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SUBJECT: Application for amend to License DPR-74,requesting relief from TS surveillances until refueling outage,currently scheduled to begin on 940806.

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Donald C. Cook Nuclear Plant Unit 2 AEP:NRC:1181
Docket No. 50-316
License No. DPR-74
SURVEILLANCE INTERVAL EXTENSION FOR UNIT 2 CYCLE 9

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Attn: T. E. Murley

April 16, 1993

Dear Dr. Murley:

This letter constitutes an application for amendment to the Technical Specifications (T/Ss) for the Donald C. Cook Nuclear Plant Unit 2. Specifically, we request an extension for certain surveillances which the T/Ss require to be performed beginning January 2, 1994. We are requesting relief from these T/S requirements until the Unit 2 refueling outage, which is currently scheduled to begin August 6, 1994. Many of these surveillances can only be performed during shutdown; therefore, to avoid unnecessary shutdown of the plant, we ask that your review of this request be performed on an expedited basis and that you respond to us by December 1, 1993.

A description of the proposed changes and our analysis concerning significant hazards considerations are contained in Attachment 1 to this letter. The proposed, revised T/S pages are contained in Attachment 2. The existing T/S pages, marked to reflect the proposed changes, are contained in Attachment 3.

All of the requested surveillance extensions are associated with surveillances normally performed during refueling outages. The current cycle will be lengthened approximately five months due to a planned power reduction to approximately 70% of rated thermal power, which is to begin in May 1993 and remain in effect until the end of the cycle. The purpose of extending the cycle is to separate the refueling outages between Unit 1 (which is scheduled for refueling in January, 1994) and Unit 2.

A001

During the last refueling outage, which was extended approximately six months due to turbine-generator rotor vibrations, an effort was made to re-perform as many surveillances as possible. A significant number of T/S surveillances (approximately 70) were re-performed, reducing the number of surveillances for which we are requesting extensions. However, our efforts were constrained because Unit 1 was in a refueling outage at the same time.

Some of the Technical Specification pages affected by this submittal are pages for which changes are pending due to prior submittals. The proposed changes contained in this submittal are in addition to our previous requests and do not supersede them. The pages included in this category and the applicable prior submittals which have not yet been processed are provided in the table below:

Letter Number	Date	T/S Page Numbers
AEP:NRC:1131A	April 19, 1991	3/4 4-33
AEP:NRC:1178	September 24, 1992	3/4 6-14
AEP:NRC:1143	May 1, 1992	3/4 7-20 & 3/4 7-40

In accordance with 10 CFR 50.92(c), our evaluation of the changes indicates no significant hazards because these changes do not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in the margin of safety.

These proposed changes have been reviewed and approved by the Plant Nuclear Safety Review Committee and by the Nuclear Safety and Design Review Committee.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to the Michigan Public Service Commission and to the Michigan Department of Public Health.

This letter is submitted pursuant to 10 CFR 50.54(f) and, as such, an oath statement is enclosed.

Sincerely,

E. E. Fitzpatrick
Vice President

dr

Attachments

cc: A. A. Blind - Bridgman
J. R. Padgett
G. Charnoff
A. B. Davis - Region III
NRC Resident Inspector - Bridgman
NFEM Section Chief

bc: S. J. Brewer
D. H. Malin/K. J. Toth
M. L. Horvath - Bridgman
J. B. Shinnock
W. G. Smith, Jr.
W. M. Dean, NRC - Washington, D. C.
AEP:NRC:1181
DC-N-6015.1

STATE OF OHIO)
COUNTY OF FRANKLIN)

E. E. Fitzpatrick, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the forgoing Technical Specifications Changes Proposed in Letter AEP:NRC:1181 and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

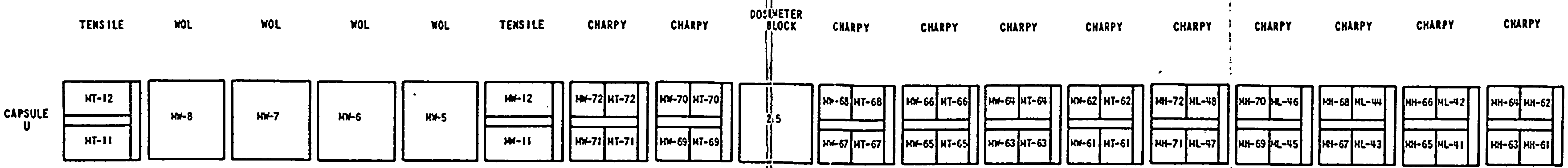
E E Fitzpatrick

Subscribed and sworn to before me this 16th

day of June, 19 93.

Rita D. Hill
NOTARY PUBLIC

RITA D. HILL
NOTARY PUBLIC, STATE OF OHIO
MY COMMISSION EXPIRES 6-28-94



SPECIMEN CODE:
MT - PLATE C5521-2 (TRANSVERSE)
ML - PLATE C5521-2 (LONGITUDINAL)
MW - WELD METAL
MH - WELD HEAT AFFECTED ZONE

Figure 4-2 Capsule U Diagram Showing Location of Specimens, Thermal Monitors and Dosimeters

SECTION 5.0
TESTING OF SPECIMENS FROM CAPSULE U

5.1 Overview

The post-irradiation mechanical testing of the Charpy V-notch and tensile specimens was performed at the Westinghouse Science and Technology Center hot cell with consultation by Westinghouse Power Systems personnel. Testing was performed in accordance with 10CFR50, Appendices G and H[2], ASTM Specification E185-82[6], and Westinghouse Remote Metallographic Facility (RMF) Procedure RMF 8402, Revision 2 as modified by RMF Procedures 8102, Revision 1 and 8103, Revision 1.

Upon receipt of the capsule at the hot cell laboratory, the specimens and spacer blocks were carefully removed, inspected for identification number, and checked against the master list in WCAP-8512[1]. No discrepancies were found.

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

The Charpy impact tests were performed per ASTM Specification E23-88[7] and RMF Procedure 8103, Revision 1 on a Tinius-Olsen Model 74, 358J machine. The tup (striker) of the Charpy machine is instrumented with a GRC 830I instrumentation system, feeding information into an IBM XT computer. 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 shown in Appendix A, 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 roughly 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 (σ_Y) was calculated from the three-point bend formula having the following expression:

$$\sigma_Y = P_{GY} * \{L/[B*(W-a)^2*C]\} \qquad (1)$$

where L = distance between the specimen supports in the impact testing machine; B = the width of the specimen measured parallel to the notch; W = height of the specimen, measured perpendicularly to the notch; a = notch depth. The constant C is dependent on the notch flank angle (ϕ), notch root radius (ρ), and the type of loading (i.e., pure bending or three-point bending).

In three-point bending a Charpy specimen in which ϕ = 45° and ρ = 0:010", Equation 1 is valid with C = 1.21. Therefore (for L = 4W),

$$\sigma_Y = P_{GY} * \{L/[B*(W-a)^2*1.21]\} = [3.3P_{GY}W]/[B(W-a)^2] \qquad (2)$$

For the Charpy specimens, B = 0.394 in., W = 0.394 in., and a = 0.079 in. Equation 2 then reduces to:

$$\sigma_Y = 33.3 \times P_{GY} \qquad (3)$$

where σ_Y is in units of psi and P_{GY} is in units of lbs. The flow stress was calculated from the average of the yield and maximum loads, also using the three-point bend formula.

Percent shear was determined from post-fracture photographs using the ratio-of-areas methods in compliance with ASTM Specification A370-89[8]. 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 Model 1115, split-console test machine, per ASTM Specification E8-89b[9] and E21-79 (1988)[10], and RMF Procedure 8102, Revision 1. All pull rods, grips, and pins were made of Inconel 718 hardened to HRC45. 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 inches per minute throughout the test.

Extension 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 E83-85[11].

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 the 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 of 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 ± 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 post-fracture 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 the Charpy V-notch impact tests performed on the various materials contained in Capsule U, which was irradiated to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV), are presented in Tables 5-1 through 5-4 and are compared with unirradiated results[1] as shown in Figures 5-1 through 5-4. The transition temperature increases and upper shelf energy decreases for the Capsule U materials are summarized in Table 5-5.

Irradiation of the reactor vessel intermediate shell plate C5521-2 Charpy specimens oriented with the longitudinal axis of the specimen parallel to the major rolling direction of the plate (longitudinal orientation) to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-1) resulted in a 30 ft-lb transition temperature increase of 95°F and in a 50 ft-lb transition temperature increase of 110°F. This resulted in a 30 ft-lb transition temperature of 120°F and a 50 ft-lb transition temperature of 165°F (longitudinal orientation).

The average Upper Shelf Energy (USE) of the intermediate shell plate C5521-2 Charpy specimens (longitudinal orientation) resulted in a energy decrease of 16 ft-lb after irradiation to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F. This results in an average USE of 111 ft-lb (Figure 5-1).

Irradiation of the reactor vessel intermediate shell plate C5521-2 Charpy specimens oriented with the longitudinal axis of the specimen normal to the major rolling direction of the plate (transverse orientation) to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-2) resulted in a 30 ft-lb transition temperature increase of 130°F and in a 50 ft-lb transition temperature increase of 135°F. This resulted in a 30 ft-lb transition temperature of 160°F and a 50 ft-lb transition temperature of 205°F (transverse orientation).

The average USE of the intermediate shell plate C5521-2 Charpy specimens (transverse orientation) resulted in an energy decrease of 14 ft-lb after irradiation to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F. This resulted in an average USE of 72 ft-lb (Figure 5-2).

Irradiation of the reactor vessel core region weld metal Charpy specimens to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-3) resulted in a 75°F increase in 30 ft-lb transition temperature and a 50 ft-lb transition temperature increase of 40°F. This resulted in a 30 ft-lb transition temperature of 85°F and the 50 ft-lb transition temperature of 110°F.

The average USE of the reactor vessel core region weld metal resulted in an energy decrease of 6 ft-lb after irradiation to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F. This resulted in an average USE of 71 ft-lb (Figure 5-3).

Irradiation of the reactor vessel weld Heat-Affected-Zone (HAZ) metal specimens to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F (Figure 5-4) resulted in a 30 ft-lb transition temperature increase of 105°F and a 50 ft-lb transition temperature increase of 110°F. This resulted in a 30 ft-lb transition temperature of 45°F and the 50 ft-lb transition temperature of 80°F.

The average USE of the reactor vessel weld HAZ metal experienced an energy decrease of 33 ft-lb after irradiation to 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV) at 550°F. This resulted in an average USE of 82 ft-lb (Figure 5-4).

The fracture appearance of each irradiated Charpy specimen from the various materials is shown in Figures 5-5 through 5-8 and show an increasingly ductile or tougher appearance with increasing test temperature.

A comparison of the 30 ft-lb transition temperature increases and upper shelf energy decreases for the various D. C. Cook Unit 2 surveillance materials with predicted values using the methods of NRC Regulatory Guide 1.99, Revision 2[3] is presented in Table 5-6. Comparison of the 30 ft-lb transition

temperature increase for the intermediate shell plate C5521-2 (transverse orientation) is 33°F greater than the Regulatory Guide prediction. However, the NRC Regulatory Guide 1.99, Revision 2 requires a 2 sigma allowance of 34°F be added to the predicted reference transition temperature to obtain a conservative upper bound value. Thus, the reference transition temperature increase is bounded by the 2 sigma allowance for shift prediction. This comparison indicates that the transition temperature increases and the upper shelf energy decreases of the Intermediate Shell Plate C5521-2 (longitudinal orientation) and surveillance weld resulting from irradiation to 1.58×10^{19} n/cm^2 (E > 1.0 MeV) are less than the Regulatory Guide predictions. This comparison also indicates that the upper shelf energy decrease of the intermediate shell plate C5521-2 (transverse orientation) resulting from irradiation to 1.58×10^{19} n/cm^2 (E > 1.0 MeV) is less than the Regulatory Guide prediction.

The end of license (32 EFPY) RT_{NDT} values for all the D. C. Cook Unit 2 beltline region materials are shown in Table 5-7. These values were predicted using Regulatory Guide 1.99, Revision 2 methodology and are projected to be within the Regulatory limits.

Photographs of the charpy and tensile specimens before testing are shown in Appendix B.

5.3 Tension Test Results

The results of the tension tests performed on the various materials contained in Capsule U irradiated to 1.58×10^{19} n/cm^2 (E > 1.0 MeV) are presented in Table 5-8 and are compared with unirradiated results[1] as shown in Figures 5-9 and 5-10.

The results of the tension tests performed on the intermediate shell plate C5521-2 (transverse orientation) indicated that irradiation to 1.58×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F caused less than a 18 ksi increase in the 0.2 percent offset yield strength and less than a 16 ksi increase in the ultimate tensile strength when compared to unirradiated data[1] (Figure 5-9).

The results of the tension tests performed on the reactor vessel core region weld metal indicated that irradiation to 1.58×10^{19} n/cm^2 (E > 1.0 MeV) at 550°F caused less than a 9 ksi increase in the 0.2 percent offset yield strength and less than a 8 ksi increase in the ultimate tensile strength when compared to unirradiated data[1] (Figure 5-10).

The fractured tension specimens for the Intermediate Shell Plate C5521-2 material are shown in Figure 5-11, while the fractured specimens for the weld metal are shown in Figure 5-12.

The engineering stress-strain curves for the tension tests are shown in Figures 5-13 and 5-14.

5.4 Wedge Opening Loading Specimens

Per the surveillance capsule testing program with the Indiana Michigan Power Company, the WOL specimens will not be tested and will be stored at the Westinghouse Science and Technology Center.

TABLE 5-1

CHARPY V-NOTCH IMPACT DATA FOR THE D. C. COOK UNIT 2 INTERMEDIATE SHELL PLATE C5521-2 IRRADIATED AT 550°F, FLUENCE 1.58 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 (%)
			Longitudinal Orientation				
ML45	75	(24)	14	(19)	10	(0.25)	10
ML48	100	(38)	27	(37)	23	(0.58)	20
ML42	125	(52)	35	(47)	27	(0.69)	30
ML44	175	(79)	62	(84)	42	(1.07)	50
ML41	200	(93)	55	(75)	42	(1.07)	50
ML43	225	(107)	103	(140)	50	(1.27)	90
ML46	250	(121)	114	(155)	79	(2.01)	100
ML47	300	(149)	115	(156)	82	(2.08)	100
			Transverse Orientation				
MT62	25	(-4)	5	(7)	2	(0.05)	5
MT61	50	(10)	14	(19)	12	(0.30)	10
MT66	75	(24)	17	(23)	10	(0.25)	15
MT71	125	(52)	19	(26)	16	(0.41)	30
MT64	150	(66)	25	(34)	21	(0.53)	35
MT72	175	(79)	33	(45)	28	(0.71)	40
MT70	200	(93)	37	(50)	30	(0.76)	45
MT69	215	(102)	39	(53)	33	(0.84)	65
MT63	225	(107)	63	(85)	52	(1.32)	95
MT68	250	(121)	67	(91)	54	(1.37)	100
MT67	275	(135)	72	(98)	50	(1.27)	100
MT65	300	(149)	77	(104)	58	(1.47)	100

TABLE 5-2

CHARPY V-NOTCH IMPACT DATA FOR THE D. C. COOK UNIT 2 REACTOR VESSEL WELD METAL AND HAZ METAL IRRADIATED AT 550°F, FLUENCE 1.58 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 (%)
			Weld Metal				
MW70	-10	(-23)	23	(31)	16	(0.41)	25
MW64	0	(-18)	29	(39)	26	(0.66)	35
MW71	25	(- 4)	17	(23)	14	(0.36)	30
MW68	50	(10)	21	(28)	21	(0.53)	40
MW63	75	(24)	26	(35)	20	(0.51)	60
MW61	100	(38)	42	(57)	37	(0.94)	80
MW72	125	(52)	60	(81)	50	(1.27)	85
MW65	150	(66)	71	(96)	57	(1.45)	100
MW66	175	(79)	47	(64)	39	(0.99)	85
MW62	185	(85)	62	(84)	49	(1.24)	100
MW67	200	(93)	78	(106)	62	(1.57)	100
MW69	250	(121)	74	(100)	62	(1.57)	100
			HAZ Metal				
MH67	-25	(-32)	4	(5)	2	(0.05)	10
MH63	25	(- 4)	21	(28)	10	(0.25)	35
MH71	50	(10)	48	(65)	30	(0.76)	55
MH69	65	(18)	18	(24)	19	(0.48)	30
MH72	75	(24)	64	(87)	44	(1.12)	90
MH70	100	(38)	39	(53)	32	(0.81)	50
MH62	125	(52)	118	(160)	71	(1.80)	100
MH64	150	(66)	72	(98)	55	(1.40)	95
MH61	175	(79)	83	(113)	61	(1.55)	100
MH65	200	(93)	103	(140)	71	(1.80)	100
MH68	225	(107)	74	(100)	53	(1.35)	100
MH66	250	(121)	69	(94)	66	(1.68)	100

TABLE 5-3

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR THE D. C. COOK UNIT 2 INTERMEDIATE SHELL PLATE C5521-2 IRRADIATED AT 550°F, FLUENCE 1.58×10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in^2)			Yield Load (lbs)	Time to Yield (msec)	Maximum Load (lbs)	Time to Maximum (msec)	Fracture Load (lbs)	Arrest Load (lbs)	Yield Stress (ksi)	Flow Stress (ksi)
			Charpy Ed/A	Maximum Em/A	Prop. Ep/A								
						Longitudinal Orientation							
ML45	75	14	113	64	48	3722	0.14	3929	0.20	3929	638	124	127
ML48	100	27	217	167	50	3670	0.14	4600	0.38	4600	858	122	137
ML42	125	35	282	124	157	3945	0.29	4258	0.38	4258	384	131	136
ML44	175	62	499	241	258	3467	0.14	4735	0.53	4563	792	115	136
ML41	200	55	443	122	321	3731	0.28	4139	0.38	3887	1340	124	131
ML43	225	103	829	308	521	3262	0.14	4507	0.54	*	*	108	129
ML46	250	114	918	234	684	3265	0.14	4546	0.54	*	*	108	130
ML47	300	115	926	147	779	3126	0.14	4071	0.38	*	*	104	120
						Transverse Orientation							
MT62	25	5	40	12	28	1107	0.10	1319	0.14	1319	55	37	40
MT61	50	14	113	77	35	3600	0.14	4154	0.22	4154	126	120	129
MT66	75	17	137	88	49	3825	0.14	4273	0.25	4273	513	127	134
MT71	125	19	153	48	105	2969	0.21	3187	0.24	3187	202	99	102
MT64	150	25	201	124	77	3381	0.14	4194	0.32	4194	1577	112	126
MT72	175	33	266	156	110	3438	0.13	4273	0.38	4273	2083	114	128
MT70	200	37	298	128	170	3681	0.28	4096	0.39	4096	2151	122	129
MT69	215	39	314	153	161	3289	0.14	4158	0.38	4158	2533	109	124
MT63	225	63	507	236	272	3359	0.15	4457	0.54	4225	3236	112	130
MT68	250	67	540	221	319	3288	0.14	4212	0.54	*	*	109	125
MT67	275	72	580	222	358	3051	0.16	4249	0.54	*	*	101	121
MT65	300	77	620	223	397	3153	0.14	4251	0.54	*	*	105	123

*Fully ductile fracture. No arrest load.

TABLE 5-4

INSTRUMENTED CHARPY IMPACT TEST RESULTS FOR THE D. C. COOK UNIT 2 WELD METAL AND HEAT-AFFECTED-ZONE (HAZ) METAL, IRRADIATED AT 550°F, FLUENCE 1.58 x 10^{19} n/cm^2 (E > 1.0 MeV)

Sample Number	Test Temp (°F)	Charpy Energy (ft-lb)	Normalized Energies (ft-lb/in^2) Charpy Ed/A	Maximum Em/A	Prop Ep/A	Yield Load (lbs)	Time to Yield (msec)	Maximum Load (lbs)	Time to Maximum (msec)	Fracture Load (lbs)	Arrest Load (lbs)	Yield Stress (ksi)	Flow Stress (ksi)
							Weld Metal						
MW70	-10	23	185	133	52	3898	0.14	4576	0.31	4576	778	129	141
MW64	0	29	234	106	127	3891	0.14	4482	0.28	4482	467	129	139
MW71	25	17	137	83	54	3776	0.14	4105	0.23	4105	978	125	131
MW68	50	21	169	85	84	3847	0.26	4021	0.30	4021	941	128	131
MW63	75	26	209	131	79	3655	0.14	4430	0.32	4430	1618	121	134
MW61	100	42	338	151	187	3712	0.26	4359	0.42	4359	463	123	134
MW72	125	60	483	239	244	3323	0.14	4581	0.54	4581	1252	110	131
MW65	150	71	572	235	337	3511	0.14	4535	0.54	*	*	117	134
MW66	175	47	378	221	158	3397	0.14	4404	0.50	4404	810	113	130
MW62	185	62	499	231	269	3280	0.13	4232	0.54	*	*	109	125
MW67	200	78	628	250	378	3754	0.28	4440	0.62	*	*	125	136
MW69	250	74	596	231	365	3283	0.14	4273	0.54	*	*	109	125
							HAZ Metal						
MH67	-25	4	32	15	17	964	0.10	1133	0.17	1133	146	32	35
MH63	25	21	169	47	122	3327	0.21	3486	0.23	3486	858	111	113
MH71	50	48	387	166	221	3797	0.14	4557	0.38	4420	2126	126	139
MH69	65	18	145	58	87	3653	0.14	3834	0.20	3834	971	121	124
MH72	75	64	515	162	354	3910	0.14	4530	0.37	4500	3802	130	140
MH70	100	39	314	114	200	2970	0.21	3618	0.38	3618	954	99	109
MH62	125	118	950	301	649	3352	0.16	4355	0.53	*	*	111	128
MH64	150	72	580	240	340	3991	0.22	4756	0.54	4756	4056	133	145
MH61	175	83	668	243	425	3622	0.14	4662	0.54	*	*	120	138
MH65	200	103	829	253	576	3642	0.15	4799	0.54	*	*	121	140
MH68	225	74	596	187	409	3066	0.17	3981	0.50	*	*	102	117
MH66	250	69	556	228	328	3330	0.14	4287	0.52	*	*	111	126

*Fully ductile fracture. No arrest load.

TABLE 5-5

EFFECT OF 550°F IRRADIATION TO 1.58×10^{19} n/cm^2 (E > 1.0 MeV)

ON THE NOTCH TOUGHNESS PROPERTIES OF THE D. C. COOK UNIT 2 REACTOR VESSEL SURVEILLANCE MATERIALS

	Average 30 ft-lb(1) Transition Temperature (°F)			Average 35 mil(1) Lateral Expansion Temperature (°F)			Average 50 ft-lb(1) Transition Temperature (°F)			Average Energy(1) Absorption at Full Shear (ft-lb)		
Material	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	ΔT	Unirradiated	Irradiated	Δ(ft-lb)
Plate C5521-2 (Longitudinal)	25	120	95	50	150	100	55	165	110	127	111	- 16
Plate C5521-2 (Transverse)	30	160	130	70	190	120	70	205	135	86	72	- 14
Weld Metal	10	85	75	50	100	50	70	110	40	77	71	- 6
HAZ Metal	- 60	45	105	- 40	85	125	- 30	80	110	115	82	- 33

(1) "AVERAGE" is defined as the value read from the curve fitted through the data points of the Charpy tests (Figures 5-1 through 5-4).

TABLE 5-6

COMPARISON OF THE D. C. COOK UNIT 2 SURVEILLANCE MATERIAL 30 FT-LB TRANSITION TEMPERATURE SHIFTS AND UPPER SHELF ENERGY DECREASES WITH REGULATORY GUIDE 1.99 REVISION 2 PREDICTIONS

			30 ft-lb Transition Temp. Shift		Upper Shelf Energy Decrease	
Material	Capsule	Fluence 10^{19} n/cm^2	Predicted (a) (°F)	Measured (°F)	Predicted (a) (%)	Measured (%)
Plate C5521-2 (Longitudinal)	T	0.264	55	55	16	13
	Y	0.683	77	90	20	19
	X	1.06	88	95	22	19
	U	1.58	97	95	24	13
Plate C5521-2 (Transverse)	T	0.264	55	80	16	14
	Y	0.683	77	100	20	20
	X	1.06	88	103	22	19
	U	1.58	97	130	24	16
Weld Metal	T	0.264	45	40	15	4
	Y	0.683	63	50	18	9
	X	1.06	72	70	21	10
	U	1.58	80	75	23	8
HAZ Metal	T	0.264	--	50	--	13
	Y	0.683	--	70	--	21
	X	1.06	--	72	--	12
	U	1.58	--	105	--	29

(a) Regulatory Guide 1.99, Revision 2

TABLE 5-7

PROJECTED END OF LICENSE (32 EFPY) RT_{NDT} AND UPPER SHELF ENERGY VALUES FOR D. C. COOK UNIT 2 BELTLINE REGION MATERIALS PER REGULATORY GUIDE 1.99, REVISION 2

MATERIAL DESCRIPTION	RT_{NDT} °F	UPPER SHELF ENERGY (ft-lbs)
Intermediate Shell Plate, C5556-2	216	76
Intermediate Shell Plate, C5521-2	171 (171)	66
Lower Shell Plate, C5540-2	100	47
Lower Shell Plate, C5592-1	128	52
Intermediate Shell Longitudinal Welds (located at 10° azimuth)	90 (56)	62
Lower Shell Longitudinal Welds (located at 90° azimuth)	84 (50)	62
Circumferential Weld	102 (67)	62

Note numbers in () are based upon surveillance capsule data.

TABLE 5-8

TENSILE PROPERTIES FOR THE D. C. COOK UNIT 2 REACTOR VESSEL SURVEILLANCE MATERIALS IRRADIATED AT 550°F TO 1.58×10^{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	MT11	150	79.5	97.4	3.60	172.0	73.3	10.5	20.1	57
Plate	MT12	550	70.3	93.7	3.65	131.8	74.4	9.6	16.7	44
Weld	MW11	125	82.5	96.8	3.20	173.9	65.2	9.0	19.5	63
Weld	MW12	550	75.9	92.7	3.50	190.2	71.3	8.1	17.1	58

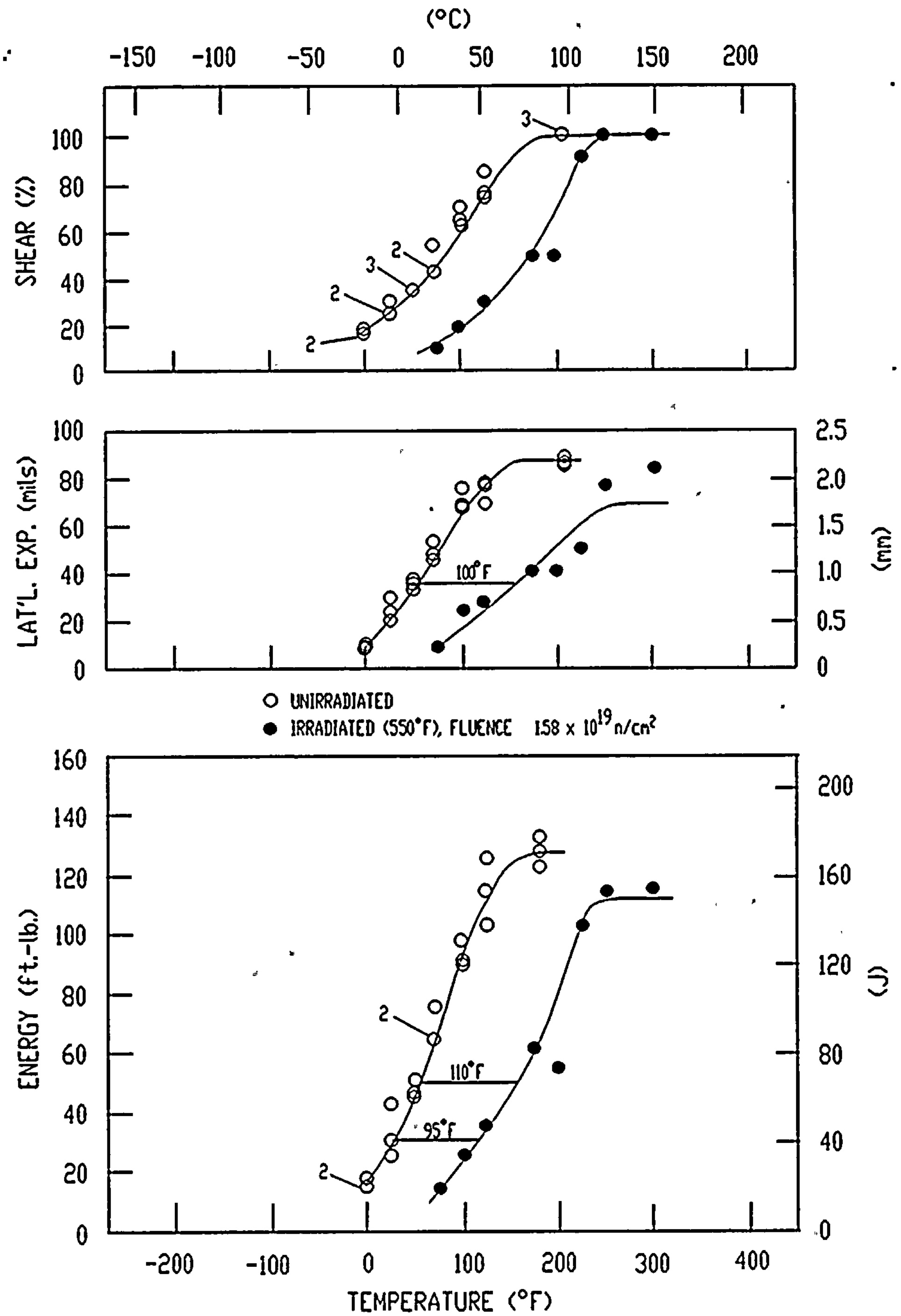


Figure 5-1. Charpy V-Notch Impact Properties for D. C. Cook Unit 2 Reactor Vessel Intermediate Shell Plate C5521-2 (Longitudinal Orientation)

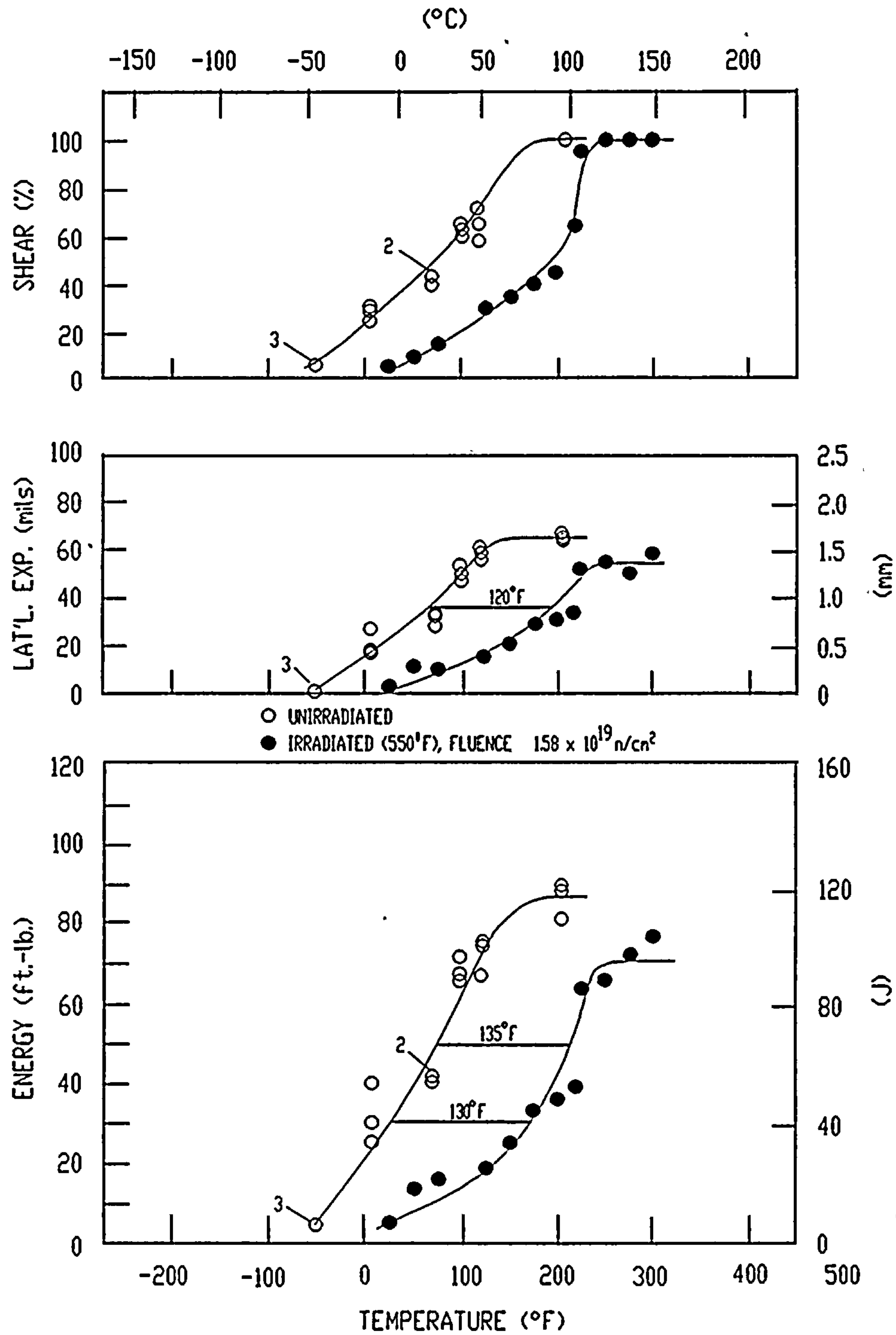


Figure 5-2. Charpy V-Notch Impact Properties for D. C. Cook Unit 2 Reactor Vessel Intermediate Shell Plate C5521-2 (Transverse Orientation)

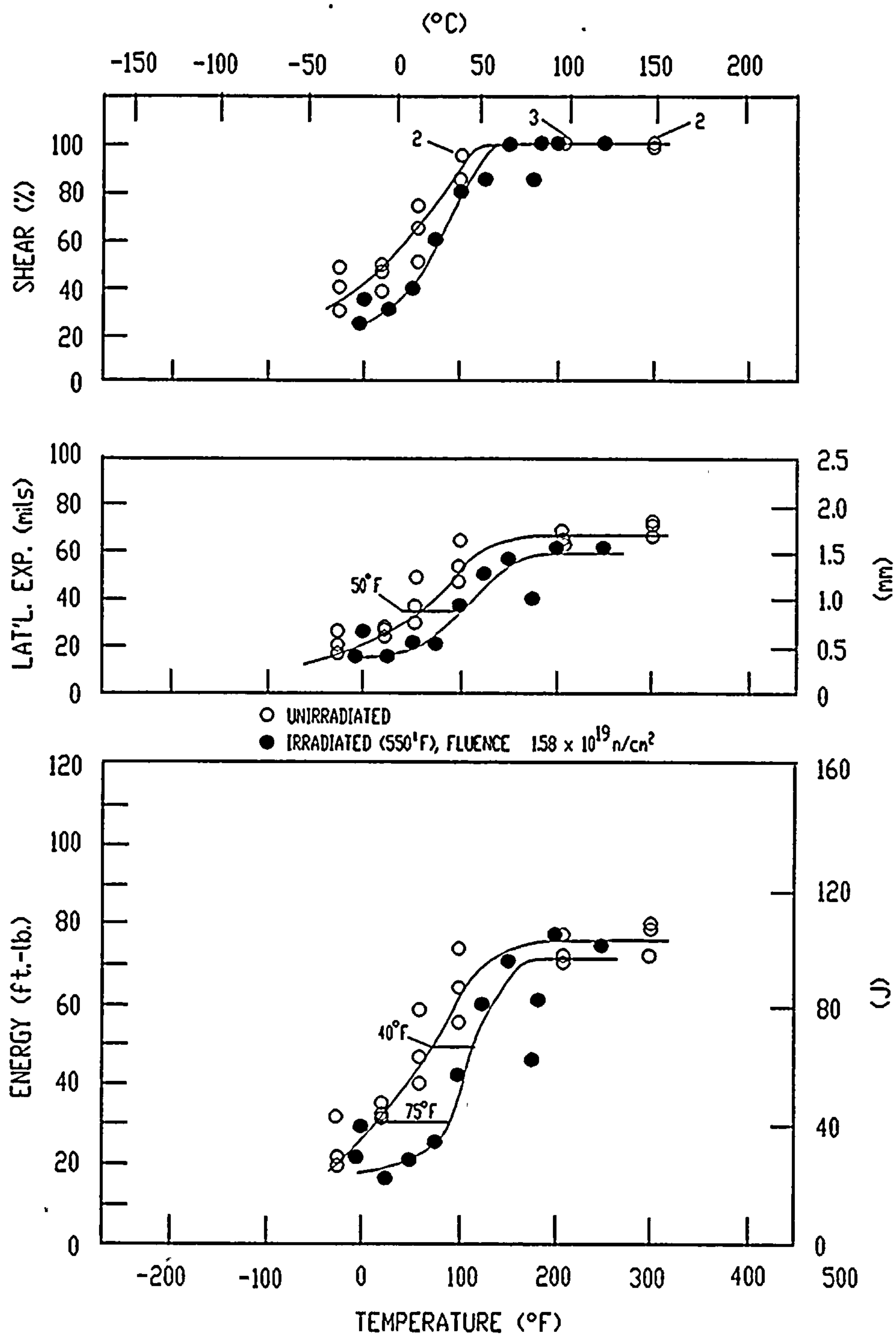


Figure 5-3. Charpy V-Notch Impact Properties for D. C. Cook Unit 2 Reactor Vessel Surveillance Weld Metal

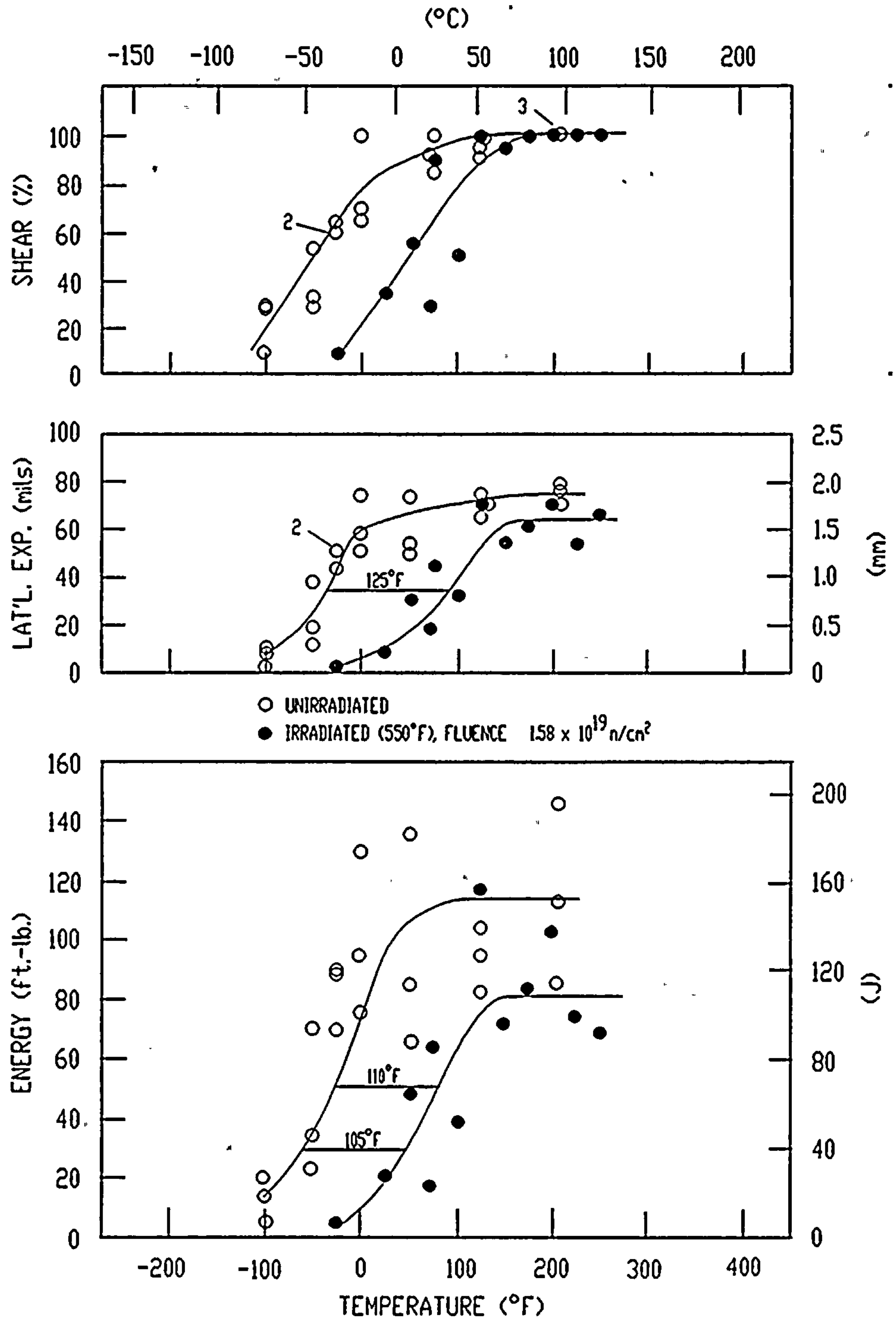
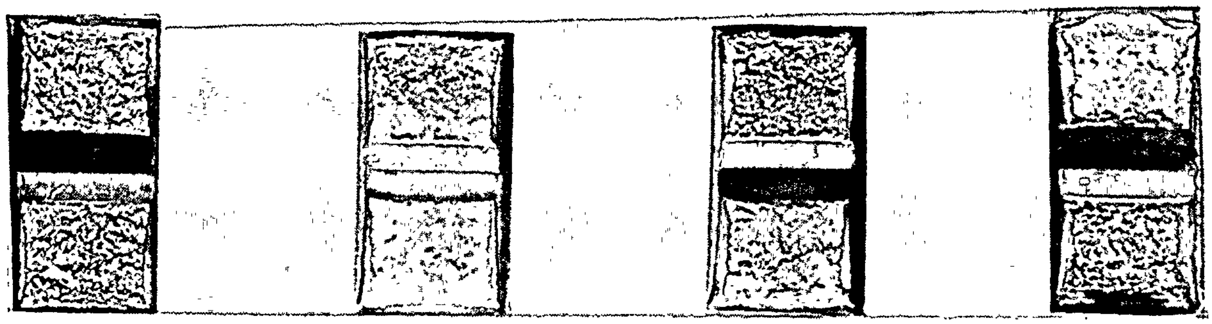
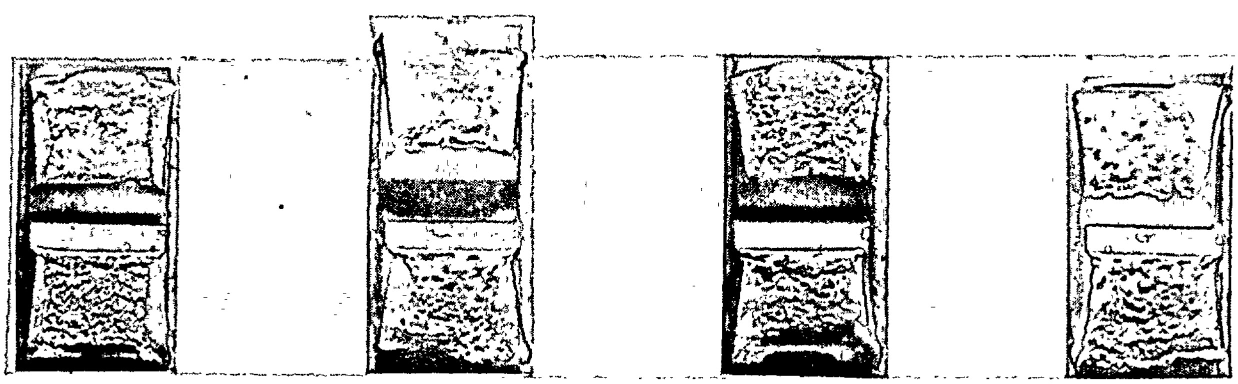


Figure 5-4. Charpy V-Notch Impact Properties for D. C. Cook Unit 2 Reactor Vessel Weld Heat-Affected-Zone Metal



ML45 ML48 ML42 ML44



ML41 ML43 ML46 ML47

Figure 5-5. Charpy Impact Specimen Fracture Surfaces for D. C. Cook Unit 2 Reactor Vessel Intermediate Shell Plate C5521-2 (Longitudinal Orientation)

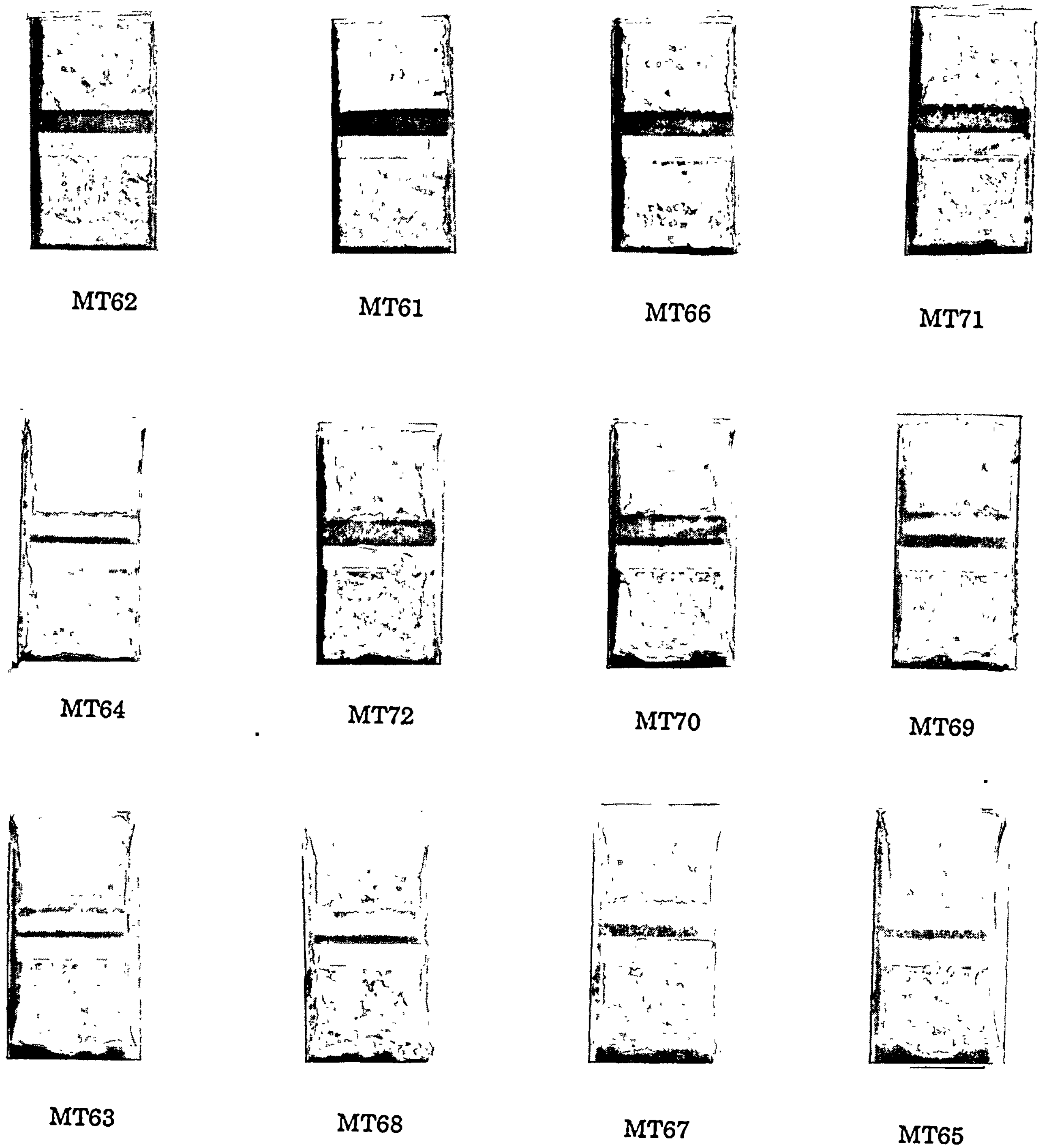


Figure 5-6. Charpy Impact Specimen Fracture Surfaces for D. C. Cook Unit 2 Reactor Vessel Intermediate Shell Plate C5521-2 (Transverse Orientation)

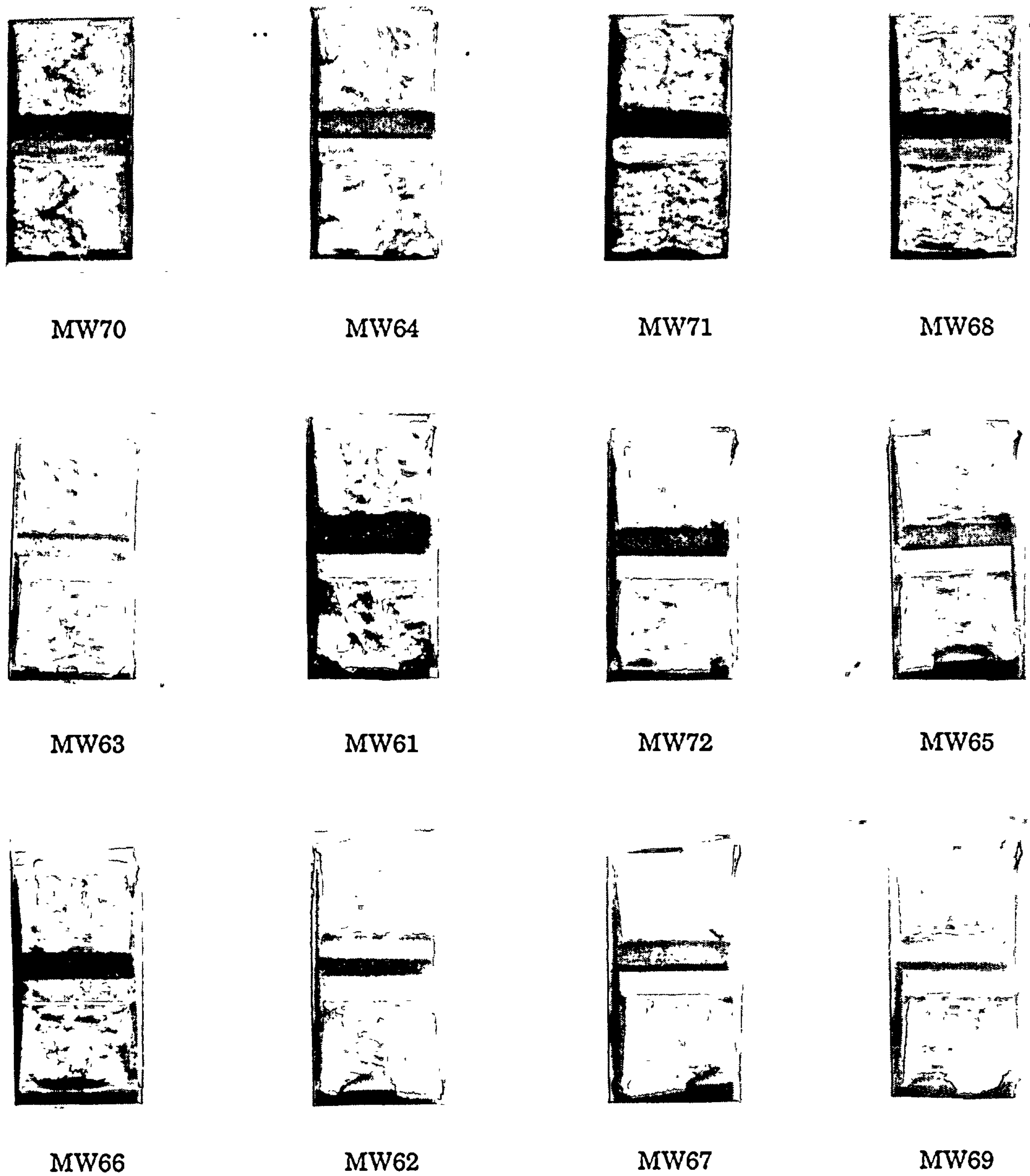


Figure 5-7. Charpy Impact Specimen Fracture Surfaces for D. C. Cook Unit 2 Reactor Vessel Surveillance Weld Metal

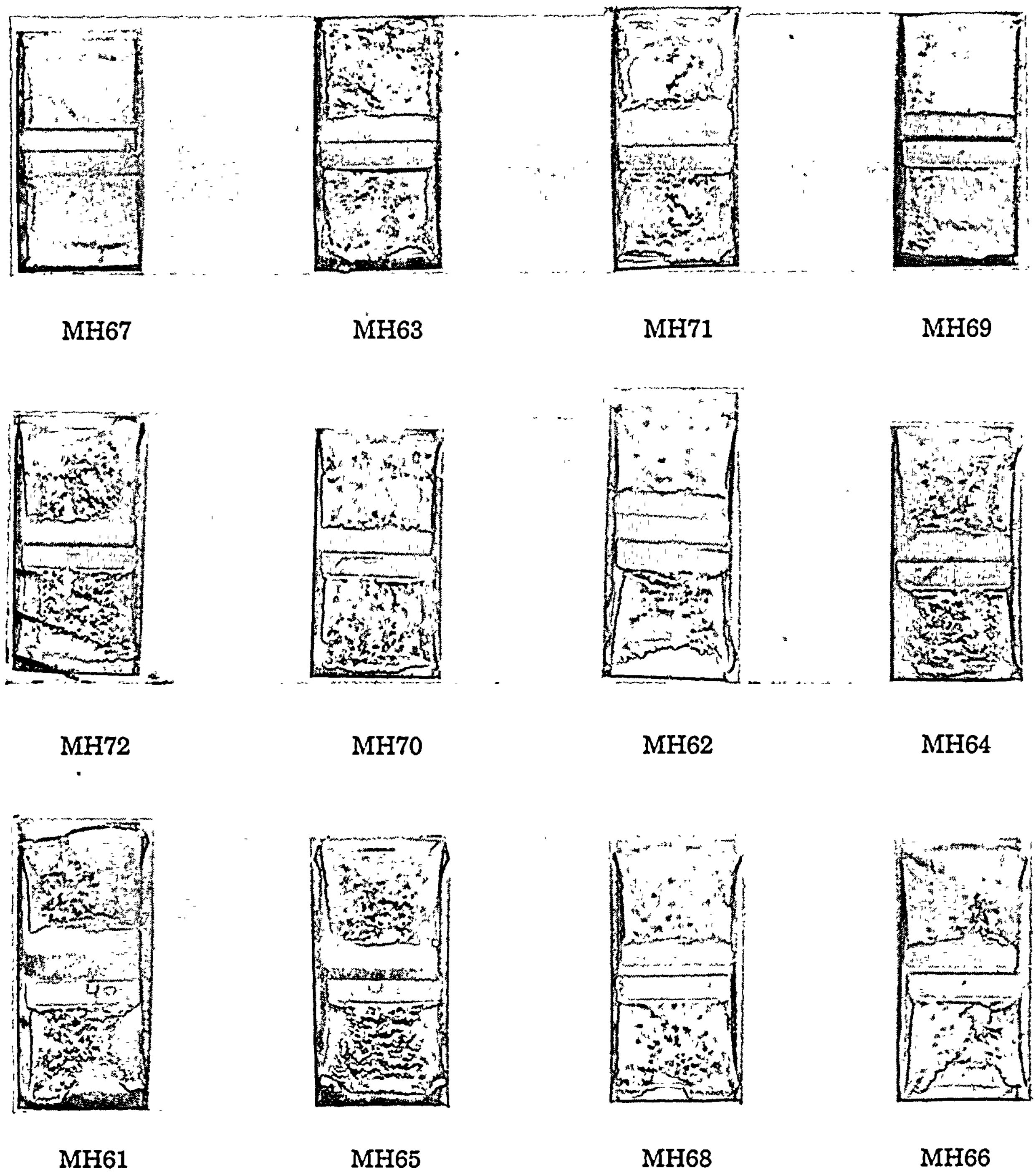


Figure 5-8. Charpy Impact Specimen Fracture Surfaces for D. C. Cook Unit 2 Reactor Vessel Weld Heat-Affected-Zone Metal

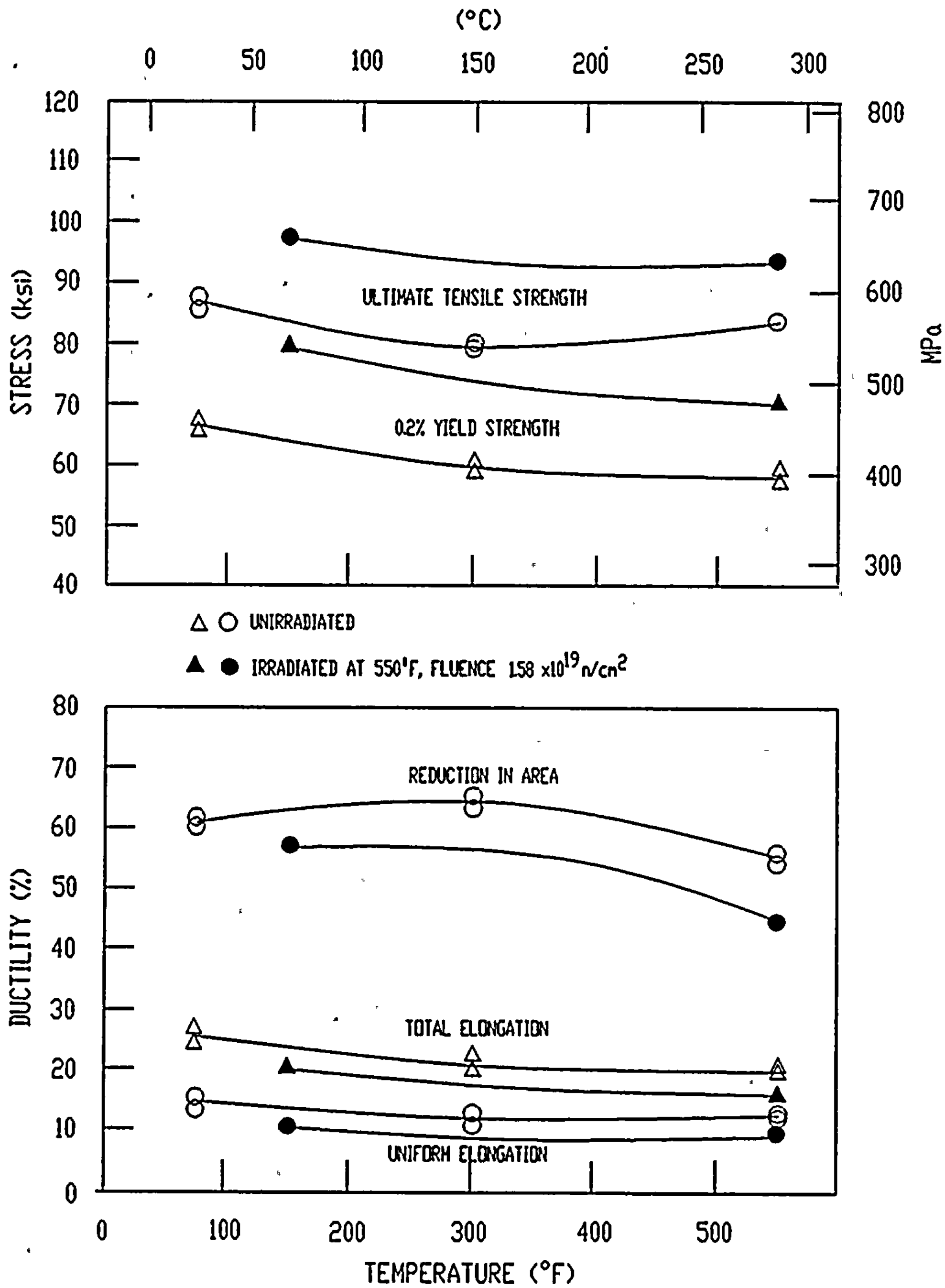


Figure 5-9. Tensile Properties for D. C. Cook Unit 2 Reactor Vessel Intermediate Shell Plate C5521-2 (Transverse Orientation)

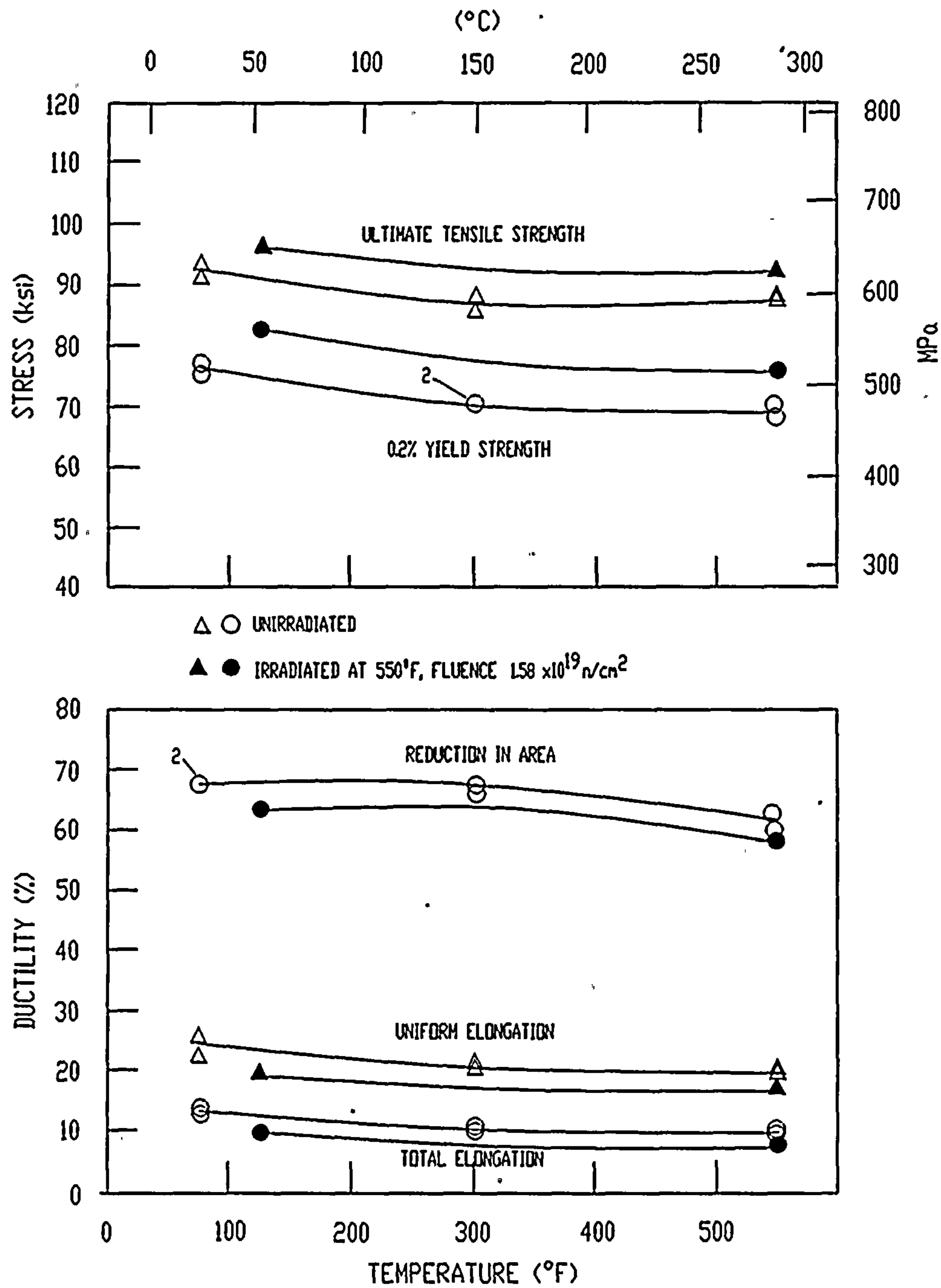
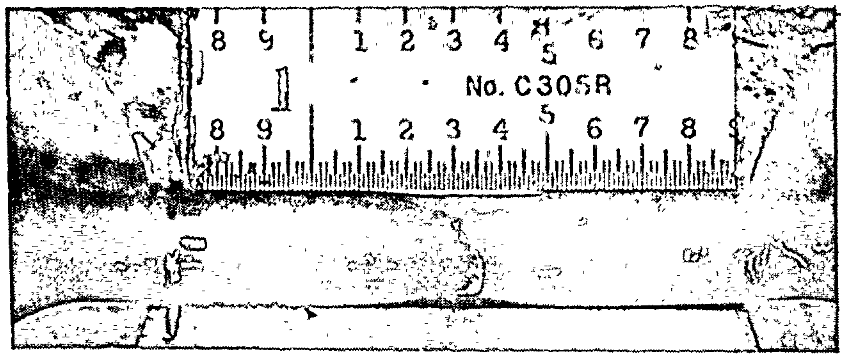
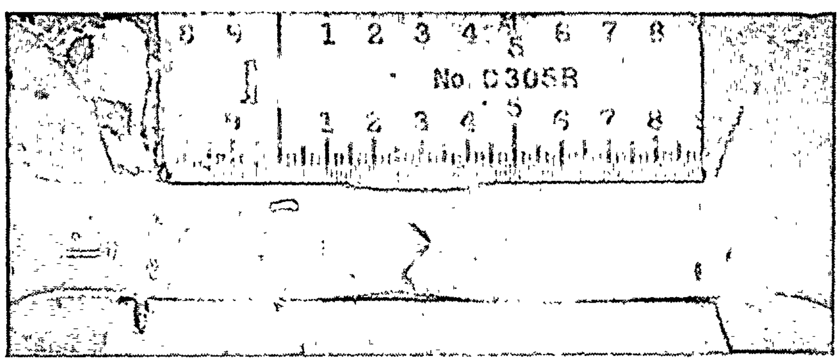


Figure 5-10. Tensile Properties for D. C. Cook Unit 2 Reactor Vessel Surveillance Weld Metal

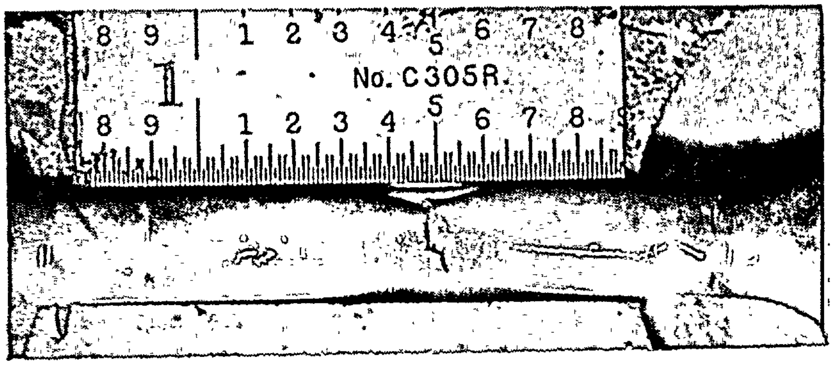


Specimen MT11 150°F

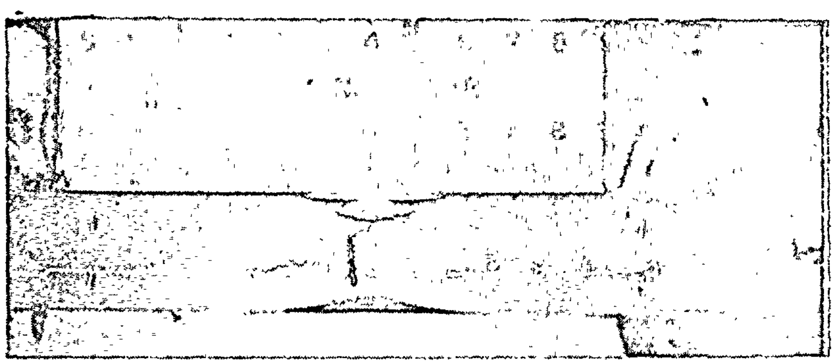


Specimen MT12 550°F

Figure 5-11. Fractured Tensile Specimens from D. C. Cook Unit 2 Reactor Vessel Intermediate Shell Plate C5521-2 (Transverse Orientation)

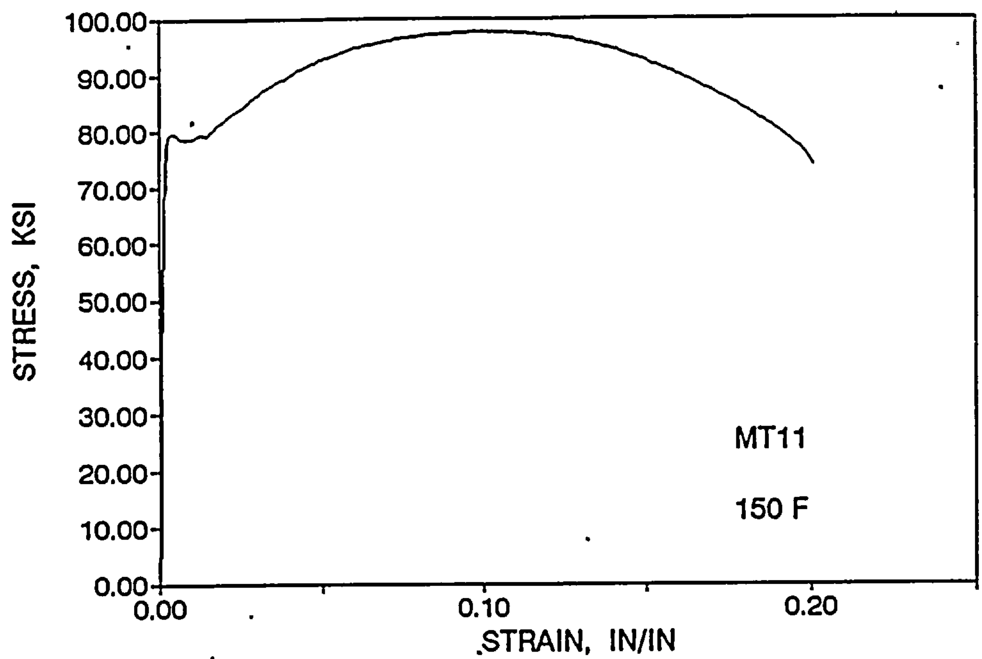


Specimen MW11 125°F



Specimen MW12 550°F

Figure 5-12. Fractured Tensile Specimens from D. C. Cook Unit 2 Reactor Vessel Surveillance Weld Metal



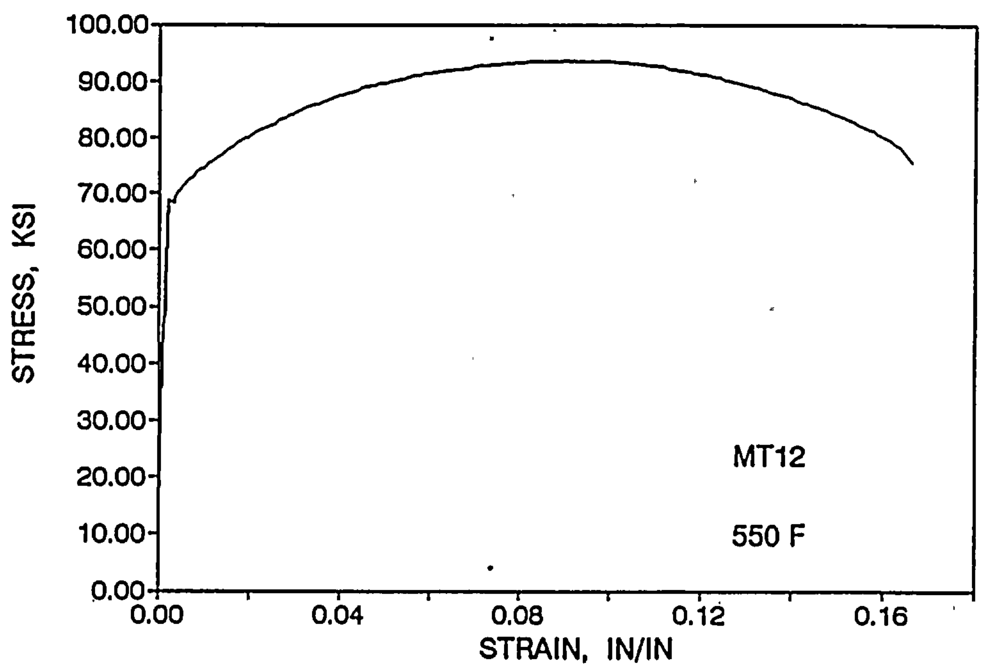
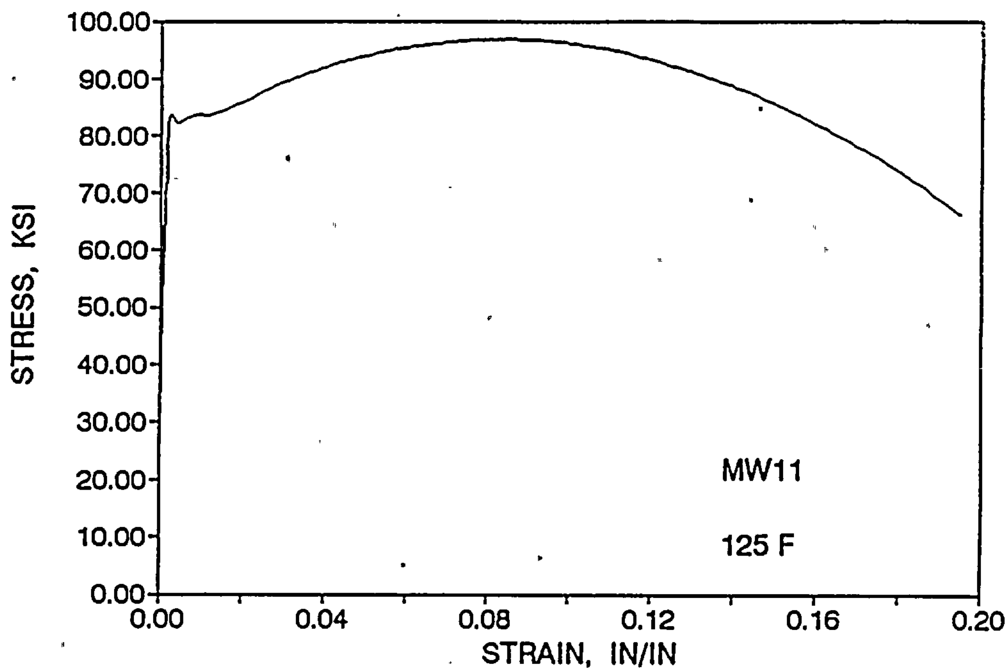


Figure 5-13. Engineering Stress-Strain Curves for Plate C5521-2 Tensile Specimens MT11 and MT12 (Transverse Orientation)



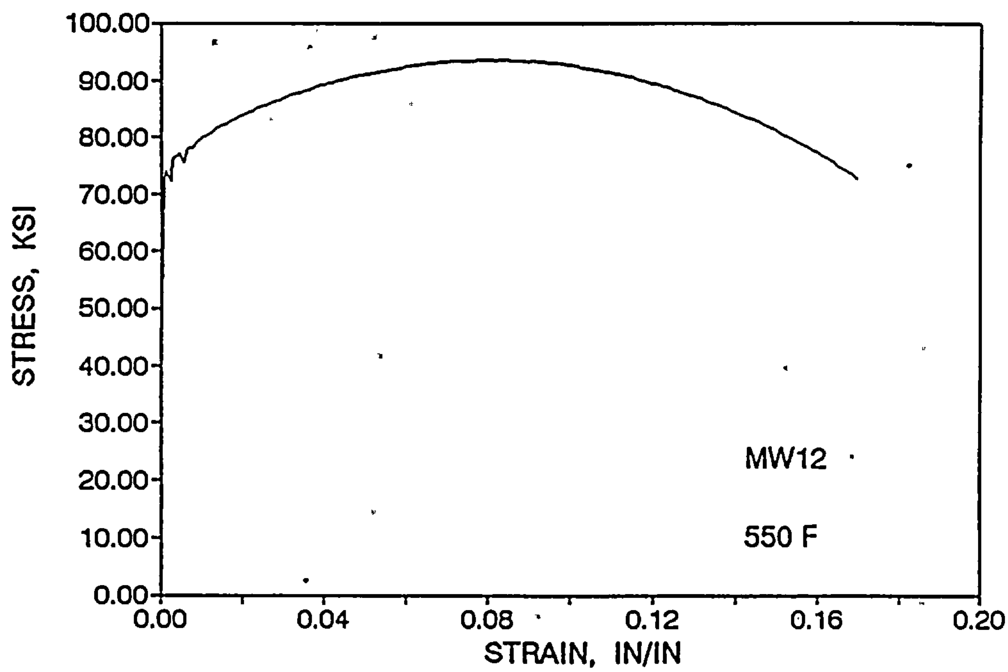


Figure 5-14. Engineering Stress-Strain Curves for Weld Metal Tensile Specimens MW11 and MW12

SECTION 6.0
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 Damage to 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 at the surveillance capsule and with the projected exposure of the pressure vessel 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. The capsules are located at azimuthal angles of 4.0°, 40.0°, 140.0°, 176.0°, 184.0°, 220.0°, 320.0°, and 356.0° 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 1.0 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 each cycle 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-in 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[15] and the SAILOR cross-section library[16]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In these analyses anisotropic 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 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 importance functions to the D C Cook Unit 2 reactor, therefore, 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 distribution used in the plant specific adjoint calculations was taken from the fuel cycle design reports for the first eight operating cycles of D C Cook Unit 2[17 through 19].

Selected results from the neutron transport analyses 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 symmetric 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 Cycles 1 through 8 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 2 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 [20 through 33]. 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 2 reactor during Cycles 1 through 8 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.

Values of key fast neutron exposure parameters were derived from the measured reaction rates using the FERRET least squares adjustment code [34]. 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:

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

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

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

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 [35]. 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:

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

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

For the a priori calculated fluxes, a short-range correlation of $\gamma = 6$ 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[36]. Maerker's results are closely duplicated when $\gamma = 6$. 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 1.58×10^{19} n/cm^2 (E > 1.0 MeV) with an associated σ uncertainty of $\pm$ 8%. 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 falls within $\pm$ 8% for all fast neutron exposure parameters listed. The thermal neutron exposure calculated for the exposure period undepredicted the measured value by approximately a factor of two.

Neutron exposure projections at key locations on the pressure vessel inner radius are given in Table 6-13. Along with the current (8.65 EFPY) exposure derived from the Capsule U measurements, projections are also provided for an exposure period of 16 EFPY and to end of vessel design life (32 EFPY). In the evaluation of the future exposure of the reactor pressure vessel the exposure rates averaged over the first eight cycles of operation were employed.

In the calculation of exposure gradients for use in the development of heatup and cooldown curves for the D C Cook Unit 2 reactor coolant system, exposure projections to 16 EFPY and 32 EFPY were also evaluated. 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 2 surveillance capsules. These data may be used as a guide in establishing future withdrawal schedules for the remaining capsules.

In order to provide a consistent data base for comparison with measured shift data, the dosimetry sets from previously withdrawn surveillance capsules (X, Y, and T) were re-evaluated using the previously described least squares adjustment methodology along with current reaction cross-sections and nuclear data. The results of those re-evaluations were as follows:

	FLUENCE [E > 1.0 MeV] (n/cm^2)
Capsule X	1.06×10^{19}
Capsule Y	6.83×10^{18}
Capsule T	2.64×10^{18}

The 1σ uncertainty associated with each of these fluence evaluations is 8%.

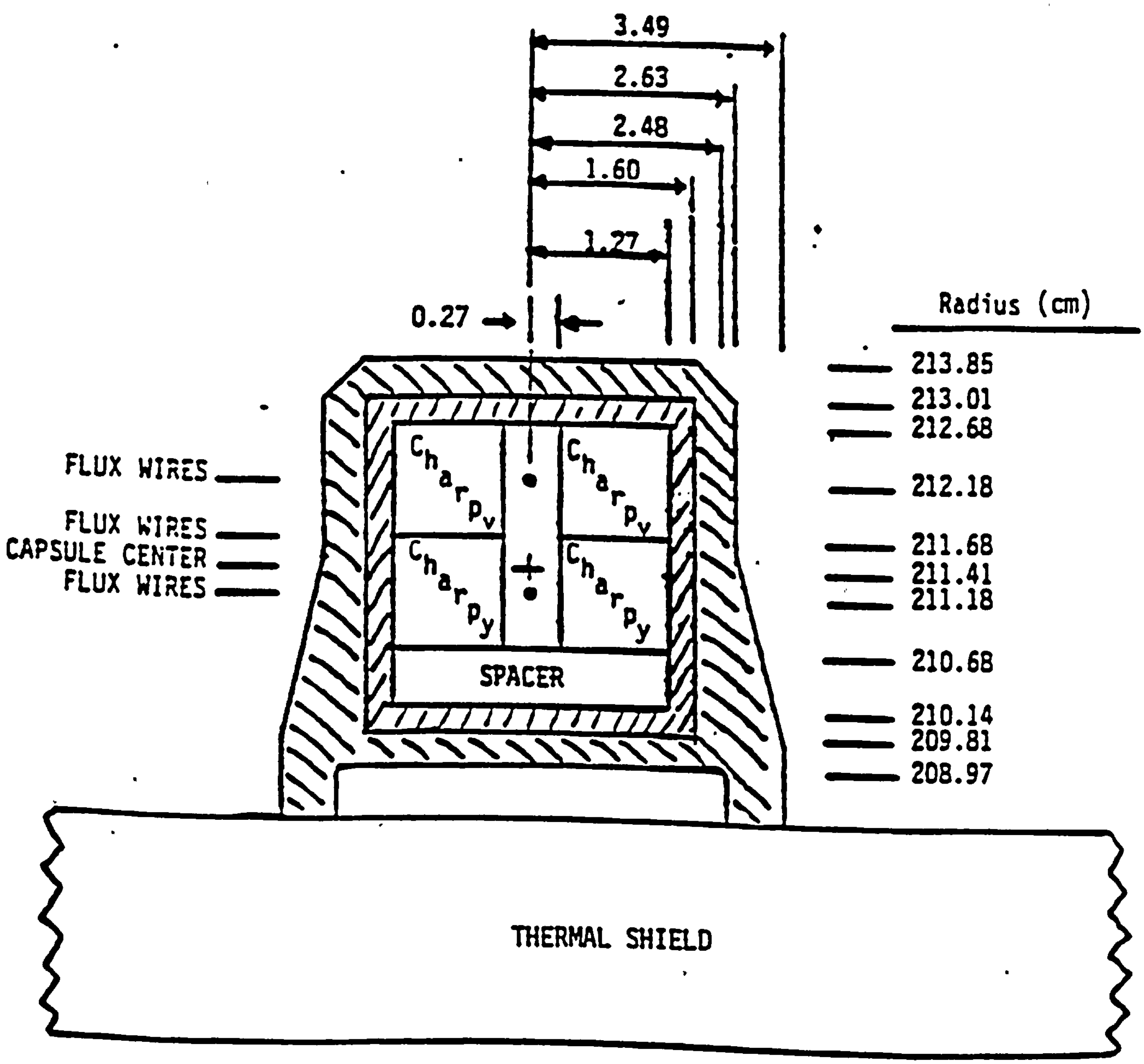


Figure 6-1. Plan View of a Reactor Vessel Surveillance Capsule

TABLE 6-1

CALCULATED FAST NEUTRON EXPOSURE PARAMETERS
AT THE SURVEILLANCE CAPSULE CENTER

	ϕ(E > 1.0MeV) [n/cm^2-sec]		ϕ(E > 0.1Mev) [n/cm^2-sec]		Iron Displacement Rate [dpa/sec]	
	4.0°	40.0°	4.0°	40.0°	4.0°	40.0°
DESIGN BASIS	2.82×10^{10}	9.05×10^{10}	8.15×10^{10}	3.04×10^{11}	4.58×10^{-11}	1.55×10^{-10}
CYCLE 1	2.12×10^{10}	6.68×10^{10}	6.13×10^{10}	2.24×10^{11}	3.43×10^{-11}	1.14×10^{-10}
CYCLE 2	2.29×10^{10}	6.62×10^{10}	6.62×10^{10}	2.22×10^{11}	3.71×10^{-11}	1.13×10^{-10}
CYCLE 3	2.21×10^{10}	5.46×10^{10}	6.39×10^{10}	1.83×10^{11}	3.58×10^{-11}	9.34×10^{-11}
CYCLE 4	2.19×10^{10}	5.47×10^{10}	6.33×10^{10}	1.84×10^{11}	3.55×10^{-11}	9.35×10^{-11}
CYCLE 5	2.24×10^{10}	4.93×10^{10}	6.47×10^{10}	1.66×10^{11}	3.63×10^{-11}	8.43×10^{-11}
CYCLE 6	1.93×10^{10}	5.19×10^{10}	5.58×10^{10}	1.74×10^{11}	3.13×10^{-11}	8.87×10^{-11}
CYCLE 7	2.12×10^{10}	4.70×10^{10}	6.13×10^{10}	1.58×10^{11}	3.43×10^{-11}	8.04×10^{-11}
CYCLE 8	1.66×10^{10}	4.76×10^{10}	4.80×10^{10}	1.60×10^{11}	2.69×10^{-11}	8.14×10^{-11}

TABLE 6-2

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

$\phi(E > 1.0MeV)$ [n/cm^2-sec]

	0.0°	15.0°	30.0°	45.0°
DESIGN BASIS	8.43×10^{09}	1.36×10^{10}	1.72×10^{10}	2.68×10^{10}
CYCLE 1	6.34×10^{09}	1.01×10^{10}	1.26×10^{10}	1.94×10^{10}
CYCLE 2	6.74×10^{09}	1.03×10^{10}	1.24×10^{10}	1.92×10^{10}
CYCLE 3	6.43×10^{09}	9.59×10^{09}	1.06×10^{10}	1.60×10^{10}
CYCLE 4	6.58×10^{09}	1.04×10^{10}	1.11×10^{10}	1.63×10^{10}
CYCLE 5	6.50×10^{09}	9.54×10^{09}	9.83×10^{09}	1.46×10^{10}
CYCLE 6	5.87×10^{09}	9.46×10^{09}	1.02×10^{10}	1.53×10^{10}
CYCLE 7	6.39×10^{09}	9.99×10^{09}	9.63×10^{09}	1.41×10^{10}
CYCLE 8	4.92×10^{09}	7.40×10^{09}	9.53×10^{09}	1.42×10^{10}

$\phi(E > 0.1MeV)$ [n/cm^2-sec]

	0.0°	15.0°	30.0°	45.0°
DESIGN BASIS	2.11×10^{10}	3.41×10^{10}	4.34×10^{10}	6.96×10^{10}
CYCLE 1	1.59×10^{10}	2.54×10^{10}	3.18×10^{10}	5.04×10^{10}
CYCLE 2	1.69×10^{10}	2.59×10^{10}	3.12×10^{10}	4.99×10^{10}
CYCLE 3	1.61×10^{10}	2.41×10^{10}	2.67×10^{10}	4.16×10^{10}
CYCLE 4	1.65×10^{10}	2.61×10^{10}	2.80×10^{10}	4.24×10^{10}
CYCLE 5	1.63×10^{10}	2.39×10^{10}	2.48×10^{10}	3.80×10^{10}
CYCLE 6	1.47×10^{10}	2.37×10^{10}	2.57×10^{10}	3.98×10^{10}
CYCLE 7	1.60×10^{10}	2.51×10^{10}	2.43×10^{10}	3.67×10^{10}
CYCLE 8	1.23×10^{10}	1.86×10^{10}	2.40×10^{10}	3.69×10^{10}

TABLE 6-2 (Continued)

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

	Iron Atom Displacement Rate [dpa/sec]			
	0.0°	15.0°	30.0°	45.0°
DESIGN BASIS	1.37×10^{-11}	2.19×10^{-11}	2.73×10^{-11}	4.26×10^{-11}
CYCLE 1	1.03×10^{-11}	1.63×10^{-11}	2.00×10^{-11}	3.08×10^{-11}
CYCLE 2	1.10×10^{-11}	1.66×10^{-11}	1.97×10^{-11}	3.05×10^{-11}
CYCLE 3	1.05×10^{-11}	1.54×10^{-11}	1.69×10^{-11}	2.54×10^{-11}
CYCLE 4	1.07×10^{-11}	1.67×10^{-11}	1.76×10^{-11}	2.59×10^{-11}
CYCLE 5	1.06×10^{-11}	1.54×10^{-11}	1.56×10^{-11}	2.32×10^{-11}
CYCLE 6	9.57×10^{-12}	1.52×10^{-11}	1.62×10^{-11}	2.43×10^{-11}
CYCLE 7	1.04×10^{-11}	1.61×10^{-11}	1.53×10^{-11}	2.24×10^{-11}
CYCLE 8	8.02×10^{-12}	1.19×10^{-11}	1.52×10^{-11}	2.26×10^{-11}

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 6-4

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	Product Half-Life	Fission 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

MONTHLY THERMAL GENERATION DURING THE FIRST EIGHT FUEL CYCLES OF THE D C COOK UNIT 2 REACTOR

MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)
3/78	53096	9/81	2430714	3/85	2461488	9/88	0
4/78	521821	10/81	284784	4/85	2449523	10/88	0
5/78	653969	11/81	2435848	5/85	2532441	11/88	0
6/78	1365478	12/81	2517865	6/85	2451623	12/88	0
7/78	1247083	1/82	2295944	7/85	1049002	1/89	0
8/78	1529472	2/82	2196190	8/85	60639	2/89	0
9/78	2178779	3/82	833555	9/85	0	3/89	775260
10/78	2231119	4/82	2391274	10/85	163249	4/89	2288800
11/78	848238	5/82	2516937	11/85	1372641	5/89	2530450
12/78	2476056	6/82	2331168	12/85	2019347	6/89	1297315
1/79	2240714	7/82	2496782	1/86	2043640	7/89	2508038
2/79	2220562	8/82	1011517	2/86	1360957	8/89	2155830
3/79	2483455	9/82	2241332	3/86	0	9/89	2452143
4/79	2164269	10/82	2293400	4/86	0	10/89	2493553
5/79	1449347	11/82	1575311	5/86	0	11/89	2355817
6/79	0	12/82	0	6/86	0	12/89	2454307
7/79	2258164	1/83	341534	7/86	980980	1/90	861130
8/79	2513690	2/83	2242228	8/86	2034055	2/90	2282581
9/79	2266726	3/83	2533602	9/86	1973118	3/90	2534067
10/79	1522346	4/83	2428234	10/86	2014579	4/90	2452883
11/79	0	5/83	2461540	11/86	1974208	5/90	2165049
12/79	0	6/83	1851461	12/86	2039056	6/90	1767222
1/80	584404	7/83	1711373	1/87	2039325	7/90	0
2/80	2209403	8/83	2343637	2/87	1776049	8/90	0
3/80	2418799	9/83	2269321	3/87	159603	9/90	0
4/80	2354329	10/83	1188161	4/87	456573	10/90	0
5/80	2483250	11/83	453959	5/87	2080553	11/90	1338296
6/80	2187611	12/83	2383687	6/87	1849770	12/90	1887935
7/80	1408949	1/84	2435731	7/87	1409763	1/91	2533040
8/80	2496594	2/84	2235476	8/87	1485651	2/91	2110364
9/80	2393783	3/84	733977	9/87	0	3/91	2241782
10/80	1414143	4/84	0	10/87	1258092	4/91	2447136
11/80	0	5/84	0	11/87	1973726	5/91	2495857
12/80	1488758	6/84	0	12/87	1995088	6/91	2410166
1/81	2505373	7/84	1417277	1/88	2039814	7/91	2435150
2/81	2271684	8/84	2341526	2/88	1900060	8/91	590907
3/81	1053877	9/84	2325725	3/88	2038466	9/91	2357045
4/81	0	10/84	2423846	4/88	1432639	10/91	2519131
5/81	449803	11/84	2056182	5/88	0	11/91	1912720
6/81	2374202	12/84	1059199	6/88	0	12/91	2474282
7/81	1775877	1/85	1361108	7/88	0	1/92	2442527
8/81	2338703	2/85	2271484	8/88	0	2/92	1219168

TABLE 6-8
MEASURED SENSOR ACTIVITIES AND REACTION RATES

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Capsule Center Reaction Rate (RPS/NUCLEUS)
Cu-63 (n,α) Co-60			
Top-Middle	1.23×10^5	2.67×10^5	
Middle	1.23×10^5	2.67×10^5	
Bottom-Middle	1.24×10^5	2.70×10^5	
Average	1.23×10^5	2.68×10^5	3.94×10^{-17}
Fe-54(n,p) Mn-54			
Top	9.19×10^5	2.15×10^6	
Top-Middle	9.45×10^5	2.21×10^6	
Middle	9.50×10^5	2.22×10^6	
Bottom-Middle	8.59×10^5	2.01×10^6	
Bottom	9.30×10^5	2.18×10^6	
Average	9.21×10^5	2.15×10^6	3.64×10^{-15}
Ni-58 (n,p) Co-58			
Top	3.93×10^6	3.21×10^7	
Middle	3.93×10^6	3.21×10^7	
Bottom	3.96×10^6	3.24×10^7	
Average	3.94×10^6	3.22×10^7	5.41×10^{-15}
U-238 (n,f) Cs-137 (Cd)			
Middle	4.88×10^5	2.94×10^6	1.94×10^{-14}

TABLE 6-8
MEASURED SENSOR ACTIVITIES AND REACTION RATES - cont'd

Monitor and Axial Location	Measured Activity (dis/sec-gm)	Saturated Activity (dis/sec-gm)	Capsule Center Reaction Rate (RPS/NUCLEUS)
Np-237(n,f) Cs-137 (Cd)			
Middle	4.26×10^{6}	2.57×10^{7}	1.61×10^{-13}
Co-59 (n,γ) Co-60			
Top	1.88×10^{7}	4.09×10^{7}	
Bottom	1.74×10^{7}	3.78×10^{7}	
Average	1.81×10^{7}	3.94×10^{7}	2.70×10^{-12}
Co-59 (n,γ) Co-60 (Cd)			
Bottom	7.32×10^{6}	1.59×10^{7}	1.20×10^{-12}

TABLE 6-9

SUMMARY OF NEUTRON DOSIMETRY RESULTS

	TIME AVERAGED EXPOSURE RATES	
ϕ (E > 1.0 MeV) {n/cm^2-sec}	5.78×10^{10}	± 8%
ϕ (E > 0.1 MeV) {n/cm^2-sec}	2.00×10^{11}	± 15%
dpa/sec	9.78×10^{-11}	± 10%
ϕ (E < 0.414 eV) {n/cm^2-sec}	6.29×10^{10}	± 20%
	INTEGRATED CAPSULE EXPOSURE	
Φ (E > 1.0 MeV) {n/cm^2}	1.58×10^{19}	± 8%
Φ (E > 0.1 MeV) {n/cm^2}	5.46×10^{19}	± 15%
dpa	2.67×10^{-2}	± 10%
Φ (E < 0.414 eV) {n/cm^2}	1.72×10^{19}	± 20%

NOTE: Total Irradiation Time = 8.65 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	3.94×10^{-17}	3.94×10^{-17}	1.00
Fe-54 (n,p) Mn-54	3.64×10^{-15}	3.74×10^{-15}	1.03
Ni-58 (n,p) Co-58	5.41×10^{-15}	5.28×10^{-15}	0.98
U-238 (n,f) Cs-137 (Cd)	1.94×10^{-14}	1.93×10^{-14}	1.00
Np-237 (n,f) Cs-137 (Cd)	1.61×10^{-13}	1.61×10^{-13}	1.00
Co-59 (n,γ) Co-60	2.70×10^{-12}	2.69×10^{-12}	1.00
Co-59 (n,γ) Co-60 (Cd)	1.20×10^{-12}	1.20×10^{-12}	1.00

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	$1.73x10^{1}$	$3.36x10^{6}$	28	$9.12x10^{-3}$	$8.36x10^{9}$
2	$1.49x10^{1}$	$8.20x10^{6}$	29	$5.53x10^{-3}$	$1.05x10^{10}$
3	$1.35x10^{1}$	$3.76x10^{7}$	30	$3.36x10^{-3}$	$3.25x10^{9}$
4	$1.16x10^{1}$	$9.51x10^{7}$	31	$2.84x10^{-3}$	$3.09x10^{9}$
5	$1.00x10^{1}$	$2.29x10^{8}$	32	$2.40x10^{-3}$	$2.98x10^{9}$
6	$8.61x10^{0}$	$4.11x10^{8}$	33	$2.04x10^{-3}$	$8.60x10^{9}$
7	$7.41x10^{0}$	$9.86x10^{8}$	34	$1.23x10^{-3}$	$8.31x10^{9}$
8	$6.07x10^{0}$	$1.42x10^{9}$	35	$7.49x10^{-4}$	$7.99x10^{9}$
9	$4.97x10^{0}$	$2.92x10^{9}$	36	$4.54x10^{-4}$	$7.73x10^{9}$
10	$3.68x10^{0}$	$3.69x10^{9}$	37	$2.75x10^{-4}$	$8.15x10^{9}$
11	$2.87x10^{0}$	$7.37x10^{9}$	38	$1.67x10^{-4}$	$8.85x10^{9}$
12	$2.23x10^{0}$	$9.24x10^{9}$	39	$1.01x10^{-4}$	$8.81x10^{9}$
13	$1.74x10^{0}$	$1.21x10^{10}$	40	$6.14x10^{-5}$	$8.79x10^{9}$
14	$1.35x10^{0}$	$1.22x10^{10}$	41	$3.73x10^{-5}$	$8.66x10^{9}$
15	$1.11x10^{0}$	$2.08x10^{10}$	42	$2.26x10^{-5}$	$8.48x10^{9}$
16	$8.21x10^{-1}$	$2.22x10^{10}$	43	$1.37x10^{-5}$	$8.31x10^{9}$
17	$6.39x10^{-1}$	$2.18x10^{10}$	44	$8.32x10^{-6}$	$8.02x10^{9}$
18	$4.98x10^{-1}$	$1.55x10^{10}$	45	$5.04x10^{-6}$	$7.55x10^{9}$
19	$3.88x10^{-1}$	$2.05x10^{10}$	46	$3.06x10^{-6}$	$7.18x10^{9}$
20	$3.02x10^{-1}$	$2.27x10^{10}$	47	$1.86x10^{-6}$	$6.72x10^{9}$
21	$1.83x10^{-1}$	$2.16x10^{10}$	48	$1.13x10^{-6}$	$5.29x10^{9}$
22	$1.11x10^{-1}$	$1.71x10^{10}$	49	$6.83x10^{-7}$	$6.28x10^{9}$
23	$6.74x10^{-2}$	$1.23x10^{10}$	50	$4.14x10^{-7}$	$1.03x10^{10}$
24	$4.09x10^{-2}$	$7.42x10^{9}$	51	$2.51x10^{-7}$	$1.05x10^{10}$
25	$2.55x10^{-2}$	$8.90x10^{9}$	52	$1.52x10^{-7}$	$1.05x10^{10}$
26	$1.99x10^{-2}$	$4.83x10^{9}$	53	$9.24x10^{-8}$	$3.16x10^{10}$
27	$1.50x10^{-2}$	$6.52x10^{9}$			

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

TABLE 6-12

COMPARISON OF CALCULATED AND MEASURED
EXPOSURE LEVELS FOR CAPSULE U

	Calculated	Measured	C/M
$\Phi(E > 1.0\ MeV)\ \{n/cm^2\}$	1.49×10^{19}	1.58×10^{19}	0.94
$\Phi(E > 0.1\ MeV)\ \{n/cm^2\}$	5.01×10^{19}	5.46×10^{19}	0.92
dpa	2.55×10^{-2}	2.67×10^{-2}	0.96
$\Phi(E < 0.414\ eV)\ \{n/cm^2\}$	8.84×10^{18}	1.72×10^{19}	0.51

TABLE 6-13
NEUTRON EXPOSURE PROJECTIONS AT KEY LOCATIONS
ON THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

8.65 EFPY

	0°	10°	15°	30°	45°
Φ (E > 1.0 Mev) [n/cm2]	1.79×10^{18}	2.44×10^{18}	2.77×10^{18}	3.09×10^{18}	4.65×10^{18}
Φ (E > 0.1 MeV) [n/cm2]	4.48×10^{18}	6.12×10^{18}	6.95×10^{18}	7.79×10^{18}	1.21×10^{19}
Iron Atom Displacements [dpa]	2.92×10^{-3}	3.93×10^{-3}	4.46×10^{-3}	4.91×10^{-3}	7.39×10^{-3}

16.0 EFPY

	0°	10°	15°	30°	45°
Φ (E > 1.0 Mev) [n/cm2]	3.31×10^{18}	4.51×10^{18}	5.12×10^{18}	5.72×10^{18}	8.58×10^{18}
Φ (E > 0.1 MeV) [n/cm2]	8.28×10^{18}	1.13×10^{19}	1.29×10^{19}	1.44×10^{19}	2.23×10^{19}
Iron Atom Displacements [dpa]	5.40×10^{-3}	7.26×10^{-3}	8.24×10^{-3}	9.09×10^{-3}	1.36×10^{-2}

32.0 EFPY

	0°	10°	15°	30°	45°
Φ (E > 1.0 Mev) [n/cm2]	6.63×10^{18}	9.02×10^{18}	1.02×10^{19}	1.14×10^{19}	1.71×10^{19}
Φ (E > 0.1 MeV) [n/cm2]	1.66×10^{19}	2.26×10^{19}	2.56×10^{19}	2.87×10^{19}	4.45×10^{19}
Iron Atom Displacements [dpa]	1.08×10^{-2}	1.45×10^{-2}	1.64×10^{-2}	1.81×10^{-2}	2.72×10^{-2}

TABLE [illegible]-14

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

16 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°	3.31×10^{18}	1.75×10^{18}	3.61×10^{17}	3.31×10^{18}	2.12×10^{18}	7.69×10^{17}
10°	4.51×10^{18}	2.40×10^{18}	4.96×10^{17}	4.51×10^{18}	2.88×10^{18}	1.04×10^{18}
15°	5.12×10^{18}	2.72×10^{18}	5.63×10^{17}	5.12×10^{18}	3.27×10^{18}	1.18×10^{18}
30°	5.72×10^{18}	3.07×10^{18}	6.46×10^{17}	5.72×10^{18}	3.73×10^{18}	1.38×10^{18}
45°	8.58×10^{18}	4.51×10^{18}	9.01×10^{17}	8.58×10^{18}	5.49×10^{18}	1.90×10^{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°	6.63×10^{18}	3.51×10^{18}	7.22×10^{17}	6.63×10^{18}	4.24×10^{18}	1.54×10^{18}
10°	9.02×10^{18}	4.80×10^{18}	9.92×10^{17}	9.02×10^{18}	5.76×10^{18}	2.07×10^{18}
15°	1.02×10^{19}	5.45×10^{18}	1.13×10^{18}	1.02×10^{19}	6.54×10^{18}	2.35×10^{18}
30°	1.14×10^{19}	6.15×10^{18}	1.29×10^{18}	1.14×10^{19}	7.46×10^{18}	2.77×10^{18}
45°	1.71×10^{19}	9.02×10^{18}	1.80×10^{18}	1.71×10^{19}	1.10×10^{19}	3.79×10^{18}

TABLE 6-15

UPDATED LEAD FACTORS FOR D C COOK UNIT 2 SURVEILLANCE CAPSULES

Capsule	Lead Factor
T	3.44[a]
X	3.41[c]
U	3.40[d]
Y	3.44[b]
S	1.30[d]
V	1.30[d]
W	1.30[d]
Z	1.30[d]

(a) Plant specific evaluation based on end of Cycle 1 calculated fluence.
(b) Plant specific evaluation based on end of Cycle 3 calculated fluence.
(c) Plant specific evaluation based on end of Cycle 5 calculated fluence.
(d) Plant specific evaluation based on end of Cycle 8 calculated fluence.

SECTION 7.0
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 2 reactor vessel:

Capsule	Location (deg.)	Capsule Lead Factor	Removal Time [a]	Estimated Fluence (n/cm^2)
T	40	3.44	1.08 (Removed)	2.64×10^{18} (Actual)
Y	320	3.44	3.24 (Removed)	6.83×10^{18} (Actual)
X	220	3.44	5.27 (Removed)	1.06×10^{19} (Actual)
U	140	3.44	8.65 (Removed)	1.58×10^{19} (Actual)
S	4	1.30	32	2.22×10^{19}
Z	356	1.30	Standby	---
W	184	1.30	Standby	---
V	176	1.30	Standby	---

(a) Effective Full Power Years (EFPY) from plant startup.

SECTION 8.0
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APPENDIX A

Load-Time Records for Charpy Specimen Tests

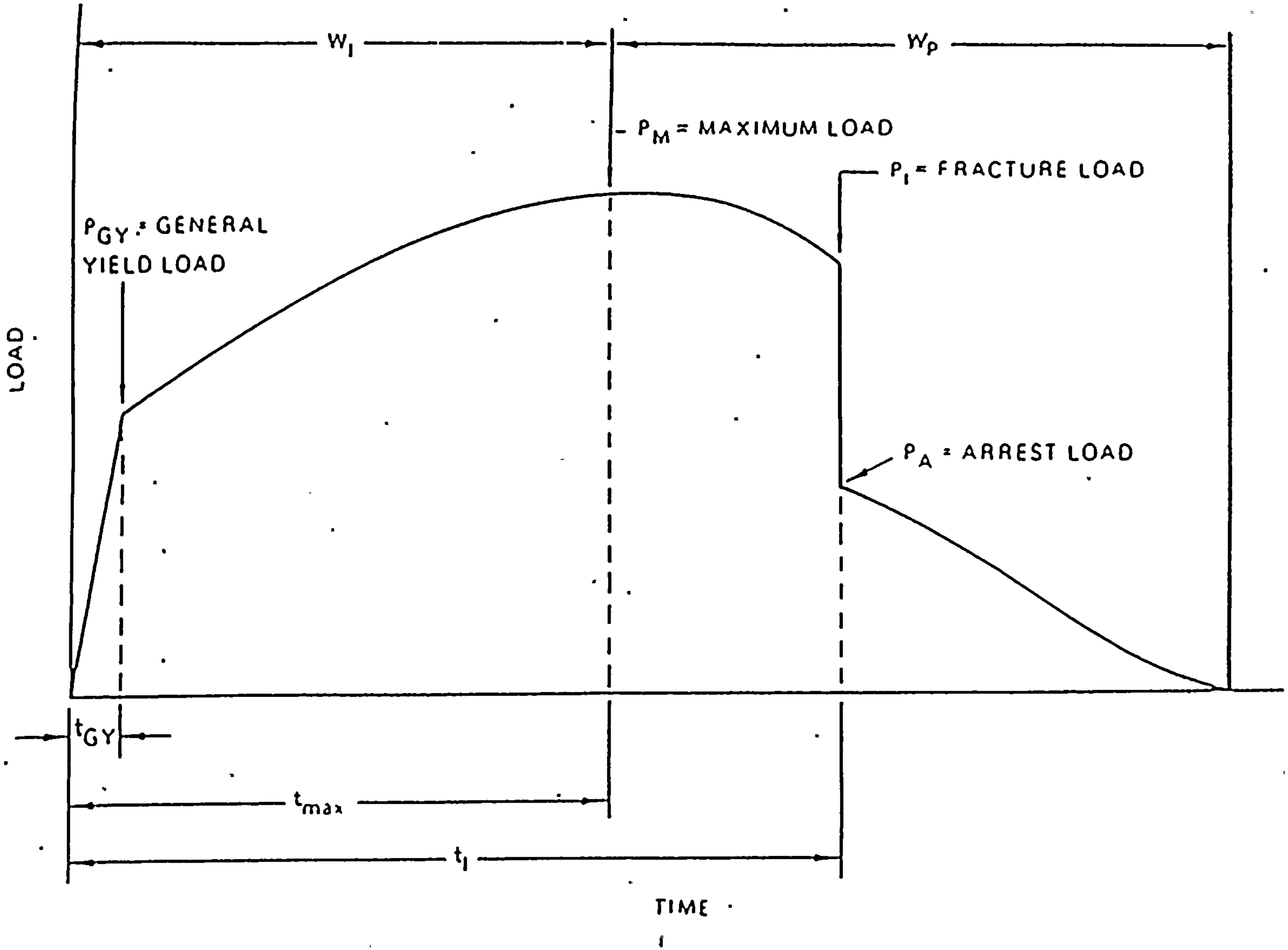
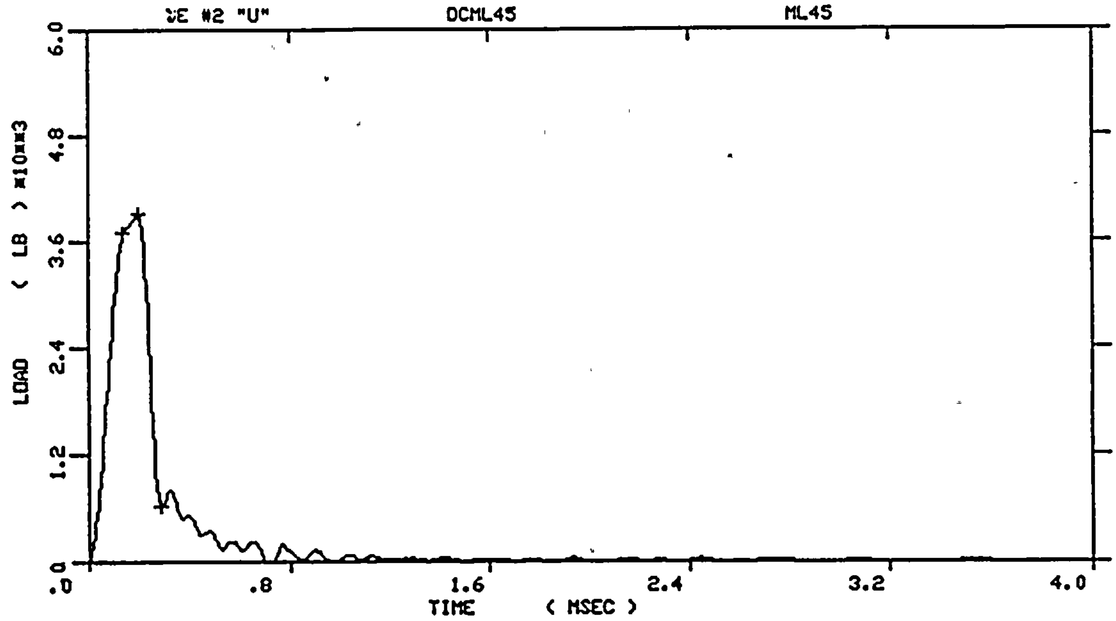


Figure A-1. Idealized load-time record



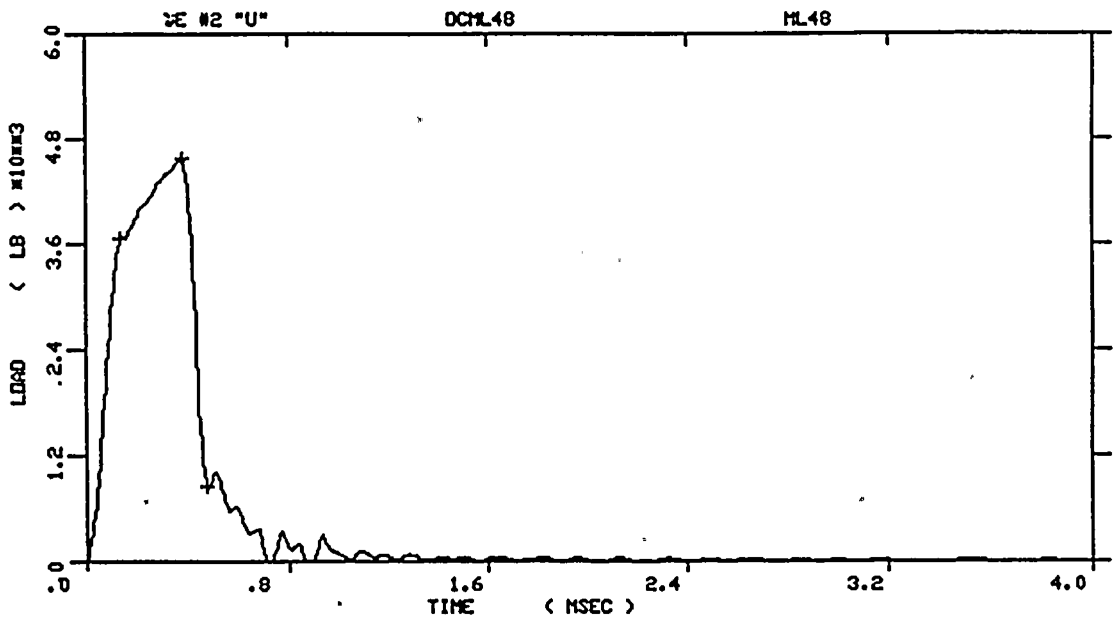
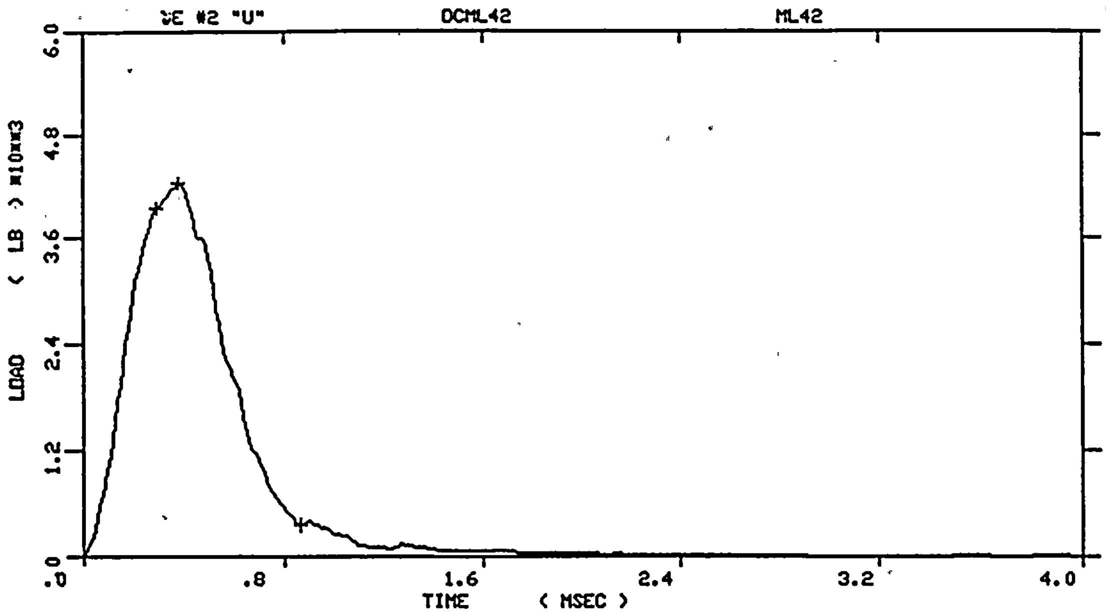


Figure A-2. Load-time records for Specimens ML45 and ML48



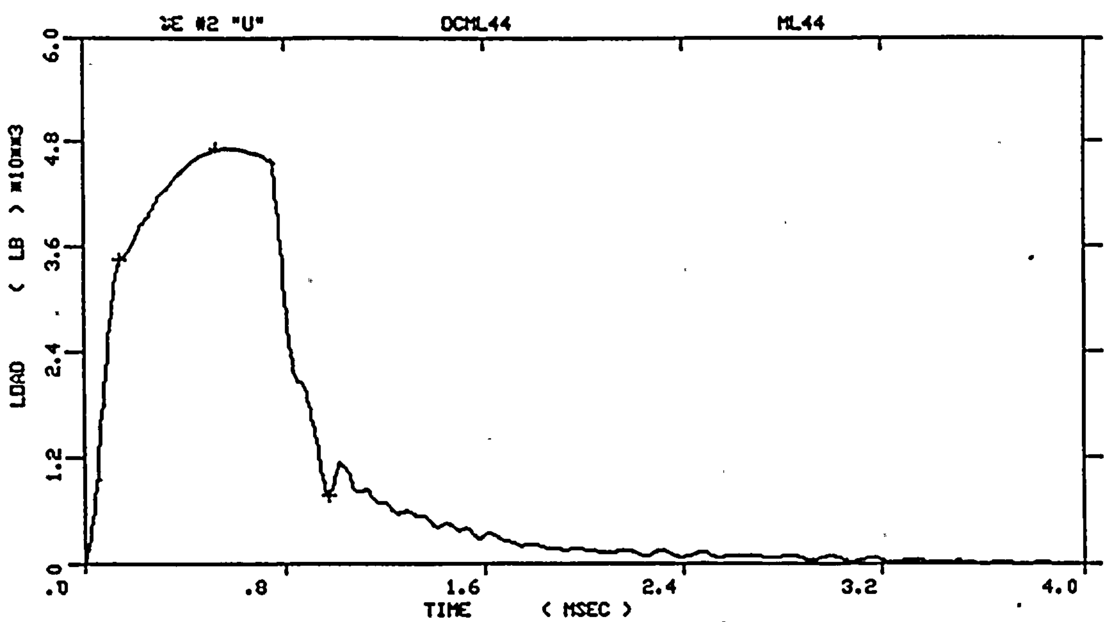
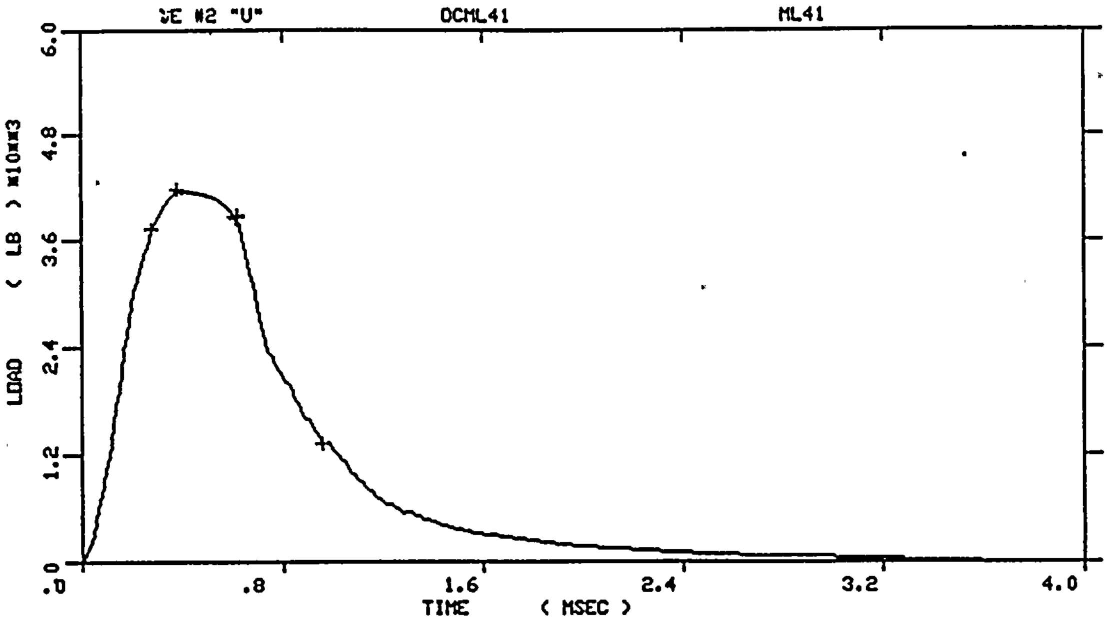


Figure A-3. Load-time records for Specimens ML42 and ML44



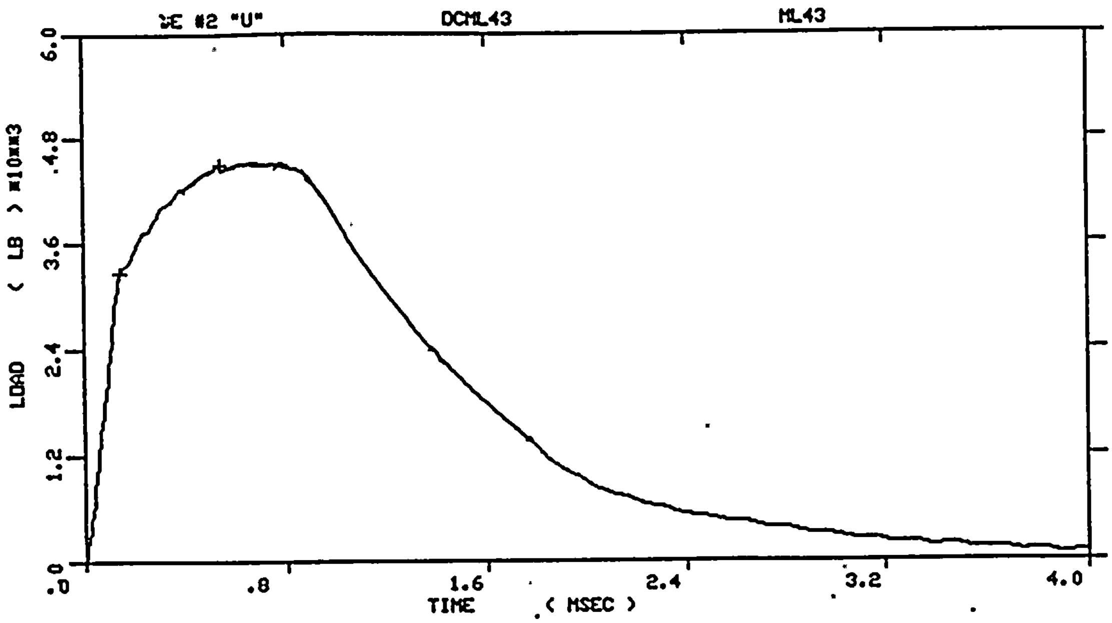
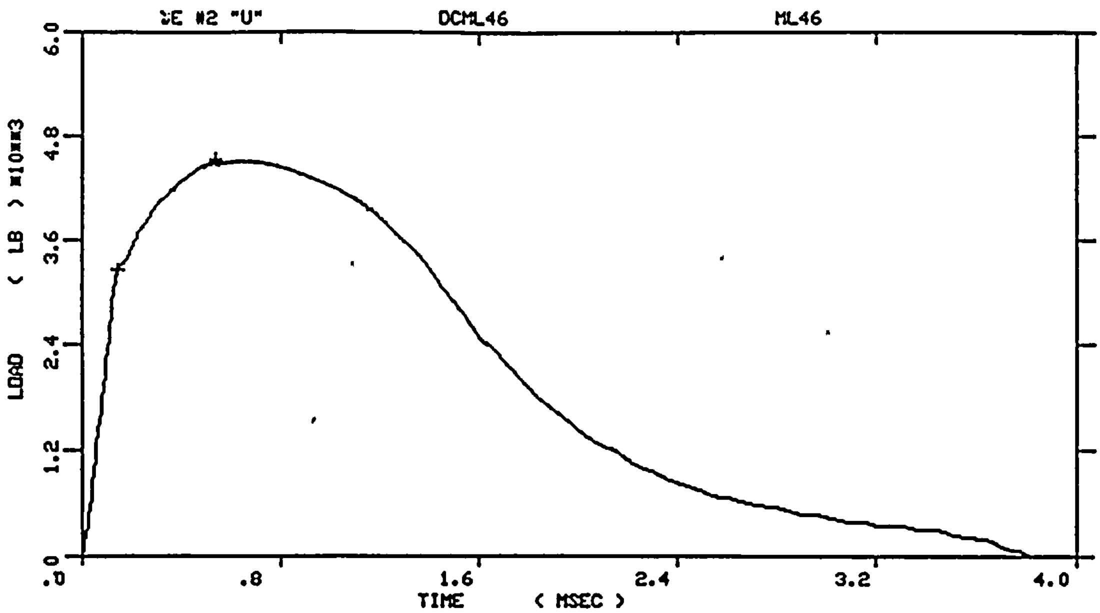


Figure A-4. Load-time records for Specimens ML41 and ML43



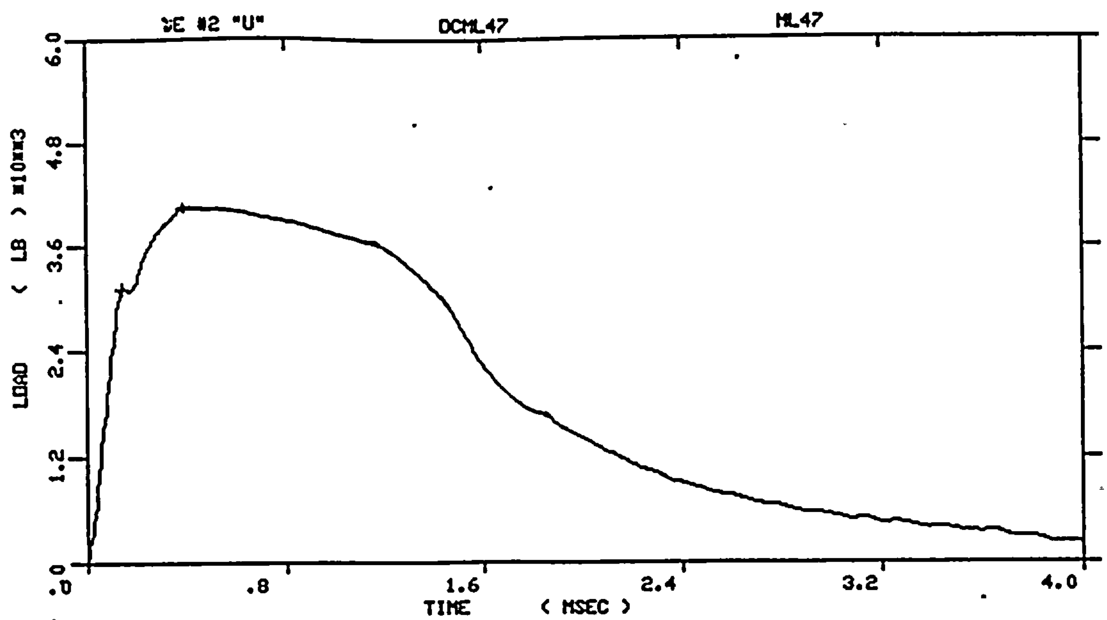
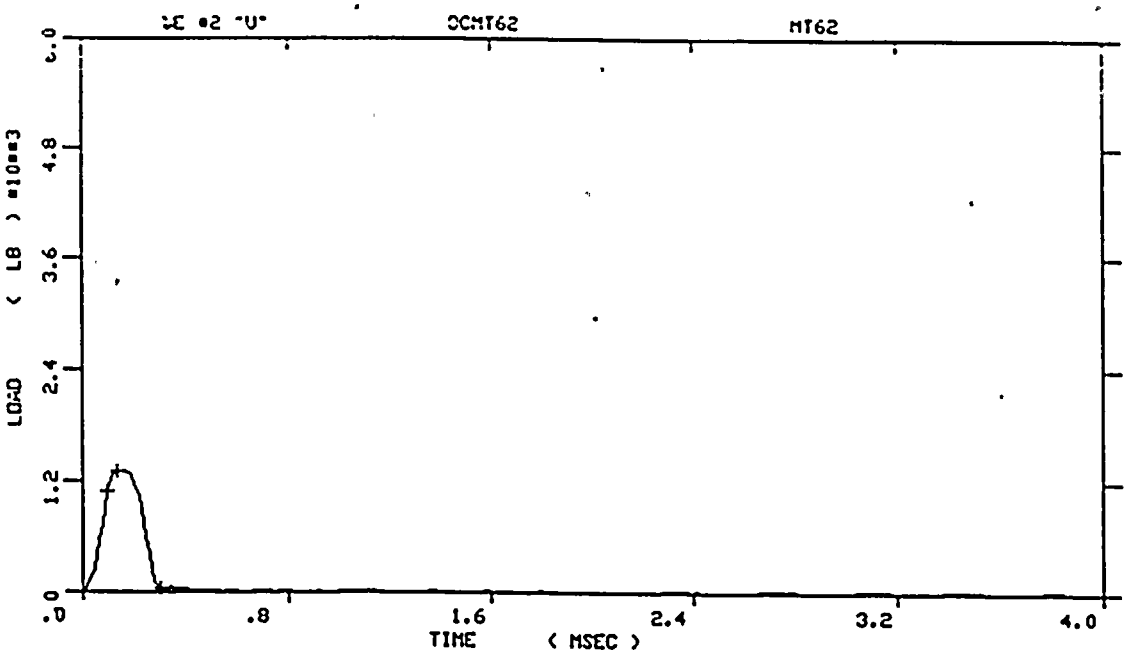


Figure A-5. Load-time records for Specimens ML46 and ML47



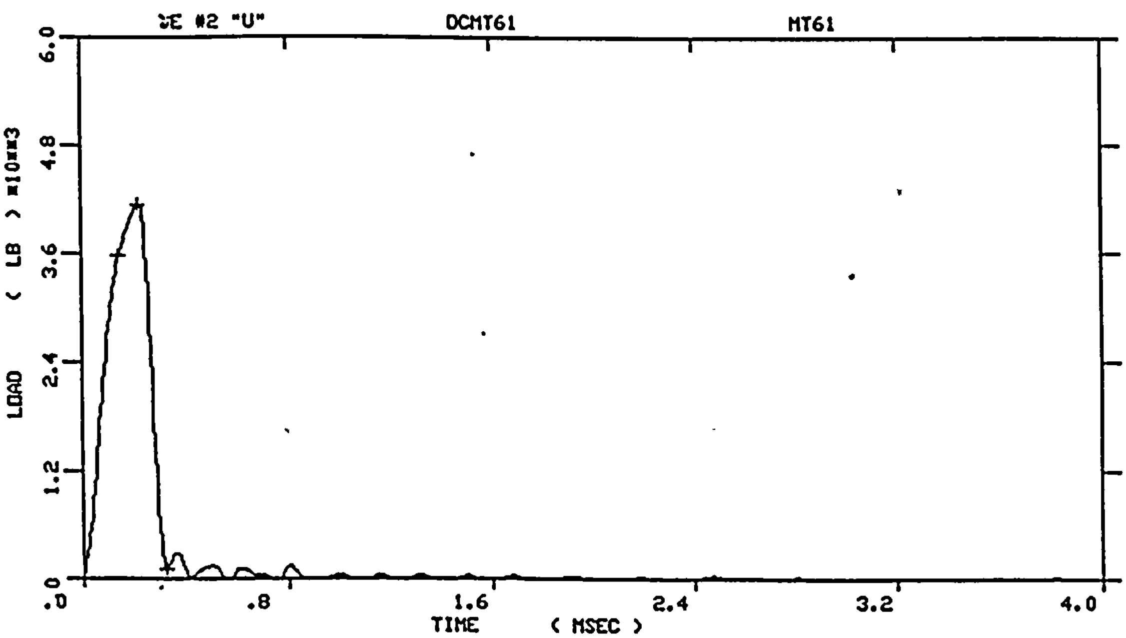
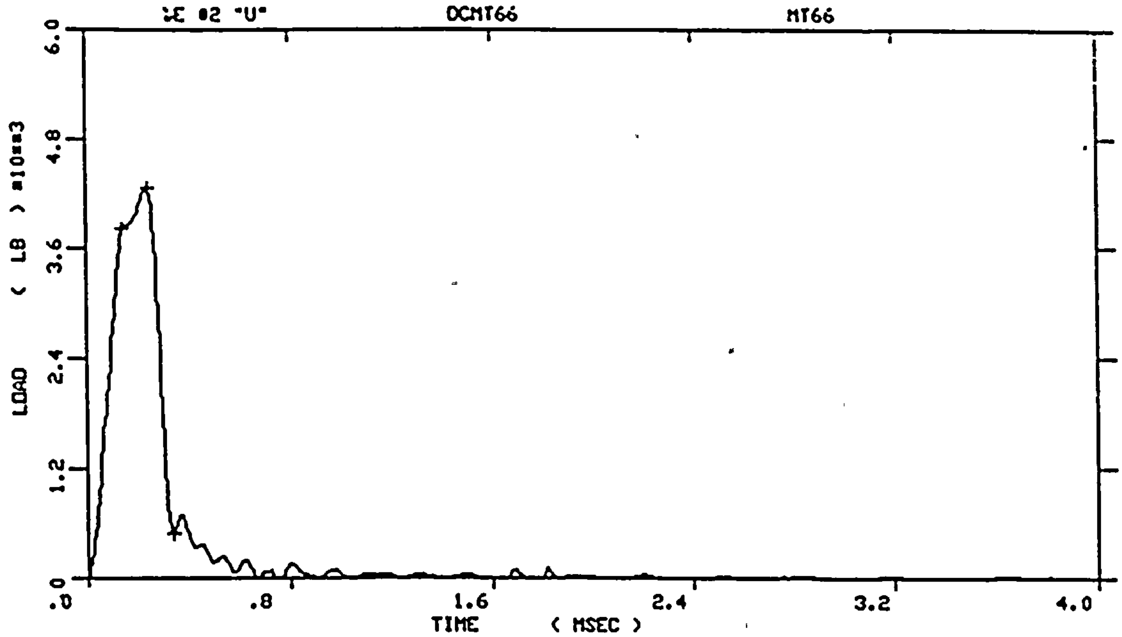


Figure A-6. Load-time records for Specimens MT62 and MT61



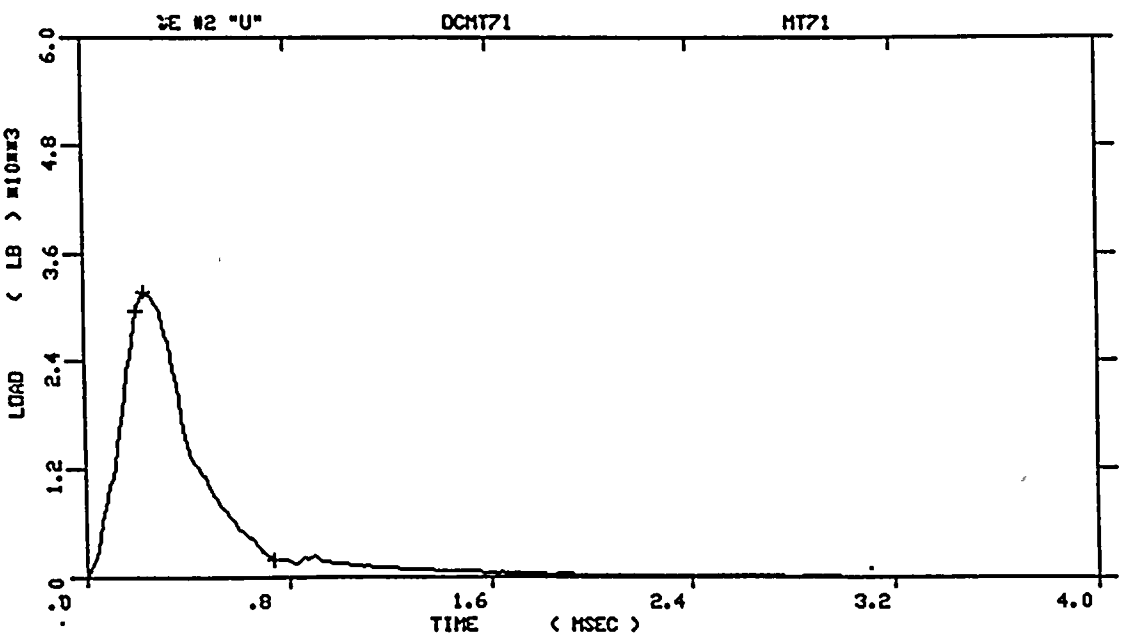
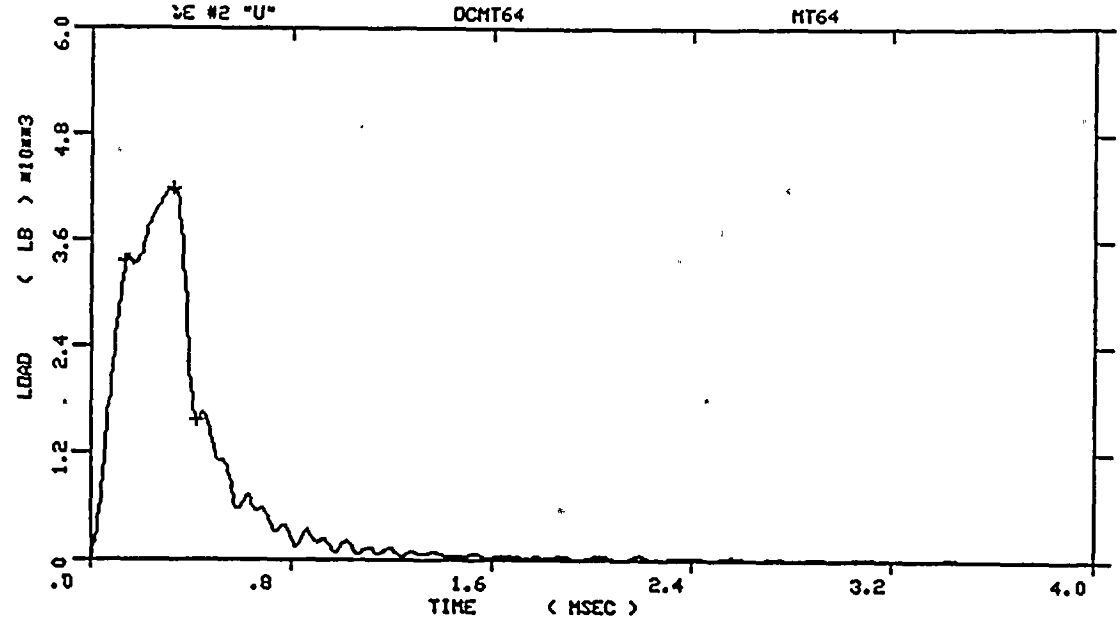


Figure A-7. Load-time records for Specimens MT66 and MT71



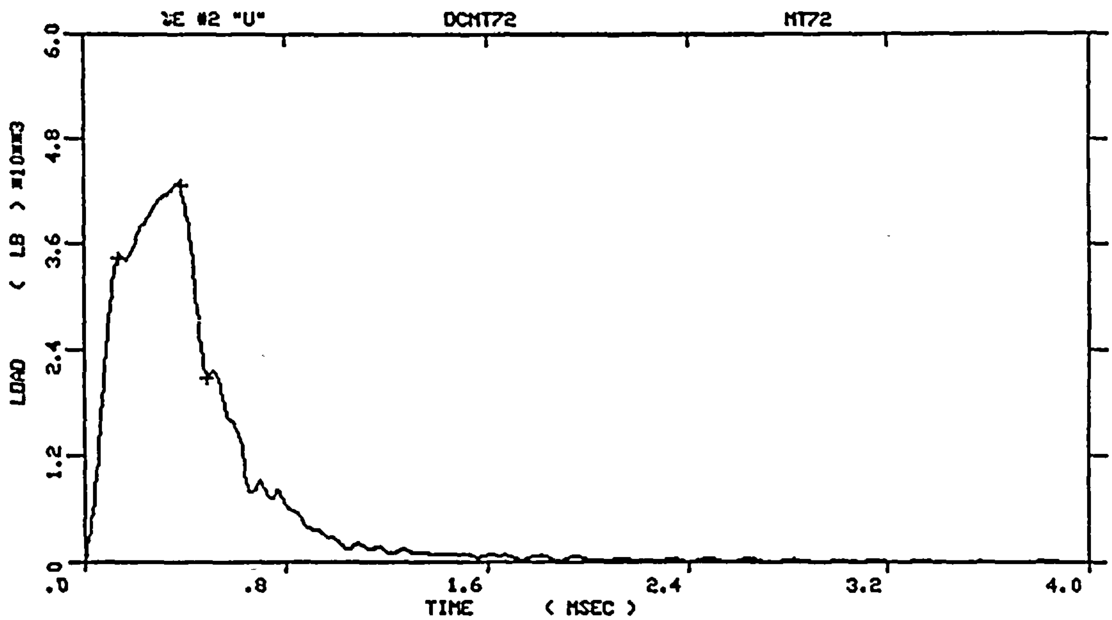
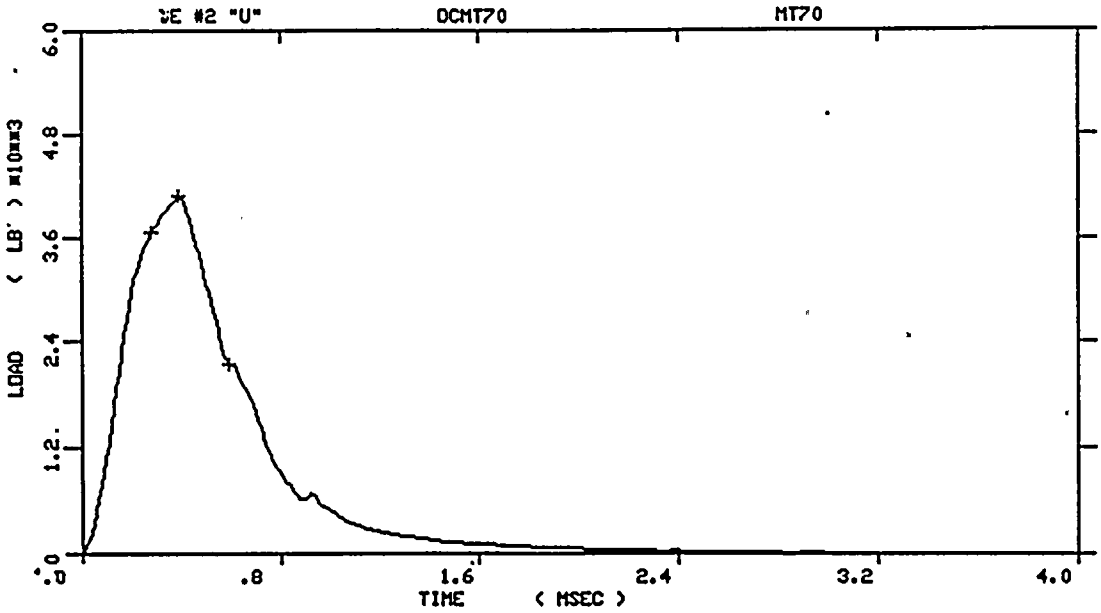


Figure A-8. Load-time records for Specimens MT64 and MT72



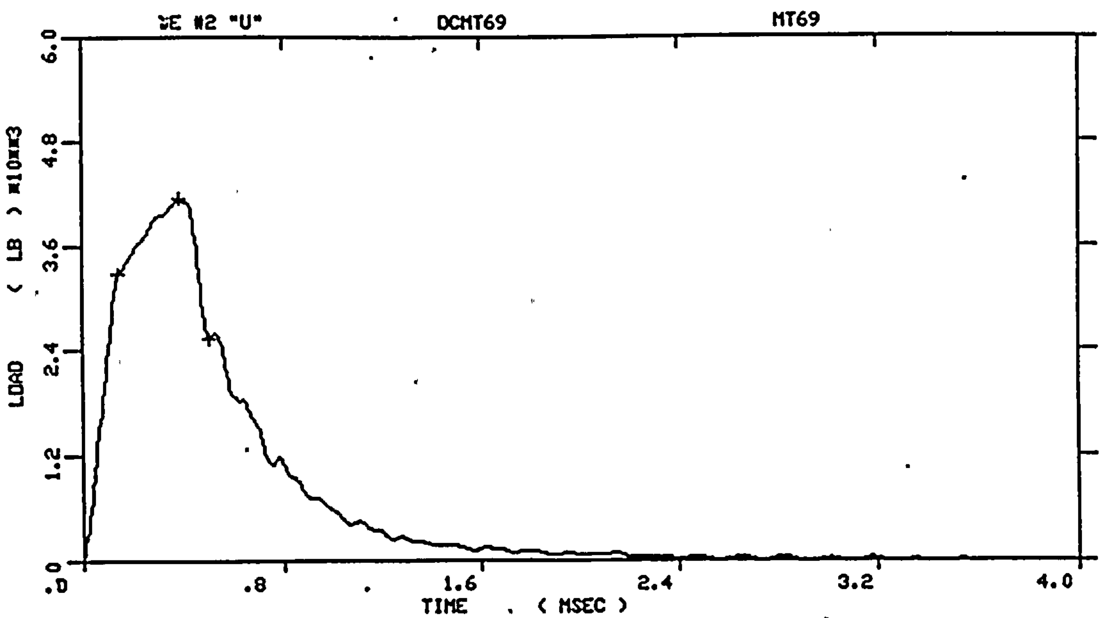
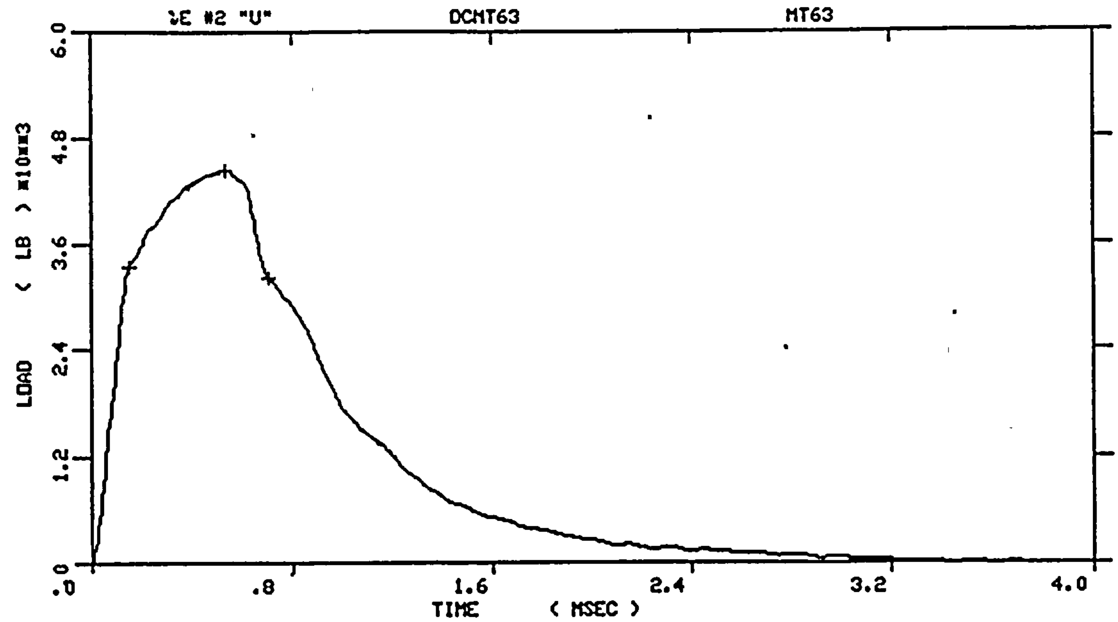


Figure A-9. Load-time records for Specimens MT70 and MT69



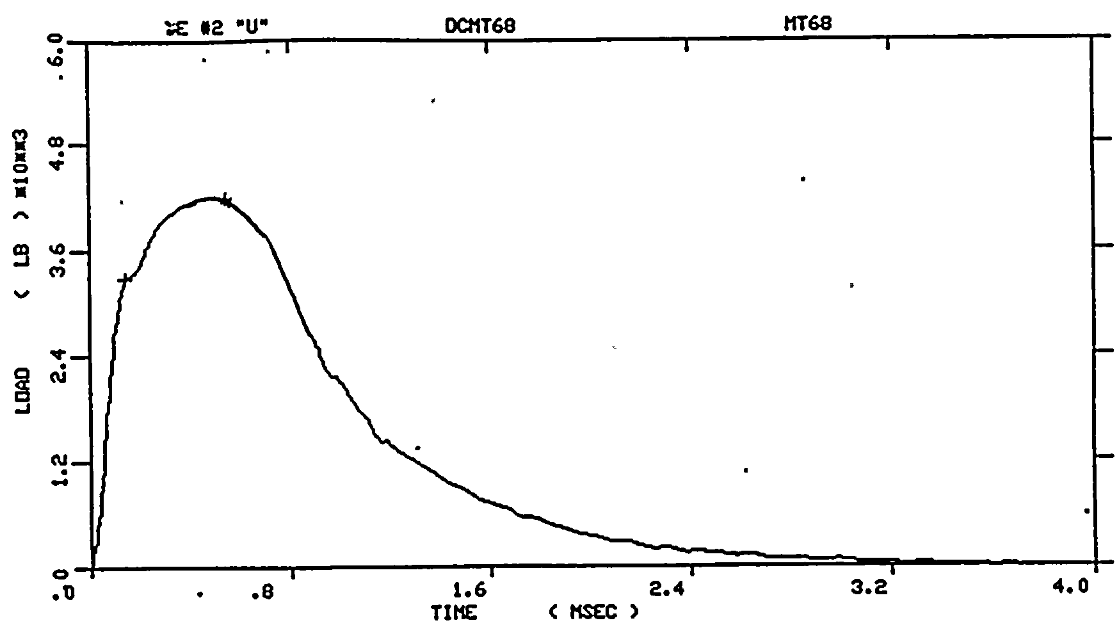
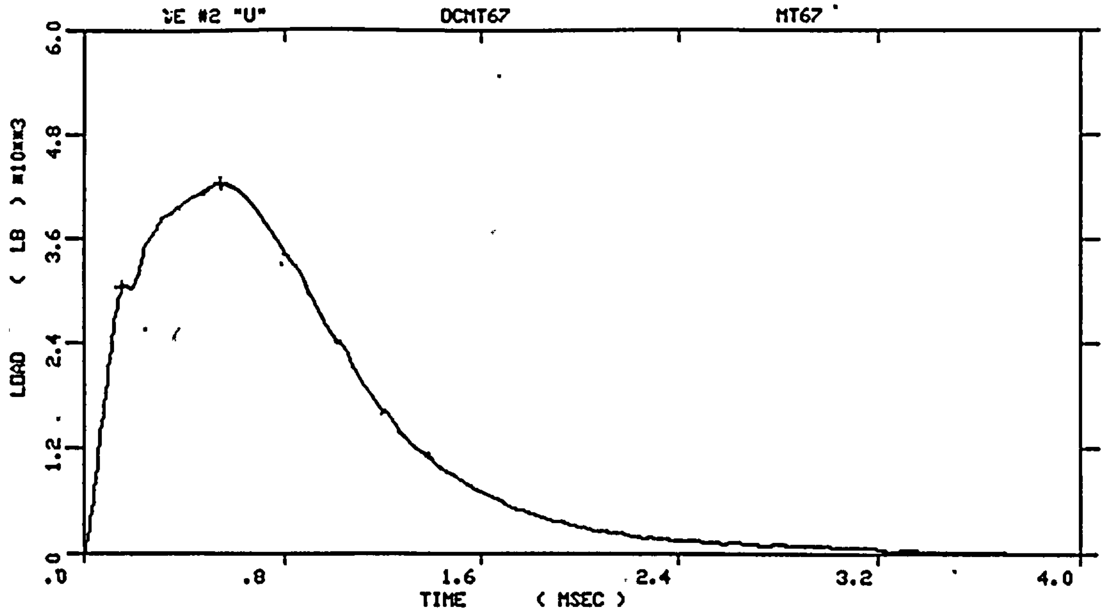


Figure A-10. Load-time records for Specimens MT63 and MT68



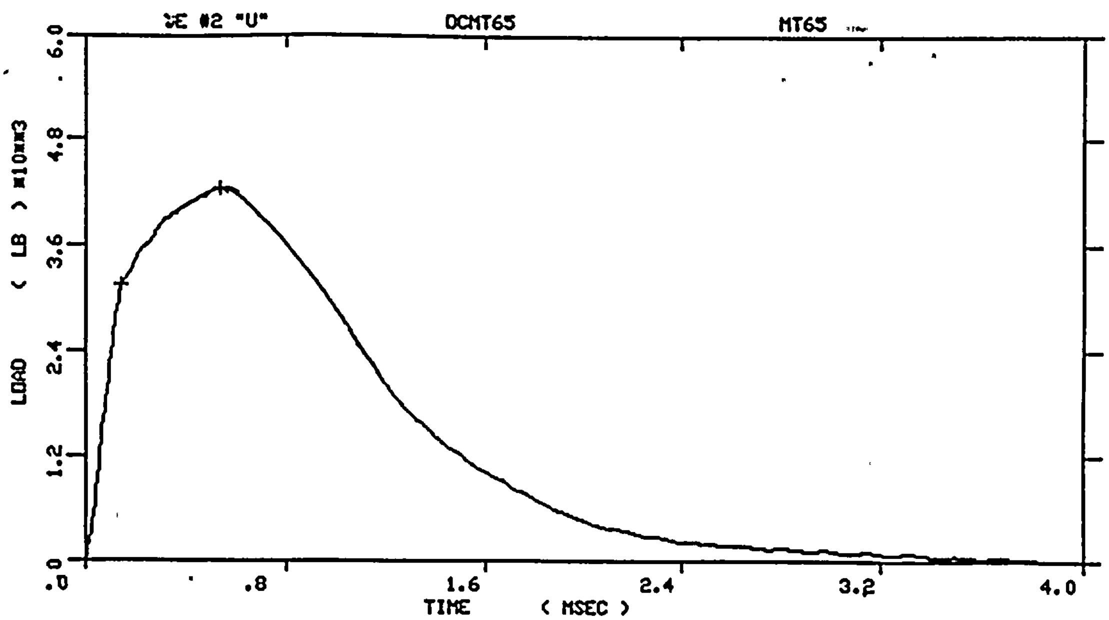
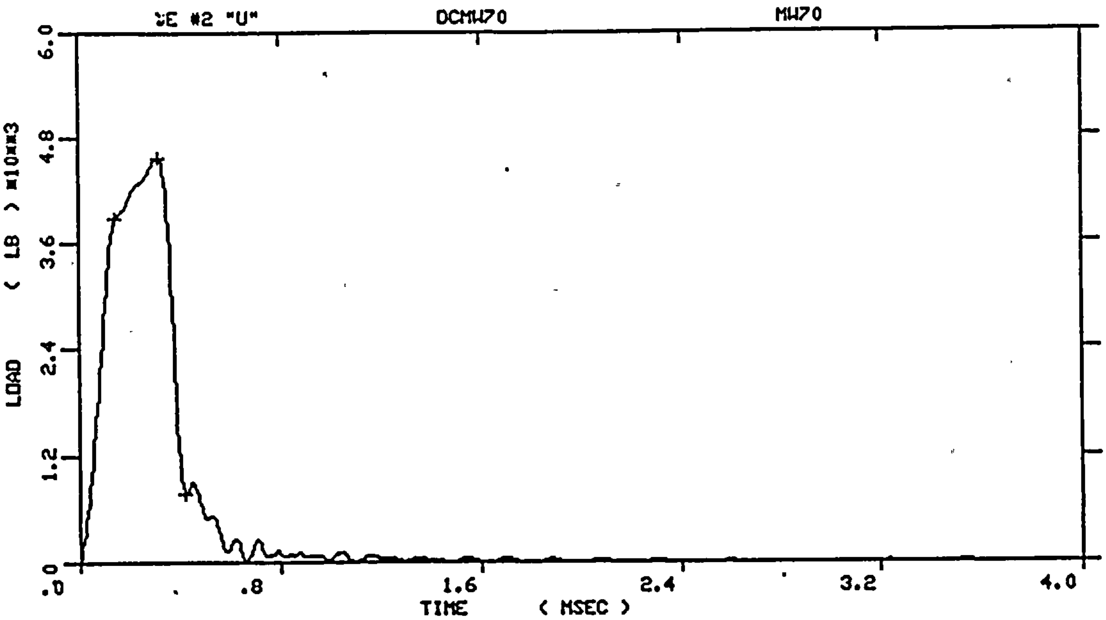


Figure A-11. Load-time records for Specimens MT67 and MT65



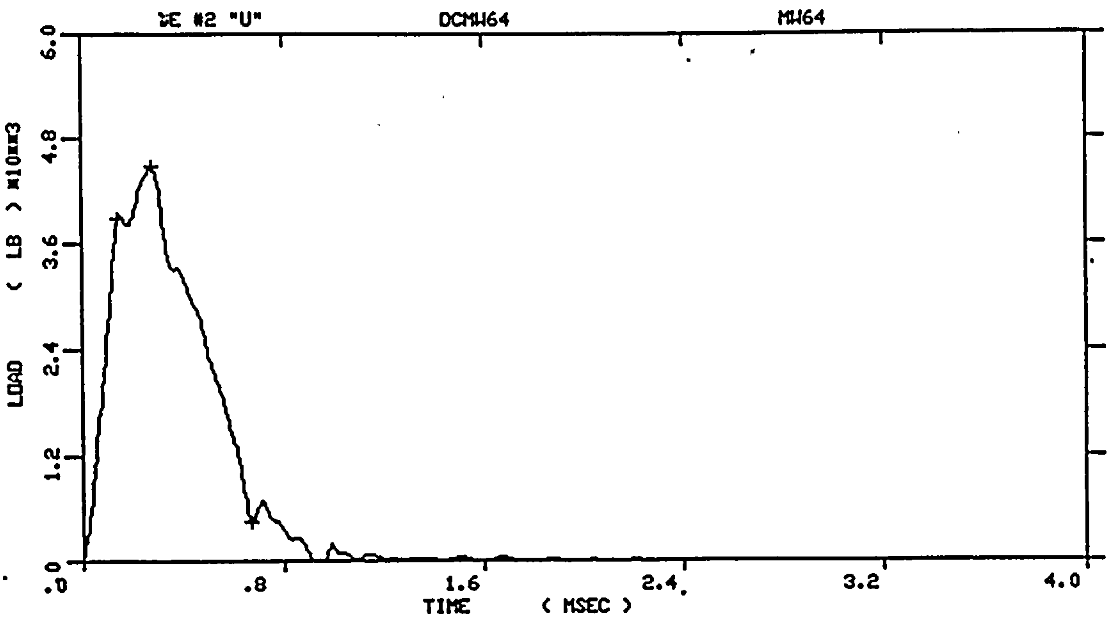
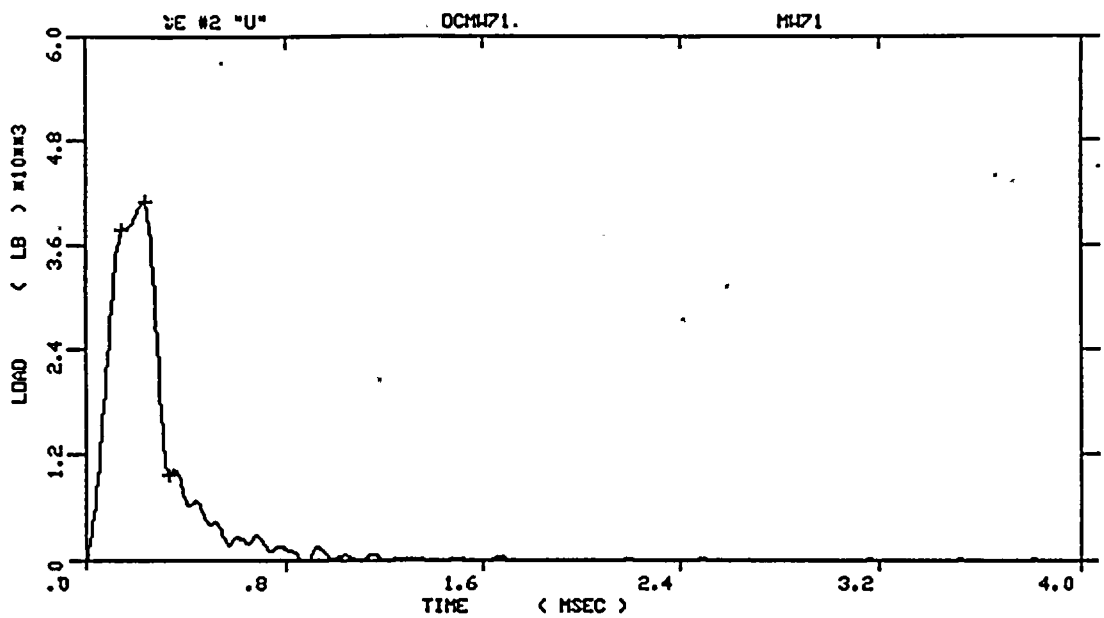


Figure A-12. Load-time records for Specimens MW70 and MW64



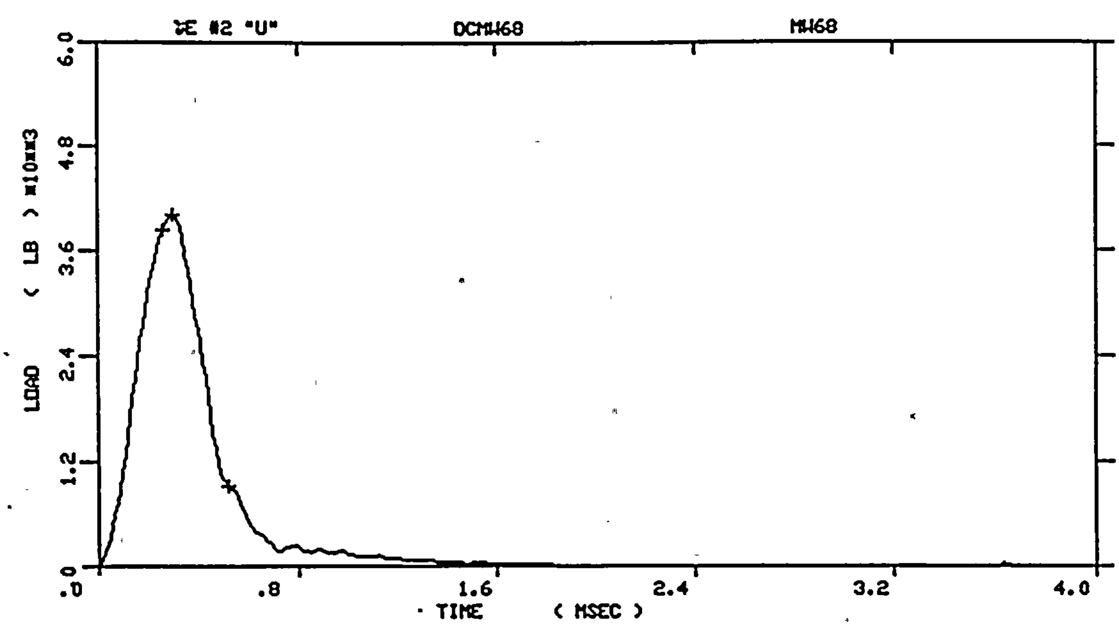
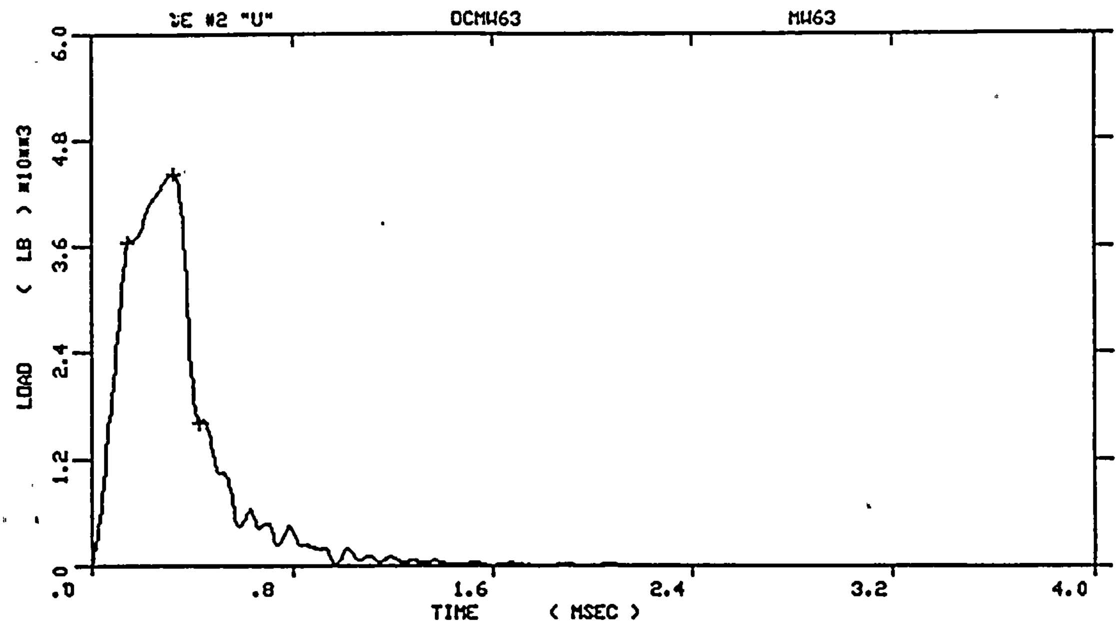


Figure A-13. Load-time records for Specimens MW71 and MW68



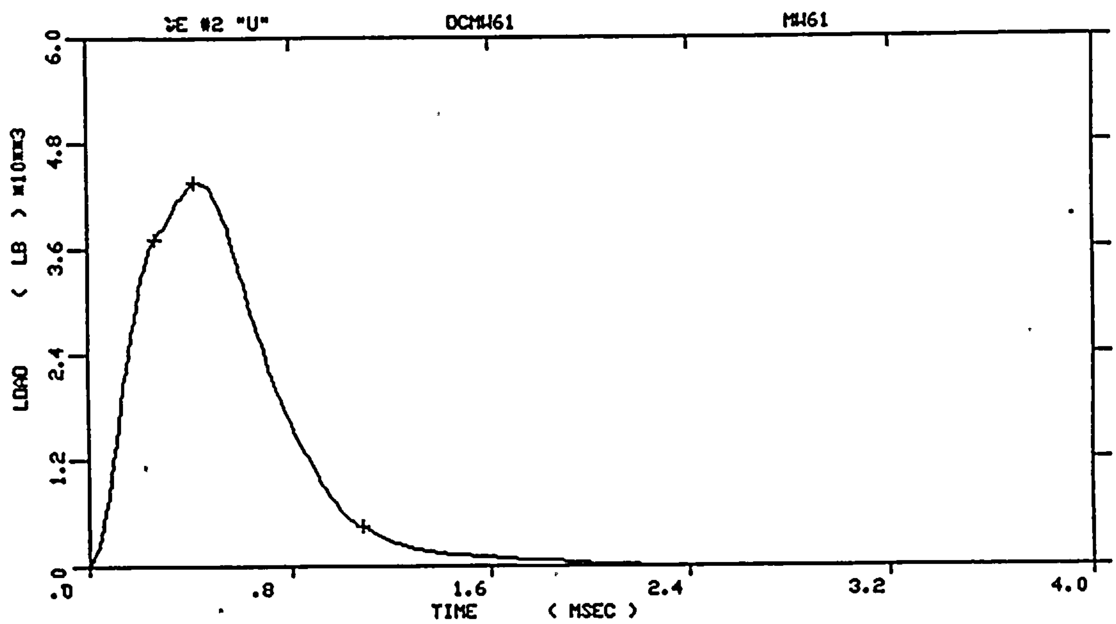
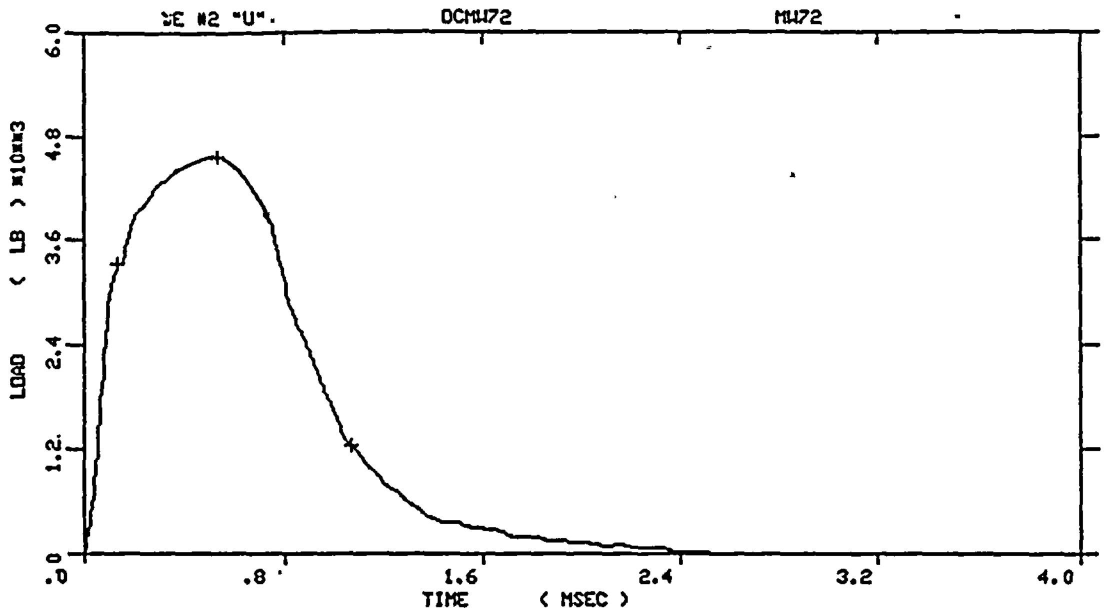


Figure A-14. Load-time records for Specimens MW63 and MW61



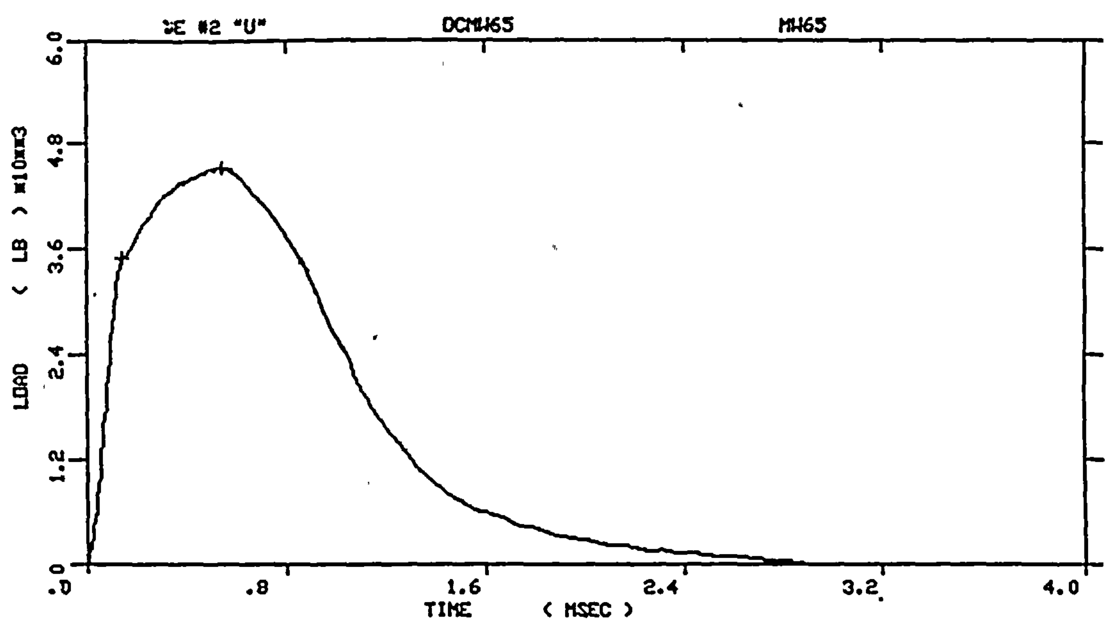
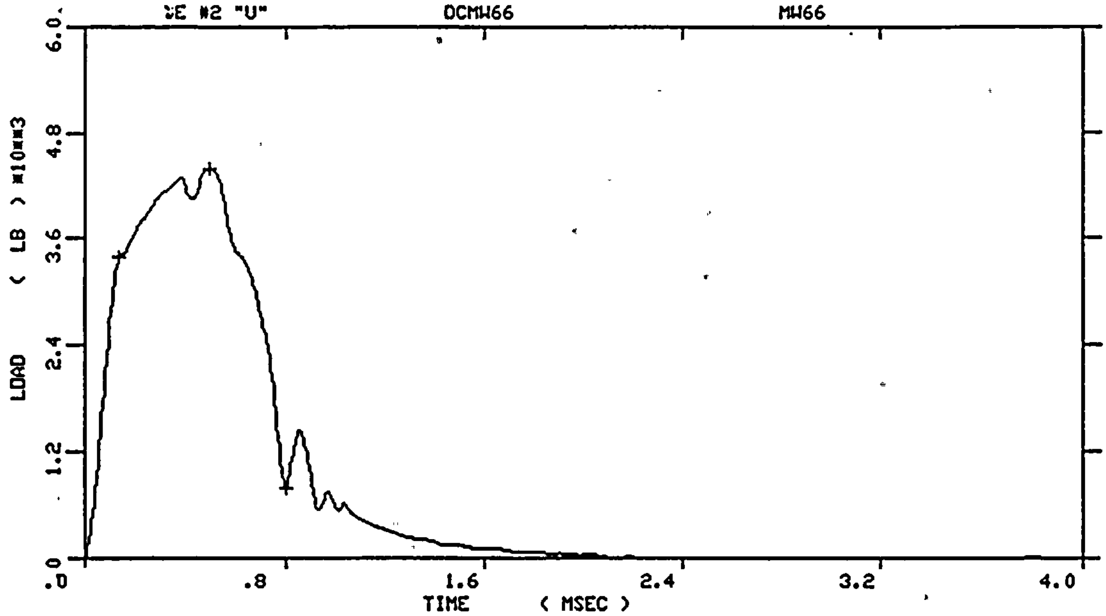


Figure A-15. Load-time records for Specimens MW72 and MW65



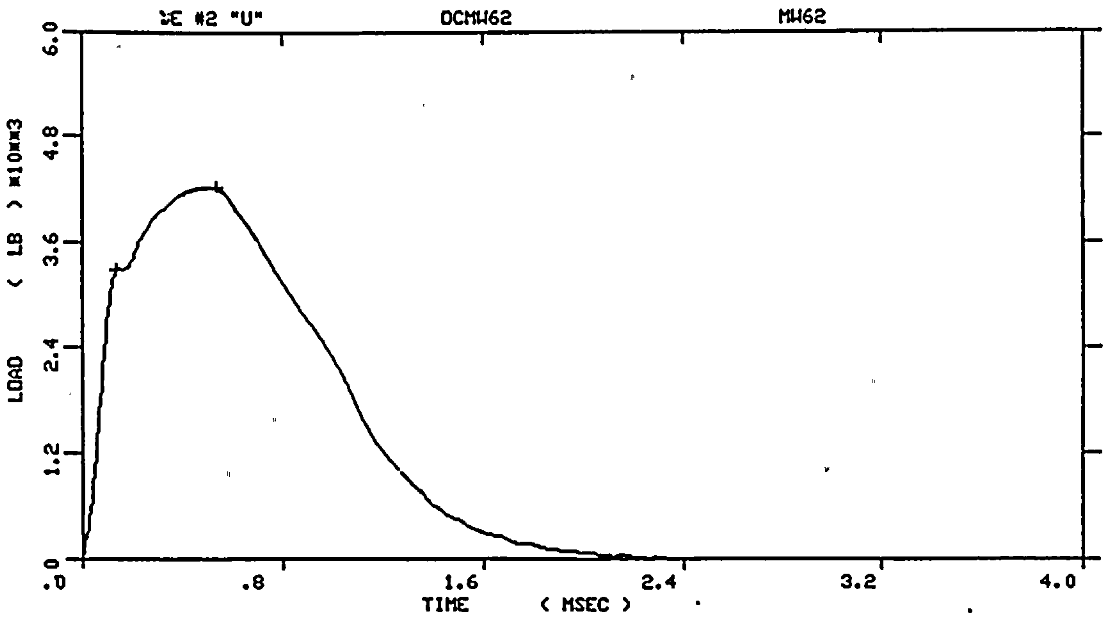
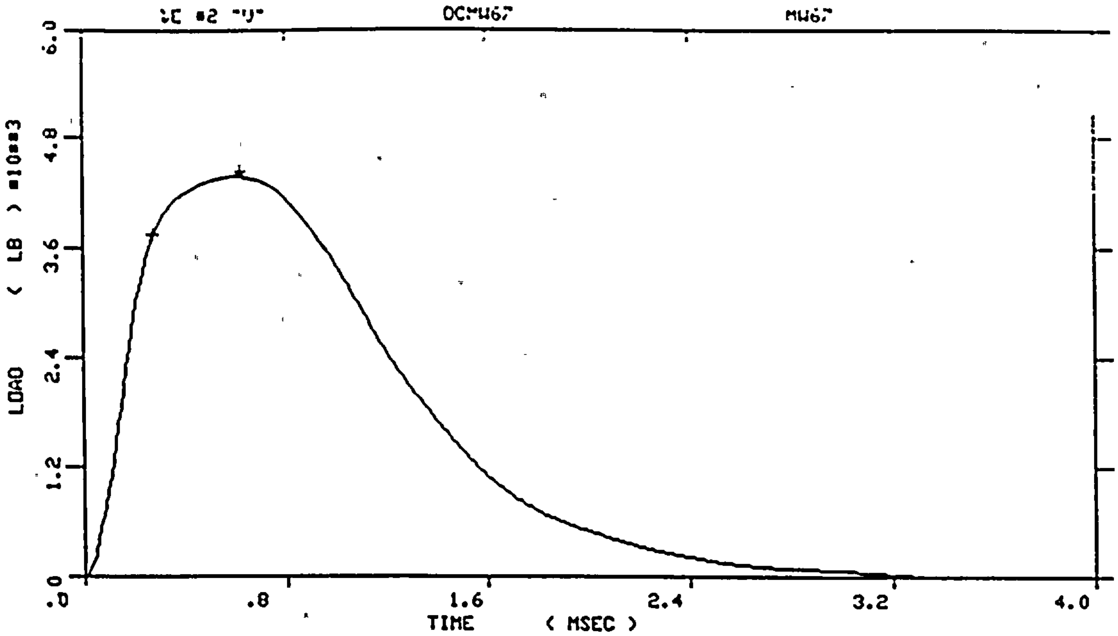


Figure A-16. Load-time records for Specimens MW66 and MW62



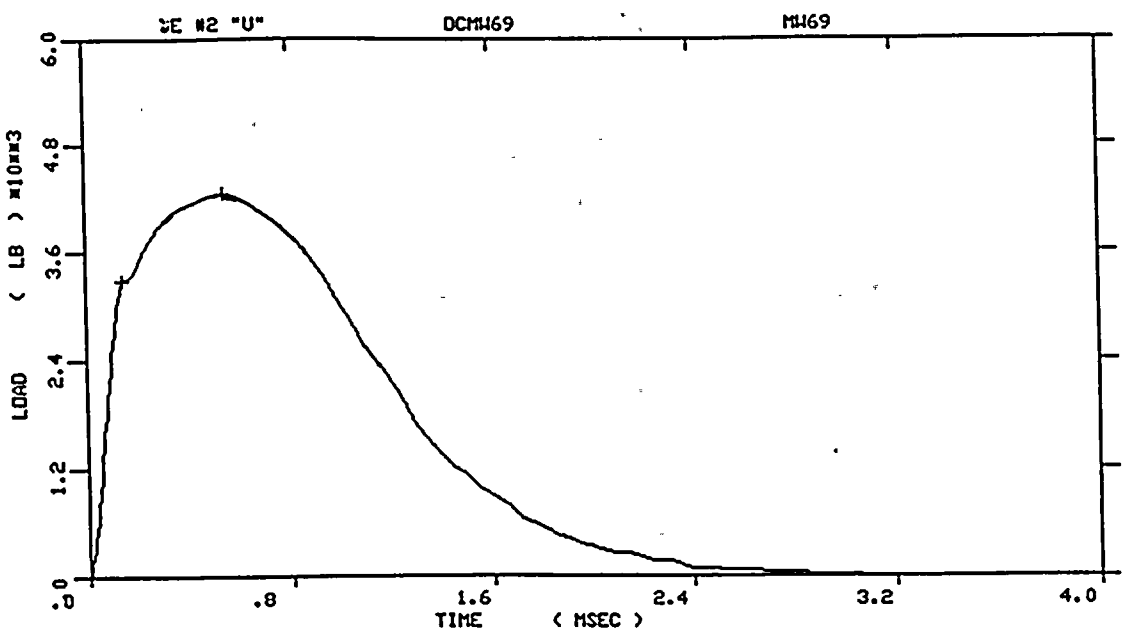
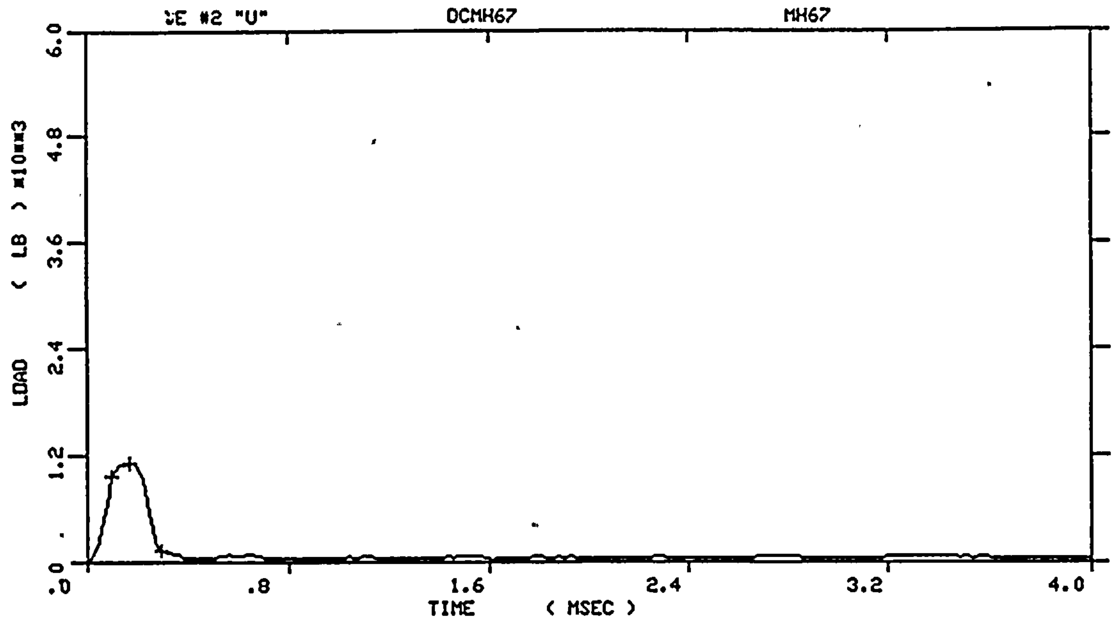


Figure A-17. Load-time records for Specimens MW67 and MW69



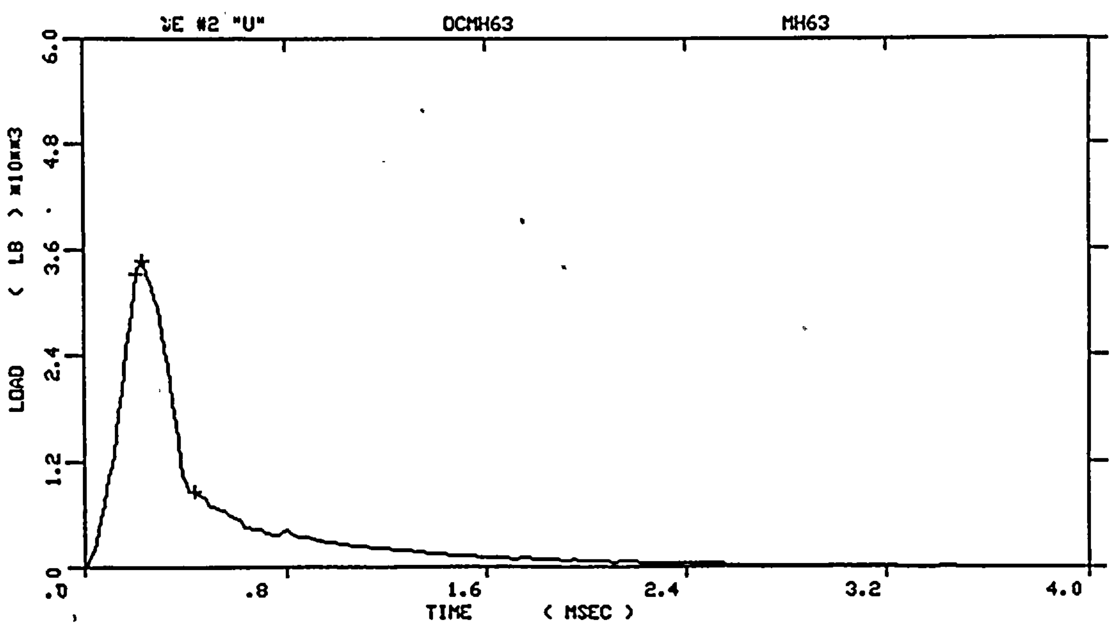
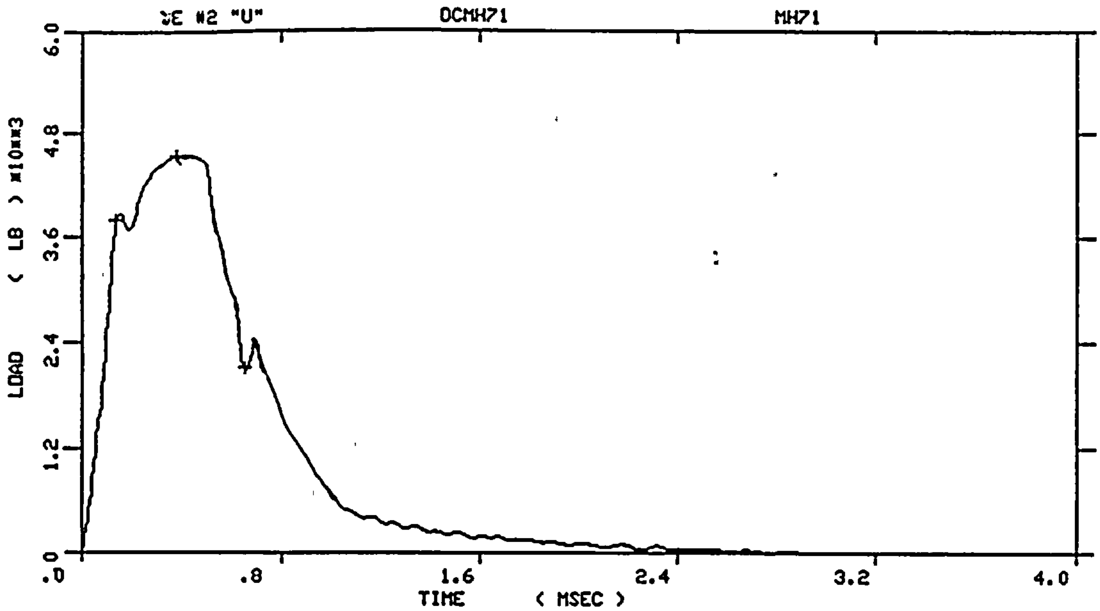


Figure A-18. Load-time records for Specimens MH67 and MH63



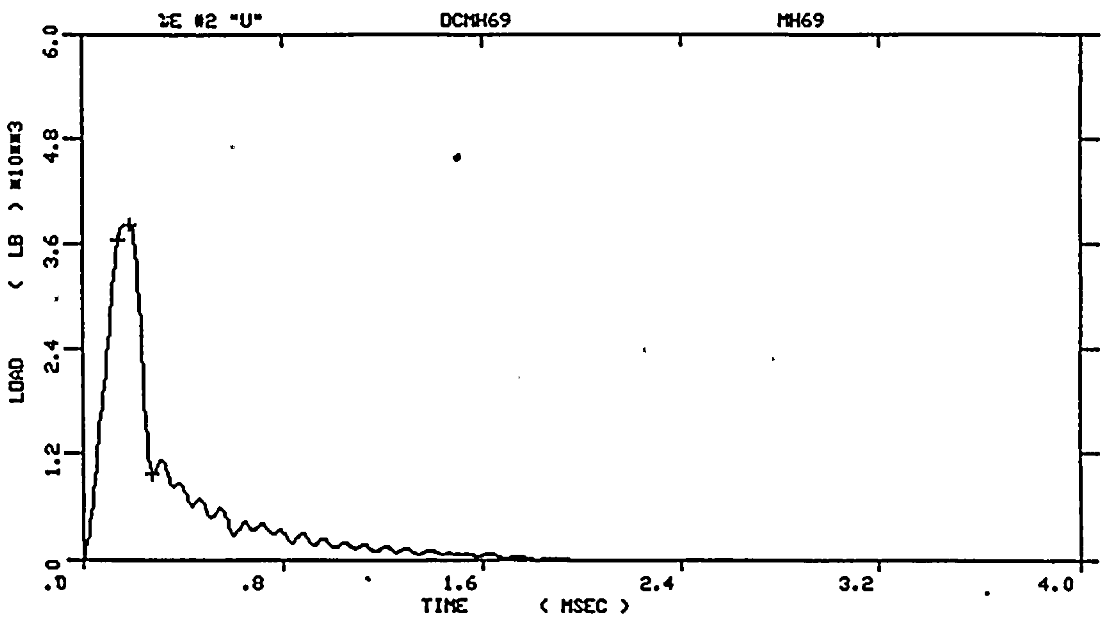
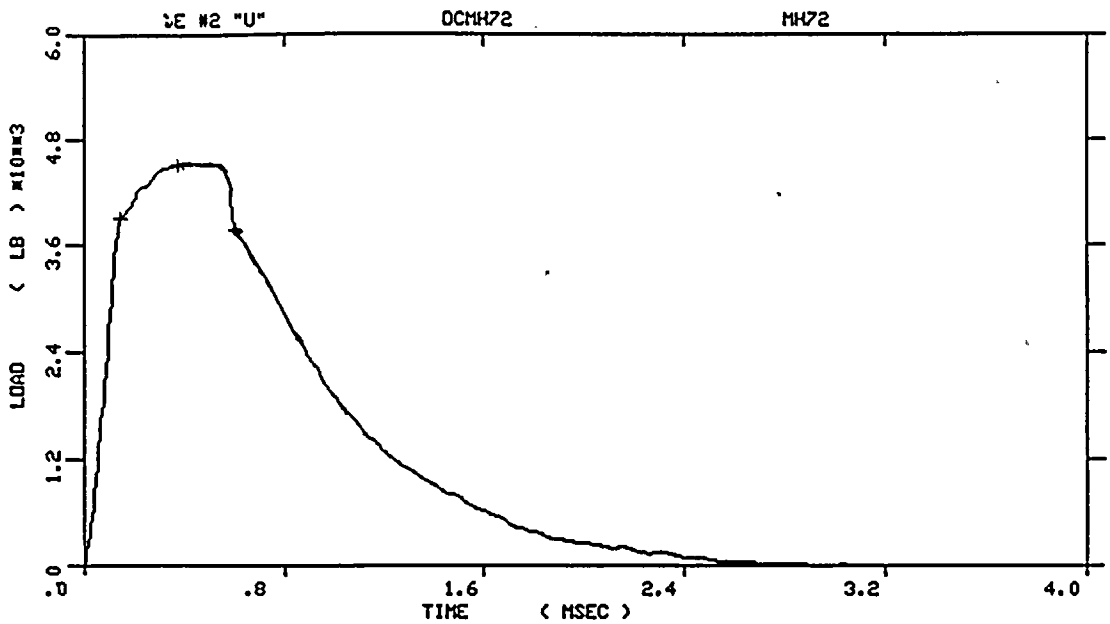


Figure A-19. Load-time records for Specimens MH71 and MH69



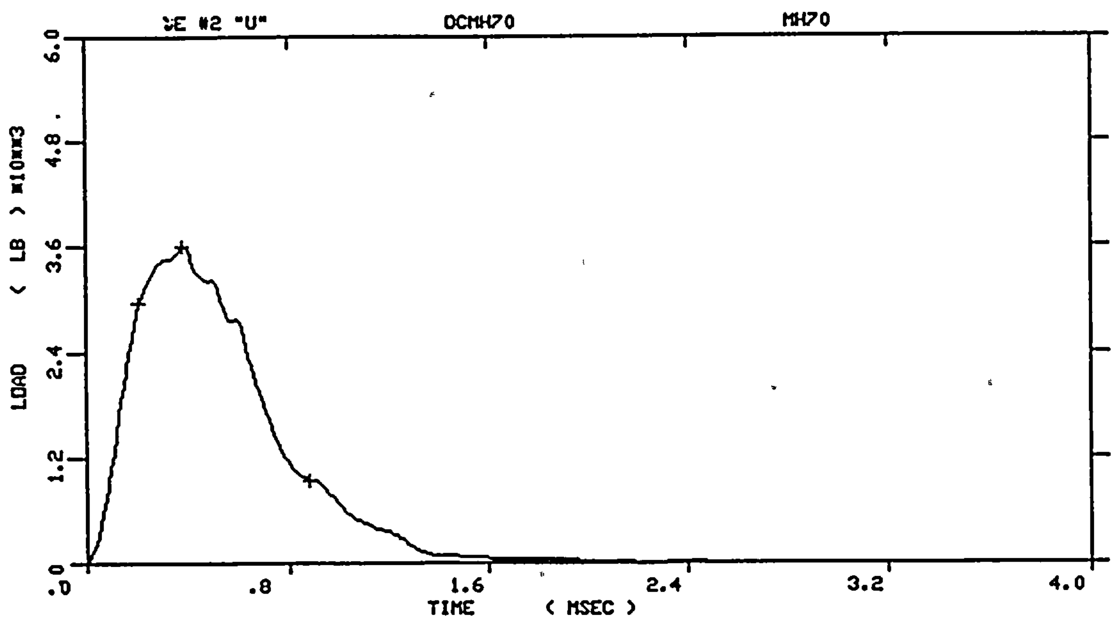
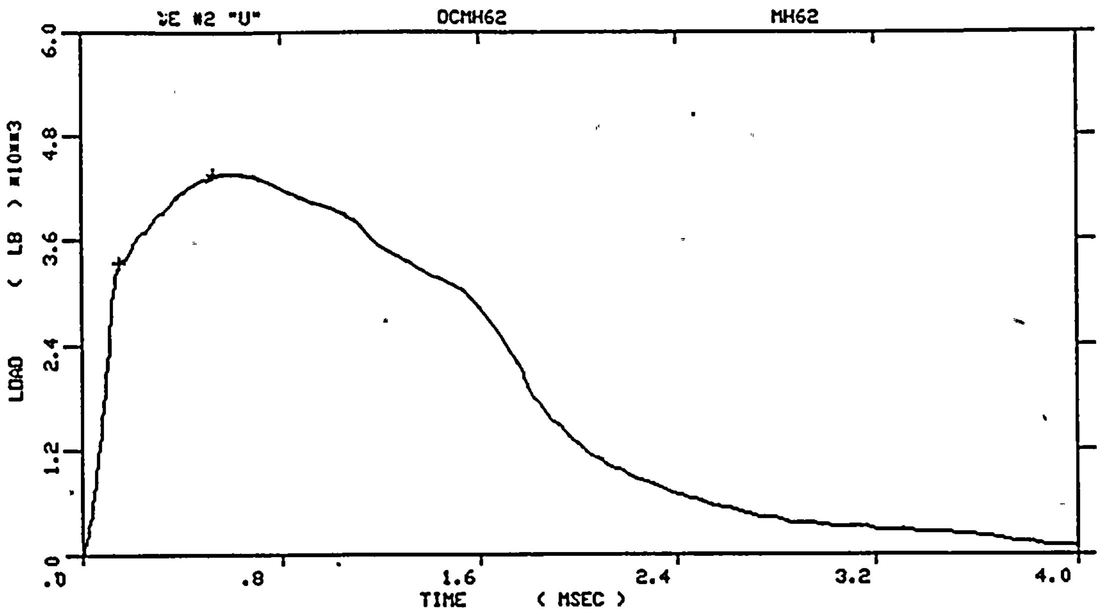


Figure A-20. Load-time records for Specimens MH72 and MH70



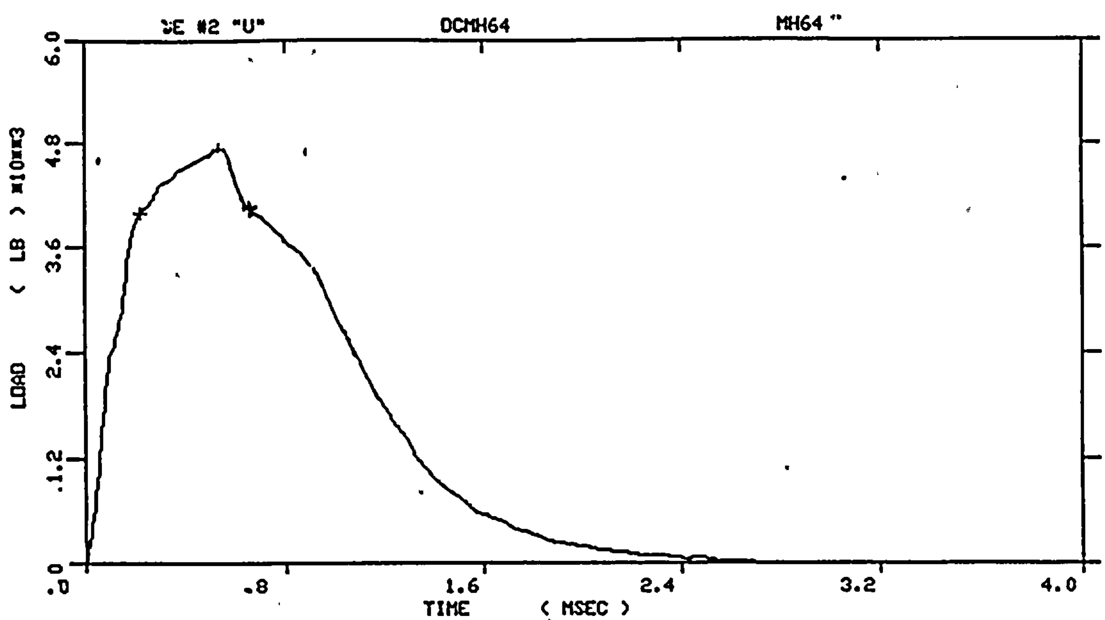
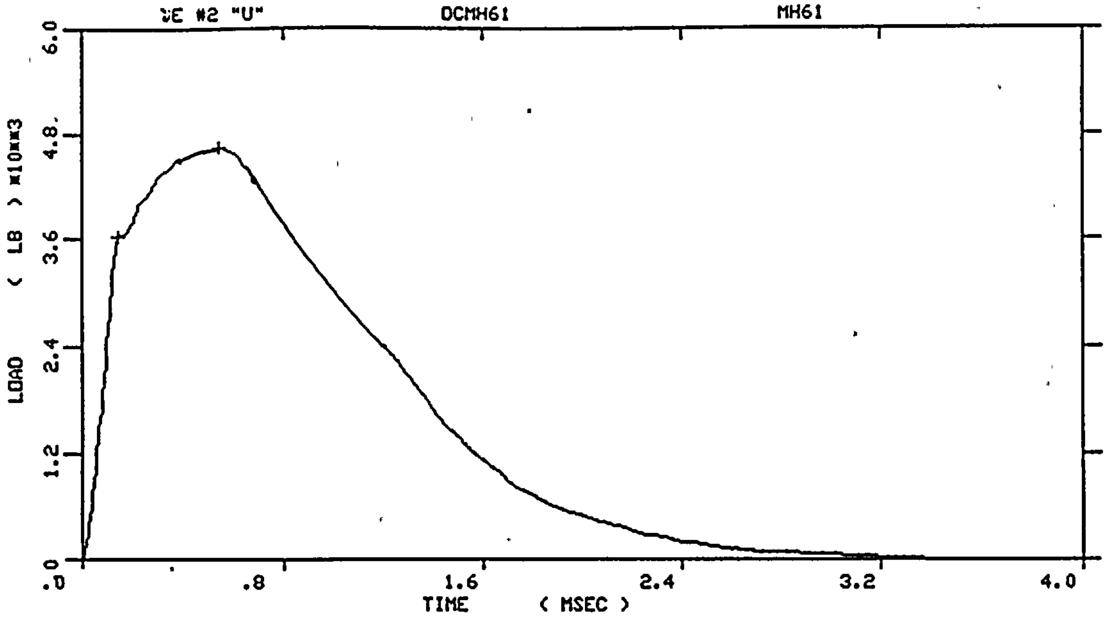


Figure A-21. Load-time records for Specimens MH62 and MH64



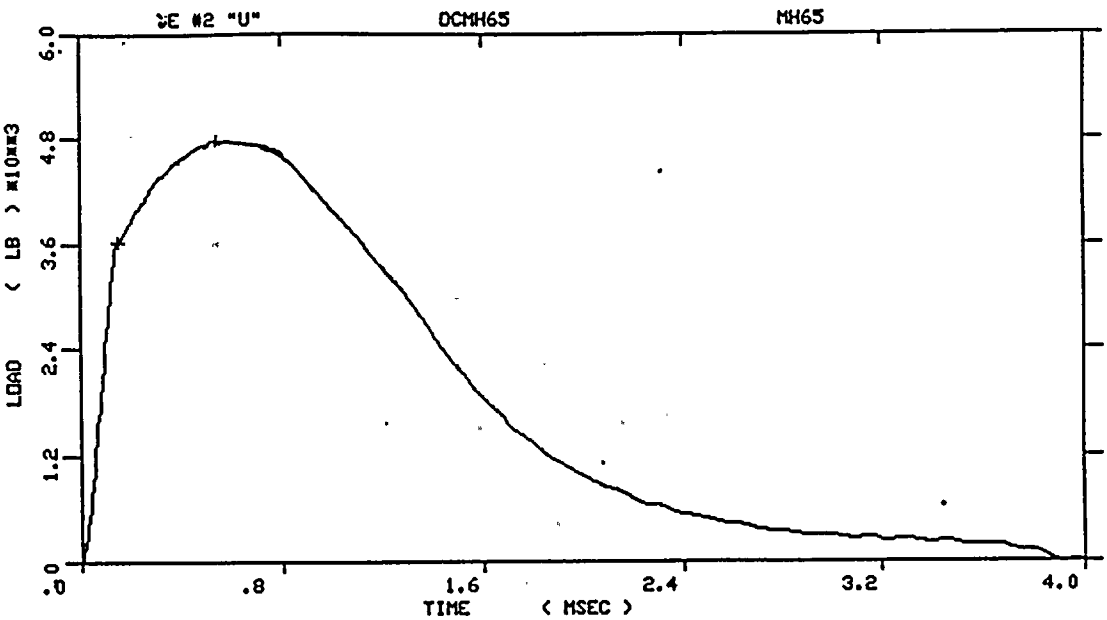
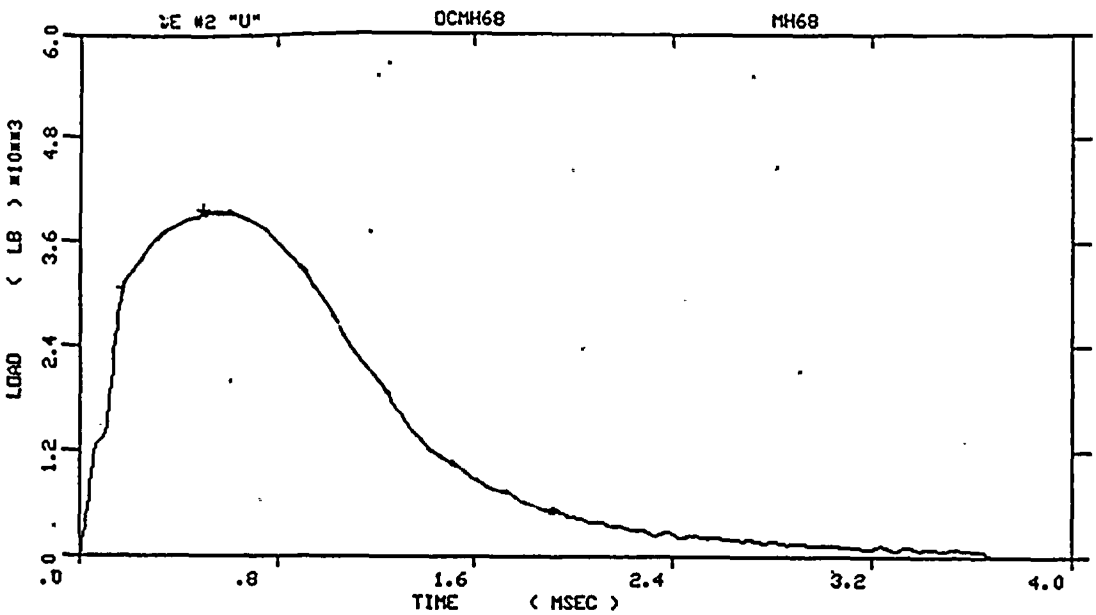


Figure A-22. Load-time records for Specimens MH61 and MH65



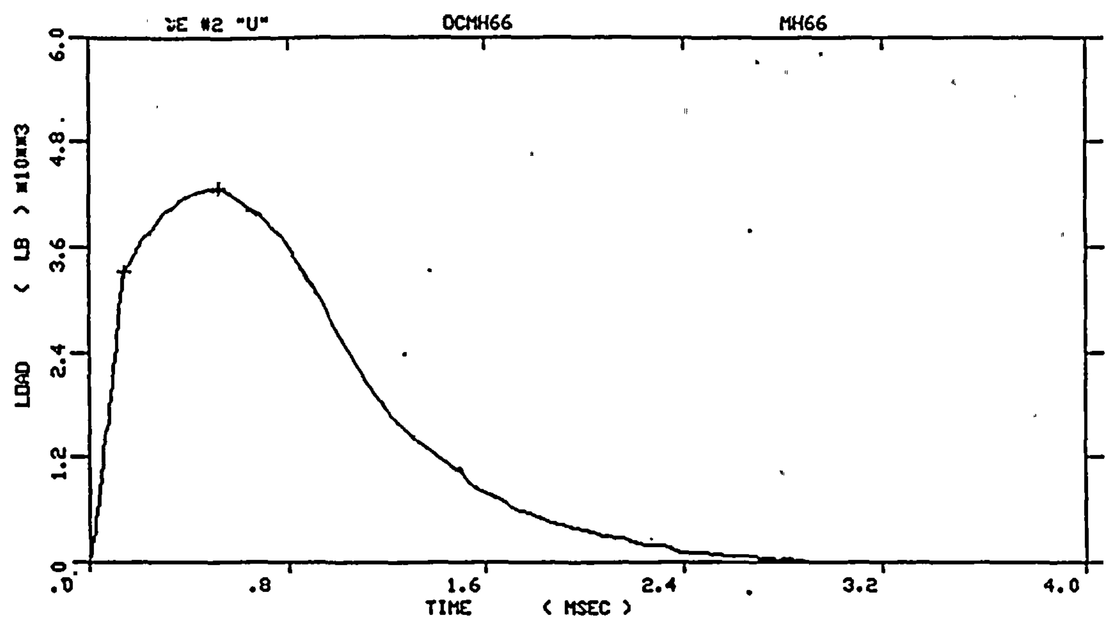


Figure A-23. Load-time records for Specimens MH68 and MH66

APPENDIX B

Photographs of Charpy, Tensile and WOL Specimens Prior to Testing

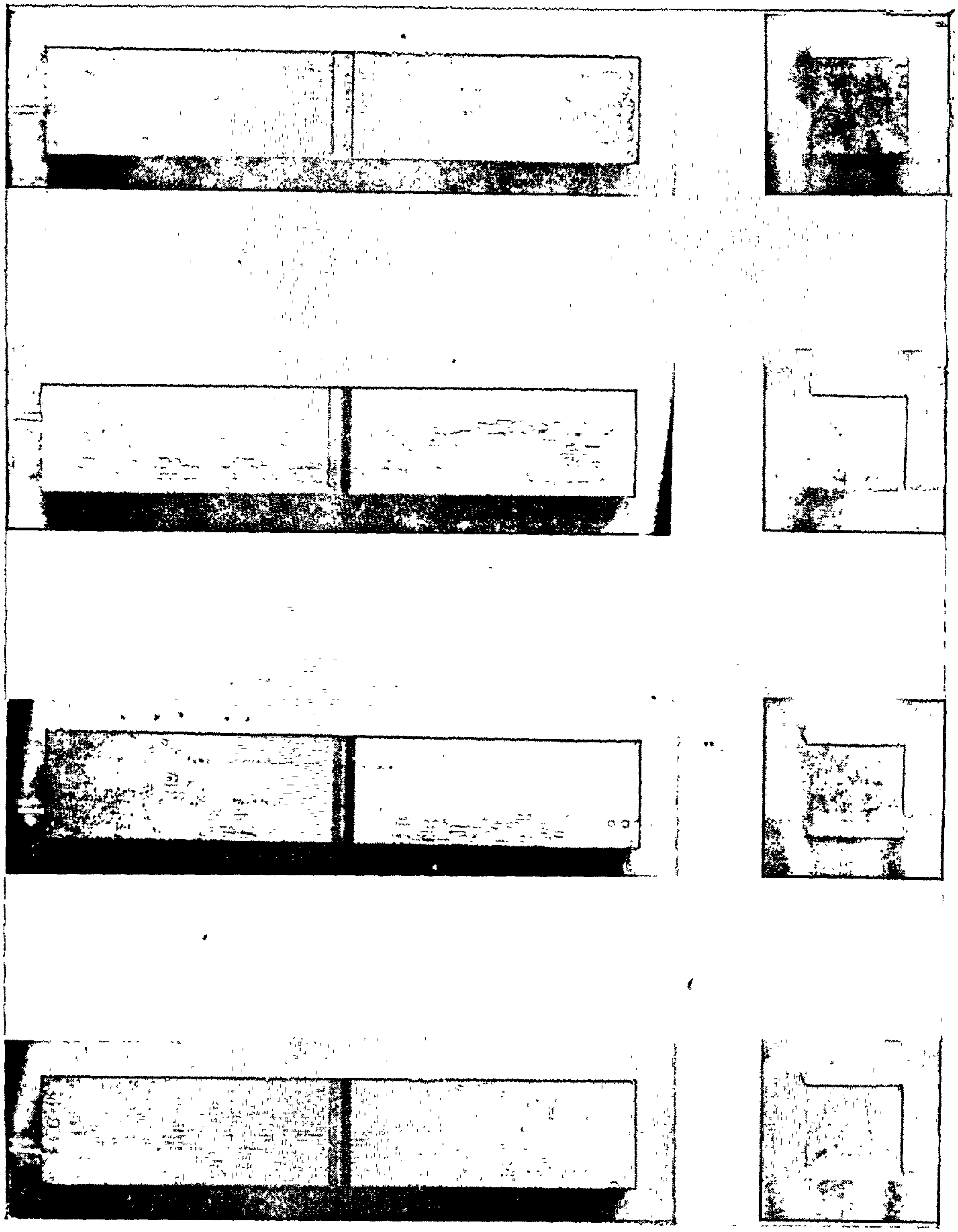


Figure B-1. Charpy impact specimens ML45, ML48, ML42, and ML44 from Intermediate Shell Plate C5521-2 (longitudinal orientation) before testing.

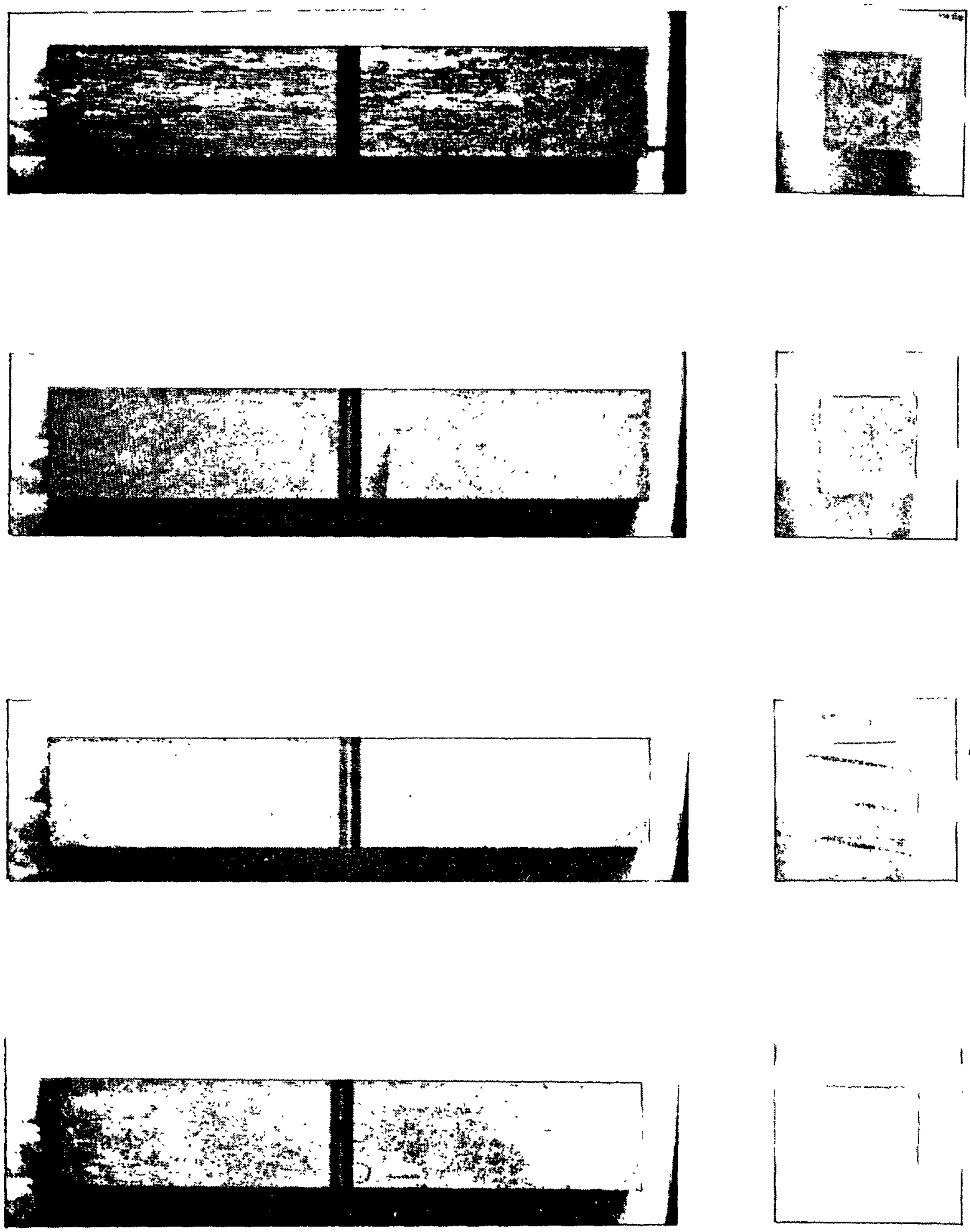


Figure B-2. Charpy impact specimens ML41, ML43, ML46, and ML47 from Intermediate Shell Plate C5521-2 (longitudinal orientation) before testing.

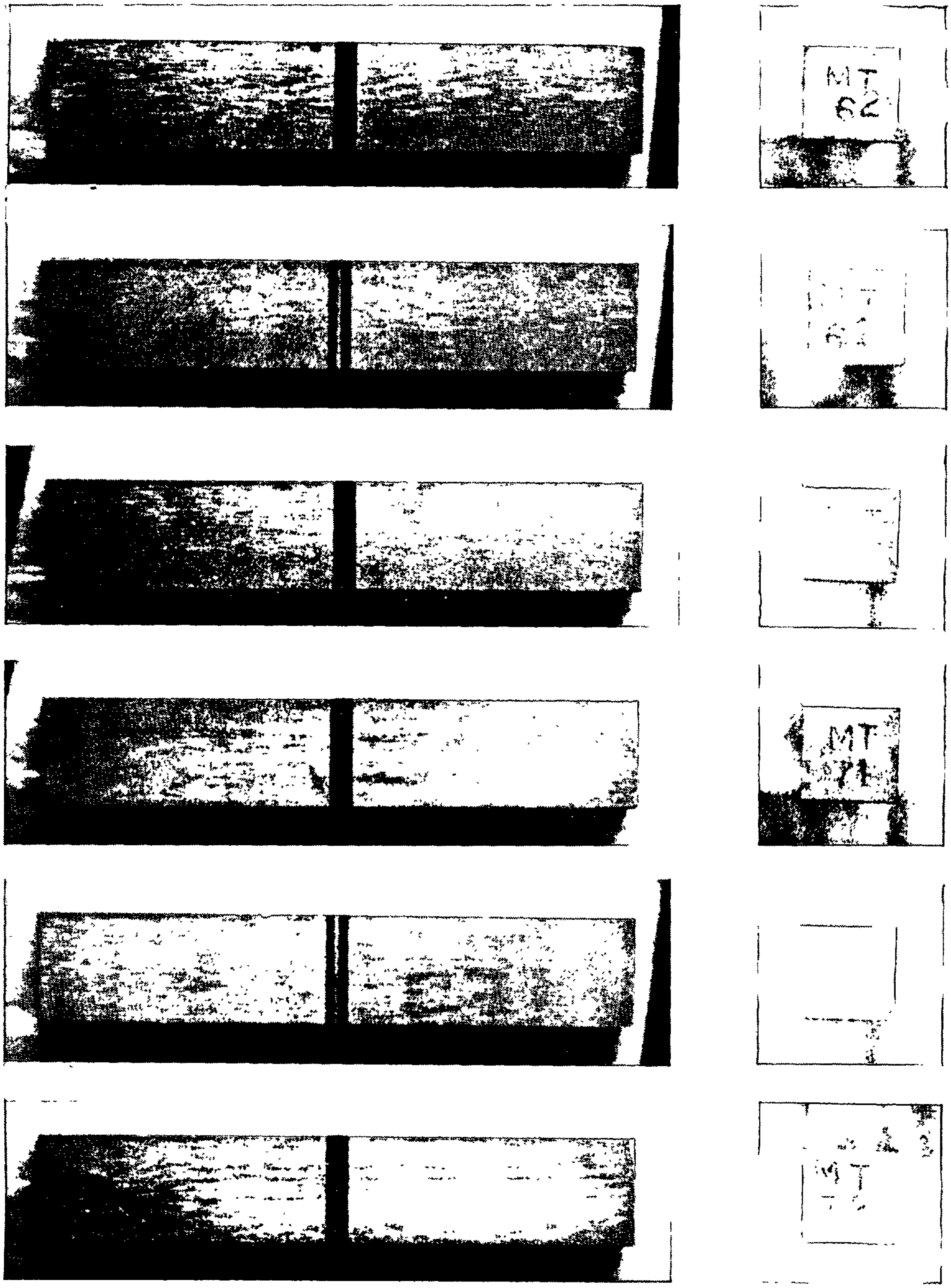


Figure B-3. Charpy impact specimens MT62, MT61, MT66, MT71, MT64, and MT72 from Intermediate Shell Plate C5521-2 (transverse orientation) before testing.

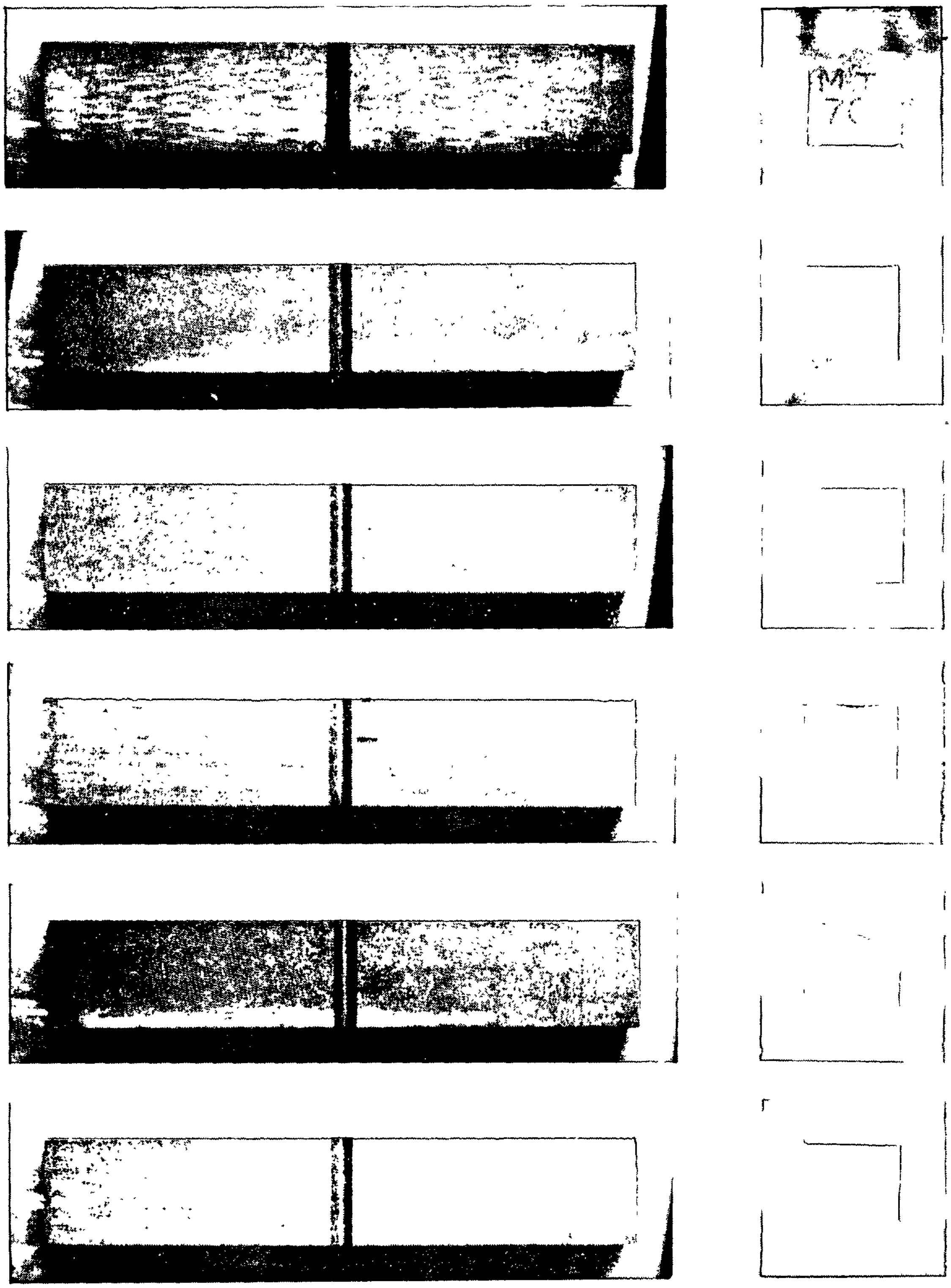


Figure B-4. Charpy impact specimens MT70, MT69, MT63, MT68, MT67, and MT65 from Intermediate Shell Plate C5521-2 (transverse orientation) before testing.

RM-28362

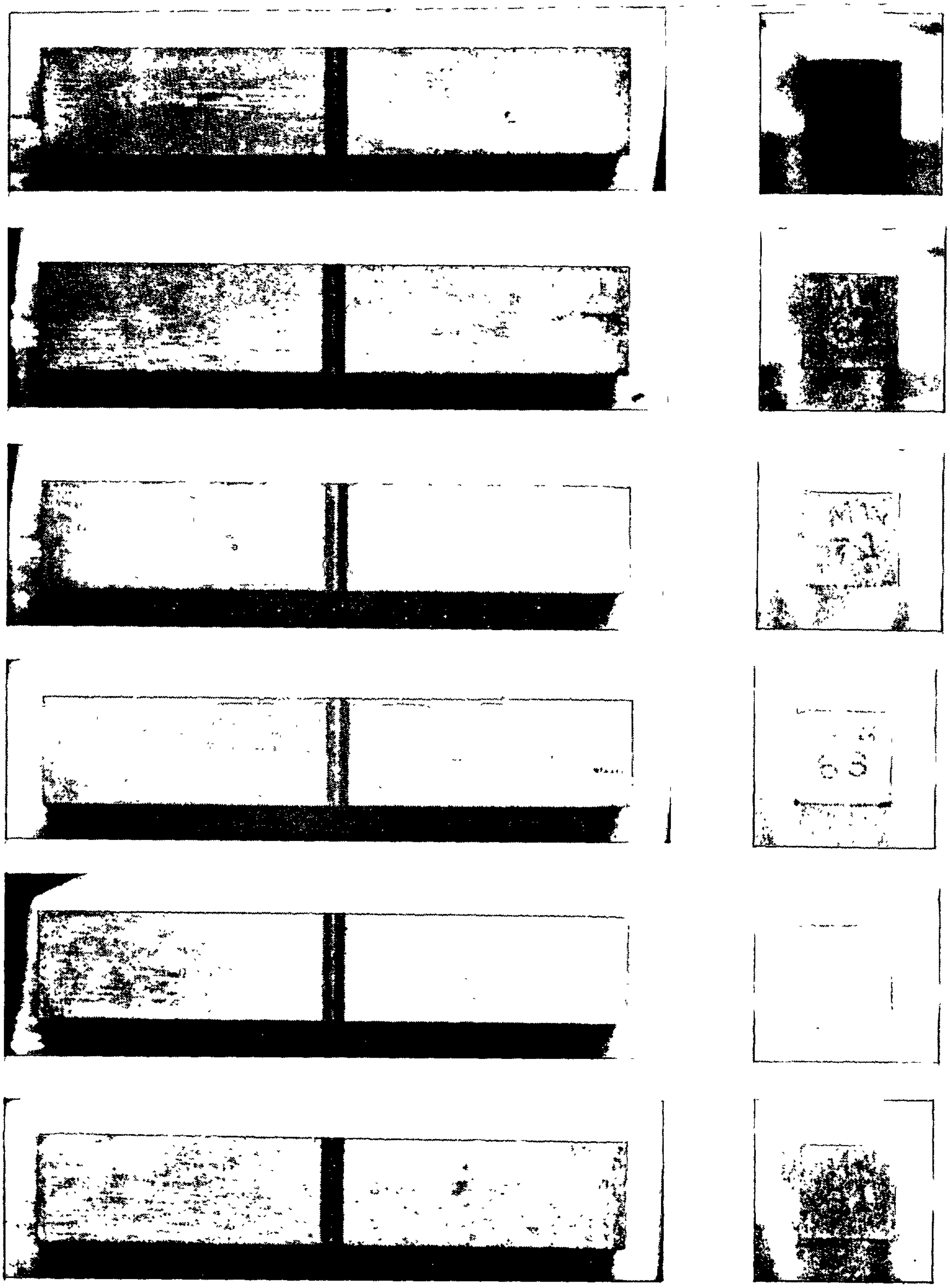


Figure B-5. Charpy impact specimens MW70, MW64, MW71, MW68, MW63, and MW61 from the weld metal, before testing.

RM-28363

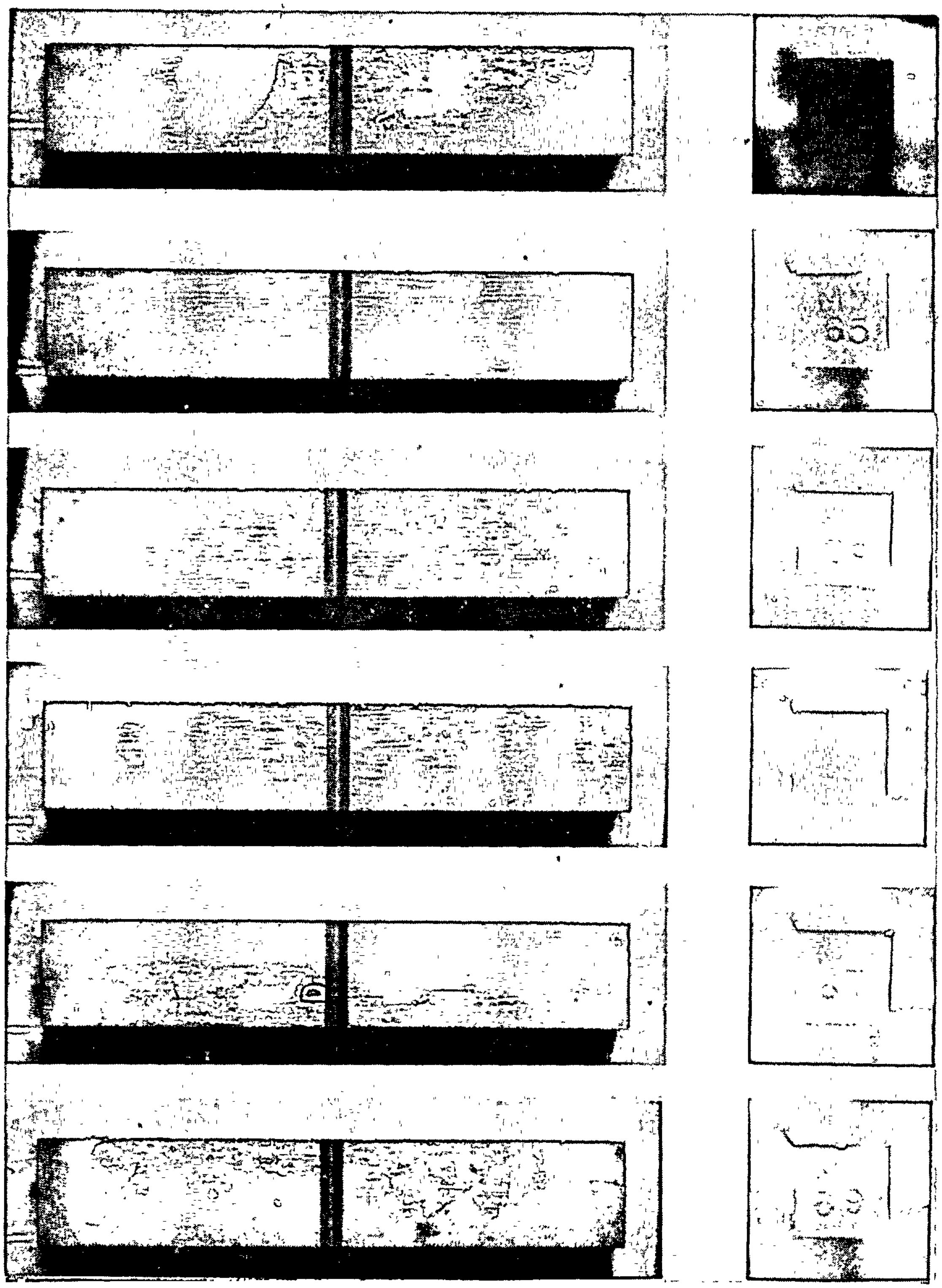


Figure B-6. Charpy impact specimens MW72, MW65, MW66, MW62, MW67, and MW69 from the weld metal, before testing.

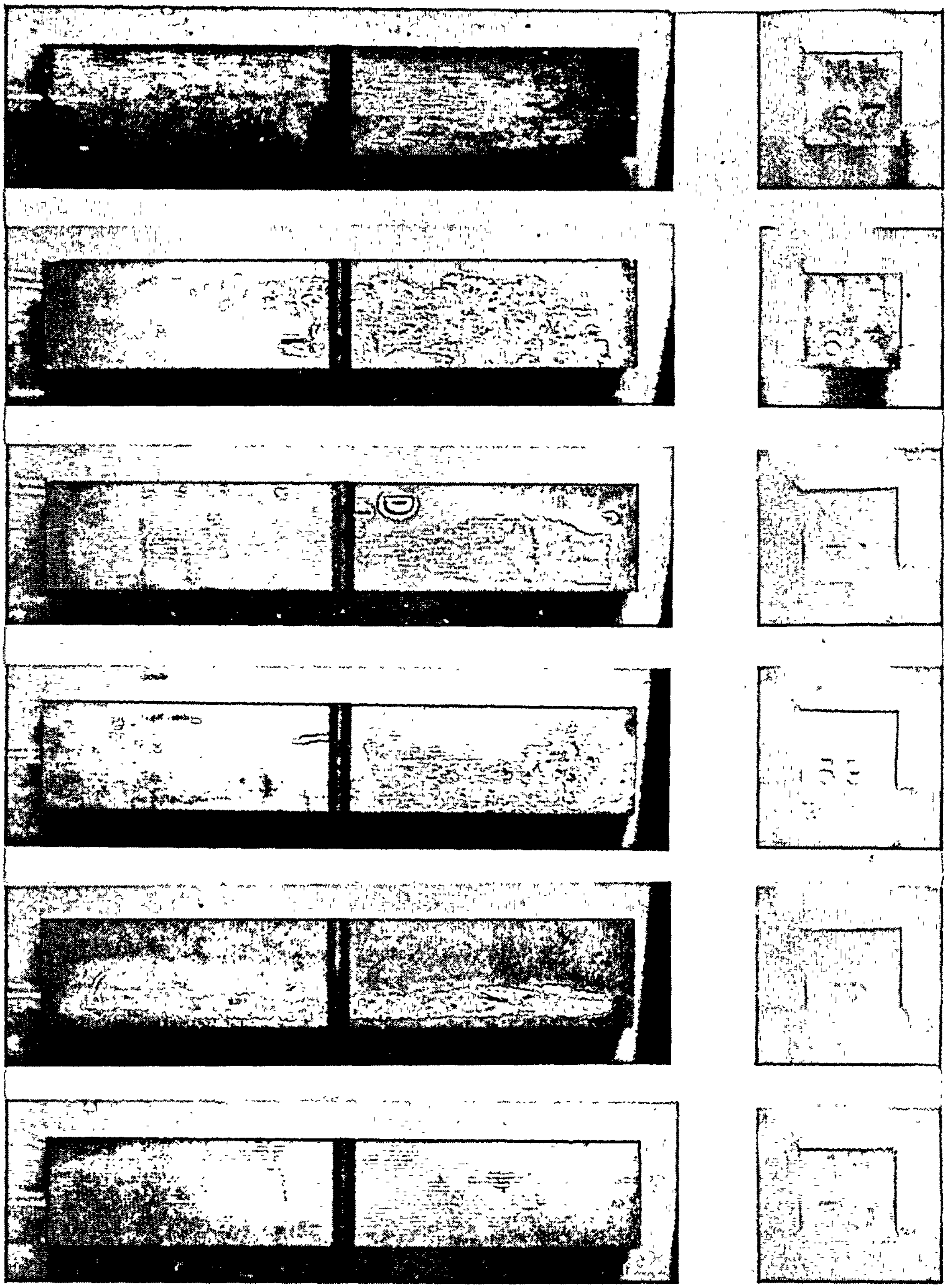


Figure B-7. Charpy impact specimens MH67, MH63, MH71, MH69, MH72, and MH70 from the heat-affected zone (HAZ), before testing.

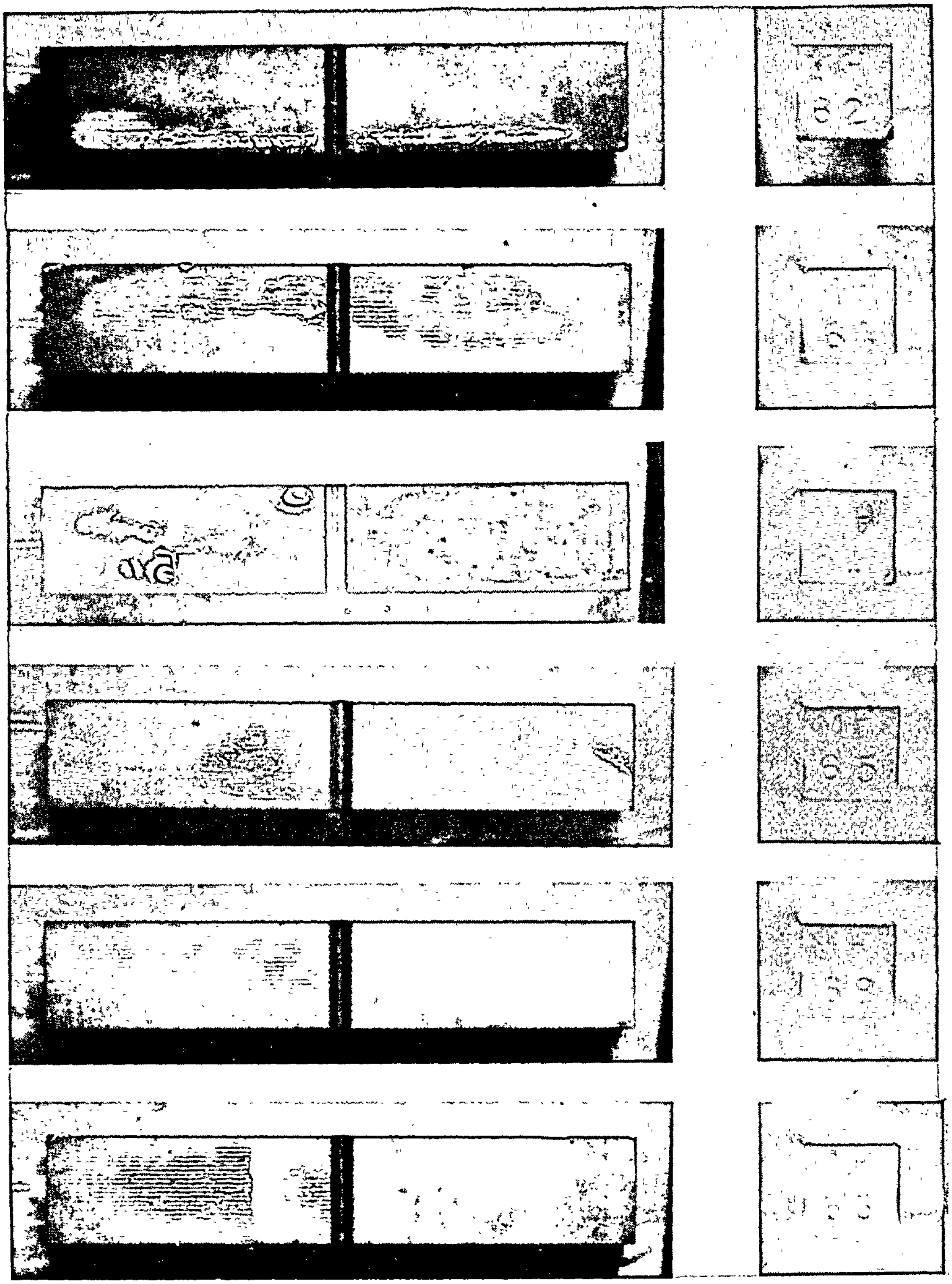


Figure B-8. Charpy impact specimens MH62, MH64, MH61, MH65, MH68, and MH66 from the heat-affected zone (HAZ), before testing.

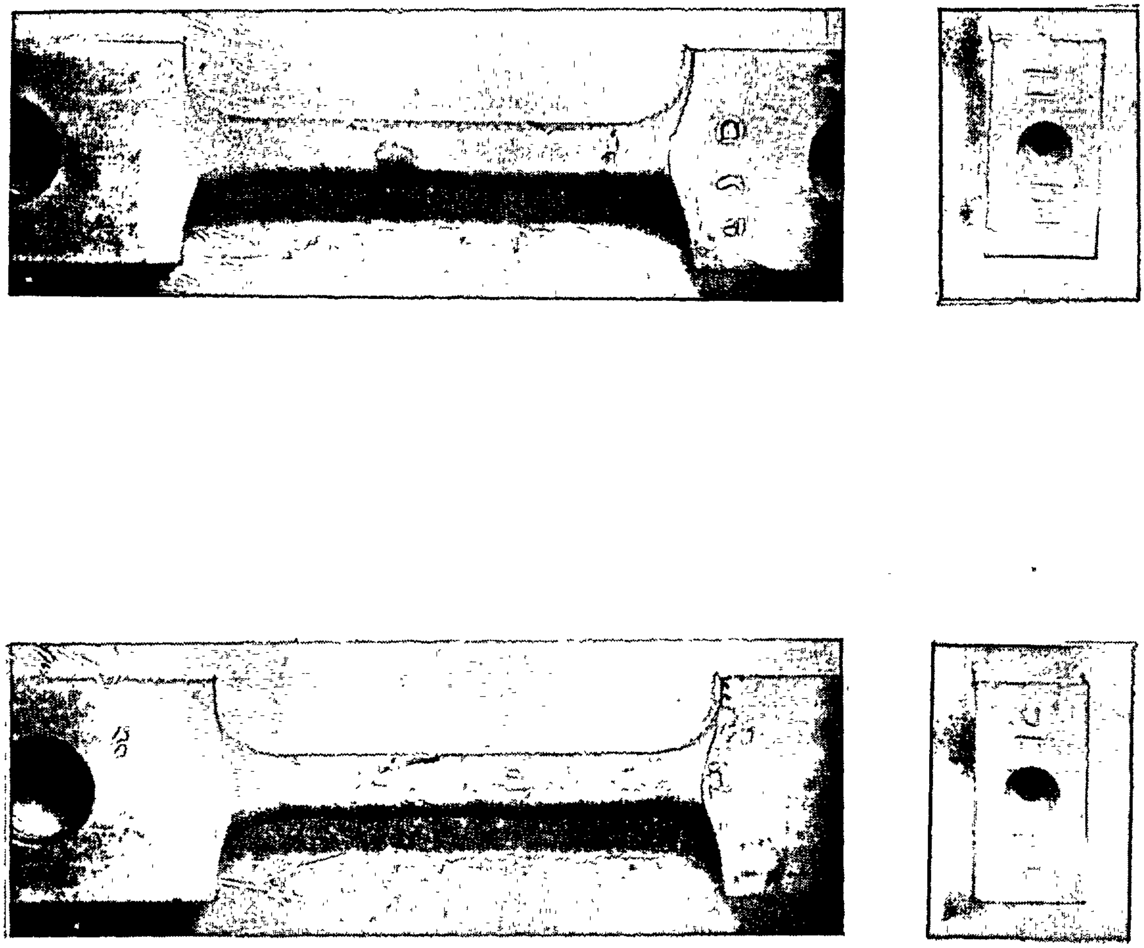


Figure B-9. Tensile specimens MT11 and MT12 from D. C. Cook Unit 2 reactor vessel Intermediate Shell Plate C5521-2 (transverse orientation) before testing.

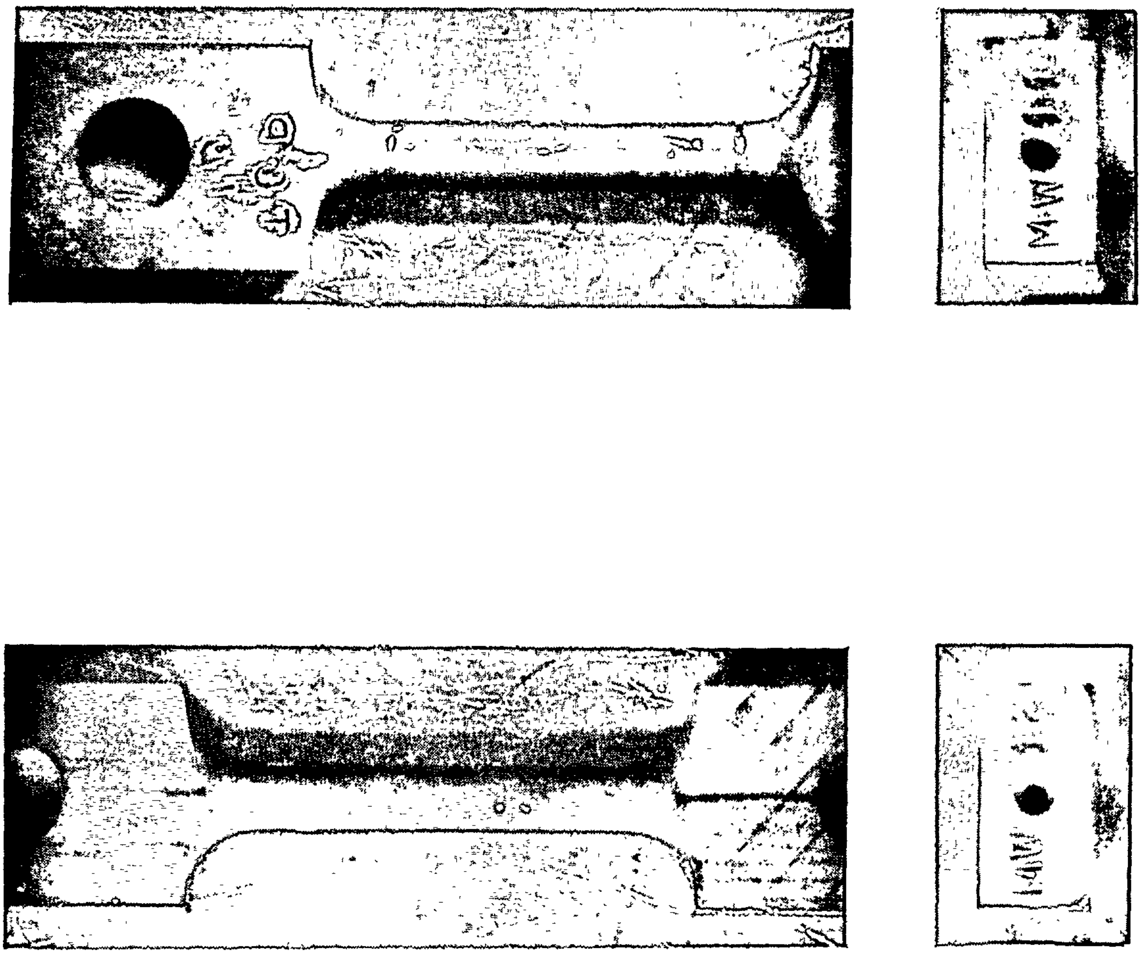


Figure B-10. Tensile specimens MW11 and MW12 from D. C. Cook Unit 2 reactor vessel weld before testing.

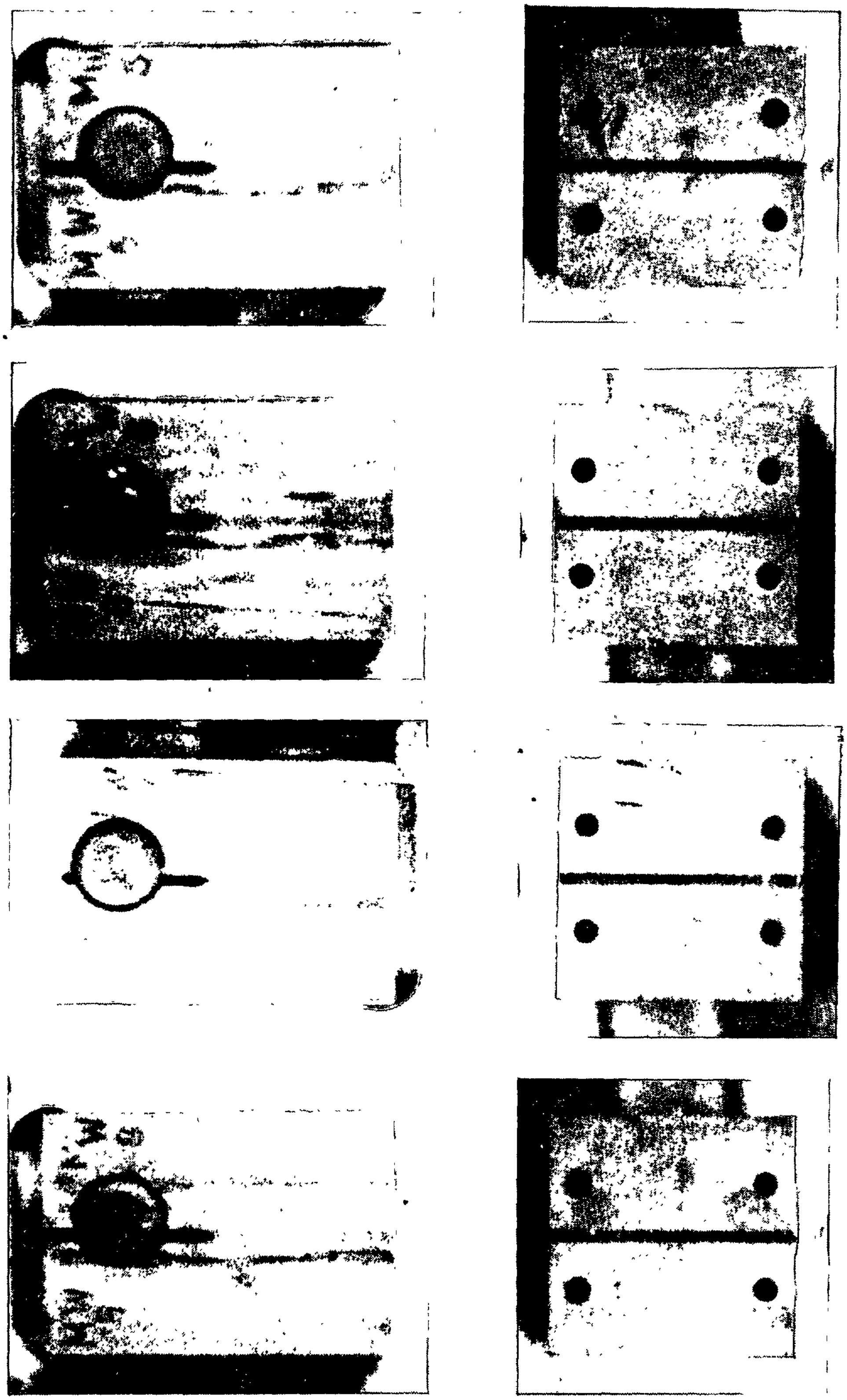


Figure B-11. WOL specimens MW5, MW6, MW7 and MW8, from D. C. Cook Unit 2 reactor vessel. The specimens were not tested, but stored for future reference.

APPENDIX C

Heatup and Cooldown Limit Curves for Normal Operation

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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]. Regulatory Guide 1.99, Revision 2 is used for the calculation of RT_{NDT} values at 1/4T and 3/4T locations (T is the thickness of the vessel at the beltline region).

2. FRACTURE TOUGHNESS PROPERTIES

The unirradiated RT_{NDT} values for the beltline region materials in the D. C. Cook Unit 2 reactor vessel were established using the guidance provided in NUREG-0800, Branch Technical Position, MTEB 5-2[2], and subarticale NB-2331 of the ASME Boiler and Pressure Vessel Code, Section III[3]. The pre-irradiation fracture-toughness properties of the D. C. Cook Unit 2 reactor vessel are presented in Table 1.

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

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

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[3] 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 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[4] 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 (621 psig without margins for instrumentation error and 561 psig with margins for D. C. Cook Unit 2).

Table 1 indicates that the limiting initial RT_{NDT} of 30°F occurs in the vessel flange of D. C. Cook Unit 2, so the minimum allowable temperature of this region is 150°F excluding margins for instrumentation error and 160°F with margins. These limits are shown in Figures 1 through 8 whenever applicable.

4. HEATUP AND COOLDOWN LIMIT CURVES

Limit curves for normal heatup and cooldown of the primary reactor pressure vessel have been calculated using the methods discussed in Section 3. If pressure readings are measured at other locations than the limiting beltline region, the pressure differences between the pressure transmitter and the limiting beltline region must be accounted for when using the pressure-temperature limit curves herein. The indicated pressure and temperature labels provided on the curves relate to the limiting beltline region of the reactor vessel.

Figures 1, 2, 5 and 6 contain the heatup curves for 60°F/hr. Figures 3, 4, 7 and 8 contain the cooldown curves up to 100°F/hr. Figures 1 and 3 are applicable for the first 32 EFPY of operation and include no margins for possible instrumentation errors. Figures 2 and 4 are applicable for the first 32 EFPY of operation and include margins of 10°F and 60 psig for possible instrumentation errors. Figures 5 and 7 are applicable for the first 15 EFPY of operation and include no margins for possible instrumentation errors. Figures 6 and 8 are applicable for the first 15 EFPY of operation and include margins of 10°F and 60 psig for possible instrumentation errors.

The current D. C. Cook Unit 2 low temperature overpressure protection system (LTOP) setpoints are valid up to the 15 EFPY pressure-temperature limit curves.

The 32 EFPY pressure-temperature limit curves cannot be used with the current LTOP setpoints.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 1 through 8. This is in addition to other criteria which must be met before the reactor is made critical.

The leak limit curve shown in Figures 1, 2, 5 and 6 represents minimum temperature requirements at the leak test pressure specified by applicable codes[2,3]. The leak test limit curve was determined by methods of References 2 and 4.

The criticality limit curve shown in Figure 1, 2, 5 and 6, specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 4. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum pressure-temperature curve for heatup and cooldown calculated as described in Section 3. The maximum temperature for the inservice hydrostatic test for the D. C. Cook Unit 2 reactor vessel for 32 EFPY is 348°F with margins for instrumentation errors and 335°F without margins for instrumentation errors. A vertical line at 348°F and 335°F on the pressure-temperature curves (with and without margins), intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel. The maximum temperature for the inservice hydrostatic test for the D. C. Cook Unit 2 reactor vessel for 15 EFPY is 324°F with margins for instrumentation errors and 311°F without margins for instrumentation errors. A vertical line at 324°F and 311°F on the pressure-temperature curves (with and without margins), intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 1 through 8 define limits for ensuring prevention of nonductile failure for the D. C. Cook Unit 2 reactor vessel.

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:

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

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB2331 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:

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

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.

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

where x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (4) to calculate ΔRT_{NDT} at the specific depth.

CF (°F) is the chemistry factor, obtained from Reference 1. All materials in the beltline region of D. C. Cook Unit 2 were considered for the limiting material. RT_{NDT} at 1/4T and 3/4T are summarized in Tables 2 and 3 for 32 and 15 EFPY respectively. From Tables 2 and 3, it can be seen that the limiting material is the intermediate shell plate C5556-2 for heatup and cooldown curves applicable up to 32 and 15 EFPY. Sample calculations for the RT_{NDT} for 32 and 15 EFPY are shown in Tables 4 and 5.

TABLE 1

D. C. COOK UNIT 2 REACTOR VESSEL TOUGHNESS TABLE (Unirradiated)

Material Description	CU (%)	NI (%)	I-RTNDT (a) (°F)
Closure Head Flange, 4437-V-1	--	--	-20 (b)
Vessel Flange, 4436-V-2	--	--	30 (b)
Intermediate Shell, C5556-2	0.15	0.57	58
Intermediate Shell, C5521-2*	0.125	0.58	38
Lower Shell, C5540-2	0.11	0.64	-20
Lower Shell, C5592-1	0.14	0.59	-20
Intermediate and Lower Shell Long. and Girth Weld Seams (Ht. S3986, Linde 124, Flux Lot No. 0934)*	0.052	0.967	-35

* % weight copper and nickel content are mean values based on the available chemistry test results as indicated below

a. The initial RT_{NDT} (I) values for the plates and welds are measured values based on actual data.

b. To be used for considering flange requirements for heatup/cooldown curves[4].

Material	Data Source	Copper (wt. %)	Nickel (wt. %)
Plate, C5521-2	Original Mill Test Report	0.14	0.58
	Surveillance Program [1]	0.11	0.58
	Mean value	0.125	0.58
Weld	Original Mill Test Report	0.05	0.97
	Surveillance Program [1]	0.055	0.97
	Surveillance Program [1]	0.05	0.96
	Mean value	0.052	0.967

TABLE 2
SUMMARY OF ADJUSTED REFERENCE TEMPERATURE (ART) AT 1/4T and 3/4T LOCATION FOR 32 EFPY

Component	32 EFPY RT_{NDT} at 1/4T (°F)	32 EFPY RT_{NDT} at 3/4T (°F)
Intermediate Shell Plate, C5556-2	201	171
Intermediate Shell Plate, C5521-2	159 (158)	135 (129)
Lower Shell Plate, C5540-2	89	68
Lower Shell Plate, C5592-1	114	86
Intermed. Shell Longitudinal Welds (a)	80	45
Lower Shell Longitudinal Welds (b)	92	68
Circumferential Weld	92 (64)	68 (40)

RT_{NDT} numbers within ( ) are based on chemistry factor calculated using capsule data.

(a) Intermediate shell longitudinal welds are located at 10°

(b) Lower shell longitudinal welds are located at 90°

TABLE 3

SUMMARY OF ADJUSTED REFERENCE TEMPERATURE (ART) AT 1/4T and 3/4T LOCATION FOR 15 EFPY

Component	15 EFPY RT_{NDT} at 1/4T (°F)	15 EFPY RT_{NDT} at 3/4T (°F)
Intermediate Shell Plate, C5556-2	178	150
Intermediate Shell Plate, C5521-2	141 (137)	118 (110)
Lower Shell Plate, C5540-2	73	54
Lower Shell Plate, C5592-1	93	67
Intermed. Shell Longitudinal Welds (a)	52	20
Lower Shell Longitudinal Welds (b)	40	11
Circumferential Weld	77 (49)	41 (28)

RT_{NDT} numbers within ( ) are based on chemistry factor calculated using capsule data.

(a) Intermediate shell longitudinal welds are located at 10°

(b) Lower shell longitudinal welds are located at 90°

TABLE 4

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR LIMITING D. C. COOK UNIT 2 REACTOR VESSEL MATERIAL - INTERMEDIATE SHELL PLATE, C5556-2 FOR 32 EFPY

Parameter	Regulatory Guide 1.99 - Revision 2	
	32 EFPY	
	1/4 T	3/4 T
Chemistry Factor, CF (°F)	108.35	108.35
Fluence, f (10^{19} n/cm^2)(a)	1.027	0.3703
Fluence Factor, ff	1.007	0.725
************************************		
ΔRT_{NDT} = CF x ff (°F)	109	79
Initial RT_{NDT}, I (°F)	58	58
Margin, M (°F) (b)	34	34
************************************		
Revision 2 to Regulatory Guide 1.99		
Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	201 *	171 *

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 1.71 at 32 EFPY. The D. C. Cook Unit 2 reactor vessel wall thickness is 8.5 inches at the beltline region.

(b) Margin is calculated as, $M = 2\,[\,\sigma_I^2 + \sigma_\Delta^2]^{0.5}$. 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} term, σ_Δ, is 17°F for the base metal, except that σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

* Limiting value used in development of heatup and cooldown limit curves.

TABLE 5

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES FOR LIMITING D. C. COOK UNIT 2 REACTOR VESSEL MATERIAL - INTERMEDIATE SHELL PLATE, C5556-2 FOR 15 EFPY

Parameter	Regulatory Guide 1.99 - Revision 2 15 EFPY 1/4 T	3/4 T
Chemistry Factor, CF (°F)	108.35	108.35
Fluence, f (10^{19} n/cm^2)[(a)]	0.483	0.1742
Fluence Factor, ff	0.797	0.537
ΔRT_{NDT} = CF x ff (°F)	86	58
Initial RT_{NDT}, I (°F)	58	58
Margin, M (°F) [(b)]	34	34
Revision 2 to Regulatory Guide 1.99		
Adjusted Reference Temperature, ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	178 *	150 *

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 Mev) = 0.804 at 15 EFPY. The D. C. Cook Unit 2 reactor vessel wall thickness is 8.5 inches at the beltline region.

(b) Margin is calculated as, $M = 2 [\sigma_I^2 + \sigma_\Delta^2]^{0.5}$. 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} term, σ_Δ, is 17°F for the base metal, except that σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

* Limiting value used in development of heatup and cooldown limit curves.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE, C5556-2

LIMITING ART AFTER 32 EFPY: 1/4T, 201°F
3/4T, 171°F

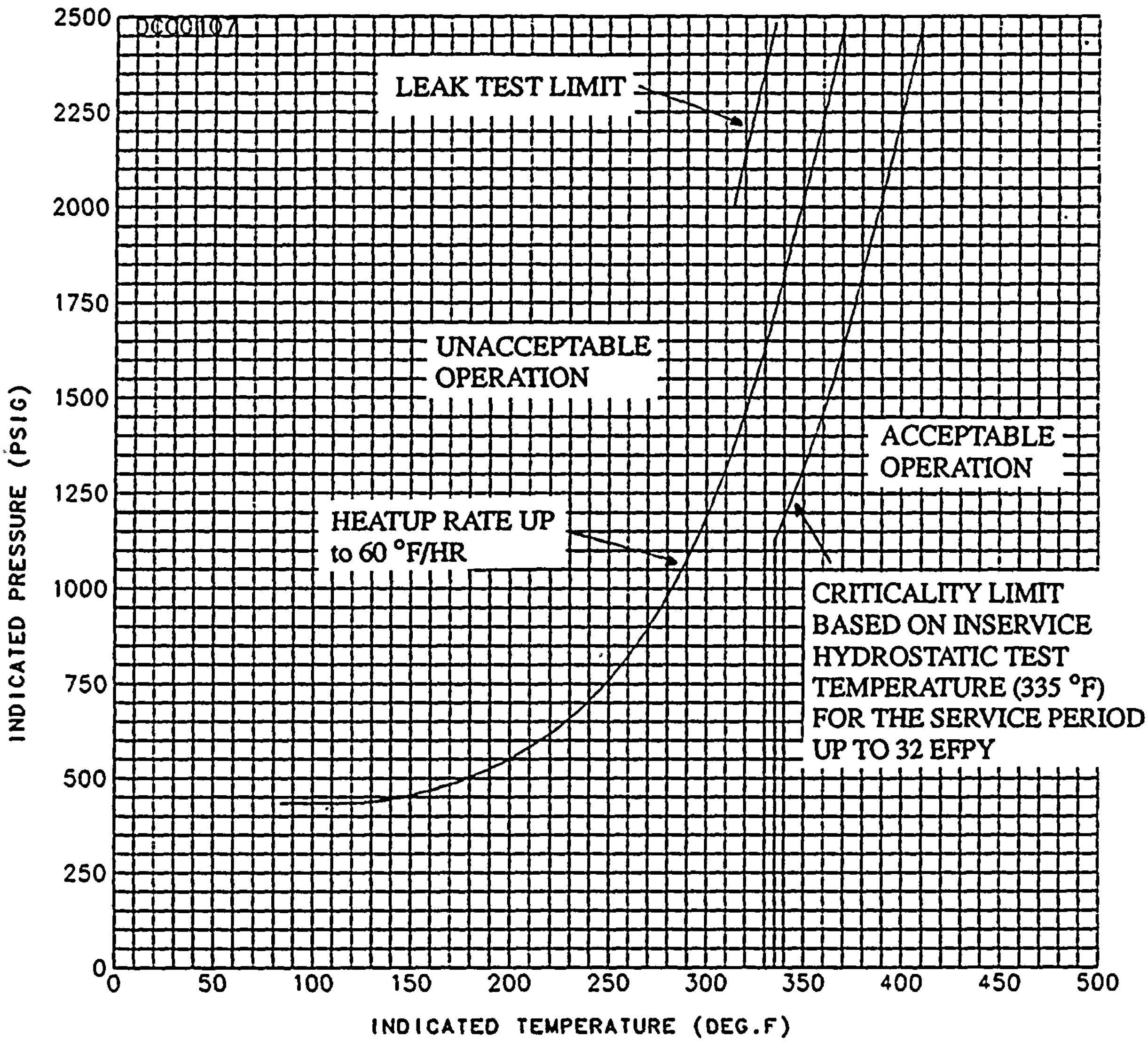

Figure 1. D. C. Cook Unit 2 Reactor Coolant System Heatup Limitations (Heat up rate up to 60°F/hr) Applicable for the First 32 EFPY (Without Margins For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE, C5556-2

LIMITING ART AFTER 32 EFPY: 1/4T, 201°F

3/4T, 171°F

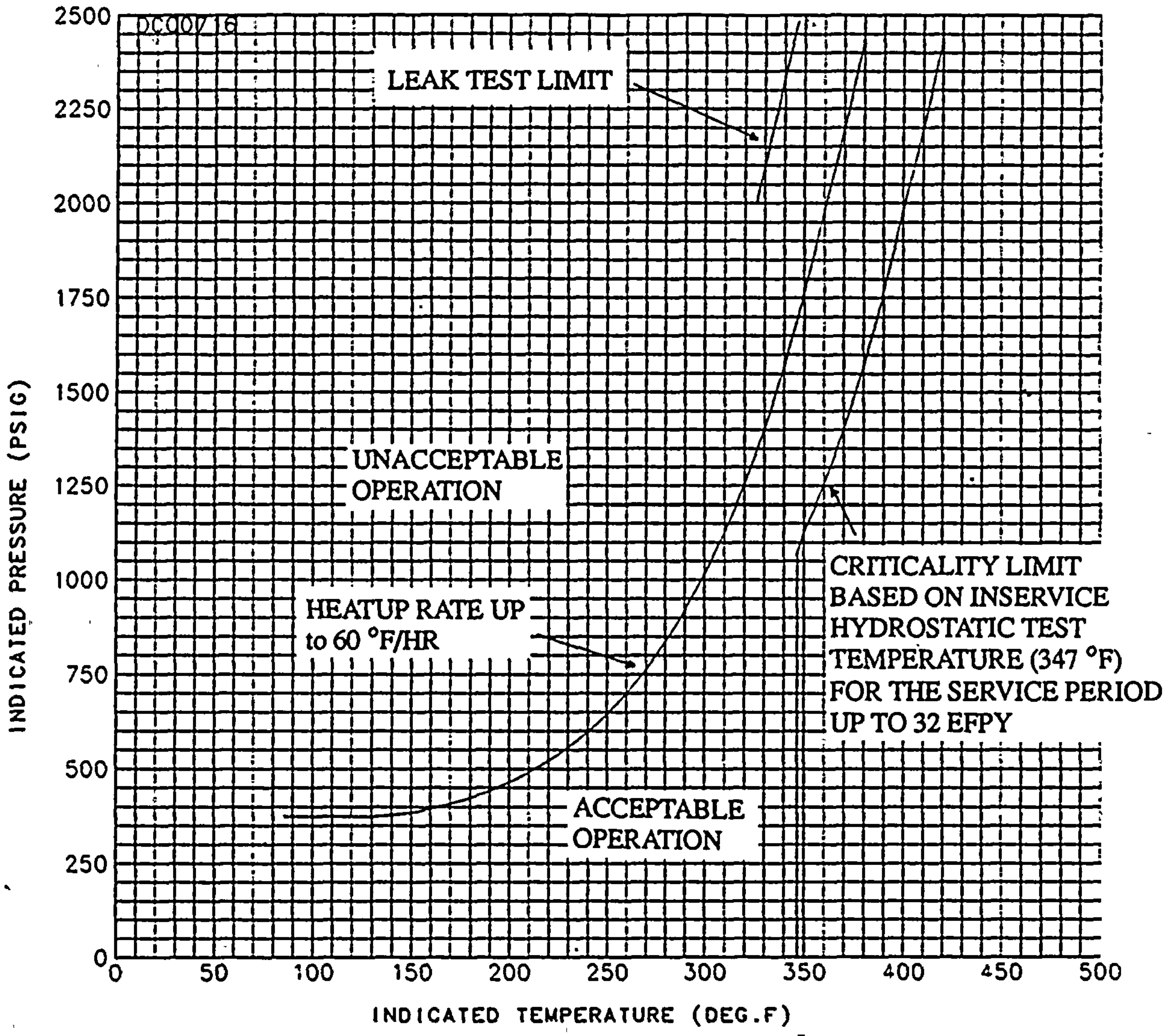

Figure 2. D. C. Cook Unit 2 Reactor Coolant System Heatup Limitations (Heat up rate up to 60°F/hr) Applicable for the First 32 EFPY (With Margins of 10°F and 60 psig For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE, C5556-2

LIMITING ART AFTER 32 EFPY: 1/4T, 201°F

3/4T, 171°F

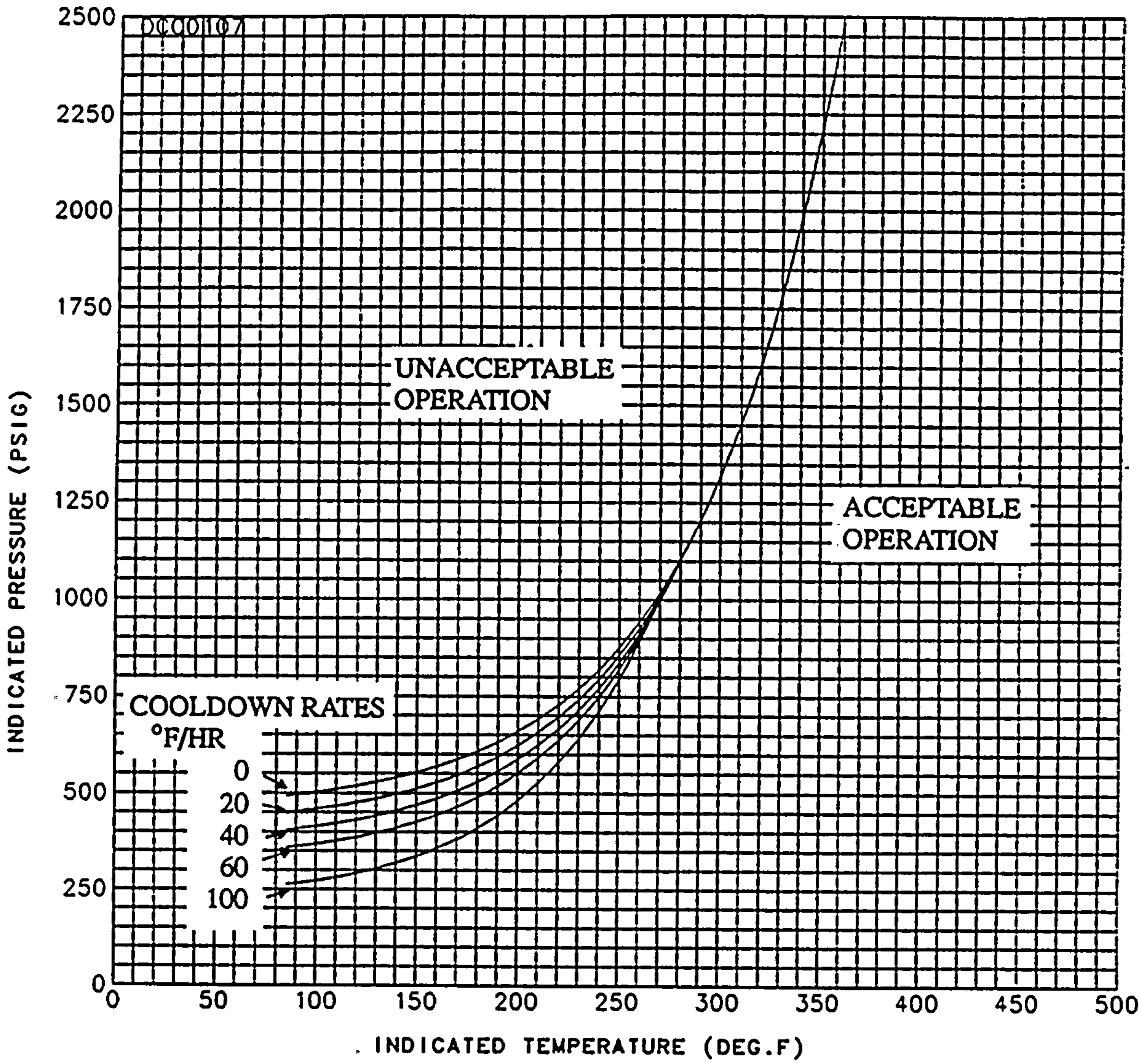

Figure 3. D. C. Cook Unit 2 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 32 EFPY (Without Margins For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE, C5556-2

LIMITING ART AFTER 32 EFPY: 1/4T, 201°F

3/4T, 171°F

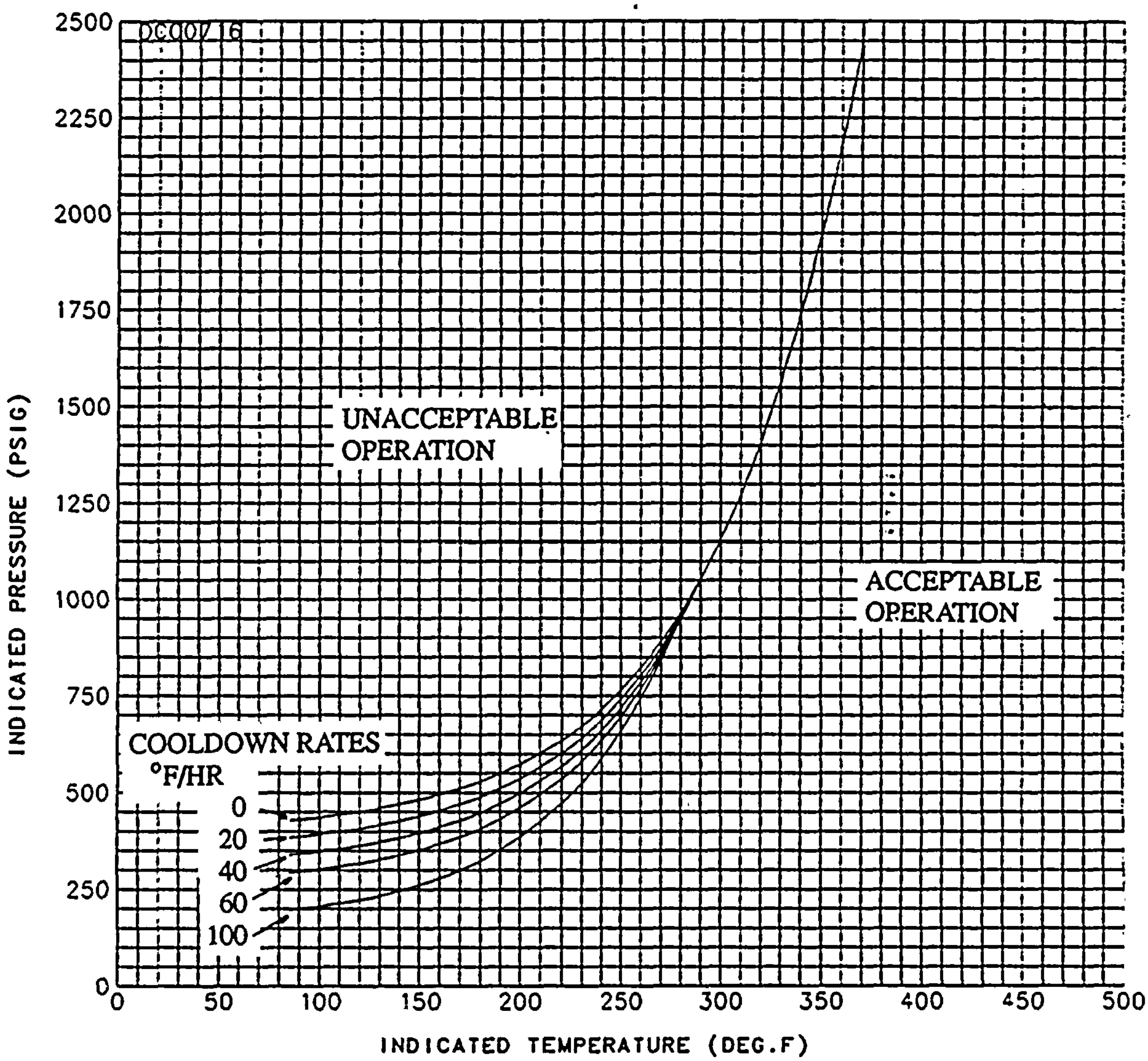

Figure 4. D. C. Cook Unit 2 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 32 EFPY (With Margins of 10°F and 60 psig for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE, C5556-2

LIMITING ART AFTER 15 EFPY: 1/4T, 178°F

3/4T, 150°F

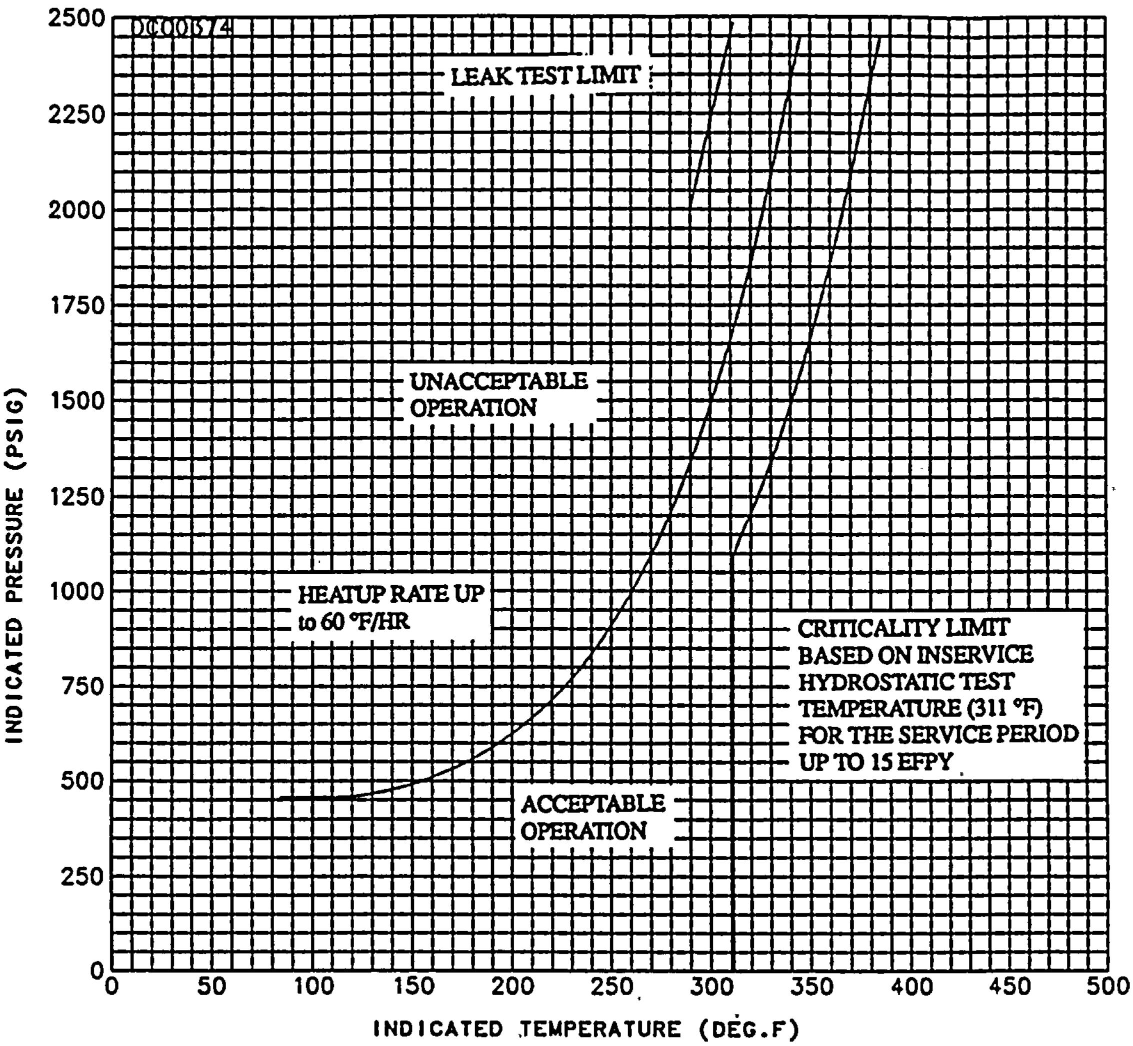

Figure 5. D. C. Cook Unit 2 Reactor Coolant System Heatup Limitations (Heat up rate up to 60°F/hr) Applicable for the First 15 EFPY (Without Margins For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE, C5556-2

LIMITING ART AFTER 15 EFPY: 1/4T, 178°F
3/4T, 150°F

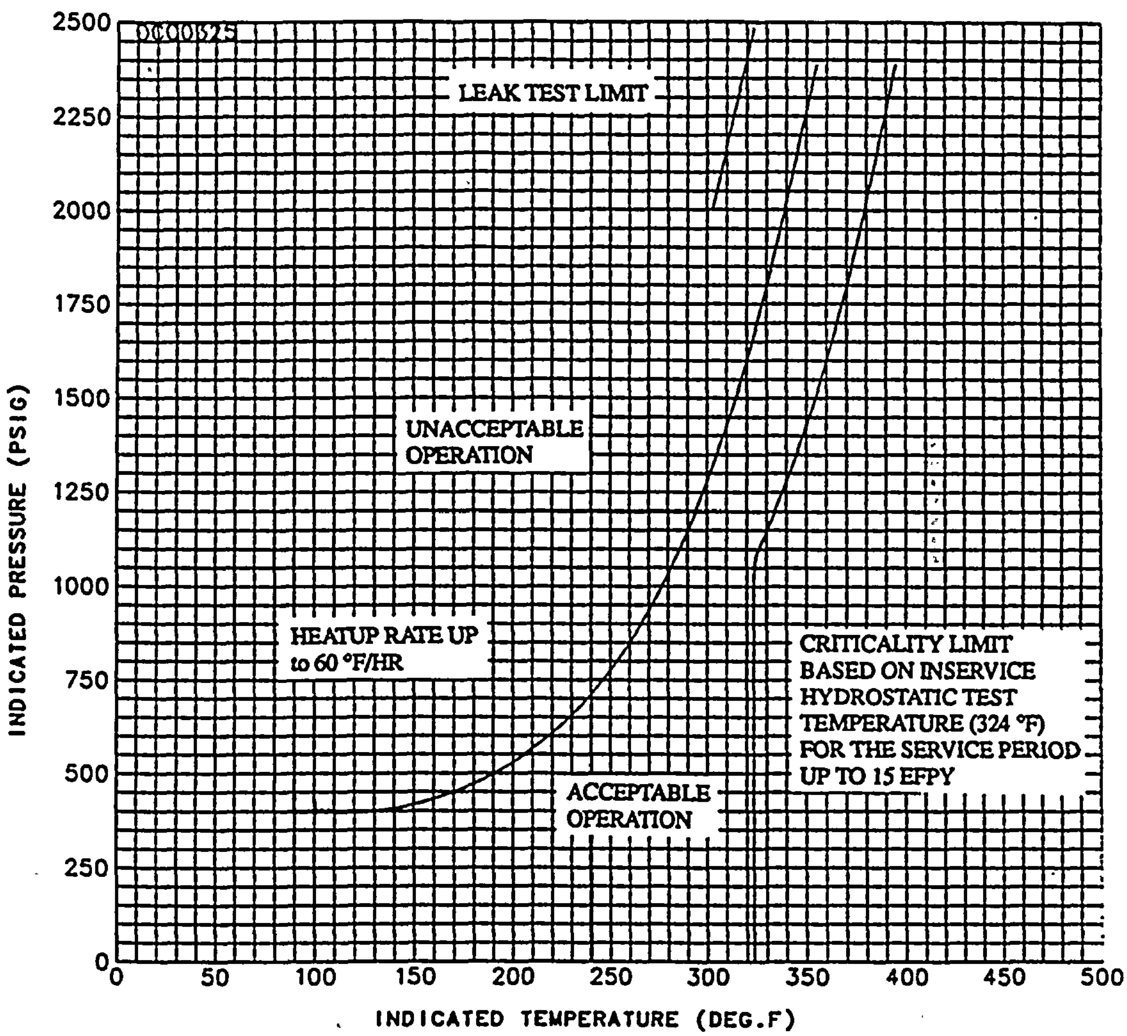

Figure 6. D. C. Cook Unit 2 Reactor Coolant System Heatup Limitations (Heat up rate up to 60°F/hr) Applicable for the First 15 EFPY (With Margins of 10°F and 60 psig For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE, C5556-2

LIMITING ART AFTER 15 EFPY: 1/4T, 178°F

3/4T, 150°F

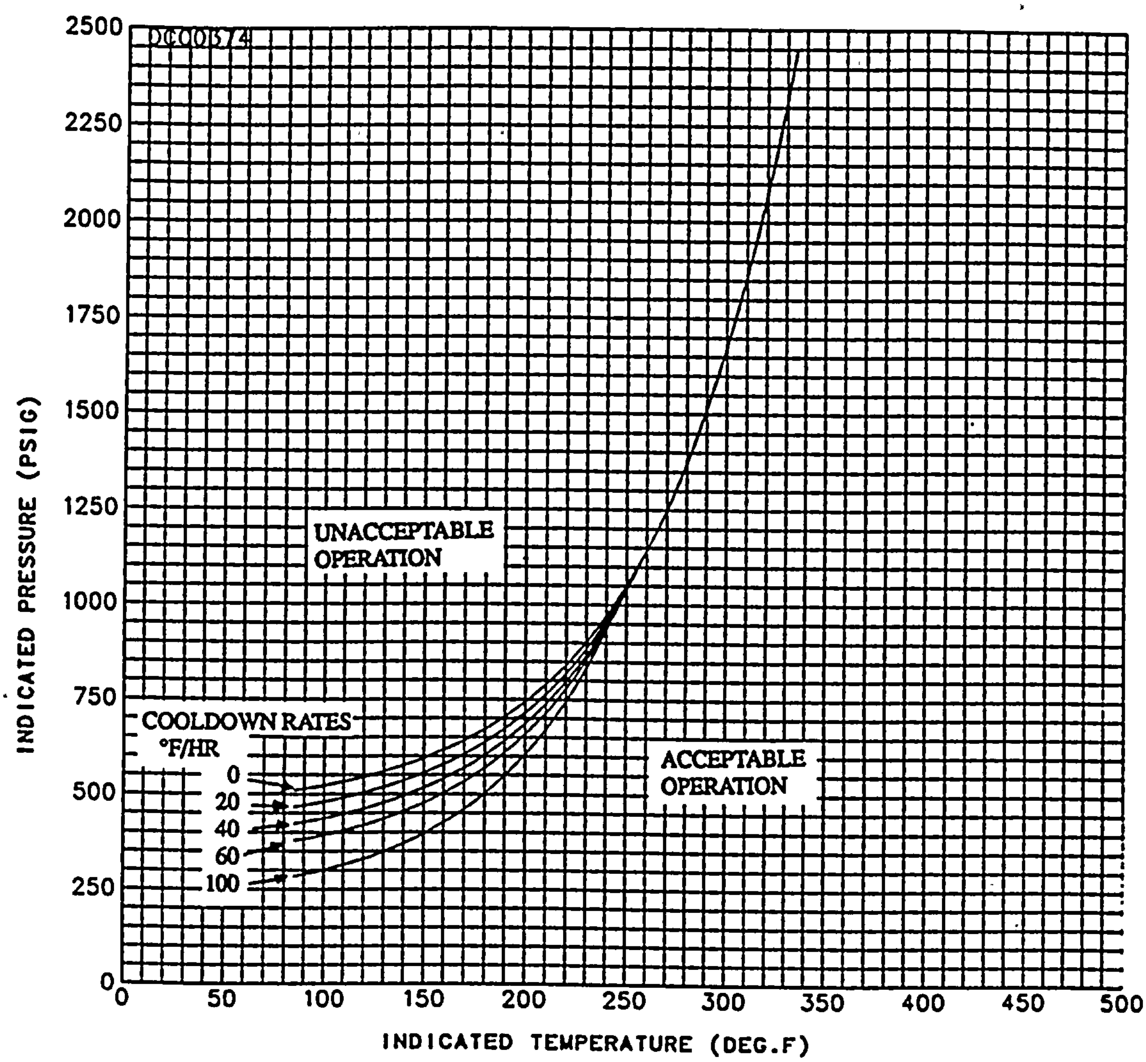

Figure 7. D. C. Cook Unit 2 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 15 EFPY (Without Margins For Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE, C5556-2

LIMITING ART AFTER 15 EFPY: 1/4T, 178°F
3/4T, 150°F

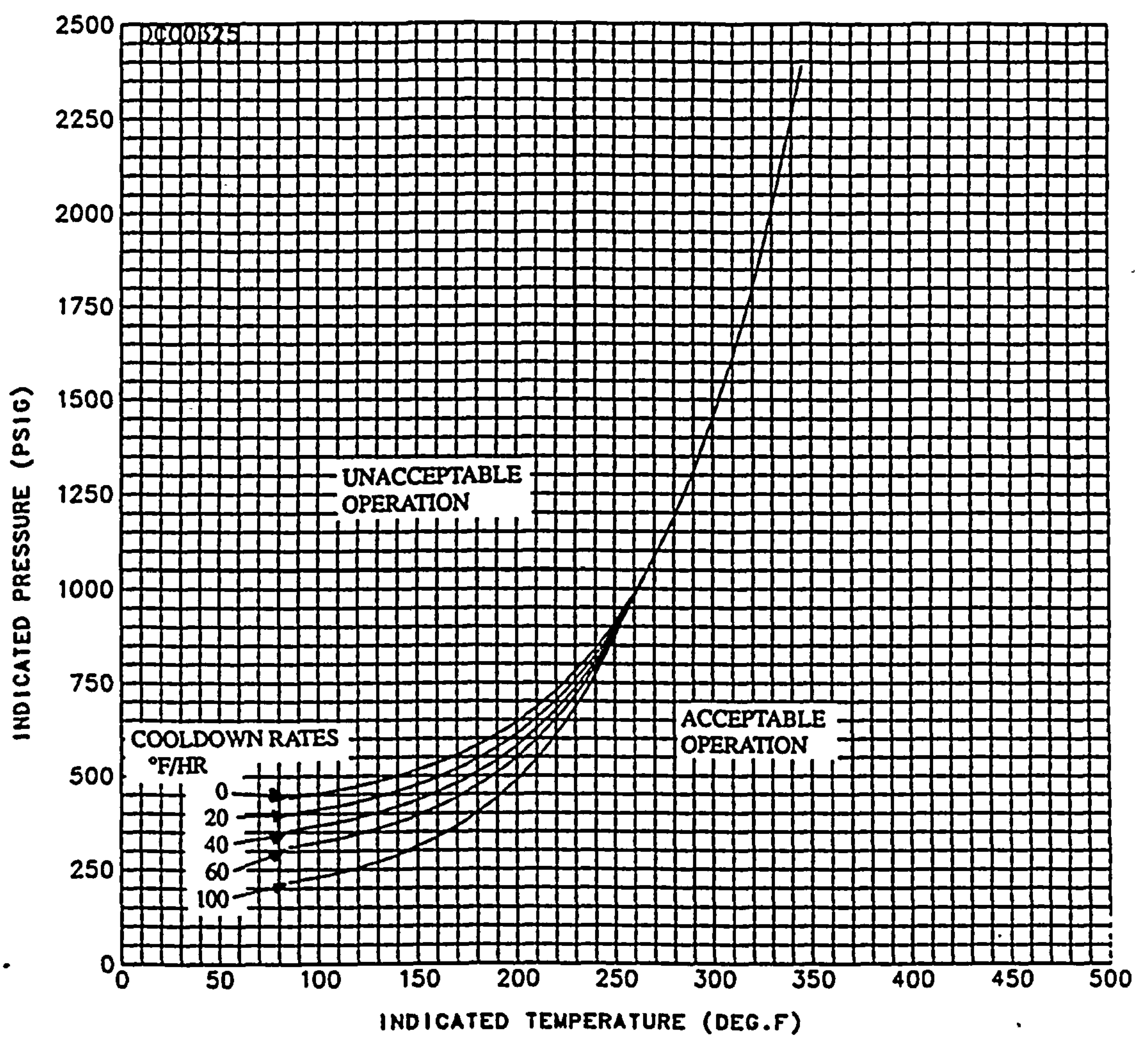

Figure 8. D. C. Cook Unit 2 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 15 EFPY (With Margins of 10°F and 60 psig for Instrumentation Errors)

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-0800, 1981.

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

4 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.

5 Letter Report, MT-SMART-090(89), "D. C. Cook Unit 2 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation", N. K. Ray, April 1989.

6 WCAP-8512, "American Electric Power Company Donald C. Cook Unit No. 2 Reactor Vessel Radiation Surveillance Program", J. A. Davidson, et al., November 1975.

7 "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", 10 CFR Part 50, Vol. 58, No. 94, May 15, 1991.

ATTACHMENT 1

DATA POINTS FOR HEATUP AND COOLDOWN CURVES

(Without Margins for Instrumentation Errors)

The data points used in the development of the heatup and cooldown curves shown in Figures 1 and 3 are contained on the attached computer printout sheets.

AMP 60 DEG-F/HR HEATUP REG.GUIDE 1.99,REV.2 WITHOUT MARGIN 09/28/92

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (32.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	314
2485	335

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

AMP 60 DEG-F/HR HEATUP REG.GUIDE 1.99,REV.2 WITHOUT MARGIN 09/28/92

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	~~463.04~~	21	185.000	513.04	40	280.000	978.92
2	90.000	~~452.83~~	22	190.000	524.65	41	285.000	1025.52
3	95.000	~~445.53~~ 433.80	23	195.000	537.13	42	290.000	1075.56
4	100.000	~~440.09~~	24	200.000	550.71	43	295.000	1129.28
5	105.000	~~436.64~~	25	205.000	565.38	44	300.000	1186.80
6	110.000	~~434.52~~	26	210.000	581.19	45	305.000	1248.53
7	115.000	433.80	27	215.000	598.11	46	310.000	1314.70
8	120.000	434.12	28	220.000	616.45	47	315.000	1385.56
9	125.000	435.53	29	225.000	636.06	48	320.000	1461.55
10	130.000	437.77	30	230.000	657.31	49	325.000	1542.87
11	135.000	440.85	31	235.000	680.15	50	330.000	1629.85
12	140.000	444.72	32	240.000	704.60	51	335.000	1723.05
13	145.000	449.38	33	245.000	730.85	52	340.000	1822.90
14	150.000	454.73	34	250.000	759.27	53	345.000	1924.64
15	155.000	460.84	35	255.000	789.61	54	350.000	2020.07
16	160.000	467.63	36	260.000	822.35	55	355.000	2121.91
17	165.000	475.18	37	265.000	857.45	56	360.000	2230.63
18	170.000	483.45	38	270.000	895.10	57	365.000	2346.74
19	175.000	492.53	39	275.000	935.52	58	370.000	2470.52
20	180.000	502.30						

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	492.98	20	180.000	607.36	39	275.000	1058.28
2	90.000	495.89	21	185.000	618.87	40	280.000	1103.35
3	95.000	499.02	22	190.000	631.24	41	285.000	1151.78
4	100.000	502.38	23	195.000	644.42	42	290.000	1203.56
5	105.000	505.99	24	200.000	658.74	43	295.000	1259.45
6	110.000	509.88	25	205.000	674.11	44	300.000	1319.45
7	115.000	514.06	26	210.000	690.49	45	305.000	1383.70
8	120.000	518.55	27	215.000	708.28	46	310.000	1452.62
9	125.000	523.38	28	220.000	727.34	47	315.000	1526.55
10	130.000	528.57	29	225.000	747.75	48	320.000	1605.82
11	135.000	534.15	30	230.000	769.81	49	325.000	1690.70
12	140.000	540.05	31	235.000	793.38	50	330.000	1781.77
13	145.000	546.50	32	240.000	818.84	51	335.000	1879.21
14	150.000	553.44	33	245.000	846.11	52	340.000	1983.42
15	155.000	560.90	34	250.000	875.34	53	345.000	2094.78
16	160.000	568.91	35	255.000	907.00	54	350.000	2214.28
17	165.000	577.54	36	260.000	940.84	55	355.000	2341.47
18	170.000	586.68	37	265.000	977.14	56	360.000	2477.36
19	175.000	596.64	38	270.000	1016.15			

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITHOUT MARGIN 09/28/92

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	449.12	15	155.000	519.26	28	220.000	695.99
2	90.000	452.03	16	160.000	527.68	29	225.000	717.97
3	95.000	455.19	17	165.000	536.76	30	230.000	741.44
4	100.000	458.58	18	170.000	546.42	31	235.000	766.88
5	105.000	462.27	19	175.000	556.96	32	240.000	794.05
6	110.000	466.23	20	180.000	568.28	33	245.000	823.42
7	115.000	470.52	21	185.000	580.49	34	250.000	854.93
8	120.000	475.13	22	190.000	593.50	35	255.000	888.77
9	125.000	480.13	23	195.000	607.65	36	260.000	925.12
10	130.000	485.49	24	200.000	622.87	37	265.000	964.44
11	135.000	491.30	25	205.000	639.12	38	270.000	1006.49
12	140.000	497.44	26	210.000	656.75	39	275.000	1051.77
13	145.000	504.19	27	215.000	675.73	40	280.000	1100.35
14	150.000	511.43						

COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITHOUT MARGIN 09/28/92

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	404.27	15	155.000	476.97	28	220.000	665.35
2	90.000	407.14	16	160.000	485.84	29	225.000	688.77
3	95.000	410.34	17	165.000	495.35	30	230.000	714.14
4	100.000	413.79	18	170.000	505.68	31	235.000	741.31
5	105.000	417.56	19	175.000	516.85	32	240.000	770.69
6	110.000	421.62	20	180.000	528.86	33	245.000	802.22
7	115.000	426.04	21	185.000	541.74	34	250.000	836.07
8	120.000	430.80	22	190.000	555.71	35	255.000	872.54
9	125.000	435.98	23	195.000	570.80	36	260.000	911.96
10	130.000	441.49	24	200.000	586.91	37	265.000	954.24
11	135.000	447.54	25	205.000	604.43	38	270.000	999.65
12	140.000	454.06	26	210.000	623.28	39	275.000	1048.51
13	145.000	461.12	27	215.000	643.47	40	280.000	1101.03
14	150.000	468.73						

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITHOUT MARGIN 09/28/92

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	358.24	14	150.000	425.29	27	215.000	611.88
2	90.000	361.20	15	155.000	433.99	28	220.000	635.24
3	95.000	364.41	16	160.000	443.30	29	225.000	660.61
4	100.000	367.93	17	165.000	453.48	30	230.000	687.75
5	105.000	371.80	18	170.000	464.44	31	235.000	717.21
6	110.000	375.99	19	175.000	476.32	32	240.000	748.73
7	115.000	380.56	20	180.000	489.12	33	245.000	782.71
8	120.000	385.50	21	185.000	502.88	34	250.000	819.44
9	125.000	390.89	22	190.000	517.80	35	255.000	858.89
10	130.000	396.71	23	195.000	533.93	36	260.000	901.34
11	135.000	403.05	24	200.000	551.19	37	265.000	947.04
12	140.000	409.83	25	205.000	569.96	38	270.000	996.21
13	145.000	417.27	26	210.000	590.05	39	275.000	1049.09

COOLDOWN CURVES REG. GUIDE 1.99,REV.2 WITHOUT MARGIN 09/28/92

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	262.50	14	150.000	336.15	27	215.000	551.05
2	90.000	265.58	15	155.000	345.99	28	220.000	578.41
3	95.000	269.01	16	160.000	356.63	29	225.000	607.88
4	100.000	272.75	17	165.000	368.16	30	230.000	639.62
5	105.000	276.90	18	170.000	380.66	31	235.000	674.05
6	110.000	281.41	19	175.000	394.26	32	240.000	711.01
7	115.000	286.36	20	180.000	408.87	33	245.000	750.89
8	120.000	291.76	21	185.000	424.79	34	250.000	793.84
9	125.000	297.70	22	190.000	441.91	35	255.000	840.17
10	130.000	304.13	23	195.000	460.53	36	260.000	890.04
11	135.000	311.18	24	200.000	480.63	37	265.000	943.83
12	140.000	318.82	25	205.000	502.30	38	270.000	1001.60
13	145.000	327.12	26	210.000	525.76			

ATTACHMENT 2

DATA POINTS FOR HEATUP AND COOLDOWN CURVES

(With Margins of 10°F and 60 psig for Instrumentation Errors)

The data points used in the development of the heatup and cooldown curves shown in Figures 2 and 4 are contained on the attached computer printout sheets.

AMP 60 DEG-F/HR HEATUP REG. GUIDE 1.99, REV.2 WITH MARGINS 10/12/92

THE FOLLOWING DATA WERE CALCULATEDFOR THE INSERVICE HYDROSTATIC LEAK TEST.

MINIMUM INSERVICE LEAK TEST TEMPERATURE (32.000 EFPY)

PRESSURE (PSI)	TEMPERATURE (DEG.F)
2000	326
2485	347

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

AMP 60 DEG-F/HR HEATUP REG. GUIDE 1.99, REV.2 WITH MARGINS 10/12/92

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	~~428.34~~	21	185.000	432.53	41	285.000	875.52
2	90.000	~~416.02~~	22	190.000	442.30	42	290.000	918.92
3	95.000	~~403.04~~	23	195.000	453.04	43	295.000	965.52
4	100.000	~~392.93~~ 373.80	24	200.000	464.65	44	300.000	1015.56
5	105.000	~~385.53~~	25	205.000	477.13	45	305.000	1069.28
6	110.000	~~380.08~~	26	210.000	490.71	46	310.000	1126.80
7	115.000	~~376.64~~	27	215.000	505.38	47	315.000	1188.53
8	120.000	~~374.52~~	28	220.000	521.19	48	320.000	1254.70
9	125.000	373.80	29	225.000	538.11	49	325.000	1325.56
10	130.000	374.12	30	230.000	556.45	50	330.000	1401.55
11	135.000	375.53	31	235.000	576.06	51	335.000	1482.87
12	140.000	377.77	32	240.000	597.31	52	340.000	1569.85
13	145.000	380.85	33	245.000	620.15	53	345.000	1663.05
14	150.000	384.72	34	250.000	644.60	54	350.000	1762.90
15	155.000	389.38	35	255.000	670.85	55	355.000	1869.46
16	160.000	394.73	36	260.000	699.27	56	360.000	1979.87
17	165.000	400.84	37	265.000	729.61	57	365.000	2082.95
18	170.000	407.63	38	270.000	762.35	58	370.000	2193.34
19	175.000	415.18	39	275.000	797.45	59	375.000	2311.05
20	180.000	423.45	40	280.000	835.10	60	380.000	2436.53

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	428.34	21	185.000	538.72	40	280.000	964.38
2	90.000	430.89	22	190.000	549.59	41	285.000	1006.91
3	95.000	433.55	23	195.000	561.28	42	290.000	1052.62
4	100.000	436.50	24	200.000	573.70	43	295.000	1101.74
5	105.000	439.67	25	205.000	587.20	44	300.000	1154.52
6	110.000	443.08	26	210.000	601.72	45	305.000	1211.18
7	115.000	446.75	27	215.000	617.30	46	310.000	1271.92
8	120.000	450.69	28	220.000	633.95	47	315.000	1336.95
9	125.000	454.93	29	225.000	651.99	48	320.000	1407.01
10	130.000	459.49	30	230.000	671.20	49	325.000	1481.90
11	135.000	464.39	31	235.000	692.05	50	330.000	1562.30
12	140.000	469.65	32	240.000	714.40	51	335.000	1648.49
13	145.000	475.32	33	245.000	738.34	52	340.000	1740.74
14	150.000	481.30	34	250.000	764.15	53	345.000	1839.56
15	155.000	487.85	35	255.000	791.84	54	350.000	1945.08
16	160.000	494.89	36	260.000	821.50	55	355.000	2058.16
17	165.000	502.45	37	265.000	853.59	56	360.000	2178.95
18	170.000	510.59	38	270.000	887.92	57	365.000	2308.09
19	175.000	519.34	39	275.000	924.79	58	370.000	2445.60
20	180.000	528.62						

AMP COOLDOWN CURVES REG. GUIDE 1.99, REV.2 WITH MARGINS 10/12/92

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	384.45	15	155.000	445.63	29	225.000	619.71
2	90.000	386.97	16	160.000	452.99	30	230.000	640.29
3	95.000	389.72	17	165.000	460.94	31	235.000	662.57
4	100.000	392.68	18	170.000	469.47	32	240.000	686.41
5	105.000	395.89	19	175.000	478.59	33	245.000	712.18
6	110.000	399.33	20	180.000	488.50	34	250.000	739.80
7	115.000	403.08	21	185.000	499.19	35	255.000	769.45
8	120.000	407.10	22	190.000	510.68	36	260.000	801.54
9	125.000	411.46	23	195.000	523.07	37	265.000	835.91
10	130.000	416.14	24	200.000	536.28	38	270.000	872.80
11	135.000	421.21	25	205.000	550.64	39	275.000	912.49
12	140.000	426.65	26	210.000	566.08	40	280.000	955.14
13	145.000	432.55	27	215.000	582.58	41	285.000	1001.02
14	150.000	438.79	28	220.000	600.47	42	290.000	1050.33

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	339.64	15	155.000	402.68	29	225.000	587.78
2	90.000	342.16	16	160.000	410.40	30	230.000	609.98
3	95.000	344.92	17	165.000	418.77	31	235.000	633.75
4	100.000	347.84	18	170.000	427.77	32	240.000	659.48
5	105.000	351.09	19	175.000	437.43	33	245.000	687.07
6	110.000	354.60	20	180.000	447.91	34	250.000	716.71
7	115.000	358.43	21	185.000	459.25	35	255.000	748.86
8	120.000	362.55	22	190.000	471.45	36	260.000	783.24
9	125.000	367.04	23	195.000	484.53	37	265.000	820.25
10	130.000	371.88	24	200.000	498.71	38	270.000	860.04
11	135.000	377.14	25	205.000	514.02	39	275.000	903.09
12	140.000	382.73	26	210.000	530.37	40	280.000	949.17
13	145.000	388.88	27	215.000	548.16	41	285.000	998.74
14	150.000	395.50	28	220.000	567.27	42	290.000	1051.84

AMP COOLDOWN CURVES REG. GUIDE 1.99, REV.2 WITH MARGINS 10/12/92

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	293.63	15	155.000	358.94	29	225.000	556.53
2	90.000	296.15	16	160.000	367.08	30	230.000	580.24
3	95.000	298.93	17	165.000	375.92	31	235.000	605.99
4	100.000	301.94	18	170.000	385.38	32	240.000	633.53
5	105.000	305.21	19	175.000	395.72	33	245.000	663.40
6	110.000	308.80	20	180.000	406.85	34	250.000	695.43
7	115.000	312.73	21	185.000	418.91	35	255.000	729.92
8	120.000	316.99	22	190.000	431.91	36	260.000	767.02
9	125.000	321.64	23	195.000	445.88	37	265.000	807.21
10	130.000	326.66	24	200.000	461.02	38	270.000	850.25
11	135.000	332.14	25	205.000	477.29	39	275.000	896.63
12	140.000	338.05	26	210.000	494.93	40	280.000	946.47
13	145.000	344.50	27	215.000	513.98	41	285.000	1000.17
14	150.000	351.38	28	220.000	534.37			

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	197.87	15	155.000	269.06	28	220.000	470.82
2	90.000	200.43	16	160.000	278.24	29	225.000	496.49
3	95.000	203.30	17	165.000	288.24	30	230.000	524.26
4	100.000	206.44	18	170.000	299.05	31	235.000	554.18
5	105.000	209.94	19	175.000	310.76	32	240.000	586.40
6	110.000	213.75	20	180.000	323.47	33	245.000	621.20
7	115.000	217.98	21	185.000	337.28	34	250.000	658.85
8	120.000	222.54	22	190.000	352.12	35	255.000	699.35
9	125.000	227.61	23	195.000	368.29	36	260.000	742.94
10	130.000	233.10	24	200.000	385.67	37	265.000	789.96
11	135.000	239.14	25	205.000	404.59	38	270.000	840.55
12	140.000	245.69	26	210.000	424.99	39	275.000	895.10
13	145.000	252.86	27	215.000	447.00	40	280.000	953.54
14	150.000	260.62						

ATTACHMENT 3

DATA POINTS FOR HEATUP AND COOLDOWN CURVES

(Without Margins for Instrumentation Errors)

The data points used in the development of the heatup and cooldown curves shown in Figures 5 and 7 are contained on the attached computer printout sheets.

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

IRRADIATION PERIOD = 15.000 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	509.08	18	170.000	641.68	35	255.000	1094.08
2	90.000	513.20	19	175.000	655.79	36	260.000	1141.82
3	95.000	517.62	20	180.000	670.96	37	265.000	1193.06
4	100.000	522.38	21	185.000	687.09	38	270.000	1248.06
5	105.000	527.50	22	190.000	704.62	39	275.000	1307.03
6	110.000	533.00	23	195.000	723.42	40	280.000	1370.49
7	115.000	538.81	24	200.000	743.52	41	285.000	1438.21
8	120.000	545.17	25	205.000	765.28	42	290.000	1511.32
9	125.000	552.01	26	210.000	788.49	43	295.000	1589.48
10	130.000	559.36	27	215.000	813.61	44	300.000	1673.05
11	135.000	567.26	28	220.000	840.46	45	305.000	1763.06
12	140.000	575.76	29	225.000	869.44	46	310.000	1858.89
13	145.000	584.89	30	230.000	900.50	47	315.000	1962.05
14	150.000	594.59	31	235.000	933.82	48	320.000	2072.05
15	155.000	605.15	32	240.000	969.60	49	325.000	2189.57
16	160.000	616.50	33	245.000	1008.27	50	330.000	2315.42
17	165.000	628.70	34	250.000	1049.65	51	335.000	2449.49

AMP COOLDOWN CURVES REG. GUIDE 1.99,REV.2 15 EFPY WITHOUT MARGINS 02/15/93

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

IRRADIATION PERIOD = 15.000 EFP YEARS
FLAW DEPTH = AQWIN 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	466.34	13	145.000	545.27	25	205.000	737.23
2	90.000	470.54	14	150.000	555.62	26	210.000	762.27
3	95.000	475.10	15	155.000	566.78	27	215.000	789.01
4	100.000	479.99	16	160.000	578.79	28	220.000	817.93
5	105.000	485.30	17	165.000	591.61	29	225.000	848.96
6	110.000	490.99	18	170.000	605.52	30	230.000	882.23
7	115.000	497.06	19	175.000	620.52	31	235.000	918.21
8	120.000	503.68	20	180.000	636.49	32	240.000	956.74
9	125.000	510.83	21	185.000	653.87	33	245.000	998.19
10	130.000	518.52	22	190.000	672.53	34	250.000	1042.74
11	135.000	526.82	23	195.000	692.49	35	255.000	1090.58
12	140.000	535.74	24	200.000	714.12			

COOLDOWN CURVES REG. GUIDE 1.99,REV.2 15 EFPY WITHOUT [illegible]NS 02/15/93

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

IRRADIATION PERIOD = 15.000 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	422.65	13	145.000	505.27	25	205.000	710.19
2	90.000	426.97	14	150.000	516.22	26	210.000	736.86
3	95.000	431.68	15	155.000	528.06	27	215.000	765.79
4	100.000	436.74	16	160.000	540.69	28	220.000	796.73
5	105.000	442.17	17	165.000	554.46	29	225.000	830.05
6	110.000	448.10	18	170.000	569.26	30	230.000	866.05
7	115.000	454.53	19	175.000	585.12	31	235.000	904.70
8	120.000	461.45	20	180.000	602.32	32	240.000	946.20
9	125.000	468.95	21	185.000	620.87	33	245.000	990.92
10	130.000	477.03	22	190.000	640.69	34	250.000	1038.91
11	135.000	485.77	23	195.000	662.22	35	255.000	1090.58
12	140.000	495.09	24	200.000	685.22			

AMP COOLDOWN CURVES REG. GUIDE 1.99 REV.2 15 EFPY WITHOUT MARGINS 02/15/93

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

IRRADIATION PERIOD = 15.000 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	377.96	13	145.000	464.78	24	200.000	657.30
2	90.000	382.43	14	150.000	476.41	25	205.000	683.97
3	95.000	387.30	15	155.000	489.00	26	210.000	712.86
4	100.000	392.56	16	160.000	502.47	27	215.000	743.83
5	105.000	398.30	17	165.000	517.14	28	220.000	777.15
6	110.000	404.50	18	170.000	532.95	29	225.000	813.27
7	115.000	411.18	19	175.000	549.93	30	230.000	851.99
8	120.000	418.45	20	180.000	568.32	31	235.000	893.66
9	125.000	426.34	21	185.000	588.07	32	240.000	938.50
10	130.000	434.86	22	190.000	609.47	33	245.000	986.80
11	135.000	444.03	23	195.000	632.54	34	250.000	1038.71
12	140.000	453.98						

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

IRRADIATION PERIOD = 15.000 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	285.32	12	140.000	370.11	23	195.000	575.84
2	90.000	290.14	13	145.000	382.38	24	200.000	604.63
3	95.000	295.46	14	150.000	395.64	25	205.000	635.73
4	100.000	301.22	15	155.000	409.97	26	210.000	669.41
5	105.000	307.55	16	160.000	425.50	27	215.000	705.62
6	110.000	314.40	17	165.000	442.28	28	220.000	744.61
7	115.000	321.81	18	170.000	460.46	29	225.000	786.70
8	120.000	329.89	19	175.000	480.16	30	230.000	832.02
9	125.000	338.85	20	180.000	501.32	31	235.000	880.91
10	130.000	348.43	21	185.000	524.32	32	240.000	933.55
11	135.000	358.88	22	190.000	549.02	33	245.000	990.30

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

IRRADIATION PERIOD = 15.000 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	480.98	19	175.000	544.43	37	265.000	1041.70
2	90.000	471.42	20	180.000	558.06	38	270.000	1092.73
3	95.000	464.74	21	185.000	572.86	39	275.000	1147.47
4	100.000	460.17	22	190.000	588.71	40	280.000	1206.00
5	105.000	457.67	23	195.000	605.98	41	285.000	1269.10
6	110.000	456.69	24	200.000	624.60	42	290.000	1336.62
7	115.000	457.27	25	205.000	644.54	43	295.000	1408.93
8	120.000	459.03	26	210.000	666.15	44	300.000	1486.22
9	125.000	462.02	27	215.000	689.27	45	305.000	1569.44
10	130.000	466.00	28	220.000	714.32	46	310.000	1658.28
11	135.000	471.04	29	225.000	741.08	47	315.000	1753.27
12	140.000	476.96	30	230.000	770.01	48	320.000	1854.90
13	145.000	483.86	31	235.000	801.00	49	325.000	1963.58
14	150.000	491.62	32	240.000	834.24	50	330.000	2079.60
15	155.000	500.24	33	245.000	870.13	51	335.000	2203.37
16	160.000	509.84	34	250.000	908.57	52	340.000	2325.82
17	165.000	520.41	35	255.000	949.85	53	345.000	2448.35
18	170.000	531.95	36	260.000	994.13			

ATTACHMENT 4

DATA POINTS FOR HEATUP AND COOLDOWN CURVES

(With Margins of 10°F and 60 psig for Instrumentation Errors)

The data points used in the development of the heatup and cooldown curves shown in Figures 6 and 8 are contained on the attached computer printout sheets.

AMP COOLDOWN CURVES REG. GUIDE 1.99, REV.2 15 EFPY WITH MARGINS 02/15/93

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

IRRADIATION PERIOD = 15.000 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	441.69	19	175.000	568.70	37	265.000	1034.08
2	90.000	445.25	20	180.000	581.68	38	270.000	1081.82
3	95.000	449.08	21	185.000	595.79	39	275.000	1133.06
4	100.000	453.20	22	190.000	610.96	40	280.000	1188.06
5	105.000	457.62	23	195.000	627.09	41	285.000	1247.03
6	110.000	462.38	24	200.000	644.62	42	290.000	1310.49
7	115.000	467.50	25	205.000	663.42	43	295.000	1378.21
8	120.000	473.00	26	210.000	683.52	44	300.000	1451.32
9	125.000	478.81	27	215.000	705.28	45	305.000	1529.48
10	130.000	485.17	28	220.000	728.49	46	310.000	1613.05
11	135.000	492.01	29	225.000	753.61	47	315.000	1703.06
12	140.000	499.36	30	230.000	780.46	48	320.000	1798.89
13	145.000	507.26	31	235.000	809.44	49	325.000	1902.05
14	150.000	515.76	32	240.000	840.50	50	330.000	2012.05
15	155.000	524.89	33	245.000	873.82	51	335.000	2129.57
16	160.000	534.59	34	250.000	909.60	52	340.000	2255.42
17	165.000	545.15	35	255.000	948.27	53	345.000	2389.49
18	170.000	556.50	36	260.000	989.65			

AMP COOLDOWN CURVES REG. GUIDE 1.99, REV.2 15 EFPY WITH MARGINS 02/15/93

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

IRRADIATION PERIOD = 15.000 EFP YEARS
FLAW DEPTH = ADWIN 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	398.81	14	150.000	475.74	26	210.000	654.12
2	90.000	402.42	15	155.000	485.27	27	215.000	677.23
3	95.000	406.34	16	160.000	495.62	28	220.000	702.27
4	100.000	410.54	17	165.000	506.78	29	225.000	729.01
5	105.000	415.10	18	170.000	518.79	30	230.000	757.93
6	110.000	419.99	19	175.000	531.61	31	235.000	788.96
7	115.000	425.30	20	180.000	545.52	32	240.000	822.23
8	120.000	430.89	21	185.000	560.52	33	245.000	858.21
9	125.000	437.06	22	190.000	576.49	34	250.000	896.74
10	130.000	443.68	23	195.000	593.87	35	255.000	938.19
11	135.000	450.83	24	200.000	612.53	36	260.000	982.74
12	140.000	458.52	25	205.000	632.49	37	265.000	1030.58
13	145.000	466.82						

AMP COOLDOWN CURVES REG. GUIDE 1.99, REV.2 15 EFPY WITH MARGINS 02/15/93

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

IRRADIATION PERIOD = 15.000 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	354.96	14	150.000	435.09	26	210.000	625.22
2	90.000	358.64	15	155.000	445.27	27	215.000	650.19
3	95.000	362.65	16	160.000	456.22	28	220.000	676.86
4	100.000	366.97	17	165.000	468.06	29	225.000	705.79
5	105.000	371.68	18	170.000	480.69	30	230.000	736.73
6	110.000	376.74	19	175.000	494.46	31	235.000	770.05
7	115.000	382.17	20	180.000	509.26	32	240.000	806.05
8	120.000	388.10	21	185.000	525.12	33	245.000	844.70
9	125.000	394.53	22	190.000	542.32	34	250.000	886.20
10	130.000	401.45	23	195.000	560.87	35	255.000	930.92
11	135.000	408.95	24	200.000	580.69	36	260.000	978.91
12	140.000	417.03	25	205.000	602.22	37	265.000	1030.58
13	145.000	425.77						

AMP COOLDOWN CURVES REG. GUIDE 1.99, REV.2 15 EFPY WITH MARGINS 02/15/93

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

IRRADIATION PERIOD = 15.000 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	310.07	13	145.000	384.03	25	205.000	572.54
2	90.000	313.84	14	150.000	393.98	26	210.000	597.30
3	95.000	317.96	15	155.000	404.78	27	215.000	623.97
4	100.000	322.43	16	160.000	416.41	28	220.000	652.86
5	105.000	327.30	17	165.000	429.00	29	225.000	683.83
6	110.000	332.56	18	170.000	442.47	30	230.000	717.15
7	115.000	338.30	19	175.000	457.14	31	235.000	753.27
8	120.000	344.50	20	180.000	472.95	32	240.000	791.99
9	125.000	351.18	21	185.000	489.93	33	245.000	833.66
10	130.000	358.45	22	190.000	508.32	34	250.000	878.50
11	135.000	366.34	23	195.000	528.07	35	255.000	926.80
12	140.000	374.86	24	200.000	549.47	36	260.000	978.71

AMP COOLDOWN CURVES REG. GUIDE 1.99, REV 2 15 EFPY WITH MARGINS 02/15/93

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

IRRADIATION PERIOD = 15.000 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	216.89	13	145.000	298.88	25	205.000	515.84
2	90.000	220.91	14	150.000	310.11	26	210.000	544.63
3	95.000	225.32	15	155.000	322.38	27	215.000	575.73
4	100.000	230.14	16	160.000	335.64	28	220.000	609.41
5	105.000	235.46	17	165.000	349.97	29	225.000	645.62
6	110.000	241.22	18	170.000	365.50	30	230.000	684.61
7	115.000	247.55	19	175.000	382.28	31	235.000	726.70
8	120.000	254.40	20	180.000	400.46	32	240.000	772.02
9	125.000	261.91	21	185.000	420.16	33	245.000	820.91
10	130.000	269.99	22	190.000	441.32	34	250.000	873.55
11	135.000	278.85	23	195.000	464.32	35	255.000	930.30
12	140.000	288.43	24	200.000	489.02			

60 DEG-F/HR HEATUP REG. GUIDE 1.99, REV 2 15 EFPY WIT[illegible]GINS 02/15/93

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

IRRADIATION PERIOD = 15.000 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	441.69	20	180.000	471.95	38	270.000	934.13
2	90.000	433.49	21	185.000	484.43	39	275.000	981.70
3	95.000	420.98	22	190.000	498.06	40	280.000	1032.73
4	100.000	411.42	23	195.000	512.86	41	285.000	1087.47
5	105.000	404.74	24	200.000	528.71	42	290.000	1146.00
6	110.000	400.17	25	205.000	545.98	43	295.000	1209.10
7	115.000	397.67	26	210.000	564.60	44	300.000	1276.62
8	120.000	396.69	27	215.000	584.54	45	305.000	1348.93
9	125.000	397.27	28	220.000	606.15	46	310.000	1426.22
10	130.000	399.03	29	225.000	629.27	47	315.000	1509.44
11	135.000	402.02	30	230.000	654.32	48	320.000	1598.28
12	140.000	406.00	31	235.000	681.08	49	325.000	1683.27
13	145.000	411.04	32	240.000	710.01	50	330.000	1794.90
14	150.000	416.86	33	245.000	741.00	51	335.000	1803.58
15	155.000	423.86	34	250.000	774.24	52	340.000	2019.60
16	160.000	431.62	35	255.000	810.13	53	345.000	2143.37
17	165.000	440.24	36	260.000	848.57	54	350.000	2265.82
18	170.000	449.84	37	265.000	889.85	55	355.000	2388.35
19	175.000	460.41						

Attachment 1 to AEP:NRC:1181

Reasons and 10 CFR 50.92 Significant Hazards
Evaluations for Changes to the Technical Specifications
for Donald C. Cook Nuclear Plant Unit 2

As discussed in the cover letter, the purpose of this proposed amendment is to prevent a surveillance outage before our next refueling outage, currently scheduled to begin August 6, 1994. This submittal requests extensions for surveillances that must be performed during shutdown or that present such operational difficulty that performing the surveillance is not practical at power. We propose to add the following Technical Specification (T/S) to Section 4.0 of the T/Ss.

4.0.8 By specific reference to this section, those surveillances which must be performed on or before August 13, 1994, and are designated as 18-month or 36-month surveillances (or required as outage-related surveillances under the provisions of Specification 4.0.5) may be delayed until the end of the cycle 9-10 refueling outage. For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 2 1994 refueling outage.

We reference this Specification by footnote in all surveillances that require this extension. This footnote will be applicable to the following T/Ss with the indicated surveillance due date. Dates given include the grace period allowed by T/S 4.0.2.

Group	T/S Affected	Description of Change	Due Date
(1)	4.3.1.1.3 4.3.2.1.3	Delay time-response testing for reactor trip and engineered safety features instrumentation	01/02/94 limiting due date
(2)	4.5.1.d 4.5.2.e 4.6.2.1.c 4.6.2.2.c 4.6.3.1.2 4.7.1.2.e 4.7.1.2.f 4.7.3.1.b 4.7.4.1.b 4.7.5.1.e.2 4.7.6.1.d.3	Delay testing for equipment response to ESF signals (safety injection, containment pressure high-high, containment isolation phase A and B and purge exhaust)	04/15/94 limiting due date
(3)	Table 4.3-2†, Item 6.d 4.7.1.2.e 4.7.1.2.f	Delay auxiliary feedwater system testing including channel functional testing of loss of main feedwater pump signal	05/05/94

Group	T/S Affected	Description of Change	Due Date
(4)	4.8.1.1.2.e 4.8.1.2 4.4.11.3 4.7.4.1.b	Delay diesel generator testing including relief valve testing and essential service water valve testing	03/25/94 limiting due date
(5)	Table 4.3-1†, Items 7 & 8 4.3.2.1.2 (P-12) Table 4.3-2†, Item 4.d Table 4.3-6A Items 5, 6, 7 & 8 Table 4.3-10, Items 2, 3, 11	Delay RTD calibrations	04/28/94
(6)	Table 4.3-1†, Items 7, 9, 10 & 11 Table 4.3-2†, Item 1.d 4.3.2.1.2 (P-11) 4.4.11.1.b	Delay pressurizer pressure & level calibrations, interlock function testing, and PORV calibrations	01/29/94
(7)	Table 4.3-10, Item 16	Delay Reactor Vessel Level Indication System Calibration	04/20/94
(8)	4.1.3.3	Delay analog rod position indication functional testing	05/03/94
(9)	4.5.2.d.1 4.5.3.1	Delay RHR auto-closure interlock testing	03/07/94
(10)	4.7.7.1.a	Delay visual inspection of inaccessible snubbers	03/19/94
(11)	Table 4.3-1†, Item 5 4.3.1.1.2 (P-6)	Delay intermediate range calibration and interlock functional testing	01/17/94
(12)	4.6.5.9	Delay divider barrier seal inspection	03/08/94
(13)	4.7.9.2.b.1	Delay RCP fire protection testing	03/30/94
(14)	Table 4.3-10, Item 18 4.5.2.d.2 4.5.3.1	Delay containment water level calibrations and sump visual inspection	01/31/94 limiting due date
(15)	4.2.5.2 Table 4.3-1†, Items 12 & 13	Delay reactor coolant flow calibrations	01/28/94

Group	T/S Affected	Description of Change	Due Date
(16)	Table 4.3-2ǂ, Items 9.a, 9.b, 9.c & 9.d	Delay ESF Manual Trip Actuating Device Operational Test	04/15/94 limiting due date

ǂ Tables 4.3-1 and 4.3-2 refer to T/S 4.3.1.1.1 and T/S 4.3.2.1.1, respectively.

A description of the proposed changes, the reasons for the changes, and our analyses concerning significant hazards considerations for each group of extension requests are given in the remainder of this attachment. It is worth noting that two similar extension requests for the Unit 2 Cycle 6-7 outage were approved by the NRC on December 28, 1987 and February 29, 1988 via Amendments 97 and 99, respectively. These two amendments granted extensions for the T/Ss described in groups 1 through 9 and 16, above.

(1) and (2) Reactor Trip and ESF Response Testing

We are requesting extensions for the time-response testing required by T/Ss 4.3.1.1.3 and 4.3.2.1.3 for the reactor trip and Engineered Safety Features (ESF) instrumentation in T/S Tables 3.3-1 and 3.3-3. In addition, we are requesting extensions for surveillance requirements involving equipment that actuates on an ESF signal (see table below). These surveillances in many cases involve the same equipment and are performed in part to satisfy the response time testing of T/Ss 4.3.1.1.3 and 4.3.2.1.3.

These additional surveillances, the affected components, and the respective ESF actuation signals are as follows:

Item	T/S	Components	ESF Signal
1.	4.5.1.d	accumulator isolation valves	SI
2.	4.5.2.e	ECCS automatic valves centrifugal charging pump safety injection pump residual heat removal pump	SI SI SI SI
3.	4.6.2.1.c	containment spray automatic valves and pumps	containment pressure high-high
4.	4.6.2.2.c	spray additive system automatic valves	containment pressure high-high
5.	4.6.3.1.2	containment isolation valves containment purge and exhaust valves	Phase A isolation Phase B isolation containment purge and exhaust isolation
6.	4.7.1.2.e,f	auxiliary feedwater automatic valves and pump starting	various See T/S Group (3)

Item	T/S	Components	ESF Signal
7.	4.7.3.1.b	component cooling water automatic valves	SI
8.	4.7.4.1.b	essential service water automatic valves	SI See T/S Group (4)
9.	4.7.5.1.e.2	control room ventilation	SI Phase A isolation
10.	4.7.6.1.d.3	ESF ventilation	containment pressure high-high

The extensions are needed from January 2, 1994 (most limiting surveillance due date), until the Unit 2 refueling outage.

At the Cook Plant, response time testing is performed in several parts. The portions of circuitry from the transmitter to the bistable, from the bistable to the master relay contact, and from the master relay contact to equipment operation are tested separately. Testing of the complete portion from the transmitter to the master relay contact cannot be performed at power without violating the T/Ss or adversely impacting plant operation, i.e., reactor trip. T/Ss 3.3.1.1, 3.3.2.1 and 3.0.3 require the plant to be shut down if sufficient reactor trip or ESF instrumentation is not operable. Both trains (all channels) of the function being tested must be taken out of service during this test because the same test signal goes into both trains, which generates a reactor trip signal or ESF actuation. Should they not be in test, each signal would initiate protective functions such as safety injection and containment spray. Therefore, the portion of the time-response tests from the bistable up to the master relay must be done during shutdown. However, testing from the transmitter to the bistable can be performed at power and will be prior to its surveillance due date. The balance of the equipment, i.e., from the master relay contact to equipment operation, is tested as part of the surveillances listed in the table above. Of these surveillances, Items 2 through 8 are specifically required by T/Ss to be performed during shutdown. Items 1, 9 and 10 are not specifically prohibited by T/Ss from being performed at power. However, to do this testing (as well as the other testing listed in the table) would require us to remove an entire train of safety equipment from operation (with the exception of the specific equipment being tested). Because this removes a layer of protection built into the plant, and because it involves operating the plant in an abnormal configuration, it is not considered prudent to perform this testing at power.

The surveillance history of these ESF systems shows that we have no reason to believe that there may be any failures in meeting the T/S requirements due to equipment degradation during the extension period. Additionally, we note that the ESF and reactor protection system channels are subjected to a T/S required surveillance program of channel checks and channel functional tests. All required channel checks and channel functional tests will continue to be performed. We believe these additional tests provide indication of the operability of the systems, and would provide indication of significant degradation.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

Based on our review of past test data, and the fact that the equipment is subject to a surveillance program which includes channel checks and channel functional tests, we believe the extensions we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability of occurrence or consequences of a previously analyzed accident, but the results of which are within limits established as acceptable. For the reasons detailed above, we believe this change falls within the scope of this example. Therefore, we believe this change does not involve significant hazards consideration as defined in 10 CFR 50.92.

(3) Auxiliary Feedwater Pump Testing

T/S Table 4.3-2 Item 6.d requires a channel functional test of the motor driven auxiliary feedwater pump start on loss of main feedwater pump signal to be performed on an 18 month basis. To perform this testing during power operations would involve tripping at least one main feed pump, which would result in a reduction of power and cause a thermal transient to be imposed on the plant. T/Ss 4.7.1.2.e & 4.7.1.2.f require testing to demonstrate that the motor- and turbine-driven auxiliary feedwater pumps start and that the associated automatic valves actuate to their correct position upon receipt of the appropriate signal as listed in T/S Table 4.3-2. Per T/Ss 4.7.1.2.e & 4.7.1.2.f, this testing must be performed during shutdown. These extensions are needed from May 5, 1994, until the Unit 2 refueling outage.

Based on the above, we cannot perform these surveillances while at power. However, in practice, the essential portions of these T/Ss (that is, startup of the auxiliary feedwater pumps when required and movement of the valves to their correct position) occur when the unit trips. The last reactor trip occurred on July 2, 1992. Prior testing experience with regard to these surveillances has indicated no significant problems when the surveillance was performed. Although we recognize that not all the actuation circuitry has been challenged as a result of the reactor trip, we feel that our recent experience, in conjunction with the excellent test history in this area, justifies our request to extend the surveillance interval.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

As discussed above, portions of the system have undergone a challenge due to a recent actuation (during a unit trip). This fact, coupled with our excellent test history for these surveillances, leads us to believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve significant hazards consideration as defined in 10 CFR 50.92.

(4) Diesel Generator Testing

An extension of the surveillance interval is requested for the surveillance requirements of T/S 4.8.1.1.2.e. These surveillances are required by T/Ss to be performed during shutdown. The requirements include subjecting the diesel to an inspection in accordance with manufacturer's recommendations, as well as testing to verify that the diesel generator and its associated circuitry are capable of energizing, sequencing and shedding the emergency loads upon receipt of the appropriate signal. An extension of the surveillance interval is also necessary for part of the requirements of T/S 4.8.1.2, since 4.8.1.1.2 is referenced there. The extension is needed from March 25, 1994 (limiting due date), through the Unit 2 refueling outage.

During the four and a half month period from March 25 until the start of the outage, each diesel generator should accumulate 5 additional starts and 5-7 additional running hours. The affect that these additional starts would have on the diesel generators is believed to be insignificant based on the wear history of each machine. Thus, we believe the additional starts do not constitute sufficient need to perform the subject surveillances prior to the proposed extended date. The history of diesel generator repairs from the past few years do not indicate any problem areas which, in our judgement, would be significantly affected by the proposed surveillance interval extension. Furthermore, conditions which have required maintenance on the diesel generators have been corrected at the time of discovery and have not required deferral until an outage (i.e., we should not be deferring any significant maintenance items through the extension period). Currently, we have a trending program for the parameters measured during our T/S required monthly testing. These trends are reviewed by our diesel generator system engineer. If an adverse trend began to develop, the preventive/corrective measures would be taken to prevent a significant problem from occurring. Also, a review of previous test results did not indicate any reasons to suspect that the diesel generator associated circuitry (i.e., energizing, sequencing, and shedding the various emergency loads) would not pass required surveillance tests with the surveillance interval extended. Based on the above, we believe that there is no reason to suspect that the diesel generators would not be capable of performing their safety functions as required by the T/Ss.

Two other extensions related to the diesel generators are also necessary to avoid a shutdown. These are for the requirements of T/Ss 4.4.11.3 and 4.7.4.1.b. T/S 4.4.11.3 requires testing of the emergency power supply for the power operated relief valves (PORVs) and their associated block valves. Since this testing involves cycling the PORVs and block valves, it is generally performed during shutdown and in conjunction with the diesel generator testing requirements of T/S 4.8.1.1.2.e, as suggested by T/S 4.4.11.3. T/S 4.7.4.1.b involves testing automatic valves in the essential service water (ESW) system. Per T/Ss, this surveillance testing must be performed during shutdown. Since some of the ESW valves which are required to be tested involve cooling water for the diesel generator and its associated equipment, this testing is generally conducted in conjunction with the diesel generator testing of T/S 4.8.1.1.2. The extension for both the ESW valves and the PORV emergency power supply are needed for the period of April 15, 1994 through the Unit 2 refueling outage. Previous test

results do not indicate any reason to suspect that the valves and their associated circuitry would not pass the required surveillance with the extended interval.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

For the diesel-generator machinery, the extension will result only in approximately 5 additional starts and 5 to 7 additional run hours. This is considered insignificant with regard to the wear history of each machine. For the diesel-associated circuitry, the ESW automatic valves, and the PORV emergency power supply, our review of previous test data has not indicated any reason to believe the equipment would not pass the required surveillance tests with the extended interval. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability of occurrence or consequences of a previously analyzed accident, but the results of which are clearly within the limits established as acceptable. We believe these changes fall within the scope of this example. Therefore we believe this change does not involve significant hazards consideration as defined in 10 CFR 50.92.

(5) RTD Calibrations

Extensions are requested for the calibration of resistance temperature detectors (RTDs). The extensions are needed from April 28, 1994, until the Unit 2 refueling outage. The T/S surveillances involving the RTD calibration are listed below.

T/S	Requirement
4.3.1.1.1, Table 4.3-1, Item 7	OTΔT Channel Calibration
4.3.1.1.1, Table 4.3-1, Item 8	OPΔT Channel Calibration
4.3.2.1.2 (P-12)	Total Interlock Function Testing
4.3.2.1.1, Table 4.3-2, Item 4.d	Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low Channel Calibration
4.3.3.5.1 Table 4.3-6A, Items 5 & 7	Calibration of Appendix R Remote Shutdown Monitoring Instrumentation Reactor Coolant Loops (2 & 4) Temperature (Cold)
4.3.3.5.1 Table 4.3-6A, Items 6 & 8	Calibration of Appendix R Remote Shutdown Monitoring Instrumentation Reactor Coolant Loops (2 & 4) Temperature (Hot)
4.3.3.6 Table 4.3-10, Item 2	Calibration of Post-Accident Monitoring Reactor Coolant Outlet Temperature - T_{HOT} Channel
4.3.3.6 Table 4.3-10, Item 3	Calibration of Post-Accident Monitoring Reactor Coolant Inlet Temperature - T_{COLD} Channel
4.3.3.6 Table 4.3-10, Item 11	Calibration of Post-Accident Monitoring Reactor Coolant System Subcooling Margin Monitor Channel

The extensions requested in this category are for the calibration of the sensors only. The calibration procedure requires data to be taken at RCS temperatures ranging from approximately 250°F through operating temperatures. This procedure cannot be performed at power because of the low temperatures necessary for the calibration and because isothermal conditions throughout the RCS are required.

The channels involved with the RTDs are subject to T/S required channel checks and/or channel functional tests. This testing, which will continue during the extension period, would be expected to provide indication of RTD drift. Also, since narrow range RTDs feed the ΔT circuits, comparisons of ΔT_{POWER} to the calorimetric calculated power or power range detectors should show drift in the narrow range RTDs. We have found RTDs at the Cook Nuclear Plant to be very stable, and have not experienced significant drifting problems. For all of these reasons, we have no reason to believe that the RTDs will not remain operable during the extension period.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

The RTDs at the Cook Nuclear Plant have traditionally been very stable. Several independent instruments are available which would allow us to notice drift of the RTDs. Also, channels involving the RTDs are subject to T/S required channel checks and/or channel functional tests, which will continue to be performed during the extension period. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(6) Pressurizer Pressure & Level Calibrations and PORV Calibrations

We are requesting an extension for the performance of some of the pressurizer channel calibrations (pressurizer pressure instruments NPS-153 & NPP-153, and pressurizer level instrument NLP-153) and interlock testing involving the pressurizer pressure instrumentation. We are also requesting relief for the calibration of the PORVs. The extensions are needed from January 29, 1994, until the Unit 2 refueling outage. The affected T/Ss are as follows:

T/S	Requirement
4.3.1.1.1, Table 4.3-1, Item 7	Calibration for OTΔT Reactor Trip.
4.3.1.1.1, Table 4.3-1, Item 9	Calibration for Pressurizer Pressure-Low Reactor Trip
4.3.1.1.1, Table 4.3-1, Item 10	Calibration for Pressurizer Pressure-High Reactor Trip
4.3.1.1.1, Table 4.3-1, Item 11	Calibration for Pressurizer Water Level-High Reactor Trip
4.3.2.1.1, Table 4.3-2, Item 1.d	Calibration for Pressurizer Pressure-Low ESF Actuation
4.3.2.1.2 (P-11)	Interlock Total Function Testing
4.4.11.1.b	Calibration of Power Operated Relief Valves

Performance of this calibration is not considered to be prudent at power due to the configuration of the pressurizer pressure and level instrumentation. Two of the pressurizer pressure instruments (NPS-153 and NPP-153) share a common sensing line with one of the pressurizer level instruments (NLP-153). Calibrating either NPS-153 or NPP-153 poses the risk of perturbing the input to the other transmitter, which could result in a trip. Calibrating NLP-153 poses the risk of perturbing the input to NPS-153 and NPP-153 transmitters, which also could result in a trip. The exemption for the PORVs is also needed because the calibrations make all three PORVs inoperable at the same time, which is contrary to the requirements of T/S 3.4.11.

As discussed in the previous paragraph, certain channels of pressurizer pressure and level instrumentation pose a threat to tripping the reactor. However, there are three channels of pressurizer level instrumentation and four channels of pressurizer pressure instrumentation of which two level and two pressure channels of instrumentation can, and will, be calibrated as required by the Technical Specifications. Thus, two of the three pressurizer level and two of the four

pressurizer pressure channels will satisfy the T/S surveillance requirements. Also, the instrumentation channels for which we are requesting surveillance interval extensions are subject to T/S required channel functional testing and/or channel checks. The channel functional tests we perform are far more stringent than required. These tests not only demonstrate channel functionality, but also verify calibration of trip setpoints, actuations and alarms. The only portion of the channel that is not tested is the sensor, which is qualitatively verified during channel checks. Thus, the testing we will continue to perform would be expected to provide indication of the operability of the systems, and would provide indication of significant degradation. Lastly, we note that based on our review of the surveillance history, we believe this equipment will remain operable during the extension period.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

Completing the required T/S surveillances on two of the three pressurizer level channels and two of the four pressurizer pressure channels will ensure that the majority of the equipment is calibrated as required. Also, the applicable channel functional tests and channel checks should ensure that these systems will perform as designed. Additionally, based on the surveillance history of the equipment, we believe that the equipment will remain operable during the extension period. We therefore believe the extension we are requesting will not result in deterioration to the extent that the equipment cannot perform its intended function. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples. (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes that may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(7) Reactor Vessel Level Indication System

An extension is requested for the channel calibration of the Reactor Vessel Level Indication System (RVLIS) required by T/S Table 4.3-10, Item 16. The required calibration cannot be performed at power because work must be performed in the lower volume of containment and reactor head area, which are only accessible when the unit is shut down. The extension is needed from April 20, 1994, until the Unit 2 refueling outage.

RVLIS has two trains of indication that are subjected to T/S required monthly channel checks which we will continue to perform during the extension period. These channel checks provide indication of the operability of the system, and would be expected to provide indication of significant degradation of the system. Our review of the maintenance history of the system gives us no reason to believe the system would be inoperable during the extension period. Additionally, indication of inadequate core cooling can be obtained by observing core exit thermocouple readings or by checking the subcooling margin monitor. These are the methods the operators would have used to assess inadequate core cooling prior to having RVLIS. We also note that there are annunciators which indicate failure of RVLIS. For these reasons, we believe that the extensions we are requesting will not adversely impact the ability of this equipment to perform its safety function.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

The equipment is subject to normal surveillances which would be expected to provide indication of significant degradation. Also, other instrumentation is available which also provides indication of inadequate core cooling. Lastly, the past maintenance history of the equipment gives us no reason to believe that the equipment would be inoperable during the extension period. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability of occurrence or consequences of a previously analyzed accident, but the results of which are within limits established as acceptable. For the reasons detailed above, we believe this change falls within the scope of this example. Therefore, we believe this change does not involve significant hazards consideration as defined in 10 CFR 50.92.

(8) Rod Position Indication System

This change would delay functional testing of the rod position indicator (RPI) channels required every 18 months by T/S 4.1.3.3. The extension is needed from May 3, 1994, until the Unit 2 refueling outage. Although T/S 4.1.3.3 is only applicable in Modes 3, 4, and 5, we believe relief is needed from this T/S to continue operation in Modes 1 and 2 since T/S 3/4.1.3.2 requires the RPI channels to be operable in these modes.

The surveillance we perform to satisfy T/S 4.1.3.3 is actually a calibration of the RPI channels over the rod insertion range. Since rods must be inserted to perform the calibration, it cannot be performed at power because to do so would violate the rod insertion limits of T/Ss 3.1.3.5 and 3.1.3.6.

The operability of the RPI channels is functionally verified once per 12 hours per T/S 4.1.3.2 by comparison to the demand position indication system. Also, during the 31 day surveillance to satisfy T/S 4.1.3.1.2, the rods are moved at least eight steps and the RPI meters are verified to track with the demand position. These comparisons would be expected to indicate significant degradation in the RPI channels. Surveillances that indicate the core is performing as designed are provided by the incore flux maps, which are taken at least once every 31 effective full power days to satisfy the requirements of T/Ss 4.2.2.2 ($F_Q(Z)$), 4.2.3 ($F^N_{\Delta H}$) and 4.2.1.4 (Axial Flux Difference Target Band). Core performance is also indicated by the excore detectors, which are used to measure the quadrant power tilt ratio per T/S 4.2.4 and axial flux difference per T/S 4.2.1.1.a. These surveillances would be expected to indicate significant discrepancies between indicated and actual rod position. Lastly, since the T/S required surveillances used to verify operability of the RPIs will continue to be performed during the extension period, there is no reason to believe that we would be operating outside the bounds of T/S 3.1.3.2.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

T/S required comparison of the RPI channels to the demand position indication system would be expected to indicate significant degradation in the RPI channels. In addition, other surveillances such as the determination of the quadrant power tilt ratio, axial flux difference and incore flux mapping surveillances, provide a comparison of core performance to design and would be expected to indicate significant deviations of the rods from their indicated position. Since operability of the RPIs will continue to be determined with our T/S required surveillances during the extension period, there is no reason to believe that we would be operating outside the bounds of T/S 3.1.3.2. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(9) RHR Auto-Closure Interlock

We are requesting an extension for the residual heat removal (RHR) auto-closure interlock test required by T/S 4.5.2.d.1. An extension is also requested for T/S 4.5.3.1 since it references T/S 4.5.2. The extensions are needed from March 7, 1994, until the Unit 2 refueling outage. The RHR auto-closure interlock automatically isolates the RHR system from the RCS if RCS pressure is above 600 psig. In order to demonstrate operability of the auto closure interlock, it is necessary to open the RHR isolation valves in the cooldown line from the hot leg in order to verify that the valves would automatically close with the RCS pressure above 600 psig. This cannot be accomplished with the unit operating (i.e., with the RCS fully pressurized) because it would result in exposing the RHR system to pressures higher than the RHR safety valves' setpoints.

Previous surveillance testing has demonstrated that the auto-closure interlock is very reliable. The previous test results give us no reason to believe the auto-closure interlock would be inoperable during the extension period. The calibration for the RCS wide-range pressure transmitters, which provide input into the interlock, can be done at power and will be performed by its September 9, 1993 due date. Thus, the only portion of the interlock for which the surveillances will not be current is the portion from the bistable of the RHR suction valves through valve operation. Additionally, we note that when the unit is operating (i.e., not on RHR), the RHR suction valves are closed and procedures require power to be removed from the valve operators. This precludes inadvertent valve opening and thus alleviates the need for the auto-closure interlock to function.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

The surveillance test history of the auto-closure interlock has shown that the system is highly reliable, and gives us no reason to believe the equipment would be inoperable during the extension period. The wide-range pressure transmitters, which provide input into the auto-closure interlock, will have a current calibration. Additionally, we note that when the RHR system is not in service, power is removed from the suction valve operators, thus preventing inadvertent valve opening and eliminating the need for the auto-closure interlock. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(10) Visual Inspection of Inaccessible Snubbers

This change would delay visual inspections of inaccessible snubbers required by T/S 4.7.7.1.a. The extension is needed from March 19, 1994, through the Unit 2 refueling outage. The extension is required because, by definition in T/S 4.7.7.1.a and Table 3.7-9, these snubbers are inaccessible during reactor operation, thus requiring the inspections to be performed during shutdown. Note that functional testing of snubbers per T/S 4.7.7.1.c is not required until after the scheduled refueling outage start date.

In the past ten years of visual inspections on Unit 2 inaccessible snubbers, we have found only one inoperable snubber. The inoperable snubber was discovered during the steam generator outage in 1988. Since then, four visual inspections have been performed on the inaccessible snubbers, in which none have been found to be inoperable. Based on these inspection results, we are allowed to perform the inspections at the maximum allowed T/S frequency of 18 months (±25%).

It should be noted that we submitted a request in our letter AEP:NRC:1143, dated May 1, 1992 to permanently change the surveillance intervals for snubber visual inspections. The submittal is based on guidance from Generic Letter 90-09, "Alternate Requirements for Snubber Visual Inspections Intervals and Corrective Actions." If we could apply the guidance of Generic Letter 90-09 or our proposed new Specifications on our current visual inspection results of inaccessible snubbers, we would have up to the maximum 48 month interval allowed for our next inspection. This would put our inspection due date beyond the scheduled refueling outage start date, thus eliminating the need for this extension. Based on the history of our inaccessible snubbers and on the guidance of Generic Letter 90-09, we believe the inaccessible snubbers will remain operable during the extension period.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

Our surveillance history of visual inspections on inaccessible snubbers has found only one inoperable snubber in the past ten years. Also, if Generic Letter 90-09 guidance is applied, our surveillance interval would be 48 months. Based on the above, we have no reason to believe the inaccessible snubbers will be inoperable during the extension period. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability of occurrence or consequences of a previously analyzed accident, but where the results are within the limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Therefore, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(11) Intermediate Range Detector Calibrations

This change would delay the calibration of the intermediate range (IR) detectors required by T/S 4.3.1.1.1, Table 4.3-1, Item 5. Also, it would delay interlock functional testing of P-6 required by T/S 4.3.1.1.2. These extensions are needed from January 17, 1994, until the Unit 2 refueling outage.

The need for this extension is because the IR detectors cannot be calibrated while at power. The calibration requires that a test signal covering the range of 10^{-11} to 10^{-3} amps be superimposed over the existing current. Since the current of the IR detectors is in the 10^{-4} amps range during power operation, a superimposed signal less than that could not be observed. Therefore, the IR detectors could not be calibrated below the actual current at which we are operating.

Past operating history in Unit 2 has shown that the IR detectors have performed without serious degradation. There is no reason to believe that the detectors would be inoperable during the extension period. The IR is subjected to a T/S required channel check every 12 hours. The IR currents are trended daily and normalized to 25% power to ensure that the high flux at low power trip set points do not become nonconservative. Through the channel checks and trending program, it is expected that any degradation in an IR detector would be noticed. Lastly, the protection provided by these detectors is required while shut down, or at low power (approximately less than 10%) and not at our normal operating power.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

Our operating history of the Unit 2 IR detectors have shown that they are highly reliable, and give us no reason to believe they would be inoperable during the extension period. Our channel checks and trending program would detect degradation in an IR detector. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(12) Divider Barrier Seal Inspection

T/S 4.6.5.9 requires a visual inspection of at least 95% of the seal's entire length. Also, it requires that two test coupons be removed from the seal for testing to ensure the physical properties are within specified limits. Per this Specification, the inspection is to be performed while shut down. The extensions are needed from March 8, 1994, until the Unit 2 refueling outage.

The divider barrier seal is a passive design feature, thus it is not subjected to any outside forces other than the environment. During the cycle 7-8 refueling outage, we replaced 100% of the divider barrier seal. Our subsequent inspection, during the last outage, revealed no degradation of the seal. Also, when the test coupons were subjected to the tensile strength and elongation tests, they satisfied the acceptable physical property requirements. Based on the facts that the divider barrier seal is passive, new, and has shown no degradation, we believe there is no reason to suspect that it would not be operable during the extension period.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

The divider barrier seal is a passive design feature which was entirely replaced in 1990. Our subsequent inspection revealed no degradation to the seal and the physical properties of the test coupons were acceptable. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(13) Reactor Coolant Pump (RCP) Fire Protection

T/S 4.7.9.2.b requires that the RCPs fire protection system be functionally tested every 18 months. In order to perform the test, the RCP fire detection instrumentation required per T/S Table 3.3-11 and the fire suppression system required by T/S 3.7.9.2 must be made inoperable, which is not considered prudent during operation of the RCPs. It is also noted that, since the RCPs are located in a high radiation area, a firewatch cannot be established per Action Statement B of T/S 3.7.9.2. Therefore, we would be forced to rely on closed circuit television coverage as a substitute for the continuous firewatch. In the event that a camera failed, we would be in non-compliance with the requirements of T/S 3.7.9.2. The extension is need from March 30, 1994, until the Unit 2 refueling outage.

Based on the past RCP sprinkler system surveillance history, there is no reason to believe that it would not be capable of performing it's intended safety function during the extension period. Also, we have seismically qualified oil collection systems on the RCPs, installed in accordance with 10 CFR 50, Appendix R. These systems are designed to mitigate the effects of a RCP lube oil leak.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

Based on the RCP fire protection system surveillance record there is no reason to believe that it would not be capable of performing it's intended safety function. Additionally, it is noted that the RCP oil collection system is designed to mitigate the effects of a RCP lube oil leak. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(14) Containment Water Level Instrumentation and Sump Visual Inspection

T/S 4.3.3.6, Item 18 requires that the containment water level instrumentation be calibrated every 18 months. T/S 4.5.2.d.2 requires that the sump and its inlets be subjected to an 18 month visual inspection. An extension is also needed for T/S 4.5.3.1 since it references T/S 4.5.2. These surveillances cannot be performed during reactor operation since they require entry into the lower volume of containment. The extensions are needed from January 31, 1994 (calibrations) and March 21, 1994 (visual), until the Unit 2 refueling outage.

Our past history on containment water level instrumentation has not shown any significant degradation. This water level instrumentation is used to measure the amount of water on the containment floor above the sump. Normally, there is no water on the floor. The instrumentation consists of RTDs, which have shown stable operation in the past. Since the instruments have a "live" zero point on the scale, a reading of zero or greater indicates that the instruments are performing correctly. In addition, there are two redundant channels that are subjected to T/S required monthly channel checks and the channels can be compared to show if drift exists. There is no reason to believe that the containment water level instrumentation would be inoperable during the extension period.

The visual inspection is performed to ensure that we have a clean system prior to start up. During reactor operation, entry into the containment sump area is restricted. Also, we have very strict material control requirements for entry into containment and at the end of an outage, a "containment closeout tour" is performed to ensure that no material is left within containment. In addition, performance of visual inspections following reactor operation has shown that very little debris ever accumulates in the sump. There is no reason to believe that the sump or it's inlets would become blocked.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

Our past history on containment water level instrumentation has not shown any significant degradation of these instruments. Typically, there is no water on the containment floor for the instruments to measure; however, the instrumentation is calibrated to read a "live" zero level. Also, we have two redundant channels that are subjected to monthly channel checks, which would show if drift exists. Therefore, there is no reason to believe that the containment water level instrumentation would not perform its intended function during the extension period. The likelihood of a significant amount of debris entering the sump is very low because we have very strict requirements for material control inside containment, restricted access into the containment sump area, and an inspection of containment is performed at the end of an outage. There is no reason to believe that the sump or its inlets could become blocked during the extension period. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(15) Reactor Coolant Flow Transmitter Calibrations

T/S 4.2.5.2 and T/S 4.3.1.1.1, Table 4.3-1, Items 12 & 13 require the reactor coolant (RC) flow instrumentation for each loop to be calibrated every 18 months. These calibrations should not be performed at power because of the possibility of a reactor trip. Each set of transmitters (3 per loop) has a common sensing line, which when valving in (or out) one of the transmitters could cause a reduced differential pressure in the other two transmitters. This could cause a reactor trip on low flow in one loop since the two out of three trip logic would be satisfied. The extension for these surveillance requirements are needed from January 28, 1994, until the Unit 2 refueling outage.

Past surveillance history has shown that the RC flow channels are very stable; very little or no drift is found during calibration and they have always been within their allowable range. Also, since there are three channels per loop, drift would be expected to be discovered during the T/S required shiftly channel checks or monthly functional checks. Since the channels have been very stable and we have three channels per loop to indicate drift, there is no reason to believe that continued operation during the extension period would cause the instrumentation to become inoperable.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

Our reactor coolant flow channels have been very stable in the past. We have three channels per loop and perform T/S required channel checks and functional tests which should show any indication of drift. Therefore, there is no reason to believe that the reactor coolant flow channels would not be operable during the extension period. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

(16) Trip Actuating Device Operational Testing (ESF manual actuation)

Extensions are requested for the Trip Actuating Device Operational Testing (ESF manual actuation switches) specified in T/S 4.3.2.1.1, Table 4.3-2, Items 9.a, 9.b, 9.c, and 9.d. These tests cannot be performed at power since they would actuate the ESF functions associated with the switches (see table below). The extensions are needed from April 15, 1994, through the Unit 2 refueling outage.

Table 4.3-2 Item No.	Description
9.a	Safety injection (ECCS) Feedwater Isolation Reactor Trip (SI) Containment Isolation Phase A Containment Purge and Exhaust Isolation Auxiliary Feedwater Pumps Essential Service Water System
9.b	Containment Spray Containment Isolation Phase B Containment Air Recirculation Fan
9.c	Containment Isolation Phase A Containment Purge and Exhaust Isolation
9.d	Steam Line Isolation

The circuitry associated with manual actuation of ESF functions is subjected to T/S required channel functional tests, monthly or bi-monthly. The only portion of the channel not tested is the manual actuation switches. Previous surveillance testing of the switches have shown them to be highly reliable; in fact, there has never been a failure of any of the ESF manual switches detected during surveillance testing of the switches in either unit. Additionally, we note that the manual circuitry serves as a backup to automatic actuation channels, which initiate the same ESF functions. The automatic channels are subjected to T/S required channel checks and channel functional tests to verify operability.

10 CFR 50.92 Criteria

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously analyzed,

(2) create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

Our evaluation of the proposed change with respect to these criteria is provided below.

Criterion 1

The surveillance test history of the ESF manual switches is excellent, indicating no failures of the switches in either unit. The majority of the manual circuitry is subject to a channel functional test on a monthly or bi-monthly basis. The channel functional testing will continue to be performed during the surveillance extension period. Additionally, we note that the manual circuitry serves as a backup to automatic circuitry, which initiates the same ESF functions. For these reasons, we believe the extension we are requesting will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability or consequences of a previously evaluated accident, but the results of which are within limits established as acceptable. We believe this change falls within the scope of this example, for the reasons cited above. Thus, we believe this change does not involve a significant hazards consideration as defined in 10 CFR 50.92.