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FACIL:50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251
AUTH.NAME AUTHOR AFFILIATION
UHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards 150-day response to NRC 810821 & 1218 questions re pressurized thermal shock to reactor pressure vessels. Emergency Operating Procedure 20002 (E-2), "Loss of Secondary Coolant," encl. Reactor integrity demonstrated.
SEE Repts.

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TITLE: Thermal Shock to Reactor Vessel

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	COM LIAW,B		1	1	ELD	12	1	0
	IE	07	2	2	IE WOODS,R		1	1
	MURLEY,T		1	1	NRR CLIFFORD		1	1
	NRR DIR		1	1	NRR GOODWIN,E		1	1
	NRR HAZELTON		1	1	NRR JOHNSON		1	1
	NRR KLECKER		1	1	NRR LOIS,L		1	1
	NRR OREILLY,P		1	1	NRR RANDALL		1	1
	NRR THROM,E		1	1	NRR VISSING,G04		1	1
	NRR/DE DIR		1	1	NRR/DHFS DEPY09		1	1
	NRR/DHFS DIR		1	1	NRR/DHFS/PTRB		1	1
	NRR/DL DIR		1	1	NRR/DL/ADSA		1	1
	NRR/DL/ORAB	11	1	0	NRR/DSI DIR		1	1
	NRR/DSI/RAB		1	1	NRR/DSI/RSB		1	1
	NRR/DST DIR		1	1	NRR/DST/GIB		1	1
	REG FILE	05	1	1	RES BASDEKAS		1	1
	RES VAGINS,M		1	1	RES/DET		1	1
	RES/DRA		1	1				
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	NRC PDR	02	1	1	NSIC	06	1	1
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FLORIDA POWER & LIGHT COMPANY

January 21, 1982
L-82-26

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Eisenhut:

Re: Turkey Point Unit 4
Docket Nos. 50-251
Pressurized Thermal Shock to Reactor Pressure Vessels

Please find attached our "150 day" response to the questions in the NRC letter dated August 21, 1981. The attachments also address the additional NRC questions and concerns which were provided in Mr. Novak's letter to FP&L dated December 18, 1981.

We have reviewed the report prepared by our NSSS vendor to respond to the NRC concerns, and we conclude that these analyses demonstrate that Turkey Point Unit 4 may continue safe operation through the design life of the reactor pressure vessel without modifications to address pressurized thermal shock.

Our responses demonstrate the integrity of the reactor vessel. Based on this we concluded that no further development of an action plan is warranted. However, work is being carried out to evaluate the current low leakage core loading pattern and to assess its effect on reducing the reactor vessel fluence for Turkey Point Unit 4.

We have also included a separate discussion of operator training and management involvement as requested in Mr. Novak's letter. It should be noted that credit for operator action was only assumed for the postulated main steam line break accident analysis. The actions required by our operators to mitigate pressurized thermal shock during a steam line break are included in our emergency procedures. (See attached procedure)

Very truly yours,

Robert E. Uhrig
Vice President
Advanced Systems and Technology

A049
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REU/PLP/mbd

Attachment

cc: Mr. James P. O'Reilly, Region II
Mr. Harold F. Reis, Esquire

Re: Turkey Point Unit 4
Docket No. 50-251
Pressurized Thermal Shock To Reactor Pressure Vessels

I. INTRODUCTION

The primary purpose of this submittal is to address the information requested of Florida Power & Light Company in U.S. Nuclear Regulatory Commission letter titled, "Pressurized Thermal Shock to Reactor Pressure Vessels", dated August 21, 1981 with regards to the Turkey Point Unit No. 4 reactor vessel. Florida Power & Light Company responded to Questions (1), (2) and (5) of this letter via FP&L Co. letter titled, "Pressurized Thermal Shock to Reactor Pressure Vessels", dated October 21, 1981. As a result of reviewing this response, the U.S. NRC identified additional information needed for Turkey Point 4 as outlined in a December 18, 1981 U.S. NRC letter to the Florida Power & Light Company. This submittal also responds to this requested additional information.

WCAP-10019, "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants"[1], provides the fundamental methods and assumptions used in the Turkey Point Unit 4 thermal shock evaluation and, therefore, is a primary reference in assessing the information contained herein. Plant specific details are provided herein to amplify Reference [1] and to address the requested information such that a more complete evaluation is available for the Turkey Point Unit 4 reactor pressure vessel.

II. IRRADIATION INFORMATION

The analytical methodology and the design basis used to predict time averaged fast neutron flux and fluence levels within the pressure vessel/surveillance capsule geometry have been discussed in some detail in WCAP-10019, "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants"[1]. The geometric, material, and power distribution information included in this submittal are fully consistent with the methodology outlined in WCAP-10019 and provide a sound basis for the prediction of the long term fast neutron environment to which the pressure vessel will be exposed.

Also included with this submittal is a summary of the results of the latest design basis neutron transport calculation performed for this vessel as well as an updated evaluation of neutron dosimetry from each of the reactor vessel surveillance capsules which have been withdrawn to date. This dosimetry re-evaluation not only reflects advances in dosimetry analysis methodology and nuclear data, but in addition establishes dosimetry results for all capsules on a consistent basis suitable for direct comparison with analytical predictions.

Geometric information for use in neutron transport calculations is provided in Figures 1 through 3. In Figure 1, a plan view of the reactor at the core midplane is depicted. This figure shows the reactor core, lower internals, pressure vessel, and the inner diameter of the primary biological shield. Pertinent dimensional information is also included on Figure 1. In Figure 2, a detailed description of the surveillance capsule geometry and associated structure is provided. This information is sufficient to allow accurate determinations of capsule lead factors as well as spectrum averaged reaction cross-sections for dosimetry applications. In Figure 3, the azimuthal locations of each of the capsules included in the reactor vessel surveillance program are illustrated.

r_1 = 170.021 cm
r_2 = 175.182 cm
r_3 = 181.134 cm
r_4 = 187.959 cm
r_5 = 197.485 cm
r_6 = 217.488 cm
r_7 = 237.808 cm

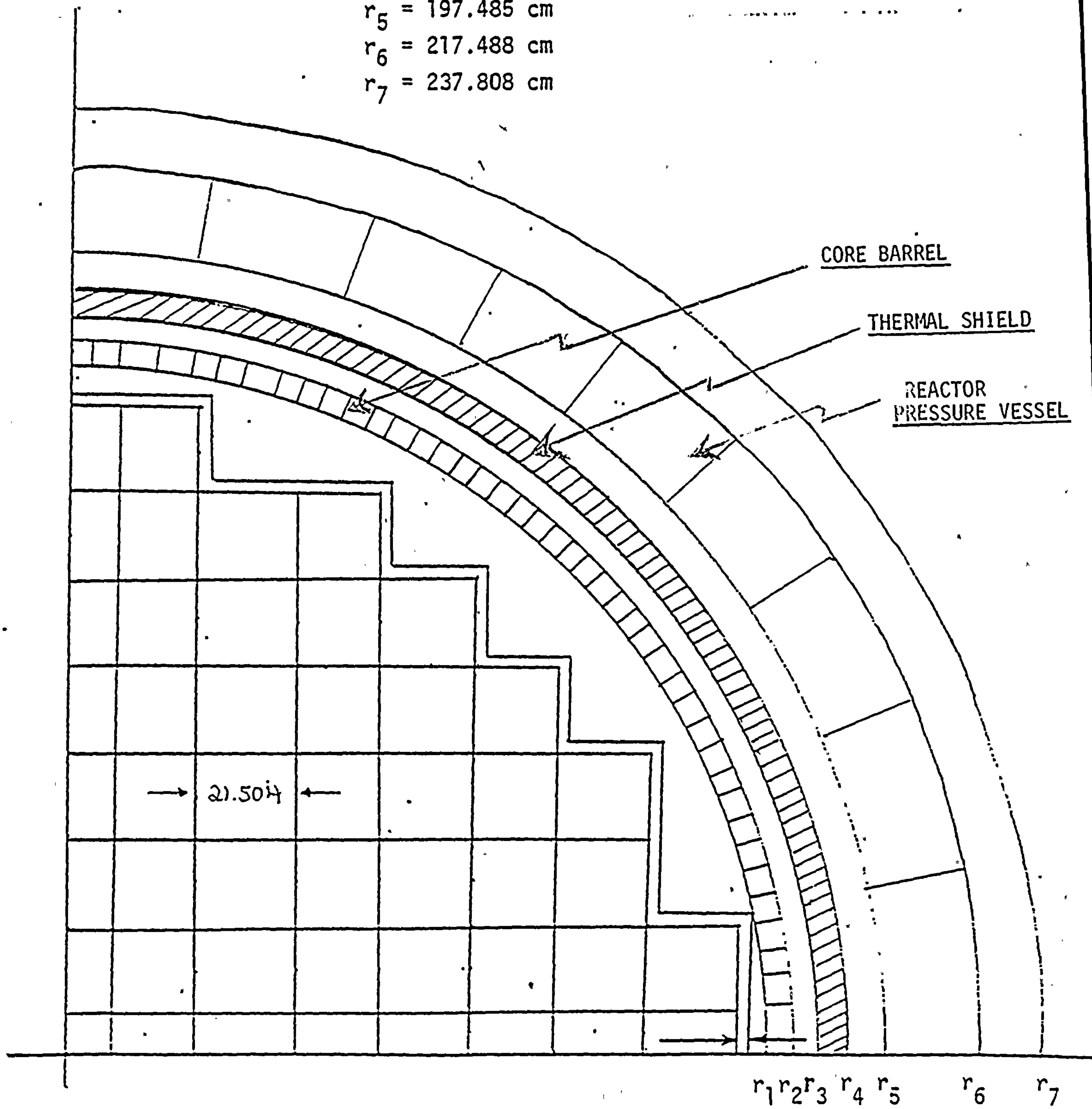


Figure 1.
(r, θ) Reactor Geometry

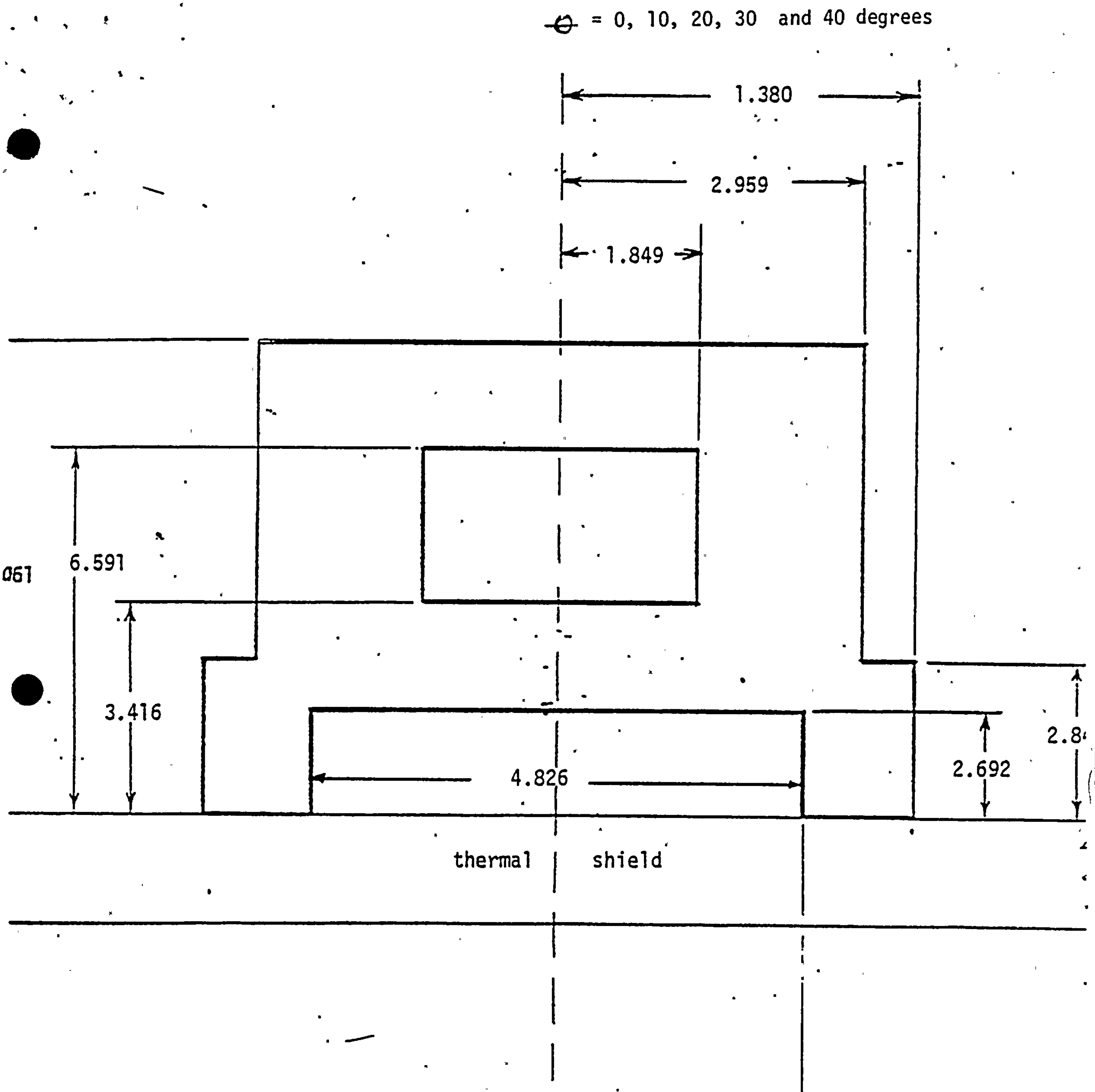


Figure 2

- Surveillance capsule dimensions (cm)

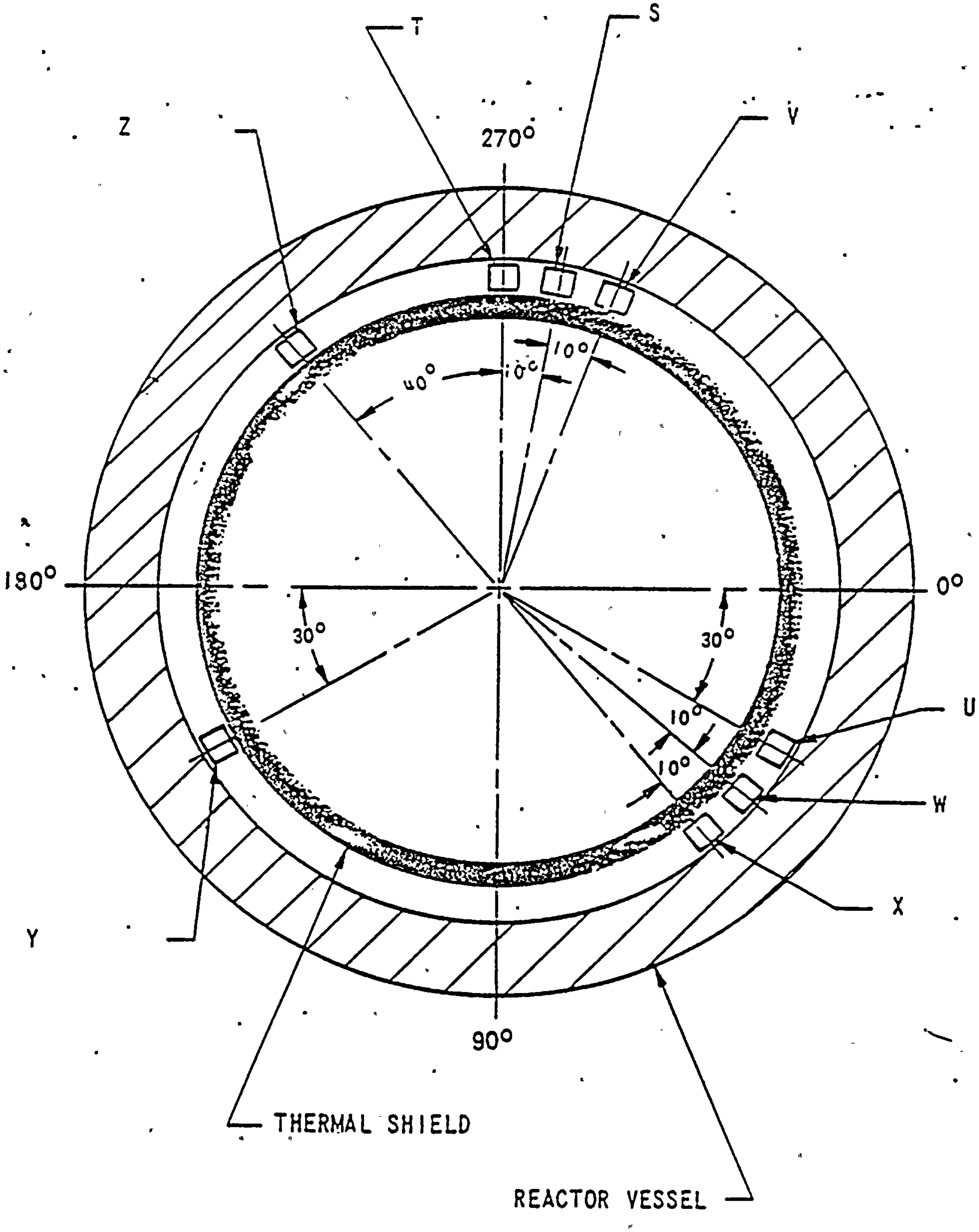


Figure 3
Arrangement of Surveillance Capsules

The material descriptions for each of the major zones shown in Figure 1 are listed in Table 1. The data is presented in terms of volume fractions of solid material in the defined zone of interest. Since neutron transport computations for fluence determinations are of the fixed source variety, fuel enrichment is of no consequence. However, for consistency the UO_2 material listed in Table 1 is taken to contain a nominal 3.2 weight percent U-235.

The core power distributions for use in the computation of time averaged neutron flux and long term neutron fluence levels are given in Figures 4 and 5 and in Table 2. In Figure 4 the relative fuel assembly power levels are given for one core octant. The information is presented relative to a core average of 1.0. Also presented in Figure 4 are a series of fuel assembly numbers which are used to relate spatial power distribution gradients listed in Table 2 with core location. All fuel assemblies labeled Number 1 are assumed to have a flat power distribution; that is, no spatial gradients exist within these assemblies. Spatial gradients for assembly types 2 through 9 are tabulated in Table 2. The data in Table 2 is oriented such that the power value in the upper left hand corner of the table represents the portion of the fuel assembly that is closest to the center of the reactor core. Values for these spatial gradients are uniformly spaced within each fuel assembly. The time averaged axial power distribution for use in neutron transport calculations is shown graphically in Figure 5. As discussed in WCAP-10019, these design basis power distributions are statistically based and have proven to yield satisfactory results for long term fluence predictions.

Results of neutron transport calculations for the geometry shown in Figure 1 are presented in Figures 6 through 8. In Figure 6, the calculated maximum neutron flux levels at the surveillance capsule centerline, pressure vessel inner radius, 1/4 thickness location, and 3/4 thickness location are presented as a function of azimuthal angle. In Figure 7, the radial distribution of maximum fast neutron flux

Table 1

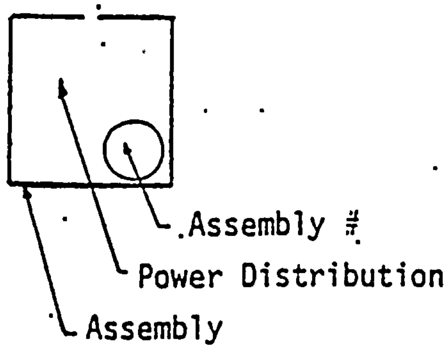
MATERIAL DESCRIPTION FOR USE IN NEUTRON TRANSPORT CALCULATIONS

Zone	Material	Volume Fraction
Reactor Core	Water	0.58864
	UO_2	0.29967
	Zirc - 4	0.10035
	Inconel - 718	0.00281
	Stainless Steel - 304	0.00062
Core Baffle	Stainless Steel - 304	1.0
Core Barrel	Stainless Steel - 304	1.0
Thermal Shield	Stainless Steel - 304	1.0
Surveillance Capsule Structure	Stainless Steel - 304	1.0
Surveillance Specimens	Low Alloy Steel	1.0
Pressure Vessel	Low Alloy Steel	1.0

Figure 4

Long Term Radial Power Distribution

	H	G	F	E	D	C	B	A
8	0.73 (1)							
9	0.83 (1)	0.97 (1)						
10	1.05 (1)	0.95 (1)	1.07 (1)					
11	0.97 (1)	1.16 (1)	1.00 (1)	1.11 (1)				
12	1.11 (1)	1.11 (1)	1.13 (1)	1.03 (1)	0.92 (9)			
13	1.11 (1)	0.97 (1)	1.02 (1)	1.04 (7)	0.85 (8)			
14	0.95 (1)	1.07 (4)	1.12 (5)	0.80 (6)				
15	0.93 (2)	0.77 (3)						



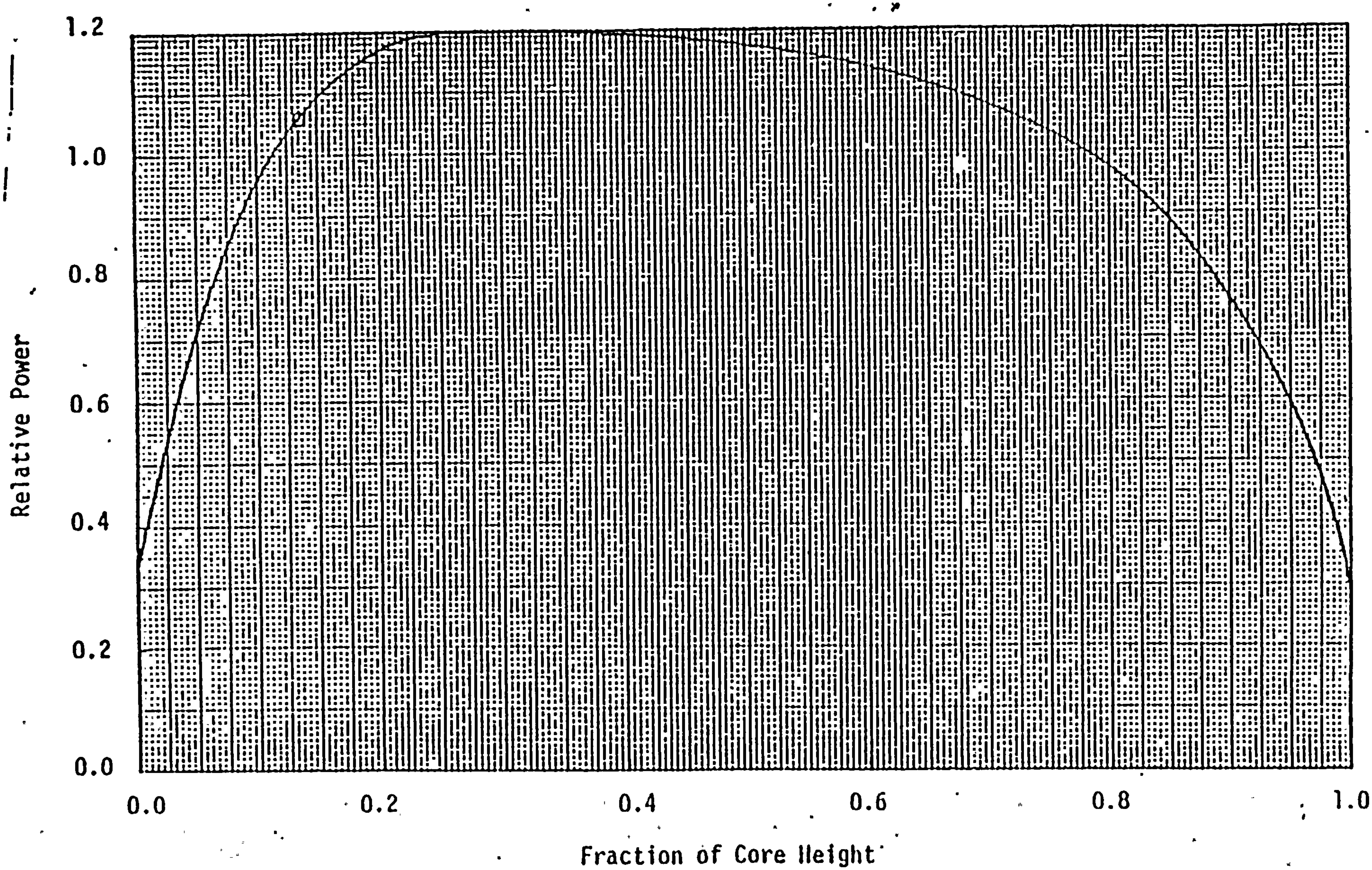


Figure 5
Time Averaged Axial Power Distribution

TABLE 2. ROD BY ROD POWER DISTRIBUTION

ASSEMBLY 2

0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.21	1.21	1.21	1.21	1.20	1.19	1.18	1.17	1.16
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.23	1.21	1.21	1.24	1.20	1.18	1.16	1.14	1.15
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.25	1.25	0.00	1.25	1.23	1.18	1.15	1.15
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.23	1.21	1.21	1.25	1.25	0.00	1.22	1.15	1.13
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.22	1.19	1.20	1.24	1.22	1.24	1.22	1.15	1.13
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.21	1.20	0.00	1.21	1.22	0.00	1.16	1.11
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.17	1.14	1.14	1.18	1.14	1.14	1.16	1.11	1.08
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.13	1.11	1.11	1.14	1.10	1.10	1.12	1.07	1.05
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.10	1.10	0.00	1.09	1.09	0.00	1.05	1.01
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.04	1.02	1.02	1.05	1.02	1.01	1.03	.98	.96
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.99	.97	.97	1.00	.97	.96	.98	.93	.91
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.96	.96	0.00	.96	.96	0.00	.92	.88
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.89	.87	.87	.90	.89	.90	.89	.83	.81
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.83	.82	.82	.85	.85	0.00	.82	.77	.74
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.77	.75	0.00	.77	.74	.70	.68	.67
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.67	.65	.65	.67	.65	.63	.62	.61	.60
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.55	.55	.55	.55	.54	.53	.52	.52	.52

ASSEMBLY 3

1.40	1.39	1.39	1.40	1.40	1.41	1.39	1.38	1.36	1.34	1.31	1.28	1.24	1.18	1.13	1.06	.96
1.37	1.36	1.37	1.38	1.40	1.43	1.38	1.36	1.37	1.32	1.29	1.29	1.21	1.15	1.08	.99	.88
1.35	1.35	1.38	1.44	1.45	0.00	1.41	1.39	0.00	1.34	1.31	0.00	1.24	1.18	1.07	.96	.85
1.34	1.35	1.41	0.00	1.44	1.42	1.34	1.32	1.33	1.27	1.24	1.25	1.22	0.00	1.07	.94	.82
1.34	1.35	1.42	1.43	1.39	1.39	1.32	1.30	1.31	1.24	1.21	1.22	1.15	1.13	1.05	.92	.81
1.32	1.36	0.00	1.38	1.37	0.00	1.32	1.29	0.00	1.23	1.20	0.00	1.12	1.07	0.00	.90	.78
1.28	1.29	1.34	1.30	1.28	1.30	1.24	1.22	1.22	1.16	1.12	1.12	1.04	.99	.95	.84	.75
1.23	1.24	1.28	1.24	1.23	1.25	1.19	1.16	1.17	1.11	1.07	1.06	.99	.93	.90	.80	.71
1.19	1.22	0.00	1.22	1.21	0.00	1.17	1.14	0.00	1.08	1.05	0.00	.96	.91	0.00	.77	.67
1.13	1.14	1.18	1.13	1.12	1.14	1.08	1.06	1.06	1.00	.95	.95	.88	.83	.79	.70	.62
1.08	1.08	1.12	1.08	1.06	1.07	1.02	1.00	1.00	.94	.90	.89	.83	.78	.74	.65	.58
1.03	1.06	0.00	1.07	1.05	0.00	1.00	.97	0.00	.92	.88	0.00	.81	.76	0.00	.63	.55
.95	.96	1.00	1.00	.96	.96	.91	.88	.88	.83	.80	.79	.74	.71	.65	.57	.4[illegible]
.88	.88	.92	0.00	.92	.90	.84	.82	.81	.77	.73	.72	.69	0.00	.59	.51	.4[illegible]
.80	.79	.80	.83	.82	0.00	.78	.75	0.00	.70	.67	0.00	.62	.57	.51	.45	.40
.70	.69	.69	.69	.69	.70	.66	.65	.64	.60	.58	.57	.52	.48	.43	.39	.36
.60	.59	.59	.58	.58	.57	.55	.54	.52	.49	.47	.45	.42	.39	.36	.33	.3[illegible]

TABLE 2. ROD BY ROD POWER DISTRIBUTION (Cont)

SSEMBLY 4

.98	.97	.98	.99	1.00	1.01	1.01	1.00	1.01	1.00	1.00	1.00	.98	.97	.96	.95	.95
.97	.96	.98	1.00	1.01	1.04	1.02	1.02	1.04	1.01	1.01	1.03	1.00	.98	.96	.94	.94
.98	.99	1.01	1.06	1.08	0.00	1.07	1.07	0.00	1.06	1.05	0.00	1.06	1.04	.99	.96	.95
.99	1.00	1.05	0.00	1.09	1.09	1.05	1.04	1.07	1.04	1.04	1.08	1.08	0.00	1.03	.97	.96
1.01	1.02	1.08	1.10	1.08	1.09	1.06	1.05	1.08	1.05	1.05	1.08	1.06	1.08	1.06	1.00	.97
1.01	1.05	0.00	1.09	1.09	0.00	1.08	1.08	0.00	1.08	1.07	0.00	1.07	1.07	0.00	1.02	.97
1.02	1.03	1.07	1.05	1.06	1.09	1.06	1.06	1.08	1.05	1.05	1.07	1.04	1.03	1.04	.99	.97
1.01	1.03	1.07	1.05	1.05	1.08	1.05	1.05	1.08	1.04	1.04	1.06	1.03	1.02	1.03	.98	.96
1.01	1.04	0.00	1.07	1.07	0.00	1.07	1.07	0.00	1.07	1.06	0.00	1.04	1.03	0.00	.99	.95
.99	1.01	1.05	1.03	1.04	1.07	1.04	1.03	1.06	1.02	1.02	1.04	1.00	.99	1.00	.95	.93
.99	1.01	1.05	1.03	1.03	1.06	1.03	1.02	1.05	1.01	1.01	1.03	.99	.98	.99	.94	.91
1.00	1.03	0.00	1.07	1.07	0.00	1.06	1.05	0.00	1.04	1.03	0.00	1.02	1.01	0.00	.95	.90
.98	.99	1.05	1.06	1.04	1.05	1.01	1.01	1.03	.99	.99	1.01	.98	.99	.96	.90	.86
.97	.98	1.03	0.00	1.06	1.05	1.01	1.00	1.02	.98	.98	1.00	.99	0.00	.93	.87	.84
.95	.95	.97	1.02	1.03	0.00	1.01	1.00	0.00	.99	.98	0.00	.95	.92	.86	.82	.79
.94	.94	.94	.96	.97	1.00	.97	.96	.97	.94	.93	.93	.89	.85	.81	.77	.75
.94	.93	.93	.93	.94	.94	.93	.92	.92	.90	.89	.87	.84	.81	.77	.74	.70

SSEMBLY 5

1.13	1.12	1.12	1.12	1.13	1.13	1.13	1.12	1.12	1.11	1.11	1.11	1.10	1.08	1.06	1.05	1.06
1.12	1.11	1.11	1.13	1.13	1.16	1.13	1.13	1.14	1.12	1.11	1.13	1.10	1.08	1.05	1.03	1.03
1.13	1.13	1.15	1.20	1.21	0.00	1.19	1.18	0.00	1.17	1.16	0.00	1.16	1.13	1.07	1.04	1.03
1.14	1.14	1.20	0.00	1.22	1.21	1.15	1.14	1.17	1.13	1.12	1.16	1.16	0.00	1.11	1.04	1.02
1.15	1.16	1.22	1.23	1.20	1.20	1.15	1.14	1.17	1.13	1.12	1.15	1.13	1.14	1.12	1.05	1.02
1.15	1.19	0.00	1.21	1.26	0.00	1.17	1.16	0.00	1.14	1.14	0.00	1.13	1.12	0.00	1.06	1.01
1.15	1.16	1.19	1.16	1.15	1.17	1.13	1.12	1.14	1.10	1.09	1.11	1.07	1.06	1.07	1.02	.99
1.13	1.14	1.17	1.14	1.13	1.16	1.11	1.09	1.12	1.07	1.06	1.08	1.04	1.03	1.04	.99	.96
1.12	1.15	0.00	1.15	1.14	0.00	1.11	1.10	0.00	1.08	1.06	0.00	1.04	1.03	0.00	.98	.94
1.09	1.10	1.13	1.09	1.08	1.09	1.05	1.04	1.06	1.01	1.00	1.02	.98	.96	.97	.92	.89
1.07	1.07	1.10	1.06	1.05	1.06	1.02	1.00	1.02	.97	.96	.98	.94	.92	.93	.88	.85
1.05	1.07	0.00	1.07	1.06	0.00	1.02	1.00	0.00	.97	.96	0.00	.94	.93	0.00	.87	.83
1.01	1.01	1.04	1.04	1.00	1.00	.94	.93	.94	.90	.88	.90	.88	.88	.86	.80	.77
.97	.96	.99	0.00	.98	.95	.90	.88	.89	.85	.84	.86	.84	0.00	.80	.73	.71
.91	.89	.89	.91	.91	0.00	.86	.84	0.00	.81	.80	0.00	.78	.74	.70	.66	.65
.85	.81	.82	.79	.79	.80	.76	.75	.75	.72	.70	.71	.67	.65	.62	.60	.59
.76	.72	.70	.69	.69	.69	.67	.65	.64	.63	.61	.60	.58	.57	.55	.53	.52

TABLE 2. ROD BY ROD POWER DISTRIBUTION (Con't)

ASSEMBLY 6

1.41	1.39	1.39	1.39	1.40	1.40	1.38	1.36	1.36	1.33	1.33	1.27	1.23	1.17	1.10	1.02	.9
1.38	1.37	1.37	1.39	1.40	1.43	1.37	1.37	1.38	1.31	1.28	1.28	1.20	1.13	1.05	.96	.8
1.36	1.36	1.39	1.44	1.45	0.00	1.42	1.39	0.00	1.33	1.30	0.00	1.23	1.15	1.04	.93	.8
1.35	1.37	1.43	0.00	1.44	1.43	1.35	1.32	1.33	1.27	1.24	1.24	1.20	0.00	1.04	.91	.7
1.35	1.36	1.42	1.42	1.39	1.39	1.32	1.29	1.30	1.23	1.20	1.20	1.13	1.09	1.01	.88	.7
1.33	1.37	0.00	1.40	1.38	0.00	1.32	1.29	0.00	1.23	1.19	0.00	1.10	1.04	0.00	.87	.7
1.29	1.29	1.35	1.30	1.28	1.30	1.24	1.21	1.22	1.14	1.10	1.09	1.01	.96	.91	.81	.7
1.25	1.27	1.30	1.25	1.24	1.26	1.19	1.16	1.17	1.10	1.06	1.04	.96	.91	.87	.77	.6
1.23	1.26	0.00	1.24	1.23	0.00	1.18	1.15	0.00	1.08	1.04	0.00	.94	.89	0.00	.75	.6
1.18	1.17	1.20	1.16	1.14	1.15	1.09	1.06	1.06	.99	.95	.94	.87	.82	.78	.69	.6
1.12	1.12	1.16	1.10	1.09	1.10	1.04	1.01	1.01	.94	.90	.89	.82	.77	.73	.64	.5
1.07	1.10	0.00	1.09	1.07	0.00	1.01	.97	0.00	.91	.88	0.00	.80	.75	0.00	.61	.5
1.00	1.00	1.04	1.03	.99	.98	.92	.88	.88	.83	.80	.79	.73	.70	.64	.55	.4
.92	.92	.95	0.00	.93	.91	.84	.81	.82	.76	.73	.72	.69	0.00	.58	.58	.4
.84	.82	.83	.85	.84	0.00	.79	.77	0.00	.71	.68	0.00	.62	.57	.50	.44	.3
.75	.73	.73	.72	.72	.72	.69	.66	.65	.61	.58	.57	.52	.48	.43	.39	.3
.66	.64	.63	.62	.61	.65	.58	.56	.54	.52	.49	.47	.44	.41	.37	.34	.3

ASSEMBLY 7

1.01	1.00	1.00	1.01	1.03	1.04	1.03	1.03	1.03	1.02	1.01	1.01	1.01	.99	.97	.96	.9
1.00	.99	1.00	1.03	1.04	1.07	1.04	1.04	1.06	1.03	1.03	1.06	1.02	1.00	.97	.95	.9
1.00	1.00	1.03	1.08	1.10	0.00	1.09	1.08	0.00	1.07	1.08	0.00	1.07	1.05	.99	.96	.9
1.01	1.02	1.00	0.00	1.11	1.11	1.06	1.05	1.08	1.06	1.05	1.09	1.08	0.00	1.03	.97	.9
1.02	1.04	1.09	1.11	1.09	1.10	1.06	1.06	1.09	1.06	1.05	1.08	1.06	1.07	1.04	.98	.9
1.03	1.07	0.00	1.10	1.10	0.00	1.10	1.09	0.00	1.08	1.08	0.00	1.07	1.06	0.00	1.00	.9
1.02	1.04	1.08	1.06	1.06	1.09	1.06	1.06	1.09	1.05	1.04	1.06	1.02	1.01	1.02	.98	.9
1.02	1.03	1.07	1.05	1.05	1.09	1.06	1.05	1.08	1.04	1.03	1.05	1.01	1.00	1.02	.96	.9
1.02	1.05	0.00	1.07	1.08	0.00	1.08	1.08	0.00	1.06	1.05	0.00	1.03	1.02	0.00	.97	.9
1.01	1.02	1.06	1.05	1.05	1.08	1.04	1.04	1.06	1.02	1.01	1.03	1.00	.98	.99	.93	.9
1.00	1.02	1.07	1.05	1.05	1.07	1.04	1.03	1.05	1.01	1.00	1.03	.98	.97	.97	.92	.8
1.00	1.04	0.00	1.08	1.07	0.00	1.06	1.05	0.00	1.03	1.03	0.00	1.00	.99	0.00	.93	.8
.99	1.01	1.06	1.07	1.05	1.06	1.02	1.00	1.02	.99	.99	1.00	.97	.98	.95	.86	.8
.97	.98	1.03	0.00	1.06	1.05	1.00	.99	1.01	.97	.96	.98	.98	0.00	.91	.84	.8
.96	.95	.98	1.02	1.03	0.00	1.01	1.00	0.00	.98	.96	0.00	.94	.91	.84	.79	.7
.94	.94	.94	.95	.96	.98	.96	.95	.96	.92	.92	.92	.87	.83	.79	.75	.7
.94	.93	.93	.93	.93	.94	.93	.92	.91	.90	.88	.86	.83	.80	.76	.71	.6

TABLE 2. ROD BY ROD POWER DISTRIBUTION (Con't)

ASSEMBLY 8

1.36	1.34	1.32	1.32	1.33	1.32	1.29	1.26	1.26	1.22	1.18	1.15	1.10	1.04	.98	.90	.81
1.34	1.32	1.32	1.33	1.33	1.35	1.29	1.28	1.28	1.22	1.18	1.17	1.09	1.02	.94	.85	.74
1.34	1.32	1.35	1.39	1.39	0.00	1.35	1.31	0.00	1.24	1.21	0.00	1.12	1.04	.93	.82	.71
1.34	1.35	1.40	0.00	1.40	1.41	1.30	1.26	1.26	1.19	1.15	1.15	1.10	0.00	.94	.80	.6[illegible]
1.35	1.36	1.41	1.40	1.37	1.36	1.28	1.24	1.25	1.17	1.13	1.12	1.05	1.00	.92	.78	.66
1.35	1.39	0.00	1.40	1.37	0.00	1.29	1.26	0.00	1.17	1.13	0.00	1.03	.96	0.00	.78	.65
1.33	1.33	1.38	1.32	1.29	1.30	1.23	1.19	1.19	1.11	1.05	1.04	.96	.89	.84	.73	.62
1.31	1.32	1.35	1.29	1.26	1.27	1.20	1.16	1.15	1.07	1.02	1.00	.92	.85	.81	.70	.59
1.31	1.33	0.00	1.29	1.27	0.00	1.20	1.15	0.00	1.07	1.02	0.00	.91	.85	0.00	.68	.56
1.27	1.26	1.29	1.23	1.20	1.20	1.13	1.08	1.07	1.00	.95	.92	.85	.79	.74	.63	.53
1.24	1.23	1.26	1.20	1.16	1.16	1.08	1.04	1.03	.95	.90	.89	.81	.75	.70	.60	.50
1.21	1.23	0.00	1.20	1.16	0.00	1.07	1.03	0.00	.94	.90	0.00	.80	.74	0.00	.58	.4[illegible]
1.17	1.15	1.18	1.15	1.09	1.07	.99	.95	.93	.87	.82	.73	.74	.69	.63	.53	.44
1.11	1.09	1.10	0.00	1.06	1.01	.93	.89	.88	.81	.77	.75	.70	0.00	.58	.48	.41
1.05	1.01	.99	1.00	.98	0.00	.89	.85	0.00	.77	.73	0.00	.64	.58	.51	.44	.37
.97	.92	.89	.87	.85	.84	.79	.75	.73	.67	.63	.61	.55	.50	.44	.39	.34
.87	.81	.77	.75	.73	.73	.69	.66	.63	.59	.55	.58	.48	.43	.39	.35	.31

ASSEMBLY 9

1.06	1.04	1.04	1.04	1.06	1.07	1.06	1.05	1.06	1.05	1.05	1.04	1.04	1.02	1.01	1.00	1.0[illegible]
1.04	1.03	1.03	1.05	1.07	1.09	1.06	1.06	1.08	1.05	1.05	1.07	1.04	1.02	.99	.98	.97
1.04	1.03	1.06	1.10	1.11	0.00	1.11	1.09	0.00	1.08	1.09	0.00	1.08	1.06	1.00	.97	.9[illegible]
1.04	1.05	1.10	0.00	1.12	1.12	1.08	1.06	1.09	1.06	1.06	1.09	1.08	0.00	1.03	.98	.9[illegible]
1.06	1.07	1.11	1.12	1.11	1.11	1.07	1.06	1.09	1.05	1.05	1.07	1.05	1.06	1.04	.98	.95
1.07	1.09	0.00	1.12	1.11	0.00	1.09	1.09	0.00	1.07	1.06	0.00	1.05	1.04	0.00	.99	.95
1.06	1.06	1.11	1.08	1.07	1.09	1.06	1.05	1.07	1.03	1.02	1.04	1.00	.99	1.00	.96	.93
1.05	1.06	1.09	1.06	1.06	1.09	1.05	1.04	1.06	1.02	1.01	1.02	.99	.97	.99	.94	.91
1.06	1.08	0.00	1.09	1.09	0.00	1.07	1.06	0.00	1.04	1.02	0.00	1.00	.99	0.00	.94	.9[illegible]
1.05	1.05	1.08	1.06	1.05	1.07	1.03	1.02	1.04	.99	.98	.99	.96	.94	.95	.90	.8[illegible]
1.05	1.05	1.09	1.06	1.05	1.06	1.02	1.01	1.02	.98	.97	.99	.95	.93	.93	.89	.8[illegible]
1.04	1.07	0.00	1.09	1.07	0.00	1.04	1.02	0.00	.99	.99	0.00	.96	.94	0.00	.88	.8[illegible]
1.04	1.04	1.08	1.08	1.05	1.05	1.00	.99	1.00	.96	.95	.96	.92	.93	.90	.84	.8[illegible]
1.02	1.02	1.06	0.00	1.06	1.04	.99	.97	.99	.94	.93	.94	.93	0.00	.86	.79	.7[illegible]
1.01	.99	1.00	1.03	1.04	0.00	1.00	.99	0.00	.95	.93	0.00	.90	.86	.80	.75	.7[illegible]
1.00	.98	.97	.98	.98	.99	.96	.94	.94	.90	.89	.88	.84	.79	.75	.71	.6[illegible]
1.00	.97	.96	.96	.95	.95	.93	.91	.90	.88	.86	.84	.80	.76	.72	.68	.6[illegible]

(E > 1.0 Mev) through the thickness of the pressure vessel is shown. The relative axial variation of neutron flux within the vessel is given in Figure 8.

The data given in Figures 6 through 8 can be used directly to develop lead factors relating each surveillance capsule to any point in the pressure vessel; or, in conjunction with appropriate full power operating times, to derive fast neutron fluence distributions within the vessel.

Since initial startup, two surveillance capsules have been withdrawn from the Turkey Point Unit 4 reactor. In 1976, Capsule T was removed from the 0° azimuthal position while in 1979 Capsule S was withdrawn from the 10° location. Neutron dosimetry from both Capsules T and S was evaluated by Southwest Research Institute and the results were documented in SWRI-02-4221 and SWRI-02-5380.

For the purposes of this submittal, the SWRI radiometric counting data have been extracted from the appropriate reports and the fluence determinations have been updated to reflect the following changes in surveillance dosimetry methodology.

1. Application of the best available nuclear data

2. Use of spectrum averaged cross-sections which include capsule perturbation effects

3. Spatial gradient corrections to measured count rates to permit neutron flux evaluating at the geometric center of the capsule.

(1) Surveillance capsules at radius = 190.900 cm

(2) Surveillance capsules at radius = 192.488 cm

(3) Reactor vessel wall inner surface at radius = 197.635 cm

(4) Reactor vessel wall, 1/4 thickness, at radious = 202.485 cm

(5) Reactor vessel wall, 3/4 thickness, at radius = 212.487 cm

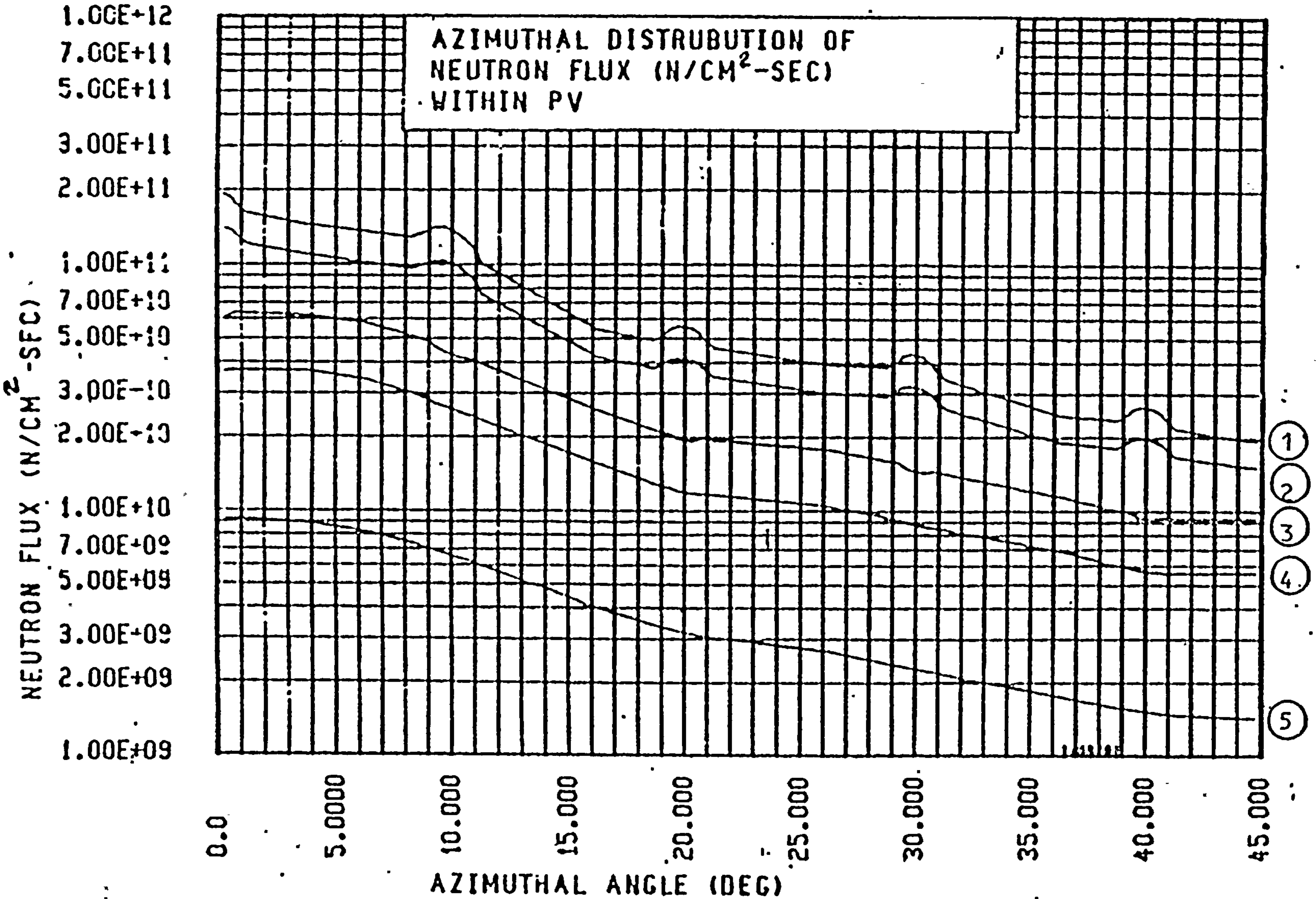
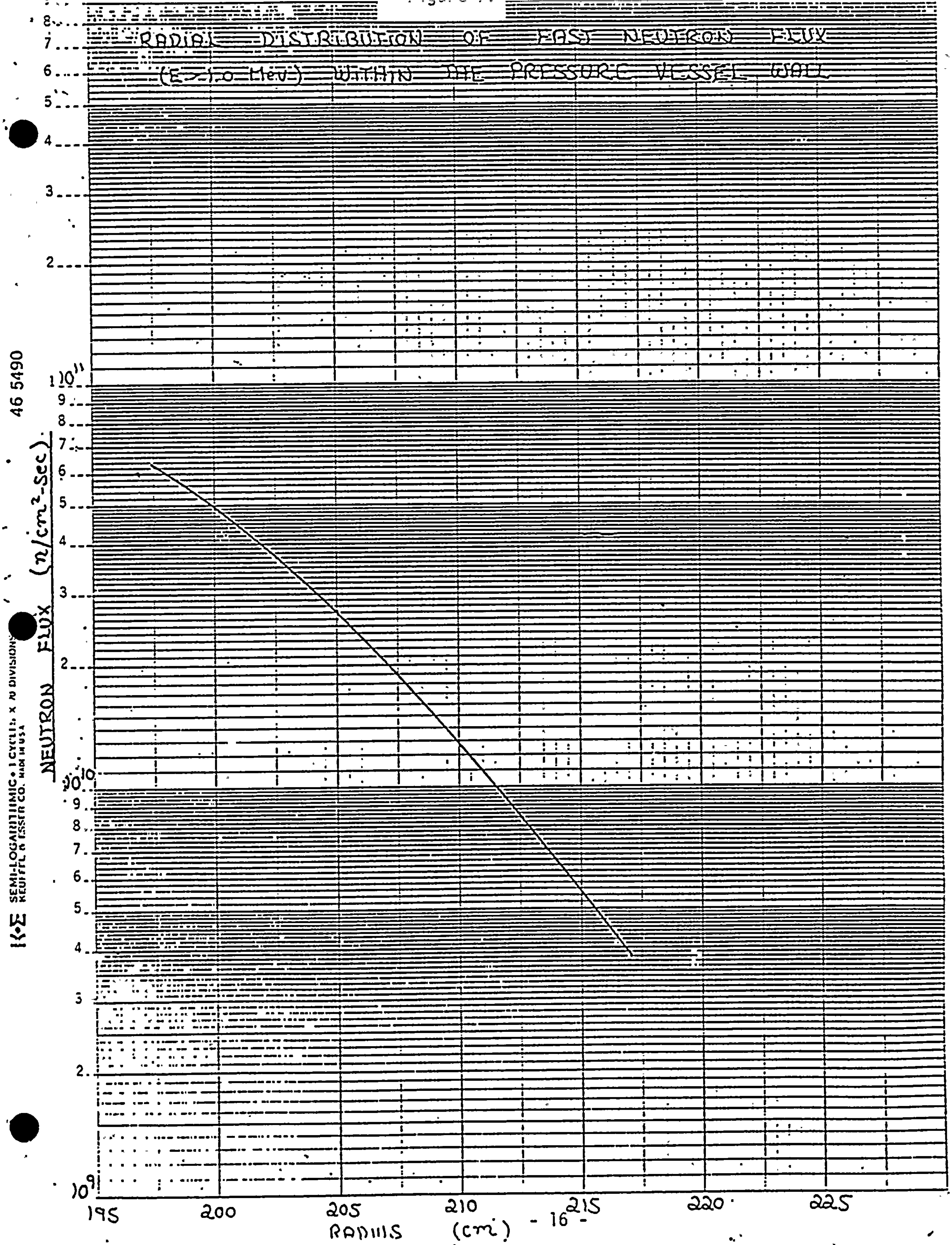
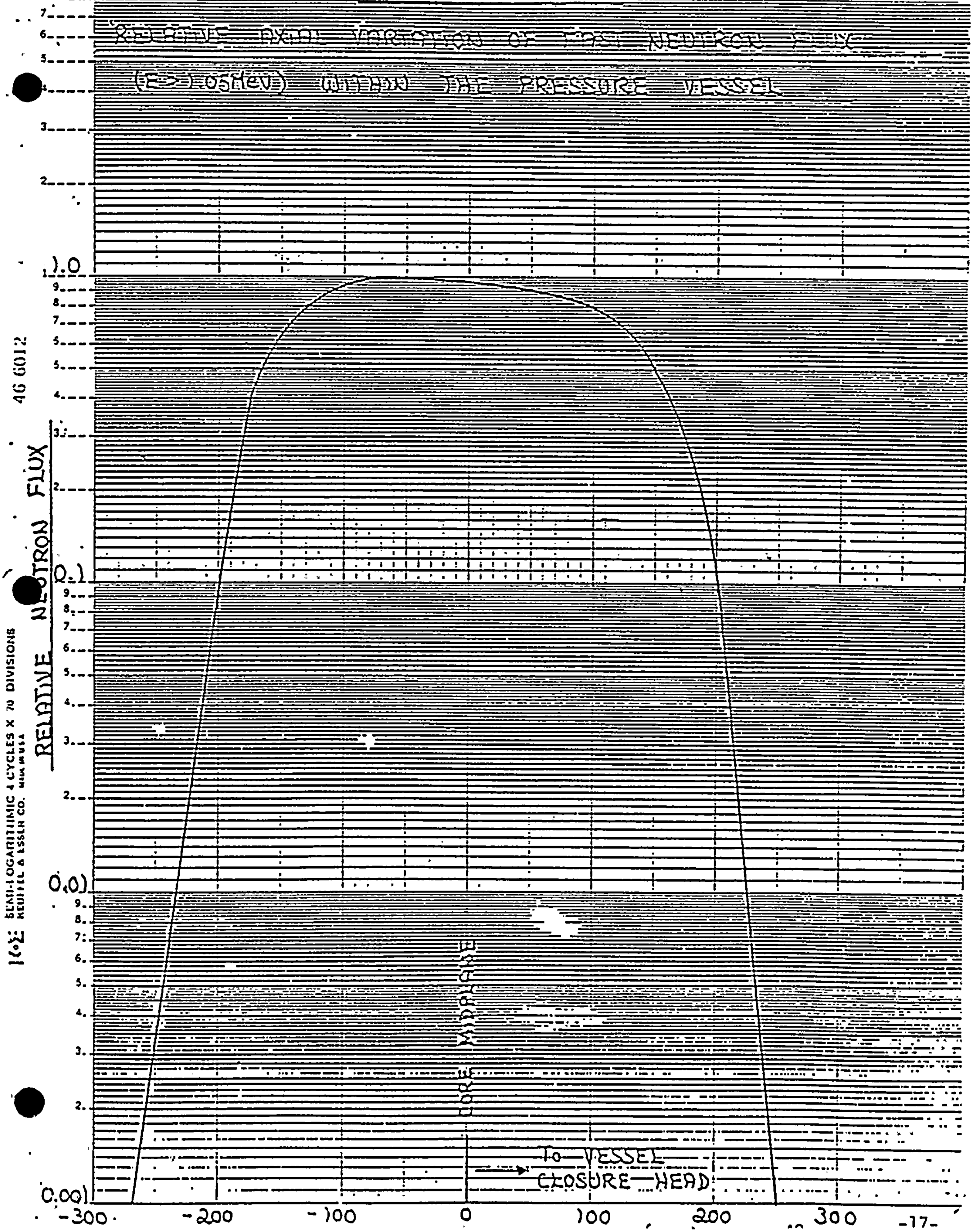


Figure 6



RADIAL DISTRIBUTION OF FAST NEUTRON FLUX
(E > 1.0 MeV) WITHIN THE PRESSURE VESSEL WALL
NEUTRON FLUX (n/cm²-sec)
RADIUS (cm)
195
200
205
210
215
220
225
10^{11}
10^{10}
10^{9}
K·E SEMI-LOGARITHMIC · 3 CYCLES X 70 DIVISIONS
KEUFFEL & ESSER CO. MADE IN USA
46 5490

Figure 8



PRESSURE VESSEL
RELATIVE NEUTRON FLUX
1.0
0.1
0.01
0.001
-300
-200
-100
0
100
200
300
CORE MIDPLANE
To VESSEL CLOSURE HEAD
46 6012
SEMI-LOGARITHMIC 4 CYCLES X 70 DIVISIONS
KEUFFEL & ESSER CO.
K&E

Updated results based on the FE^{54} (n,p) Mn^{54} reaction are summarized in Tables 3 and 4 for Capsules T and S, respectively. The Capsule T data is in excellent agreement with prediction with the average measured value deviating from the calculation by only 7%. However, the Capsule S data indicates a larger mismatch with prediction with the measurement exceeding calculation by some 37%. A disagreement of this magnitude is not typical for Westinghouse Plants. Furthermore, the Capsule S data appears to be inconsistent with Capsule T data in that the measured fluxes agree within 10%, yet flux levels at the 10° position should be 35 - 40% below the 0° flux level. A recheck of the dosimetry or supplementary measurements will verify that the Capsule S data are unrealistically high.

Over the last several fuel cycles beginning with fuel cycle No. 5 (August 1978) the Turkey Point Unit 4 reactor pressure vessel has used forms of a low leakage core configuration. A qualitative assessment of the effect of these core patterns on vessel fluence indicates that the current rate of flux may be as much as 30% below design values. Further work is being carried out to evaluate the current low leakage pattern and to assess more exact core configurations for reducing the rate of flux as much as possible.

TABLE 3

COMPARISON OF MEASURED AND CALCULATED FAST NEUTRON FLUX

MONITOR SATURATED ACTIVITIES FOR CAPSULE T (FLA)

Reaction and Axial Location	Radial Location (cm)	Saturated Activity (DPS/mg)	Adjusted Saturated Activity (DPS/mg)	Fast Neutron Flux (n/cm^2-sec) Capsule T	Calculated
$Fe^{54}(n,p)Mn^{54}$					
W-17	191.46	7.52 (3)	7.14 (3)	1.64 (11)	
W-21	191.46	8.36 (3)	7.94 (3)	1.82 (11)	
S-67	191.46	8.80 (3)	8.36 (3)	1.91 (11)	
S-69	191.46	8.61 (3)	8.18 (3)	1.87 (11)	
S-73	191.46	8.83 (3)	8.39 (3)	1.92 (11)	
Average				1.83 (11)	1.67×10^{11}

TABLE 4

COMPARISON OF MEASURED AND CALCULATED FAST NEUTRON FLUX

MONITOR SATURATED ACTIVITIES FOR CAPSULE S (FLA)

Reaction and Axial Location	Radial Location (cm)	Saturated Activity (DPS/mg)	Adjusted Saturated Activity (DPS/mg)	Fast Neutron Flux (n/cm^2-sec) Capsule S	Calculated
$Fe^{54}(n,p)Mn^{54}$					
Top	191.46	7.37 (3)	7.03 (3)	1.61 (11)	
Middle	191.46	7.15 (3)	6.82 (3)	1.56 (11)	
Bottom	191.46	7.31 (3)	6.97 (3)	1.60 (11)	
Average				1.59 (11)	1.19 (11)
Top	192.46	6.39 (3)	7.58 (3)	1.74 (11)	
Middle	192.46	5.85 (3)	6.94 (3)	1.59 (11)	
Bottom	192.46	6.25 (3)	7.41 (3)	1.70 (11)	
Average				1.68 (11)	1.19 (11)

III. VESSEL WELD MATERIAL INFORMATION

A. Weld Locations and Chemistry

Figure 9 provides the axial locations of the Turkey Point Unit #4 reactor vessel weld seams with respect to the core. End-of-life fluences are identified with each weld location using the flux distributions given in Figures 6 through 8. These indicated values do not account for the low leakage core configuration patterns employed at Turkey Point Unit 4 since August 1978.

The weld chemistry along with the weld wire heat number is given in Table 5 for the critical weld seam, the intermediate shell to lower shell weld. The estimated mean copper content, including the range and standard deviation, are also reported in Table 5 based upon all reported measurements for the critical weld heat.

B. RT_{NDT} Values Based Upon Turkey Point Unit 3 Surveillance Test Results

The surveillance weld for Turkey Point Unit 3 had the same weld wire heat number and the same flux lot number as the beltline weld in Unit 4. Although it has been suggested that the Unit 4 Capsule T surveillance result should be applied to predict ΔRT_{NDT}, the Unit 3 surveillance information is still more representative of both Units 3 and 4 beltline welds for the reasons provided below.

Figure 10 provides a plot of thirteen (13) surveillance results for several B&W reactor vessels, including those for Turkey Point Units 3 & 4, against the NRC Reg. Guide 1.99 trend curves. The copper content of these data range from 0.25 to 0.35% wt, which are representative of the Turkey Point beltline welds. All of the points are of high nickel content and have experienced fluence levels equal to or greater than the subject weld. As can be seen from Figure 10, 12 of the 13 surveillance results show RT_{NDT} values less than those predicted by the Reg. Guide 1.99 trend curves.

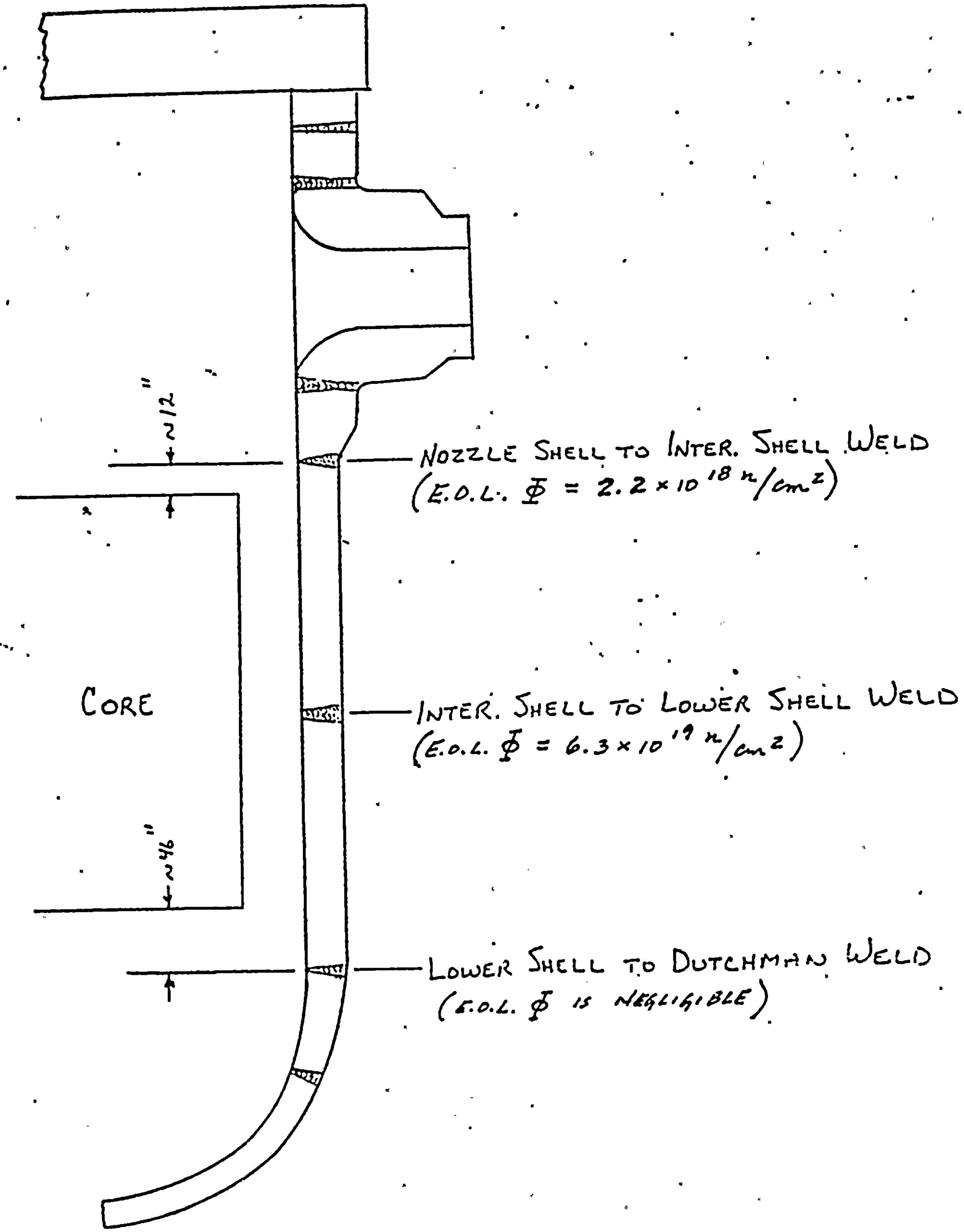


Figure 9. Turkey Point Unit #4 Reactor Vessel Weld Locations and Fluences

TABLE 5

TURKEY POINT UNIT 4 WELD CHEMISTRY

Critical Weld Seam		Chemical Composition (Wt.%) C	Mn	P	S	Si	CR.	Ni	Mo	Cu
Inter Shell to Lower	(b)	.07	1.28	.021	.014	.52	.17	.57	.36	.21
Shell Girth Weld[a]	(c)	.076	1.26	.011	.018	.66	.14	.57	.42	.31

NOTES:

(a) B&W Weld Code No. SA1101 (Wire Heat No. 71249 - Linde 80 Flux Lot No. 8445)

(b) B&W Weld Metal Qualification Analyses

(c) W Unit 3 Surveillance Program Analyses (Same as B&W Weld Code No. SA 1101)

o

Copper analysis on as deposited weld metal for Weld Wire Heat No. 71249

Cu (%)	
.23	B&W Weld Metal Qualification Tests
.21	
.19	
.31	W Surveillance Test
.30	
.35	
.34	
.32	

Mean = 0.28

Range = 0.19 - 0.35

Standard Deviation = 0.06

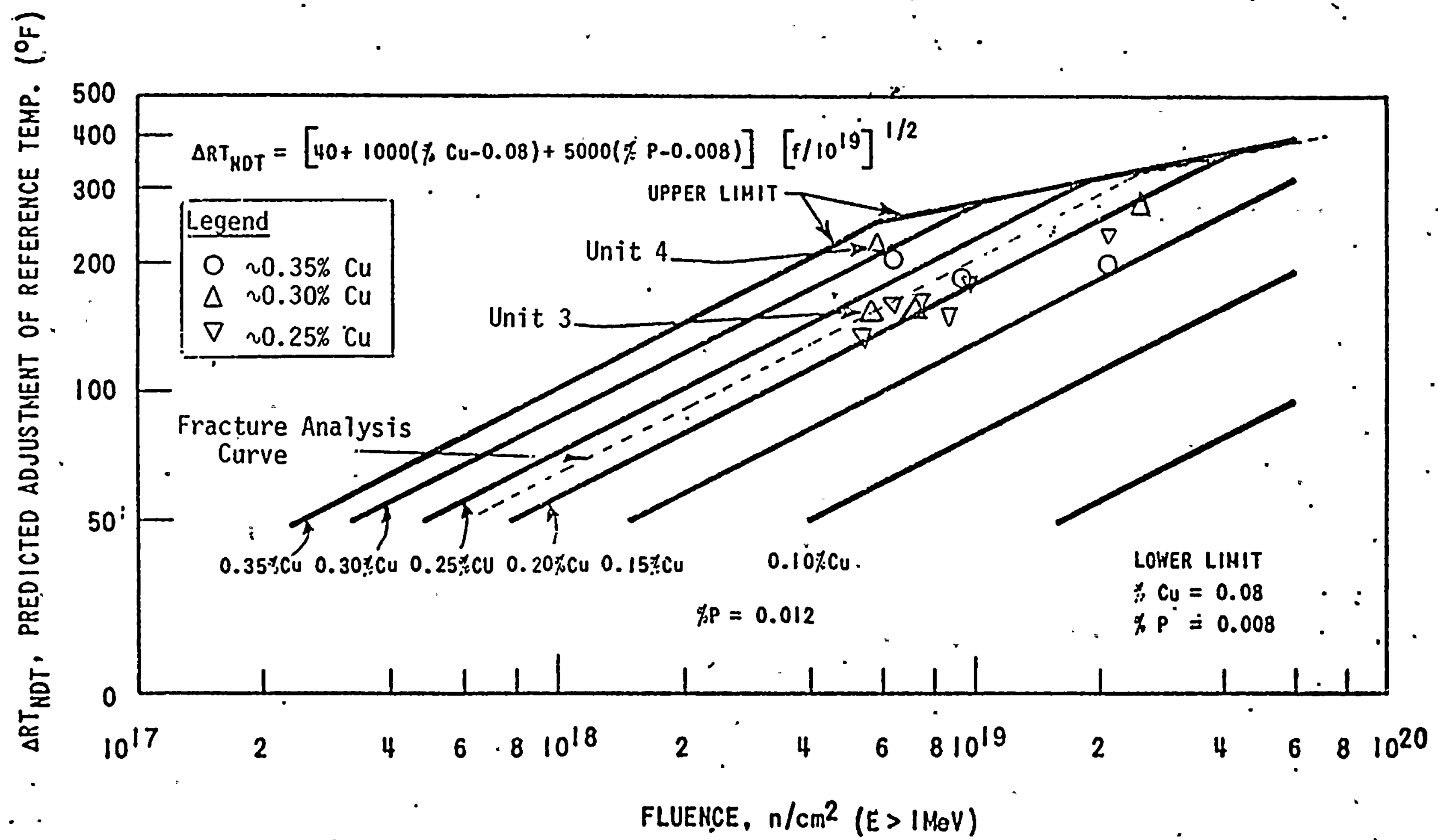
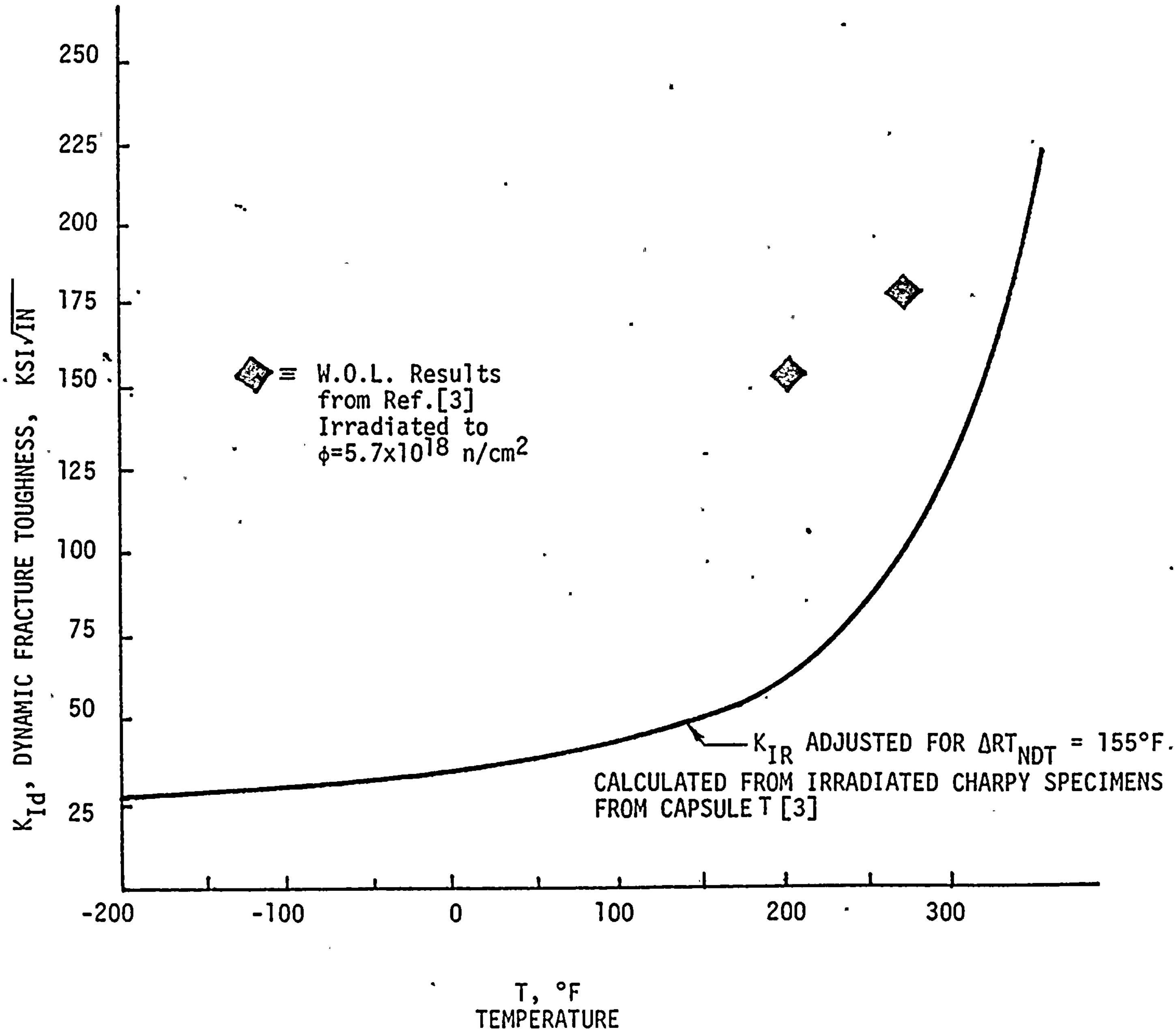


Figure 10 Plot of Surveillance Weld Data for B&W Reactor Vessels Against Regulatory Guide 1.99 Trend Curves

The Unit 4 surveillance result is the only point showing a positive increase in RT_{NDT} when compared against the trend curve prediction and, thus, appears to be an outlier. A review of the irradiated Charpy V notch data for the Unit 4 weld metal[2] indicates that a sufficient number of tests were not performed in the transition region to adequately define the transition temperature. This lack of sufficient data probably accounts for the higher transition temperature shift reported for this weld metal. The Turkey Point Unit 3 surveillance result appears to be near the mean of the 12 points showing a lower shift. Therefore, the Unit 3 surveillance result is considered to be representative of the beltline weld in Unit 4 and has been used in recent fracture analyses for pressurized thermal shock.

To further support the use of the Unit 3 result, Figure 11 shows the results of two dynamic fracture toughness tests on IX-WOL fracture mechanics specimens from the Unit 3 weld metal, which were irradiated to 5.7 to 10^{18} n/cm^2 in the Unit 3 surveillance capsule T[3]. The comparison of these results with the adjusted K_{IR} curve show that use of an adjusted K_{IR} curve based on Charpy transition temperature shifts is very conservative for this material.

The fracture analysis results presented in the following section for the small loss-of-coolant-accident, small steam break, and Rancho-Seco transient are based upon the Unit 3 surveillance result and the NRC Reg. Guide 1.99 trend curve. It was assumed that the trend curve for Unit 4 passes through the Unit 3 surveillance data point and follows the slope of the Reg. Guide trend curve until it intersects the upper limit line following this lower slope until end-of-life as shown in Figure 10. The initial RT_{NDT} value is 3°F. In conjunction with this information, the predicted shift is 211°F for a fluence of 1.1 x 10^{19} n/cm^2 (current total fluence on the inner wall) using Reg. Guide 1.99.



TURKEY POINT UNIT 3 SURVEILLANCE DATA (WELDMENT)

FIGURE 11.

C. Rate of Increase of RT_{NDT}

Future rates of RT_{NDT} increase have been previously generated for the Turkey Point Unit 4 reactor vessel using the slope of prediction curves presented in proposed ASTM Standard, "Predicting Neutron Radiation Damage to Reactor Vessel Material". However, based upon the information given above in combination with the Reg. Guide 1.99 trend curve, RT_{NDT} is increasing at the rate of ∿13.5°F/EFPY for the next 10 effective full power years and ∿3.5°F from then to the end of design life. These values, which are averages since the rate of increase in RT_{NDT} does not vary linearly with time, apply to the inner wall of the reactor vessel beltline weld.

The low leakage core patterns, which have been employed since August 1978, are currently under evaluation for Turkey Point #4. The rates of increase of RT_{NDT} per EFPY taking these configurations into consideration are unavailable at this time. However, the values are anticipated to be lower than those presented above.

D. RT_{NDT} Limit and Criteria for Continued Operation

RT_{NDT} should not be utilized as a sole parameter to determine the acceptability of the integrity of the reactor vessel for any specific plant. RT_{NDT} also should not be used as the sole parameter to compare the relative acceptability of different vessels. However, if a limiting RT_{NDT} is to be defined for a specific vessel, it should be qualified to the specific methodology utilized to calculate an acceptable lifetime for the specific vessel.

The basic methodology that has been utilized in calculating acceptable lifetimes for specific vessels is given in Reference [1]. This report describes the various operational and non-operational transients considered, the thermal and hydraulic methods, the fracture mechanics methods, and the acceptance criteria. In addition to the Owners' Group work, plant specific analyses have been performed to evaluate the impact of key plant specific parameters such as fluence, transient characteristics, material properties, vessel geometry and weld locations on the acceptable vessel lifetimes for Turkey Point Unit 4.

In summary, the most appropriate criterion for continued operation should be based upon fracture analysis results. These results incorporate all of the variables which can affect vessel integrity.

IV. LIMITING TRANSIENT FRACTURE ANALYSES SHOWING BASIS FOR CONTINUED OPERATION

Detailed integrity assessments have been carried out for the Turkey Point Unit 4 reactor vessel postulating the occurrence of the four most limiting thermal shock events applicable to the subject plant:

- Large loss-of-coolant-accident (LOCA)
- Small LOCA
- Large steamline break
- Small steam break

The Rancho-Seco transient was also arbitrarily evaluated although it is an event which is not directly representative of the Turkey Point Unit 4 plant design.

A. Transient Development

A complete discussion of the applied transients relative to the analytical methods and assumptions used in the transient development and to the description of the associated temperature, pressure, and flow rate histories are provided in Reference [1]. Additional similar transient information for the large LOCA and large steamline break events, which were analyzed on a plant specific basis for the Turkey Point Unit 4 reactor vessel in late 1976 and early 1977, are given in Reference [4]. Since this transient information is well documented in Reference [1] and [4], it will not be repeated here. However, some conservatisms inherent in the applied transients for the Turkey Point Unit 4 design are discussed in Section V.

1. Probability of Transient Occurrence

The transient events which have been postulated for evaluation in this report range from small breaks to the instantaneous complete severance of a primary or steam pipe. The large breaks are very unlikely to occur, and have been assigned probabilities in the well-known Rasmussen report [5]. The smaller breaks which have been analyzed herein will be discussed separately.

The small loss of coolant accident transient analyzed results from a sequence of events which can be assigned a probability of 2 x 10^{-2} per vessel per year of operation.

The small steam break transient chosen for analysis is actually one of three scenarios which could occur, and represents the most severe transient of the three, that of a stuck-open safety valve. The probability of this occurring is about 10^{-3} per vessel per year. The other two scenarios, either a stuck open steam dump valve or stuck open atmospheric dump valve, are somewhat more probable, but result in less severe transients.

The Rancho-Seco transient was caused by a control system failure that resulted in an excessive feedwater addition transient in the primary system. This transient produced a significant reactor vessel thermal shock; however, a similar excessive feedwater addition event in a Westinghouse PWR would not produce a primary system transient of significance to reactor vessel integrity. This is due to the fact that the Westinghouse PWR has a large secondary system thermal interia which would minimize the primary system cooldown rate. The primary system pressure and temperature transient that occurred in Rancho-Seco would be similar to the primary system transient that could occur in a Westinghouse PWR due to a low probability small steam break, Also, the repressurization in the Rancho-Seco transient ($\sim$2,100 psi) is not possible for the Turkey Point Unit 4 plant since the capability of the high pressure safety injection (HPSI) pumps is much less (1,400 psi) than the applied values.

B. Fracture Mechanics Analysis

1. General Methods and Acceptance Criteria

A detailed discussion of the basis for the thermal, stress, and fracture analyses completed for the Turkey Point 4 reactor vessel has been previously provided in Reference [1], so it will not be repeated here. Although the acceptance criteria were also provided in Reference [1], it is deemed worthy to provide this information below.

The results of the fracture mechanics analyses of postulated longitudinal and circumferential flaws are presented in Section IV.C below in terms of the maximum number of calendar years the reactor vessel will conform to the following criteria:

1. Minimum critical flaw depth for crack initiation is greater than 1.0 inch, or
2. Crack arrest occurs within 75 percent of the vessel wall thickness.

The initiation criterion is based on the ultrasonic inspection limitations, and the arrest criterion is set to be consistent with Appendix A of Section XI, ASME Code.

It should be noted that the acceptable vessel lifetimes given in the report are based on these acceptance criteria, and therefore, the defined acceptable lifetime does not indicate catastrophic failure of the vessel.

2. Warm Prestressing

The results obtained for the Turkey Point Unit 4 reactor vessel made use of the principle of warm prestressing to demonstrate integrity for the remaining design lifetime for some of the limiting transients provided in the following section.

The technical basis for the use of warm prestressing in demonstrating vessel integrity has been given in detail in References [1] and [6], so it will not be repeated here. The application of warm prestressing to some of the transients results in an excellent behavior with regard to vessel integrity because warm prestressing occurs very early in each transient. Specific details are provided in the following section.

C. Results of Analyses

The minimum number of additional years of vessel operation without violation of acceptance criteria for the above transients relative to vessel thermal shock consideration were provided in Reference [1] for the Turkey Point Unit 4 plant. However, further analyses have been recently completed using plant specific property information and the benefits of warm prestressing for the large LOCA, large steamline break, and Rancho Seco transients. These new results are provided in Table 6 along with the previously submitted small LOCA transient result. Additional information with respect to fundamental inputs to each analysis are also provided in Table 6. Corresponding plots giving specific results of warm prestressing are shown in Figures 12, 13, and 14 for the applicable limiting transients where this benefit was used. The updated results in Table 6 are considered to be more appropriate to the Turkey Point 4 reactor vessel, and therefore should replace the results previously submitted in Reference [1].

The results obtained on all four of the limiting transients applicable to the Turkey Point 4 design demonstrated that vessel integrity would be maintained throughout the design lifetime of the plant. An end-of-life value would also be expected for the Rancho Seco transient, which is not applicable to the Turkey Point 4 plant, if plant specific information would be taken into account.

D. Summary

Detailed analyses have been carried out for a number of postulated transients which result in thermal shock to the reactor pressure vessel. The results of these analyses are summarized in Table 6 and show that reactor vessel integrity will be maintained for the Turkey Point Unit 4 vessel throughout its design lifetime.

TABLE 6

TURKEY POINT UNIT #4 MINIMUM NUMBER OF ADDITIONAL YEARS OF VESSEL OPERATION WITHOUT VIOLATION OF ACCEPTANCE CRITERIA FOR THERMAL SHOCK TRANSIENTS

Transient	Minimum Number of Add'l Years*	Remarks: Vessel Geometry	Weld Location	Material Properties	Transient Characteristic	Fluence Profile	Trend Curve	Benefit of Warm-Prestressi
Large LOCA	33	Plant Specific	Plant Specific	Plant Specific as given in Table 5, Item (c)	Plant Specific	Plant Specific	R.G. 1.99	YES (See Figure 1
Small LOCA	33	"	"	Plant Specific as given in Section III.B	Limiting Generic 3 Loop 2" Break - No Mixing Case	"	"	YES (See Figure 13
Large Steam Break	33	"	"	Plant Specific as given in Table 5, Item (c)	Plant Specific	"	"	YES (See Figure 14
Small Steam Break	33	"	"	Plant Specific as given in Section III.B	Generic	"	"	NO
Rancho-Seco Transient**	25	Generic 3-Loop	Longitudinal Weld in Peak Fluence Location	Generic Category in combination with properties given in Section III.B.	——	Generic 3-Loop	"	YES

*Accumulated EFPY as of 10/31/81 is 5.66 years. The values shown reflect the number of years before conservative acceptance criteria [1] are exceeded (does not indicate actual vessel failure) with the use of a 0.8 plant usage factor (capacity factor).

**This transient is not applicable to the Turkey Point 4 design. However, it was arbitrarily applied.

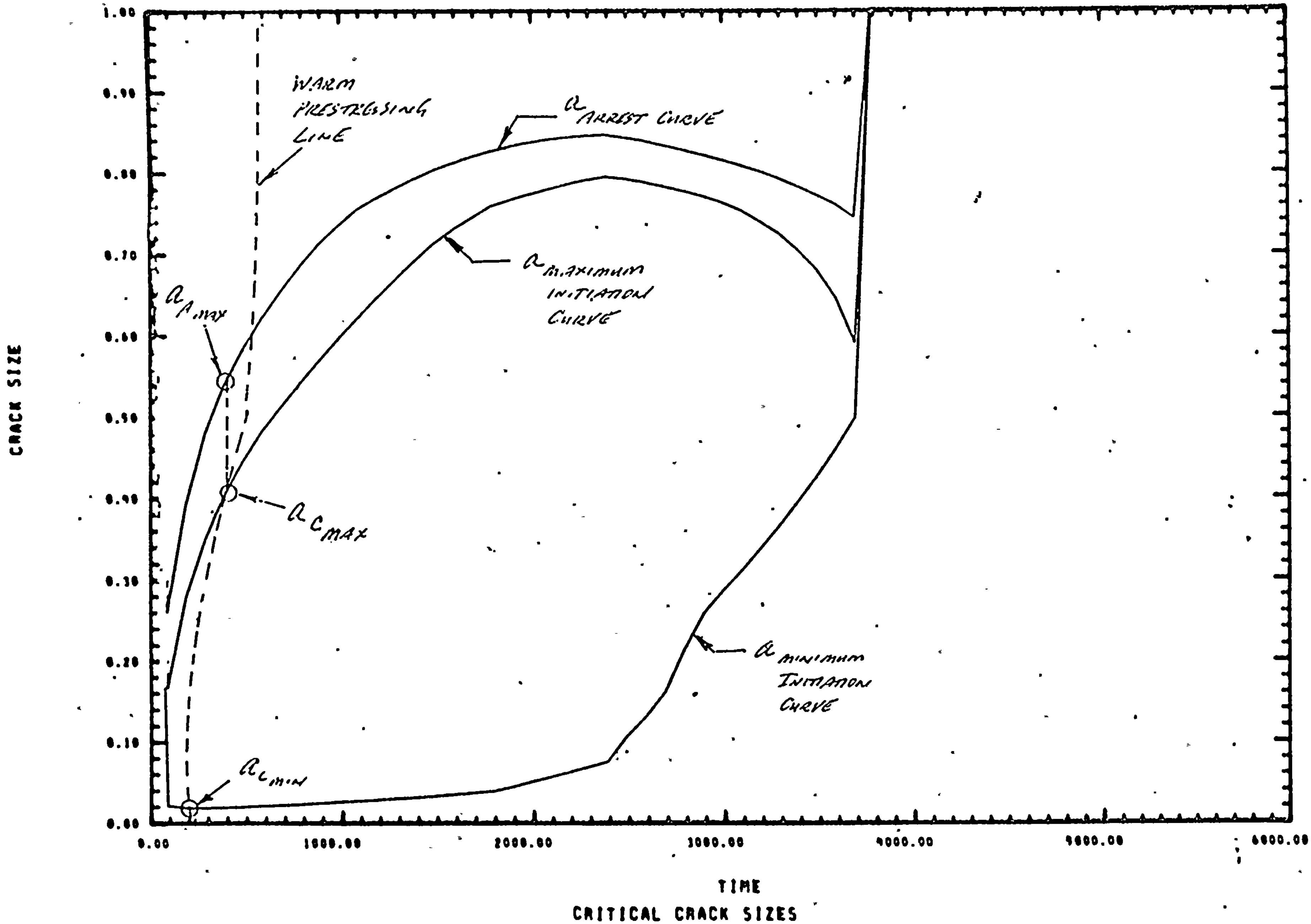


Figure 12 Warm Prestressing Plot for Unit 4 Large LOCA Transient

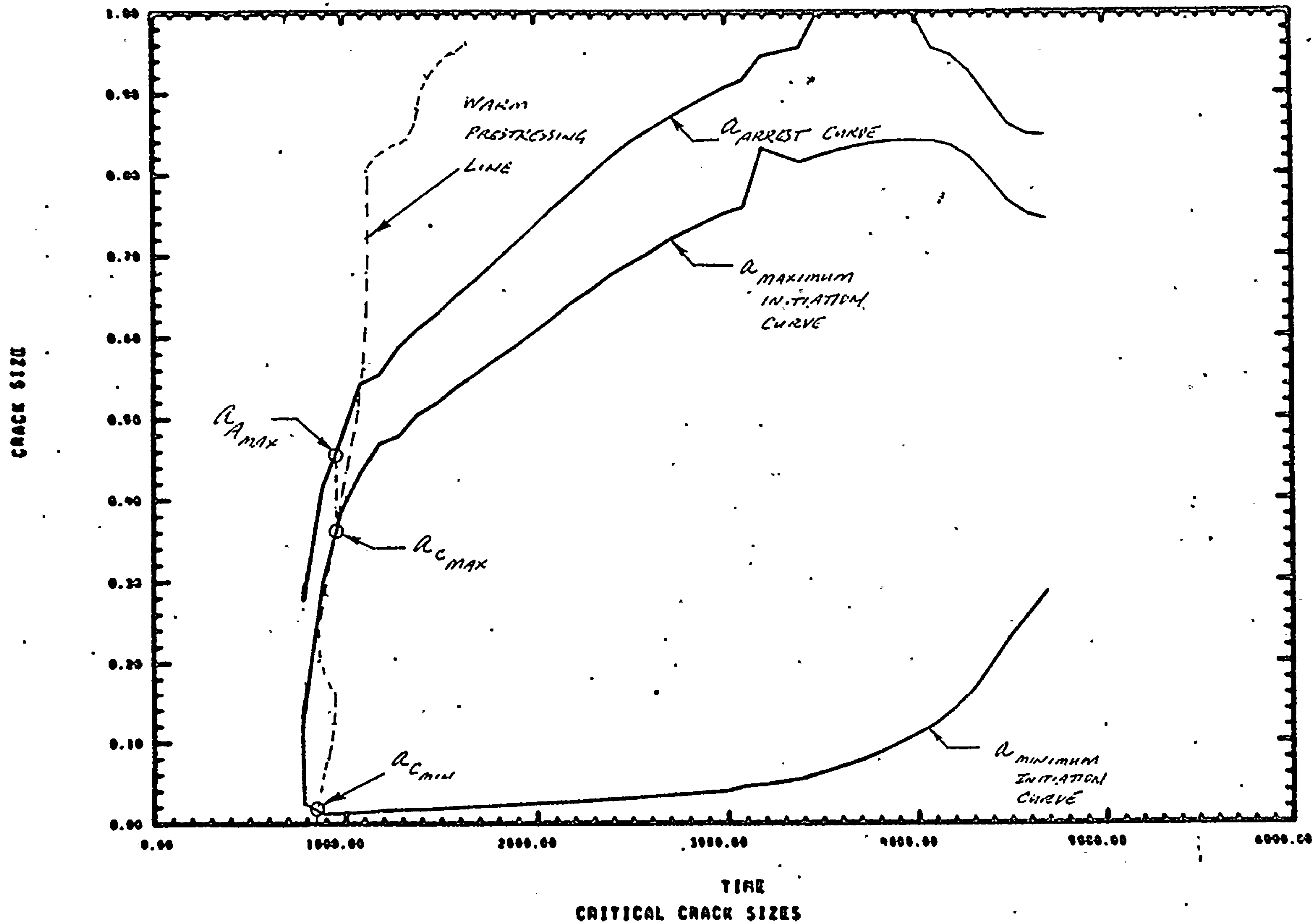


Figure 13 Warm Prestressing Plot for Unit 4 Small LOCA Transient

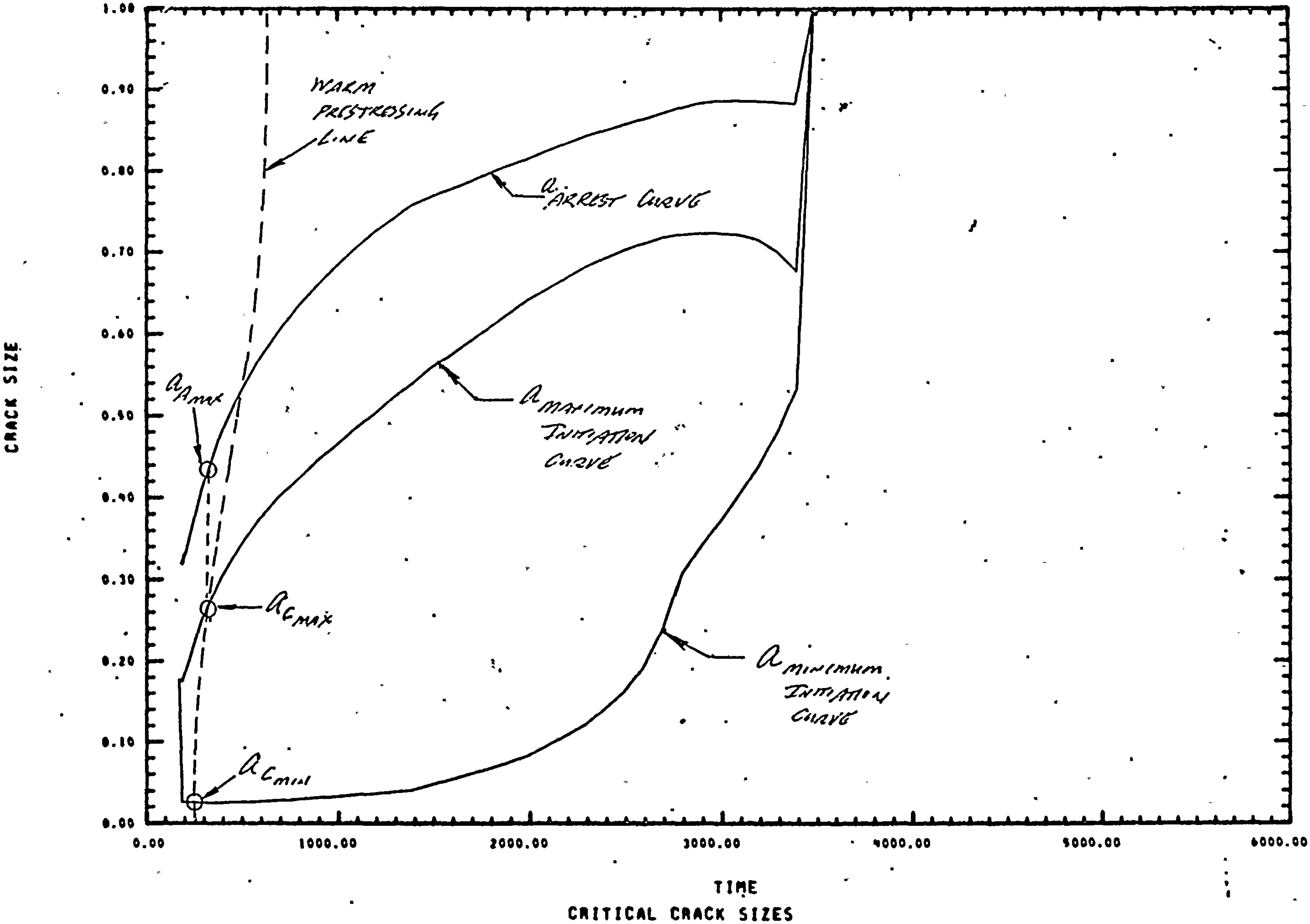


Figure 14 Warm Prestressing Plot for Unit 4 Large Steamline Break Transient

V. OTHER CONSIDERATIONS

A. Uncertainties and Margins

Although a quantified assessment of the sensitivity of fracture analyses recently completed for the Turkey Point Unit 4 beltline weld has not been evaluated at this time relative to uncertainties in input values (e.g. initial crack size, copper content, fluence, and initial RT_{NDT}), a detailed discussion of the conservatisms inherent in these analyses is provided in Reference [1]. Conservatisms due to the generic analytical approach and margins associated with transient development, fluence calculations, and stress and fracture mechanics analyses are outlined in the December 1981 report on reactor vessel integrity [1].

To further quantify the margins inherent in the analyses for Turkey Point Unit 4, the following plant specific conservatisms are noted:

- The large and small LOCA transients development assumed a refueling water storage tank temperature of approximately 40°F. Considering the warm climate of the plant location, this water temperature value is considered to be quite conservative.

- The small steam break transient (and Rancho Seco transient) have included repressurization values (∿2100 psi) well beyond the capability of the Turkey Point Unit 4 high pressure safety injection (HPSI) pumps (1,400 psi).

- Forms of a low leakage core pattern have been in place since fuel cycle No. 5 (Aug. 1978) and this information has not been utilized in the evaluation of vessel integrity. True dosage to the reactor vessel is expected to be somewhat lower than assumed in the analysis and the results obtained are conservative.

B. Remedial Actions

The Westinghouse Owner's Group report [1] provides a qualitative assessment of the feasibility and/or usefulness of the following remedial actions that could be used to resolve vessel integrity concerns, including a reduction in rate of further vessel embrittlement:

(1) Increasing the ECC water temperature via heating of the refueling water storage tank

(2) Limiting auxiliary feedwater flow

(3) Design of control systems to mitigate challenges to reactor vessel integrity.

(4) Core modifications to reduce further neutron radiation damage at the beltline

(5) Recovery of material toughness by in-place annealing of the reactor vessel

Work is being carried out to evaluate the current low leakage core pattern and to assess more exact core configurations for reducing the rate of flux as much as possible for Turkey Point Unit 4.

C. Action Plan

Since integrity has been demonstrated for the design lifetime of Turkey Point Unit 4, there is no need for an action plan.

VI. REFERENCES

[1]. Meyer, T. A., "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants", WCAP-10019, December 1981.

[2]. Norris, E. B., "Reactor Vessel Material Surveillance Program for Turkey Point Unit No. 4, Analysis of Capsule T," Final Report, Southwest Research Institute Project 02-4221, June 14, 1976.

[3]. Yanichko, S.E., et. al, "Analysis of Capsule T from The Florida Power and Light Company Turkey Point Unit No. 3 Reactor Vessel Radiation Surveillance Program", December 1975.

[4]. Meeuwis, O. et al, "Fracture Mechanics Evaluation of the Florida Power and Light Company Turkey Point Unit No. 4 Reactor Pressure Vessel Subjected to Postulated Accident Transients", WCAP-8945, May 1977.

[5]. "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, August 1974.

[6]. McGowan, J. J., "Application of Warm Prestressing Effects to Fracture Mechanics Analyses of Nuclear Reactor Vessels During Severe Thermal Shock," Journal of Nuclear Engineering and Design, V51, No. 3, Feb. 1979.

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

Robert E. Uhrig, being first duly sworn, deposes and says:

That he is Vice President of Florida Power & Light Company, the herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said

Robert E. Uhrig

Subscribed and sworn to before me this

21 day of January, 1982

Cheryl L. Fredrick
NOTARY PUBLIC, in and for the County of Dade,
State of Florida

My commission expires: Notary Public, State of Florida at Large
My Commission Expires October 30, 1983
Bonded thru Maynard Bonding Agency