Attachment 3

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WESTINGHOUSE CLASS 3

ANALYSIS OF CAPSULE U FROM THE AMERICAN ELECTRIC POWER COMPANY D. C. COOK UNIT 1 REACTOR VESSEL RADIATION SURVEILLANCE PROGRAM

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PREFACE

This report has been technically reviewed and verified.

	Reviewer	
Sections 1 through 5 and 7, 8	N. K. Ray	
Section 6	E. P. Lippincott	

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SECTION 1
SUMMARY OF RESULTS

The analysis of the reactor vessel material contained in Capsule U, the fourth surveillance capsule to be removed from the American Electric Power Company D. C. Cook Unit 1 reactor pressure vessel, resulted in the following conclusions:

- Capsule U was pulled at 9.17 EFPY and had a lead factor of 3.43. With a lead factor of 3.43 at 9.17 EFPY capsule U received an average fast neutron fluence (E > 1.0 MeV) of 1.88×10^{19} n/cm^2 at the geometric center of the capsule.

- Irradiation of the reactor vessel intermediate shell Plate B4406-3, to 1.88×10^{19} n/cm^2, resulted in 30 and 50 ft-lb transition temperature increases of 115 and 120°F, respectively, for specimens oriented normal to the major working direction (transverse orientation) and temperature increases of 115 and 125°F, respectively, for specimens oriented parallel to the major working direction (longitudinal orientation).

- Weld metal irradiated to 1.88×10^{19} n/cm^2 resulted in a 205 and 245°F increase in the 30 and 50 ft-lb transition temperatures, respectively. This results in a 30 ft-lb transition temperature of 115°F and a 50 ft-lb transition temperature of 175°F.

- Irradiation to 1.88×10^{19} n/cm^2 resulted in an average upper shelf energy decrease of 1 ft-lb for plate B4406-3 (transverse orientation) and an average upper shelf energy decrease of 16 ft-lbs for the weld metal. Both materials exhibit an upper shelf energy level greater than 50 ft-lb for plant operation through 32 EFPY.

- o Comparison of the 30 ft-lb transition temperature increases for the D. C. Cook Unit 1 surveillance material with predicted increases using the methods of NRC Regulatory Guide 1.99, Revision 2, demonstrated that the Plate B4406-3 (longitudinal orientation) material transition temperature increase was 9°F greater than predicted. This increase is bounded by the 2 sigma allowance for shift prediction of 34°F. The weld metal showed a transition temperature increase that was 37°F less than the prediction.

- o Since capsule U received a fluence of 1.88 x 10^{19} n/cm^2, which is greater than the maximum calculate EOL (32 EFPY) fluence at the vessel inner surface of 1.41 x 10^{19} n/cm^2 (table 6-14), and based on the results presented in Tables 5-7 and 5-8 it is recommended that the remaining untested surveillance capsules be held on standby and remain in their current locations. See Section 7 for the current recommended surveillance capsule removal schedule.

SECTION 2
INTRODUCTION

This report presents the results of the examination of Capsule U, the fourth capsule to be removed from the reactor in the continuing surveillance program which monitors the effects of neutron irradiation on the American Electric Power Company D. C. Cook Unit 1 reactor pressure vessel materials under actual operating conditions.

The surveillance program for the American Electric Power Company D. C. Cook Unit 1 reactor pressure vessel materials was designed and recommended by the Westinghouse Electric Corporation. A description of the surveillance program and the preirradiation mechanical properties of the reactor vessel materials are presented by Yanichko and Lege.[1] The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E-185-70, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors"[29]. Westinghouse Energy Systems personnel were contracted to aid in the preparation of procedures for removing the capsule from the reactor and its shipment to the Westinghouse Science and Technology Center 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 U removed from the American Electric Power Company D. C. Cook Unit 1 reactor vessel and discusses the analysis of the data. The data are also compared to capsules T,[2] X,[3] and Y[4] which were previously removed from the reactor.

SECTION 3
BACKGROUND

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as SA533 Grade B Class 1 (base material of the D. C. Cook Unit 1 reactor pressure vessel beltline) are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and a decrease in ductility and toughness under certain conditions of irradiation.

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

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT per ASTM E-208)[30] 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 projected 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 D. C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program,[1] in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is used to index the material to the K_{IR} curve and, in turn, to set operating limits for the nuclear power plant which take into account the projected effects of irradiation on the reactor vessel materials.

SECTION 4
DESCRIPTION OF PROGRAM

Eight surveillance capsules for monitoring the effects of neutron exposure on the D. C. Cook Unit 1 reactor pressure vessel core region material were inserted in the reactor vessel prior to initial plant startup. The capsules were positioned in the reactor vessel between the thermal shield and the vessel wall at locations shown in Figure 4-1. The vertical center of the capsules is opposite the vertical center of the core.

Capsule U (Figure 4-2) was removed after 9.17 effective full power years of plant operation. This capsule contained Charpy V-notch impact, tensile, and 1X-WOL fracture mechanics specimens from the reactor vessel intermediate shell Plate B4406-3, weld metal representative of the beltline region weld seams, Charpy V-notch specimens from weld heat-affected zone (HAZ) material and Charpy V-notch specimens from ASTM correlation material. All heat-affected zone specimens were obtained from the weld heat-affected zone of Plate B4406-3.

The chemistry and heat treatment of the surveillance material are presented in Table 4-1 and Table 4-2, respectively. The chemical analyses reported in table 4-1 were obtained from unirradiated material used in the surveillance program.

All test specimens were machined from the 1/4 thickness location. Test specimens represent material taken at least one plate thickness from the quenched end of the plate. All base metal Charpy V-notch impact and tensile specimens were oriented with the longitudinal axis of the specimen both normal to (transverse orientation) and parallel to (longitudinal orientation) the principal working direction of the plate. 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 specimens normal to the welding direction. The 1X-WOL fracture mechanics test specimens in Capsule U were machined such that the simulated crack in the specimen would propagate parallel to the major working direction and the major surfaces of the shell plate. All specimens were fatigue precracked per ASTM E399-70T.

Capsule U contained dosimeters of pure iron, copper, nickel, and aluminum-cobalt wire (cadmium-shielded and unshielded), and Neptunium (Np^{237}) and Uranium (U^{238}) which measure the integrated flux at specific neutron energy levels.

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

2.5% Ag, 97.5% Pb	Melting Point 579°F (304°C)
1.75% Ag, 0.75% Sn, 97.5% Pb	Melting Point 590°F (310°C)

The arrangement of the various mechanical test specimens, dosimeters and thermal monitors contained in Capsule U are shown in Figure 4-2.

TABLE 4-1

CHEMICAL COMPOSITION OF
THE D. C. COOK UNIT 1 REACTOR VESSEL
SURVEILLANCE MATERIALS

Element	Plate B4406-3 (Wt. %)	Weld Metal (Wt. %)
C	0.24	0.26
S	0.015	0.014
N_2	0.008	0.010
Co	<0.001	<0.001
Cu	0.14	0.27
Si	0.25	0.18
Mo	0.46	0.44
Ni	0.49	0.74
Mn	1.40	1.33
Cr	0.068	0.022
V	<0.001	0.001
P	0.009	0.023
Sn	0.010	0.006
Ti	<0.001	<0.001
AL	0.024	0.006
Zn	<0.001	0.002
As	0.010	0.009
B	<0.003	<0.003
Sb	0.001	0.001

TABLE 4-2

HEAT TREATMENT OF THE D. C. COOK UNIT 1
REACTOR VESSEL SURVEILLANCE MATERIALS

Material	Temperature (°F)	Time (hr)	Coolant
Intermediate Shell	1600	4	Water quenched
Course Plate B4406-3	1225	4	Air cooled
	1150	40	Furnace cooled
Weldment	1150	40	Furnace cooled

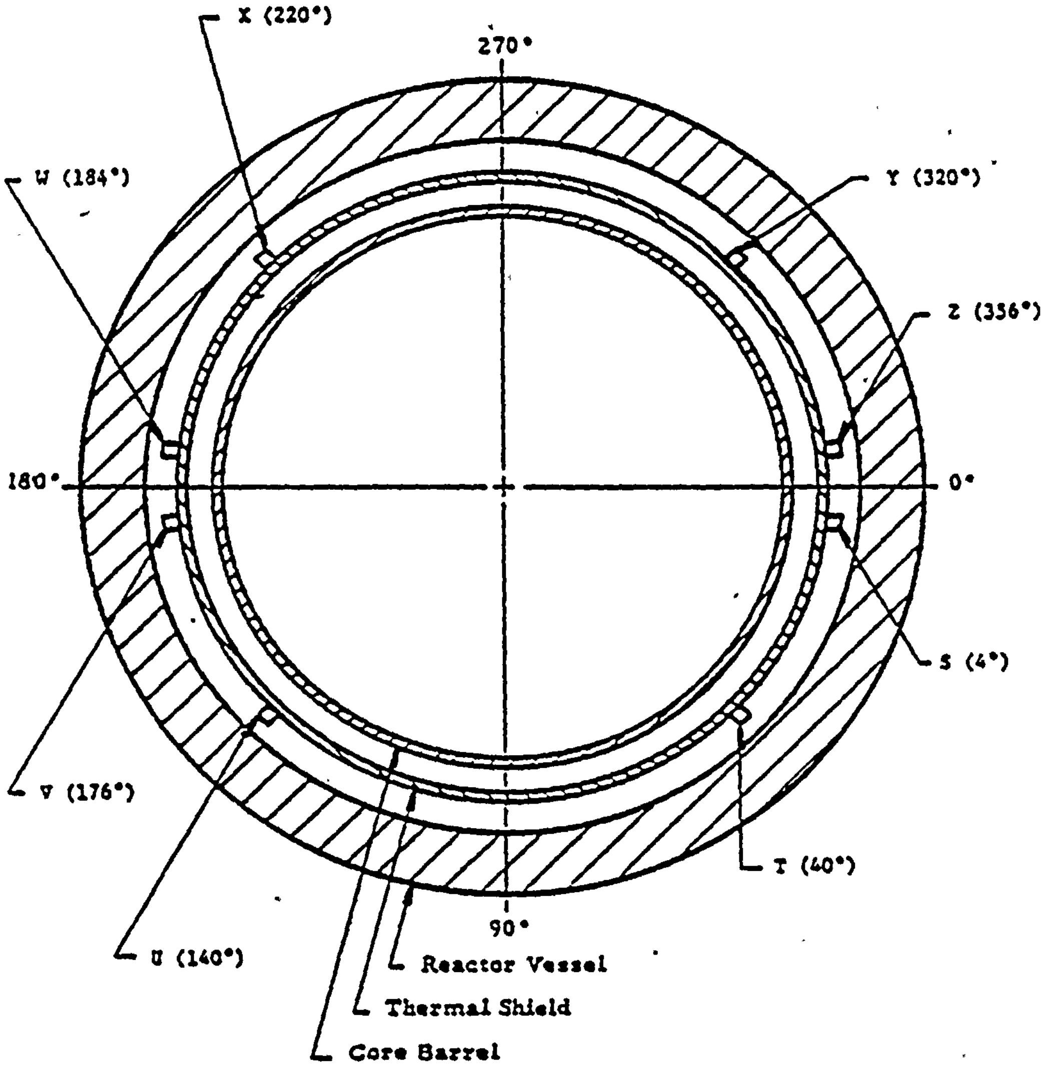
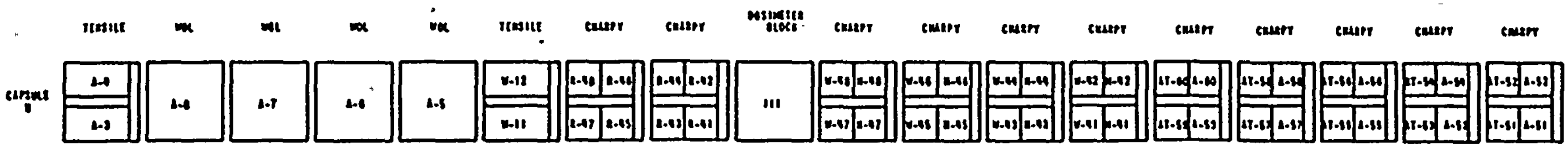


Figure 4-1. Arrangement of Surveillance Capsules in the D. C. Cook Unit 1 Reactor Vessel



SPECIMEN NUMBERING CODE

A - PLATE B4406-3 (LONGITUDINAL DIRECTION)
AT - PLATE B4406-3 (TRANSVERSE DIRECTION)
R - ASTM CORRELATION MONITOR
H - WELD HEAT-AFFECTED ZONE
W - WELD METAL

Figure 4-2. Capsule U Diagram Showing Location [illegible]pecimens, Thermal Monitors, and Dosimeters

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SECTION 5
TESTING OF SPECIMENS FROM CAPSULE U

5-1. OVERVIEW

The postirradiation mechanical testing of the Charpy V-notch and tensile specimens was performed at the Westinghouse Science and Technology Center Laboratory with consultation by Westinghouse Energy Systems personnel. Testing was performed in accordance with 10CFR50, Appendices G and H[5], ASTM Specification E185-82[31] and Westinghouse Procedure MHL 8402, Revision 1 as modified by Westinghouse RMF Procedures 8102, Revision 1 and 8103, Revision 1.

Upon receipt of the capsule at the laboratory, the capsule was visually examinated and photographed for identification purposes. The specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in WCAP-8047.[1] No discrepancies were found.

Examination of the two low-melting 304°C (579°F) and 310°C (590°F) eutectic alloys indicated no melting of either type of thermal monitor. Based on this examination, the maximum temperature to which the test specimens were exposed was less than 304°C (579°F).

The Charpy impact tests were performed per ASTM Specification E23-86[32] and RMF Procedure 8103, Revision 1 on a Tinius-Olsen Model 74, 358J machine. The tup (striker) of the Charpy machine is instrumented with an Effects Technology model 500 instrumentation system. With this system, load-time and energy-time signals can be recorded in addition to the standard measurement of Charpy energy (E_D). From the load-time curve, the load of general yielding (P_{GY}), the time to general yielding (t_{GY}), the maximum load (P_M), and the time to maximum load (t_M) can be determined. Under some test conditions, a sharp drop in load indicative of fast fracture was observed. The load at which fast fracture was initiated is identified as the fast fracture load (P_F), and the load at which fast fracture terminated is identified as the arrest load (P_A).

The energy at maximum load (E_M) was determined by comparing the energy-time record and the load-time record. The energy at maximum load is approximately equivalent to the energy required to initiate a crack in the specimen. Therefore, the propagation energy for the crack (E_p) is the difference between the total energy to fracture (E_D) and the energy at maximum load.

The yield stress (σy) is calculated from the three point bend formula. The flow stress is calculated from the average of the yield and maximum loads, also using the three point bend formula.

Percentage shear was determined from postfracture photographs using the ratio-of-areas methods in compliance with ASTM Specification A370-88[33]. The lateral expansion was measured using a dial gage rig similar to that shown in the same specification.

Tension tests were performed on a 20,000-pound Instron, split-console test machine (Model 1115) per ASTM Specifications E8-83[34] and E21-79[35], and RMF Procedure 8102, Revision 1. All pull rods, grips, and pins were made of Inconel 718 hardened to Rc45. The upper pull rod was connected through a universal joint to improve axiality of loading. The tests were conducted at a constant crosshead speed of 0.05 inch per minute throughout the test.

Deflection measurements were made with a linear variable displacement transducer (LVDT) extensometer. The extensometer knife edges were spring-loaded to the specimen and operated through specimen failure. The extensometer gage length is 1.00 inch. The extensometer is rated as Class B-2 per ASTM E83-67[36].

Elevated test temperatures were obtained with a three-zone electric resistance split-tube furnace with a 9-inch hot zone. All tests were conducted in air.

Because of the difficulty in remotely attaching a thermocouple directly to the specimen, the following procedure was used to monitor specimen temperature. Chromel-alumel thermocouples were inserted in shallow holes in the center and each end of the gage section of a dummy specimen and in each grip. In test configuration, with a slight load on the specimen, a plot of specimen

temperature versus upper and lower grip and controller temperatures was developed over the range room temperature to 550°F (288°C). The upper grip was used to control the furnace temperature. During the actual testing the grip temperatures were used to obtain desired specimen temperatures. Experiments indicated that this method is accurate to plus or minus 2°F.

The yield load, ultimate load, fracture load, total elongation, and uniform elongation were determined directly from the load-extension curve. The yield strength, ultimate strength, and fracture strength were calculated using the original cross-sectional area. The final diameter and final gage length were determined from postfracture photographs. The fracture area used to calculate the fracture stress (true stress at fracture) and percent reduction in area was computed using the final diameter measurement.

5.2. CHARPY V-NOTCH IMPACT TEST RESULTS

The results of Charpy V-notch impact tests performed on the various materials contained in Capsule U irradiated to approximately 1.88 x 10^{19} n/cm^2 at 550°F are presented in Tables 5-1 through 5-6 and Figures 5-1 through 5-5. The transition temperature increases and upper shelf energy decreases for the Capsule U material are shown in Table 5-7.

All materials contained in Capsule U exhibited an average Charpy upper shelf energy greater than 50 ft-lb at a fluence of 1.88 x 10^{19} n/cm^2 which is greater than the projected fluence of 1.41 x 10^{19} n/cm^2 at 32 EFPY.

Irradiation of the vessel intermediate shell Plate B4406-3 material (transverse orientation) specimens to 1.88 x 10^{19} n/cm^2 (Figure 5-1) resulted in 30 and 50 ft-lb transition temperature increases of 115 and 120°F respectively, and an upper shelf energy decrease of 1 ft-lb when compared to the unirradiated data.[1]

Irradiation of the vessel intermediate shell Plate B4406-3 material (longitudinal orientation) specimens to 1.88 x 10^{19} n/cm^2 (Figure 5-2) resulted in a 30 and 50 ft-lb transition temperature increase of 115 and 125°F, respectively, and an upper shelf energy decrease of 17 ft-lbs when compared to the unirradiated data.[1]

Weld metal irradiated to 1.88 x 10^{19} n/cm^2 (Figure 5-3) resulted in a 30 and 50 ft-lb transition temperature increase of 205 and 245°F respectively and an upper shelf energy decrease of 16 ft-lb.

Weld HAZ metal irradiated to 1.88 x 10^{19} n/cm^2 (Figure 5-4) resulted in a 30 and 50 ft-lb transition temperature increase of 175 and 190°F respectively and an upper shelf energy decrease of 9 ft-lb.

ASTM correlation material irradiated to 1.88 x 10^{19} n/cm^2 (Figure 5-5) resulted in a 30 and 50 ft-lb transition temperature increase of 120°F and an upper shelf energy decrease of 10 ft-lb.

The fracture appearance of each irradiated Charpy specimen from the various materials is shown in Figures 5-6 through 5-10 and show an increasing ductile or tougher appearance with increasing test temperature.

Table 5-8 shows a comparison of the 30 ft-lb transition temperature (ΔRT_{NDT}) increases for the various D. C. Cook Unit 1 surveillance materials with predicted increases using the methods of NRC Regulatory Guide 1.99, Revision 2.[6] This comparison shows that the transition temperature increase resulting from irradiation to 1.88 x 10^{19} n/cm^2 is 9°F higher than predicted by the Guide for Plate B4406-3 longitudinal specimens and 4°F higher than predicted for transverse specimens. The weld metal transition temperature increase resulting from 1.88 x 10^{19} n/cm^2 is less than the Guide prediction.

5-3. TENSION TEST RESULTS

The results of tension tests performed on Plate B4406-3 (longitudinal orientation) and weld metal irradiated to 1.88 x 10^{19} n/cm^2 are shown in Table 5-9 and Figures 5-11 and 5-12, respectively. These results show that irradiation produced a 12 to 15 Ksi increase in 0.2 percent yield strength for Plate B4406-3 and 18 to 20 Ksi increase for the weld metal. Fractured tension specimens for each of the materials are shown in Figures 5-13 and 5-14. A typical stress-strain curve for the tension specimens is shown in Figure 5-15.

5-4. COMPACT TENSION TESTS

Per the surveillance capsule testing contract with the American Electric Power Company, the 1X-WOL Fracture Mechanics specimens will not be tested and will be stored at the Hot Cell at the Westinghouse Science and Technology Center.

TABLE 5-1

CHARPY V-NOTCH IMPACT DATA FOR THE D. C. COOK UNIT 1
REACTOR VESSEL SHELL PLATE B4406-3
IRRADIATED AT 550°F, FLUENCE 1.88 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample No.	Temperature (°F)	(°C)	Impact Energy (ft-lb)	(J)	Lateral (mils)	Expansion (mm)	Shear (%)
			Longitudinal Orientation				
A60	82	(28)	16.0	(21.5)	13.0	(0.33)	10
A54	100	(38)	37.0	(50.0)	24.0	(0.61)	15
A56	125	(52)	43.0	(58.0)	33.0	(0.84)	25
A52	150	(66)	34.0	(46.0)	30.0	(0.76)	35
A55	175	(79)	63.0	(85.5)	43.0	(1.09)	40
A58	225	(107)	121.0	(164.0)	78.0	(1.98)	100
A51	250	(121)	100.0	(135.5)	77.0	(1.96)	100
A57	300	(149)	124.0	(168.0)	82.0	(2.08)	100
A53	300	(149)	113.0	(153.0)	81.0	(2.06)	100
A59	350	(177)	110.0	(149.0)	73.0	(1.85)	100
			Transverse Orientation				
AT60	50	(10)	19.0	(26.0)	15.0	(0.38)	10
AT57	82	(28)	29.0	(39.5)	20.0	(0.51)	15
AT59	100	(38)	22.0	(30.0)	22.0	(0.56)	20
AT58	125	(52)	32.0	(43.5)	21.0	(0.53)	20
AT54	150	(66)	37.0	(50.0)	32.0	(0.81)	25
AT51	200	(93)	56.0	(76.5)	42.0	(1.07)	75
AT56	240	(116)	82.0	(111.0)	65.0	(1.65)	100
AT52	250	(121)	95.0	(129.0)	71.0	(1.80)	100
AT55	300	(149)	95.0	(129.0)	61.0	(1.55)	100
AT53	350	(177)	103.0	(139.5)	64.0	(1.63)	100

TABLE 5-2

CHARPY V-NOTCH IMPACT DATA FOR THE D. C. COOK UNIT 1
REACTOR VESSEL WELD METAL AND HAZ METAL IRRADIATED AT 550°F
FLUENCE 1.88 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample No.	Temperature (°F)	(°C)	Impact Energy (ft-lb)	(J)	Lateral (mils)	Expansion (mm)	Shear (%)
			Weld Metal				
W47	50	(10)	21.0	(28.5)	14.0	(0.36)	15
W42	74	(23)	53.0	(71.8)	35.0	(0.89)	65
W44	82	(28)	63.0	(85.5)	41.0	(1.04)	60
W45	125	(52)	38.0	(51.5)	31.0	(0.79)	65
W43	150	(66)	48.0	(65.0)	35.0	(0.89)	70
W48	200	(93)	61.0	(82.5)	50.0	(1.27)	100
W46	250	(121)	90.0	(122.0)	63.0	(1.60)	100
W41	350	(177)	95.0	(129.0)	68.0	(1.73)	100
			HAZ Metal				
H45	0	(- 18)	10.0	(13.5)	10.0	(0.25)	20
H47	50	(10)	48.0	(65.0)	36.0	(0.91)	35
H46	82	(28)	75.0	(101.5)	42.0	(1.07)	70
H44	150	(66)	64.0	(87.0)	46.0	(1.17)	80
H48	200	(93)	95.0	(129.0)	72.0	(1.83)	100
H43	250	(121)	121.0	(164.0)	67.0	(1.70)	100
H41	275	(135)	112.0	(152.0)	72.0	(1.83)	100
H42	350	(177)	78.0	(106.0)	63.0	(1.60)	100

TABLE 5-3

CHARPY V-NOTCH IMPACT DATA FOR THE D. C. COOK UNIT 1
ASTM CORRELATION MONITOR MATERIAL IRRADIATED AT 550°F,
FLUENCE 1.88 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample No.	Temperature (°F)	(°C)	Impact Energy (ft-lb)	(J)	Lateral Expansion (mils)	(mm)	Shear (%)
R44	100	(38)	14.0	(19.0)	12.0	(0.30)	10
R42	175	(79)	26.0	(35.5)	20.0	(0.51)	25
R43	200	(93)	31.0	(42.0)	25.0	(0.64)	30
R48	225	(107)	68.0	(92.5)	49.0	(1.24)	90
R45	250	(121)	81.0	(110.0)	59.0	(1.50)	100
R46	275	(135)	89.0	(120.5)	67.0	(1.70)	100
R47	300	(149)	112.0	(152.0)	70.0	(1.78)	100
R41	350	(177)	108.0	(146.5)	74.0	(1.88)	100

TABLE 5-4

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR D. C. COOK UNIT 1

REACTOR VESSEL SHELL PLATE B4406-3

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in²) Charpy Ed/A	Maximum Em/A	Prop Ep/A	Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
						Longitudinal Orientation							
A60	82	16.0	129	102	26	3.10	100	4.20	260	4.15	0.20	102	121
A54	100	37.0	298	189	108	2.75	75	4.25	435	4.15	-	91	116
A56	125	43.0	346	233	113	3.40	125	4.60	530	4.55	0.30	113	132
A52	150	34.0	274	301	-27	4.45	90	6.00	500	6.00	1.40	148	173
A55	175	63.0	507	313	195	2.85	80	4.55	670	4.50	1.80	95	122
A58	225	121.0	974	293	682	2.55	100	4.25	690	-	-	84	112
A51	250	100.0	805	286	519	2.70	125	4.30	695	-	-	90	116
A53	300	113.0	910	382	528	4.05	100	5.75	660	-	-	133	162
A57	300	124.0	998	284	715	2.55	80	4.05	675	-	-	84	109
A59	350	110.0	886	250	636	2.70	110	4.15	605	-	-	89	113
						Transverse Orientation							
AT60	50	19.0	153	101	52	2.65	70	3.90	260	3.90	-	87	108
AT57	82	29.0	234	164	69	3.00	85	4.60	370	4.45	0.25	99	125
AT59	100	22.0	177	207	-30	4.45	135	5.75	395	5.65	0.25	151	171
AT58	125	32.0	258	146	111	2.95	100	4.25	355	4.15	0.95	97	118
AT54	150	37.0	298	174	124	2.40	90	4.05	445	3.90	1.20	80	106
AT51	200	56.0	451	220	230	2.95	85	4.30	505	4.20	1.90	98	120
AT56	240	82.0	660	288	372	3.90	105	5.65	510	-	-	129	159
AT52	250	95.0	765	249	516	2.50	95	4.05	600	-	-	83	109
AT55	300	95.0	765	240	525	2.50	100	3.90	600	-	-	82	106
AT53	350	103.0	829	211	619	2.60	110	4.10	525	-	-	86	110

TABLE 5-5

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR D. C. COOK UNIT 1

REACTOR VESSEL WELD METAL AND HAZ METAL

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in²) Charpy Ed/A	Maximum Em/A	Prop Ep/A	Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
							Weld Metal						
W47	50	21.0	169	160	9	4.35	80	5.80	275	5.75	0.15	144	168
W42	74	53.0	427	COMPUTER MALFUNCTION			-	-	-	-	-	-	-
W44	82	63.0	507	244	263	3.15	95	4.60	520	4.35	1.25	103	128
W45	125	38.0	306	308	-2	4.40	90	5.90	505	5.70	2.50	145	171
W43	150	48.0	387	197	198	2.75	70	4.35	435	4.35	1.35	91	118
W48	200	61.0	491	244	248	4.45	125	4.70	450	-	-	148	169
W46	250	90.0	725	206	518	2.60	85	3.90	510	-	-	86	108
W41	350	95.0	765	200	565	2.55	75	3.75	505	-	-	85	104
							HAZ Metal						
H45	0	10.0	81	52	29	5.20	140	5.50	155	5.50	1.60	172	177
H47	50	48.0	387	384	2	4.86	125	6.60	605	6.25	0.40	161	190
469	82	75.0	604	245	359	3.00	45	5.00	460	4.50	2.00	99	132
H44	150	64.0	515	246	269	3.2	90	4.60	520	4.45	2.25	105	129
H48	200	95.0	765	307	458	2.35	95	6.10	505	-	-	144	174
H43	250	121.0	974	218	756	2.7	75	4.15	505	-	-	89	113
H41	275	112.0	902	350	552	4.3	125	6.00	610	-	-	142	170
H42	350	78.0	628	211	417	2.55	115	3.80	545	-	-	84	105

TABLE 5-6

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR D. C. COOK UNIT 1

ASTM CORRELATION MONITOR MATERIAL

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in²) Charpy Ed/A	Maximum Em/A	Prop Ep/A	Yield Load (kips)	Time to Yield (μsec)	Maximum Load (kips)	Time to Maximum (μsec)	Fracture Load (kips)	Arrest Load (kips)	Yield Stress (ksi)	Flow Stress (ksi)
R44	100	14.0	113	98	15	4.00	115	5.05	220	4.95	0.50	132	149
R42	175	26.0	209	108	101	2.90	125	3.80	310	3.80	3.80	96	110
R43	200	31.0	250	54	54	3.80	100	5.05	380	4.85	4.85	126	147
R48	225	68.0	548	126	126	4.25	130	5.50	775	5.50	5.50	141	162
R45	250	81.0	652	435	435	2.7	95	4.15	520	-	-	89	113
R46	275	89.0	717	436	436	3.80	95	5.45	510	-	-	126	154
R47	300	112.0	902	539	539	3.70	85	5.35	665	-	-	123	150
R41	350	108.0	870	624	624	2.30	40	3.80	605	-	-	76	101

TABLE 5-7
THE EFFECT OF 550°F IRRADIATION AT 1.88 x 10^{19} n/cm^2 (E > 1.0 MeV) ON THE NOTCH TOUGHNESS PROPERTIES OF THE D. C. COOK UNIT 1 SURVEILLANCE CAPSULE MATERIALS

	Average 30 ft-lb Temp (°F)			Average 35 mil Lateral Expansion Temp (°F)			Average 50 ft-lb Temp (°F)			Average Energy Absorption at Full Shear (ft-lb)		
Material	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	Δ(ft-lb)
Plate B4406-3 (Longitudinal)	5	120	115	20	140	120	30	155	125	130	113	-17
Plate B4406-3 (Transverse)	15	130	115	25	140	115	65	185	120	96	95	-1
Weld Metal	-90	115	205	-80	125	205	-70	175	245	110	94	-16
HAZ Metal	-100	70	175	-75	75	150	-65	120	190	126	117	-9
Correlation Material	45	165	120	60	190	130	80	200	120	120	110	-10

Note: All unirradiated data presented here was taken from WCAP-8047[1].

TABLE 5-8

COMPARISON OF D. C. COOK UNIT 1
REACTOR VESSEL SURVEILLANCE CAPSULE CHARPY IMPACT TEST RESULTS
WITH REGULATORY GUIDE 1.99 REVISION 2 PREDICTIONS

			ΔRT_{NDT} (°F)		Δ USE DECREASE (%)	
Material	Capsule	Fluence 10^{19} n/cm^2	Meas.	R.G 1.99 Pred.	Meas.	R.G 1.99 Pred.
Plate B4406-3 (Longitudinal)	T	0.18	60	51.4	18	15.4
	X	0.77	90	88.4	21	21.1
	Y	1.34	105	102.7	21	24.5
	U	1.88	115	111.3	13	26.5
Plate B4406-3 (Transverse)	T	0.18	70	51.4	14	15.4
	X	0.62	110	82.7	19	20.5
	Y	1.06	115	97.0	21	23.0
	U	1.88	115	111.3	1	26.5
Weld Metal	T	0.18	80	111.5	27	28.0
	X	0.62	165	179.6	33	37.0
	Y	1.06	200	210.5	37	41.0
	U	1.88	205	241.5	15	47.0
HAZ Metal	T	0.18	120	-	25	-
	X	0.77	160	-	36	-
	Y	1.34	165	-	38	-
	U	1.88	170	-	7	-
Correlation Material	T	0.18	60	-	15	-
	X	0.69	100	-	33	-
	Y	1.20	110	-	26	-
	U	1.88	120	-	8	-

TABLE 5-9

TENSILE PROPERTIES FOR D. C. COOK UNIT 1

REACTOR VESSEL MATERIAL IRRADIATED TO 1.88 x 10^{19} n/cm^2 (E > 1.0 MeV)

Material	Sample Number	Test Temp. (°F)	0.2% Yield Strength (ksi)	Ultimate Strength (ksi)	Fracture Load (kip)	Fracture Stress (ksi)	Fracture Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	Reduction in Area (%)
Plate	A3	74	83.0	103.9	3.40	199.8	69.3	11.3	23.1	65
Plate	A4	600	70.8	95.7	3.20	162.6	65.2	10.5	21.6	60
Weld	W11	74	83.5	97.8	3.30	167.7	67.3	12.8	23.9	60
Weld	W12	600	79.5	96.8	3.85	171.6	78.4	10.5	19.4	54

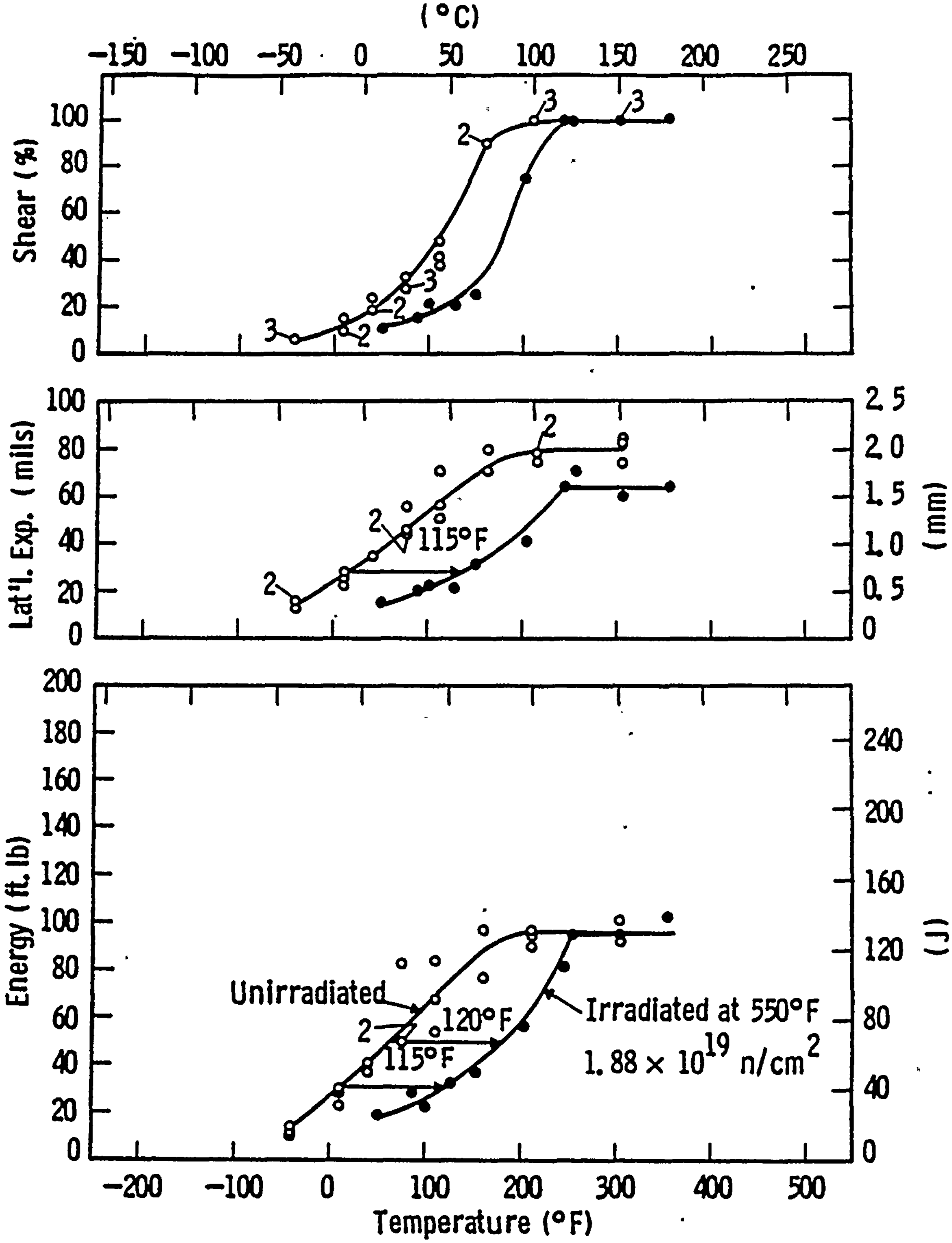


FIGURE 5-1 CHARPY V-NOTCH IMPACT DATA FOR D. C. COOK UNIT 1 REACTOR VESSEL SHELL PLATE B4406-3 (TRANSVERSE ORIENTATION)

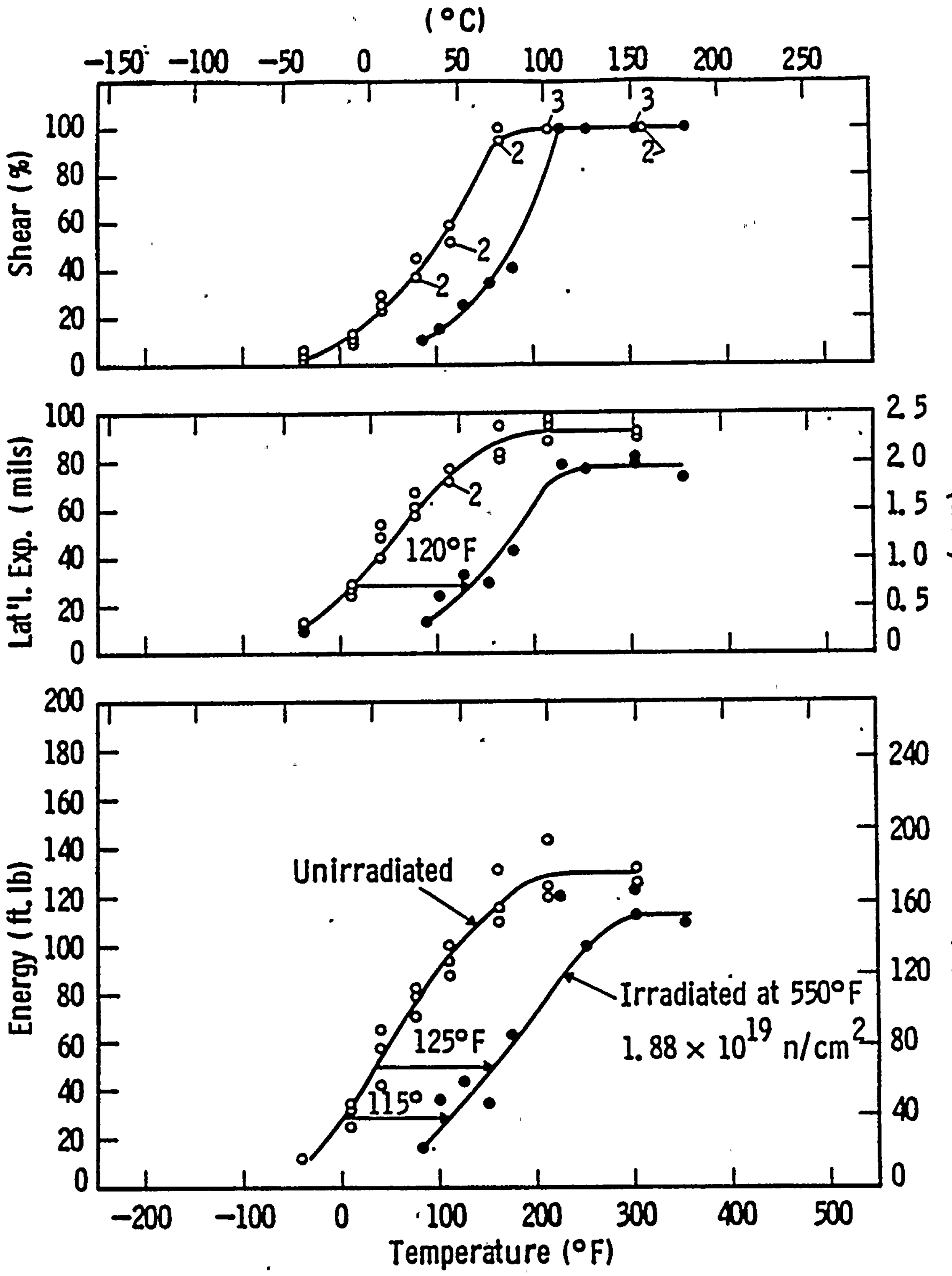


FIGURE 5-2 CHARPY V-NOTCH IMPACT DATA FOR D. C. COOK UNIT 1 REACTOR VESSEL SHELL PLATE B4406-3 (LONGITUDINAL ORIENTATION)

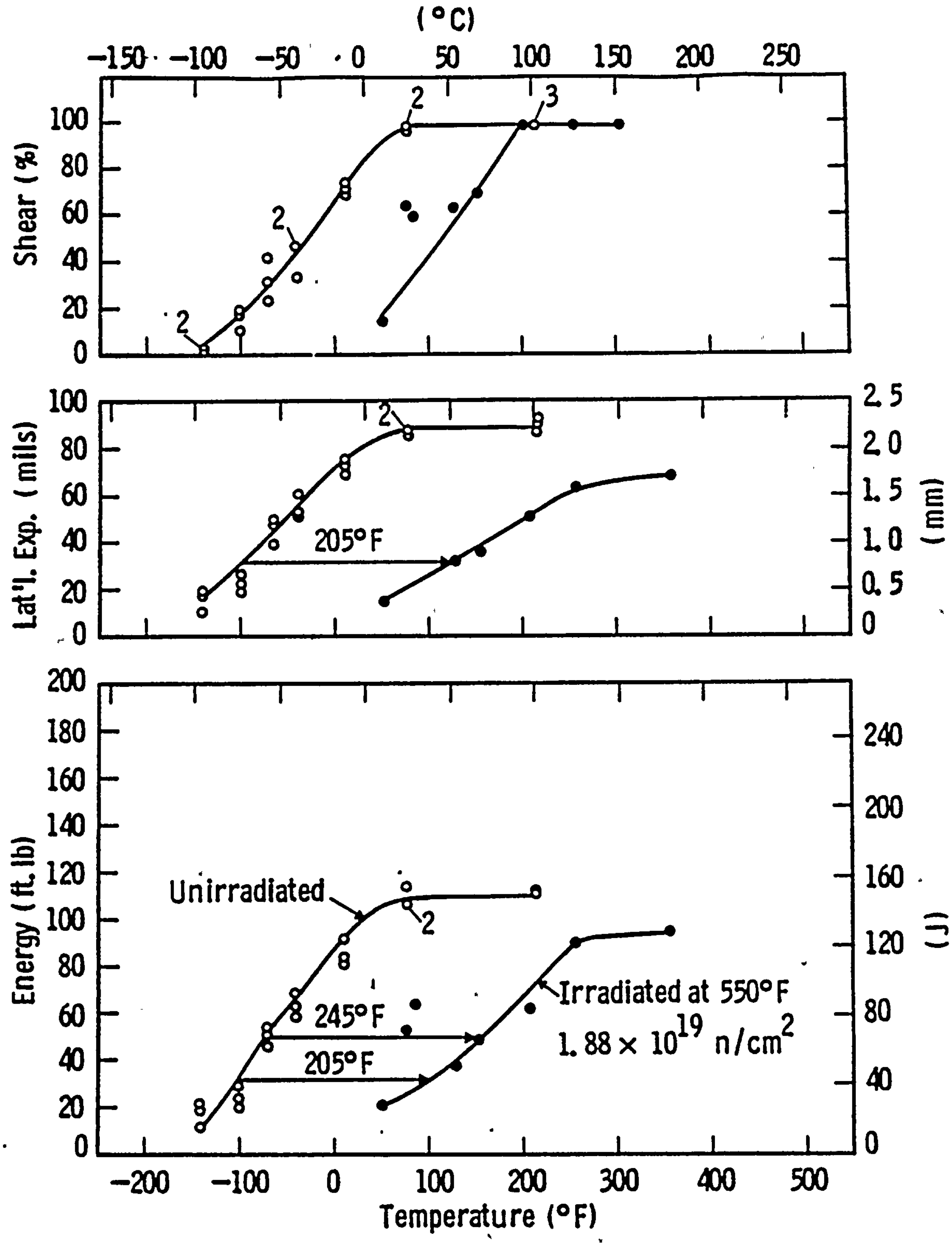


FIGURE 5-3 CHARPY V-NOTCH IMPACT DATA FOR D. C. COOK UNIT 1 REACTOR VESSEL WELD METAL

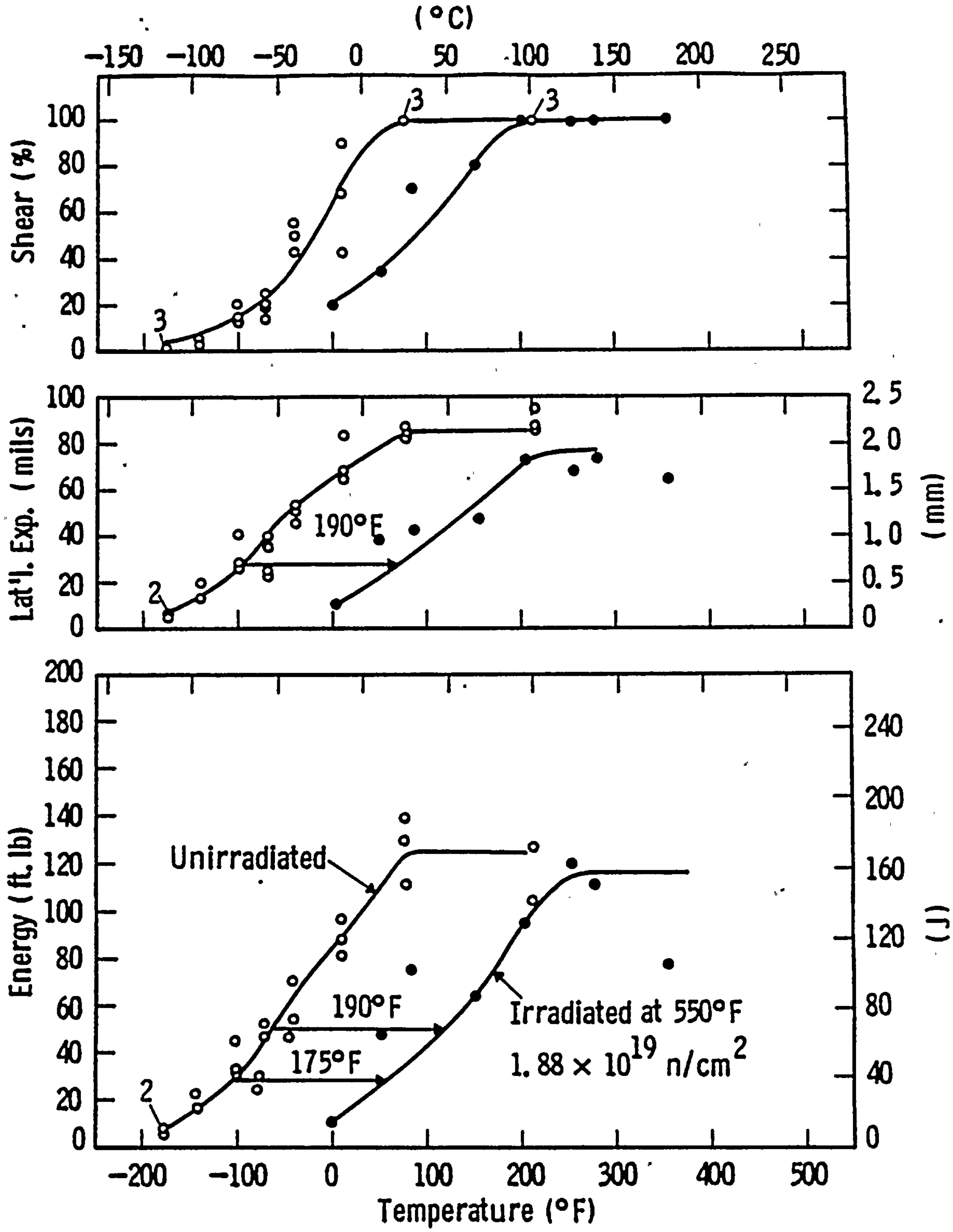


FIGURE 5-4 CHARPY V-NOTCH IMPACT DATA FOR D. C. COOK UNIT 1 REACTOR VESSEL WELD HEAT AFFECTED ZONE METAL

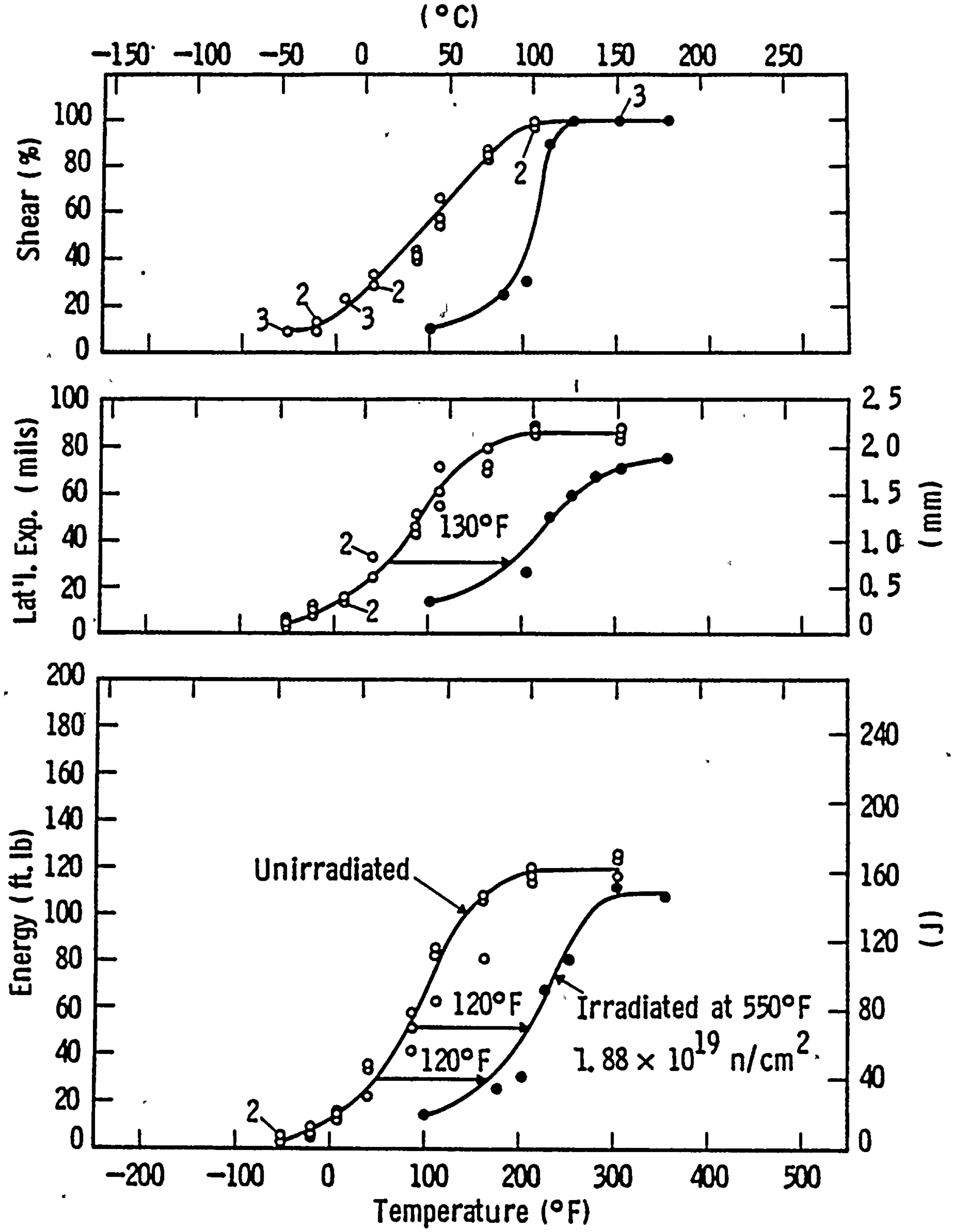


FIGURE 5-5 CHARPY V-NOTCH IMPACT DATA FOR D. C. COOK UNIT 1 ASTM CORRELATION MATERIAL

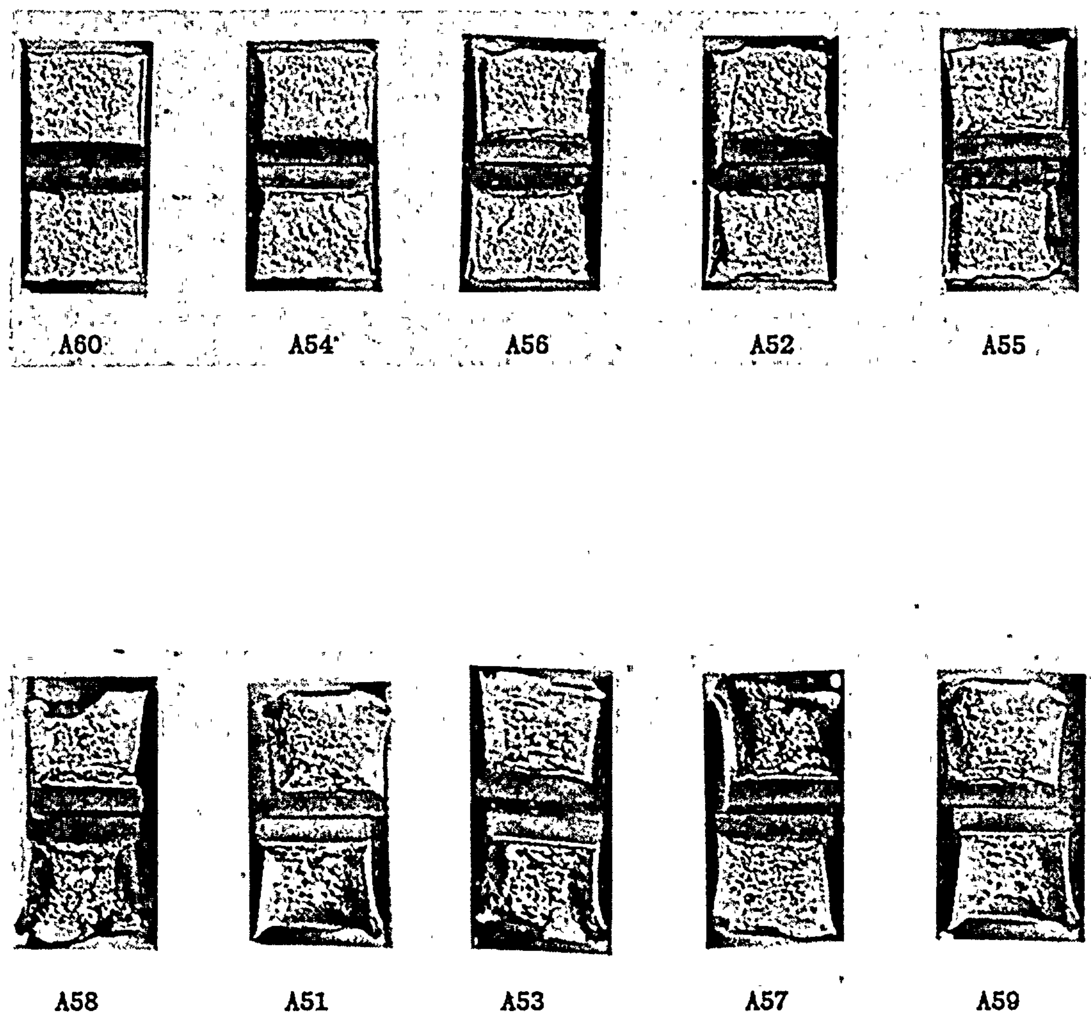


FIGURE 5-6 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR D. C. COOK UNIT 1 REACTOR VESSEL SHELL PLATE B4406-3 (LONGITUDINAL ORIENTATION)

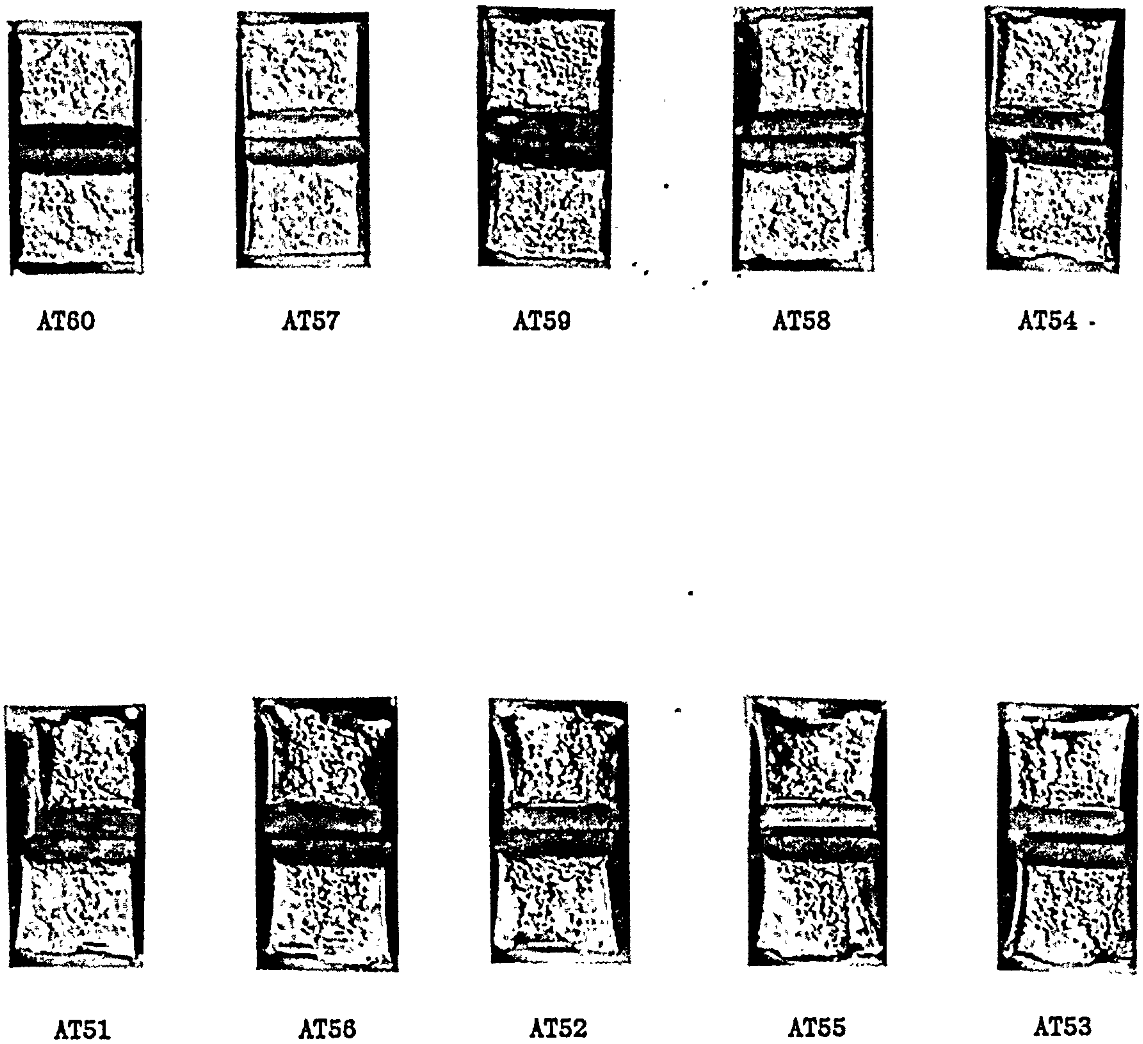


FIGURE 5-7 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR D. C. COOK UNIT 1 REACTOR VESSEL SHELL PLATE B4406-3 (TRANSVERSE ORIENTATION)

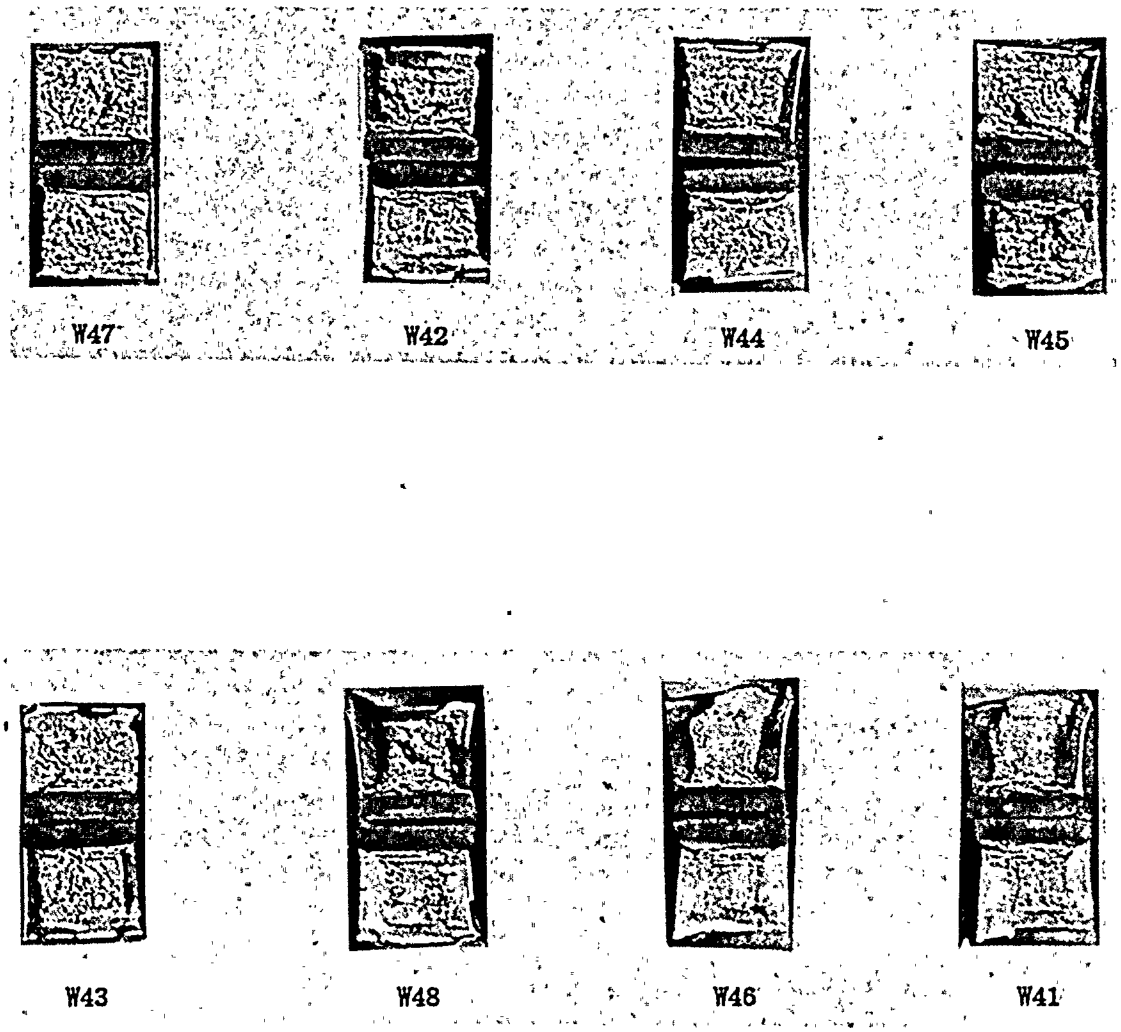


FIGURE 5-8 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR D. C. COOK UNIT 1 REACTOR VESSEL WELD METAL

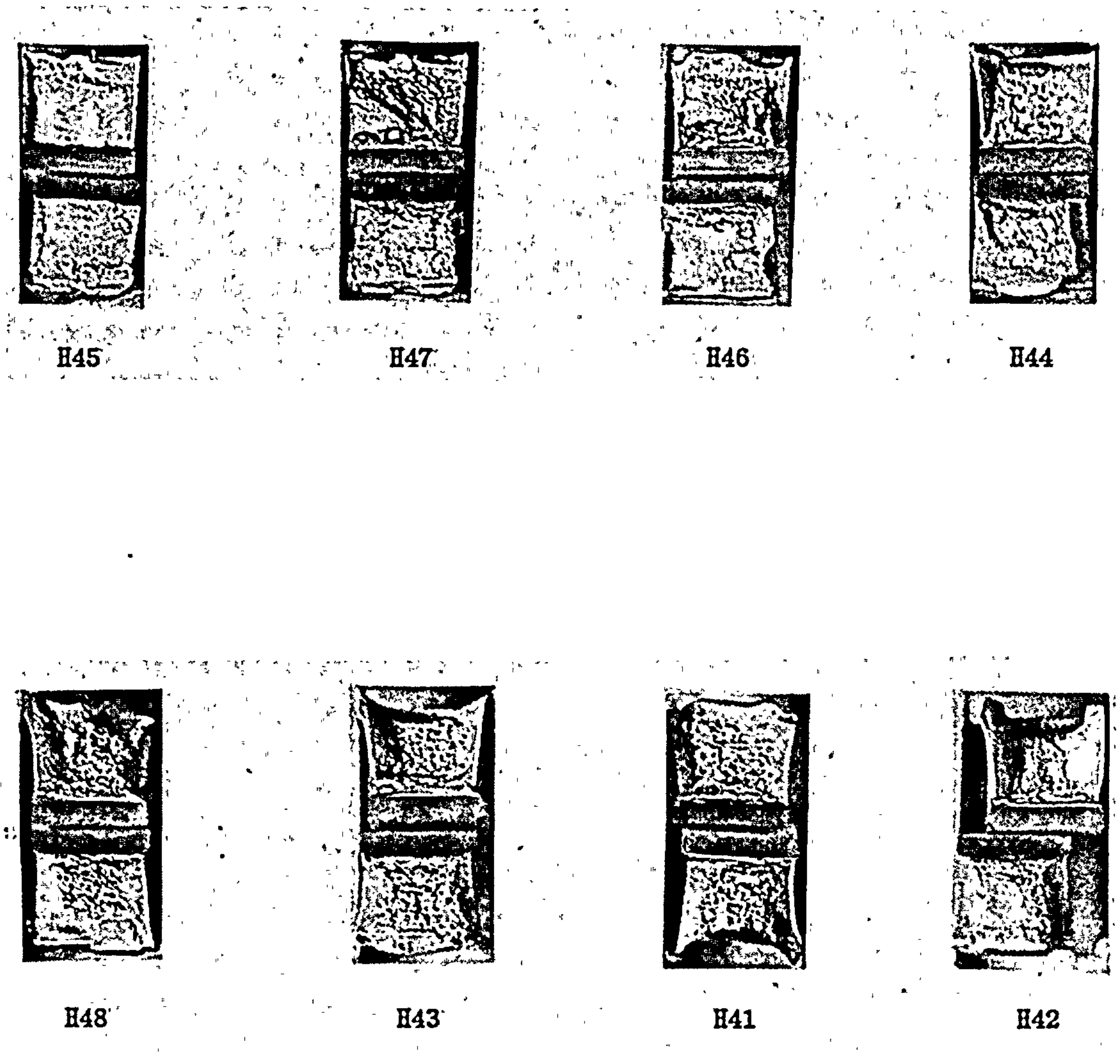


FIGURE 5-9 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR D. C. COOK UNIT 1 REACTOR VESSEL WELD HAZ METAL.

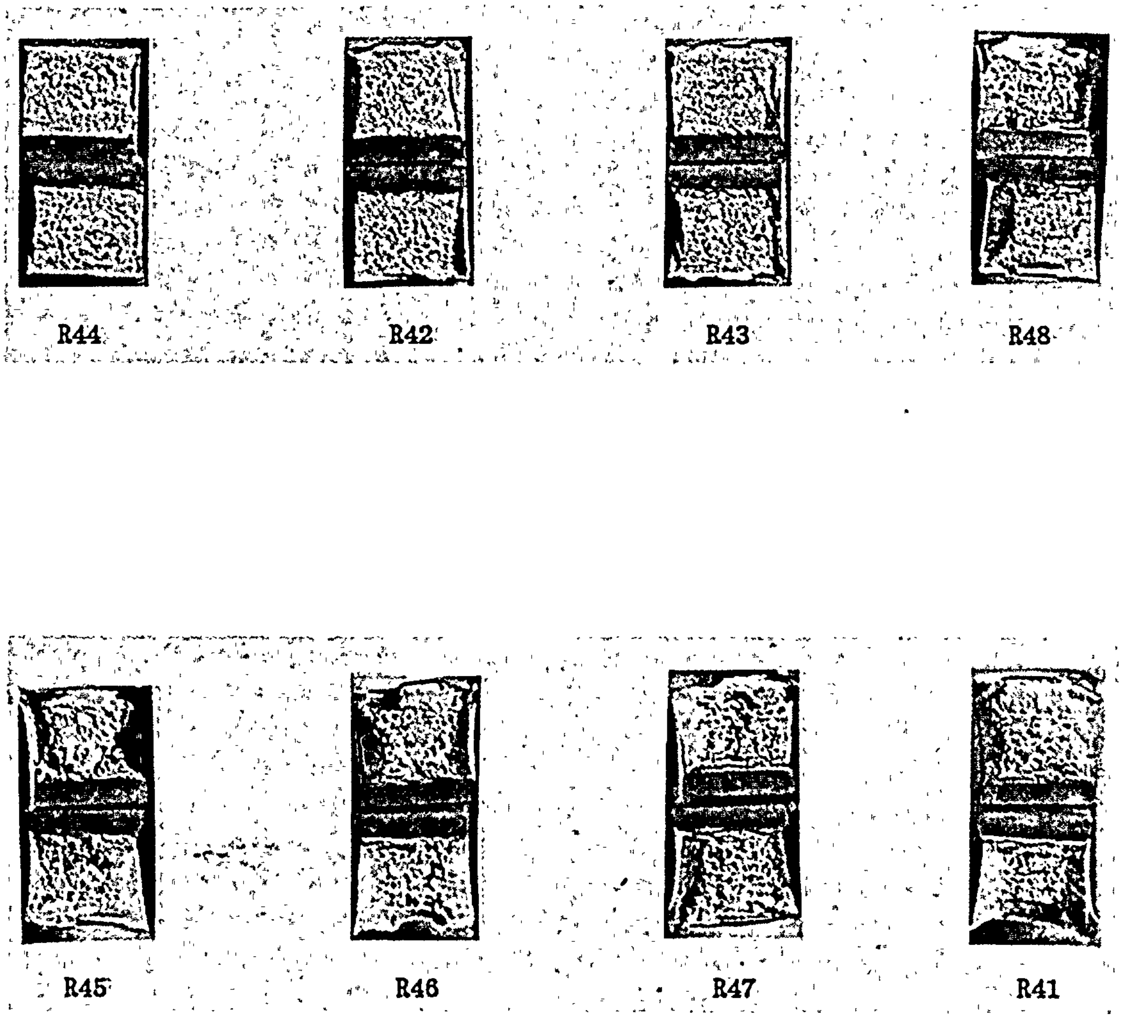


FIGURE 5-10 CHARPY IMPACT SPECIMEN FRACTURE SURFACES FOR ASTM CORRELATION MATERIAL

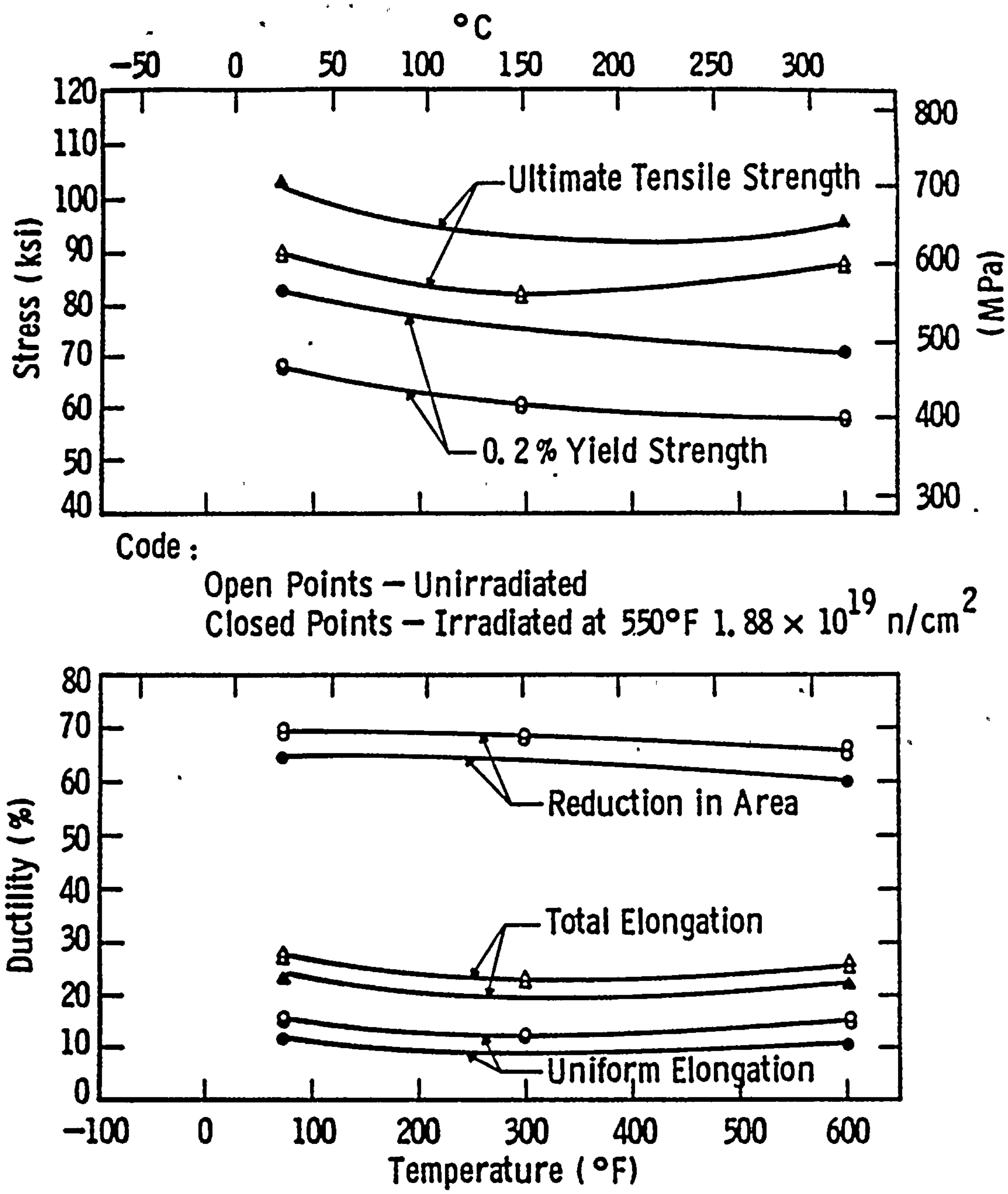


FIGURE 5-11 TENSILE PROPERTIES FOR D. C. COOK UNIT 1 REACTOR VESSEL SHELL PLATE B4406-3 (LONGITUDINAL ORIENTATION)

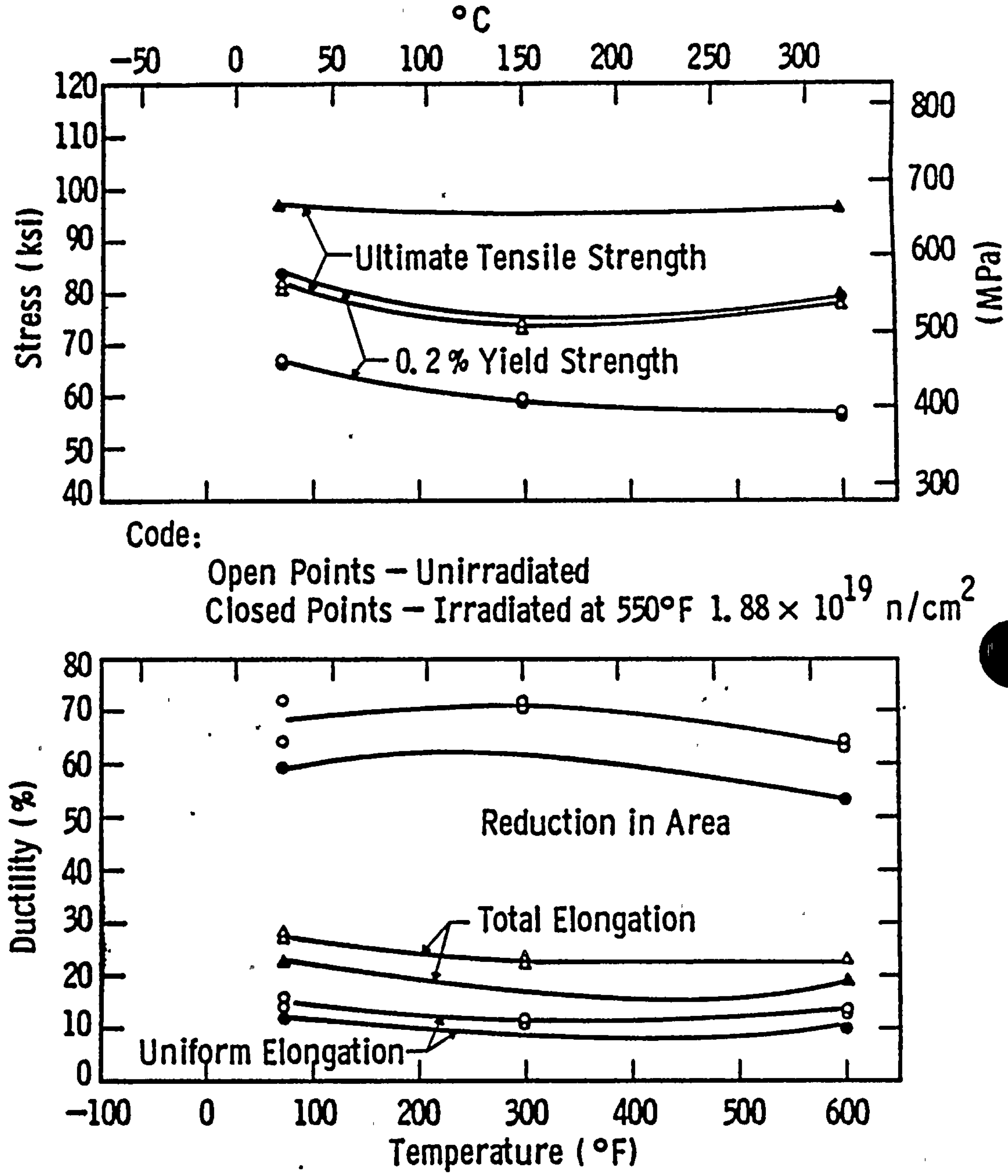
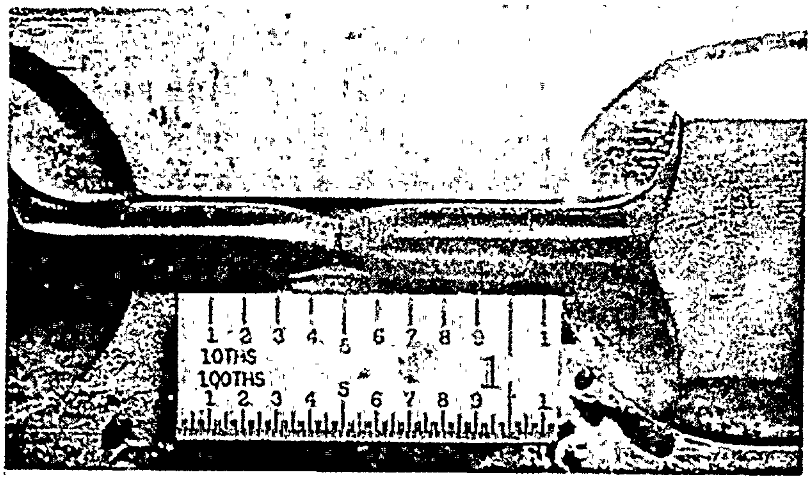
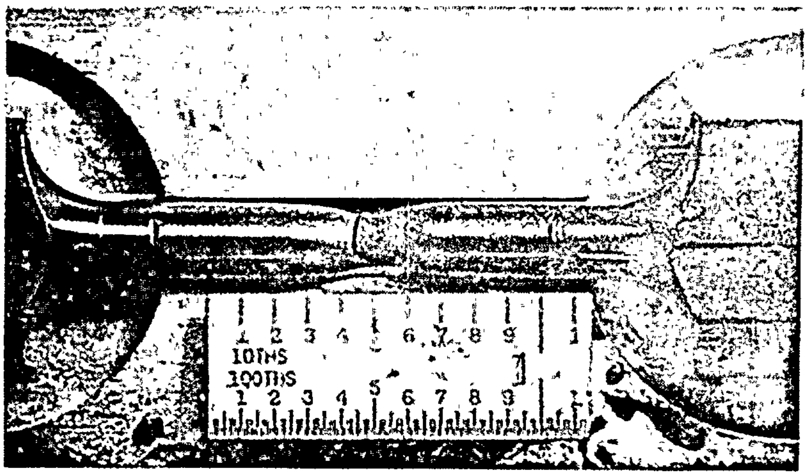


FIGURE 5-12 TENSILE PROPERTIES FOR D. C. COOK UNIT 1 REACTOR VESSEL WELD METAL

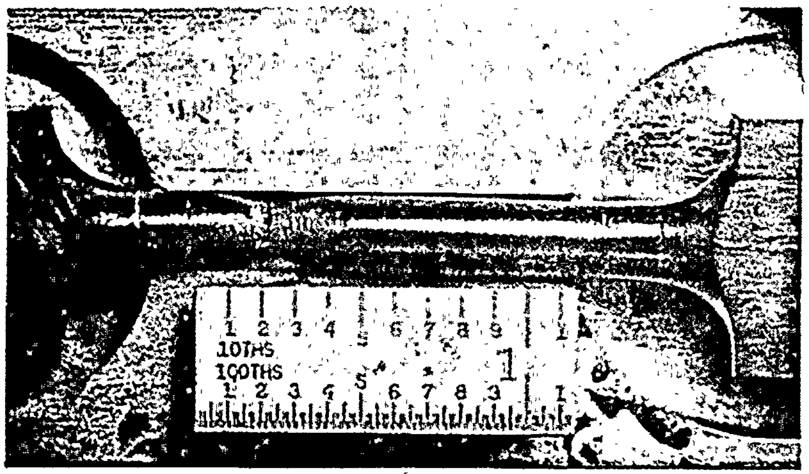


Specimen A3 74°F

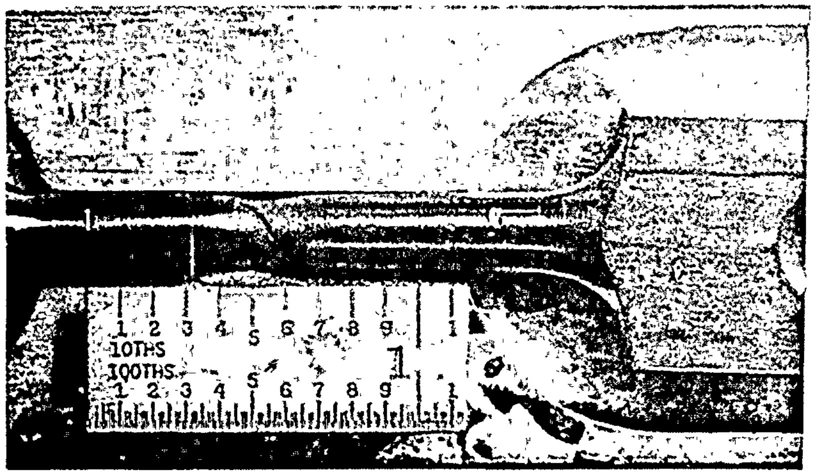


Specimen A4 600°F

FIGURE 5-13 FRACTURED TENSILE SPECIMENS FOR D. C. COOK UNIT 1 REACTOR VESSEL SHELL PLATE B4406-3 (LONGITUDINAL ORIENTATION)



Specimen W11 74°F



Specimen W12 600°F

FIGURE 5-14 FRACTURED TENSILE SPECIMENS FOR D. C. COOK UNIT 1 REACTOR VESSEL WELD METAL

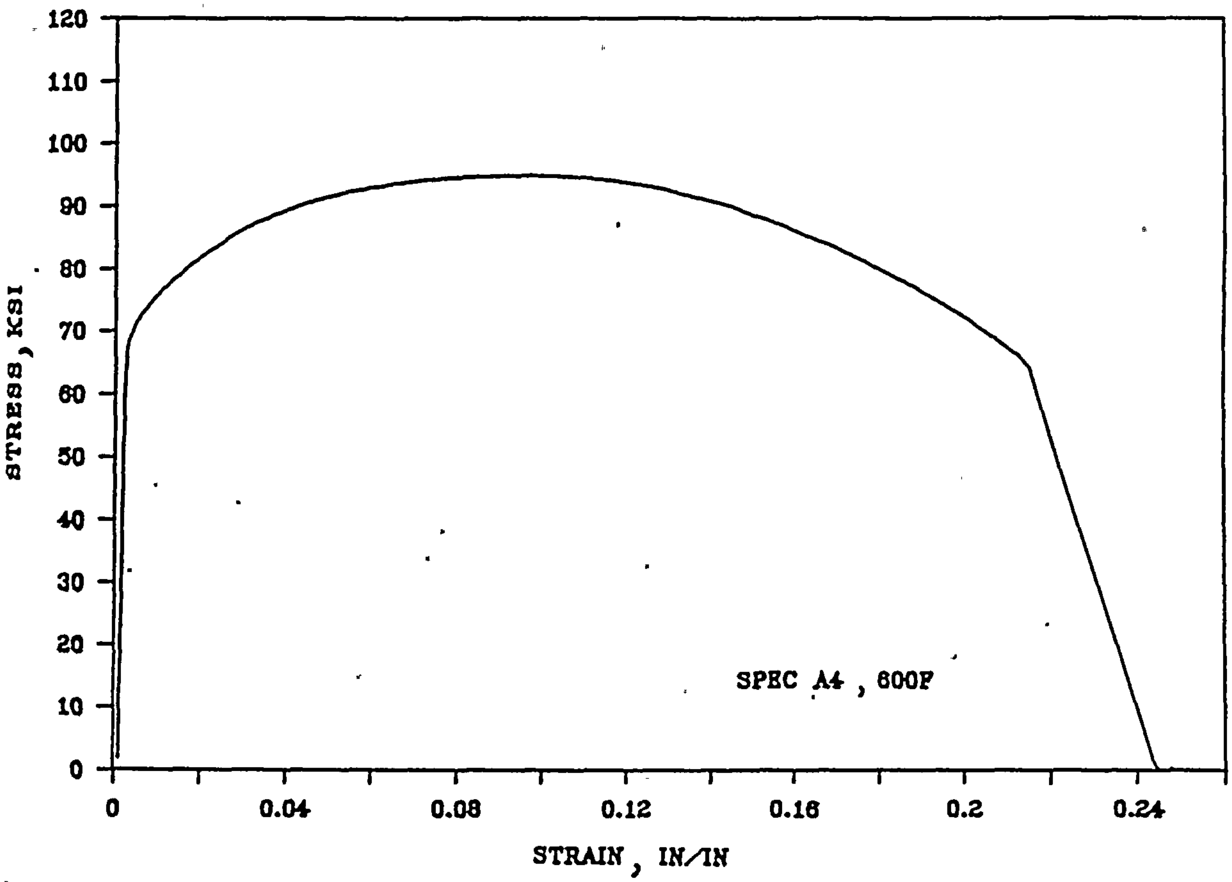


FIGURE 5-15 TYPICAL STRESS-STRAIN CURVE FOR TENSION SPECIMENS

SECTION 6
RADIATION ANALYSIS AND NEUTRON DOSIMETRY

6.1 INTRODUCTION

Knowledge of the neutron environment within the reactor pressure vessel and surveillance capsule geometry is required as an integral part of LWR reactor pressure vessel surveillance programs for two reasons. First, in order to interpret the neutron radiation-induced material property changes observed in the test specimens, the neutron environment (energy spectrum, flux, fluence) to which the test specimens were exposed must be known. Second, in order to relate the changes observed in the test specimens to the present and future condition of the reactor vessel, a relationship must be established between the neutron environment at various positions within the reactor vessel and that experienced by the test specimens. The former requirement is normally met by employing a combination of rigorous analytical techniques and measurements obtained with passive neutron flux monitors contained in each of the surveillance capsules. The latter information is derived solely from analysis.

The use of fast neutron fluence (E > 1.0 MeV) to correlate measured materials properties changes to the neutron exposure of the material for light water reactor applications has traditionally been accepted for development of damage trend curves as well as for the implementation of trend curve data to assess vessel condition. In recent years, however, it has been suggested that an exposure model that accounts for differences in neutron energy spectra between surveillance capsule locations and positions within the vessel wall could lead to an improvement in the uncertainties associated with damage trend curves as well as to a more accurate evaluation of damage gradients through the pressure vessel wall.

Because of this potential shift away from a threshold fluence toward an energy dependent damage function for data correlation, ASTM Standard Practice E853, "Analysis and Interpretation of Light Water Reactor Surveillance Results," recommends reporting displacements per iron atom (dpa) along with fluence (E > 1.0 MeV) to provide a data base for future reference. The energy

dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693, "Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements per Atom." The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the pressure vessel wall has already been promulgated in Revision 2 to the Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

This section provides the results of the neutron dosimetry evaluations performed in conjunction with the analysis of test specimens contained in surveillance capsule U. Fast neutron exposure parameters in terms of fast neutron fluence (E > 1.0 MeV), fast neutron fluence (E > 0.1 Mev), and iron atom displacements (dpa) are established for the capsule irradiation history. The analytical formalism relating the measured capsule exposure to the exposure of the vessel wall is described and used to project the integrated exposure of the vessel itself. Also, uncertainties associated with the derived exposure parameters are provided.

6.2 DISCRETE ORDINATES ANALYSIS

A plan view of the reactor geometry at the core midplane is shown in Figure 4-1. Eight irradiation capsules attached to the thermal shield are included in the reactor design to constitute the reactor vessel surveillance program. Four capsules are located symetrically at azimuthal angles of 4° and 40° relative to the core cardinal axes as shown in Figure 4-1.

A plan view of a surveillance capsule holder attached to the thermal shield is shown in Figure 6-1. The stainless steel specimen containers are 1-inch square and approximately 38 inches in height. The containers are positioned axially such that the specimens are centered on the core midplane, thus spanning the central 3 feet of the 12-foot high reactor core.

From a neutron transport standpoint, the surveillance capsule structures are significant. They have a marked effect on both the distribution of neutron flux and the neutron energy spectrum in the water annulus between the thermal shield and the reactor vessel. In order to properly determine the neutron environment at the test specimen locations, the capsules themselves must be included in the analytical model.

In performing the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, two distinct sets of transport calculations were carried out. The first, a single computation in the conventional forward mode, was used primarily to obtain relative neutron energy distributions throughout the reactor geometry as well as to establish relative radial distributions of exposure parameters (ϕ(E > 1.0 Mev,) ϕ(E > 0.1 Mev), and dpa) through the vessel wall. The neutron spectral information was required for the interpretation of neutron dosimetry withdrawn from the surveillance capsule as well as for the determination of exposure parameter ratios; i.e., dpa/ϕ(E > 1.0 MeV), within the pressure vessel geometry. The relative radial gradient information was required to permit the projection of measured exposure parameters to locations interior to the pressure vessel wall; i.e., the 1/4T, 1/2T, and 3/4T locations.

The second set of calculations consisted of a series of adjoint analyses relating the fast neutron flux (E > 1.0 MeV) at surveillance capsule positions, and several azimuthal locations on the pressure vessel inner radius to neutron source distributions within the reactor core. The importance functions generated from these adjoint analyses provided the basis for all absolute exposure projections and comparison with measurement. These importance functions, when combined with cycle specific neutron source distributions, yielded absolute predictions of neutron exposure at the locations of interest for the first 10 cycles of irradiation; and established the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles. It is important to note that the cycle specific neutron source distributions utilized in these analyses included not only spatial variations of fission rates within the reactor core; but, also accounted for the effects of varying neutron yield per fission and fission spectrum introduced by the build-up of plutonium as the burnup of individual fuel assemblies increased.

The absolute cycle specific data from the adjoint evaluations together with relative neutron energy spectra and radial distribution information from the forward calculation provided the means to:

1. Evaluate neutron dosimetry obtained from surveillance capsule locations.

2. Extrapolate dosimetry results to key locations at the inner radius and through the thickness of the pressure vessel wall.

3. Enable a direct comparison of analytical prediction with measurement.

4. Establish a mechanism for projection of pressure vessel exposure as the design of each new fuel cycle evolves.

The forward transport calculation for the reactor model summarized in Figures 4-1 and 6-1 was carried out in R, Θ geometry using the DOT two-dimensional discrete ordinates code [7] and the SAILOR cross-section library [8]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In these analyses anisotopic scattering was treated with a P_3 expansion of the cross-sections and the angular discretization was modeled with an S_8 order of angular quadrature.

The reference core power distribution utilized in the forward analysis was derived from statistical studies of long-term operation of Westinghouse 4-loop plants. Inherent in the development of this reference core power distribution is the use of an out-in fuel management strategy; i.e., fresh fuel on the core periphery. Furthermore, for the peripheral fuel assemblies, a 2σ uncertainty derived from the statistical evaluation of plant to plant and cycle to cycle variations in peripheral power was used. Since it is unlikely that a single reactor would have a power distribution at the nominal $+2\sigma$ level for a large number of fuel cycles, the use of this reference distribution is expected to yield somewhat conservative results.

All adjoint analyses were also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the SAILOR library. Adjoint source locations were chosen at several azimuthal locations along the pressure vessel inner radius as well as the geometric center of each surveillance capsule. Again, these calculations were run in R, Θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case, ϕ (E > 1.0 MeV). Having the importance functions and appropriate core source distributions, the response of interest could be calculated as:

$$R(r, \Theta) = \int_r \int_\Theta \int_E I(r, \Theta, E)\ S(r, \Theta, E)\ r\ dr\ d\Theta\ dE$$

where: $R(r, \Theta)$ = ϕ (E > 1.0 MeV) at radius r and azimuthal angle Θ

$I(r, \Theta, E)$ = Adjoint importance function at radius, r, azimuthal angle Θ, and neutron source energy E.

$S(r, \Theta, E)$ = Neutron source strength at core location r, Θ and energy E.

Although the adjoint importance functions used in the D. C. Cook Unit 1 analysis were based on a response function defined by the threshold neutron flux (E > 1.0 MeV), prior calculations have shown that, while the implementation of low leakage loading patterns significantly impact the magnitude and the spatial distribution of the neutron field, changes in the relative neutron energy spectrum are of second order. Thus, for a given location the ratio of dpa/ϕ (E > 1.0 MeV) is insensitive to changing core source distributions. In the application of these adjoint important functions to the D. C. Cook Unit 1 reactor, therefore, calculation of the iron displacement rates (dpa) and the neutron flux (E > 0.1 MeV) were computed on a cycle specific basis by using dpa/ϕ (E > 1.0 MeV) and ϕ (E > 0.1 MeV)/ϕ (E > 1.0 MeV) ratios from the forward analysis in conjunction with the cycle specific ϕ (E > 1.0 MeV) solutions from the individual adjoint evaluations.

The reactor core power distributions used in the plant specific adjoint calculations were taken from fuel cycle design studies for the first ten operating cycle of D. C. Cook Unit 1 [9 thru 12].

Selected results from the neutron transport analyses performed for the D. C. Cook Unit 1 reactor are provided in Tables 6-1 through 6-5. The data listed in these tables establish the means for absolute comparisons of analysis and measurement for the capsule irradiation period and provide the means to correlate dosimetry results with the corresponding neutron exposure of the pressure vessel wall.

In Table 6-1, the calculated exposure parameters {ϕ (E > 1.0 MeV), ϕ (E > 0.1 MeV), and dpa} are given at the geometric center of the two surveillance capsule positions for both the design basis and the plant specific core power distributions. The plant specific data, based on the adjoint transport analysis, are meant to establish the absolute comparison of measurement with analysis. The design basis data derived from the forward calculation are provided as a point of reference against which plant specific fluence evaluations can be compared. Similar data is given in Table 6-2 for the pressure vessel inner radius. Again, the three pertinent exposure parameters are listed for both the design basis and the cycle 1 through 10 plant specific power distributions. It is important to note that the data for the vessel inner radius were taken at the clad/base metal interface; and, thus, represent the maximum exposure levels of the vessel wall itself.

Radial gradient information for neutron flux (E > 1.0 MeV), neutron flux (E > 0.1 MeV), and iron atom displacement rate is given in Tables 6-3, 6-4, and 6-5, respectively. The data, obtained from the forward neutron transport calculation, are presented on a relative basis for each exposure parameter at several azimuthal locations. Exposure parameter distributions within the wall may be obtained by normalizing the calculated or projected exposure at the vessel inner radius to the gradient data given in Tables 6-3 through 6-5.

For example, the neutron flux (E > 1.0 MeV) at the 1/4T position on the 45° azimuth is given by:

$$\phi_{1/4T}(45°) = \phi(220.27, 45°)\ F\ (225.75, 45°)$$

where $\phi_{1/4T}(45°)$ = Projected neutron flux at the 1/4T position on the 45° azimuth

ϕ (220.27, 45°) = Projected or calculated neutron flux at the vessel inner radius on the 45° azimuth.

F (225.75, 45°) = Relative radial distribution function from Table 6-3.

Similar expressions apply for exposure parameters in terms of ϕ(E > 0.1 MeV) and dpa/sec.

6.3 NEUTRON DOSIMETRY

The passive neutron sensors included in the D. C. Cook Unit 1 surveillance program are listed in Table 6-6. Also given in Table 6-6 are the primary nuclear reactions and associated nuclear constants that were used in the evaluation of the neutron energy spectrum within the capsule and the subsequent determination of the various exposure parameters of interest (ϕ (E > 1.0 Mev), ϕ (E > 0.1 MeV), dpa).

The relative locations of the neutron sensors within the capsules are shown in Figure 4-2. The iron, nickel, copper, and cobalt-aluminum monitors, in wire form, were placed in holes drilled in spacers at several axial levels within the capsules. The cadmium-shielded neptunium and uranium fission monitors were accommodated within the dosimeter block located near the center of the capsule.

The use of passive monitors such as those listed in Table 6-6 does not yield a direct measure of the energy dependent flux level at the point of interest.

Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The specific activity of each monitor.
- The operating history of the reactor.
- The energy response of the monitor.
- The neutron energy spectrum at the monitor location.
- The physical characteristics of the monitor.

The specific activity of each of the neutron monitors was determined using established ASTM procedures [13 through 25]. Following sample preparation and weighing, the activity of each monitor was determined by means of a lithium-drifted germanium, Ge(Li), gamma spectrometer. The irradiation history of the D. C. Cook Unit 1 reactor during cycles 1 through 10 was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" for the applicable period.

The irradiation history applicable to capsule U is given in Table 6-7. Measured and saturated reaction product specific activities as well as measured full power reaction rates are listed in Table 6-8. Reaction rate values were derived using the pertinent data from Tables 6-6 and 6-7.

Relative to the measured reaction rates listed in Table 6-8, it should be noted that the U-238 and Np-237 results were 50-70 present low relative to the other radiometric sensors from capsule U as well as when compared to data from other capsules withdrawn from the 40 degree capsule position in other 4-loop plants. These low results may have been due to incomplete recovery of cesium (C_s) during the radiochemical processing. However, the actual cause of the low results cannot be definitively determined at this time. Because of this discrepancy, these fission monitor reaction rates were not used in the least squares adjustment of the capsule U dosimetry data.

Values of key fast neutron exposure parameters were derived from the measured reaction rates using the FERRET least squares adjustment code [26]. The FERRET approach used the measured reaction rate data and the calculated neutron energy spectrum at the the center of the surveillance capsule as input and proceeded to adjust a priori (calculated) group fluxes to produce a best fit (in a least squares sense) to the reaction rate data. The exposure parameters along with associated uncertainties where then obtained from the adjusted spectra.

In the FERRET evaluations, a log normal least-squares algorithm weights both the a priori values and the measured data in accordance with the assigned uncertainties and correlations. In general, the measured values f are linearly related to the flux ϕ by some response matrix A:

$$f_i^{(s,a)} = \sum_g A_{ig}^{(s)} \phi_g^{(\alpha)}$$

where i indexes the measured values belonging to a single data set s, g designates the energy group and a delineates spectra that may be simultaneously adjusted. For example,

$$R_i = \sum_g a_{ig} \phi_g$$

relates a set of measured reaction rates R_i to a single spectrum ϕ_g by the multigroup cross section σ_{ig}. (In this case, FERRET also adjusts the cross-sections.) The log normal approach automatically accounts for the physical constraint of positive fluxes, even with the large assigned uncertainties.

In the FERRET analysis of the dosimetry data, the continuous quantities (i.e., fluxes and cross-sections) were approximated in 53 groups. The calculated fluxes from the discrete ordinates analysis were expanded into the FERRET group structure using the SAND-II code [27]. This procedure was carried out by first expanding the a priori spectrum into the SAND-II 620 group structure using a SPLINE interpolation procedure for interpolation in regions where group boundaries do not coincide. The 620-point spectrum was then easily collapsed to the group scheme used in FERRET.

The cross-sections were also collapsed into the 53 energy-group structure using SAND II with calculated spectra (as expanded to 620 groups) as weighting functions. The cross sections were taken from the ENDF/B-V dosimetry file. Uncertainty estimates and 53 x 53 covariance matrices were constructed for each cross section. Correlations between cross sections were neglected due to data and code limitations, but are expected to be unimportant.

For each set of data or a priori values, the inverse of the corresponding relative covariance matrix M is used as a statistical weight. In some cases, as for the cross sections, a multigroup covariance matrix is used. More often, a simple parameterized form is used:

$$M_{gg'} = R_N^2 + R_g\, R_{g'}\, P_{gg'}$$

where R_N specifies an overall fractional normalization uncertainty (i.e., complete correlation) for the corresponding set of values. The fractional uncertainties R_g specify additional random uncertainties for group g that are correlated with a correlation matrix:

$$P_{gg'} = (1 - \theta)\, \delta_{gg'} + \theta \exp \frac{[-\,(g-g')^2]}{[2\,\gamma^2]}$$

The first term specifies purely random uncertainties while the second term describes short-range correlations over a range Y (θ specifies the strength of the latter term.)

For the a priori calculated fluxes, a short-range correlation of Y = 6 groups was used. This choice implies that neighboring groups are strongly correlated when θ is close to 1. Strong long-range correlations (or anticorrelations) were justified based on information presented by R. E. Maerker [28]. Maerker's results are closely duplicated when Y = 6. For the integral reaction rate covariances, simple normalization and random uncertainties were combined as deduced from experimental uncertainties.

Results of the FERRET evaluation of the capsule U dosimetry are given in Table 6-9 The data summarized in Table 6-9 indicated that the capsule received an integrated exposure of 1.88×10^{19} n/cm^2 (E > 1.0 MeV) with an associated uncertainty of ± 13%. Also reported are capsule exposures in terms of fluence (E > 0.1 MeV) and iron atom displacements (dpa). Summaries of the fit of the adjusted spectrum are provided in Table 6-10. In general, excellent results were achieved in the fits of the adjusted spectrum to the individual experimental reaction rates. The adjusted spectrum itself is tabulated in Table 6-11 for the FERRET 53 energy group structure. A summary of the measured and calculated neutron exposure of capsule U is presented in Table 6-12. The agreement between calculation and measurement is good for all exposure parameters.

Neutron exposure projections at key locations on the pressure vessel inner radius are given in Table 6-13. Along with the current (9.17 EFPY) exposure, projections are also provided for an exposure period of 23 EFPY and to end of vessel design life (32 EFPY). The time averaged exposure rates for the low leakage fuel cycles (8, 9, 10) were used to perform projections beyond the end of the cycle 1 through 10 exposure period.

In the calculation of exposure gradients for use in the development of heatup and cooldown curves for the D. C. Cook Unit 1 reactor coolant system, exposure projections to 23 EFPY and 32 EFPY were employed. Data based on both a fluence (E > 1.0 MeV) slope and a plant specific dpa slope through the vessel wall are provided in Table 6-14. In order to access RT_{NDT} vs. fluence trend curves, dpa equivalent fast neutron fluence levels for the 1/4T and 3/4T positions were defined by the relations

$$\Phi' \ 1/4T = \Phi \ (\text{Surface}) \left\{ \frac{\text{dpa } (1/4T)}{\text{dpa (Surface)}} \right\}$$

$$\Phi' \ 3/4T = \Phi \ (\text{Surface}) \left\{ \frac{\text{dpa } (3/4T)}{\text{dpa (Surface)}} \right\}$$

Using this approach results in the dpa equivalent fluence values listed in Table 6-14.

In Table 6-15 updated lead factors are listed for each of the D. C. Cook Unit 1 surveillance capsules. These data may be used as a guide in establishing future withdrawal schedules for the remaining capsules.

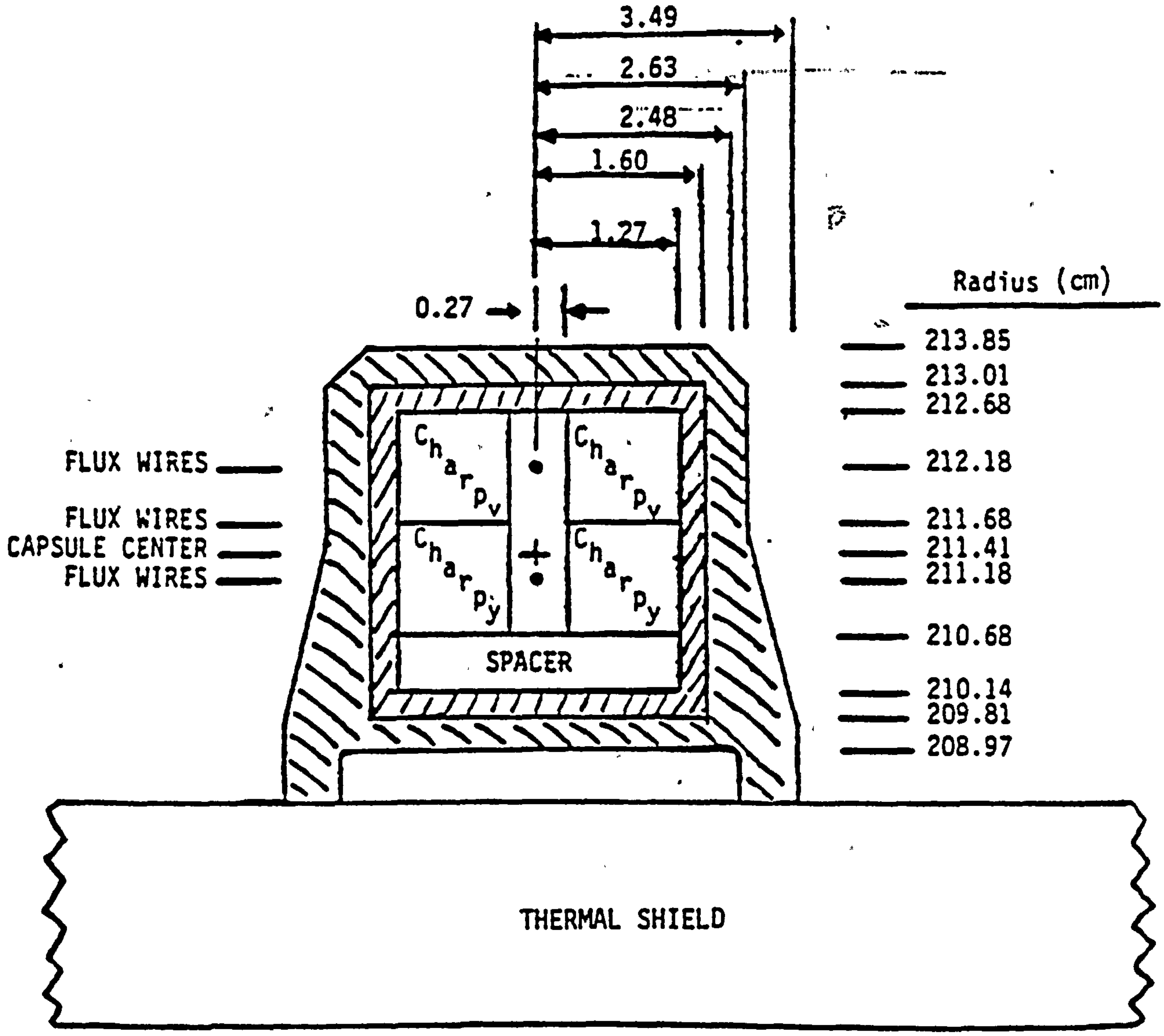


Figure 6-1. Plan View of a Reactor Vessel Surveillance Capsule

TABLE 6-1

CALCULATED FAST NEUTRON EXPOSURE RATES
AT THE CENTER OF SURVEILLANCE CAPSULES

	FLUX (E > 1.0 MEV) (N/CM2-SEC)	
	4 DEG	40 DEG
CYCLE 1	2.06×10^{10}	6.45×10^{10}
CYCLE 2	2.33×10^{10}	7.94×10^{10}
CYCLE 3	2.26×10^{10}	7.73×10^{10}
CYCLE 4	2.20×10^{10}	7.41×10^{10}
CYCLE 5	2.22×10^{10}	7.80×10^{10}
CYCLE 6	2.22×10^{10}	7.29×10^{10}
CYCLE 7	2.24×10^{10}	7.35×10^{10}
CYCLE 8	2.18×10^{10}	4.08×10^{10}
CYCLE 9	2.18×10^{10}	4.11×10^{10}
CYCLE 10	1.84×10^{10}	3.87×10^{10}
DESIGN	2.49×10^{10}	7.54×10^{10}

	FLUX (E > 0.1 MEV) (N/CM2-SEC)	
	4 DEG	40 DEG
CYCLE 1	5.96×10^{10}	2.17×10^{11}
CYCLE 2	6.74×10^{10}	2.67×10^{11}
CYCLE 3	6.54×10^{10}	2.59×10^{11}
CYCLE 4	6.36×10^{10}	2.49×10^{11}
CYCLE 5	6.42×10^{10}	2.62×10^{11}
CYCLE 6	6.42×10^{10}	2.45×10^{11}
CYCLE 7	6.48×10^{10}	2.47×10^{11}
CYCLE 8	6.31×10^{10}	1.37×10^{11}
CYCLE 9	6.31×10^{10}	1.38×10^{11}
CYCLE 10	5.32×10^{10}	1.30×10^{11}
DESIGN	7.20×10^{10}	2.53×10^{11}

TABLE 6-1 (Continued)

CALCULATED FAST NEUTRON EXPOSURE RATES
AT THE CENTER OF SURVEILLANCE CAPSULES

	dpa	
	4 DEG	40 DEG
CYCLE 1	3.35×10^{-11}	1.10×10^{-10}
CYCLE 2	3.78×10^{-11}	1.36×10^{-10}
CYCLE 3	3.67×10^{-11}	1.32×10^{-10}
CYCLE 4	3.57×10^{-11}	1.27×10^{-10}
CYCLE 5	3.61×10^{-11}	1.33×10^{-10}
CYCLE 6	3.61×10^{-11}	1.25×10^{-10}
CYCLE 7	3.64×10^{-11}	1.26×10^{-10}
CYCLE 8	3.54×10^{-11}	6.98×10^{-11}
CYCLE 9	3.54×10^{-11}	7.03×10^{-11}
CYCLE 10	2.99×10^{-11}	6.62×10^{-11}
DESIGN	4.04×10^{-11}	1.29×10^{-10}

TABLE 6-2

CALCULATED FAST NEUTRON EXPOSURE RATES AT
THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

FLUX (E > 1.0 Mev) [n/cm2-Sec]

	0 DEG	15 DEG	30 DEG	45 DEG
CYCLE 1	6.16×10^{9}	9.90×10^{9}	1.23×10^{10}	1.88×10^{10}
CYCLE 2	7.04×10^{9}	1.14×10^{10}	1.45×10^{10}	2.29×10^{10}
CYCLE 3	6.74×10^{9}	1.09×10^{10}	1.41×10^{10}	2.22×10^{10}
CYCLE 4	6.58×10^{9}	1.06×10^{10}	1.37×10^{10}	2.15×10^{10}
CYCLE 5	6.62×10^{9}	1.07×10^{10}	1.41×10^{10}	2.24×10^{10}
CYCLE 6	6.64×10^{9}	1.07×10^{10}	1.35×10^{10}	2.12×10^{10}
CYCLE 7	6.68×10^{9}	1.08×10^{10}	1.36×10^{10}	2.13×10^{10}
CYCLE 8	6.36×10^{9}	9.27×10^{9}	8.46×10^{9}	1.22×10^{10}
CYCLE 9	6.22×10^{9}	8.53×10^{9}	8.33×10^{9}	1.22×10^{10}
CYCLE 10	5.46×10^{9}	8.16×10^{9}	7.91×10^{9}	1.16×10^{10}
DESIGN	8.02×10^{9}	1.30×10^{10}	1.61×10^{10}	2.49×10^{10}

FLUX (E > 0.1 Mev) [n/cm2-Sec]

	0 DEG	15 DEG	30 DEG	45 DEG
CYCLE 1	1.55×10^{10}	2.48×10^{10}	3.16×10^{10}	5.00×10^{10}
CYCLE 2	1.77×10^{10}	2.86×10^{10}	3.73×10^{10}	6.09×10^{10}
CYCLE 3	1.69×10^{10}	2.73×10^{10}	3.63×10^{10}	5.90×10^{10}
CYCLE 4	1.65×10^{10}	2.66×10^{10}	3.52×10^{10}	5.72×10^{10}
CYCLE 5	1.66×10^{10}	2.68×10^{10}	3.63×10^{10}	5.96×10^{10}
CYCLE 6	1.67×10^{10}	2.68×10^{10}	3.47×10^{10}	5.64×10^{10}
CYCLE 7	1.68×10^{10}	2.71×10^{10}	3.50×10^{10}	5.66×10^{10}
CYCLE 8	1.60×10^{10}	2.32×10^{10}	2.18×10^{10}	3.24×10^{10}
CYCLE 9	1.56×10^{10}	2.14×10^{10}	2.14×10^{10}	3.24×10^{10}
CYCLE 10	1.37×10^{10}	2.04×10^{10}	2.03×10^{10}	3.08×10^{10}
DESIGN	2.01×10^{10}	3.26×10^{10}	4.14×10^{10}	6.62×10^{10}

TABLE 6-3

RELATIVE RADIAL DISTRIBUTIONS OF NEUTRON FLUX (E > 1.0 MeV) WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0°	15°	30°	45°
220.27(1)	1.00	1.00	1.00	1.00
220.64	0.977	0.978	0.979	0.977
221.66	0.884	0.887	0.889	0.885
222.99	0.758	0.762	0.765	0.756
224.31	0.641	0.644	0.648	0.637
225.63	0.537	0.540	0.545	0.534
226.95	0.448	0.451	0.455	0.443
228.28	0.372	0.373	0.379	0.367
229.60	0.309	0.310	0.315	0.303
230.92	0.255	0.257	0.261	0.250
232.25	0.211	0.212	0.216	0.206
233.57	0.174	0.175	0.178	0.169
234.89	0.143	0.144	0.147	0.138
236.22	0.117	0.118	0.121	0.113
237.54	0.0961	0.0963	0.0989	0.0912
238.86	0.0783	0.0783	0.0807	0.0736
240.19	0.0635	0.0632	0.0656	0.0584
241.51	0.0511	0.0501	0.0519	0.0454
242.17(2)	0.0483	0.0469	0.0487	0.0422

NOTES: 1) Base Metal Inner Radius
2) Base Metal Outer Radius

TABLE 6-2 (Continued)

CALCULATED FAST NEUTRON EXPOSURE RATES AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

	dpa/sec			
	0 DEG	15 DEG	30 DEG	45 DEG
CYCLE 1	1.00×10^{-11}	1.59×10^{-11}	1.99×10^{-11}	3.06×10^{-11}
CYCLE 2	1.15×10^{-11}	1.84×10^{-11}	2.35×10^{-11}	3.73×10^{-11}
CYCLE 3	1.10×10^{-11}	1.75×10^{-11}	2.29×10^{-11}	3.61×10^{-11}
CYCLE 4	1.07×10^{-11}	1.71×10^{-11}	2.22×10^{-11}	3.50×10^{-11}
CYCLE 5	1.08×10^{-11}	1.72×10^{-11}	2.29×10^{-11}	3.65×10^{-11}
CYCLE 6	1.08×10^{-11}	1.72×10^{-11}	2.19×10^{-11}	3.45×10^{-11}
CYCLE 7	1.09×10^{-11}	1.74×10^{-11}	2.20×10^{-11}	3.47×10^{-11}
CYCLE 8	1.04×10^{-11}	1.49×10^{-11}	1.37×10^{-11}	1.99×10^{-11}
CYCLE 9	1.01×10^{-11}	1.37×10^{-11}	1.35×10^{-11}	1.99×10^{-11}
CYCLE 10	8.90×10^{-12}	1.31×10^{-11}	1.28×10^{-11}	1.89×10^{-11}
DESIGN	1.31×10^{-11}	2.09×10^{-11}	2.61×10^{-11}	4.05×10^{-11}

TABLE 6-4

RELATIVE RADIAL DISTRIBUTIONS OF NEUTRON FLUX (E > 0.1 MeV) WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0°	15°	30°	45°
220.27(1)	1.00	1.00	1.00	1.00
220.64	1.00	1.00	1.00	1.00
221.66	1.00	0.996	1.00	0.994
222.99	0.965	0.958	0.968	0.953
224.31	0.916	0.906	0.919	0.898
225.63	0.861	0.849	0.865	0.838
226.95	0.803	0.790	0.809	0.777
228.28	0.746	0.732	0.752	0.717
229.60	0.689	0.675	0.695	0.657
230.92	0.633	0.619	0.640	0.600
232.25	0.578	0.565	0.586	0.544
233.57	0.525	0.513	0.534	0.490
234.89	0.474	0.463	0.483	0.437
236.22	0.424	0.414	0.433	0.387
237.54	0.375	0.367	0.385	0.338
238.86	0.328	0.322	0.338	0.291
240.19	0.283	0.277	0.292	0.244
241.51	0.239	0.232	0.245	0.196
242.17(2)	0.229	0.220	0.232	0.183

NOTES: 1) Base Metal Inner Radius
2) Base Metal Outer Radius

TABLE 6-5

RELATIVE RADIAL DISTRIBUTIONS OF IRON DISPLACEMENT RATE (dpa) WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0°	15°	30°	45°
220.27(1)	1.00	1.00	1.00	1.00
220.64	0.983	0.983	0.984	0.983
221.66	0.913	0.914	0.918	0.915
222.99	0.818	0.819	0.827	0.820
224.31	0.728	0.728	0.739	0.730
225.63	0.647	0.646	0.659	0.647
226.95	0.574	0.573	0.587	0.573
228.28	0.510	0.507	0.523	0.507
229.60	0.453	0.450	0.466	0.449
230.92	0.402	0.399	0.414	0.397
232.25	0.356	0.353	0.368	0.349
233.57	0.315	0.312	0.327	0.307
234.89	0.277	0.275	0.289	0.269
236.22	0.243	0.241	0.254	0.233
237.54	0.212	0.210	0.222	0.201
238.86	0.182	0.181	0.192	0.170
240.19	0.155	0.154	0.164	0.141
241.51	0.131	0.128	0.137	0.113
242.17(2)	0.125	0.122	0.130	0.106

NOTES: 1) Base Metal Inner Radius
2) Base Metal Outer Radius

TABLE 6-6

NUCLEAR PARAMETERS FOR NEUTRON FLUX MONITORS

Monitor Material	Reaction of Interest	Target Weight Fraction	Response Range	Fission Product Half-Life	Yield (%)
Copper	$Cu^{63}(n,\alpha)Co^{60}$	0.6917	E> 4.7 MeV	5.272 yrs	
Iron	$Fe^{54}(n,p)Mn^{54}$	0.0582	E> 1.0 MeV	312.2 days	
Nickel	$Ni^{58}(n,p)Co^{58}$	0.6830	E> 1.0 MeV	70.90 days	
Uranium-238*	$U^{238}(n,f)Cs^{137}$	1.0	E> 0.4 MeV	30.12 yrs	5.99
Neptunium-237*	$Np^{237}(n,f)Cs^{137}$	1.0	E> 0.08 MeV	30.12 yrs	6.50
Cobalt-Aluminum*	$Co^{59}(n,\gamma)Co^{60}$	0.0015	0.4ev <E< 0.015 MeV	5.272 yrs	
Cobalt-Aluminum*	$Co^{59}(n,\gamma)Co^{60}$	0.0015	E < 0.015 MeV*	5.272 yrs	

*Denotes that monitor is cadmium shielded.

TABLE 6-7

IRRADIATION HISTORY OF NEUTRON SENSORS
CONTAINED IN CAPSULE U

Month	Year	PJ (MW)	PJ/PMAX	Irradiation Time (days)	Decay Time (days)
1	1975	3.1	0.0010	14	5366
2	1975	27.2	0.0084	28	5338
3	1975	768.8	0.2366	31	5307
4	1975	2260.1	0.6954	30	5277
5	1975	2598.8	0.7996	31	5246
6	1975	2365.2	0.7277	30	5216
7	1975	901.0	0.2772	31	5185
8	1975	2485.4	0.7647	31	5154
9	1975	2587.9	0.7963	30	5124
10	1975	2080.2	0.6400	31	5093
11	1975	1339.7	0.4122	30	5063
12	1975	2451.4	0.7543	31	5032
1	1976	2261.2	0.6957	31	5001
2	1976	2420.2	0.7447	29	4972
3	1976	2515.8	0.7741	31	4941
4	1976	1277.8	0.3932	30	4911
5	1976	1478.3	0.4549	31	4880
6	1976	3164.3	0.9736	30	4850
7	1976	2487.4	0.7654	31	4819
8	1976	3226.1	0.9926	31	4788
9	1976	2258.2	0.6948	30	4758
10	1976	3249.0	0.9997	31	4727
11	1976	2657.2	0.8176	30	4697
12	1976	2338.9	0.7197	31	4666
1	1977	0.0	0.0000	31	4635
2	1977	2759.3	0.8490	28	4607
3	1977	164.4	0.0506	31	4576
4	1977	2347.3	0.7222	30	4546
5	1977	1830.0	0.5631	31	4515
6	1977	1675.4	0.5155	30	4485
7	1977	2088.2	0.6425	31	4454
8	1977	2347.0	0.7222	31	4423
9	1977	1743.1	0.5364	30	4393
10	1977	2142.6	0.6593	31	4362
11	1977	1460.4	0.4494	30	4332
12	1977	2780.8	0.8556	31	4301
1	1978	2584.8	0.7953	31	4270
2	1978	2907.2	0.8945	28	4242
3	1978	3129.4	0.9629	31	4211
4	1978	608.7	0.1873	30	4181
5	1978	0.0	0.0000	31	4150
6	1978	320.0	0.0985	30	4120

TABLE 6-7 (Continued)

IRRADIATION HISTORY OF NEUTRON SENSORS CONTAINED IN CAPSULE U

Month	Year	PJ (MW)	PJ/PMAX	Irradiation Time (days)	Decay Time (days)
7	1978	3032.3	0.9330	31	4089
8	1978	2992.7	0.9208	31	4058
9	1978	2982.4	0.9176	30	4028
10	1978	3078.3	0.9472	31	3997
11	1978	3203.6	0.9857	30	3967
12	1978	2259.6	0.6953	31	3936
1	1979	3090.6	0.9510	31	3905
2	1979	3112.5	0.9577	28	3877
3	1979	2875.5	0.8848	31	3846
4	1979	571.9	0.1760	30	3816
5	1979	0.0	0.0000	31	3785
6	1979	0.0	0.0000	30	3755
7	1979	1024.6	0.3152	31	3724
8	1979	3221.0	0.9911	31	3693
9	1979	3213.1	0.9886	30	3663
10	1979	2779.1	0.8551	31	3632
11	1979	2115.2	0.6508	30	3602
12	1979	2316.9	0.7129	31	3571
1	1980	1318.1	0.4056	31	3540
2	1980	3174.4	0.9767	29	3511
3	1980	3240.2	0.9970	31	3480
4	1980	3072.7	0.9455	30	3450
5	1980	3099.2	0.9536	31	3419
6	1980	0.0	0.0000	30	3389
7	1980	0.0	0.0000	31	3358
8	1980	2018.2	0.6210	31	3327
9	1980	2917.4	0.8977	30	3297
10	1980	3107.6	0.9562	31	3266
11	1980	3185.7	0.9802	30	3236
12	1980	2464.9	0.7584	31	3205
1	1981	2546.0	0.7834	31	3174
2	1981	3238.8	0.9965	28	3146
3	1981	3240.6	0.9971	31	3115
4	1981	3244.7	0.9984	30	3085
5	1981	3030.9	0.9326	31	3054
6	1981	0.0	0.0000	30	3024
7	1981	0.0	0.0000	31	2993
8	1981	2406.1	0.7403	31	2962
9	1981	3240.6	0.9971	30	2932
10	1981	3241.8	0.9975	31	2901
11	1981	1777.3	0.5469	30	2871
12	1981	3022.1	0.9299	31	2840
1	1982	2109.6	0.6491	31	2809

TABLE 6-7 (Continued)

IRRADIATION HISTORY OF NEUTRON SENSORS
CONTAINED IN CAPSULE U

Month	Year	PJ (MW)	PJ/PMAX	Irradiation Time (days)	Decay Time (days)
2	1982	0.0	0.0000	28	2781
3	1982	2720.5	0.8371	31	2750
4	1982	3116.2	0.9588	30	2720
5	1982	2947.5	0.9069	31	2689
6	1982	3235.2	0.9954	30	2659
7	1982	195.7	0.0602	31	2628
8	1982	0.0	0.0000	31	2597
9	1982	17.7	0.0055	30	2567
10	1982	2642.2	0.8130	31	2536
11	1982	3160.3	0.9724	30	2506
12	1982	2930.5	0.9017	31	2475
1	1983	3187.6	0.9808	31	2444
2	1983	3206.1	0.9865	28	2416
3	1983	2978.0	0.9163	31	2385
4	1983	3192.3	0.9822	30	2355
5	1983	2789.7	0.8584	31	2324
6	1983	3031.5	0.9328	30	2294
7	1983	1268.4	0.3903	31	2263
8	1983	0.0	0.0000	31	2232
9	1983	0.0	0.0000	30	2202
10	1983	434.3	0.1336	31	2171
11	1983	2050.4	0.6309	30	2141
12	1983	991.3	0.3050	31	2110
1	1984	2342.8	0.7208	31	2079
2	1984	2787.6	0.8577	29	2050
3	1984	3064.2	0.9428	31	2019
4	1984	2576.9	0.7929	30	1989
5	1984	3224.1	0.9920	31	1958
6	1984	2585.0	0.7954	30	1928
7	1984	2818.5	0.8672	31	1897
8	1984	1814.1	0.5582	31	1866
9	1984	3096.8	0.9529	30	1836
10	1984	2729.0	0.8397	31	1805
11	1984	3038.6	0.9349	30	1775
12	1984	2842.9	0.8747	31	1744
1	1985	940.4	0.2894	31	1713
2	1985	3089.0	0.9505	28	1685
3	1985	3093.1	0.9517	31	1654
4	1985	474.7	0.1461	30	1624
5	1985	0.0	0.0000	31	1593
6	1985	0.0	0.0000	30	1563
7	1985	0.0	0.0000	31	1532
8	1985	0.0	0.0000	31	1501

TABLE 6-7 (Continued)

IRRADIATION HISTORY OF NEUTRON SENSORS CONTAINED IN CAPSULE U

Month	Year	PJ (MW)	PJ/PMAX	Irradiation Time (days)	Decay Time (days)
9	1985	0.0	0.0000	30	1471
10	1985	0.0	0.0000	31	1440
11	1985	413.5	0.1272	30	1410
12	1985	1491.8	0.4590	31	1379
1	1986	2932.9	0.9024	31	1348
2	1986	2927.5	0.9008	28	1320
3	1986	2930.5	0.9017	31	1289
4	1986	2640.9	0.8126	30	1259
5	1986	2837.1	0.8730	31	1228
6	1986	0.0	0.0000	30	1198
7	1986	1100.8	0.3387	31	1167
8	1986	2670.6	0.8217	31	1136
9	1986	2928.8	0.9012	30	1106
10	1986	2931.8	0.9021	31	1075
11	1986	2784.4	0.8567	30	1045
12	1986	2949.2	0.9074	31	1014
1	1987	2889.1	0.8889	31	983
2	1987	2846.8	0.8759	28	955
3	1987	2821.6	0.8682	31	924
4	1987	1346.4	0.4143	30	894
5	1987	2764.2	0.8505	31	863
6	1987	2272.0	0.6991	30	833
7	1987	0.0	0.0000	31	802
8	1987	0.0	0.0000	31	771
9	1987	0.0	0.0000	30	741
10	1987	1782.7	0.5485	31	710
11	1987	2933.6	0.9026	30	680
12	1987	2818.3	0.8672	31	649
1	1988	2791.6	0.8590	31	618
2	1988	2870.1	0.8831	29	589
3	1988	2697.8	0.8301	31	558
4	1988	2902.5	0.8931	30	528
5	1988	2898.6	0.8919	31	497
6	1988	2760.1	0.8492	30	467
7	1988	2933.6	0.9026	31	436
8	1988	3091.8	0.9513	31	405
9	1988	2011.5	0.6189	30	375
10	1988	2493.2	0.7671	31	344
11	1988	2669.8	0.8215	30	314
12	1988	2935.2	0.9031	31	283
1	1989	2481.3	0.7635	31	252
2	1989	2327.0	0.7160	28	224
3	1989	1586.1	0.4880	19	205

TABLE 6-8

MEASURED SENSOR ACTIVITIES AND REACTION RATES

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Adjusted Saturated Acticity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Cu-63 (n,α) Co-60			
Top-Middle	1.30×10^5	2.74×10^5	
Middle	1.26×10^5	2.66×10^5	
Bottom-Middle	1.33×10^5	2.80×10^5	
Average	1.30×10^5	2.73×10^5	4.17×10^{-17}
Fe-54(n,p) Mn-54			
Top	7.10×10^5	2.50×10^6	
Top-Middle	7.34×10^5	2.59×10^6	
Middle	7.31×10^5	2.57×10^6	
Bottom	7.10×10^5	2.50×10^6	
Average	7.21×10^5	2.54×10^6	4.05×10^{-15}
Ni-58 (n,p) Co-58			
Top-Middle	2.30×10^6	4.21×10^7	
Middle	2.27×10^6	4.15×10^7	
Bottom-Middle	2.36×10^6	4.32×10^7	
Average	2.31×10^6	4.23×10^7	6.03×10^{-15}
U-238 (n,f) Cs-137 (Cd)			
Middle	3.67×10^5	2.10×10^6	1.39×10^{-14}

TABLE 6-8 (Continued)

MEASURED SENSOR ACTIVITIES AND REACTION RATES

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Adjusted Saturated Acticity (dis/sec-gm)	Reaction Rate (RPS/NUCLEUS)
Np-237(n,f) Cs-137 (Cd)			
Middle	2.87×10^6	1.65×10^7	9.94×10^{-14}
Co-59 (n,γ) Co-60 (Cd)			
Top	8.66×10^6	2.21×10^7	
Bottom	8.01×10^6	2.04×10^7	
Average	8.34×10^6	2.12×10^7	1.39×10^{-12}
Co-59 (n,Y) Co-60			
Top	2.06×10^7	4.41×10^7	
Bottom	1.93×10^7	4.13×10^7	
Average	2.00×10^7	4.27×10^7	2.79×10^{-12}

TABLE 6-9

SUMMARY OF NEUTRON DOSIMETRY RESULTS

	TIME AVERAGED EXPOSURE RATES	
ϕ (E> 1.0 MeV) {n/cm^2-sec}	6.50×10^{10}	$\pm$ 13%
ϕ (E> 0.1 MeV) {n/cm^2-sec}	2.21×10^{11}	$\pm$ 22%
dpa/sec	1.09×10^{-10}	$\pm$ 16%
	INTEGRATED CAPSULE EXPOSURE	
Φ (E> 1.0 MeV) {n/cm^2}	1.88×10^{19}	$\pm$ 13%
Φ (E> 0.1 MeV) {n/cm^2}	6.40×10^{19}	$\pm$ 22%
dpa	3.16×10^{-2}	$\pm$ 16%

NOTE: Total Irradiation Time = 9.17 EFPY

TABLE 6-10

COMPARISON OF MEASURED AND FERRET CALCULATED
REACTION RATES AT THE SURVEILLANCE CAPSULE CENTER

Reaction	Measured	Adjusted Calculation	C/M
Cu-63 (n,α) Co-60	4.17×10^{-17}	4.18×10^{-17}	1.00
Fe-54 (n,p) Mn-54	4.05×10^{-15}	4.14×10^{-15}	1.02
Ni-58 (n,p) Co-58	6.03×10^{-15}	5.88×10^{-15}	0.97
Co-59 (n,γ) Co-60 (Cd)	1.39×10^{-12}	1.37×10^{-12}	0.99
Co-59 (n,γ) Co-60	2.79×10^{-12}	2.83×10^{-12}	1.01

TABLE 6-11

ADJUSTED NEUTRON ENERGY SPECTRUM AT THE SURVEILLANCE CAPSULE CENTER

Group	Energy (Mev)	Adjusted Flux (n/cm^2-sec)	Group	Energy (Mev)	Adjusted Flux (n/cm^2-sec)
1	$1.73x10^{1}$	$3.47x10^{6}$	28	$9.12x10^{-3}$	$8.92x10^{9}$
2	$1.49x10^{1}$	$8.50x10^{6}$	29	$5.53x10^{-3}$	$1.12x10^{10}$
3	$1.35x10^{1}$	$3.92x10^{7}$	30	$3.36x10^{-3}$	$3.48x10^{9}$
4	$1.16x10^{1}$	$9.98x10^{7}$	31	$2.84x10^{-3}$	$3.32x10^{9}$
5	$1.00x10^{1}$	$2.42x10^{8}$	32	$2.40x10^{-3}$	$3.22x10^{9}$
6	$8.61x10^{0}$	$4.40x10^{8}$	33	2.04x10-3	$9.37x10^{9}$
7	$7.41x10^{0}$	$1.07x10^{9}$	34	$1.23x10^{-3}$	$9.14x10^{9}$
8	$6.07x10^{0}$	$1.56x10^{9}$	35	$7.49x10^{-4}$	$8.88x10^{9}$
9	$4.97x10^{0}$	$3.23x10^{9}$	36	$4.54x10^{-4}$	$8.68x10^{9}$
10	$3.68x10^{0}$	$4.14x10^{9}$	37	$2.75x10^{-4}$	$9.20x10^{9}$
11	$2.87x10^{0}$	$8.31x10^{9}$	38	$1.67x10^{-4}$	$1.05x10^{10}$
12	$2.23X10^{0}$	$1.04x10^{10}$	39	$1.01X10^{-4}$	$9.91X10^{9}$
13	$1.74x10^{0}$	$1.37x10^{10}$	40	$6.14x10^{-5}$	$9.77x10^{9}$
14	$1.35x10^{0}$	$1.37x10^{10}$	41	$3.73x10^{-5}$	$9.47x10^{9}$
15	$1.11x10^{0}$	$2.34x10^{10}$	42	$2.26x10^{-5}$	$9.11x10^{9}$
16	$8.21x10^{-1}$	$2.48x10^{10}$	43	$1.37x10^{-5}$	$8.76x10^{9}$
17	$6.39x10^{-1}$	$2.43x10^{10}$	44	$8.32x10^{-6}$	$8.30x10^{9}$
18	$4.98x10^{-1}$	$1.71x10^{10}$	45	$5.04x10^{-6}$	$7.68x10^{9}$
19	$3.88x10^{-1}$	$2.25x10^{10}$	46	$3.06x10^{-6}$	$7.18x10^{9}$
20	$3.02x10^{-1}$	$2.48x10^{10}$	47	$1.86x10^{-6}$	$6.62x10^{9}$
21	$1.83x10^{-1}$	$2.34x10^{10}$	48	$1.13x10^{-6}$	$5.13x10^{9}$
22	$1.11x10^{-1}$	$1.84x10^{10}$	49	$6.83x10^{-7}$	$4.37x10^{9}$
23	$6.74x10^{-2}$	$1.32x10^{10}$	50	$4.14x10^{-7}$	$5.26x10^{9}$
24	$4.09x10^{-2}$	$7.94x10^{9}$	51	$2.51x10^{-7}$	$4.81x10^{9}$
25	$2.55x10^{-2}$	$9.49x10^{9}$	52	$1.52x10^{-7}$	$4.84x10^{9}$
26	$1.99x10^{-2}$	$5.15x10^{9}$	53	$9.24x10^{-8}$	$1.09x10^{10}$
27	$1.50x10^{-2}$	$6.95x10^{9}$			

NOTE: Tabulated energy levels represent the upper energy of each group.

TABLE 6-12

COMPARISON OF CALCULATED AND MEASURED
EXPOSURE LEVELS FOR SURVEILLANCE CAPSULE U

	Calculated	Measured	C/M
ϕ (E> 1.0 MeV) {n/cm^2}	1.76×10^{19}	1.88×10^{19}	0.94
ϕ (E> 0.1 MeV) {n/cm^2}	5.91×10^{19}	6.40×10^{19}	0.92
dpa/sec	3.01×10^{-2}	3.16×10^{-2}	0.95

TABLE 6-13

NEUTRON EXPOSURE PROJECTINS AT KEY LOCATIONS
ON THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

	AZIUTHAL ANGLE			
	0°	15°	30°	45°
9.17 EFPY				
ϕ(E> 1.0 MeV) (n/cm2)	1.96×10^{18}	3.04×10^{18}	3.54×10^{18}	5.46×10^{18}
ϕ(E> 0.1 MeV) (n/cm2)	5.02×10^{18}	7.79×10^{18}	9.30×10^{18}	1.48×10^{19}
dpa	3.16×10^{-3}	4.84×10^{-3}	5.68×10^{-3}	8.79×10^{-3}
23.0 EFPY				
ϕ(E> 1.0 MeV) (n/cm2)	4.56×10^{18}	6.83×10^{18}	7.14×10^{18}	1.05×10^{19}
ϕ(E> 0.1 MeV) (n/cm2)	1.15×10^{19}	1.73×10^{19}	1.85×10^{19}	2.86×10^{19}
dpa	7.42×10^{-3}	1.09×10^{-2}	1.16×10^{-2}	1.74×10^{-2}
32.0 EFPY				
ϕ(E> 1.0 MeV) (n/cm2)	6.26×10^{18}	9.30×10^{18}	9.48×10^{18}	1.41×10^{19}
ϕ(E> 0.1 MeV) (n/cm2)	1.58×10^{19}	2.35×10^{19}	2.46×10^{19}	3.76×10^{19}
dpa	1.02×10^{-2}	1.49×10^{-2}	1.53×10^{-2}	2.28×10^{-2}

TABLE 6-14

NEUTRON EXPOSURE VALUES FOR USE IN THE GENERATION OF HEATUP/COOLDONW CURVES

23 EFPY

	NEUTRON FLUENCE (E > 1.0 MeV) SLOPE (n/cm^2)			dpa SLOPE (equivalent n/cm^2)		
	Surface	1/4 T	3/4 T	Surface	1/4 T	3/4 T
0°	4.56×10^{18}	2.41×10^{18}	4.96×10^{17}	4.56×10^{18}	2.92×10^{18}	1.06×10^{18}
15°	6.83×10^{18}	3.64×10^{18}	7.51×10^{17}	6.83×10^{18}	4.37×10^{18}	1.57×10^{18}
30°	7.14×10^{18}	3.84×10^{18}	8.06×10^{17}	7.14×10^{18}	4.66×10^{18}	1.73×10^{18}
45°	1.05×10^{19}	5.63×10^{18}	1.12×10^{18}	1.07×10^{19}	6.84×10^{18}	2.36×10^{18}

32 EFPY

	NEUTRON FLUENCE (E > 1.0 MeV) SLOPE (n/cm^2)			dpa SLOPE (equivalent n/cm^2)		
	Surface	1/4 T	3/4 T	Surface	1/4 T	3/4 T
0°	6.26×10^{18}	3.31×10^{18}	6.82×10^{17}	6.26×10^{18}	4.00×10^{18}	1.45×10^{18}
15°	9.30×10^{18}	4.94×10^{18}	1.02×10^{18}	9.30×10^{18}	5.94×10^{18}	2.13×10^{18}
30°	9.48×10^{18}	5.09×10^{18}	1.07×10^{18}	9.48×10^{18}	6.18×10^{18}	2.29×10^{18}
45°	1.41×10^{19}	7.42×10^{18}	1.48×10^{18}	1.41×10^{19}	9.00×10^{18}	3.11×10^{18}

TABLE 6-15
UPDATED LEAD FACTORS FOR D. C COOK
UNIT 1 SURVEILLANCE CAPSULES

Capsule	Lead Factor
T	3.43
Y	3.43
X	3.43
U	3.43
V	0.96
S	0.96
Z	0.96
W	0.96

-. Plant specific evaluation integrated through cycle 10 (9.17 EFPY)

SECTION 7
SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following removal schedule meets ASTM E185-82 and is recommended for future capsules to be removed from the D. C. Cook Unit 1 reactor vessel:

Capsule	Vessel Location (deg)	Lead Factor	Removal Time(a)	Actual Capsule Fluence (n/cm^2)
T	40	3.43	1.13 (removed)	0.18×10^{19}
X	40	3.43	3.48 (removed)	0.77×10^{19} (b)
Y	40	3.43	4.94 (removed)	1.34×10^{19} (c)
U	40	3.43	9.17 (removed)	1.88×10^{19} (d)
V	4	0.96	Standby	--
S	4	0.96	Standby	--
Z	4	0.96	Standby	--
W	4	0.96	Standby	--

a) Effective full power years from plant startup
b) Approximate EOL (32 EFPY) fluence at 1/4T location (at 45°F) and corresponds to fluence value midway between 1st and 3rd capsule.
c) Approximate EOL (32 EFPY) fluence at vessel inner wall (at 45°)
d) Not less than once or greater than twice the estimated peak EOL (32 EFPY) vessel fluence.

SECTION 8
REFERENCES

1. Yanichko, S. E., and Lege, D. J., "American Electric Power Service Corp. Donald C. Cook Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-8047, March 1973.

2. "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1 Analysis of Capsule T," Final Report SWRI Project 02-4770, December 1977.

3. "Reactor Vessel Material Surveillance Program For Donald C. Cook Unit No. 1 Analysis of Capsule X," Final Report SWRI Project 02-6159, June 1981.

4. "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1 Analysis of Capsule Y," Final Report SWRI Project No. 06-7244-001, January 1984.

5. Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements" and Appendix H, "Reactor Vessel Material Surveillance Program Requirements", U. S. Nuclear Regulatory Commission, Washington, D.C.

6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May, 1988.

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

8. "ORNL RSCI Data Library Collection DLC-76, SAILOR Coupled Self-Shielded, 47 Neutron, 20 Gamma-Ray, P3, Cross Section Library for Light Water Reactors".

9. AEP Letter RBB 88-005/4, R. B. Bennett (AEP) to H. C. Wells (Westinghouse), January 29, 1988.

10. WCAP-10376, B. Y. Hubbard, et al., "Core Physics Characteristic of the Donald C. Cook Station Nuclear Plant (Unit 1, Cycle 8), July 1983.

11. WCAP-10862, F. J. Silva, et al., "Core Physics Characteristics of the Donald C. Cook Station Nuclear Plant (Unit 1, Cycle 9), August 1985

12. WCAP-11586, B. J. Johansen et al., "Nuclear Parameters and Operations Package for the Donald C. Cook Station Nuclear Plant (Unit 1, Cycle 10), October 1987

13. ASTM Designation E482-82, "Standard Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

14. ASTM Designation E560-77, "Standard Recommended Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

15. ASTM Designation E706-81a, "Standard Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standard", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

16. ASTM Designation E853-84, "Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

17. ASTM Designation E261-77, "Standard Method for Determining Neutron Flux, Fluence, and Spectra by Radioactivation Techniques", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

18. ASTM Designation E262-77, "Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

19. ASTM Designation E263-82, "Standard Method for Determining Fast-Neutron Flux Density by Radioactivation of Iron", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

20. ASTM Designation E264-82, "Standard Method for Determining Fast-Neutron Flux Density by Radioactivation of Nickel", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

21. ASTM Designation E481-78, "Standard Method for Measuring Neutron-Flux Density by Radioactivation of Cobalt and Silver", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

22. ASTM Designation E523-82, "Standard Method for Determining Fast-Neutron Flux Density by Radioactivation of Copper", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

23. ASTM Designation E704-84, "Standard Method for Measuring Reaction Rates by Radioactivation of Uranium-238", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

24. ASTM Designation E705-79, "Standard Method for Measuring Fast-Neutron Flux Density by Radioactivation of Neptunium-237", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

25. ASTM Designation E1005-84, "Standard Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance", in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1984.

26. F. A. Schmittroth, FERRET Data Analysis Core, HEDL-TME 79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

27. W. N. McElroy, S. Berg and T. Crocket, A Computer-Automated Iterative Method of Neutron Flux Spectra Determined by Foil Activation, AFWL-TR-67-41, Vol. I-IV, Air Force Weapons Laboratory, Kirkland AFB, NM, July 1967.

28. EPRI-NP-2188, "Development and Demonstration of an Advanced Methodology for LWR Dosimetry Applications", R. E. Maerker, et al., 1981.

29. ATSM Designation E185-70, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors".

30. ASTM Designation E208-87, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels".

31. ASTM.Designation E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706 (IF)".

32. ASTM Designation E23-86, "Standard Test Methods for Notched Bar Impact Testing of Metallic Materials".

33. ASTM Designation A370-88, "Standard Test Methods and Definitions for Mechanical Testing of Steel Products".

34. ASTM Designation E8-83, "Standard Test Methods of Tension Testing of Metallic Materials".

35. ASTM Designation E21-79, "Standard Practice for Elevated Temperature Tension Tests of Metallic Materials".

36. ASTM Designation E83-67, "Standard Practice for Verification and Classification of Extensometers".

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) for the reactor vessel. 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 fracture toughness properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either 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. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, 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 and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Rev. 2 (Radiation Embrittlement of Reactor Vessel Materials)[1]. The value, "f", given in figure A-1 is the calculated value of the neutron fluence at the location of interest (inner surface, 1/4T, or 3/4T) in the vessel at the location of the postulated defect, n/cm^2 ($E > 1$ MeV) divided by 10^{19}. The fluence factor is determined from figure A-1.

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 NRC Regulatory Standard Review Plan[2].

Fast neutron irradiation embrittlement of reactor vessel materials is largely dependent on the chemical composition, particularly the copper concentration. The variability in irradiation-induced property changes, which exist in general, is compounded by the variability of copper concentration within the weldments.

Westinghouse normally uses the method identified in Section C-1.1 of revision 2 of Regulatory Guide 1.99 [1] for determining best estimate values of copper and nickel content of reactor vessel welds and base metal using this method. The best estimate chemical content for welds is established from the mean of the measured values for weld samples made with the weld wire heat number that matches the critical vessel weld.

D. C. Cook Unit 1 Reactor Vessel weld is identical to Kewaunee and Maine Yankee vessel weld. Kewaunee recently submitted a report to the Nuclear Regulatory Commission about their modified Cu and Ni contents of their critical vessel weld. NRC evaluated and approved these revised values [3].

Since the Kewaunee vessel weld is identical to D. C. Cook Unit 1 vessel weld, the values (Cu, Ni and initial RT_{NDT}) will be used for the calculation of RT_{NDTS}. Cu, Ni content and initial RT_{NDT} for the D. C. Cook capsule (weld) is different than the actual vessel weld. These differences in material chemistry has been considered while developing chemistry factors using surveillance capsule data. This is as per recommendation of Regulatory Guide 1.99, Revision 2.

All material properties for the plates in the belt line region identified in Table A-1 are plant specific.

TABLE A-1

D. C. UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

Component	Plate No.	Cu (Wt/.)	Ni (Wt/.)	Initial RT_{NDT} (°F)
Intermediate Shell Plate	B4406-1	.12	.52	5
Intermediate Shell Plate	B4406-2	.15	.50	33
Intermediate Shell Plate	B4406-3	.15	.49	40
Lower Shell Plate	B4407-1	.14	.55	28
Lower Shell Plate	B4407-2	.12	.59	-12
Lower Shell Plate	B4407-3	.14	.50	38
Longitudinal Welds		.28[a]	.74[a]	-56[a]
Circumferential Welds		.28[a]	.74[a]	-56[a]
Closure Head Flange		---	---	60[b]
Vessel Flange		---	---	28[b]

(a) Reference (3)
(b) To be used for considering flange requirements for heatup/cooldown curves [5]

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[4]. The K_{IR} curve is given by the following equation:

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

where

K_{IR} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code[4] as follows:

$$C\, K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

where

K_{IM} = stress intensity factor caused by membrane (pressure) stress

K_{IT} = stress intensity factor caused by the thermal gradients

K_{IR} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak 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 the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From equation 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 reference flaw of Appendix G to the ASME Code 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 the 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 1/4 T 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 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so 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 1/4 T location and, 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 ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup 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 1/4 T defect 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 1/4 T crack during heatup is lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same time coolant temperature. During heatup, especially at the end of the transient, conditions may exist so 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 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to ensure 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 the pressure-temperature limitations for the case in which a 1/4 T 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 therefore 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 by constructing a composite curve 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 wherein, over the course of the heatup 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 1983 Amendment to 10CFR50[5] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure.

Table A-1 indicates that the limiting RT_{NDT} of 60°F occurs in the closure head flange of D. C. Cook Unit 1, so the minimum allowable temperature of this region is 180°F. These limits are shown on Figures A-2 through A9, whenever applicable.

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 section A-3. Figures A2 through A5 are applicable for the first 23 EFPY and Figures A6 through A9 are applicable for the first 32 EFPY. Instrumentation error margin of 10°F and 60 psig are included in developing heatup and cooldown curves for Figures A2, A3, A6 and A7. No instrumentation error margins are considered for Figures A4, A5, A8 and A9.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in figures A-2 through A-9. This is in addition to other criteria which must be met before the reactor is made critical.

The leak limit curve shown in figure A-2, A-4, A-6, and A-8 represent minimum temperature requirements at the leak test pressure specified by applicable codes[2,4]. The leak test limit curve was determined by methods of references 2 and 5.

Figures A-2 through A-9 define limits for ensuring prevention of nonductile failure for the D. C. Cook Unit 1 Primary Reactor Coolant System.

A-5. ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99 Rev. 2 [1] the adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (3)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = [CF]f^{(0.28-0.10 \log f)} \quad (4)$$

f = Neutron fluence, n/cm^2 (E > 1 MeV), divided by 10^{19}

CF = Chemistry factor from tables for welds and for base metal (plates and forgings) (if no data use 0.35% Cu and 1.0% Ni)

$$\text{Margin} = 2\,[\sigma_I^{\,2} + \sigma_\Delta^{\,2}]^{0.5} \text{ where,} \tag{5}$$

σ_I = standard deviation of initial RT_{NDT}. If the initial RT_{NDT} is measured, σ_I is to be estimated from the precision of the test method; otherwise, σ_I is obtained from the same set of data that is used to get initial RT_{NDT}

σ_Δ = Standard deviation of ΔRT_{NDT}; 28°F for welds and 17°F for base metal

[σ_Δ need not exceed 1/2 times ΔRT_{NDT}]

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(\text{depth } X)} = f_{\text{surface}}{}^{(e^{-.24x})} \tag{5}$$

where x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface. The resultant fluence is then put into equation (4) to calculate ΔRT_{NDT} at the specific depth.

CF (°F) is the chemistry factor, obtained using the methods given in reference 1. The Adjusted Reference Temperatures (ART) for all belt line region materials for D. C. Cook Unit 1 are provided in Table A-2 for 23 and 32 EFPYS. Samples of calculations for ART are shown in Tables A-3 and A-4. From Table A-2 the limiting material is found to be the intermediate plate B4406-3.

TABLE A-2

SUMMARY OF ADJUSTED REFERENCE TEMPERATURES (ART)
AT 1/4T and 3/4T LOCATION

Component	23 EFPY RT_{NDT} At		32 EFPY RT_{NDT} at	
	1/4T (°F)	3/4T (°F)	1/4T (°F)	3/4T (°F)
Intermediate Plate, B4406-1	110	88	116	94
Intermediate Plate, B4406-2	159	131	167	138
Intermediate Plate, B4406-3	165(161)*	137(131)*	173(171)**	145(138)**
Lower Plate, B4407-1	148	121	156	129
Lower Plate, B4407-2	94	72	101	78
Lower Plate, B4407-3	156	130	164	137
Longitudinal Weld	169(130)	117(83)	186(144)	131(95)
Circumferential Weld	192(149)	136(100)	209(164)	151(114)

Number within ( ) are using chemistry factor based on surveillance capsule data.

* These RT_{NDT} numbers used to generate heatup and cooldown curves applicable up to 23 EFPY

** These RT_{NDT} numbers used to generate heatup and cooldown curves applicable up to 32 EFPY

TABLE A-3

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR D. C. COOK UNIT 1 REACTOR VESSEL MATERIAL - CIRCUMFERENTIAL WELD

Parameter	Regulatory Guide 1.99 - Revision 2			
	23 EFPY		32 EFPY	
	1/4 T	3/4 T	1/4 T	3/4 T
Chemistry Factor, CF (°F)	208.7 (184.7)	208.7 (184.7)	208.7 (184.7)	208.7 (184.7)
Fluence, f (10^{19} n/cm^2)[(a)]	.634	.232	.852	.311
Fluence Factor, ff	.872	.605	.955	.679
******				
ΔRT_{NDT} = CF x ff (°F)	182.1 (161.2)	126.2 (111.8)	199.3 (176.4)	141.8 (125.5)
Initial RT_{NDT}, I (°F) [(b)]	-56	-56	-56	-56
Margin, M (°F) [(c)]	65.5 (44)	65.5 (44)	65.5 (44)	65.5 (44)
******				
Revision 2 to Regulatory Guide 1.99				
Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	192 (149)	136 (100)	209 (164)	151 (114)

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 MeV) = 1.05 at 23 EFPY, and 1.41 at 32 EFPY [Table 6-14 of the text]. The D. C. Cook Unit 1 reactor vessel wall thickness is 8.5 inches at the beltline region.

(b) The initial RT_{NDT} (I) value for the weld is a generic value (from Table A-1).

(c) Margin is calculated as, $M = 2\,[\sigma_1^2 + \sigma_\Delta^2]^{0.5}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is assumed to be 17°F since the initial RT_{NDT} is a generic mean value. The standard deviation for ΔRT_{NDT}, ($\sigma\Delta$) is 28°F for the weld. σ_Δ is 14°F for weld (cut into half) when surveillance data is used.

( ) numbers in parenthesis were calculated using surveillance capsule data.

TABLE A-4

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR D. C. COOK UNIT 1 REACTOR VESSEL MATERIAL - INTERMEDIATE PLATE, B4406-3

Parameter	Regulatory Guide 1.99 - Revision 2			
	23 EFPY		32 EFPY	
	1/4 T	3/4 T	1/4 T	3/4 T
Chemistry Factor, CF (°F)	104 (119.55)	104 (119.55)	104 (119.55)	104 (119.55)
Fluence, f (10^{19} n/cm^2)[(a)]	.634	.236	.852	.311
Fluence Factor, ff	.872	.609	.955	.679

****

ΔRT_{NDT} = CF x ff (°F)	90.7 (104.3)	63.4 (72.8)	99.3 (114.2)	70.7 (81.2)
Initial RT_{NDT}, I (°F) [(b)]	40	40	40	40
Margin, M (°F) [(c)]	34 (17)	34 (17)	34 (17)	34 (17)

****

Revision 2 to Regulatory Guide 1.99

Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	165 (161)	137 (131)	173 (171)	145 (138)

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 MeV) = 1.05 at 23 EFPY, and 1.41 at 32 EFPY [Table 6-14 of the text]. The D. C. Cook Unit 1 reactor vessel wall thickness is 8.5 inches at the beltline region.

(b) The initial RT_{NDT} (I) value for the plate is a measured value (from Table A-1).

(c) Margin is calculated as, $M = 2\,[\sigma_1^{\,2} + \sigma_\Delta^{\,2}]^{0.5}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is assumed to be 0°F since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT}, ($\sigma\Delta$) is 17°F for the plate. σ_Δ is 8.5°F for the plate (cut into half) when surveillance data is used.

( ) numbers in parenthesis were calculated using surveillance capsule data.

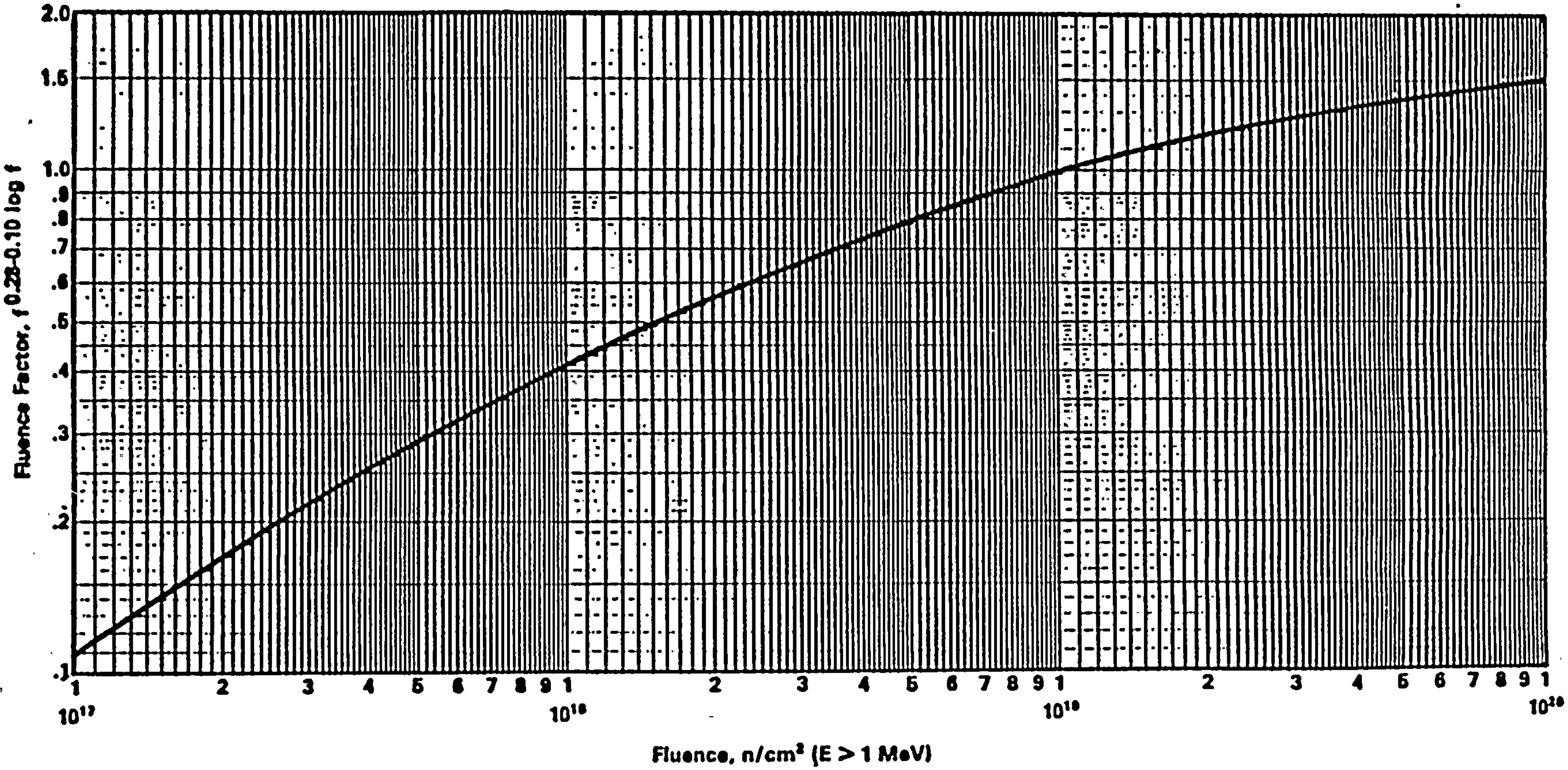


Figure A-1. Fluence Factor for Use in the Expression for ΔRT_{NDT}

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE PLATE, B4406-3

INITIAL RT_{NDT}: 40°F

RT_{NDT} AFTER 23 EFPY: 1/4T, 161°F

3/4T, 131°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 23 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

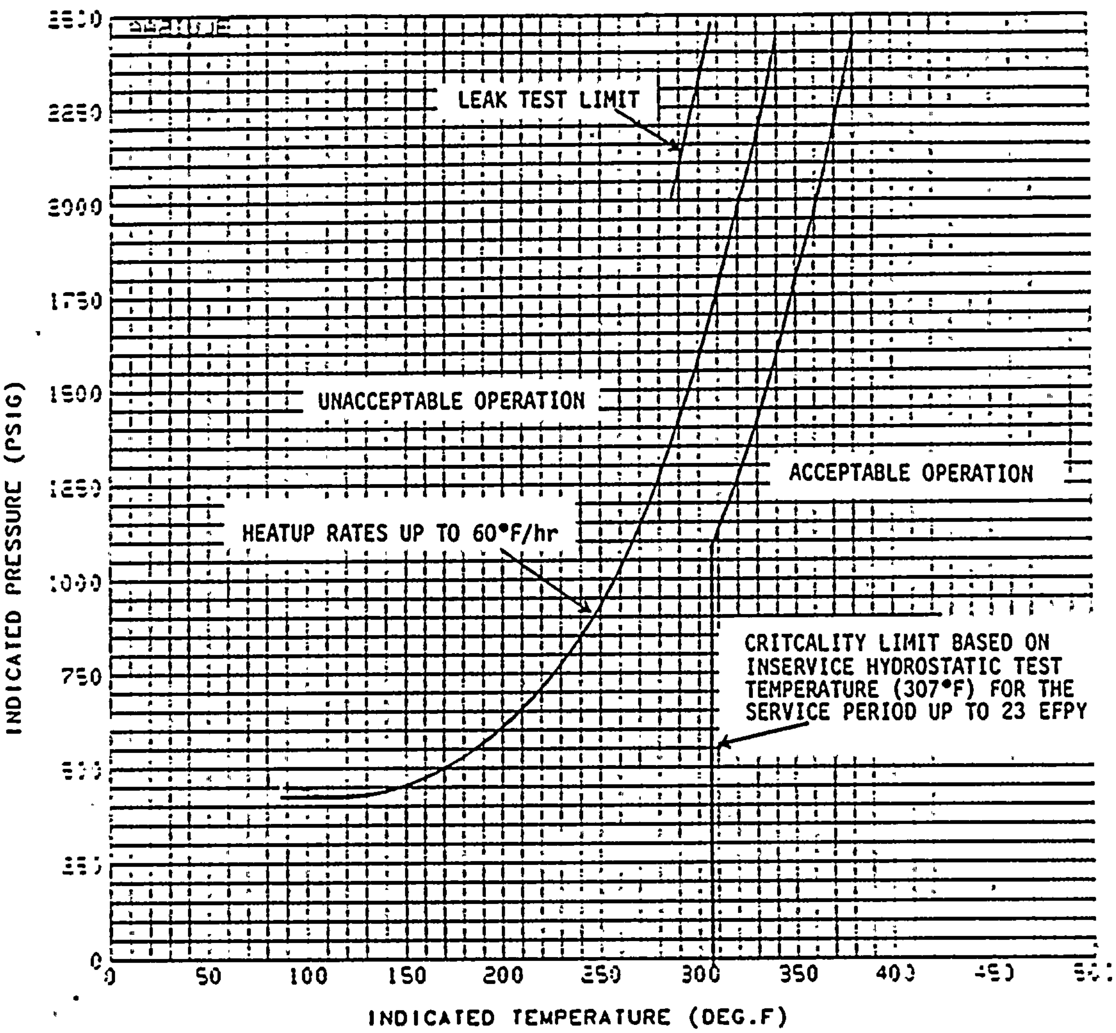


Figure A-2. D. C. Cook Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 23 EFPY (With Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE PLATE, B4406-3
INITIAL RT_{NDT}: 40°F
RT_{NDT} AFTER 23 EFPY: 1/4T, 161°F
3/4T, 131°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 23 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

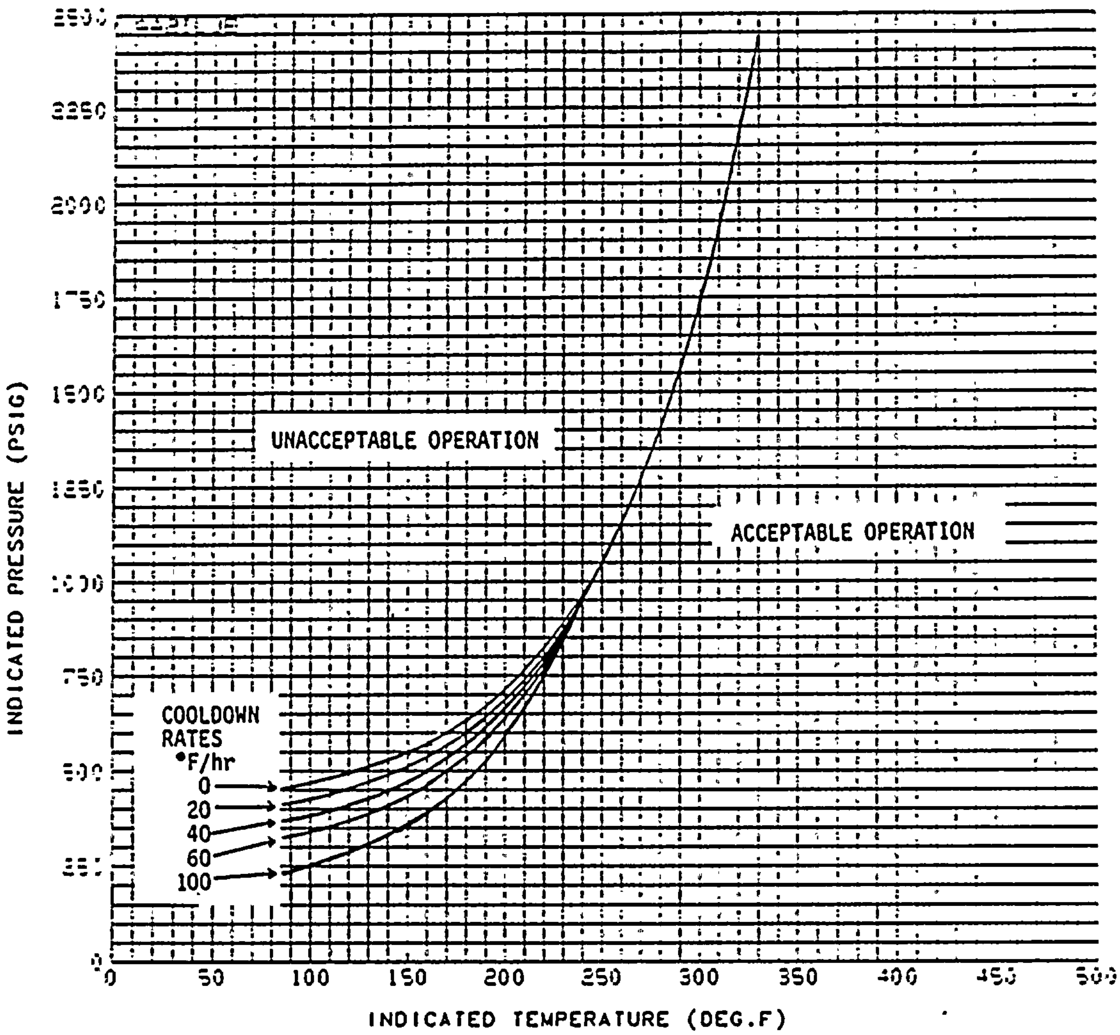


Figure A-3. D. C. Cook Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 23 EFPY (With Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE PLATE, B4406-3

INITIAL RT_{NDT}: 40°F

RT_{NDT} AFTER 23 EFPY: 1/4T, 161°F

3/4T, 131°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 23 EFPY. CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

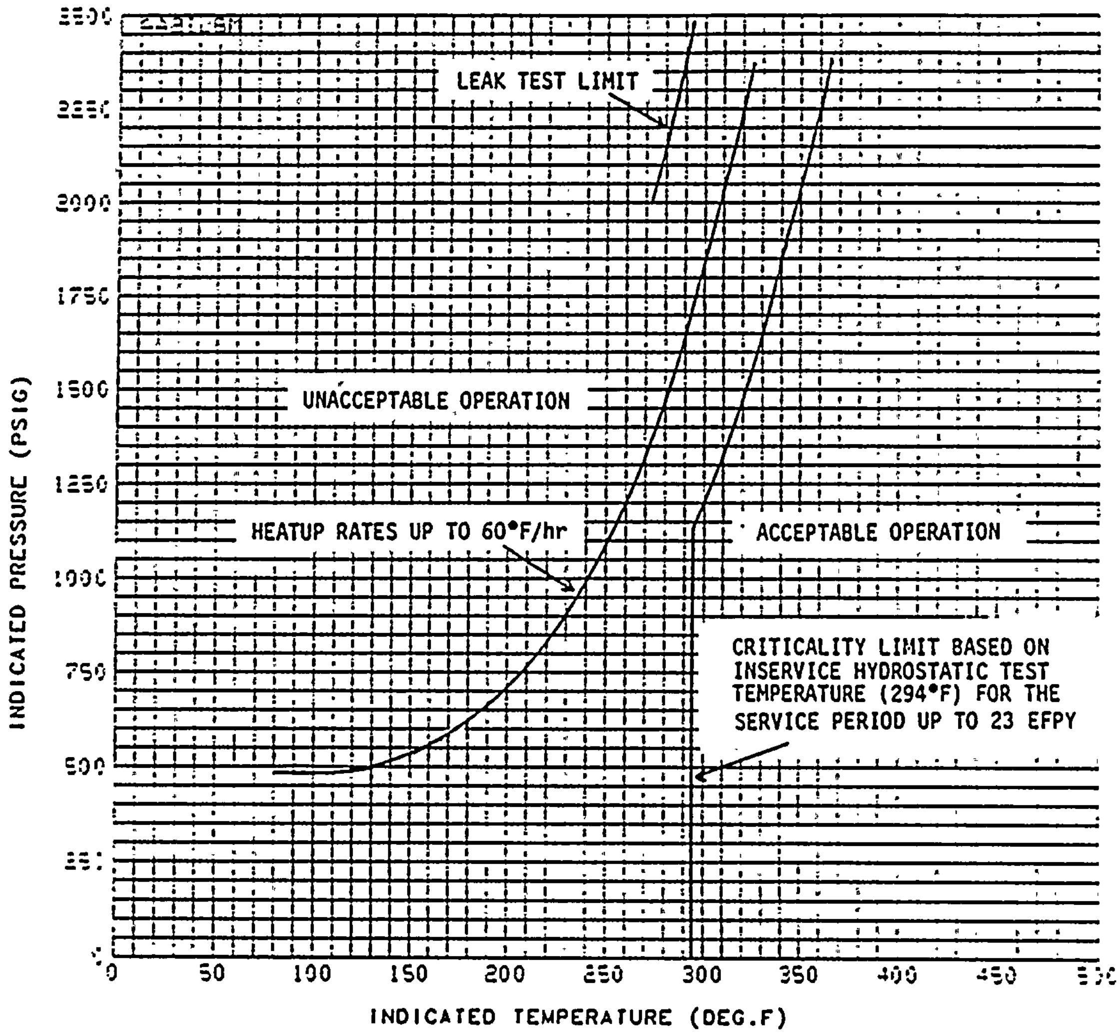


Figure A-4. D. C. Cook Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 23 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE PLATE, B4406-3
INITIAL RT_{NDT}: 40°F
RT_{NDT} AFTER 23 EFPY: 1/4T, 161°F
3/4T, 131°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 23 EFPY. CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

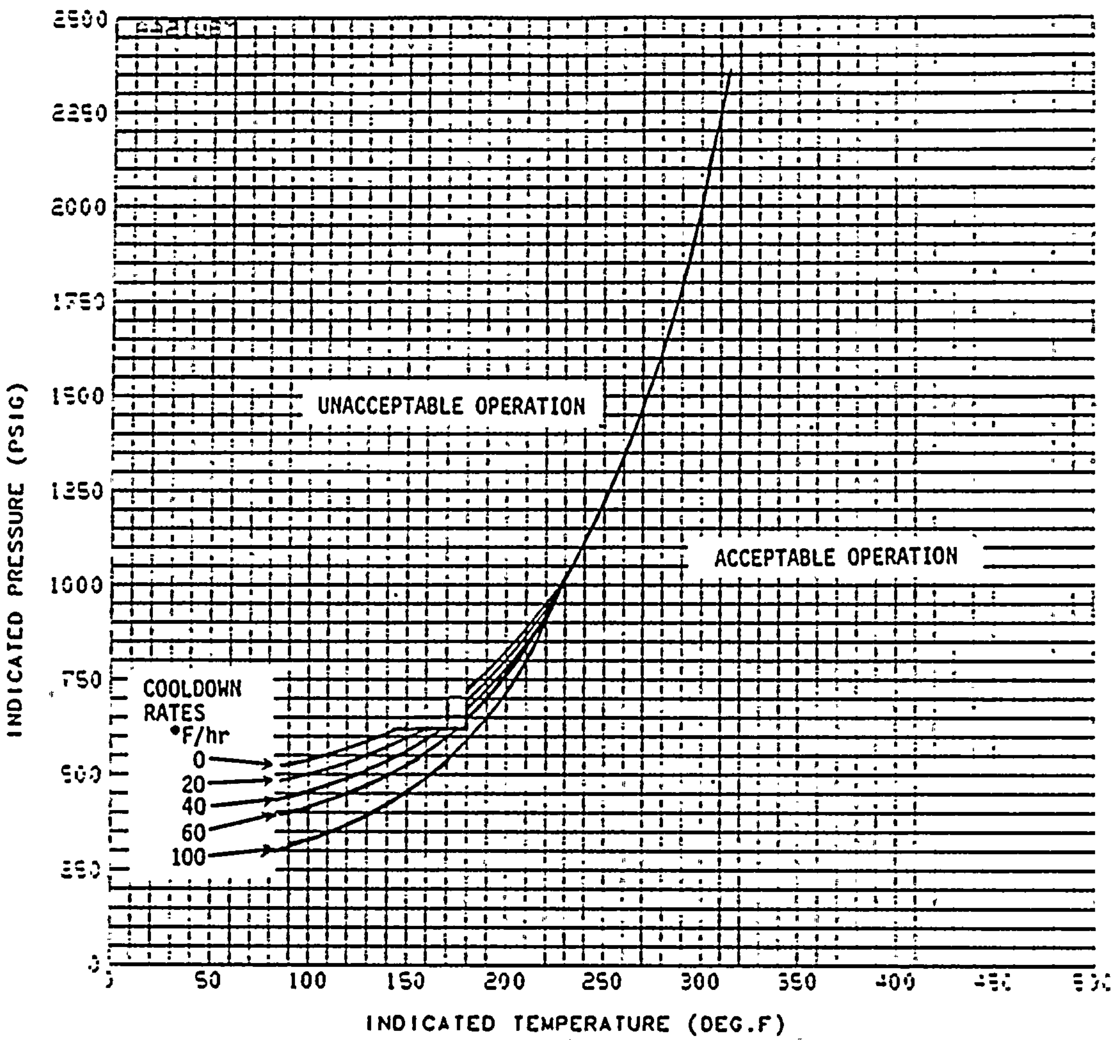


Figure A-5. D. C. Cook Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 23 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE PLATE, B4406-3

INITIAL RT_{NDT}: 40°F

RT_{NDT} AFTER 32 EFPY: 1/4T, 171°F

3/4T, 138°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 32 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

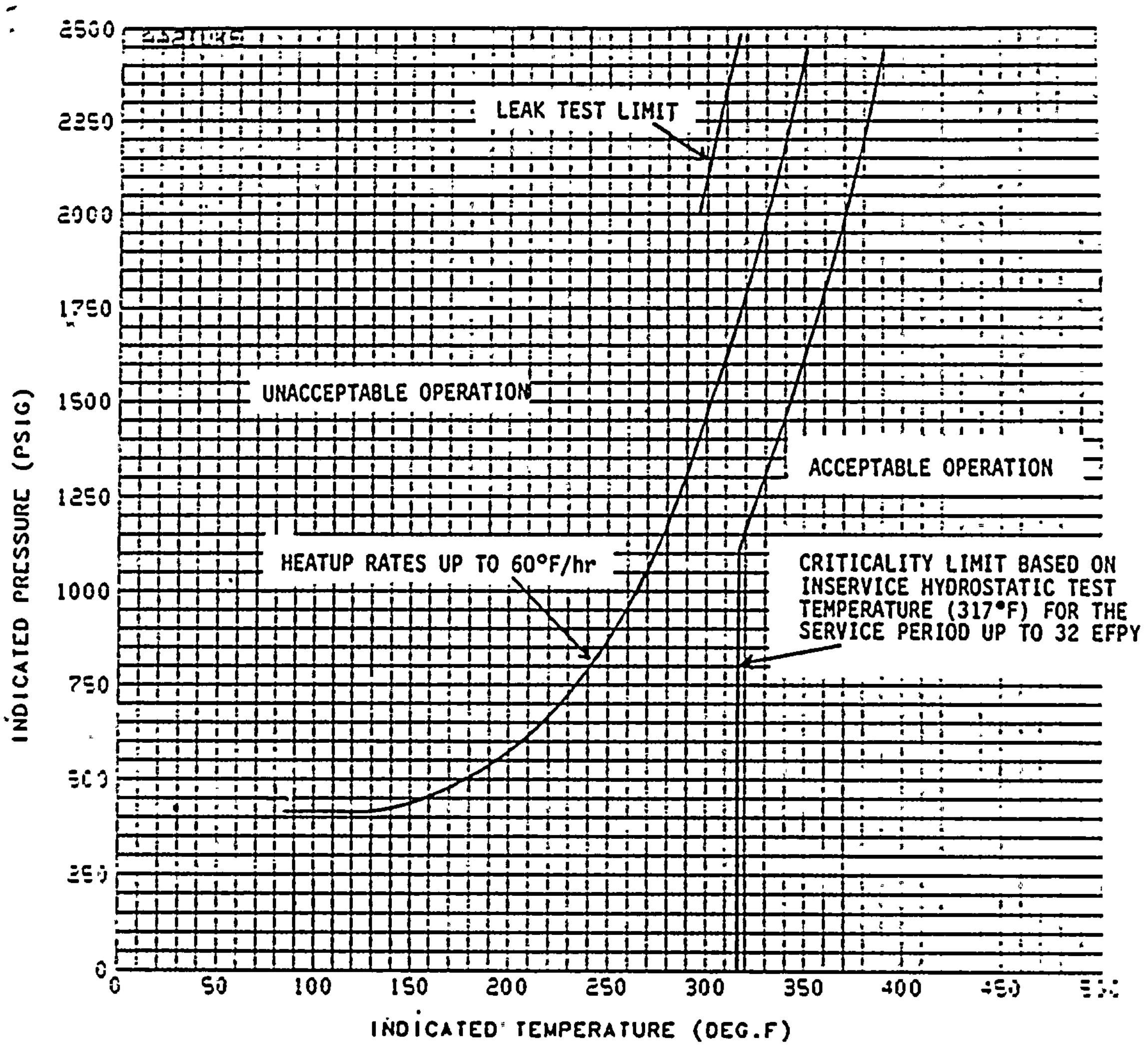


Figure A-6. D. C. Cook Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 32 EFPY (With Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE PLATE, B4406-3

INITIAL RT_{NDT}: 40°F

RT_{NDT} AFTER 32 EFPY: 1/4T, 171°F

3/4T, 138°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 32 EFPY. CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

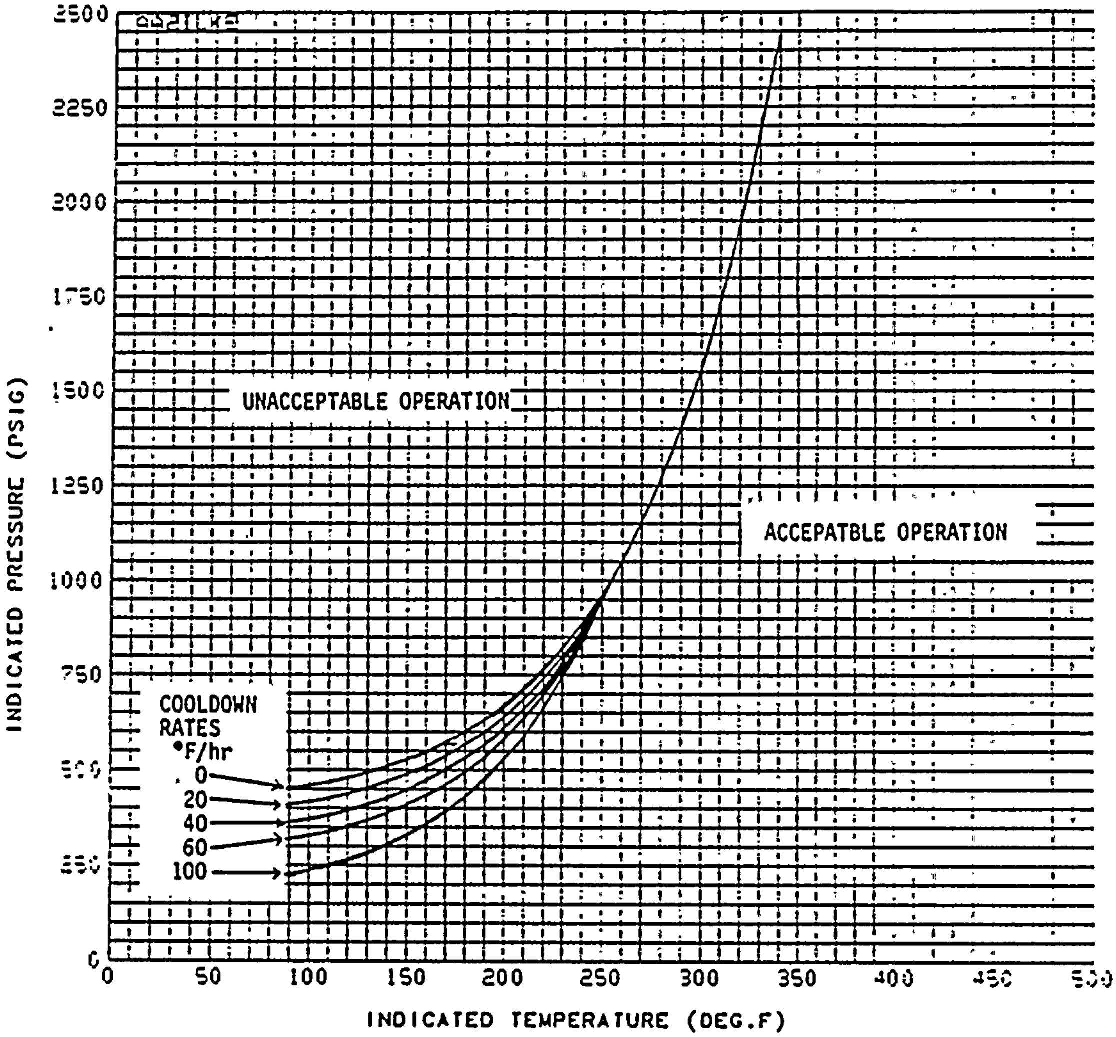


Figure A-7. D. C. Cook Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 32 EFPY (With Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE PLATE, B4406-3

INITIAL RT_{NDT}: 40°F

RT_{NDT} AFTER 32 EFPY: 1/4T, 171°F

3/4T, 138°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 32 EFPY. CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

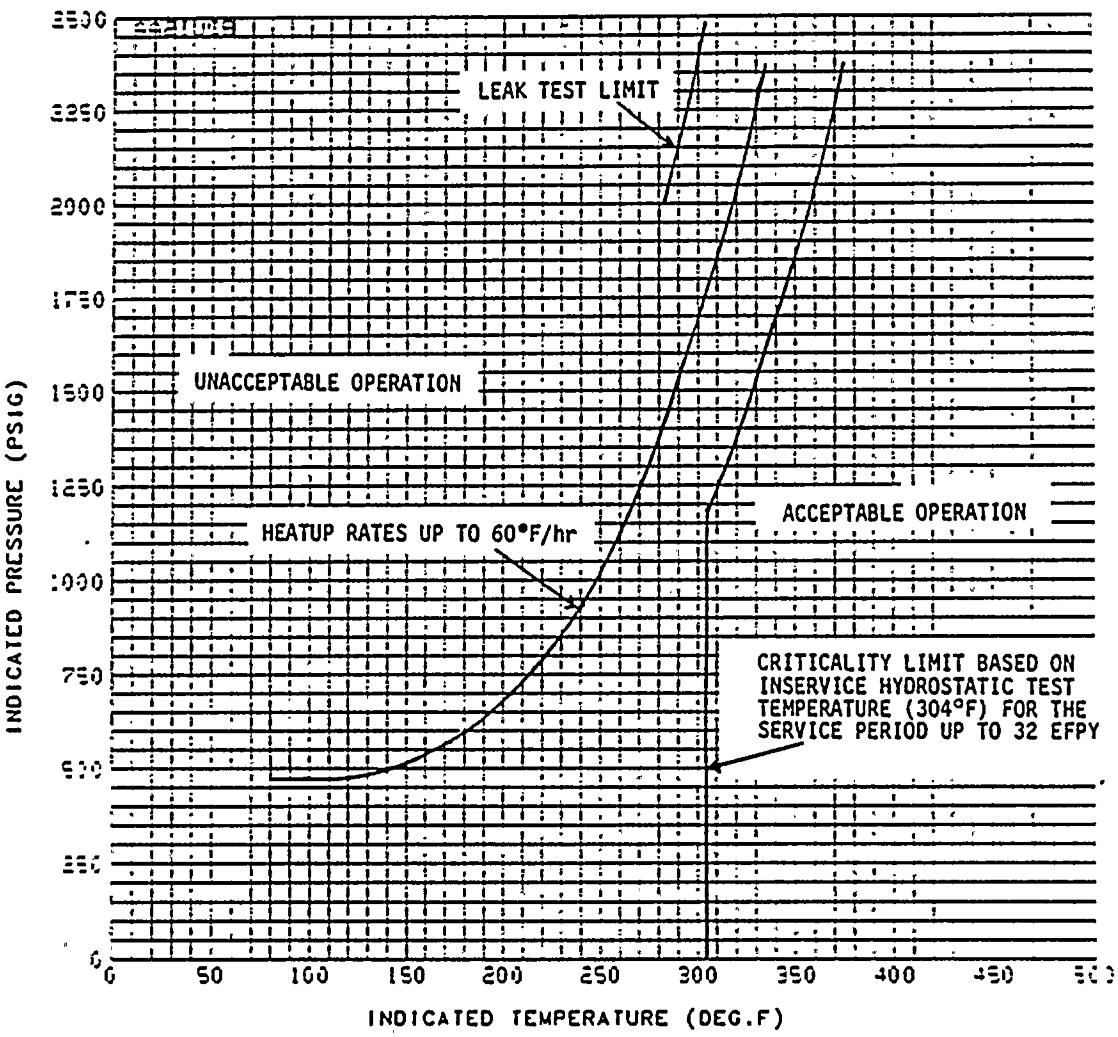


Figure A-8. D. C. Cook Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE PLATE, B4406-3

INITIAL RT_{NDT}: 40°F

RT_{NDT} AFTER 32 EFPY: 1/4T, 171°F

3/4T, 138°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 32 EFPY. CONTAINS NO MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

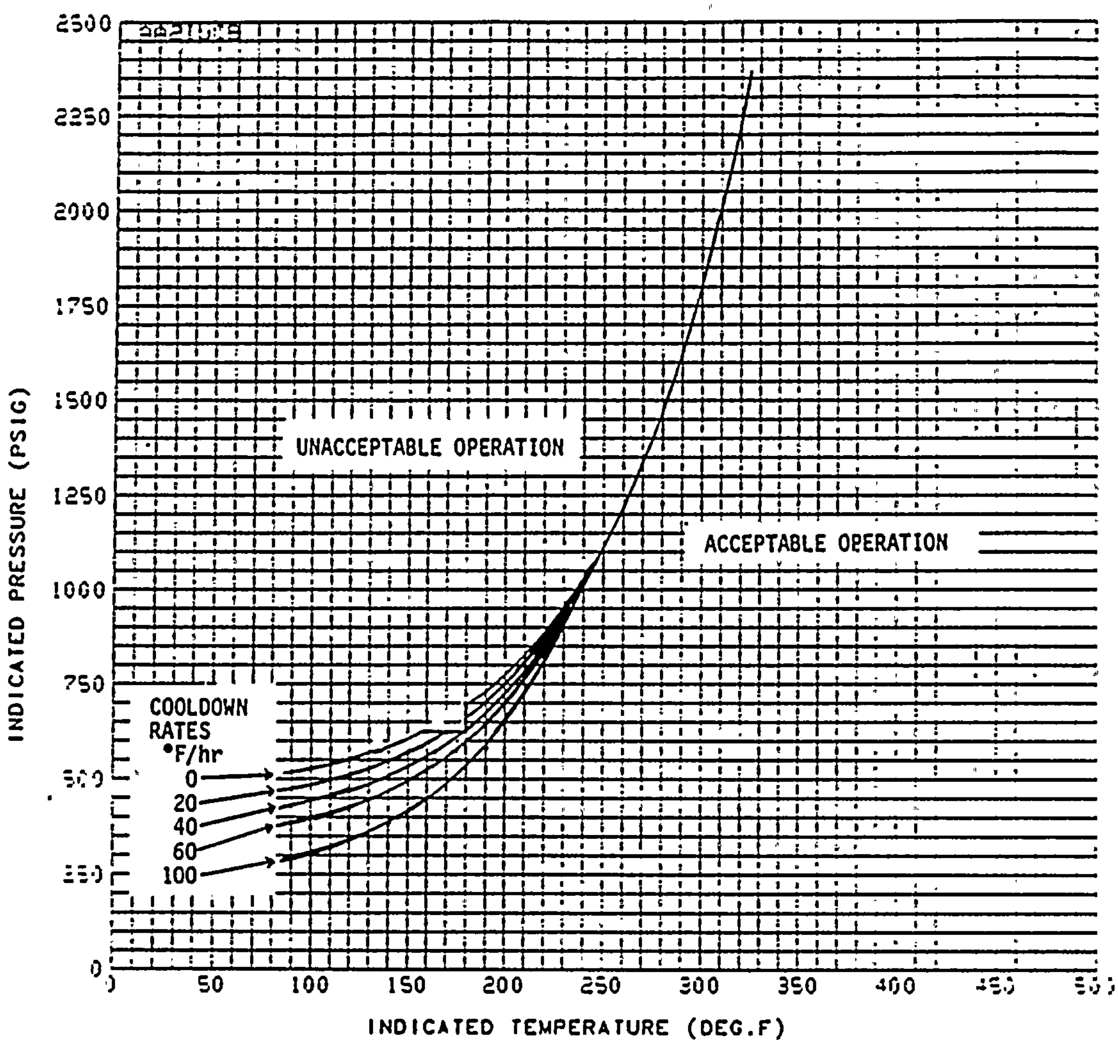


Figure A-9. D. C. Cook Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

A-6. REFERENCES

[1] Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May, 1988.

[2] "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-800, 1981.

[3] Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to Fast Neutron Fluence for Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, 10CFR 50.61, Wisconsin Public Service Corporation Kewaunee Nuclear Power Station Dockett No. 50--305.

[4] ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendixes, "Rules for Construction of Nuclear Power Plan Components, Appendix G, Protection Against Nonductile Failure," pp. 558-563, 1986 Edition, American Society of Mechanical Engineers, New York, 1986.

[5] Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 48 No. 104, May 27, 1983.

A-7. DATA POINTS FOR HEATUP AND COOLDOWN CURVES

Data Points for Heatup and Cooldown
Curves for up to 23 EFPY and
With Margins for Instrumentation Error

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	454.83	18	170.000	601.72	35	255.000	1101.74
2	90.000	459.49	19	175.000	617.30	36	260.000	1154.52
3	95.000	464.39	20	180.000	~~633.95~~ 621*	37	265.000	1211.18
4	100.000	469.65	21	185.000	651.99	38	270.000	1271.92
5	105.000	475.32	22	190.000	671.20	39	275.000	1336.95
6	110.000	481.30	23	195.000	692.05	40	280.000	1407.01
7	115.000	487.85	24	200.000	714.40	41	285.000	1481.90
8	120.000	494.89	25	205.000	738.34	42	290.000	1562.30
9	125.000	502.45	26	210.000	764.15	43	295.000	1648.49
10	130.000	510.59	27	215.000	791.84	44	300.000	1740.74
11	135.000	519.34	28	220.000	821.50	45	305.000	1839.56
12	140.000	528.62	29	225.000	853.59	46	310.000	1945.08
13	145.000	538.72	30	230.000	887.92	47	315.000	2058.16
14	150.000	549.59	31	235.000	924.79	48	320.000	2178.95
15	155.000	561.28	32	240.000	964.38	49	325.000	2308.09
16	160.000	573.70	33	245.000	1006.91	50	330.000	2445.60
17	165.000	587.20	34	250.000	1052.62			

* FLANGE REQUIREMENT

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	412.97	13	145.000	500.58	24	200.000	687.46
2	90.000	417.65	14	150.000	512.05	25	205.000	713.19
3	95.000	422.71	15	155.000	524.41	26	210.000	740.75
4	100.000	428.15	16	160.000	537.60	27	215.000	770.35
5	105.000	433.95	17	165.000	551.94	28	220.000	802.38
6	110.000	440.27	18	170.000	567.34	29	225.000	836.69
7	115.000	447.10	19	175.000	583.82	30	230.000	873.50
8	120.000	454.45	20	180.000	601.68	31	235.000	913.11
9	125.000	462.38	21	185.000	620.89	32	240.000	955.68
10	130.000	470.91	22	190.000	641.43	33	245.000	1001.47
11	135.000	480.01	23	195.000	663.67	34	250.000	1050.68
12	140.000	489.90						

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	370.12	13	145.000	462.04	24	200.000	661.54
2	90.000	374.94	14	150.000	474.20	25	205.000	689.02
3	95.000	380.12	15	155.000	487.22	26	210.000	718.54
4	100.000	385.76	16	160.000	501.36	27	215.000	750.57
5	105.000	391.89	17	165.000	516.62	28	220.000	784.82
6	110.000	398.49	18	170.000	532.91	29	225.000	821.68
7	115.000	405.65	19	175.000	550.63	30	230.000	861.31
8	120.000	413.35	20	180.000	569.67	31	235.000	904.19
9	125.000	421.69	21	185.000	590.10	32	240.000	950.09
10	130.000	430.66	22	190.000	612.20	33	245.000	999.46
11	135.000	440.29	23	195.000	635.90	34	250.000	1052.35
12	140.000	450.74						

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	326.31	12	140.000	411.11	23	195.000	609.13
2	90.000	331.31	13	145.000	423.11	24	200.000	636.52
3	95.000	336.77	14	150.000	435.95	25	205.000	666.21
4	100.000	342.66	15	155.000	449.93	26	210.000	698.04
5	105.000	349.01	16	160.000	464.99	27	215.000	732.33
6	110.000	355.93	17	165.000	481.17	28	220.000	769.21
7	115.000	363.44	18	170.000	498.71	29	225.000	809.13
8	120.000	371.55	19	175.000	517.66	30	230.000	851.94
9	125.000	380.27	20	180.000	537.93	31	235.000	898.03
10	130.000	389.76	21	185.000	559.96	32	240.000	947.58
11	135.000	400.04	22	190.000	583.53	33	245.000	1000.73

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	235.61	12	140.000	330.55	23	195.000	568.86
2	90.000	241.06	13	145.000	344.21	24	200.000	590.75
3	95.000	247.04	14	150.000	358.91	25	205.000	625.18
4	100.000	253.53	15	155.000	374.91	26	210.000	662.44
5	105.000	260.63	16	160.000	392.11	27	215.000	702.54
6	110.000	268.29	17	165.000	410.83	28	220.000	745.69
7	115.000	276.68	18	170.000	431.02	29	225.000	792.25
8	120.000	285.77	19	175.000	452.79	30	230.000	842.35
9	125.000	295.67	20	180.000	476.37	31	235.000	896.34
10	130.000	306.33	21	185.000	501.77	32	240.000	954.27
11	135.000	317.97	22	190.000	529.14			

AEP60 F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (23.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	286
2485	307

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	22088	92529
2485	27288	115366

AEP60 F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~454.93~~	19	175.000	515.05	36	260.000	1028.28
2	90.000	~~454.71~~	20	180.000	530.40	37	265.000	1082.53
3	95.000	~~442.56~~	21	185.000	547.22	38	270.000	1140.71
4	100.000	~~433.68~~ 423.15	22	190.000	565.41	39	275.000	1203.10
5	105.000	~~427.05~~	23	195.000	584.96	40	280.000	1270.04
6	110.000	~~424.43~~	24	200.000	606.21	41	285.000	1341.72
7	115.000	423.15	25	205.000	628.97	42	290.000	1418.60
8	120.000	423.57	26	210.000	653.67	43	295.000	1500.87
9	125.000	425.71	27	215.000	680.09	44	300.000	1588.97
10	130.000	429.21	28	220.000	708.68	45	305.000	1683.27
11	135.000	434.03	29	225.000	739.32	46	310.000	1784.05
12	140.000	440.10	30	230.000	772.20	47	315.000	1885.89
13	145.000	447.42	31	235.000	807.76	48	320.000	1983.07
14	150.000	455.83	32	240.000	845.79	49	325.000	2086.75
15	155.000	465.41	33	245.000	886.67	50	330.000	2197.61
16	160.000	476.08	34	250.000	930.58	51	335.000	2315.77
17	165.000	487.83	35	255.000	977.71	52	340.000	2442.19
18	170.000	500.82						

Data Points for Heatup and Cooldown
Curves for up to 23 EFPY and
Without Margins for Instrumentation Error

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	524.39	17	165.000	~~677.30~~	33	245.000	1161.74
2	90.000	529.65	18	170.000	~~693.95~~	34	250.000	1214.52
3	95.000	535.32	19	175.000	~~711.99~~	35	255.000	1271.18
4	100.000	541.30	20	180.000	~~731.20~~	36	260.000	1331.92
5	105.000	547.86	21	185.000	752.05	37	265.000	1396.95
6	110.000	554.89	22	190.000	774.40	38	270.000	1467.01
7	115.000	562.45	23	195.000	798.34	39	275.000	1541.90
8	120.000	570.59	24	200.000	824.15	40	280.000	1622.30
9	125.000	579.34	25	205.000	851.84	41	285.000	1708.49
10	130.000	588.62	26	210.000	881.50	42	290.000	1800.74
11	135.000	598.72	27	215.000	913.59	43	295.000	1899.56
12	140.000	609.59	28	220.000	947.92	44	300.000	2005.08
13	145.000	~~621.28~~	29	225.000	984.79	45	305.000	2118.16
14	150.000	~~633.70~~	30	230.000	1024.38	46	310.000	2238.95
15	155.000	~~647.20~~	31	235.000	1066.91	47	315.000	2368.09
16	160.000	~~661.72~~	32	240.000	1112.62			

* 621 PSI. FLANGE REQUIREMENT

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	482.71	12	140.000	572.05	23	195.000	773.19
2	90.000	488.15	13	145.000	584.41	24	200.000	800.75
3	95.000	493.85	14	150.000	597.60	25	205.000	830.35
4	100.000	500.27	15	155.000	611.94	26	210.000	862.38
5	105.000	507.10	16	160.000	~~627.34~~	27	215.000	896.69
6	110.000	514.45	17	165.000	~~643.82~~ *	28	220.000	933.50
7	115.000	522.38	18	170.000	~~661.68~~	29	225.000	973.11
8	120.000	530.91	19	175.000	~~680.89~~	30	230.000	1015.68
9	125.000	540.01	20	180.000	~~701.42~~	31	235.000	1061.47
10	130.000	549.90	21	185.000	723.67	32	240.000	1110.68
11	135.000	560.58	22	190.000	747.46			

* 621 PSI, FLANGE REQUIREMENT

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	440.12
2	90.000	445.76
3	95.000	451.89
4	100.000	458.49
5	105.000	465.65
6	110.000	473.35
7	115.000	481.69
8	120.000	490.66
9	125.000	500.29
10	130.000	510.74
11	135.000	522.04
12	140.000	534.20
13	145.000	547.22
14	150.000	561.36
15	155.000	576.62
16	160.000	592.91
17	165.000	610.63
18	170.000	~~629.87~~ *
19	175.000	~~650.10~~ *
20	180.000	~~672.20~~ *
21	185.000	695.90
22	190.000	721.54
23	195.000	749.02
24	200.000	778.54
25	205.000	810.67
26	210.000	844.82
27	215.000	881.68
28	220.000	921.31
29	225.000	964.19
30	230.000	1010.09
31	235.000	1059.46
32	240.000	1112.35

* FLANGE REQUIREMENT

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	396.77	12	140.000	495.95	22	190.000	696.52
2	90.000	402.66	13	145.000	509.93	23	195.000	726.21
3	95.000	409.01	14	150.000	524.99	24	200.000	758.04
4	100.000	415.93	15	155.000	541.17	25	205.000	792.33
5	105.000	423.44	16	160.000	558.71	26	210.000	829.21
6	110.000	431.55	17	165.000	577.66	27	215.000	869.13
7	115.000	440.27	18	170.000	597.93	28	220.000	911.94
8	120.000	449.76	19	175.000	619.96	29	225.000	958.03
9	125.000	460.04	20	180.000	~~642.52~~ *	30	230.000	1007.58
10	130.000	471.11	21	185.000	669.13	31	235.000	1060.73
11	135.000	483.11						

*621 PSI, FLANGE REQUIREMENT

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	307.04	11	135.000	404.21	21	185.000	618.86
2	90.000	313.53	12	140.000	418.91	22	190.000	650.75
3	95.000	320.63	13	145.000	434.91	23	195.000	685.18
4	100.000	328.29	14	150.000	452.11	24	200.000	722.44
5	105.000	336.68	15	155.000	470.83	25	205.000	762.54
6	110.000	345.77	16	160.000	491.02	26	210.000	805.69
7	115.000	355.67	17	165.000	512.79	27	215.000	852.25
8	120.000	366.33	18	170.000	536.37	28	220.000	902.35
9	125.000	377.97	19	175.000	561.77	29	225.000	956.34
10	130.000	390.55	20	180.000	589.14	30	230.000	1014.27

AEP60 F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (23.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	273
2485	294

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	21444	89745
2485	26645	112605

AEP60 F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 01/12/90

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 23.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~503.56~~
2	90.000	~~493.68~~
3	95.000	~~487.05~~ 483.15
4	100.000	~~484.43~~
5	105.000	483.15
6	110.000	483.57
7	115.000	485.71
8	120.000	489.21
9	125.000	494.03
10	130.000	500.10
11	135.000	507.42
12	140.000	515.83
13	145.000	525.41
14	150.000	536.08
15	155.000	547.83
16	160.000	560.82
17	165.000	575.05
18	170.000	590.40
19	175.000	607.22
20	180.000	~~625.41~~ *
21	185.000	644.96
22	190.000	666.21
23	195.000	688.97
24	200.000	713.67
25	205.000	740.09
26	210.000	768.68
27	215.000	799.32
28	220.000	832.20
29	225.000	867.76
30	230.000	905.79
31	235.000	946.67
32	240.000	990.58
33	245.000	1037.71
34	250.000	1088.28
35	255.000	1142.53
36	260.000	1200.71
37	265.000	1263.10
38	270.000	1330.04
39	275.000	1401.72
40	280.000	1478.60
41	285.000	1560.87
42	290.000	1648.97
43	295.000	1743.27
44	300.000	1844.05
45	305.000	1945.89
46	310.000	2043.07
47	315.000	2146.75
48	320.000	2257.61
49	325.000	2375.77

* 62 PSI, FLANGE REQUIREMENT

Data Points for Heatup and Cooldown
Curves for up to 32 EFPY and
With Margins for Instrumentation Error

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	446.75	19	175.000	587.20	36	260.000	1052.62
2	90.000	450.69	20	180.000	601.72	37	265.000	1101.74
3	95.000	454.83	21	185.000	617.30	38	270.000	1154.52
4	100.000	459.49	22	190.000	633.95	39	275.000	1211.18
5	105.000	464.39	23	195.000	651.99	40	280.000	1271.92
6	110.000	469.65	24	200.000	671.20	41	285.000	1336.95
7	115.000	475.32	25	205.000	692.05	42	290.000	1407.01
8	120.000	481.30	26	210.000	714.40	43	295.000	1481.90
9	125.000	487.85	27	215.000	738.34	44	300.000	1562.30
10	130.000	494.89	28	220.000	764.15	45	305.000	1648.49
11	135.000	502.45	29	225.000	791.84	46	310.000	1740.74
12	140.000	510.59	30	230.000	821.50	47	315.000	1839.56
13	145.000	519.34	31	235.000	853.59	48	320.000	1945.08
14	150.000	528.62	32	240.000	887.92	49	325.000	2058.16
15	155.000	538.72	33	245.000	924.79	50	330.000	2178.95
16	160.000	549.59	34	250.000	964.38	51	335.000	2308.09
17	165.000	561.28	35	255.000	1006.91	52	340.000	2445.60
18	170.000	573.70						

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	404.22	13	145.000	479.66	25	205.000	663.40
2	90.000	408.24	14	150.000	489.55	26	210.000	687.20
3	95.000	412.59	15	155.000	500.23	27	215.000	712.94
4	100.000	417.27	16	160.000	511.71	28	220.000	740.51
5	105.000	422.34	17	165.000	524.08	29	225.000	770.13
6	110.000	427.78	18	170.000	537.27	30	230.000	802.17
7	115.000	433.58	19	175.000	551.62	31	235.000	836.50
8	120.000	439.90	20	180.000	567.02	32	240.000	873.33
9	125.000	446.74	21	185.000	583.51	33	245.000	912.96
10	130.000	454.09	22	190.000	601.38	34	250.000	955.54
11.	135.000	462.02	23	195.000	620.60	35	255.000	1001.36
12	140.000	470.55	24	200.000	641.15	36	260.000	1050.59

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN

01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	360.75	13	145.000	439.58	25	205.000	635.37
2	90.000	364.87	14	150.000	450.04	26	210.000	661.03
3	95.000	369.35	15	155.000	461.34	27	215.000	688.53
4	100.000	374.18	16	160.000	473.51	28	220.000	718.09
5	105.000	379.36	17	165.000	486.55	29	225.000	750.15
6	110.000	385.01	18	170.000	500.70	30	230.000	784.43
7	115.000	391.15	19	175.000	515.97	31	235.000	821.33
8	120.000	397.75	20	180.000	532.28	32	240.000	860.99
9	125.000	404.91	21	185.000	550.01	33	245.000	903.92
10	130.000	412.62	22	190.000	569.08	34	250.000	949.86
11	135.000	420.86	23	195.000	589.53	35	255.000	999.29
12	140.000	429.95	24	200.000	611.66	36	260.000	1052.23

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	316.28	13	145.000	398.96	25	205.000	608.35
2	90.000	320.52	14	150.000	410.06	26	210.000	635.78
3	95.000	325.15	15	155.000	422.07	27	215.000	665.51
4	100.000	330.16	16	160.000	434.92	28	220.000	697.39
5	105.000	335.62	17	165.000	448.92	29	225.000	731.73
6	110.000	341.52	18	170.000	464.01	30	230.000	768.67
7	115.000	347.88	19	175.000	480.21	31	235.000	808.64
8	120.000	354.80	20	180.000	497.77	32	240.000	851.52
9	125.000	362.32	21	185.000	516.74	33	245.000	897.68
10	130.000	370.44	22	190.000	537.05	34	250.000	947.30
11	135.000	379.18	23	195.000	559.11	35	255.000	1000.75
12	140.000	388.67	24	200.000	582.72			

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED. PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	224.03	13	145.000	316.17	24	200.000	527.88
2	90.000	228.59	14	150.000	328.78	25	205.000	557.68
3	95.000	233.62	15	155.000	342.48	26	210.000	589.65
4	100.000	239.08	16	160.000	357.21	27	215.000	624.18
5	105.000	245.08	17	165.000	373.26	28	220.000	661.54
6	110.000	251.58	18	170.000	390.50	29	225.000	701.73
7	115.000	258.70	19	175.000	409.27	30	230.000	744.99
8	120.000	266.37	20	180.000	429.51	31	235.000	791.66
9	125.000	274.79	21	185.000	451.34	32	240.000	841.89
10	130.000	283.89	22	190.000	474.98	33	245.000	896.02
11	135.000	293.82	23	195.000	500.44	34	250.000	954.08
12	140.000	304.50						

AEP60 F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (32.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	296
2485	317

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	22088	92529
2485	27288	115366

AEP60 F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 WITH MARGIN 01/12/90

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~448.75~~	19	175.000	493.11	37	265.000	1005.26
2	90.000	~~446.21~~	20	180.000	507.00	38	270.000	1057.96
3	95.000	~~433.87~~ 412.80	21	185.000	522.12	39	275.000	1114.56
4	100.000	~~424.81~~	22	190.000	538.37	40	280.000	1175.14
5	105.000	~~418.64~~	23	195.000	556.10	41	285.000	1239.91
6	110.000	~~414.71~~	24	200.000	575.09	42	290.000	1309.73
7	115.000	~~412.93~~	25	205.000	595.76	43	295.000	1384.19
8	120.000	412.80	26	210.000	617.99	44	300.000	1464.20
9	125.000	414.31	27	215.000	641.85	45	305.000	1549.80
10	130.000	417.11	28	220.000	667.61	46	310.000	1630.38
11	135.000	421.25	29	225.000	695.27	47	315.000	1709.42
12	140.000	426.49	30	230.000	724.95	48	320.000	1794.44
13	145.000	432.80	31	235.000	757.06	49	325.000	1885.24
14	150.000	440.22	32	240.000	791.42	50	330.000	1982.27
15	155.000	448.72	33	245.000	828.30	51	335.000	2085.80
16	160.000	458.23	34	250.000	867.91	52	340.000	2196.69
17	165.000	468.82	35	255.000	910.47	53	345.000	2314.48
18	170.000	480.35	36	260.000	956.17	54	350.000	2440.72

Data Points for Heatup and Cooldown
Curves for up to 32 EFPY and
Without Margins for Instrumentation Error

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	514.83
2	90.000	519.49
3	95.000	524.39
4	100.000	529.65
5	105.000	535.32
6	110.000	541.30
7	115.000	547.85
8	120.000	554.89
9	125.000	562.45
10	130.000	570.59
11	135.000	579.34
12	140.000	588.62
13	145.000	598.72
14	150.000	609.59
15	155.000	~~621.28~~
16	160.000	~~633.70~~ *
17	165.000	~~647.20~~
18	170.000	~~661.72~~
19	175.000	~~677.90~~ *
20	180.000	~~693.95~~
21	185.000	711.99
22	190.000	731.20
23	195.000	752.05
24	200.000	774.40
25	205.000	798.34
26	210.000	824.15
27	215.000	851.84
28	220.000	881.50
29	225.000	913.59
30	230.000	947.92
31	235.000	984.79
32	240.000	1024.38
33	245.000	1066.91
34	250.000	1112.62
35	255.000	1161.74
36	260.000	1214.52
37	265.000	1271.18
38	270.000	1331.92
39	275.000	1396.95
40	280.000	1467.01
41	285.000	1541.90
42	290.000	1622.30
43	295.000	1708.49
44	300.000	1800.74
45	305.000	1899.56
46	310.000	2005.08
47	315.000	2118.16
48	320.000	2238.95
49	325.000	2368.09

* 62 PSI, FLANGE REQUIREMENT

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 2 (20 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	472.59	13	145.000	560.23	24	200.000	747.20
2	90.000	477.27	14	150.000	571.71	25	205.000	772.94
3	95.000	482.34	15	155.000	584.08	26	210.000	800.51
4	100.000	487.78	16	160.000	597.27	27	215.000	830.13
5	105.000	493.68	17	165.000	611.62	28	220.000	862.17
6	110.000	499.90	18	170.000	~~627.02~~	29	225.000	896.50
7	115.000	506.74	19	175.000	~~648.51~~ *	30	230.000	933.33
8	120.000	514.09	20	180.000	~~661.38~~	31	235.000	972.96
9	125.000	522.02	21	185.000	680.60	32	240.000	1015.54
10	130.000	530.55	22	190.000	701.15	33	245.000	1061.36
11	135.000	539.66	23	195.000	723.40	34	250.000	1110.59
12	140.000	549.55						

* FLANGE REQUIREMENT

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 3 (40 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	429.35	13	145.000	521.34	24	200.000	721.03
2	90.000	434.18	14	150.000	533.51	25	205.000	748.53
3	95.000	439.36	15	155.000	546.55	26	210.000	778.09
4	100.000	445.01	16	160.000	560.70	27	215.000	810.15
5	105.000	451.15	17	165.000	575.97	28	220.000	844.43
6	110.000	457.75	18	170.000	592.28	29	225.000	881.33
7	115.000	464.91	19	175.000	610.01	30	230.000	920.99
8	120.000	472.62	20	180.000	~~629.08~~ *	31	235.000	963.92
9	125.000	480.86	21	185.000	649.53	32	240.000	1009.86
10	130.000	489.95	22	190.000	671.66	33	245.000	1059.29
11	135.000	499.58	23	195.000	695.37	34	250.000	1112.23
12	140.000	510.04						

* FLANGE REQUIREMENT

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 4 (60 DEG-F / HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	385.15	12	140.000	470.06	23	195.000	668.35
2	90.000	390.16	13	145.000	482.07	24	200.000	695.78
3	95.000	395.62	14	150.000	494.92	25	205.000	725.51
4	100.000	401.52	15	155.000	508.92	26	210.000	757.39
5	105.000	407.88	16	160.000	524.01	27	215.000	791.73
6	110.000	414.80	17	165.000	540.21	28	220.000	828.67
7	115.000	422.32	18	170.000	557.77	29	225.000	868.64
8	120.000	430.44	19	175.000	576.74	30	230.000	911.52
9	125.000	439.18	20	180.000	597.05	31	235.000	957.68
10	130.000	448.67	21	185.000	619.11	32	240.000	1007.30
11	135.000	458.86	22	190.000	642.72	33	245.000	1060.75

AEP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 5 (100 DEG-F/HR COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	293.62	12	140.000	388.78	23	195.000	617.68
2	90.000	299.08	13	145.000	402.48	24	200.000	649.65
3	95.000	305.08	14	150.000	417.21	25	205.000	684.18
4	100.000	311.58	15	155.000	433.26	26	210.000	721.54
5	105.000	318.70	16	160.000	450.50	27	215.000	761.73
6	110.000	326.37	17	165.000	469.27	28	220.000	804.99
7	115.000	334.79	18	170.000	489.51	29	225.000	851.66
8	120.000	343.89	19	175.000	511.34	30	230.000	901.89
9	125.000	353.82	20	180.000	534.98	31	235.000	956.02
10	130.000	364.50	21	185.000	560.44	32	240.000	1014.08
11	135.000	376.17	22	190.000	587.88			

AEP60 F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 01/12/90

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (32.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	283
2485	304

PRESSURE (PSI)	PRESSURE STRESS (PSI)	1.5 K1M (PSI SQ.RT.IN.)
2000	21444	89745
2485	26645	112505

AEP60 F/HR HEATUP CURVE REG. GUIDE 1.99,REV.2 01/12/90

COMPOSITE CURVE PLOTTED FOR HEATUP PROFILE 2 HEATUP RATE(S) (DEG.F/HR) = 60.0

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = (1-AOWIN)T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	~~[illegible]~~	18	170.000	567.00	35	255.000	1065.26
2	90.000	~~484.81~~ 472.8	19	175.000	582.12	36	260.000	1117.96
3	95.000	~~478.64~~	20	180.000	598.37	37	265.000	1174.56
4	100.000	~~474.71~~	21	185.000	616.10	38	270.000	1235.14
5	105.000	~~472.83~~	22	190.000	635.09	39	275.000	1299.91
6	110.000	472.80	23	195.000	655.76	40	280.000	1369.73
7	115.000	474.31	24	200.000	677.99	41	285.000	1444.19
8	120.000	477.11	25	205.000	701.85	42	290.000	1524.20
9	125.000	481.25	26	210.000	727.61	43	295.000	1609.80
10	130.000	486.49	27	215.000	755.27	44	300.000	1690.38
11	135.000	492.80	28	220.000	784.95	45	305.000	1769.42
12	140.000	500.22	29	225.000	817.06	46	310.000	1854.44
13	145.000	508.72	30	230.000	851.42	47	315.000	1945.24
14	150.000	518.23	31	235.000	888.30	48	320.000	2042.27
15	155.000	528.82	32	240.000	927.91	49	325.000	2145.80
16	160.000	540.35	33	245.000	970.47	50	330.000	2256.69
17	165.000	553.11	34	250.000	1016.17	51	335.000	2374.48

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