BAW-2082

March 1990

ANALYSIS OF CAPSULE Y
COMMONWEALTH EDISON COMPANY
ZION NUCLEAR PLANT UNIT 1

-- Reactor Vessel Material Surveillance Program --

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-- Reactor Vessel Material Surveillance Program --

by

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B&W Document No. 77-1176466-00
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SUMMARY

This report describes the results of the examination of the fourth capsule (Capsule Y) of the Commonwealth Edison Company, Zion Nuclear Plant Unit 1 reactor vessel surveillance program. The objective of the program is to monitor the effects of neutron irradiation on the tensile and fracture toughness properties of the reactor vessel materials by the testing and evaluation of tension and Charpy impact specimens. The program was designed in accordance with the requirements of ASTM Specification E185-70.

The capsule received an average fast fluence of 1.56×10^{19} n/cm^2 (E > 1.0 MeV) and the predicted fast fluence for the reactor vessel T/4 location at the end of the tenth cycle is 2.90×10^{18} n/cm^2 (E > 1 MeV). Based on the calculated fast flux at the vessel wall, an 80% load factor, and the planned fuel management, the projected fast fluence that the Zion Nuclear Plant Unit 1 reactor pressure vessel inside surface will receive in 40 calendar years of operation is 1.60×10^{19} n/cm^2 (E > 1 MeV) and the corresponding T/4 fluence is calculated to be 8.89×10^{18} n/cm^2 (E > 1 MeV). The peak calculated RT_{NDT} at T/4 vessel wall location is 257F at EOL per Regulatory Guide 1.99, Rev. 2 but appears to be no greater than 200F based on data from this surveillance capsule. Likewise, the T/4 vessel wall upper-shelf energy is calculated to decrease to 41 ft-lbs per Regulatory Guide 1.99, Rev. 2 but based on data from this surveillance capsule should not decrease below 50 ft-lbs prior to EOL.

The results of the tension tests indicated that the materials exhibited normal behavior relative to neutron fluence exposure. The Charpy impact data results exhibited the characteristic shift to higher temperature for the 30 ft-lb transition temperature and a decrease in upper-shelf energy. These results demonstrated that the current techniques used for predicting the change in both the increase in the RT_{NDT} and the decrease in upper-shelf properties due to irradiation are conservative.

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1. INTRODUCTION

This report describes the results of the examination of the fourth capsule (Capsule Y) of the Commonwealth Edison Company, Zion Nuclear Plant Unit 1 (Zion-1) reactor vessel material surveillance program (RVSP). The capsule was removed and evaluated after being irradiated in the Zion Nuclear Plant Unit-1 as part of the reactor vessel materials surveillance program (WCAP-8064).[1] The capsule experienced a fluence of 1.56×10^{19} n/cm^2 (E > 1 MeV), which is the equivalent of approximately 31 effective full power years' (EFPY) operation of the Zion Nuclear Plant Unit 1 reactor vessel inside surface. The first capsule (Capsule T) from this program was removed and examined after the first year of operation; the results are reported in BCL-585-4.[2] The second capsule (Capsule U) was removed and examined after irradiation for 4 cycles, or 3.5 EFPY; the results are reported in WCAP-9890.[3] The third capsule (Capsule X) of the program was removed and evaluated after 6 cycles, or 5 EFPY and the results reported by Southwest Research Institute in SWRI 06-7484-001.[4]

The objective of the program is to monitor the effects of neutron irradiation on the tensile and impact properties of reactor pressure vessel materials under actual operating conditions. The surveillance program for Zion Nuclear Plant Unit-1 was designed and furnished by Westinghouse Electric Corporation (W) as described in WCAP-8064[1] and conducted in accordance with 10CFR50, Appendix H. The program was planned to monitor the effects of neutron irradiation on the reactor vessel materials for the 40-year design life of the reactor pressure vessel.

2. BACKGROUND

The ability of the reactor pressure vessel to resist fracture is the primary factor in ensuring the safety of the primary system in light water-cooled reactors. The beltline region of the reactor vessel is the most critical region of the vessel because it is exposed to neutron irradiation. The general effects of fast neutron irradiation on the mechanical properties of such low-alloy ferritic steels as SA533, Grade B, used in the fabrication of the Zion Unit 1 reactor vessel, are well characterized and documented in the literature. The low-alloy ferritic steels used in the beltline region of reactor vessels exhibit an increase in ultimate and yield strength properties with a corresponding decrease in ductility after irradiation. The most significant mechanical property change in reactor pressure vessel steels is the increase in temperature for the transition from brittle to ductile fracture accompanied by a reduction in the Charpy upper shelf energy value.

Appendix G to 10CFR50, "Fracture Toughness Requirements,"[6] specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of water-cooled power reactors, and provides specific guidelines for determining the pressure-temperature limitations on operation of the RCPB. The toughness and operational requirements are specified to provide adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. Although the requirements of Appendix G to 10CFR50 became effective on August 13, 1973, the requirements are applicable to all boiling and pressurized water-cooled nuclear power reactors, including those under construction or in operation on the effective date.

Appendix H to 10CFR50, "Reactor Vessel Materials Surveillance Program Requirements,"[7] defines the material surveillance program required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water-cooled reactors resulting from exposure to neutron irradiation and the thermal environment. Fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. These data will permit determination of the conditions under which the vessel can be operated with adequate safety margins against fracture throughout its service life.

A method for guarding against brittle fracture in reactor pressure vessels is described in Appendix G to the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."[8] This method utilizes fracture mechanics concepts and the reference nil-ductility temperature, RT_{NDT}, which is defined as the greater of the drop weight nil-ductility transition temperature (per ASTM E-208) or the temperature that is 60F below that at which the material exhibits 50 ft-lbs and 35 mils lateral expansion. 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 ASME Section III. The K_{IR} curve is a lower bound of dynamic, static, and crack arrest 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 using these allowable stress intensity factors.

The RT_{NDT} and, in turn, the operating limits of a nuclear power plant, can be adjusted to account for the effects of radiation on the properties of the reactor vessel materials. The radiation embrittlement and the resultant changes in mechanical properties of a given pressure vessel steel can be monitored by a surveillance program in which a surveillance capsule containing prepared specimens of the reactor vessel materials is periodically removed from the operating nuclear reactor and the specimens are tested. The increase in the Charpy V-notch 30 ft-lb temperature is added to the original RT_{NDT} to adjust it for radiation embrittlement. This adjusted RT_{NDT} is used to index the material to the K_{IR} curve which, in turn, is used to set operating limits for the nuclear power plant. These new limits take into account the effects of irradiation on the reactor vessel materials.

Appendix G, 10CFR50, also requires a minimum Charpy V-notch upper-shelf energy of 75 ft-lbs for all beltline region materials unless it is demonstrated that lower values of upper-shelf fracture energy will provide an adequate margin for deterioration as the result of neutron radiation. No action is required for a material that does not meet the 75 ft-lb requirement provided the irradiation deterioration does not cause the upper-shelf energy to drop below 50 ft-lbs. The regulations specify that if the upper-shelf energy drops below 50 ft-lbs, it must be demonstrated in a manner approved by the Office of Nuclear Regulation that the lower values will provide adequate margins of safety.

When a reactor vessel fails to meet the 50 ft-lb requirement, a program must be submitted for review and approval at least three years prior to the time the predicted fracture toughness will no longer satisfy the regulatory requirements. The program must address the following:

A. A volumetric examination of 100 percent of the beltline materials that do not meet the requirement.

B. Supplemental fracture toughness data as evidence of the fracture toughness of the irradiated beltline materials.

C. Fracture toughness analysis to demonstrate the existence of equivalent margins of safety for continued operation.

If these procedures do not indicate the existence of an adequate margin of safety, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness properties of the materials.

3. SURVEILLANCE PROGRAM DESCRIPTION

The surveillance program for Zion Unit 1 comprises eight surveillance capsules designed to monitor the effects of neutron and thermal environment on the materials of the reactor pressure core region. The capsules, which were inserted into the reactor vessel before initial plant startup, were positioned inside the reactor vessel between the thermal shield and the vessel wall at the locations shown in Figure 3-1. The eight capsules, designed to be placed in holders attached to the thermal shield are positioned near the peak axial and azimuthal neutron flux. WCAP-8064 includes a full description of the capsule locations and design. During the ten cycles of operation, capsule Y was irradiated in the 320^{o} position as shown in Figure 3-1.

Capsule Y was removed during the tenth refueling shutdown of Zion Unit 1. The capsule contained Charpy V-notch impact test specimens fabricated from the one base metal (SA533, Grade B1), one heat-affected-zone, a weld metal and a correlation monitor. Tension test specimens were fabricated from the base metal and the weld metal only. In addition, specimens were included for determining the fracture toughness of the weld metal. The number of specimens of each material contained in the capsule are described in Table 3-1, and the location of the individual specimens within the capsule are described in Figure 3-2. The chemical composition and heat treatment of the surveillance material in capsule Y are described in Table 3-2.

All test specimens were machined from the 1/4-thickness (1/4T) location of the plate material. Charpy V-notch and tension test specimens were cut from the surveillance material such that they were oriented with their longitudinal axes either parallel or perpendicular to the principal working direction.

Capsule Y contained dosimeter wires, described as follows:

Dosimeter	Shielding
U_3O_8	CdO
NpO_2	CdO
Ni	None
Co-Al	Cd
Co-Al	None
Fe	None
Cu	None

Thermal monitors of low-melting alloys were included in the capsule. The alloys and their melting points are as follows:

Alloy	Melting Point, F
97.5% Pb, 2.5% Ag	579
97.5% Pb, 1.75% Ag, 0.75% Sn	590

Table 3-1. Specimens in Surveillance Capsule Y

	Number of Test Specimens Per Capsule		
Material Description	Tension	CVN[a] Impact	WOL[b]
Plate: B7835-1			
Longitudinal	2	10	--
Transverse	--	10	--
HAZ	--	8	--
Weld Metal: WF-209-1[c]	2	8	4
Correlation Monitor			
HSST Plate 02	--	8	--
Total per Capsule	4	44	4

[a] CVN denotes Charpy V-notch.

[b] WOL denotes wedge opening loading.

[c] Per BAW-1543, Rev. 3.

Table 3-2. Chemical Composition and Heat Treatment of Surveillance Materials[9]

Element	Chemical Composition, w/o[(d)] Heat No. B7835-1[(a)]	HSST Plate 02[(b)]	Weld Metal (WF-209-1)[(c)]
C	0.20	0.22	0.09
Mn	1.30	1.48	1.45
P	0.010	0.012	0.020
S	0.011	0.018	0.013
Si	0.20	0.25	0.68
Ni	0.49	0.68	0.57
Cr	---	---	0.063
Mo	0.47	0.52	0.39
Cu	0.11	0.14	0.35

Heat No.	Heat Treatment[(e)] Temp, F	Time, h	Cooling
Plate B7835-1	1625	9.75	Brine Quenched
	1212	9.75	Brine Quenched
	1125	25	Furnace Cooled
Weld Metal	1125	23	Furnace Cooled
Correlation Monitor	1650 - 1700	4	Air Cooled
	1575 - 1625	4	Water-Quenched
	1200 - 1250	4	Furnace Cooled
	1125 - 1175	40	Furnace Cooled to 600°F

(a) Chemical analysis by Westinghouse of surveillance program test plate B7835-1.[3]

(b) Chemical analysis from ORNL-4313.

(c) Chemical analysis by Westinghouse of surveillance program test weld metal.[3]

(d) Expanded chemical analysis is contained in WCAP-8064.[1]

(e) Post weld heat treatment data.[1]

Figure 3-1. Reactor Vessel Cross Section Showing Location of Capsule Y in Zion Unit 1

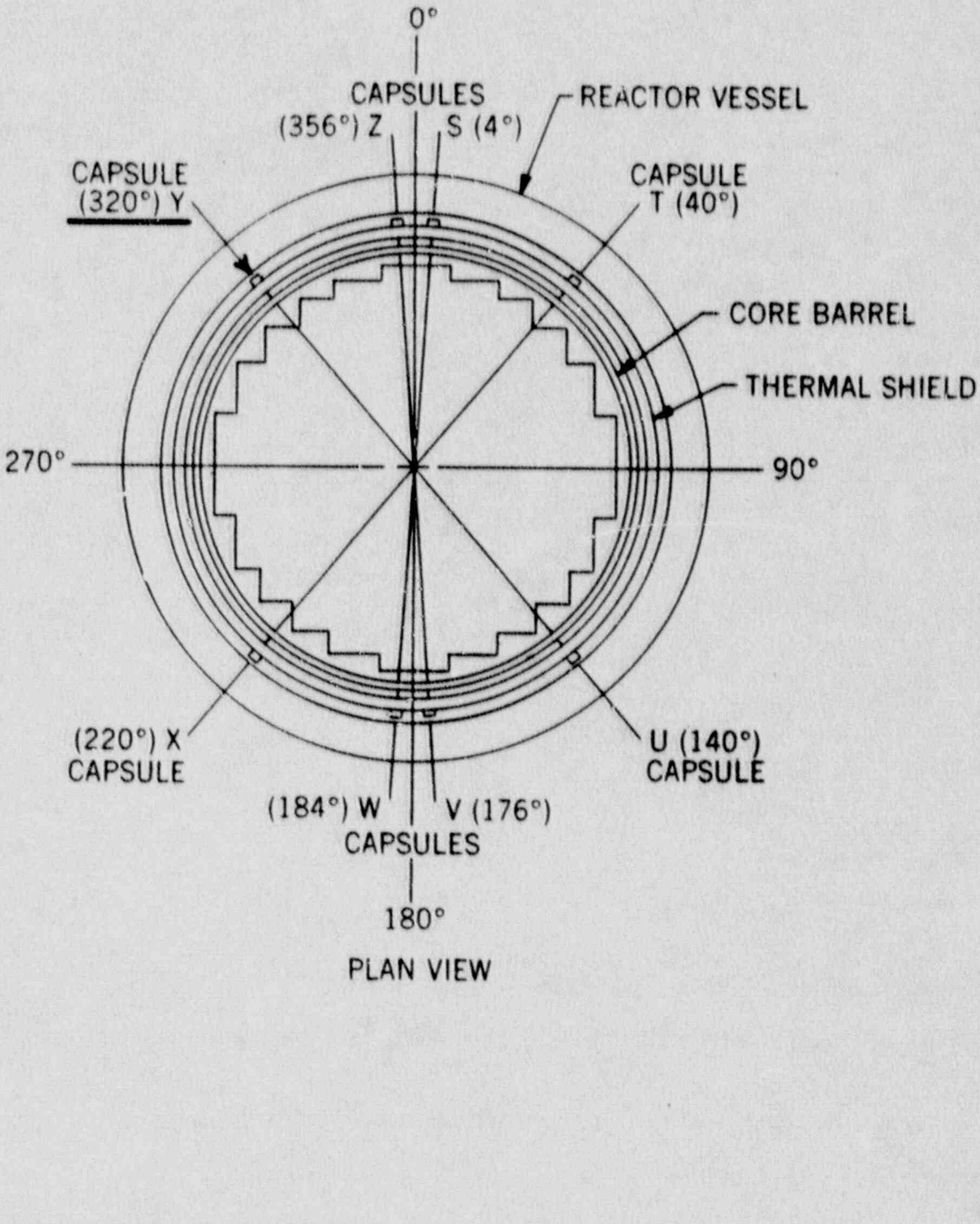
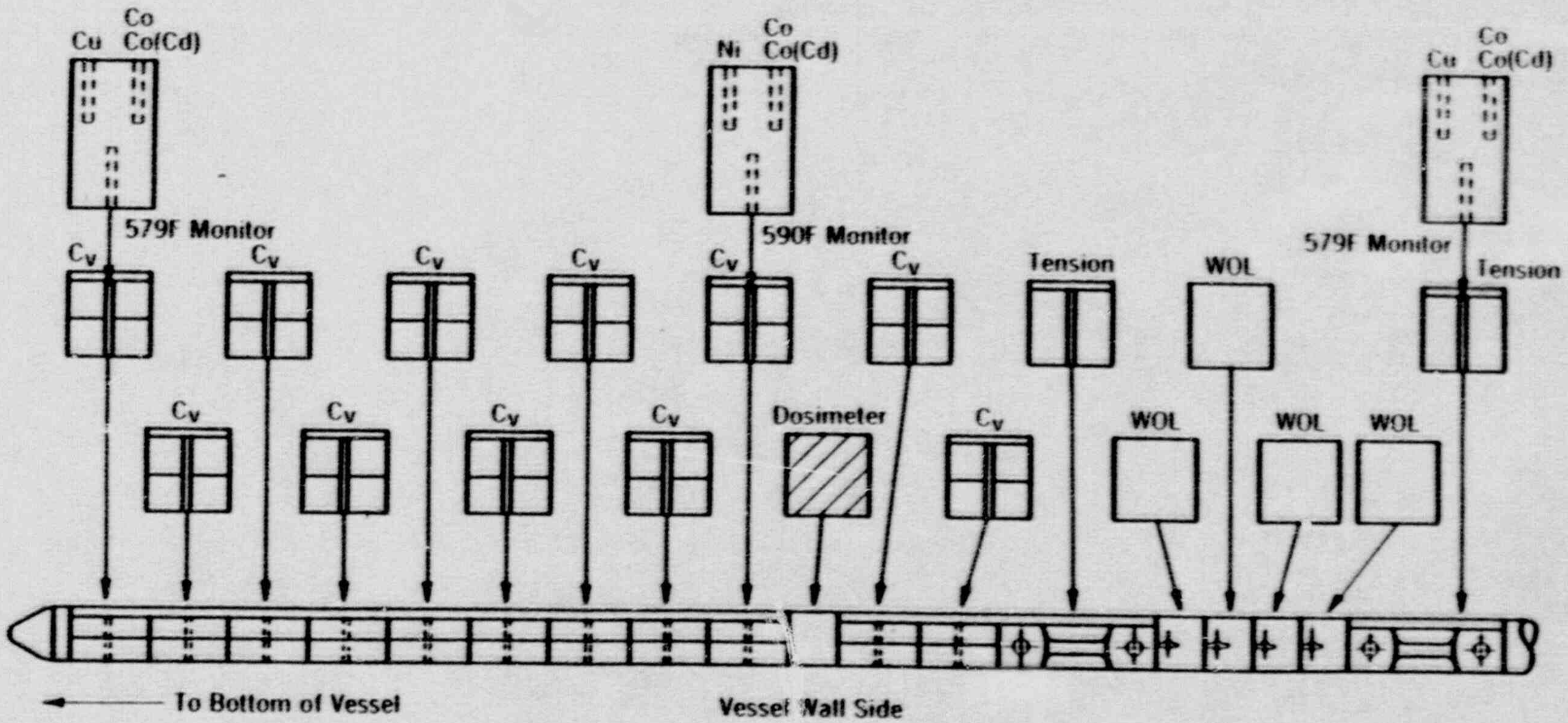


Figure 3-2. Loading Diagram for Test Specimens in Capsule Y



4. PRE-IRRADIATION TESTS

Unirradiated material was evaluated for two purposes: (1) to establish a baseline of data to which irradiated properties data could be referenced, and (2) to determine those materials properties to the extent practical from available material, as required for compliance with Appendixes G and H to 10CFR50.

The pre-irradiated specimens were tested by Westinghouse as part of the development of the Zion Unit 1 surveillance program. The details of the testing procedures are described in WCAP-8064 and are summarized here to provide continuity.

4.1. Tension Tests

Tension test specimens were fabricated from the reactor vessel shell plate and weld metal. The specimens were 4.23 inches long with a reduced section 1.255 inches long by 0.250 inch in diameter. They were tested on a universal test machine. An extension device with a strain gaged extensometer was used to determine the 0.2% yield point. Test conditions were in accordance with the applicable requirements of ASTM A370-68.[10] For each material type and/or condition, six specimens in groups of two were tested at room temperature, 300 and 600F. All test data for the pre-irradiation tensile specimens are given in Appendix B.

4.2. Impact Tests

Charpy V-notch impact tests were conducted in accordance with the requirements of ASTM E23-72[11] on an impact tester certified to meet Watertown standards.[12] Test specimens were of the Charpy V-notch type, which were nominally 0.394 inch square and 2.115 inches long.

Impact test data for the unirradiated baseline reference materials are presented in Appendix C. Tables C-1 through C-5 contain the basis data that are plotted in Figures C-1 through C-5. These data were replotted and re-evaluated to be consistent with the irradiated Charpy curves and evaluations.

5. POST-IRRADIATION TESTING

5.1. Visual Examination and Inventory

The capsule was inspected and photographed upon receipt. Figure 5-1 confirms the markings as those of Capsule Y. The contents of the capsule were inventoried and found to be consistent with the surveillance program report inventory. All specimens were visually examined and no signs of abnormalities were found. There was no evidence of rust or of the penetration of reactor coolant into the capsule. The weld metal wedge opening loading (WOL) specimens were stored for future disposition.

5.2. Thermal Monitors

Surveillance capsule Y contained two temperature monitors in two holder blocks which also contained dosimeters. The upper holder block contained a thermal monitor designed to melt at 590F and the lower holder block contained a thermal monitor designed to melt at 570F. The holder blocks were radiographed for evaluation. None of the thermal monitors exhibited any signs of melting. From these data, it was concluded that the irradiated specimens had been exposed to a maximum temperature of less than 579F during the reactor vessel operating period. This is not significantly greater than the nominal inlet temperature of 558F, and is considered acceptable for inclusion of the data in the general pool of irradiated surveillance data. There appeared to be no significant signs of a temperature gradient along the capsule length.

5.3. Tension Test Results

The results of the post-irradiation tension tests are presented in Table 5-1. Tests were performed on specimens at both room temperature and 550F. They were tested on a 55,000-lb load capacity universal test machine at a cross-head speed of 0.005 inch per minute to yield point and thereafter 0.050 inch per minute. A 4-pole extension device with a strain gaged extensometer was used to determine the 0.2% yield point. Test conditions were in accordance with the applicable requirements of ASTM A370-77.[13] For each material type and/or condition, specimens were tested at both room temperature and 550F. The tension-compression load cell used had a certified accuracy of better than +0.5% of full scale (25,000 lb). Photographs of the tension test specimen fractured surfaces are presented in Figure 5-7.

In general, the ultimate strength and yield strength of the material increased with a corresponding slight decrease in ductility as compared to the unirradiated values; both effects were the result of neutron radiation damage. The type of behavior observed and the degree to which the material properties changed in within the range of changes to be expected for the radiation environment to which the specimens were exposed.

The results of the pre-irradiation tension tests are presented in Appendix B.

5.4. Charpy V-Notch Impact Test Results

The test results from the irradiated Charpy V-notch specimens of the reactor vessel beltline material are presented in Tables 5-2 through 5-5 and Figures 5-2 through 5-6. Photographs of the Charpy specimen fracture surfaces are presented in Figures 5-8 through 5-12. The Charpy V-notch impact tests were conducted in accordance with the requirements of ASTM E23-86[14] on an impact tester certified to meet Watertown standards.[12]

The data show that the materials exhibited a sensitivity to irradiation within the values to be expected based on their chemical composition and the fluence to which they were exposed. Detailed discussion of the results are provided in Section 7.

The results of the pre-irradiation Charpy V-notch impact tests are given in Appendix C.

5.5. Chemical Composition Check Analysis

Nine tested irradiated Charpy specimens from the base metal and the weld metal were analyzed by emission spectrograph to determine the nickel and copper content of the individual specimens. The results of these analysis are presented in Table 5-6. The National Institute of Standards and Technology (NIST) Standard Reference Materials (SRM) used and their nominal composition for the elements of interest are shown in Table 5-8. Also presented, for comparison, is the nominal composition for Mn-Mo-Ni/Linde 80 Submerged-arc weld metal. This comparison shows that all elements are well bracketed.

5.6. Wedge Opening Loading Specimens

The weld metal wedge opening loading specimens were not tested. Two specimens were stored for testing at a future date. Two specimens were transferred to the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program for further irradiation.[29]

Table 5-1. Tensile Properties of Capsule Y Base Metal and Weld Metal Irradiated to 1.56×10^{19} n/cm^2 (E > 1 MeV)

Specimen No.	Test Temp, F	Strength, psi: Yield	Strength, psi: Ultimate	Fracture: Load, lbs	Fracture: Stress, psi	Fracture: Strength, psi	Elongation, %: Uniform	Elongation, %: Total	Reduction in Area, %
Base Metal, B7835-1, Longitudinal									
E5	70	77,300	98,000	3,190	190,700	65,000	10.9	24.1	65.9
E6	550	67,700	91,500	3,350	153,000	68,300	8.3	19.5	55.4
Weld Metal, Transverse									
W13	70	93,100	108,400	3,920	185,600	79,900	11.1	22.8	57.0
W14	550	84,700	103,900	3,930	165,400	80,100	8.7	18.6	51.6

Table 5-2. Charpy Impact Results for Capsule Y Base Metal Longitudinal (LT) Orientation, Heat No. B7835-1, 1.56 x 10^{19} n/cm^2

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion inch	Shear Fracture %
E62	40	14.0	0.013	5
E66	70	28.0	0.023	10
E63	110	35.0	0.030	30
E64	140	46.5	0.040	50
E70	175	72.0	0.060	50
E69	225	102.5	0.078	90
E68	275	111.5	0.080	100
E65	350	114.0	0.082	100
E67	400	119.0	0.090	100
E61	550	109.5	0.083	100

Table 5-3. Charpy Impact Results for Capsule Y Base Metal Transverse (TL) Orientation, Heat No. B7835-1, 1.56 x 10^{19} n/cm^2

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion inch	Shear Fracture %
ET68	70	17.5	0.016	10
ET70	110	30.5	0.025	30
ET63	140	32.0	0.029	40
ET66	150	42.5	0.039	50
ET65	175	55.5	0.049	60
ET61	225	82.5	0.068	85
ET62	275	96.0	0.078	100
ET67	350	95.0	0.076	100
ET69	400	87.0	0.070	100
ET64	550	95.5	0.080	100

Table 5-4. Charpy Impact Results for Capsule Y Heat Affected Zone Metal, Heat No. B7835-1, 1.56 x 10^{19} n/cm^2

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion inch	Shear Fracture %
H55	-50	20.0	0.016	10
H49	-25	45.0	0.033	20
H54	0	46.0	0.036	50
H52	40	57.5	0.048	60
H51	70	99.0	0.073	85
H56	150	106.0	0.079	100
H53	275	116.0	0.091	100
H50	400	118.5	0.087	100

Table 5-5. Charpy Impact Results for Capsule Y Weld Metal, WF-209-1, 1.56 x 10^{19} n/cm^2

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion inch	Shear Fracture %
W53	70	7.5	0.005	5
W55	140	20.0	0.018	40
W52	175	25.5	0.022	40
W54	225	30.0	0.028	70
W49	290	43.0	0.041	100
W50	350	43.5	0.046	100
W56	400	44.5	0.042	100
W51	550	46.5	0.046	100

Table 5-6. Charpy Impact Results for Capsule Y Correlation Monitor Material, HSST PL-02 Longitudinal (LT) Orientation, 1.56 x 10^{19} n/cm^2

Specimen ID	Test Temperature F	Impact Energy ft-lbs	Lateral Expansion inch	Shear Fracture %
R53	150	22.0	0.018	20
R55	175	37.0	0.032	40
R50	200	40.5	0.034	40
R52	225	54.5	0.044	60
R56	260	76.0	0.061	100
R51	300	80.5	0.066	100
R49	350	93.0	0.077	100
R54	550	98.0	0.084	100

Table 5-7. Chemical Analysis Results of Selected Charpy Specimens*

Charpy Specimen	Chemical Composition, w/o									
	C	Mn	P	S	Si	Ni	Cr	Mo	V	Cu
Base Metal, B7835-1										
E64	0.24	1.45	0.007	0.011	0.22	0.53	0.099	0.47	0.001	0.12
Weld Metal, WF-209-1										
W49	0.09	1.52	0.019	0.008	0.70	0.55	0.080	0.38	0.007	0.25
W50	0.11	1.48	0.020	0.009	0.72	0.55	0.078	0.39	0.007	0.22
W51	0.13	1.52	0.020	0.009	0.70	0.54	0.085	0.38	0.006	0.22
W52	0.09	1.51	0.020	0.009	0.69	0.54	0.083	0.38	0.007	0.23
W53	0.09	1.48	0.020	0.009	0.68	0.54	0.080	0.39	0.007	0.23
W54	0.10	1.50	0.020	0.009	0.70	0.54	0.080	0.40	0.007	0.22
W55	0.10	1.54	0.020	0.010	0.72	0.55	0.083	0.38	0.007	0.24
W56	0.09	1.50	0.021	0.009	0.70	0.53	0.073	0.38	0.007	0.24
Mean	0.10	1.51	0.020	0.009	0.70	0.54	0.080	0.39	0.007	0.23
Std. Dev.	0.014	0.021	0.001	0.001	0.014	0.007	0.004	0.008	0.000	0.011

*Analysis performed with an emission spectrograph (see paragraph 5.5).

Table 5-8. Chemical Composition of NIST Standard Reference Materials

Standard Reference Material Number	Nominal Chemical Composition, Weight %									
	C	Mn	P	S	Si	Ni	Cr	Mo	V	Cu
1261	0.38	0.66	0.015	0.017	0.223	1.99	0.69	0.19	0.011	0.042
1262	0.16	1.04	0.042	0.037	0.39	0.59	0.30	0.07	0.04	0.50
1263	0.62	1.50	0.029	0.008	0.74	0.32	1.31	0.03	0.31	0.10
1264	0.87	0.25	0.010	0.028	0.067	0.14	0.06	0.49	0.10	0.25
1265	0.0067	0.057	0.002	0.006	0.008	0.041	0.007	0.005	0.001	0.006
Nominal Weld Metal	0.08	1.52	0.016	0.014	0.48	0.59	0.09	0.40	---	0.29

Figure 5-1. Photograph of Capsule Y Identifier

Figure 5-2. Charpy Impact Data for Irradiated Plate Material, Longitudinal Orientation, Heat No. B7835-1

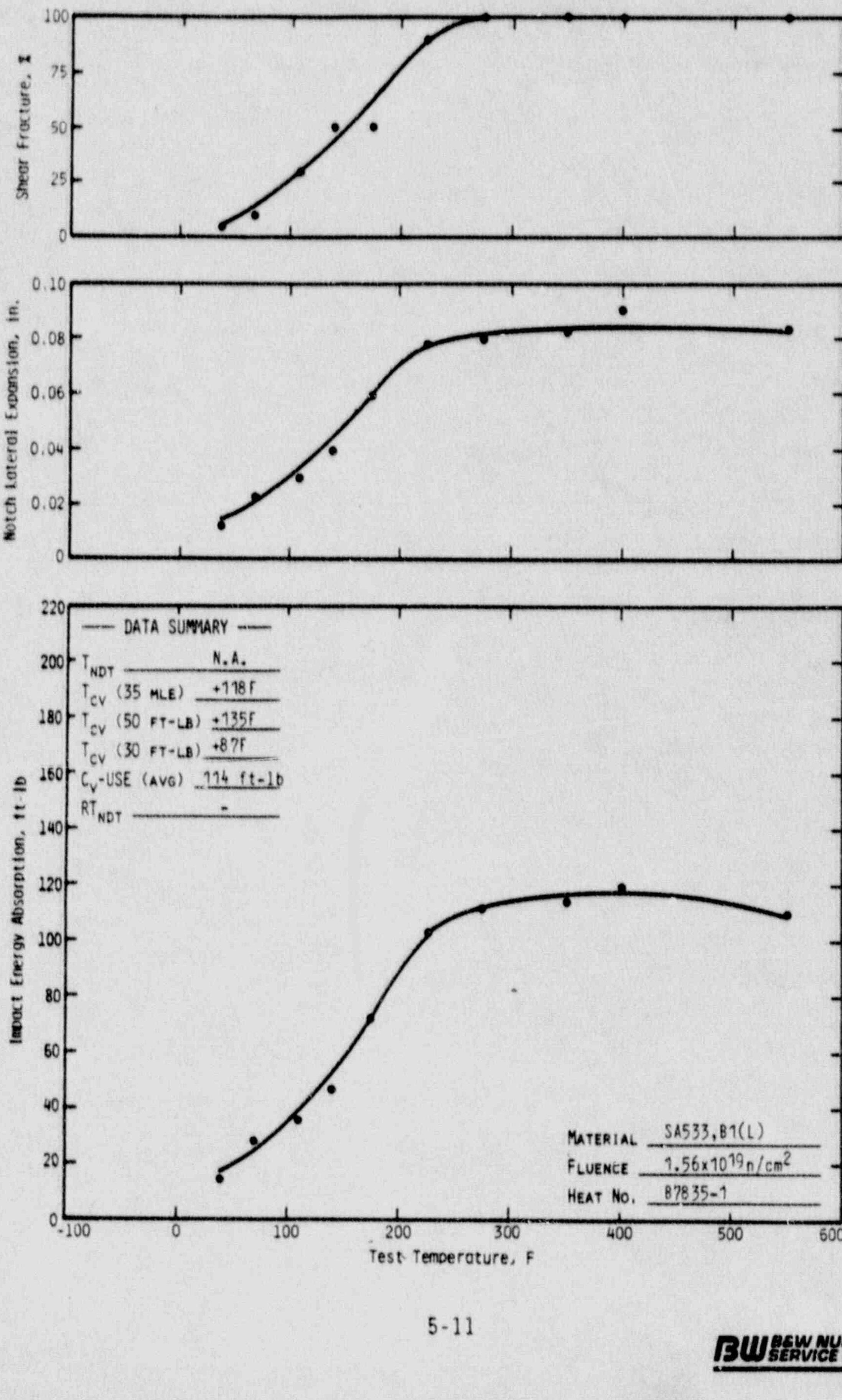


Figure 5-3. Charpy Impact Data for Irradiated Plate Material, Transverse Orientation, Heat No. B7835-1

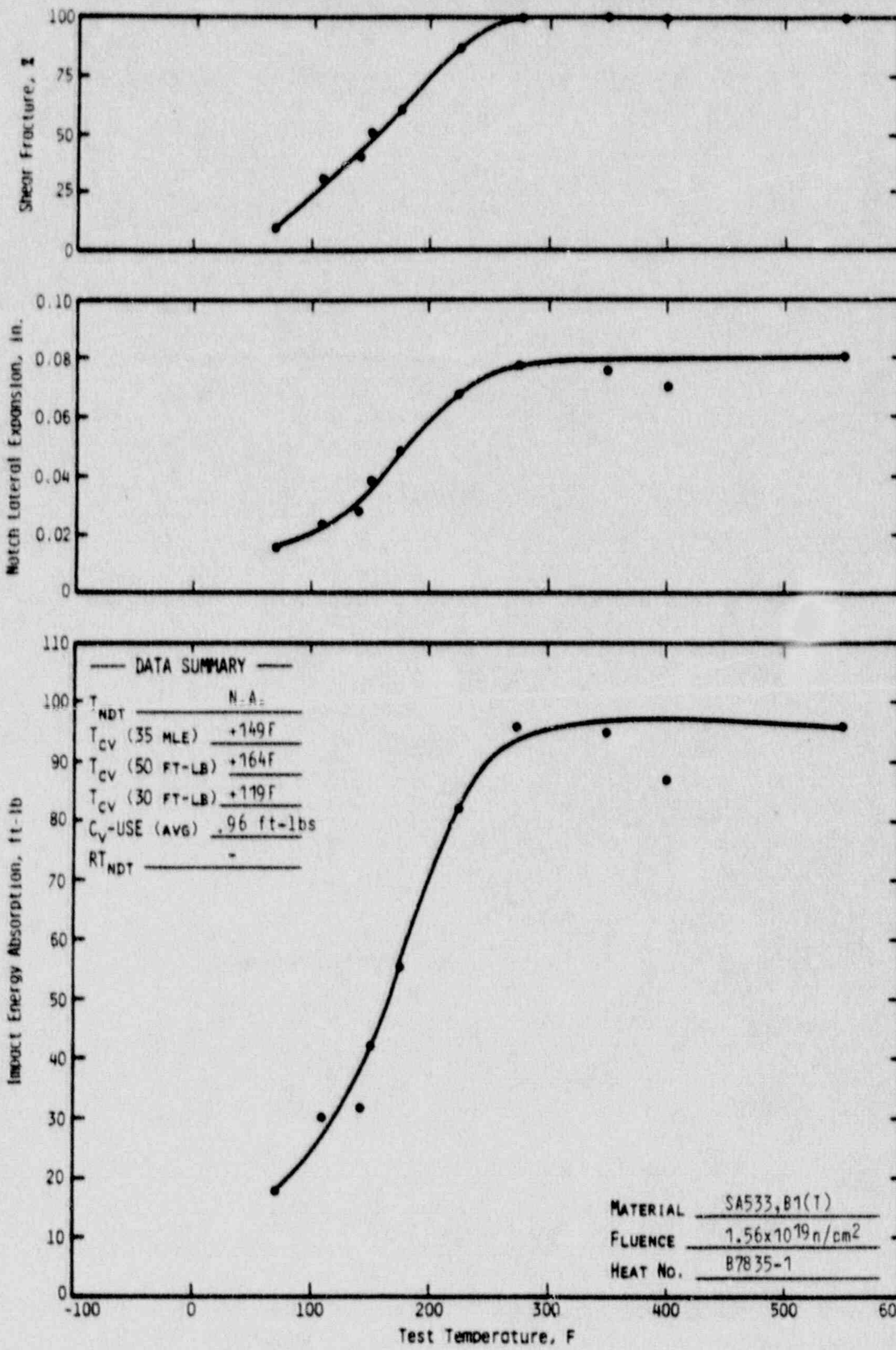


Figure 5-4. Charpy Impact Data for Irradiated Plate Material, Heat-Affected Zone, Heat No. B7835-1

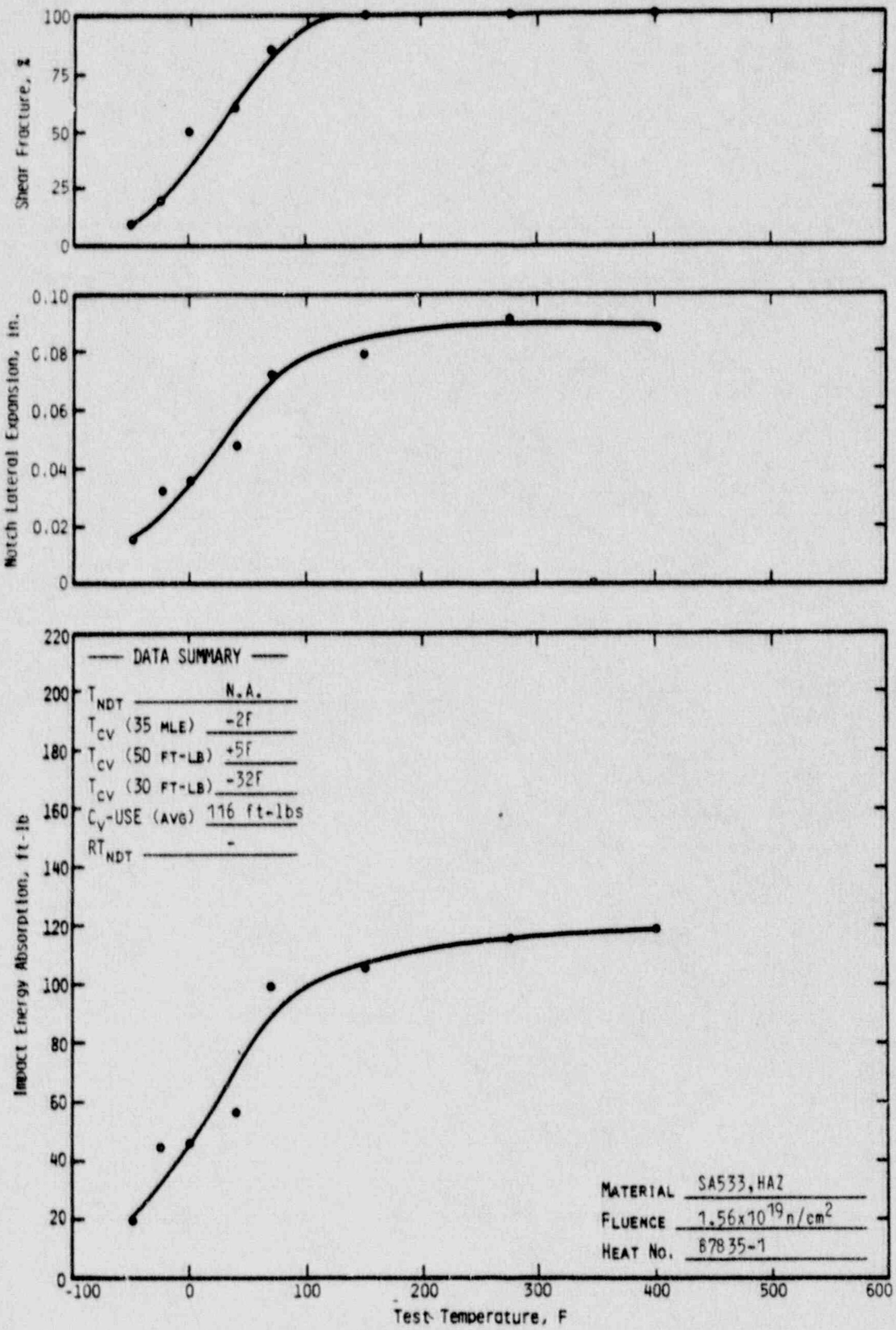


Figure 5-5. Charpy Impact Data for Irradiated Weld Metal, WF-209-1

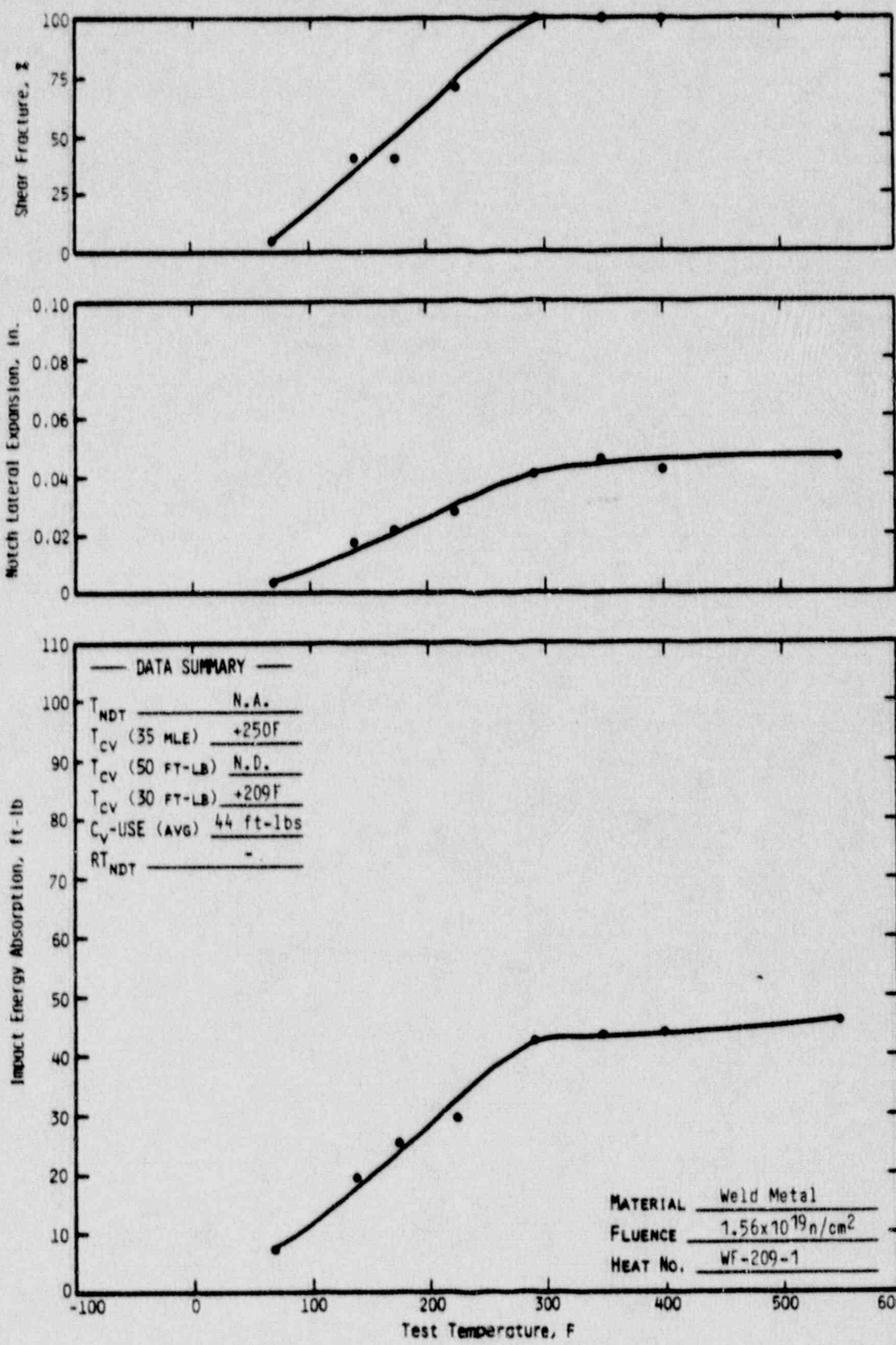


Figure 5-6. Charpy Impact Data for Unirradiated Correlation Monitor Material, HSST PL-02, Heat No. A1195-1

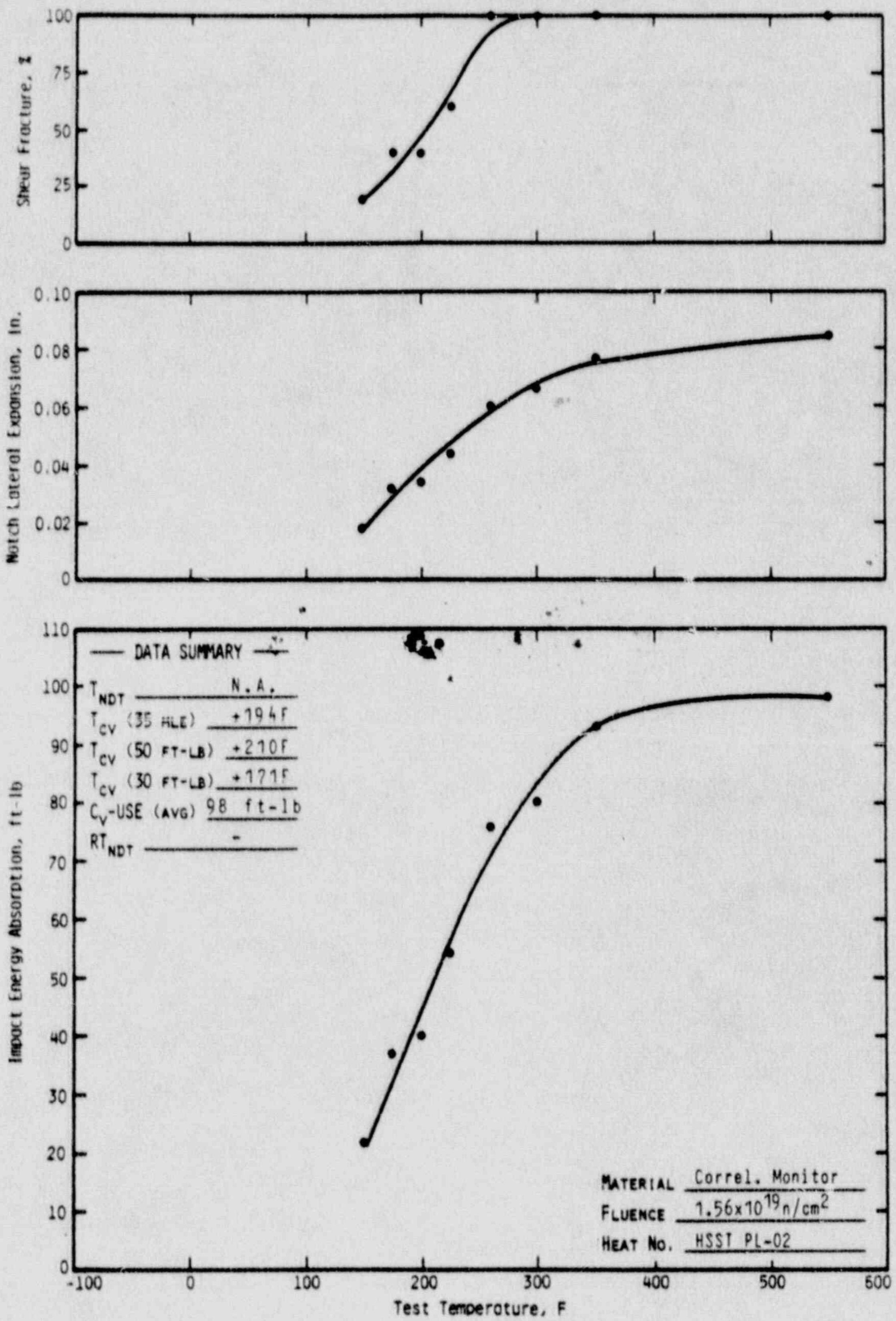


Figure 5-7. Photographs of Tension Test Specimens Fracture Surfaces

Base Metal

Specimen E5 (70F)

Specimen E6 (550F)

Weld Metal

Specimen W13 (70F)

Specimen W14 (550F)

Figure 5-8. Photographs of Charpy Impact Specimen Fracture Surfaces - Plate Material Longitudinal Orientation, Heat No. B7835-1

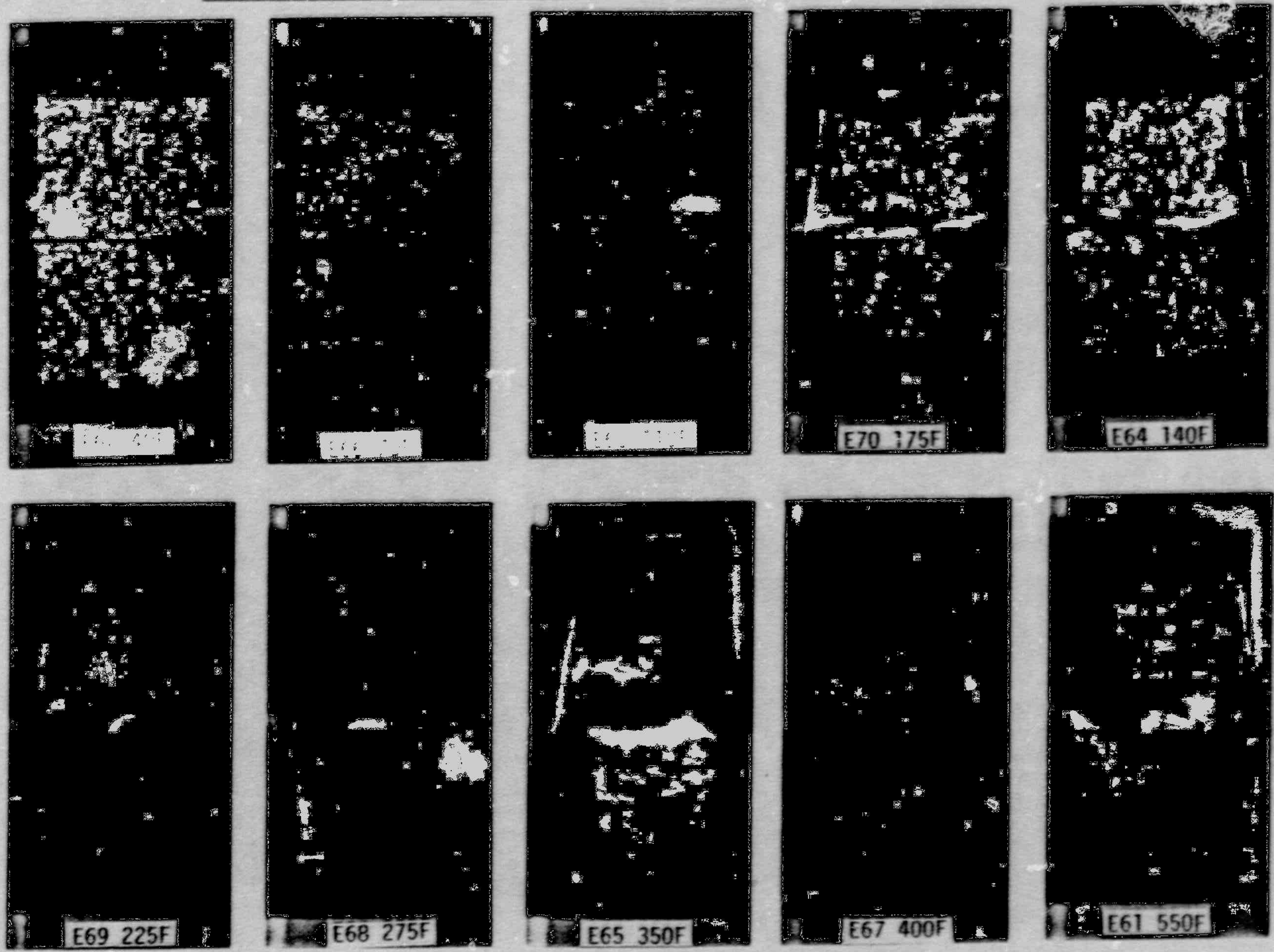


Figure 5-9. Photographs of Charpy Impact Specimen Fracture Surfaces - Plate Material Transverse Orientation, Heat No. B7835-1

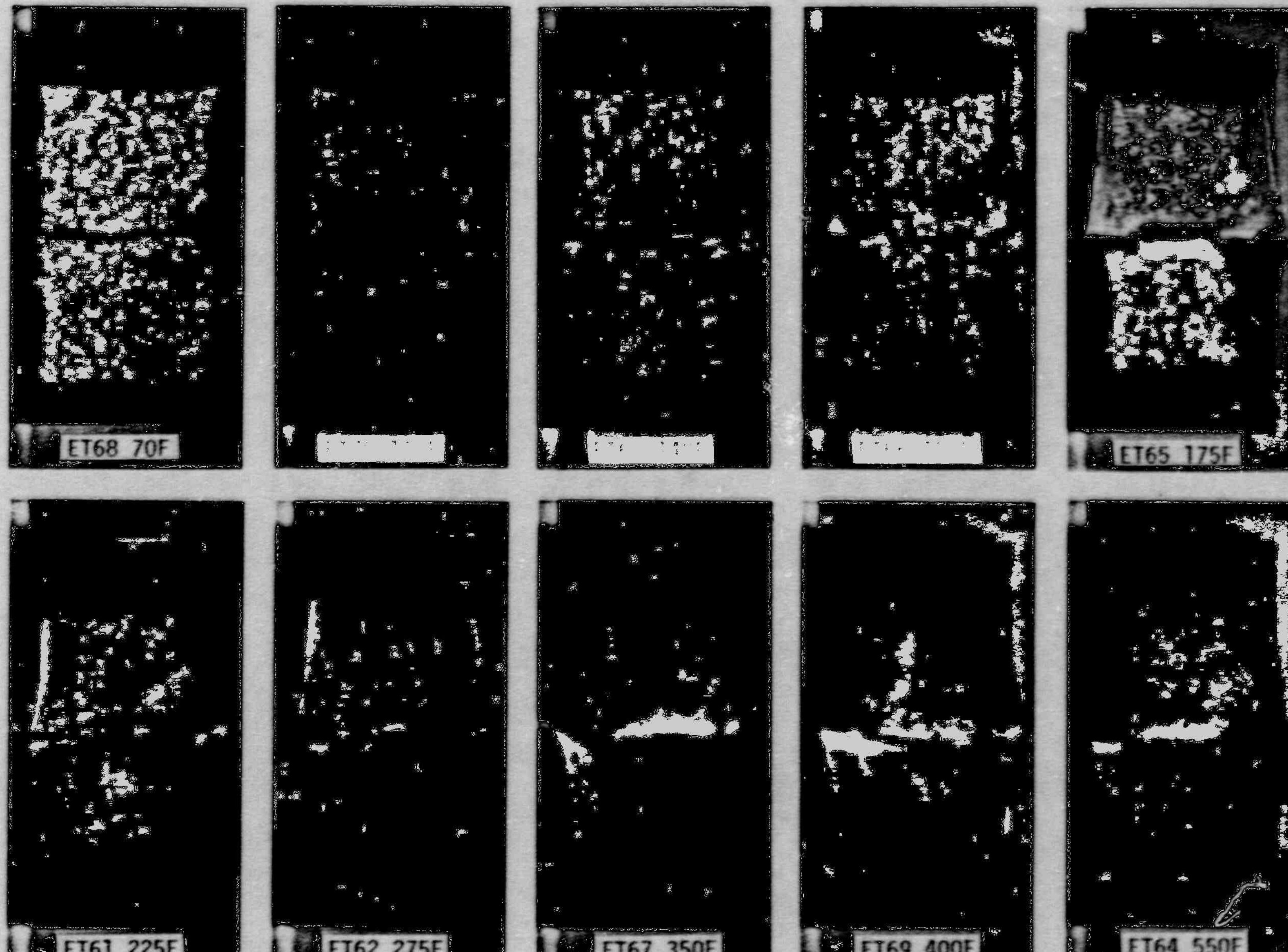


Figure 5-10. Photographs of Charpy Impact Specimen Fracture Surfaces - Plate Material Heat-Affected Zone, Heat No. B7835-1

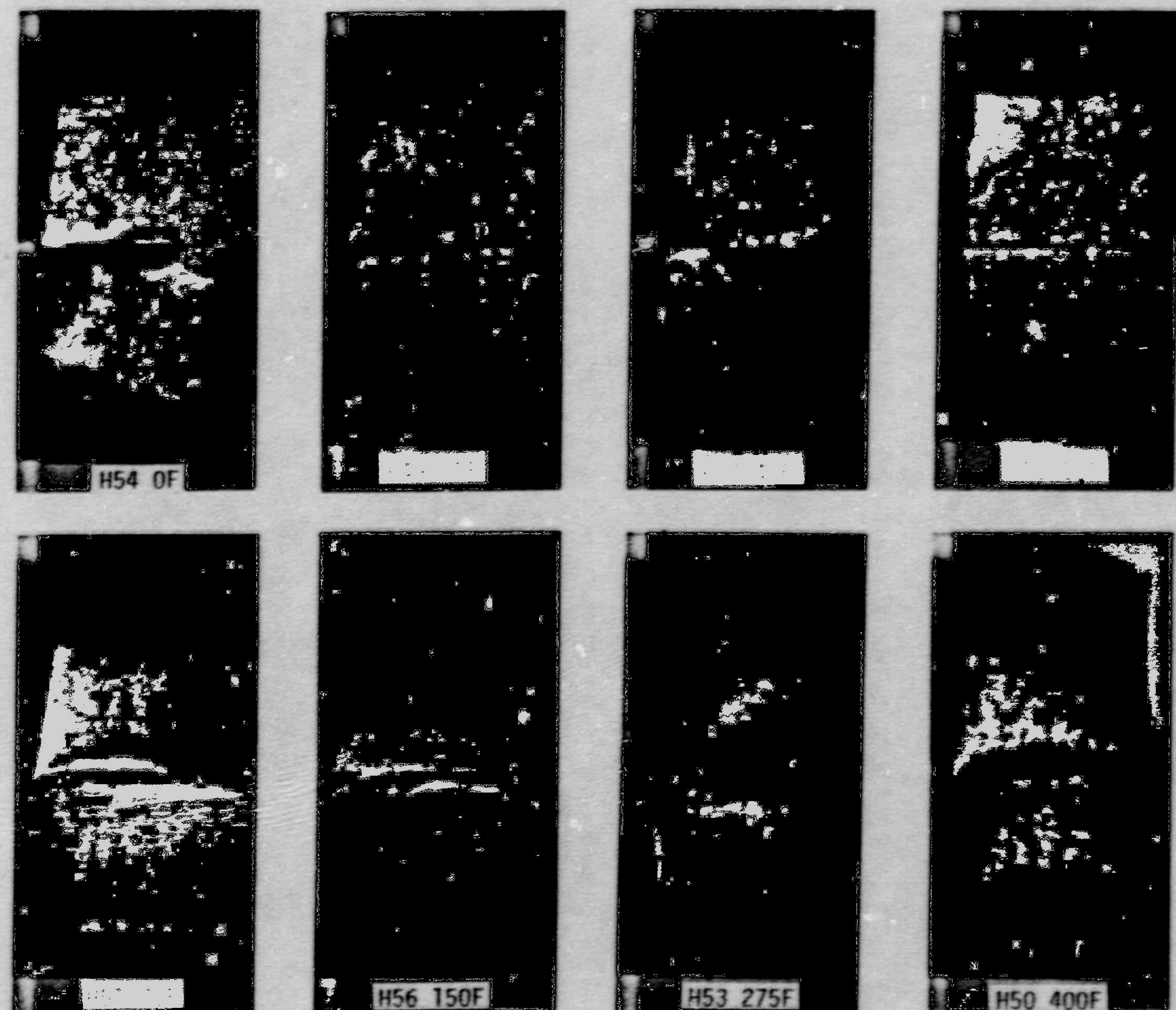


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Figure 5-11. Photographs of Charpy Impact Specimen Fracture Surfaces - Weld Metal, WF-209-1

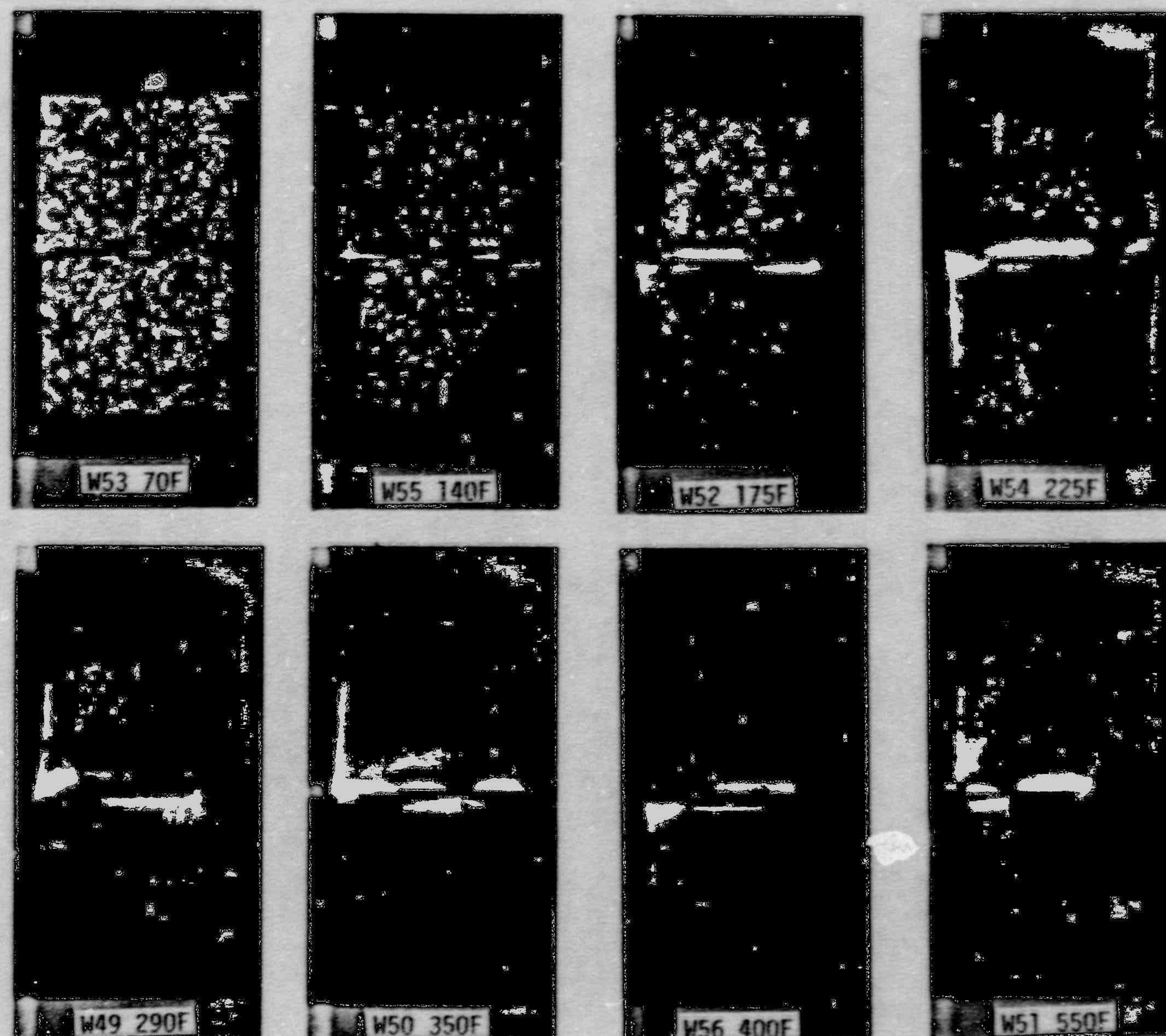


Figure 5-12. Photographs of Charpy Impact Specimen Fracture Surfaces - Correlation Monitor Material, HSST PL-02

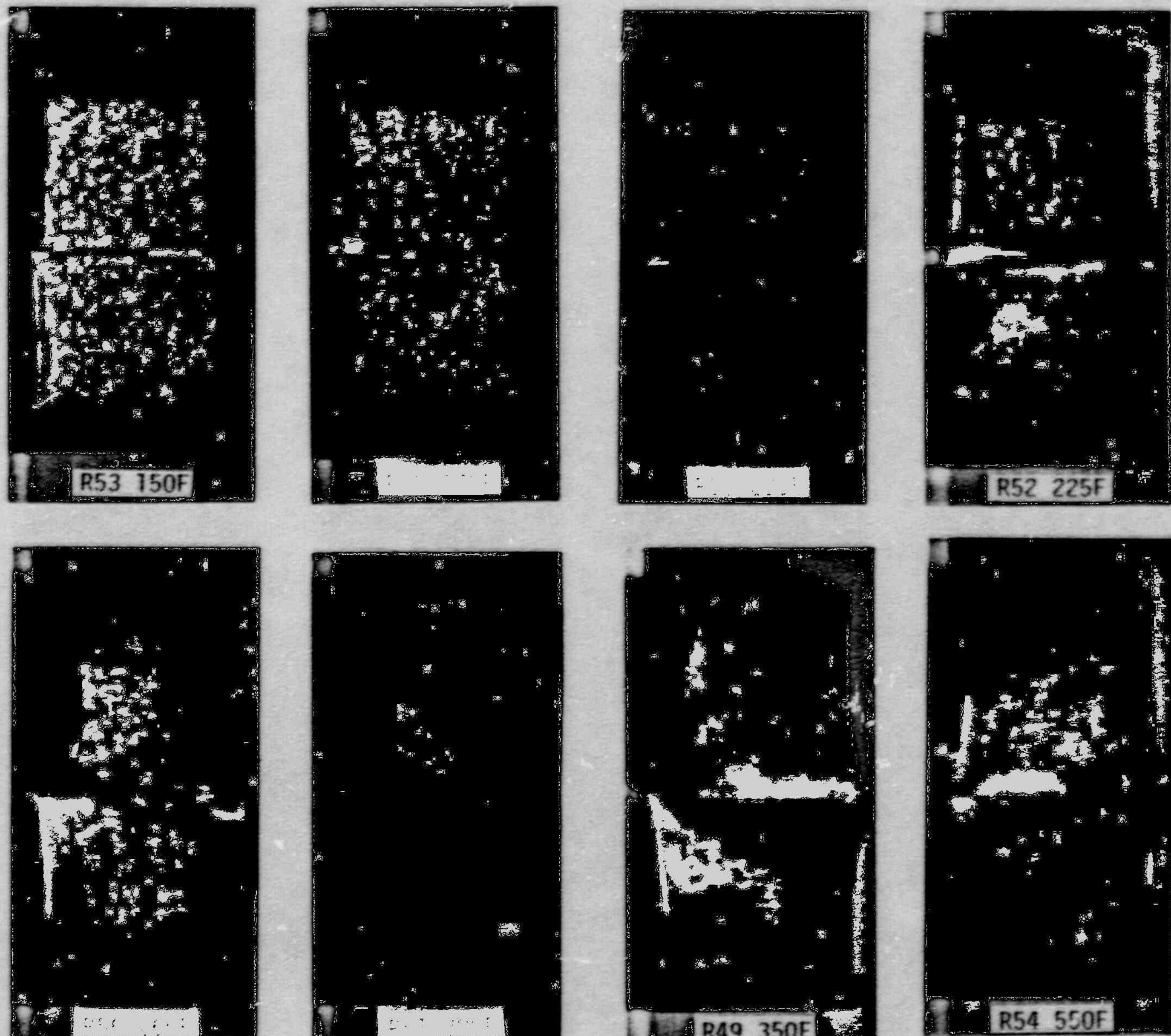
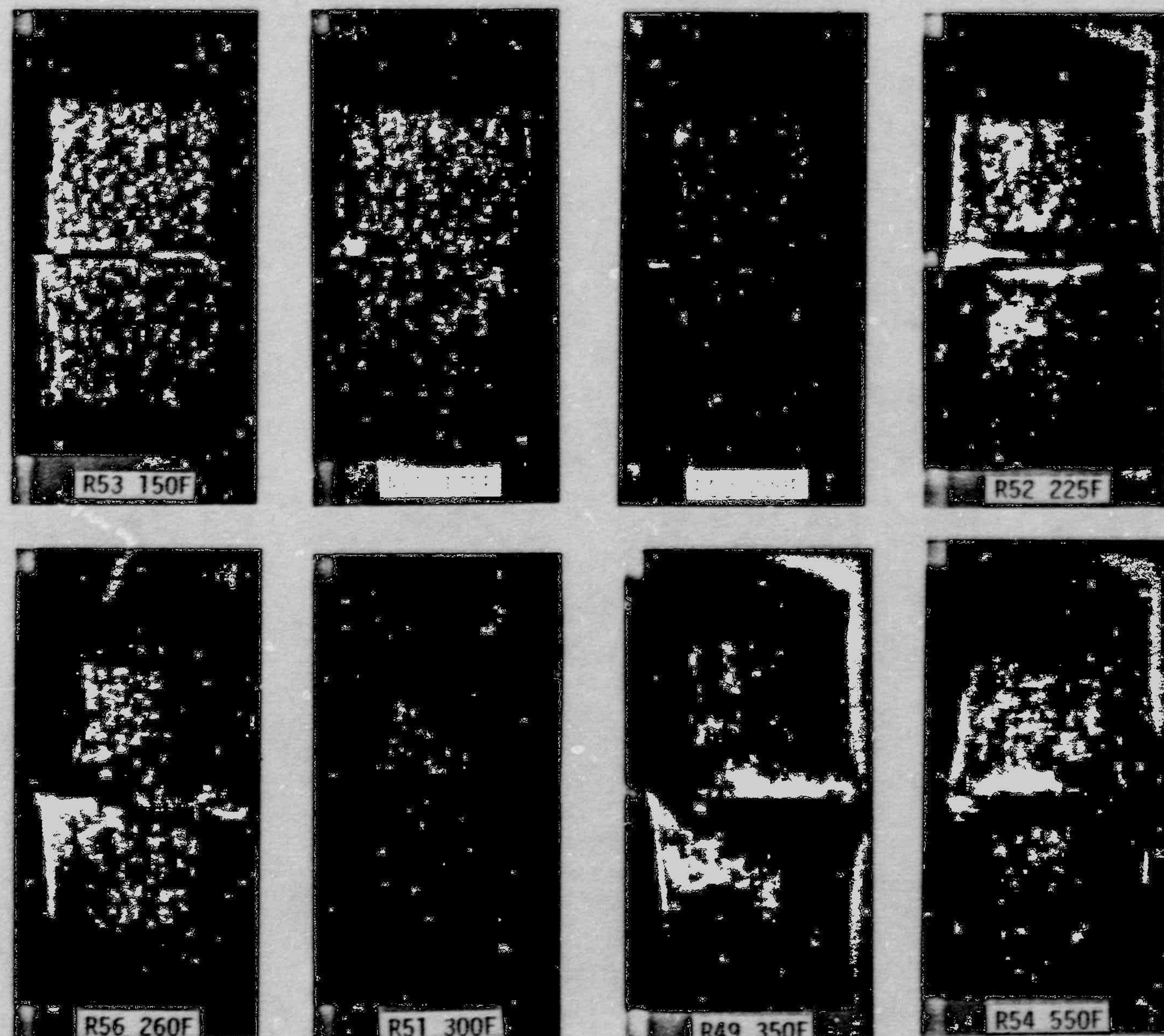


Figure 5-12. Photographs of Charpy Impact Specimen Fracture Surfaces - Correlation Monitor Material, HSST PL-02



6. NEUTRON FLUENCE

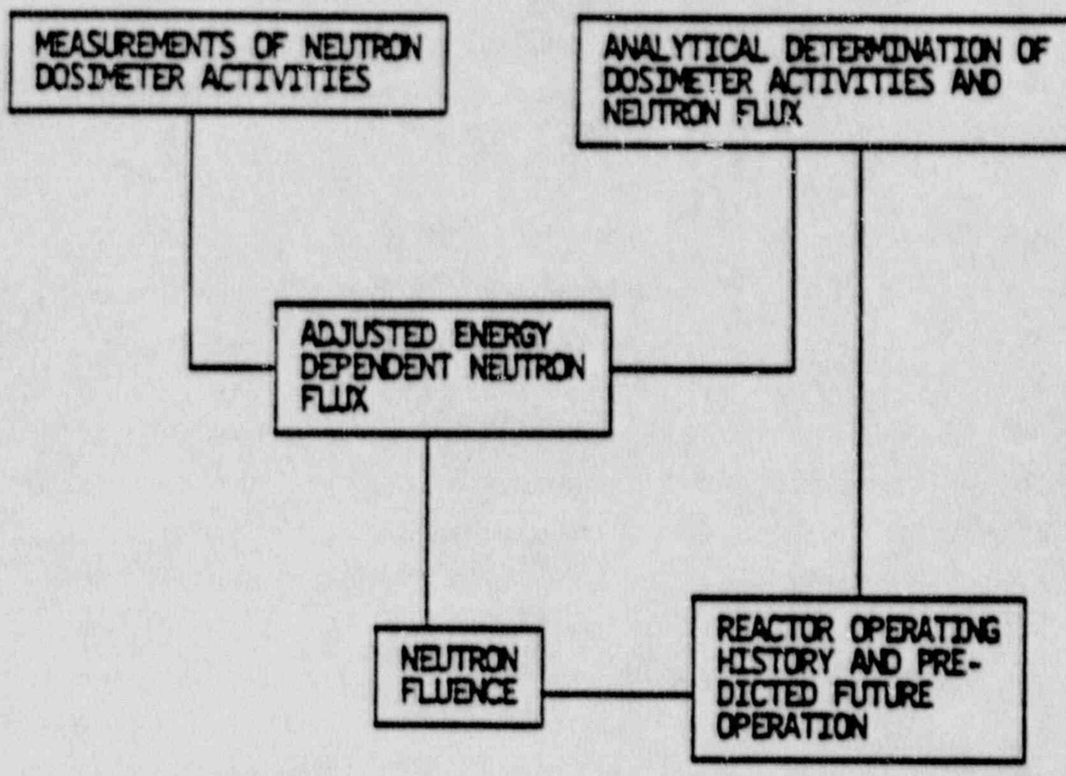
6.1. Introduction

The neutron fluence (time integral of flux) is a quantitative way of expressing the cumulative exposure of a material to a pervading neutron flux over a specific period of time. Fast neutron fluence, defined as the fluence of neutrons having energies greater than 1 MeV, is the parameter that is presently used to correlate radiation induced changes in material properties. Accordingly, the fast fluence must be determined at two locations: (1) in the test specimens located in the surveillance capsule, and (2) in the wall of the reactor vessel. The former is used in developing the correlation between fast fluence and changes in the material properties of specimens, and the latter is used to ascertain the point of maximum fluence in the reactor vessel, the relative radial and azimuthal distribution of the fluence, the fluence gradient through the reactor vessel wall, and the corresponding material properties.

The accurate determination of neutron flux is best accomplished through the simultaneous consideration of neutron dosimeter measurements and analytically derived flux spectra. Dosimeter measurements alone cannot be used to predict the fast fluence in the vessel wall or in the test specimens because (1) they cannot measure the fluence at the points of interest, and (2) they provide only rudimentary information about the neutron energy spectrum. Conversely, reliance on calculations alone to predict fast fluence is not prudent because of the length and complexity of the analytical procedures involved. In short, measurements and calculations are necessary complements of each other and together they provide assurance of accurate results.

Therefore, the determination of the fluence is accomplished using a combined analytical-empirical methodology which is outlined in Figure 6-1 and described in the following paragraphs. The details of the procedures and methods are presented in general terms in Appendix D and in BAW-1485P.[15]

Figure 6-1. General Fluence Determination Methodology



Analytical Determination of Dosimeter Activities and Neutron Flux

The analytical calculation of the space and energy dependent neutron flux in the test specimens and in the reactor vessel is performed with the two dimensional discrete ordinates transport code, DOT-IV.[16] The calculations employ an angular quadrature of 48 sectors (S8), a third order LeGendre polynomial scattering approximation (P3), the CASK23E cross section set[17] with 22 neutron energy groups and a fixed distributed source corresponding to the time weighted average power distribution for the applicable irradiation period.

In addition to the flux in the test specimens, the DOT-IV calculation determines the saturated specific activity of the various neutron dosimeters located in the surveillance capsule using the ENDF/B5 dosimeter reaction cross sections.[18] The saturated activity of each dosimeter is then adjusted by a factor which corrects for the fraction of saturation attained during the dosimeter's actual (finite) irradiation history. Additional corrections are normally made to account for the following effects:

- Photon induced fissions in U and Np dosimeters (without this correction the results underestimate the measured activity).
- Fissile impurities in U dosimeters (without this correction the results underestimate the measured activity).
- Short half-life of isotopes produced in iron and nickel dosimeters (303-day Mn-54 and 71-day Co-58, respectively). (Without this correction, the results could be biased high or low depending on the long term versus short term power histories.)
- Pu-239 generated in the U-238 dosimeter. Subsequent fissions in the Pu would generate Cs-137 activity that was not a result of U-238 fast fission, but which would be present in the dosimeter. (Without this correction the results underestimate the measured activity.)

Measurement of Neutron Dosimeter Activities

The neutron dosimeters located in the surveillance capsules are listed in Table 6-1 along with their characteristic reactions. Both activation type and fission type dosimeters are listed. The ratio of measured dosimeter activity to calculated dosimeter activity (M/C) is determined for each dosimeter, as discussed in Appendix D. These M/C ratios are evaluated on a case-by-case basis to assess the dependability or veracity of each individual dosimeter response. After carefully evaluating all factors known to affect the calculations or the measurements, an average M/C ratio is calculated and defined as the "normalization factor." The normalization factor is applied as an adjustment factor to the DOT-calculated flux at all points of interest.

Neutron Fluence

The determination of the neutron fluence from the time averaged flux requires only a simple multiplication by the time in EFPS (effective full-power seconds) over which the flux was averaged, i.e.

$$f_{ij}(\Delta T) = \sum_g \phi_{ijg} \Delta T$$

where

$f_{ij}(\Delta T)$ = Fluence at (i,j) accumulated over time ΔT (n/cm^2),

g = Energy group index,

ϕ_{ijg} = Time-average flux at (i,j) in energy group g, (n/cm^2-sec),

ΔT = Irradiation time, EFPS.

Neutron fluence was calculated in this analysis for the following components over the indicated operating time:

Test Specimens: Capsule irradiation time in EFPS

Reactor Vessel: Vessel irradiation time in EFPS

Reactor Vessel: Maximum point on inside surface extrapolated to 32 effective full power years

The neutron exposure to the reactor vessel and the material surveillance specimens was also determined in terms of the iron atom displacements per atom of iron (DPA). The iron DPA is an exposure index giving the fraction of iron atoms in an iron specimen which would be displaced during an irradiation. It is considered to be an appropriate damage exposure index since displacements of atoms from their normal lattice sites is a primary source of neutron radiation damage. DPA was calculated based on the ASTM Standard E693-79 (reapproved 1985).[9] A DPA cross section for iron is given in the ASTM Standard in 641 energy groups. DPA per second is determined by multiplying the cross section at a given energy by the neutron flux at that energy and integrating over energy. DPA is then the integral of DPA per second over the time of the irradiation. For the DPA values reported herein, the ASTM DPA cross sections were first collapsed to the 22 neutron group structure of CASK-23E; the DPA was then determined by summing the group flux times the DPA cross section over the 22 energy groups and multiplying by the time of the irradiation.

6.2. Vessel Fluence

The maximum fluence (E > 1.0 MeV) exposure of the reactor vessel during Zion 1 Cycles 1-10, was determined to be 5.22×10^{18} n/cm^2 based on a maximum neutron flux of 1.93×10^{10} n/cm^2-s (Tables 6-2 and 6-3). The maximum fluence occurs at the cladding/vessel interface at an azimuthal location of 45 degrees from a major horizontal axis of the core. The above value also includes an axial peaking factor at the reactor vessel of 1.21.

The end-of-life reactor vessel fluence value was calculated by taking the calculated cumulative fluence for cycles 1-10 and linearly extrapolating to 32 EFPY. This extrapolation was done using a flux of 1.46×10^{10} n/cm^2-sec for cycles beyond cycle 10. The flux was calculated as follows: (1) the cycle 7 vessel flux from a Westinghouse report WCAP-10962[20] was assumed to be constant for cycles 7 and beyond, (2) a time-weighted average flux was then calculated for cycles 1-10, (3) a ratio of the average flux beyond cycle 10 to the average flux of cycles 1-10 was taken, and finally (4) this factor was applied to the average flux obtained in this analysis to give us the average flux value used in the extrapolation. At 32 EFPY, the cumulative fluence of the reactor vessel inside surface was determined to be 1.60E+19 n/cm^2.

Relative fluence and displacements per atom (DPA) as a function of radial location in the reactor vessel wall is shown in Figure 6-2. Reactor vessel neutron fluence lead factors, which are the ratio of the neutron flux at the clad interface to that in the vessel wall at the T/4, T/2 and 3T/4 locations, are 1.8, 3.8, and 8.5, respectively. DPA lead factors at the same locations are 1.5, 2.6, and 4.7, respectively. The relative fluence as a function of azimuthal angle is shown in Figure 6-3. The peak occurs in the fast flux (E > 1.0 MeV) at 45° degrees with a corresponding value of 1.93×10^{10} n/cm^2-s.

6.3. Capsule Fluence

Capsule Y was irradiated in Zion for Cycles 1-10 (3131.9 EFPD) at a location 40 degrees off a major horizontal axis. The cumulative fast fluence of the surveillance capsule was calculated to be 1.56×10^{19} n/cm^2. This fluence value represents an average value for the center location of the Charpy specimens in the capsule. It includes an axial peaking factor in the capsule of 1.19 and a normalization factor of 0.967. The fluence is approximately 11% higher at the center of the Charpy specimens closest to the core and approximately 11% lower at the center of the Charpy specimens away from the core.

6.4. Fluence Uncertainties

Uncertainties were estimated for the fluence values reported herein. The results are shown in Table 6-5 and are based on comparisons to benchmark experiments, when available; estimated and measured variations in input data;

and on engineering judgement. The values in Table 6-5 represent best estimate values based on past experience with reactor vessel fluence analyses.

Table 6-1. Surveillance Capsule Dosimeters

Dosimeter Reactions[a]	Lower Energy Limit for Reaction, MeV	Isotope Half-Life
$^{54}Fe(n,p)^{54}Mn$	2.5	312.5 days
$^{58}Ni(n,p)^{58}Co$	2.3	70.85 days
$^{238}U(n,f)^{137}Cs$	1.1	30.03 years
$^{237}Np(n,f)^{137}Cs$	0.5	30.03 years
$^{63}Cu(n,\alpha)^{60}Co$	6.0	5.27 years

[a]Reaction activities measured for capsule flux evaluation.

Table 6-2. Zion Unit 1 Reactor Vessel Fast Flux

Cycle	Fast Flux (E > 1 MeV), n/cm^2-s: Inside Surface (Max Location)	T/4	T/2	3T/4	Flux n/cm^2-s (E > 0.1 MeV) Inside Surface (Max Location)
Cycles 1-10, 3131.9 EFPD	1.93E+10*	1.07E+10*	5.08E+9*	2.27E+9*	4.86E+10*
15 EFPY	1.46E+10***	8.11E+9**	3.84E+9**	1.72E+9**	---
24 EFPY	1.46E+10***	8.11E+9**	3.84E+9**	1.72E+9**	---
32 EFPY	1.46E+10***	8.11E+9**	3.84E+9**	1.72E+9**	---

*Includes axial peaking factor of 1.21.

**Divide flux at inside surface by the appropriate lead factors on p. 6-5 to obtain these T/4, T/2 and 3T/4 fast flux values.

***Assumes future fuel designs are similar to low leakage cycles 7-10.

Table 6-3. Calculated Zion Unit 1 Reactor Vessel Fluence

Cumulative Irradiation Time	Fast Fluence, n/cm^2 (E > 1 MeV) Inside Surface (Max Location)	T/4	T/2	3T/4
End of Cycle 10 (8.57 EFPY)	5.22E+18	2.90E+18	1.37E+18	6.14E+17
15 EFPY	8.17E+18***	4.54E+18*	2.15E+18*	9.61E+17*
24 EFPY	1.23E+19***	6.83E+18*	3.24E+18*	1.45E+18*
32 EFPY	1.60E+19***	8.89E+18*	4.21E+18*	1.88E+18*
*Calculated using these lead factors	1.0	1.8	3.8	8.5
Conversion Factors				
Fluence (E > 1 MeV) to DPA	1.50E-21**	1.85E-21**	2.20E-21**	2.70E-21**

**Multiply fast fluence values (E > 1 MeV) in units of n/cm^2 by these factors to obtain the corresponding DPA values.

***Assumes future fuel designs are similar to low leakage cycles 7-10.

Table 6-4. Zion Unit 1 Surveillance Capsule Y Fluence, Flux, and DPA

Irradiation Time	Flux (E > 1 MeV), n/cm^2-s	Fluence, n/cm^2	DPA
Cycles 1-10 (3131.9 EFPD)	5.78E+10	1.56E+19	2.41E-2

Table 6-5. Estimated Fluence Uncertainty

Calculated Fluence	Estimated Uncertainty	Basis of Estimate
In the capsule	± 15%	Activity measurements, cross section fission yields, saturation factor, deviation from average fluence value
In the reactor vessel at maximum location	± 21%	Activity measurements, cross sections, fission yields, factors, axial factor, capsule location, radial/azimuthal extrapolation, normalization factor
In the reactor vessel at the maximum location for end-of-life extrapolation	± 23%	Factors in vessel fluence above plus uncertainties for extrapolation to 32 EFPY

Figure 6-2. Fast Flux, Fluence and DPA Distribution Through Reactor Vessel Wall

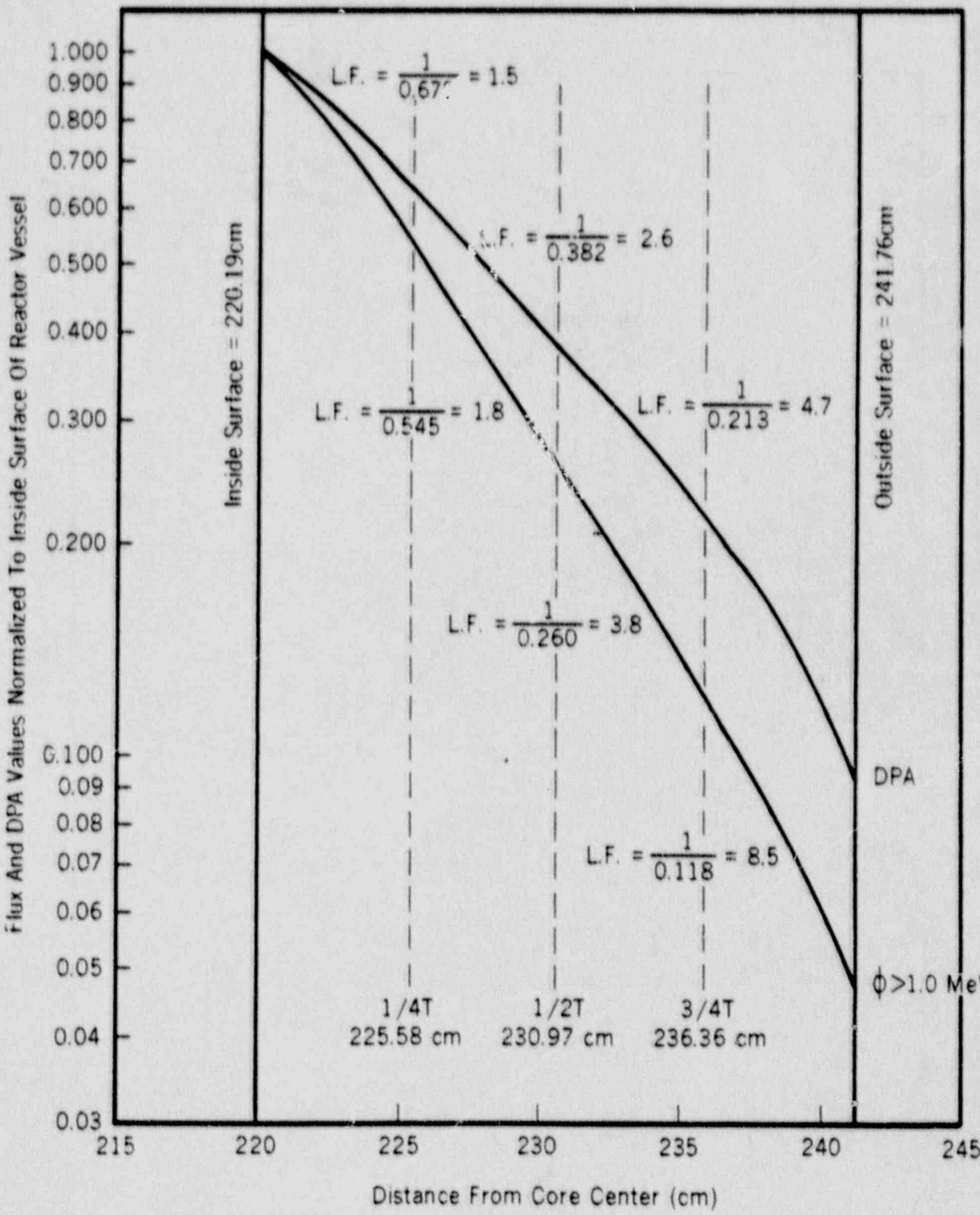


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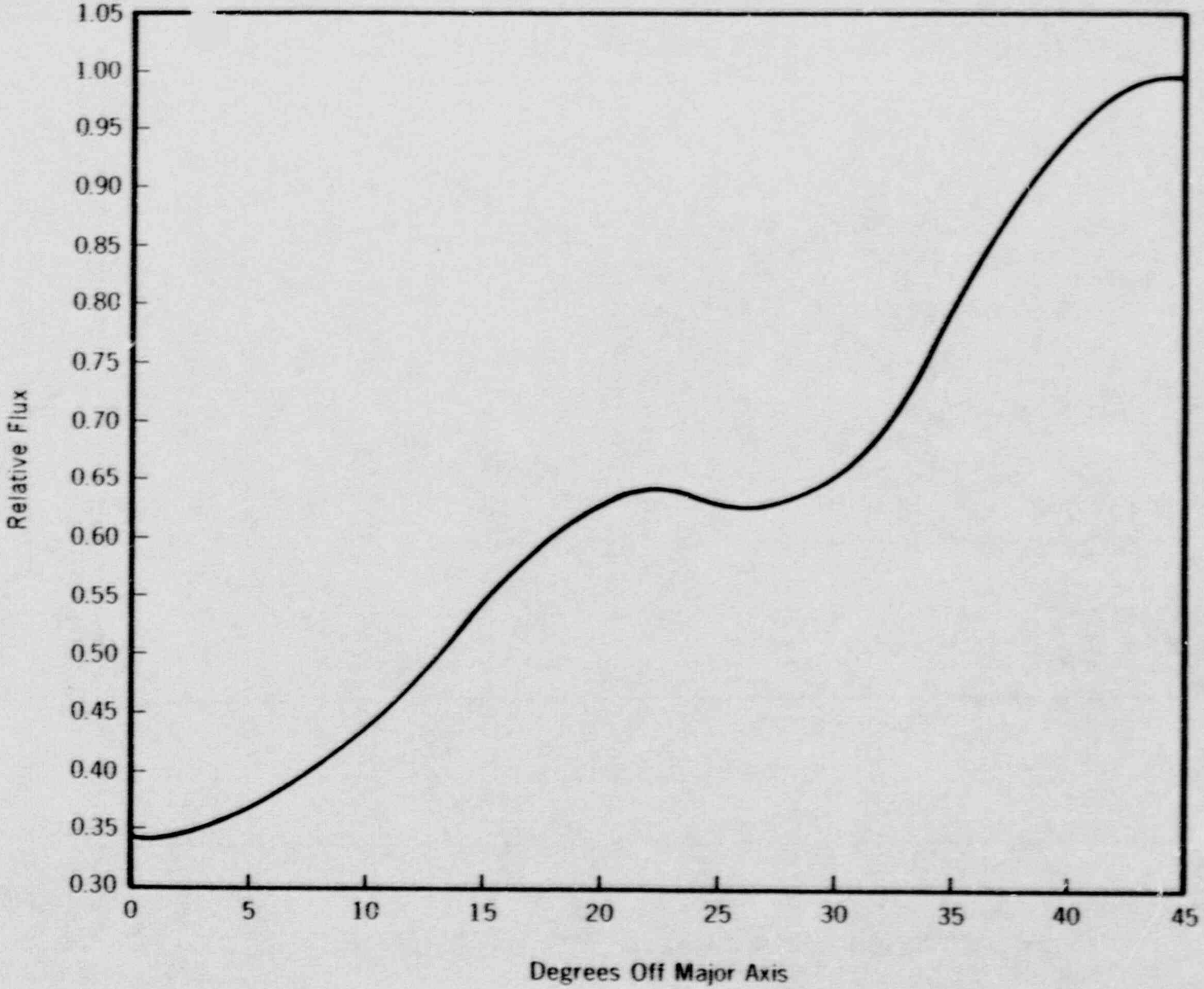
Figure 6-3. Azimuthal Flux and Fluence Distributions at Reactor Vessel Wall Inside Surface



7. DISCUSSION OF CAPSULE RESULTS

7.1. Pre-Irradiation Property Data

The weld metal and base metals were selected for inclusion in the surveillance program in accordance with the criteria in effect at the time the program was designed for Zion Unit 1. The applicable selection criterion was based on the unirradiated properties only. A review of the original unirradiated properties of the reactor vessel core beltline region materials indicated no significant deviation from expected properties except in the case of the upper shelf properties of the weld metal which was below the current required 75 ft-lbs. Based on the design end-of-service peak neutron fluence value at the 1/4T vessel wall location and the copper content of the weld metal, it was predicted that the end-of-service Charpy upper-shelf energy (USE) will be below 50 ft-lb.

7.2. Irradiated Property Data

7.2.1. Tensile Properties

Table 7-1 compares irradiated properties from Capsule Y with the unirradiated tensile properties. At both room temperature and elevated temperature, the ultimate and yield strength changes in the base metal as a result of irradiation and the corresponding changes in ductility are within the limits observed for similar materials. There is some strengthening, as indicated by increases in ultimate and yield strengths and decreases in ductility properties. All changes observed in the base metal are such as to be considered within acceptable limits. The changes, at both room temperature and 550-600F, in the properties of the base metal are not as large as those observed for the weld metal, indicating a lesser sensitivity of the base metal to irradiation damage. In either case, the changes in tensile properties are insignificant relative to the analysis of the reactor vessel materials at this time period in the reactor vessel service life.

A comparison of the tensile data from previously evaluated capsules (Capsules T, U and X) with the corresponding data from the capsule reported in this report is shown in Table 7-2. The currently reported capsule experienced a fluence that is approximately six times greater than the first capsule.

The general behavior of the tensile properties as a function of neutron irradiation is an increase in both ultimate and yield strength and a decrease in ductility as measured by both total elongation and reduction of area. The most significant observation from these data is that the weld metal exhibited greater sensitivity to neutron radiation than the base metal.

7.2.2. Impact Properties

The behavior of the Charpy V-notch impact data is more significant to the calculation of the reactor system's operating limitations. Table 7-3 compares the observed changes in irradiated Charpy impact properties with the predicted changes.

The 30 ft-lb transition temperature shift for the base metal is in relatively good agreement with the value predicted using Regulatory Guide 1.99, Rev. 2[21] and the predicted value is conservative. It would be expected that these values would exhibit good agreement when it is considered that the data used to develop Regulatory Guide 1.99, Rev. 2, was taken at the 30 ft-lb temperature.

The transition temperature measurements at 30 ft-lbs for the weld metal is in good agreement with the predicted shift using Regulatory Guide 1.99, Revision 2 and the predicted value is also conservative. The shift being in good agreement with the predicted value which indicates that the estimating technique based on the Regulatory Guide 1.99, Rev. 2, are conservative for predicting the 30 ft-lb transition temperature shifts since the method requires that a margin be added to the calculated value to provide a conservative value.

The data for the decrease in Charpy USE with irradiation showed good agreement with predicted values for the base metal in both the longitudinal and transverse directions. The weld metal decrease in Charpy USE was over predicted. However, the poor comparison of the measured weld metal data with the predicted value is to be expected in view of the lack of data for medium-, or high-copper-content materials at medium fluence values that were used to develop the estimating curves.

A comparison of the Charpy impact data from the previously evaluated capsules from Zion Unit-1 with the corresponding data from the capsule reported in this report is shown in Table 7-4. The currently reported data experienced a fluence that is six times greater than the first capsule.

The base metal exhibited transition temperature shifts at the 30 ft-lb levels for the latest capsule that were similar in magnitude to those of the previous capsule. The corresponding data for the weld metal also showed a further increase at the 30 ft-lb level as compared to the previously reported increase at the 30 ft-lb level. This may be related, in part, to a further decrease in the upper-shelf energy and the change in the slope of the Charpy curve in the transition region.

Both the base metal and the weld metal exhibited decreases in the upper-shelf values similar to the previous capsules. The weld metal in this capsule exhibited a decrease similar to the weld metal in the previous capsule. These data confirm that the upper-shelf drop for this weld metal may have reached a stabilized condition, or "saturation" as observed in the results of capsules evaluated by others.[22] This behavior of Charpy USE drop for this weld metal should not be considered indicative of a similar behavior of upper-shelf region fracture toughness properties. This behavior indicates that other reactions may be taking place within the material besides simple neutron damage. Verification of this relationship must await the testing and evaluation of the data from compact fracture toughness test specimens.

Results from other surveillance capsules also indicate that RT_{NDT} estimating curves have greater inaccuracies than originally thought. These inaccuracies are a function of a number of parameters related to the basic data available at the time the estimating curves are established. These parameters may include inaccurate fluence values, inaccurate chemical composition values, and variations in data interpretation. The change in the regulations requiring the shift measurement to be based on the 30 ft-lb value has minimized the errors that resulted from using the 30 ft-lb data base to predict the shift behavior at 50 ft-lbs.

The design curves for predicting the shift will continue to be modified as more data become available; until that time, the design curves for predicting the RT_{NDT} shift as given in Regulatory Guide 1.99, Revision 2, are considered adequate for predicting the RT_{NDT} shift of those materials for which data are not available. These curves will be used to establish the pressure-temperature operational limitations for the irradiated portions of the reactor vessel until the time that improved prediction curves are developed and approved.

The relatively good agreement of the change in Charpy upper-shelf energy is in support of the accuracy of the prediction curves for medium copper content materials. However, for high copper content materials such as weld metal the predicted values may be too conservative. Although the prediction curves are conservative in that they generally predict a larger decrease in upper-shelf energy than is observed for a given fluence and copper content, the conservatism can unduly restrict the operational limitations. These data support the contention that the upper-shelf energy drop curves will have to be revised as more reliable data become available; until that time the design curves used to predict the decrease in upper-shelf energy of the controlling materials are considered conservative.

7.3. Reactor Vessel Fracture Toughness

An evaluation of the reactor vessel end-of-life fracture toughness and the pressurized thermal shock criterion was made and the results are presented in Table 7-5.

The fracture toughness evaluation shows that the controlling weld metal may have a T/4 wall location end-of-life RT_{NDT} of 265F based on Regulatory Guide 1.99, Revision 2, including a margin of 56F. This predicted shift is excessive since data from an Integrated Reactor Vessel Surveillance Program surveillance capsules exhibit measured RT_{NDT} significantly less for comparable fluence values. It is estimated that the end-of-life RT_{NDT} shift will be significantly less than the value predicted using Regulatory Guide 1.99, Revision 2. This reduced shift will permit the calculation of less restrictive pressure temperature operating limitations than if Regulatory Guide 1.99, Revision 2, was used.

An evaluation of the reactor vessel end-of-life upper-shelf energy for each of the materials used in the fabrication was made and the results are presented in Table 7-6. This evaluation was made because the weld metals used to fabricate the reactor vessel are characterized by relatively low-upper-shelf-energy and high copper contents; and, consequently, are expected to be sensitive to neutron radiation damage. Two methods were used to evaluate the radiation induced decrease in upper-shelf energy; the method of Regulatory Guide 1.99, Revision 2, which is the same procedure used in Revision 1, and the method presented in BAW-1803[19] which was developed specifically to address the need for an estimating method for this class of weld metals (Automatic Submerged-Arc: Mn-Mo-Ni Wire/Linde 80 Flux).

The method of Regulatory Guide 1.99, Revision 2, shows that all of the weld metals used in the fabrication of the beltline region of the reactor vessel will have an upper-shelf energy below 50 ft-lbs prior to the 32 EFPY design life based on the T/4 wall location. Regulatory Guide 1.99 method predicts a decrease below 50 ft-lbs for the controlling weld metal at the vessel inside wall. However, based on surveillance data and the prediction techniques presented in BAW-1803, it is calculated that none of the reactor vessel material upper-shelf energies will decrease to below 50 ft-lbs during the vessel design life.

7.4. Neutron Fluence Analysis

The neutron fluence analysis shows a sharp reduction in the neutron flux as the result of improved fuel management schemes to lower core leakage. These new analysis calculated an end-of-life fluence value of 1.60×10^{19} n/cm^2 (E > 1 MeV) at the reactor vessel inside surface peak location. The corresponding value for the vessel wall T/4 location is calculated to be 8.89×10^{18} n/cm^2 (E > 1 MeV). These values represent a 9 percent reduction compared to the values calculated for the pressurized thermal shock analysis.[20]

Table 7-1. Comparison of Zion Unit 1, Capsule Y Tension Test Results

	Room Temp Test		Elevated Temp Test	
	Unirr	Irrad	Unirr*	Irrad**
Base Metal -- B7835-1, Longitudinal				
Fluence, 10^{19} n/cm^2 (E > 1 MeV)	0	1.56	0	1.56
Ultimate tensile strength, ksi	83.8	98.0	80.9	91.5
0.2% yield strength, ksi	63.0	77.3	54.3	67.7
Uniform elongation, %	15.3	10.9	14.0	8.3
Total elongation, %	27.6	24.1	25.0	19.5
Reduction of area, %	71.5	65.9	63.1	55.4
Weld Metal -- WF-209-1				
Fluence, 10^{19} n/cm^2 (E > 1 MeV)	0	1.56	0	1.56
Ultimate tensile strength, ksi	89.4	108.4	87.7	103.9
0.2% yield strength, ksi	72.7	93.1	66.5	84.7
Uniform elongation, %	15.0	11.1	13.3	8.7
Total elongation, %	25.8	22.8	21.0	18.6
Reduction of area, %	62.7	57.0	50.3	51.6

*Test temperature is 600F.

**Test temperature is 550F.

Table 7-2. Summary of Zion Unit 1 Reactor Vessel Surveillance Capsules Tensile Test Results

Material	Cap. I.D.	Fluence, 10^{18} n/cm^2	Test Temp, F	Strength, ksi: Ultimate	% (a)	Yield	% (a)	Ductility, %: Total Elon.	% (a)	Reduction of Area	% (a)
Base metal Longitudinal (B7835-1)	-	00.	RT	83.8	--	63.0	--	28	--	72	--
			300	76.3	--	56.0	--	24	--	70	--
			600	80.9	--	54.3	--	25	--	63	--
	T	02.5	77	90.5	+ 8.0	68.4	+ 8.6	28	0.0	68	- 5.6
			550(c)	85.4	+ 5.6	64.9	+19.5	23	- 8.0	62	- 1.6
	U	08.5	250(b)	88.0	+15.3	69.3	+23.8	22	- 8.3	64	- 8.6
			580(c)	89.6	+10.8	63.1	+16.2	20	-20.0	62	- 1.6
	Y	15.6	70	98.0	+16.9	77.3	+22.7	24	-14.3	66	- 8.3
			550(c)	91.5	+13.1	67.7	+24.7	20	-20.0	55	-12.7
Weld metal (WF-209-1)	-	00.	RT	89.4	--	72.7	--	26	--	63	--
			300	83.0	--	67.1	--	23	--	64	--
			600	87.7	--	66.5	--	21	--	50	--
	T	02.5	77	103.5	+15.8	86.2	+18.6	24	- 7.7	58	- 7.9
			570(c)	97.1	+10.7	79.9	+20.2	19	- 9.5	52	+ 4.0
	U	08.5	275(b)	98.8	+19.0	83.5	+24.4	21	- 8.7	56	-12.5
			580(c)	101.8	+16.1	82.5	+24.1	17	-19.0	42	-16.0
	X	12.6	300	102.3	+23.3	84.8	+26.4	20	-13.0	54	-15.6
			550(c)	101.6	+15.8	83.4	+25.4	17	-19.0	54	+ 8.0
	Y	15.6	70	108.4	+21.3	93.1	+28.1	23	-11.5	57	- 9.5
			550(c)	103.9	+18.5	84.7	+27.4	19	- 9.5	52	+ 4.0

(a) Change relative to unirradiated.

(b) Compared to 300F unirradiated data.

(c) Compared to 600F unirradiated data.

Table 7-3. Observed Vs. Predicted Changes for Capsule Y Irradiated Charpy Impact Properties - 1.56 x 10^{19} n/cm^2 (E > 1 MeV)

Material	Observed Unirrad.	Observed Irrad.	Observed Diff.	Predicted - RG 1.99/2[b] Without Margin[a]	Predicted - RG 1.99/2[b] With Margin[b]
Increase in 30 ft-lb Trans. Temp., F					
Base Material (B7835-1)					
Longitudinal	- 5	87	92	82	116
Transverse	25	119	94	82	116
Heat-Affected Zone (B7835-1)	-100	-32	68	82	116
Weld Metal	4	209	205	234	290
Correlation Material (HSST PL-02)	42	171	129	115	149
Decrease in Charpy USE, ft-lb					
Base Material (B7835-1)					
Longitudinal	140	114	26	32	N.A.
Transverse	117	96	21	26	N.A.
Heat-Affected Zone (B7835-1)	150	116	34	34	N.A.
Weld Metal	64	44	20	29	N.A.
Correlation Material (HSST PL-02)	124	98	26	31	N.A.

[a] Mean value per Regulatory Guide 1.99, Revision 2, May 1988.

[b] Mean value per Regulatory Guide 1.99, Revision 2, May 1988, plus margin.

N.A. - Not applicable.

Table 7-4. Summary of Zion Unit 1 Reactor Vessel Surveillance Capsules Charpy Impact Test Results

Material	Fluence, 10^{18} n/cm^2	Transition Temperature Increase, F: 30 ft-lb Observed	Predicted 30 ft-lb[a]: W/O Margin	Predicted 30 ft-lb[a]: W/Margin	Upper-Shelf Energy, ft-lb: Observed USE	Observed ΔUSE	Predicted[a] USE	Predicted[a] ΔUSE
Base material (B7835-1)								
Longitudinal	2.5[b]	60	46	80	132	8	119	21
	8.5[b]	85	70	104	120	20	112	28
	12.6[b]	90	78	112	107	33	109	31
	15.6	92	82	116	114	26	108	32
Transverse	2.5[b]	25	46	80	114	3	99	18
	8.5[b]	60	70	104	102	15	94	23
	12.6[b]	80	78	112	93	24	91	26
	15.6	94	82	116	96	21	90	26
Heat-affected zone	2.5[b]	57	46	80	135	15	127	23
	8.5[b]	40	70	104	120	30	120	30
	12.6[b]	70	78	112	125	25	117	33
	15.6	68	82	116	116	34	115	34
Weld metal	2.5[b]	112	130	186	56	8	42	22
	8.5[b]	199	199	255	52	12	38	26
	12.6[b]	199	221	277	44	20	36	28
	15.6	205	234	290	44	20	35	29
Correlation material (HSST plate 02)	2.5[b]	66	64	98	106	18	104	20
	8.5[b]	130	97	131	88	36	97	27
	12.6[b]	140	109	143	84	40	94	30
	15.6	129	115	149	98	26	93	31

[a] Per RG 1.99, Revision 2, May 1988.

[b] Prior capsule fluence values are the calculated values per WCAP-10926.[20]

Table 7-5. Evaluation of Reactor Vessel End-of-Life Fracture Toughness - Zion Unit 1

Material Description			Material Chemical Composition, w/o[c]		Estimated EOL Fluence		End-of-Life RT_{NDT}, F[a]		
Reactor Vessel Beltline Region Location	Heat Number	Type	Copper	Nickel	Inside Surface n/cm^2	1/4 Wall Location n/cm^2	Initial RT_{NDT}, F	Inside Surface	1/4 Wall Location
Intermediate Shell	C3795-1	SA533, B1	0.12	0.49	1.60E+19	8.89E+18	+ 1[f]	154	141
Intermediate Shell	B7835-1	SA533, B1	0.12	0.49	1.60E+19	8.89E+18	+ 5[e]	131	117
Lower Shell	B7823-1	SA533, B1	0.13	0.48	1.38E+19	7.69E+18	- 4[e]	125	111
Lower Shell	C3799-1	SA533, B1	0.15	0.50	1.38E+19	7.69E+18	+20[e]	168	151
Upper Circum. Weld (I.D. 82%)	WF-154	ASA/Linde 80	0.31	0.59	1.04E+19	5.78E+18	- 6[d]	261	229
Upper Circum. Weld (O.D. 18%)	SA-1769	ASA/Linde 80	0.26	0.61	N.A.	N.A.	---	N.A.	N.A.
Middle Circum. Weld (100%)	WF-70	ASA/Linde 80	0.35	0.59	1.38E+19	7.69E+18	- 6[d]	292	257
Interm. Longit. Weld (100%)	WF-4	ASA/Linde 80	0.29	0.55	5.52E+18	3.07E+18	- 6[d]	216	187
Interm. Longit. Weld (I.D. 39%)	WF-8	ASA/Linde 80	0.29	0.55	5.52E+18	3.07E+18	- 6[d]	216	187
Interm. Longit. Weld (O.D. 61%)	WF-4	ASA/Linde 80	0.29	0.55	N.A.	N.A.	---	N.A.	N.A.
Lower Longit. Welds (Both 100%)	WF-8	ASA/Linde 80	0.29	0.55	4.77E+18	2.65E+18	- 6[d]	208	180
Middle Circum. Weld (100%)[b]	Atypical	ASA/Linde 80	0.41	0.10	1.38E+19	7.69E+18	+90[b]	253	233

(a) Per Regulatory Guide 1.99, Revision 2, May 1988.[21]

(b) Per Licensing Documents BAW-10144A, February 1980,[23] and BAW-1895, January 1986.[24]

(c) Materials chemical compositions per WCAP-10962, December 1985, and BAW-1799, July 1983.[25]

(d) Per BAW-1803, January 1984.[26]

(e) Per Commonwealth Edison submittal.[30]

(f) Estimated mean value per BAW-10046P, March 1976.[27]

NA - Not Applicable

Table 7-6. Evaluation of Reactor Vessel End-of-Life Upper Shelf Energy - Zion Unit 1

Material Description: Reactor Vessel Beltline Region Location	Heat Number	Type	Material Chemical Composition, w/o[d]: Copper	Nickel	Estimated EOL Fluence: Inside Surface n/cm^2	T/4 Wall Location n/cm^2	Initial Use[a] ft-lbs	Estimated EOL Use Per R.G. 1.99/2[a]: Inside Surface	T/4 Wall Location	Estimated EOL Use Per BAW-1803[b]: Inside Surface	T/4 Wall Location	Estimated EFPY to 50 ft-lbs at T/4: R.G. 1.99/2	BAW-1803
Intermediate Shell	C3795-1	SA533, B1	0.12	0.49	1.60E+19	8.89E+18	(9?)[f]	71	74	N.A.	N.A.	>32	N.A.
Intermediate Shell	B7835-1	SA533, B1	0.12	0.49	1.60E+19	8.89E+18	117[g]	90	94	N.A.	N.A.	>32	N.A.
Lower Shell	B7823-1	SA533, B1	0.13	0.48	1.38E+19	7.69E+18	(92)[f]	70	73	N.A.	N.A.	>32	N.A.
Lower Shell	C3799-1	SA533, B1	0.15	0.50	1.38E+19	7.69E+18	(92)[f]	68	71	N.A.	N.A.	>32	N.A.
Upper Circum. Weld (ID 82%)	WF-154	ASA/Linde 80	0.31	0.59	1.04E+19	5.78E+18	(70)[c]	40	43	53	53	?	>32
Upper Circum. Weld (OD 18%)	SA-1769	ASA/Linde 80	0.26	0.61	N.A.	N.A.	(70)[c]	N.A.	N.A.	N.A.	N.A.	N.A.	N.A.
Middle Circum. Weld (100%)	WF-70	ASA/Linde 80	0.35	0.59	1.38E+19	7.69E+18	(70)[c]	39	41	52	53	4	>32
Interm. Longit. Weld (100%)	WF-4	ASA/Linde 80	0.29	0.55	5.52E+18	3.07E+18	(70)[c]	44	47	55	56	18	>32
Interm. Longit. Weld (ID 39%)	WF-8	ASA/Linde 80	0.29	0.55	5.52E+18	3.07E+18	(70)[c]	44	47	55	56	18	>32
Interm. Longit. Weld (OD 61%)	WF-4	ASA/Linde 80	0.29	0.55	N.A.	N.A.	(70)[c]	N.A.	N.A.	N.A.	N.A.	N.A.	N.A.
Lower Longit. Welds (Both 100%)	WF-8	ASA/Linde 80	0.29	0.55	4.77E+18	2.65E+18	(70)[c]	45	48	56	56	21	>32
Middle Circum. Weld (100%)[e]	Atypical	ASA/Linde 80	0.41	0.10	1.38E+19	7.69E+18	79	44	47	N.A.	N.A.	2	N.A.

[a]Per Regulatory Guide 1.99, Revision 2, May 1988.[21]

[b]Per BAW-1803, January 1984.[26]

[c]Mean value per BAW-1803, January 1984.[26]

[d]Materials chemical compositions per WCAP-10962, December 1985,[20] and BAW-1799, July 1983.[25]

[e]Per Licensing Document BAW-10144, February 1980.[23]

[f]Estimated mean value per BAW-10046P, March 1976.[27]

[g]Per WCAP-8064.[1]

NA - Not Applicable

Figure 7-1. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Plate Material Longitudinal Orientation, Heat No. B7835-1

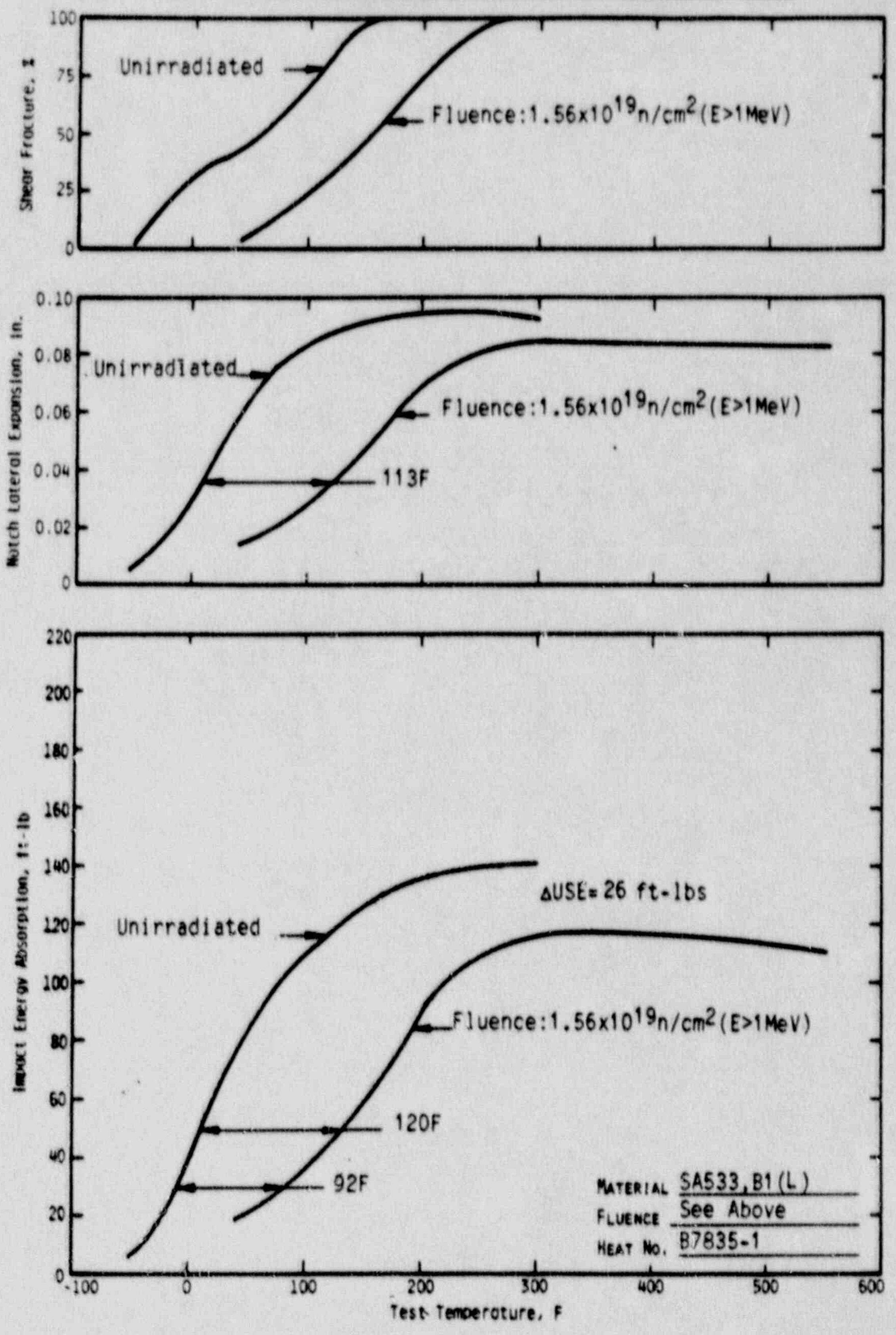


Figure 7-2. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Plate Material Transverse Orientation, Heat No. B7835-1

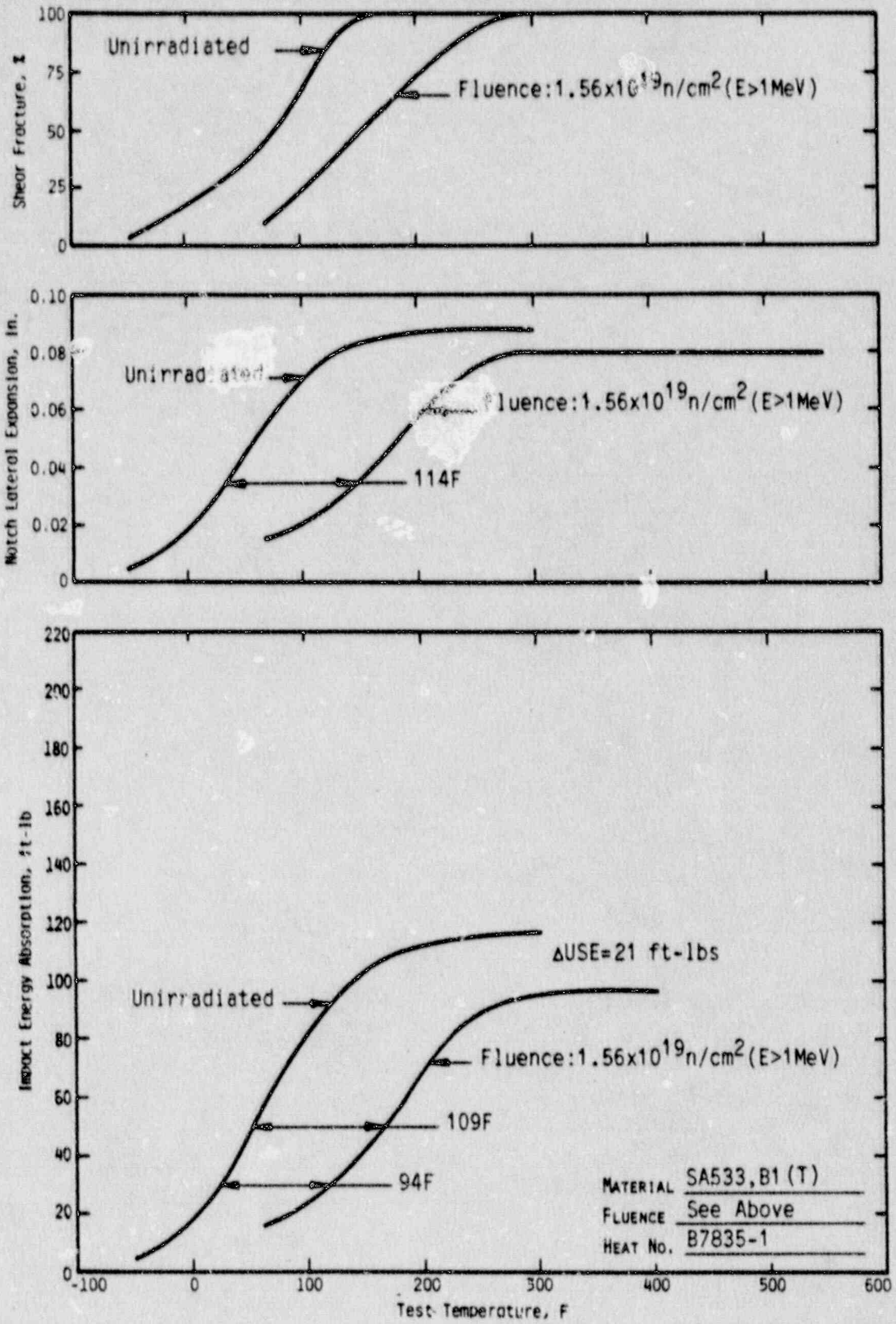


Figure 7-3. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Heat-Affected Zone, Heat No. B7835-1

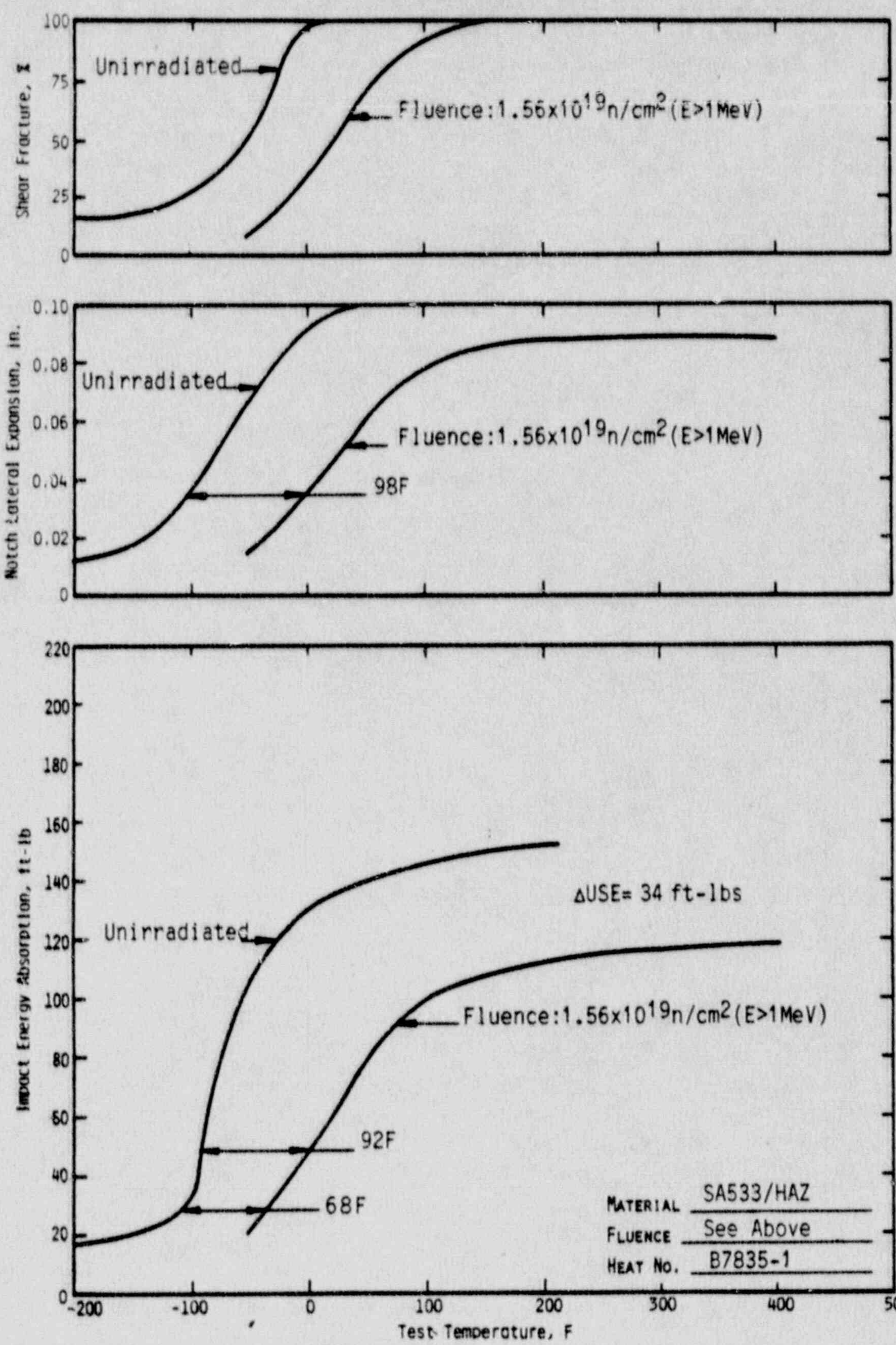


Figure 7-4. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Weld Metal WF-209-1

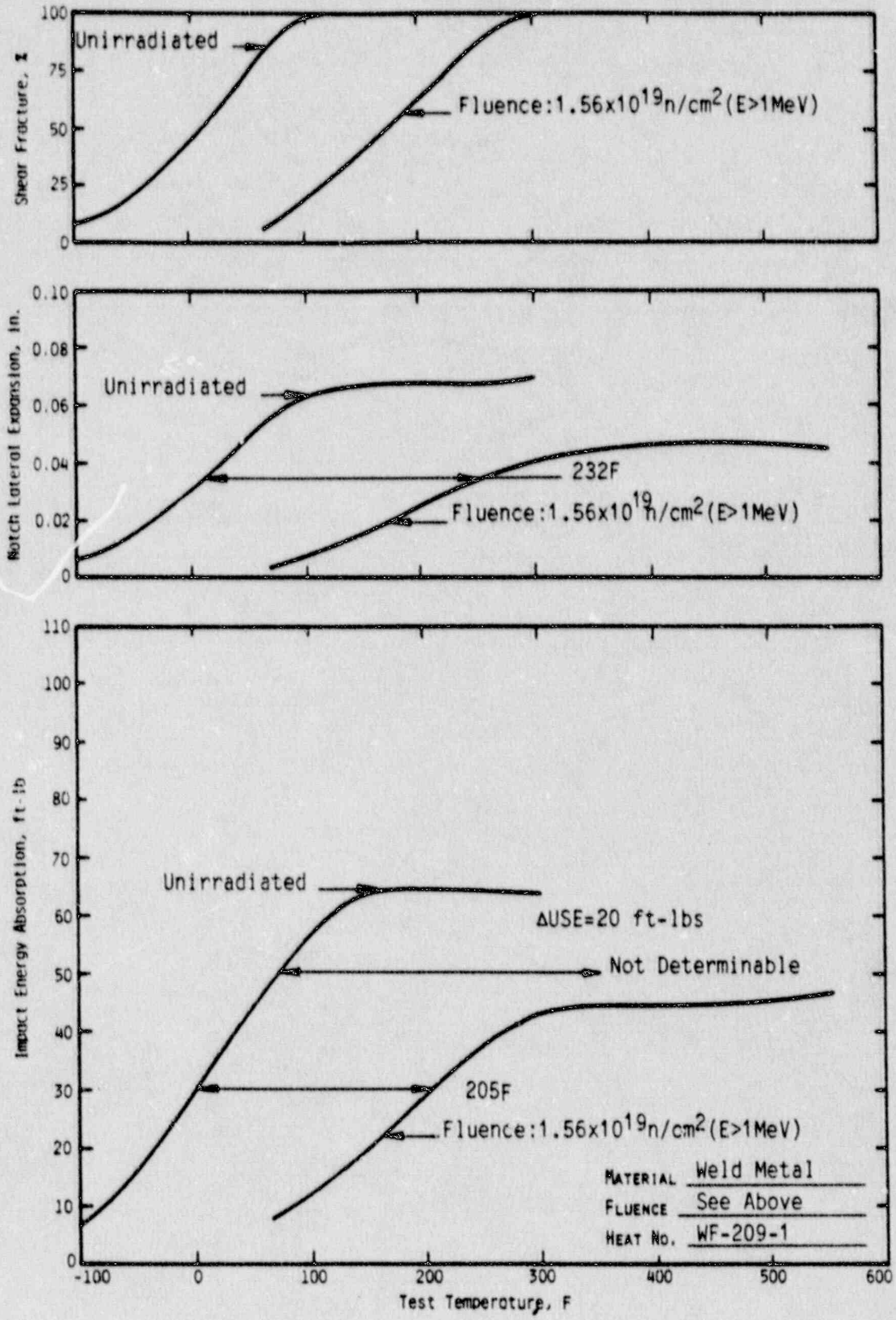
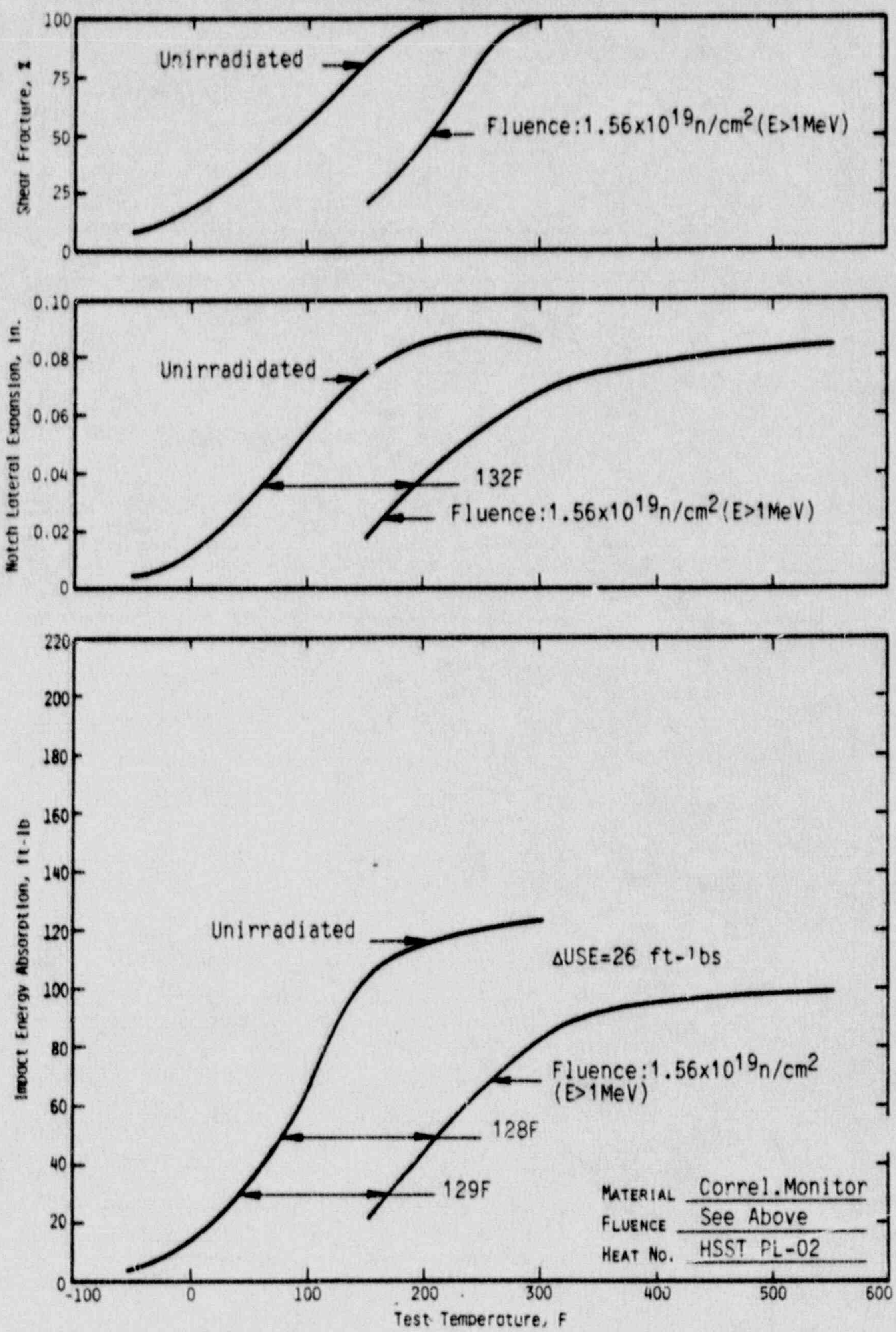


Figure 7-5. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Correlation Monitor Material, HSST PL-02, Heat No. A1195-1



8. SUMMARY OF RESULTS

The analysis of the reactor vessel material contained in the fourth surveillance capsule (Capsule Y) removed for evaluation as part of the Zion Nuclear Plant Unit 1 Reactor Vessel Surveillance Program, led to the following conclusions:

1. The capsule received an average fast fluence of 1.56×10^{19} n/cm^2 (E > 1.0 MeV). The predicted fast fluence for the reactor vessel T/4 location at the end of the tenth fuel cycle is 2.90×10^{18} n/cm^2 (E > 1 MeV).

2. The fast fluence of 1.56×10^{19} n/cm^2 (E > 1 MeV) increased the RT_{NDT} of the capsule reactor vessel core region shell materials a maximum of 205F.

3. Based on the calculated fast flux at the vessel wall, an 80% load factor and the planned fuel management, the projected fast fluence that the Zion Nuclear Plant Unit 1 reactor pressure vessel inside surface will receive in 40 calendar year's operation is 1.60×10^{19} n/cm^2 (E > 1 MeV).

4. The increase in the RT_{NDT} for the shell plate material was in good agreement with that predicted by the currently used design curves of RT_{NDT} versus fluence (i.e., Regulatory Guide 1.99, Revision 2), and the prediction techniques are conservative.

5. The increase in the RT_{NDT} for the weld metal was in good agreement with that predicted.

6. The weld metal upper-shelf energy at the T/4 location, based on surveillance capsule results, will not decrease below 50 ft-lbs prior to 32 EFPY.

7. The current techniques (i.e., Regulatory Guide 1.99, Revision 2) used to predict the change in weld metal Charpy upper-shelf properties due to irradiation are conservative.

8. The analysis of the neutron dosimeters demonstrated that the analytical techniques used to predict the neutron flux and fluence were accurate.

9. SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Based on the postirradiation test results of Capsule Y the following schedule is recommended for the examination of the remaining capsules in the Zion Nuclear Plant Unit 1 RVSP:

Evaluation Schedule[a]

Capsule Identification	Location of Capsules[a]	Lead Factor[b]	Removal Time	Expected Capsule Fluence (n/cm^2)[c]
S	4°	1.13	Standby	1.70×10^{19}
V	176°	1.13	Standby	1.70×10^{19}
W	184°	1.13	Standby	1.70×10^{19}
Z	356°	1.13	Cycle 21	2.20×10^{19}

(a) Reference reactor vessel irradiation locations, Figure 3-1.

(b) The factor by which the capsule fluence leads the vessels maximum inner wall fluence.

(c) Based on current capsule analysis and BAW-1543, Rev. 3.[29]

10. CERTIFICATION

The specimens were tested, and the data obtained from Commonwealth Edison Company Zion Nuclear Plant Unit 1, reactor vessel surveillance Capsule Y were evaluated using accepted techniques and established standard methods and procedures in accordance with the requirements of 10CFR50, Appendixes G and H.

A L Lowe Jr, P.E. 1/19/90

A. L. Lowe, Jr., P.E. — Date
Project Technical Manager

This report has been reviewed for technical content and accuracy.

L B Gross 2/19/90

L. B. Gross, P.E. (Material Analysis) — Date
M&SA Unit

Donald A. Nitti 2/19/90

D. A. Nitti (Fluence Analysis) — Date
Performance Analysis Unit

Verification of independent review.

A D McKim 2/19/90

A. D. McKim, Manager — Date
M&SA Unit

This report is approved for release.

J. F. Walters 3/7/90

J. F. Walters — Date
Program Manager

APPENDIX A

Reactor Vessel Surveillance Program
Background Data and Information

1. Material Selection Data

The data used to select the materials for the specimens in the surveillance program, in accordance with E185-82, are shown in Table A-1. The locations of these materials within the reactor vessel are shown in Figures A-1 and A-2.

2. Definition of Beltline Region

The beltline region of Zion Unit 1 was defined in accordance with the definition given in ASTM E185-82.

3. Capsule Identification

The capsules used in the Zion Unit 1 surveillance program are identified below by identification, location, and lead factor.

		Lead Factors[b]	
Capsule Identification	Capsule Location[a]	Previous Analysis[c]	Current Analysis
T	40°	3.74	3.00
U	140°	3.74	3.00
X	220°	3.74	3.00
Y	320°	3.74	3.00
S	4°	1.09	1.13
V	176°	1.09	1.13
W	184°	1.09	1.13
Z	356°	1.09	1.13

[a] Reference irradiation capsule locations as shown in Figure A-3.

[b] The factor by which the capsule fluence leads the vessels maximum inner wall fluence.

[c] Previous analysis as reported in WCAP-9890.

4. Specimens per Surveillance Capsule

Material	Capsules S, V, W, and X		
	Charpy	Tensile	WOL
Plate B7835-1			
Longitudinal	10	-	-
Transverse	10	2	4
Weld metal	8	2	-
HAZ	8	-	-
Correlation Monitor	8	-	-

Material	Capsules T and U		
	Charpy	Tensile	WOL
Plate B7835-1			
Longitudinal	10	2	4
Transverse	10	-	-
Weld metal	8	2	-
HAZ	8	-	-
Correlation monitor	8	-	-

Material	Capsules Y and Z		
	Charpy	Tensile	WOL
Plate B7835-1			
Longitudinal	10	2	-
Transverse	10	-	-
Weld metal	8	2	4
HAZ	8	-	-
Correlation monitor	8	-	-

Figure A-1. Location and Identification of Materials Used in the Fabrication of Zion Unit 1 Reactor Pressure Vessel

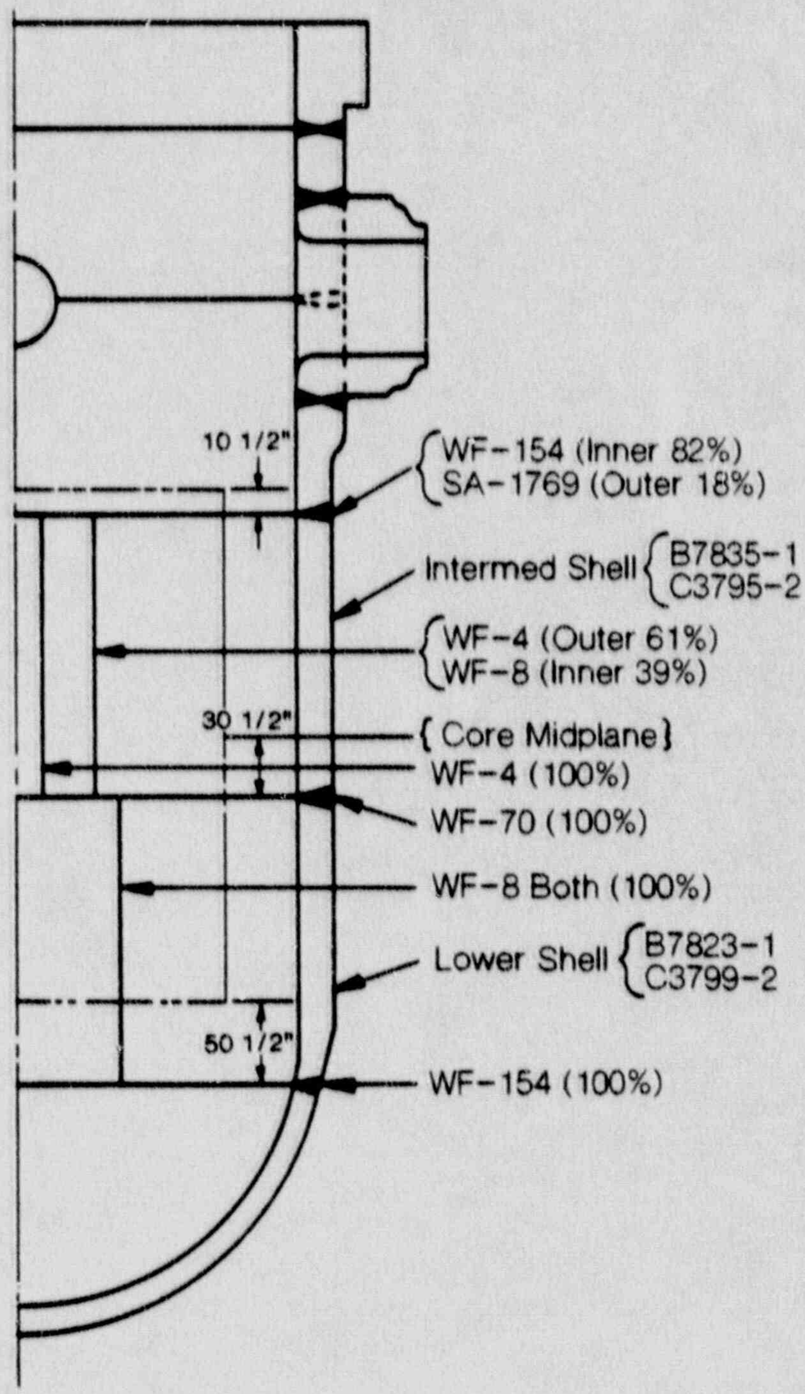


Figure A-2. Location of Longitudinal Welds in Zion Unit 1 Upper and Lower Shell Courses

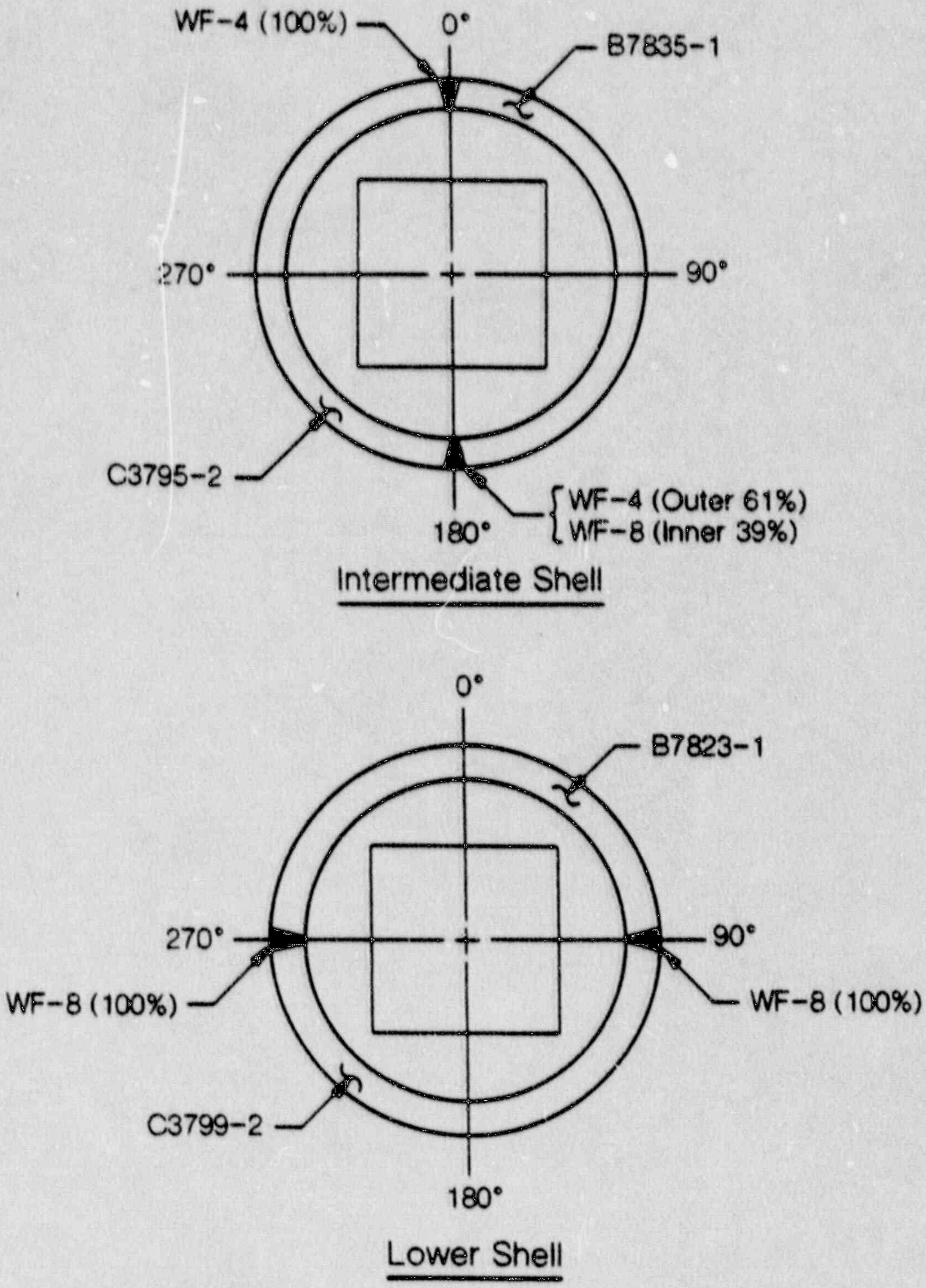
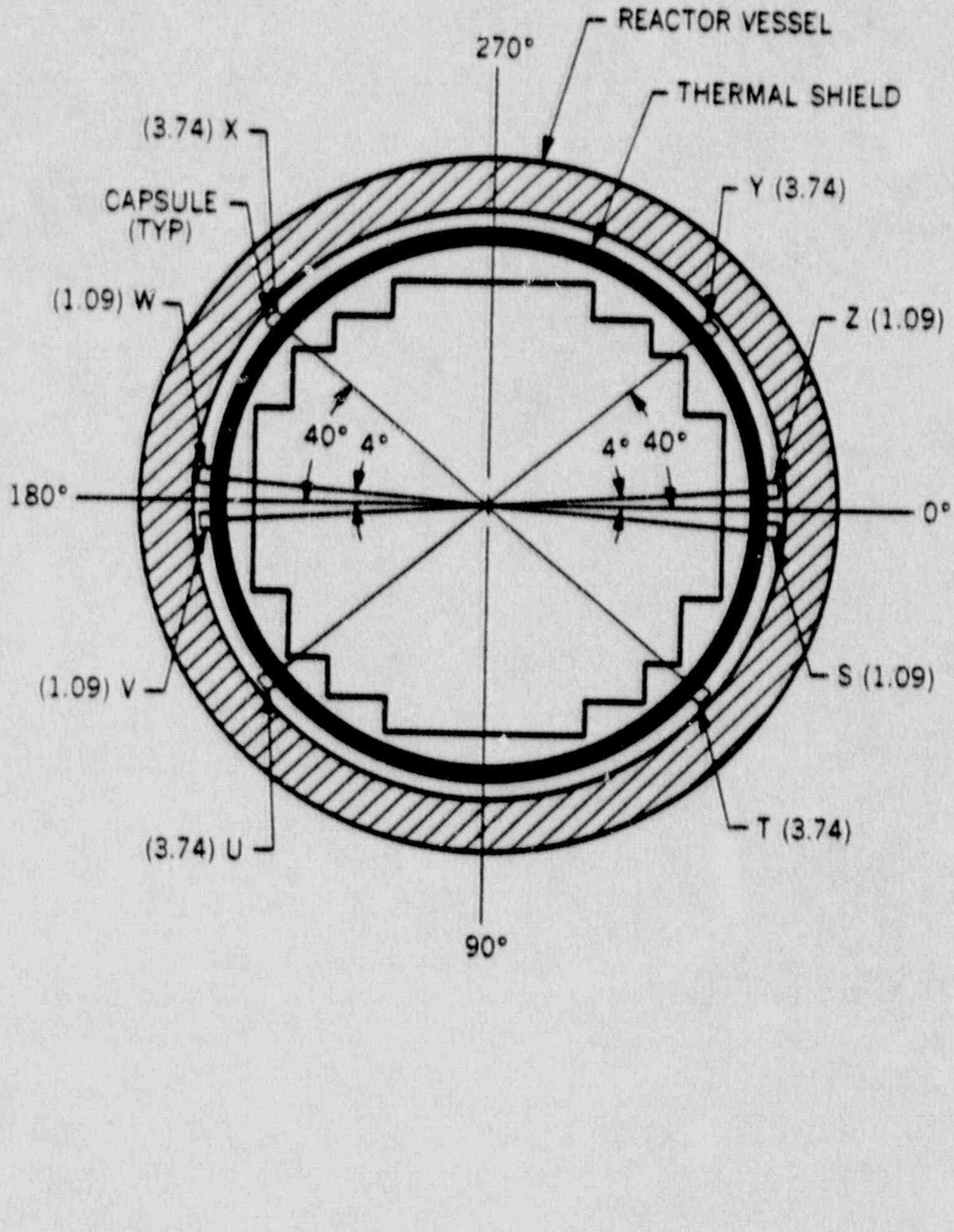


Figure A-3. Location of Surveillance Capsule Irradiation Sites in Zion Unit 1 Reactor Vessel (Lead Factors for the Capsules Shown in Parentheses are for the Original Fuel Management)



APPENDIX B

Pre-Irradiation Tensile Data

Table B-1. Tensile Properties of Unirradiated Shell Plate Material, Heat No. B7835-1, Longitudinal[1]

Specimen No.	Test Temp, F	Strength, psi Yield	Strength, psi Ultimate	Elongation, % Uniform	Elongation, % Total	Reduction of Area, %
--	RT	63.40	83.75	15.2	27.6	71.7
--	RT	62.60	83.85	15.4	27.5	71.3
--	300	56.05	76.30	14.8	23.7	71.3
--	300	56.00	76.25	13.6	24.1	69.6
--	600	54.30	80.90	14.0	25.0	63.1

Table B-2. Tensile Properties of Unirradiated Shell Plate Material, Heat No. B7835-1, Transverse[1]

Specimen No.	Test Temp, F	Strength, psi Yield	Strength, psi Ultimate	Elongation, % Uniform	Elongation, % Total	Reduction of Area, %
--	RT	61.55	82.95	15.5	27.6	69.1
--	RT	61.10	82.50	15.4	27.1	69.1
--	300	55.80	75.70	13.8	23.0	63.1
--	300	55.70	76.15	13.2	23.7	67.6
--	600	52.60	73.40	14.8	24.5	64.4
--	600	53.40	81.40	14.0	23.3	62.6

Table B-3. Tensile Properties of Unirradiated Weld Metal[1]

Specimen No.	Test Temp, F	Strength, psi Yield	Strength, psi Ultimate	Elongation, % Uniform	Elongation, % Total	Reduction of Area, %
--	RT	69.85	87.15	15.5	26.7	63.5
--	RT	75.55	91.60	14.4	24.8	61.9
--	300	65.50	82.05	13.4	23.3	64.1
--	300	68.60	83.85	13.6	23.5	63.6
--	600	66.30	87.45	13.3	21.0	51.4
--	600	66.75	87.95	13.2	21.0	49.1

APPENDIX C

Pre-Irradiation Charpy Impact Data

Table C-1. Charpy Impact Data From Unirradiated Base Material, Longitudinal Orientation, Heat No. B7835-1

Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
300	175.0	92	100
300	134.0	94	100
300	131.0	93	100
210	132.5	93	100
210	136.0	97	100
210	142.0	96	100
140	122.0	89	93
140	127.0	93	96
140	115.0	87	93
75	103.0	80	59
75	100.0	79	57
75	90.0	73	53
40	63.0	53	38
40	68.0	57	43
40	69.0	59	47
10	57.0	49	43
10	39.5	36	35
10	37.0	32	33
- 20	12.5	13	21
- 20	21.0	19	23
- 20	20.0	20	23
- 50	6.5	6	5
- 50	8.0	8	9
- 50	6.5	6	5

Table C-2. Charpy Impact Data From Unirradiated Base Material, Transverse Orientation, Heat No. B7835-1

Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
300	119.0	91	100
300	115.0	92	100
300	116.5	85	100
210	113.0	89	100
210	115.0	90	100
210	113.0	89	100
140	102.0	82	96
140	103.0	82	96
140	88.0	76	91
75	65.0	58	47
75	69.0	60	52
75	58.5	51	43
40	44.0	40	33
40	42.5	40	33
40	25.0	27	23
10	33.5	30	29
10	32.5	30	29
10	25.0	25	27
- 20	15.0	14	13
- 20	10.5	10	9
- 20	10.5	11	9
- 50	7.0	8	5
- 50	6.5	5	5
- 50	5.0	4	3

Table C-3. Charpy Impact Data from Unirradiated Base Metal, HAZ, Longitudinal Orientation, Heat No. B7835-1

Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
210	155.0	96	100
210	142.5	94	100
210	168.0	94	100
110	139.5	95	100
110	157.0	97	100
110	123.5	97	100
10	153.0	95	100
10	120.0	89	100
10	133.0	--	100
- 50	87.0	66	52
- 50	130.5	90	71
- 50	49.0	38	30
-100	61.5	44	34
-100	17.0	15	16
-100	75.0	56	51
-125	29.0	22	17
-125	21.0	15	18
-125	16.0	12	18
-200	40.5	25	25
-200	8.5	6	5
-200	29.0	21	25

Table C-4. Charpy Impact Data from Unirradiated Weld Metal, WF-209-1

Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
300	68.0	73	100
300	61.5	72	100
300	61.5	67	100
210	62.5	66	100
210	63.5	66	100
210	65.5	67	100
110	65.0	70	100
110	59.0	64	95
80	46.5	52	90
80	51.5	54	90
80	52.5	56	91
10	27.5	31	43
10	37.0	37	47
10	31.0	34	43
- 20	23.0	24	37
- 20	18.0	20	32
- 20	25.5	26	37
- 50	9.0	11	12
- 50	17.5	17	21
- 50	16.5	17	17
-100	5.0	7	5
-100	6.5	7	9
-100	6.5	7	5

Table C-5. Charpy Impact Data from Unirradiated Correlation Monitor Material, Longitudinal Orientation, HSST Plate 02

Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
300	127.0	84	100
300	117.5	83	100
300	125.0	87	100
210	121.0	87	100
210	115.0	88	98
210	117.0	84	98
160	109.0	79	87
160	81.0	69	85
160	108.5	72	84
110	63.5	54	55
110	85.5	71	67
110	82.5	60	58
85	52.0	45	42
85	41.5	42	41
85	58.5	51	43
40	35.0	32	29
40	36.0	32	29
40	22.0	23	33
10	13.5	14	23
10	14.5	14	23
10	12.0	15	23
-20	6.0	9	13
-20	9.0	10	13
-20	6.5	6	9
-50	3.0	4	9
-50	5.0	5	9
-50	5.0	3	9

Figure C-1. Charpy Impact Data From Unirradiated Base Metal (Plate), Longitudinal Orientation, Heat No. B7835-1

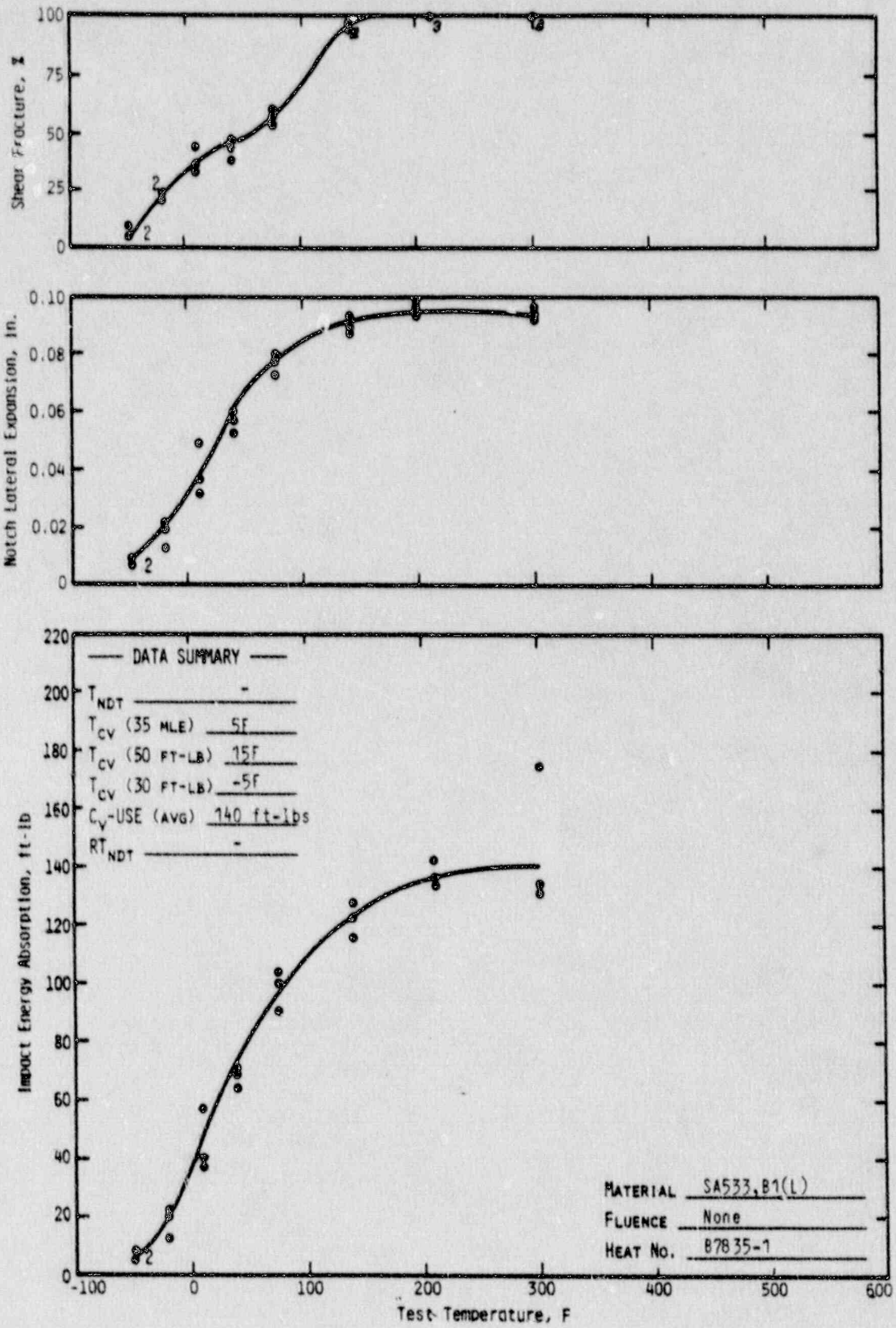


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Figure C-2. Charpy Impact Data From Unirradiated Base Metal (Plate), Transverse Orientation, Heat No. B7835-1

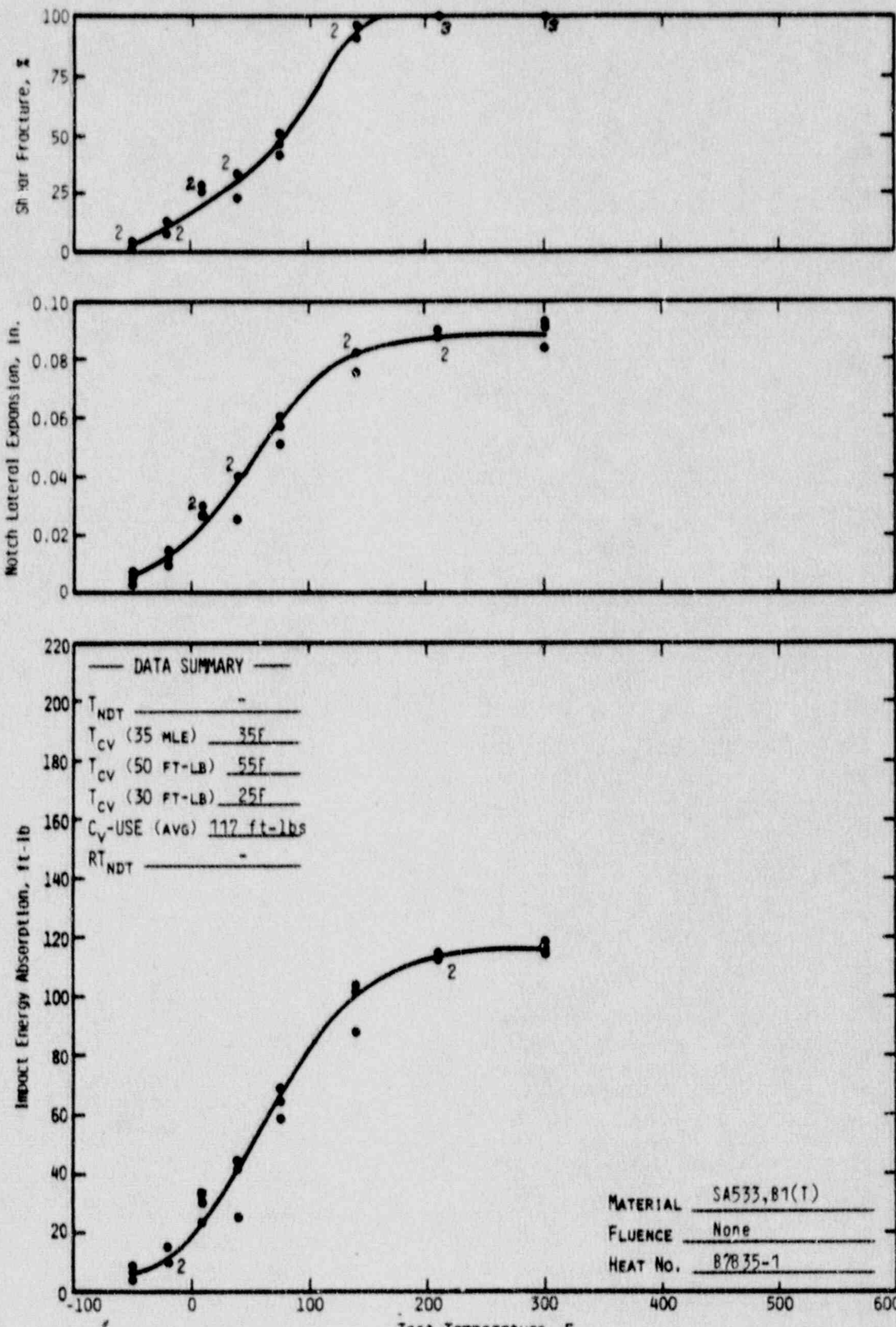


Figure C-3. Charpy Impact Data From Unirradiated Heat-Affected-Zone Base Metal, Heat No. B7835-1

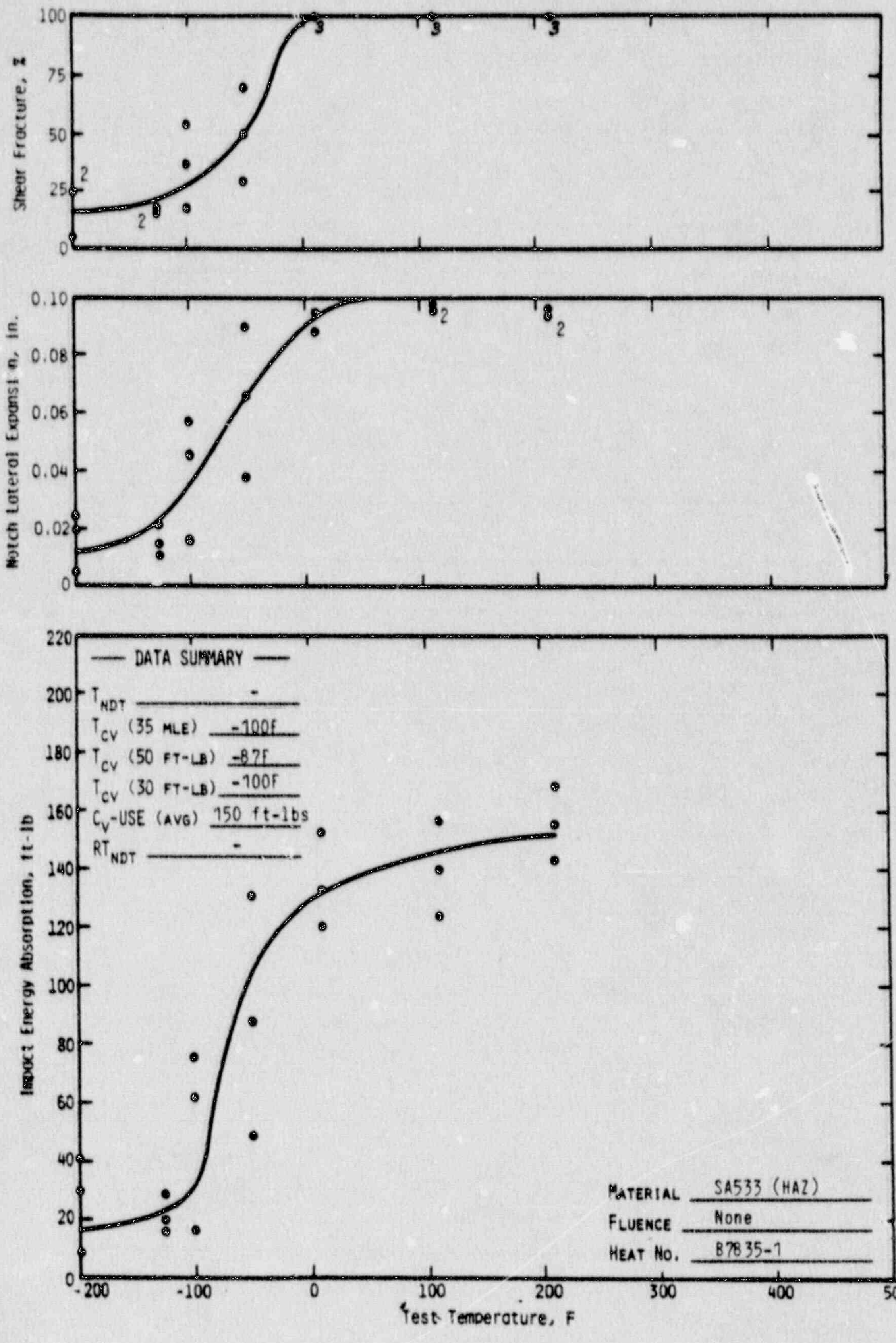


Figure C-4. Charpy Impact Data From Unirradiated Weld Metal, WF-209-1

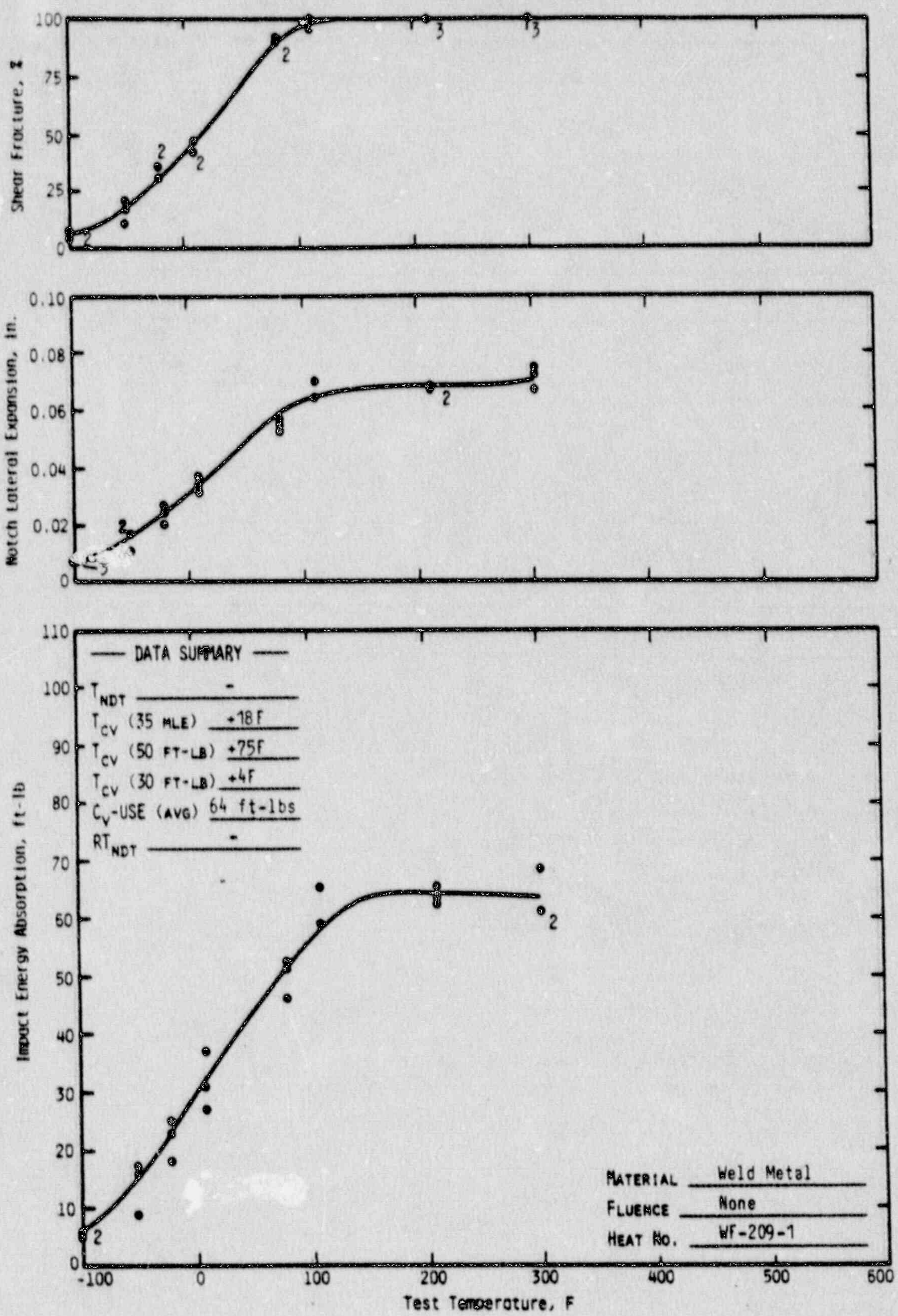
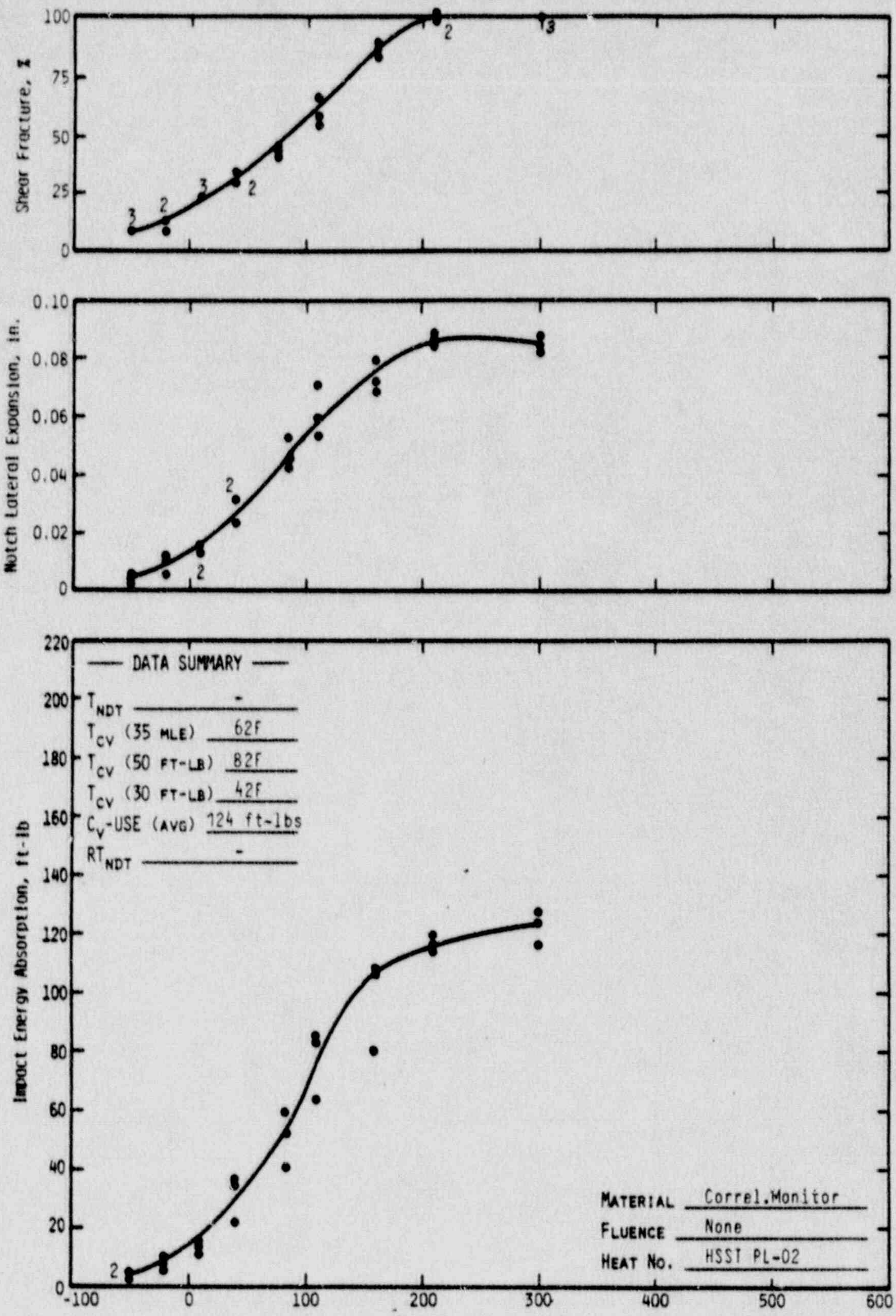


Figure C-5. Charpy Impact Data for Unirradiated Correlation Monitor Material, HSST PL-02



APPENDIX D

Fluence Analysis Methodology

1. Analytical Method

A semiempirical method is used to calculate the capsule and vessel flux. The method employs explicit modeling of the reactor vessel and internals and uses an average core power distribution in the discrete ordinates transport code DOTIV, version 4.3. DOTIV calculates the energy and space dependent neutron flux for the specific reactor under consideration. This semiempirical method is conveniently outlined in Figures D-1 (capsule flux) and D-2 (vessel flux).

The two-dimensional transport code DOTIV was used to calculate the energy- and space-dependent neutron flux at all points of interest in the reactor system. DOTIV uses the discrete ordinates method of solution of the Boltzmann transport equation and has multi-group and asymmetric scattering capability. The reference calculational model is an R-θ geometric representation of a plan view through the reactor core midplane which includes the core, core liner, coolant, core barrel, neutron pad, pressure vessel, and concrete. The material and geometry model, represented in Figure D-3, uses one-eight core symmetry. In order to include reasonable geometric detail within the computer memory limitations, the code parameters are specified as P_3 order of scattering, S_8 quadrature, and 40 energy groups. The P_3 order of scattering adequately describes the predominately forward scattering of neutrons observed in the deep penetration of steel and water media, as demonstrated by the close agreement between measured and calculated dosimeter activities. The S_8 symmetric quadrature has generally produced accurate results in discrete ordinates solutions for similar problems, and is used routinely in the B&W R-θ DOT analyses.

Flux generation in the core was represented by a fixed distributed source which the code derives based on a ^{235}U fission spectrum, the input relative power distribution, and a normalization factor to adjust flux level to the desired power density.

Geometrical Configuration

For modeling purposes, the actual geometrical configuration is divided into three parts, as shown in Figure D-3. The first part, Model "A," is used to generate the energy-dependent angular flux at the inner boundary of Model "B," which begins at the inside surface of the core barrel. Model A includes a detailed representation of the core baffle (or liner) in R-θ geometry that has been checked for both metal thickness and total metal volume to ensure that the DOT approximation to the actual geometry is as accurate as possible for these two very important parameters. The second, Model B, contains an explicit representation of the surveillance capsule and associated components. The B&W Owners Group's Flux Perturbation Experiment[28] verified that the surveillance capsule must be explicitly included in the DOT models used for capsule and vessel flux calculations in order to obtain the desired accuracy. The magnitude of the perturbations in the fast flux due to the presence of the capsule was determined in the Perturbation Experiment to be as high as 47% at the center of B&W capsules and as high as 10% at the inner surface of the reactor vessel. Detailed explicit modeling of the capsule, capsule holder tube, and internal components is therefore incorporated into the DOT calculational models. The third, Model "C," is similar to Model B except that no capsule is included. Model C is used in determining the vessel flux in quadrants that do not contain a surveillance capsule; typically these quadrants contain the azimuthal flux peak on the inside surface of the reactor vessel.

An overlap region of approximately [illegible]3.47 cm is specified between Model A and Models B or C. The width of this overlap region, which is fixed by the placement of the Model A vacuum boundary and the Model B boundary source, is greater than the distance determined by an iterative process that resulted in close agreement between the overlap region flux as predicted by Models A and B or C. The outer boundary was placed sufficiently far into the concrete shield (cavity wall) that the use of a "vacuum" boundary condition does not cause a perturbation in the flux at the points of interest.

Macroscopic Cross Sections

Macroscopic cross sections, required for transport analyses, are obtained with the mixing code GIP. Nominal compositions are used for the structural metals. Coolant compositions were determined using the average boron concentration over a fuel cycle and the bulk temperature of the region. The core region is a homogeneous mixture of fuel, fuel cladding, structure, and coolant.

The cross-section library presently used is the (22-neutron group and 18-gamma group) CASK 23E 40-group set. The dosimeter reaction cross sections are based on the ENDF/B5 library, and are listed in Table E-3. The measured and calculated dosimeters activities are compared in Table D-1.

Distributed Source

The neutron population in the core during full power operation is a function of neutron energy, space, and time. The time dependence is accounted for in the analysis by calculating the time-weighted average neutron source, i.e. the neutron source corresponding to the time-weighted average power distribution. The effects of the other two independent variables, energy and space, are accounted for by using a finite but appropriately large number of discrete intervals in energy and space. In each of these intervals the neutron source is assumed to be invariant and independent of all other variables. The space and energy dependent source function can be considered as the product of a discretely expressed "spatial function" and a magnitude coefficient, i.e.

$$Sv_{ijg} = \underbrace{[\upsilon/K\ P_D]}_{\text{magnitude}} \times \underbrace{[RPD_{ij}X_g]}_{\text{spatial}} \quad \text{(D-1)}$$

where:

Sv_{ijg} = Energy-and space-dependent neutron source, n/cc-sec,

υ/K = Fission neutron production rate, n/w-sec,

P_D = Average power density in core, w/cc,

RPD_{ij} = Relative power density at interval (i,j), unitless,

X_g = Fission spectrum, fraction of fission neutrons having energy in group "g,"

i = Radial coordinate index,

j = Azimuthal coordinate index,

g = Energy group index.

The spatial dependence of the flux is directly related to the RPD distribution. Even though the entire (eighth-core symmetric) RPD distribution is modeled in the analysis, only the peripheral fuel assemblies contribute significantly to the ex-core flux. The axial average RPD distribution is calculated on a quarter-core symmetric basis for the entire capsule irradiation period. The time-weighted average RPD distribution is used to generate the normalized space and energy dependency of the neutron source. Calculations for the energy and space dependent, time-averaged flux were performed for the midpoint of each DOT interval throughout the model. Since the reference model calculation produced fluxes in the R-θ plane that averaged over the core height, an axial correction factor was required to adjust these fluxes to the capsule elevation. This factor was calculated to be 1.19.

1.1. Capsule Flux and Fluence Calculation

As discussed above, the DOTIV code was used to explicitly model the capsule assembly and to calculate the neutron flux as a function of energy within the capsule. The calculated fluxes were used in the following equation to obtain calculated activities for comparison with the measured data. The calculated activity for reaction product D_i, in (μCi/gm) is:

$$D_i = \frac{N \quad f_i}{(3.7 \times 10^4)An} \sum_E \sigma_n(E)\, \phi(E) \sum_j F_j \left(1 - e^{-\lambda_i t_j}\right) e^{-\lambda_i (T - \tau_j)} \qquad \text{(D-2)}$$

where:

N = Avogadro's number,

An = Atomic weight of target material n,

f_i = Either weight fraction of target isotope in n-th material or the fission yield of the desired isotope,

$\sigma_n(E)$ = Group-averaged cross sections for material n (listed in Table E-3)

$\phi(E)$ = Group averaged fluxes calculated by DOTIV analysis,

F_j = Fraction of full power during j-th time interval, t_j

λ_i = Decay constant of the ith isotope,

T = Sum of total irradiation time, i.e., residual time in reactor, and the wait time between reactor shutdown and counting times,

τ_j = Cumulative time from reactor startup to end of j-th time period.

t_j = Length of the j-th time period

Adjustments were made to the calculated dosimeter activities to correct for the effects listed below:

Short half-life adjustments to Ni and Fe dosimeter activities.

Photofission adjustments to ^{238}U and ^{237}Np dosimeter activities.

Fissile impurity adjustments to ^{238}U dosimeter activities.

After making these adjustments the calculated dosimeter activities were used with the corresponding measured activities to obtain the flux normalization factors:

$$C_i = \frac{D_i \text{ (measured)}}{D_i \text{ (calculated)}}.$$

These normalization factors were evaluated, averaged, and then used to adjust the calculated test specimen flux and fluence to be consistent with the dosimeter measurements. The ratio of measured to calculated dosimeter activities are listed in Table D-1.

Table D-1. Capsule Normalization Constant

	Measured Activity,[a] μCi/g	Calculated Activity,[b] μCi/g	M/C[c]
$^{54}Fe(n,p)^{54}Mn$	667.86	884.542	1.050[d]
$^{58}Ni(n,p)^{58}Co$	1111.99	1333.805	1.121[d]
$^{238}U(n,f)^{137}Cs$	14.30	16.434	0.870
$^{237}Np(n,f)^{137}Cs$	120.07	128.119	0.937
$^{63}Cu(n,\gamma)^{60}Co$	5.71	6.931	0.824
$^{59}Co(n,\gamma)^{60}Co$ (Cd covered)	156283.76	156018.016	1.002

Average Normalization Constant (M/C Value) = 0.967

[a] Average of dosimeters from Table E-2 with impurity and photofission corrections for U-238 and Np-237.

[b] Average of calculated activities.

[c] Ratio of average measured activity to average calculated activity.

[d] Includes short half-life correction for Fe and Ni dosimeters.

Figure D-1. Rationale for the Calculation of Dosimeter Activities and Neutron Flux in the Capsule

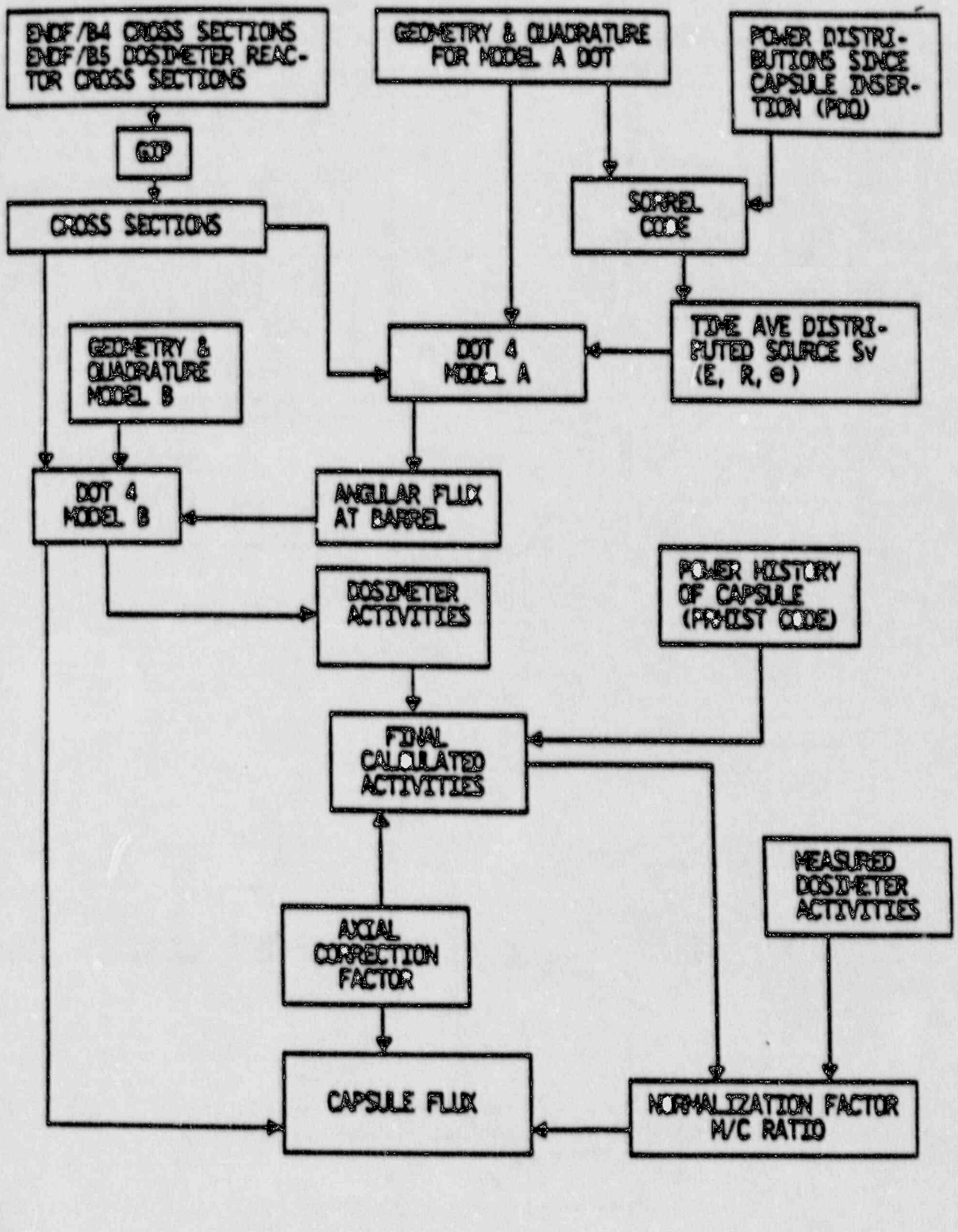


Figure D-2. Rationale for the Calculation of Neutron Flux in the Reactor Vessel

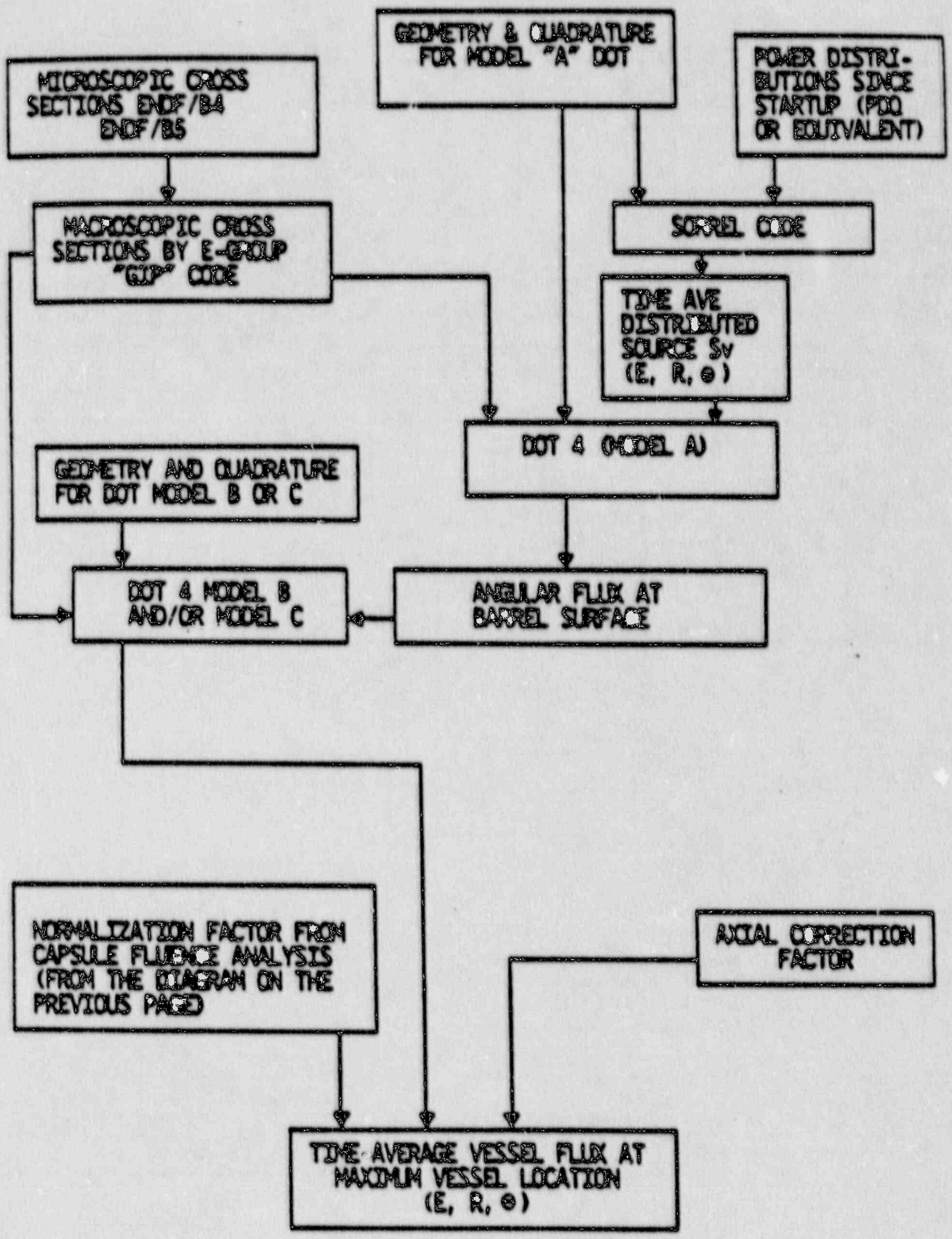
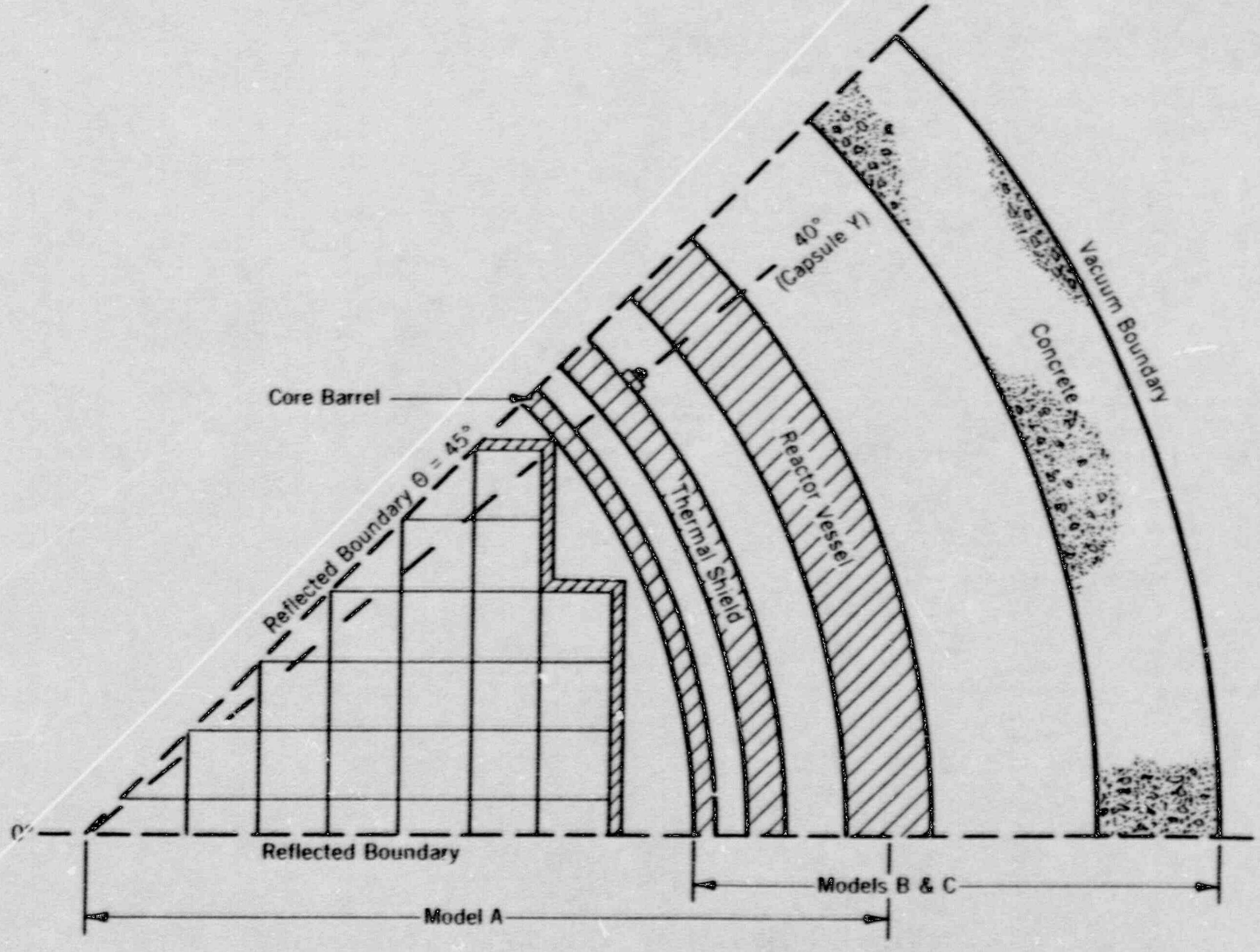


Figure D-3. Plan View Through Reactor Core Midplane (Reference R-θ Calculation Model)



APPENDIX E

Capsule Dosimetry Data

Table E-1 lists the characteristics of the neutron dosimeters. Table E-2 shows the measured activity per gram of target material (i.e., per gram of uranium, nickel, etc.) for the capsule dosimeters. Activation cross sections for the various materials were flux-weighted with the ^{235}U fission spectrum shown in Table E-3.

Table E-1. Detector Composition and Shielding

Detector Material	% Target	Shielding	Reaction
U_3O_8	84.80% ^{238}U	CdO	$^{238}U(n,f)^{137}Cs$
NpO_2	88.11% ^{237}Np	CdO	$^{237}Np(n,f)^{137}Cs$
Ni	68.27% ^{58}Ni	None	$^{58}Ni(n,p)^{58}Co$
Co-Al	0.15% ^{59}Co	Cd	$^{59}Co(n,\gamma)^{60}Co$
Co-Al	0.15% ^{59}Co	None	$^{59}Co(n,\gamma)^{60}Co$
Fe	5.82% ^{54}Fe	None	$^{54}Fe(n,p)^{54}Mn$
Cu	69.20% ^{63}Cu	None	$^{63}Cu(n,\gamma)^{60}Co$

Table E-2. Measured Specific Activities (Unadjusted) for Dosimeters in Capsule Y

Detector Material	Dosimeter Reaction	Dosimeter Activity, (μCi/gm of Target)					
		Top	Upper	Middle	Lower	Bottom	Fission
Ni	$^{58}Ni(n,p)^{58}Co$	---	1108.80	1092.20	1134.97	--	--
Fe	$^{54}Fe(n,p)^{54}Mn$	683.88	664.50	661.74	670.50	658.69	--
U_3O_8	$^{238}U(n,f)^{137}Cs$	---	---	---	---	--	1[illegible].81
NpO_2	$^{237}Np(n,f)^{137}Cs$	---	---	---	---	--	122.[illegible]
Cu	$^{63}Cu(n,\alpha)^{60}Co$	---	5.71	5.59	5.84	--	--
Co-Al	$^{59}Co(n,\alpha)^{60}Co$ (Cd covered)	195411.42	---	---	---	117156.10	--

Table E-3. Dosimeter Activation Cross Sections, b/atom[(a)]

G	Energy Range, MeV	^{237}Np(n,f)	^{238}U(n,f)	^{58}Ni(n,p)	^{54}Fe(n,p)	^{63}Cu(n,)
1	12.2 - 15	2.323	1.051E+0	4.830E-1	4.133E-1	4.478E-2
2	10.0 - 12.2	2.341	9.851E-1	5.735E-1	4.728E-1	5.361E-2
3	8.18 - 10.0	2.309	9.935E-1	5.981E-1	4.772E-1	3.378E-2
4	6.36 - 8.18	2.093	9.110E-1	5.921E-1	4.714E-1	1.246E-2
5	4.96 - 6.36	1.542	5.777E-1	5.223E-1	4.321E-1	3.459E-3
6	4.06 - 4.96	1.532	5.454E-1	4.146E-1	3.275E-1	6.348E-4
7	3.01 - 4.06	1.614	5.340E-1	2.701E-1	2.193E-1	7.078E-4
8	2.46 - 3.01	1.689	5.325E-1	1.445E-1	1.080E-1	3.702E-6
9	2.35 - 2.46	1.695	5.399E-1	9.154E-2	5.613E-2	6.291E-7
10	1.83 - 2.35	1.676	5.323E-1	4.856E-2	2.940E-2	1.451E-7
11	1.11 - 1.83	1.596	2.608E-1	1.180E-2	2.948E-3	1.317E-9
12	0.55 - 1.11	1.241	9.845E-3	1.336E-3	6.999E-5	0
13	0.111 - 0.55	2.352E-1	2.436E-4	5.013E-4	6.419E-8	0
14	0.0033 - 0.111	1.200E-2	6.818E-5	1.512E-5	0	0

(a) ENDF/B5 values that have been flux weighted (over CASK energy groups) based on a ^{235}U fission spectrum in the fast energy range plus a 1/E shape in the intermediate energy range.

APPENDIX F

References

1. S. E. Yanichko and D. J. Lege, Commonwealth Edison Company, Zion Unit No. 1 Reactor Vessel Radiation Surveillance Program, WCAP-8064, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, March 1973.

2. J. S. Perrin, et al., Zion Nuclear Plant Reactor Pressure Vessel Surveillance Program: Unit No. 1 Capsule T, and Unit No. 2 Capsule U, BCL-585-4, Battelle Columbus Laboratories, Columbus, Ohio, March 25, 1978.

3. S. E. Yanichko, et al., Analysis of Capsule U from the Commonwealth Edison Company Zion Nuclear Plant Unit 1 Reactor Vessel Radiation Surveillance Program, WCAP-9890, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, March 1981.

4. E. B. Norris, Reactor Vessel Material Surveillance Program for Zion Unit No. 1 Analysis of Capsule X, Project No. 06-7484-001, Southwest Research Institute, San Antonio, Texas, March 1984.

5. American Society for Testing and Materials, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, E185-70, July 1970.

6. Code of Federal Regulation, Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix G, Fracture Toughness Requirements.

7. Code of Federal Regulation, Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix H, Reactor Vessel Material Surveillance Program Requirements.

8. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure (G-2000).

9. C. E. Childress, "Fabrication History of the First Two 12-inch-thick ASTM A-533, Grade B, Class 1 Steel Plates of the Heavy Section Steel Technology Program, Documentary Report 1," ORNL-4313, February 1969.

10. American Society for Testing and Materials, Methods and Definitions for Mechanical Testing of Steel Products, A370-68.

11. American Society for Testing and Materials, Methods for Notched Bar Impact Testing of Metallic Materials, E23-72.

12. Standardized Specimens for Certification of Charpy Impact Specimens from the Army Materials and Mechanics Research Center, Watertown, Mass. 02172, Attn: DRXMR-MQ.

13. American Society for Testing and Materials, Methods and Definitions for Mechanical Testing of Steel Products, A370-77.

14. American Society for Testing and Materials, Methods for Notched Bar Impact Testing of Metallic Materials, E23-86.

15. S. Q. King, Pressure Vessel Fluence Analysis for 177-FA Reactors, BAW-1485P, Revision 1, Babcock & Wilcox, Lynchburg, Virginia, April 1988.

16. B&W's Version of DOTIV Version 4.3, One- and Two-Dimensional Transport Code System," Oak Ridge National Laboratory, Distributed by the Radiation Shielding Information Center as CC-429, November 1, 1983.

17. "CASK-40-Group Coupled Neutron and Gamma-Ray Cross Section Data," Radiation Shielding Information Center, DLC-23E.

18. Dosimeter File ENDF/B5 Tape 531, distributed March 1984, National Neutron Data Center, Brookhaven National Laboratory, Upton, Long Island, NY.

19. American Society of Testing Materials, Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements Per Atom (DPA), E693-79 (Reapproved 1985).

20. E. L. Furchi, et al., Zion Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations, WCAP-10962, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, December 1985.

21. U.S. Nuclear Regulatory Commission, Radiation Damage to Reactor Vessel Material, Regulatory Guide 1.99, Revision 2, May 1988.

22. S. E. Yanisko and J. N. Chirigos, Observations of a Steady State Effect Limiting Radiation Damage in Reactor Vessel Steels, Nuclear Engineering and Design 56 (1980) p. 297-307.

23. K. E. Moore, et al., Evaluation of the Atypical Weldment, BAW-10144A, Babcock & Wilcox, Lynchburg, Virginia, February 1980.

24. A. L. Lowe, Jr., et al., Pressurized Thermal Shock Evaluations in Accordance with 10CFR50.61 for Babcock & Wilcox Owners Group Reactor Pressure Vessel, BAW-1895, Babcock & Wilcox, Lynchburg, Virginia, January 1986.

25. K. E. Moore and A. S. Heller, B&W 177-FA Reactor Vessel Beltline Weld Chemistry Study, BAW-1799, Babcock & Wilcox, Lynchburg, Virginia, July 1983.

26. A. S. Heller and A. L. Lowe, Jr., Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds, BAW-1803, Babcock & Wilcox, Lynchburg, Virginia, January 1984.

27. H. S. Palme, H. W. Behnke, and W. J. Keyworth, Methods of Compliance With Fracture Toughness and Operational Requirements of 10CFR50, Appendix G, BAW-10046P, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, March 1976.

28. N. L. Snider and L. A. Hassler, B&WOG Flux Perturbation Experiment at ORNL, Measured and Calculated Dosimeter Results, BAW-1886, Babcock & Wilcox, Lynchburg, Virginia, September 1985.

29. S. Fyfitch, L. B. Gross and A. L. Lowe, Jr., Master Integrated Reactor Vessel Surveillance Program, BAW-1543, Rev. 3, Babcock & Wilcox, Lynchburg, Virginia, September 1989.

30. Commonwealth Edison Company letter from D. E. O Brien to Mr. A. Schwenur of the Nuclear Regulatory Commission, "Zion Station Units 1 and 2, NRC Docket Numbers 50-295 and 50-304," September 7, 1977.