating license, illustrating the effect of a 20 year life extension.

TABLE 6-10

SURRY UNIT 1

CALCULATED FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 45° SURVEILLANCE CAPSULE CENTER

Irradiation Interval	Elapsed Irradiation Time (EFPY)	Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2)
CY-1	1.1	2.79×10^{10}	9.45×10^{17}
CY-2	1.6	3.43×10^{10}	1.55×10^{18}
CY-3	2.3	2.95×10^{10}	2.21×10^{18}
CY-4	3.4	2.83×10^{10}	3.15×10^{18}
CY-5	4.6	2.29×10^{10}	3.98×10^{18}
CY-6	5.9	2.21×10^{10}	4.91×10^{18}
CY-7	6.8	1.97×10^{10}	5.50×10^{18}
CY-8[(a)]	8.0	2.10×10^{10}	6.28×10^{18}
CY-9 + CY-10[(b)]	10.3	2.10×10^{10}	7.78×10^{18}
CY-11 → 6/25/2008[(c)]	25.6	2.19×10^{10}	1.84×10^{19}
6/25/2008 → 5/25/2012[(d)]	28.8	2.19×10^{10}	2.06×10^{19}
5/25/2012 → 5/25/2032[(e)]	44.8	2.19×10^{10}	3.16×10^{19}

(a) Current neutron fluences are defined at the end of CY-8.

(b) At the beginning of CY-11, the core thermal power will be uprated to 2546 MWth. Beyond the end of CY-8 a 80% capacity factor is assumed.

(c) Exposure period from the onset of the uprating to the original license expiration date.

(d) 5/25/2012 corresponds to 40 calendar years beyond issuance of the operating license and is the license expiration date.

(e) 5/25/2032 corresponds to 60 calendar years beyond issuance of the operating license, illustrating the effect of a 20 year life extension.

TABLE 6-11

IRRADIATION HISTORY OF SURRY UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE V

MONTH	YEAR	P_J (MW)	P_{MAX} (MW)	P_J/P_{MAX}	IRRADIATION TIME (DAY)	DECAY TIME (DAY)	
7	1972	83	2441	0.034	31	5202	
8	1972	83	2441	0.034	31	5171	
9	1972	436	2441	0.178	30	5141	
10	1972	0	2441	0.000	31	5110	
11	1972	1	2441	0.000	30	5080	
12	1972	1129	2441	0.463	31	5049	
1	1973	285	2441	0.117	31	5018	
2	1973	1649	2441	0.676	28	4990	
3	1973	1416	2441	0.580	31	4959	
4	1973	1371	2441	0.562	30	4929	
5	1973	1361	2441	0.557	31	4898	
6	1973	1181	2441	0.484	30	4868	
7	1973	1838	2441	0.753	31	4837	
8	1973	1777	2441	0.728	31	4806	
9	1973	1250	2441	0.512	30	4776	
10	1973	674	2441	0.276	31	4745	
11	1973	2073	2441	0.849	30	4715	
12	1973	0	2441	0.000	31	4684	
1	1974	0	2441	0.000	31	4653	
2	1974	0	2441	0.000	28	4625	
3	1974	1047	2441	0.429	31	4594	
4	1974	2192	2441	0.898	30	4564	
5	1974	2073	2441	0.849	31	4533	
6	1974	2224	2441	0.911	30	4503	
7	1974	1400	2441	0.574	31	4472	
8	1974	2384	2441	0.977	31	4441	
9	1974	2095	2441	0.858	30	4411	
10	1974	1129	2441	0.462	31	4380	Capsule T Removed
11	1974	0	2441	0.000	30	4350	
12	1974	0	2441	0.000	31	4319	
1	1975	1	2441	0.000	31	4288	
2	1975	1979	2441	0.811	28	4260	
3	1975	1833	2441	0.751	31	4229	
4	1975	2049	2441	0.839	30	4199	
5	1975	2302	2441	0.943	31	4168	
6	1975	2176	2441	0.891	30	4138	
7	1975	1799	2441	0.737	31	4107	
8	1975	2180	2441	0.893	31	4076	
9	1975	1971	2441	0.807	30	4046	
10	1975	0	2441	0.000	31	4015	
11	1975	0	2441	0.000	30	3985	
12	1975	1209	2441	0.495	31	3954	

TABLE 6-11 (Continued)

IRRADIATION HISTORY OF SURRY UNIT 1

REACTOR VESSEL SURVEILLANCE CAPSULE V

MONTH	YEAR	P_J (MW)	P_{MAX} (MW)	P_J/P_{MAX}	IRRADIATION TIME (DAY)	DECAY TIME (DAY)	
1	1976	2430	2441	0.995	31	3923	
2	1976	2399	2441	0.983	29	3894	
3	1976	1634	2441	0.669	31	3863	
4	1976	1894	2441	0.776	30	3833	
5	1976	2022	2441	0.828	31	3802	
6	1976	2423	2441	0.992	30	3772	
7	1976	1777	2441	0.728	31	3741	
8	1976	1834	2441	0.751	31	3710	
9	1976	1914	2441	0.784	30	3680	
10	1976	1258	2441	0.516	31	3649	
11	1976	0	2441	0.000	30	3619	
12	1976	0	2441	0.000	31	3588	
1	1977	600	2441	0.246	31	3557	
2	1977	2143	2441	0.878	28	3529	
3	1977	2414	2441	0.989	31	3498	
4	1977	861	2441	0.353	30	3468	
5	1977	1319	2441	0.540	31	3437	
6	1977	2441	2441	1.000	30	3407	
7	1977	2389	2441	0.979	31	3376	
8	1977	2213	2441	0.906	31	3345	
9	1977	2402	2441	0.984	30	3315	
10	1977	2441	2441	1.000	31	3284	
11	1977	1623	2441	0.665	30	3254	
12	1977	1906	2441	0.781	31	3223	
1	1978	2436	2441	0.998	31	3192	
2	1978	2434	2441	0.997	28	3164	
3	1978	2437	2441	0.998	31	3133	
4	1978	1703	2441	0.698	30	3103	Capsule W Removed
5	1978	0	2441	0.000	31	3072	
6	1978	0	2441	0.000	30	3042	
7	1978	1708	2441	0.700	31	3011	
8	1978	2383	2441	0.976	31	2980	
9	1978	2148	2441	0.880	30	2950	
10	1978	2430	2441	0.996	31	2919	
11	1978	2425	2441	0.993	30	2889	
12	1978	785	2441	0.322	31	2858	
1	1979	2287	2441	0.937	31	2827	
2	1979	2443	2441	1.001	28	2799	
3	1979	1097	2441	0.450	31	2768	
4	1979	0	2441	0.000	30	2738	
5	1979	0	2441	0.000	31	2707	
6	1979	0	2441	0.000	30	2677	

TABLE 6-11 (Continued)

IRRADIATION HISTORY OF SURRY UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE V

MONTH	YEAR	P_J (MW)	P_{MAX} (MW)	P_J/P_{MAX}	IRRADIATION TIME (DAY)	DECAY TIME (DAY)
7	1979	0	2441	0.000	31	2646
8	1979	0	2441	0.000	31	2615
9	1979	0	2441	0.000	30	2585
10	1979	488	2441	0.200	31	2554
11	1979	2428	2441	0.995	30	2524
12	1979	1463	2441	0.599	31	2493
1	1980	1798	2441	0.736	31	2462
2	1980	1582	2441	0.648	29	2433
3	1980	0	2441	0.000	31	2402
4	1980	0	2441	0.000	30	2372
5	1980	1520	2441	0.623	31	2341
6	1980	2361	2441	0.967	30	2311
7	1980	2341	2441	0.959	31	2280
8	1980	1405	2441	0.576	31	2249
9	1980	859	2441	0.352	30	2219
10	1980	0	2441	0.000	31	2188
11	1980	0	2441	0.000	30	2158
12	1980	0	2441	0.000	31	2127
1	1981	0	2441	0.000	31	2096
2	1981	0	2441	0.000	28	2068
3	1981	0	2441	0.000	31	2037
4	1981	0	2441	0.000	30	2007
5	1981	0	2441	0.000	31	1976
6	1981	0	2441	0.000	30	1946
7	1981	1427	2441	0.584	31	1915
8	1981	2340	2441	0.959	31	1884
9	1981	1391	2441	0.570	30	1854
10	1981	1981	2441	0.812	31	1823
11	1981	2214	2441	0.907	30	1793
12	1981	1441	2441	0.590	31	1762
1	1982	2141	2441	0.877	31	1731
2	1982	1120	2441	0.459	28	1703
3	1982	2332	2441	0.955	31	1672
4	1982	2021	2441	0.828	30	1642
5	1982	2441	2441	1.000	31	1611
6	1982	2437	2441	0.998	30	1581
7	1982	2354	2441	0.964	31	1550
8	1982	2311	2441	0.947	31	1519
9	1982	2426	2441	0.994	30	1489
10	1982	997	2441	0.408	31	1458
11	1982	2104	2441	0.862	30	1428
12	1982	2374	2441	0.973	31	1397

TABLE 6-11 (Continued)

IRRADIATION HISTORY OF SURRY UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE V

MONTH	YEAR	P_J (MW)	P_{MAX} (MW)	P_J/P_{MAX}	IRRADIATION TIME (DAY)	DECAY TIME (DAY)
1	1983	2373	2441	0.972	31	1366
2	1983	510	2441	0.209	28	1338
3	1983	0	2441	0.000	31	1307
4	1983	0	2441	0.000	30	1277
5	1983	0	2441	0.000	31	1246
6	1983	554	2441	0.227	30	1216
7	1983	2370	2441	0.971	31	1185
8	1983	2430	2441	0.996	31	1154
9	1983	1574	2441	0.645	30	1124
10	1983	1950	2441	0.799	31	1093
11	1983	2330	2441	0.955	30	1063
12	1983	2561	2441	1.049	31	1032
1	1984	2264	2441	0.927	31	1001
2	1984	1928	2441	0.790	29	972
3	1984	1595	2441	0.654	31	941
4	1984	1047	2441	0.429	30	911
5	1984	1956	2441	0.801	31	880
6	1984	1335	2441	0.547	30	850
7	1984	1387	2441	0.568	31	819
8	1984	1910	2441	0.783	31	788
9	1984	1672	2441	0.685	30	758
10	1984	0	2441	0.000	31	727
11	1984	0	2441	0.000	30	697
12	1984	62	2441	0.025	31	666
1	1985	1737	2441	0.711	31	635
2	1985	2426	2441	0.994	28	607
3	1985	2440	2441	1.000	31	576
4	1985	2169	2441	0.888	30	546
5	1985	1248	2441	0.511	31	515
6	1985	2441	2441	1.000	30	485
7	1985	2325	2441	0.952	31	454
8	1985	860	2441	0.352	31	423
9	1985	2017	2441	0.826	30	393
10	1985	2232	2441	0.914	31	362
11	1985	2321	2441	0.951	30	332
12	1985	2325	2441	0.952	31	301
1	1986	1579	2441	0.647	31	270
2	1986	1704	2441	0.698	28	242
3	1986	2439	2441	0.999	31	211
4	1986	2394	2441	0.981	30	181
5	1986	1947	2441	0.798	10	171

Decay time is referenced to 10/28/86.

TABLE 6-12

MEASURED FLUX MONITOR ACTIVITIES FROM
SURRY UNIT 1, CAPSULE T+

Reaction and Axial Location	Radial Location (cm)	Saturated Activity (DPS/gm)	Adjusted Saturated Activity (DPS/gm)	Measured ϕ (E>1.0 Mev) (n/cm^2-sec)
$Fe^{54}(n,p)Mn^{54}$				
Top		4.44×10^6		9.63×10^{10}
Middle		3.71×10^6		8.01×10^{10}
Bottom		3.45×10^6		7.45×10^{10}
Average				8.36×10^{10}
$Ni^{58}(n,p)Co^{58}$				
Middle	190.92	6.11×10^7	5.84×10^7	8.24×10^{10}
$Cu^{63}(n,\alpha)Co^{60}$				
Top	190.92	4.15×10^5	3.98×10^5	8.84×10^{10}
Bottom	190.92	4.24×10^5	4.06×10^5	9.03×10^{10}
Average				8.94×10^{10}
$Np^{237}(n,f)Cs^{137}$				
Middle	191.15	3.23×10^7		7.01×10^{10}
$U^{238}(n,f)Cs^{137}$*				
Middle	191.15	4.40×10^6		8.19×10^{10}

*$U^{238}(n,f)Cs^{137}$ activities corrected by a factor of 0.89 for 300 ppm U-235 impurity and Pu-239 buildup.

+Neither measured activities nor radial locations were reported in Reference 3. Saturated activities were determined from the reported nuclear constants and resultant flux levels. Adjusted saturated activities are not given for the iron monitors because the radial and azimuthal position of the charpy chips is uncertain.

TABLE 6-13

MEASURED FLUX MONITOR ACTIVITIES FROM
SURRY UNIT 1, CAPSULE W[(a)]

Reaction and Axial Location	Radial Location (cm)	Saturated Activity (DPS/gm)	Adjusted Saturated Activity (DPS/gm)	Measured ϕ (E>1.0 Mev) (n/cm^2-sec)
$Fe^{54}(n,p)Mn^{54}$				
Mid Top		1.99×10^6		3.69×10^{10}
Mid Bottom		1.94×10^6		3.60×10^{10}
Average				3.64×10^{10}
$Ni^{58}(n,p)Co^{58}$				
Middle	190.92	3.47×10^7	3.31×10^7	4.10×10^{10}
$Cu^{63}(n,\alpha)Co^{60}$				
Top	190.92	2.38×10^5	2.28×10^5	3.92×10^{10}
Bottom	190.92	2.42×10^5	2.32×10^5	3.98×10^{10}
Average				3.95×10^{10}

a) Neither measured activities nor radial locations were reported in Reference 4. Saturated activities were determined from the reported nuclear constants and resultant flux levels. Adjusted saturated activities are not given for the iron monitors because the radial and azimuthal position of the tensile specimen chips is uncertain.

TABLE 6-14

MEASURED FLUX MONITOR ACTIVITIES FROM
SURRY UNIT 1, CAPSULE V

Reaction and Axial Location	Radial Location (cm)	Saturated Activity (DPS/gm)	Adjusted Saturated Activity (DPS/gm)	Measured ϕ (E>1.0 Mev) (n/cm^2-sec)
$Fe^{54}(n,p)Mn^{54}$				
R-41	191.92	3.13×10^6	3.54×10^6	7.49×10^{10}
H-10	191.92	3.16×10^6	3.74×10^6	7.89×10^{10}
V-50	190.92	3.86×10^6	3.77×10^6	7.95×10^{10}
Average				7.78×10^{10}
$Ni^{58}(n,p)Co^{58}$				
Middle	190.92	5.45×10^7	5.21×10^7	7.30×10^{10}
$Cu^{63}(n,\alpha)Co^{60}$				
Top	190.92	3.84×10^5	3.68×10^5	8.17×10^{10}
$Np^{237}(n,f)Cs^{137}$				
Middle	191.15	3.22×10^7	3.22×10^7	7.02×10^{10}
$U^{238}(n,f)Cs^{137}$*				
Middle	191.15	4.13×10^6	4.13×10^6	7.69×10^{10}

*$U^{238}(n,f)Cs^{137}$ activities corrected by a factor of 0.84 for 300 ppm U-235 impurity and Pu-239 buildup.

TABLE 6-15

CALCULATED NEUTRON ENERGY SPECTRA AT THE CENTER OF SURRY UNIT 1 SURVEILLANCE CAPSULES

	Neutron Flux (n/cm^2-sec)	
Group No.	15° Capsules T & V	35° Capsule W
1	2.64×10^7	1.62×10^7
2	9.64×10^7	5.85×10^7
3	3.34×10^8	1.95×10^8
4	6.08×10^8	3.48×10^8
5	1.01×10^9	5.62×10^8
6	2.26×10^9	1.23×10^9
7	3.11×10^9	1.63×10^9
8	6.17×10^9	3.01×10^9
9	5.40×10^9	2.48×10^9
10	6.97×10^9	1.99×10^9
11	5.16×10^9	2.30×10^9
12	2.57×10^9	1.14×10^9
13	7.81×10^8	3.45×10^8
14	3.82×10^9	1.68×10^9
15	9.81×10^9	4.29×10^9
16	1.25×10^{10}	5.30×10^9
17	1.18×10^{10}	7.73×10^9
18	3.52×10^{10}	1.43×10^{10}
19	2.52×10^{10}	9.97×10^9
20	1.22×10^{10}	4.82×10^9
21	3.78×10^{10}	1.44×10^{10}

Note: These spectra were obtained from the forward DOT calculation using a design basis core power distribution with a core thermal power rating of 2900 MWt.

TABLE 6-15 (Continued)

CALCULATED NEUTRON ENERGY SPECTRA AT THE CENTER OF SURRY UNIT 1 SURVEILLANCE CAPSULES

	Neutron Flux (n/cm^2-sec)	
Group No.	15° Capsules T & V	35° Capsule W
22	2.90×10^{10}	1.09×10^{10}
23	3.40×10^{10}	1.29×10^{10}
24	3.09×10^{10}	1.16×10^{10}
25	3.99×10^{10}	1.50×10^{10}
26	3.87×10^{10}	1.44×10^{10}
27	3.07×10^{10}	1.14×10^{10}
28	2.27×10^{10}	8.37×10^{9}
29	7.49×10^{9}	2.77×10^{9}
30	4.19×10^{9}	1.55×10^{9}
31	9.18×10^{9}	3.36×10^{9}
32	5.52×10^{9}	2.01×10^{9}
33	1.29×10^{10}	4.74×10^{9}
34	1.97×10^{10}	7.24×10^{9}
35	3.03×10^{10}	1.04×10^{10}
36	2.52×10^{10}	9.21×10^{9}
37	3.81×10^{10}	1.39×10^{10}
38	2.18×10^{10}	7.94×10^{9}
39	2.35×10^{10}	8.52×10^{9}
40	3.17×10^{10}	1.15×10^{10}
41	3.87×10^{10}	1.39×10^{10}
42	2.22×10^{10}	7.95×10^{9}
43	2.69×10^{10}	9.63×10^{9}
44	1.78×10^{10}	6.37×10^{9}
45	1.50×10^{10}	5.37×10^{9}
46	2.88×10^{10}	1.03×10^{10}
47	5.30×10^{10}	1.91×10^{10}

TABLE 6-16

SPECTRUM AVERAGED REACTION CROSS-SECTIONS AT THE CENTER OF SURRY UNIT 1 SURVEILLANCE CAPSULES

	$\bar{\sigma}$ (barns)	
Reaction	Capsules T & V (15°)	Capsule W (35°)
$Cu^{63}(n,\alpha)Co^{60}$	0.00068	0.00088
$Fe^{54}(n,p)Mn^{54}$	0.074	0.086
$Ni^{58}(n,p)Co^{58}$	0.100	0.114
$U^{238}(n,f)$	0.354	
$Np^{237}(n,f)$	2.79	

$$\bar{\sigma} = \frac{\int_0^\infty \sigma(E)\,\phi(E)\,dE}{\int_{1.0\ \mathrm{Mev}}^\infty \phi(E)\,dE}$$

TABLE 6-17

THERMAL NEUTRON FLUX DATA FROM CAPSULES T, W, AND V

Capsule	Axial Location	Saturated Activity (DPS/gm) Bare	Cd-Covered	ϕth (n/cm^2-sec)	Adjusted ϕth (n/cm^2-sec)
T	Top	7.32×10^{7}	2.76×10^{7}	8.04×10^{10}	6.47×10^{10}
	Middle	6.34×10^{7}	2.27×10^{7}	7.18×10^{10}	5.77×10^{10}
	Bottom	6.87×10^{7}	2.57×10^{7}	7.59×10^{10}	6.10×10^{10}
W	Top	2.35×10^{7}	1.05×10^{7}	2.29×10^{10}	1.83×10^{10}
	Middle	2.35×10^{7}	7.56×10^{6}	2.81×10^{10}	2.24×10^{10}
	Bottom	2.63×10^{7}	9.60×10^{6}	2.95×10^{10}	2.34×10^{10}
V	Top	5.30×10^{7}	2.42×10^{7}	5.08×10^{10}	4.08×10^{10}
	Middle	4.33×10^{7}	1.66×10^{7}	4.71×10^{10}	3.78×10^{10}
	Bottom	4.47×10^{7}	1.70×10^{7}	4.89×10^{10}	3.93×10^{10}

TABLE 6-18

COMPARISON OF MEASURED AND CALCULATED FAST NEUTRON FLUENCES FOR CAPSULES T, W, AND V*

Capsule	Reaction	Irradiation Time (EFPS)	Φ (E>1.0 MeV) (n/cm^2) Measured	Calculated
T	$Fe^{54}(n,p)Mn^{54}$	3.39×10^{7}	2.83×10^{18}	
	$Ni^{58}(n,p)Co^{58}$		2.79×10^{18}	
	$Cu^{63}(n,\alpha)Co^{60}$		3.03×10^{18}	
	$Np^{237}(n,f)Cs^{137}$		2.38×10^{18}	
	$U^{238}(n,f)Cs^{137}$		2.78×10^{18}	
	Average		2.76×10^{18}	2.81×10^{18}
W	$Fe^{54}(n,p)Mn^{54}$	1.07×10^{8}	3.89×10^{18}	
	$Ni^{58}(n,p)Co^{58}$		4.39×10^{18}	
	$Cu^{63}(n,\alpha)Co^{60}$		4.23×10^{18}	
	Average		4.17×10^{18}	3.97×10^{18}
V	$Fe^{54}(n,p)Mn^{54}$	2.53×10^{8}	1.97×10^{19}	
	$Ni^{58}(n,p)Co^{58}$		1.85×10^{19}	
	$Cu^{63}(n,\alpha)Co^{60}$		2.07×10^{19}	
	$Np^{237}(n,f)Cs^{137}$		1.78×10^{19}	
	$U^{238}(n,f)Cs^{137}$		1.94×10^{19}	
	Average		1.92×10^{19}	1.94×10^{19}

* Measured data have been adjusted to the radial and azimuthal center of the capsule where possible. In addition, corrections were made to the U^{238} monitor activites to account for the U^{235} impurity and build-in of Pu^{239}.

TABLE 6-19

$dPa/\phi(E > 1.0\ MeV)$ RATIOS FOR SURRY UNIT 1

Location	$dPa/\phi(E > 1.0\ MeV)$
15° CAPSULE	1.69×10^{-21}
25° CAPSULE	1.65×10^{-21}
35° CAPSULE	1.63×10^{-21}
45° CAPSULE	1.62×10^{-21}
VESSEL INNER RADIUS (0°)	1.62×10^{-21}
VESSEL 1/4 THICKNESS (0°)	1.87×10^{-21}
VESSEL 3/4 THICKNESS (0°)	2.95×10^{-21}
VESSEL INNER RADIUS (45°)	1.66×10^{-21}
VESSEL 1/4 THICKNESS (45°)	1.92×10^{-21}
VESSEL 3/4 THICKNESS (45°)	3.08×10^{-21}

NOTE: RATIOS ARE IN UNITS OF
[DISPLACEMENTS PER ATOM]/[n/cm^2]

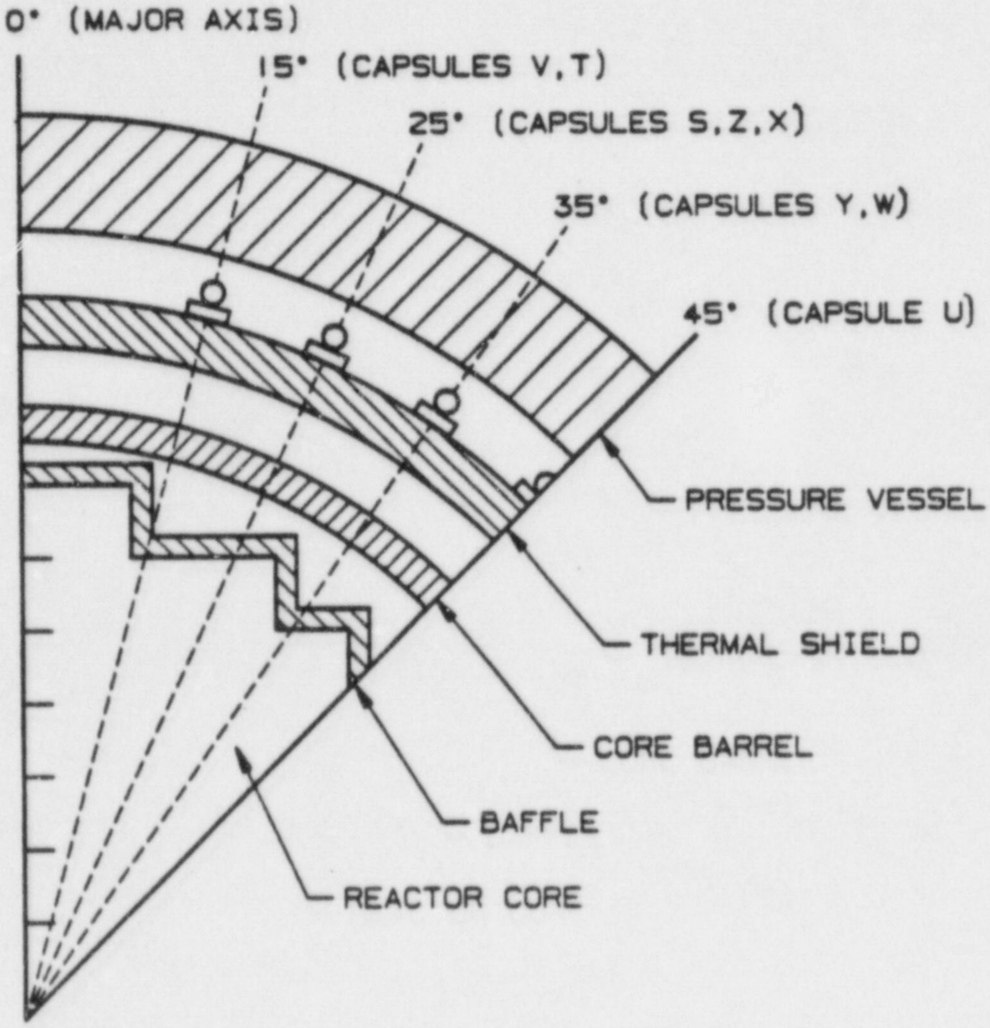

Figure 6-1 Surry Unit 1 Reactor Geometry

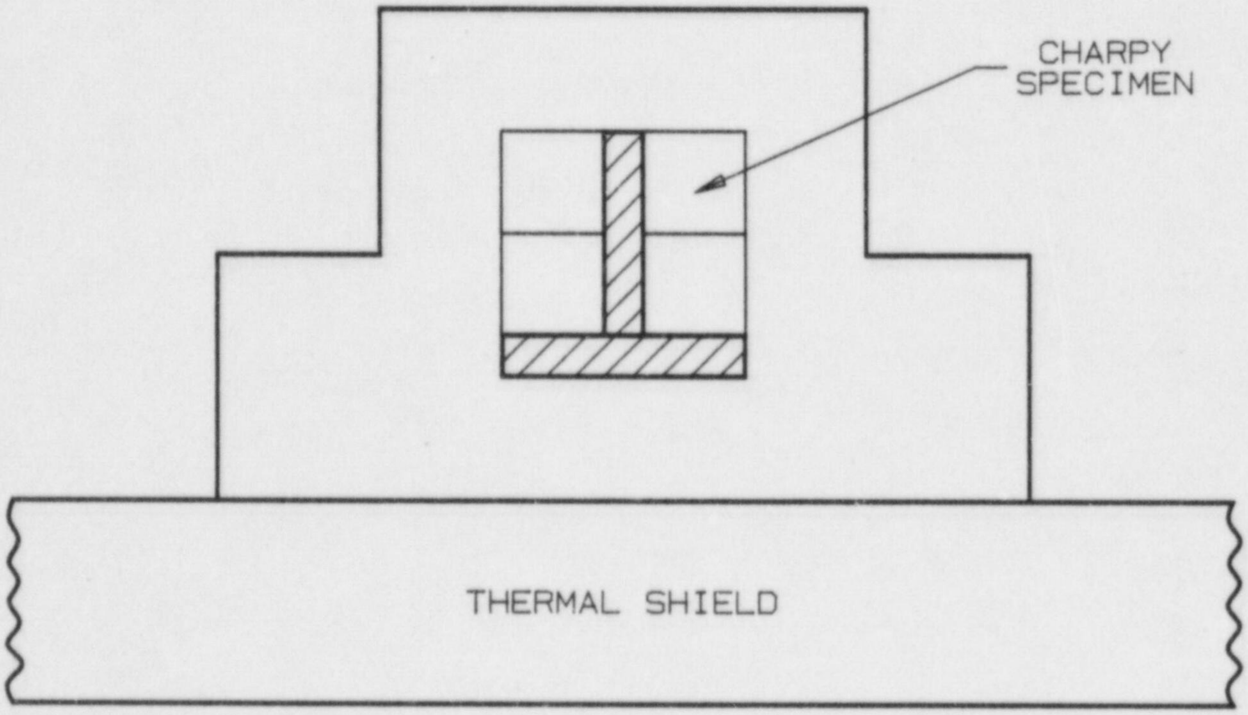

Figure 6-2. Reactor Vessel Surveillance Capsule

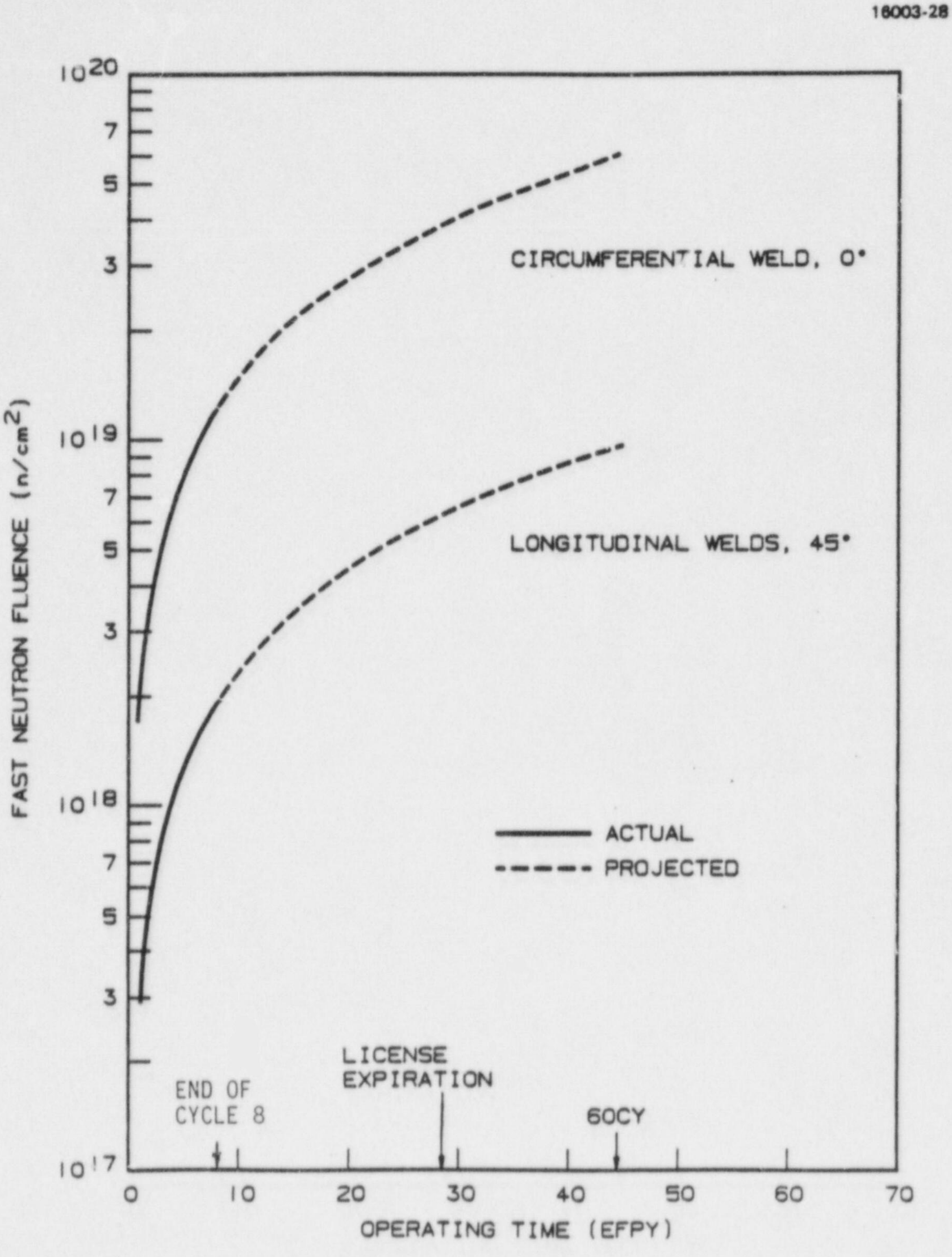

Figure 6-3 Surry Unit 1 Maximum Fast Neutron ($E > 1.0$ MeV) Fluence at the Beltline Weld Locations as a Function of Full Power Operating Time

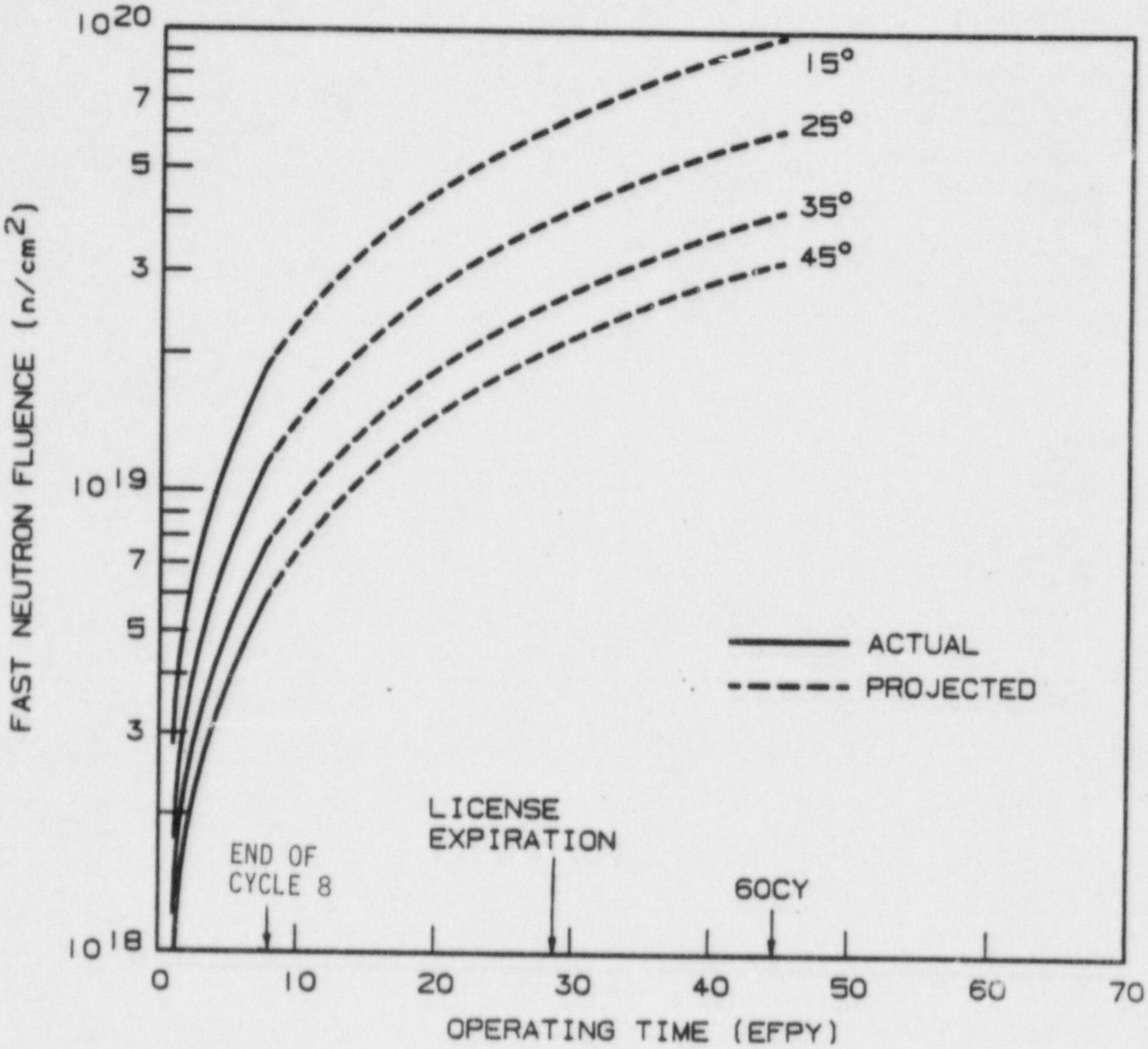

Figure 6-4 Surry Unit 1 Maximum Fast Neutron (E > 1.0 MeV) Fluence at the Center of the Surveillance Capsules as a Function of Full Power Operating Time

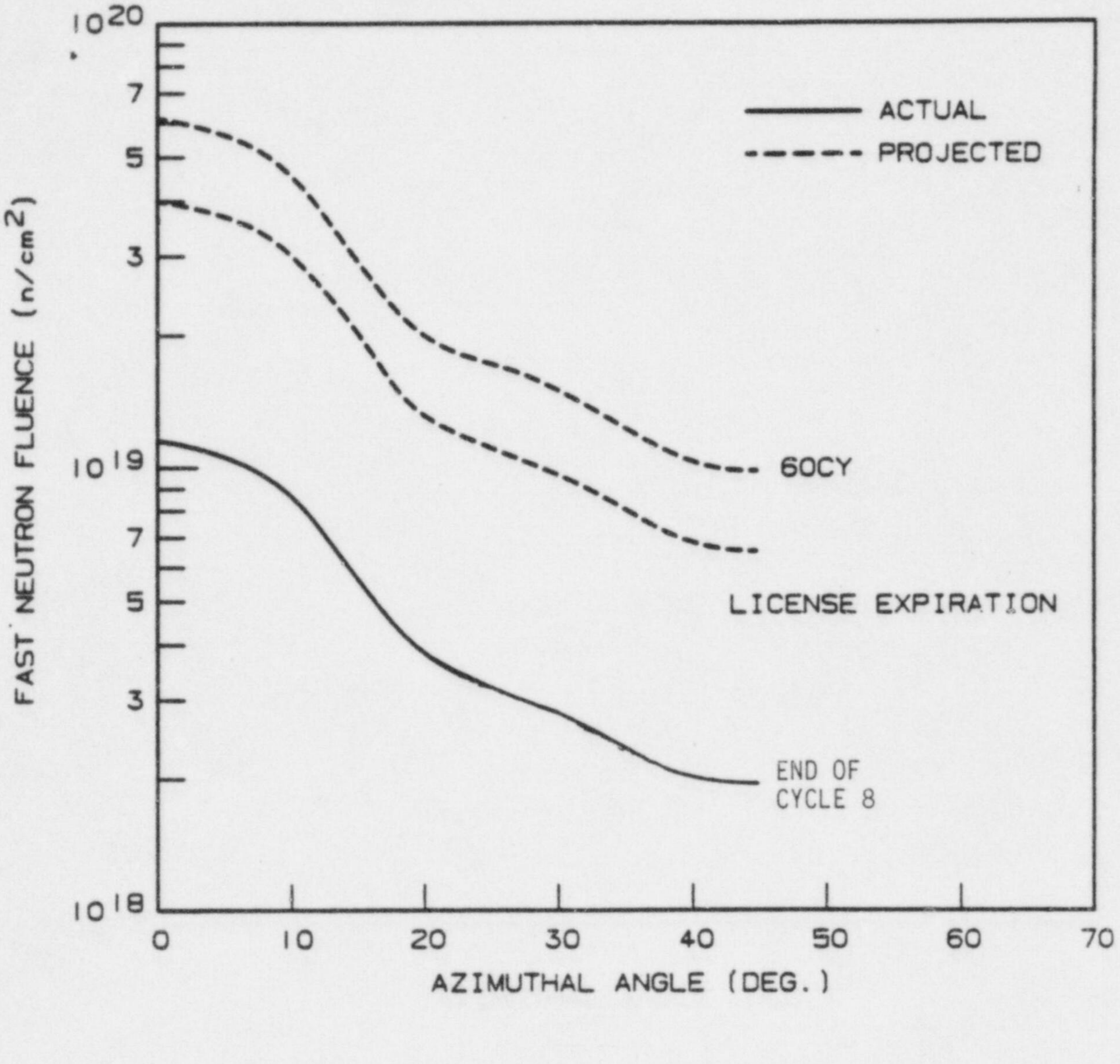

Figure 6-5 Surry Unit 1 Maximum Fast Neutron (E > 1.0 MeV) Fluence at the Pressure Vessel Inner Radius as a Function of Azimuthal Angle

16003-34

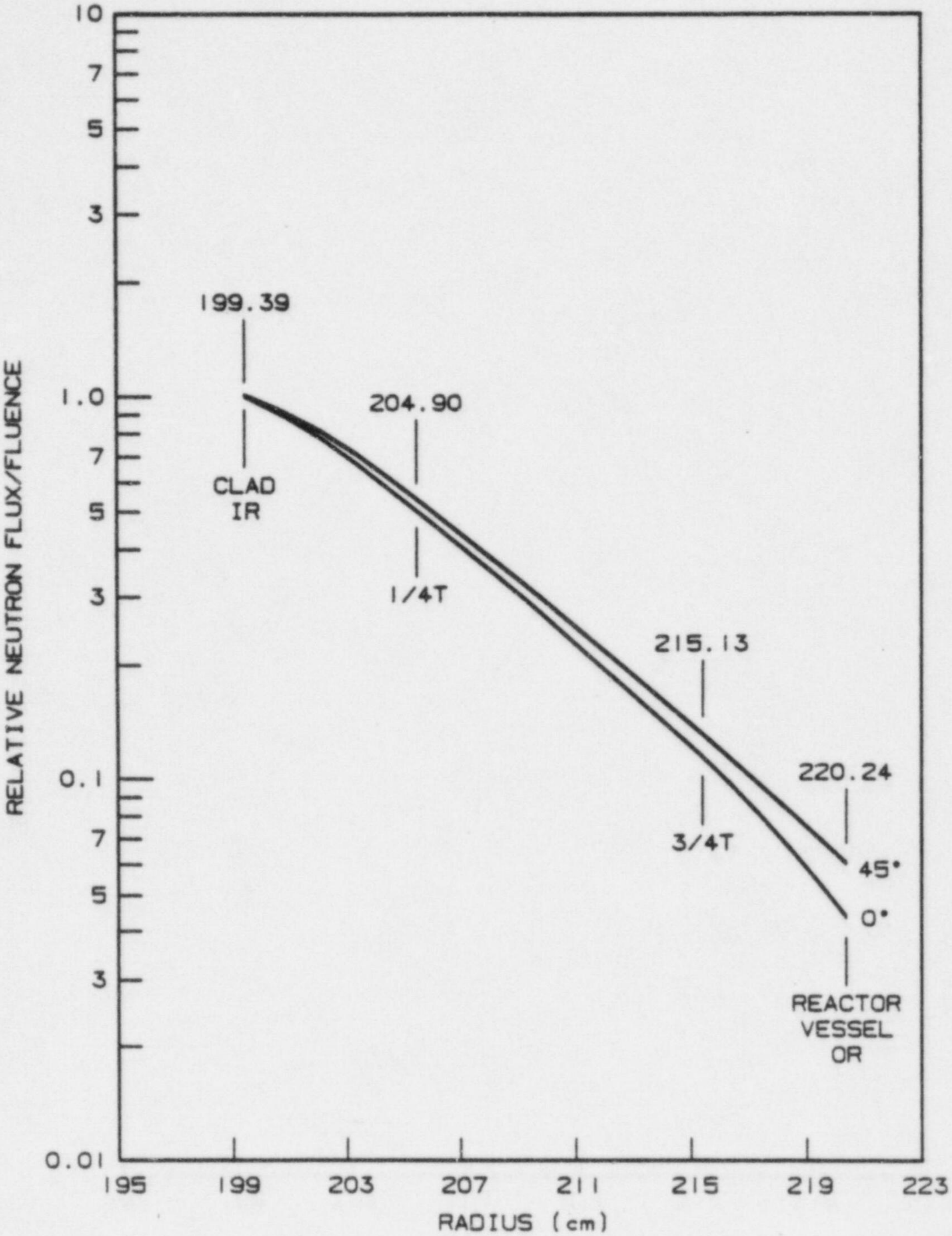

Figure 6-6 Surry Unit 1 Relative Radial Distribution of Fast Neutron (E > 1.0 MeV) Flux and Fluence within the Pressure Vessel Wall

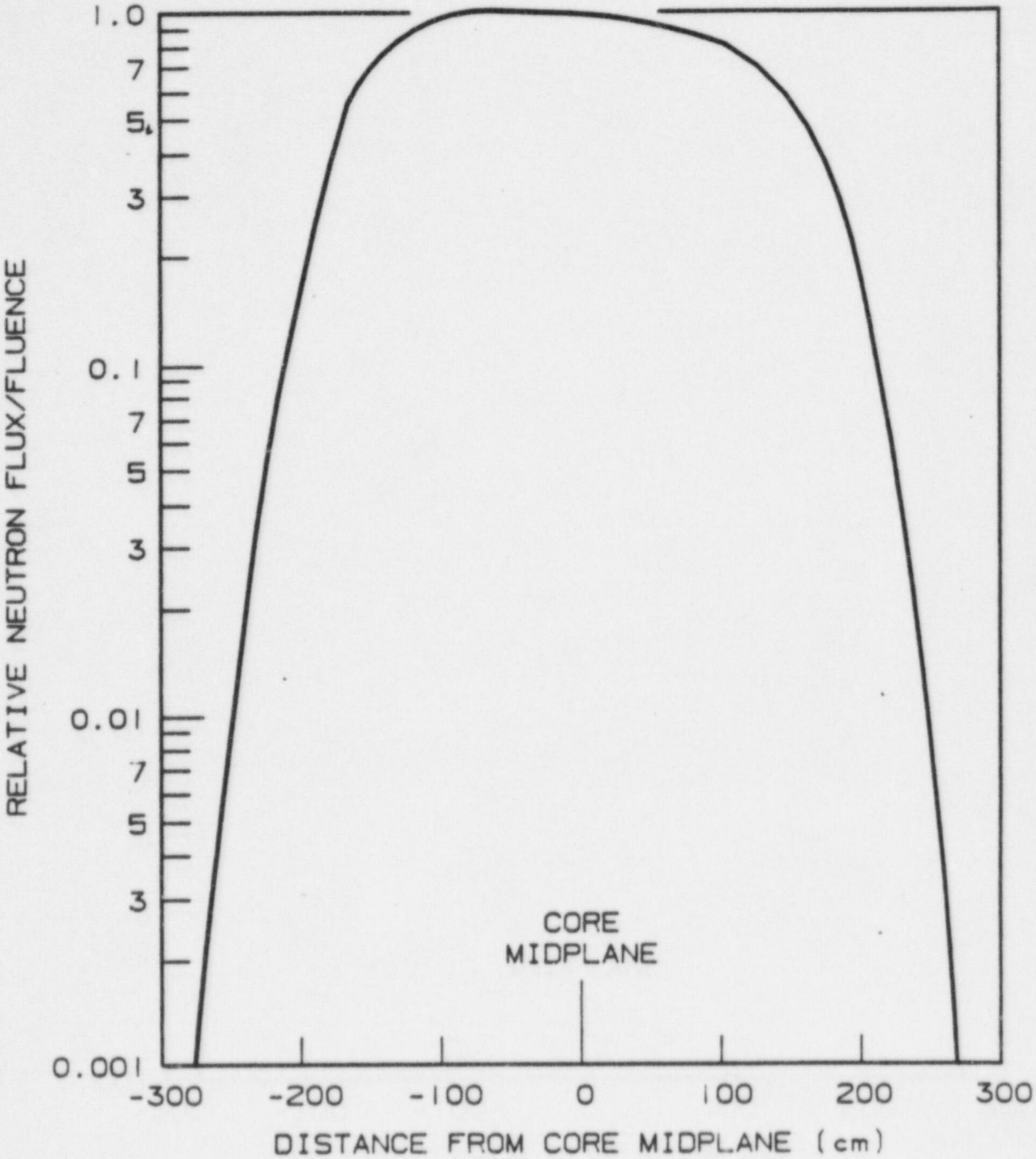

Figure 6-7 Surry Unit 1 Relative Axial Variation of Fast Neutron (E > 1.0 MeV) Flux and Fluence within the Pressure Vessel Wall

SECTION 7
SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE

The following withdrawal schedule per ASTM E185-82 is recommended for future capsules to be removed from the Surry Unit 1 reactor vessel.

Capsule	Vessel Location (deg)	Withdrawal Time[a]	Estimated Fluence (n/cm^2)
T	285°	1.07 (removed)	2.81×10^{18}[b]
W	55°	3.39 (removed)	3.97×10^{18}[b]
V	165°	8.02 (removed)	1.94×10^{19}[b]
X	65°	21.2	2.78×10^{19}
Z	245°	28.8	3.78×10^{19}[c]
S	295°	standby (d)	--
Y	305°	standby (d)	--
U	45°	standby (d)	--

a. Effective full power years from plant startup

b. Actual Fluence

c. Approximate maximum neutron fluence on vessel inner wall at 32 EFPY.

d During 20 year inservice inspection capsules should possibly be transferred to higher flux capsule location.

The surveillance weld represents the Surry Unit 1 reactor vessel lower shell vertical weld seam (L2) which is located at 45° azimuthal position where the maximum neutron fluence after long operating times (30 to 40 EFPY) will be approximately 6 to 8 x 10^{18} n/cm^2. The data obtained for capsule V therefore represents a fluence for seam (L2) which will be well beyond any fluence experienced by seam (L2). Since the surveillance weld is typical of other welds in the beltline region because it was fabricated with Linde 80 flux, data from this weld to be obtained at 21.2 and 28.8 EFPY will be useful in further assessing this type of weld for normal operating life and plant life extension.

It should be noted that the Point Beach Unit 1 vessel surveillance program contains a surveillance weld which was fabricated with the same heat of weld wire (72445) and type of flux (Linde 80) as the Surry Unit 1 intermediate to lower shell girth weld. Data from this weld[20] shows a ΔRT_{NDT} of 165 to 180°F at a neutron fluence of approximately 2.1×10^{19} n/cm^2 and an upper shelf energy of approximately 55 ft-lbs. It is recommended that future weld test results from the Point Beach Unit 1 surveillance program be considered when evaluating the Surry Unit 1 reactor vessel at later times in life.

SECTION 8
REFERENCES

1. Yanichko, S. E., "Virginia Electric and Power Company Surry Unit No. 1 Reactor Vessel Radiation Surveillance Program" WCAP-7723, July 1971.

2. ASTM E185-73 "Practice for Surveillance Tests for Nuclear Reactor Vessels" in ASTM Standards, Part 10 (1973), American Society for Testing and Materials, Philadelphia, Pa., 1973.

3. Perrin, J. S., etal, "Surry Unit No. 1 Pressure Vessel Irradiation Capsule Program: Examination and Analysis of Capsule T," Battelle Columbus Laboratories, June 24, 1975.

4. Perrin, J. S., etal, "Surry Unit No. 1 Nuclear Plant Reactor Pressure Vessel Surveillance Program: Examination and Analysis of Capsule W," Battelle Columbus Laboratories Report BCL-585-8R, March 30, 1979.

5. Lowe, A. L. "Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for Surry Units 1 and 2," Babcock and Wilcox Report BAW-1909, March, 1986.

6. Proposed Revision 2 to Regulatory Guide 1.99 "Radiation Damage to Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, February, 1986.

7. Soltesz, R. G., Disney, R. K., Jedruch, J., and Zeigler, S. L., "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation. Vol. 5 - Two-Dimensional Discrete Ordinates Transport Technique," WANL-PR(LL)034, Vol. 5, August 1970.

8. SAILOR RSIC Data Library Collection "DLC-76, Coupled, Self-shielded, 47 Neutron, 20 Gamma-ray, P3, Cross Section Library for Light Water Reactors."

9. Furchi, E. L., Perone, V. A., Weaver, M., and Wrights, G. N., "Surry Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations," WCAP-11015, December 1985.

10. Benchmark Testing of Westinghouse Neutron Transport Analysis Methodology - to be published.

11. ASTM Designation E261-77, Standard Method for Determining Neutron Flux, Fluence, and Spectra by Radioactivation Techniques," in ASTM Standards (1983), Section 12, Nuclear Standards, pp. 76-87, American Society for Testing and Materials, Philadelphia, Pa., 1983.

12. ASTM Designation E262-77, "Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques," in ASTM Standards (1983), Section 12, Nuclear Standards, pp. 88-96, American Society for Testing and Materials, Philadelphia, Pa., 1983.

13. ASTM Designation E263-82, "Standard Method for Determining Fast-Neutron Flux Density by Radioactivation of Iron," in ASTM Standards (1983), Section 12, Nuclear Standards, pp. 97-102, American Society for Testing and Materials, Philadelphia, Pa., 1983.

14. ASTM Designation 481-78, "Standard Method of Measuring Neutron-Flux Density by Radioactivation of Cobalt and Silver," in ASTM Standards (1983), Section 12, Nuclear Standards, pp. 228-235, American Society for Testing and Materials, Philadelphia, Pa., 1983.

15. ASTM Designation E264-82, "Standard Method for Determining Fast-Neutron Flux Density by Radioactivation of Nickel," in ASTM Standards (1983), Section 12, Nuclear Standards, pp. 103-107, American Society for Testing and Materials, Philadelphia, Pa., 1983.

16. Nucleonics Week, October 1972 and November 1972 issues.

17. Docket 50280-160, "Surry Power Station Units 1 and 2 Semi-Annual Operating Report, SOR-2, for the period January 1, 1973 - June 30, 1973" (Virginia Electric and Power Company, Richmond).

18. "Licensed Operating Reactors Status Summary Report," NUREG 0020, May 1974 through June 1986.

19. Private communication from Vivian Jones of Virginia Electric and Power Company to V. A. Perone of Westinghouse (December 18, 1986).

20. Yanichko, S. E., etal, "Analysis of Capsule T from the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-10736, Decembe- 1984.