Entergy

Entergy Operations, Inc.
17265 River Road
Killona, LA 70066
Tel 504 739 6650

W3F1-2004-0075

September 13, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Reissue of Report BAW-2177, "Analysis of Capsule W-97 Entergy Operations, Inc. Waterford Generating Station Unit 3 – Reactor Vessel Material Surveillance Program"
Docket No. 50-382
License No. NPF-38
Waterford 3

Reference:

1. Letter Number W3F192-0369, dated November 25, 1992, Reactor Vessel Material Surveillance Program Requirements – Report of Test Results

2. Attachment to Letter Number W3F192-0369, BAW-2177, "Analysis of Capsule W-97 Entergy Operations, Inc. Waterford Generating Station Unit 3 – Reactor Vessel Material Surveillance Program," dated November 1992.

Dear Sir or Madam:

The purpose of this letter is to provide a revised copy of report BAW-2177, "Analysis of Capsule W-97 Entergy Operations, Inc. Waterford Generating Station Unit 3 – Reactor Vessel Material Surveillance Program," which contained an editorial error. This report was originally submitted as an attachment to Letter Number W3F192-0369, dated November 25, 1992, Reactor Vessel Material Surveillance Program Requirements – Report of Test Results. This error was communicated to the Waterford 3 Senior Resident Inspector and NRC Project Manager. Following these discussions, Waterford 3 committed to submit a revised report which includes a correction of the editorial error.

This error was discovered during a recent review of report BAW-2177 by Westinghouse while assembling Charpy data for development of the new Charpy-irradiation correlation for the ASTM E900 standard. In review of the report by Areva, the vendor who provided the report, it was noted that the values in Table 5.6, "Charpy Impact Results for Capsule W-97 Weld Metal, 88114/0145, 6.47 x 1018 n/cm^2" were incorrect in Revision 0 of the report.

A008

Figure 5-8, "Charpy Impact Data for Irradiated Weld Metal, 88114/0145" had been plotted in Revision 0 using the correct data even though Table 5.6 had incorrect data. A "cut and paste" mistake had occurred from a previous draft.

Section 7.0 "Discussion of Capsule Results" and Table 7.3, "Observed vs. Predicted Changes for Capsule W-97 Irradiated Charpy Impact Properties – 6.47 x 10^{18} n/cm^2 (E> 1 MEV)" used the correct data from Figure 5-8 for the comparison of observed vs. predicted property changes. Since the actual numbers used in the comparison of the transition temperature and upper shelf energy changes are correct, the conclusions that the calculated property changes are conservative relative to the observed properties therefore remain unchanged.

In summary, the error in Table 5.6 of report BAW-2177 has no impact on the conclusions contained in the report. The calculated reactor vessel material properties remain conservative in relation to the observed, via specimen testing, material properties.

There are no new commitments contained in this submittal.

Should you have questions regarding this report please contact Mrs. Stacie Fontenot at 504-739-6656.

Sincerely,

R.A. Dodds III

R.A. Dodds
Manager, Licensing

RAD/STF/ssf
Attachment: Report BAW-2177-01, dated February 2004, "Analysis of Capsule W-97 Entergy Operations, Inc. Waterford Generating Station Unit 3 – Reactor Vessel Material Surveillance Program"

cc: Mr. Bruce S. Mallett
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Waterford Steam Electric Station Unit 3
P.O. Box 822
Killona, LA 70066-0751

U. S. Nuclear Regulatory Commission
Attn: Mr. N. Kalyanam
Mail Stop O-07D1
Washington, DC 20555-0001

Wise, Carter, Child & Caraway
ATTN: J. Smith
P.O. Box 651
Jackson, MS 39205

Winston & Strawn
ATTN: N.S. Reynolds
1400 L Street, NW
Washington, DC 20005-3502

Attachment

W3F1-2004-0075

Report BAW-2177-01, dated February 2004, "Analysis of Capsule W-97 Entergy Operations, Inc. Waterford Generating Station Unit 3 – Reactor Vessel Material Surveillance Program"

The B&W
Owners Group

BAW-2177-01
February 2004

ANALYSIS OF CAPSULE W-97
ENTERGY OPERATIONS, INC.
WATERFORD GENERATING STATION, UNIT NO. 3

-- Reactor Vessel Material Surveillance Program --

AREVA

BAW-2177-01
February 2004

ANALYSIS OF CAPSULE W-97
ENTERGY OPERATIONS, INC.
WATERFORD GENERATING STATION, UNIT NO. 3

-- Reactor Vessel Material Surveillance Program --

by

A. L. Lowe, Jr., PE
R. E. Napolitano
D. M. Spaar
W. R. Stagg

FRAMATOME ANP

Document No. 77-2177-01
(See Section 10 for document signatures)

Framatome ANP, Inc.
3315 Old Forest Road
P. O. Box 10935
Lynchburg, Virginia 24506-0935

FRAMATOME ANP

RECORD OF REVISIONS

Revision	Description	Date
00	Original Release	11/92
01	Page 5-6 was replaced correcting the data in Table 5-6. All subsequent reporting of the weld metal Charpy results were correct. The conclusions are not affected. Page 10-2 was added containing the Revision 1 signatures. Both cover pages were replaced reflecting revision change.	2/04

SUMMARY

This report describes the results of the examination of the first capsule (Capsule W-97) of the Entergy Operations, Inc., Waterford Generating Station, Unit No. 3 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 E185-73.

The capsule received an average fast fluence of 6.47×10^{18} n/cm² (E > 1.0 MeV) and the predicted fast fluence for the reactor vessel T/4 location at the end of the fourth cycle is 2.74×10^{18} n/cm² (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 Waterford Generating Station, Unit No. 3 reactor pressure vessel inside surface will receive in 40 calendar years of operation is 3.69×10^{19} n/cm² (E > 1 MeV) and the corresponding T/4 fluence is calculated to be 1.97×10^{19} n/cm² (E > 1 MeV).

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.

CONTENTS

Contents (Cont'd)

APPENDIXES | Page

List of Tables

Table

Tables (Cont'd)

List of Figures

Figures (Cont'd)

Figures (Cont'd)

Page Intentionally Left Blank

1. INTRODUCTION

This report describes the results of the examination of the first capsule (Capsule W-97) of the Entergy Operations, Inc., Waterford Generating Station, Unit No. 3 (Waterford Unit 3) reactor vessel material surveillance program (RVSP). The capsule was removed and evaluated after being irradiated in the Waterford Unit 3 reactor as part of the reactor vessel materials surveillance program (Combustion Engineering (C-E) Report C-NLM-003[1]). The capsule experienced a fluence of 6.47×10^{18} n/cm^2 (E > 1 MeV), which is the equivalent of approximately six effective full power years' (EFPY) operation of the Waterford Unit 3 reactor vessel inside surface.

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 Waterford Generating Station Unit No. 3 was designed and furnished by Combustion Engineering, Incorporated (C-E) as described in TR-C-MCS-001[2] and conducted in accordance with 10CFR50, Appendix H[3]. 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.

Page Intentionally Left Blank

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 low-alloy ferritic steels such as SA533, Grade B, used in the fabrication of the Waterford Unit 3 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,"[4] 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 for 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 16, 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,"[3] 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 (B&PV) Code, Section III, "Nuclear Power Plant Components."[5] 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[6]) 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 B&PV Code 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.

Page Intentionally Left Blank

3. SURVEILLANCE PROGRAM DESCRIPTION

The surveillance program for Waterford Unit 3 comprises six surveillance capsules designed to monitor the effects of neutron and thermal environment on the materials of the reactor pressure vessel core region. The capsules, which were inserted into the reactor vessel before initial plant startup, were positioned near the inside wall of the reactor vessel at the locations shown in Figure 3-1. The six capsules, designed to be placed in holders attached to the reactor vessel wall are positioned near the peak axial and azimuthal neutron flux. During the four cycles of operation, Capsule W-97 was irradiated in the 97° position adjacent to the reactor vessel wall as shown in Figure 3-1.

Capsule W-97 was removed during the fourth refueling shutdown of Waterford Unit 3. The capsule contained Charpy V-notch impact test specimens fabricated from the one base metal (SA533, Grade B1) both longitudinal and transverse orientation, one heat-affected-zone, and a weld metal. Tension test specimens were fabricated from the base metal, heat-affected-zone, and 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 Figures 3-2 through 3-4. The chemical composition and heat treatment of the surveillance material in Capsule W-97 are described in Table 3-2.

All plate and heat-affected-zone specimens were machined from the 1/4-thickness (1/4T) location of the plate material. Weld metal specimens were machined throughout the thickness of the weldment. 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.

The neutron dosimeters contained in Capsule W-97 are as follows:

Material	Shielding	Reaction	Threshold Energy (Mev)	Half-Life
Uranium	None/Cd	U^{238} (n,f) Sr^{90}	0.7	28.0 years
Sulfur	None	S^{32} (n,p) p^{32}	2.9	14.3 days
Iron	None	Fe^{54} (n,p) Mn^{54}	4.0	312.5 days
Nickel	Cd	Ni^{58} (n,p) Co^{58}	5.0	70.9 days
Copper	Cd	Cu^{63} (n,α) Co^{60}	7.0	5.27 years
Titanium	None	Ti^{46} (n,p) Sc^{46}	8.0	83.8 days
Cobalt	None/Cd	Co^{59} (n,γ) Co^{60}	Thermal	5.27 years

Four thermal monitors of low-melting alloys were included in the W-97 capsule. The eutectic alloys and their melting points are as follows:

Alloy Composition, wt%	Melting Point, F
80.0 Au, 20.0 Sn	536
90.0 Pb, 5.0 Sn, 5.0 Ag	558
97.5 Pb, 2.5 Ag	580
97.5 Pb, 0.75 Sn, 1.75 Ag	590

Table 3-1. Specimens in Surveillance Capsule W-97

Material Description	Number of Test Specimens: Tension	CVN Impact
Base Metal (M-1004-2)		
Longitudinal	-	12
Transverse	3	12
Heat-Affected Zone	3	12
Weld Metal (88114/0145)	3	12
Total Per Capsule	9	48

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

Element	Chemical Composition, w/o: Heat No.[(a)] M-1004-2	Weld Metal 88114/0145[(b)]
C	0.23	0.23
Mn	1.38	1.35
P	0.005	0.008
S	0.005	0.006
Si	0.23	0.16
Ni	0.58	0.22
Cr	0.01	0.05
Mo	0.57	0.57
Cu	0.03	0.04

Heat No.	Heat Treatment: Temp, F	Time, h	Cooling
Plate (M-1004-2)	1575±50	4	Water Quenched
	1220±25	4	Furnace Cooled
	1150±25	40	Furnace Cooled to 600F
Weld Metal (88114/0145)	1100-1175	40 1/2	Furnace Cooled to 600F

(a) Chemical analysis by Combustion Engineering of surveillance program test plate.[7]

(b) Chemical analysis by Combustion Engineering of surveillance program test weld metal.[7]

Figure 3-1. Reactor Vessel Cross Section Showing Location of Capsule W-97 in Waterford Unit 3

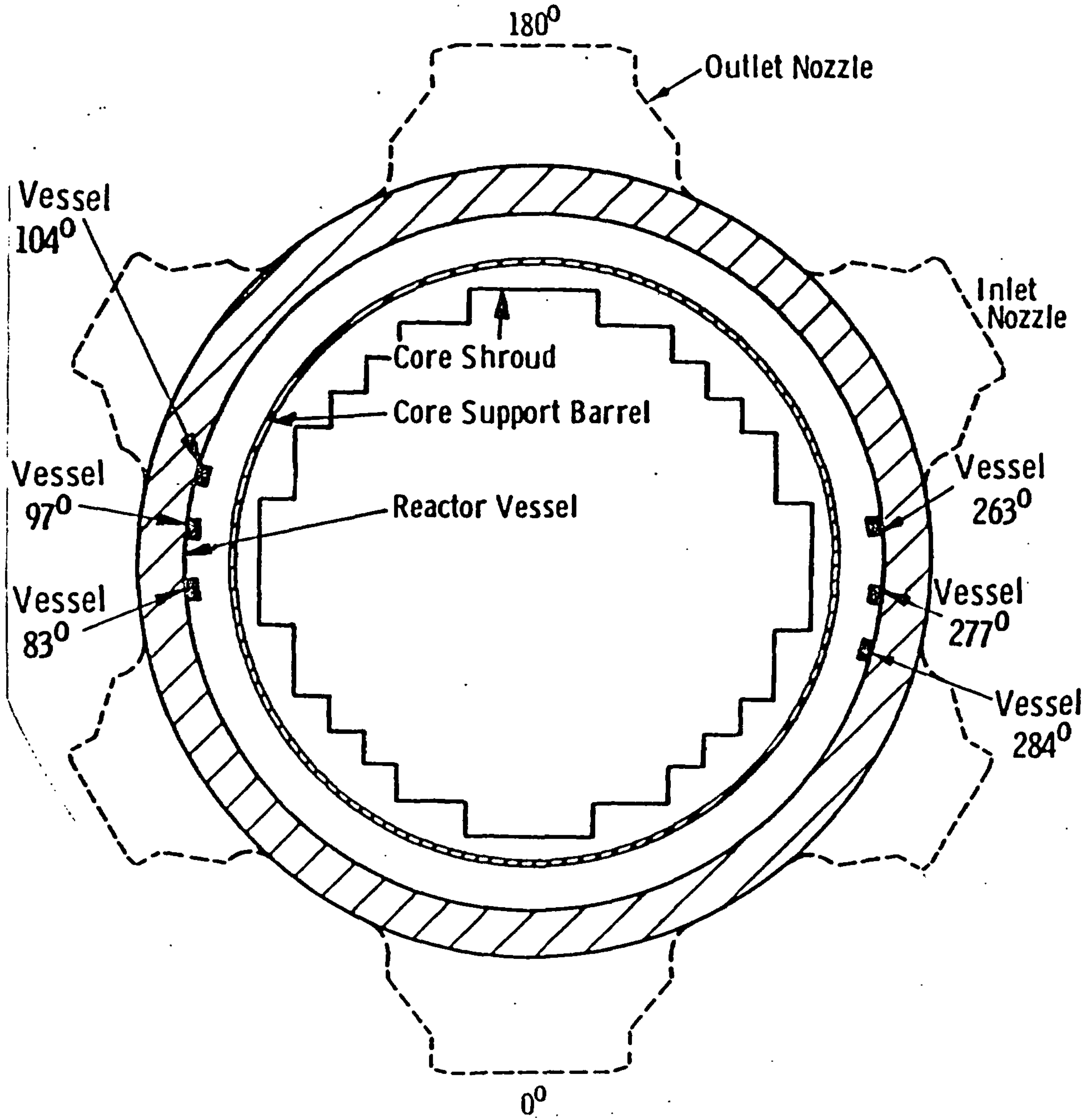


Figure 3-2. Typical Surveillance Capsule Assembly Showing Location of Specimens and Monitors

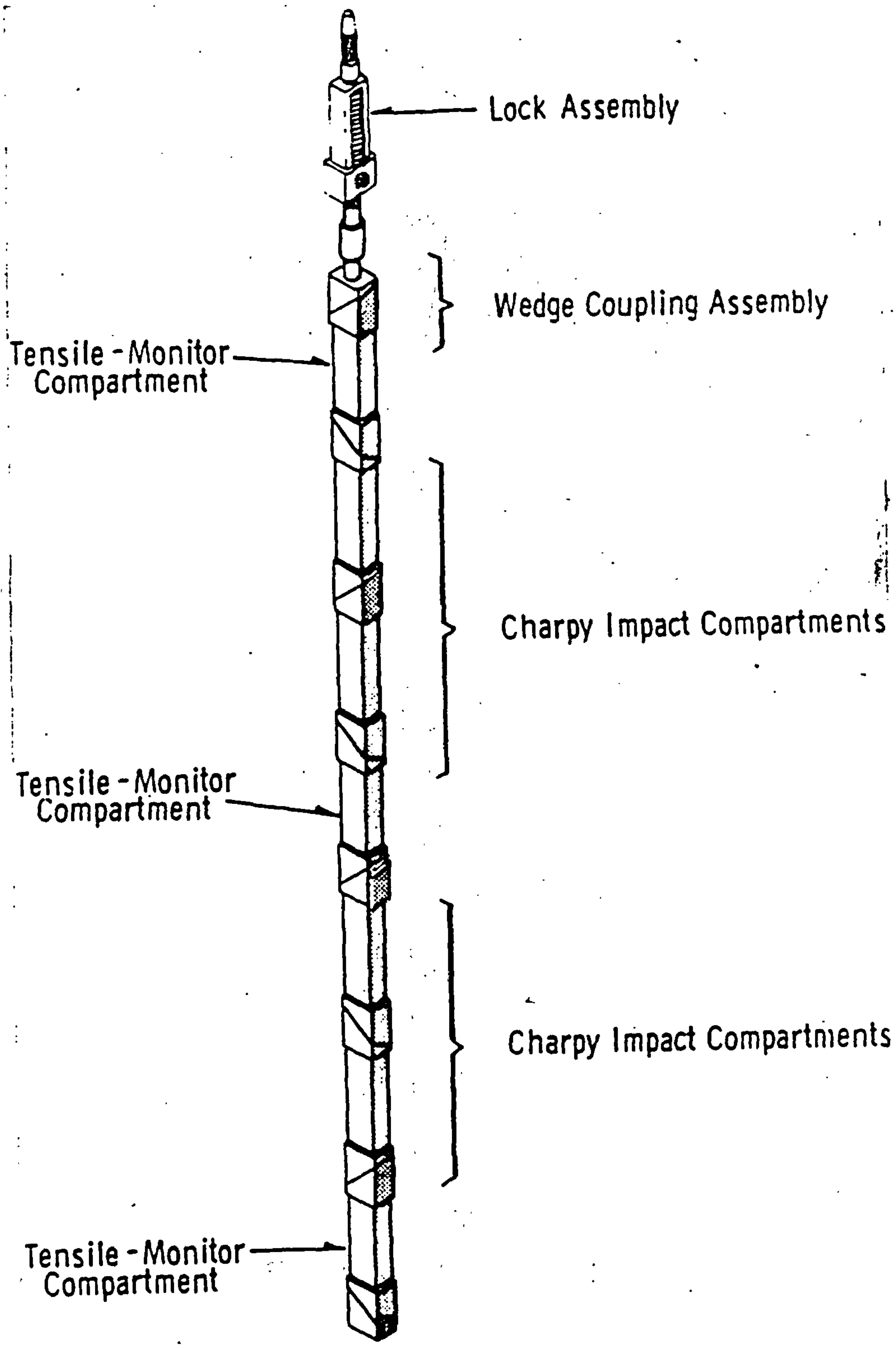


Figure 3-3. Typical Surveillance Capsule Tensile - Monitor Compartment Assembly (Three per Capsule)

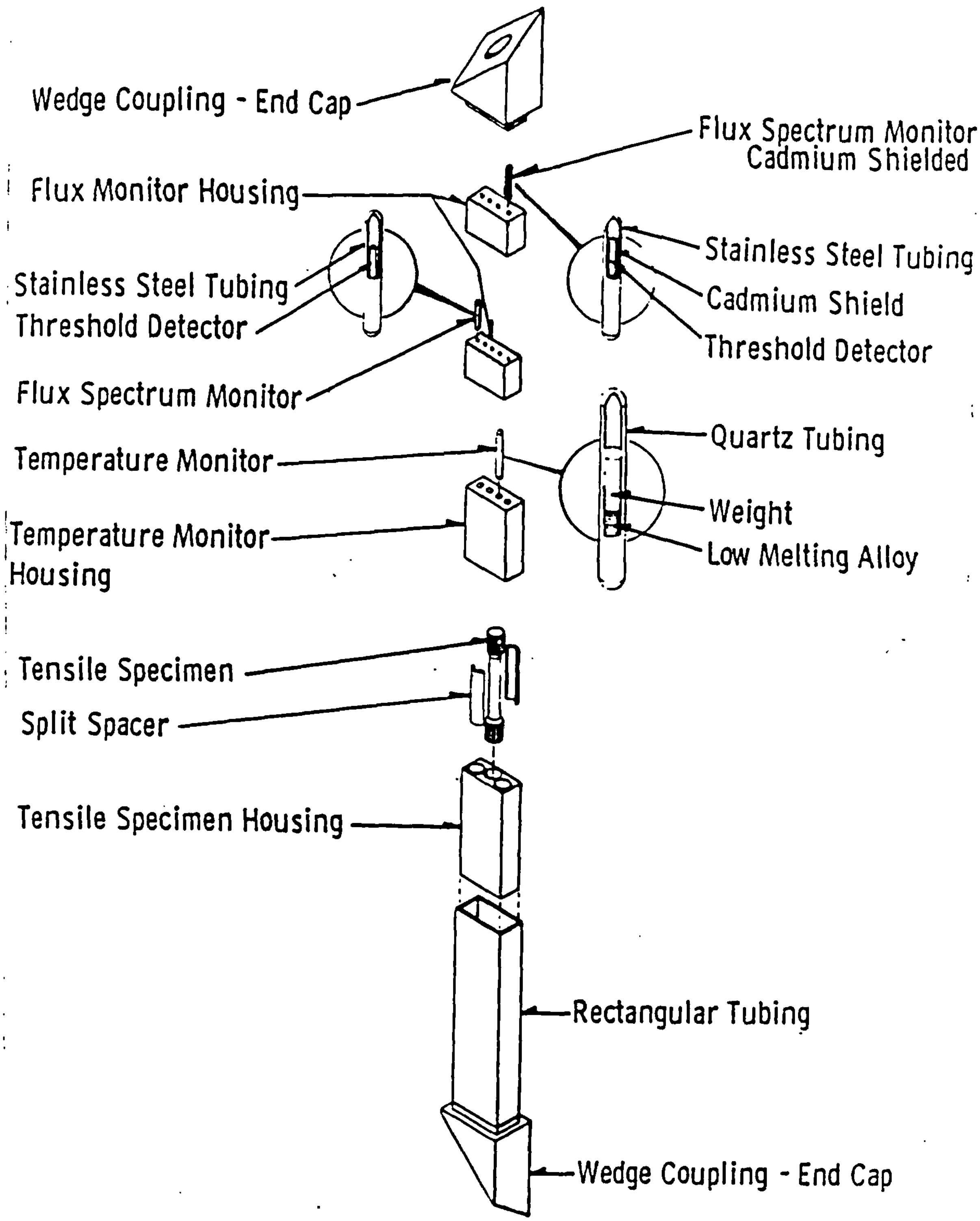
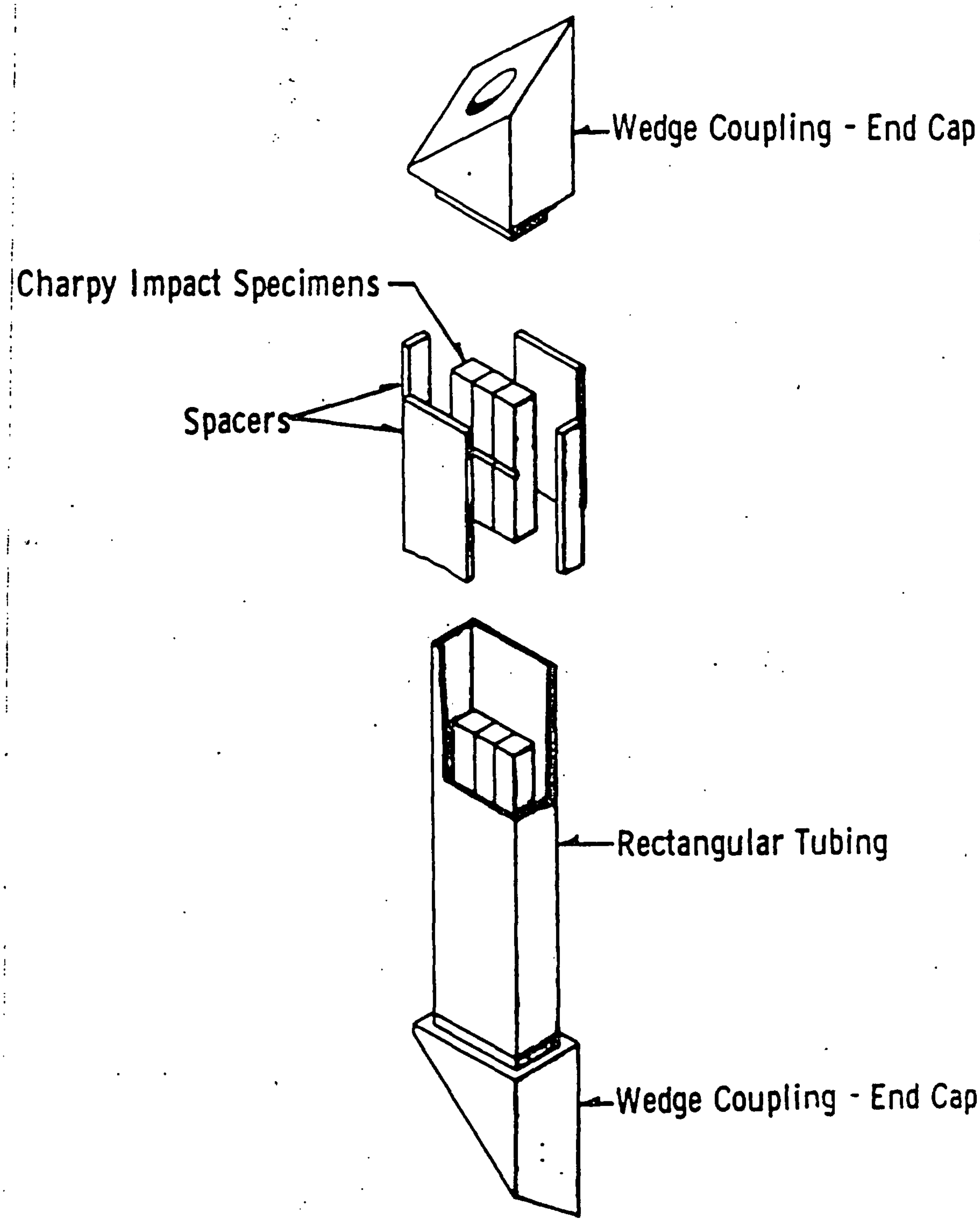


Figure 3-4. Typical Surveillance Capsule Charpy Impact Compartment Assembly (Four per Capsule)



Page Intentionally Left Blank

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 material 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 Combustion Engineering as part of the development of the Waterford Unit 3 surveillance program. The details of the testing procedures are described in C-E Report TR-C-MCS-002[7] and are summarized here to provide continuity.

4.1. Tension Tests

Tension test specimens were fabricated from the reactor vessel shell plate, HAZ metal, and weld metal. The specimens were 3.00 inches long with a reduced section 1.50 inches long by 0.250 inch in diameter. The tensile tests were performed using a Riehle universal screw testing machine with a maximum capacity of 30,000 lb and separate scale ranges between 50 lb and 30,000 lb. The machine is capable of constant cross head rate or constant strain rate operation.

Elevated temperature tests were performed in a 2-1/2" ID x 18" long high temperature tensile testing furnace with a temperature limit of 1800°F. A Riehle high temperature, dual range extensometer was used for monitoring specimen elongation.

Tensile testing was conducted in accordance with ASTM E-8, "Tension Tests of Metallic Materials:"[8] and/or Recommended Practice E-21, "Short-Time Elevated Temperature Tension Tests of Materials,"[9] except as modified by Section 6.1 of Recommended Practice E-184, "Effects of High-Energy Radiation on the Mechanical Properties of Metallic Matereials."[10]

For each material type and/or condition, nine specimens in groups of three were tested at room temperature, 250 and 550F. 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 a Model SI-1 BLH Sonntag Universal Impact Machine certified to meet Watertown standards.[12] Test specimens were of the Charpy V-notch type, which were nominally 0.394 inch square and 2.165 inches long.

Impact test data for the unirradiated baseline reference materials are presented in Appendix C. Tables C-1 through C-4 contain the basis data that are plotted in Figures C-1 through C-4. 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 and confirmed that the markings as those of Capsule W-97. 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.

5.2. Thermal Monitors

Surveillance Capsule W-97 contained three temperature monitor holder blocks each containing four fusible alloys with different melting points. Each of the thermal monitors was inspected and the results are tabulated in Table 5-1. Photographs of the monitors are shown in Figure 5-1.

From these data, it can be concluded that the irradiated specimens had been exposed to a maximum temperature no greater than 580F during the reactor vessel operating period. This is not significantly greater than the nominal inlet temperature of 550F, and is considered acceptable. However, the partly melted or slumped appearance of the 558F monitor is probably due to an irradiation induced creep mechanism and not the result of actual melting. This being the case, then the maximum temperature was no greater than 558F which is the most likely case. This behavior has been seen in other surveillance capsules. There appeared to be no signs of a significant temperature gradient along the capsule length.

5.3. Tension Test Results

The results of the post-irradiation tension tests are presented in Table 5-2. Tests were performed on specimens at room temperature, 250, and 550F. They were tested on a 55,000-lb load capacity MTS servohydraulic computer-controlled universal test machine. All tests were run using stroke control with an initial actuator travel rate of 0.005 inch per minute through yield point. Past specimen yielding an actuator speed of 0.040 inch per minute was used. 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 room temperature, 250 and 550F. The data for both the heat-affect zone specimen and the weld metal specimen, tested at 250F, were lost because of a test machine malfunction. 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 Figures 5-2 through 5-4.

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 is 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 5-3 through 5-6 and Figures 5-5 through 5-8. Photographs of the Charpy specimen fracture surfaces are presented in Figures 5-9 through 5-12. The Charpy V-notch impact tests were conducted in accordance with the requirements of ASTM E23-88[14] on a Satec SI-1K impact tester certified to meet Watertown standards.[10]

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.

Table 5-1. Conditions of Thermal Monitors in Capsule W-97

Capsule Segment	Melt Temperature	Post-Irradiation Condition
A1	536F	Melted
(Top)	558F	Melted (slumped?)
	580F	Unmelted
	590F	Unmelted
A4	536F	Melted
(Middle)	558F	Melted (slumped?)
	580F	Unmelted
	590F	Unmelted
A7	536F	Melted
(Bottom)	558F	Melted (slumped?)
	580F	Unmelted
	590F	Unmelted

Table 5-2. Tensile Properties of Capsule W-97 Base Metal and Weld Metal Irradiated to 6.47 x 10^{18} n/cm^2 (E > 1 MeV)**

Specimen No.	Test Temp, F	Strength, psi		Fracture			Elongation, %		Reduction in Area, %
		Yield	Ultimate	Load, lbs	Stress, psi	Strength, psi	Uniform	Total	
Base Metal, M-1004-2, Transverse									
2L6	70	70,400	92,600	3,097	173,000	63,100	11.7	26.2	63.5
2K5	250	65,500	85,800	2,834	175,300	57,700	10.2	23.1	67.1
2K2	550	63,500	90,000	2,994	162,900	61,000	10.2	23.0	62.5
Base Metal Heat-Affected Zone, M-1004-2									
4K3	70	69,500	93,500	2,844	184,700	57,500	7.0	20.3	68.9
4KK	---	----	----	---	----	---	----	----	----
4J4	550	69,600	91,000	2,913	194,700	59,300	6.4	18.5	69.5
Weld Metal 88114/0145									
3JM	70	84,500	95,900	3,351	187,100	68,300	7.3	7.9*	63.5
3KK	---	----	----	----	----	----	----	----	----
3KY	550	74,000	93,200	2,766	187,700	56,400	7.9	22.6	70.0

*Results not valid - specimen necked and fractured outside extensometer gage length.

**Stress-strain curves are presented in Appendix F.

Table 5-3. Charpy Impact Results for Capsule W-97 Base Metal Longitudinal (LT) Orientation, Heat No. M-1004-2, 6.47 x 10^{18} n/cm^2

Specimen ID	Test Temperature, F	Impact Energy, ft-lbs	Lateral Expansion, inch	Shear Fracture, %
14Y	-50	7.5	0.006	0
11D	-25	24.0	0.021	0
13D	0	22.5	0.025	10
133	20	36.0	0.032	10
15C	30	84.5	0.061	40
12P	35	76.0	0.054	40
14C	50	72.5	0.057	40
15K	70	90.0	0.069	80
132	100	113.0	0.075	70
11E	150	156.0*	0.093	100
14K	200	152.0*	0.093	100
11Y	550	157.0	0.082	100

*Values used to determine upper-shelf energy value per ASTM E-185.[15]

Table 5-4. Charpy Impact Results for Capsule W-97 Base Metal Transverse (TL) Orientation, Heat No. M-1004-2, 6.47 x 10^{18} n/cm^2

Specimen ID	Test Temperature, F	Impact Energy, ft-lbs	Lateral Expansion, inch	Shear Fracture, %
216	-50	7.0	0.006	0
214	-25	11.0	0.011	5
25A	0	15.0	0.017	5
21C	10	36.0	0.031	10
22P	20	53.0	0.044	20
26K	35	54.0	0.047	40
25K	50	73.0	0.057	50
217	70	71.5	0.057	100
22L	100	88.5	0.070	70
245	150	121.5*	0.086	100
244	200	125.0*	0.086	100
22M	550	124.0	0.081	100

*Values used to determine upper-shelf energy value per ASTM E-185.[15]

Table 5-5. Charpy Impact Results for Capsule W-97 Base Metal Heat-Affected Zone Material, Heat No. M-1004-2, 6.47 x 10^{18} n/cm^2

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, inch	Shear Fracture, %
415	-100	21.0	0.014	5
46Y	-85	35.0	0.026	25
455	-65	53.5	0.036	15
45J	-50	90.0	0.060	60
42C	0	115.5	0.071	80
43D	10	101.0	0.067	70
44U	20	121.0	0.073	85
45K	50	119.5	0.077	85
45Y	70	155.0*	0.083	100
41M	100	163.5*	0.090	100
474	150	150.0*	0.071	100
414	550	>240.0	-----	-----

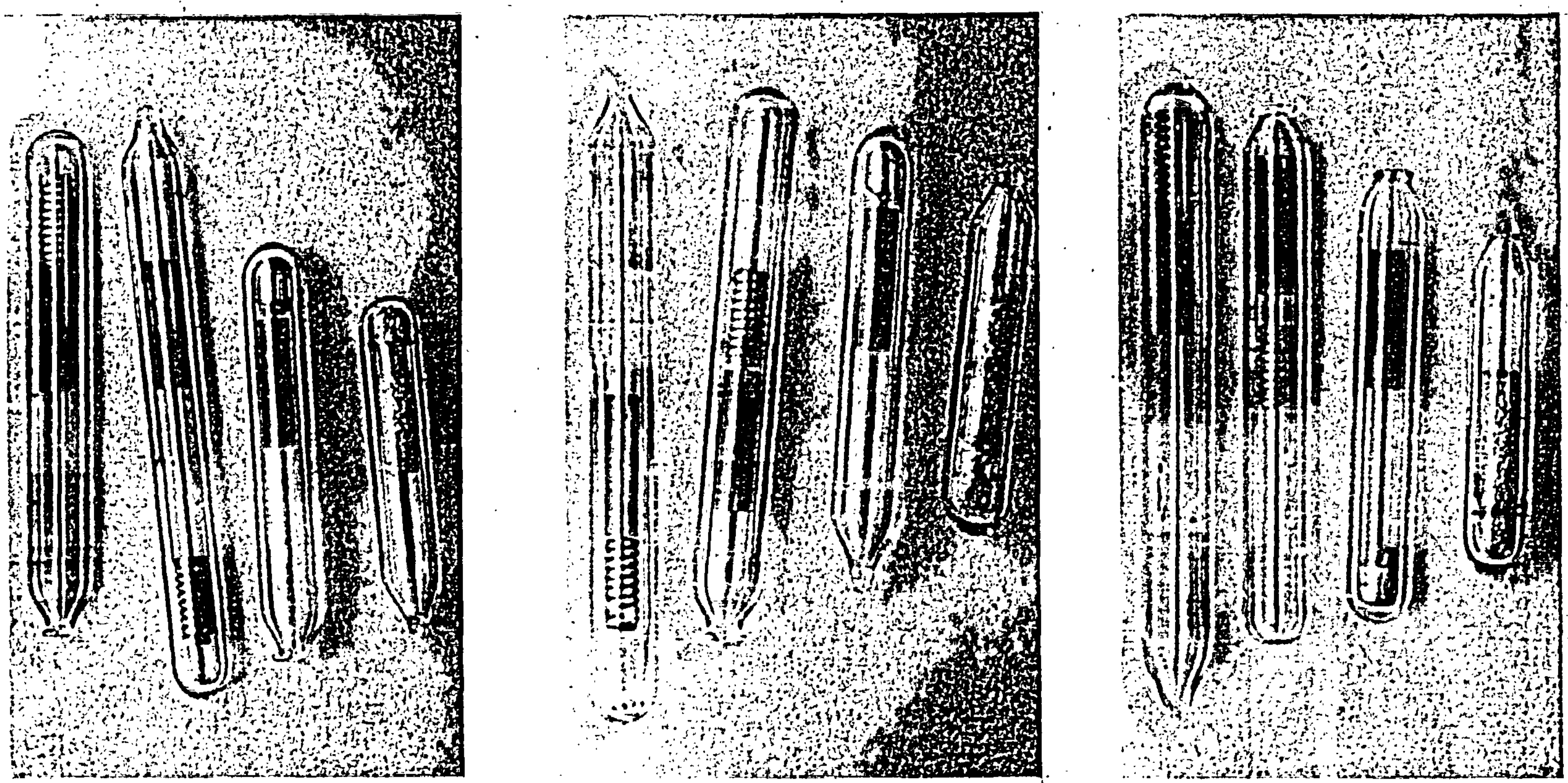
*Values used to determine upper-shelf energy value per ASTM E185.[15]

Table 5-6. Charpy Impact Results for Capsule W-97 Weld Metal, 88114/0145, 6.47 x 10^{18} n/cm^2

Specimen ID	Test Temperature, F°	Impact Energy, ft-lbs	Lateral Expansion, inch	Shear Fracture, %
35J	-50	14.5	0.015	40
32T	-40	39.5	0.033	45
31M	-35	67	0.050	50
32K	-20	93	0.068	65
362	-15	64.5	0.051	50
34L	0	78	0.060	60
31T	20	108	0.078	80
326	50	131	0.085	90
35K	70	134.5	0.091	90
33C	100	147*	0.093	100
374	200	139*	0.094	100
33B	550	176.5	0.079	100

*Values used to determine upper-shelf energy value per ASTM E185.[15]

Figure 5-1. Photographs of Thermal Monitor Melt Wire Capsules as Removed From Surveillance Capsule

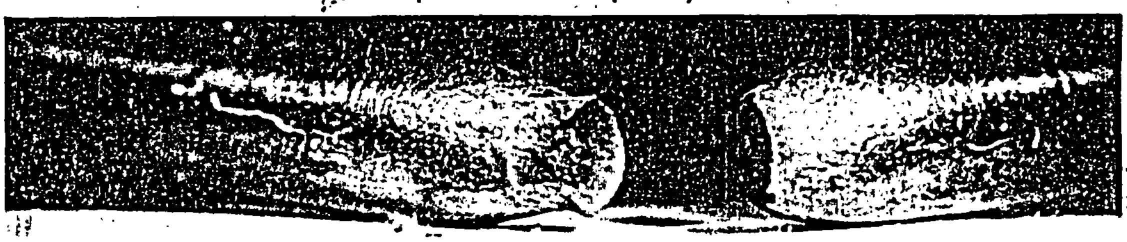


Bottom Group | Middle Group | Top Group

Figure 5-2. Photographs of Tested Tension Test Specimens and Corresponding Fractured Surfaces - Base Metal, Transverse Orientation

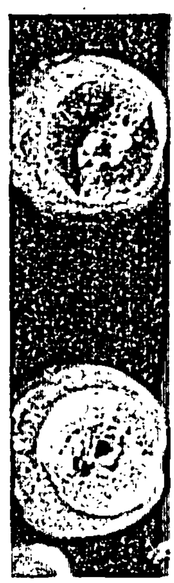
Specimen 2L6 (70F)

Specimen 2K5 (250F)



Specimen 2K2 (550F)

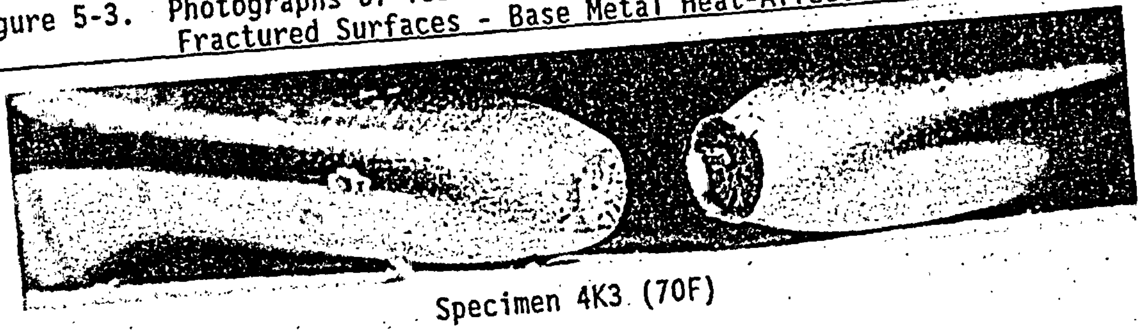
Specimen 2L6 (70F)



Specimen 2K5 (250F)

Specimen 2K2 (550F)

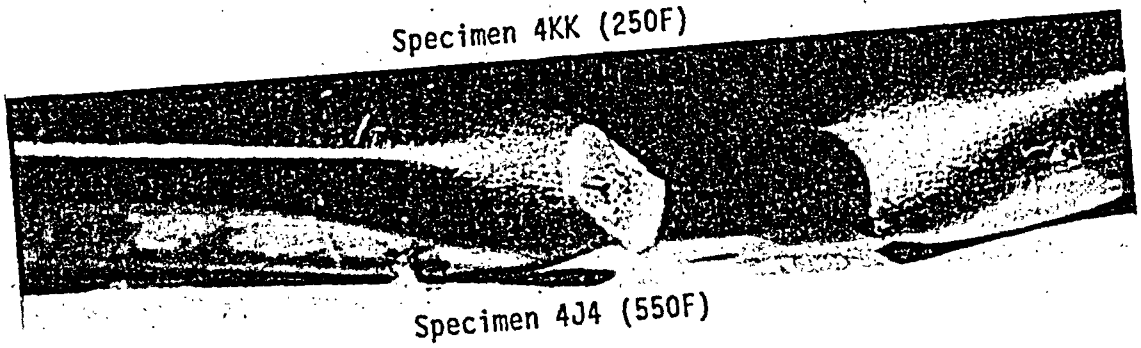
Figure 5-3. Photographs of Tested Tension Test Specimens and Corresponding Fractured Surfaces - Base Metal Heat-Affected Zone



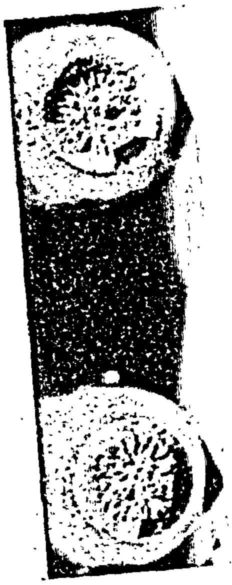
Specimen 4K3 (70F)

No Photograph

Specimen 4KK (250F)



Specimen 4J4 (550F)



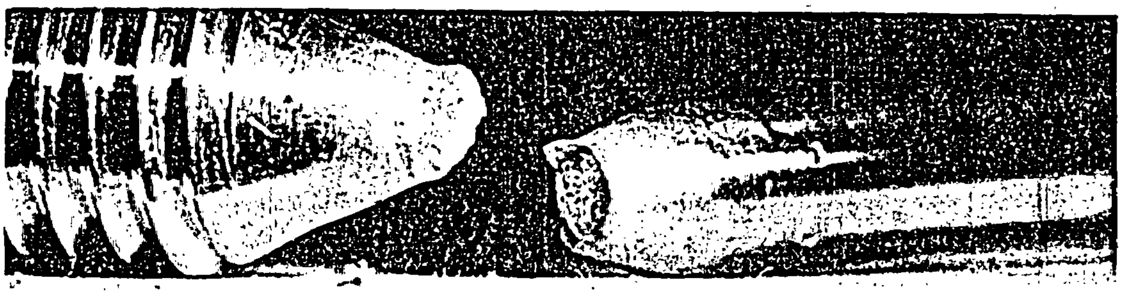
Specimen 4K3 (70F)

No Photograph

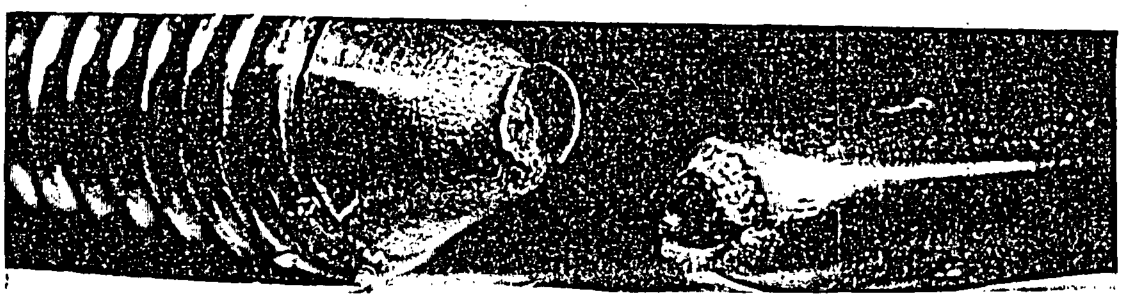
Specimen 4KK (250F)

Specimen 4J4 (550F)

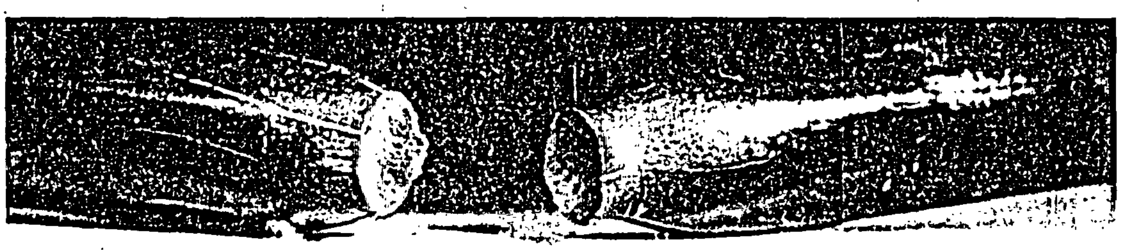
Figure 5-4. Photographs of Tested Tension Test Specimens and Corresponding Fractured Surfaces - Weld Metal 88114/0145



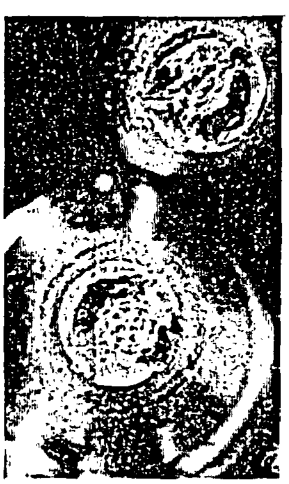
Specimen 3JM (70F)



Specimen 3KK (250F)



Specimen 3KY (550F)



Specimen 3JM (70F)

Specimen 3KK (250F)

Specimen 3KY (550F)

Figure 5-5. Charpy Impact Data for Irradiated Base Metal, Longitudinal Orientation, Heat No. M-1004-2

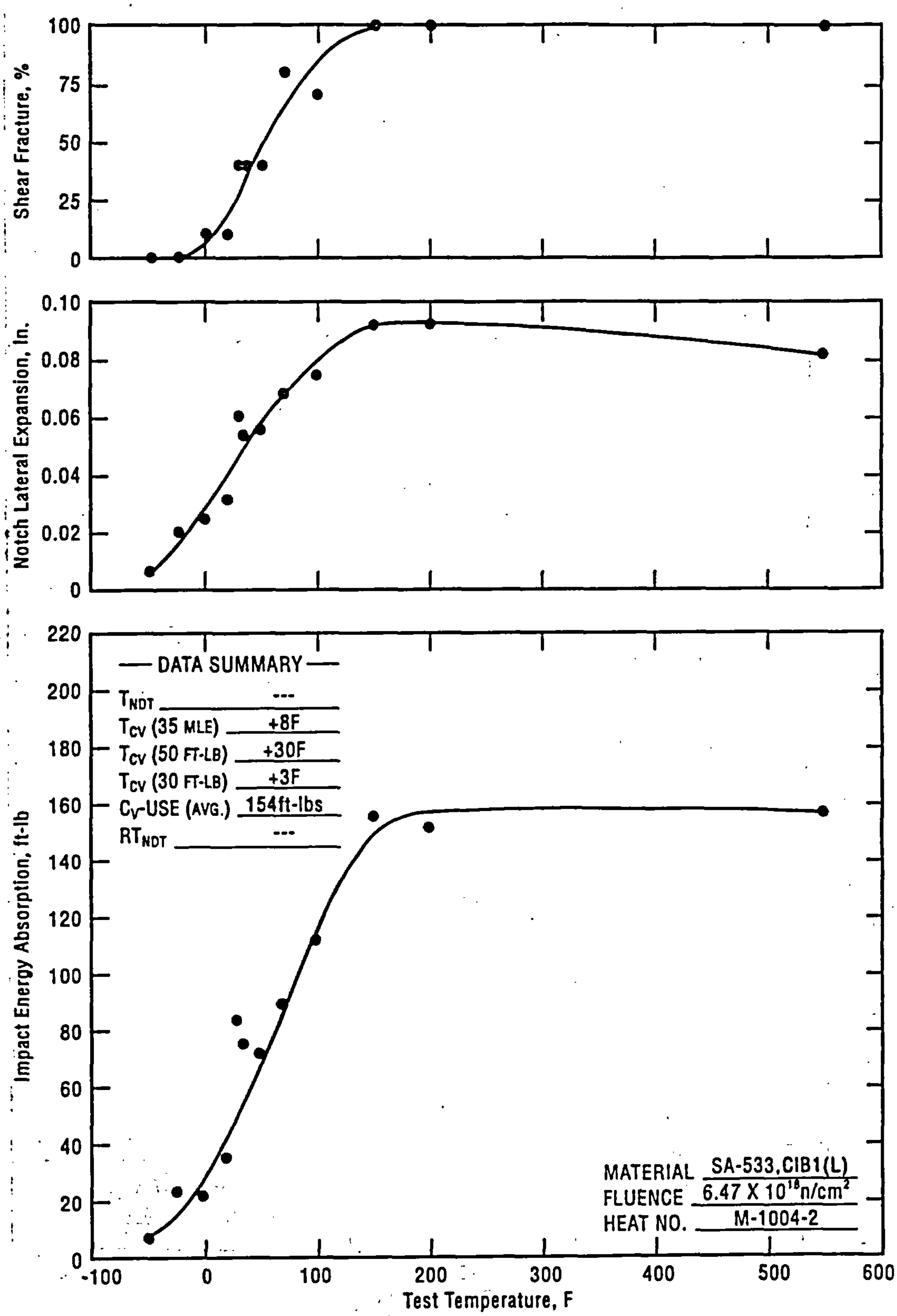


Figure 5-6. Charpy Impact Data for Irradiated Base Metal, Transverse Orientation, Heat No. M-1004-2

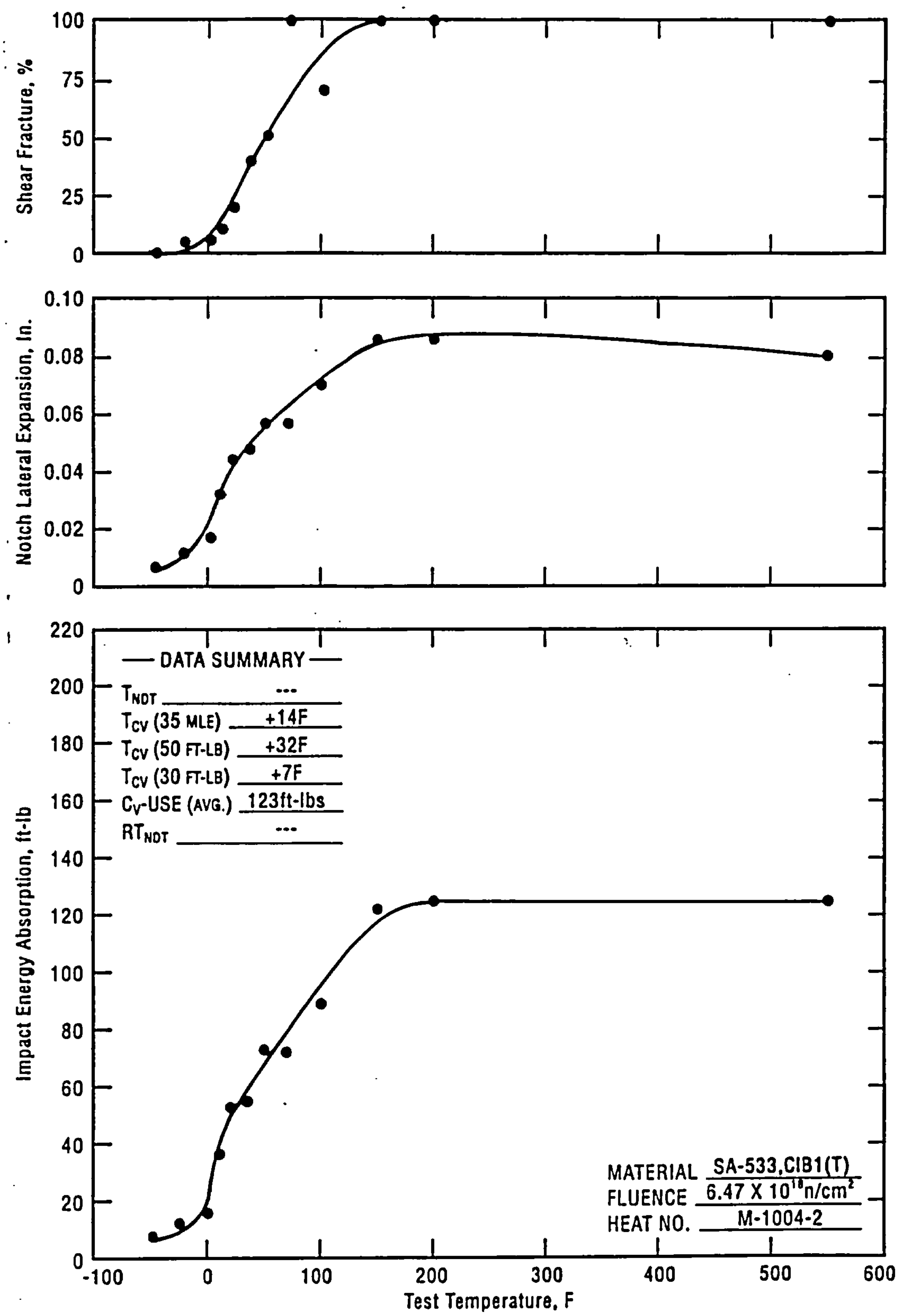


Figure 5-7. Charpy Impact Data for Irradiated Base Metal, Heat-Affected Zone, Heat No. M-1004-2

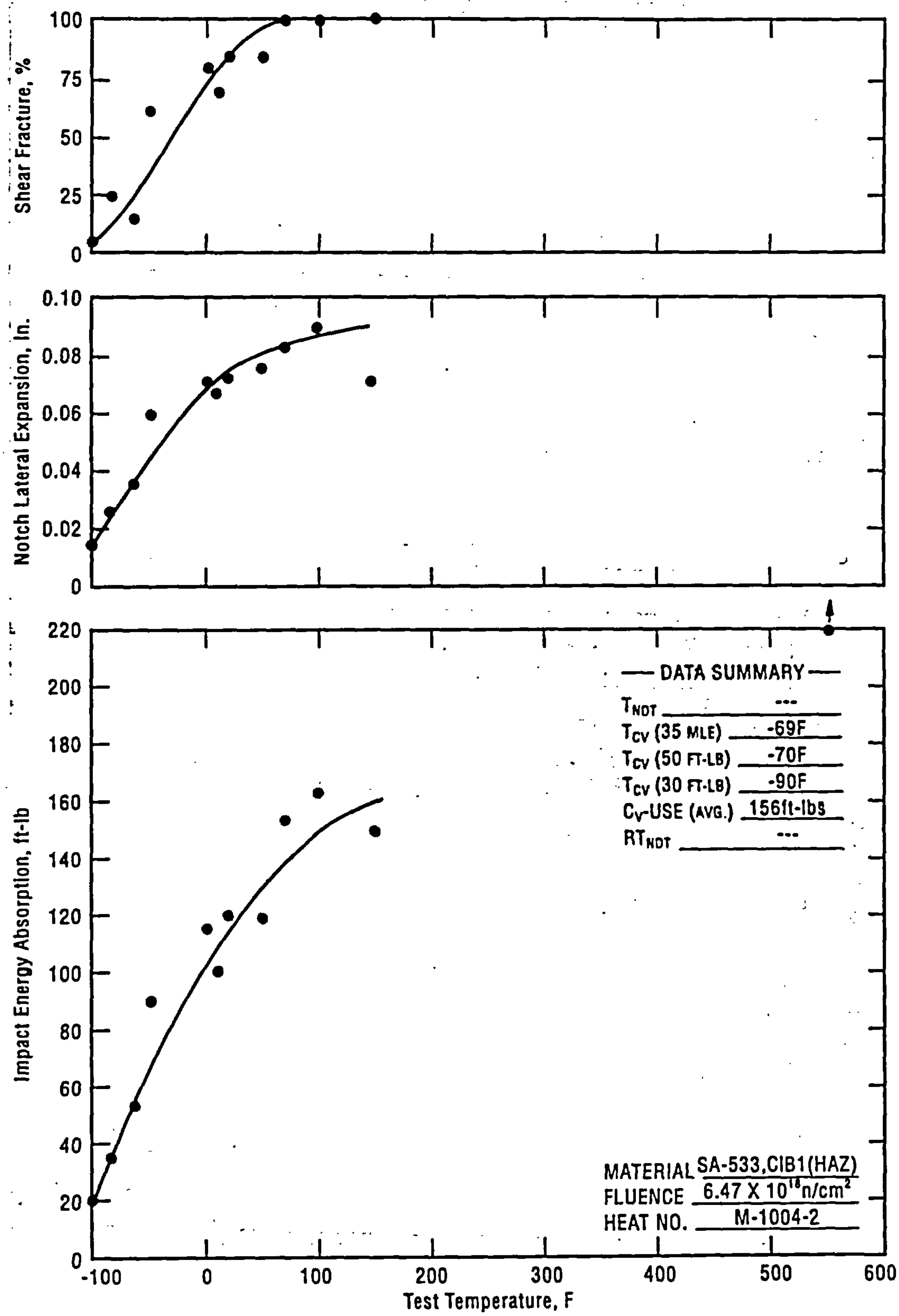


Figure 5-8. Charpy Impact Data for Irradiated Weld Metal, 88114/0145

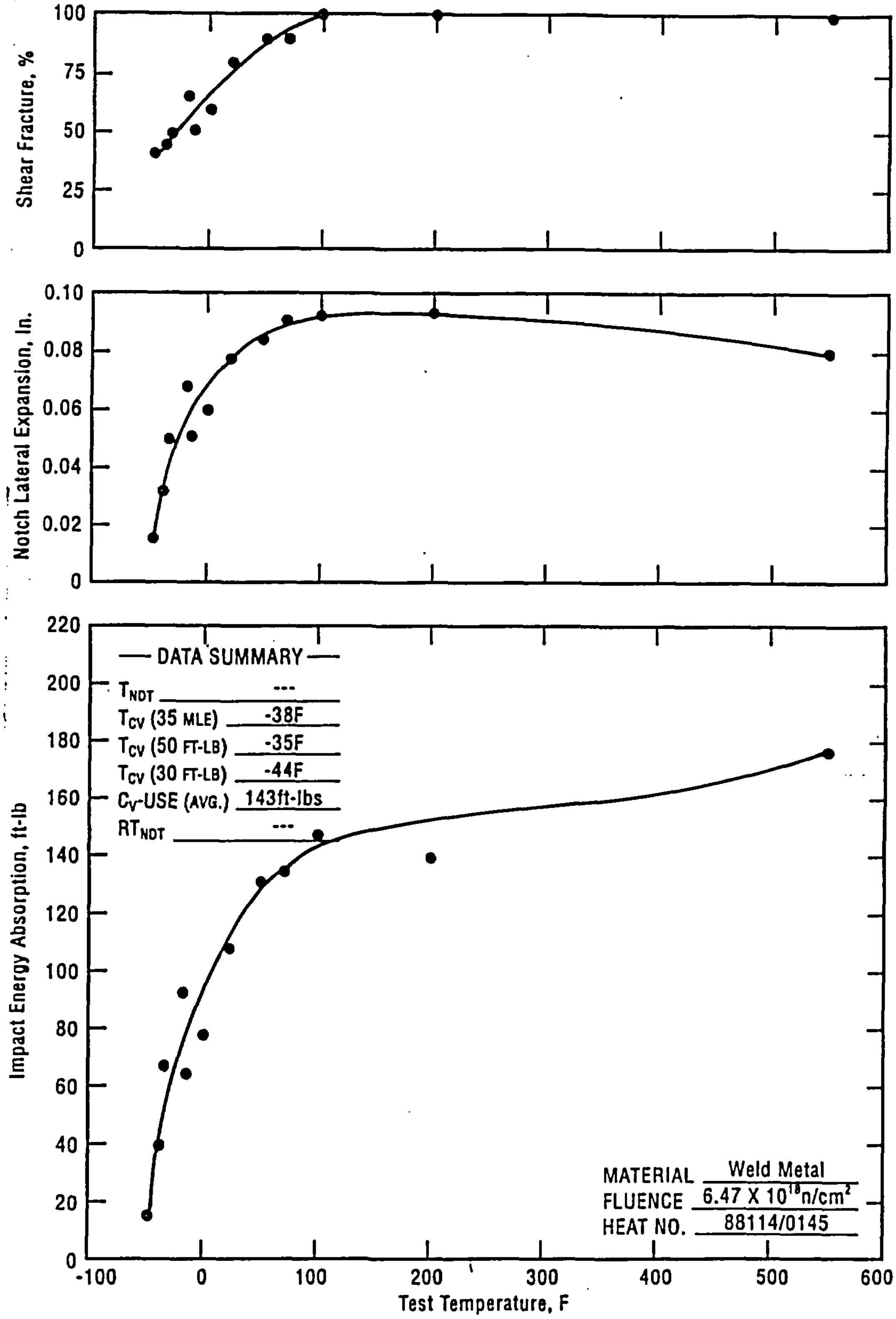


Figure 5-9. Photographs of Charpy Impact Specimen Fracture Surfaces - Base Metal, Longitudinal

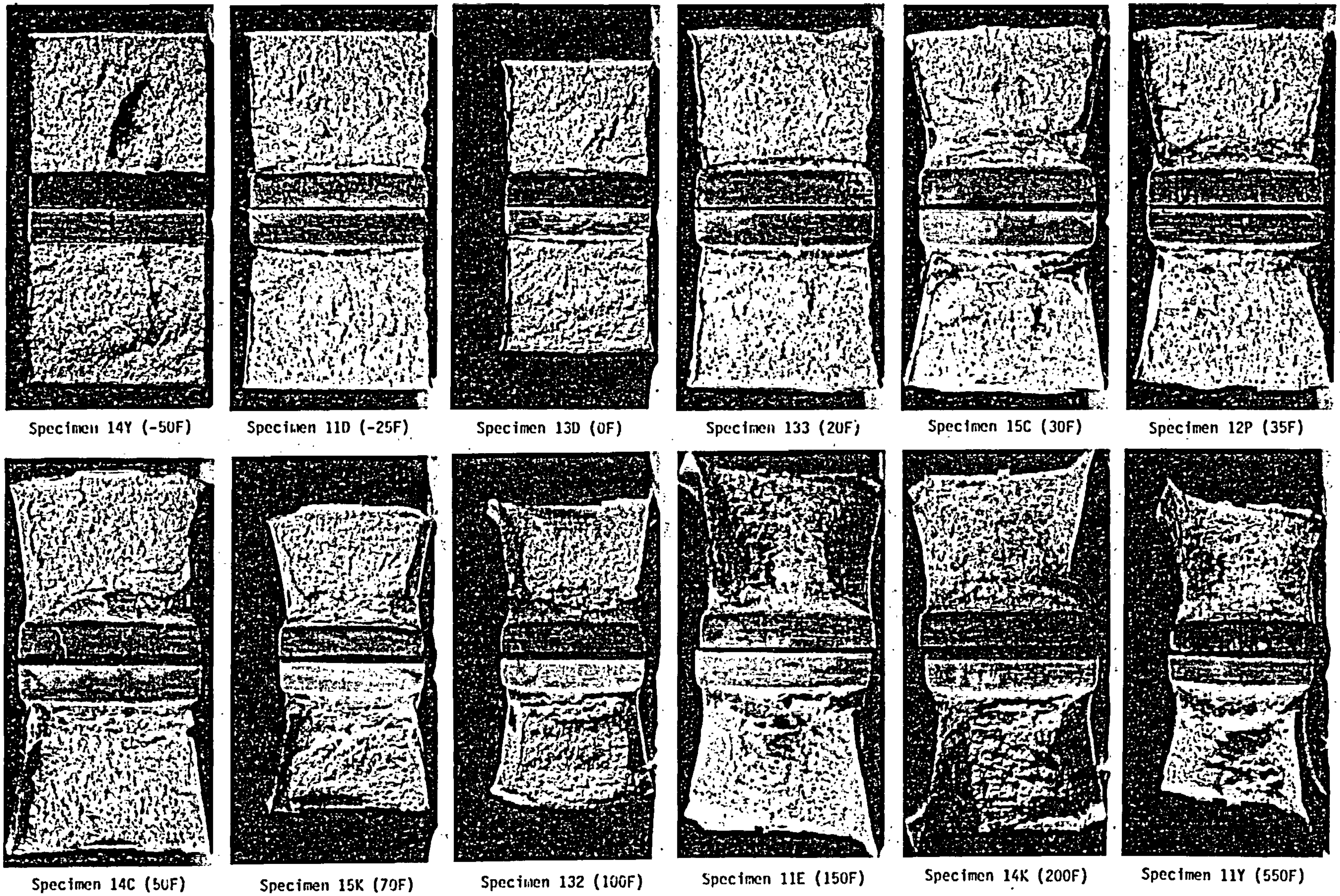
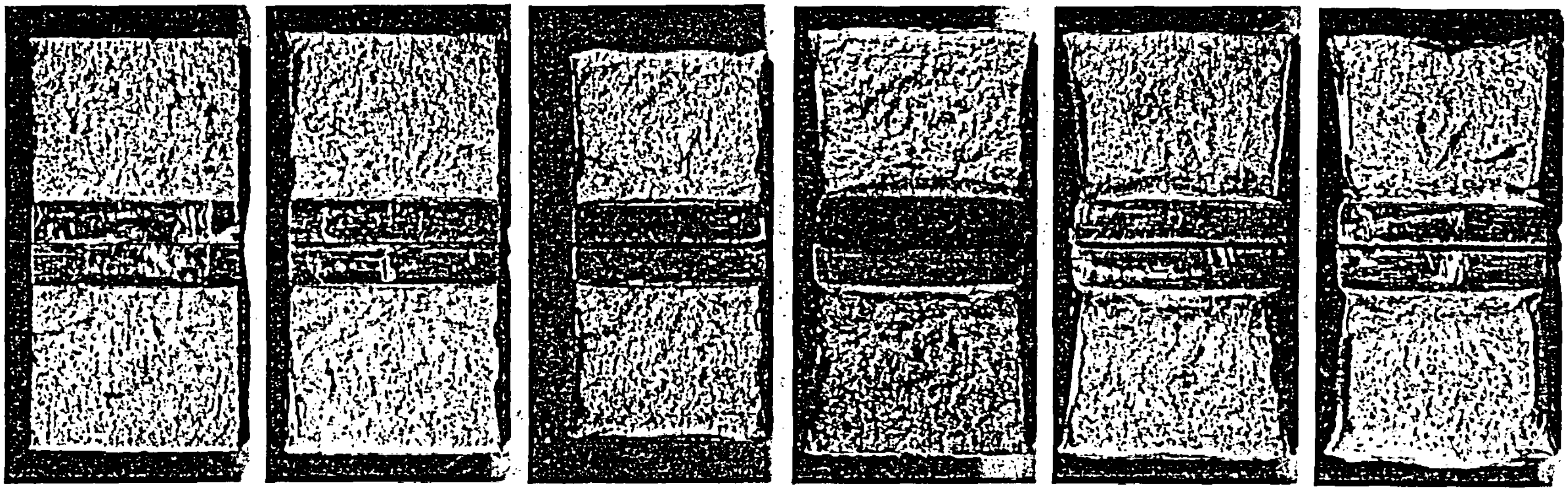
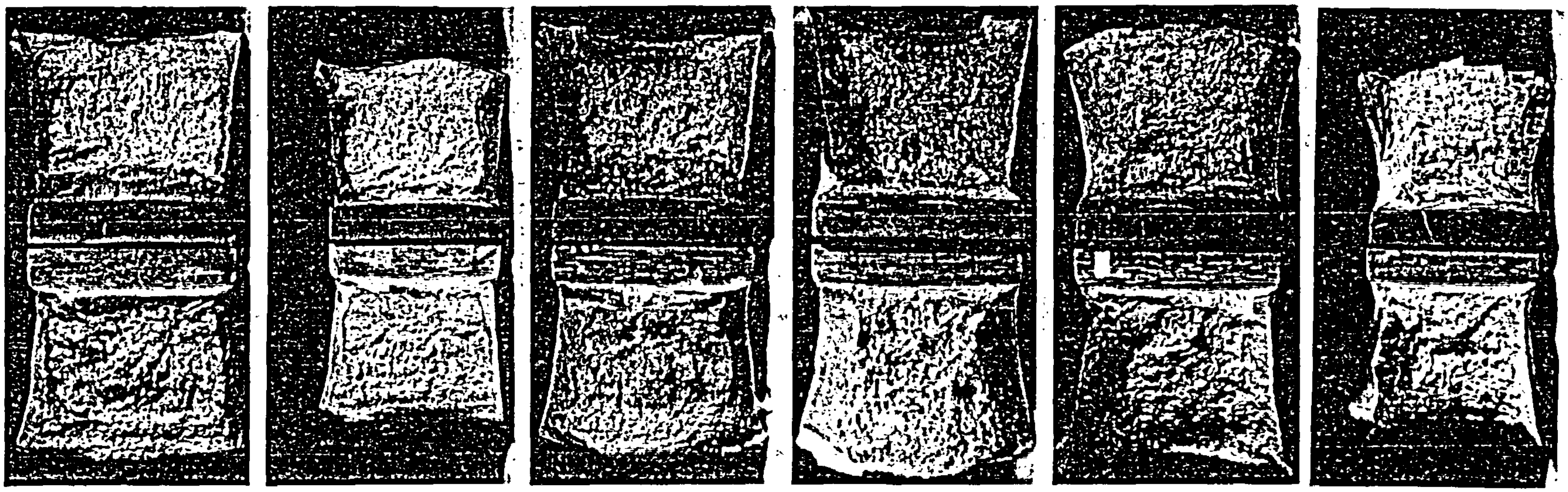


Figure 5-10. Photographs of Charpy Impact Specimen Fracture Surfaces - Base Metal, Transverse

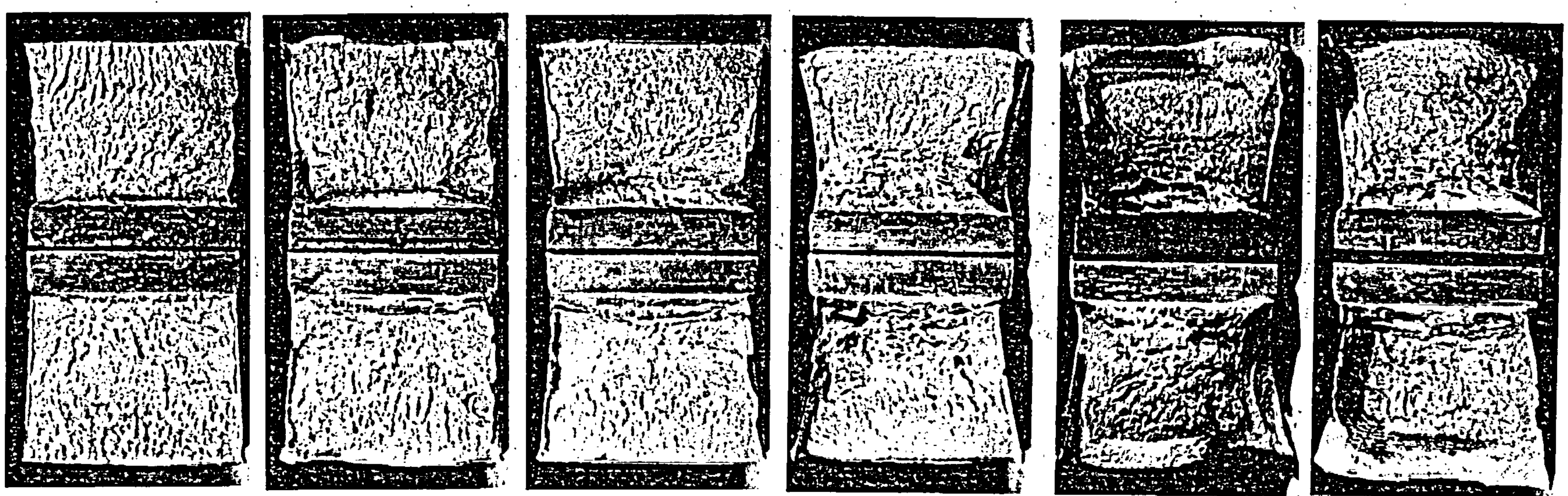


Specimen 216 (-50F) | Specimen 214 (-25F) | Specimen 25A (0F) | Specimen 21C (10F) | Specimen 22P (20F) | Specimen 26K (35F)

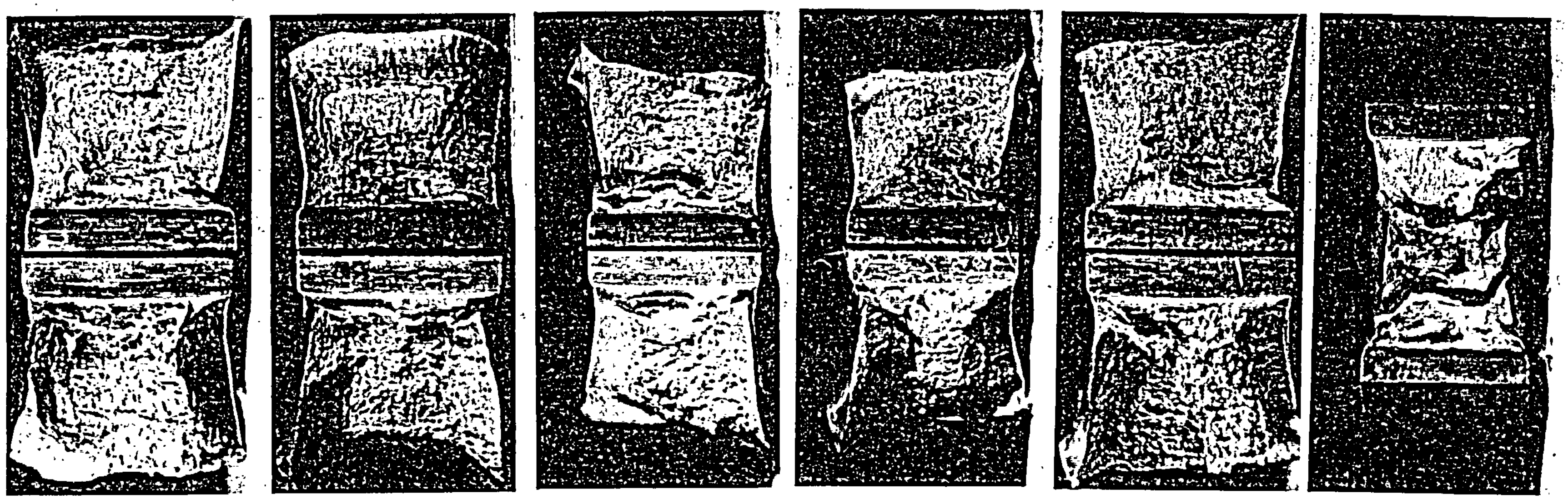


Specimen 25K (50F) | Specimen 217 70F) | Specimen 22L (100F) | Specimen 245 (150F) | Specimen 244 (200F) | Specimen 22H (550F)

Figure 5-11. Photographs of Charpy Impact Specimen Fracture Surfaces - Base Metal, Heat-Affected Zone

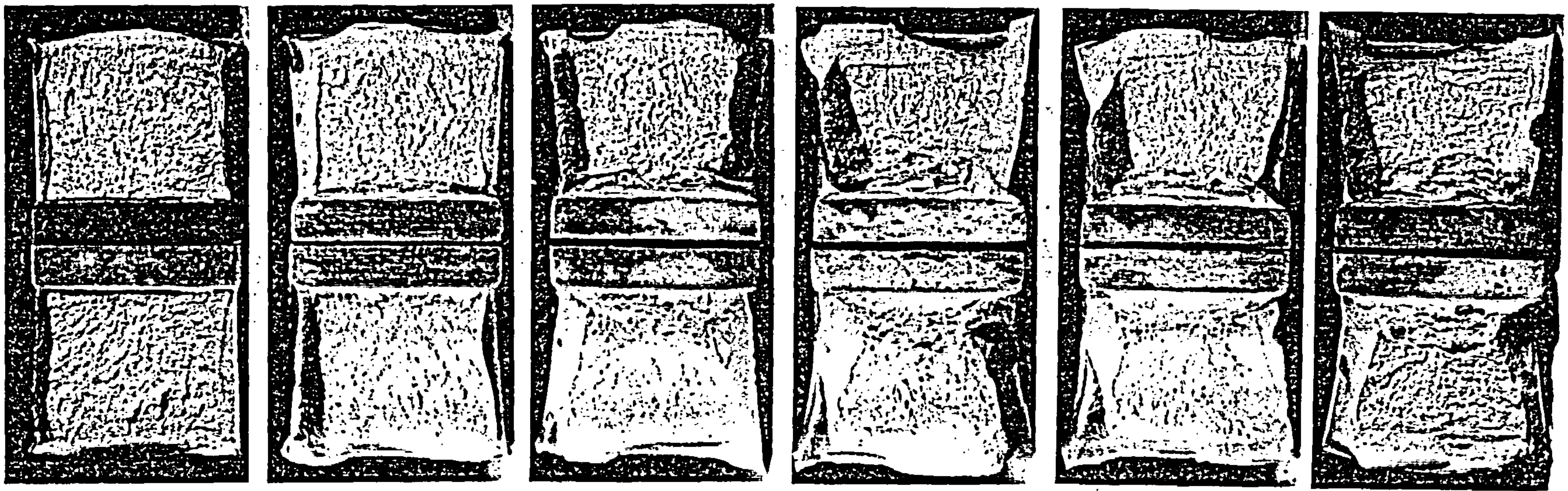


Specimen 415 (-100F) | Specimen 46Y (-85F) | Specimen 455 (-65F) | Specimen 45J (-50F) | Specimen 42C (0F) | Specimen 43D (10F)

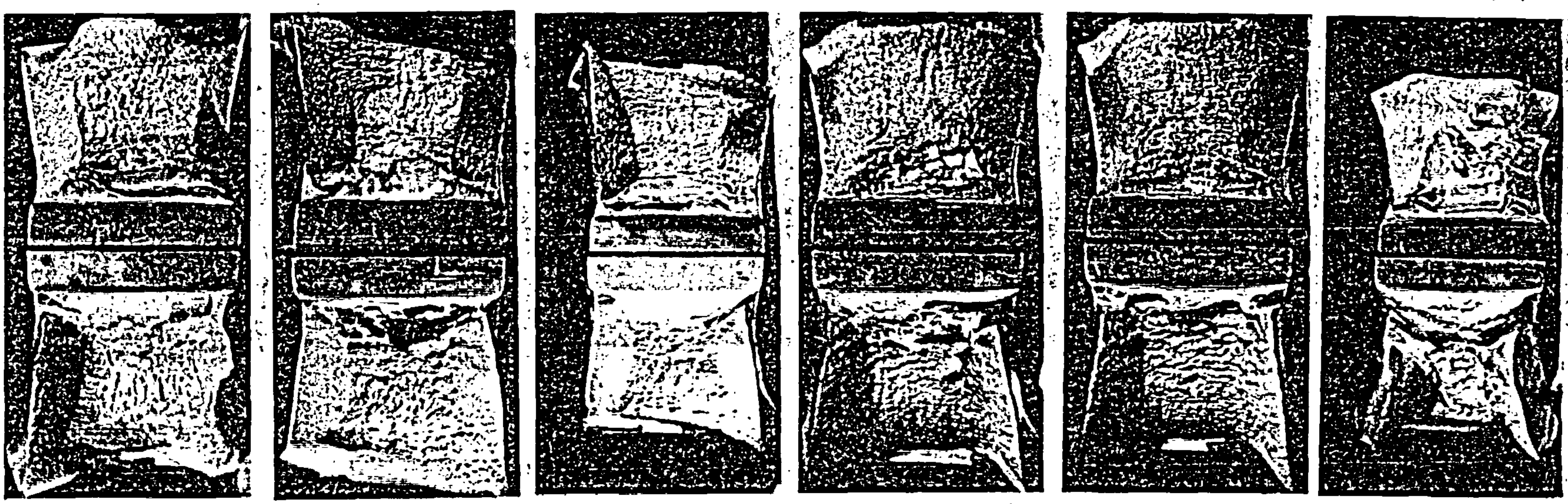


Specimen 44U (20F) | Specimen 45K (50F) | Specimen 45Y (70F) | Specimen 41M (100F) | Specimen 474 (150F) | Specimen 414 (550F)

Figure 5-12. Photographs of Charpy Impact Specimen Fracture Surfaces - Weld Metal 88114/0145



Specimen 35J (-50F) Specimen 32T (-40F) Specimen 31M (-35F) Specimen 32K (-20F) Specimen 362 (-15K) Specimen 34L (0K)



Specimen 31T (20F) Specimen 32G (50F) Specimen 35K (70F) Specimen 33C (100F) Specimen 374 (200F) Specimen 33B (550F)

6. NEUTRON FLUENCE

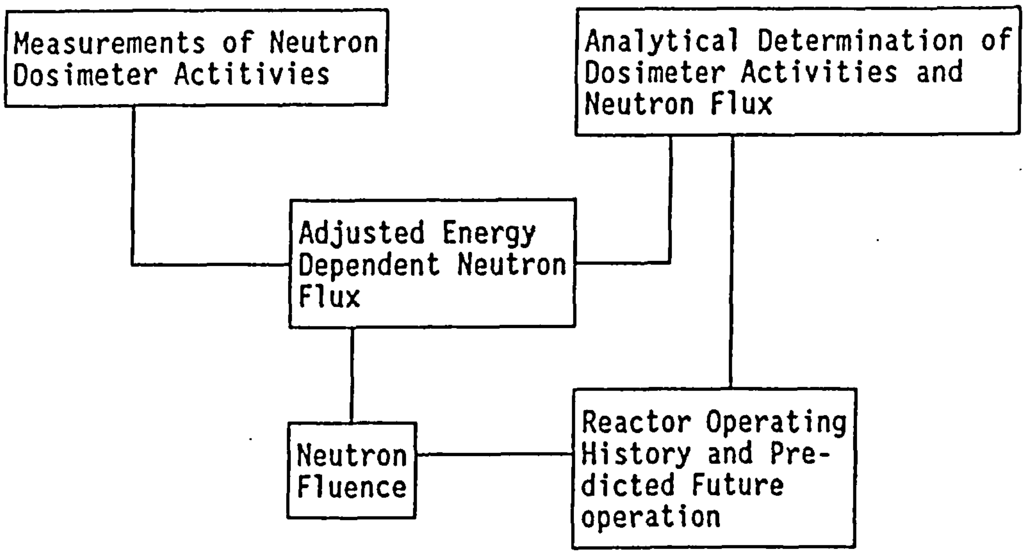
6.1. Introduction

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.[16]

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, DOTIV.[17] The calculations employ an angular quadrature of 48 sectors (S8), a third order LeGendre polynomial scattering approximation (P3), the BUGLE cross section set[18] with 47 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 DOTIV calculation determines the saturated specific activity of the various neutron dosimeters located in the surveillance capsule using the ENDF/B5 dosimeter reaction cross sections.[19] 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 made to account for the following effects:

- Photon induced fissions in U dosimeters (without this correction the results underestimate the measured activity).
- Short half-life of isotopes produced in nickel, iron and titanium dosimeters (71 day Co-58, 312 day Mn-54 and 84 day Sc-46 respectively). (Without this correction, the results could be biased high or low depending on the long term versus short term power histories.)

Measurement of Neutron Dosimeter Activities

The accuracy of neutron fluence predictions is improved if the calculated neutron flux is compared with neutron dosimeter measurements adjusted for the effects noted above. The neutron dosimeters located in the surveillance capsules are listed in Table 6-1. Both activation type and fission type dosimeters were used.

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 T (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
Fluence Monitors:	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 (DPA) of iron. 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).[20] 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. In the DPA calculations reported herein, the ASTM DPA cross sections were first collapsed to the 47 neutron group structure of BUGLE; the DPA was then determined by summing the group flux times the DPA cross section over the 47 energy groups and multiplying by the time of the irradiation.

6.2. Vessel Fluence

The maximum fluence (E > 1 MeV) exposure of the Waterford Unit 3 reactor vessel during Cycles 1 to 4 was determined to be 5.13×10^{18} n/cm^2 based on a maximum neutron flux of 3.66×10^{10} n/cm^2-s. The maximum fluence occurred at the cladding/vessel interface at an azimuthal location of approximately 1 degree from a major horizontal axis of the core (Figure 6-3). Cumulative DPA results were calculated at the quarter T positions and are presented in Table 6-4.

Fluence data were extrapolated to 32 EFPY of operation based on two assumptions: (1) the future fuel cycle operations do not differ significantly from the cycles 1 to 4 design, and (2) the latest calculated (or extrapolated) flux remains constant from EOC4 through 32 EFPY. The extrapolation was carried out from EOC 4 to 32 EFPY. The cycle averaged fluxes for future cycles are assumed to be the average flux experienced during cycles 1 to 4.

Fast fluence and DPA (displacements per atom) gradients relative to the inside surface of the vessel wall are 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 1.87, 4.03, and 9.17, respectively. DPA lead factors at the same locations are 1.62, 2.79, and 5.00, respectively. The relative fluence as a function of azimuthal angle is shown in Figure 6-3. The peak average flux from cycles 1 to 4 occurred at about 1 degree with a corresponding value of 3.66×10^{10} n/cm^2-s.

The flux and fluence results were corrected using the final measured to calculated activity ratio (M/C) derived from the capsule (0.958) and were also corrected to account for an axial power peak (1.08). The M/C ratio is detailed in Appendix D. The axial fluence, which was normalized over the height of the core and assumed to be proportional to the axial power distributions in the peripheral assemblies, was averaged over cycles 1 to 4. Table 6-7 shows the nodal values used to obtain the axial factors. These values were based on time-averaged nodal values obtained from the customer. Figure 6-4 shows the axial flux variation, overlaid by an image of the capsule showing the axial factors in each dosimeter compartment.

6.3. Capsule Fluence

The 97° capsule was irradiated in Waterford Unit 3 for Cycles 1 to 4, 4.44 EFPY, at a location 7 degrees off a major horizontal axis. The cumulative fast fluence at the center of the surveillance capsule was calculated to be 6.47×10^{18} n/cm^2. This fluence value represents an average value for the various locations in the capsule. It includes an axial peaking factor of 1.08 and a normalization factor of 0.958. The fluence is approximately 6% higher at the center of the charpy specimens closest to the core and approximately 6% lower at the center of the charpy specimens away from the core. Figure 6-5 shows a sketch of the capsule and pressure vessel, which includes the radial dimensions from the core center supplied by the customer, although the dimensions have been converted from the inches in which the information was supplied to the centimeters which were used in the modelling.

6.4. Fluence Uncertainties

Surveillance capsules provide neutron dosimetry information as well as materials data at various points during the lifetime of power reactors. The dosimetry results, measured-to-calculated ratios, obtained from numerous analyses utilizing the same methodology provide a measure of confidence in the analytical techniques and a benchmarking for the methodology used to determine vessel fluence. Table 6-6 presents a comparison of the results of fourteen surveillance capsule analyses which utilized B&W's methodology.

Table 6-1. Surveillance Capsule Dosimeters

Dosimeter Reactions[a]	Lower Energy Limit for Reaction, MeV	Isotope Half-Life
$^{58}Ni(n,p)^{58}Co$	2.3	70.8 days
$^{54}Fe(n,p)^{54}Mn$	2.5	312.5 days
$^{63}Cu(n,\alpha)^{60}Co$	3.7	5.27 years
$^{48}Ti(n,p)^{48}Sc$	1.9	83.81 days
$^{238}U(n,f)^{137}Cs$	1.1	30.0 years
$^{59}Co(n,\gamma)^{60}Co$	thermal	5.27 years

[a]Reaction activities measured for capsule flux evaluation.

Table 6-2. Waterford Unit 3 Reactor Vessel Fast Flux

Cycle	Fast Flux (E > 1 MeV), n/cm²-s Inside Surface** (Max Location)	T/4	3T/4	Flux n/cm²-s (E > 0.1 MeV) Inside Surface (Max Location)
Cycles 1 to 4	3.66E+10	1.96E+10	3.99E+9	7.91E+10
5 EFPY	3.66E+10	1.96E+10*	3.99E+9*	
6 EFPY	3.66E+10	1.96E+10*	3.99E+9*	
7 EFPY	3.66E+10	1.96E+10*	3.99E+9*	
8 EFPY	3.66E+10	1.96E+10*	3.99E+9*	
16 EFPY	3.66E+10	1.96E+10*	3.99E+9*	
24 EFPY	3.66E+10	1.96E+10*	3.99E+9*	
32 EFPY	3.66E+10	1.96E+10*	3.99E+9*	

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

**Clad/Base metal interface at 221.54 cm from core center.

Table 6-3. Calculated Waterford Unit 3 Reactor Vessel Fluence

Cummulative Irradiation Time	Fast Fluence, n/cm² (E > 1 MeV) Inside Surface[a] (Max Location)	T/4	T/2	3T/4
End of Cycle 4	5.13E+18	2.74E+18	1.27E+18	5.59E+17
5 EFPY	5.76E+18	3.08E+18*	1.43E+18*	6.29E+17*
6 EFPY	6.92E+18	3.70E+18*	1.72E+18*	7.54E+17*
7 EFPY	8.07E+18	4.32E+18*	2.00E+18*	8.80E+17*
8 EFPY	9.22E+18	4.93E+18*	2.29E+18*	1.01E+18*
16 EFPY	1.84E+19	9.86E+18*	4.58E+18*	2.01E+18*
24 EFPY	2.77E+19	1.48E+19*	6.87E+18*	3.02E+18*
32 EFPY	3.69E+19	1.97E+19*	9.15E+18*	4.02E+18*
*Calculated using these lead factors.	1.00	1.87	4.03	9.17

[a]Clad/Base metal interface at 221.54 cm from core center.

Conversion Factors

Fluence (E > 1 MeV) to DPA.	1.50E-21**	1.72E-21**	2.16E-21**	2.73E-21**

**Multiply fast fluence values (E > 1 MeV) in units of n/cm² by these factors to obtain the corresponding DPA values.

Table 6-4. Calculated Waterford Unit 3 Reactor Vessel DPA

Cummulative Irradiation Time	DPA, Displacements/Atom (Total) Inside Surface[a] (Max Location)	T/4	T/2	3T/4
End of Cycle 4	7.67E-3*	4.73E-3*	2.76E-3*	1.53E-3*
5 EFPY	8.62E-3*	5.31E-3*	3.10E-3*	1.72E-3*
6 EFPY	1.03E-2*	6.38E-3*	3.72E-3*	2.06E-3*
7 EFPY	1.21E-2*	7.44E-3*	4.33E-3*	2.41E-3*
8 EFPY	1.38E-2*	8.50E-3*	4.95E-3*	2.75E-3*
16 EFPY	2.76E-2*	1.70E-2*	9.91E-3*	5.50E-3*
24 EFPY	4.14E-2*	2.55E-2*	1.49E-2*	8.25E-3*
32 EFPY	5.51E-2*	3.40E-2*	1.98E-2*	1.10E-2*

[a]Clad/Base metal interface at 221.54 cm from core center.

*Calculated using these lead factors	1.00	1.62	2.79	5.00
Conversion Factors				
Fluence (E > 1 MeV) to DPA.	1.50E-21**	1.72E-21**	2.16E-21**	2.73E-21**

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

Table 6-5. Fluence, Flux, and DPA for 97° Surveillance Capsule

Capsule	Irradiation Time	E > 1.0 MeV Flux, n/cm^2	E > 1.0 MeV Fluence, n/cm^2	E > 1.0 MeV DPA	E > 0.1 MeV Flux, n/cm^2
W-97	Cycles 1 to 4 (4.44 EFPY)	4.62E+10	6.47E+18	9.25E-3	8.63E+10

Table 6-6. Surveillance Capsule Measurements

Plant	Capsule	Measured/ Calculated
Arkansas One, Unit 1	AN1-C	1.04
Rancho Seco	RS1-F	1.03
Crystal River-3	CR3-F	0.99
Oconee Unit 1	OC1-C	1.01
Oconee Unit 2	OC2-E	0.98
Davis-Besse	DB1-LG1	1.08
Crystal River-3	CR3-LG1	1.06
Oconee Unit 3	OC3-D	1.00
Davis-Besse	TE1-D	1.03
St. Lucie	W-83	1.08
Shearon Harris	U	0.88
Zion Unit 1	Y	1.11
Millstone Unit 2	W-104	0.99
Millstone Unit 2	W-97	0.94

Average M/C for 14 surveillance data points = 1.02

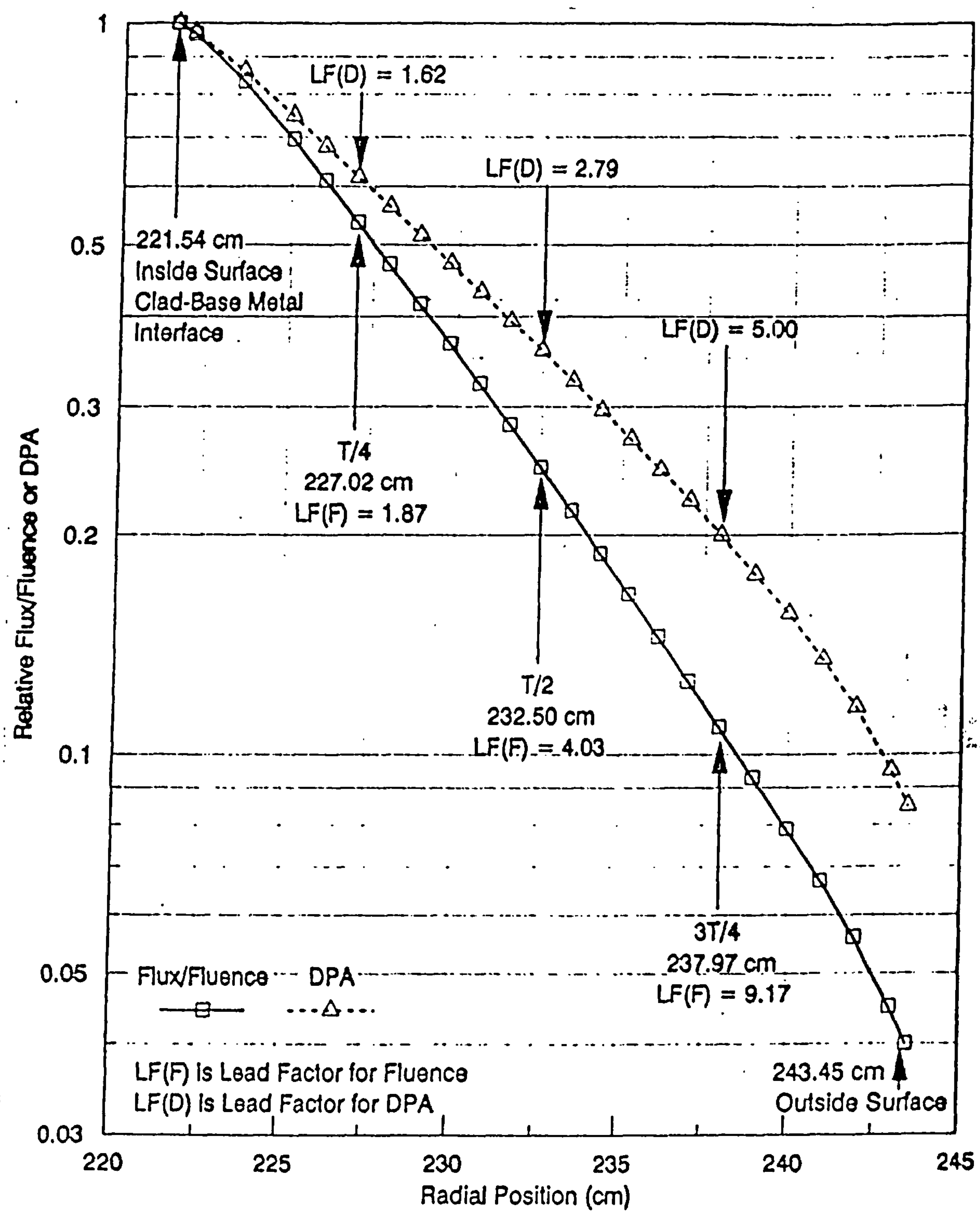
1 Sigma standard deviation of data base = 0.06

Table 6-7. Axial Power Data Affecting Flux

Node	Exposure*	Relative Exp.
1	3.272	0.750
2	4.360	1.000
3	4.666	1.070
4	4.731	1.085
5	4.726	1.084
6	4.704	1.079
7	4.679	1.073
8	4.649	1.066
9	4.602	1.055
10	4.491	1.030
11	4.179	0.958
12	3.269	0.750
Avg	4.361	

*These exposure values are based upon the nodal values for assemblies [9,1] and [9,2] supplied by the customer and time-averaged for cycles 1 through 4.

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



B&W NUCLEAR SERVICE COMPANY

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

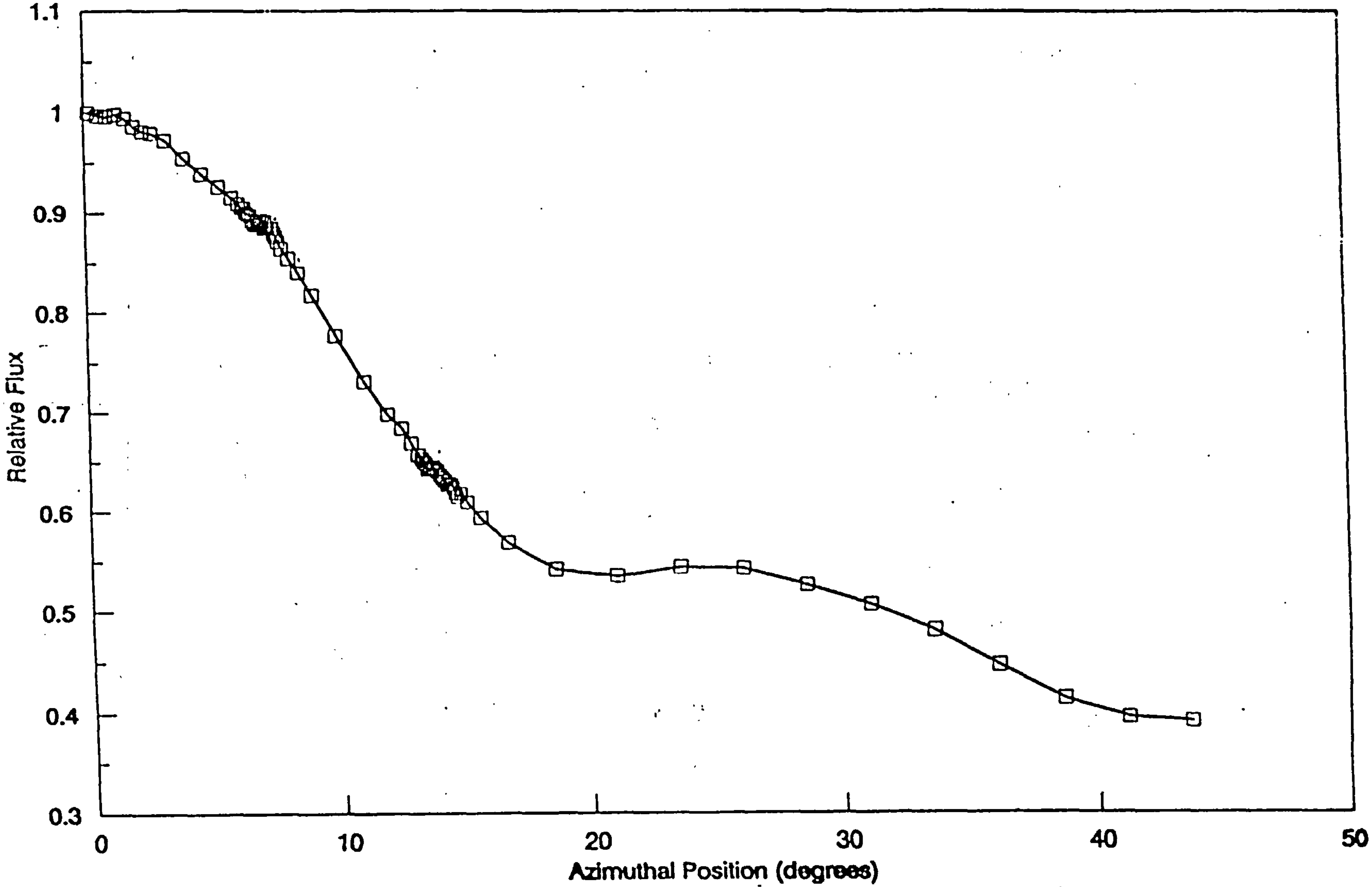


Figure 6-4. Relative Axial Variation of E > 1 MeV Flux/Fluence

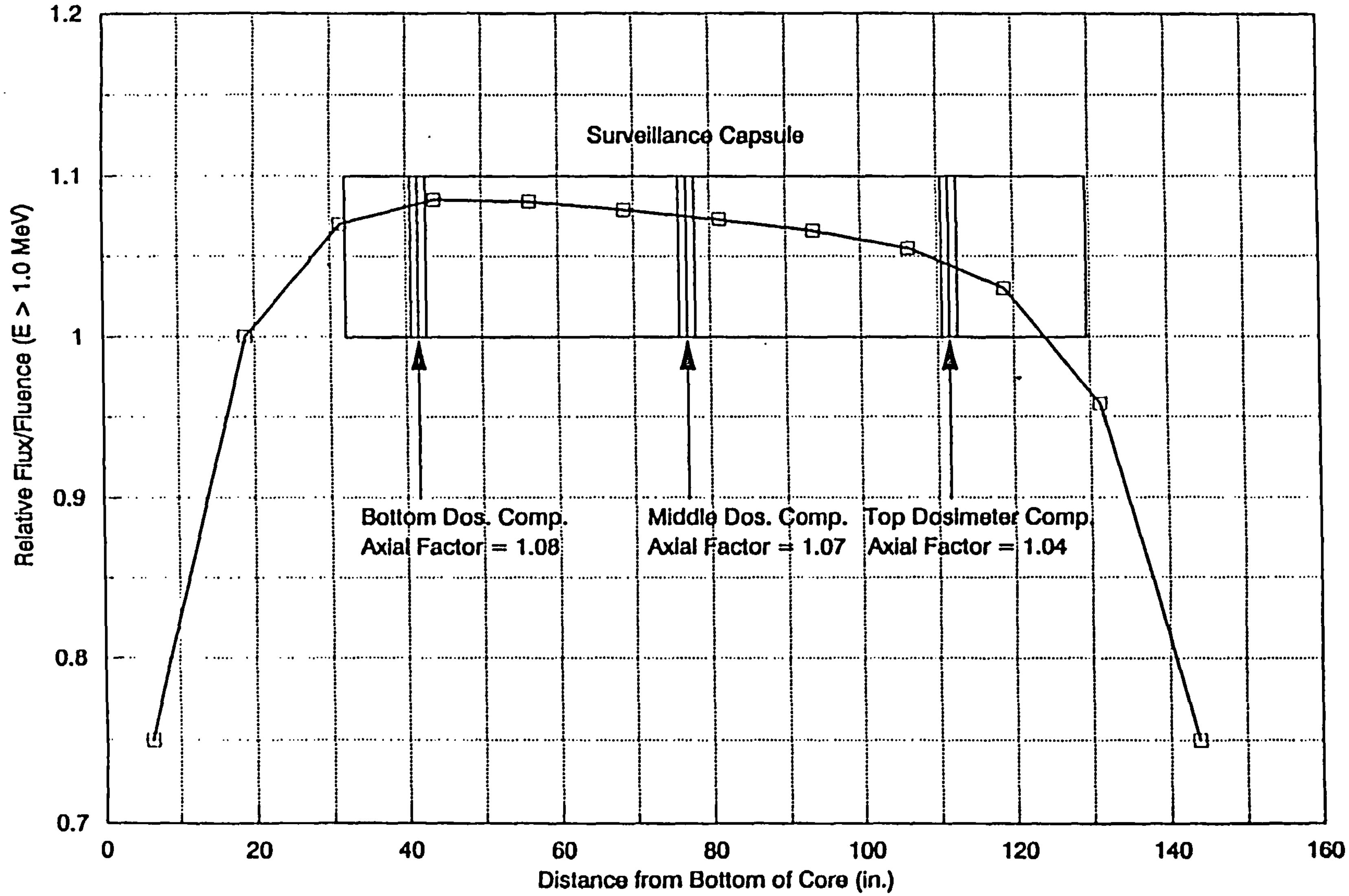
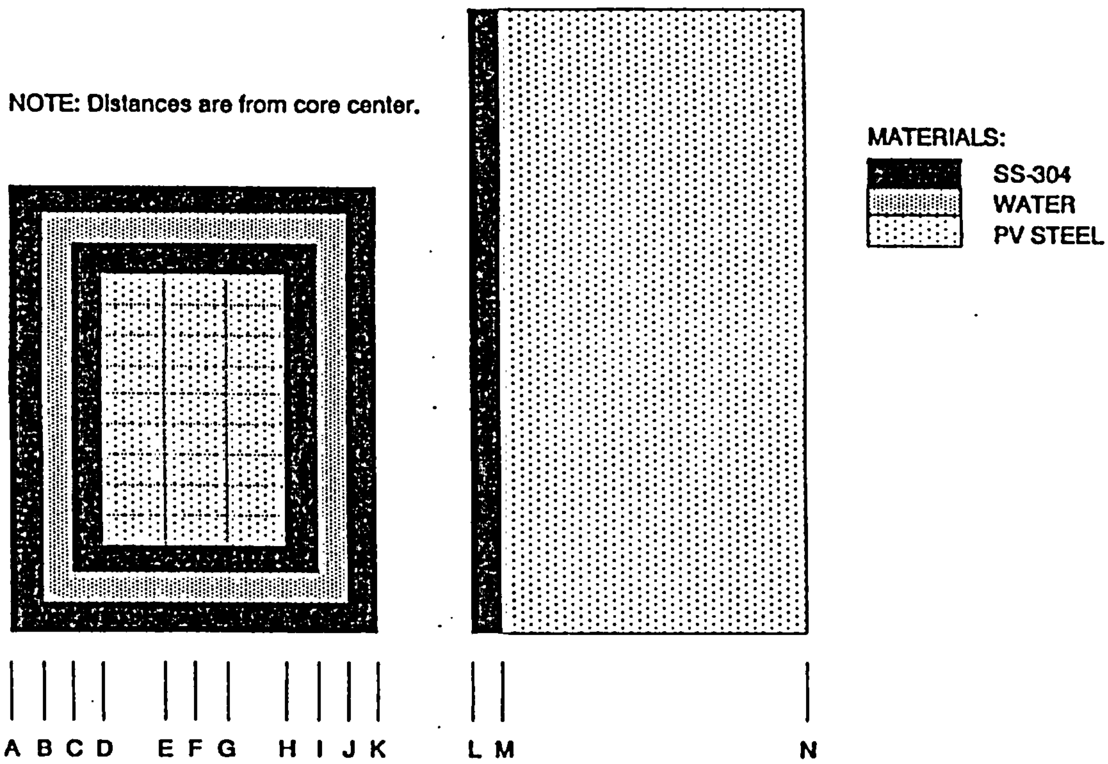


Figure 6-5. Radial Dimensions Used in Modeling Capsule and Pressure Vessel Regions



POSITION	DISTANCE(cm)	POSITION	DISTANCE(cm)	POSITION	DISTANCE(cm)
A	215.31	F	217.27	K	219.22
B	215.67	G	217.50	L	220.98
C	216.32	H	217.95	M	221.54
D	216.59	I	218.22	N	243.45
E	217.04	J	218.87		

Page Intentionally Left Blank

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 Waterford Unit 3. 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. Based on the design end-of-service peak neutron fluence value at the 1/4T vessel wall location and the copper content of the base metals, it was predicted that the end-of-service Charpy upper-shelf energy (USE) will not be below 50 ft-lb.

7.2. Irradiated Property Data

7.2.1. Tensile Properties

Tables 7-1 and 7-2 compare irradiated properties from Capsule W-97 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. The changes observed in the base metal are such as to be considered within acceptable limits. The changes, at both room temperature and 550F, in the properties of the base metal are equivalent to those observed for the weld metal, indicating a similar sensitivity of both the base metal and the weld 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.

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 slightly 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. A comparison of the Charpy data curves are presented in Figures 7-1 through 7-4.

The 30 ft-lb transition temperature shift for the base metal in the longitudinal orientation is conservative compared to the value predicted using Regulatory Guide 1.99, Rev. 2[21] and when the margin is added the predicted value is very conservative. However, the 30 ft-lb transition temperature shift in the transverse orientation is not in good agreement with the predicted using Regulatory Guide 1.99, Rev. 2, and when the margin is added the predicted value equals the measured value without any margin for conservatism. It would be expected that these values for the longitudinal orientation would exhibit good agreement when it is considered that the data used to develop Regulatory Guide 1.99, Rev. 2, was taken from data obtained from longitudinal oriented specimens.

The transition temperature measurements at 30 ft-lbs for the weld metal is in relatively good agreement with the predicted shift using Regulatory Guide 1.99, Revision 2 but the predicted value is not conservative. The predicted shift being slightly under estimated indicates that the estimating technique based on the Regulatory Guide 1.99, Rev. 2, is not overly conservative for predicting the 30 ft-lb transition temperature shift. Since the method requires that a margin be added to the calculated value to provide a conservative value, the final shift value using Regulatory Guide 1.99, Revision 2, is conservative, and future evaluations should be based on Position 2 when additional data are available which will help to account for some of the over-conservatism in the application of Regulatory Guide 1.99, Position 1.

The data for the decrease in Charpy USE due to irradiation showed relatively good agreement with predicted values for the base metal. The weld metal decrease in Charpy USE was over predicted by 200 percent. 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 low copper-content materials at medium fluence values that were used to develop the estimating curves.

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 were 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 poor agreement of the change in Charpy upper-shelf energy for the weld metal does support the conservatism of the prediction curves for low copper-content materials. However, for low copper-content base materials the predicted values are not conservative. Although the prediction curves are conservative for the weld metal 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 was made and the results are presented in Table 7-4.

The fracture toughness evaluation shows that the controlling base metal will have a T/4 wall location end-of-life RT_{NDT} of 69F based on Regulatory Guide 1.99, Revision 2, including a margin of 24F. The controlling weld metal will have a T/4 wall location end-of-life RT_{NDT} of 34F based on Regulatory Guide 1.99, Revision 2, including a margin of 52F. These predicted shifts may be excessive since data from the first surveillance capsule exhibited measured RT_{NDT} values that are comparable to the Regulatory Guide mean values. It is estimated that the end-of-life RT_{NDT} shift for both the controlling base metal and weld metal will be significantly less than the value predicted using Regulatory Guide 1.99, Revision 2 because the use of future surveillance data will permit a reduction in the applied margin. 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 reactor vessel fabrication was made and the results are presented in Table 7-5. This evaluation was made because the base metals used to fabricate the reactor vessel are characterized by upper-shelf energies measured only in the longitudinal orientation. Consequently, when adjusted for the transverse orientation are expected to be sensitive to neutron radiation damages and exhibit values significantly lower than the longitudinal value. The method used to evaluate the radiation induced decrease in upper-shelf energy is the method defined in Regulatory Guide 1.99, Revision 2, which is the same procedure used in Revision 1.

The method of Regulatory Guide 1.99, Revision 2, shows that the base metals used in the fabrication of the beltline region of the reactor vessel will have an upper-shelf energy greater than 50 ft-lbs through the 32 EFPY design life based on the T/4 wall location. Regulatory Guide 1.99 method also predicts an upper-shelf energy above 50 ft-lbs for the controlling base metal at the vessel inside wall. The weld metal upper-shelf energies unirradiated values are so high as to preclude any change of the values decreasing below 50 ft-lbs during the 32 EFPY design life. Based on the first surveillance capsule data, it is estimated that the controlling vessel base metal upper-shelf energy will remain above the required 50 ft-lbs during the vessel design life.

7.4. Operating Limitations

The current normal pressure-temperature operating limitations are designed for operation through 8 EFPY. Based on the fluence calculations performed for Capsule W-97 and the results of the Charpy impact test results, the current operating limitations may be extended to 10.5 EFPY. However, any changes must be verified by confirmatory calculations and, in addition, any changes in the fuel cycle designs will require a review and possible verification for extension from the original 8 EFPY limit.

7.5. Pressurized Thermal Shock (PTS) Evaluation

The pressurized thermal shock evaluation shown in Table 7-6 demonstrates that the Waterford Unit 3 reactor pressure vessel is well below the screening criterion limits and, therefore, need not take any additional corrective action as required by the regulation.

7.6. Neutron Fluence Analysis

These new analyses calculated an end-of-life fluence value of 3.69×10^{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 1.97×10^{19} n/cm^2 (E > 1 MeV). These values do not represent a reduction compared to the values calculated based on the design basis fluence values.

Table 7-1. Comparison of Waterford Unit 3, Capsule W-97 Tension Test Results

	Room Temp Test		Elevated Temp Test*	
	Unirr**	Irrad	Unirr**	Irrad
Base Metal -- M-1004-2, Transverse				
Fluence, 10^{18} n/cm² (E > 1 MeV)	0	6.47	0	6.47
Ultimate tensile strength, ksi	89.0	92.6	87.0	90.0
0.2% yield strength, ksi	68.1	70.4	64.5	63.5
Uniform elongation, %	11.0	11.7	9.9	10.2
Total elongation, %	27.3	26.2	22.3	23.0
Reduction of area, %	68.2	63.5	65.3	62.5
Base Metal -- Heat-Affected Zone				
Fluence, 10^{18} n/cm² (E > 1 MeV)	0	6.47	0	6.47
Ultimate tensile strength, ksi	91.3	93.5	86.7	91.0
0.2% yield strength, ksi	68.2	69.5	60.2	69.6
Uniform elongation, %	6.8	7.0	6.6	6.4
Total elongation, %	21.3	20.3	20.3	18.5
Reduction of area, %	69.4	68.9	66.6	69.5
Weld Metal -- 88114/0145				
Fluence, 10^{18} n/cm² (E > 1 MeV)	0	6.47	0	6.47
Ultimate tensile strength, ksi	92.2	95.9	88.3	93.2
0.2% yield strength, ksi	81.0	84.5	72.2	74.0
Uniform elongation, %	9.6	7.3	9.2	7.9
Total elongation, %	27.7	--***	23.7	22.6
Reduction of area, %	70.7	63.5	69.4	70.0

*Test temperature is 550F.
**Average of the lower yield strength data in Appendix B.
***See footnote Table 5-2.

Table 7-2. Summary of Waterford Unit 3 Reactor Vessel Surveillance Capsule Tensile Test Results

Material	Cap. I.D.	Fluence, 10^{18} n/cm^2	Test Temp, F	Strength, ksi: Ultimate	Strength, ksi: Δ%[a]	Strength, ksi: Yield	Strength, ksi: Δ%[a]	Ductility, %: Total Elon.	Ductility, %: Δ%[a]	Ductility, %: Reduction of Area	Ductility, %: Δ%[a]
Base metal Transverse (M-1004-2)	--	0.00	71	89.0	--	68.1	--	27.3	--	68.2	--
			550	87.0	--	64.5	--	22.3	--	65.3	--
	W-97	6.47	70	92.6	+ 4	70.4	+ 3	26.2	- 4	63.5	- 7
			550	90.0	+ 3	63.5	- 2	23.0	+ 3	62.5	- 4
Base metal Heat-affected zone (M-1004-2)	--	0.00	71	91.3	--	68.2	--	21.3	--	69.4	--
			550	86.7	--	60.2	--	20.3	--	66.6	--
	W-97	6.47	70	93.5	+ 2	69.5	+ 2	20.3	- 5	68.9	- 1
			550	91.0	+ 5	69.6	+16	18.5	- 9	69.5	+ 4
Weld metal (88114/0145)	--	0.00	71	92.2	--	81.0	--	27.7	--	70.7	--
			550	88.3	--	72.2	--	23.7	--	69.4	--
	W-97	6.47	70	95.9	+ 4	84.5	+ 4	--[b]	--	63.5	-10
			550	93.2	+ 6	74.0	+ 2	22.6	- 5	70.0	+ 1

[a] Change relative to unirradiated.

[b] See footnote Table 5-2.

Table 7-3. Observed Vs. Predicted Changes for Capsule W-97 Irradiated Charpy Impact Properties - 6.47 x 10^{18} n/cm^2 (E > 1 MeV)

Material	Observed Unirrad.	Observed Irrad.	Observed Diff.	Difference Predicted Per R.G. 1.99/2 Without Margin[(a)]	Difference Predicted Per R.G. 1.99/2 With Margin[(b)]
Increase in 30 ft-lb Trans. Temp., F					
Base Material (M-1004-2)					
Longitudinal	0	+ 3	+ 3	18	35
Transverse	- 29	+ 7	+36	18	35
Heat-Affected Zone (M-1004-2)	-106	-90	+16	18	35
Weld Metal (88114/0145)	- 80	-44	+36	39	78
Decrease in Charpy USE, ft-lb					
Base Material (M-1004-2)					
Longitudinal	170	154	-16	N.A.	19[(c)]
Transverse	141	123	-18	N.A.	16[(c)]
Heat-Affected Zone (M-1004-2)	170	156	-14	N.A.	19[(c)]
Weld Metal (88114/0145)	156	143	-13	N.A.	26[(c)]

(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.

(c) Bounding value per Regulatory Guide 1.99, Revision 2, May 1988 (includes margin).

N.A. - Not applicable.

Table 7-4. Evaluation of Reactor Vessel End-of-Life (32 EFPY) Fracture Toughness - Waterford Unit 3

Material Description				Material Chemical Composition, w/o[c]		Estimated EOL Fluence[b]		End-of-Life RT_{NDT}, F[a]		
Fab. Mat'l. Code	Reactor Vessel Beltline Location	Heat Number[e]	Type	Copper	Nickel	Inside Surface n/cm^2	T/4 Wall Location n/cm^2	Initial RT_{NDT}[e]	Inside Surface	T/4 Wall Location
M-1003-1	Intermed. Shell	56488-1	SA533, Gr. B	0.02	0.71	3.64E+19	1.95E+19	-30	+23	+17
M-1003-2	Intermed. Shell	56512-1	SA533, Gr. B	0.02	0.67	3.64E+19	1.95E+19	-50	+ 3	- 3
M-1003-3	Intermed. Shell	56484-1	SA533, Gr. B	0.02	0.70	3.64E+19	1.95E+19	-42	+11	+ 5
M-1004-1	Lower Shell	57326-1	SA533, Gr. B	0.03	0.62	3.69E+19	1.97E+19	-15	+39	+32
M-1004-2	Lower Shell	57286-1	SA533, Gr. B	0.03	0.58	3.69E+19	1.97E+19	+22	+76	+69
M-1004-3	Lower Shell	57359-1	SA533, Gr. B	0.03	0.62	3.69E+19	1.97E+19	-10	+44	+37
101-171	Mid. Circum. Weld	WW88114/ FL0145	ASA Weld/ Linde 0091	0.05	0.16	3.64E+19	1.95E+19	-70	+45	+34
101-124-A,-B,-C	Intermed. Longit. Weld	CE Lots BOLA, HODA	MMA Weld/ Type 8018	0.02	0.96	3.64E+19[d]	1.95E+19[d]	-60	+12	+ 4
101-142-A,-B,-C	Lower Longit. Weld	WW83653/ FL3536	ASA Weld/ Linde 0091	0.03	0.20	3.69E+19[d]	1.97E+19[d]	-80	+14	+ 3

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

[b]Per Section 6 of this report using neutron transport calculation methods.

[c]Materials chemical compositions per response to Generic Letter 92-01.

[d]Fluence value for longitudinal weld with maximum value.

[e]Per response to Generic Letter 92-01.

Table 7-5. Evaluation of Reactor Vessel End-of-Life (32 EFPY) Upper-Shelf Energy - Waterford Unit 3

Material Description				Material Chemical Composition, w/o[c]		Estimated EOL Fluence[b]		Initial	Estimated EOL-USE Per RG 1.99/2[a]		Estimated EFPY to 50 ft-lbs	
Fab. Mat'l. Code	Reactor Vessel Beltline Location	Heat Number[a]	Type	Copper	Nickel	Inside Surface n/cm^2	T/4 Wall Location n/cm^2	USE ft-lbs	Inside Surface	T/4 Wall Location	Inside Surface	T/4 Wall Location
M-1003-1	Intermed. Shell	56488-1	SA533, Gr. B	0.02	0.71	3.64E+19	1.95E+19	94[f]	79	81	>32	>32
M-1003-2	Intermed. Shell	56512-1	SA533, Gr. B	0.02	0.67	3.64E+19	1.95E+19	97[f]	81	84	>32	>32
M-1003-3	Intermed. Shell	56484-1	SA533, Gr. B	0.02	0.70	3.64E+19	1.95E+19	90[f]	76	78	>32	>32
M-1004-1	Lower Shell	57326-1	SA533, Gr. B	0.03	0.62	3.69E+19	1.97E+19	106[f]	87	90	>32	>32
M-1004-2	Lower Shell	57286-1	SA533, Gr. B	0.03	0.58	3.69E+19	1.97E+19	94[f]	78	80	>32	>32
M-1004-3	Lower Shell	57359-1	SA533, Gr. B	0.03	0.62	3.69E+19	1.97E+19	94[f]	78	80	>32	>32
101-171	Mid. Circum. Weld	WW88114/ FL0145	ASA Weld/ Linde 0091	0.05	0.16	3.64E+19	1.95E+19	156[g]	115	122	>32	>32
101-124-A,-B,-C	Intermed. Longit. Weld	CE Lots BOLA, HODA	MMA Weld/ Type 8018	0.02	0.96	3.64E+19[d]	1.95E+19[d]	N.A.	>50[e]	>50[e]	>32	>32
101-142-A,-B,-C	Lower Longit. Weld	WW83653/ FL3536	ASA Weld/ Linde 0091	0.03	0.20	3.69E+19[d]	1.97E+19[d]	N.A.	>50[e]	>50[e]	>32	>32

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

[b]Per Section 6 of this report using neutron transport calculation methods.

[c]Materials chemical compositions per response to Generic Letter 92-01.

[d]Fluence value for longitudinal weld with maximum value.

[e]Per response to Generic Letter 92-01.

[f]Based on 0.65 of longitudinal upper-shelf energy data.

[g]Estimate based on value given for the surveillance weld metal fabricated with the same weld wire and Linde 0091 weld flux.

Table 7-6. Evaluation of Reactor Vessel End-of-Life Pressurized Thermal Shock Criterion - Waterford Unit 3

Material Description			Material Chemical Composition, w/o[c]		Estimated Inside Surface EOL Fluence	PTS Evaluation, F[a]		
Reactor Vessel Beltline Location	Heat Number[e]	Type	Copper	Nickel	n/cm^2	Initial RT_{NDT},[e]	Inside Surface RT_{PTS}	Screening Criteria
Intermed. Shell	56488-1	SA533, Gr. B	0.02	0.71	3.64E+19	-30	+31	270
Intermed. Shell	56512-1	SA533, Gr. B	0.02	0.67	3.64E+19	-50	+11	270
Intermed. Shell	56484-1	SA533, Gr. B	0.02	0.70	3.64E+19	-42	+19	270
Lower Shell	57326-1	SA533, Gr. B	0.03	0.62	3.69E+19	-15	+46	270
Lower Shell	57286-1	SA533, Gr. B	0.03	0.58	3.69E+19	+22	+83	270
Lower Shell	57359-1	SA533, Gr. B	0.03	0.62	3.69E+19	-10	+51	270
Mid. Circum. Weld	WW88114/ FL0145	ASA Weld/ Linde 0091	0.05	0.16	3.64E+19	-70	+45	300
Intermed. Longit. Weld	CE Lots BOLA/HODA	MMA Weld/ Type 8018	0.02	0.96	3.64E+19[d]	-60	+32	270
Lower Longit. Weld	WW83653/ FL3536	ASA Weld/ Linde 0091	0.03	0.20	3.69E+19[d]	-80	+23	270

[a]Per 10CFR50, Section 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.[22]

[b]Per Section 6 of this report using neutron transport calculation methods.

[c]Materials chemical compositions per response to Generic Letter 92-01.

[d]Fluence value for longitudinal weld with maximum value.

[e]Per response to Generic Letter 92-01.

Figure 7-1. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Plate Material Longitudinal Orientation, Heat No. M-1004-2

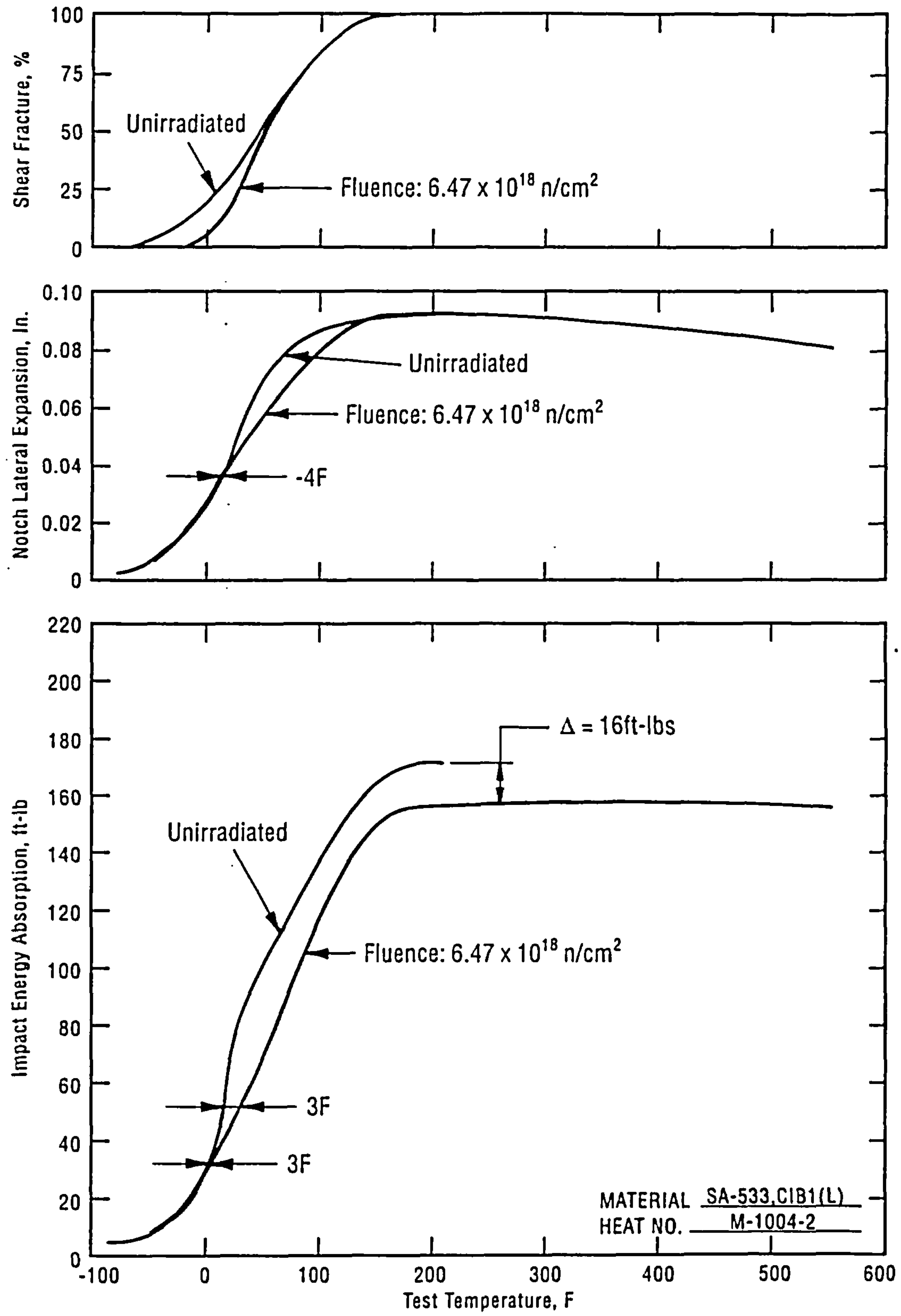


Figure 7-2. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Plate Material Transverse Orientation, Heat No. M-1004-2

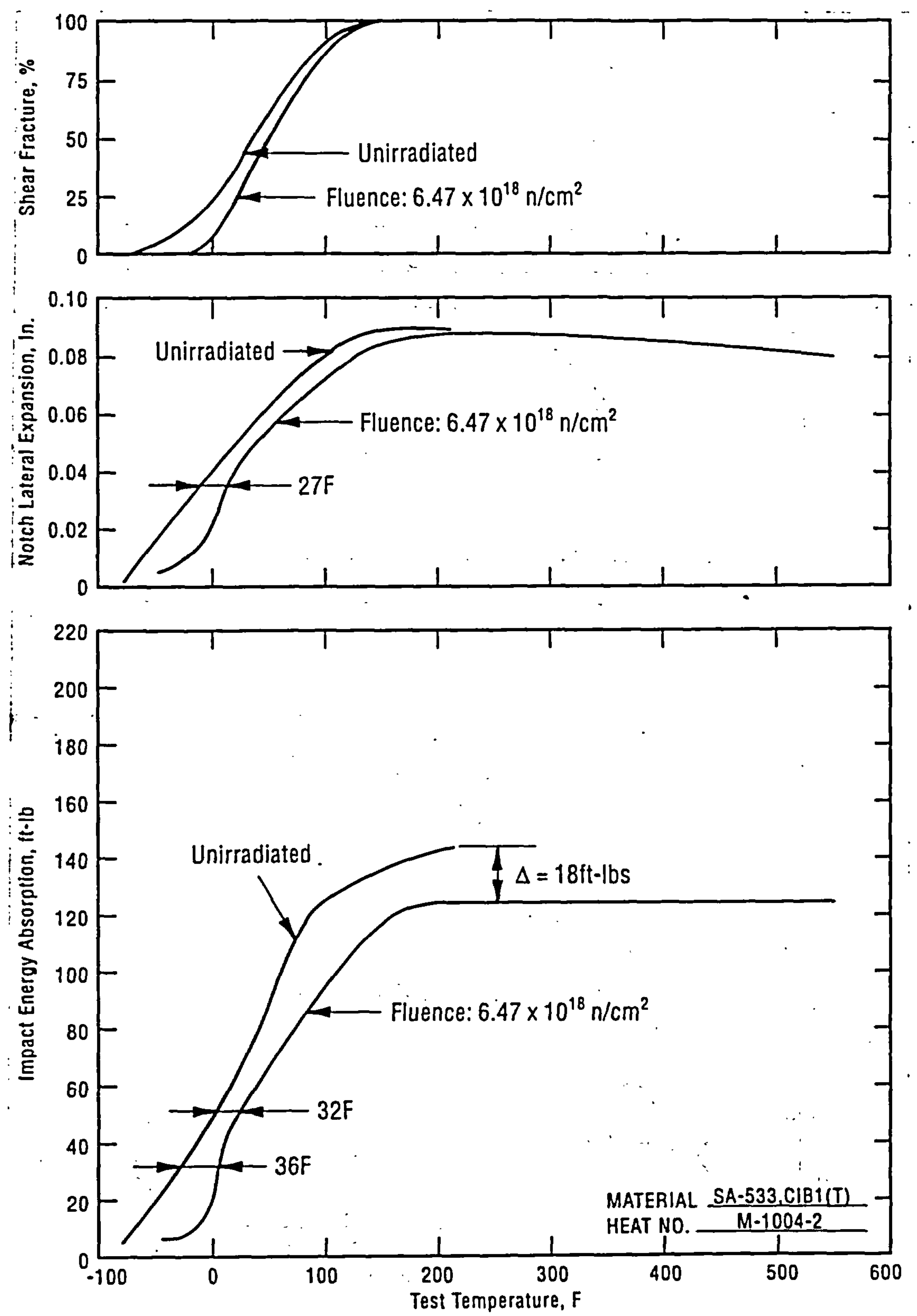


Figure 7-3. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Base Metal, Heat-Affected-Zone, Heat No. M-1004-2

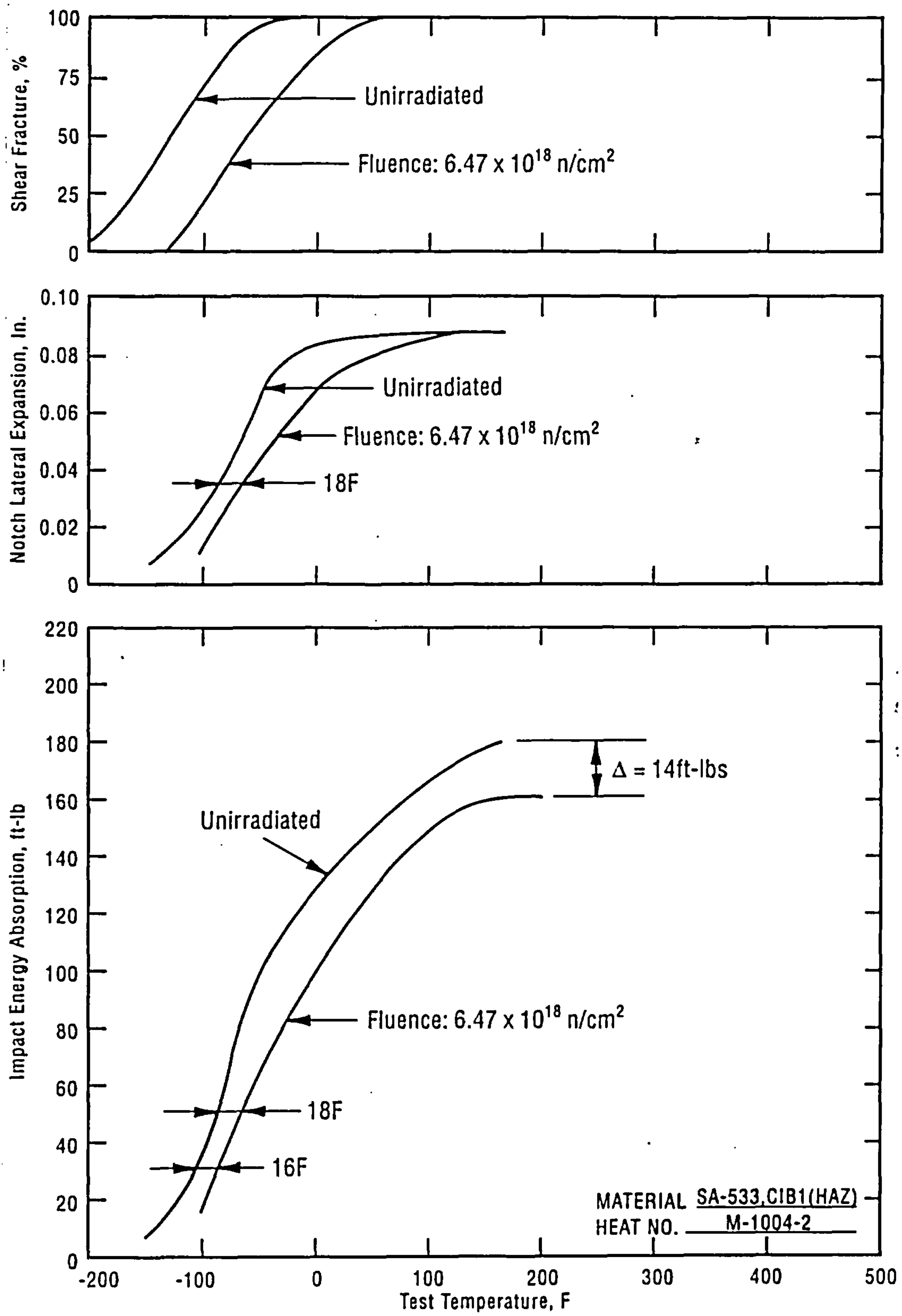
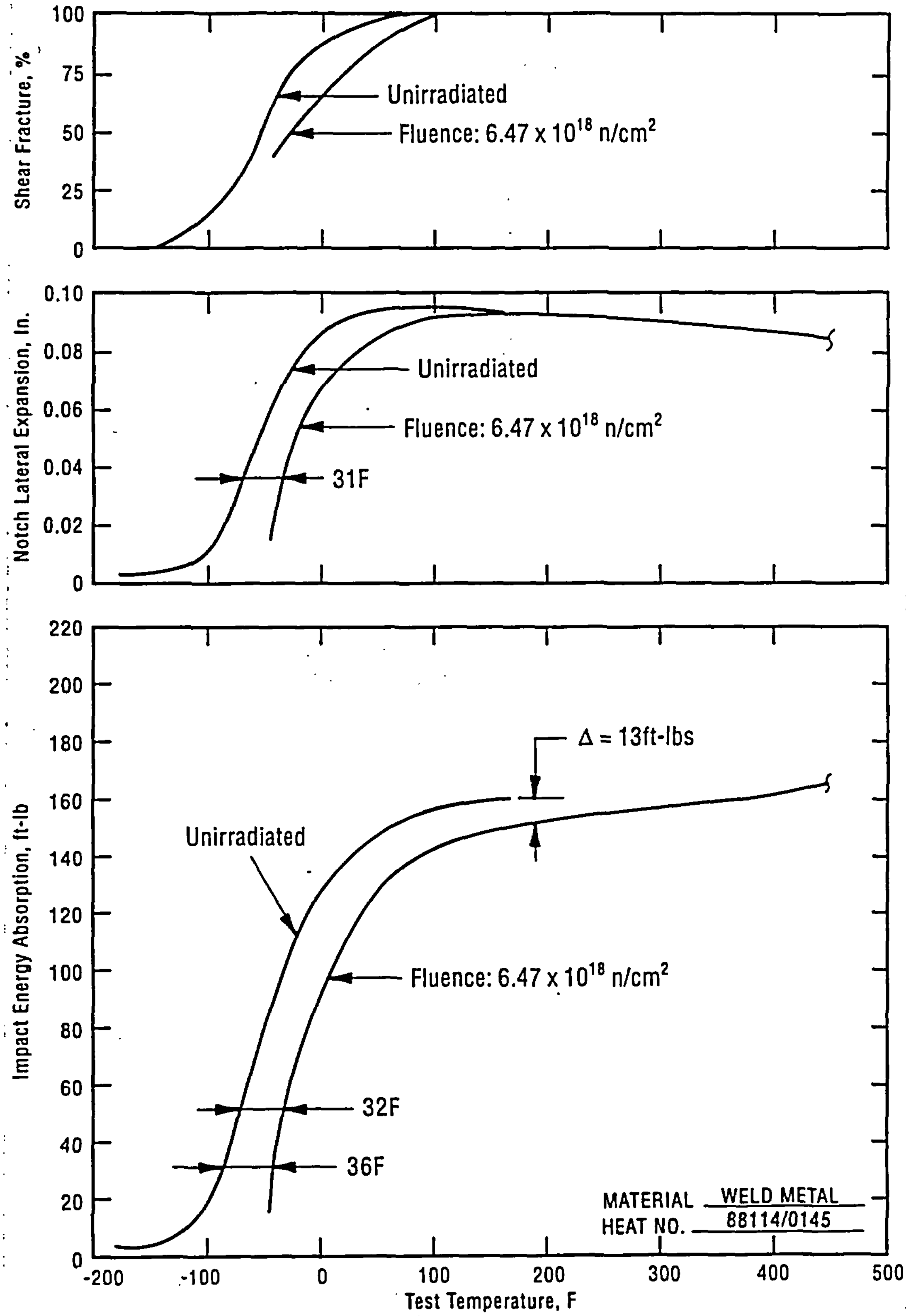


Figure 7-4. Comparison of Unirradiated and Irradiated Charpy Impact Data Curves for Weld Metal 88114/0145



Page Intentionally Left Blank

8. SUMMARY OF RESULTS

The analysis of the reactor vessel material contained in the first surveillance capsule (Capsule W-97) removed for evaluation as part of the Waterford Generating Station Unit No. 3 Reactor Vessel Surveillance Program, led to the following conclusions:

1. The capsule received an average fast fluence of 6.47×10^{18} n/cm^2 (E > 1.0 MeV). The predicted fast fluence for the reactor vessel T/4 location at the end of the fourth fuel cycle is 2.74×10^{18} n/cm^2 (E > 1 MeV).

2. The fast fluence of 6.47×10^{18} n/cm^2 (E > 1 MeV) increased the RT_{NDT} of the capsule reactor vessel core region shell materials by a maximum of 40F.

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 Waterford Generating Station Unit No. 3 reactor pressure vessel inside surface will receive in 40 calendar year's operation is 3.69×10^{19} n/cm^2 (E > 1 MeV).

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

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

6. Neither the base metal nor the weld metal upper-shelf energies at the T/4 location, based on surveillance capsule results, are predicted to 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 the base metal and the weld metal RT_{NDT} properties due to irradiation are conservative except for the base metal transverse properties.

8. The current techniques (i.e., Regulatory Guide 1.99, Revision 2) used to predict the change in the base metal and the weld metal Charpy

upper-shelf properties due to irradiation are in good agreement with the base metal and conservative for the weld metal.

9. 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 post-irradiation test results of Capsule W-97 and the recommended withdrawal schedule of Table 1 of E185[15] the following schedule is recommended for the examination of the remaining capsules in the Waterford Generating Station, Unit No. 3 RVSP:

Capsule Identification	Location of Capsules[a]	Lead Factor[b]	Removal Time	Expected Capsule Fluence (n/cm^2)[c]
		Evaluation Schedule[a]		
W-83	83°	1.26	15 EFPY	2.19×10^{19}
W-104	104°	0.81	Spare[d]	(2.99×10^{19})
W-263	263°	1.26	26 EFPY	3.69×10^{19}
W-277	277°	1.26	Spare[d]	(4.65×10^{19})
W-284	284°	0.81	Spare[d]	(2.99×10^{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]Estimated fluence values based on current fuel cycle designs.

[d]Spare capsule to be irradiated and available for an intermediate evaluation, if data needed, to support licensing requirements or provide data for license renewal. Capsule withdrawal at 32 EFPY will have estimated fluence as defined in brackets ().

10. CERTIFICATION

The specimens were tested, and the data obtained from Entergy Operations, Inc., Waterford Generating Station, Unit No. 3, reactor vessel surveillance Capsule W-97 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. — 11/16/92 (Date)
Project Technical Manager

This report has been reviewed for technical content and accuracy.

M. J. Devan (Material Analysis) — 11/17/92 (Date)
M&SA Unit

L. Petrusha (Fluence Analysis) — 11/17/92 (Date)
Performance Analysis Unit

Verification of independent review.

JFShepard for KE Moore — 11/17/92

K. E. Moore, Manager — Date
M&SA Unit

This report is approved for release.

T. L. Baldwin, P.E. — 11/16/92 (Date)
Program Manager

REVISION 1 CERTIFICATION

The revision to the document is technically accurate and conforms to accepted techniques, established standard methods and procedures in accordance with the requirements of 10 CFR 50, Appendices G and H.

J B Hall 2-5-04
J. B. Hall (Materials Analysis) Date
Materials & Structural Analysis Unit

This report has been reviewed for technical content and accuracy.

2-5-04
B. R. Grambau (Materials Analysis) Date
Materials & Structural Analysis Unit

Verification of independent review.

2/13/2004
A D. McKim, Manager Date
Materials & Structural Analysis Unit

This report is approved for release.

2/23/04
W. R. Gray 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 E185-73, are shown in Table A-1. The locations of these materials within the reactor vessel are shown in Figures A-1 through A-4.

2. Definition of Beltline Region

The beltline region of Waterford Unit 3 was defined in accordance with the definition given in ASTM E185-73.

3. Capsule Identification

The capsules used in the Waterford Unit 3 surveillance program are identified below by identification, location, and original target fluence.[1]

Capsule Removal	Capsule Identification	Capsule Location[(a)]	Approximate Refueling	Target Fluence, n/cm²
1	W-97	97°	7	6.0×10^{18}
2	W-104	104°	19	1.6×10^{19}
3	W-284	284°	30	2.5×10^{19}
4	W-263	263°	Standby	---
5	W-277	277°	Standby	---
6	W-83	83°	Standby	---

4. Specimens Per Surveillance Capsule

The type and quantity of each material contained in each surveillance capsule is shown in Table A-2.

Table A-1. Unirradiated Impact Properties and Residual Element Content Data of Beltline Region Materials Used for Selection of Surveillance Program Materials - Waterford Unit No. 3[21,22,23]

Fabricator Material Code	Material Ident., Heat No.	Material Type	Beltline Region Location	Drop wt T_{NDT}, F	Charpy Impact Data, Longitudinal: 30 ft-lb, F	50 ft-lb, F	35 MLE, F	USE, ft-lb	RT_{NDT} F	Chemistry, wt%: Cu	Ni	P	S
M-1003-1	56488-1	SA533, Gr. B	Intermed. Shell	-30	-30	-10	-10	144	-30	0.02	0.71	0.004	0.010
M-1003-2	56512-1	SA533, Gr. B	Intermed. Shell	-50	-55	-12	-15	149	-50	0.02	0.67	0.006	0.007
M-1003-3	56484-1	SA533, Gr. B	Intermed. Shell	-50	-22	- 2	-10	138	-42	0.02	0.70	0.007	0.009
M-1004-1	57326-1	SA533, Gr. B	Lower Shell	-50	+10	+25	+20	163	-15	0.03	0.62	0.006	0.008
M-1004-2	57286-1	SA533, Gr. B	Lower Shell	-20	+37	+62	+55	144	22	0.03	0.58	0.005	0.005
M-1004-3	57359-1	SA533, Gr. B	Lower Shell	-50	+12	+30	+25	145	-10	0.03	0.62	0.007	0.007
101-171	88114/0145	ASA Weld/Linde 0091	Middle Circum.	-70	---	---	---	---	-70	0.05	0.16	0.008	0.008
101-124-A,-B,-C	BOLA/HODA	MMA Weld/Type 8018	Intermed. Longit.	-60	---	---	---	---	-60	0.02	0.96	0.010	0.016
101-142-A,-B,-C	83653/3536	ASA Weld/Linde 0091	Lower Longit.	-80	---	---	---	---	-80	0.03	0.20	0.007	0.009

Table A-2. Type and Quantity of Specimens Contained in Each Irradiation Capsule Assembly

Capsule Location	Target Fluence[a] (n/cm²)	Base Metal (Heat No M-1004-2) Impact			Weld Metal (88114/0145)[c]		HAZ (Heat No. M-1004-2)		Correl. Material[b]	Total Specimens	
		L	T	Tensile	Impact	Tensile	Impact	Tensile	Impact	Impact	Tensile
Vessel 97°	6.0×10^{18}	12	12	3	12	3	12	3	--	48	9
Vessel 104°	1.6×10^{19}	--	12	3	12	3	12	3	12	48	9
Vessel 284°	2.5×10^{19}	12	12	3	12	3	12	3	--	48	9
Vessel 263°	Standby	--	12	3	12	3	12	3	12	48	9
Vessel 277°	Standby	12	12	3	12	3	12	3	--	48	9
Vessel 83°	Standby	12	12	3	12	3	12	3	--	48	9
TOTALS		48	72	18	72	18	72	18	24	288	54

[a]Adjusted to nearest value attainable during scheduled refueling.

[b]Reference material correlation monitors.

[c]Weld wire/weld flux lot combination.

L = Longitudinal

T = Transverse

Figure A-1. Location and Identification of Materials Used in the Fabrication of Waterford Unit 3 Reactor Pressure Vessel

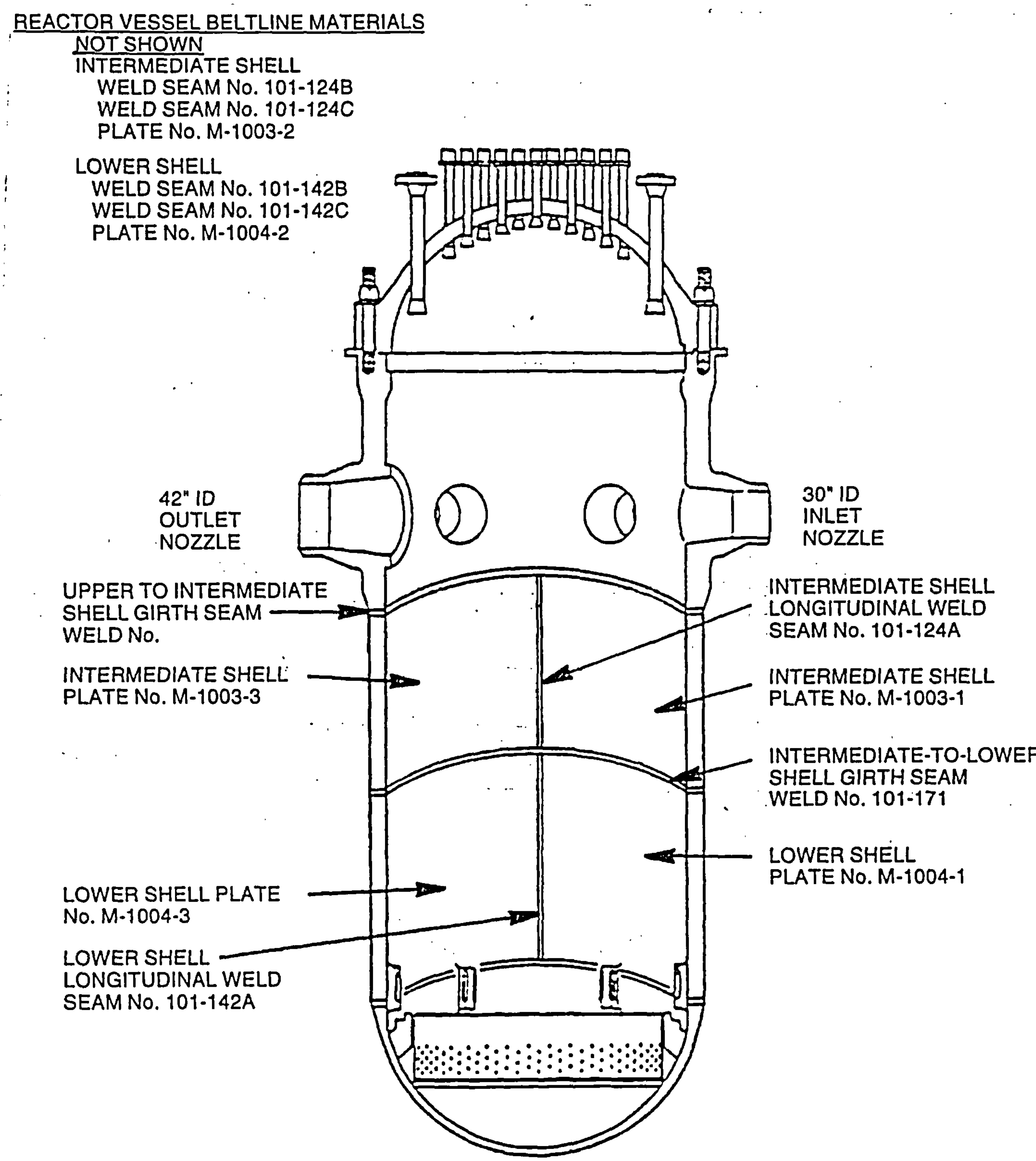


Figure A-2. Location of Beltline Region Materials in Relationship to the Reactor Vessel Core

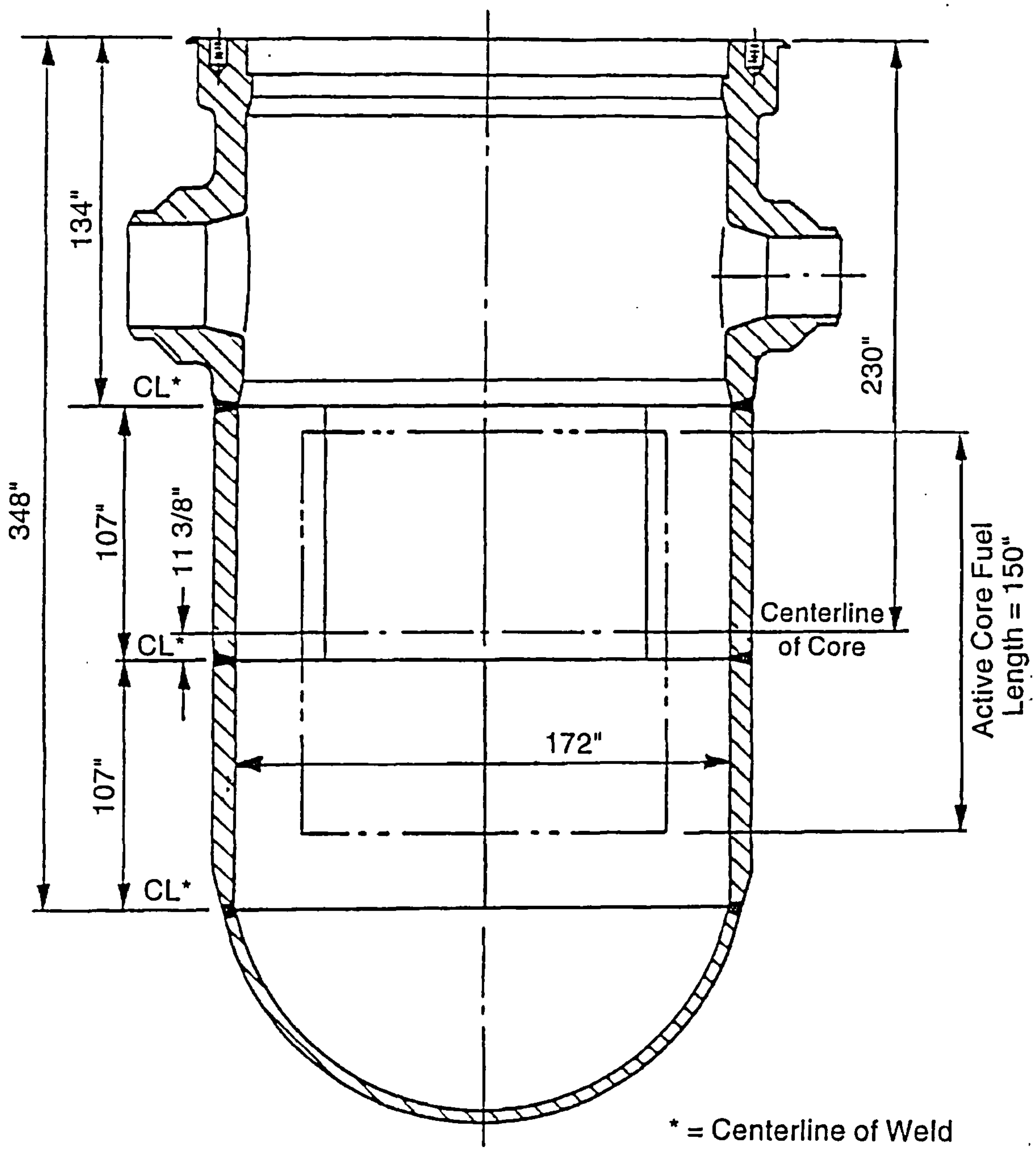


Figure A-3. Location of Longitudinal Welds in Waterford Unit 3 Upper and Lower Shell Courses

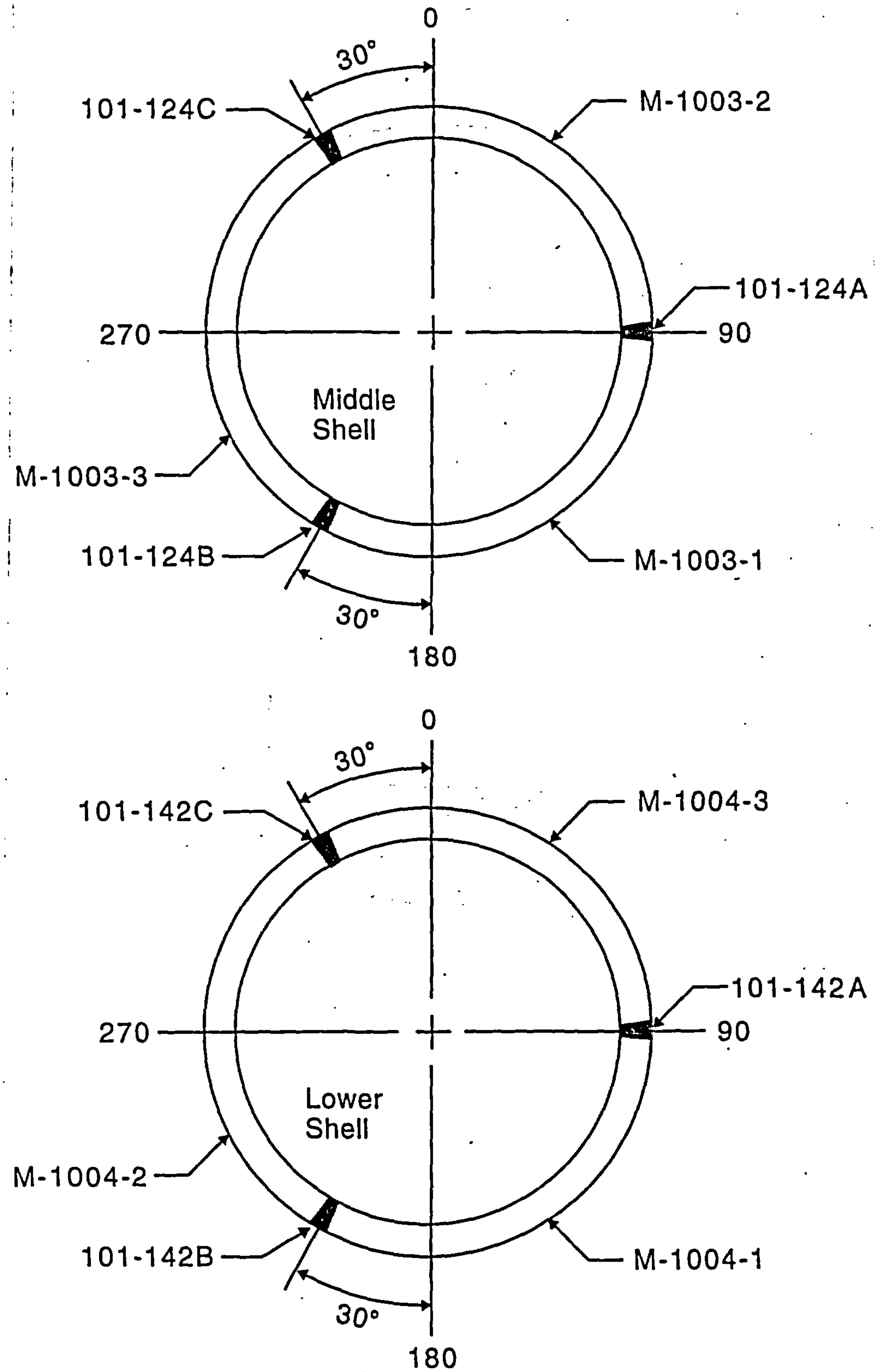
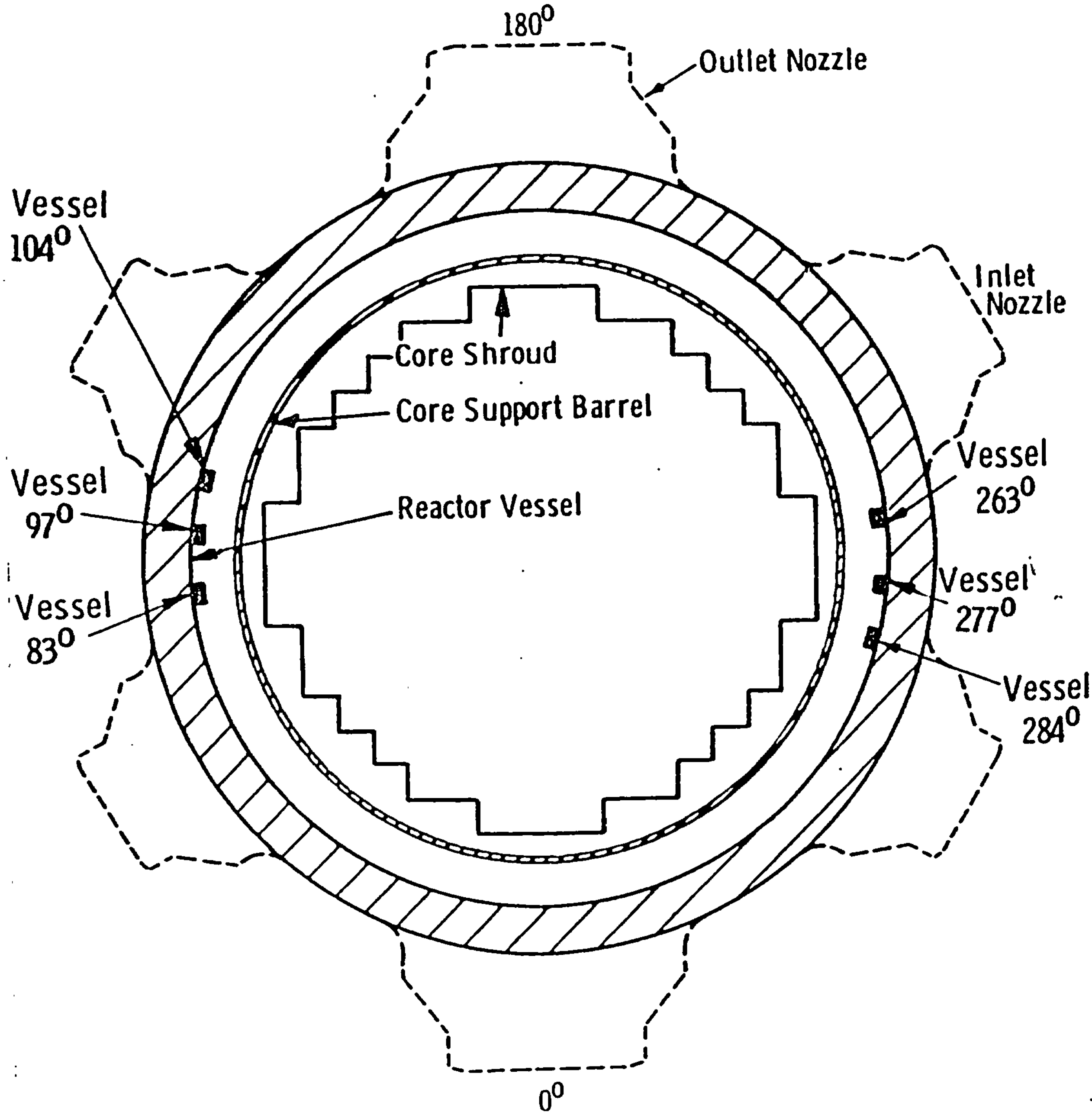


Figure A-4. Location of Surveillance Capsule Irradiation Sites in Waterford Unit 3



APPENDIX B

Pre-Irradiation Tensile Data

Table B-1. Tensile Properties of Unirradiated Shell Plate Material, Heat No. M-1004-2, Longitudinal

Specimen No.	Test Temp, F	Strength, ksi Yield*	Strength, ksi Ultimate	Fracture Load, lb	Fracture, ksi Strength	Fracture, ksi Stress	Reduction of Area, %	Elongation, Total/Unif. %
1J2	71	68.6/66.7	88.5	2640	53.9	189	71.4	29/11.3
1J1	71	70.4/67.4	88.4	2700	55.1	180	69.4	27/11.3
1K2	71	70.0/68.6	90.1	2700	55.1	193	71.4	30/11.7
1JA	250	63.7/63.1	82.5	2640	53.9	176	69.4	24/ 9.2
1K3	250	66.1/64.9	84.1	2640	53.9	176	69.4	24/ 9.3
1JL	250	63.7/63.1	83.3	2700	55.1	180	69.4	26/ 9.3
1J6	550	63.7/----	85.5	2700	55.1	193	71.4	23/ 9.3
1JC	550	63.1/----	85.9	2700	55.1	208	73.4	26/ 9.8
1J3	550	62.5/----	85.6	2760	56.3	173	67.3	25/10.2

*Lower and upper yield strengths.

Table B-2. Tensile Properties of Unirradiated Shell Plate Material, Heat No. M-1004-2, Transverse

Specimen No.	Test Temp, F	Strength, ksi Yield*	Strength, ksi Ultimate	Fracture Load, lb	Fracture, ksi Strength	Fracture, ksi Stress	Reduction of Area, %	Elongation, Total/Unif. %
2KC	71	69.2/68.6	89.7	2880	58.8	192	69.4	27/10.8
2KT	71	68.4/67.2	88.4	2820	58.8	188	70.0	29/11.3
2KB	71	69.8/68.6	89.0	2940	60.0	196	65.3	26/10.8
2KD	250	65.5/64.3	83.7	2700	55.1	180	69.4	23/ 9.7
2JE	250	64.6/64.6	83.9	2820	57.6	188	69.4	21/ 9.3
2L2	250	64.9/64.3	82.3	2940	60.0	163	63.3	23/ 9.3
2J7	550	64.9/----	87.2	2880	58.8	169	65.3	23/10.2
2KP	550	63.7/----	86.9	3000	61.2	188	67.3	22/ 9.8
2KU	550	64.9/----	87.0	2880	58.8	160	63.3	22/ 9.8

*Lower and upper yield strengths.

Table B-3. Tensile Properties of Unirradiated Shell Plate HAZ Material, Heat No. M-1004-2, Transverse

Specimen No.	Test Temp, F	Strength, ksi Yield*	Strength, ksi Ultimate	Fracture Load, lb	Fracture, ksi Strength	Fracture, ksi Stress	Reduction of Area, %	Elongation, Total/Unif. %
4KT	71	71.0/68.6	91.0	2820	57.6	188	69.4	22/ 7.3
4JJ	71	68.6/68.0	90.9	2820	57.6	188	69.4	21/ 7.0
4K4	71	69.8/68.0	91.9	2820	57.6	188	69.4	21/ 6.2
4KP	250	64.3/63.7	84.3	2640	53.9	176	69.4	21/ 5.4
4J5	250	63.1/63.1	84.0	2640	53.9	165	67.3	19/ 5.8
4KE	250	63.7/63.7	84.8	2640	53.9	176	69.4	21/ 5.4
4JT	550	60.6/60.0	86.3	2820	57.6	166	65.3	21/ 6.7
4JE	550	63.1/61.8	86.9	2820	57.6	176	67.3	20/ 6.8
4J4	550	60.0/58.7	86.9	2820	57.6	176	67.3	20/ 6.3

*Lower and upper yield strengths.

Table B-4. Tensile Properties of Unirradiated Weld Metal 88114/0145

Specimen No.	Test Temp, F	Strength, ksi Yield*	Strength, ksi Ultimate	Fracture Load, lb	Fracture, ksi Strength	Fracture, ksi Stress	Reduction of Area, %	Elongation, Total/Unif. %
3KE	71	85.7/82.0	92.9	2760	56.3	184	69.4	27/ 9.3
3J3	71	84.5/80.2	91.6	2760	56.3	197	71.4	27/ 9.3
3K1	71	84.5/80.8	92.0	2700	55.1	193	71.4	29/10.2
3L2	250	79.6/75.9	86.9	2700	55.1	180	69.4	22/ 7.7
3J5	250	80.2/75.9	87.2	2760	56.3	197	71.4	21/ 7.8
3JC	250	79.6/73.5	85.4	2580	52.6	184	71.4	23/ 7.3
3K4	550	72.2/----	88.8	2580	52.6	172	69.4	24/ 9.5
3KA	550	72.2/----	88.2	2640	53.9	189	71.4	24/ 9.3
3KU	550	72.2/----	87.9	2820	57.6	176	67.3	23/ 8.8

*Lower and upper yield strengths.

Page Intentionally Left Blank

APPENDIX C

Pre-Irradiation Charpy Impact Data

Table C-1. Charpy Impact Data From Unirradiated Base Material, Longitudinal Orientation, Heat No. M-1004-2

Specimen ID	Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
13T	-80	7.0	3	0
156	-40	12.0	10	10
152	-40	11.5	10	10
15J	0	14.5	14	15
123	0	48.5	41	20
112	40	86.0	65	40
11P	40	105.0	78	65
153	80	107.0	72	75
11A	80	130.0	90	80
12D	120	131.0	89	85
14P	120	147.0	92	90
147	140	158.0	91	100
11M	160	169.5	94	100
137	160	177.5	90	100
127	210	168.5	95	100
126	210	175.0	90	100

Table C-2. Charpy Impact Data From Unirradiated Base Material, Transverse Orientation, Heat No. M-1004-2

Specimen ID	Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
264	-80	9.0	7	0
21L	-60	10.0	6	0
22E	-40	20.5	18	10
23B	-40	28.5	23	10
25P	0	44.0	37	20
26B	0	65.5	50	25
21K	40	68.5	55	40
23L	40	73.0	57	60
26A	80	118.0	82	75
23A	80	130.0	77	75
212	120	123.5	81	90
255	120	141.0	90	100
21P	160	136.0	88	100
254	160	138.5	92	100
25T	210	143.5	88	100
23C	210	145.0	90	100

Table C-3. Charpy Impact Data from Unirradiated Base Metal, Heat-Affected Zone, Heat No. M-1004-2

Specimen ID	Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
417	-150	6.5	9	0
44C	-135	10.5	8	0
451	-120	21.5	17	10
42J	- 80	44.5	33	30
46T	- 80	76.5	50	45
45P	- 40	113.5	73	70
443	- 40	116.5	77	60
466	0	118.0	77	75
46D	0	138.0	88	80
42M	40	126.0	84	75
47T	40	162.0	88	100
45C	80	163.5	91	100
43M	80	177.0	84	100
41K	120	152.5	90	100
45D	120	183.5	89	100
427	160	164.5	87	100
41A	160	183.5	89	100

Table C-4. Charpy Impact Data from Unirradiated Weld Metal, 88114/0145

Specimen ID	Test Temp, F	Absorbed Energy, ft-lb	Lateral Expansion, 10^{-3} in.	Shear Fracture, %
354	-180	3.5	4	0
316	-150	5.5	3	0
36A	-120	8.0	6	10
36D	- 80	13.5	12	20
31D	- 80	45.5	36	30
32A	- 40	83.5	61	50
313	- 40	96.5	69	75
357	0	122.5	80	80
341	0	130.5	95	85
31A	40	142.0	95	90
344	40	149.0	97	100
33T	80	146.0	94	100
32M	80	158.0	96	100
37L	120	155.5	96	100
32B	120	162.5	97	100
343	160	148.5	94	100
36J	160	171.0	94	100

Figure C-1. Charpy Impact Data From Unirradiated Base Metal (Plate), Longitudinal Orientation, Heat No. M-1004-2

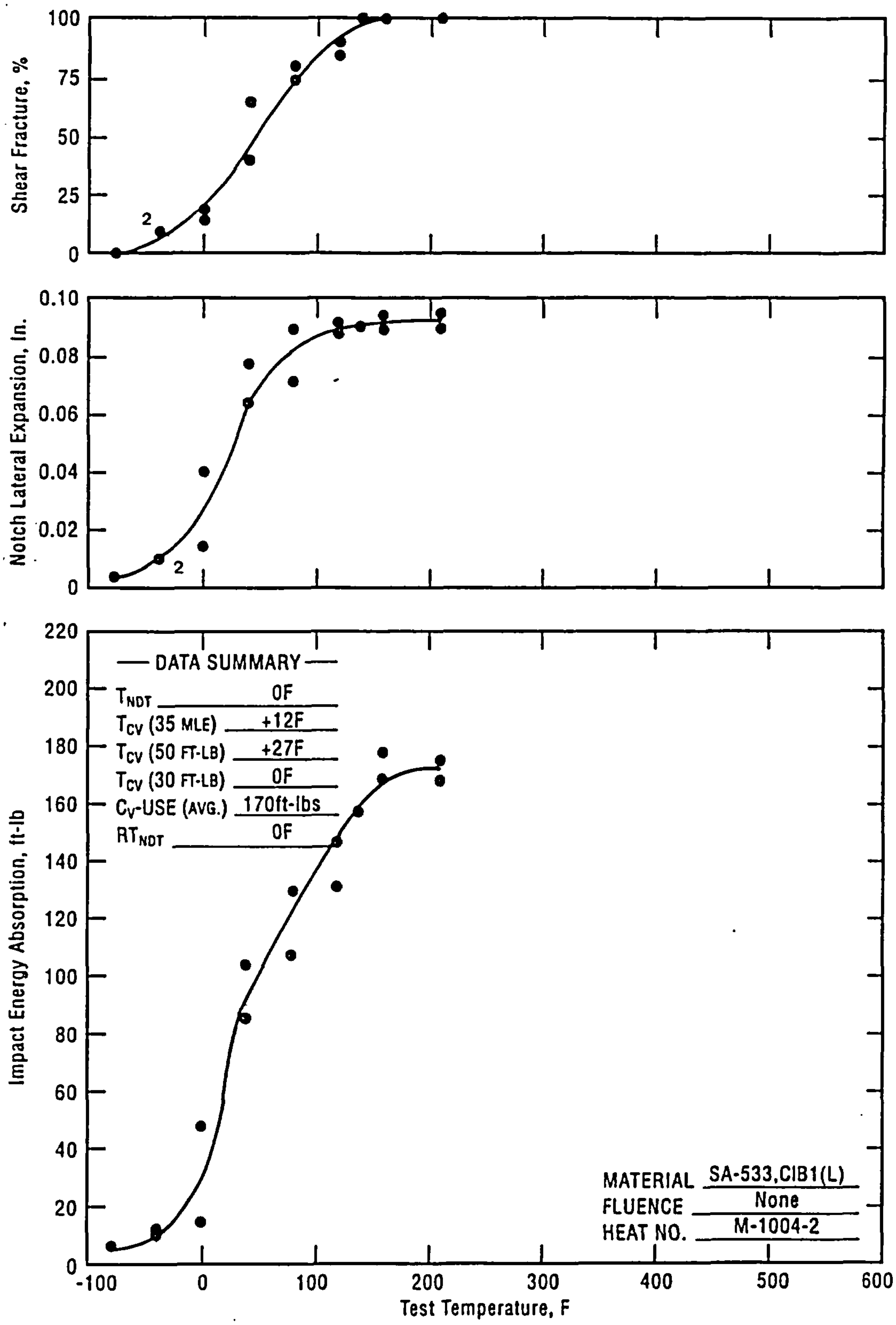


Figure C-2. Charpy Impact Data From Unirradiated Base Metal (Plate), Transverse Orientation, Heat No. M-1004-2

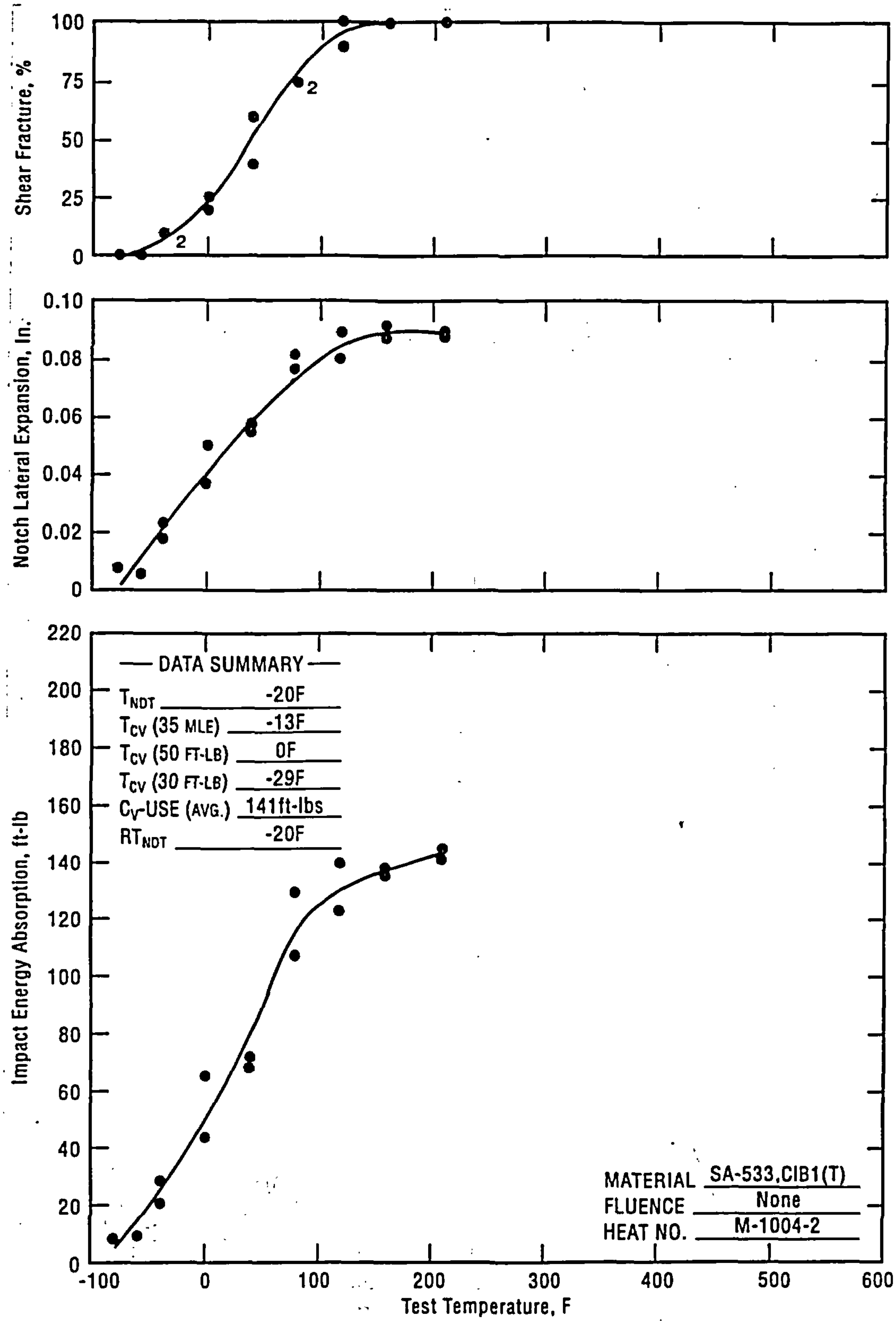


Figure C-3. Charpy Impact Data From Unirradiated Heat-Affected-Zone Base Metal, Heat No. M-1004-2

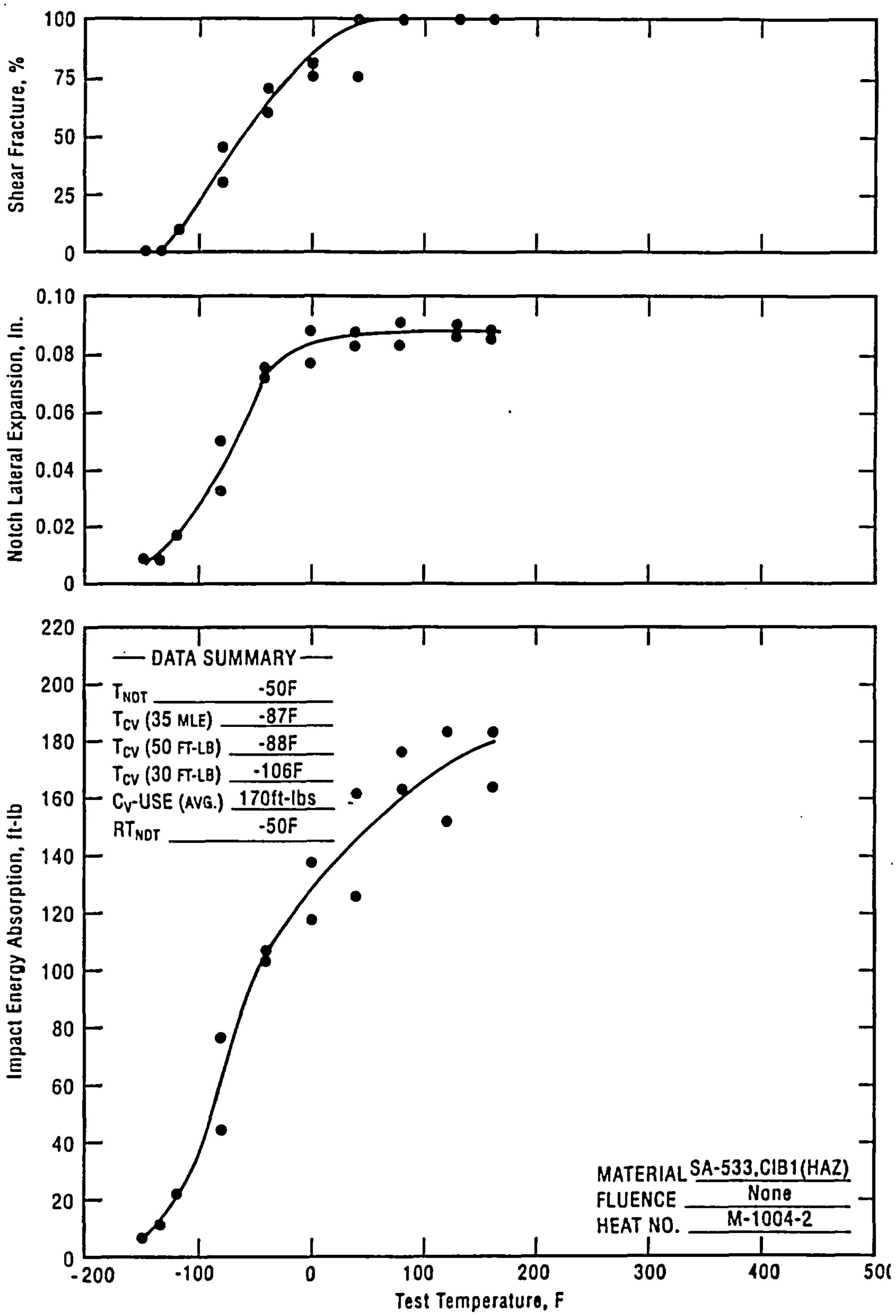
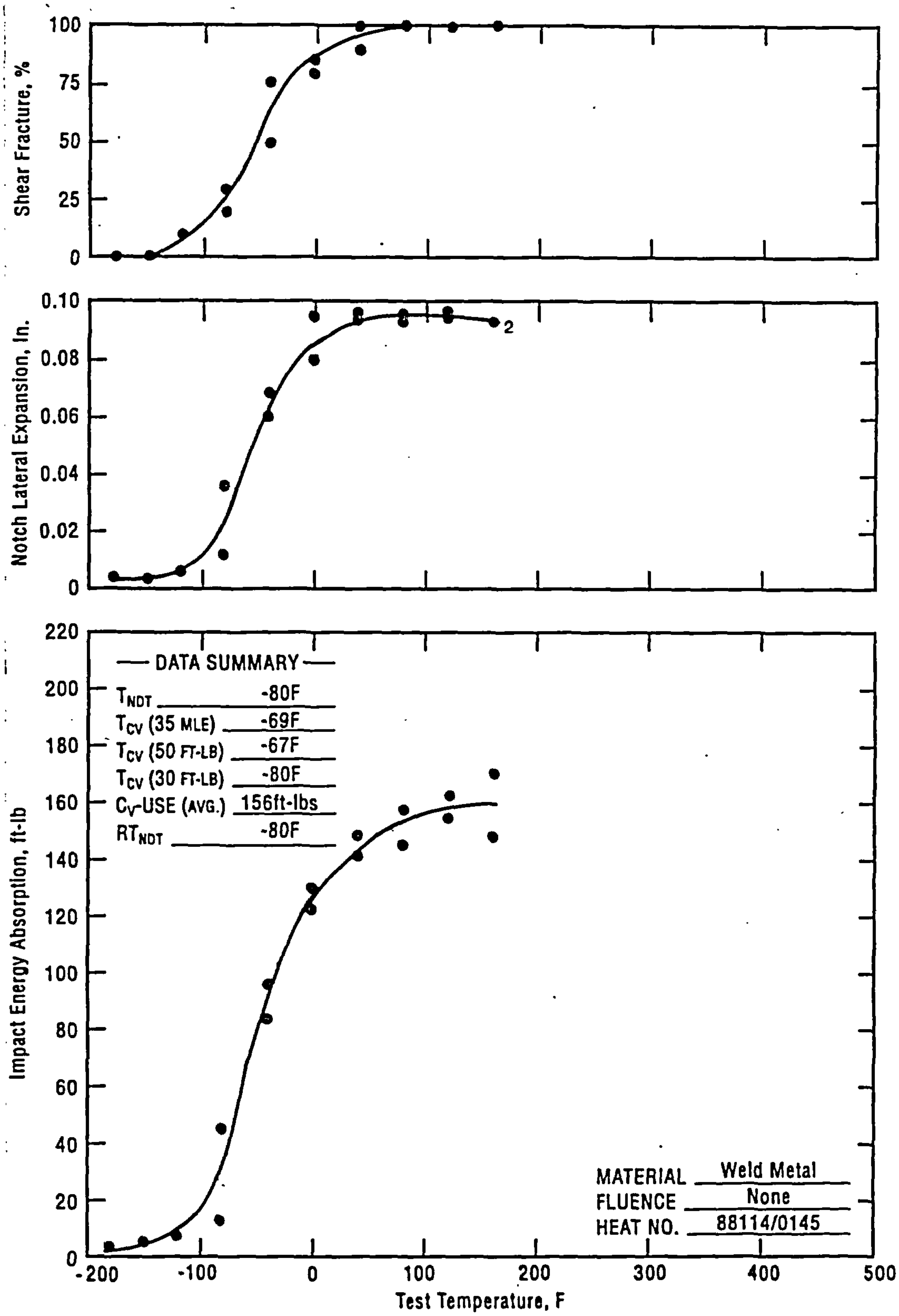


Figure C-4. Charpy Impact Data From Unirradiated Weld Metal, 88114/0145



Page Intentionally Left Blank

APPENDIX D

Fluence Analysis Methodology

1. Analytical Method

A semi-empirical 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 semi-empirical 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, thermal shield, pressure vessel, and concrete. The material and geometry model, represented in Figure D-3, uses one-eighth 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 47 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 derived based on a combined ^{235}U and ^{239}Pu 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 was divided into three parts, as shown in Figure D-3. The first part, Model "A," was used to generate the energy-dependent angular flux at the inner boundary of Model "B," which began at the inner surface of the core barrel. Model A included 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 was as accurate as possible for these two very important parameters. The second, Model B, contained an explicit representation of both surveillance capsules and associated components for the applicable time periods. The B&W Owners Group's Flux Perturbation Experiment[16] 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. Detailed explicit modeling of the capsule, capsule holder tube, and internal components were therefore incorporated into the DOT calculational models. The third, Model "C," was similar to Model B except that no capsule was included. Model C was used in determining the vessel flux in quadrants that did 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 52.95 cm was specified between Model A and Models B or C. The width of this overlap region, which was fixed by the placement of the Model A vacuum boundary and the Model B boundary source, was 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 did not cause a perturbation in the flux at the points of interest.

Macroscopic Cross Sections

Macroscopic cross sections, required for transport analyses, were obtained with the mixing code GIP. Nominal compositions were 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 was a homogeneous mixture of fuel, fuel cladding, structure, and coolant.

The cross-section library presently used is the (47-neutron group and 20-gamma group) BUGLE coupled 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 was 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, were 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 was 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.

$$Sv_{ijg} = \underbrace{[\frac{\nu}{K} P_D]}_{\text{magnitude}} \, x \, \underbrace{[RPD_{ij} X_g]}_{\text{spatial}}$$

where:

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

ν/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. Even though the entire (eighth-core symmetric) RPD was modeled in the analysis, only the peripheral fuel assemblies contributed 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 1.08.

A pin-by-pin RPD was provided by the customer, and was subsequently used to produce the source for use in the DOTIV code.

1.1. Capsule Flux and Fluence Calculation

As discussed above, the DOTIV code was used to explicitly model the capsule assemblies and to calculate the neutron flux as a function of energy within the capsules. 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:

$$D_i = \frac{N\, f_i}{(3.7x10^4) A_n\, E} \sum_E \sigma_n(E)\, \phi(E) \sum_j F_j\, (1-e^{-\lambda_i t_j})\; e^{-\lambda_i (T-\tau_j)}$$

where:

N = Avogadro's number,

A_n = 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:

Photofission adjustments to ^{238}U dosimeter activities

Axial correction factor to adjust for axial power distribution

After making these adjustments the calculated dosimeter activities were used with the corresponding measured activities to obtain the measured to calculated activity ratios or 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 for each capsule to be consistent with the dosimeter measurements. The flux normalization factors are given in Table D-1. Note that the Co-60 dosimeters are typically not used in the determination of the final normalization factor to be applied to the calculated flux due to the fact that they do not respond in the regions of interest, E > 1.0 MeV and E > 0.1 MeV, and the thermal region is not accurately calculated in the DOT analysis.

2. Vessel Fluence Extrapolation

For past core cycles, fluence values in the pressure vessel were calculated as described above. Extrapolation to future cycles was required to predict the useful vessel life. Two time periods were considered in the extrapolation: 1) operation to date for which vessel fluence has been calculated, 2) future fuel cycles which no analyses exist.

For the Waterford Unit 3 analysis, time period 1 was through cycle 4, and time period 2 covered cycles from the end of cycle 4 through 32 EFPY. The flux and fluence for time period 2 was estimated by assuming that the flux at the inside surface of the pressure vessel (PVIS) for future cycles was the same as that calculated for cycles 1 through 4. This was a reasonable assumption because the first four cycles were similar in that fresh fuel was loaded in the peripheral locations in each of the cycles.

It was found in the Waterford Unit 3 analysis and is shown in figure 6-3 that the peak fluence at the PVIS occurred at approximately 1 degree off the major axis for cycles 1 to 4. For this reason, the flux used to extrapolate from EOC 4 to 32 EFPY was the flux calculated at 1 degree. Future analyses will ascertain the actual effects.

Table D-1. Flux Normalization Factor for 97°Capsule

Dosimeter Reaction	Measured Activity,[a] μCi/g	Calculated Activity,[b] μCi/g	Flux Normalization[c] Factor
$^{58}Ni(n,p)^{58}Co$	1835.0	1900.7	0.965
$^{46}Ti(n,p)^{46}Sc$	365.9	342.6	1.067
$^{54}Fe(n,p)^{54}Mn$	1427.0	1442.0	0.990
$^{63}Cu(n,\alpha)^{60}Co$	8.377	8.847	0.947
$^{238}U(n,f)^{137}Cs$	8.157[d]	9.957	0.819
$^{59}Co(n,\gamma)^{60}Co$[e]	4.380E+5	3.409E+5	1.286
$^{59}Co(n,g)^{60}Co$[f]	4.860E+4	5.643E+4	0.862
		Averaged:	0.958[g]

[a]Average of three dosimeter wires, except for U powder capsules.

[b]Each listed activity was determined as the average of three calculated activities.

[c]Average of measured to calculated activity ratios for each dosimeter type.

[d]The U dosimeters were powder capsules, three shielded and three bare. Since the U-238 dosimeter response in the thermal range is negligible, the shielding should have virtually no effect on the response of the dosimeters. The three shielded U samples showed good agreement with the calculations. However, of the three bare U-238 dosimeters, one was unrecoverable and one gave unreasonable results. Therefore, the U values in this table are for four dosimeters, three shielded and one bare, averaged together.

[e]Bare dosimeters.

[f]Cd-shielded dosimeters.

[g]Average of all dosimeters except Co.

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

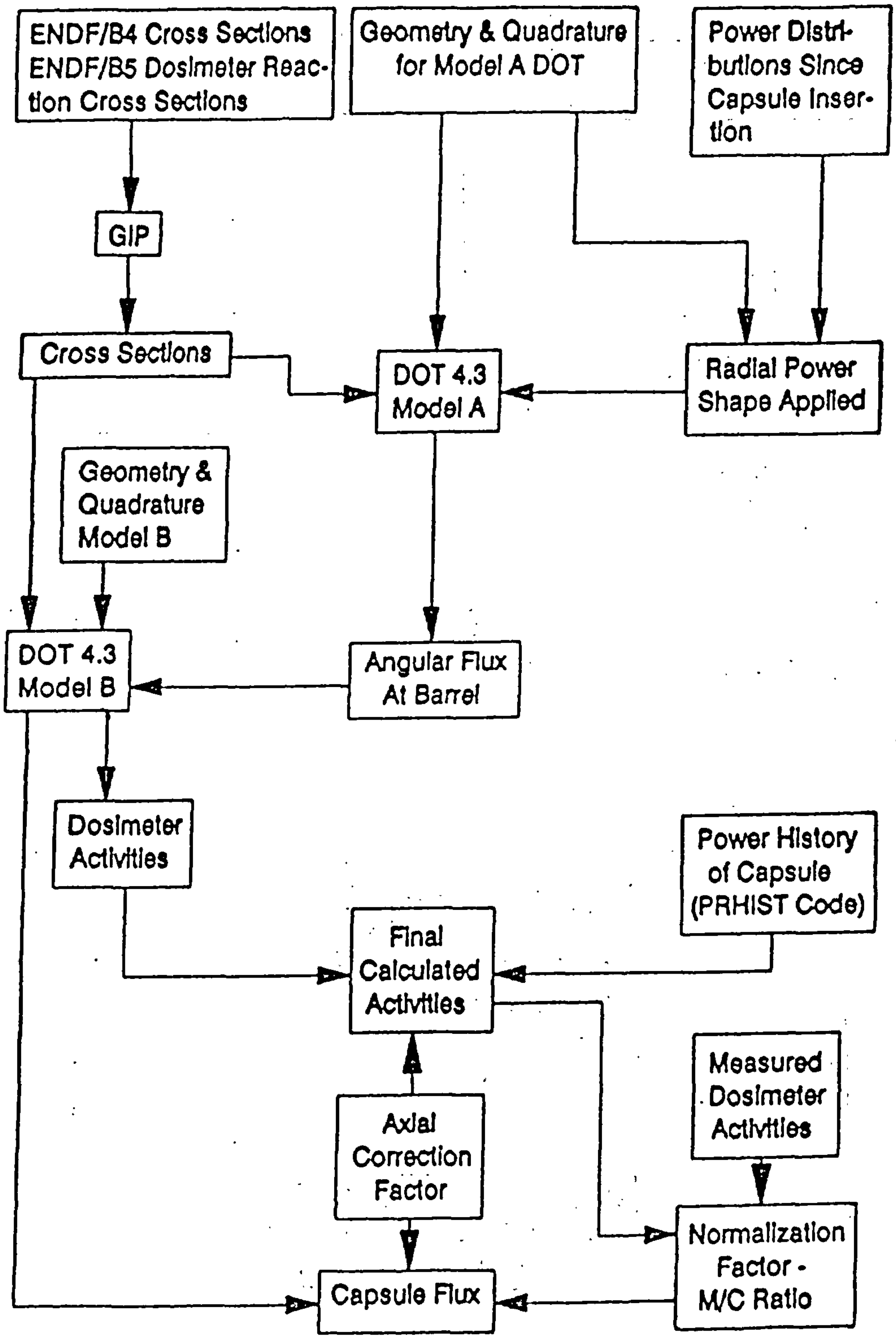


Figure D-2. Rationale for the Calculation of Neutron Flux in the Reactor Vessel

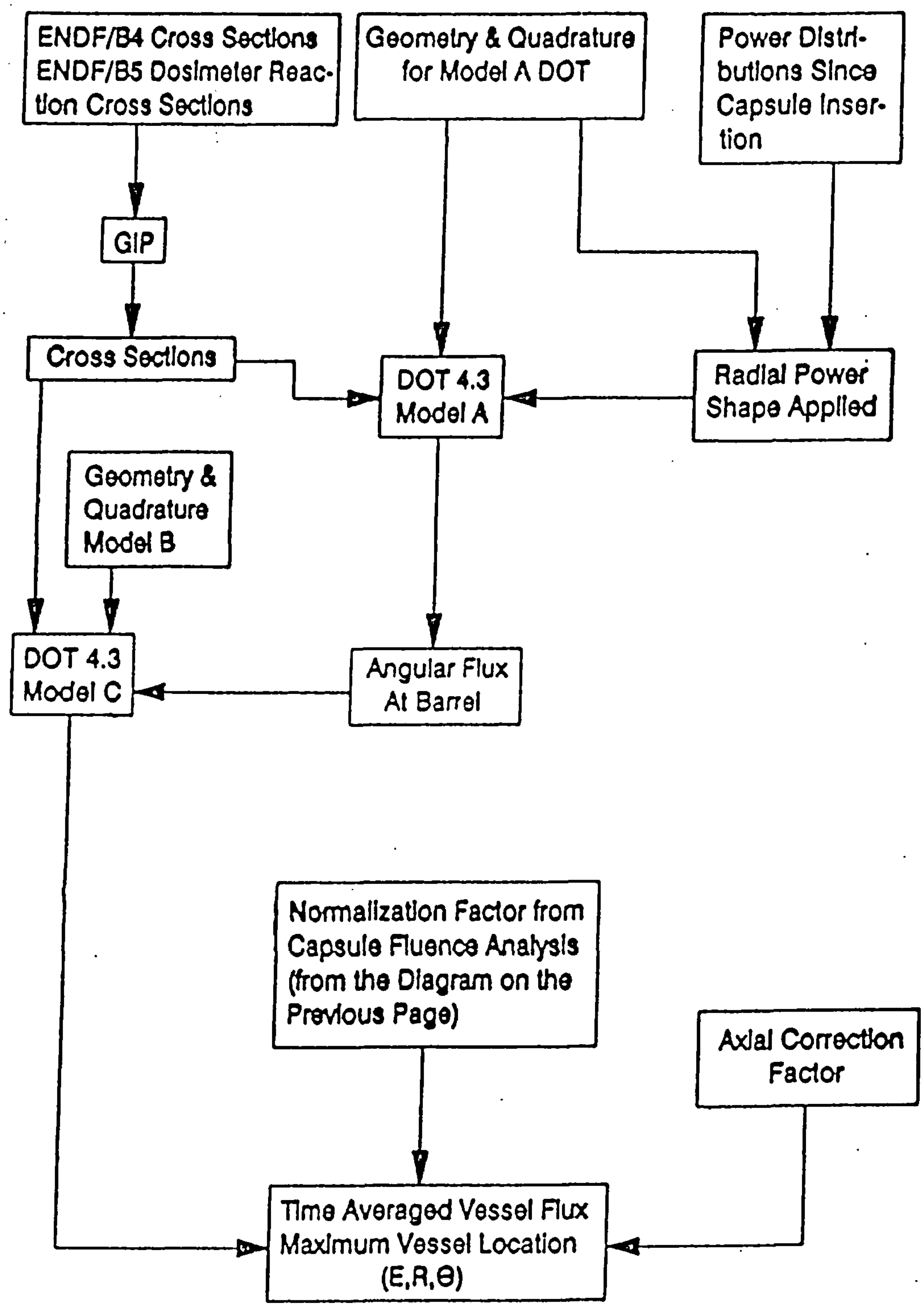
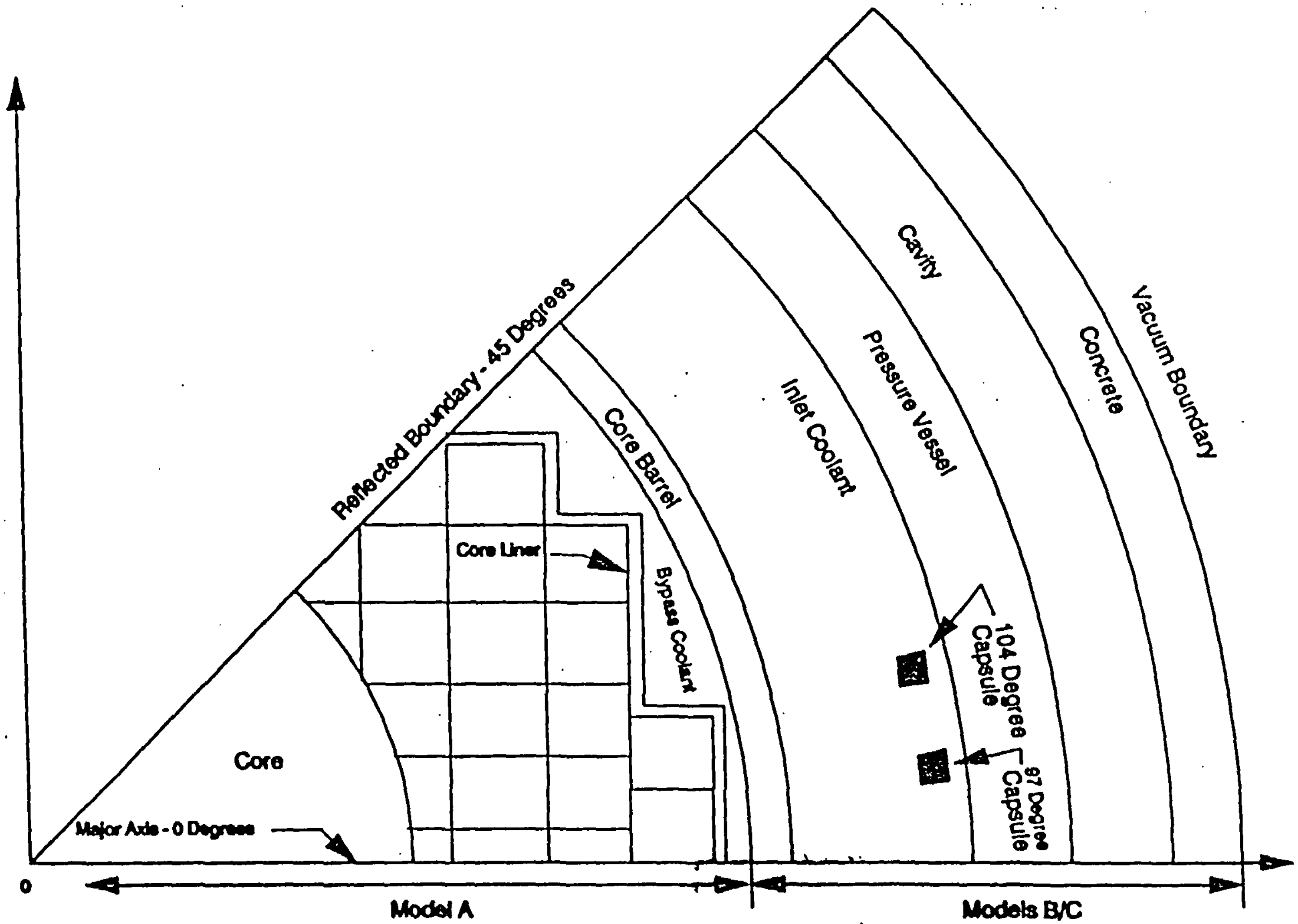


Figure D-3. Plan View Through Reactor Core Midplane (Reference R-Θ Calculation Model)



APPENDIX E

Capsule Dosimetry Data

Table E-1 lists the characteristics of the neutron dosimeters. Tables E-2 and E-3 show the measured activity per gram of target material (i.e., per gram of uranium, nickel, etc.) for each capsule's dosimeters. Activation cross sections for the various materials were flux-weighted with the ^{235}U fission spectrum shown in Table E-4.

Table E-1. Detector Composition and Shielding

Detector Material	Shielding	Reaction
Ni Wire	Cd	$^{58}Ni(n,p)^{58}Co$
Co Wire	Bare	$^{59}Co(n,\gamma)^{60}Co$
Co Wire	Cd	$^{59}Co(n,\gamma)^{60}Co$
Fe Wire	Bare	$^{54}Fe(n,p)^{54}Mn$
Cu Wire	Bare	$^{63}Cu(n,\alpha)^{60}Co$
U_3O_8	Cd	$^{238}U(n,f)^{137}Cs$
U_3O_8	Bare	$^{238}U(n,f)^{137}Cs$
Ti Wire	Cd	$^{46}Ti(n,p)^{46}Sc$

Table E-2. Measured Specific Activities (Unadjusted) for Dosimeters in 97° Capsule

Detector Material	Dosimeter Reaction	Dosimeter Activity, (μCi/gm of Target) Upper	Center	Lower
Ni Wire	$^{58}Ni(n,p)^{58}Co$	1808	1768	1929
Co Wire(b)	$^{59}Co(n,\gamma)^{60}Co$	4.357E+5	4.895E+5	3.887E+5
Co Wire(sh)	$^{59}Co(n,\gamma)^{60}Co$	5.163E+4	4.442E+4	4.976E+4
Fe Wire	$^{54}Fe(n,p)^{54}Mn$	1462	1374	1445
Cu Wire	$^{63}Cu(n,\alpha)^{60}Co$	8.519	7.907	8.706
U Powder(sh)	$^{238}U(n,f)^{137}Cs$	7.946	7.941	8.424
U Powder(b)	$^{238}U(n,f)^{137}Cs$	16.00	--	8.317
Ti Wire	$^{46}Ti(n,p)^{46}Sc$	332.3	354.9	410.5

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

Group No.	Upper Energy (eV)	$^{46}Ti(n,p)$	$^{238}U(n,f)$	$^{54}Fe(n,p)$	$^{58}Ni(n,p)$	$^{63}Cu(n,\alpha)$	$^{59}Co(n,\gamma)$
1	1.733+7	2.407-1	1.215+0	2.803-1	3.215-1	3.641-2	7.968-4
2	1.419+7	2.667-1	1.033+0	4.260-1	4.980-1	4.535-2	8.380-4
3	1.221+7	2.600-1	9.851-1	4.728-1	5.734-1	5.360-2	7.633-4
4	1.000+7	2.356-1	9.933-1	4.769-1	5.971-1	3.842-2	6.978-4
5	8.607+6	2.043-1	9.898-1	4.759-1	5.988-1	1.926-2	9.431-4
6	7.408+6	1.555-1	8.240-1	4.687-1	5.845-1	9.389-3	2.214-3
7	6.065+6	9.645-2	5.588-1	4.266-1	5.141-1	2.956-3	2.455-3
8	4.966+6	3.766-2	5.452-1	3.041-1	3.847-1	4.568-4	2.871-3
9	3.679+5	5.573-3	5.292-1	1.998-1	2.424-1	3.600-5	3.269-3
10	3.012+6	4.747-4	5.282-1	1.371-1	1.674-1	5.844-6	3.523-3
11	2.723+6	6.816-6	5.365-1	8.061-2	1.232-1	1.692-6	3.772-3
12	2.466+6	1.100-6	5.398-1	5.715-2	9.340-2	6.645-7	3.938-3
13	2.365+6	3.770-7	5.404-1	5.134-2	8.278-2	4.712-7	4.006-3
14	2.346+6	3.427-7	5.410-1	4.564-2	7.227-2	3.305-7	4.090-3
15	2.231+6	2.326-7	5.358-1	2.892-2	4.600-2	1.124-7	4.337-3
16	1.921+6	8.518-8	4.799-1	8.181-3	2.440-2	1.500-8	4.931-3
17	1.653+6	0.000-0	3.154-1	2.933-3	1.206-2	0.000-0	6.222-3
18	1.353+6	0.000-0	4.480-2	6.824-4	3.758-3	0.000-0	8.205-3
19	1.003+6	0.000-0	1.296-2	5.308-5	1.362-3	0.000-0	7.473-3
20	8.209+5	0.000-0	3.820-3	4.367-6	1.156-3	0.000-0	6.519-3
21	7.427+5	0.000-0	1.553-3	6.842-7	9.891-4	0.000-0	6.905-3
22	6.081+5	0.000-0	6.233-4	1.097-7	7.958-4	0.000-0	7.598-3
23	4.393+5	0.000-0	2.846-4	8.051-8	6.086-4	0.000-0	9.233-3
24	2.688+5	0.000-0	1.635-4	5.615-8	4.483-4	0.000-0	8.724-3
25	2.972+5	0.000-0	1.001-4	3.448-8	3.058-4	0.000-0	1.058-2
26	1.835+5	0.000-0	7.720-5	1.197-8	1.577-4	0.000-0	1.322-2
27	1.111+5	0.000-0	6.115-5	0.000-0	6.464-5	0.000-0	1.780-2
28	6.738+4	0.000-0	6.174-5	0.000-0	7.780-6	0.000-0	3.155-2
29	4.087+4	0.000-0	6.984-5	0.000-0	0.000-0	0.000-0	3.211-2
30	3.183+4	0.000-0	7.894-5	0.000-0	0.000-0	0.000-0	3.892-2

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

Group No.	Upper Energy (eV)	^{46}Ti(n,p)	^{238}U(n,f)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{63}Cu(n,α)	^{59}Co(n,γ)
31	2.606+4	0.000-0	8.361-5	0.000-0	0.000-0	0.000-0	9.668-2
32	2.418+4	0.000-0	8.624-5	0.000-0	0.000-0	0.000-0	3.587-2
33	2.188+4	0.000-0	9.269-5	0.000-0	0.000-0	0.000-0	5.816-2
34	1.505+4	0.000-0	9.681-5	0.000-0	0.000-0	0.000-0	9.916-2
35	7.108+3	0.000-0	3.211-5	0.000-0	0.000-0	0.000-0	1.906-1
36	3.355+3	0.000-0	3.380-9	0.000-0	0.000-0	0.000-0	4.447-2
37	1.585+3	0.000-0	8.094-4	0.000-0	0.000-0	0.000-0	2.462-2
38	4.540+2	0.000-0	1.279-5	0.000-0	0.000-0	0.000-0	2.424-1
39	2.145+2	0.000-0	1.857-3	0.000-0	0.000-0	0.000-0	7.332+1
40	1.013+2	0.000-0	2.814-5	0.000-0	0.000-0	0.000-0	2.782+0
41	3.727+1	0.000-0	1.518-4	0.000-0	0.000-0	0.000-0	1.730+0
42	1.068+1	0.000-0	7.968-5	0.000-0	0.000-0	0.000-0	2.361+0
43	5.044+0	0.000-0	5.481-7	0.000-0	0.000-0	0.000-0	3.533+0
44	1.855+0	0.000-0	5.600-7	0.000-0	0.000-0	0.000-0	5.344+0
45	8.764-1	0.000-0	1.100-6	0.000-0	0.000-0	0.000-0	7.722+0
46	4.140-1	0.000-0	2.000-6	0.000-0	0.000-0	0.000-0	1.464+1
47	1.000-1	0.000-0	4.300-6	0.000-0	0.000-0	0.000-0	2.922+1

[a]ENDF/B5 values that have been flux weighted (over BUGLE 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

Tension Test Stress-Strain Curves

Figure F-1. Tension Test Stress-Strain Curve for Base Metal Plate Heat M-1004-2, Specimen No. 2L6, Tested at 70F

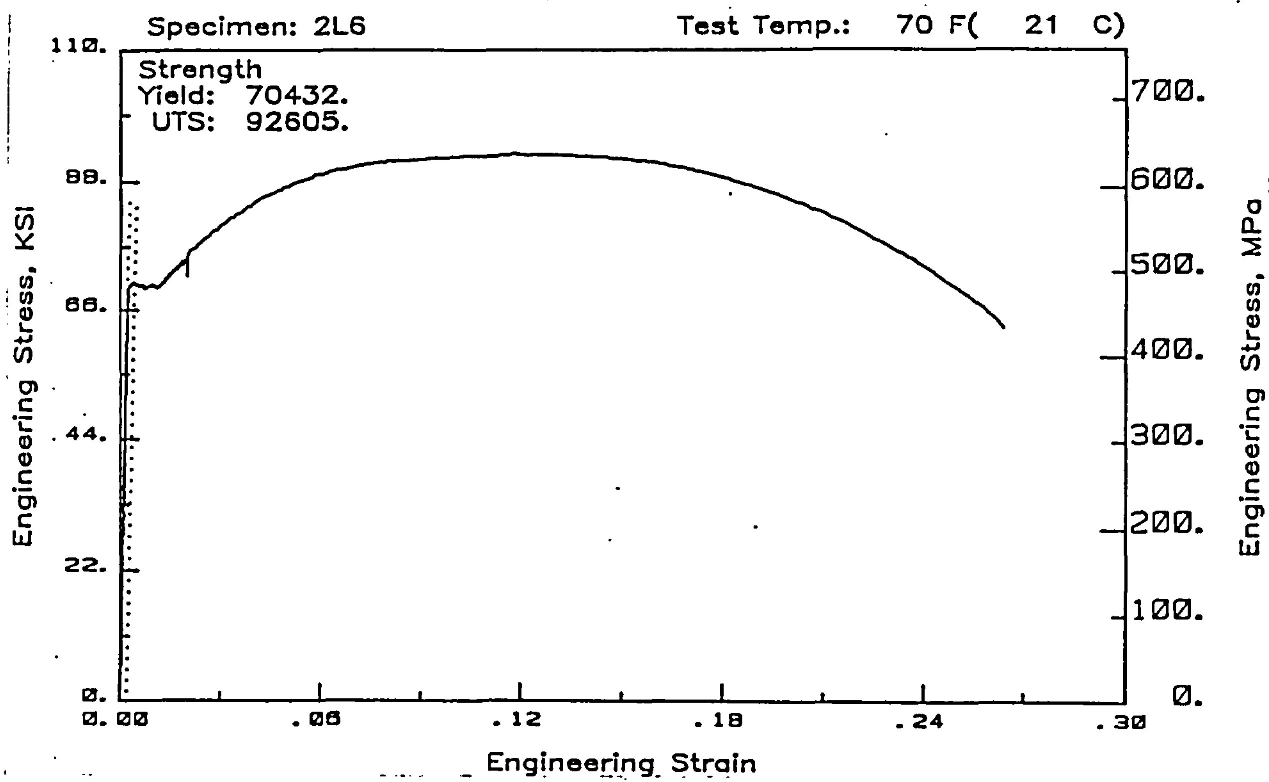


Figure F-2. Tension Test Stress-Strain Curve for Base Metal Plate Heat M-1004-2, Specimen No. 2K5, Tested at 250F

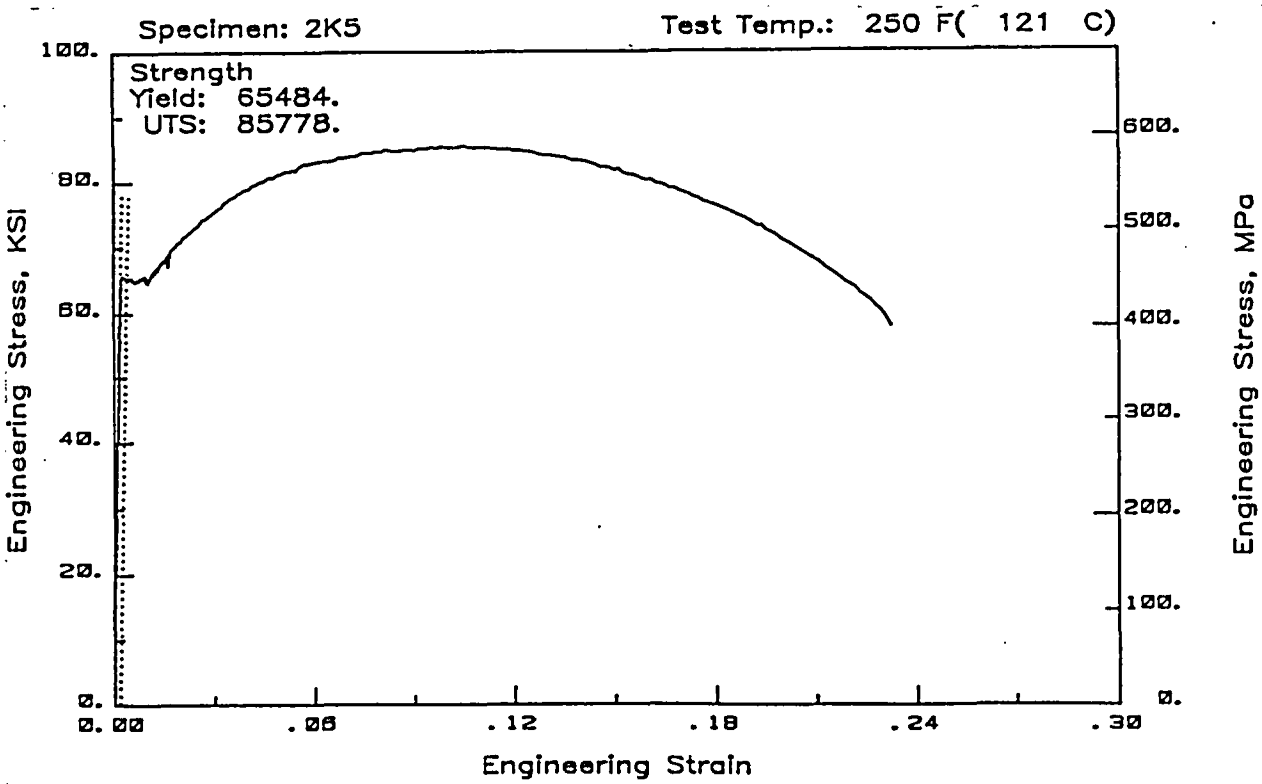


Figure F-3. Tension Test Stress-Strain Curve for Base Metal Plate Heat M-1004-2, Specimen No. 2K2, Tested at 550F

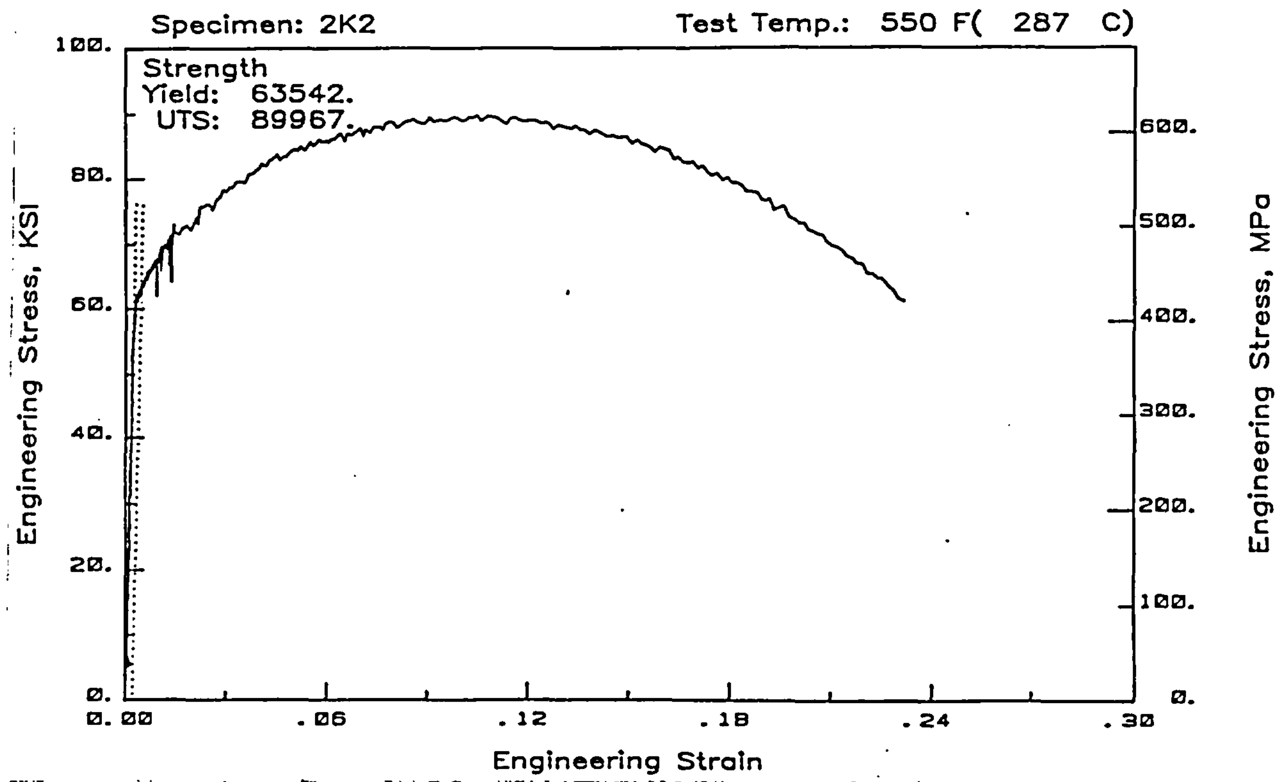


Figure F-4. Tension Test Stress-Strain Curve for Base Metal Heat-Affected Zone, Heat M-1004-2, Specimen No. 4K3, Tested at 70F

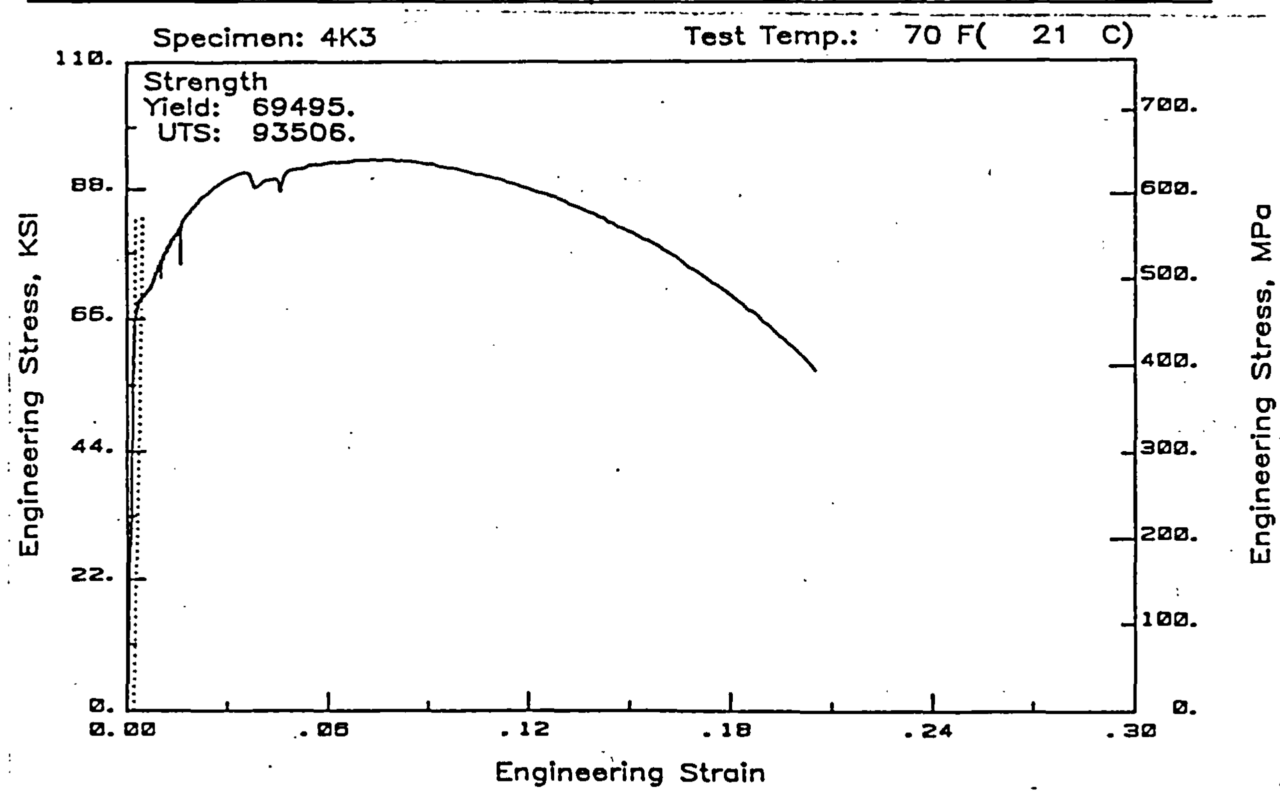


Figure F-5. Tension Test Stress-Strain Curve for Base Metal Heat-Affected Zone, Heat M-1004-2, Specimen No. 4KK, Tested at 250F

NO DATA - SEE SECTION 5.3

Figure F-6. Tension Test Stress-Strain Curve for Base Metal Heat-Affected Zone, Heat M-1004-2, Specimen No. 4J4, Tested at 550F

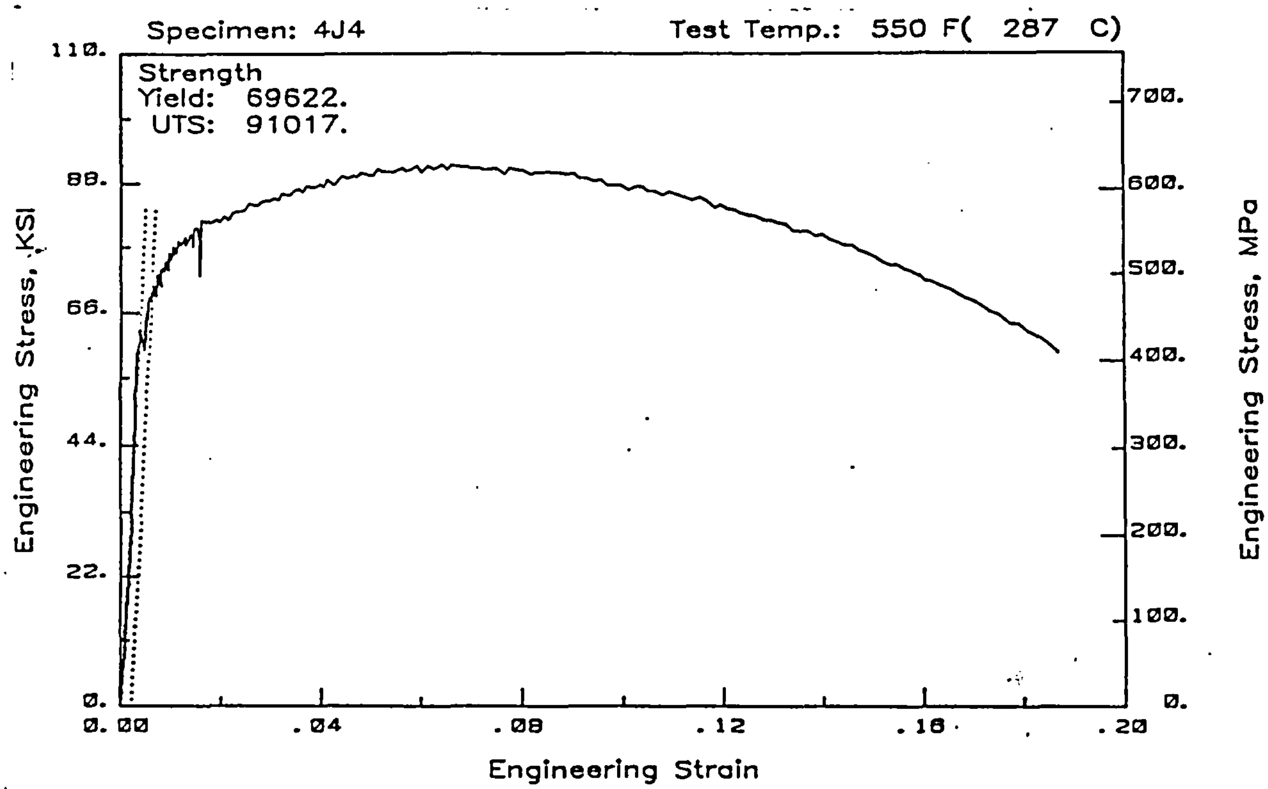


Figure F-7. Tension Test Stress-Strain Curve for Weld Metal 88114/0145, Specimen No. 3JM, Tested at 70F

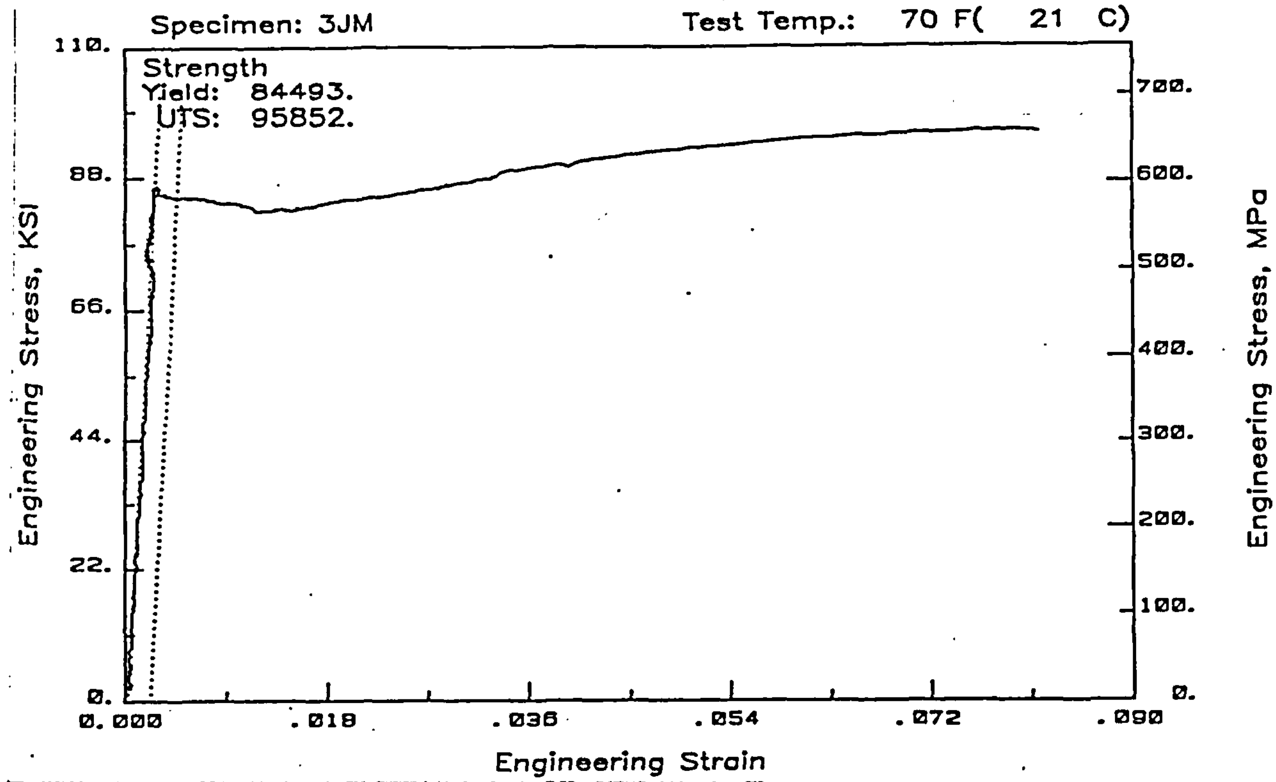
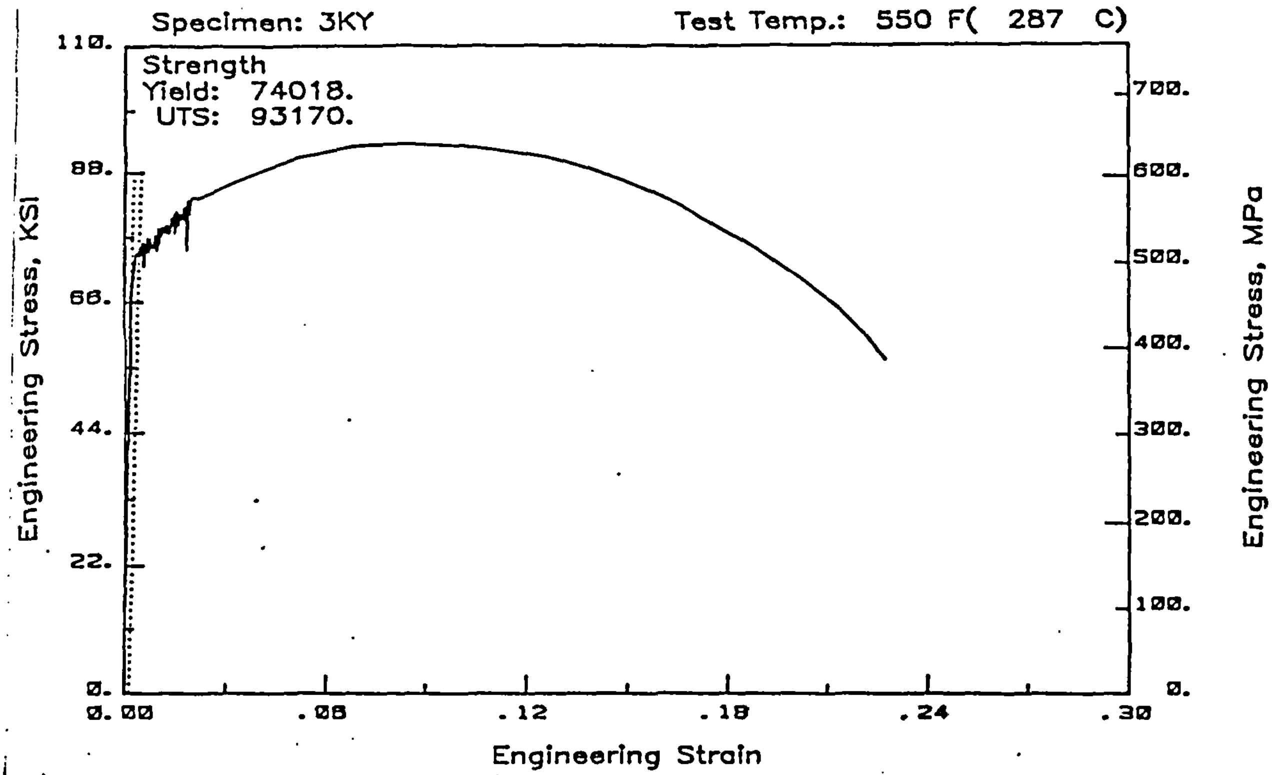


Figure F-8. Tension Test Stress-Strain Curve for Weld Metal 88114/0145, Specimen No. 3KK, Tested at 250F

NO DATA - SEE SECTION 5.3

Figure F-9. Tension Test Stress-Strain Curve for Weld Metal 88114/0145, Specimen No. 3KY, Tested at 550F



APPENDIX G

References

1. Program for Irradiation Surveillance of Waterford Unit Three Reactor Vessel Materials, C-NLM-003, Revision 1, October 30, 1974.

2. Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Waterford Unit 3 Reactor Vessel Materials, TR-C-MCS-001, Combustion Engineering, Inc., Windsor, Connecticut.

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

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

5. 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).

6. ASTM Standard E208, "Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

7. A. G. Ragl, et al., Louisiana Power and Light Waterford Steam Electric Station Unit No. 3, Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program, TR-C-MCS-002, Combustion Engineering, Inc., Windsor, Connecticut, August 1977.

8. ASTM Standard E8, "Standard Methods of Tension Testing of Metallic Materials," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

9. ASTM Standard E21, "Standard Recommended Practice for Elevated Temperature Tension Tests of Metallic Materials," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

10. ASTM Standard E184, "Standard Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

11. ASTM Designation E23-72, "Method for Notched Bar Impact Testing of Metallic Materials," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

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

13. ASTM Designation A370-77, "Methods and Definitions for Mechanical Testing of Steel Products," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

14. ASTM Designation E23-86, "Methods for Notched Bar Impact Testing of Metallic Materials," in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

15. ASTM Designation E185-XX (to be released), Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

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

17. B&W's Version of DOTIV Version 4.3, Filepoint 2A4, "One- and Two-Dimensional Transport Code System," Oak Ridge National Laboratory, Distributed by the Radiation Shielding Information Center as CC-429, November 1, 1983.

18. "Bugle - 80 Coupled 47 Neutron, 20 Gamma-Ray, P_3, Cross Section Library for LWR Shielding Calculations," Radiation Information Shielding Center, DLC-75.

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

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

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

22. Code of Federal Regulations, Title 10, Part 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.