WCAP-9357

WESTINGHOUSE CLASS 3
CUSTOMER DESIGNATED DISTRIBUTION

ANALYSIS OF CAPSULE R FROM THE WISCONSIN ELECTRIC POWER COMPANY POINT BEACH NUCLEAR PLANT UNIT NO. 1 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

S. E. Yanichko
S. L. Anderson

August 1978

APPROVED: J. N. Chirigos, Manager
Structural Materials Engineering

Prepared by Westinghouse for the Wisconsin Electric Power Company

Work Performed Under EIZP-101

Although the information contained in this report is nonproprietary, no distribution shall be made outside Westinghouse or its Licensees without the customer's approval.

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

7811140131
50-266-P
50-301-P

TABLE OF CONTENTS

Section	Title	
1	SUMMARY	1-1
2	INTRODUCTION	2-1
3	BACKGROUND	3-1
4	DESCRIPTION OF PROGRAM	4-1
5	TESTING OF SPECIMENS FROM CAPSULE R	5-1
	5-1 Charpy V-Notch Impact Test Results	5-2
	5-2 Tensile Test Results	5-4
	5-3 Wedge Opening Loading Tests	5-4
6	NEUTRON DOSIMETRY ANALYSIS	6-1
	6-1 Description of Neutron Flux Monitors	6-1
	6-2 Analytical Procedures	6-4
	6-3 Results of Analysis	6-7

LIST OF ILLUSTRATIONS

Figure	Title	Page
4-1	Arrangement of Surveillance Capsules in the Point Beach Unit No. 1 Reactor Vessel (Lead Factors for the Capsules Shown in Parentheses)	4-3
4-2	Capsule R Schematic Diagram Showing Designed Arrangement of Specimens, Thermal Monitors, and Dosimeter Placement and Orientation With Respect to the Core and Vessel Wall	4-7/4-8
5-1	Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811	5-5
5-2	Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423	5-6
5-3	Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Weld Metal	5-7
5-4	Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Weld-Heat-Affected-Zone Metal	5-8
5-5	Charpy V-Notch Impact Data for SA302 Grade B ASTM Correlation Monitor Material	5-9
5-6	Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811	5-10
5-7	Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423	5-11
5-8	Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 Weld Metal	5-12
5-9	Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 Weld Heat-Affected-Zone Metal	5-13
5-10	Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 ASTM Correlation Monitor Material	5-14
5-11	Point Beach Unit No. 1 Material 30 FT-LB Transition Temperature Increases as Compared to Westinghouse Predictions	5-15
5-12	Tensile Properties for the Point Beach Unit No. 1 Pressure Vessel Weld Metal	5-16
5-13	Tensile Properties for the Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811	5-17
5-14	Tensile Properties for the Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423	5-18

LIST OF ILLUSTRATIONS (cont)

Figure	Title	Page
5-15	Fractured Tensile Specimens From Point Beach Unit No. 1 Weld Metal	5-19
5-16	Fractured Tensile Specimens From Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811	5-20
5-17	Fractured Tensile Specimens From Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423	5-21
5-18	Typical Stress-Strain Curve for Tension Specimens (Tension Specimen No. A7)	5-22
6-1	Point Beach Unit No. 1 Reactor Geometry	6-2
6-2	Calculated Azimuthal Distribution of Maximum Fast Neutron Flux (E > 1 Mev) Within the Point Beach Unit No. 1 Reactor Vessel	6-9
6-3	Relative Axial Variation of Fast Neutron Flux (E > 1.0 Mev) Incident on the Point Beach Unit No. 1 Reactor Vessel	6-10
6-4	Calculated Maximum End-of-Life Fast Neutron Fluence (E > 1 Mev) as a Function of Radius Within the Point Beach Unit No. 1 Reactor Vessel	6-11

LIST OF TABLES

Table	Title	Page
4-1	Chemistry and Heat Treatment of Material Representing the Core Region Shell Plates and Weld Metal From the Point Beach Unit No. 1 Reactor Vessel	4-4
4-2	Chemistry and Heat Treatment of Surveillance Material Representing 6-Inch-Thick A302B ASTM Correlation Monitor Material	4-5
5-1	Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811 Irradiated at 550°F, Fluence 2.22×10^{19} n/cm^2 (E > 1 Mev)	5-23
5-2	Charpy-V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423 Irradiated at 550°F, Fluence 2.22×10^{19} n/cm^2 (E > 1 Mev)	5-23
5-3	Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Weld Metal Irradiated at 550°F, Fluence 2.22×10^{19} n/cm^2 (E > 1 Mev)	5-24
5-4	Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Weld-Heat-Affected-Zone Metal Irradiated at 550°F, Fluence 2.22×10^{19} n/cm^2 (E > 1 Mev)	5-24
5-5	Charpy V-Notch Impact Data for the Point Beach Unit No. 1 ASTM SA302 Grade B Correlation Monitor Material Irradiated at 550°F, Fluence 2.22×10^{19} n/cm^2 (E > 1 Mev)	5-25
5-6	The Effect of 550°F Irradiation at 2.22×10^{19} n/cm^2 (E > 1 Mev) on the Notch Toughness Properties of the Point Beach Unit No. 1 Reactor Vessel Impact Test Specimens	5-26
5-7	Summary of Point Beach Unit No. 1 Reactor Vessel Surveillance Capsule Charpy Impact Test Results	5-27
5-8	Irradiated Tensile Properties for the Point Beach Unit No. 1 Pressure Vessel Materials	5-28
6-1	Neutron Flux Monitors Contained Within Capsule R	6-3
6-2	Irradiation History of Capsule R	6-12
6-3	Spectrum-Averaged Reaction Cross Sections Used in Fast Neutron Flux Derivation	6-15

LIST OF TABLES (cont)

Table	Title	Page
6-4	Results of Fast Neutron Dosimetry for Capsule R	6-16
6-5	Results of Thermal Neutron Dosimetry for Capsule R	6-17
6-6	Calculated Fast Neutron Flux and Lead Factors for Capsule R	6-17

SECTION 1
SUMMARY

The analysis which compared unirradiated with irradiated material properties of the reactor vessel material contained in the third surveillance capsule, designated R, from the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 1 reactor pressure vessel led to the following conclusions:

- The capsule received an average fast fluence of 2.22×10^{19} n/cm^2 (E > 1 Mev). The predicted fast fluence for the capsule was 1.80×10^{19} n/cm^2 (E > 1 Mev).

- The fast fluence of 2.22×10^{19} n/cm^2 resulted in a 205°F increase in the 50 ft-lb reference nil-ductility transition temperature (RT_{NDT}) of the weld metal, which is representative of the most limiting material in the core region of the reactor vessel. The intermediate pressure vessel shell plate A9811 and lower shell plate C1423 exhibited a 50 ft-lb transition temperature increase of 105°F and 50°F, respectively (specimens oriented parallel to the rolling direction of the plates). The weld-heat-affected-zone material exhibited a 50 ft-lb transition increase of 60°F.

- The average upper shelf impact energy of the weld metal decreased from 65 to 51 ft-lb.

- An increase of 110°F in the 30 ft-lb transition temperature was determined for the ASTM A302B reference correlation monitor material contained in the capsule.

- Transition temperature increases for the surveillance materials were essentially the same as those obtained from the second capsule irradiated to 7.05×10^{18} n/cm^2, indicating that a limiting or steady state value has been reached at a level well below predicted trend curves.

- The following end-of-life projected fast neutron fluences for the reactor vessel, based on 32 full-power years of operation at 1518 Mw as derived from both calculated and measured surveillance capsule results, were determined.

	Fast Neutron Fluence (n/cm^2)	
Vessel Location	**Calculated**	**Measured**
Inner surface	3.9×10^{19}	4.90×10^{19}
1/4 Thickness	2.4×10^{19}	3.05×10^{19}
3/4 Thickness	7.5×10^{18}	9.50×10^{18}

The difference of approximately 25 percent in the calculated versus the measured fast neutron fluence is due in part to the ±10 percent uncertainty in the measured activities of the fast neutron iron monitors. The remainder of the difference may be attributed to uncertainties in the monitor reaction cross sections, analytical approximation and differences between the actual core spatial power distribution in the peripheral fuel assemblies and that assumed in the analysis.

SECTION 2
INTRODUCTION

This report presents the results of the examination of Capsule R, the third capsule of the continuing surveillance program, which monitors the effects of neutron irradiation on the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 1 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the Point Beach Unit No. 1 reactor pressure vessel materials was designed and recommended by the Westinghouse Electric Corporation. A description of the surveillance program and the preirradiation mechanical properties of the reactor vessel materials are presented in WCAP-7513.[1] The surveillance program, which was planned to cover the 40-year life of the reactor pressure vessel, was based on ASTM E-185-66, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."[2]

Post-irradiation data have been obtained from the third material surveillance capsule (Capsule R) removed from the Point Beach Unit No. 1 reactor vessel. This report summarizes the tests and the results, and discusses the analysis of the data.

---

1. Yanichko, S. E., "Wisconsin Michigan Power Co. Point Beach Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-7513, 1970.

2. ASTM Designation E185-66, "Surveillance Tests on Structural Materials in Nuclear Reactors," in ASTM Standards (1967), Part 31, Physical and Mechanical Testing of Metals – Metallography, Nondestructive Testing, Fatigue, Effect of Temperature, pp. 638-642, Am. Soc. for Testing and Materials, Philadelphia, Pa. 1967.

SECTION 3
BACKGROUND

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as SA302 Grade B (base material of the Unit No. 1 reactor pressure vessel beltline) are well-documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness under certain conditions of irradiation.

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

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

RT_{NDT}, and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel steel can be

monitored by a reactor surveillance program such as the Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program,[1] in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the Charpy V-notch temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is used to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effect of irradiation on the reactor vessel materials.

1. Yanichko, S. E., "Wisconsin Michigan Power Co. Point Beach Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-7513, 1970.

SECTION 4
DESCRIPTION OF PROGRAM

Six surveillance capsules for monitoring the effects of neutron exposure on the Point Beach Nuclear Plant Unit No. 1 reactor pressure vessel core region material were inserted in the reactor vessel prior to initial plant startup. The six capsules were positioned in the reactor vessel between the thermal shield and the vessel wall at locations shown in figure 4-1. The vertical center of the capsules is opposite the vertical center of the core.

Capsule R was removed in the autumn of 1977 after approximately 7 calendar years (5.1 effective full-power years) of plant operation. This capsule contained Charpy V-notch impact, tensile, and WOL specimens (shown in WCAP-7513) from the intermediate and lower shell ring plates, weld metal representative of the core region of the reactor vessel, and Charpy V-notch specimens from weld-heat-affected-zone (HAZ) material. The capsule also contained Charpy V-notch specimens from the 6-inch-thick ASTM correlation monitor material (A302 Grade B) furnished by the U. S. Steel Corporation. The chemistry and heat treament of the surveillance material is presented in tables 4-1 and 4-2.

All test specimens were machined from the 1/4 thickness location of the plates. Test specimens represent material taken at least one plate thickness from the quenched end of the plate. All base metal Charpy V-notch and tensile specimens were oriented with the longitudinal axis of the specimen parallel to the principal rolling direction of the plates. The WOL test specimens were machined with the simulated crack of the specimen perpendicular to the surfaces and rolling direction of the plates.

Charpy V-notch specimens from the weld metal were oriented with the longitudinal axis of the specimens transverse to the weld direction. Tensile specimens were oriented with the longitudinal axis of the speciment parallel to the weld.

Capsule R contained dosimeter wires of copper, nickel, and aluminum-cobalt (cadmium-shielded and unshielded). In addition, the capsule contained cadmium-shielded dosimeters of Np^{237} and U^{238}, located as shown in figure 4-2.

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

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

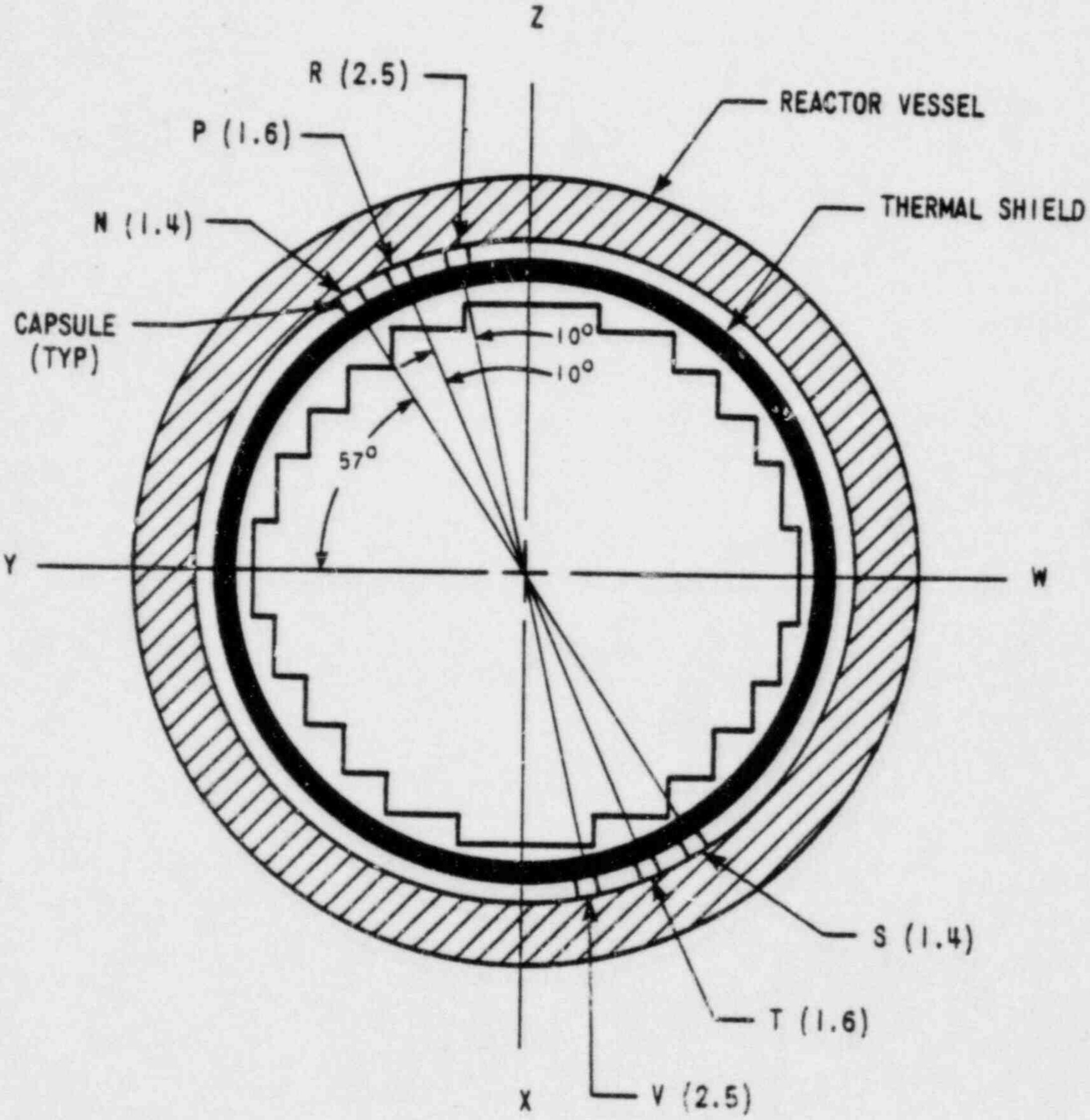


Figure 4-1. Arrangement of Surveillance Capsules in the Point Beach Unit No. 1 Reactor Vessel (Lead Factors for the Capsules Shown in Parentheses)

TABLE 4-1

CHEMISTRY AND HEAT TREATMENT OF MATERIAL REPRESENTING THE CORE REGION SHELL PLATES AND WELD METAL FROM THE POINT BEACH UNIT NO. 1 REACTOR VESSEL

CHEMICAL ANALYSES (PERCENT)

Element	Plate A9811	Plate C1423	Weld Metal
C	0.19	0.21	0.09
Mn	1.42	1.37	1.47
P	0.010	0.014	0.019
S	0.020	0.019	0.024
Si	0.25	0.25	0.49
Mo	0.48	0.46	0.39
Cu	0.19	0.11	0.18-0.24
Ni	---	---	0.57
Cr	---	---	0.13
Al	---	---	0.035
N_2	---	---	0.016
V	---	---	0.001
Sn	---	---	0.004
Ti	---	---	0.001
As	---	---	0.004
Co	---	---	0.001 [a]
Zr	---	---	0.001 [a]
Sb	---	---	0.001 [a]
Zn	---	---	0.001 [a]
B	---	---	0.003 [a]

HEAT TREATMENT

Plate A9811	Heated at 1650°F, 7 hours, water-quenched Tempered at 1225°F, 7 hours, aircooled Stress-relieved at 1125°F, 11-1/4 hours, furnace-cooled
Plate C1423	Heated at 1650°F, 7 hours, water-quenched Tempered at 1225°F, 7 hours, aircooled Stress-relieved at 1125°F, 10-1/2 hours, furnace-cooled
Weldment	Stress-relieved at 1125°F, 11-1/4 hours, furnace-cooled

a. Not detected; the number indicates the minimum limit of detection.

TABLE 4-2

CHEMISTRY AND HEAT TREATMENT OF SURVEILLANCE MATERIAL REPRESENTING 6-INCH-THICK A302B ASTM CORRELATION MONITOR MATERIAL

CHEMICAL ANALYSIS (PERCENT)								
C	Mn	P	S	Mo	Si	Cu	Ni	Cr
0.24	1.34	0.011	0.023	0.51	0.23	0.20	0.18	0.11

HEAT TREATMENT

The 6-inch-thick plate was charged into a furnace operating at 110°F heated at a maximum rate of 63°F per hour to 1650°F, held at temperature for 4 hours, and water-quenched to 300°F. The plate was then recharged into a furnace operating at 700° to 750°F, heated at a maximum rate of 63°F per hour to 1200°F, and held at that temperature for 6 hours.

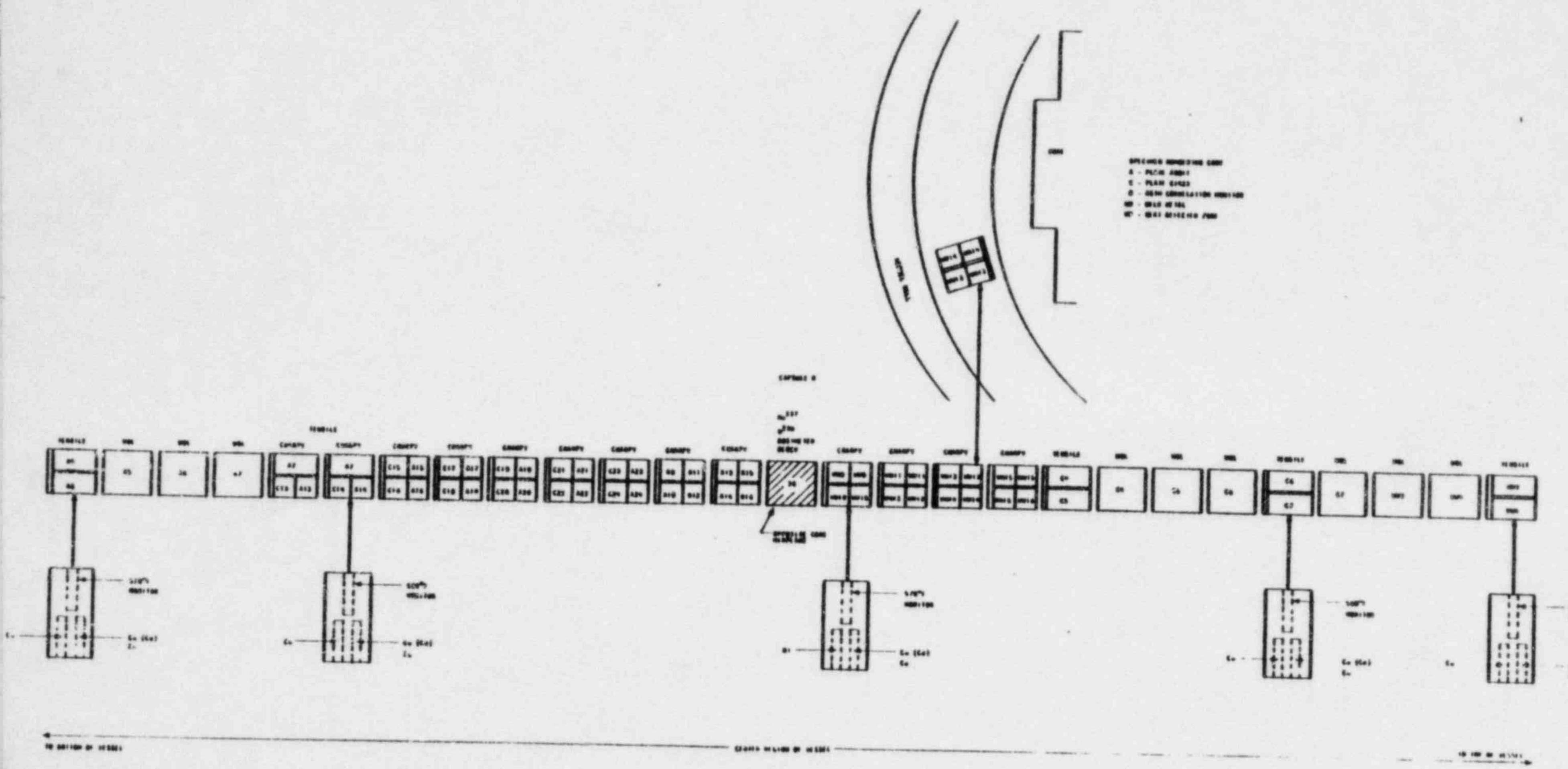


Figure 4-2. Capsule R Schematic Diagram Showing Designed Arrangement of Specimens, Thermal Monitors, and Dosimeter Placement and Orientation With Respect to the Core and Vessel Wall

SECTION 5
TESTING OF SPECIMENS FROM CAPSULE R

The post-irradiation mechanical testing of the Charpy V-notch and tensile specimens was performed at the Westinghouse Research and Development Laboratory with consultation by Westinghouse Nuclear Energy Systems personnel. Testing was performed in accordance with 10CFR50, Appendices G and H.

Upon receipt of the capsule at the laboratory, the specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in WCAP-7513.[1] No discrepancies were found.

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

A Tinius Olsen Model 74 impact test machine was used to test the irradiated Charpy V-notch specimens per ASTM E23-72, "Notched Bar Impact Testing of Metallic Materials." Before initiating tests on the irradiated Charpy-V specimens, the accuracy of the impact machine was checked with a set of standard specimens obtained from the Army Material and Mechanics Research Center in Watertown, Massachusetts. The results of the calibration testing showed that the machine was certified for Charpy V-notch impact testing.

The tensile tests were conducted on a screw-driven Instron testing machine of 20,000 lb capacity per ASTM E8-69, "Tension Testing of Metallic Materials" and ASTM E21-70, "Elevated Temperature Tension Tests of Metallic Materials." The crosshead speed was 0.5 inch per minute. The deformation of the specimen was measured with a strain gage extensometer. The extensometer was calibrated before testing with a Sheffield high magnification drum-type extensometer calibrator.

Elevated-temperature tensile tests were conducted in a split-tube furnace. The specimens were held at temperature a minimum of 20 minutes to stabilize the temperature prior to testing. Temperature was monitored with a chromel-alumel thermocouple in contact with the clevis-pin-type upper and lower specimen grips. Temperature was controlled within ±3°F.

---

1. Yanichko, S. E., "Wisconsin Michigan Power Co. Point Beach Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-7513, 1970.

The load-extension data were recorded on the testing-machine strip chart. The yield strength, ultimate tensile strength, and uniform elongation were determined from these charts. The reduction in area and total elongation were determined from specimen measurements.

5-1. CHARPY V-NOTCH IMPACT TEST RESULTS

The irradiated Charpy V-notch specimens represented the Point Beach Nuclear Plant Unit No. 1 reactor pressure vessel beltline plate material, weld and heat-affected zone (HAZ) material, and the ASTM reference correlation monitor material. The results are presented in figures 5-1 through 5-5. Table 5-6 summarizes the increase in the 30 and 50 ft-lb energy and 35-mil lateral expansion transition temperature, and the decrease in the upper shelf energy resulting from irradiation to 2.22×10^{19} n/cm^2.

Figures 5-1 and 5-2 give the test results obtained on the vessel beltline shell plate material. Tables 5-1 and 5-2 show a 30 and 50 ft-lb transition temperature increase of 105°F for plate A9811 and 50°F for plate C1423. Plate A9811 showed a 115°F increase in the 35-mil lateral expansion temperature which was in reasonable agreement with the 105°F increase in the 50 ft-lb transition temperature. Plate C1423 showed a 60°F increase in the 35-mil lateral expansion temperature as compared to a 50°F increase in the 50 ft-lb transition temperature. Plate A9811 exhibited a 7 ft-lb or 7 percent decrease in upper shelf energy while plate C1423 showed a 15 ft-lb or 12 percent increase in upper shelf energy.

Figure 5-3 and table 5-3 give the test results obtained on the weld metal. These results show that, in 30 and 50 ft-lb testing, the weld metal exhibited respective increases of 165°F and 205°F in transition temperature. The 200°F increase in the 35-mil lateral expansion temperature was nearly identical to the 205°F increase in the 50 ft-lb transition temperature. The upper shelf energy of the weld metal decreased 14 ft-lb or 21 percent.

The test results for the HAZ material are shown in table 5-4 and figure 5-4. A 70°F and 60°F transition temperature increase was obtained at the 30 and 50 ft-lb temperatures, respectively. The 35-mil lateral expansion temperature increased 90°F as compared to a 60°F increase in the 50 ft-lb transition temperature. The upper shelf of the HAZ decreased 23 ft-lb or 17 percent.

Figure 5-5 and table 5-5 present the test results obtained on the A302B ASTM reference correlation monitor material. Respective increases of 110° and 123°F in the 30 and 50 ft-lb transition temperatures were obtained for this material. The upper shelf energy of the correlation monitor material increased 5 ft-lb or 6 percent.

Charpy impact specimen fracture surfaces of the various Point Beach Unit No. 1 vessel material and the correlation monitor material are presented in figures 5-6 through 5-10.

Table 5-7 summarizes the Charpy impact test results for the first and second capsules [1,2] and the third capsule. The results show that, as a result of additional irradiation from 7.05×10^{18} to 2.22×10^{19} n/cm^2, (1) essentially no additional transition temperature increase resulted for plate C1423, the weld metal, and the weld HAZ, and (2) only a 15°F increase resulted for plate A9811 and the correlation monitor material. A comparison of the 30 ft-lb transition temperature increases with predicted temperature increases (figure 5-11) indicates that long-time irradiations (5 to 7 years) received by the second and third capsules do not result in as much radiation damage as predicted from trend curves developed primarily from experimental data from short time irradiations (< 1 year) and first time surveillance capsule removals (< 2 years). The 110°F increase in the 30 ft-lb transition temperature of the correlation monitor material irradiated to 2.22×10^{19} n/cm^2 falls below the trend band established for this material [3], thus also seeming to indicate that irradiation-induced increases in transition temperature reach a limiting or steady state value well below predicted trend curves.

The weld metal used in the surveillance program is representative of but not identical with any of the core region welds in the Point Beach Unit No. 1 vessel [4]. This is because the particular heat of weld wire and lot of flux used to fabricate the surveillance weldment differs from that used in fabricating the vessel core region welds. A recent weld metal study by Westinghouse establishes that the surveillance weldment for the Florida Power and Light Co. Turkey Point Unit No. 3 surveillance program was fabricated using the same heat of weld wire and lot of flux as that used to fabricate the core region girth weld of the Point Beach Unit No. 1 vessel. Post-irradiation tests performed on the Turkey Point No. 3 surveillance weld [5] resulted in respective increases of 155° and 190°F in the 30 and 50 ft-lb transition temperatures after irradiation to 5.68×10^{18} n/cm^2.

The fact that these increases are in agreement with those obtained on the Point Beach Unit No. 1 surveillance weld, and that the weld metal is the controlling vessel material for normal heatup and cooldown operations of the plant, gave a basis for constructing limit curves for those operations. [4] The curves, applicable for up to 10 Effective Full-Power Years (EFPY), and prepared with the use of Westinghouse trend curves for adjusting the reference transition temperature, are considered adequate as bounds for continued safe operation of the plant.

---

1. Perrin, J. S., et al., Final Report on "Point Beach Nuclear Plant Unit No. 1 Pressure Vessel Surveillance Program: Evaluation of Capsule V," to Wisconsin Electric Power Company by Battelle Laboratories, Columbus, Ohio, June 1973.
2. Yanichko, S. E. and Anderson, S. L., "Analysis of Capsule S from the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-8739, 1976.
3. ASTM DS 54, Radiation Effects Information Generated on the ASTM Reference Correlation Monitor Steels, ASTM, Philadelphia, 1974.
4. Phillips, J. H., "Heatup and Cooldown Limit Curves for the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Unit No. 1 Nuclear Power Plant," WCAP-8743, 1976.
5. Yanichko, S. E., et al., "Analysis of Capsule T from the Florida Power and Light Company Turkey Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-8631, 1975.

Because the surveillance data indicate that the Westinghouse curves conservatively predict the 30 ft-lb transition temperature increase, predicted adjusted reference temperatures based on the use of Westinghouse curves will be applied in various vessel analyses.

5-2. TENSILE TEST RESULTS

Table 5-8 and figures 5-12 through 5-14 give the results of the tensile test. The weld metal and plates A9811 and C1423 were tested at a range of temperatures bounded by the ambient and 550°F. Increases in yield strength caused by irradiation were as follows:

Increase in Yield Strength From Irradiation at Several Temperatures (ksi)

Material	Ambient, 200, & 400°F	550°F	Ambient	300°F
Plate A9811	>5			
Plate C1523	~20	30		
Weld Metal			25	20

These increases in yield strength are essentially the same as those resulting from irradiation at 7.05×10^{18} n/cm^2, thus also indicating that a limiting or steady state condition has been reached.

Photographs of the fractured tensile specimens are shown in figures 5-15 through 5-17. A typical stress strain curve for the tensile tests is shown in figure 5-18.

5-3. WEDGE OPENING LOADING TESTS

Wedge Opening Loading (WOL) fracture mechanics specimens which were contained in the surveillance capsule have been stored, on the recommendation of the U.S. Nuclear Regulatory Commission and at the request of the Wisconsin Electric Power Company, at the Westinghouse R&D Center. They will be tested and the results reported at a later date.

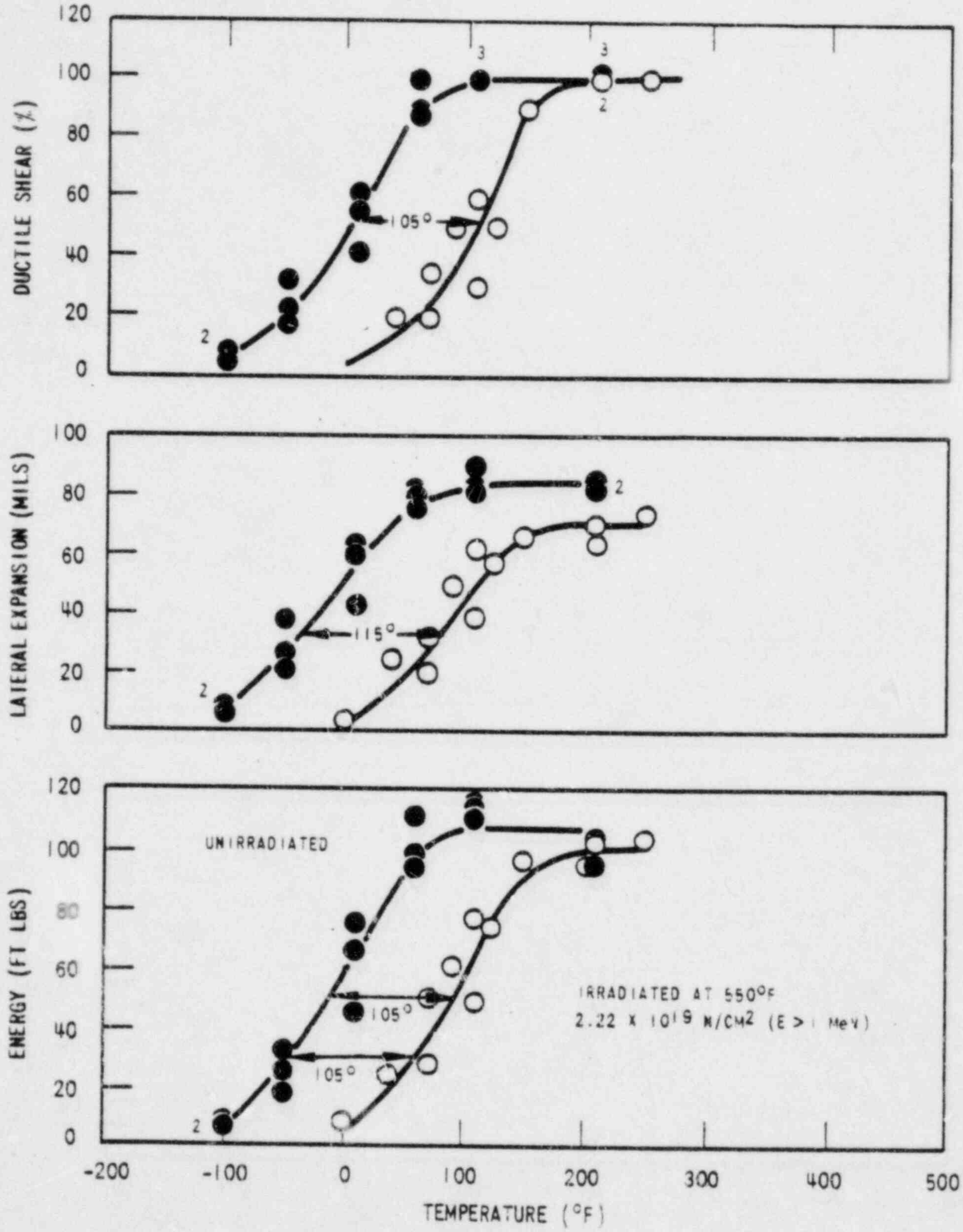


Figure 5-1. Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811

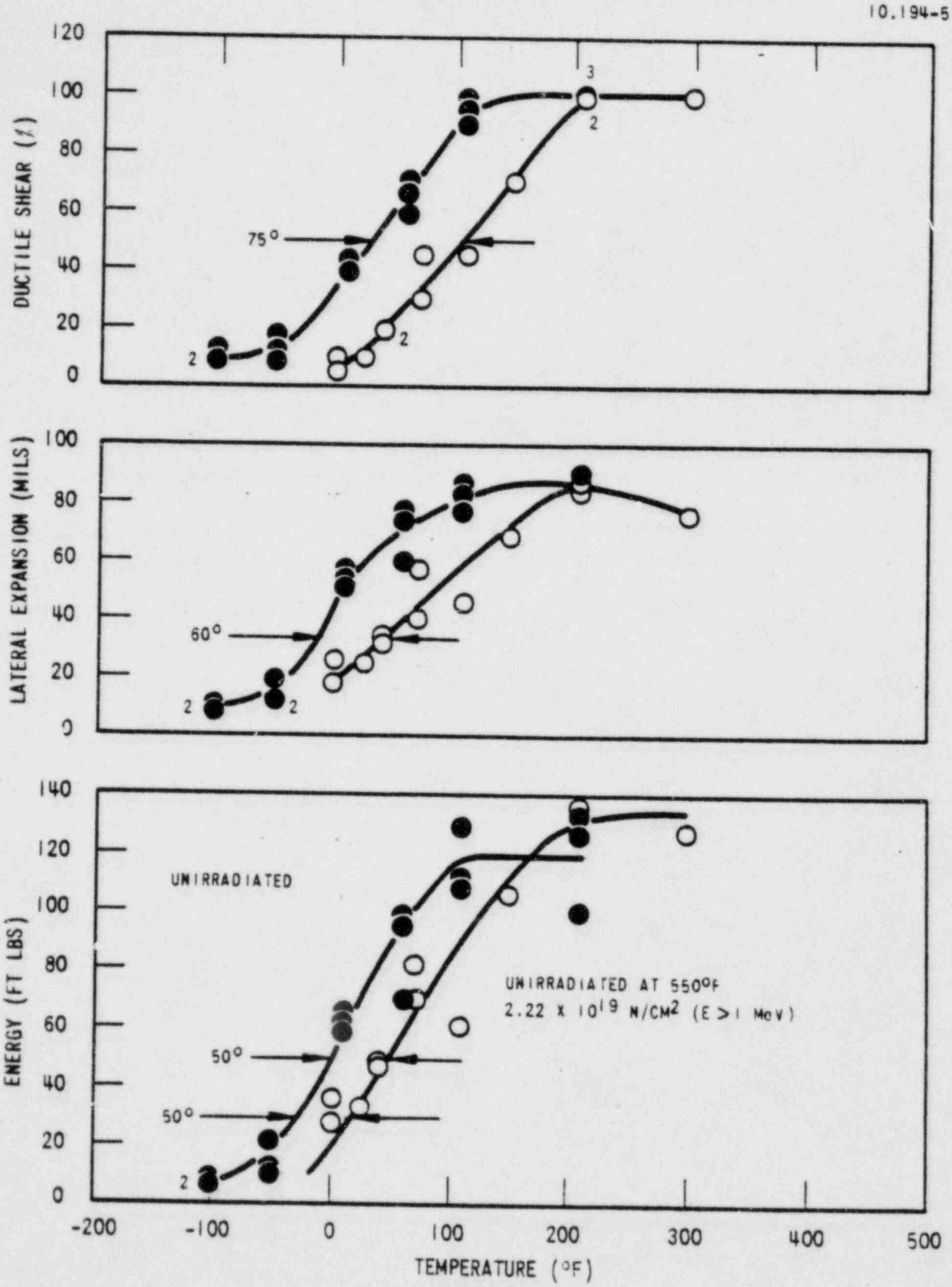


Figure 5-2. Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423

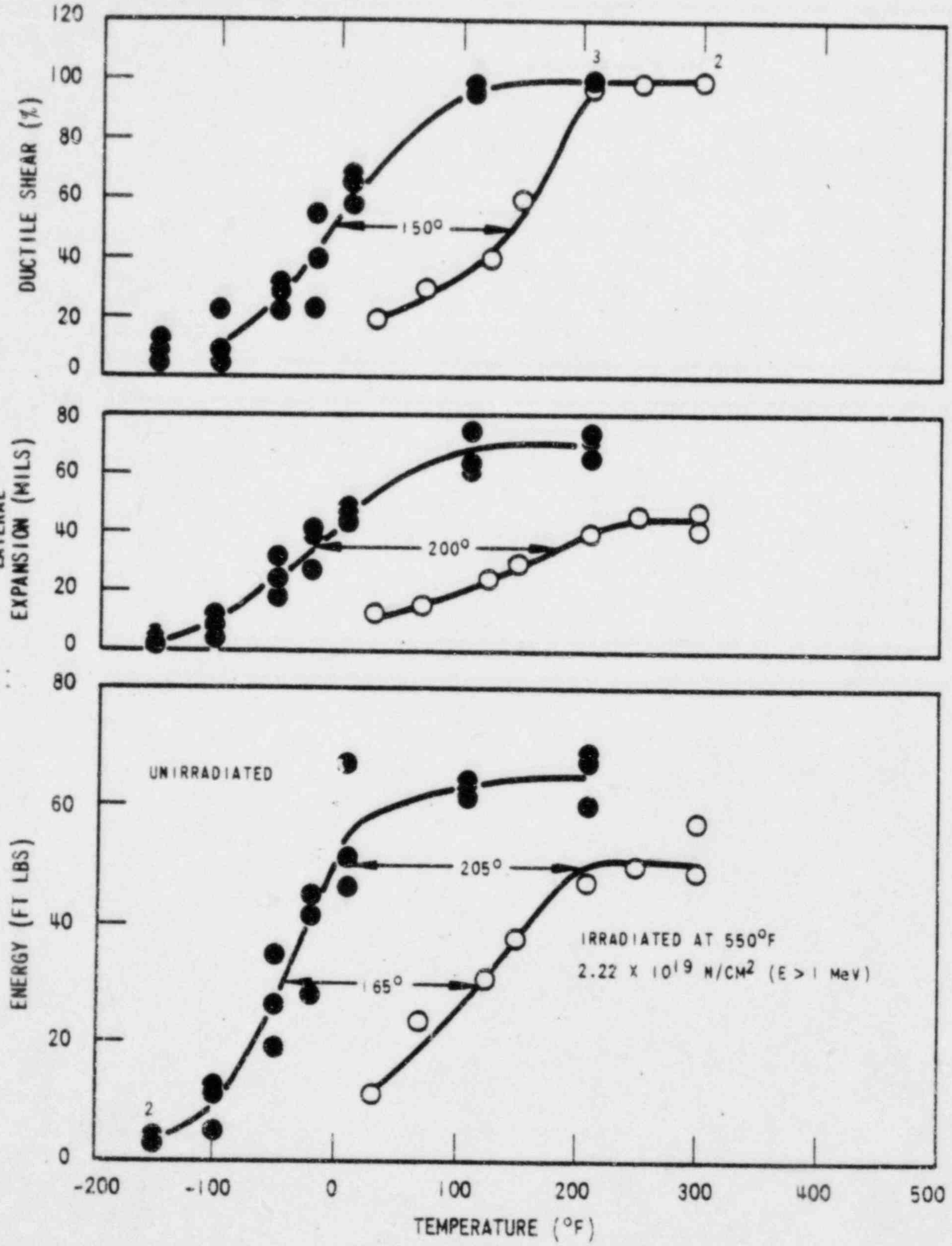


Figure 5-3. Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Pressure Vessel Weld Metal

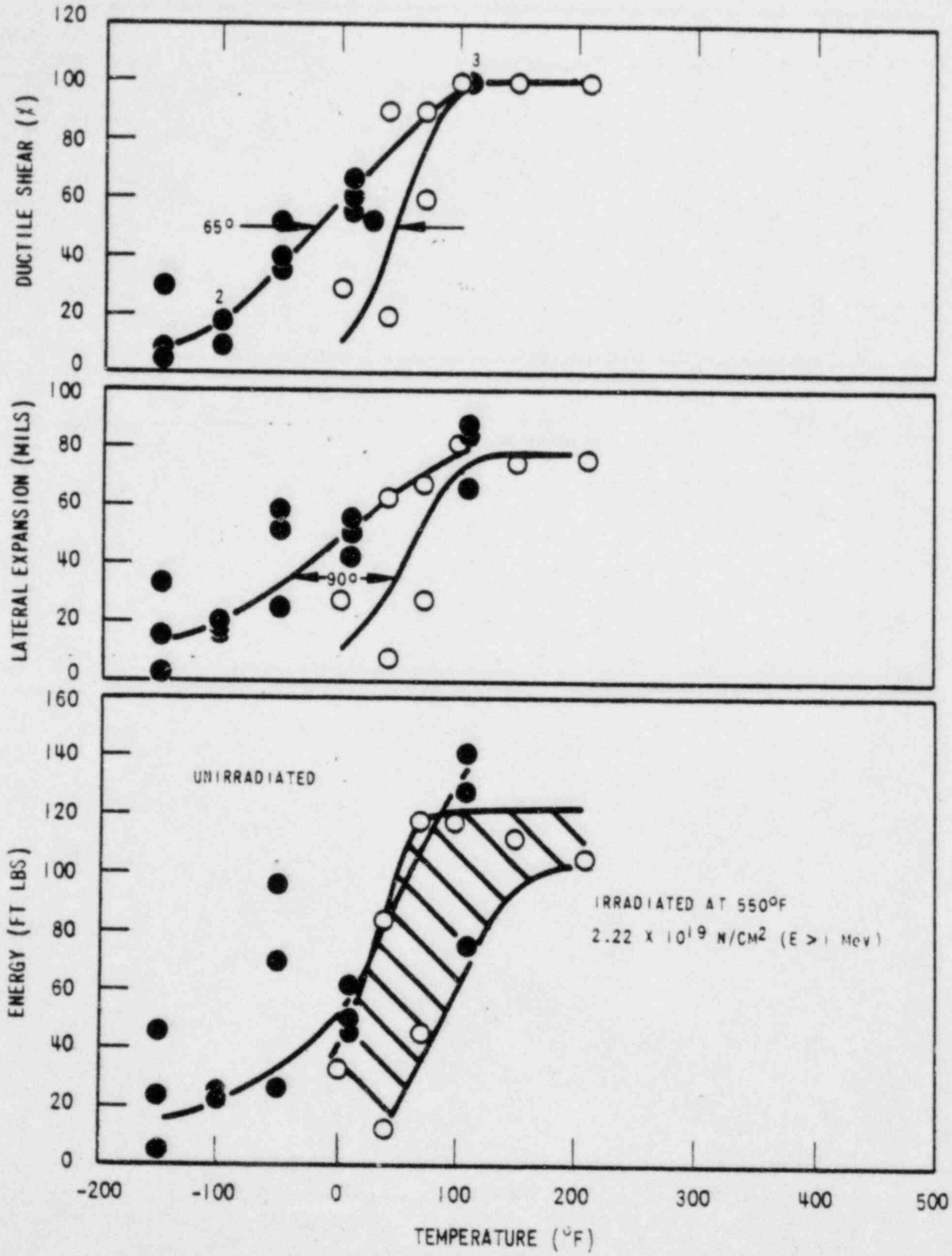


Figure 5-4. Charpy V-Notch Impact Data for the Point Beach Unit No. 1 Weld-Heat-Affected-Zone Metal

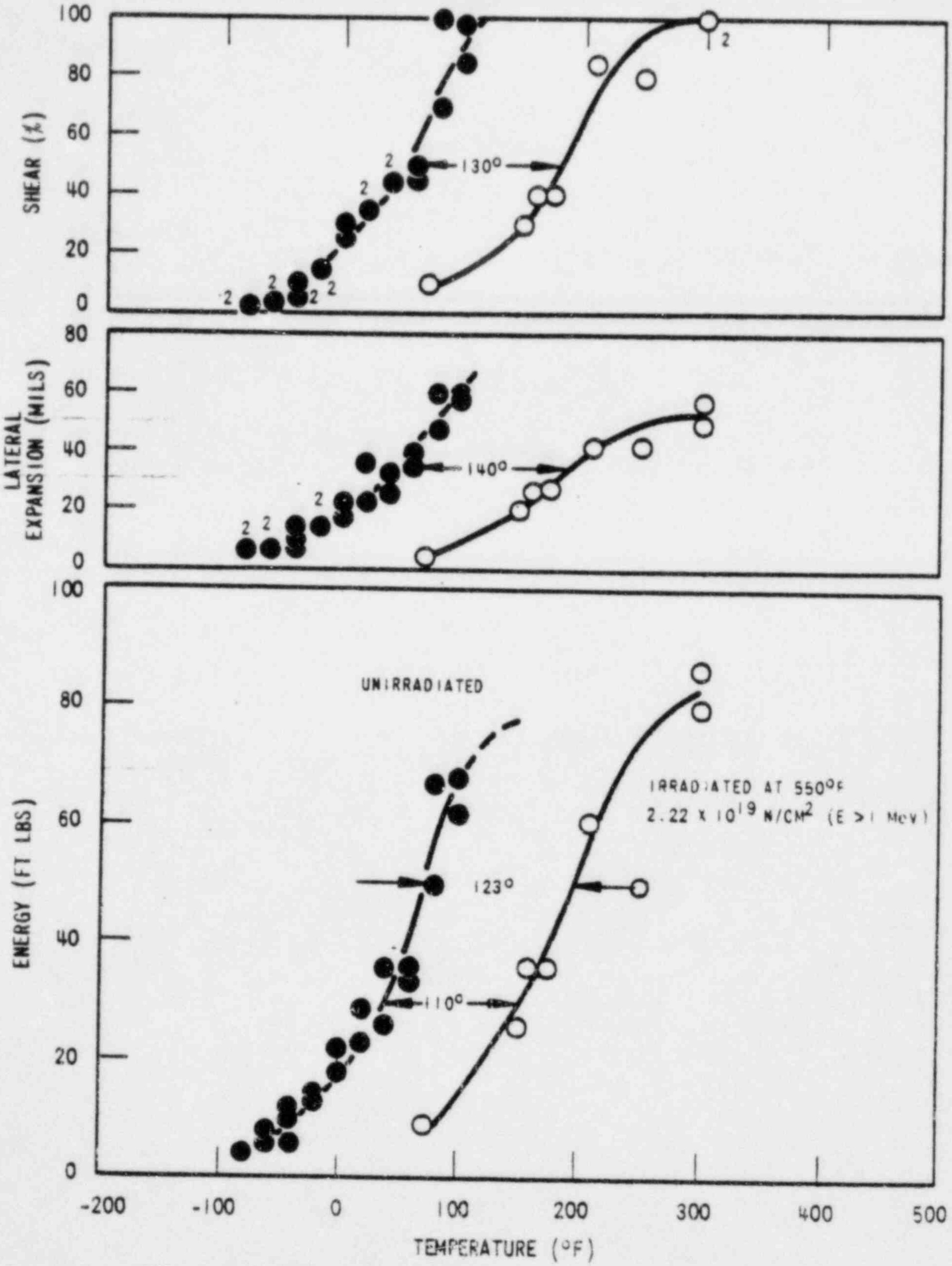
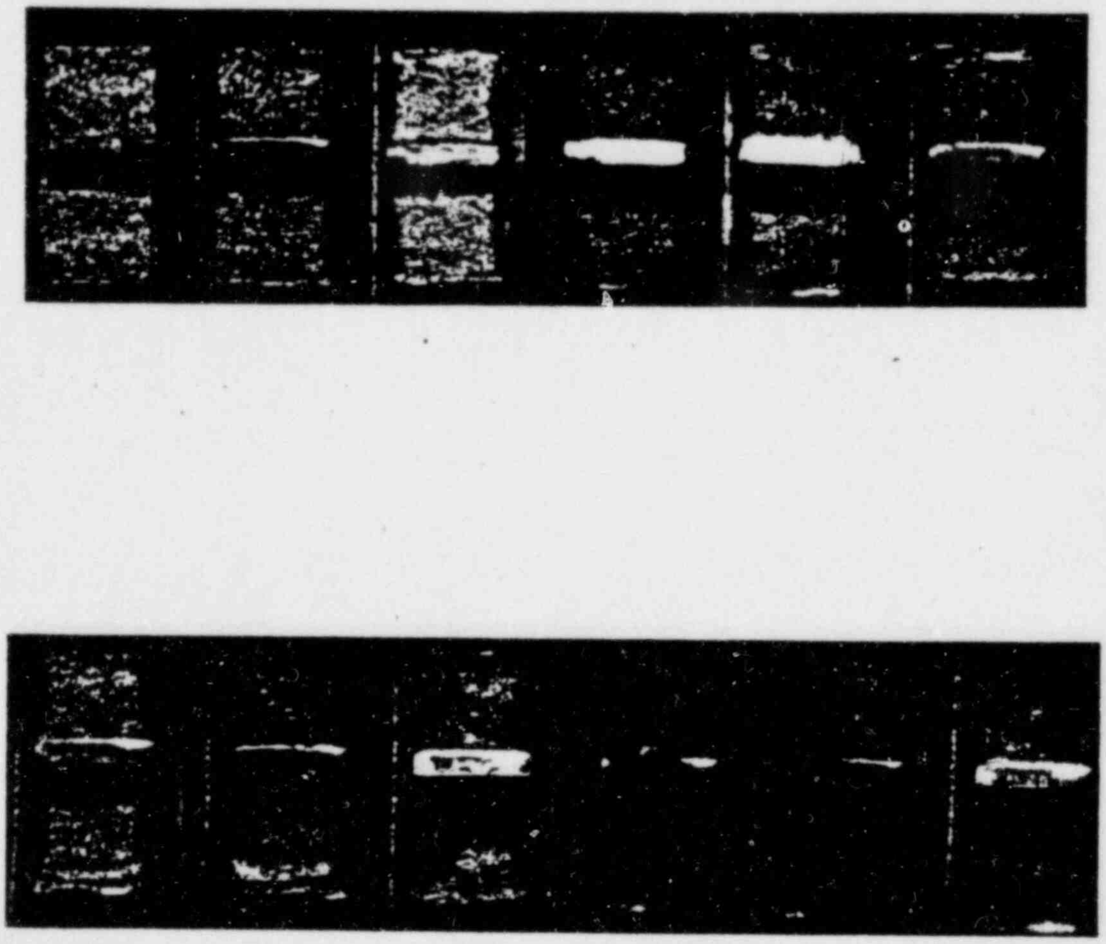


Figure 5-5. Charpy V-Notch Impact Data for SA302 Grade B ASTM Correlation Monitor Material

A22 A18 A13 A23 A16 A14



A21 A19 A17 A15 A24 A20

Figure 5-6. Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811

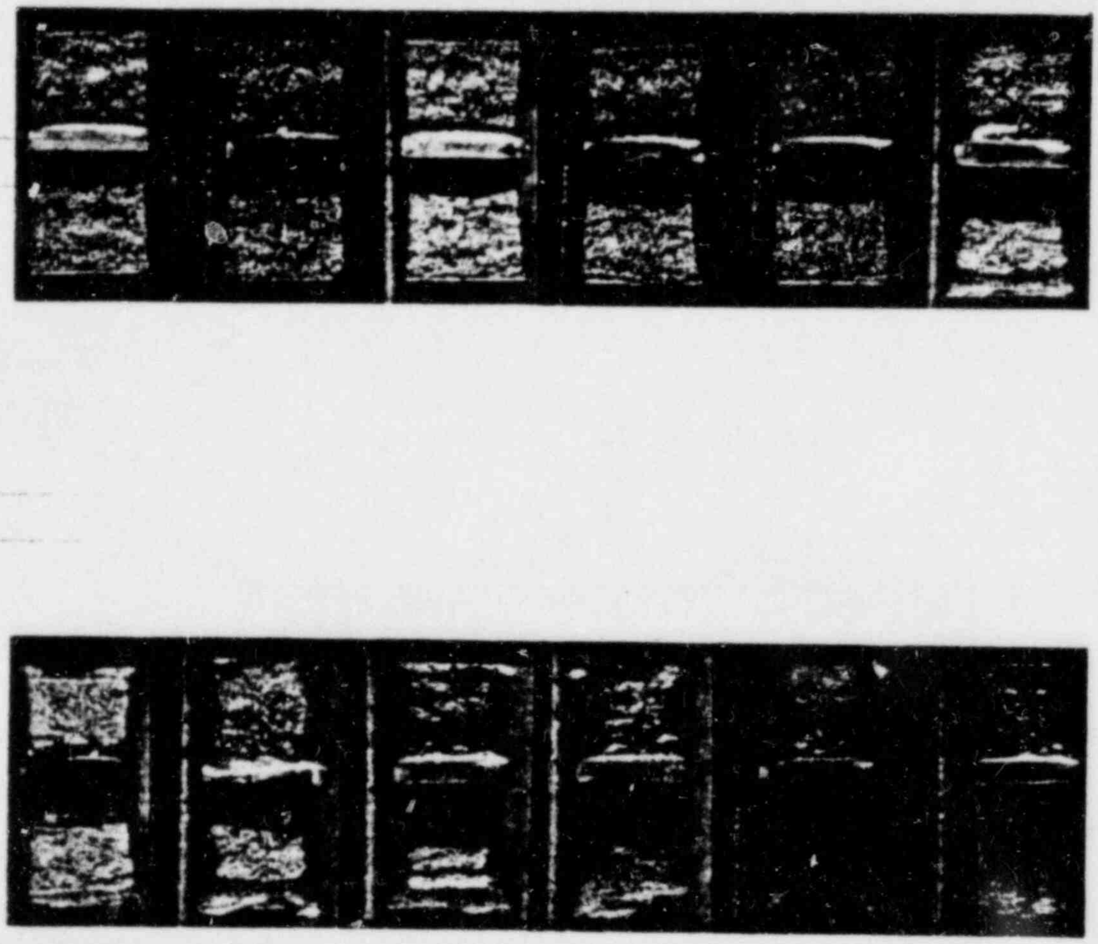


Figure 5-7. Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423

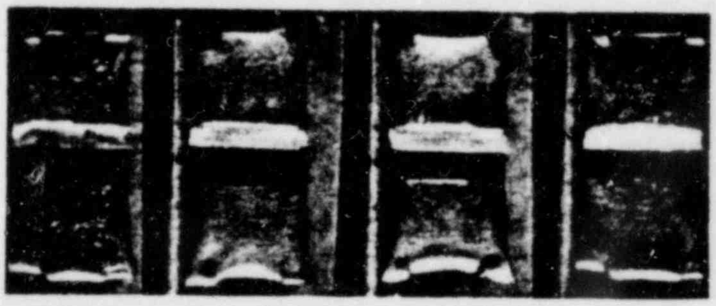


Figure 5-8. Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 Weld Metal

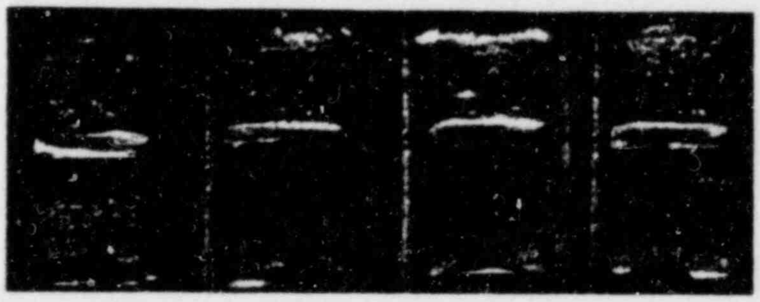


Figure 5-9. Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 Weld-Heat-Affected-Zone Metal

R10 R13 R15 R9

R11 R12 R14 R16

Figure 5-10. Charpy Impact Specimen Fracture Surfaces for Point Beach Unit No. 1 ASTM Correlation Monitor Material

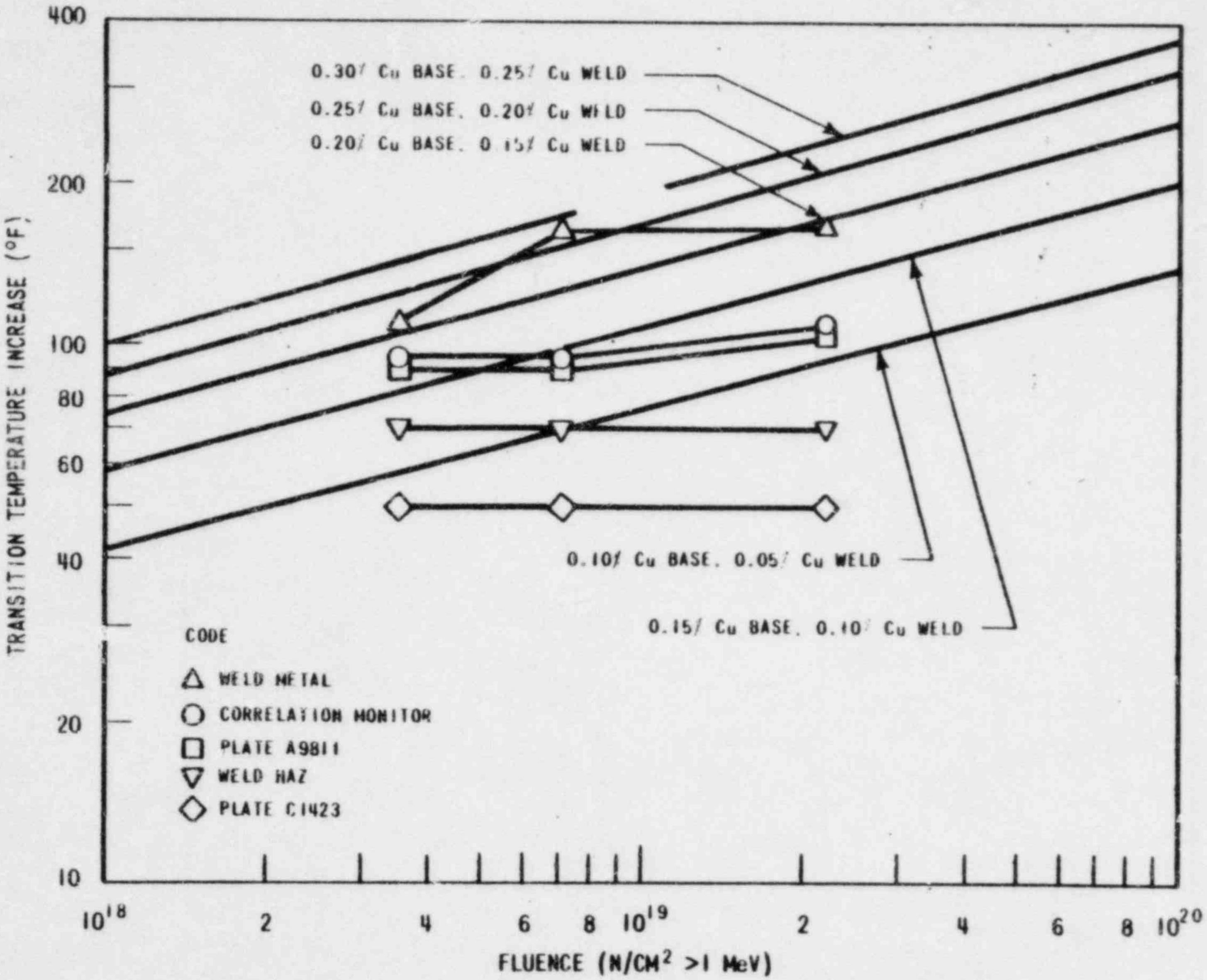


Figure 5-11. Point Beach Unit No. 1 Material 30 FT-LB Transition Temperature Increases as Compared to Westinghouse Predictions

10,194-14

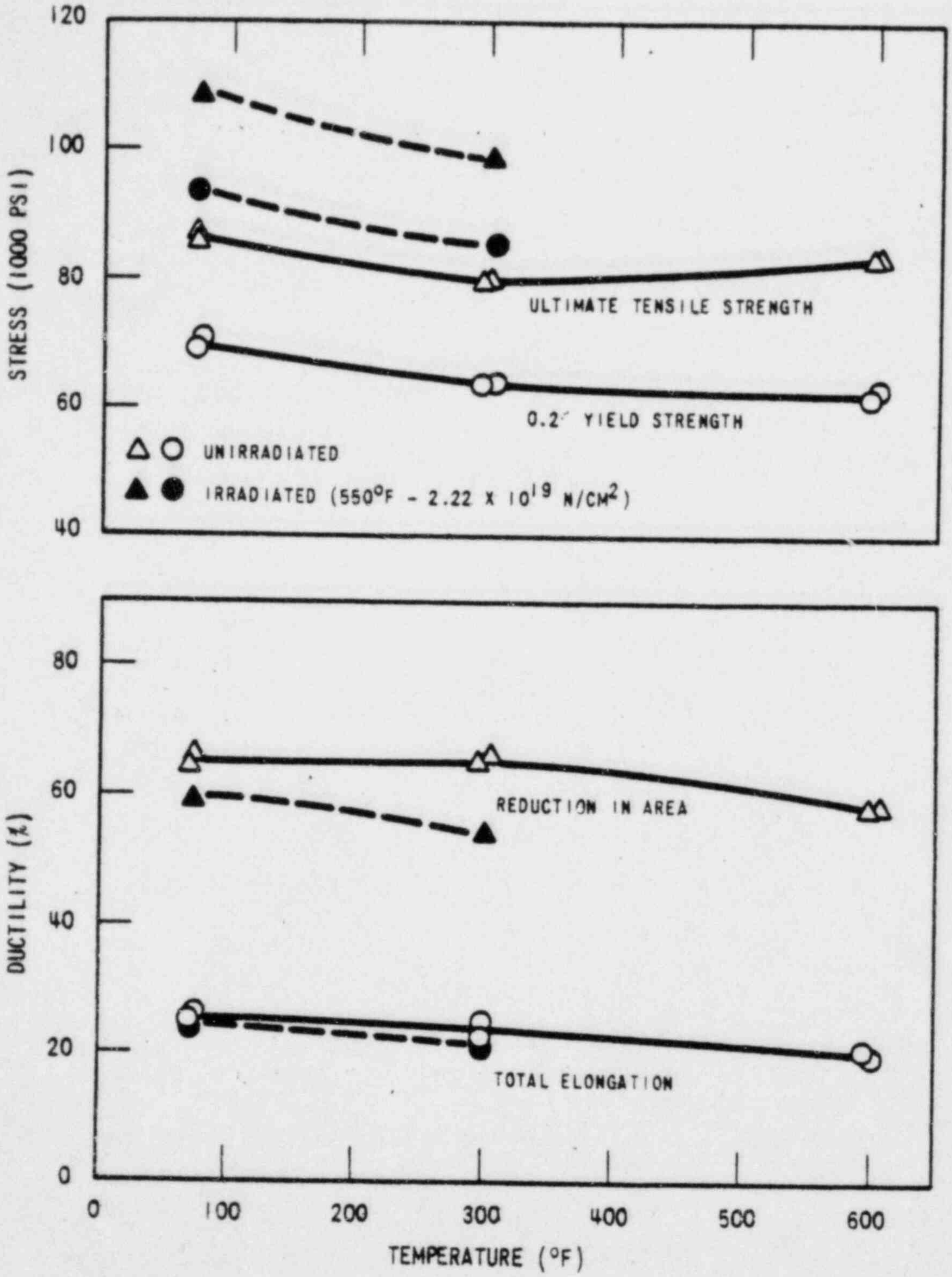


Figure 5-12. Tensile Properties for the Point Beach Unit No. 1 Pressure Vessel Weld Metal

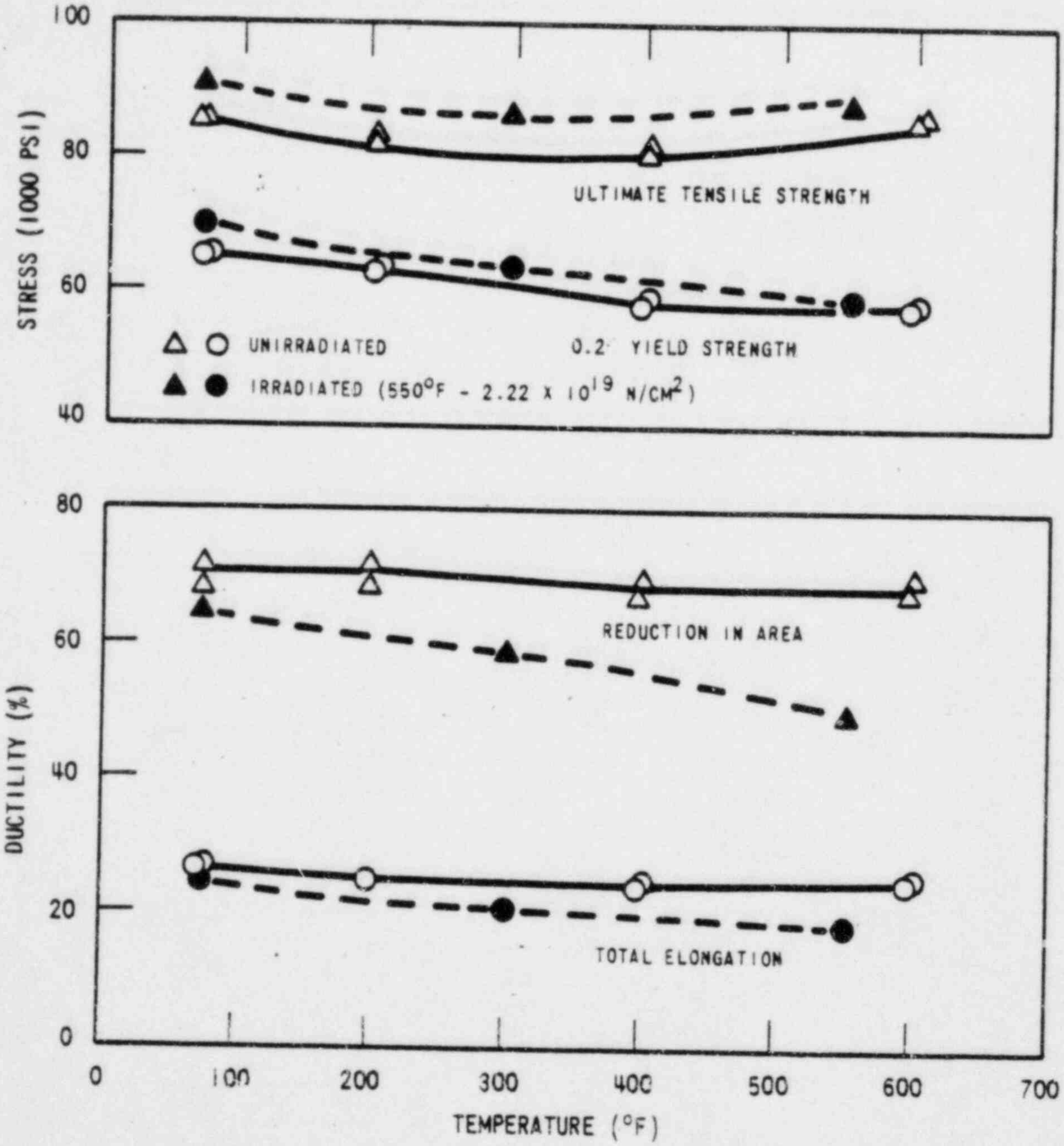


Figure 5-13. Tensile Properties for the Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811

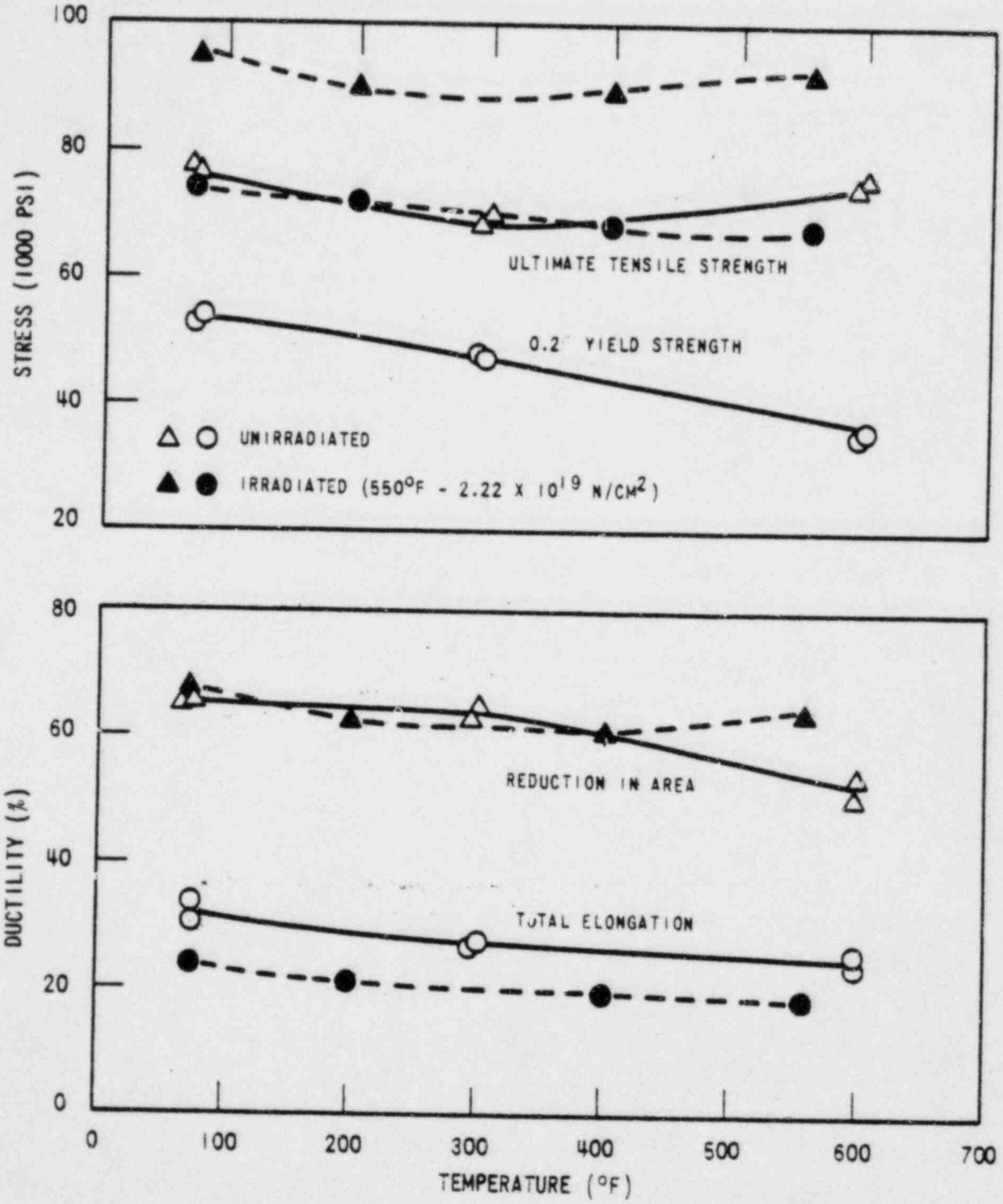


Figure 5-14. Tensile Properties for the Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423

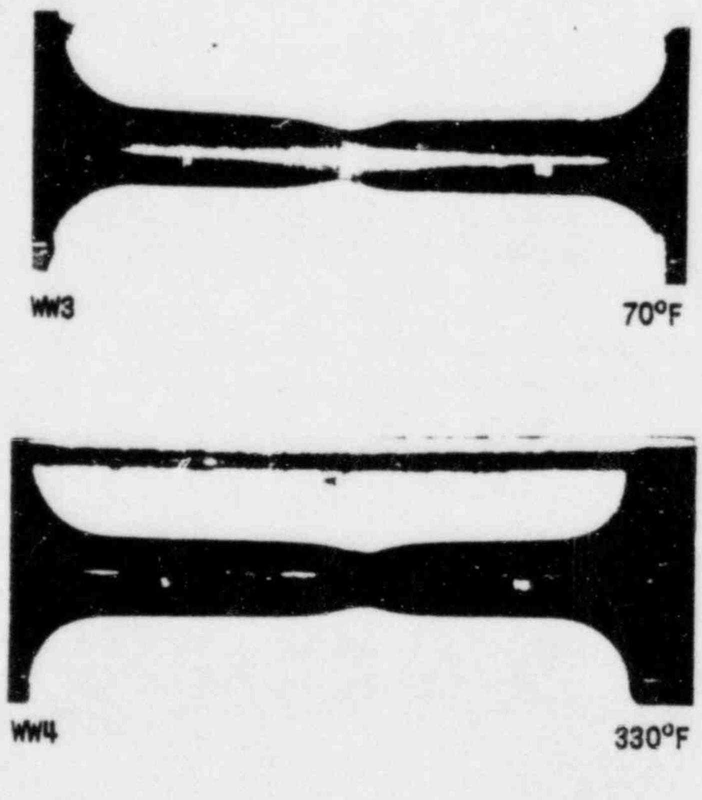


Figure 5-15. Fractured Tensile Specimens From Point Beach Unit No. 1 Weld Metal

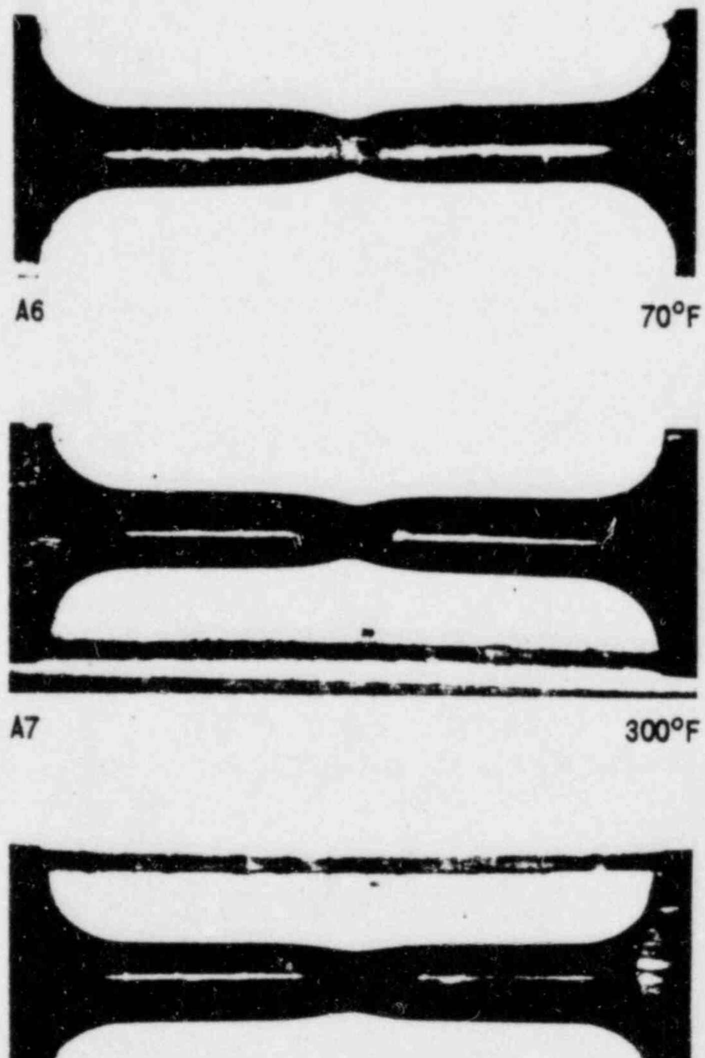


Figure 5-16. Fractured Tensile Specimens From Point Beach Unit No. 1 Pressure Vessel Shell Plate A9811

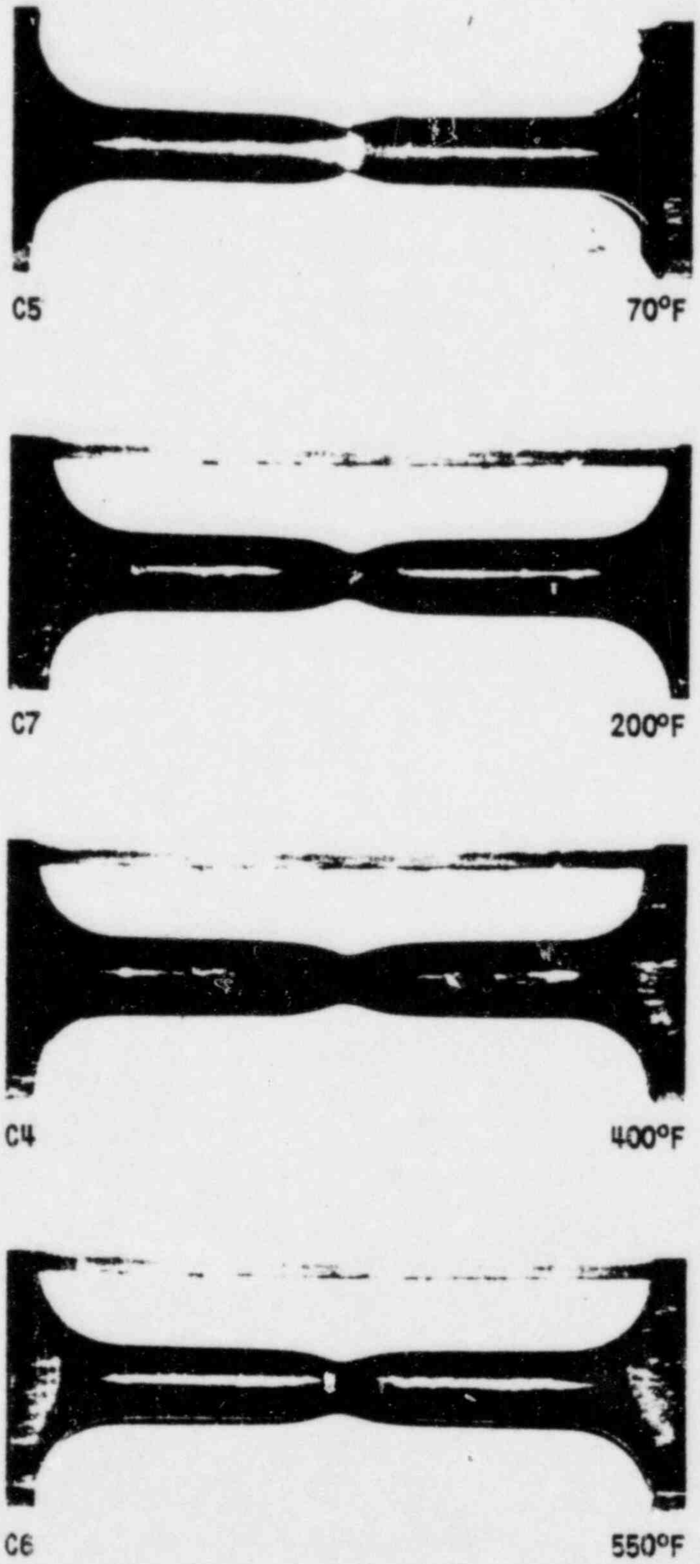


Figure 5-17. Fractured Tensile Specimens From Point Beach Unit No. 1 Pressure Vessel Shell Plate C1423

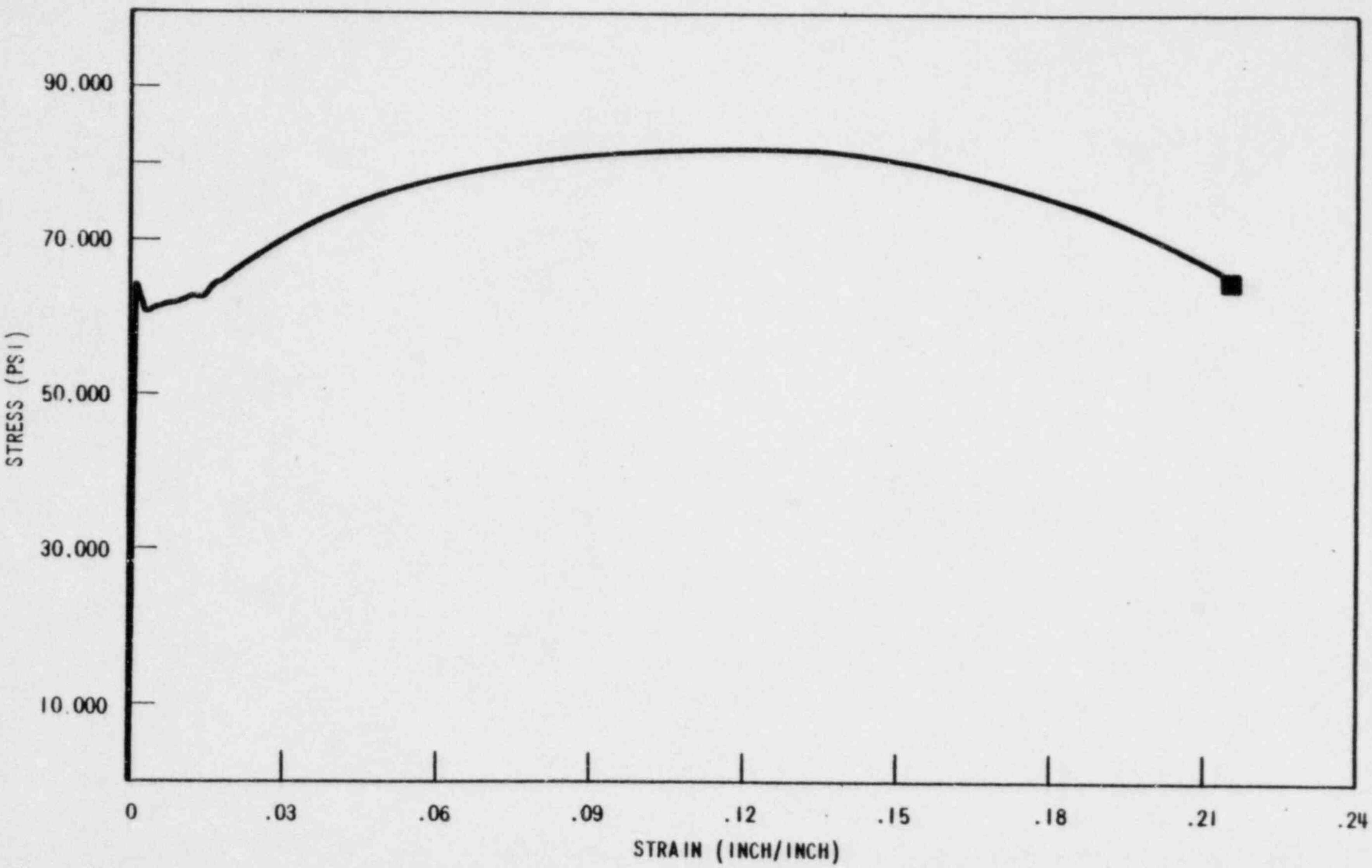


Figure 5-18. Typical Stress-Strain Curve for Tension Specimens (Tension Specimen No. A7)

10,194-19

TABLE 5-1

CHARPY V-NOTCH IMPACT DATA FOR THE POINT BEACH UNIT NO. 1
PRESSURE VESSEL SHELL PLATE A9811 IRRADIATED AT
550°F, FLUENCE 2.22 x 10^{19} n/cm^2 (E > 1 Mev)

Specimen Number	Test Temp (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
A22	0	7.5	4	5
A18	40	24.0	25	20
A13	70	28.0	20	20
A23	70	49.0	32	35
A16	90	61.0	50	50
A14	110	77.0	62	60
A21	110	49.0	39	30
A19	125	74.0	57	50
A17	150	97.0	66	90
A15	210	102.0	70	100
A24	210	95.0	64	100
A20	250	104.0	74	100

TABLE 5-2

CHARPY V-NOTCH IMPACT DATA FOR THE POINT BEACH UNIT NO. 1
PRESSURE VESSEL SHELL PLATE C1423 IRRADIATED AT
550°F, FLUENCE 2.22 x 10^{19} n/cm^2 (E > 1 Mev)

Specimen Number	Test Temp (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
C23	0	28.0	18	5
C23	0	36.0	26	10
C19	25	34.0	25	10
C15	40	48.0	34	20
C21	40	47.0	32	20
C18	70	82.0	57	50
C13	70	70.0	40	30
C20	110	61.0	46	45
C14	150	106.0	68	70
C16	210	135.0	84	100
C17	210	137.0	88	100
C24	300	128.0	76	100

TABLE 5-3

CHARPY V-NOTCH IMPACT DATA FOR THE POINT BEACH UNIT NO. 1 PRESSURE VESSEL WELD METAL IRRADIATED AT 550°F, FLUENCE 2.22 x 10^{19} n/cm^2 (E > 1 Mev)

Specimen Number	Test Temp (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
WW11	30	11.5	13	20
WW15	70	24.0	16	30
WW10	125	31.0	24	40
WW16	150	38.0	30	60
WW14	210	47.0	40	98
WW9	250	50.0	46	100
WW13	300	49.0	42	100
WW12	300	58.0	48	100

TABLE 5-4

CHARPY V-NOTCH IMPACT DATA FOR THE POINT BEACH UNIT NO. 1 PRESSURE VESSEL WELD-HEAT-AFFECTED-ZONE METAL IRRADIATED AT 550°F, FLUENCE 2.22 x 10^{19} n/cm^2 (E > 1 Mev)

Specimen Number	Test Temp (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
WH11	0	34.0	28	30
WH10	40	84.0	63	90
WH15	40	13.0	8	20
WH16	70	118.0	68	90
WH14	70	45.0	28	60
WH9	100	117.0	82	100
WH13	150	112.0	75	100
WH12	210	105.0	76	100

TABLE 5-5

CHARPY V-NOTCH IMPACT DATA FOR THE POINT BEACH UNIT NO. 1 ASTM SA302 GRADE B CORRELATION MONITOR MATERIAL IRRADIATED AT 550°F, FLUENCE 2.22×10^{19} n/cm^2 (E > 1 Mev)

Specimen Number	Test Temp (°F)	Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
R10	70	9.0	4	10
R13	150	26.0	21	30
R15	160	36.0	27	40
R9	175	36.0	28	40
R11	210	60.5	42	85
R12	250	50.0	42	80
R14	300	87.0	58	100
R16	300	80.0	50	100

TABLE 5-6

THE EFFECT OF 550°F IRRADIATION AT 2.22 x 10^{19} n/cm^2 (E > 1 Mev) ON THE NOTCH TOUGHNESS PROPERTIES OF THE POINT BEACH UNIT NO. 1 REACTOR VESSEL IMPACT TEST SPECIMENS

Material	Transition Temp (°F)						Δ Temp (°F)			Average Energy Absorption at Full Shear (ft-lb)		
	Unirradiated			Irradiated						Unirradiated	Irradiated	ΔEnergy
	50 ft-lb	30 ft-lb	35 mils	50 ft-lb	30 ft-lb	35 mils	50 ft-lb	30 ft-lb	35 mils			
A9811	-10	-45	-30	95	60	85	105	105	115	107	100	7
C1423	0	-30	-12	50	20	48	50	50	60	119	134	+15
Weld Metal	0	-45	-18	205	120	182	205	165	200	65	51	14
HAZ Metal (A9811)	0	-60	-40	60	10	50	60	70	90	135	112	23
Correlation Material	72	40	50	195	150	190	123	110	140	78	83	+5

TABLE 5-7

SUMMARY OF POINT BEACH UNIT NO. 1 REACTOR VESSEL SURVEILLANCE CAPSULE CHARPY IMPACT TEST RESULTS

Material	Fluence 10^{18} n/cm^2	Trans. Temp Increase (°F) 30 ft lb	Trans. Temp Increase (°F) 50 ft lb	Upper Shelf Decrease (ft lb)
Plate A9811	3.50	90	90	18
Plate A9811	7.05	90	90	15
Plate A9811	22.20	105	105	7
Plate C1423	3.50	50	50	None
Plate C1423	7.05	50	50	None
Plate C1423	22.20	50	50	None
Weld Metal	3.50	110	140	12
Weld Metal	7.05	165	200	13
Weld Metal	22.20	165	205	14
HAZ Metal (A9811)	3.50	70	25	28
HAZ Metal (A9811)	7.05	70	60	25
HAZ Metal (A9811)	22.20	70	60	23
Correlation Monitor	3.50	95	100	14
Correlation Monitor	7.05	95	105	10
Correlation Monitor	22.20	110	123	None

TABLE 5-6

IRRADIATED TENSILE PROPERTIES FOR THE POINT BEACH UNIT NO. 1 PRESSURE VESSEL MATERIALS

Material Iden.	Specimen Number	Test Temp (°F)	0.2% Yield Strength (ksi)	Ultimate Tensile Strength (ksi)	Fracture Load (lb)	Fracture Stress (ksi)	Fracture Strength (ksi)	Uniform Elong. (%)	Total Elong. (%)	Reduction in Area (%)
A9811	A6	70	69.5	90.0	3250	181.6	69.5	13.4	24.9	60
	A7	300	62.2	86.1	3250	166.3	66.5	11.8	21.4	59
	A5	550	58.3	86.7	3500	137.8	71.6	11.0	18.9	48
C1423	C5	70	75.0	95.1	3200	195.1	65.4	11.6	24.0	66
	C7	200	71.6	89.8	2860	159.8	58.5	9.5	22.1	63
	C4	400	67.5	88.8	3110	161.1	63.6	9.6	20.7	61
	C6	550	66.1	91.0	3200	188.2	65.4	9.4	20.5	65
Weld Metal	WW3	70	94.9	108.6	3980	203.1	81.4	10.2	21.8	60
	WW4	300	83.8	99.0	3700	165.2	75.7	10.5	20.9	54

SECTION 6
NEUTRON DOSIMETRY ANALYSIS

6-1. DESCRIPTION OF NEUTRON FLUX MONITORS

To effect a correlation between neutron exposure and the radiation-induced property changes observed in the test specimens, a number of neutron flux monitors were included as an integral part of the Reactor Vessel Surveillance Program. Table 6-1 lists the particular monitors contained within Capsule R, along with the nuclear reaction of interest and the energy range of each monitor.

The first five reactions listed in table 6-1 are used as fast neutron monitors to relate neutron fluence (E > 1.0 Mev) to the measured shift in RT_{NDT}. To properly account for burnout of the product isotope generated by the fast neutron reactions, it is necessary to also determine the magnitude of the thermal neutron flux at the monitor location. Therefore, bare and cadmium-covered cobalt-aluminum monitors were included within Capsule R.

Figure 4-2 shows the relative locations of the various monitors within Capsule R. Figure 6-1 shows the radial and azimuthal positions of the capsule with respect to the nuclear core, reactor internals, and pressure vessel. The nickel, copper, and cobalt-aluminum monitors (in wire form) were placed in holes drilled in spacers at several axial levels within the capsule. The iron monitors were obtained by drilling samples from selected Charpy test specimens. The cadmium-shielded neptunium and uranium fission monitors were accommodated within the dosimeter block located near the center of the capsule.

The use of activation monitors, such as those listed in table 6-1, does not yield a direct measure of the energy-dependent neutron flux level at the point of interest. Rather, the activation process is a measure of the integrated effect that the time and energy-dependent neutron flux has on the target material. An accurate estimate of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The operating history of the reactor
- The energy response of the monitor

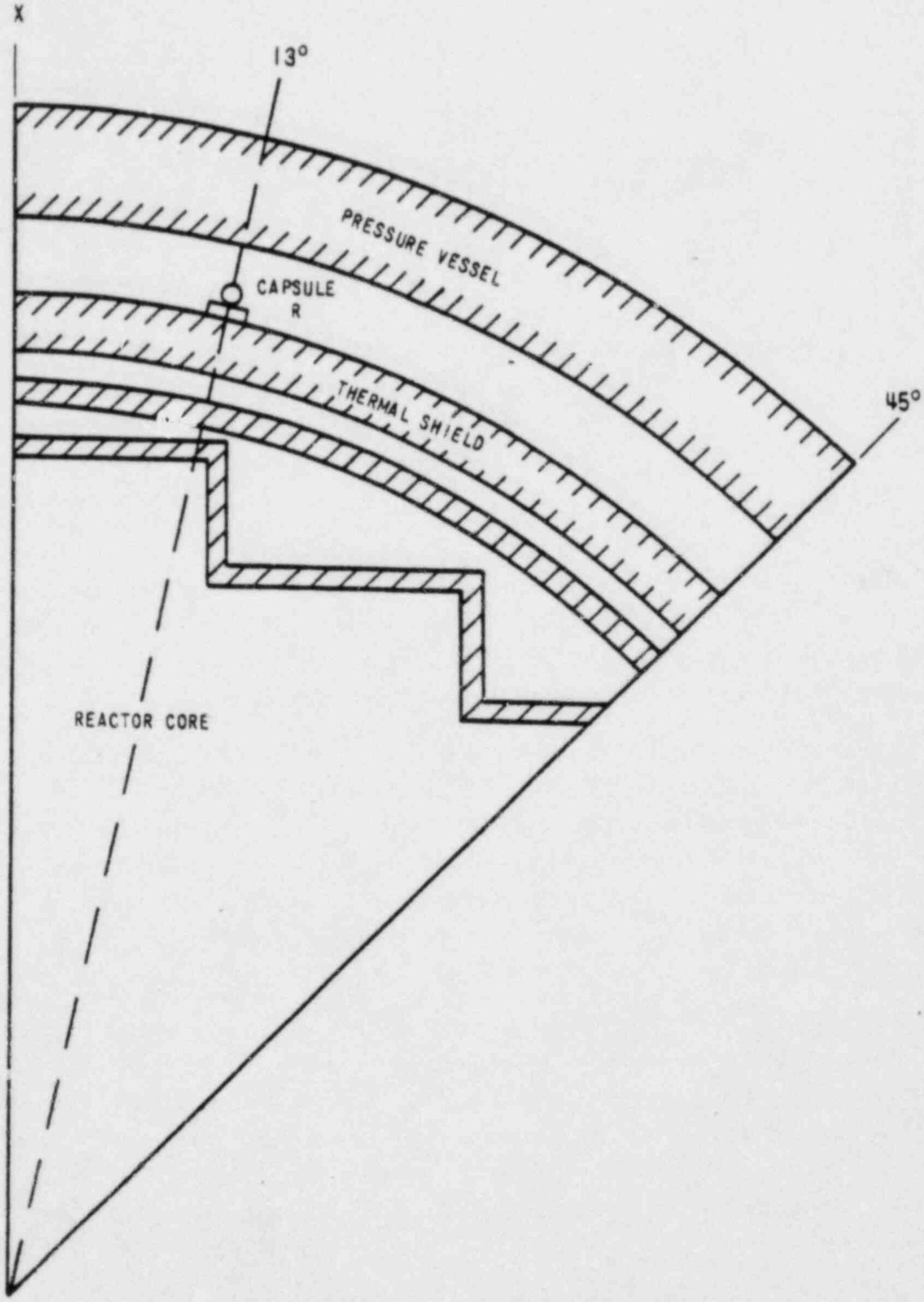


Figure 6-1. Point Beach Unit No. 1 Reactor Geometry

TABLE 6-1

NEUTRON FLUX MONITORS CONTAINED WITHIN CAPSULE R

Monitor Material	Reaction of Interest	Wt% of Target $\frac{gm_{target}}{gm_{monitor}}$	Response Range	Product Half-Life
Copper	Cu^{63} (n,α) Co^{60}	0.6917	E > 4.7 Mev	5.27 years
Iron	Fe^{54} (n,p) Mn^{54}	0.0585	E > 1.0 Mev	314 days
Nickel	Ni^{58} (n,p) Co^{58}	0.6777	E > 1.0 Mev	71.4 days
Uranium^{238} [a]	U^{238} (n,f) Cs^{137}	1.0	E > 0.4 Mev	30.2 years
Neptunium^{237} [a]	$N_p{}^{237}$ (n,f) Cs^{137}	1.0	E > 0.36 Mev	30.2 years
Cobalt-aluminum[a]	Co^{59} (n,λ) Co^{60}	0.0015	0.4ev < E < 0.015 Mev	5.27 years
Cobalt-aluminum	Co^{59} (n,λ) Co^{60}	0.0015	E < 0.015 Mev	5.27 years

a. Cadmium shielded monitors.

- The neutron energy spectrum at the monitor location
- The physical characteristics of the monitor

6-2. ANALYTICAL PROCEDURES

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

The energy and spatial distribution of neutron flux within the Point Beach Nuclear Plant Unit No. 1 reactor geometry was obtained with the DOT[1] two-dimensional S_n transport code. The radial and azimuthal distributions were obtained from an R,θ computation wherein the reactor core, reactor internals, surveillance capsule, water annuli, pressure vessel, and primary shield concrete were described on the analytical model. These analyses employed 21 neutron energy groups, an S_8 angular quadrature, and a P_1 cross-section expansion. The reactor core power distributions used in the calculations, which were representative of time-averaged conditions over an equilibrium fuel cycle, accounted for rod-by-rod spatial variations in the peripheral fuel assemblies. The analytical geometries described a 45° sector of the reactor, assuming one-eighth symmetry. Relative axial variations of neutron flux incident on the reactor vessel were obtained from R,Z DOT calculations based on the equivalent cylindrical core concept.

The specific activity of each of the activation monitors was determined in accordance with established ASTM procedures.[2,3,4,5,6]. Following sample preparation, the activity of each monitor was determined by means of a lithium drifted germanium Ge (Li) gamma spectrometer. The overall standard deviation of the measured data is a function of the precision of

---

1. Soltesz, R. G., Disney, R. K., Jedruch, J. and Ziegler, S. L., "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation. Vol. 5 – Two-Dimensional, Discrete Ordinates Transport Technique," WANL-PR(LL)034, Vol. 5, August 1970.
2. ASTM Designation E261-70, "Standard Method for Measuring Neutron Flux by Radioactivation Techniques," in ASTM Standards (1975), Part 45, Nuclear Standards, pp. 745-755, Am. Society for Testing and Materials, Philadelphia, Pa. 1975.
3. ASTM Designation E262-70, "Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques," in ASTM Standards (1975), Part 45, Nuclear Standards, pp. 756-763, Am. Society for Testing and Materials, Philadelphia, Pa. 1975.
4. ASTM Designation E263-70, "Standard Method for Measuring Fast-Neutron Flux by Radioactivation of Iron," in ASTM Standards (1975), Part 45, Nuclear Standards, pp. 764-769, Am. Society for Testing and Materials, Philadelphia, Pa., 1975.
5. ASTM Designation E481-73T, "Tentative Method of Measuring Neutron – Flux Density by Radioactivation of Cobalt and Silver," in ASTM Standards (1975), Part 45, Nuclear Standards, pp. 887-894, Am. Society for Testing and Materials, Philadelphia, Pa. 1975.
6. ASTM Designation E264-70, "Standard Method for Measuring Fast-Neutron Flux by Radioactivation of Nickel," in ASTM Standards (1975), Part 45, Nuclear Standards, pp. 770-774, Am. Society for Testing and Materials, Philadelphia, Pa. 1975.

sample weighing, the uncertainty in counting, and the acceptable error in detector calibration. For the samples removed from Capsule R, the overall 2σ deviation in all of the measured data was determined to be ±10 percent.

Having the measured activity of the monitors and the neutron energy spectrum at the location of interest, the calculation of the fast neutron flux proceeded as follows. The reaction product activity in the monitor was expressed as

$$D = \frac{No}{A} f_i \, y \int_E \sigma(E) \, \theta(E) \sum_{j=1}^{n} \frac{P_j}{P_{max}} (1 - e^{-\lambda\tau_j}) \, e^{-\lambda\tau_d} \qquad (6\text{-}1)$$

where

D = induced product activity

No = Avogadro's number

A = atomic weight of the target isotope

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

y = number of product atoms produced per reaction

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

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

P_j = average core power level during irradiation period j

P_{max} = maximum or reference core power level

λ = decay constant of the product isotope

τ_j = length of irradiation period j

τ_d = decay time following irradiation period j

Since neutron flux distributions were calculated by means of multigroup transport methods, and further, since the prime interest was in the fast neutron flux above 1 Mev, spectrum-averaged reaction cross sections were defined such that the integral term in equation (6-1) could be replaced by the following relation

$$\int_E \sigma(E) \, \phi(E) = \bar{\sigma}\phi \; (E > 1 \text{ Mev})$$

where

$$\sigma = \frac{\int_{o}^{\infty} \sigma(E)\, \phi(E)}{\int_{1.0\ \mathrm{Mev}}^{\infty} \phi(E)} = \frac{\sum_{G=1}^{n} \sigma_g\, \phi_g}{\sum_{G=G_{1.0\ \mathrm{Mev}}}^{n} \phi_g}$$

Thus, equation (6-1) was rewritten

$$D = \frac{No}{A} \cdot f_i\ Y\ \bar{\sigma}\phi\ (E > 1.0\ \mathrm{Mev}) \sum_{j=1}^{n} \frac{P_j}{P_{max}}\ (\ -\ e^{-\lambda\tau_j})\ e^{-\lambda\tau_d}$$

or, solving for the neutron flux

$$\phi\ (E > 1.0\ \mathrm{Mev}) = \frac{D}{\frac{No}{A}\ f_i\ Y\ \bar{\sigma} \sum \frac{P_j}{P_{max}}\ (1 - e^{-\lambda\tau_j})\ e^{-\lambda\tau_d}} \qquad (6\text{-}2)$$

The total fluence above 1 Mev was then given by

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

where

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

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

$$\phi_{th} = \frac{D_{bare} \left\{\frac{R - 1}{R}\right\}}{\frac{No}{A}\ f_i\ Y\ \sigma \sum_{j=1}^{n} \frac{P_j}{P_{max}}\ (1 - e^{-\lambda\tau_j})\ e^{-\lambda\tau_d}} \qquad (6\text{-}4)$$

where R is defined as $D_{bare}/D_{Cd\text{-}covered}$.

The irradiation history of the flux monitors removed from Capsule R is listed in table 6-2. The data were obtained from the Point Beach semi-annual operating reports.[1] The spectrum-averaged reaction cross sections derived for each of the fast neutron flux monitors are listed in table 6-3.

6-3. RESULTS OF ANALYSIS

Table 6-4 lists the fast neutron ($E > 1$ Mev) flux and fluence levels derived from the monitors taken from Capsule R. Table 6-5 summarizes the thermal neutron flux obtained from the cobalt-aluminum monitors. Due to the relatively low thermal neutron flux at the capsule location, no burnup correction was made to any of the measured activities. The maximum error introduced by this assumption is estimated to be less than 1 percent for the Ni^{58} (n, p) Co^{58} reaction and even less significant for all of the other fast reactions.

Figures 6-2 through 6-4 and table 6-6 summarize results of the S_n transport calculations for the Point Beach Unit No. 1 Reactor. In figure 6-2, the calculated maximum fast neutron flux levels at the pressure vessel inner radius, 1/4 thickness location, and 3/4 thickness location are presented as a function of azimuthal angle. Figure 6-3 shows the relative axial variation of neutron flux. Absolute axial variations of fast neutron flux may be obtained by multiplying the levels given in figure 6-2 by the appropriate values from figure 6-3. In figure 6-4, the calculated maximum end-of-life fast neutron exposure of the Point Beach Reactor Vessel is given as a function of radial position within the vessel wall. Table 6-6 lists the calculated fast neutron flux levels interior to Capsule R along with the lead factors (LF) relating capsule exposure to vessel exposure. The lead factor is defined as the ratio of the calculated flux at the monitor location to the calculated peak neutron flux incident on the reactor vessel.

Based on the iron data in table 6-4, the average fast neutron fluence incident on the front row of Charpy specimens is determined to be 2.43×10^{19} n/cm^2, while that on the back row of the specimens is 2.02×10^{19} n/cm^2. These measured values correspond to analytical values of 1.99×10^{19} and 1.60×10^{19} n/cm^2, respectively. A comparison of these values shows the calculations to be 22 to 26 percent low.

1. Point Beach Nuclear Units 1 and 2 Semi-Annual Operating Reports, 1970 through 1977.

With the use of the lead factors listed in table 6-6, a comparison of the end-of-life peak fast neutron exposure of the Point Beach reactors as derived from both calculations and measured surveillance capsule results may be made as follows:

FAST NEUTRON FLUENCE (n/cm^2)

Vessel Location	Calculated	Based on Iron From Front Charpys	Based on Iron From Back Charpys
Inner Surface	3.9×10^{19}	4.8×10^{19}	5.0×10^{19}
1/4 Thickness	2.4×10^{19}	3.0×10^{19}	3.1×10^{19}
3/4 Thickness	7.5×10^{18}	9.3×10^{18}	9.7×10^{18}

These data are based on 32 full-power years of operation at 1518 Mw.

Based on the results of the tests performed to date, the following recommended removal schedule for the remaining surveillance capsules, developed by the Wisconsin Electric Power Company, has been approved by the NRC.

Capsule Identification	Lead Factor	Removal Date
T	1.6	Fall of 1985
P	1.6	Fall of 1989
N	1.4	Standby

The actual removal dates should correspond to normal plant refueling and/or major plant shutdowns for the year identified.

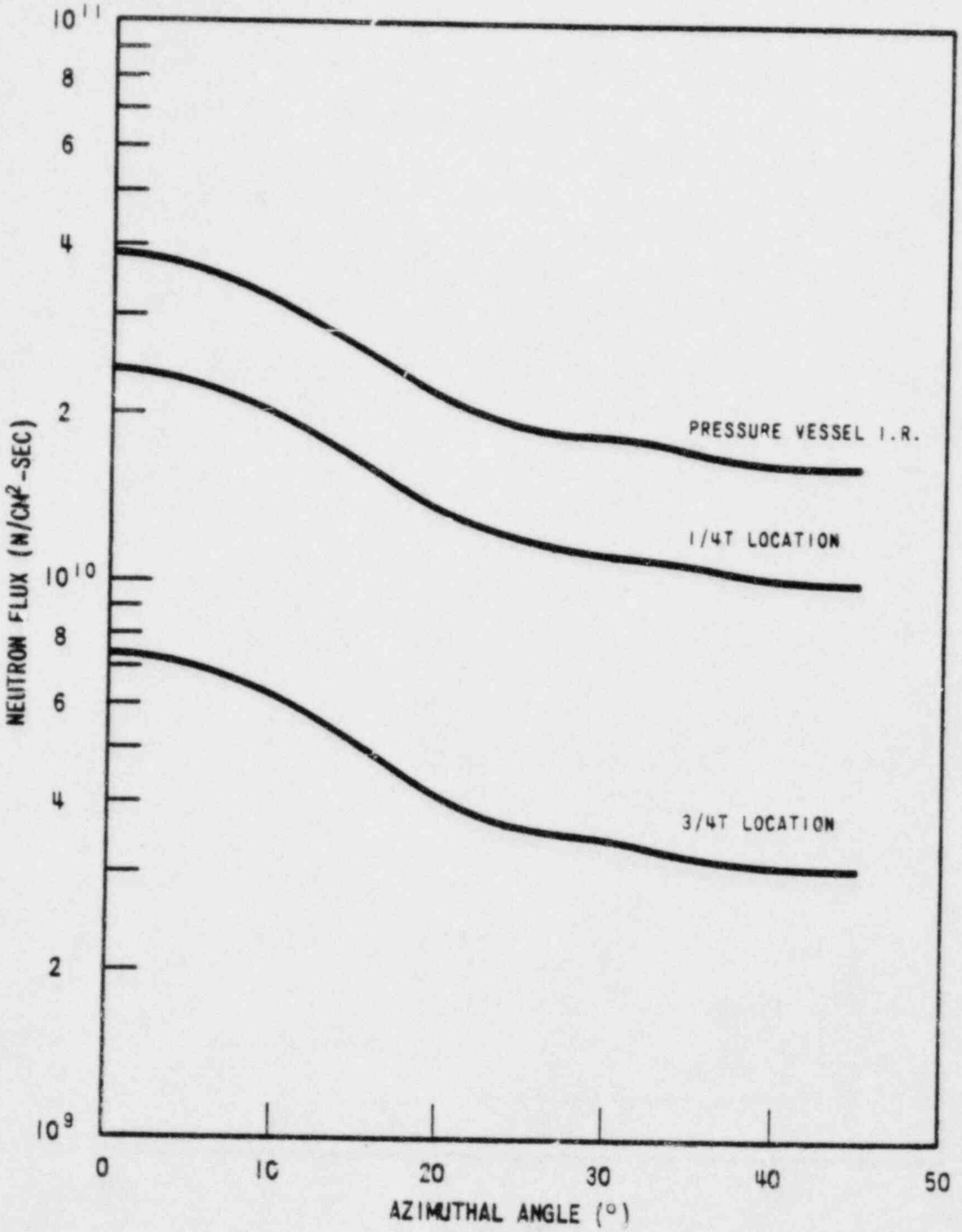


Figure 6-2. Calculated Azimuthal Distribution of Maximum Fast Neutron Flux (E > 1.0 Mev) Within the Point Beach Unit No. 1 Reactor Vessel

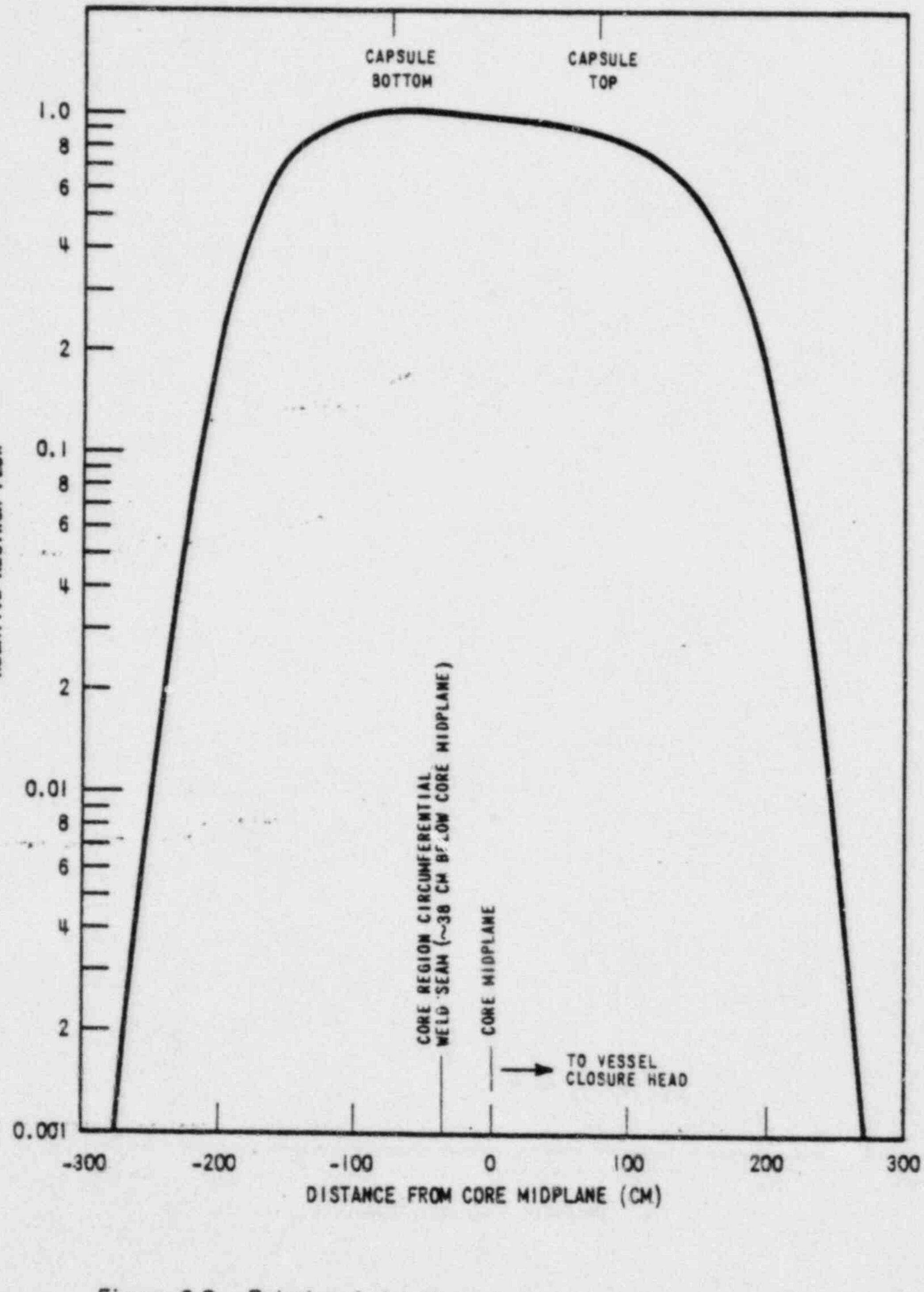


Figure 6-3. Relative Axial Variation of Fast Neutron Flux (E > 1.0 Mev) Incident on the Point Beach Unit No. 1 Reactor Vessel

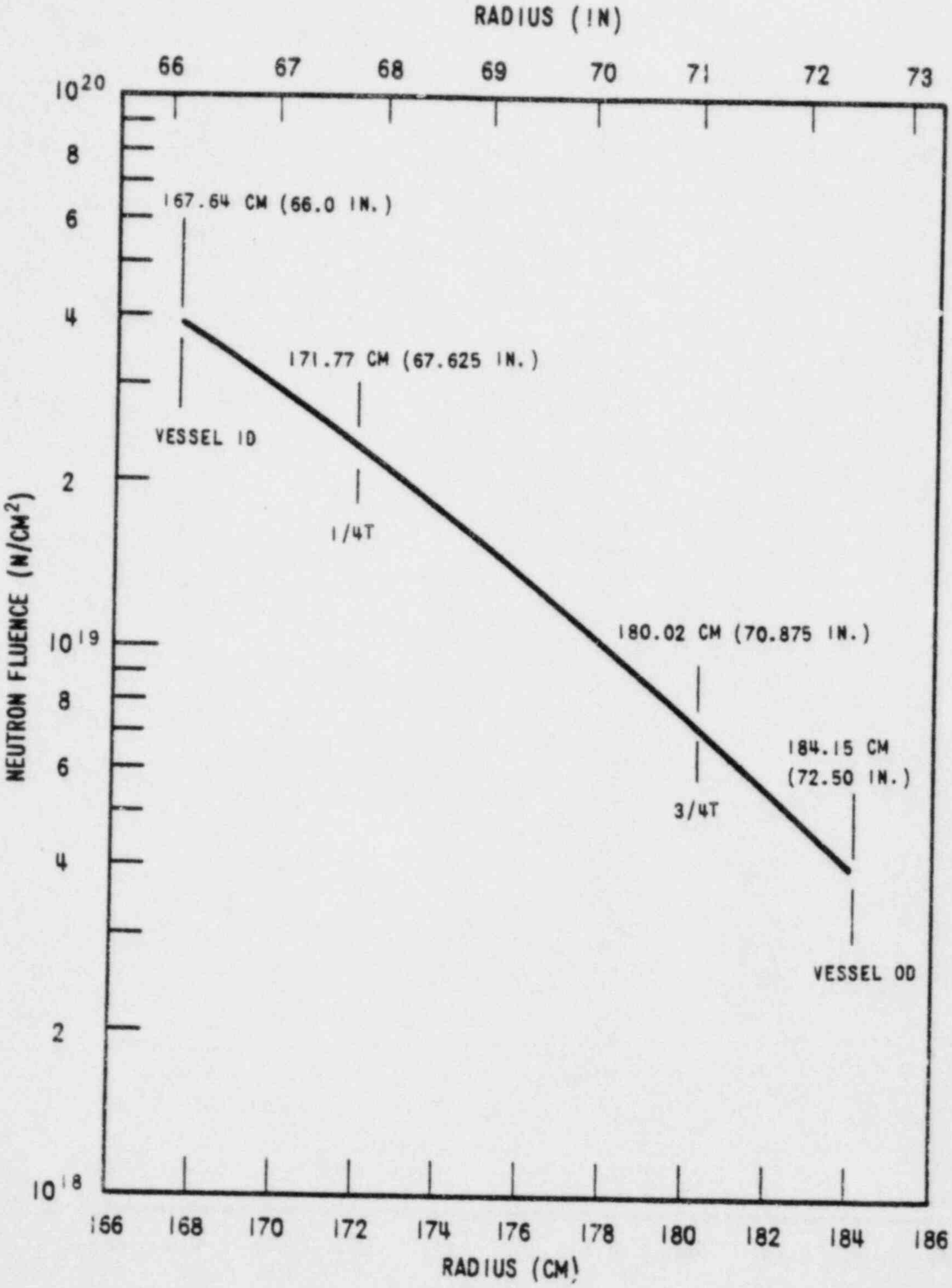


Figure 6-4. Calculated Maximum End of Life Fast Neutron Fluence ($E > 1.0$ Mev) as a Function of Radius Within the Point Beach Unit No. 1 Reactor Vessel

TABLE 6-2
IRRADIATION HISTORY OF CAPSULE R

Month	P_J (MW)	P_{max} (MW)	P_J/P_{max}	Irradiation Time (days)	Decay Time[a] (days)
11/70	247	1518	0.163	30	2695
12/70	610	1518	0.402	31	2664
1/71	971	1518	0.639	31	2633
2/71	978	1518	0.643	28	2605
3/71	1249	1518	0.821	31	2574
4/71	609	1518	0.401	30	2544
5/71	1114	1518	0.733	31	2513
6/71	1233	1518	0.811	30	2483
7/71	1254	1518	0.825	31	2452
8/71	1372	1518	0.903	31	2421
9/71	1238	1518	0.815	30	2391
10/71	1299	1518	0.855	31	2360
11/71	1071	1518	0.705	30	2330
12/71	1317	1518	0.867	31	2299
1/72-6/72	1393	1518	0.916	182	2117
7/72	1375	1518	0.905	31	2086
8/72	1485	1518	0.977	31	2055
9/72	1296	1518	0.853	30	2025
10/72-12/72	0	1518	0.000	92	1933
1/73-2/73	0	1518	0.000	59	1874
3/73	854	1518	0.562	31	1843
4/73	1084	1518	0.713	30	1813
5/73	1092	1518	0.718	31	1782
6/73	1236	1518	0.813	30	1752
7/73	1439	1518	0.947	31	1721
8/73	1345	1518	0.885	31	1690
9/73	1401	1518	0.922	30	1660

a. Decay time is referenced to the counting date of the Fe, Ni, Cu, and Co monitors (4/17/78); the Np and U monitors were counted on 4/25/78.

TABLE 6-2 (cont)
IRRADIATION HISTORY OF CAPSULE R

Month	P_J (MW)	P_{max} (MW)	P_J/P_{max}	Irradiation Time (days)	Decay Time[a] (days)
10/73	1275	1518	0.839	31	1629
11/73	1136	1518	0.747	30	1599
12/73	1268	1518	0.834	31	1568
1/74	1360	1518	0.895	31	1537
2/74	1449	1518	0.953	28	1509
3/74	1481	1518	0.975	31	1478
4/74	260	1518	0.171	30	1448
5/74	0	1518	0.000	31	1417
6/74	937	1518	0.616	30	1387
7/74	1451	1518	0.955	31	1356
8/74	1385	1518	0.911	31	1325
9/74	1379	1518	0.907	30	1295
10/74	1394	1518	0.917	31	1264
11/74	1303	1518	0.857	30	1234
12/74	1482	1518	0.975	31	1203
1/75	671	1518	0.442	31	1172
2/75	1334	1518	0.878	28	1144
3/75	0	1518	0.000	31	1113
4/75	1136	1518	0.748	30	1083
5/75	1481	1518	0.974	31	1052
6/75	1414	1518	0.930	30	1022
7/75	1488	1518	0.979	31	991
8/75	1488	1518	0.979	31	960
9/75	1455	1518	[illegible].957	30	930
10/75	1479	1518	0.973	31	899
11/75	1425	1518	0.937	16	883

a. Decay time is referenced to the counting date of the Fe, Ni, Cu, and Co monitors (4/17/78); the Np and U monitors were counted on 4/25/78.

TABLE 6-2 (cont)
IRRADIATION HISTORY OF CAPSULE R

Month	P_J (MW)	P_{max} (MW)	P_J/P_{max}	Irradiation Time (days)	Decay Time[a] (days)
11/75-12/75	0	1518	0	45	838
1/76	974	1518	0.642	31	807
2/76	1448	1518	0.954	29	778
3/76	1473	1518	0.970	31	747
4/76	1465	1518	0.965	30	717
5/76	1395	1518	0.919	31	686
6/76	1468	1518	0.967	30	656
7/76	1472	1518	0.970	31	625
8/76	1479	1518	0.974	31	594
9/76	1495	1518	0.985	30	564
10/76	36	1518	0.024	31	533
11/76	190	1518	0.125	30	503
12/76	1346	1518	0.887	31	472
1/77	1485	1518	0.978	31	441
2/77	1404	1518	0.925	28	413
3/77	1478	1518	0.974	31	382
4/77	1477	1518	0.973	30	352
5/77	1437	1518	0.947	31	321
6/77	1016	1518	0.669	30	291
7/77	1413	1518	0.931	31	260
8/77	1495	1518	0.985	31	229
9/77	1372	1518	0.904	30	199
10/77	164	1518	0.108	31	168

a. Decay time is referenced to the counting date of the Fe, Ni, Cu, and Co monitors (4/17/78); the Np and U monitors were counted on 4/25/78.

TABLE 6-3

SPECTRUM-AVERAGED REACTION CROSS SECTIONS USED IN FAST NEUTRON FLUX DERIVATION

Reaction	$\overline{\sigma}$ (barns)
Fe^{54} (n,p) Mn^{54} [a]	0.0561
Fe^{54} (n,p) Mn^{54} [b]	0.0587
Ni^{58} (n,p) Co^{58}	0.0758
Cu^{63} (n,α) Co^{60}	0.000427
U^{238} (n,f) F.P.	0.320
Np^{237} (n,f) F.P.	3.00

a. Applicable to samples taken from core-side Charpy specimens

b. Applicable to samples taken from pressure-vessel-side Charpy specimens

TABLE 6-4
RESULTS OF FAST NEUTRON DOSIMETRY FOR CAPSULE R

Reaction and Monitor Location	Measured[a] Activity (dps/gm)	ϕ (E > 1 Mev)[b] (n/cm^2-sec)	Φ (E > 1 Mev)[b] (n/cm^2)
Fe^{54} (n,p) Mn^{54}			
Core-Side			
C-13 (Bottom Center)	2.84×10^6	1.46×10^{11}	2.34×10^{19}
R-10 (Center)	3.11×10^6	1.60×10^{11}	2.56×10^{19}
WH-16 (Top Center)	2.90×10^6	1.49×10^{11}	2.38×10^{19}
Vessel Side			
A-13 (Bottom Center)	2.54×10^6	1.25×10^{11}	2.00×10^{19}
R-11 (Center)	2.54×10^6	1.25×10^{11}	2.00×10^{19}
WW-15 (Top Center)	2.60×10^6	1.28×10^{11}	2.05×10^{19}
Ni^{58} (n,p) Co^{58}			
Center	1.17×10^7	1.62×10^{11}	2.59×10^{19}
Cu^{63} (n,α) Co^{60}			
Bottom	2.19×10^5	1.86×10^{11}	2.98×10^{19}
Bottom-Center	2.14×10^5	1.81×10^{11}	2.90×10^{19}
Top-Center	1.79×10^5	1.52×10^{11}	2.43×10^{19}
Top	2.04×10^5	1.73×10^{11}	2.77×10^{19}
Np^{237} (n,f) Cs^{137}			
Center	7.47×10^6	1.40×10^{11}	2.24×10^{19}
U^{238} (n,f) Cs^{137}			
Center	9.92×10^5	1.67×10^{11}	2.67×10^{19}

a. Mn^{54}, Co^{58}, and Co^{60} activities are referenced to 12:00 4/17/78. Cs^{137} activities are referenced to 12:00 4/25/78.

b. Derived flux and fluence levels are subject to ±10 percent measurement error.

TABLE 6-5

RESULTS OF THERMAL NEUTRON DOSIMETRY FOR CAPSULE R

Monitor Location	Bare Activity[a] (dps/gm)	Cd Covered Activity[a] (dps/gm)	ϕ_{th} (n/cm^2-sec)[b]
Bottom	5.00×10^7	2.67×10^7	9.84×10^{10}
Bottom Center	5.62×10^7	2.85×10^7	1.17×10^{11}
Center	5.24×10^7	2.46×10^7	1.17×10^{11}
Top-Center	5.56×10^7	2.27×10^7	1.39×10^{11}
Top	5.77×10^7	2.32×10^7	1.46×10^{11}

a. Co^{60} activities are referenced to 12:00 4/17/78.

b. Derived flux levels are subject to ±10 percent measurement error.

TABLE 6-6

CALCULATED FAST NEUTRON FLUX AND LEAD FACTORS FOR CAPSULE R

Location Within Capsule R	ϕ (E > 1 Mev) (n/cm^2-sec)	Lead Factor
Front Charpy (Core Side)	1.24×10^{11}	3.18
Dosimeter Block and Flux Wires	1.18×10^{11}	3.03
Back Charpy (Vessel Side)	9.91×10^{10}	2.52