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WCAP-9491

ANALYSIS OF CAPSULE T FROM THE INDIAN POINT UNIT NO. 3 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

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April 1979

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Work performed under Shop Order NYP-77858

Prepared by Westinghouse for the Power Authority, State of New York

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SECTION 1
SUMMARY OF RESULTS AND CONCLUSIONS

The analysis of the reactor vessel material contained in surveillance Capsule T from Indian Point Unit No. 3 reactor pressure vessel led to the following conclusions:

- The capsule received an average fast fluence of 2.92×10^{18} neutrons/cm^2 (E > 1 Mev).

- The fast fluence of 2.92×10^{18} n/cm^2 (E > 1 Mev) resulted in an increase in the average 50 ft lb reference nil-ductility transition temperature (RT_{NDT}). Lower shell plate B2803-3 exhibited a shift of 125°F while the weld metal exhibited a shift of 175°F.

- The irradiated properties of the plate and the weld metal exceeded those predicted by the Westinghouse copper trend curves. Lower shell plate B2803-3 exceeded the predicted shift by 8°F while the weld metal shift in RT_{NDT} exceeded the predicted shift by 41°F.

- Although the shift in RT_{NDT} of Indian Point Unit No. 3 core region material exceeded that predicted, the average upper shelf impact energy remained above 50 foot pounds, thereby providing adequate toughness for continued safe operation.

- Heatup and cooldown limit operating curves were calculated using the actual shifts of the ductile to brittle transition temperature seen by the limiting core region plate and weld metal material.

SECTION 2
INTRODUCTION

This report presents the results of the examination of Capsule T, the first capsule to be removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on Indian Point Unit No. 3 reactor pressure vessel materials under actual operating conditions.

The surveillance program for Indian Point Unit No. 3 reactor pressure vessel materials was designed and recommended by the Westinghouse Electric Corporation. A description of the surveillance program and the preirradiation mechanical properties of the reactor vessel materials are presented by Yanichko.[1] The surveillance program was planned to cover the 40-year life of the reactor pressure vessel and is based on ASTM E-185-62, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors".[2] Westinghouse Nuclear Energy Systems personnel were contracted for the preparation of procedures for removing the capsule from the reactor and its shipment to the Westinghouse Research and Development Laboratory, where the postirradiation mechanical testing of the Charpy V-notch impact and tensile surveillance specimens was performed.

This report summarizes testing and the postirradiation data obtained from surveillance Capsule T removed from Indian Point Unit No. 3 reactor vessel and discusses the analysis of these data.

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1. Yanichko S.E., Davidson J.A., "Consolidated Edison Co. of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP 8475, January, 1975.
2. ASTM Standard E185-62, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors" in ASTM Standards, Part 10, (1962), American Society for Testing and Materials, Philadelphia, Pa., 1962.

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 SA 302 Grade B modified (base material of the Indian Point Unit No. 3 reactor pressure vessel beltline) are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness under certain conditions of irradiation.

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

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

RT_{NDT} and, in turn, the operating limits of nuclear power plants can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel steel can be monitored by a reactor surveillance program such as the Indian Point Unit No. 3 Reactor Vessel

Radiation Surveillance Program,[1] in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 50 ft lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is used to index the material to the K_{IR} curve and, in turn, to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials.

1. Yanichko S.E., Davidson J.A., "Consolidated Edison Co. of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP 8475, January, 1975.

SECTION 4
DESCRIPTION OF PROGRAM

Eight surveillance capsules for monitoring the effects of neutron exposure on the Indian Point Unit No. 3 reactor pressure vessel core region material were inserted in the reactor vessel prior to initial plant startup. The eight capsules were positioned in the reactor vessel between the thermal shield and the vessel wall. The vertical center of the capsules is opposite the vertical center of the core.

Capsule T was removed after approximately 2 years (1.34 effective full power years) of plant operation. This capsule contained Charpy V-notch impact, tensile, and wedge opening loading (WOL) fracture mechanics specimens from intermediate and lower shell plates (B2802-1 and B2803-3, respectively). The capsule also contained Charpy V-notch specimens from the core region weld metal. The chemistry and heat treatment of the surveillance material in Capsule T is presented in table 4-1.

All test specimens were machined from the 1/4-thickness location of the plate. Test specimens represent material taken at least one plate thickness from the quenched end of the plate. Lower shell plate B2803-3 Charpy V-notch specimens were oriented with the longitudinal axis of the specimen both normal (transverse) to and parallel (longitudinal) to the principal working direction of the plate. Charpy V-notch specimens from the intermediate shell plate were oriented longitudinally as were the tensile specimens from both lower shell plate B2803-3 and intermediate shell plate B2802-1. The weld metal Charpy specimens were removed with the notch at the center of the weld and oriented in the welding direction.

The WOL specimens in Capsule T are from intermediate shell plate B2802-1 and were machined such that the simulated crack of the specimen would propagate normal to the major working direction of the plate. All WOL specimens were fatigue precracked per ASTM E399-73 requirements.

Capsule T contained dosimeter wires of pure copper nickel, cadmium and aluminum-0.15 Wt%-cobalt (cadmium-shielded and unshielded). The test specimens served as iron dosimeters.

TABLE 4-1

CHEMISTRY AND HEAT TREATMENT OF THE SURVEILLANCE MATERIAL FROM INDIAN POINT UNIT NO. 3 SURVEILLANCE CAPSULE T

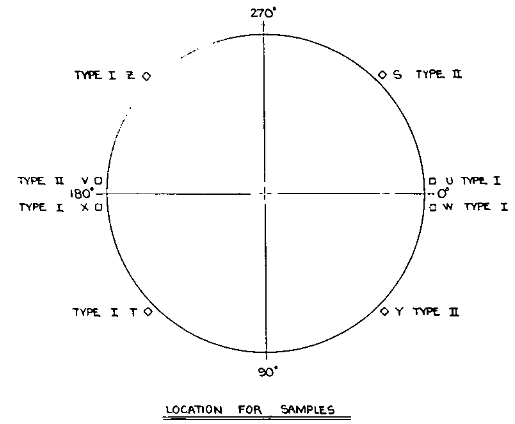
Element	Chemical Analyses (Percent)		
	Intermediate Shell Plate B2802-1[a]	Lower Shell Plate B2803-3[a]	Weld Metal[b]
C	0.22	0.22	0.08
Mn	1.41	1.30	1.18
P	0.010	0.012	0.019
S	0.023	0.024	0.016
Si	0.28	0.28	0.17
Ni	0.50	0.52	1.02
Cr	0.08	0.08	0.04
Mo	0.46	0.45	0.53
Cu	0.18	0.24	0.15
Al	0.036	0.03	< 0.01
V	< 0.01	< 0.01	< 0.01
Sn	0.014	< 0.01	0.007
Cb	< 0.01	< 0.01	< 0.01
Zr	< 0.01	< 0.01	< 0.01
Ti	< 0.01	< 0.01	< 0.01

a. Heat treatment – Heated 1550-1650°F for 4 hours, water-quenched; tempered at 1225°F for 4 hours, furnace-cooled; stress-relieved at 1150°F for 40 hours, furnace-cooled

b. Stress relieved at 1150°F for 40 hours, furnace-cooled

Thermal monitors made from two low melting eutectic alloys and sealed in quart tubes were included in the capsule and were located as shown in figure 4-1. 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



NOTES: ASSEMBLY & TEST PROCEDURE

1. CLEAN ALL PARTS AS PER WPS 292722-1 PRIOR TO ASSEMBLY.
2. INSERT ITEMS 21 THRU 26 INTO ITEM 8 AND WELD PLUGS.
3. INSTALL SPECIMEN, SPACERS, ETC, (ITEMS 7 THRU 20) IN ENCLOSURE HALF.
4. WELD ENCLOSURE HALVES ITEM 6 TOGETHER.
5. WELD END PLUGS (ITEMS 3 & 4) TO ENCLOSURE HALVES (ITEM 6) ORIENT (ITEM 3) AS SHOWN ON DRAWING.
6. TEMPORARILY INSTALL PLUG (ITEM 5) IN PLACE. DO NOT WELD PLUG OR EXTENSION TUBE (ITEM 2) PLACE ASSEMBLY IN WELD BOX.
7. EVALUATE ENCLOSURE & BACK FILL WITH HELIUM AT ONE ATMOS. PRESSURE. WELD PLUG (ITEM 5) TO END PLUG (ITEM 3) IN HELIUM ATMOSPHERE.
8. HELIUM LEAKTEST SEALED ENCLOSURE. MAXIUM ALLOWABLE LEAK RATE 1×10^{-6} CC/SEC. AT ONE ATMOSPHERE PRESSURE.
9. AUTOCLAVE ENCLOSURE IN DIMINERALIZED WATER AT 2800 PSIA & 550°F FOR A PERIOD NOT TO EXCEED 24 HOURS AT TEMPERATURE & PRESSURE.
10. HELIUM LEAK TEST ENCLOSURE AS IN STEP 8 ABOVE AFTER PRESSURIZATION WITH HELIUM FOR 15 MIN AT 40 PSI.
11. WELD PER WPS 82148PF EXCEPT WELDER IDENTIFICATION (PARA. 8) NOT REQUIRED.
12. FLUID PENETRANT TEST, WELDS MADE IN NOTE 5 ABOVE PER WPS 595139. ALL WELDS SHALL BE COMPLETELY FREE OF INDICATIONS.

BILL OF MATERIAL

NOTE	ITEM	TITLE	DRAWING & GR. OR IT.	MATERIAL SPECIFICATION	EQUIVALENT SPECIFICATION	NO. REQ.
	1	TOP SUPPORT PLUG	883D952-1T-1			1
	2	EXTENSION TUBE		SEE NOTE A		1
	3	TOP END PLUG	499B424-1T-1			1
	4	BOTTOM END PLUG	676C145-1T-1			1
	5	PLUG	676C145-1T-2			1
	6	ENCLOSURE HALF	676C181-1T-2			2
	7	IMPACT SPECIMEN	500B315-1T-1			32
	8	SPACER	676C182-1T-1			3
	9	SPACER	676C182-1T-3			4
	10	SPACER	676C182-1T-4			2
	11	SPACER	676C182-1T-7			1
	12	SPACER	676C182-1T-8			2
	13	SPACER	676C182-1T-9			2
	14	TENSILE SPECIMEN	674C487-1T2			3
	15	TENSILE COVER	499B826-1T-1			6
	16	WOL SPECIMEN	500B775-1T-1			6
	17	.375x1.000LG DOWEL PIN		CARBON STL		6
	18	.375x.375LG DOWEL PIN		CARBON STL		6
	19	.187x.875LG DOWEL PIN		COML SST		1
	20	.500-20UNC x .375LG FLAT PT. SOC. SET SCR		CARBON STL		6
	21	COBALT SPECIMEN		SEE NOTE B		6
	22	CADMIUM SPECIMEN		CADMIUM		3
	23	COPPER SPECIMEN		COPPER		2
	24	NICKEL SPECIMEN		NICKEL		1
	25	579°F THERMAL MONITOR		2.5AG, 97.5PB		2
	26	590°F THERMAL MONITOR		1.75AG, 0.75SN, 97.5PB		1
	27	SPACER		TO BE SPEC BY ENGR.		1

A - ASTM A213, TYPE 304, SEAMLESS
B - ALUM. COBALT WIRE 0.15% COBALT

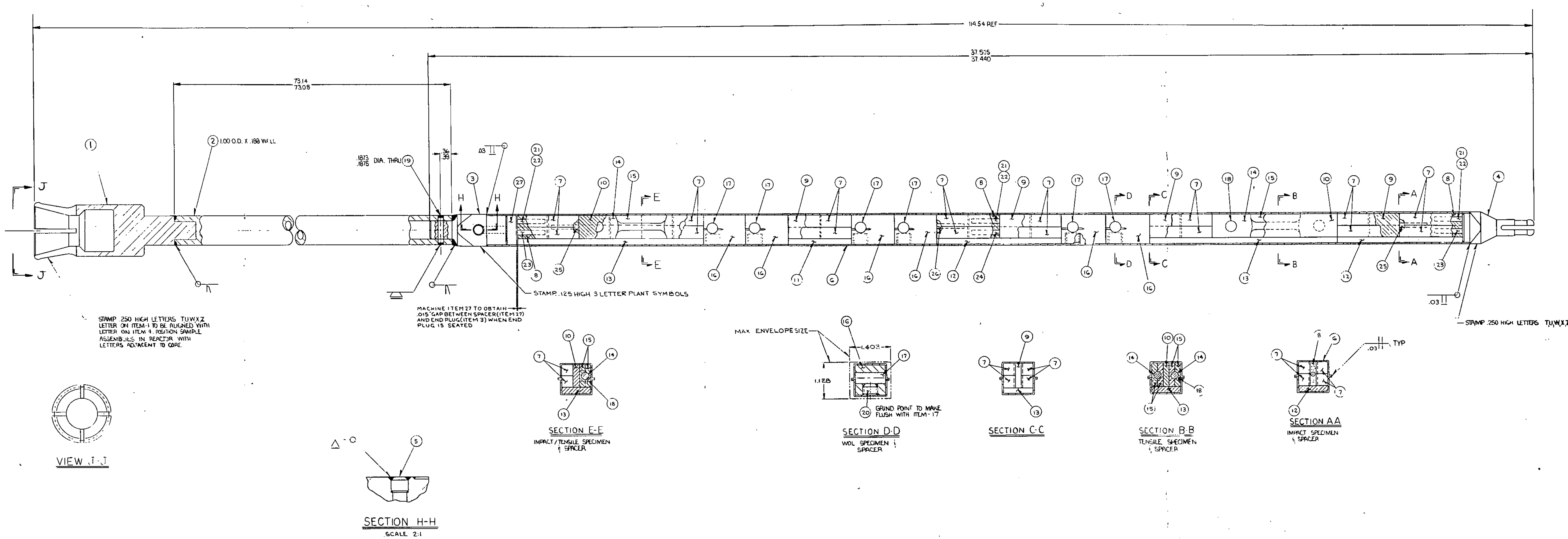


Figure 4-1. Schematic Diagram of Capsule T and Arrangement of Surveillance Capsules in the Reactor Vessel

SECTION 5
TESTING SPECIMENS FROM CAPSULE T

5-1. TEST PROCEDURE

The postirradiation mechanical testing of the Charpy V-notch and tensile specimens was performed at the Westinghouse Research and Development Hot Laboratory with consultation by Westinghouse Nuclear Energy Systems personnel. Testing was performed in accordance with 10CFR50, appendixes 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-8475.[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 during irradiation was less than 579°F.

A Tinius Olsen Model 74 impact test machine was used to perform tests on the irradiated Charpy V-notch specimens. 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 having a 20,000-pound capacity. A crosshead speed of 0.5 in./min was used. The deformation of the specimen was measured using a strain gage extensometer. The extensometer was calibrated before testing with a Sheffield high magnification drum-type extensometer calibrator. 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.

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1. Yanichko S.E., Davidson J.A., "Consolidated Edison Co. of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP 8475, January, 1975.

5-2. CHARPY V-NOTCH IMPACT TEST RESULTS

The irradiated Charpy V specimens represented reactor pressure vessel beltline plate material and weld metal. The results are presented in tables 5-1 and 5-2 and figures 5-1 through 5-4. The unirradiated data are also shown in figures 5-1 through 5-4 for comparison with the postirradiation data. A summary of the increase in the 50 ft lb fix transition temperature and the decrease in the upper shelf energy is presented in table 5-3. The fracture and specimen appearance of all the Charpy specimens tested in the Indian Point Unit No. 3 surveillance Capsule T are shown in figures 5-5 through 5-8.

5-3. TENSILE TEST RESULTS

The results of the tensile tests are presented in table 5-4 and figures 5-9 and 5-10. Tests were performed on specimens from the plate material at room temperature and 300°F. Irradiation increased the 300°F yield and ultimate tensile strength of intermediate shell plate B2802-1 by 26.7 and 18 percent, respectively. The increase in the lower shell plate B2803-3 yield and ultimate tensile strength was 33.6 and 21.8 percent, respectively, at room temperature and 30.6 and 22 percent, respectively, at 300°F. A typical load-displacement curve obtained for the tensile tests is shown in figure 5-11. The fractured tensile specimens from the surveillance capsule are shown in figure 5-12.

5-4. WEDGE OPENING LOADING TEST RESULTS

The wedge opening loading (WOL) fracture mechanics specimens which were contained in the surveillance capsule have been stored at the Westinghouse Research Laboratory on the recommendation of the U. S. Nuclear Regulatory Commission and will be tested at a later date. The results of these tests will be reported upon their completion.

5-5. DISCUSSION OF RESULTS

The shift in RT_{NDT} of lower shell plate B2803-3 of 125°F exceeded the shift predicted by the Westinghouse trend curve by 8°F. (Refer to appendix A.) A similar comparison to the NRC Regulatory Guide 1.99 trend curves shows an excess of 5°F of that predicted. The shift of 175°F in RT_{NDT} for the weld metal exceeded the upper limit predicted shift of the Westinghouse trend curve by 41°F. Regulatory Guide 1.99 predicts a shift similar to the 175°F observed. The upper limit curves were used in the comparison because the weld procedure used in the Indian Point Unit No. 3 vessel did not specify a control on copper content. However, the copper content of the weld metal was analyzed from a broken weld Charpy as a double check and found to be 0.14 Wt%. This is essentially the same percent as that originally reported (0.15 Wt%). Since the copper content was verified to be 0.14 to 0.15 Wt%, another explanation for the large shift in RT_{NDT} for the weld metal was examined.

TABLE 5-1

CHARPY V-NOTCH IMPACT DATA OF INDIAN POINT UNIT NO. 3 PRESSURE VESSEL SHELL PLATES IRRADIATED AT 550°F, FLUENCE 2.92 x 10^{18} n/cm^2 (E > 1 Mev)

Specimen Number	Test Temp (°F)	Energy (ft lb)	Lateral Expansion (mils)	Shear (%)
Intermediate Plate B2802-1 (Longitudinal)				
B16	15	9.0	5	5
B11	65	30.0	24	20
B9	70	23.5	21	20
B12	125	45.0	32	30
B13	165	61.0	49	40
B10	190	79.0	60	50
B14	210	101.0	73	85
B15	300	119.0	82	100
Lower Plate B2803-3 (Longitudinal)				
A32	70	12.0	9	10
A30	135	19.5	17	20
A31	175	33.0	28	25
A29	200	46.0	41	40
A27	210	50.0	39	50
A28	250	92.0	71	99
A26	300	88.0	63	100
A25	400	96.0	60	100
Lower Plate B2803-3 (Transverse)				
AT58	70	13.5	10	10
AT60	120	22.5	17	25
AT54	150	22.0	19	20
AT59	175	30.0	28	30
AT55	210	36.5	32	45
AT57	225	47.5	39	60
AT61	250	56.0	49	100
AT56	300	58.0	55	100

TABLE 5-2

CHARPY V-NOTCH IMPACT DATA OF INDIAN POINT UNIT NO. 3 SURVEILLANCE WELD METAL MATERIAL IRRADIATED AT 550°F, FLUENCE 2.92 x 10^{18} n/cm^2 (E > 1 Mev)

Specimen Number	Test Temp (°F)	Energy (ft lb)	Lateral Expansion (mils)	Shear (%)
W37	0	13.0	9	15
W40	70	17.5	17	20
W34	110	48.0	45	55
W36	150	55.5	46	60
W39	150	53.0	35	55
W38	165	66.0	55	60
W33	210	78.0	73	98
W35	300	90.5	74	100

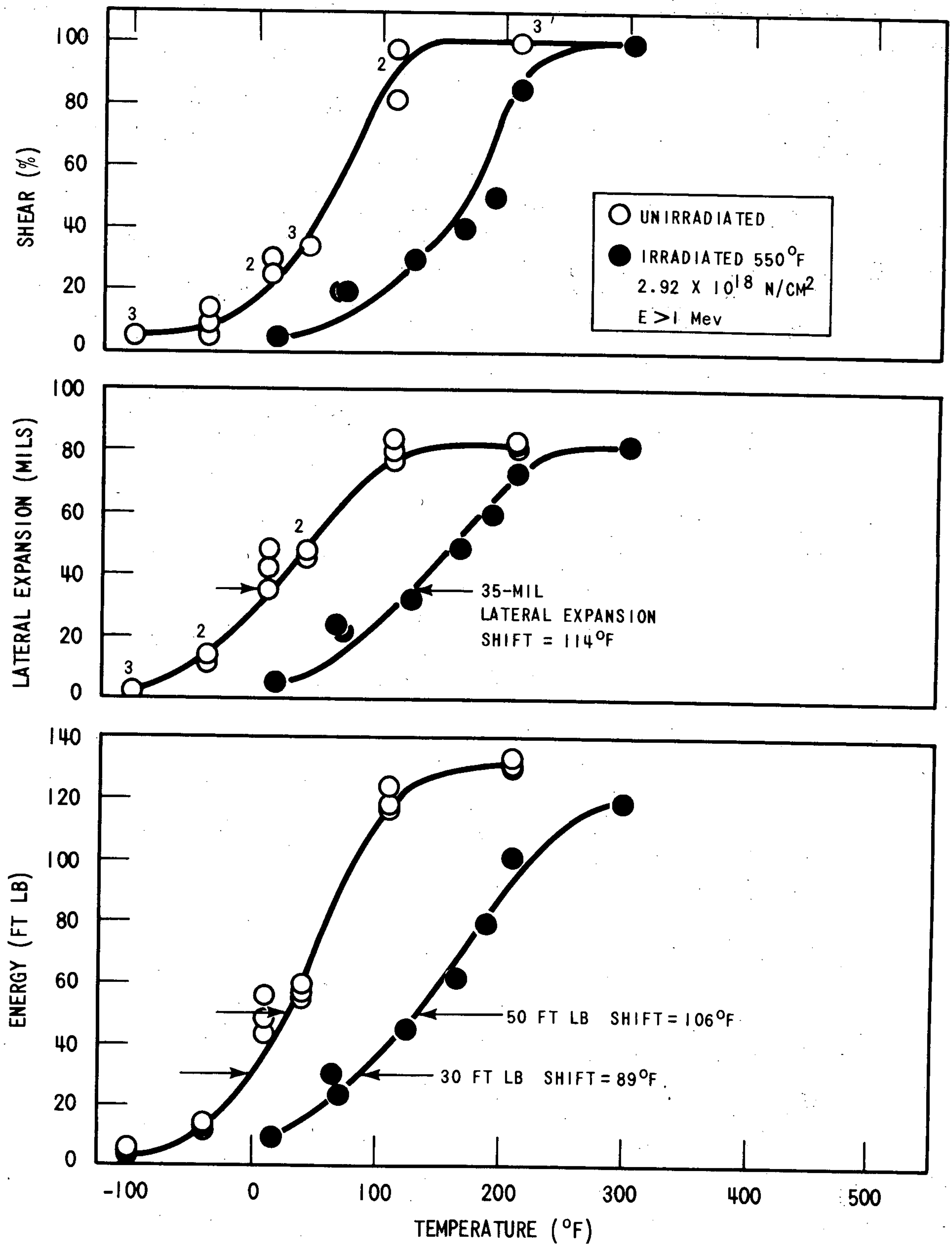


Figure 5-1. Irradiated Charpy V-Notch Impact Properties for Indian Point Unit No. 3 Reactor Vessel Intermediate Shell Plate B2802-1 (Longitudinal)

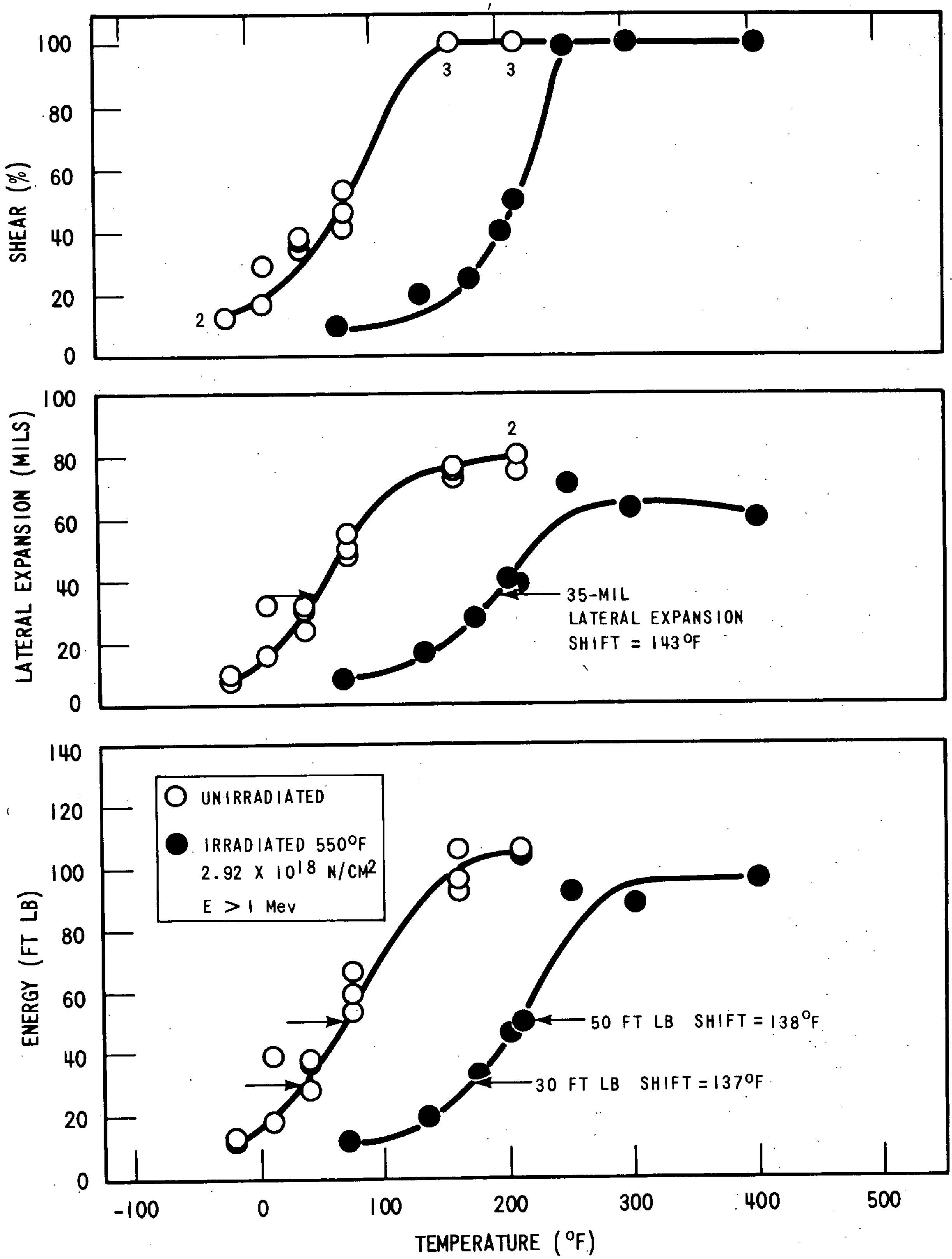


Figure 5-2. Irradiated Charpy V-Notch Impact Properties for Indian Point Unit No. 3 Reactor Vessel Lower Shell Plate B2803-3 (Longitudinal)

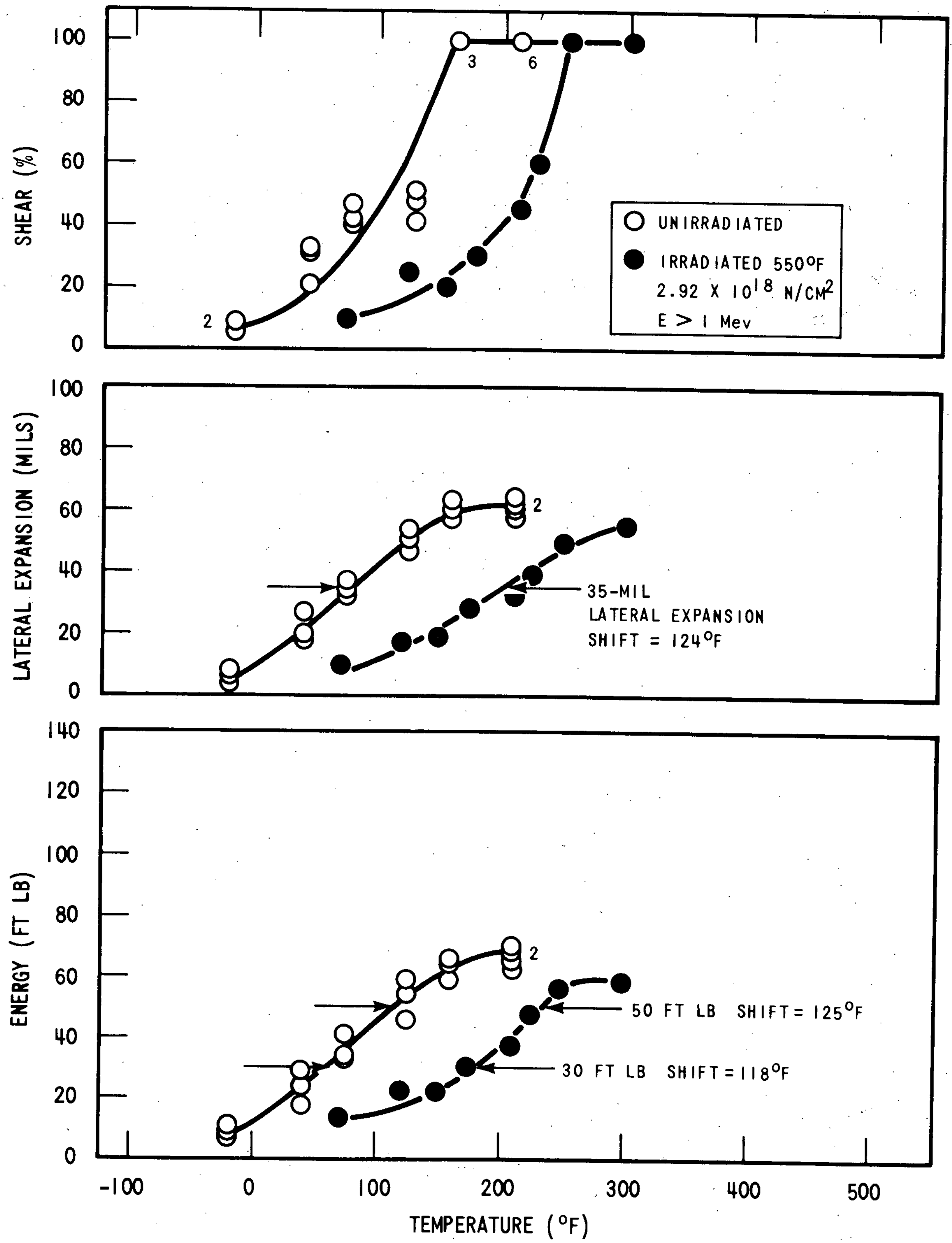


Figure 5-3. Irradiated Charpy V-Notch Impact Properties for Indian Point Unit No. 3 Reactor Vessel Lower Shell Plate B2803-3 (Transverse)

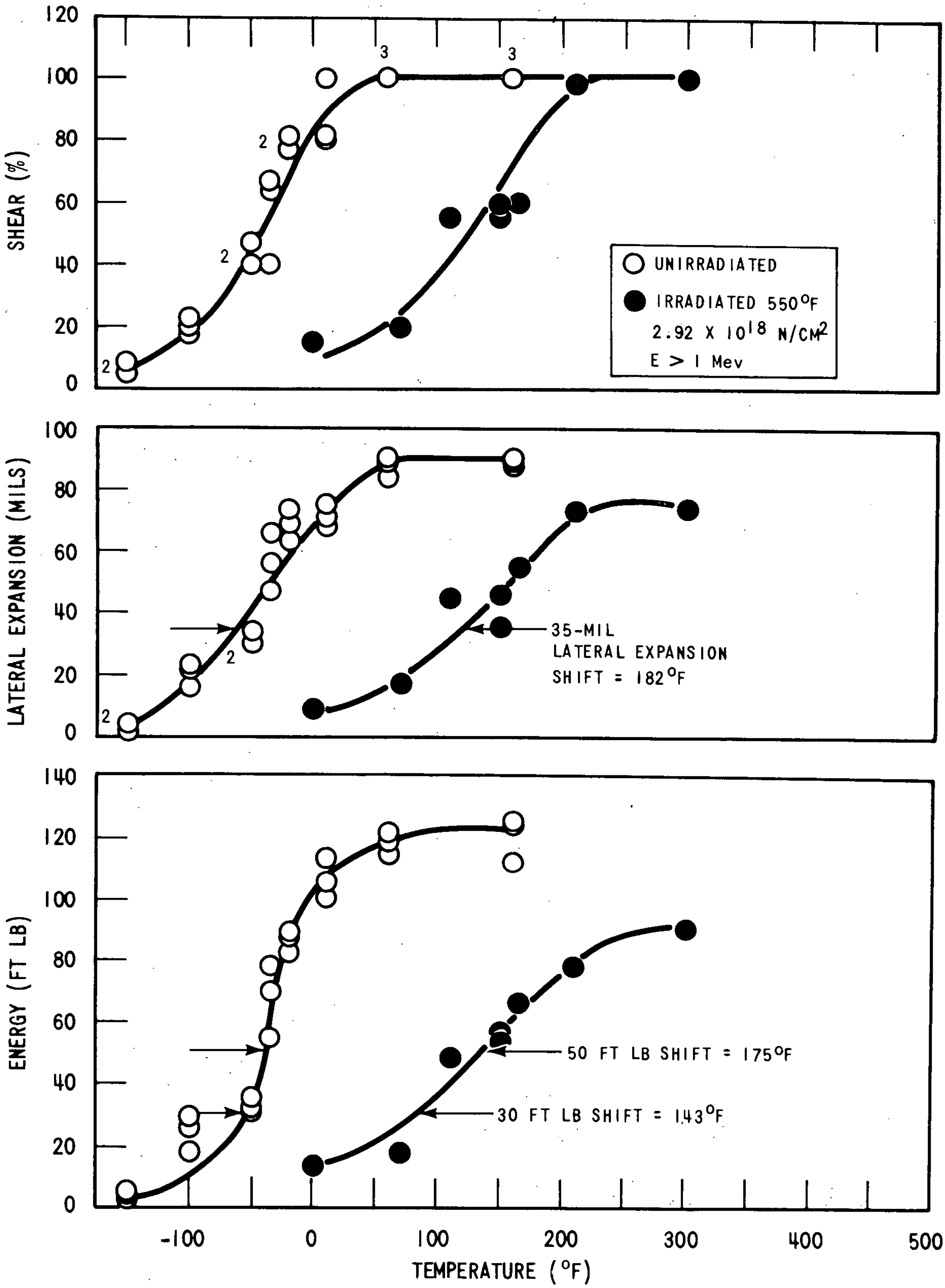
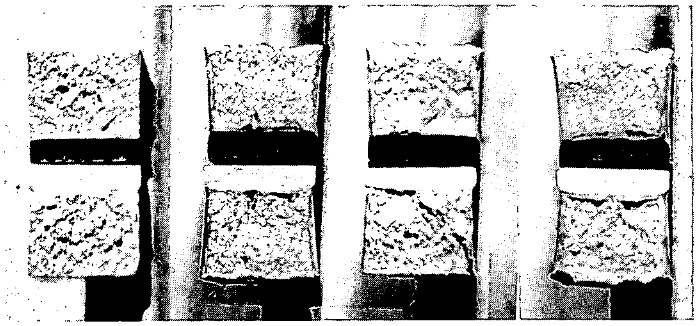


Figure 5-4. Irradiated Charpy V-Notch Impact Properties for Indian Point Unit No. 3 Weld Metal Material

TABLE 5-3

EFFECT OF 550°F IRRADIATION AT 2.92 x 10^{18} n/cm^2 (E > 1 Mev) ON NOTCH TOUGHNESS PROPERTIES OF INDIAN POINT UNIT NO. 3 REACTOR VESSEL IMPACT TEST SPECIMENS

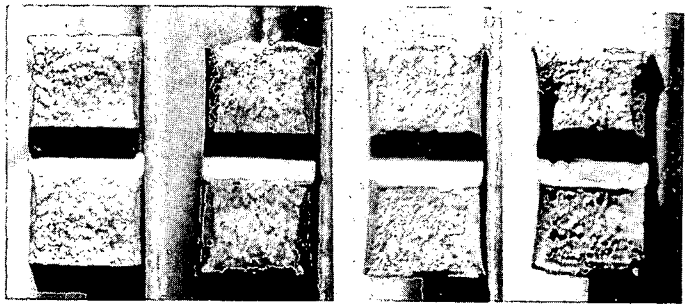
	Average 50 ft lb Temp (°F)			Average 35 mil Lateral Expansion Temp (°F)			Average 30 ft lb Temp (°F)			Average Energy Absorbtion at Full Shear (ft lb)		
Material	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	Δ (ft lb)
Plate B2802-1 (Longitudinal)	28	134	106	11	125	114	-2	87	89	132	119	13
Plate B2803-3 (Longitudinal)	89	207	138	50	193	143	33	170	137	105	96	9
Plate B2803-3 (Transverse)	110	235	125	78	202	124	60	178	118	67	58	9
Weld Metal	61	136	175	-62	120	182	-55	88	143	120	91	29



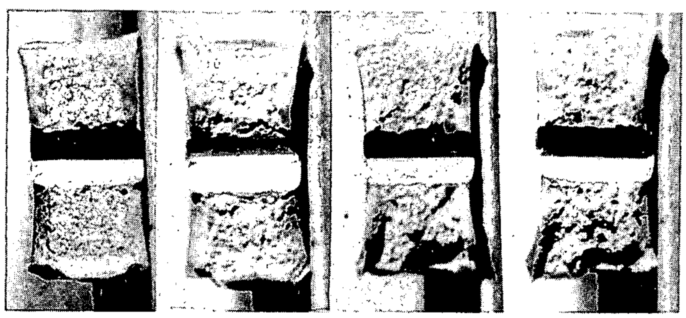
B-16 B-11 B-9 B-12

B-13 B-10 B-14 B-15

Figure 5-5. Charpy Impact Specimen Fracture Surfaces for Indian Point Unit No. 3 Reactor Vessel Intermediate Shell Plate B2802-1 (Longitudinal)

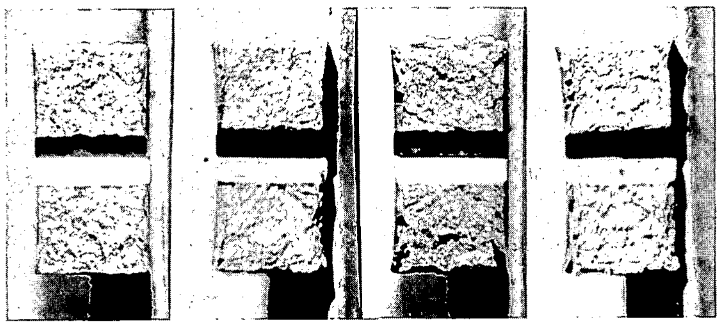


A-29 A-31 A-30 A-32

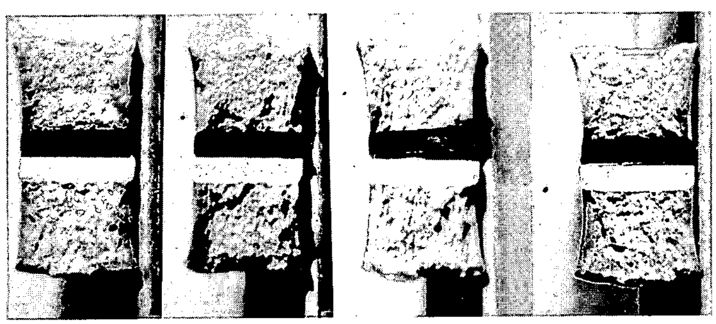


A-27 A-28 A-26 A-25

Figure 5-6. Charpy Impact Specimen Fracture Surfaces for Indian Point Unit No. 3 Reactor Vessel Lower Shell Plate B2803-3 (Longitudinal)

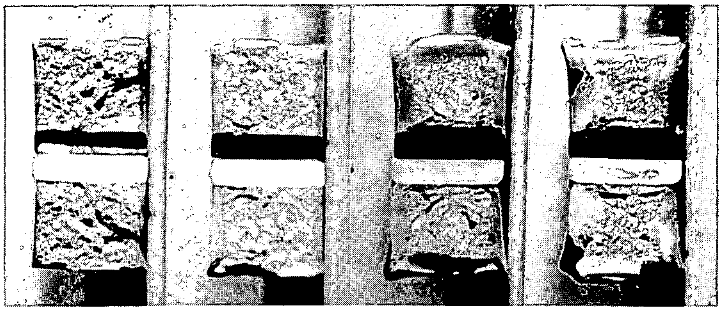


AT-58 AT-60 AT-54 AT-59

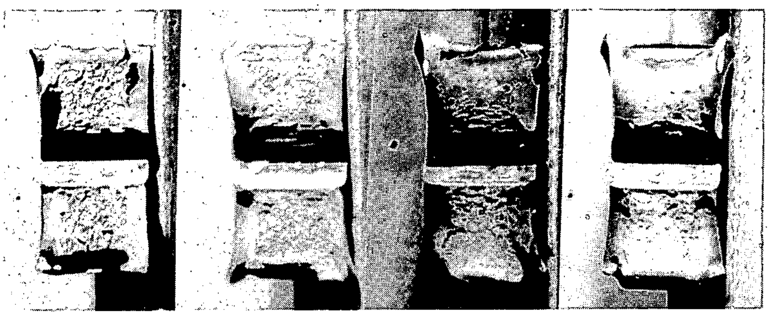


AT-55 AT-57 AT-61 AT-56

Figure 5-7. Charpy Impact Specimen Fracture Surfaces for Indian Point Unit No. 3 Reactor Vessel Lower Shell Plate B2803-3 (Transverse)



W-37 W-40 W-34 W-36



W-39 W-38 W-33 W-35

Figure 5-8. Charpy Impact Specimen Fracture Surfaces for Indian Point Unit No. 3 Weld Metal Material

TABLE 5-4

TENSILE PROPERTIES FOR INDIAN POINT UNIT NO. 3 PRESSURE VESSEL SHELL PLATES IRRADIATED AT 550°F, FLUENCE 2.92×10^{18} n/cm^2 (E > 1 Mev)

Material	Specimen Number	Test Temp (°F)	0.2% Yield Strength (psi)	Ultimate Tensile Strength (psi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Intermediate Shell Plate B2802-1	B2	300	68.4	85.7	9.5	20.8	67
Lower Shell Plate B2803-3	A3 A4	R. T. (74) 300	86.1 76.7	105.1 95.7	11.0 10.3	22.0 20.2	62 62

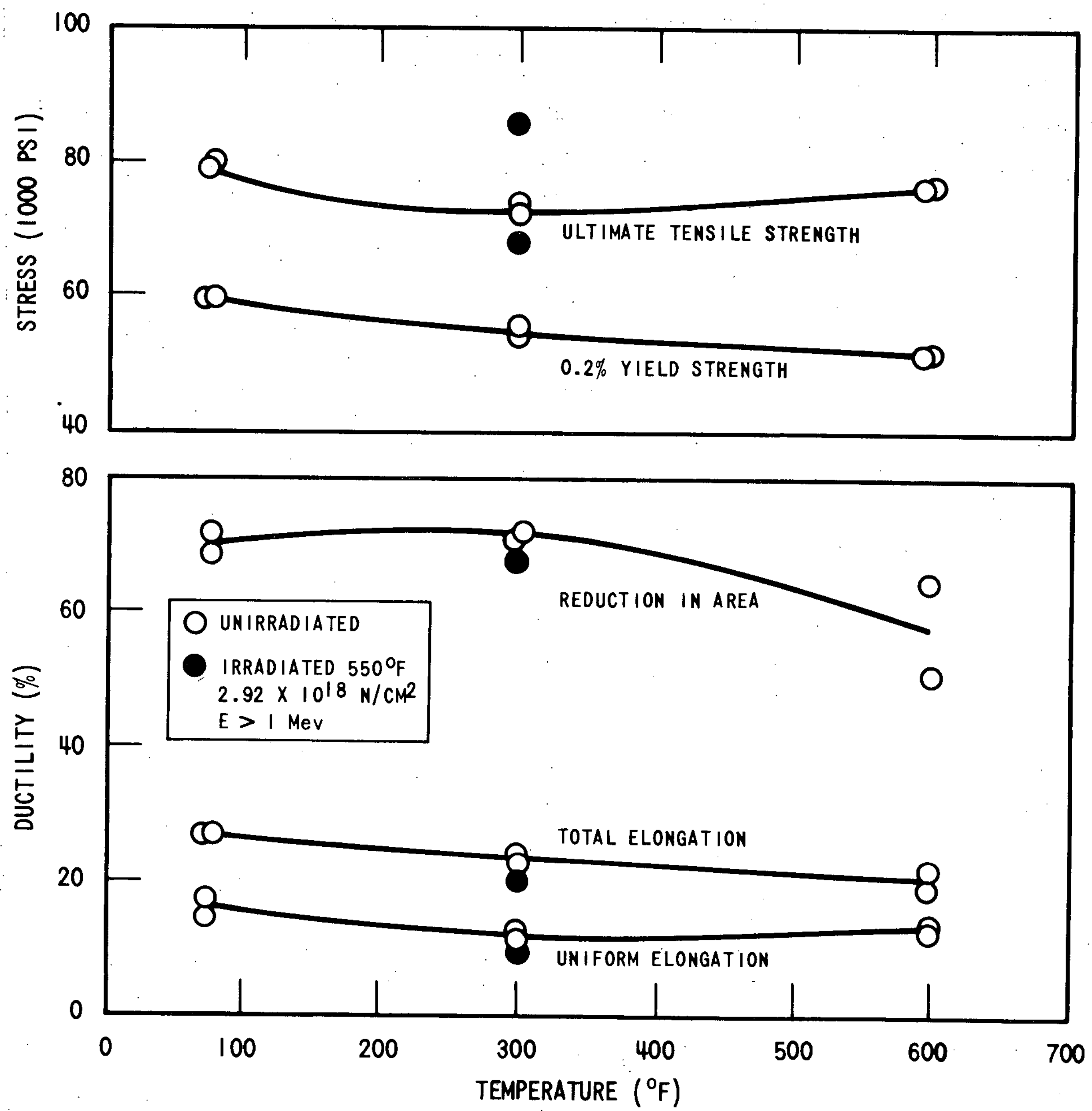


Figure 5-9. Irradiated Tensile Properties for Indian Point Unit No. 3 Reactor Vessel Intermediate Shell Plate B2802-1

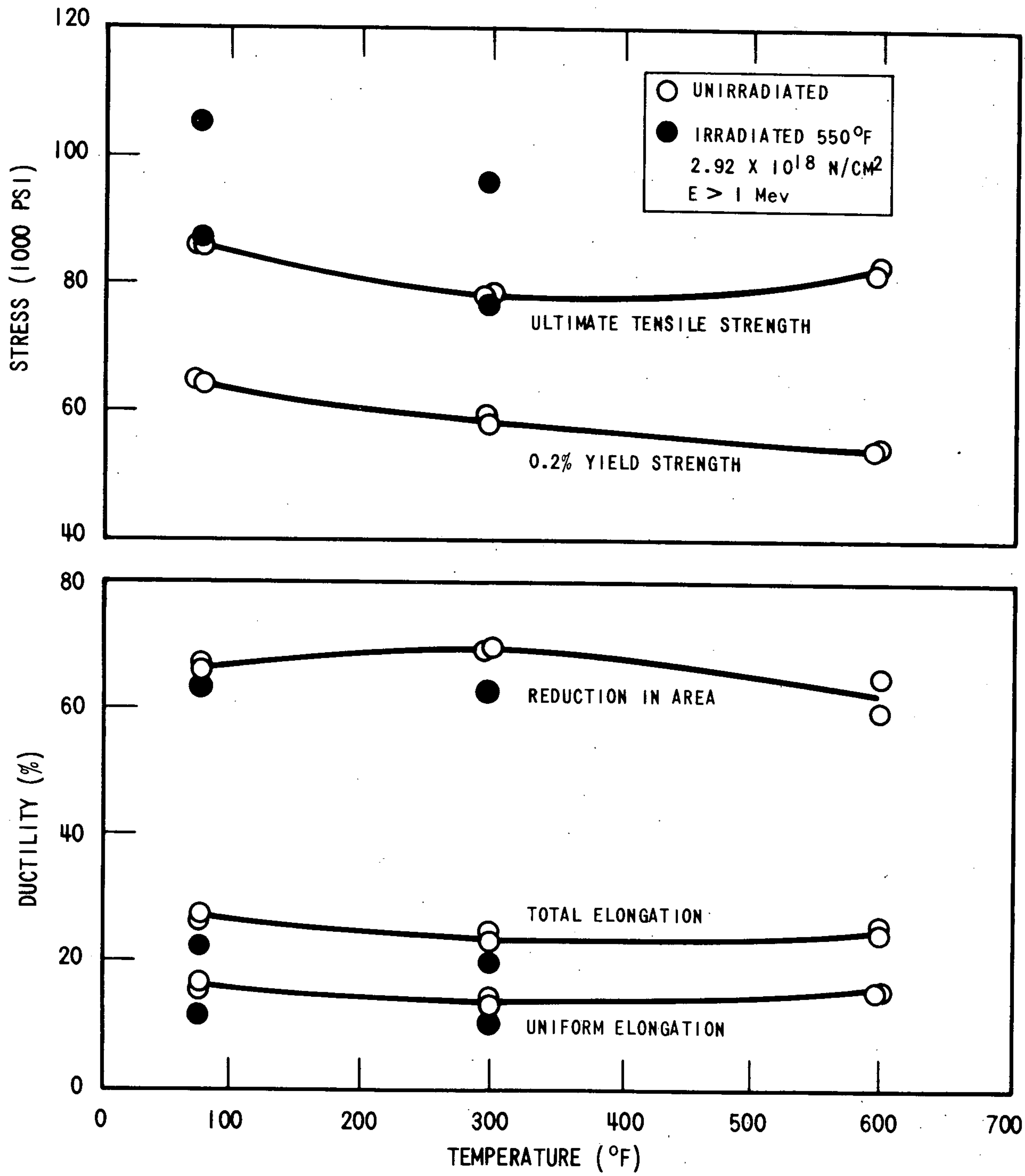


Figure 5-10. Irradiated Tensile Properties for Indian Point Unit No. 3 Reactor Vessel Lower Shell Plate B2803-3

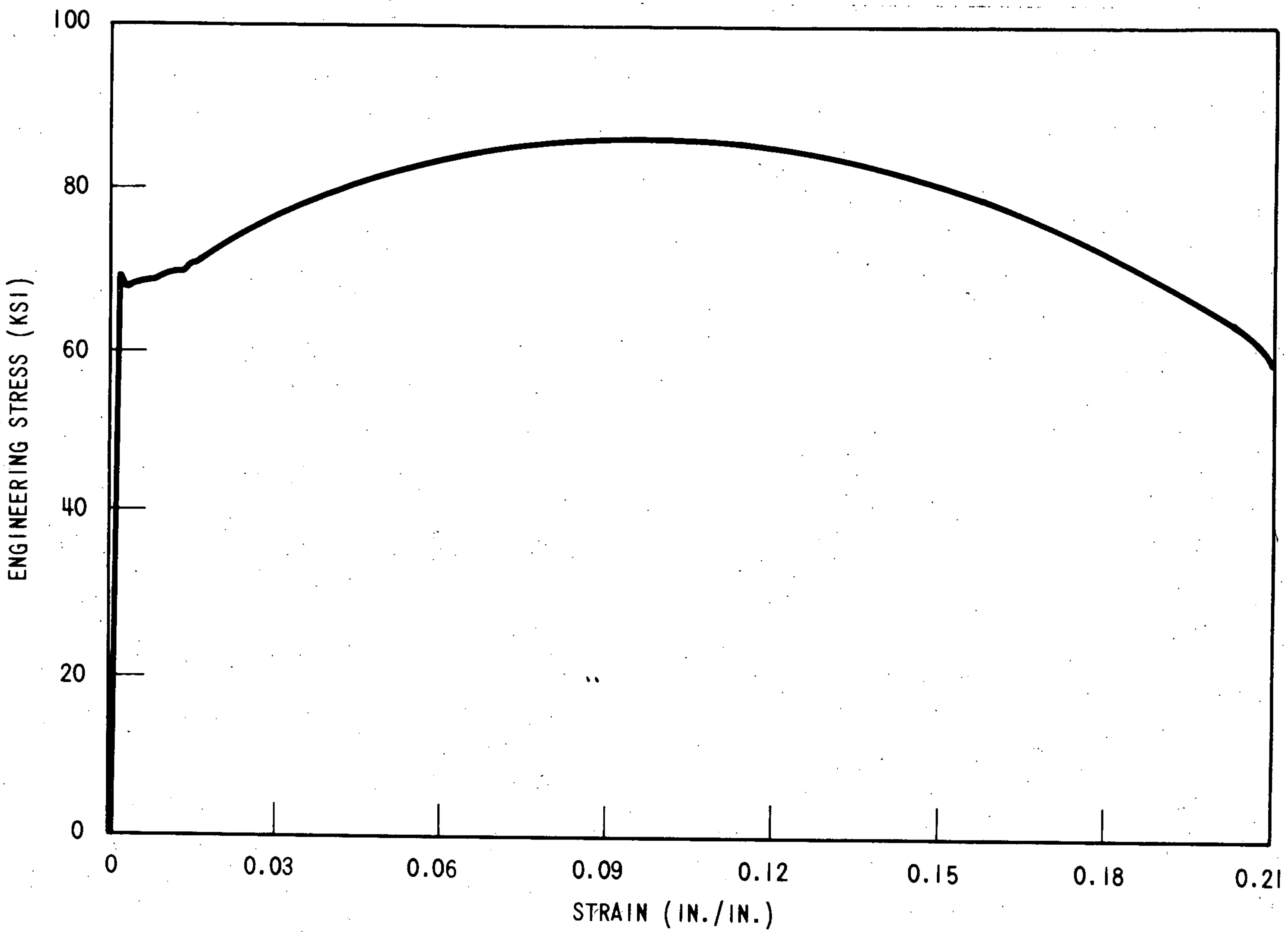
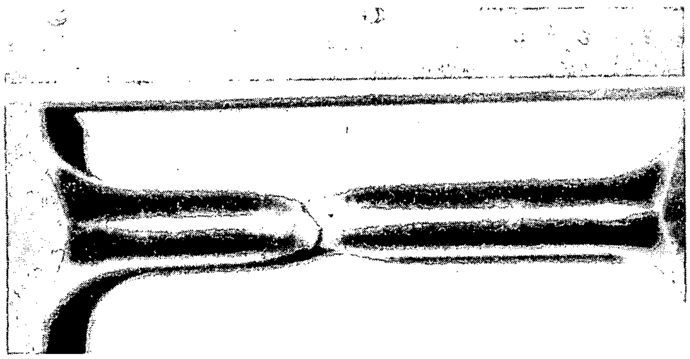
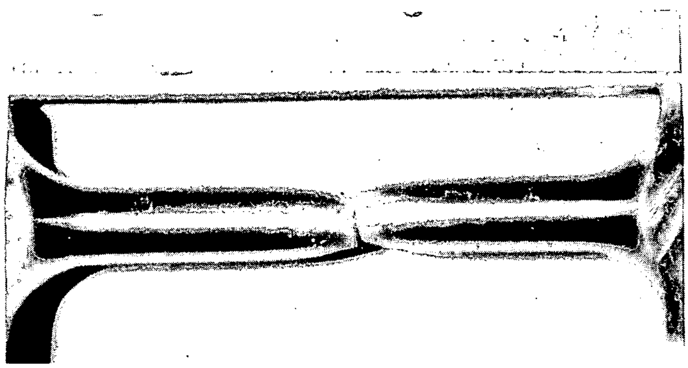


Figure 5-11. Typical Stress-Strain Curve for Tension Specimens

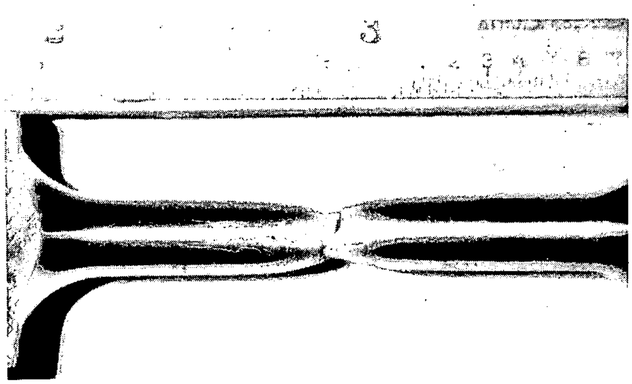
14,179-12



SPECIMEN B-2



SPECIMEN A-3



SPECIMEN A-4

Figure 5-12. Fractured Tensile Specimens from Indian Point Unit No. 3 Reactor Vessel Intermediate and Lower Shell Plates B2802-1 and B2803-3

Previous power reactor irradiated weld data of similar copper content, but containing a relatively high nickel content (greater than 0.75 Wt%), showed a similar shift behavior. The Indian Point Unit No. 3 weld metal, containing 1.02 Wt% nickel, is consistent with these other high nickel shifts. However, the data show that the magnitude of the shift stays the same at higher fluences, thereby falling below the Westinghouse trend curves later in plant life. This can be verified following removal and testing of the next irradiation surveillance capsule from the Indian Point Unit No. 3 reactor.

With regard to the shift in RT_{NDT} of lower shell plate B2803-3 of 8°F greater than predicted, other power reactor data from similar plate material showed that, although shifts exceeded the Westinghouse trend curves at fluences less than 7×10^{18} n/cm^2 (E > 1 Mev), the magnitude of the shift remained approximately constant and fell below the Westinghouse trend curve at higher fluence levels. It is anticipated that the Indian Point Unit No. 3 plate will behave in a similar manner. However, as in the case with the weld metal, a second capsule must be removed and tested to confirm this.

The upper shelf energy from the Charpy V-notch specimen tests showed that the limiting core region material from the Indian Point Unit No. 3 reactor vessel has adequate toughness for continued safe operation.

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. The particular monitors contained within Capsule T, along with the nuclear reaction of interest and the energy range of each monitor, are listed in table 6-1.

The first three 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 T.

The relative locations of the various monitors within Capsule T are shown in figure 4-1 while the radial and azimuthal positions of the capsule with respect to the nuclear core, reactor internals, and pressure vessel are illustrated in figure 6-1. The copper, nickel, aluminum, 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 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

TABLE 6-1
NEUTRON FLUX MONITORS CONTAINED WITHIN CAPSULE T

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
Cobalt-aluminum[a]	Co^{59} (n,γ) Co^{60}	0.0015	$0.4ev < E < 0.015$ Mev	5.27 years
Cobalt-aluminum	Co^{59} (n,γ) Co^{60}	0.0015	$E < 0.015$ Mev	5.27 years

a. Denotes that monitor is cadmium shielded.

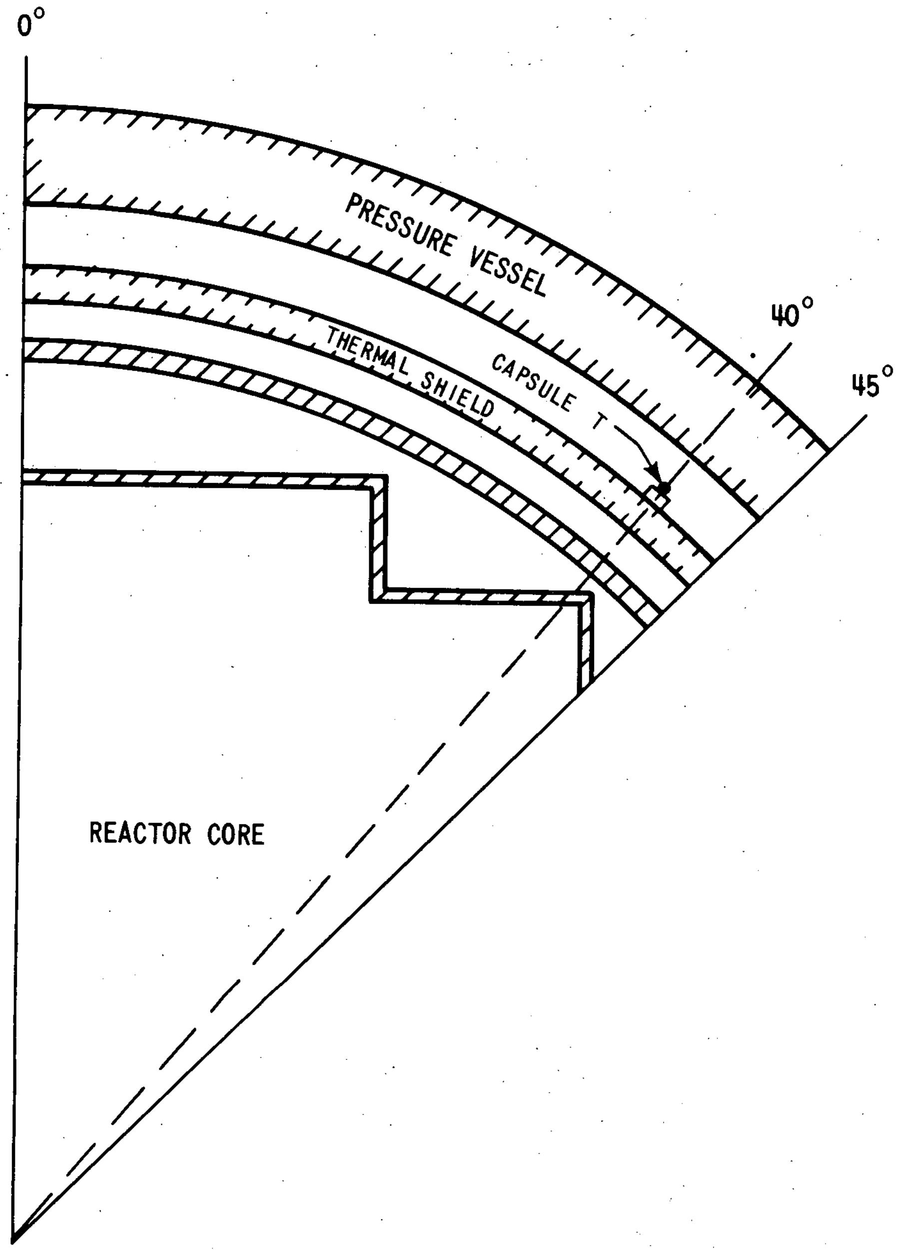


Figure 6-1. Indian Point Unit No. 3 Reactor Geometry

- 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 Indian Point Unit No. 3 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 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 analytical geometries described a 45-degree 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 using the equivalent cylindrical core concept.

The specific activity of each of the activation monitors was determined using established ASTM procedures.[2,3,4,5,6]

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1. Soltesz, R.G., et al., "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation Volume 5 – Two-Dimensional, Discrete Ordinates Transport Technique, WANL-PR-(LL)-034, August 1970.
2. ASTM Standard E261-70, "Standard Method for Measuring Neutron Flux by Radioactivation Techniques," in ASTM Standards, Part 45, (1975), pp. 745-755, American Society for Testing and Materials, Philadelphia, Pa., 1975.
3. ASTM Standard E262-70, "Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques," in ASTM Standards, Part 45, (1975), pp. 756-763, American Society for Testing and Materials, Philadelphia, Pa., 1975.
4. ASTM Standard E263-70, "Standard Method for Measuring Fast-Neutron Flux by Radioactivation of Iron," in ASTM Standards, Part 45, (1975), pp. 764-769, American Society for Testing and Materials, Philadelphia, Pa., 1975.
5. ASTM Standard E481-73T, "Tentative Method of Measuring Neutron-Flux Density by Radioactivation of Cobalt and Silver," in ASTM Standards, Part 45, (1975), pp. 887-894, American Society for Testing and Materials, Philadelphia, Pa. 1975.
6. ASTM Standard E264-70, "Standard Method of Measuring Fast-Neutron Flux by Radioactivation of Nickel, in ASTM Standards, Part 45, pp. 685-689, American Society for Testing and Materials, Philadelphia, Pa., 1975.

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

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

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 using 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

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

where

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

Thus, equation (6-1) was rewritten

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

or, solving for the neutron flux

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

The total fluence above 1 Mev was then given by

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

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 T was obtained from the bare and cadmium-covered Co^{59} (n,γ) Co^{60} data by means of cadmium ratios and the use of a 37-barn 2200 m/sec cross section. Thus

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

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

The spectrum-averaged reaction cross sections derived for each of the fast neutron flux monitors are listed in table 6-2. The irradiation history of the flux monitors removed from Capsule T is listed in table 6-3. The data were obtained from the Indian Point Unit No. 3 operating plant status report NUREG-0020.

TABLE 6-2
SPECTRUM-AVERAGED REACTION CROSS SECTIONS USED IN FAST NEUTRON FLUX DERIVATION

Reaction	$\bar{\sigma}$ (Barns)
Fe^{54} (n,p) Mn^{54}	0.0670
Ni^{58} (n,p) Co^{58}	0.0900
Cu^{63} (n,α) Co^{60}	0.000490

6-3. RESULTS OF ANALYSIS

The fast neutron (E >1.0 Mev) flux and fluence levels derived from the monitors taken from Capsule T are presented in table 6-4. In examining the data listed in table 6-4, it should be noted that the Fe^{54} monitors were positioned within the surveillance capsule at a radius of 211.10 cm and 212.10 cm relative to the core centerline. The corresponding radius of the Ni^{58} and Cu^{63} monitors was 211.10 cm. Thus, it should be expected that the measured neutron flux levels reflect the flux gradient caused by attenuation within the test specimens.

The thermal neutron flux obtained from the cobalt-aluminum monitors is summarized in table 6-5. Due to the relatively low thermal neutron flux at the monitor locations, no burnup correlation was made to any of the measured activities. The maximum error introduced by this assumption is estimated to be less than 1 percent for all fast reactions.

Results of the Sn transport calculations for the Indian Point Unit No. 3 reactor are summarized in figures 6-2 through 6-4, and in table 6-6. 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. The relative axial variation of neutron flux is shown in figure 6-4. Absolute axial variations of fast neutron 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 predicted maximum end-of-life fast neutron exposure of the Indian Point Unit No. 3 reactor vessel is given as a function of radial position within the vessel wall. The

TABLE 6-3

IRRADIATION HISTORY OF CAPSULE T REMOVED FROM INDIAN POINT UNIT NO. 3

Month	P_J (MWt)	P_{max} (MWt)	P_J/P_{max}	Irradiation Time τ_J (days)	Decay Time[a] τ_d (days)
4/76-8/76	31	2760	0.011	146	798
9/76	841	2760	0.305	30	768
10/76	2168	2760	0.786	31	737
11/76	2583	2760	0.936	30	707
12/76	2268	2760	0.822	31	676
1/77	1557	2760	0.564	31	645
2/77	2687	2760	0.974	28	617
3/77	2640	2760	0.957	31	586
4/77	2054	2760	0.744	30	556
5/77	2713	2760	0.983	31	525
6/77	2688	2760	0.974	30	495
7/77	2609	2760	0.945	31	464
8/77	2684	2760	0.972	31	433
9/77	2635	2760	0.955	30	403
10/77	500	2760	0.181	31	372
11/77	0	2760	0.000	30	342
12/77	1120	2760	0.406	31	311
1/78	2623	2760	0.950	31	280
2/78	1729	2760	0.626	28	252
3/78	2628	2760	0.952	31	221
4/78	2245	2760	0.813	30	191
5/78	2534	2760	0.918	31	160
6/78	2619	2760	0.949	7	153

a. Decay time is referenced to the counting date of the neutron flux monitors (11/7/78).

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

Reaction and Monitor Location	Measured Activity[a] (dps/gm)	$\overline{\phi}$ (E > 1.0 Mev)[b] (n/cm^2/sec)	ϕ (E > 1.0 Mev)[b] (n/cm^2)
Fe^{54} (n,p) Mn^{54}			
W40 (front)	1.37×10^6	7.69×10^{10}	3.24×10^{18}
A32 (rear)	1.17×10^6	6.57×10^{10}	2.76×10^{18}
AT58 (rear)	1.09×10^6	6.12×10^{10}	2.58×10^{18}
W37 (front)	1.25×10^6	7.01×10^{10}	2.95×10^{18}
B9 (front)	1.34×10^6	7.52×10^{10}	3.16×10^{18}
AT54 (rear)	1.18×10^6	6.62×10^{10}	2.79×10^{18}
Ni^{58} (n,p) Co^{58}			
Middle 24	8.11×10^6	7.13×10^{10}	3.00×10^{18}
Cu^{63} (n,α) Co^{60}			
Middle 23	4.87×10^4	1.01×10^{11}	4.25×10^{18}
Bottom 23	4.59×10^4	9.55×10^{10}	4.02×10^{18}

a. Monitor activities are referenced to 00:00 on 11/7/78.

b. Derived flux and fluence are subject to plus or minus 10 percent measurement uncertainty.

calculated fast neutron flux levels interior to Capsule T along with the lead factors (LF) relating capsule exposure to vessel exposure are listed in table 6-6. 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. Lead factors shown in table 6-6 are higher than those previously reported.[1] These differences are discussed in paragraph 6-4.

TABLE 6-5
RESULTS OF THERMAL NEUTRON DOSIMETRY FOR CAPSULE T

Monitor Location	Bare[a] Activity (dps/gm)	Cadmium Covered[a] Activity (dps/gm)	ϕth[b] (n/cm^2 - sec)
Bottom	7.68×10^6	3.02×10^6	5.57×10^{10}
Middle	6.84×10^6	3.03×10^6	4.55×10^{10}
Top	7.63×10^6	3.08×10^6	5.44×10^{10}

a. Co^{60} activities are referenced to 11/7/78.
b. Derived flux levels are subjected to plus or minus 10 percent measurement error.

TABLE 6-6
CALCULATED FAST NEUTRON FLUX AND LEAD FACTORS FOR CAPSULE T

Location Within Capsule T	ϕ (E > 1 Mev) (n/cm^2 - sec)	Lead Factor
Front Monitors	6.99×10^{10}	3.89
Rear Monitors	5.66×10^{10}	3.15

1. Yanichko, S. E., Davidson, J. A., "Consolidated Edison Co. of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-8475, January, 1975.

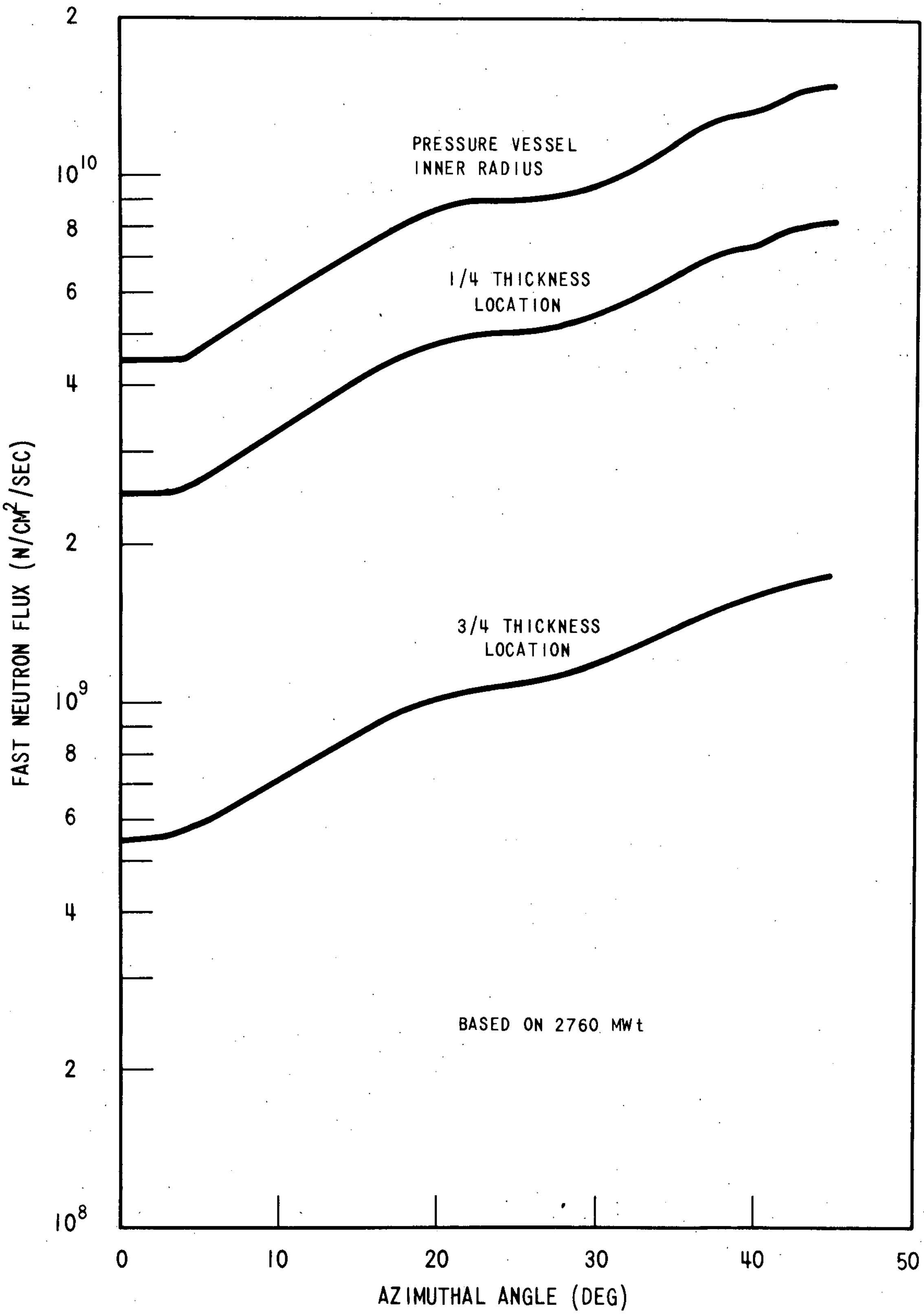

Figure 6-2. Calculated Azimuthal Distribution of Maximum Fast Neutron Flux (E > 1.0 Mev) Within Indian Point Unit No. 3 Reactor Vessel

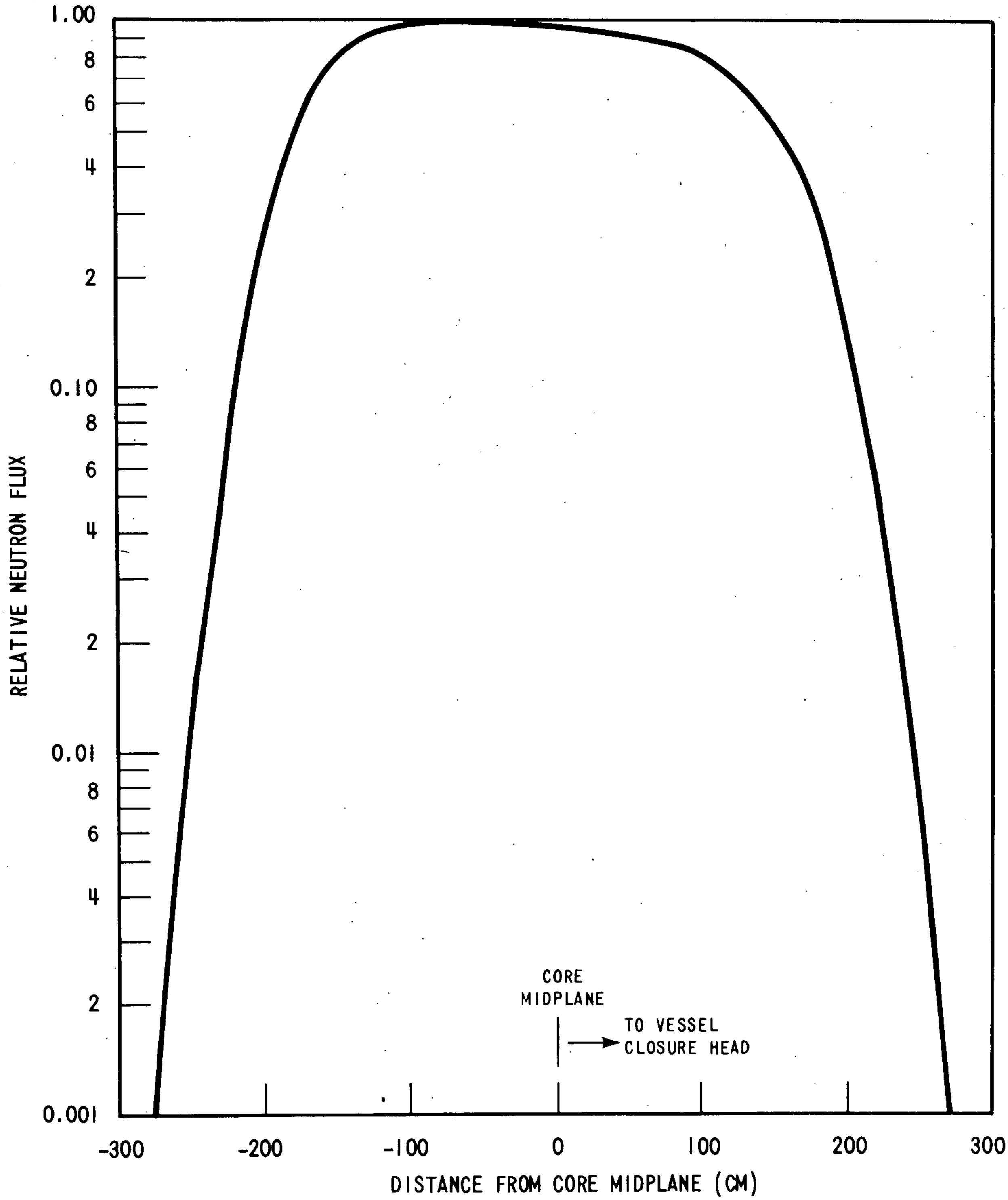

Figure 6-3. Relative Axial Variation of Fast Neutron Flux (E >1.0 Mev) Incident on Indian Point Unit No. 3 Reactor Vessel

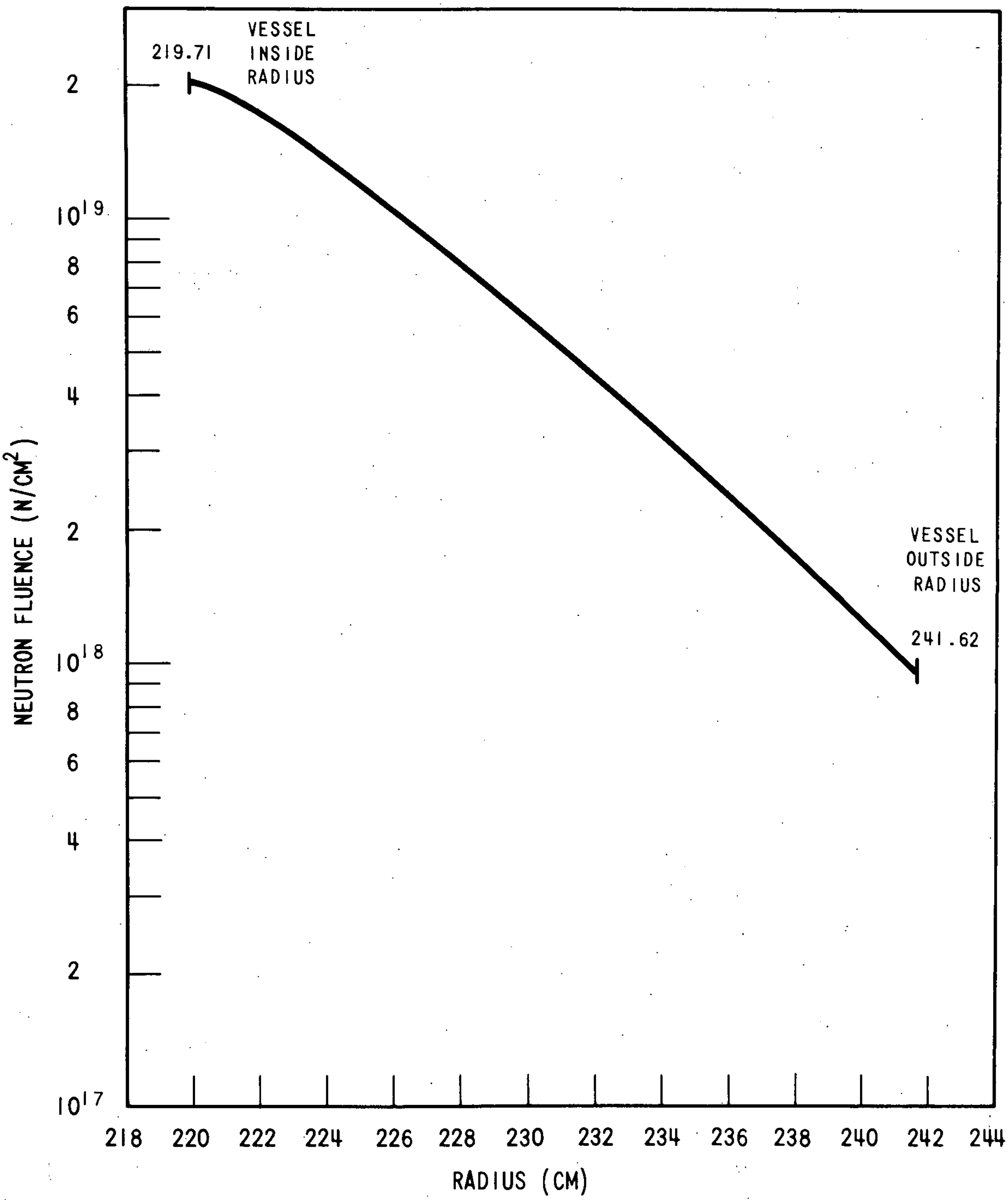

Figure 6-4. Calculated Maximum End of Life Fast Neutron Fluence ($E > 1.0$ Mev) as a Function of Radius Within Indian Point Unit No. 3 Reactor Vessel

Table 6-7 presents comparisons of calculated and measured fast neutron flux levels at the front monitor, dosimeter block, and rear monitor locations within Capsule T.

TABLE 6-7
COMPARISON OF MEASURED AND CALCULATED FAST NEUTRON FLUX LEVELS WITHIN CAPSULE T

	ϕ (E > 1.0 Mev) (n/cm^2 - sec)	
	Front Monitor Location	Rear Monitor Location
Calculated	6.99×10^{10}	5.66×10^{10}
Fe^{54} (n,p) Mn^{54}	7.41×10^{10}	6.44×10^{10}
Ni^{58} (n,p) Co^{58}	7.13×10^{10}	—
Cu^{63} (n,α) Co^{60}	9.83×10^{10}	—

6-4. DISCUSSION OF RESULTS

From the iron data presented in table 6-4, the average fast neutron fluence at the front flux monitor location is determined to be 3.12×10^{18} n/cm^2; the average fast neutron fluence at the rear flux monitor location is 2.71×10^{18} n/cm^2. These measured values correspond to analytical values of 2.94×10^{18} and 2.38×10^{18} at the front and rear locations, respectively, and result in an average fluence for the capsule of 2.92×10^{18} n/cm^2. Agreement between calculation and measurement is excellent.

By employing the lead factors listed in table 6-6, a comparison of the end-of-life peak fast neutron exposure of the Indian Point Unit No. 3 reactor vessel, as derived from both calculations and measured surveillance capsule results, may be made as follows:

$$\Phi_{\text{calculated}} = 1.8 \times 10^{19} \text{ n/cm}^2$$

$$\Phi_{\text{front monitors}} = 1.9 \times 10^{19} \text{ n/cm}^2$$

$$\Phi_{\text{rear monitors}} = 2.1 \times 10^{19} \text{ n/cm}^2$$

These data are based on 32 full power years of operation at 2760 MWt.

Following removal of Capsule T, the Indian Point Unit No. 3 reactor was granted a power upgrade to 3025 MWt. Operation of the reactor at the upgraded level for the remaining 30.66 effective full power years of plant life will result in projected end of life fluence values which are higher than those derived from Capsule T dosimetry. Reactor vessel end of life fast neutron fluence values based on the upgraded mode of operation are as follows:

$$\Phi_{surface} = 2.0 \times 10^{19}$$

$$\Phi_{1/4 T} = 1.1 \times 10^{19}$$

$$\Phi_{3/4 T} = 2.5 \times 10^{18}$$

These data are based on calculated neutron flux levels within the reactor vessel and on plant operation at 2760 MWt for 1.34 full power years followed by operation at 3025 MWt for 30.66 full power years.

As mentioned briefly in paragraph 6-3, the lead factors calculated based on Capsule T are greater than previously reported (2.9 lead factor for Capsules T, Y, Z, and S and 1.0 for Capsules V, W, U, and X). Since the time the original calculations were made, the analytical technique used in determining lead factors has been improved by including a refined description of the reactor core power distribution as well as a more detailed geometric representation of the surveillance capsules and holders. The application of this updated analytical approach has indicated that significant perturbations in both the neutron flux magnitude and energy spectra occur in the vicinity of the surveillance capsules. As a result of these local effects, the lead factors for the Indian Point Unit No. 3 plant must be updated as follows:

Capsule	Lead Factor[1]
T, Y, Z, S	3.7
V, W, U, X	1.1

Although these new calculated lead factors are greater than that recommended by 10 CFR Part 50, Appendix H (lead factors should be between one and three), they are still relatively close and should pose no problem with regard to information obtained through the Indian Point Unit No. 3 Radiation Surveillance Program. The recent revision of ASTM Standard E185 (1979 version) reflects this position, as lead factors are recommended to be in the range of one to three.

1. Calculated at the capsule center midway between the front and rear faces of the capsule.

6-5. SURVEILLANCE CAPSULE REMOVAL SCHEDULE

To date, Capsule T has been removed and the encapsulated specimens have been tested. Based on removal and testing of this capsule, two things have come to light. First, the damage rate of the plate and weld metal due to irradiation is in excess of that predicted by the Westinghouse trend curves. Second, the calculation of the capsule lead factors resulted in lead factors slightly higher than originally estimated.

The increased rate of embrittlement of the plate (ΔRT_{NDT} of 125°F versus 117°F) and weld metal (ΔRT_{NDT} of 175°F versus 134°F) has been considered in development of updated operation heatup and cooldown limit curves, and is discussed in detail in appendix A. It is also important that the higher calculated capsule lead factors be considered in determining an updated surveillance capsule removal schedule.

As mentioned previously, ASTM Standard E185-73 on surveillance tests for nuclear reactor vessels has recently (1979) been revised. Revisions include the removal schedule criteria for surveillance capsules contained in reactor vessels. Based on this revision, the following removal schedule is recommended for future capsules removed from the Indian Point Unit No. 3 reactor vessel.

Capsule	Lead Factor	Removal Time[a]	Estimated Fluence, n/cm^2 (E > 1 Mev) x 10^{19}
T	3.7	Removed (1.34)	
Y	3.7	3	0.6
Z	3.7	5	1.1[b]
S	3.7	9	2.1[c]
W	1.1	32	2.3
V	1.1	Standby	––
U	1.1	Standby	––
X	1.1	Standby	––

a. Effective full power years from plant startup
b. Approximates vessel end of life 1/4 thickness wall location fluence
c. Approximates vessel end of life surface wall location fluence

APPENDIX A
HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

A-1. INTRODUCTION

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

RT_{NDT} increases as the material is exposed to fast neutron radiation. Thus, to find the most limiting RT_{NDT} at any time period in the reactor's life, a ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper) present in reactor vessel steels. Design curves which show the effect of fluence and copper content on ΔRT_{NDT} for reactor vessel steels exposed to 550°F are shown in figure A-1.

Given the copper content of the most limiting material, the radiation-induced ΔRT_{NDT} can be estimated from figure A-1. Fast neutron fluence (E > 1 Mev) at the 1/4T (wall thickness) and 3/4T (wall thickness) vessel locations are given as a function of full-power service life in figure A-2. The data for all other ferritic materials in the reactor coolant pressure boundary are examined to assure that no other component will be limiting with respect to RT_{NDT}.

A-2. FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code,[1] and the calculation methods of reference [2]. The preirradiation fracture toughness properties of the

1. ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, Summer 1972 Addenda, Non-Mandatory Appendix G – "Protection Against Non-ductile Failure."
2. Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-70, 1975 Book of ASTM Standards, Part 45, pp. 756-763.

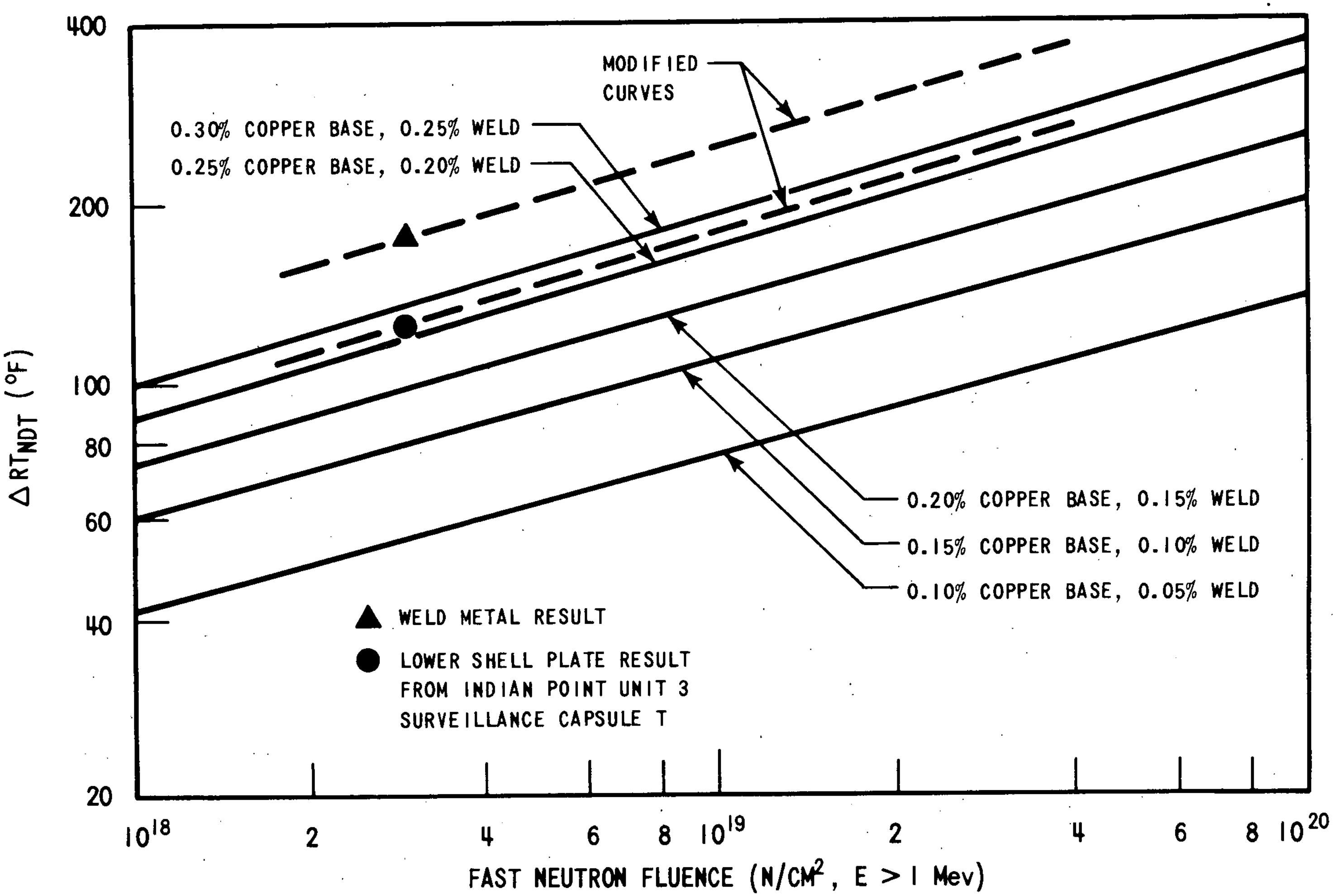

Figure A-1. Effect of Fluence and Copper on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to Irradiation at 550°F

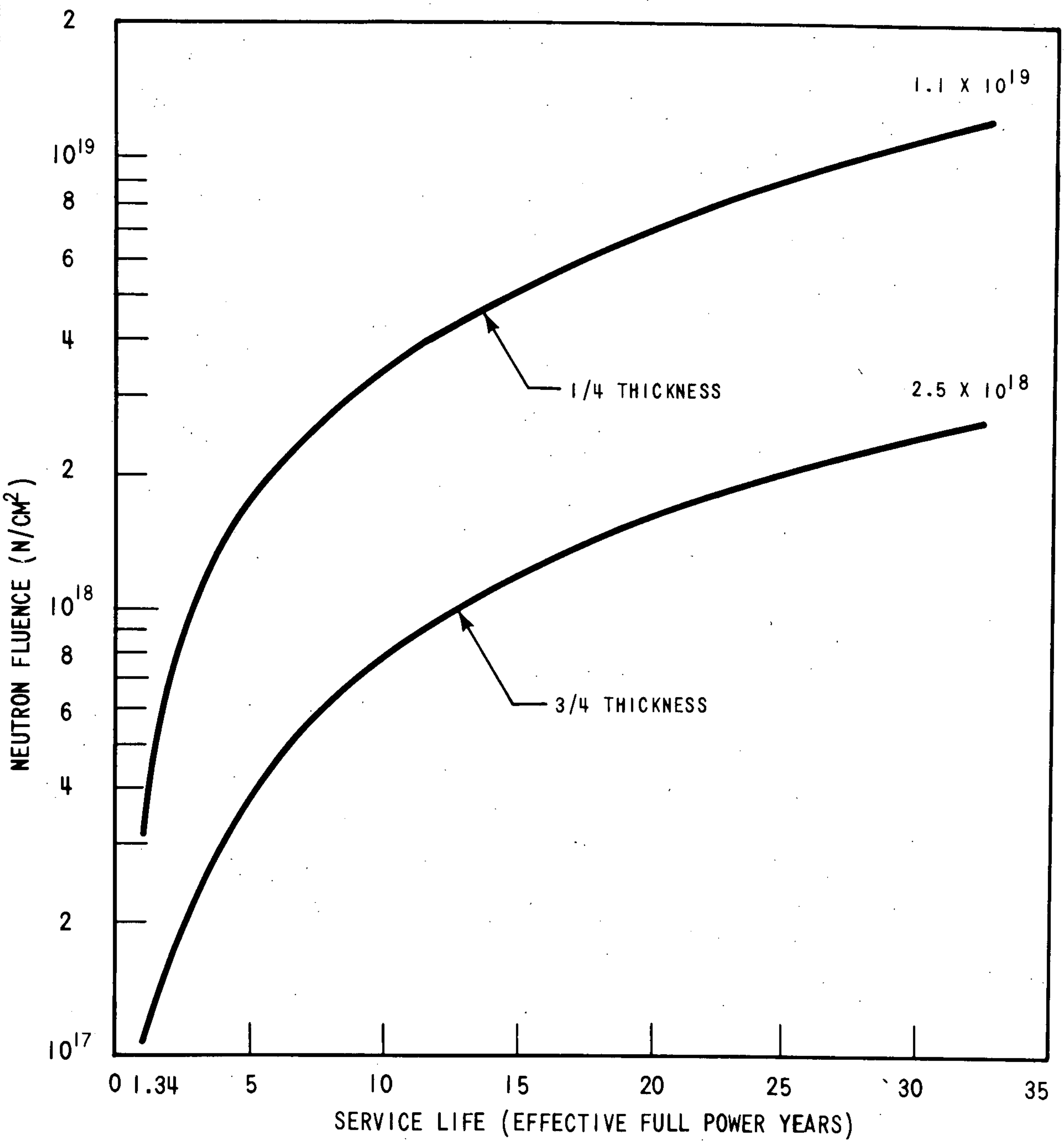

Figure A-2. Fast Neutron Fluence (E > 1 Mev) as a Function of Full-Power Service Life

Indian Point Unit No. 3 reactor vessel materials are presented in table A-1. The postirradiation fracture toughness properties of the reactor vessel belt line material were obtained directly from the Indian Point Unit No. 3 Vessel Material Surveillance Program.

A-3. CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code.[1] The K_{IR} curve is given by the equation:

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

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is:

$$C\, K_{IM} + K_{It} \leqslant K_{IR} \qquad \text{(A-2)}$$

where

K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients.

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for normal and upset conditions per Appendix G of the ASME Code.[1]

C = 1.5 for test conditions during which the reactor core is not critical.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients

1. Appendix G to the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NA, "Protection Against Non-Ductile Failure," American Society of Mechanical Engineers, New York, N.Y., 1977 Edition and Summer 1978 Addenda.

TABLE A-1
INDIAN POINT UNIT NO. 3 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Heat No.	Material Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-lb/35 mil Temp (°F)		RT_{NDT} (°F)	Minimum Upper Shelf Energy (ft-lb)	
						Parallel to Major Working Direction	Normal to Major Working Direction		Parallel to Major Working Direction	Normal to Major Working Direction
Closure Head Dome	C1533-5	SA 302B, MOD	0.13	.011	10	93	113[a]	53	85	55[a]
Closure Head Segment	C1533-3	SA 302B, MOD	0.14	.011	10	51	71[a]	11	128	83[a]
Closure Head Segment	C1533-1	SA 302B, MOD	0.14	.010	0	83	103[a]	43	108	70[a]
Closure Head Segment	C1533-4	SA 302B, MOD	0.13	.012	10	78	98[a]	38	117	76[a]
Closure Head Flange	122P240VA1	A508, CL 2	—	.010	3[a]	-22	-2[a]	3	117	76[a]
Vessel Shell Flange	4P1124/3P1102	A508, CL 2	—	.008	38[a]	-5	15[a]	38	141	92[a]
Inlet Nozzle	123P435VA1	A508, CL 2	—	.010	20[a]	-28	-8[a]	20	154	100[a]
Inlet Nozzle	123P435VA2	A508, CL 2	—	.010	45[a]	-8	12[a]	45	120	78[a]
Inlet Nozzle	123P421VA1	A508, CL 2	—	.010	40[a]	-7	13[a]	40	158	103[a]
Inlet Nozzle	123P421VA2	A508, CL 2	—	.010	12[a]	-14	6[a]	12	155	101[a]
Outlet Nozzle	ZT 2550-1	A508, CL 2	—	.011	60[a]	60	80[a]	60	72	47[a]
Outlet Nozzle	ZT 2585-1	A508, CL 2	—	.011	60[a]	4	24[a]	60	105	69[a]
Outlet Nozzle	ZT 2585-1	A508, CL 2	—	.013	60[a]	4	24[a]	60	96	62[a]
Outlet Nozzle	ZT 2600	A508, CL 2	—	.017	60[a]	-2	18[a]	60	123	80[a]
Upper Shell	B5391-1	SA 302B, MOD	0.21	.010	-50	74	94[a]	34	90	58[a]
Upper Shell	B5394-1	SA 302B, MOD	0.20	.011	-40	84	104[a]	44	100	65[a]
Upper Shell	A0516-1	SA 302B, MOD	0.22	.015	-40	43	63[a]	3	127	82[a]
Intermediate Shell	B5394-2	SA 302B, MOD	0.20	.010	-50	24	65[b]	5[b]	134	97[b]
Intermediate Shell	A0516-2	SA 302B, MOD	0.22	.015	-50	23	56[b]	-4[b]	113	86[b]
Intermediate Shell	B5391-2	SA 302B, MOD	0.20	.011	-40	42	77[b]	17[b]	113	85[b]
Lower Shell	A0495-2	SA 302B, MOD	0.19	.012	0	74	109[b]	49[b]	90	64.5[b]
Lower Shell	C1397-3	SA 302B, MOD	0.22	.011	-20	6	55[b]	-5[b]	134	89[b]
Lower Shell	A0512-2	SA 302B, MOD	0.24	.012	-10	92	134[b]	74[b]	69	62[b]
Bottom Head Segment	B5549-1	SA 302B, MOD	0.13	.013	-40	23	43[a]	-17	103	67[a]
Bottom Head Segment	B5549-2	SA 302B, MOD	0.16	.018	-40	33	53[a]	-7	108	70[a]
Bottom Head Segment	B5549-3	SA 302B, MOD	0.13	.015	-40	38	58[a]	-2	106	69[a]
Bottom Head Segment	B5549-4	SA 302B, MOD	0.13	.016	-30	60	80[a]	20	80	52[a]
Weld	—	Weld	0.15	.019	0[a]	—	5[b]	0	—	112[b]
Heat-Affected Zone	—	HAZ	—	—	—	—	10[b]	—	—	111[b]

a. Estimated using methods identified in NRC Standard Review Plan Section 5.3.2 pressure temperature limits

b. From data obtained through the Indian Point Unit No. 3 Radiation Surveillance Program through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From equation (A-2) the pressure stress intensity factors are obtained, and from these the allowable pressures are calculated.

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

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

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

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

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

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heaup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

A-4. HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated using the methods discussed in paragraph A-3. The derivation of the limit curves is presented in the NRC Regulatory Standard Review Plan.[1]

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program.

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1. NRC Regulatory Standard Review Plan, Directorate of Licensing, Section 5.3.2, "Pressure-Temperature Limits," 1974.

Charpy test specimens from Capsule T indicate that the core region base and weld metal exhibit the actual shifts listed in table A-2. Capsule T experienced a dose (2.92×10^{18} n/cm^2, E > 1 Mev) equivalent to 8.67 effective full power years at the vessel 1/4T location. Table A-2 shows that the shift in RT_{NDT} for both the weld and base metal exceeds the shift predicted by the trend curves in figure A-1. Modified trend curves were developed by drawing lines through the surveillance data points which are parallel to the present trend curves. Figure A-1 shows the modified trend curves as dashed lines, and these curves yield the modified adjusted RT_{NDT}'s given in tables A-3 and A-4.

Tables A-3 and A-4 show that the highest adjusted RT_{NDT} occurs in the lower shell plate material for 9.26 EFPY through 29.1 EFPY. Therefore, the heatup and cooldown curves are based on the lower shell plate material. Test results from the next capsule scheduled for removal will provide additional information necessary to better quantify the materials' response to irradiation.

The periods 9.26 and 29.1 EFPY at the new power rating of 3025 MWt would have been equivalent to 10 and 32 EFPY had there been no power rating increase.

Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in figures A-3 through A-6 and represent operational time periods of 9.26 and 29.1 effective full-power years.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown on the heatup and cooldown curves. The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit lines shown in figures A-3 and A-5. In addition, there are other criteria which must be met before the reactor is made critical.

The leak test limit curves shown in figures A-3 and A-5 represent minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curves were determined by methods of Reference [1] as well as Reference [2].

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1. ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components" Section III, Summer 1972 Addenda, Non-Mandatory Appendix G — "Protection Against Non-ductile Failure."
2. NRC Regulatory Standard Review Plan, Directorate of Licensing, Section 5.3.2, "Pressure-Temperature Limits," 1974.

TABLE A-2

SURVEILLANCE CAPSULE DATA AT 8.67 EFFECTIVE FULL POWER YEARS

Material	Copper Content (%)	Initial RT_{NDT} (°F)	Predicted Shift[a] for 2.92×10^{18} n/cm^2 (°F)	Actual Shift for 2.92×10^{18} n/cm^2 (°F)	Actual Adjusted RT_{NDT} (°F)
Lower Shell Plate B2803-3	0.24	74	117	125	199
Weld	uncontrolled	0	134	175	175

a. Based on trend curves in figure A-1

TABLE A-3

MODIFIED SURVEILLANCE CAPSULE DATA FOR 9.26 EFFECTIVE FULL POWER YEARS

Material	Copper Content (%)	Initial RT_{NDT} (°F)	Modified Actual Shift[a] for 3.12×10^{18} n/cm^2 (°F)	Modified Adjusted RT_{NDT} (°F)
Lower Shell Plate B2803-3	0.24	74	127	201
Weld	uncontrolled	0	177	177

a. Based on modified trend curves in figure A-1; fluence at 1/4 T location

TABLE A-4

MODIFIED SURVEILLANCE CAPSULE DATA FOR 29.1 EFFECTIVE FULL POWER YEARS

Material	Copper Content (%)	Initial RT_{NDT} (°F)	Modified Actual Shift[a] for 1 x 10^{19} n/cm^2 (°F)	Modified Adjusted RT_{NDT} (°F)
Lower Shell Plate B2803-3	0.24	74	176	250
Weld	uncontrolled	0	250	250

a. Based on modified trend curves in figure A-1; fluence at 1/4 T location

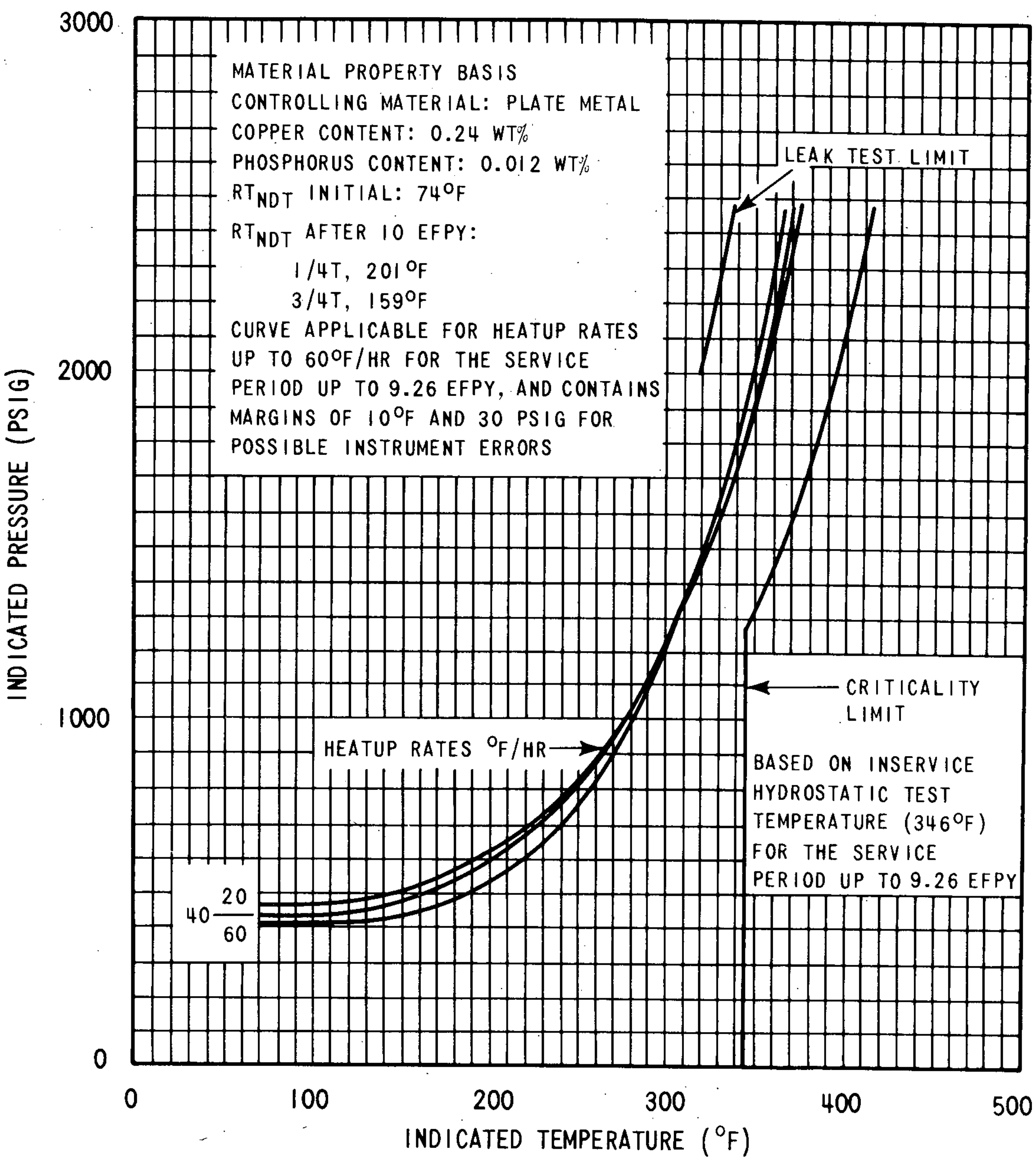

Figure A-3. Indian Point Unit No. 3 (INT) Reactor Coolant System Heatup Limitations Applicable for the First 9.26 EFPY (Curves Based on Test Results of Surveillance Capsule T)

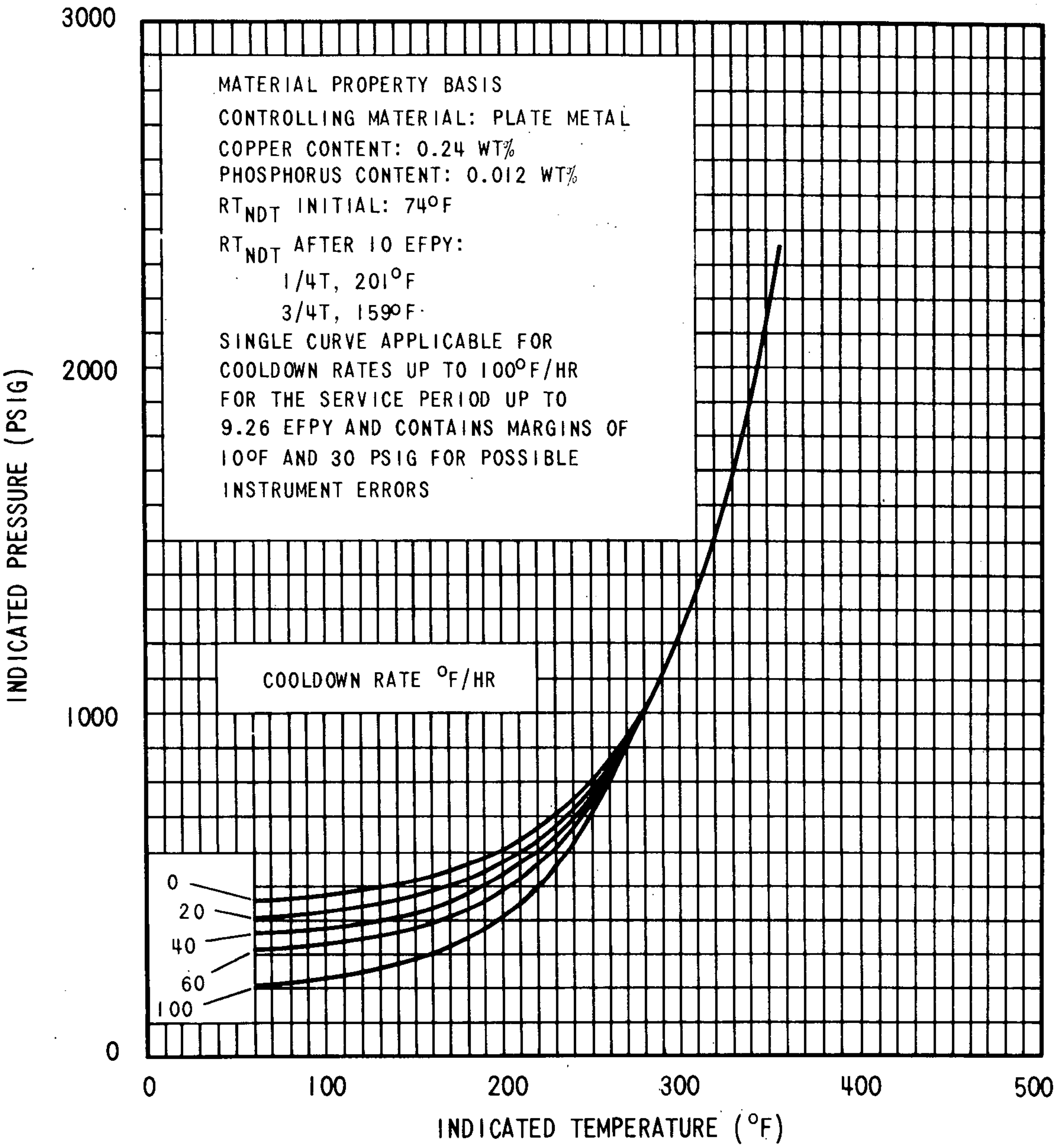

Figure A-4. Indian Point Unit No. 3 (INT) Reactor Coolant System Cooldown Limitations Applicable for the First 9.26 EFPY (Based on Test Results of Surveillance Capsuel T)

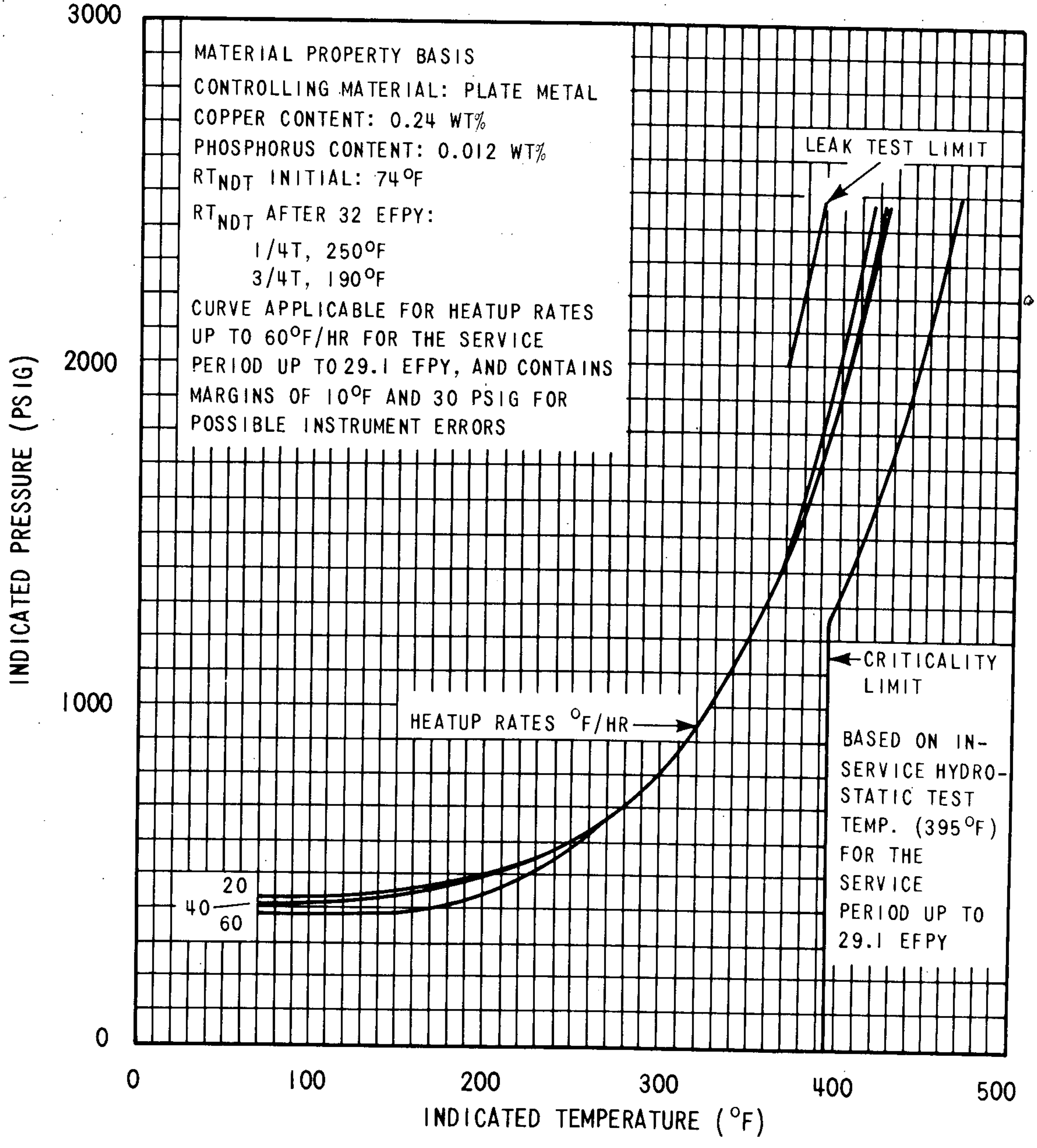

Figure A-5. Indian Point Unit No. 3 (INT) Reactor Coolant System Heatup Limitations Applicable for the First 29.1 EFPY (Curves Based on Test Results of Surveillance Capsule T)

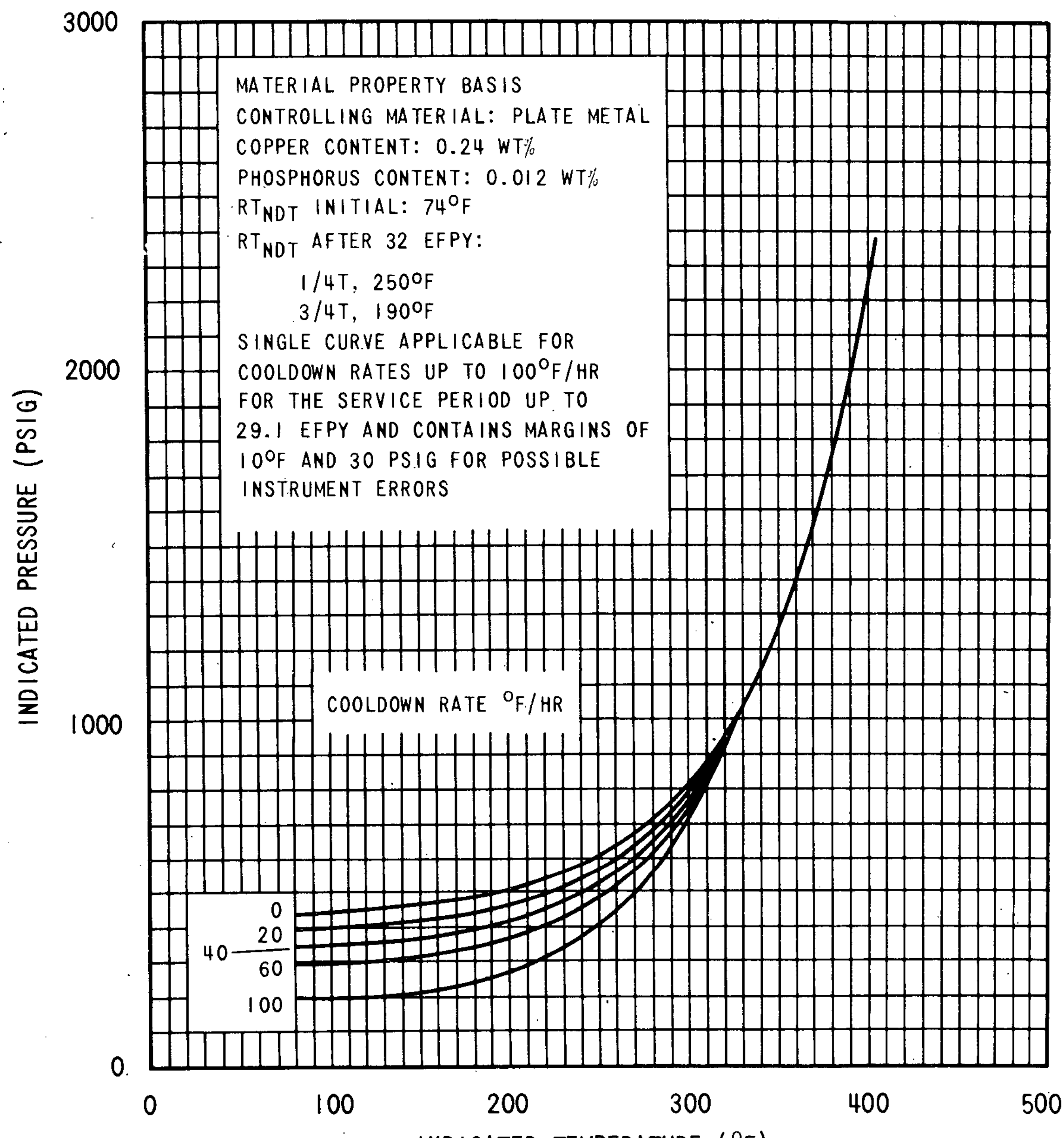

Figure A-6. Indian Point Unit No. 3 (INT) Reactor Coolant System Cooldown Limitations Applicable for the First 29.1 EFPY (Curves Based on Test Results of Surveillance Capsule T)

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