Table 3-1

Calvert Cliffs Unit 2 Cycle 5
Core Loading

Assembly Designation	Number of Assemblies	Initial Enrichment wt% U-235	Batch Average Burnup MWD/T (EOC4 = 17,[illegible])	Poison Rods Per Assembly	Initial Poison Loading wt% B_4C (2)	Total Number of Poison Rods	Total Number of Fuel Rods
D (1)	13	3.03	20,8[illegible]	0	0	0	2,288
F	40	3.65	13,100	0	0	0	7,040
F/	88	3.03	19,700	8	3.03	704	14,784
G	48	4.00	0	0	0	0	8,448
G/	28	3.55	0	8	3.03	224	4,704
TOTALS	217					928	37,264

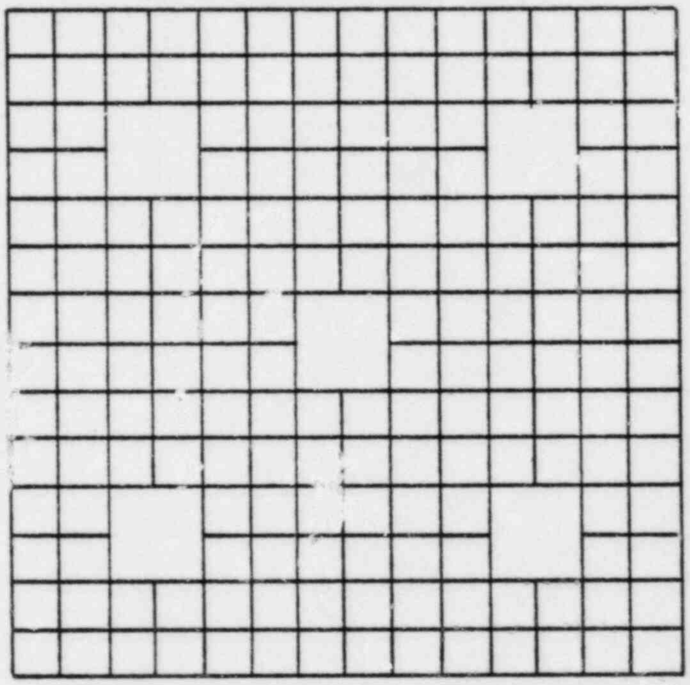
(1) Discharged from Calvert Cliffs Unit 2 Cycle 3.

(2) Shim B^{10} concentration equals .02685 gms B^{10}/inch.

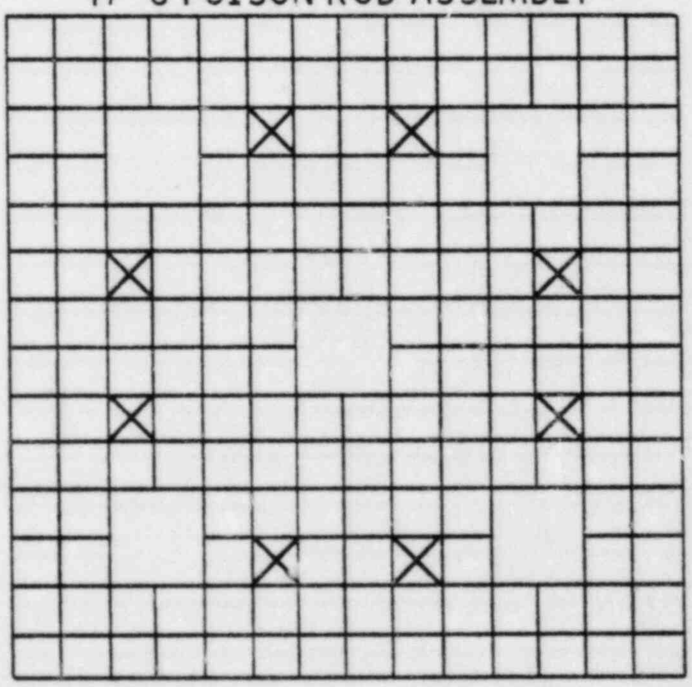
							1 G	2 G
				3 G	4 G	5 G	6 F/	7 F/
			8 G	9 F	10 F/	11 F/	12 F	13 D
		14 G	15 G/	16 F/	17 G/	18 F/	19 G/	20 F/
	21 G	22 F	23 F/	24 F/	25 F/	26 F	27 F/	28 F
	29 G	30 F/	31 G/	32 F/	33 F	34 F/	35 G/	36 D
	37 G	38 F/	39 F/	40 F	41 F/	42 F	43 F/	44 D
45 G	46 F/	47 F	48 G/	49 F/	50 G/	51 F/	52 F/	53 F
54 G	55 F/	56 D	57 F/	58 F	59 D	60 D	61 F	62 D

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 CORE MAP	Figure 3-1

UNSHIMMED ASSEMBLY



F/ -8 POISON ROD ASSEMBLY

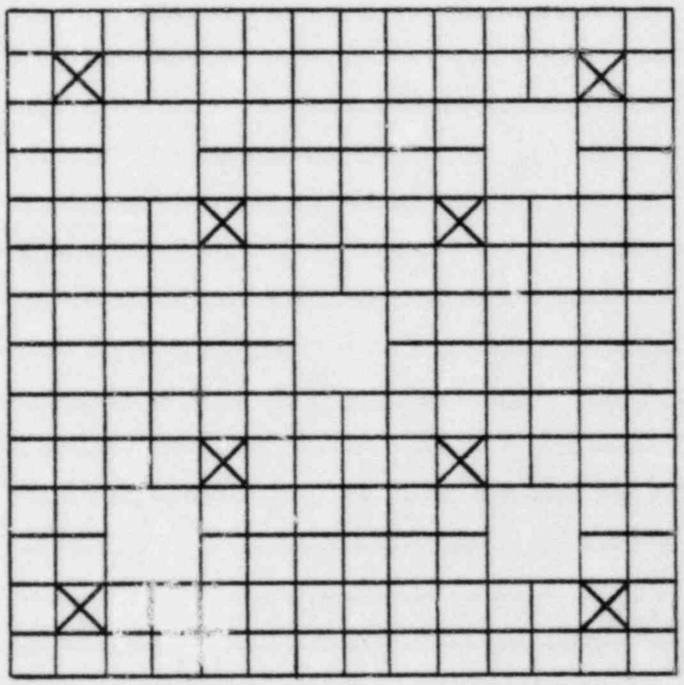


FUEL ROD LOCATION

POISON ROD LOCATION

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY FUEL AND OTHER ROD LOCATIONS	Figure 3-2

G/ -8 POISON ROD ASSEMBLY



☐ FUEL ROD LOCATION

☒ POISON ROD LOCATION

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY FUEL AND OTHER ROD LOCATIONS	Figure 3-3

INITIAL ENRICHMENT, W/O U-235
BOC5 BURNUP (MWD/T), EOC4 = 17,300 MWD/T

Assembly	Initial Enrichment (W/O U-235)	BOC5 Burnup (MWD/T)
1	4.00	0
2	4.00	0
3	4.00	0
4	4.00	0
5	4.00	0
6	3.03	20,600
7	3.03	20,300
8	4.00	0
9	3.65	11,500
10	3.03	20,700
11	3.03	20,700
12	3.65	15,400
13	3.03	22,400
14	4.00	0
15	3.55	0
16	3.03	18,000
17	3.55	0
18	3.03	20,500
19	3.55	0
20	3.03	21,000
21	4.00	0
22	3.65	11,500
23	3.03	18,000
24	3.03	16,400
25	3.03	18,800
26	3.65	12,900
27	3.03	20,600
28	3.65	14,600
29	4.00	0
30	3.03	20,700
31	3.55	0
32	3.03	18,400
33	3.65	11,000
34	3.03	20,200
35	3.55	0
36	3.03	19,400
37	4.00	0
38	3.03	20,700
39	3.03	20,600
40	3.65	12,900
41	3.03	20,200
42	3.65	14,600
43	3.03	19,400
44	3.03	20,100
45	4.00	0
46	3.03	20,600
47	3.65	15,400
48	3.55	0
49	3.03	20,700
50	3.55	0
51	3.03	19,100
52	3.03	16,300
53	3.65	11,100
54	4.00	0
55	3.03	20,300
56	3.03	22,400
57	3.03	21,000
58	3.65	14,600
59	3.03	19,400
60	3.03	20,100
61	3.65	11,100
62	3.03	22,300

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY AVERAGE BURNUP AT BOC AND INITIAL ENRICHMENT DISTRIBUTION	Figure 3-4

						1 10,200	2 12,500	
				3 10,500	4 14,000	5 15,800	6 32,500	7 31,900
			8 12,000	9 25,800	10 33,700	11 33,800	12 29,900	13 34,100
		14 12,000	15 16,700	16 32,100	17 17,900	18 34 400	19 18,200	20 34,700
	21 10,500	22 25,800	23 32,100	24 30,500	25 33,000	26 29,000	27 34,800	28 30,300
	29 14,000	30 33,800	31 18,000	32 32,800	33 27,300	34 34,200	35 18,000	36 32,700
45 10,200	37 15,800	38 33,800	39 34,400	40 29,000	41 34,300	42 29,700	43 32,600	44 32,400
54 12,500	46 32,500	47 29,900	48 18,100	49 34,800	50 18,000	51 32,400	52 30,000	53 26,500
	55 31,900	56 34,100	57 34,700	58 30,300	59 32,700	60 32,400	61 26,500	62 35,200

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY AVERAGE BURNUP AT EOC (MWD/T)	Figure 3-5

4.0 FUEL SYSTEM DESIGN

4.1 Mechanical Design

The mechanical design for the standard Batch G reload fuel is identical to that of the standard Batch H fuel described in the reference cycle submittal (Calvert Cliffs Unit 1 Cycle 6, Reference 1). It is also identical to that of the standard Batch F fuel used in Calvert Cliffs Unit 2 Cycle 4 (Reference 3) with the exception of a .200 inch reduction in the overall length of the fuel rods. The length reduction will provide additional clearance for fuel rod length increase during the lifetime of the Batch G fuel.

The mechanical designs of the Batch D, and F fuel assemblies were described in References 7 and 3, respectively.

C-E has performed analytical predictions of cladding creep collapse time for all Calvert Cliffs Unit 2 fuel batches that will be irradiated in Cycle 5 and has concluded that the collapse resistance of all standard fuel rods is sufficient to preclude collapse during their design lifetime. This lifetime will not be exceeded by the Cycle 5 duration (Table 4-1). These analyses utilized the CEPAN computer code (Reference 8) and the analysis procedures described in Reference 9. The analysis procedures described in Reference 9 were approved in Reference 10.

TABLE 4-1

Batch	Minimum Collapse Time	EOC 5 Exposure
D	> 35,000 EFPH	26,000 EFPH
F	> 35,000	23,200
G	> 27,500	10,400

All batches of fuel were also reviewed for dimensional changes using the SIGREEP model described in Reference 11. All clearances were found to be adequate during Cycle 5. The SIGREEP model described in Reference 11 was approved in Reference 10.

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for the Batch G fuel are identical to those of the Batch D and F fuels to be included in Cycle 5. Thus, the chemical or metallurgical performance of the Batch G fuel will remain unchanged from the performance of the Cycle 4 fuel.

4.2 Hardware Modifications to Mitigate Guide Tube Wear

All standard fuel assemblies which will be placed in CEA locations in Cycle 5 will have stainless steel sleeves installed in the guide tubes to prevent guide tube wear. A detailed discussion of the design of the sleeves and their effect on reactor operation is contained in Reference 12.

4.3 Thermal Design

The thermal performance of composite fuel pins that envelopes the various fuel assemblies present in Cycle 5 (fuel batches D, F and G) has been evaluated using the FATES3 version of the fuel evaluation model (References 13 and 14). The analysis was performed with a power history that modeled the power and burnup levels representative of the peak pin at each burnup interval, from beginning of cycle to end of cycle burnups. The burnup range analyzed is in excess of that expected at the end of Cycle 5.

The FATES3 power-to-centerline melt limit was determined for Cycle 5 by taking some credit for the decrease in power peaking which is characteristic of highly burned fuel. Since a gradual decrease in the calculated power-to-melt (due to a decrease in the fuel melt temperature) also accompanies burnup, the most limiting power-to-centerline melt has been found to occur within an intermediate burnup range. Using conservative estimates of the burnup point at which the power peaking begins to decrease and the rate at which it decreases for Cycle 5, the most limiting power-to-centerline melt has been determined to be in excess of 22 kw/ft and to occur at a rod average burnup of approximately 33,000 MWD/MTU.

5.0 NUCLEAR DESIGN

Cycle 5 of Unit 2 will be the second fuel cycle for either Calvert Cliffs Unit for which the average discharge exposure will be as high as 33,700 MWD/T. All analyses address fuel exposure explicitly and, as in the reference cycle, the small increase in average discharge exposure over that of the previous cycles has not yielded, of itself, significant changes in core parameters. Furthermore, those assemblies with burnups in excess of 24,000 MWD/T at EOC5 contain maximum 1-pin peaks which are substantially below the maximum 1-pin peak in the core (See Section 6.2).

5.1 Physics Characteristics

5.1.1 Fuel Management

The Cycle 5 fuel management employs a mixed central region as described in Section 3, Figure 3-1. The fresh Batch G fuel is comprised of two sets of assemblies, each having a unique enrichment in order to minimize radial power peaking. There are 48 assemblies with an enrichment of 4.00 wt% U-235 and 28 assemblies with an enrichment of 3.55 wt% U-235 and 8 poison shims per assembly. With this loading, the Cycle 5 burnup capacity for full power operation is expected to be between 13,100 MWD/T and 13,700 MWD/T, depending on the final Cycle 4 termination point. The Cycle 5 core characteristics have been examined for Cycle 4 terminations between 16,300 and 17,300 MWD/T and limiting values established for the safety analyses. The loading pattern (see Section 3) is applicable to any Cycle 4 termination point between the stated extremes.

Physics characteristics including reactivity coefficients for Cycle 5 are listed in Table 5-1 along with the corresponding values from the reference cycle. Please note that the values of parameters actually employed in safety analyses are different from those displayed in Table 5-1 and are typically chosen to conservatively bound predicted values with accommodation for appropriate uncertainties and allowances.

Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for the end of Cycle 5 zero power steam line break accident and a comparison to reference cycle data. The EOC zero power steam line break was selected since it is the most limiting zero power steam line break accident and, thus, provides the basis for establishing the Technical Specification required shutdown margin.

The power dependent insertion limit (PDIL) for bank 2 insertion has changed relative to Cycle 4. The new PDIL curve, which is the same as the curve for the reference cycle, is discussed in Section 9. Table 5-3 shows the reactivity worths of various CEA groups calculated at full power conditions for Cycle 5 and the reference cycle.

5.1.2 Power Distribution

Figures 5-1 through 5-3 illustrate the all rods out (ARO) planar radial power distributions at BOC5, MOC5 and EOC5, respectively, that are characteristic of the high burnup end of the Cycle 4 shutdown window. These planar radial power peaks are characteristic of the major portion of the active core length between about 20 and 80 percent of the fuel height. The high burnup end of the Cycle 4 shutdown window tends to increase the power peaking in this axial central region of the core for Cycle 5. The planar radial power distributions for the above region with CEA Group 5 fully inserted at beginning and end of Cycle 5 are shown in Figures 5-4 and 5-5, respectively, for the high burnup end of the Cycle 4 shutdown window.

The radial power distributions described in this section are calculated data without uncertainties or other allowances. However, the single rod power peaking values do include the increased peaking that is characteristic of fuel rods adjoining the water holes in the fuel assembly lattice. For both DNB and kw/ft safety and setpoint analyses in either rodded or unrodded configurations, the power peaking values actually used are higher than those expected to occur at any time during Cycle 5. These conservative values, which are used in Section 7 of this document, establish the allowable limits for power peaking to be observed during operation.

The range of allowable axial peaking is defined by the limiting conditions for operation covering axial shape index (ASI). Within these ASI limits, the necessary DNBR and kw/ft margins are maintained for a wide range of possible axial shapes. The maximum three-dimensional or total peaking factor anticipated in Cycle 5 during normal base load, all rods out operation at full power is 1.88, not including uncertainty allowances and augmentation factors.

5.1.3 Safety Related Data

The Cycle 5 safety related data for this section are identical to the safety related data used in the Reference Cycle, as presented in Section 5.1.3 of Reference 1.

5.2 Analytical Input to In-Core Measurements

In-core detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in the manner described in Reference 15, which is the same method used for the reference cycle.

5.3 Nuclear Design Methodology

Analyses have been performed in the same manner and with the same methodologies used for the reference cycle analyses.

5.4 Uncertainties in Measured Power Distributions

The power distribution measurement uncertainties contained in Reference 15 which are applied to Cycle 5 are:

Total 3-D peaking factor (Fq) uncertainty = 6.2 percent
where $Fq = F_{xy} \times F_z$, local power density

Integrated radial peaking factor (Fr) uncertainty = 6.0 percent

These values are to be used for monitoring power distribution parameters during operation.

Table 5-1

Calvert Cliffs Unit 2 Cycle 5
Nominal Physics Characteristics

	Units	Reference Cycle**	Cycle 5
Dissolved Boron			
Dissolved Boron content for Criticality, CEAs Withdrawn			
Hot Full Power, Equilibrium Xenon, BOC	PPM	1025	1032
Boron Worth			
Hot Full Power BOC	PPM/%$\Delta\rho$	106	105
Hot Full Power EOC	PPM/%$\Delta\rho$	85	85
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients, Hot Full Power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\ \Delta\rho/^{o}F$	-0.2	-0.1
End of Cycle	$10^{-4}\Delta\rho/^{o}F$	-2.1	-2.1
Doppler Coefficient			
Hot Zero Power BOC	$10^{-5}\Delta\rho/^{o}F$	-1.48	-1.48
Hot Full Power BOC	$10^{-5}\Delta\rho/^{o}F$	-1.24	-1.27
Hot Full Power EOC	$10^{-5}\Delta\rho/^{o}F$	-1.47	-1.47
Total Delayed Neutron Fraction, βeff			
BOC		.00609^{+}	.00609
EOC		.00522^{+}	.00521
Neutron Generation Time, ℓ^*			
BOC	10^{-6}sec	23.8	24.0
EOC	10^{-6}sec	30.2	30.5

$^{+}$These values are corrections to those reported in Reference 1 for Unit 1 Cycle 6.

**Unit 1 Cycle 6

Table 5-2

Calvert Cliffs Unit 2 Cycle 5 Limiting Values of Reactivity Worths and Allowances for Hot Zero Power Steam Line Break, %$\Delta\rho$ End-of-Cycle (EOC)

	Reference Cycle	Cycle 5
1. Worth of all CEA's Inserted	10.2	9.4
2. Stuck CEA Allowance	2.6	1.8
3. Worth of all CEA's Less Worth of CEA Stuck Out**	7.6	7.6
4. Zero Power Dependent Insertion Limit CEA Bite	1.7	1.8
5. Calculated Scram Worth	5.9	5.8
6. Physics Uncertainty (10% of Item 5)	0.6	0.6
7. Net Ava'lable Scram Worth (Item 5 minus Item 6)	5.3	5.2
8. Technical Specification Shutdown Margin	5.3	5.2
9. Margin in Excess of Technical Specification Shutdown Margin	0.0	0.0

---

* Unit 1 Cycle 6

**Stuck CEA is one which yields worst results for HZP SLB, i.e., worst combination of scram worth and reactivity insertion with cooldown.

Table 5-3

Calvert Cliffs Unit 2 Cycle 5
Reactivity Worth of CEA Regulating
Groups at Hot Full Power, % $\Delta\rho$

	Beginning of Cycle		End of Cycle	
Regulating CEA's	Reference Cycle*	Cycle 5	Reference Cycle*	Cycle 5
Group 5	0.48	0.49	0.65	0.63
Group 4	0.31	0.27	0.33	0.36
Group 3	0.84	0.91	1.04	1.16

Notes

Values shown assume sequential group insertion

*Unit 1 Cycle 6

						0.77	0.93	
				0.78	1.08	1.23 X	0.86	0.82
			0.89	1.08	0.94	0.93	1.03	0.79
		0.89	1.21	1.01	1.26	0.95	1.25	0.92
	0.78	1.08	1.01	1.00	0.99	1.15	0.96	1.10
	1.08	0.94	1.26	1.01	1.19	0.94	1.20	0.87
0.76	1.23	0.93	0.95	1.16	0.95	1.05	0.87	0.80
0.93	0.86	1.03	1.25	0.95	1.20	0.88	0.93	1.09
	0.82	0.79	0.92	1.10	0.87	0.80	1.09	0.89

NOTE: X=MAXIMUM 1-PIN PEAK = 1.57

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY RELATIVE POWER DENSITY AT BOC, EQUILIBRIUM XENON	Figure 5-1

							0.74	0.90
				0.75	1.00	1.13	0.84	0.82
			0.87	1.03	0.91	0.91	1.03	0.83
		0.87	1.21	1.00	1.29	0.97	1.31 X	0.96
	0.75	1.03	1.00	1.01	0.99	1.15	0.99	1.13
	1.00	0.91	1.29	1.00	1.16	0.97	1.29	0.93
	1.13	0.91	0.97	1.15	0.97	1.08	0.92	0.86
0.74	0.84	1.03	1.31	0.99	1.29	0.93	0.97	1.11
0.89	0.82	0.83	0.96	1.13	0.93	0.86	1.11	0.91

NOTE: X=MAXIMUM 1-PIN PEAK= 1.47

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY RELATIVE POWER DENSITY AT 7 GWD/T EQUILIBRIUM XENON	Figure 5-2

							0.76	0.91
				0.77	0.99	1.11	0.85	0.34
			0.90	1.03	0.92	0.92	1.03	0.85
		0.90	1.26	1.00	1.32	0.97	1.34 X	0.98
	0.77	1.03	1.00	0.99	0.97	1.11	0.99	1.11
	0.99	0.92	1.32	0.98	1.10	0.95	1.31	0.94
0.76	1.11	0.92	0.97	1.11	0.95	1.04	0.91	0.86
0.91	0.85	1.03	1.34	0.99	1.31	0.92	0.93	1.05
	0.84	0.85	0.98	1.11	0.94	0.86	1.05	0.87

NOTE: X=MAXIMUM 1-PIN PEAK= 1.51

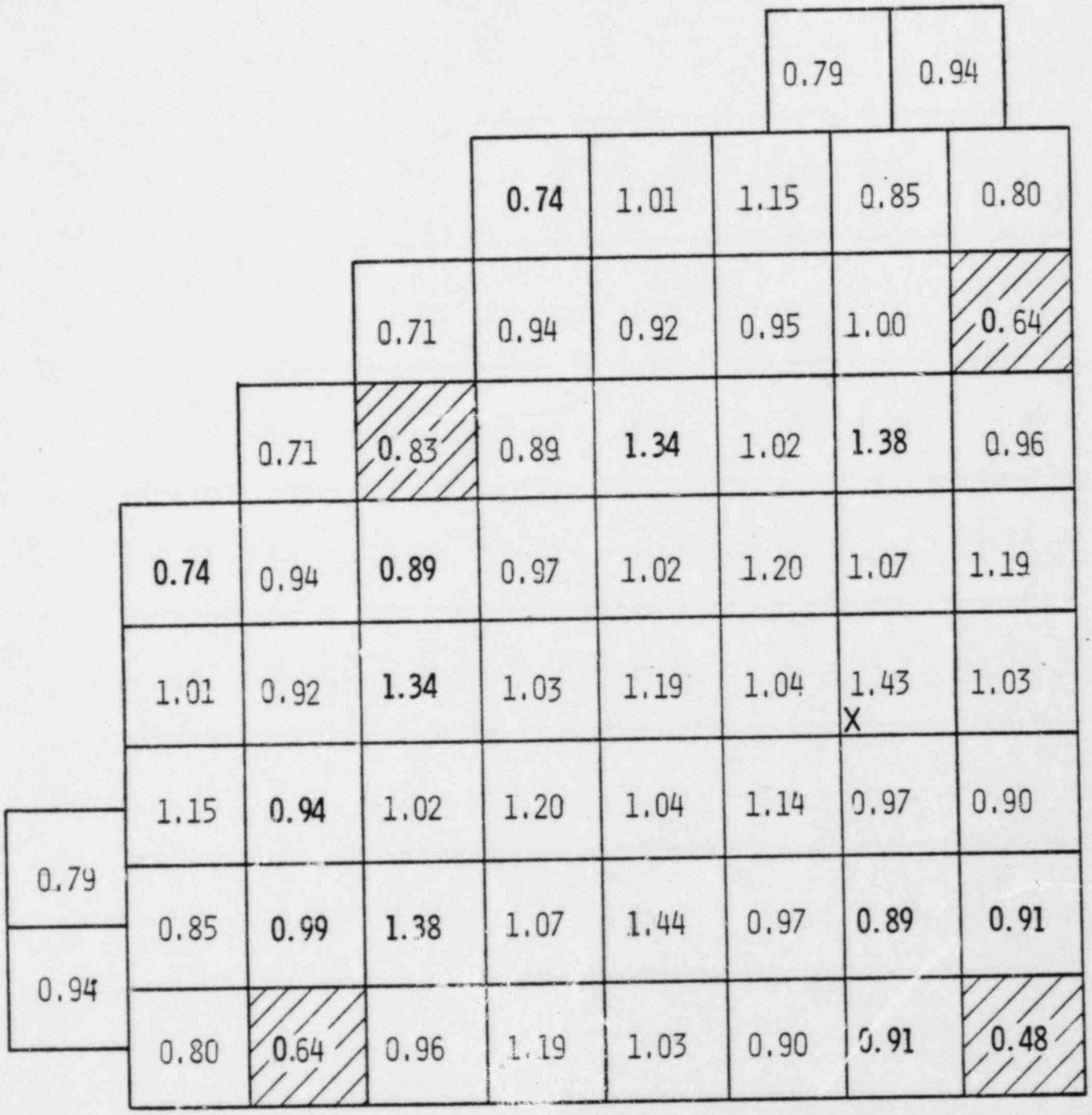
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY RELATIVE POWER DENSITY AT EOC, EQUILIBRIUM XENON	Figure 5-3

						0.81	0.99	
				0.80	1.14	1.30 X	0.88	0.80
			0.74	1.01	0.97	0.97	1.00	0.63
		0.74	0.86	0.91	1.28	1.00	1.28	0.92
	0.80	1.01	0.91	0.97	1.04	1.21	1.02	1.13
	1.14	0.97	1.28	1.05	1.25	1.02	1.28	0.92
0.81	1.30	0.97	1.00	1.22	1.03	1.09	0.88	0.79
0.99	0.88	1.00	1.28	1.02	1.28	0.88	0.82	0.87
	0.80	0.63	0.92	1.13	0.92	0.79	0.87	0.48

NOTE: X=MAXIMUM 1-PIN PEAK= 1.65

CEA BANK 5 LOCATIONS

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, BOC	Figure 5-4



NOTE: X=MAXIMUM 1-PIN PEAK= 1.64

CEA BANK 5 LOCATIONS

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 5 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, EOC	Figure 5-5

6.0 THERMAL-HYDRAULIC DESIGN

6.1 DNBR Analysis

Steady state DNBR analyses of Cycle 5 at the rated power level of 2700 MWt have been performed using the TORC computer code described in Reference 1, the CE-1 critical heat flux correlation described in Reference 2, and the simplified modeling methods described in Reference 3.

A variant of TORC called CETOP, optimized for simplified modeling applications, was used in this cycle to develop the "design thermal margin model" described generically in Reference 3. Details of CETOP are discussed in Reference 4. CETOP was approved for use on Calvert Cliffs Units in Reference 5. CETOP is used only because it reduces computer costs significantly; no margin gain is expected or taken credit for.

Table 6-1 contains a list of pertinent thermal-hydraulic design parameters used for both safety analyses and for generating reactor protective system setpoint information. The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) listed in Table 6-1 have been combined statistically with other uncertainty factors at the 95/95 confidence/probability level to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6 and approved by the NRC in Reference 5. The applicability of this minimum DNBR limit has been verified for Cycle 5.

Investigations have been made to ascertain the effect of the CEA guide tube wear problem and the sleeving repair on DNBR margins. The findings were reported to the NRC in Reference 7 which concluded that the wear problem and the sleeving repair do not adversely affect DNBR margin.

6.2 Effects of Fuel Rod Bowing on DNBR Margin

The fuel rod bowing effects on DNB margin for Calvert Cliffs Unit 2 Cycle 5 have been evaluated according to the guidelines set forth in Reference 8.

A total of 141 fuel assemblies will exceed the NRC specified DNB penalty threshold burnup of 24 GWD/T (Reference 8) during Cycle 5, the maximum assembly burnup reaching 35.2 GWD/T by the end of cycle. For those assemblies which will experience a burnup of between 24 and 35.2 GWD/T at any time during Cycle 5, the minimum best estimate margin available relative to more limiting peaking values present in other assemblies is greater than 11%. The DNB rod bow penalty for this burnup range, as determined via an interpolation of the data contained in Reference 8, varies from 0 to 3.8%. In summary, the magnitude of the margin available is considerably in excess of the corresponding DNB rod bow penalty and, consequently, no power penalty for fuel rod bowing is required in Cycle 5.

Table 6-1

Calvert Cliffs Unit 2

Thermal-Hydraulic Parameters at Full Power

General Characteristics	Unit	Reference+ Unit 1, Cycle 6**	Cycle 5**
Total Heat Output (core only)	MWT 10^6 Btu/hr	2700 9215	2700 9215
Fraction of Heat Generated in Fuel Rod		.975	.975
Primary System Pressure (Nominal)	psia	2250	2250
Inlet Temperature	°F	548	543
Total Reactor Coolant Flow (steady state)	gpm 10^6 lb/hr	381,600 143.8	381,600 143.8
Coolant Flow Through Core	10^6 lb/hr	138.5	138.5
Hydraulic Diameter (nominal channel)	ft	0.044	0.044
Average Mass Velocity	10^6 lb/hr-ft^2	2.59	2.59
Pressure Drop Across Core (steady state flow irreversible Δp over entire fuel assembly)	psi	11.1	11.1
Total Pressure Drop Across Vessel (based on steady state flow and nominal dimensions)	psi	34.7	34.7
Core Average Heat Flux (Accounts for above fraction of heat generated in fuel rod and axial densification factor)	Btu/hr-ft^2	184,266***	185,532****
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	48,748***	48,415****
Film Coefficient at Average Conditions	Btu/hr-ft^2-°F	5930	5930
Average Film Temperature Difference	°F	31	31

Table 6-1 (continued)

General Characteristics	Unit	Reference+ Unit 1, Cycle 6**	Cycle 5**
Average Linear Heat Rate of Undensified Fuel Rod (accounts for above fraction of heat generated in fuel rod)	kw/ft	6.16***	6.20****
Average Core Enthalpy Rise	Btu/lb	66.5	66.5
Maximum Clad Surface Temperature	°F	657	657

Calculational Factors	Reference Unit 1, Cycle 6	Cycle 5
Engineering Heat Flux Factor	1.03*	1.03*
Engineering Factor on Hot Channel Heat Input	1.02*	1.02*
Rod Pitch and Clad Diameter Factor	1.065*	1.065*
Fuel Densification Factor (axial)	1.01	1.01

NOTES

* These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level (Reference 6) to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6 and approved by the NRC in Reference 5. This limit has been verified to be applicable to Cycle 5.

** Due to the statistical combination of uncertainties described in References 6, 9, and 10, the nominal inlet temperature and nominal primary system pressure were used to calculate some of these parameters.

***Based on Unit 1, Cycle 6 specific value of 672 shims.

****Based on Unit 2, Cycle 5 specific value of 928 shims.

+ Reference cycle (Unit 1, Cycle 6) analysis is contained in Reference 11.

7.0 Transient Analysis

This section presents the results of the Baltimore Gas & Electric Calvert Cliffs Unit 2, Cycle 5 Non-LOCA safety analysis at 2700 MWt.

The Design Bases Events (DBEs) considered in the safety analysis are listed in Table 7-1. These events were categorized in the following groups:

1. Anticipated Operational Occurrences (AOOs) for which the intervention of the Reactor Protection System (RPS) is necessary to prevent exceeding acceptable limits.

2. AOOs for which the intervention of the RPS trips and/or initial steady state thermal margin, maintained by Limiting Conditions for Operation (LCO), are necessary to prevent exceeding acceptable limits.

3. Postulated Accidents

4. Postulated Occurrences

The Design Basis Events (DBEs) considered in the Unit 2, Cycle 5 safety analyses are listed in Table 7-1. Core parameters input to the safety analyses for evaluating approaches to DNB and centerline temperature to melt fuel design limits are presented in Table 7-2.

As indicated in Table 7-1, no reanalysis was performed for the DBEs for which key transient input parameters are within the bounds (conservative with respect to) of the reference cycle values (Unit 1, Cycle 6, Reference 1). For these DBEs the results and conclusions quoted in the reference cycle analysis are valid for Unit 2, Cycle 5.

A re-evaluation of a number of DBEs, as indicated in Table 7-1, was performed to determine the impact of lowering the Technical Specification minimum indicated pressure from 2225 psia to 2200 psia. For these DBEs, DNBR is the primary criterion; thus, they were re-evaluated to ensure that the transient minimum DNBR does not exceed the design limit of 1.23. The evaluation of these DBEs showed that sufficient initial steady state thermal margin is maintained by the Technical Specification DNB LCOs to assure that the transient minimum DNBR is equal to, or greater, than the design limit of 1.23. Therefore, the conclusions reached in the reference cycle analysis remain valid.

A re-evaluation of all DBEs was also performed to determine the impact of extended burnup. The evaluation showed that effects of extended burnup are no more severe than those reported in Reference 1. Thus, the conclusions reached about extended burnup effects in Reference 1 are valid for Unit 2, Cycle 5.

For the events reanalyzed, Table 7-3 shows the reason for the reanalysis, the acceptance criterion to be used in judging the results and a summary of the results obtained. Detailed presentations of the results of the reanalysis are provided in the appropriate sections.

TABLE 7-1

CALVERT CLIFFS UNIT 2, CYCLE 5
DESIGN BASIS EVENTS CONSIDERED IN THE NON-LOCA SAFETY ANALYSIS

		Analysis Status
7.1	Anticipated Operational Occurrences for which intervention of the RPS is necessary to prevent exceeding acceptable limits:	
7.1.1	Boron Dilution	Not Reanalyzed
7.1.2	Startup of an Inactive Reactor Coolant Pump	Not Reanalyzed[1]
7.1.3	Loss of Load	Reanalyzed
7.1.4	Excess Load	Reanalyzed
7.1.5	Loss of Feedwater Flow	Not Reanalyzed
7.1.6	Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed
7.1.7	Reactor Coolant System Depressurization	Not Reanalyzed
7.1.8	Excessive Charging Event	Analyzed
7.2	Anticipated Operational Occurrences for which RPS trips and/or sufficient initial steady state thermal margin, maintained by the LCOs, are necessary to prevent exceeding the acceptable limits:	
7.2.1	Sequential CEA Group Withdrawal[2]	Re-evaluated
7.2.2	Loss of Coolant Flow[3]	Re-evaluated
7.2.3	Full Length CEA Drop	Re-evaluated
7.2.4	Transients Resulting from the Malfunction of One Steam Generator[4]	Re-evaluated
7.2.5	Loss of AC Power[3]	Re-evaluated
7.3	Postulated Accidents	
7.3.1	CEA Ejection	Reanalyzed
7.3.2	Steam Line Rupture	Reanalyzed
7.3.3	Steam Generator Tube Rupture	Reanalyzed
7.3.4	Seized Rotor[3]	Re-evaluated
7.4	Postulated Occurrences	
7.4.1	Fuel Handling	Reanalyzed

[1]Technical Specifications preclude this event during operation.

[2]Requires High Power and Variable High Power Trip.

[3]Requires Low Flow Trip.

[4]Requires trip on high differential steam generator pressure.

TABLE 7-2

CALVERT CLIFFS UNIT 2, CYCLE 5
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Physics Parameters	Units	Reference Cycle Values (Unit 1, Cycle 6)	Unit 2, Cycle 5 Values
Radial Peaking Factors			
For DNB Margin Analyses (F_r^T)			
Unrodded Region		1.70+,*	1.70+,*
Bank 5 Inserted		1.87+,*	1.87+,*
For Planar Radial Compoent (F_{xy}^T) of 3-D Peak (CTM Limit Analyses)			
Unrodded Region		1.70+,*	1.70*
Bank 5 Inserted		1.87+,*	1.87*
Maximum Augmentation Factor		1.055	1.055
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^{\circ}F$	-2.5**→+.5	-2.5**→+.5
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)	$\%\Delta\rho$	-5.3	-5.2
Tilt Allowance	%	3.0	3.0

*For DNBR and CTM calculations, effects of uncertainties on these parameters were accounted for statistically. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2a, 2b, 2c. These procedures have been approved by NRC for the Calvert Cliffs Units in Reference 3.

**The effective initial MTC assumed for the SLB event is $-2.2\times10^{-4}\Delta\rho/^{\circ}F$.

+The values assumed are conservative with respect to the Technical Specification limits.

TABLE 7-2
(continued)

Safety Parameters	Units	Reference Cycle Values (Unit 1, Cycle 6)	Unit 2, Cycle 5 Values
Power Level	MWt	2700*	2700*
Maximum Steady State Temperature	^oF	548*	548*
Minimum Steady State RCS Pressure	psia	2225*	2200*
Reactor Coolant Flow	10^6lbm/hr	138.5*	138.5*
Negative Axial Shape LCO Extreme Assumed at Full Power (Ex-Cores)	I_p	-.15*	-.15*
Maximum CEA Insertion at Full Power	% Insertion of Bank 5	25	25
Maximum Initial Linear Heat Rate for Transient Other than LOCA	KW/ft	16.0	16.0
Steady State Linear Heat Rate for Fuel CTM Assumed in the Safety Analysis	KW/ft	21.3	22.0
CEA Drop Time from Removal of Power to Holding Coils to 90% Insertion	sec	3.1	3.1
Minimum DNBR (CE-1)		1.23*	1.23*

*For DNBR and CTM calculations, effects of uncertainties on these parameters were accounted for statistically. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2a, 2b, 2c. These procedures have been approved by NRC for the Calvert Cliffs Units in Reference 3.

TABLE 7-3

DESIGN BASIS EVENT REANALYZED FOR UNIT 2, CYCLE 5

Event	Reason for Reanalysis (changes relative to reference cycle)	Acceptance Criterion	Summary of Results
Loss of Load	Increase in initial pressurizer liquid level and decrease in initial pressurizer pressure.	Peak pressure less than upset pressure limit of 2750 psia.	Peak pressure calculated to be 2617 psia. Further details in Section 7.1.3.
Excess Load	Decrease in initial liquid level and decrease in initial pressurizer pressure.	Impact of Reactor Vessel Upper Head voiding on reactor coolant circulation and DNBR limits not exceeded.	Results acceptable. Further details in Section 7.1.4.
Excessive Charging	Increase in initial pressurizer liquid level.	Time to fill pressurizer.	Results show that operator has at least 15 minutes from the initiation of high pressurizer level alarm to terminate the event. Further details in Section 7.1.8.
CEA Ejection	Lower available scram worth at trip. Changes in post ejected 3-D peak and ejected CEA worth.	Fuel failure small fraction of 10CFR100.	Results show that no pin experiences clad damage. Further details in Section 7.3.1.
Steam Line Rupture	Changes in moderator cooldown curve and available scram worth at trip.	Post trip DNBR greater than 1.3 with MacBeth correlation.	Results show minimum DNBR is equal to 1.35 for HFP and 2.0 for HZP. Further details in Section 7.3.2.

TABLE 7-3
(continued)

Event	Reason for Reanalysis	Acceptance Criterion	Summary of Results
	(changes relative to reference cycle)		
Steam Generator Tube Rupture	Increase in initial pressurizer liquid level.	Site boundary dose less than 10CFR100 limits.	Results acceptable. Further details in Section 7.3.3.
Fuel Handling	High burnup.	Site boundary dose less than 10CFR100 limits.	Results acceptable. Further details in Section 7.4.1.

7.1.3 Loss of Load Event

The Loss of Load event was reanalyzed for Cycle 5 to determine that the RCS pressure upset limit of 2750 psia is not exceeded. The transient DNBR was also evaluated to ensure that the results are within the design limit of 1.23.

The methods used to analyze this event are consistent with those reported in the reference cycle (Reference 1).

The assumptions used to maximize RCS pressure during the transient are:

a. The event is assumed to result from the sudden closure of the turbine stop valves without a simultaneous reactor trip. This assumption causes the greatest reduction in the rate of heat removal from the reactor coolant system and thus results in the most rapid increase in primary pressure and the closest approach to the RCS pressure upset limit.

b. The steam dump and bypass system, the pressurizer spray system, and the power operated pressurizer relief valves are assumed not to be operable. This too maximizes the primary system pressure reached during the transient.

The Loss of Load event was initiated at the conditions shown in Table 7.1.3-1. The combination of parameters shown in Table 7.1.3-1 maximizes the calculated peak RCS pressure. As can be inferred from the table, the key parameters for this event are the initial primary and secondary pressures and the moderator and fuel temperature coefficients of reactivity.

The initial core average axial power distribution for this analysis was assumed to be a bottom peaked shape. This distribution is assumed because it minimizes the negative reactivity inserted during the initial portion of the scram following a reactor trip and maximizes the time required to mitigate the pressure and heat flux increases. The Moderator Temperature Coefficient (MTC) of $+0.5\times10^{-4}$ $\Delta\rho/^{\circ}F$ was assumed in this analysis. This MTC, in conjunction with the increasing coolant temperatures, maximizes the rate of change of heat flux and the pressure at the time of reactor trip. A Fuel Temperature Coefficient (FTC) corresponding to beginning of cycle conditions was used in the analysis. This FTC causes the least amount of negative reactivity feedback to mitigate the transient increases in both the core heat flux and the pressure. The uncertainty on the FTC used in the analysis is shown in Table 7.1.3-1. The lower limit on initial RCS pressure is used to maximize the rate of change of pressure, and thus peak pressure, following trip.

The Loss of Load event, initiated from the conditions given in Table 7.1.3-1, results in a high pressurizer pressure trip signal at 6.8 seconds. At 10.1 seconds, the primary pressure reaches its maximum value of 2617 psia. The increase in secondary pressure is limited by the opening of the main steam safety valves, which open at 5.8 seconds. The secondary pressure reaches its maximum value of 1047 psia at 10.6 seconds after initiation of the event.

Table 7.1.3-2 presents the sequence of events for this event. Figures 7.1.3-1 to 7.1.3-4 show the transient behavior of power, heat flux, RCS pressure, and RCS coolant temperatures.

The event was also reanalyzed with the initial conditions listed in Table 7.1.3-3 to determine that the acceptable DNBR limit is not exceeded. The minimum transient DNBR calculated for the event is 1.34, compared to the design limit of 1.23.

The results of this analysis demonstrate that during a Loss of Load event the peak RCS pressure and the minimum DNBR do not exceed their respective design limits.

TABLE 7.1.3-1

KEY PARAMETERS ASSUMED IN THE LOSS OF LOAD ANALYSIS
TO MAXIMIZE CALCULATED RCS PEAK PRESSURE

Parameter	Units	Reference* Cycle	Cycle 5
Initial Core Power Level	MWt	2754	2754
Initial Core Inlet Coolant Temperature	^oF	550	550
Core Coolant Flow	$x10^6$lbm/hr	133.9	133.9
Initial Reactor Coolant System Pressure	psia	2200^+	2154^{++}
Initial Pressurizer Liquid Level at Full Power	ft^3	800	975
Initial Steam Generator Pressure	psia	864	864
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^oF$	+0.5	+0.5
Doppler Coefficient Multiplier		0.85	0.85
CEA Worth at Trip	$\%\Delta\rho$	-4.7	-4.7
Time to 90% Insertion of Scram Rods	sec	3.1	3.1
Reactor Regulating System	Operating Mode	Manual	Manual
Steam Dump and Bypass	Operating Mode	Inoperative	Inoperative

*Unit 1, Cycle 6 (Reference 1)

$^+$Corresponds to Technical Specification minimum indicated pressure of 2225 psia. The value includes an uncertainty of 25 psia.

$^{++}$Corresponds to Technical Specification minimum indicated pressure of 2200 psia. The value includes an uncertainty of 46 psia.

TABLE 7.1.3-2

SEQUENCE OF EVENTS FOR THE LOSS OF LOAD EVENT
TO MAXIMIZE CALCULATED RCS PEAK PRESSURE

Time (sec)	Event	Setpoint or Value
0.0	Loss of Secondary Load	---
5.8	Steam Generator Safety Valves Open	1000 psia
6.8	High Pressurizer Pressure Analysis Trip Setpoint is Reached	2422 psia
8.2	CEAs Begin to Drop Into Core	---
7.9	Pressurizer Safety Valves Open	2500 psia
10.1	Maximum RCS Pressure	2617 psia
10.6	Maximum Steam Generator Pressure	1047 psia
12.3	Pressurizer Safety Valves are Fully Closed	2500 psia

TABLE 7.1.3-3

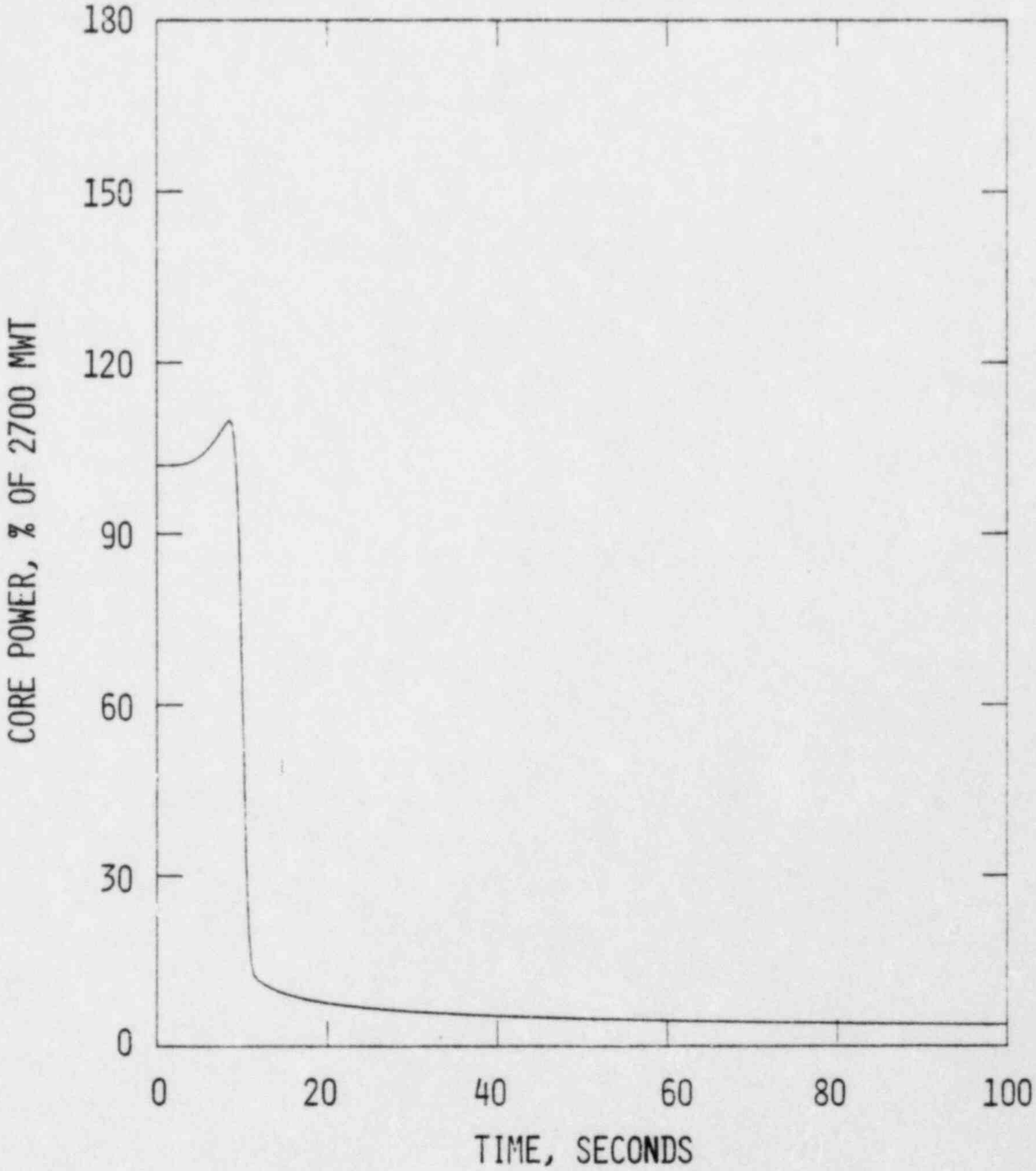
KEY PARAMETERS ASSUMED IN THE LOSS OF LOAD ANALYSIS
TO CALCULATE TRANSIENT MINIMUM DNBR

Parameter	Units	Reference* Cycle	Unit 2, Cycle 5
Initial Core Power Level	MWt	2700**	2700**
Initial Core Inlet Coolant Temperature	^{o}F	548**	548**
Core Coolant Flow	$x10^{6}$ lbm/hr	138.5**	138.5**
Initial Reactor Coolant System Pressure	psia	2225**	2200**
Initial Steam Generator Pressure	psia	864	864
Integrated Radial Peaking Factors, F_r^T (Bank 5 Inserted 25%)		1.75**,+	1.75**,+
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^{o}F$	+0.5	+0.5
Doppler Coefficient Multiplier		0.85	0.85
CEA Worth at Trip	$\%\Delta\rho$	-4.7	-4.7
Time to 90% Insertion of Scram Rods	sec	3.1	3.1
Reactor Regulating System	Operating Mode	Manual	Manual
Steam Dump and Bypass System	Operating Mode	Inoperative	Inoperative

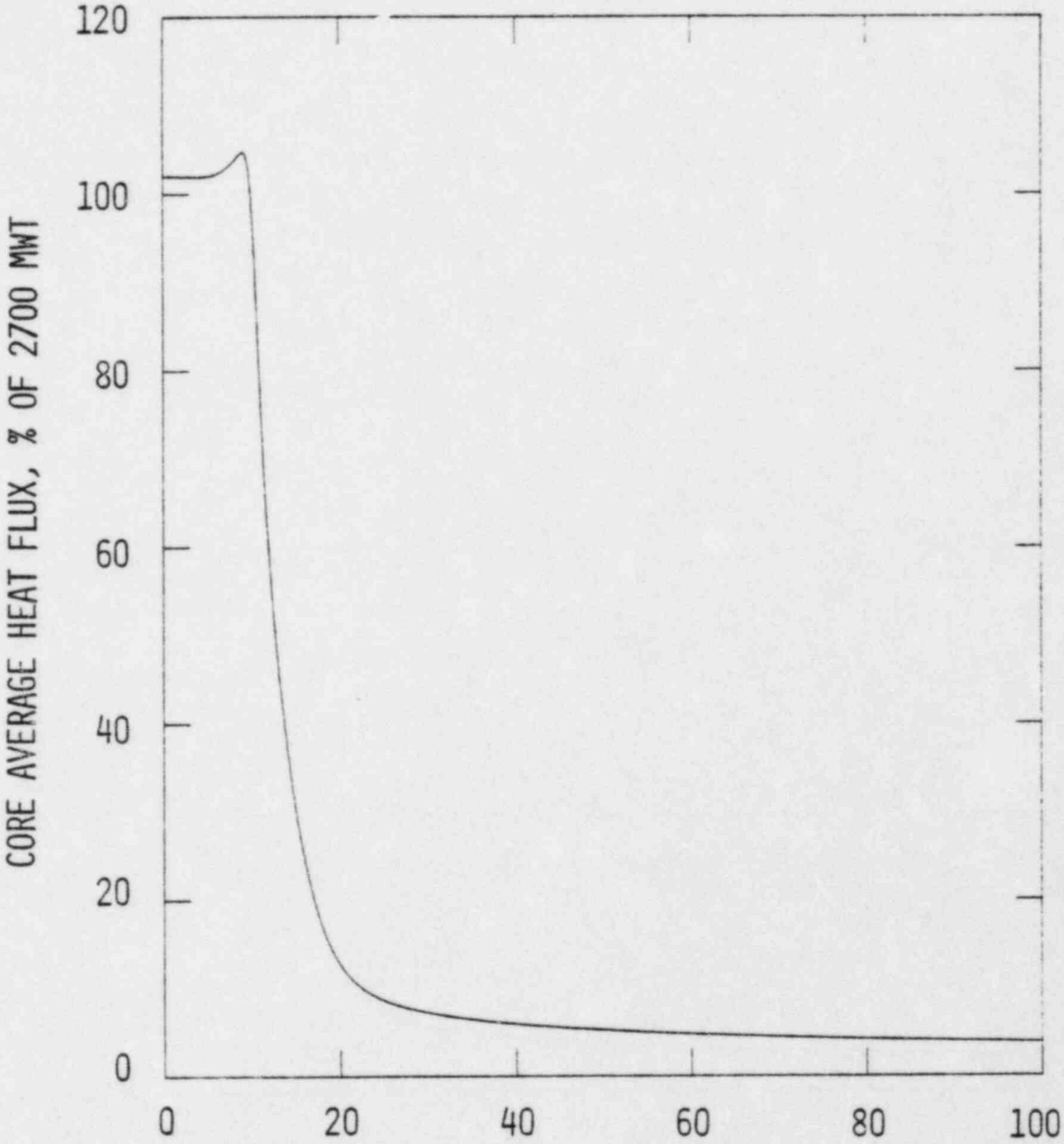
*Unit 1, Cycle 6 (Reference 1)

**Effects of uncertainties on these parameters were accounted for statistically. (See Reference 2)

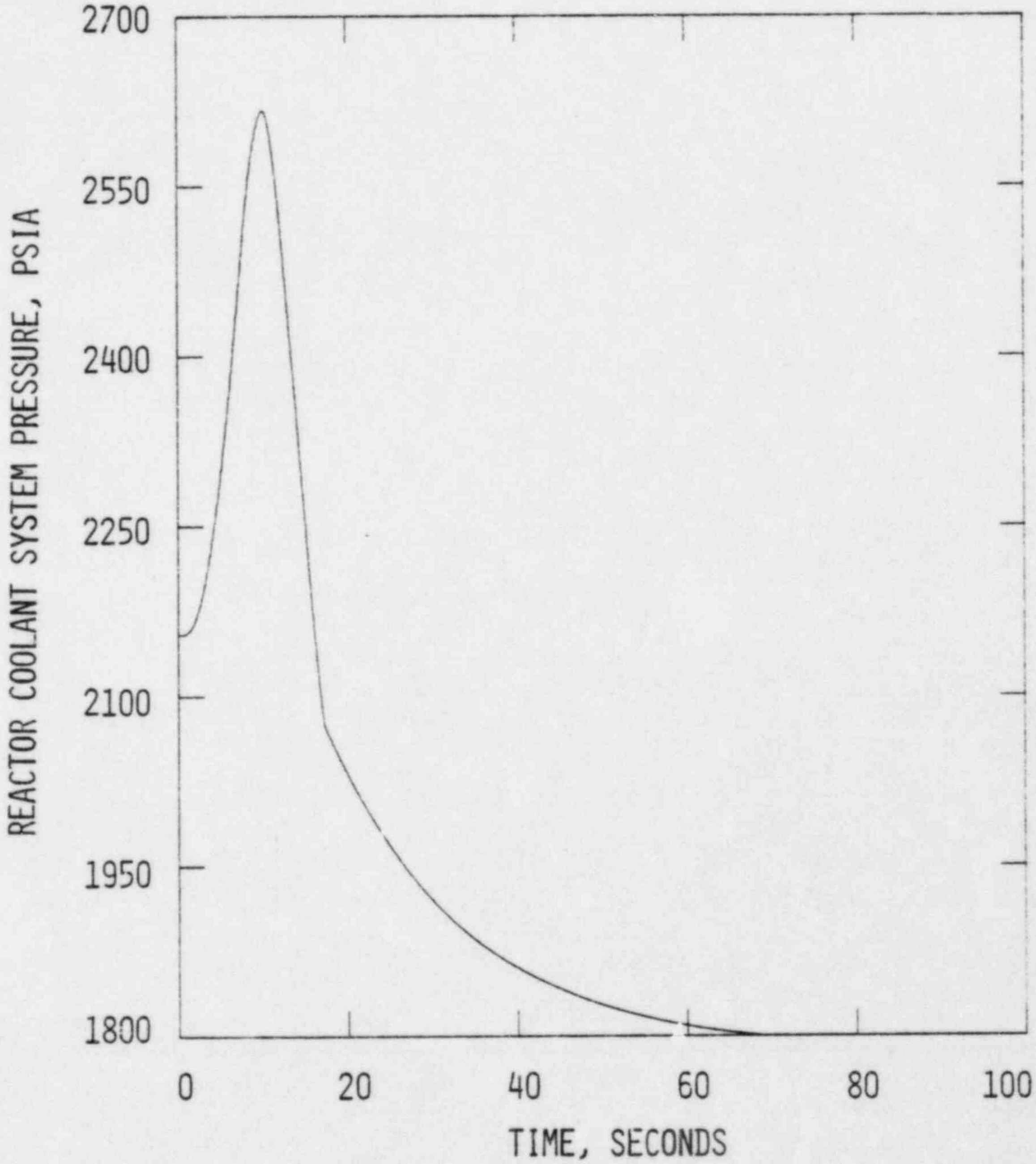
+The values assumed are conservative with respect to the Technical Specification limits.



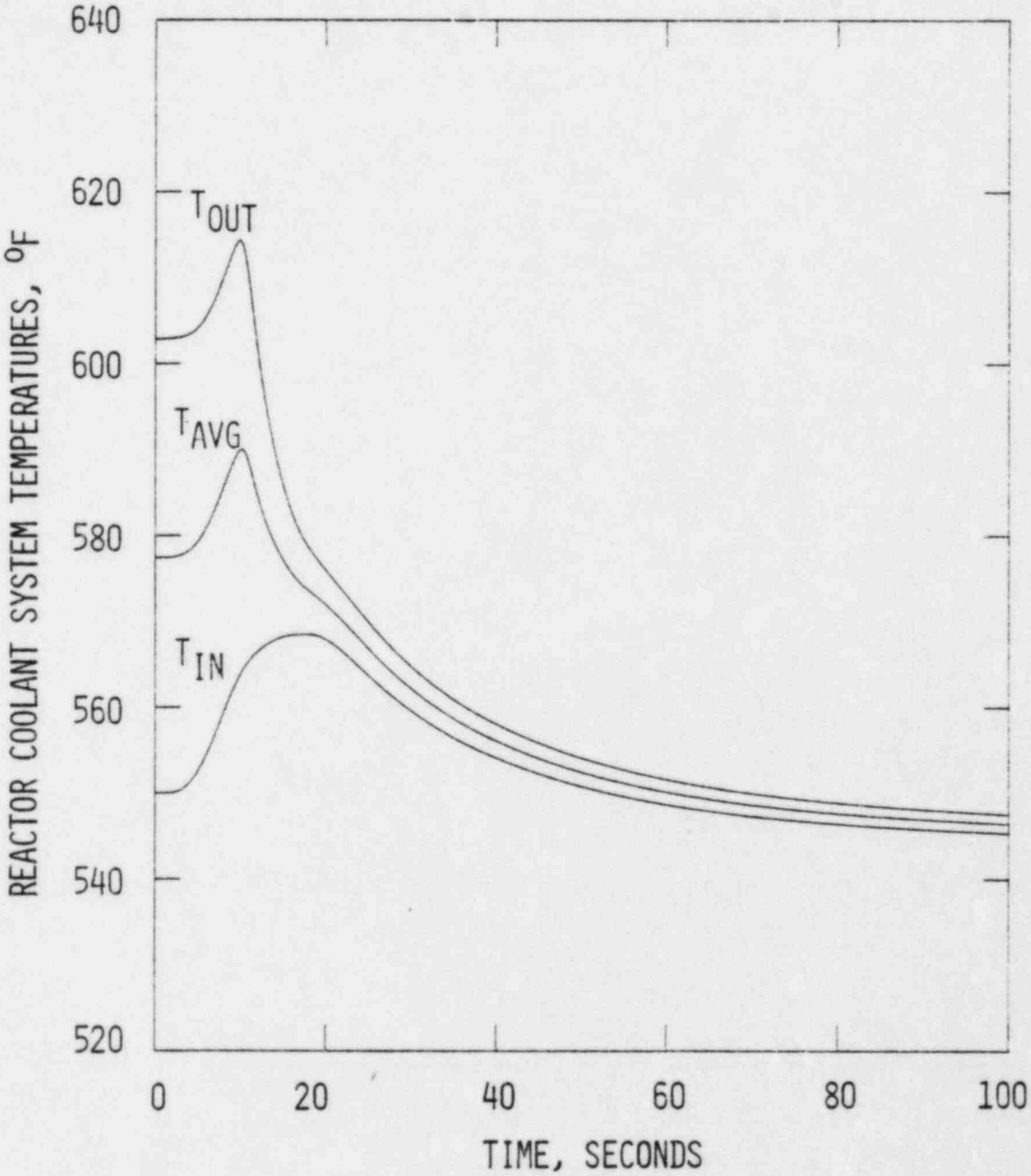
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT CORE POWER VS TIME	FIGURE 7.1.3-1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT CORE AVERAGE HEAT FLUX VS TIME	FIGURE 7.1.3-2



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7.1.3-3



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7.1.3-4

7.1.4 Excess Load Event

The Excess Load event was reanalyzed for Unit 2, Cycle 5 to evaluate the impact of initiating the event from a pressurizer level of 270 ft^3. The analysis evaluated the impact of reactor vessel upper head (RVUH) voiding on reactor coolant circulation and fuel design limits. The analysis included the automatic initiation of auxiliary feedwater three minutes after initiation of reactor trip signal and a manual trip of the reactor coolant pumps (RCPs) following a safety injection actuation signal (SIAS) due to low pressurizer pressure. The RCP coastdown results in a proportionately reduced RVUH flow until natural circulation is established; at that time all flow to the RVUH is assumed to terminate.

The Excess Load event is initiated by the instantaneous opening of steam dump and bypass valves which have a combined capacity of 45% of nominal full power steam flow. This Excess Load event persists until the steam generators are isolated on steam generator isolation signal (SGIS) due to low secondary pressure. The full power event maximizes primary cooldown, shrinkage and consequently RVUH voiding.

The magnitude of RVUH voiding at full power is significantly greater than at zero power because of the higher RVUH coolant temperatures and the larger primary coolant shrinkage that occurs at the higher system temperatures. Therefore, only the full power excess load transient results are described herein.

The key parameters assumed in the analysis to maximize RVUH is given in Table 7.1.4-1. The key parameters assumed to minimize DNBR is given in Table 7.1.4-2.

The analysis to maximize RVUH voiding conservatively assumed a Moderator Temperature Coefficient of $+0.5\times10^{-4}\Delta\rho/^{\circ}F$. This MTC in combination with decreasing coolant temperatures inserts negative reactivity and causes the core power to decrease. The decreasing core power does not allow either the High Power trip or TM/LP trip to be initiated and thus the time of reactor trip is delayed until a low pressurizer pressure trip (i.e., floor of the TM/LP trip) is generated. The longer time required to initiate reactor trip causes the pressurizer to drain and thus maximizes RVUH voiding.

The analysis conservatively assumed that all three charging pumps were inoperable and that one High Pressure Safety Injection (HPSI) pump fails to start on SIAS due to low pressurizer pressure.

The effect of auxiliary feedwater was explicitly evaluated by analyzing the event both with and without auxiliary feedwater initiated three minutes after reactor trip signal is generated. An auxiliary feedwater flow of 172 lbm/sec to each steam generator is conservatively assumed (i.e., 10.5% of full power main feedwater flow per generator). The maximum auxiliary feedwater flow causes the fastest primary cooldown and thus enhances the bubble formation in the upper head.

This analysis shows that some RVUH voiding will occur as a result of the RCS depressurization caused by an Excess Load transient. Voiding in the RVUH starts when RCS pressure control is lost due to the primary coolant shrinkage which drains the pressurizer. During the period of RVUH voiding, RCS pressure is controlled by the saturation pressure within the upper head. This reduces the RCS depressurization in the latter part of the transient as compared to analyses which do not explicitly model RVUH voids.

RVUH coolant temperatures initially follow the core outlet temperature and therefore decrease immediately following a reactor trip. The decrease in RVUH coolant flow reduces the convective heat transfer and inhibits upper head cooldown. This leads to elevated RVUH coolant temperatures which raise the upper head saturation pressure and therefore increase RCS pressure during the period of voiding. The increased RCS pressure does not adversely impact the approach to any SAFDL; however, safety injection flow is decreased. The reduced safety injection flow does not result in a return to criticality. However, the decreased flow diminishes the mitigating effect of safety injection on coolant shrinkage and, therefore, enhances voiding. Subsequent RVUH cooldown is accomplished through an exchange of coolant between the RVUH and the core outlet plenum. This exchange of coolant is driven by the expansion and contraction of the steam bubble. Additional RVUH cooling is accomplished through heat conduction across the upper guide structure.

At the time of maximum RVUH voiding approximately 63% of the head is occupied by steam. Since this steam bubble does not expand beyond the upper head, primary coolant circulation is unaffected. Tables 7.1.4-3 and 7.1.4-4 present the sequence of events for the event initiated without and with auxiliary feedwater flow. Figures 7.1.4-1 through 7.1.4-14 present the transient behavior of the system variables during the event. The analysis demonstrates that the addition of auxiliary feedwater prolongs the duration of RVUH voiding and delays repressurization of the RCS. However, since auxiliary feedwater is delivered after the time of maximum voiding, the peak void fraction is unchanged.

The Excess Load event initiated from the conditions given in Table 7.1.4-2 resulted in a minimum DNBR of 1.46 compared to the design limit of 1.23 (CE-1 correlation).

In conclusion, the potential RVUH voiding associated with an Excess Load transient initiated from a pressurizer level of 270 ft^3 does not extend beyond the upper head and therefore will not affect primary coolant circulation. In addition, approach to SAFDLs are not impacted by RVUH voiding. The results of the analysis also shows that the NSSS achieves stable conditions and that shutdown cooling procedures can be initiated if deemed necessary.

TABLE 7.1.4-1

KEY PARAMETERS ASSUMED IN
THE EXCESS LOAD EVENT ANALYSIS TO MAXIMIZE RVUH VOIDING

Parameter	Units	Value
Initial Core Power Level	MWt	2754
Core Inlet Temperature	^{o}F	550
Reactor Coolant System Pressure	psia	2300
Core Mass Flow Rate	$x10^{6}$ lbm/hr	133.6
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^{o}F$	+.5
CEA Worth Available at Trip	$\%\Delta\rho$	-4.7
Auxiliary Feedwater Flow Rate	lbm/sec	172.0/S.G.
Low Pressurizer Pressure Analysis Trip Setpoint	psia	1728
SIAS Analysis Setpoint	psia	1556
SGIS Analysis Setpoint	psia	548
Initial Pressurizer Level	ft^3	270

TABLE 7.1.4-2

KEY PARAMETERS ASSUMED IN THE EXCESS LOAD EVENT ANALYSIS
TO CALCULATE TRANSIENT MINIMUM DNBR

Parameter	Units	Reference Cycle*	Unit 2, Cycle 5
Initial Core Power Level	MWt	2700+	2700+
Core Inlet Temperature	^oF	548+	548+
Reactor Coolant System Pressure	psia	2225+	2200+
Core Mass Flow Rate	$x10^6$lbm/hr	138.5+	138.5+
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^oF$	-2.5	-2.5
CEA Worth Available at Trip	$\%\Delta\rho$	-4.3	-4.3
Doppler Multiplier		0.85	0.85
Inverse Boron Worth	PPM/%	105	105
Auxiliary Feedwater Flow Rate	lbm/sec	175.0/S.G.	175.0/S.G.
High Power Level Trip Setpoint	% of full power	110	110
Low S.G. Water Level Trip Setpoint	ft	30.9	30.9
RTD Response Time	sec	12.0	12.0

*Reference cycle is Unit 1 Cycle 6, Reference 1.

+For DNBR calculations, effects of uncertainties on these parameters were combined statistically (see Reference 2).

TABLE 7.1.4-3

SEQUENCE OF EVENTS FOR THE EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER

Time (sec)	Event	Setpoint or Value
0.0	Steam Dump and Bypass Valves Fully Open	---
24.2	Pressurizer Empties	---
28.9	Low Pressurizer Pressure Analysis Trip Setpoint is Reached	1728 psia
29.8	Trip Breakers Open	
30.3	CEAs Begin to Drop Into Core Feedwater Starts Rampdown	---
32.6	SIAS is Initiated Reactor Coolant Pumps Manually Tripped	1556 psia
67.1	SGIS is Generated	548 psia
68.0	Main Steam Isolation Valves Begin to Close	---
80.0	Main Steam Isolation Valves are Closed	---
90.3	Feedwater Rampdown to 5% is Completed	
107.6	Maximum RVUH Void is Reached	63%
210.3*	Main Feedwater Isolated	
521.6	Minimum RCS Pressure	728.50
717.8	Upper Head Void is Zero Pressurizer Starts to Refill	

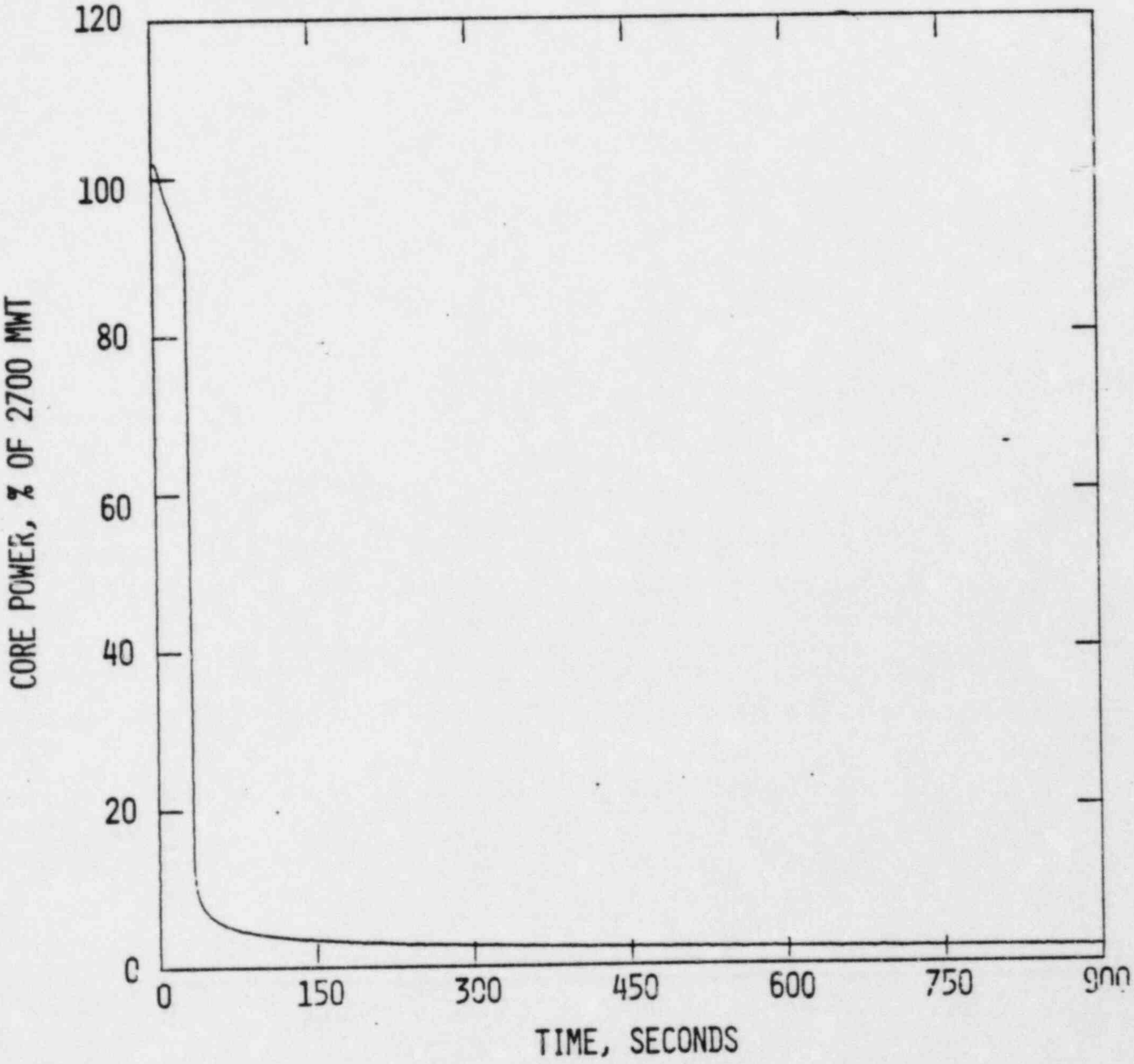
*Main feedwater would have been isolated 80 seconds after SGIS is initiated (i.e., at 147.1 seconds). The analysis conservatively assumed that main feedwater is isolated at 210.3 seconds to prolong the duration of RVUH voiding.

TABLE 7.1.4-4

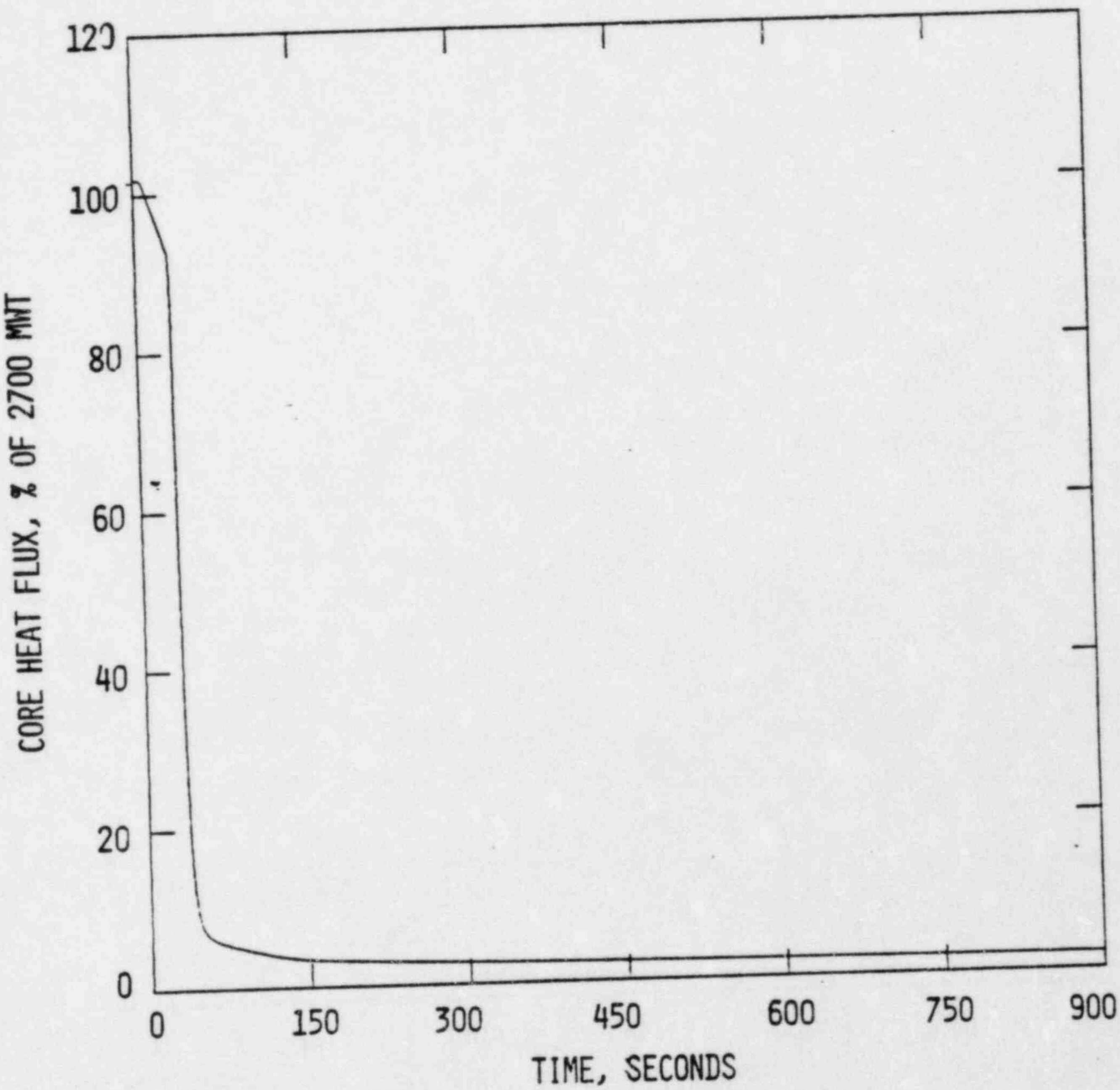
SEQUENCE OF EVENTS FOR THE EXCESS LOAD EVENT
WITH AUXILIARY FEEDWATER

Time (sec)	Event	Setpoint or Value
0.0	Steam Dump and Bypass Valves Fully Open	---
24.2	Pressurizer Empties	---
28.9	Low Pressurizer Pressure Analysis Trip Setpoint is Reached	1728 psia
29.8	Trip Breakers Open	
30.3	CEAs Begin to Drop Into Core Feedwater Starts Rampdown Turbine Valves Begin to Close	---
32.6	SIAS is Initiated Reactor Coolant Pumps Manually Tripped	1556 psia
67.1	SGIS is Generated	548 psia
68.0	Main Steam Isolation Valves Begin to Close	
80.0	Main Steam Isolation Valves are Closed	
90.3	Feedwater Rampdown to 5% is Completed	
107.6	Maximum RVUH Void is Reached	63%
210.3*	Main Feedwater Isolated Auxiliary Feedwater is Initiated	172 lbm/S.G.
900.0	Operator Action To Isolate Auxilary Feedwater to Steam Generators	

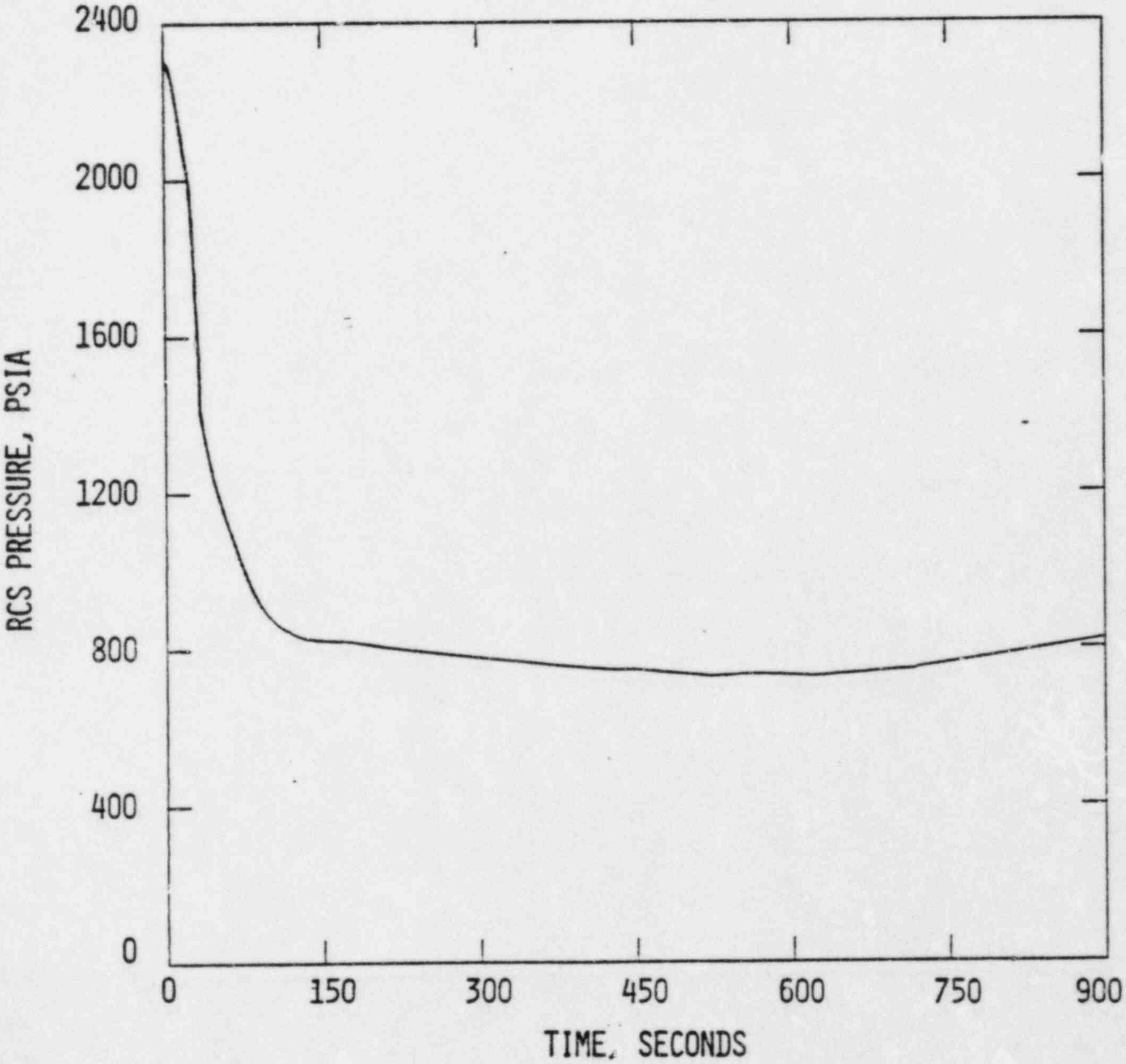
*Main feedwater would have been isolated 80 seconds after SGIS is initiated (i.e., at 147.1 seconds). The analysis conservatively assumed that main feedwater is isolated at 210.3 seconds to prolong the duration of RVUH voiding.



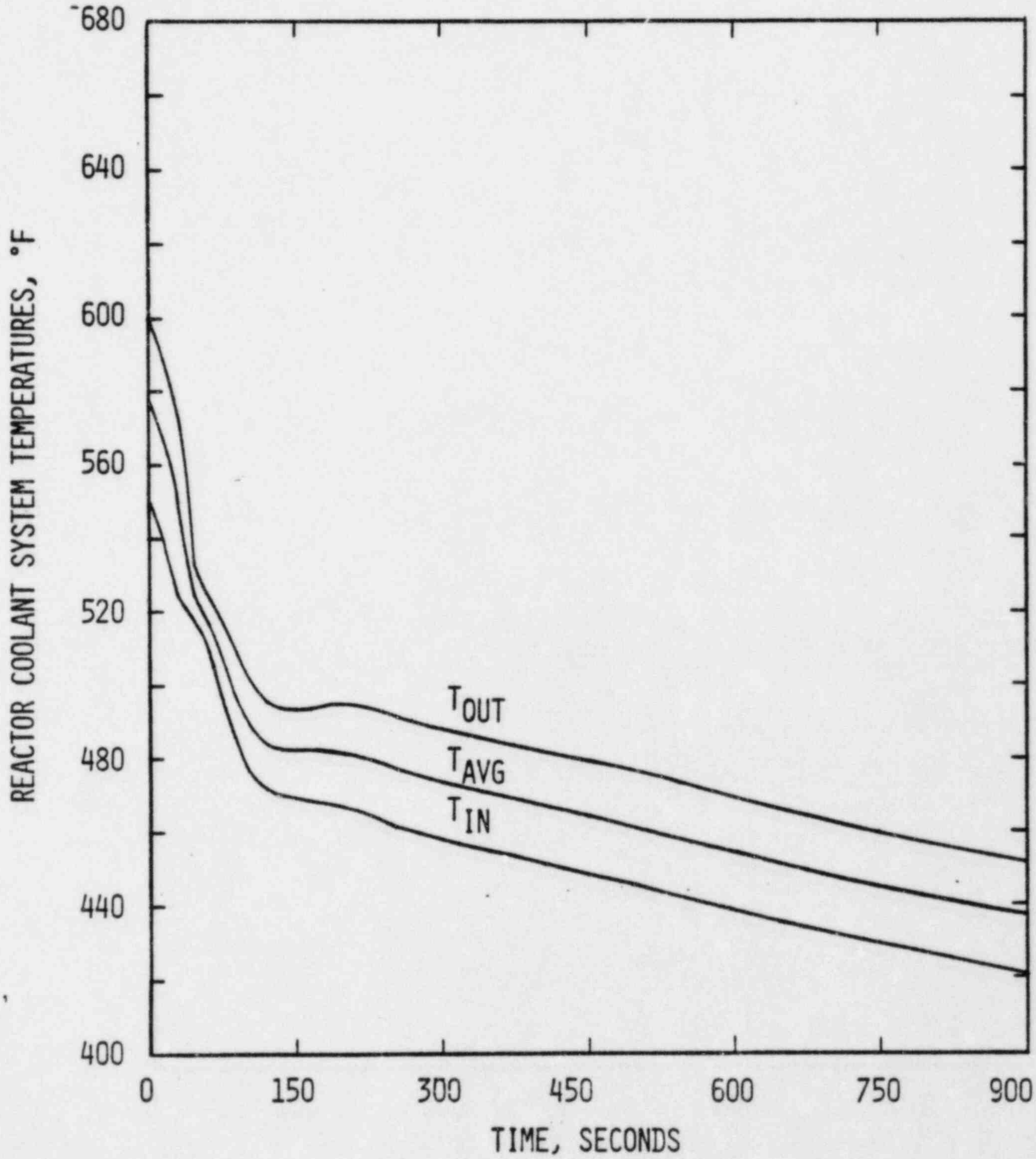
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER CORE POWER VS TIME	FIGURE 7.1.4-1



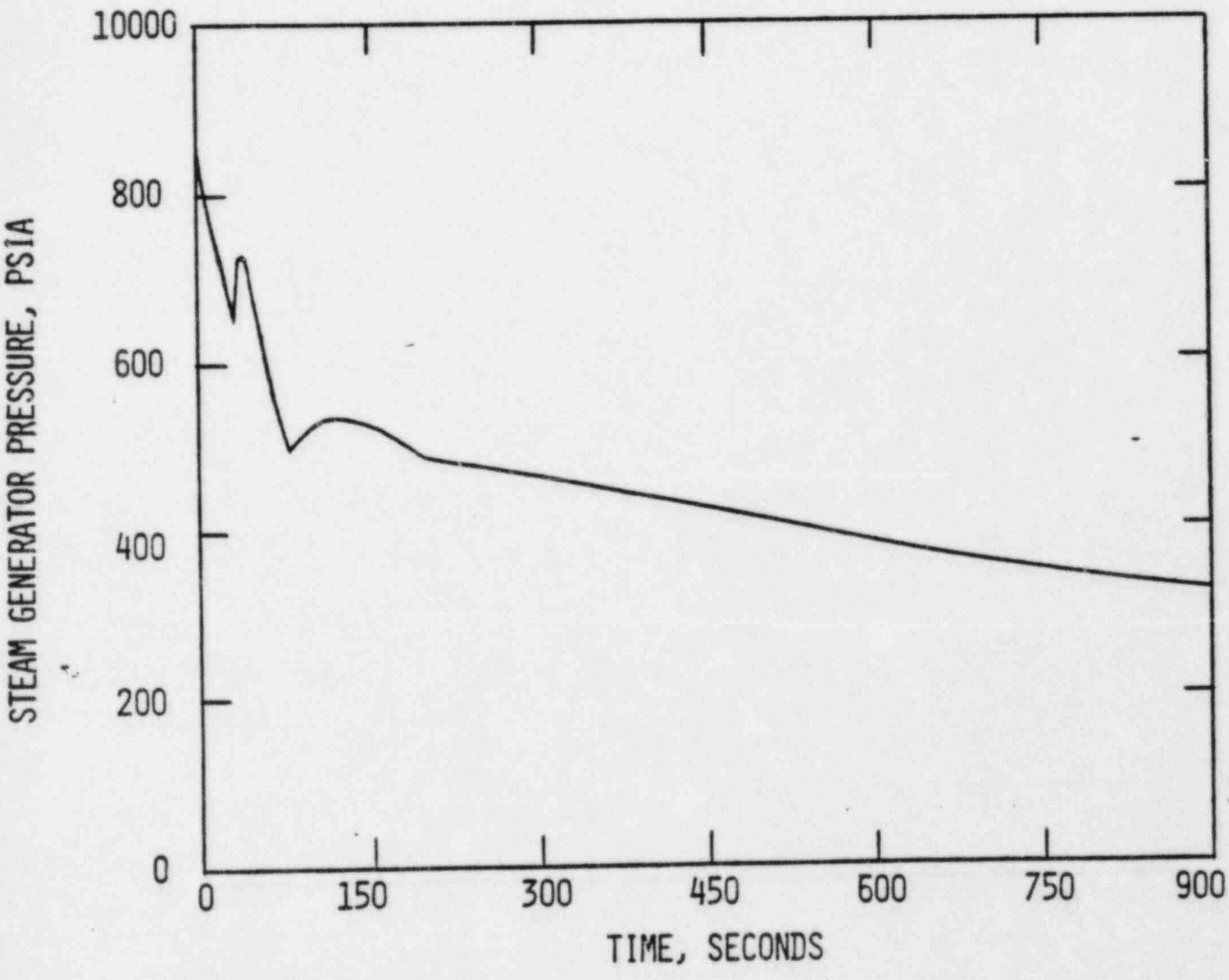
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER CORE HEAT FLUX VS TIME	FIGURE 7.1.4-2



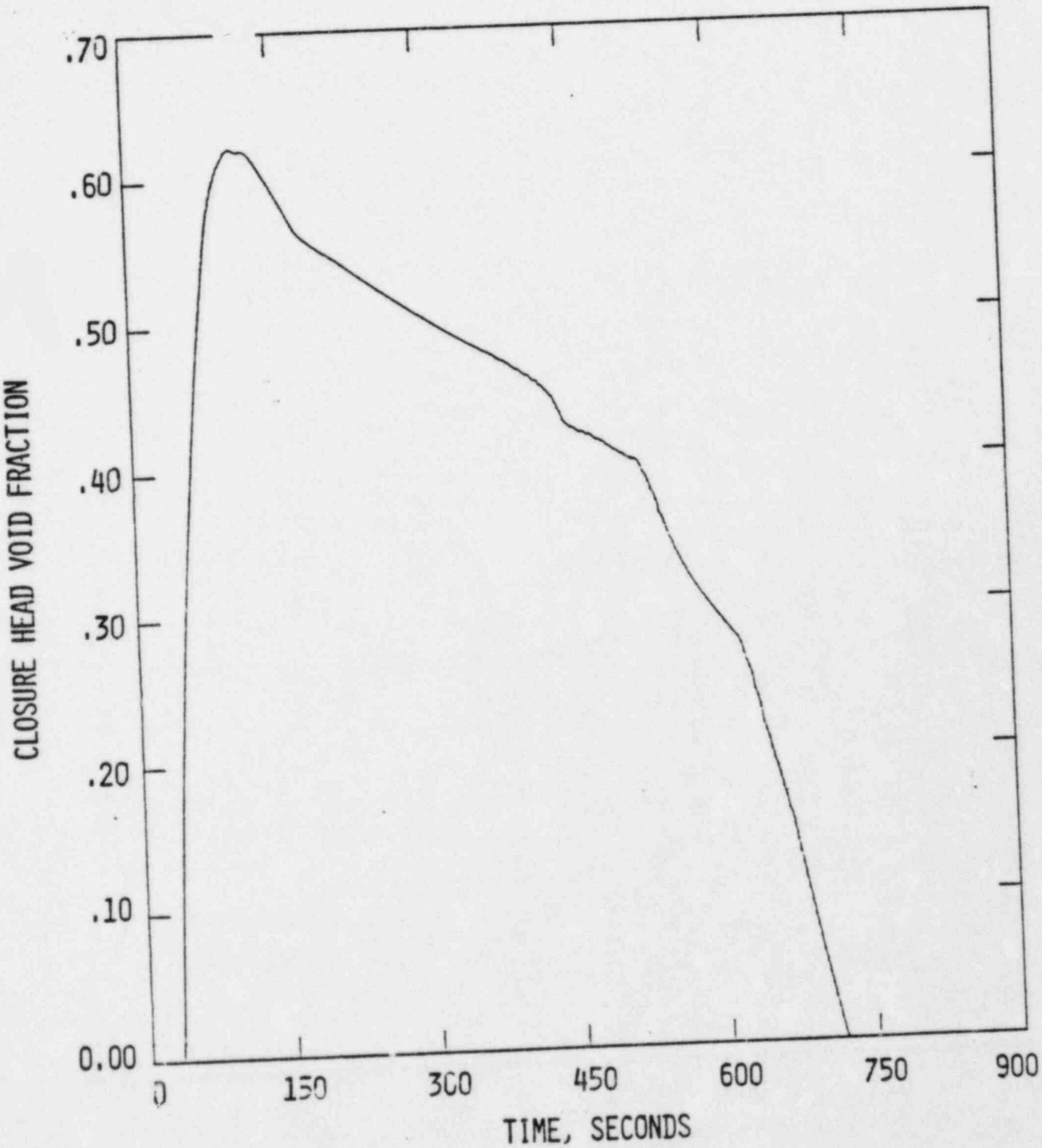
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER RCS PRESSURE VS TIME	FIGURE 7.1.4-3



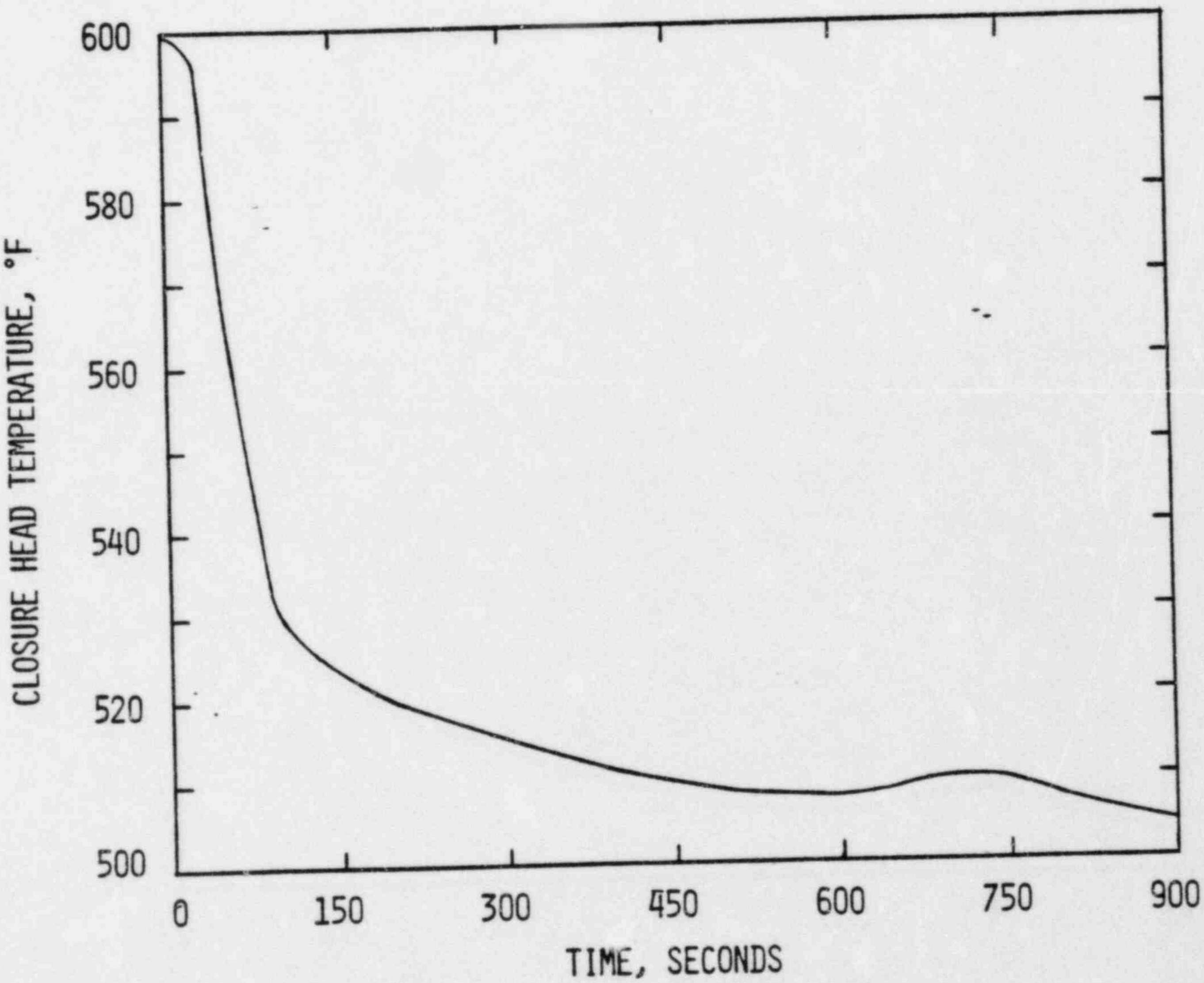
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7.1.4-4



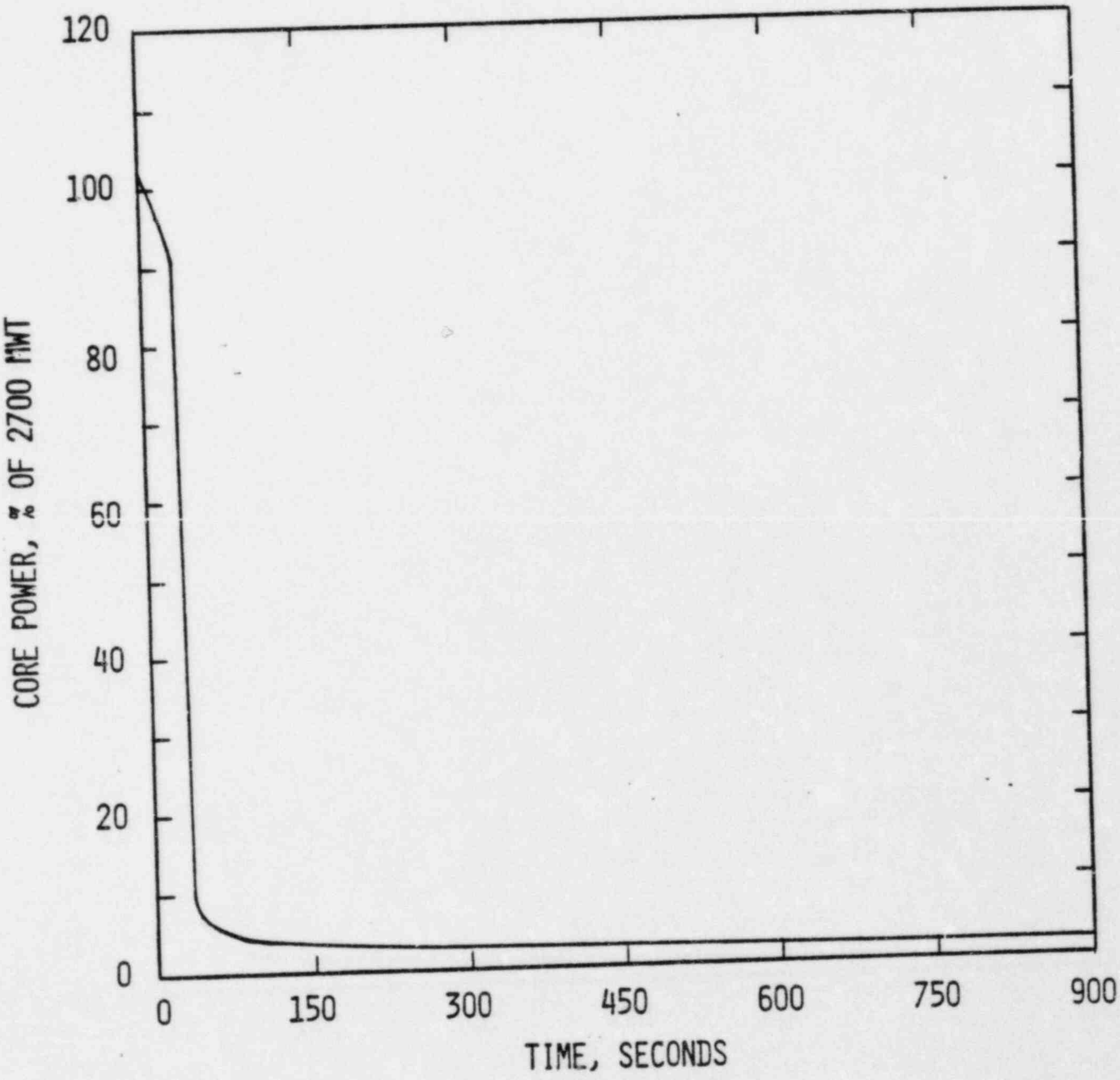
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER STEAM GENERATOR PRESSURE VS TIME	FIGURE 7.1.4-5



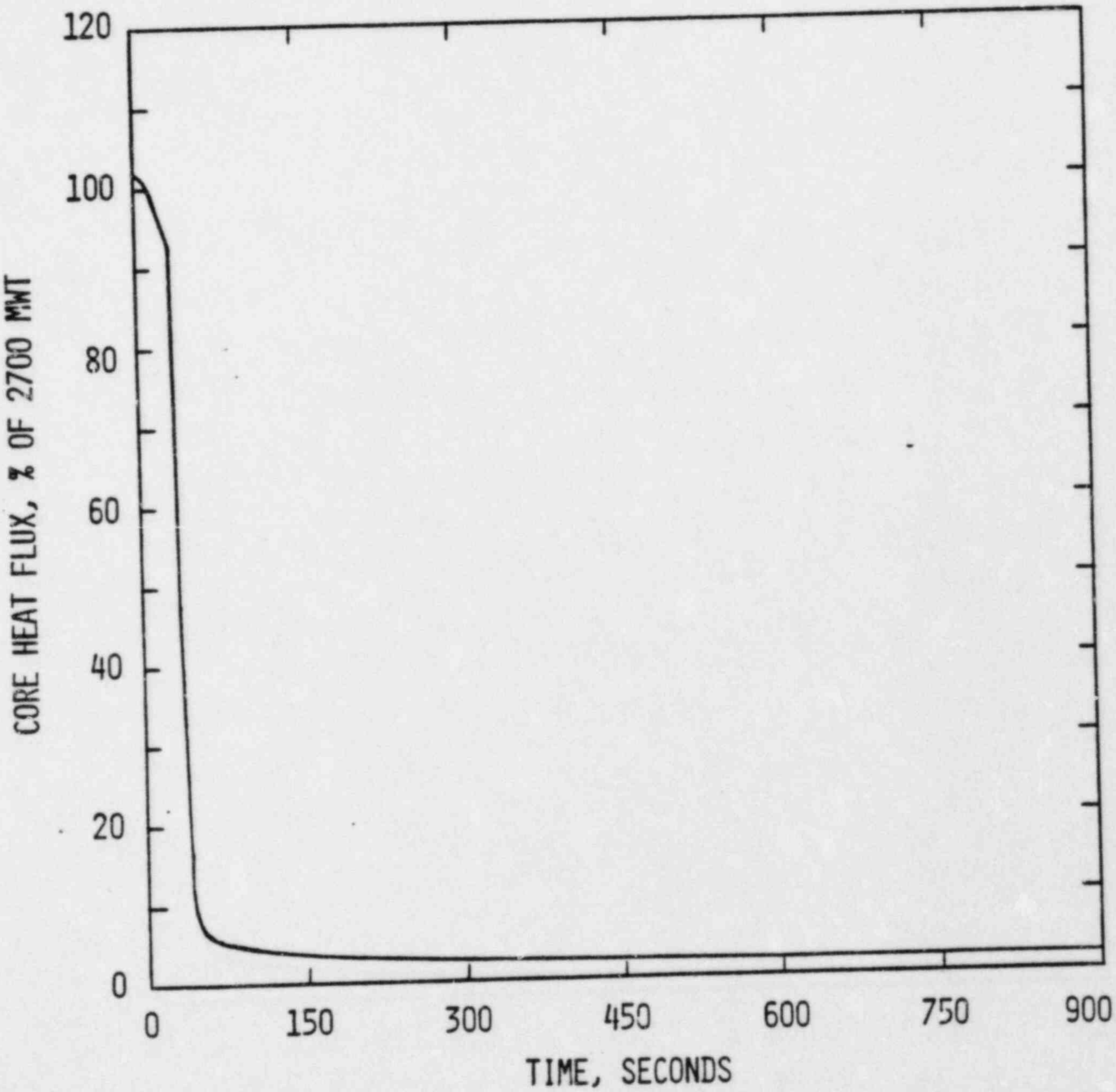
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER CLOSURE HEAD VOID FRACTION VS TIME	FIGURE 7.1.4-6



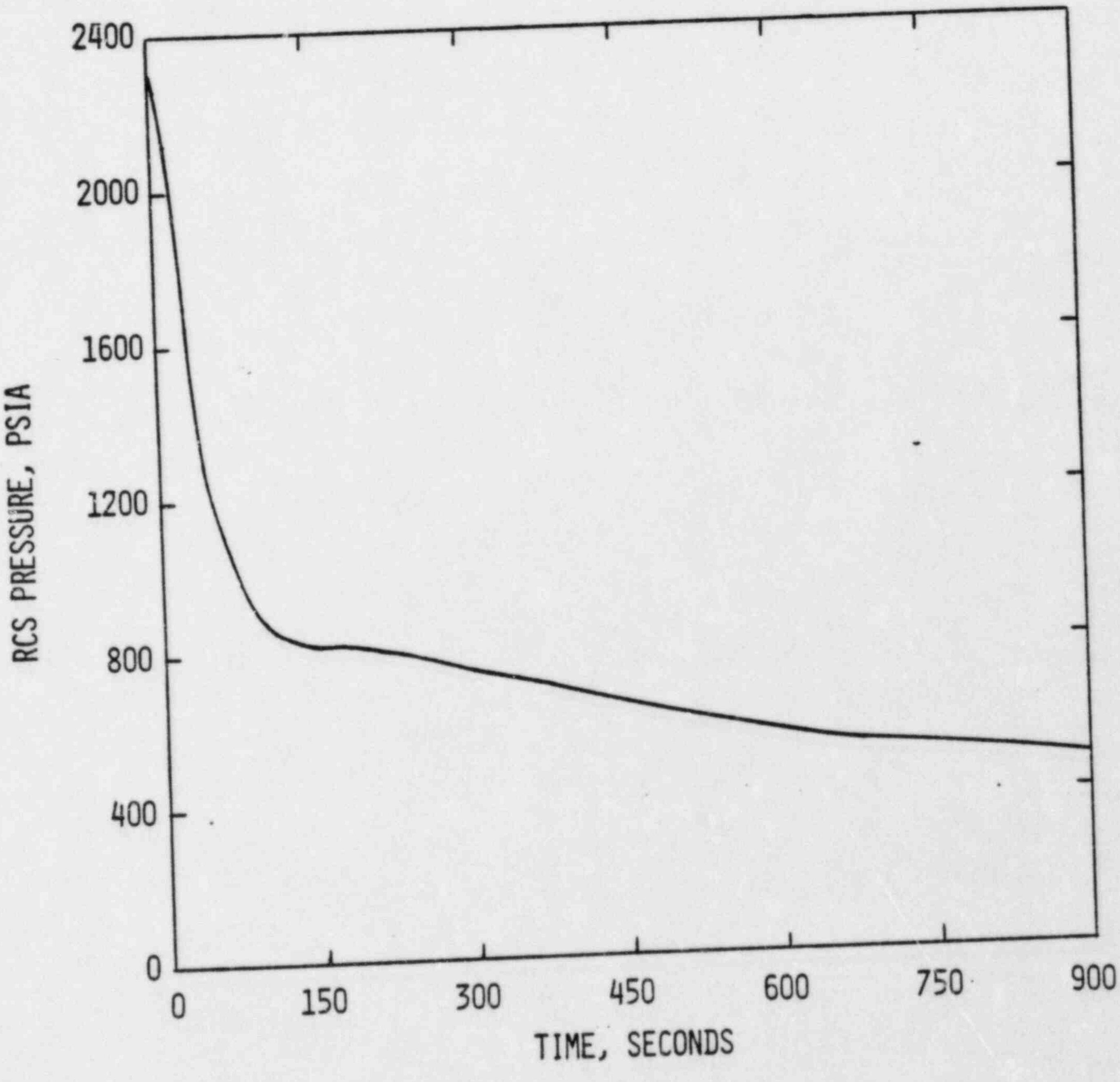
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER CLOSURE HEAD TEMPERATURE VS TIME	FIGURE 7.1.4-7



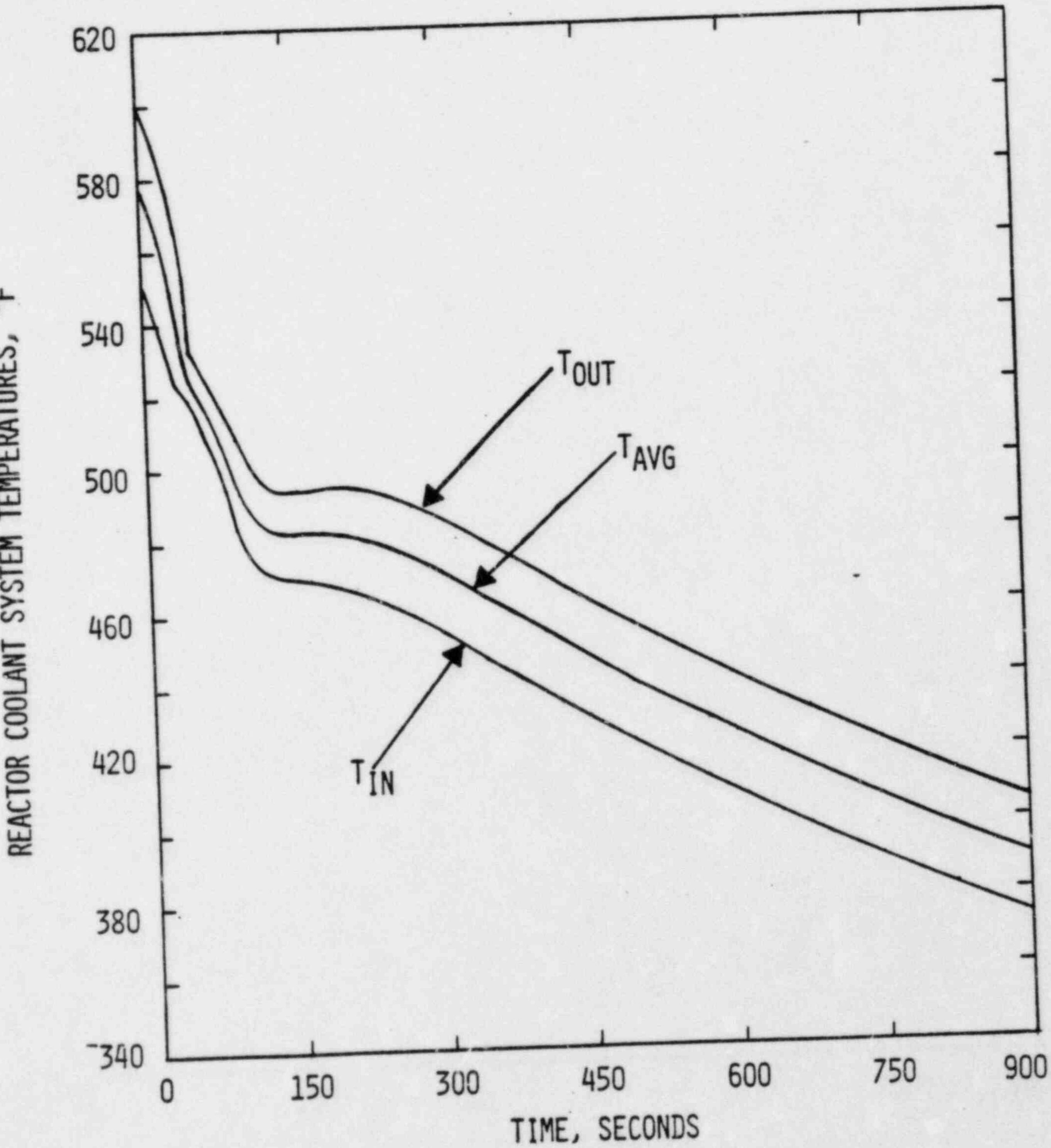
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER CORE POWER VS TIME	FIGURE 7.1.4-8



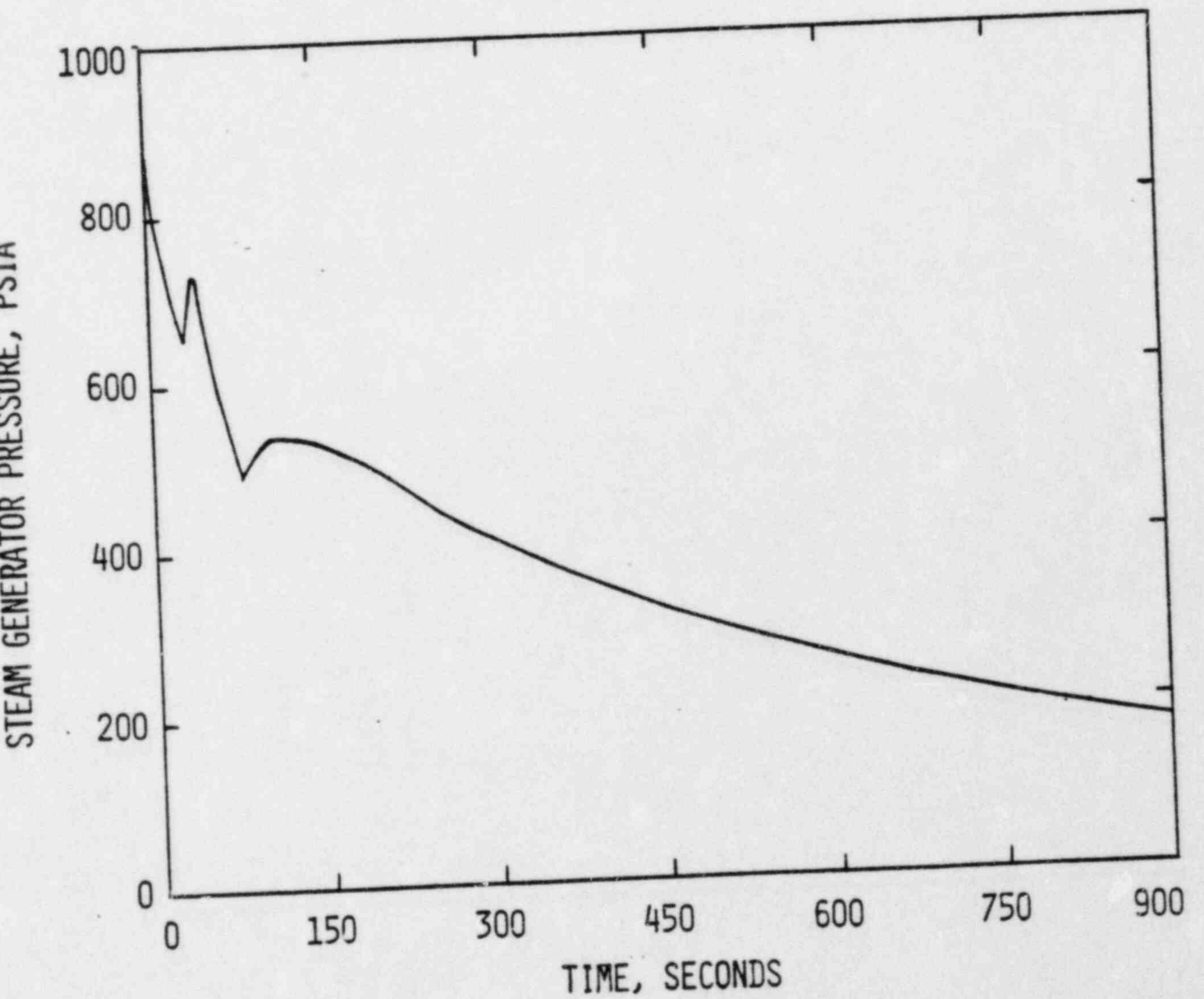
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER CORE HEAT FLUX VS TIME	FIGURE 7.1.4-9



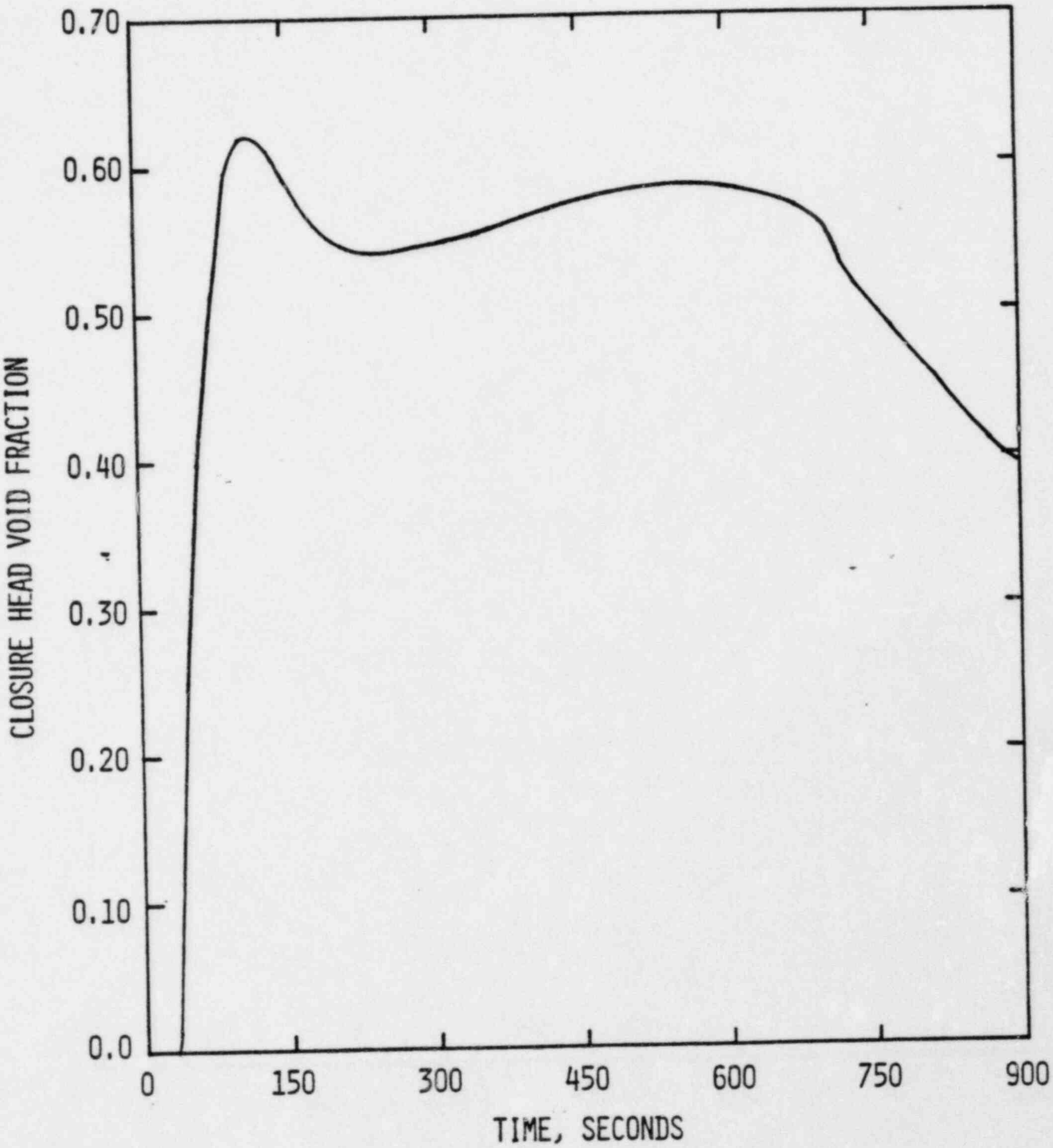
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER RCS PRESSURE VS TIME	FIGURE 7.1.4-10



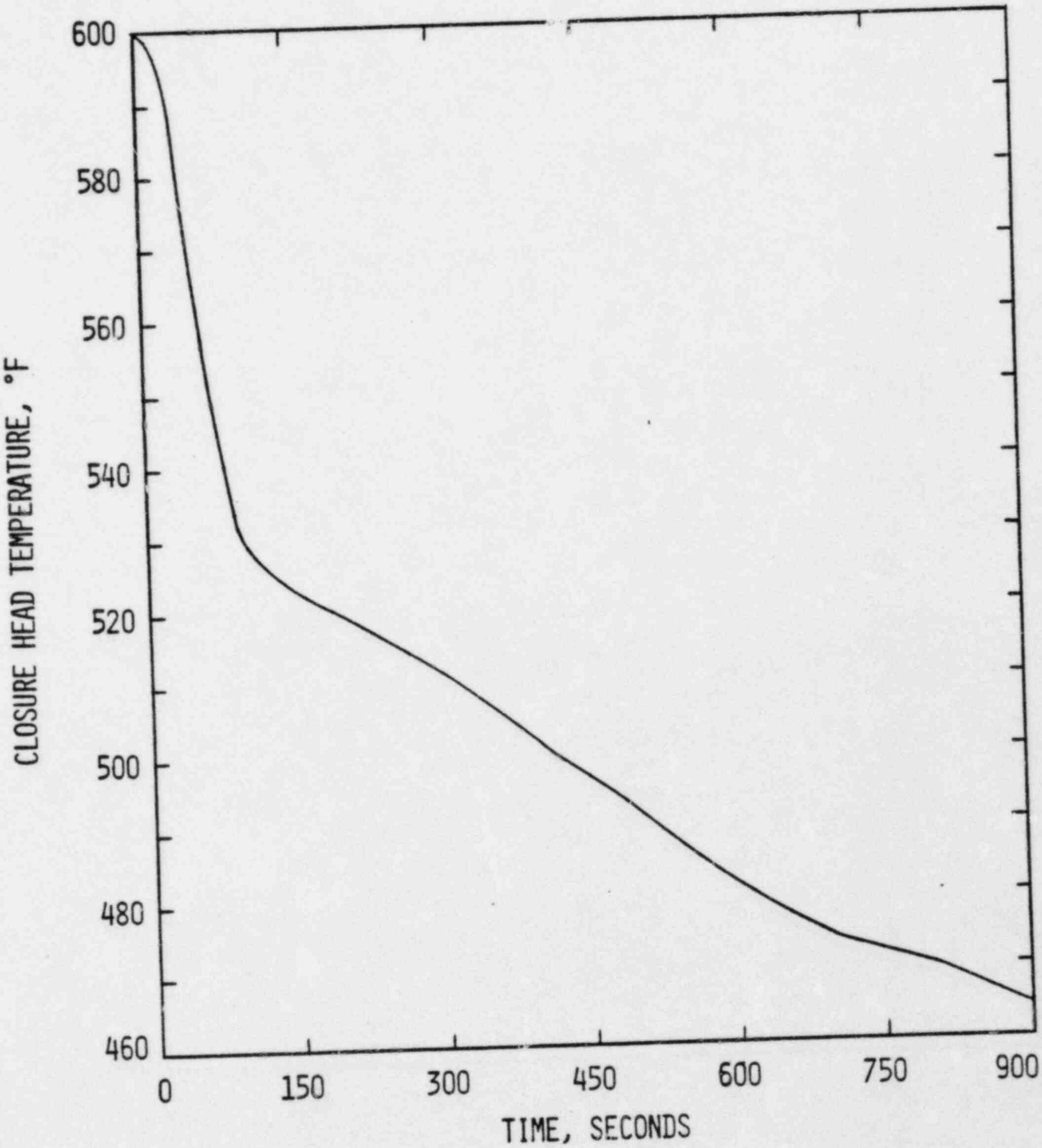
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE ?.1.4-11



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER STEAM GENERATOR PRESSURE VS TIME	FIGURE 7.1.4-12



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER CLOSURE HEAD VOID FRACTION VS TIME	FIGURE 7. .4-13



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER CLOSURE HEAD TEMPERATURE VS TIME	FIGURE 7.1.4-14

7.1.8 Excessive Charging Event

The Excessive Charging event initiated from maximum pressurizer level was analyzed for Unit 2 Cycle 5 to assure that the operator has at least fifteen (15) minutes from initiation of a high pressurizer level alarm to take corrective action and terminate the event prior to filling the pressurizer solid. The Excessive Charging event is assumed to occur by inadvertent initiation of charging flow.

The time required to fill the pressurizer solid was calculated using Equation 7.1.8-1.

$$T = \frac{V_S - V_{SL} - V_T}{(F_{CH} - F_{LD}) \; \frac{\nu_2}{\nu_1}} \qquad \text{Eq. 7.1.8-1}$$

where:

V_S = steam volume in the pressurizer

V_{SL} = equivalent saturated liquid volume of pressurizer steam volume

V_T = volume above the spray nozzles

F_{CH} = charging flow rate

F_{LD} = letdown flow rate

ν_1 = specific volume of liquid at charging and letdown conditions

ν_2 = specific volume of liquid at pressurizer conditions

The analysis was performed for three combinations of charging and letdown flows. Table 7.1.8-1 presents the initial conditions assumed in the analysis and the results of the analysis. The initial conditions assumed in the analysis are consistent with Technical Specification limits on initial pressurizer level including appropriate uncertainties. As seen from the table, all three combinations of charging and letdown flows analyzed provide at least fifteen minutes after initiation of high level alarm for the operator to take corrective actions and terminate the event prior to filling the pressurizer solid.

TABLE 7.1.8-1

Volume Control Assumptions		Maximum Initial Pressurizer Level Assumed in Analysis		High Level Alarm Analysis Setpoint		Time to Fill[1] Pressurizer
Charging Flow (GPM)	Letdown Flow (GPM)	Liquid Volume (ft^3)	Level[3] (in)	Liquid Volume (ft^3)	Level[3] (in)	(minutes)
1. 132	0	915	227	920	228	15
2. 132	29	975[2]	242	1040	257	15
3. 88	0	975[2]	242	1100	272	15

(1) From time of initiation of high pressurizer level alarm

(2) Maximum limit based on Loss of Load event

(3) Referenced to the 1" level nozzle at the bottom of the pressurizer

7.3.1 CEA Ejection Event

The CEA Ejection event was reanalyzed for Unit 2 Cycle 5 to determine the fraction of fuel pins that exceed the criterion for clad damage.

The analytical method employed in the reanalysis of this event is the NRC approved Combustion Engineering CEA Ejection method which is described in CENPD-190-A, (Reference 4).

The key parameters used in this event are listed in Table 7.3.1-1. With these key parameters, selected to add conservatism, the procedure outlined in Figure 2.1 of Reference 4 is then used to determine the average and centerline enthalpies in the hottest spot of the rod. The calculated enthalpy values are compared to threshold enthalpy values to determine the amount of fuel exceeding these thresholds. These threshold enthalpy values are (References 5, 6, and 7).

Clad Damage Threshold:	
Total Average Enthalpy	= 200 cal/gm
Incipient Centerline Melting Threshold:	
Total Centerline Enthalpy	= 250 cal/gm
Fully Molten Centerline Threshold:	
Total Centerline Enthalpy	= 310 cal/gm

To bound the most adverse conditions during the cycle, the most limiting of either the Beginning of Cycle (BOC) or End of Cycle (EOC) parameter values were used in the analysis. A BOC Doppler defect was used since it produces the least amount of negative reactivity feedback to mitigate the transient. A BOC moderator temperature coefficient of $+0.5 \times 10^{-4}\ \Delta\rho\ /^{\circ}F$ was used because a positive MTC results in positive reactivity feedback and thus increases coolant temperatures. An EOC delayed neutron fraction was used in the analysis to produce the highest power rise during the event.

The zero power CEA Ejection event was analyzed assuming the core is initially operating at 1 MWt. At zero power, a Variable High Power trip is conservatively assumed to initiate at 40% (30% + 10% uncertainty) of 2700 MWt and terminates the event.

The full and zero power cases were analyzed, assuming the value of 0.05 seconds for the total ejection time, which is consistent with the FSAR and previous reload submittals.

The power transient produced by a CEA ejection initiated at the maximum allowed power is shown in Figure 7.3.1-1. Similar results for the zero power case are shown in Figure 7.3.1-2.

The results of the two CEA ejection cases analyzed (Table 7.3.1-2) show that the maximum total energy deposited during the event is less than the criterion for clad damage (i.e., 200 cal/gm). Also, an acceptably small fraction of the fuel reaches incipient centerline melt threshold.

TABLE 7.3.1-1

KEY PARAMETERS ASSUMED IN THE CEA EJECTION ANALYSES

Parameter	Units	Reference Cycle*	Unit 2 Cycle 5
Full Power			
Core Power Level	MWt	2754	2754
Core Average Linear Heat Generation Rate at 2754 MWt	KW/ft	6.12	6.36
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^{o}F$	+.5	+.5
Ejected CEA Worth	$\%\Delta\rho$	.32	.28
Delayed Neutron Fraction,		.0047	.0044
Post-Ejected Radial Power Peak		3.36	3.6
Axial Power Peak		1.39	1.44
CEA Bank Worth at Trip	$\%\Delta\rho$	-3.88	-3.0
Tilt Allowance		1.03	1.03
Doppler Multiplier		0.85	0.85
Zero Power			
Core Power Level	MWt	1.0	1.0
Ejected CEA Worth	$\%\Delta\rho$	.60	.63
Post-Ejected Radial Power Peak		9.83	9.40
Axial Power Peak		1.60	1.60
CEA Bank Worth at Trip	$\%\Delta\rho$	-2.58	-1.50
Tilt Allowance		1.10	1.10
CEA Drop Time	sec	3.1	3.1
Doppler Multiplier		0.85	0.85

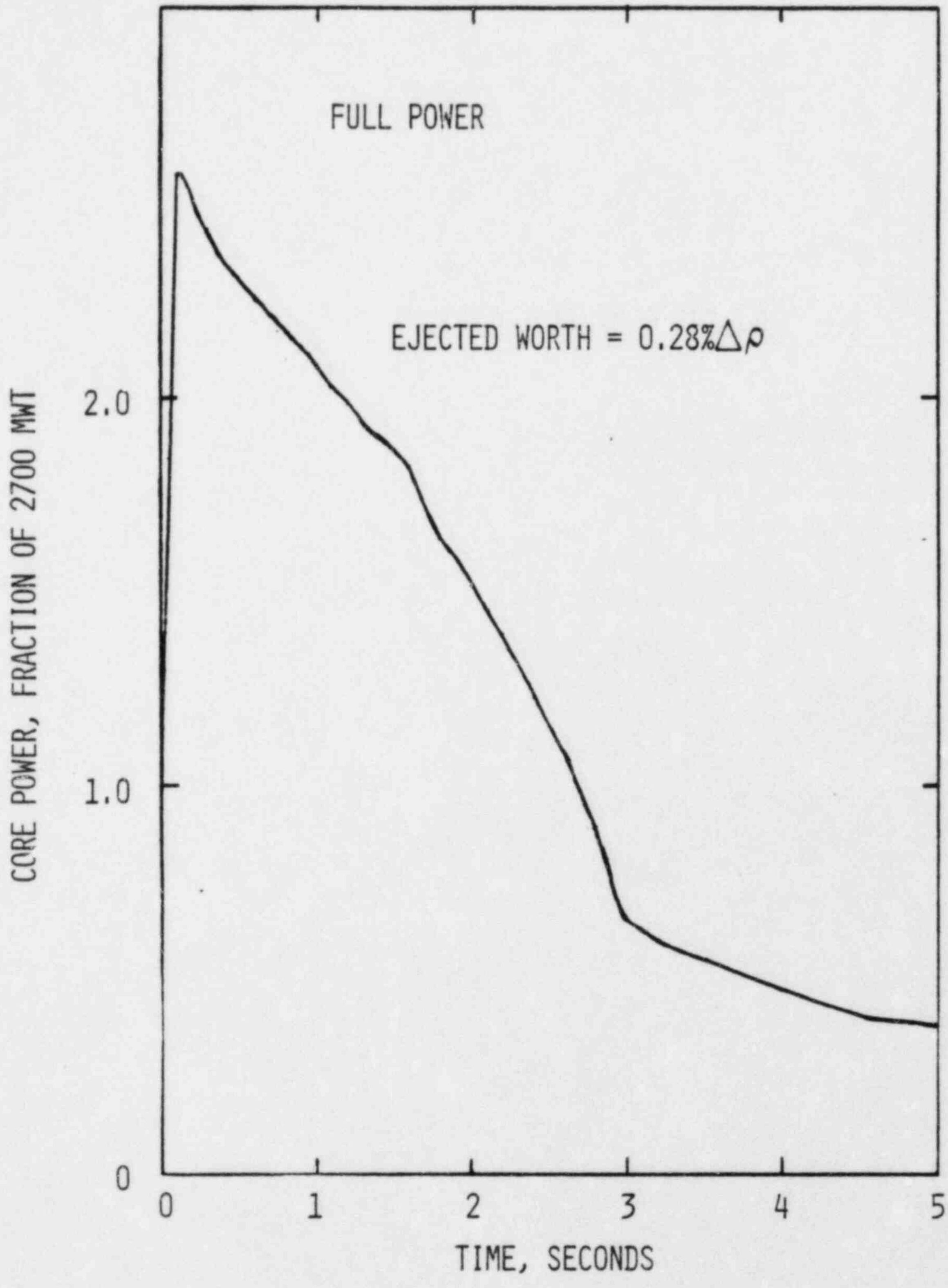
*Reference cycle is Unit 1, Cycle 4 (Reference 8).

TABLE 7.3.1-2

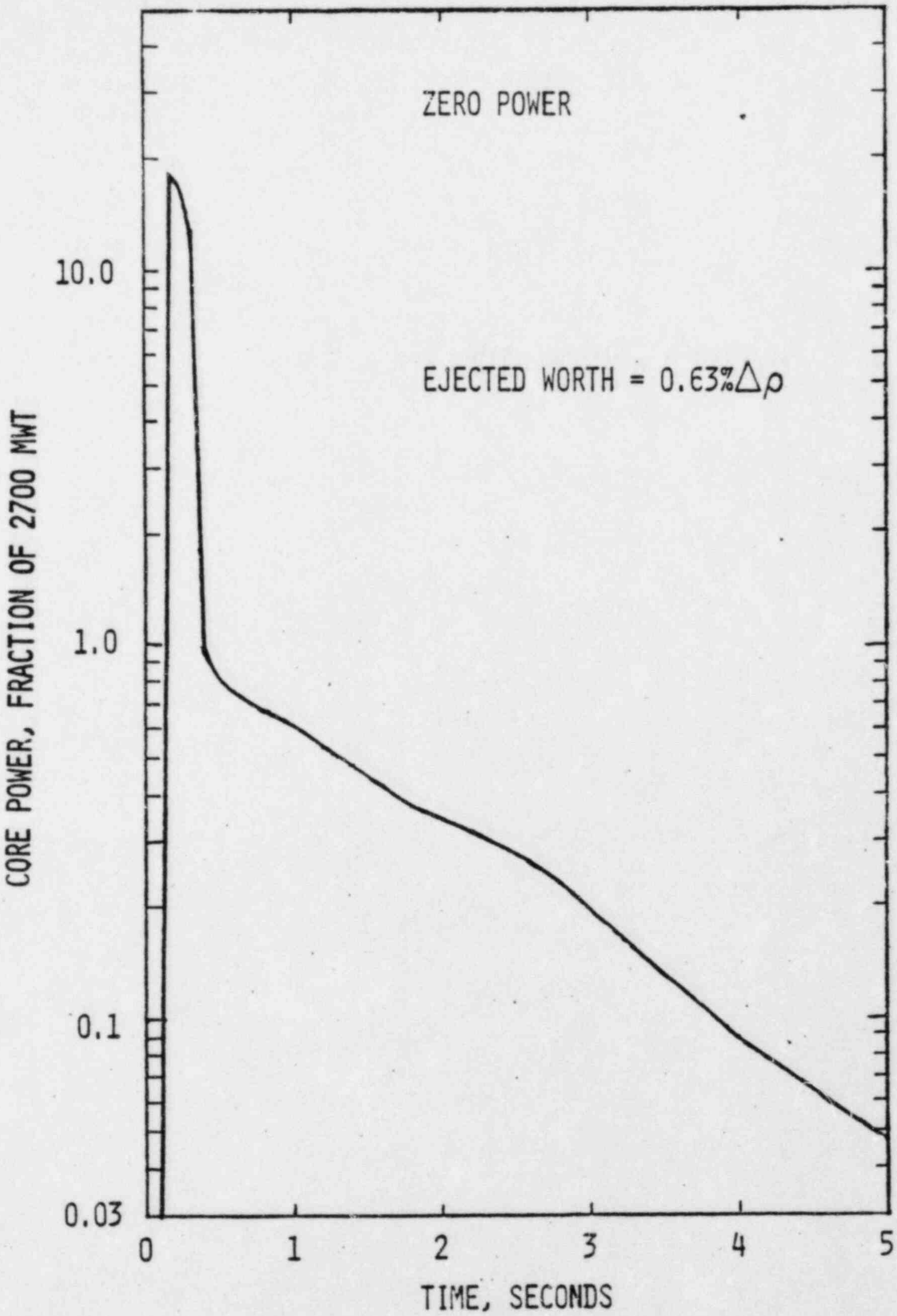
CEA EJECTION EVENT RESULTS

Full Power	Reference Cycle Unit 1, Cycle 4	Unit 2 Cycle 5
Total Average Enthalpy of Hottest Fuel Pellet (cal/gm)	198.	185.
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	268.	293.
Fraction of Rods that Suffer Clad Damage (Average Enthalpy $\geq$ 200 cal/gm)	0	0
Fraction of Fuel Having a Least Incipient Centerline Melting (Centerline Enthalpy $\geq$ 250 cal/gm)	.01	.08
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy $\geq$ 310 cal/gm)	0	0

Zero Power	Reference Cycle Unit 1, Cycle 4	Unit 2 Cycle 5
Total Average Enthalpy of Hottest Fuel Pellet (cal/gm)	177.	145.
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	177.	199.
Fraction of Rods that Suffer Clad Damage (Average Enthalpy $\geq$ 200 cal/gm)	0	0
Fraction of Fuel Having a Least Incipient Centerline Melting (Centerline Enthalpy $\geq$ 250 cal/gm)	0	0
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy $\geq$ 310 cal/gm)Total	0	0



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CEA EJECTION EVENT CORE POWER VS TIME	FIGURE 7.3.1-1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CEA EJECTION EVENT CORE POWER VS TIME	FIGURE 7.3.1-2

7.3.2 Steam Line Rupture Event

The Steam Line Rupture (SLB) event was analyzed for Cycle 5 to determine that the critical heat flux is not exceeded during this event.

The analysis included the effect of automatic initiation of auxiliary feedwater flow in three (3) minutes from the initiation of a Low S.G. level signal, a manual trip of the Reactor Coolant Pumps on Safety Injection Actuation Signal due to low pressurizer pressure, and an MSIV closure time of 12 seconds.*

The analysis assumed that the event is initiated by a circumferential rupture of a 34-inch (inside diameter) steam line at the steam generator main steam line nozzle. This break size is the most limiting, since it causes the greatest rate of temperature reduction in the reactor core region.

The SLB event was analyzed with the assumption of a three minute delay between the time of reactor trip on Low S.G. pressure and the time when Auxiliary Feedwater (AFW) flow is delivered to the affected steam generator. This is conservative with respect to the expected time of AFW initiation since the generation of the AFW signal actually occurs at the time of the Low Steam Generator Water Level trip signal which occurs later than the Low S.G. Pressure trip. The analysis assumes, therefore, that AFW flow is delivered to the steam generator sooner than the flow is actually available resulting in a conservative prediction of the resulting cooldown.

A conservatively high value of the AFW flow was calculated assuming that all auxiliary feedwater pumps are operable. An AFW flow value of 21% of full power feedwawter flow was used in the analysis. This value accounts for pump run-out due to reduced back pressure. In addition, the analysis conservatively assumed that all the AFW flow is fed only to the damaged steam generator.

The analyses assumed that the main feedwater flow is ramped down to 8% of full power feedwater flow in 20 seconds and that the main feedwater isolation valves are completely closed in 80 seconds after a low steam generator pressure or a main steam isolation signal. These assumptions are consistent with Technical Specification limits (see Table 3.3-5).

During a Return-To-Power, negative reactivity credit was assumed in the analysis. This negative reactivity credit is due to the local heatup of the inlet fluid in the hot channel, which occurs near the location of the stuck CEA. This credit is based on three-dimensional coupled neutronic-thermal-hydraulic calculations performed with the HERMITE/TORC code (References 9 and 10) for St. Lucie Unit 2, Cycle 1 (Reference 11). It should be noted that only a small fraction of the negative reactivity credit justified for St. Lucie Unit 2 was included in the SLB event analysis for Calvert Cliffs Unit 2, Cycle 5.

The manual trip of the RCPs is assumed to result in no flow mixing at the core inlet plenum. Thus, cold edge temperatures were used to calculate the moderator reactivity insertion during the cooldown of the RCS following a SLB.

The two SLB cases considered in conjunction with the automatic initiation of auxiliary feedwater flow and manual trip of RCPs are:

*Conservative with respect to Technical Specification limit of 6.0 seconds.

1. 2 Loop - Full Load (2754 MWt)

2. 2 Loop - No Load (1 MWt)

The 1 Loop - Full Load and 1 Loop - No Load cases were not analyzed since Technical Specifictions prohibit operation in these modes.

The Two Loop-2764 MWt case was initiated at the conditions listed in Table 7.3.2-1. The Moderator Temperature Coefficient (MTC) of reactivity assumed in the analysis corresponds to end of life, since this MTC results in the greatest positive reactivity change during the RCS cooldown caused by the Steam Line Rupture. Since the reactivity change associated with moderator feedback varies significantly over the moderator density covered in the analysis, a curve of reactivity insertion versus density rather than a single value of MTC, is assumed in the analysis. The moderator cooldown curve assumed is given in Figure 7.3.2-1. The moderator cooldown curve given in Figure 7.3.2-1 was conservatively calculated assuming that on reactor scram, the Control Element Assembly is stuck in the fully withdrawn position which yields the most severe combination of scram worth and reactivity insertion.

The reactivity defect associated with the fuel temperature decrease is also based on an end of life Doppler defect. The Doppler defect based on an end of life Fuel Temperature Coefficient (FTC), in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the Steam Line Rupture event. The uncertainty on the FTC assumed in the analysis is given in Table 7.3.2-1. The β fraction assumed is the maximum absolute value including uncertainties for end of life conditions. This too is conservative since it maximizes subcritical multiplication and, thus, enhances the potential for Return-To-Power (R-T-P).

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip at the maximum allowed power level is $-6.89\%\ \Delta\rho$. This available scram worth was calculated for the stuck rod which produced the moderator cooldown curve in Figure 7.3.2-1.

The analysis conservatively assumed that on Safety Injection Actuation Signal one High Pressure Safety Injection pump and one Low Pressure Safety Injection pump fail to start. The analysis also assumed a conservatively low value of boron reactivity worth at $-1.0\%\ \Delta\rho$ per 95 PPM.

The conservative assumptions on feedwater flow were discussed previously. The feedwater flow and enthalpy as a function of time are presented in Figures 7.3.2-2 and 7.3.2-3, respectively.

Table 7.3.2-2 presents the sequence of events for the full power case initiated at the conditions given in Table 7.3.2-1. The reactivity insertion as a function of time is presented in Figure 7.3.2-4. The response of the NSSS during this event is given in Figures 7.3.2-5 through 7.3.2-9.

The results of the analyses show that SIAS is actuated at 16.7 seconds, at which time the Reactor Coolant Pumps are manually tripped by the operator. The manual trip of RCPs slows down the rate of primary heat removal and, thus, delays the time when the affected steam generator blows dry. The affected steam generator blows dry at 94.0 seconds and terminates the cooldown of the

RCS. The peak reactivity attained prior to delivery of Auxiliary feedwater flow is -.052% $\Delta\rho$ at 123.0 seconds. A peak R-T-P of 9.3%, consisting of 3.5% decay heat and 5.8% fission power is produced at 127.5 seconds. The continued production of decay heat from the fuel after termination of blowdown, causes the reactor coolant temperatures to increase. This in turn reduces the magnitude of the positive moderator reactivity inserted and, thus, the total reactivity becomes more negative.

The delivery of auxiliary feedwater flow to the affected steam generator at 183.7 seconds initiates a further cooldown of the RCS which results in more positive reactivity insertion. The positive reactivity insertion causes the core to approach criticality. The peak criticality attained is +.044% $\Delta\rho$ at 584.3 seconds. The reactivity transient is terminated by the boron injected via the High Pressure Safety Injection Pumps. A peak R-T-P of 3.1%, consisting of 2.3% decay heat and 0.8% fission power is produced at 624.0 seconds.

The MacBeth correlation (Reference 12) with the Lee non-uniform mixing correlation factor (Reference 13) results in a post-trip minimum DNBR of 1.35 compared to the DNBR limit of 1.30 during a SLB event initiated from hot full power conditions. Thus, critical heat fluxes are not exceeded during this event.

Two Loop-No Load case was initiated at the conditions given in Table 7.3.2-3. The moderator cooldown curve is given in Figure 7.3.2-10. The cooldown curve corresponds to an end of life MTC. An end of life FTC was also used for the reasons previously discussed in connection with the Two Loop-2754 MWt case.

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip at the zero power level is -5.2% $\Delta\rho$. This available scram worth was calculated for the stuck rod which produced the moderator cooldown curve in Figure 7.3.2-10. A maximum inverse boron worth of 90 PPM/% $\Delta\rho$ was conservatively assumed for the safety injection during the no load case. The feedwater flow and the enthalpy used in the analysis are presented in Figures 7.3.2-11 and 7.2.3-12, respectively.

Table 7.3.2-4 presents the sequence of events for the Two Loop-No Load case initiated from the conditions given in Table 7.3.2-3. The reactivity insertion as a function of time is presented in Figure 7.3.2-13. The NSSS response during this event are given in Figures 7.3.2-14 to 7.3.2-18.

The results of the analysis show that SIAS is actuated at 20.9 seconds, at which time the RCPs are manually tripped by the operator. Auxiliary feedwater flow is initiated at 183.3 seconds which continues the cooldown of the RCS. Thus, the total core reactivity approaches criticality. The peak reactivity attained is +0.34% $\Delta\rho$ at 450.8 seconds and a peak power of 3.13% occurs at 472.5 seconds. The addition of boron via High Pressure Safety Injection mitigates the reactivity transient.

The MacBeth correlation with the Lee non-uniform mixing correlation factor results in a post-trip minimum DNBR of 2.00 compared to the DNBR limit of 1.30, during a SLB event initiated from hot zero power conditions. Thus, critical heat fluxes are not exceeded during this event.

The Steam Line Rupture event initiated from HFP and HZP conditions with automatic initiation of auxiliary feedwater flow and manual trip of RCPs on SIAS due to low pressurizer pressure shows that the DNBR limits are not exceeded. Since the DNBR limits are not exceeded and no fuel pins are predicted to fail, it is concluded that the consequences of the SLB event are acceptable for Cycle 5 operation.

TABLE 7.3.2-1

KEY PARAMETERS ASSUMED IN THE STEAM LINE RUPTURE ANALYSIS 2-LOOP-2754 MWT

Parameter	Units	Reference Cycle*	Unit 2 Cycle 5
Initial Core Power	MWt	2754	2754
Initial Core Inlet Temperature	°F	550	550
Initial RCS Pressure	psia	2300	2300
Initial Steam Generator Pressure	psia	871	860**
Low Steam Generator Pressure Analysis Trip Setpoint	psia	548	548
Safety Injection Actuation Signal	psia	1556	1645
Minimum CEA Worth Available at Trip	%Δρ	-7.02	-6.89
Doppler Multiplier		1.15	1.15
Moderator Cooldown Curve	%Δρ vs. density	See Figure 7.3.2-1 of Reference 1	See Figure 7.3.2-1
Inverse Boron Worth	PPM/% Δρ	95	95
Effective MTC	$x10^{-4}\Delta\rho/°F$	-2.2	-2.2
β Fraction (including uncertainty)		.0060	.006C

* Unit 1 Cycle 6 (Reference 1)

**The difference in initial steam generator pressure has negliable impact on the results.

TABLE 7.3.2-2

SEQUENCE OF EVENTS FOR STEAM LINE RUPTURE EVENT
WITH AUTOMATIC INITIATION OF AUXILIARY FEEDWATER
AND MANUAL TRIP OF REACTOR COOLANT PUMP
2-LOOP-2754 MWT

Time (sec)	Event	Setpoint or Value
0.0	Steam Line Rupture Occurs	---
2.3	Low Steam Generator Pressure Analysis Trip Setpoint is Reached; Steam Generator Isolation Analysis Setpoint is Reached	548.0 psia
3.2	Trip Breakers Open; Main Steam Isolation Valves Begin to Close	---
3.7	CEAs Enter Core; Main Feedwater Rampdown Begins	---
15.2	Main Steam Isolation Valves Completely Closed	---
16.7	Safety Injection Actuation Signal	1645.0 psia
16.8	Reactor Coolant Pumps Manually Tripped	---
17.2	Pressurizer Empties	---
23.7	Main Feedwater Rampdown Completed	8% of full power feedwater flow
46.7	High Pressure Safety Injection Pumps at Full Speed	---
83.2	Main Feedwater Isolation	---
94.0	Affected Steam Generator Blows Dry	---
123.0	Peak Reactivity, Prior to Auxiliary Feedwater Flow	-.052%
127.5	Peak Return to Power	9.3%
183.7	Auxiliary Feedwater Flow Initiated to Ruptured Steam Generator	340 lbm/sec
584.3	Peak Reactivity Post Auxiliary Feedwater Flow	+.044%

TABLE 7.3.2-2
(continued)

Time (sec)	Event	Setpoint or Value
624.0	Peak Return to Power Post Auxiliary Feedwater Flow	3.1%
1200.0	Operator Isolates Ruptured Steam Genrator and Terminates Auxiliary Feedwater Flow	---

TABLE 7.3.2-3

KEY PARAMETERS ASSUMED IN THE STEAM LINE RUPTURE ANALYSIS 2-LOOP NO LOAD

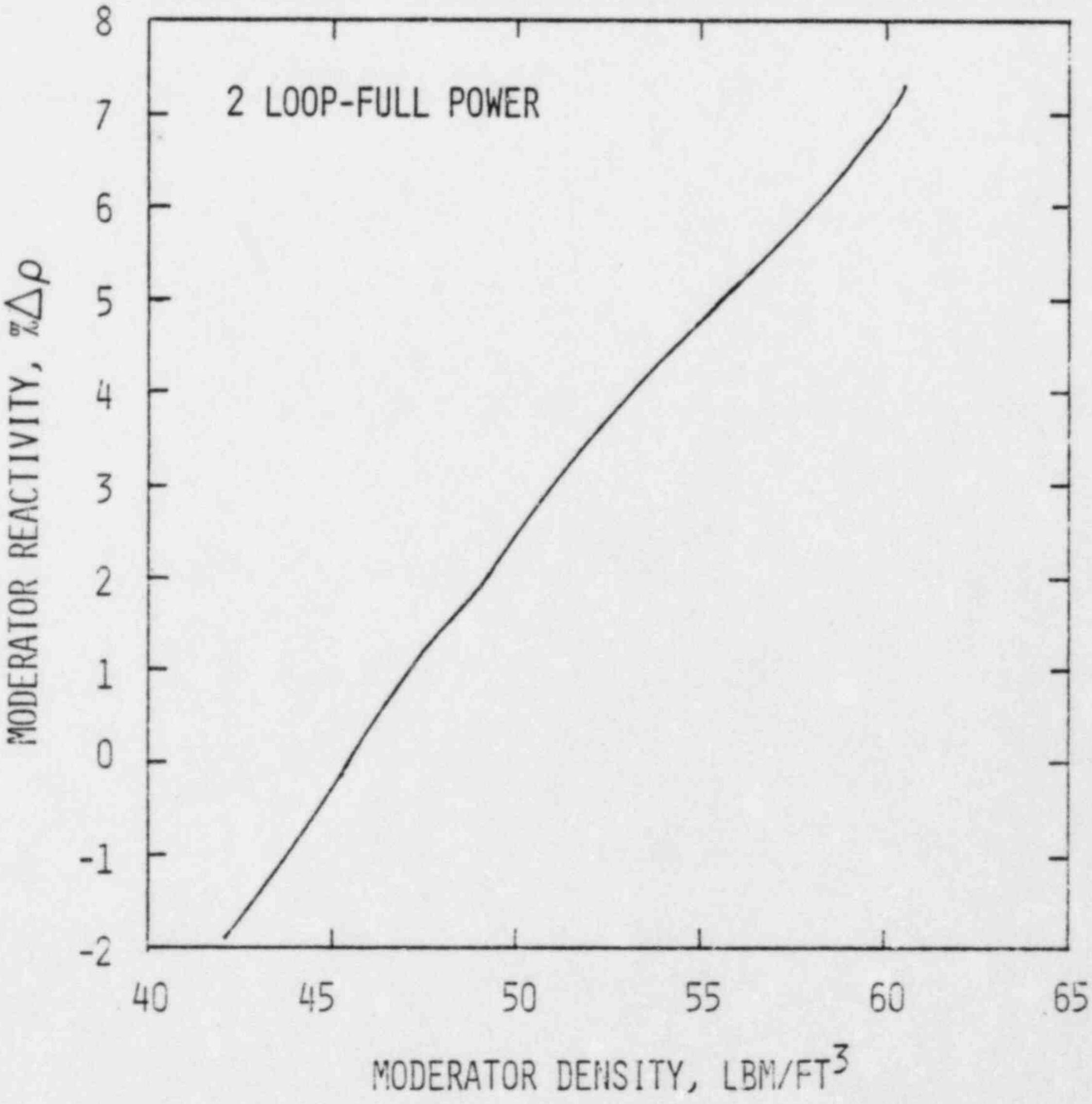
Parameter	Units	Reference Cycle*	Unit 2 Cycle 5
Initial Core Power	MWt	1.0	1.0
Initial Core Inlet Temperature	^{o}F	532	532
Initial RCS Pressure	psia	2300	2300
Initial Steam Generator Pressure	psia	900	900
Low Steam Generator Pressure Analysis Trip Setpoint	psia	548	548
Safety Injection Actuation Signal	psia	1556	1645
Minimum CEA Worth Available at Trip	$\%\Delta\rho$	-5.3	-5.2
Doppler Multiplier		1.15	1.15
Moderator Cooldown Curve	% vs. density	See Figure 7.3.2.10 of Reference 1	See Figure 7.3.2-10
Inverse Boron Worth	PPM/$\%\Delta\rho$	90	90
Effective MTC	$x10^{-4}\Delta\rho/^{o}F$	-2.2	-2.2
β Fraction (including uncertainty)		.0060	.0060

*Unit 1 Cycle 6 (Reference 1)

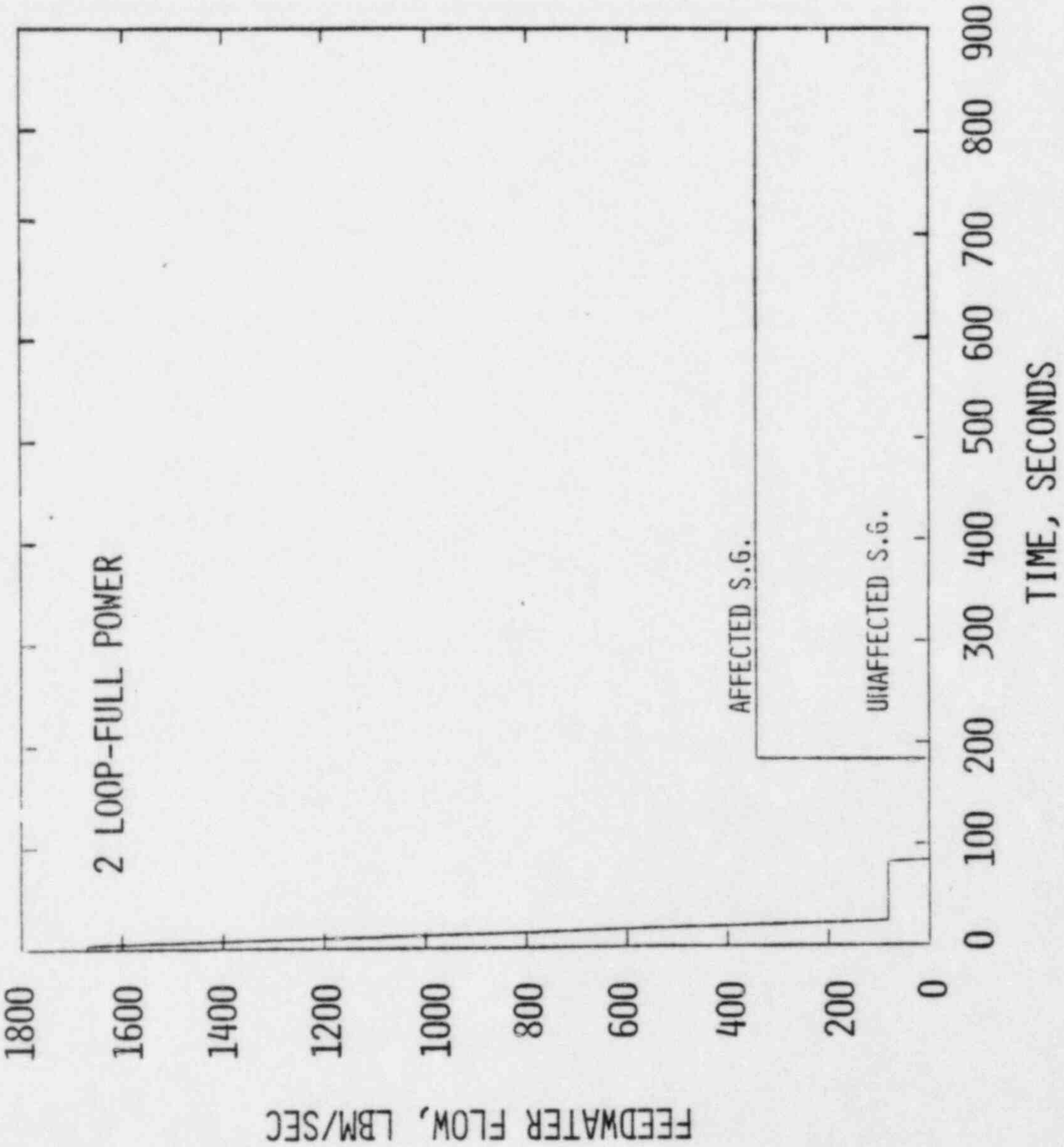
TABLE 7.3.2-4

SEQUENCE OF EVENTS FOR STEAM LINE RUPTURE EVENT
WITH AUTOMATIC INITIATION OF AUXILIARY FEEDWATER
AND MANUAL TRIP OF REACTOR COOLANT PUMP
2-LOOP-NO LOAD

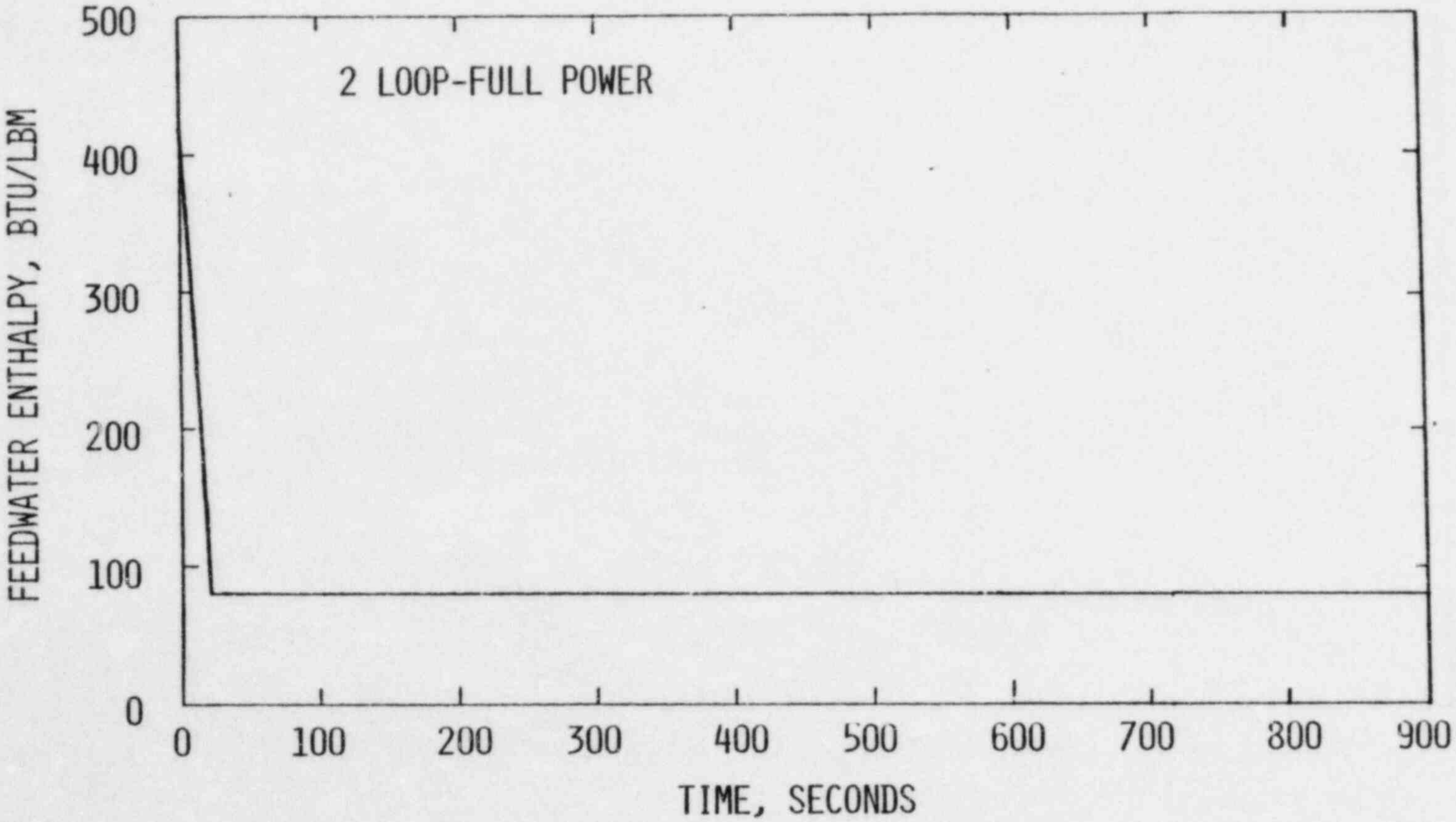
Time (sec)	Event	Setpoint or Value
0.0	Steam Line Rupture Occurs	6.305 ft^2
1.9	Low Steam Generator Pressure Analysis Trip Setpoint is Reached; Steam Generator Isolation Analysis Setpoint is Reached	548.0 psia
2.8	Trip Breakers Open; Main Steam Isolation Valves Completely Closed	---
3.3	CEAs Enter Core	---
14.8	Main Steam Isolation Valves Completely Closed	---
20.9	Safety Injection Actuation Signal	1645.0 psia
21.0	Reactor Coolant Pumps Manually Tripped	---
24.3	Pressurizer Empties	---
50.9	High Pressure Safety Injection Pumps at Full Speed	---
82.8	Main Feedwater Isolation	---
183.3	Auxiliary Feedwater Flow Initiated to Ruptured Steam Generator	340 lbm/sec
450.8	Peak Reactivity	+0.34%
472.5	Peak Power	3.13%
1200.0	Operator Isolates Ruptured Steam Generator and Terminates Auxiliary Feedwater Flow	---



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT MODERATOR REACTIVITY VS MODERATOR DENSITY	FIGURE 7.3.2-1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT FEEDWATER FLOW VS TIME	FIGURE 7.3.2-2



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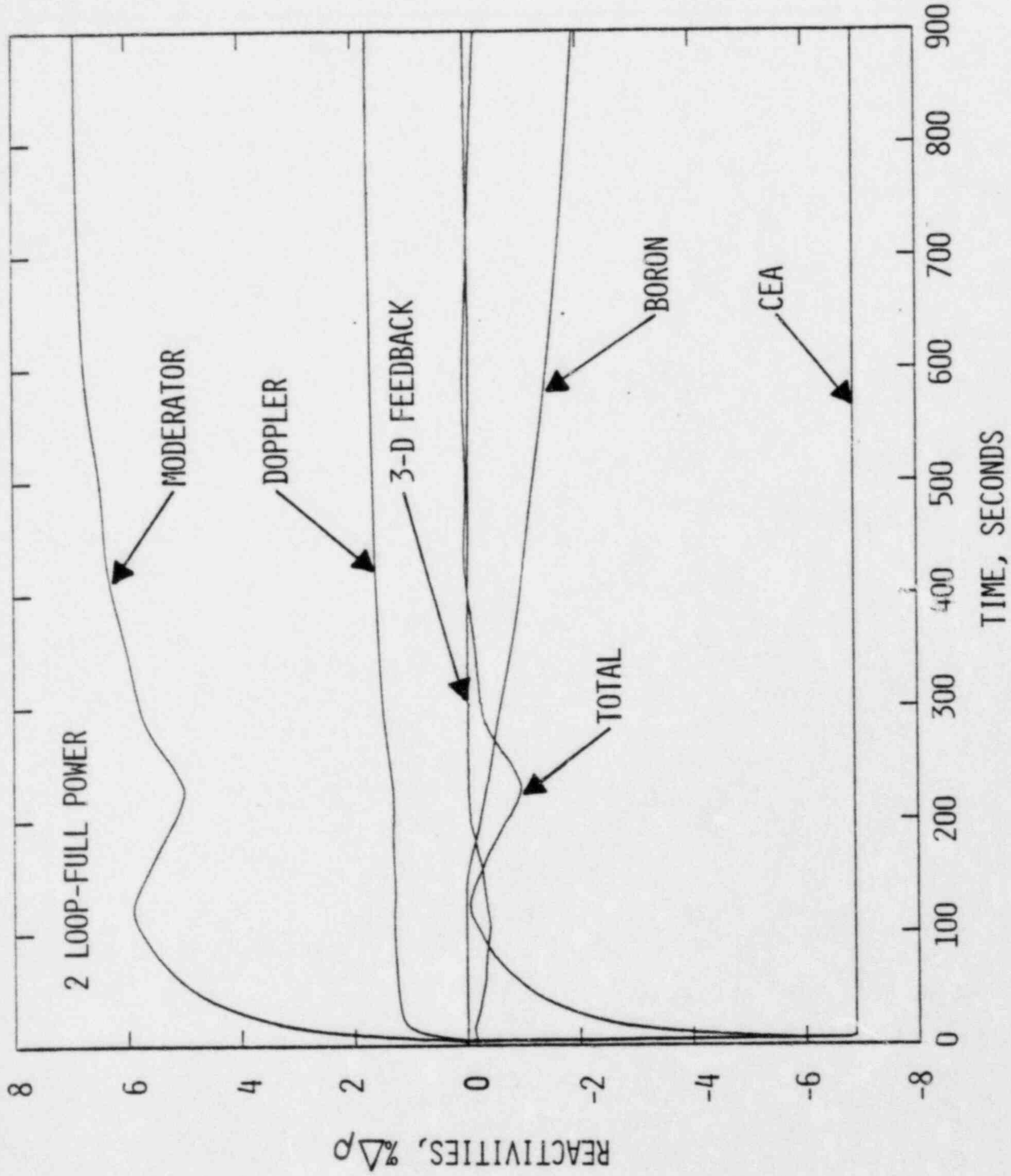
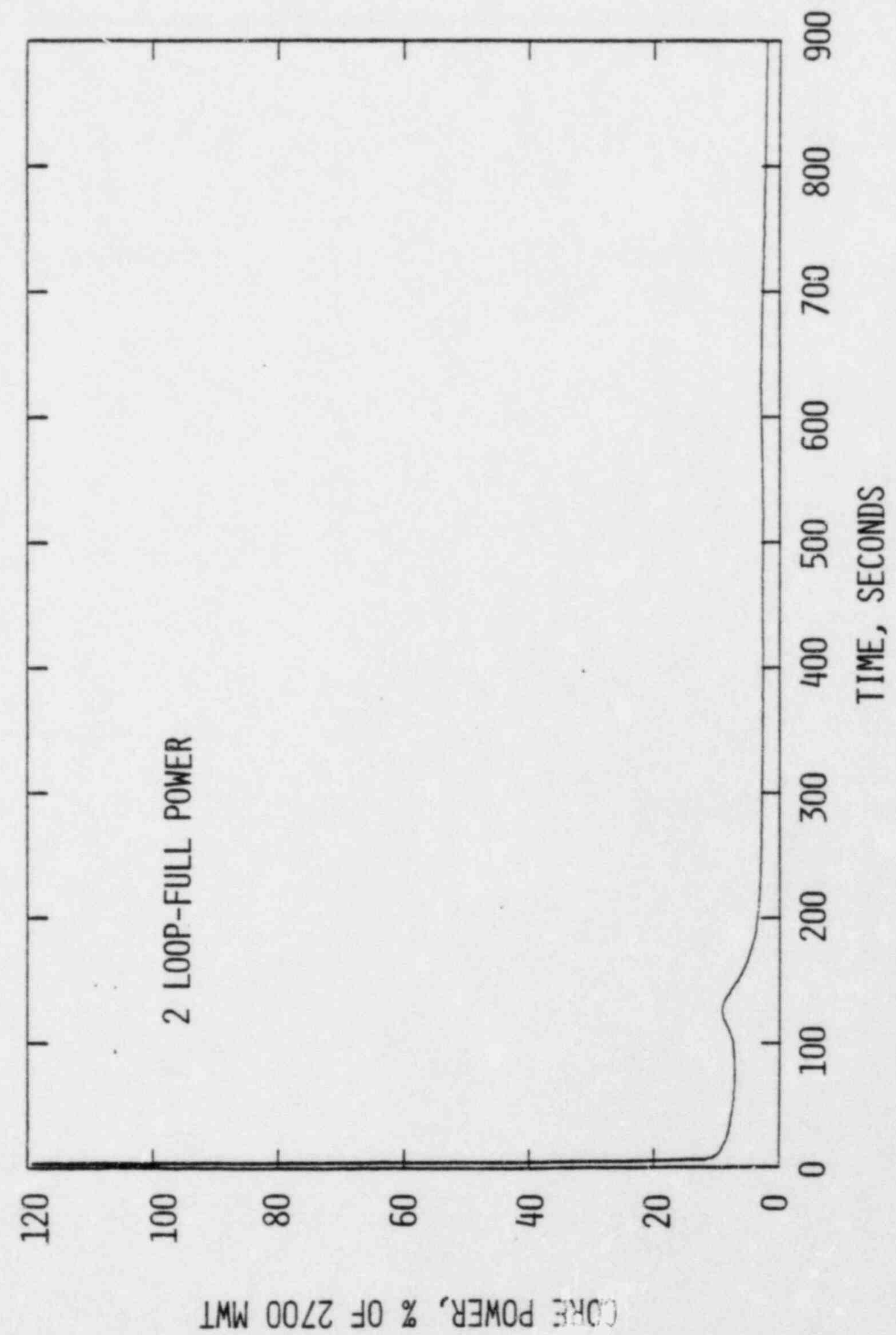
STEAM LINE BREAK EVENT
FEEDWATER ENTHALPY VS TIME

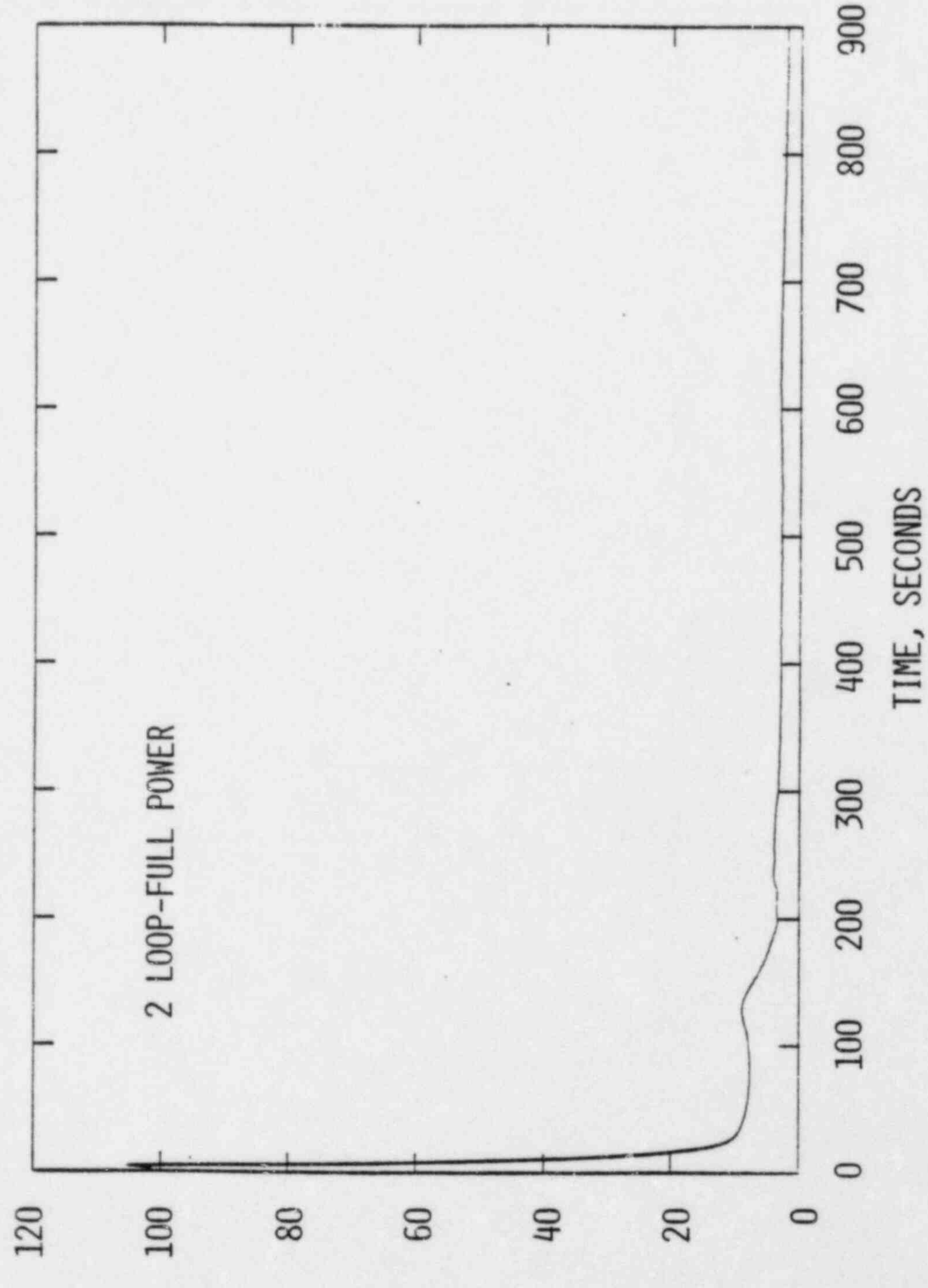
FIGURE
7.3.2-3



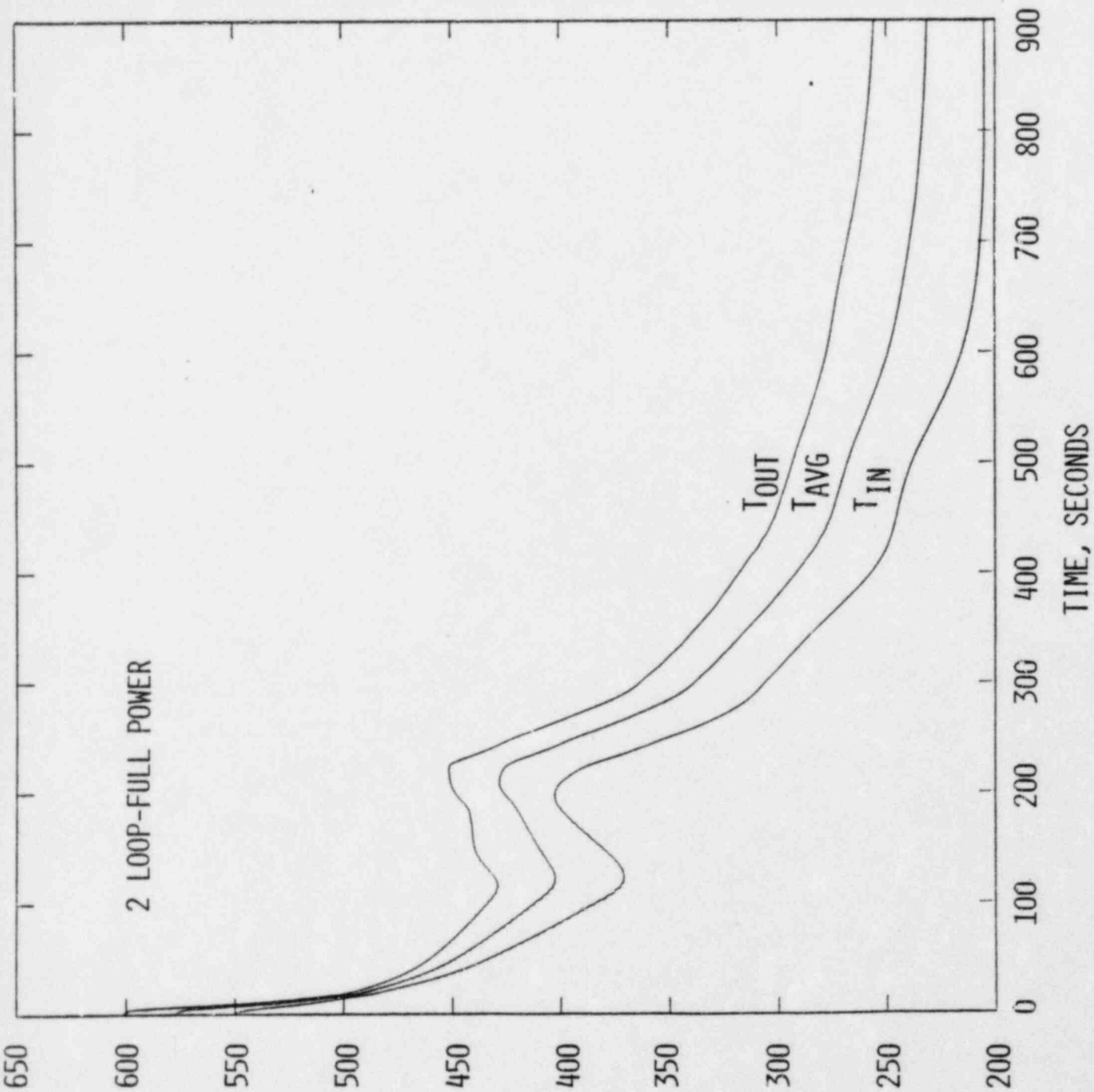
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT REACTIVITIES VS TIME	FIGURE 7.3.2-4



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT CORE POWER VS TIME	FIGURE 7.3.2-5



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT CORE HEAT FLUX VS TIME	FIGURE 7.3.2-6



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7.3.2-7

BALTIMORE
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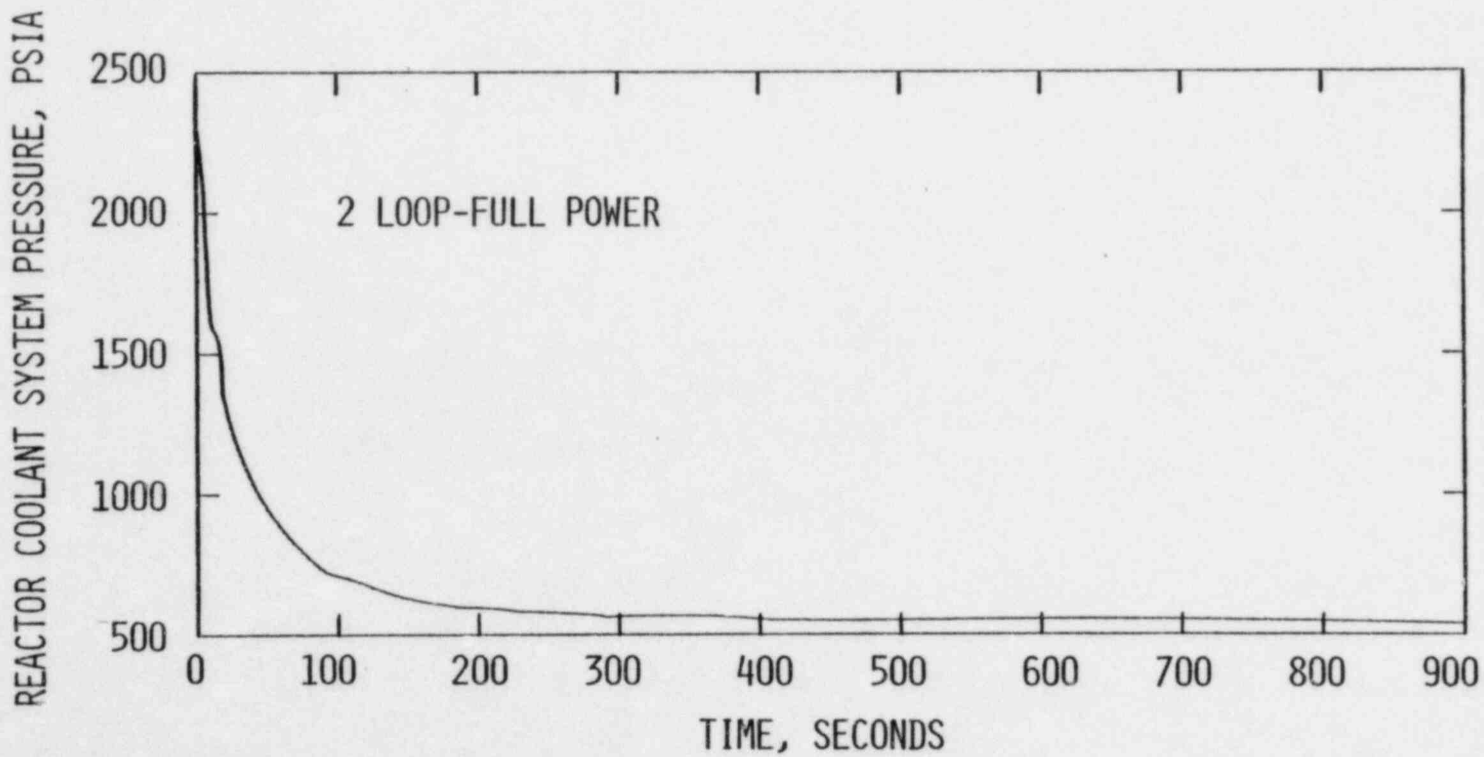
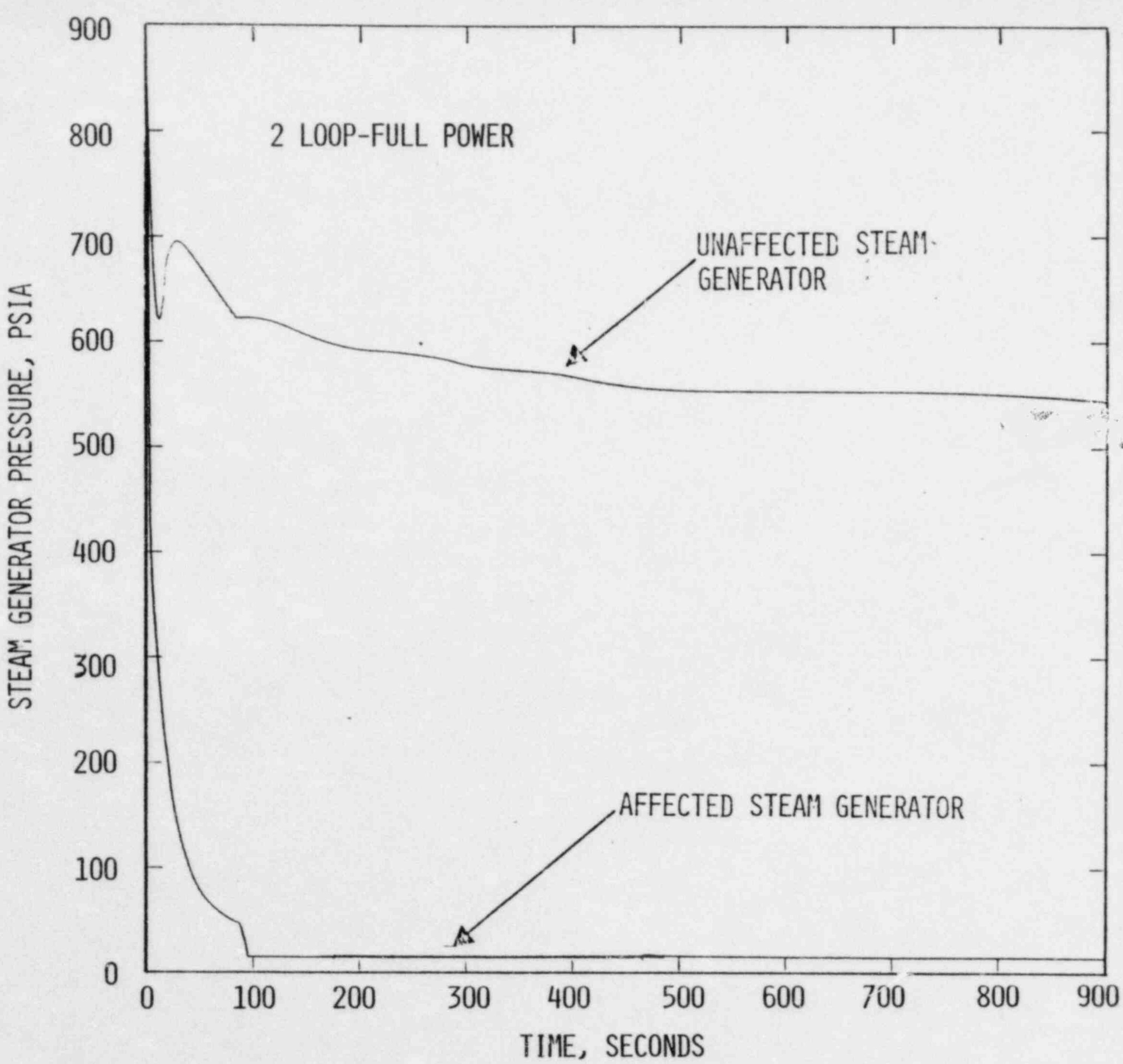
Nuclear Power Plant

STEAM LINE BREAK EVENT
REACTOR COOLANT SYSTEM PRESSURE VS TIME

FIGURE
7.3.2-8





BALTIMORE GAS & ELECTRIC CO.
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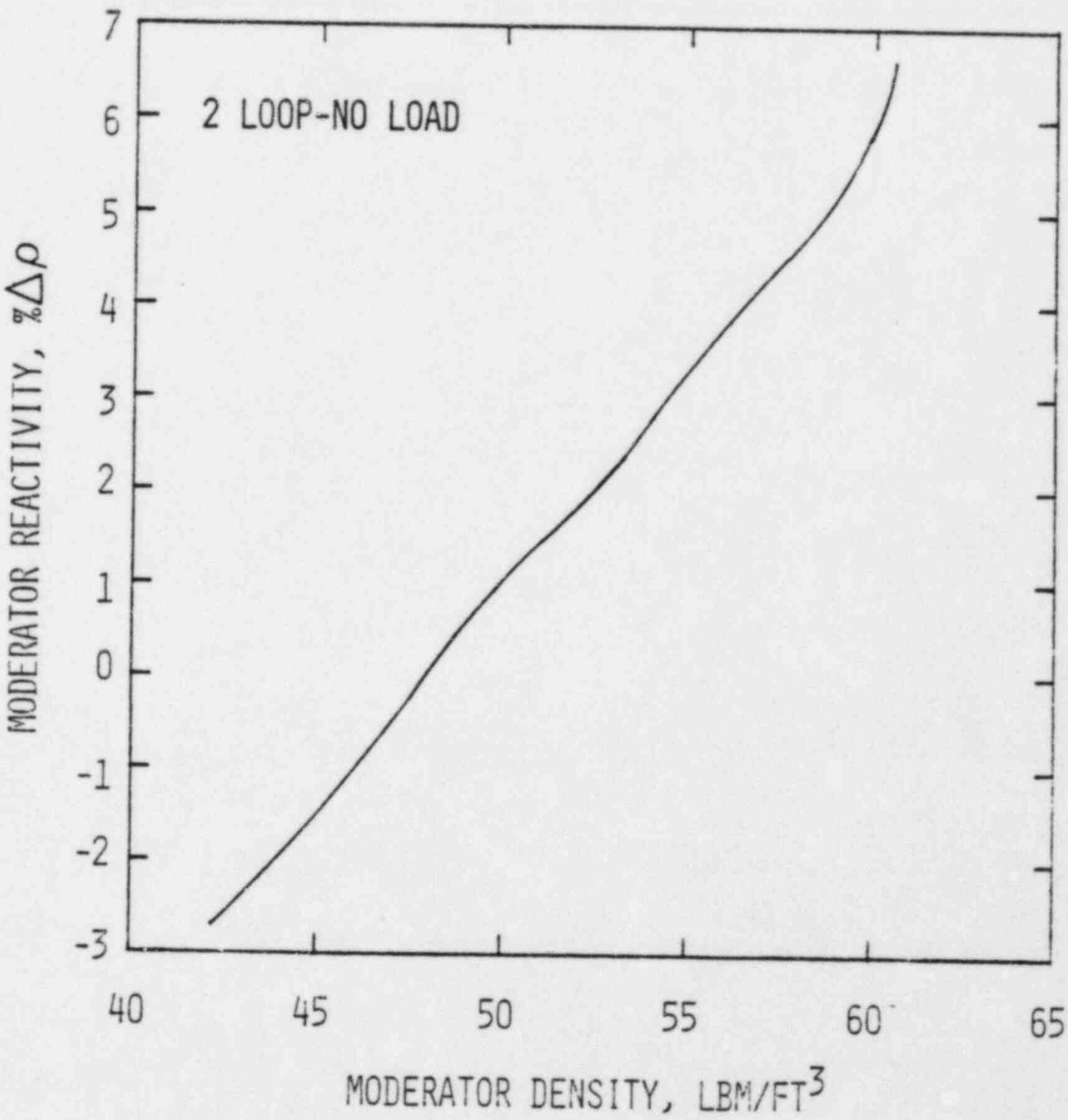
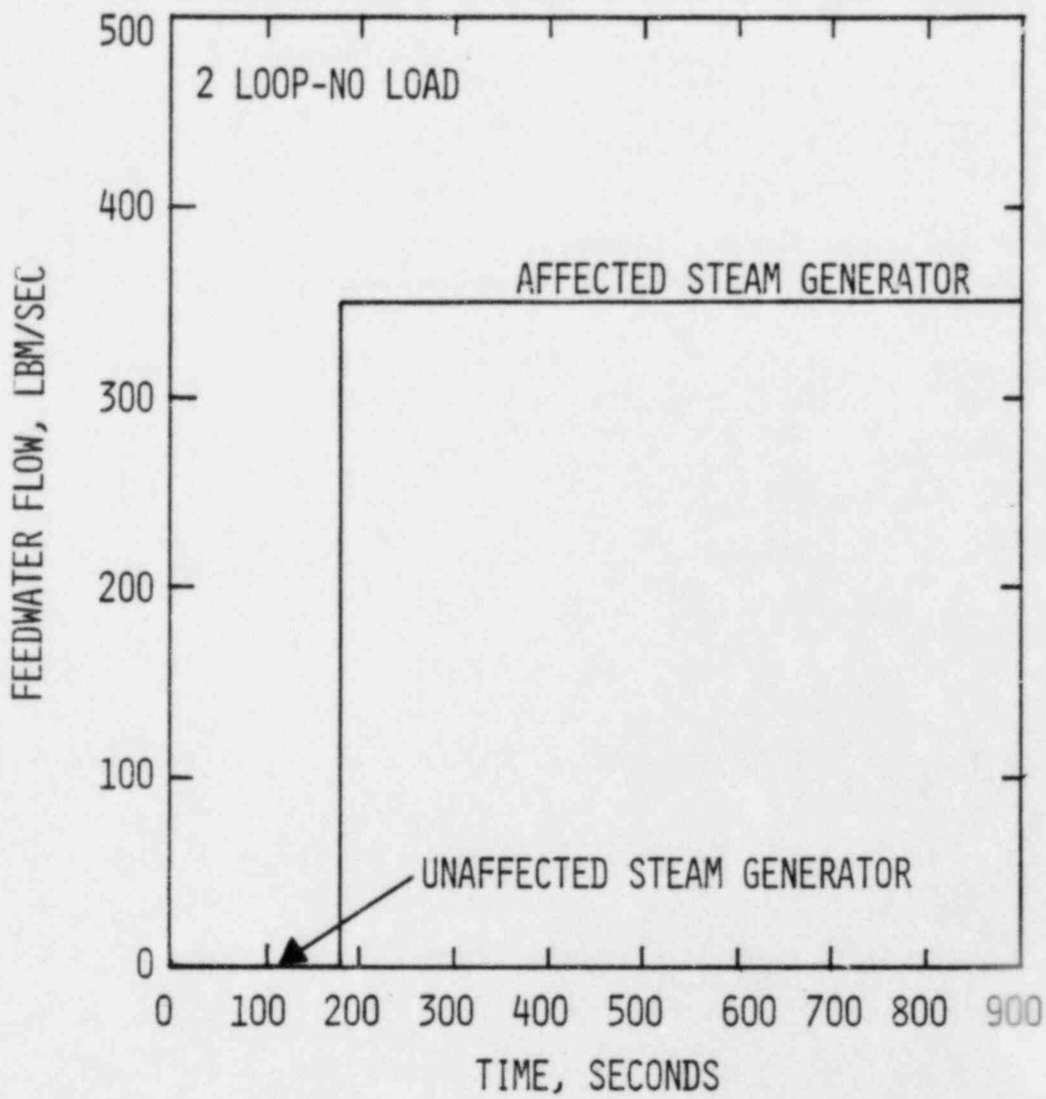
STEAM LINE BREAK EVENT
STEAM GENERATOR PRESSURE VS TIME

FIGURE 7.3.2-9



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT MODERATOR REACTIVITY VS MODERATOR DENSITY	FIGURE 7.3.2-10



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT FEEDWATER FLOW VS TIME	FIGURE 7.3.2-11

BALTIMORE
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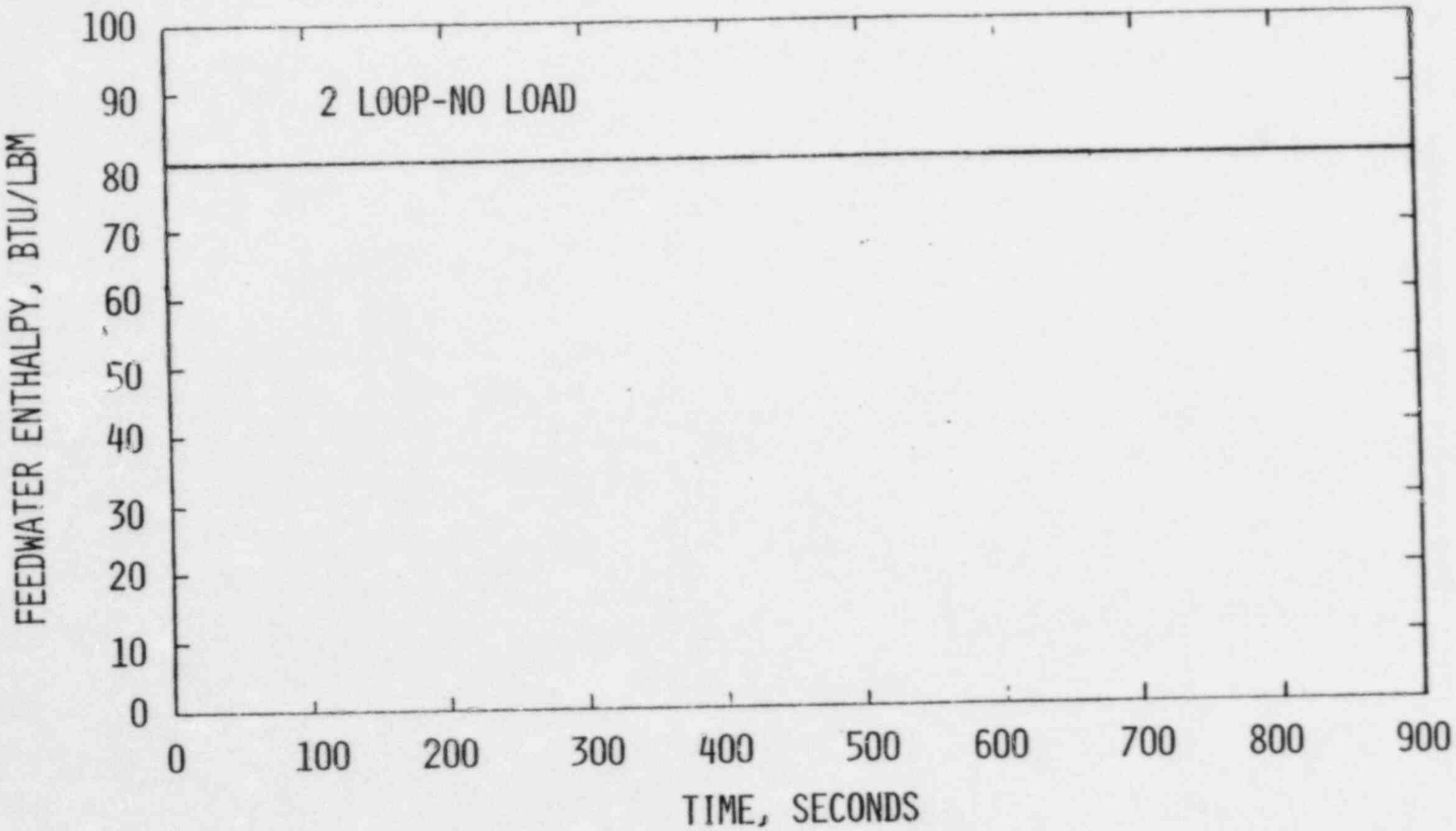
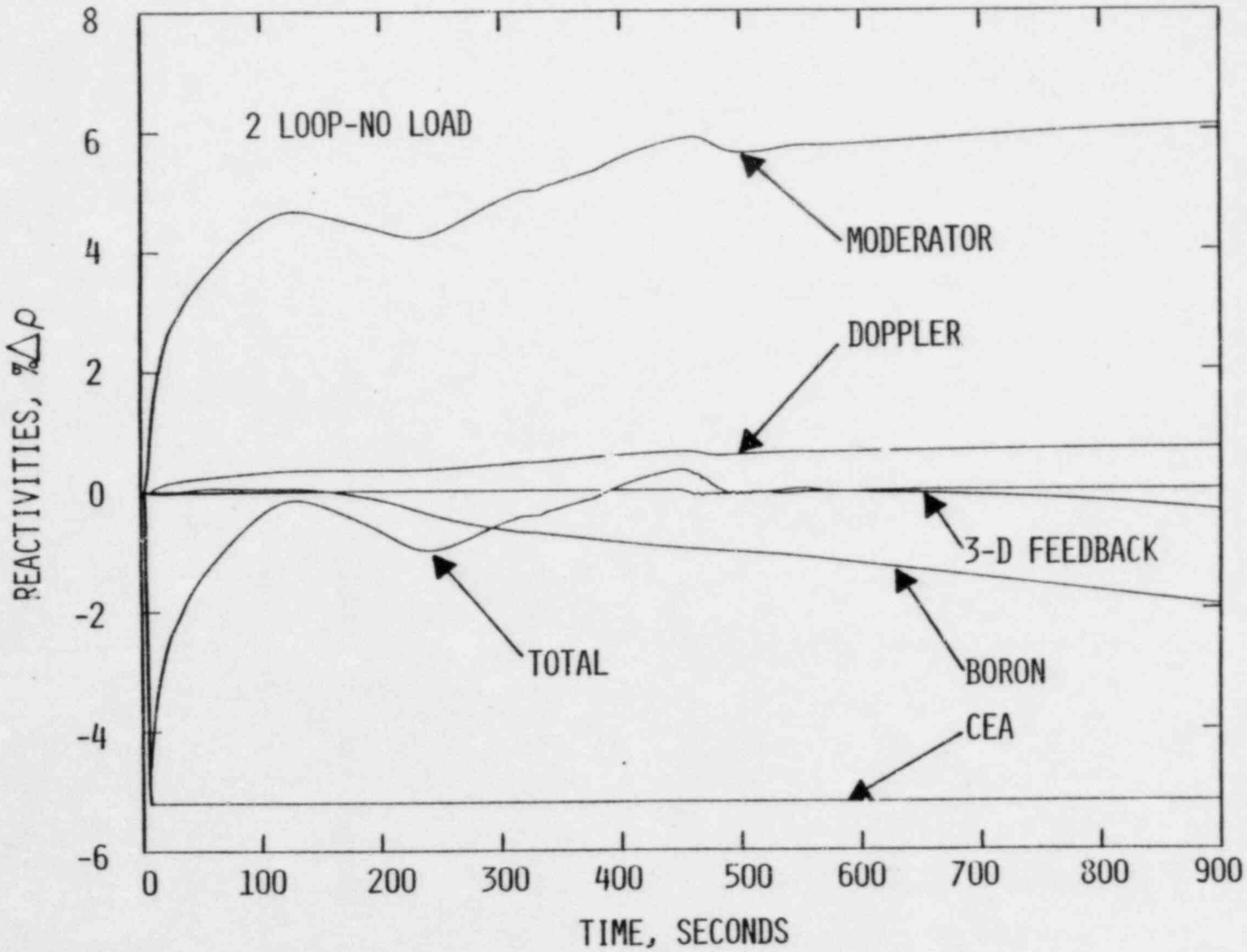
STEAM LINE BREAK EVENT
FEEDWATER ENTHALPY VS TIME

FIGURE
7.3.2-12





BALTIMORE GAS & ELECTRIC CO.
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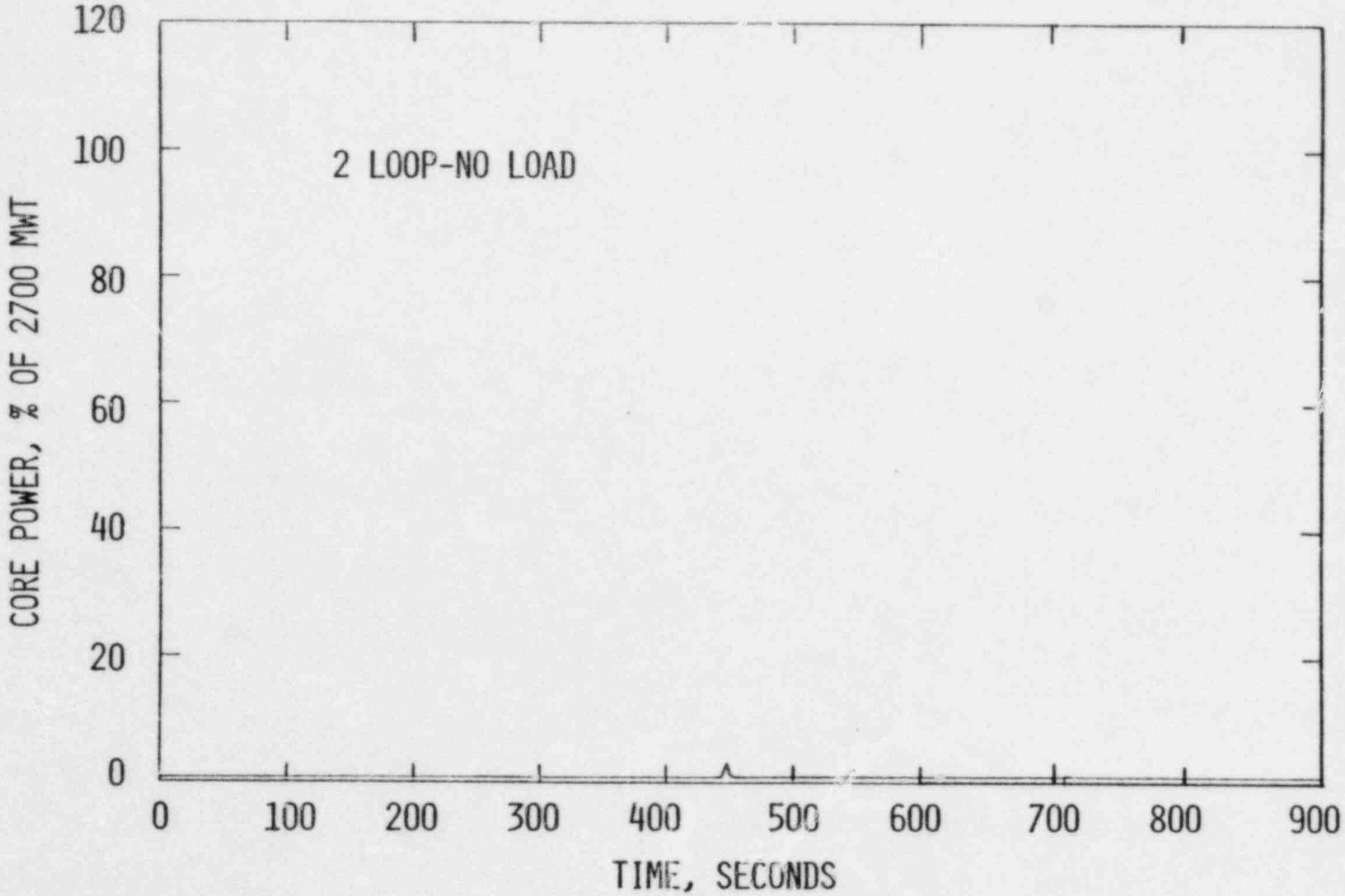
STEAM LINE BREAK EVENT
REACTIVITIES VS TIME

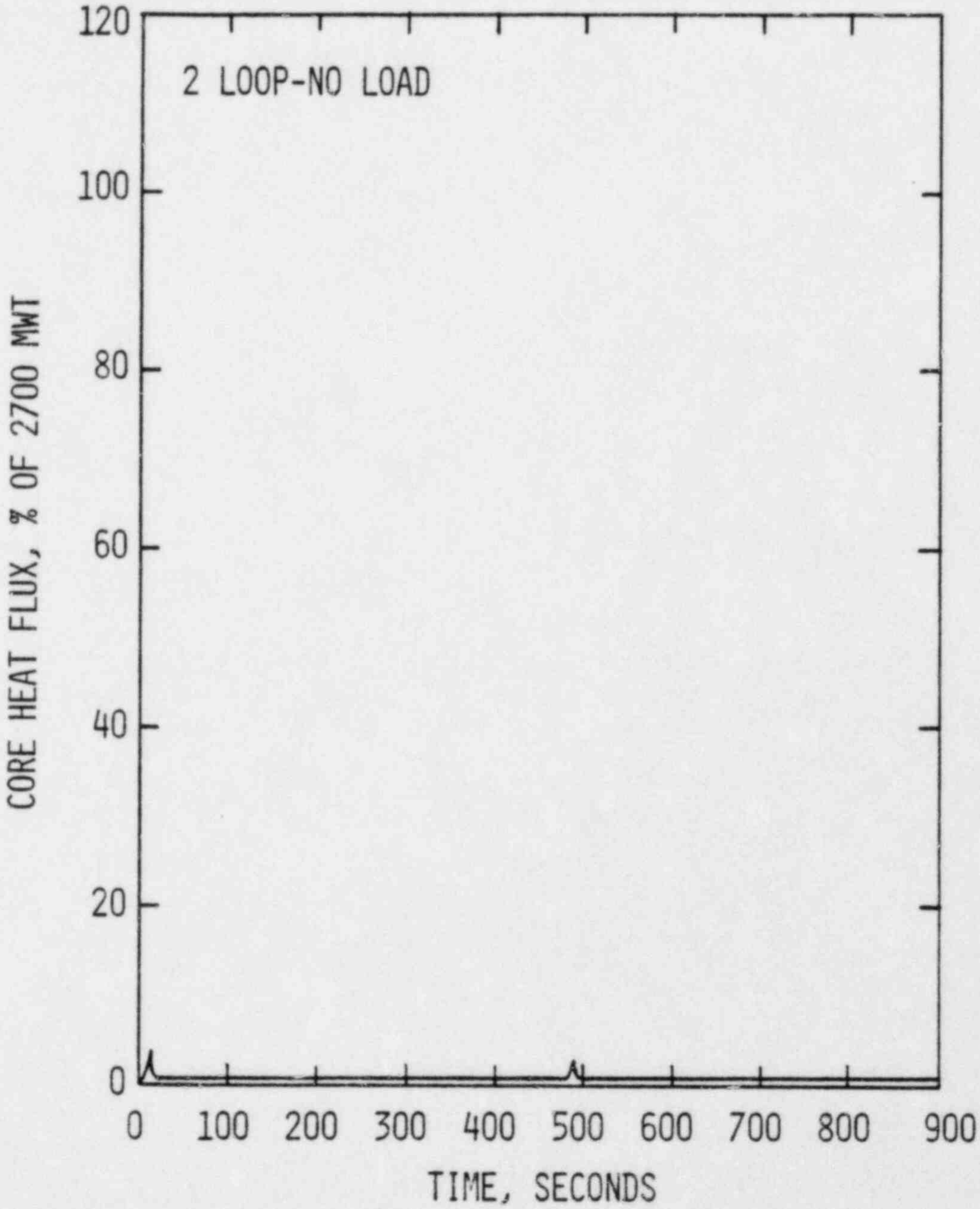
FIGURE 7.3.2-13



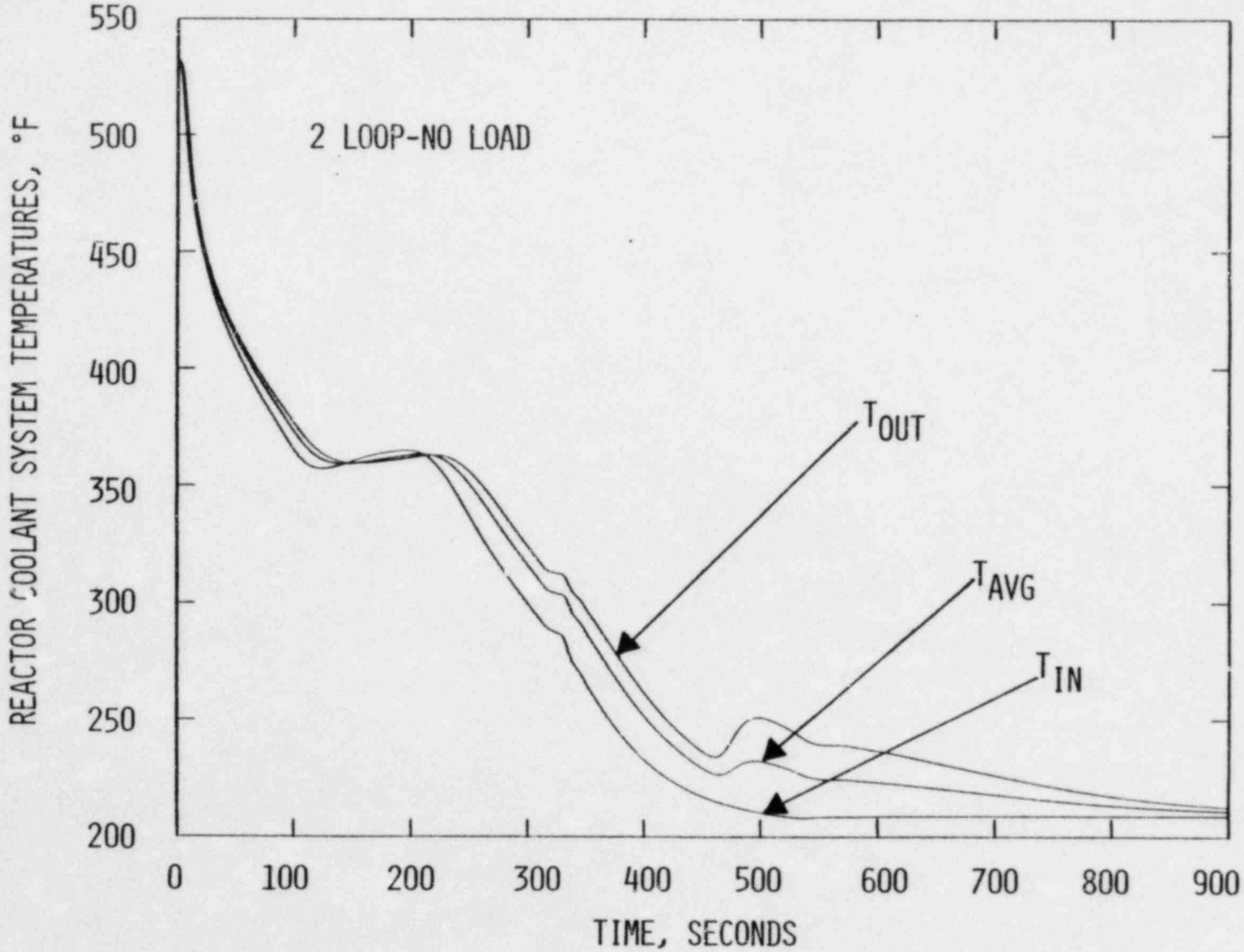
BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

STEAM LINE BREAK EVENT
CORE POWER VS TIME

FIGURE 7.3.2-14



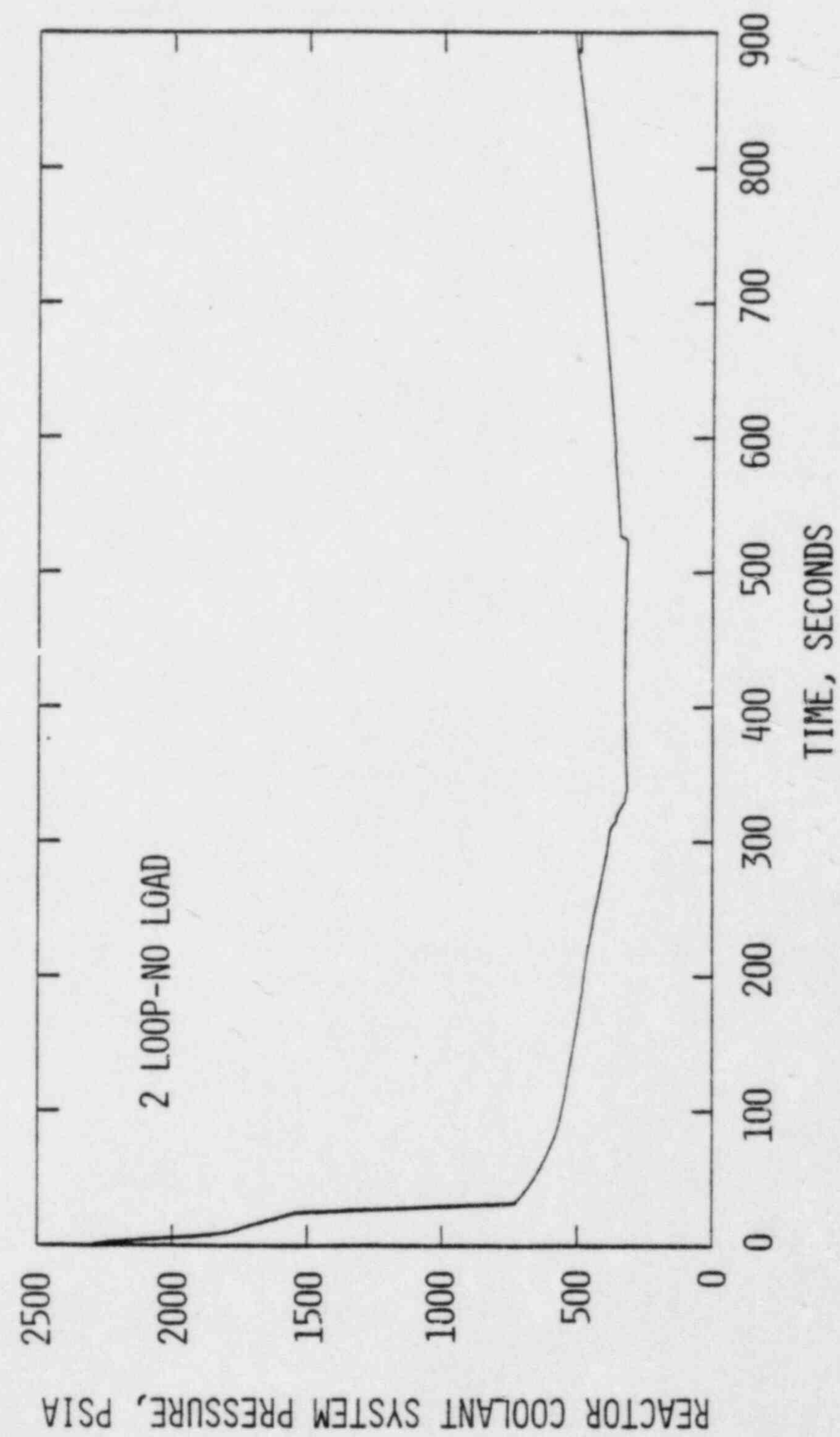
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT CORE HEAT FLUX VS TIME	FIGURE 7.3.2-15



BALTIMORE GAS & ELECTRIC CO.
Calvert Cliffs Nuclear Power Plant

STEAM LINE BREAK EVENT
REACTOR COOLANT SYSTEM TEMPERATURES VS TIME

FIGURE 7.3.2-15



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7.3.2-17

BALTIMORE
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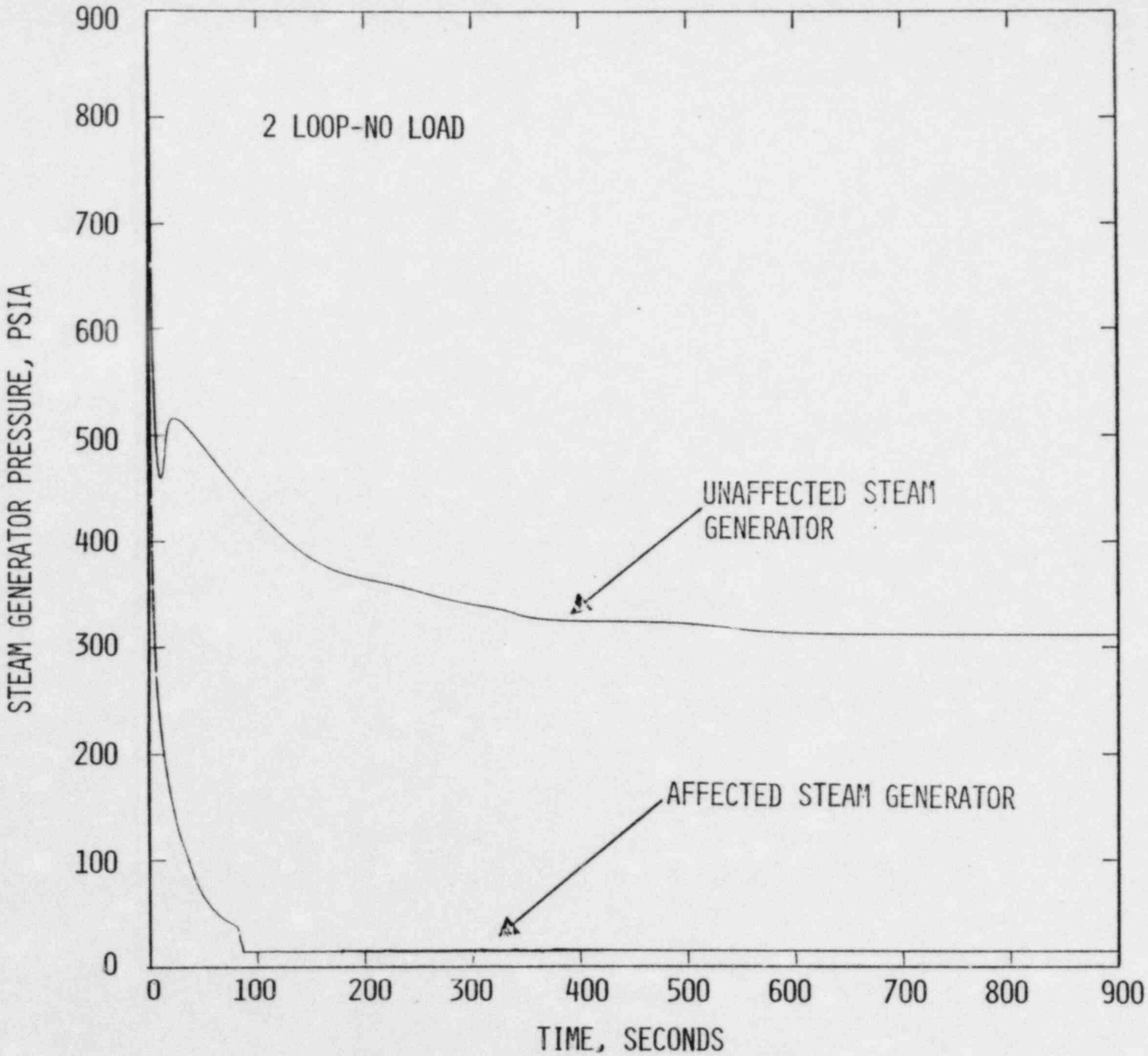
Calvert Cliffs
Nuclear Power Plant

STEAM LINE BREAK EVENT
STEAM GENERATOR PRESSURE VS TIME

FIGURE
7.3.2-18



7.3.3 Steam Generator Tube Rupture Event

The Steam Generator Tube Rupture (SGTR) event was analyzed for Unit 2, Cycle 5 to verify that the site boundary doses will not exceed the guidelines of 10CFR100 for the event initiated from a pressurizer level of 975 ft^3.

The analysis included the effects of manually tripping the Reactor Coolant Pumps on SIAS due to low pressurizer pressure.

The design basis SGTR is a double ended break of one steam generator U-tube. Table 7.3.3-1 lists the key transient related parameters used in this analysis. In the analysis, it is assumed that the initial RCS pressure is as high as 2300 psia. This initial RCS pressure maximizes the amount of primary coolant transported to the secondary steam system since the leak rate is directly proportional to the difference between the primary and secondary pressure. In addition, the higher pressure delays the lower pressurizer pressure trip which prolongs the transient and, therefore, maximizes the total primary to secondary mass and activities transported.

For this event, the acceptable DNBR limit is not exceeded due to the action of the Thermal Margin/Low Pressure (TM/LP) trip which provides a reactor trip to maintain the DNBR above 1.23. Since the SGTR event does not significantly affect the core power distribution, the PLHGR SAFDL is not approached.

The Thermal Margin/Low Pressure trip, with conservative coefficients which account for the limiting radial and axial peaks, maximum inlet temperature, RCS pressure, core power, and conservative CEA scram characteristics, would be the primary RPS trip intervening during the course of the transient. However, to maximize the coolant transported from the primary to the secondary and, thus, the radioactive steam releases to the atmosphere, the analysis was performed assuming that the reactor does not trip until the minimum setpoint (floor) of the Thermal Margin/Low Pressure trip is reached. This prolongs the steam releases to the atmosphere and, thus, maximizes the site boundary doses.

The Steam Generator Tube Rupture was analyzed assuming a manual trip of the reactor coolant pumps on Safety Injection Actuation Signal (SIAS). The Steam Generator Tube Rupture (SGTR) with RCS trip on SIAS results in higher site boundary doses because: (1) RCP coastdown increases pressure difference between the primary and the secondary, which increases the leak rate, and (2) RCP coastdown decreases the rate of decay heat removal, which increases the steam flow through the atmospheric dump valves.

The sequence of events for the SGTR event with manual trip of RCP on SIAS is presented in Table 7.3.3-2. Figures 7.3.3-1 through 7.3.3-5 present the transient behavior of core power, heat flux, RCS pressure, RCS temperatures, and steam generator pressure.

I-131 activity release is based on the primary to secondary leak and on the steam flow required to reach cold shutdown conditions. This release is calculated as the product of steam flow, the time dependent steam activity and the decontamination factors applicable to each release pathway.

The 0 to 2 hour I-131 site boundary dose is calculated from:

$$\text{DOSE (REM)} = R_{total} * BR * DCF * \chi/Q$$

where:

R_{total} = the total activity released to the atmosphere, Ci

BR = breathing rate, m^3/sec

χ/Q = atmospheric dispersion coefficient, sec/m^3

DCF = Dose Conversion Factor in equivalent I-131, Rem/Ci

In determining the whole body dose, the major assumption made is that all noble gases leaked through the ruptured tube will be released to the atmosphere. Therefore, the whole body dose is proportional to the total primary to secondary leak and is calculated using the following equation.

$$\text{Whole Body Dose} = [K_\gamma\ (\overline{E}_\gamma + \frac{K_\gamma}{K_\beta}\ \overline{E}_\beta)] * R * \chi/Q * \Delta t$$

where:

$\overline{E}_\gamma$ = the average γ energy (MEV/dis) for the halogen isotopes of concern

$\overline{E}_\beta$ = the average β energy (MEV/dis) for the halogen isotopes of concern

R = the activity released to the atmosphere, Ci/sec

χ/Q = atmospheric dispersion coefficient, sec/m^3

K_γ = $\frac{.25 \text{ Rem x } m^3 \text{ x dis}}{\text{Mev x sec x Ci}}$

K_β = $\frac{.23 \text{ Rem x } m^3 \text{ x dis}}{\text{Mev x sec x Ci}}$

Δt = time period (i.e., 0 to 2 hours), sec

The results of the analysis show that 85616 lbs. of primary coolant are transported to the steam generator secondary side. Based on this mass transport and the values in Table 7.3.3-3, the site boundary doses calculated are:

Thyroid (DEQ I-131) = 0.34 REM
Whole Body (DEQ Xe-133) = 0.18 REM

The reactor protective system (TM/LP) is adequate to protect the core from exceeding the DNBR limit. The doses resulting from the activity released as a consequence of a double-ended rupture of one steam generator tube, assuming the maximum allowable Tech. Spec. activity for the primary concentration at a core power of 2754 MWt, are significantly below the guidelines of 10CFR100.

TABLE 7.3.3-1

KEY PARAMETERS ASSUMED IN THE STEAM GENERATOR TUBE RUPTURE EVENT

KEY TRANSIENT RELATED PARAMETERS:

Parameter	Units	Reference Cycle*	Unit 2 Cycle 5
Power	MWt	2754	2754
MTC	$x10^{-4} \Delta\rho/^{o}F$	-2.5	-2.5
Doppler Coefficient Multiplier		1.15	1.15
Scram Worth	$\%\Delta\rho$	-4.3	-4.7
T_{in}	^{o}F	550	550
RCS Pressure	psia	2300	2300
Initial Core Mass Flow Rate	$x10^{6}$lbm/hr	133.9	133.9
Initial Secondary Pressure	psia	815	815
Tube ID	inches	.654	.654
Flow Constant		1.17	1.17
ASI (for scram)		+.41	+.41

*Reference cycle is Unit 1, Cycle 5 (Reference 14).

TABLE 7.3.3-2

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT
WITH RCP COASTDOWN ON SIAS

Time (sec)	Event	Setpoint or Value
0.0	Tube Rupture Occurs	---
786.5	Low Pressurizer Pressure Analysis Trip Setpiont is Reached	1728 psia
787.6	Dump Valves Open	---
787.9	CEAs Begin to Drop Into Core	---
791.3	Pressurizer Empties	---
792.6	Safety Injection Actuation Signal Generated, RCPs Manually Tripped	1556 psia
795.5	Bypass Valves Open	---
797.1	Maximum Steam Generator Pressure	906 psia
854.0	Minimum RCS Pressure	1064 psia
1800.0	Operator Isolates Damaged Steam Generator and Begins Cooldown to 300°F	---
12797.0	Operator Initiates Shutdown Cooling (T_{AV} = 300°F)	

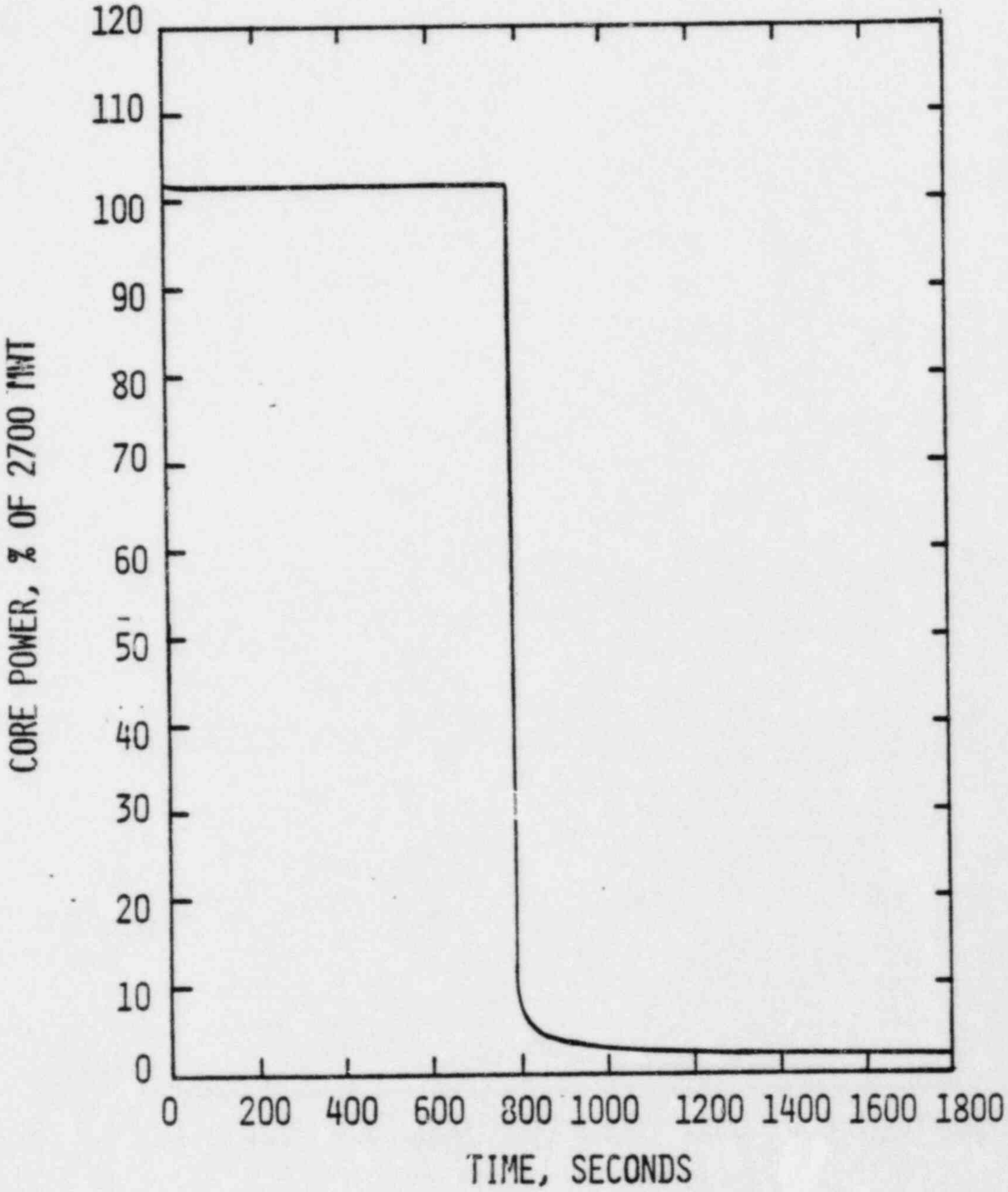
TABLE 7.3.3-3

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE STEAM GENERATOR TUBE RUPTURE

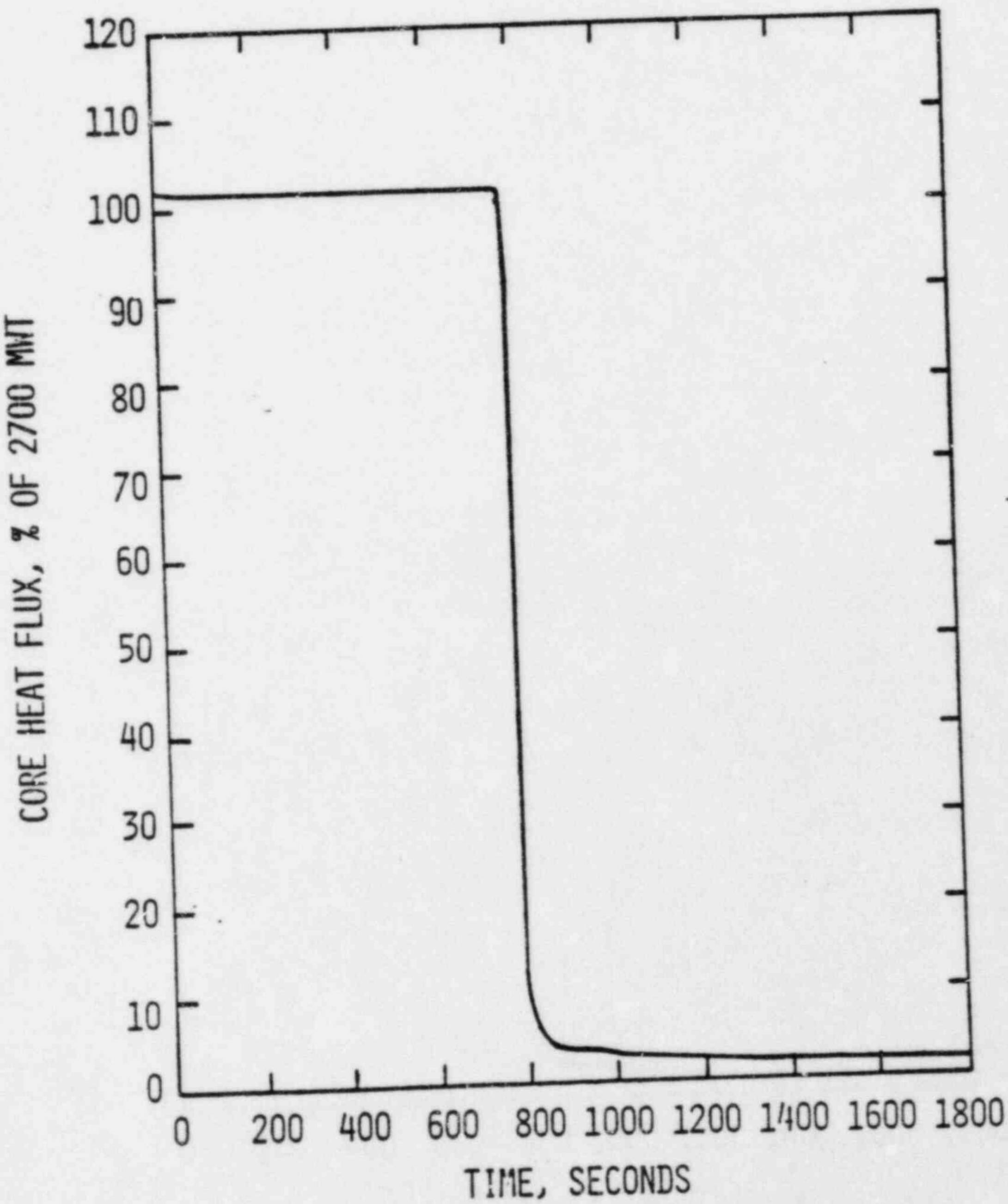
Parameter	Units	Value
Reactor Coolant System Maximum Allowable Concentration (DEQ I-131)[1]	μCi/gm	1.0
Steam Generator Maximum Allowable Concentration (DEQ I-131)[1]	μCi/gm	.1
Reactor Coolant System Maximum Allowable Concentration of Noble Gases (DEQ Xe-133)[1]	μCi/gm	$100/\bar{E}$
Steam Generator Partition Factor	---	.1
Air Ejector Partition Factor	---	.0005
Atmospheric Dispersion Coefficient[2]	sec/M^3	1.80×10^{-4}
Breathing Rate	M^3/sec	3.47×10^{-4}
Dose Conversion Factor (I-131)	REM/Ci	1.48×10^{-6}

[1]Tech. Spec. limits

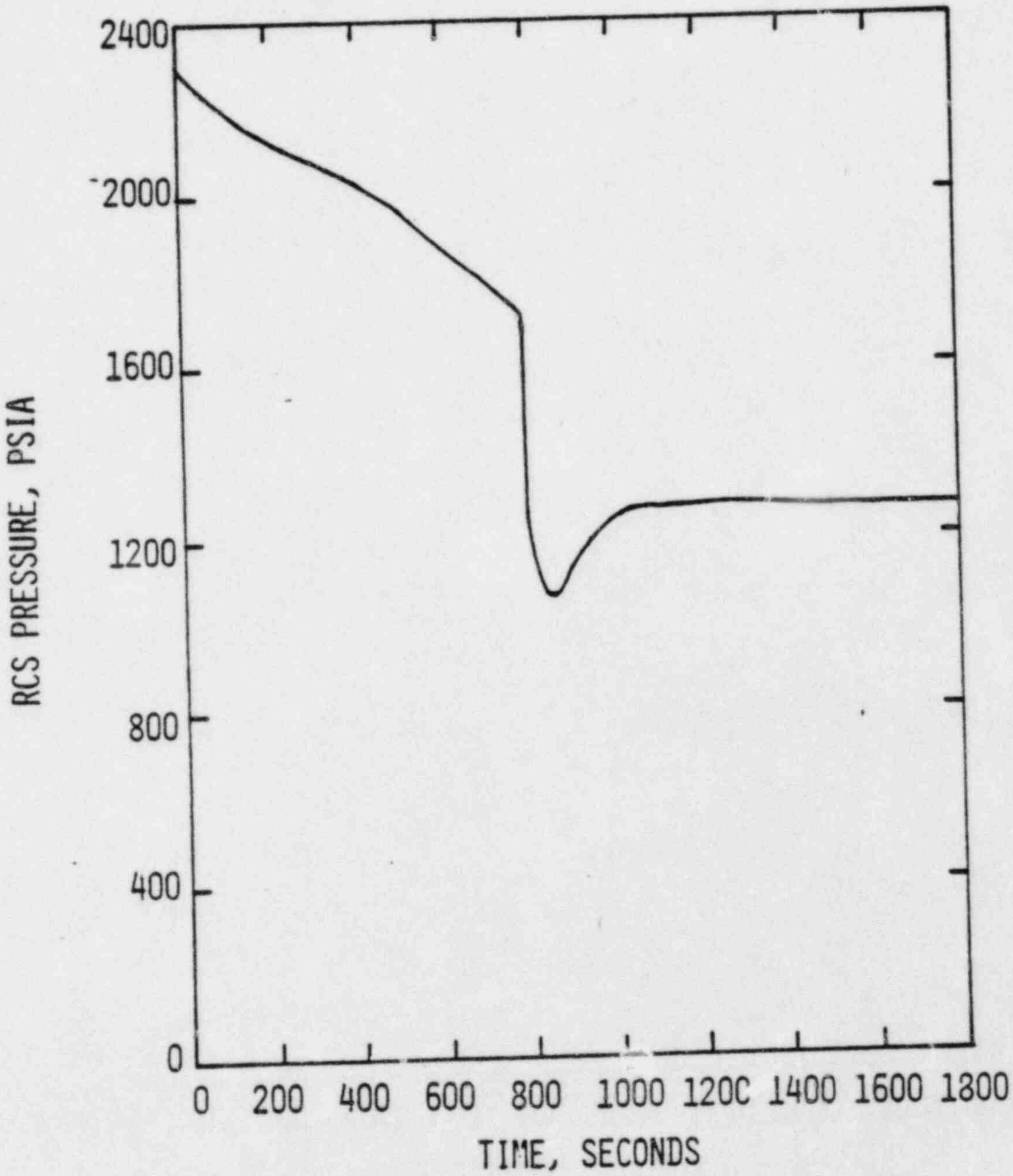
[2]0-2 hour accident condition



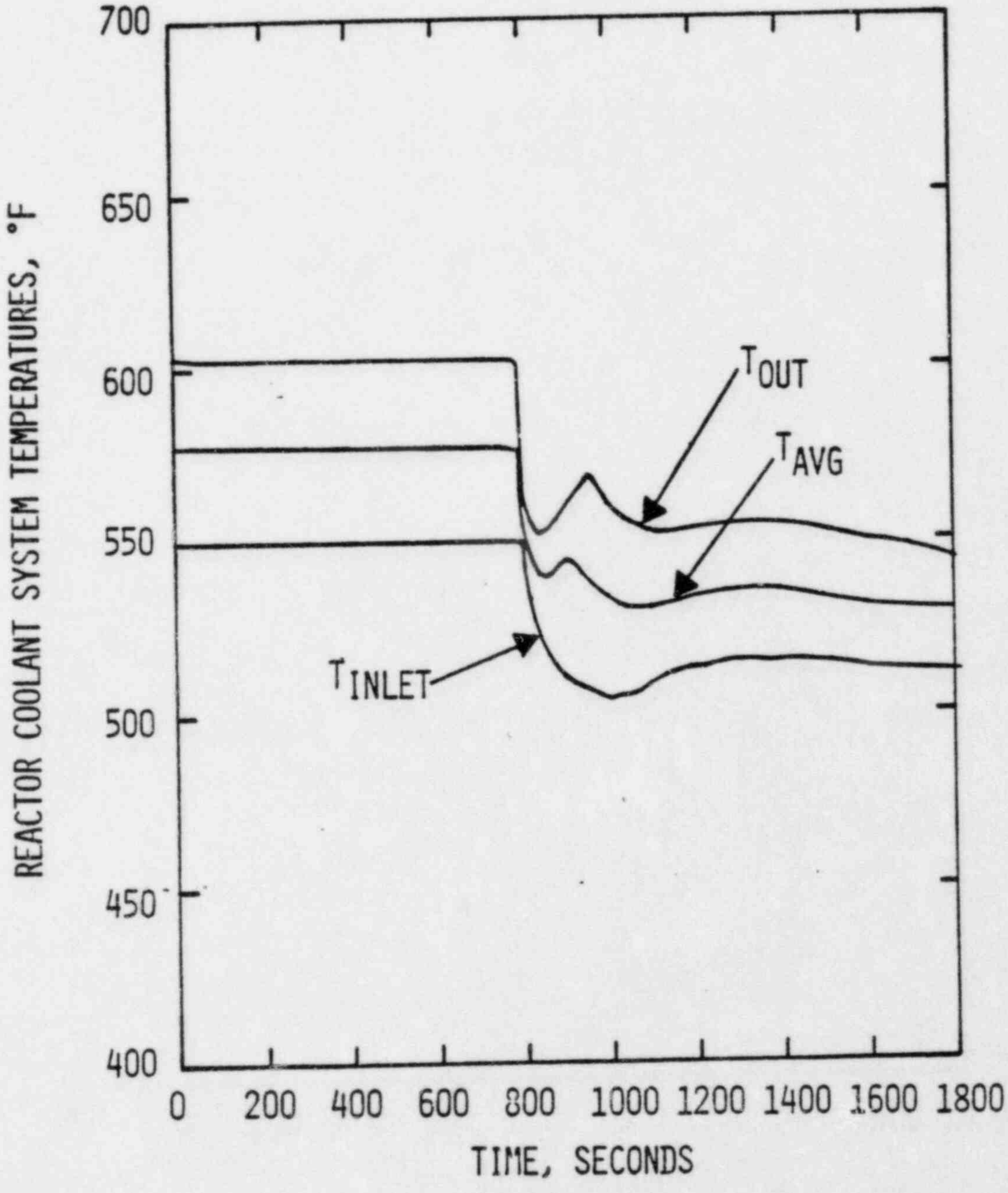
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT CORE POWER VS TIME	FIGURE 7.3.3-1



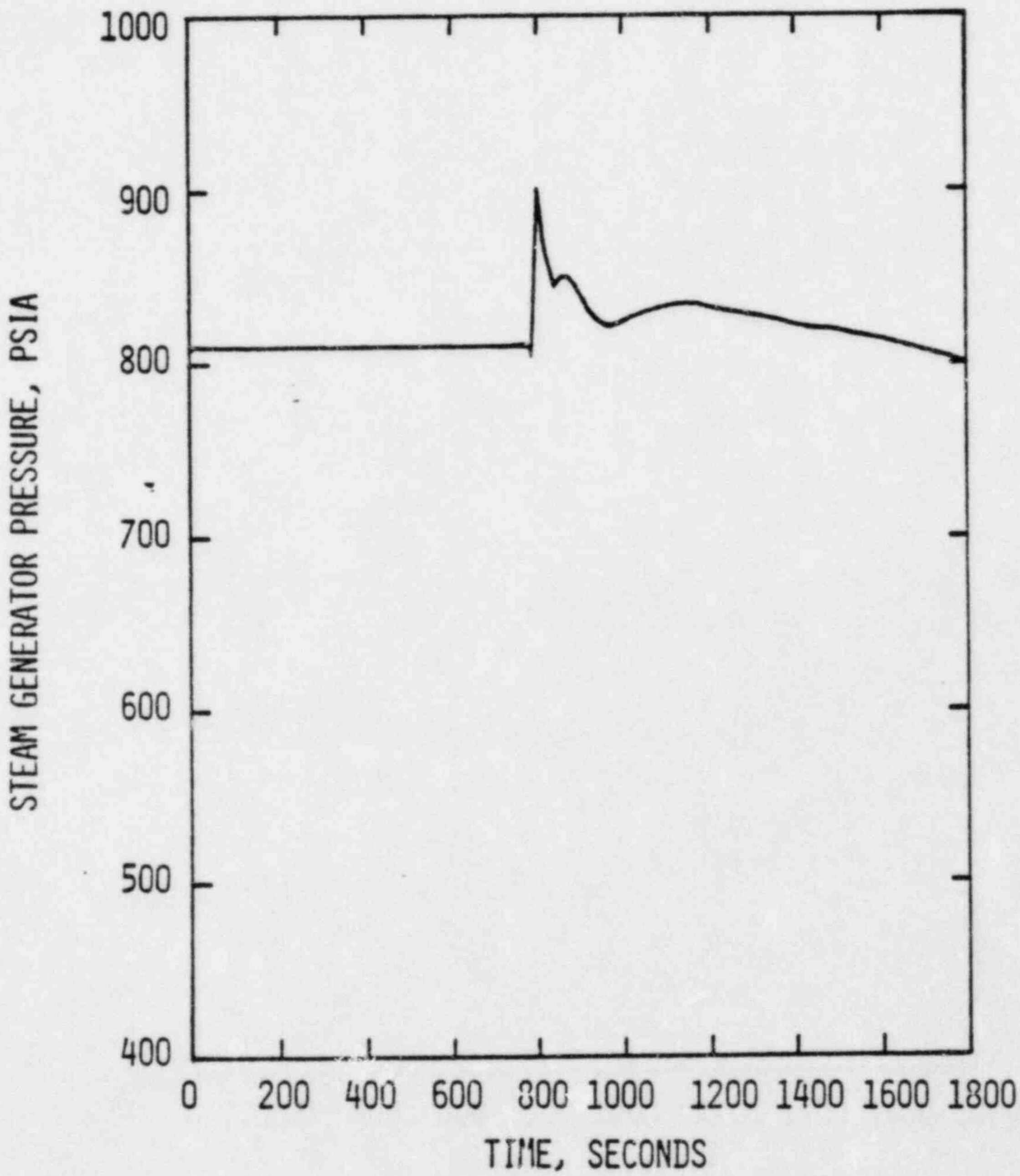
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT CORE AVERAGE HEAT FLUX VS TIME	FIGURE 7.3.3-2



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7.3.3-3



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT REACTOR COOLANT SYSTEM TEMPERATURE VS TIME	FIGURE 7.3.3-4



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 7.3.3-5

7.4.1 Fuel Handling Event

The consequences of a fuel handling incident due to increasing the burnup to 40,000 MWD/T has been investigated for Calvert Cliffs fuel. The results of this investigation demonstrate that the results shown in Reference 15 will not be changed by increasing the burnup.

The dose rate at the site boundary will not increase because the gas gap inventory will be less than the gas gap inventory shown in Reference 15. The Reference 15 gas gap activity is based on the hottest fuel assembly in the core, independent of time (burnup) during the fuel cycle. Since the radioactive fission products which contribute significantly to the dose rate at the site boundary reach maximum concentrations at relatively low burnups, the only significant influence of burnup is the increased release from the fuel pellet for a given fuel temperature, commonly called "enhancement". The fuel temperature, in turn, is principally dependent on linear heat rate.

The predicted linear heat rate for Calvert Cliffs Unit 2, Cycle 5 fuel rods has been calculated to determine the radioactive fission product release to the gas gap. The maximum linear heat rate for rods with burnups between 33,000 MWD/T and 40,000 MWD/T is about 7.5 KW/ft; this maximum occurs at 33,000 MWD/T and the linear heat rate decreases to less than 6.5 KW/ft for the maximum fuel rod exposure at EOC5. The maximum fuel temperature is not high enough to have significant diffusion-type release from the fuel; the method of release will be primarily from knock-out or recoil. Consequently, the radioactive fission product release to the gas gap will be less than 1% of the inventory; this release is based on the ANS 5.4 Standard, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel." The release of less than 1% is over a factor of ten lower than that assumed in the fuel handling accident in Reference 15.

8.0 ECCS Analysis

An ECCS performance analysis was performed for Calvert Cliffs Unit 2 Cycle 5 to demonstrate compliance with 10CFR50.46 which presents the NRC Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Reactors (1). The analysis justifies an allowable peak linear heat generation rate (PLHGR) of 15.5 kw/ft which is equal to the existing limit for Unit 2.

The ECCS analysis performed for Unit 2 Cycle 5 used input data, including the reload fuel performance parameters, which apply to both Unit 2 Cycle 5 and Unit 1 Cycle 6. Reference 2 contains the details of the method, results and conclusions of the analysis. They are directly applicable to both Unit 2 Cycle 5 and Unit 1 Cycle 6.

The results of the analysis identified the peak clad temperature to be 2038°F, compared to the acceptance criteria limit of 2200°F. The peak local clad oxidation was 8.5% and the peak core wide clad oxidation was less than 0.51% versus the acceptance criteria limits of 17% and 1.0%, respectively. Hence, Unit 2 Cycle 5 operation at a peak linear heat generation rate of 15.5 kw/ft and at a power level of 2754 MWt (102% of 2700 MWt) will result in acceptable ECCS performance.

9.0 Technical Specifications

The Technical Specification changes which must be made in order to make the Calvert Cliffs Unit 2 Technical Specifications valid for the operation of Cycle 5 are presented in this section. Table 9-1 presents a summary of the Technical Specification changes. Table 9-2 presents the explanations for the changes summarized in Table 9-1.

The requested Technical Specification modifications for Unit 2 Cycle 5 (Table 9-1) are very similar to those changes requested for the reference cycle (Unit 1 Cycle 6, References 1 and 2). The differences between the Technical Specification modifications being requested herein for Unit 2 Cycle 5 and those requested for Unit 1 Cycle 6 are explained in Table 9-3. The most noteworthy of these differences are:

1. The centerline melt limit is being increased to 22.0 kw/ft, compared to the Unit 1 Cycle 6 value of 21.3 kw/ft, to increase operating margins and flexibility.

2. The MTC surveillance requirements are being modified, compared to no change for Unit 1 Cycle 6, to allow the use of MTC determinations made during power ascension startup measurements for the purpose of satisfying surveillance requirements.

3. The minimum pressurizer pressure is being decreased from 2225 psia to 2200 psia, compared to no change for Unit 1 Cycle 6, to improve operating flexibility.

4. The F_{xy}^{T} Technical Specification is being increased to 1.70, compared to the Unit 1 Cycle 6 value of 1.65, to increase operating margins and flexibility.

5. A relaxed pressurizer level band is being incorporated, compared to no change for Unit 1 Cycle 6, to improve operating flexibility.

Following Table 9-3, for each Technical Specification which must be modified, either the existing page with the intended modification or the already modified page with a new figure is provided.

Table 9-1

Calvert Cliffs II Cycle 5
Technical Specification Changes

Change	Tech Spec #	Action
1	Figure 2.1-1, page 2-2	Replace Figure 2.1-1 with Figure 2.1-1
2	Table 2.2-1 page 2-9	Change Steam Generator Low Pressure Trip setting from 570 psia to 635 psia
3	Table 2.2-1 page 2-10	Change Steam Generator Low Pressure Trip Bypass limit from 685 psia to 710 psia
4	Figure 2.2-1 page 2-11	Replace Figure 2.2-1 with Figure 2.2-1
5*	B.2.1.1 page B2-1	Change LHGR to centerline melt limit from 21 kw/ft to 22.0 kw/ft.
6	B.2.1.1, B.2.2.1 pages B2-1, B2-3 B2-5, B2-6	Change minimum DNBR value from 1.195 to 1.23 as indicated on noted pages
7	B.2.1.1, B.2.2.1 pages B2-1, B2-4	Change high power level trip and maximum high power level trip actuation from 112% of rated thermal power to 110%
8	B.2.2.1 page B2-5	Change Steam Generator Low Pressure Trip setting from 570 psia to 635 psia, change uncertainty from 22 psia to 87 psia, and revise description of uncertainty.
9	B.2.2.1 page B 2-6	Change the minimum trip setting (floor) of the Thermal Margin/Low Pressure (TM/LP) Trip from 1750 psia to 1875 psia
10	B.2.2.1 page B2-7	Revise description of TM/LP trip, and change allowance from 92 psia to 40 psia
11*	3/4.1.1.1 page 3/4 1-1	Change shutdown margin, T_{avg} >200°F, from 4.3% Δk/k to 5.2% Δk/k
12*	4.1.1.4.2 page 3/4 1-6	Change MTC surveillance Item (b)
13	Figure 3.1-2 page 3/4 1-27	Replace Figure 3.1-2 with Figure 3.1-2
14	4.2.1.3 page 3/4 2-2	Change Figure 3.2-3 to Figure 3.2-3b

Table 9-1 (continued)

Change	Tech Spec #	Action
15	Figure 3.2-2 page 3/4 2-4	Replace Figure 3.2-2 with Figure 3.2-2
16*	Figure 3.2-3b page 3/4 2-4a (new)	Insert new Figure 3.2-3b after Page 3/4 2-4
17*	3/4.2.2.1 (new) 3/4.2.2 (old) pages 3/4 2-6, 3/4 2-7	Change calculated value of F_{xy}^T from 1.620 to 1.700, change Figure 3.2-3 to Figure 3.2-3a and change Tech Spec numbers from 3/4.2.2 to 3/4.2.2.1
18*	Figure 3.2-3a (new) 3.2-3 (old) page 3/4 2-7a (new) 2-8 (old)	Replace Figure 3.2-3 with new Figure 3.2-3a and change page number from 3/4 2-8 to 3/4 2-7a
19*	3/4.2.2.2 (new) page 3/4 2-8 (new)	Insert new Page 3/4 2-8 after Page 3/4 2-7a
20*	3.2.3 page 3/4 2-9	Change calculated value of F_r^T from 1.620 to 1.650, insert Action Item (b) and change Figure 3.2-3 to 3.2-3c
21*	Figure 3.2-3c page 3/4 2-10a (new)	Insert new Figure 3.2-3c after Page 3/4 2-10
22	Figure 3.2-4 page 3/4 2-11	Replace Figure 3.2-4 with Figure 3.2-4
23*	3.2.5 page 3/4 2-13	Add ",Core Power" to Item (d)
24*	Table 3.2-1 page 3/4 2-14	Change minimum pressurizer pressure from 2225 psia to 2200 psia, change AXIAL SHAPE INDEX limit from Figure 3.2-4 to "***" and add footnote for "***"
25	Table 3.3-1 page 3/4 3-4	Change Steam Generator Low Pressure Trip Bypass limit from 685 psia to 710 psia
26	Table 3.3-2 page 3/4 3-6	Change RTD response time from 8.0 seconds to 12.0 seconds
27	Table 3.3-3 page 3/4 3-15	Change Safety Injection (SIAS) Pressurizer Low Pressure Trip Bypass limit from 1700 psia to 1800 psia
28	Table 3.3-3 page 3/4 3-15	Change Main Steam Line Isolation (SGIS) Steam Generator Low Pressure Trip Bypass limit from 685 psia to 710 psia

Table 9-1 (continued)

Change	Tech Spec #	Action
29	Table 3.3-4 page 3/4 3-17	Change Safety Injection (SIAS) Pressurizer Low Pressure Trip setting from 1578 psia to 1725 psia
30	Table 3.3-4 page 3/4 3-17	Change Main Steam Line Isolation (SGIS) Steam Generator Low Pressure Trip setting from 570 psia to 635 psia
31*	3/4.4.4 page 3/4 4-5	Change the description and limits of the pressurizer level operating band
32*	B 3/4.1.1.1 and B 3/4.1.1.2 page B 3/4 1-1	Change EOC shutdown margin, T_{avg} $>200^{o}F$, from 4.3% $\Delta k/k$ to 5.2% $\Delta k/k$, and change BOC shutdown requirement to 4.5% $\Delta k/k$
33	B 3/4.2.5 page B 3/4 2-2	Change minimum DNBR of 1.195 to minimum DNBR of 1.23
34*	B 3/4.4.4 page B 3/4 4-2	Change the description concerning pressurizer level
35*	5.3.1 page 5-4	Increase limit on enrichment in description of reload fuel assemblies from 3.7 w/o U-235 to 4.1 w/o U-235.

*Added or modified request relative to Reference 1 or 2. Explanation for addition or modification is contained in Table 9-3.

Table 9-2

Explanations for Cycle 5 Tech Spec Changes

Change	Tech Spec #	Explanation
1	Figure 2.1-1	Thermal Limit Lines have been changed to reflect higher radial peaking factors and the implementation of margin recovery programs.
2	Table 2.2-1	The Steam Generator Low Pressure Trip setting is being raised to accommodate the larger uncertainties associated with the new pressure transmitters.
3	Table 2.2-1	The Steam Generator Low Pressure Trip Bypass limit is being raised to reflect the change in the trip setting (See Change No. 2)
4	Figure 2.2-1	The LHR LSSS has been changed to reflect higher radial peaking factors and the implementation of margin recovery programs
5*	B.2.1.1	LHGR to centerline melt is being raised to increase operating margins and flexibility
6	B.2.1.1, B.2.2.1	The minimum DNBR has been increased to 1.23 to be consistent with Statistical Combination of Uncertainties
7	B.2.1.1, B.2.2.1	Statistical Combination of Uncertainties has removed the 2% power uncertainty from the transient analyses
8	B.2.2.1	See Change No. 2
9	B.2.2.1	The minimum trip setting (floor) of the Thermal Margin/Low Pressure (TM/LP) Trip is being raised to accommodate the larger uncertainties associated with the new pressurizer pressure transmitters.
10	B.2.2.1	The TM/LP basis has been adjusted to be consistent with Statistical Combination of Uncertainties (SCU) and the bias has been changed as a result of the implementation of SCU and the recatagorization of CEAW (A conservative bias value relative to the Transient Analysis results has been incorporated)

Table 9-2 (continued)

Change	Tech Spec #	Explanation
11*	3/4.1.1.1	The shutdown margin has been increased to yield acceptable results for the EOC HZP SLB event.
12*	4.1.1.4.2	The surveillance requirements on MTC are being modified to allow the use of MTC determinations made during power ascension startup measurements for the purpose of satisfying surveillance requirements. This change is consistent with the objective of assuring that the most positive MTC at power conditions, which occurs at the highest boron concentration, meets Tech Spec. #3.1.1.4.b.
13	Figure 3.1-2	The PDIL is being changed to increase available SCRAM worth and produce acceptable results for the EOC HZP SLB event; the allowable BASSS operating region is being indicated.
14	4.2.1.3	A new figure is being added to take credit, in terms of maximum allowable fraction of RATED THERMAL POWER, for the calculated value of F_{xy}^T when monitoring the LHR LCO with the ex-core detector system. This change (#14) and Change Nos. 15 through 19 are being made to avoid unnecessary power level changes resulting from temporary on-line computer outages.
15	Figure 3.2-2	The LHR LCO is being changed as a result of higher radial peaks, the implementation of margin recovery programs, and the addition of Figure 3.2-3b to take credit for the calculated value of F_{xy}^T when monitoring the LHR LCO with the excore detector system.
16*	Figure 3.2-3b	Figure 3.2-3b is being added to take credit for the calculated value of F_{xy}^T when monitoring the LHR LCO with the excore detector system.
17*	3/4.2.2.1	The planar radial peaking factor, F_{xy}^T, is being raised for Cycle 5 to increase operating margins and flexibility, and the Tech Spec and figure numbers are being changed to facilitate other Tech Spec changes which are being made to take credit for the calculated value of F_{xy}^T when monitoring the LHR LCO with the ex-core detector system.

Table 9-2 (continued)

Change	Tech Spec #	Explanation
18*	Figure 3.2-3a	The planar radial peaking factor, F_{xy}^T, is being raised for Cycle 5 to increase operating margins and flexibility, the F_{xy}^T curve is being separated from the F_r^T curve to accommodate different F_{xy}^T and F_r^T values, and F_{xy}^T limits, based upon ex-core monitoring of the LHR LCO, are being separated from this figure to facilitate other Tech Spec changes which are being made to take credit for the calculated value of F_{xy}^T when monitoring the LHR LCO with the excore detector system.
19*	3/4 2.2.2	A new Tech Spec is being added to take credit for the calculated value of F_{xy}^T when monitoring the LHR LCO with the ex-core detector system.
20*	3.2.3	The integrated radial peaking factor, F_r^T, is being raised for Cycle 5 to increase operating margins and flexibility, BASSS is being implemented, and the F_r^T curve is being separated from the F_{xy}^T curve to accommodate different F_{xy}^T and F_r^T values.
21*	Figure 3.2-3c	The integrated radial peaking factor, F_r^T, is being raised for Cycle 5 to increase operating margins and flexibility, and the F_r^T curve is being separated from the F_{xy}^T curve to accommodate different F_{xy}^T and F_r^T values.
22	Figure 3.2-4	The DNB LCO has been changed to reflect higher radial peaking factors and the implementation of margin recovery programs.
23*	3.2.5	The phrase "Core Power" is being added to reflect the implementation of BASSS
24*	Table 3.2-1	The minimum steady state pressurizer pressure has been lowered to increase operating flexibility and the ASI limits have been modified for the implementation of BASSS
25	Table 3.3-1	See Change No. 3

Table 9-2 (continued)

Change	Tech Spec #	Explanation
26	Table 3.3-2	The RTD delay time used in the Cycle 5 analysis has been increased from 8 to 12 seconds to increase the acceptance test criteria
27	Table 3.3-3	The Safety Injection (SIAS) Pressurizer Low Pressure Trip Bypass limit is being raised to reflect the change in the trip setting (See Change No. 29)
28	Table 3.3-3	The Main Steam Line Isolation (SGIS) Steam Generator Low Pressure Trip Bypass limit is being raised to reflect the change in the trip setting (See Change No. 30)
29	Table 3.3-4	The Safety Injection (SIAS) Pressurizer Low Pressure Trip setting is being raised to accommodate the larger uncertainties associated with the new pressure transmitters.
30	Table 3.3-4	The Main Steam Line Isolation (SGIS) Steam Generator Low Pressure Trip setting is being raised to accommodate the larger uncertainties associated with the new pressure transmitters.
31*	3/4.4.4	A relaxed pressurizer level operating band is being incorporated to improve operating flexibility.
32*	B 3/4.1.1.1 and B 3/4.1.1.2	The shutdown margin has been increased to make it consistent with Specification 3/4.1.1.1
33	B 3/4.2.5	The minimum DNBR has been increased to be consistent with Tech Spec B.2.1.1 and B.2.2.1.
34*	B 3/4.4.4	The bases for the pressurizer level operating band are being modified to support the expanded limits.
35*	5.3.1	The specification of the enrichment limit in reload fuel assemblies has been increased to permit the use of higher enrichment fuel.

*Added or modified request relative to Reference 1 or 2. Explanation for addition or modification is contained in Table 9-3.

Table 9-3

Explanation for Changes Relative to Those Requested for Unit 1 Cycle 6

Change	Explanation
5	The centerline melt limit is being increased to 22.0 kw/ft, compared to the Unit 1 Cycle 6 value of 21.3 kw/ft, to increase operating margins and flexibility.
11	The Unit 2 Cycle 5 analyses support a shutdown margin of 5.2% $\Delta k/k$, compared to the Unit 1 Cycle 6 value of 5.3% $\Delta k/k$.
12	The MTC surveillance requirements are being modified, compared to no change for Unit 1 Cycle 6, to allow the use of MTC determinations made during power ascension startup measurements for the purpose of satisfying surveillance requirements. This change is consistent with the objective of assuring that the most positive MTC at power conditions, which occurs at the highest boron concentration, meets Tech Spec #3.1.1.4.b.
16	Figure 3.2-3b is being modified to be consistent with an F_{xy}^T value of 1.70, compared to the Unit 1 Cycle 6 F_{xy}^T value of 1.65; the requested page number has been changed to be consistent with the issued page number in the Unit 1 Cycle 6 SER (Reference 3).
17	The F_{xy}^T is being increased to 1.70, compared to the Unit 1 Cycle 6 value of 1.65, to increase operating margins and flexibility.
18	Figure 3.2-3a is being modified to be consistent with an F_{xy}^T value of 1.70, compared to the Unit 1 Cycle 6 value of 1.65, and to separate the F_{xy}^T curve from the F_r^T curve; the requested page number has been changed to be consistent with the issued page number in the Unit 1 Cycle 6 SER (Reference 3).
19	Requested page number has been changed to be consistent with the issued page number in the Unit 1 Cycle 6 SER (Reference 3).
20	Figure 3.2-3 is being changed to Figure 3.2-3c, compared to Figure 3.2-3a for Unit 1 Cycle 6, to facilitate the separation of the F_{xy}^T and F_r^T curves.
21	Figure 3.2-3c is being added to facilitate the separation of the F_{xy}^T and F_r^T curves due to the difference between the Unit 2 Cycle 5 F_{xy}^T and F_r^T values, compared to Unit 1 Cycle 6 which grouped both parameters into one curve on a single figure due to their similarity.

Table 9-3 (continued)

Explanation for Changes Relative to Those Requested for Unit 1 Cycle 6

Change	Explanation
23	The Technical Specification No. and page number are being changed to be consistent with existing Technical Specification (Unit 1 Cycle 6 T.S. No. and page number were out of date in request).
24	The minimum pressurizer pressure is being decreased from 2225 psia to 2200 psia, compared to no change for Unit 1 Cycle 6, to improve operating flexibility; the page number has been changed to be consistent with existing Technical Specifications (Unit 1 Cycle 6 page number was out of date in request).
31	A relaxed pressurizer level operating band is being incorporated, compared to no change for Unit 1 Cycle 6, to improve operating flexibility.
32	Same as No. 11
34	The bases for the pressurizer level operating band are being modified, compared to no change for Unit 1 Cycle 6, to support the expanded limits.
35	The Unit 2 Cycle 5 request has been modified, relative to the Unit 1 Cycle 6 written request, to conform to the verbally agreed upon and subsequently issued change for Unit 1 Cycle 6 (Reference 3). The initial written request asked for the removal of the enrichment limit; the agreed upon change simply raised the enrichment limit from 3.7 w/o U-235 to 4.1 w/o U-235.

FIGURE 2.1-1

REACTOR CORE THERMAL MARGIN SAFETY LIMIT-
FOUR REACTOR COOLANT PUMPS OPERATING

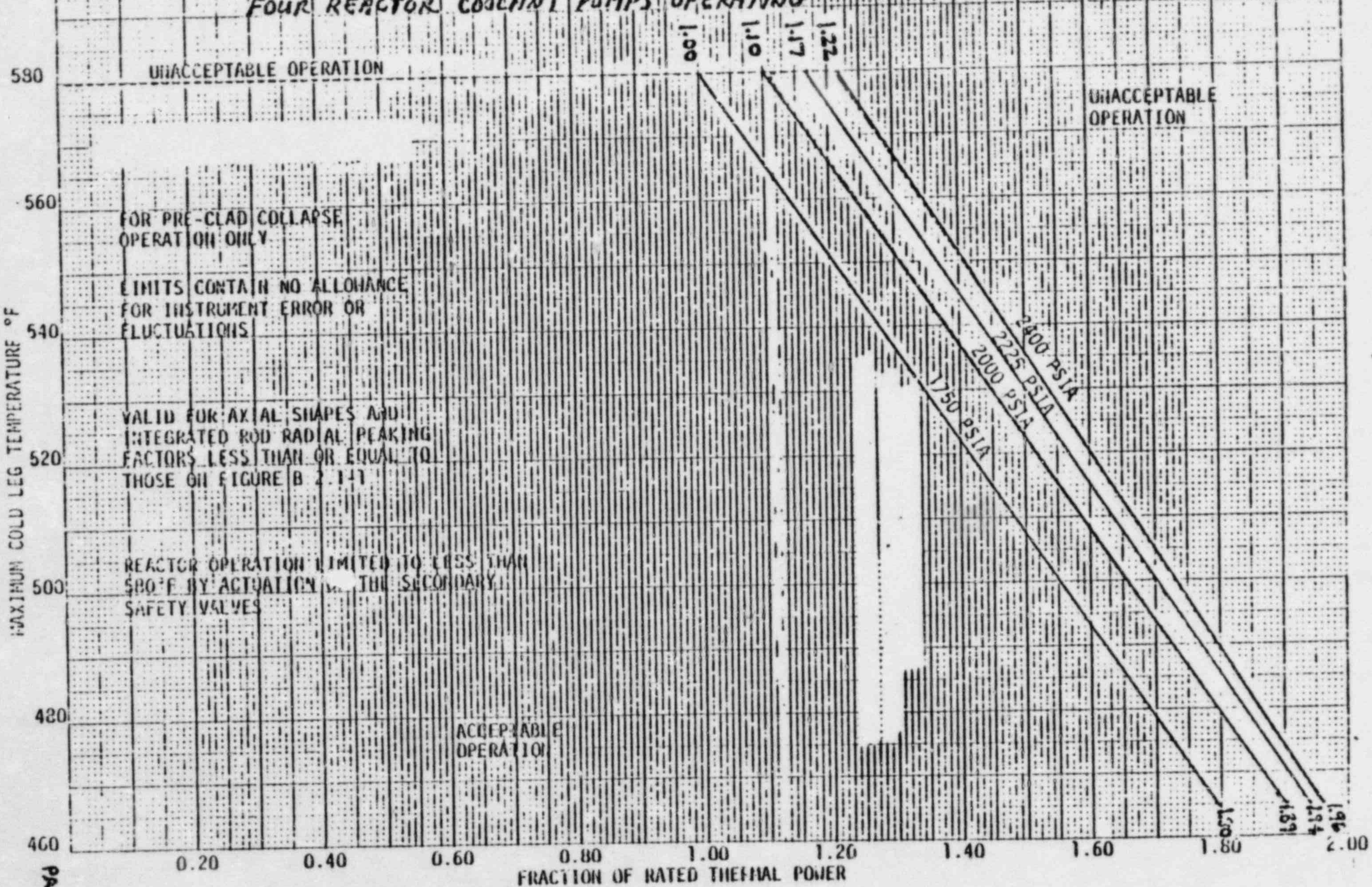


TABLE 2.2-1 (Cont'd)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Pressurizer Pressure - High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure - High	≤ 4 psig	≤ 4 psig
6. Steam Generator Pressure - Low (2)	≥ 570 → 635 psia	≥ 570 → 635 psia
7. Steam Generator Water Level - Low	≥ 10 inches below top of feed ring.	≥ 10 inches below top of feed ring.
8. Axial flux offset (3)	Trip setpoint adjusted to not exceed the limit lines of Figure 2.2-1.	Trip setpoint adjusted to not exceed the limit lines of Figure 2.2-1.
9. Thermal Margin/Low Pressure (1)		
a. Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-2 and 2.2-3.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-2 and 2.2-3.
b. Steam Generator Pressure Difference - High (1)	≤ 135 psid	≤ 135 psid
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	≥ 1100 psig	≥ 1100 psig
11. Rate of Change of Power - High (4)	≤ 2.6 decades per minute	≤ 2.6 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 10^{-4}% of RATED THERMAL POWER.

<u>TABLE 2.2-1</u> (Cont'd)

<u>TABLE NOTATIONS</u> (Cont'd)

(2) Trip may be manually bypassed below 685 → 710 psia; bypass shall be automatically removed at or above 685 → 710 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 12% of RATED THERMAL POWER.

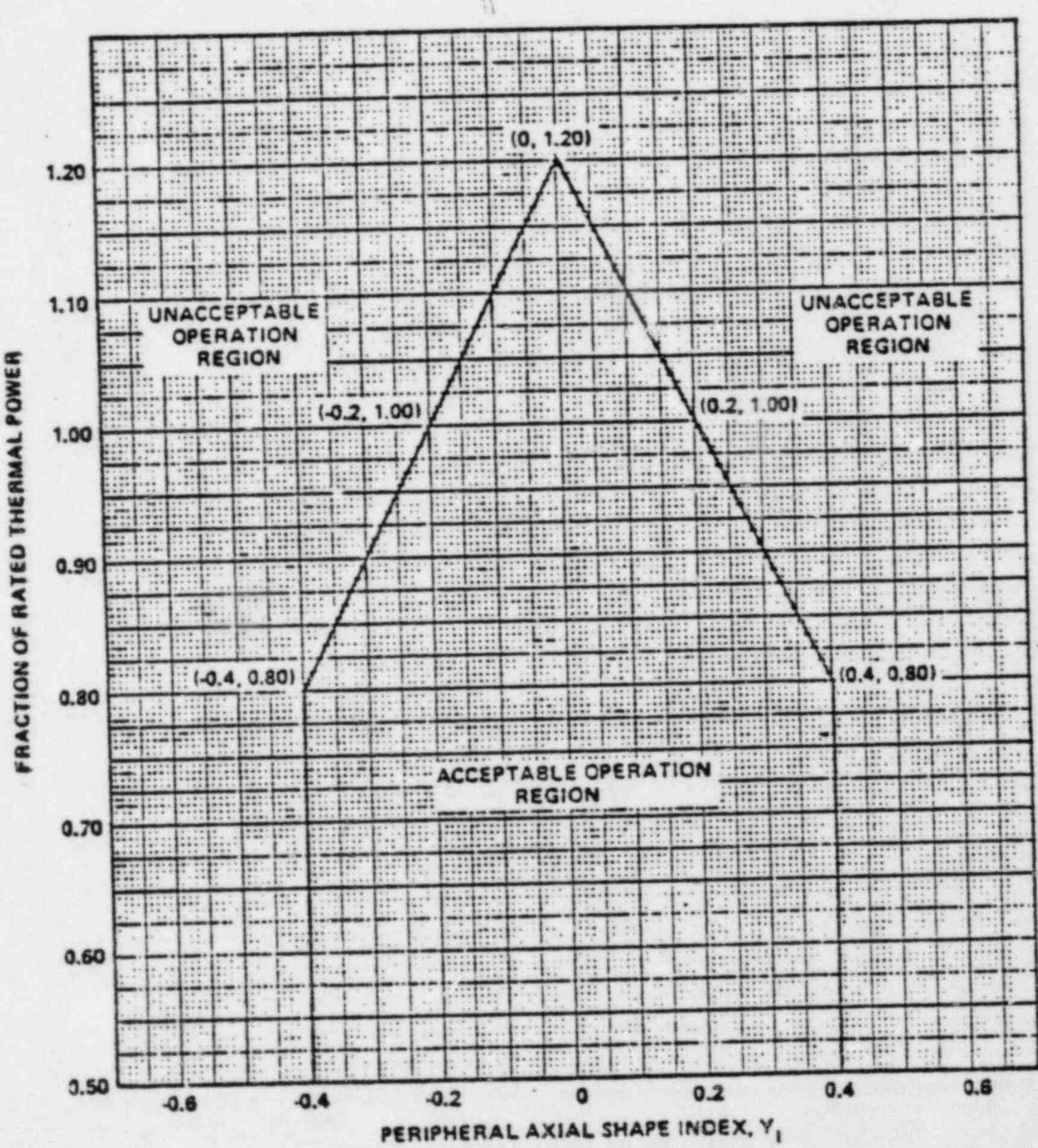


FIGURE 2.2-1
Peripheral Axial Shape Index, Y_I Versus Fraction of RATED THERMAL POWER

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

22.0

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ration of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.195 → 1.23 This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

1.23

The curves of Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature of various pump combinations for which the minimum DNBR is no less than 1.195 for the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. The limits in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in

110%

Table 2.1-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 to be valid are shown on the figures.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than 1.195 and preclude the existence of flow instabilities.

→ 1.23

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III, 1967 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I, 1969 Edition, which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 10% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL power decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 30% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the safety analyses.

110%

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit

1.23

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.195 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.195 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating. 1.23

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 570 psia is sufficiently below the full-load operating point of 850 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 22 psi, ~~in the accident analyses.~~ 635 87
which was based on the Main Steam Line Break Event.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the reactor coolant system will not exceed its Safety Limit. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 13 minutes before auxiliary feedwater is required.

Axial Flux Offset

1.23

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than 1.195 nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore axial flux offset relationship.

Thermal Margin/Low Pressure

1.23

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.195.

1875

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1750 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

LIMITING SAFETY SYSTEM SETTINGS

BASES

40

INCLUDE

The Thermal Margin/Low Pressure trip setpoints ~~are derived from the core safety limits through application of appropriate~~ allowances for equipment response time, measurement uncertainties, ~~and~~ processing error. ~~A safety margin is provided which includes: an allowance of 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2°F to compensate for potential temperature measurement uncertainty;~~ and a further allowance of 92 psia to compensate for The ~~pressure measurement error, trip [illegible] processing error, and~~ time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. ~~The 92 psia allowance is made up of a 22 psia pressure measurement allowance and a 70 psia time delay allowance~~.

Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF)

The ASGTPTF utilizes steam generator pressure inputs to the TM/LP calculator, which causes a reactor trip when the difference in pressure between the two steam generators exceeds the trip setpoint. The ASGTPTF is designed to provide a reactor trip for those Anticipated Operational Occurrences associated with secondary system malfunctions which result in asymmetric primary loop coolant temperatures. The most limiting event is the loss of load to one steam generator caused by a single Main Steam Isolation Valve closure.

The equipment trip setpoint and allowable values are calculated to account for instrument uncertainties, and will ensure a trip at or before reaching the analysis setpoint.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

5.2%

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ 4.3%* Δk/k.

APPLICABILITY: MODES 1, 2**, 3 and 4.

ACTION:

5.2%

With the SHUTDOWN MARGIN < 4.3%* Δk/k, immediately initiate and continue boration at ≥ 40 gpm of 2300 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

5.2%

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ 4.3%* Δk/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

---

* Adherence to Technical Specification 3.1.3.6 as specified in Surveillance Requirements 4.1.1.1.1 assures that there is sufficient available shutdown margin to match the shutdown margin requirements of the safety analyses.

** See Special Test Exception 3.10.1.

\# With K_{eff} ≥ 1.0.

\#\# With K_{eff} < 1.0.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any $\underbrace{\text{THERMAL POWER}}_{}$ above 90% of RATED THERMAL POWER, within 7 EFPD after ~~reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.~~ initially reaching an equilibrium condition at or above 90% of RATED THERMAL POWER.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

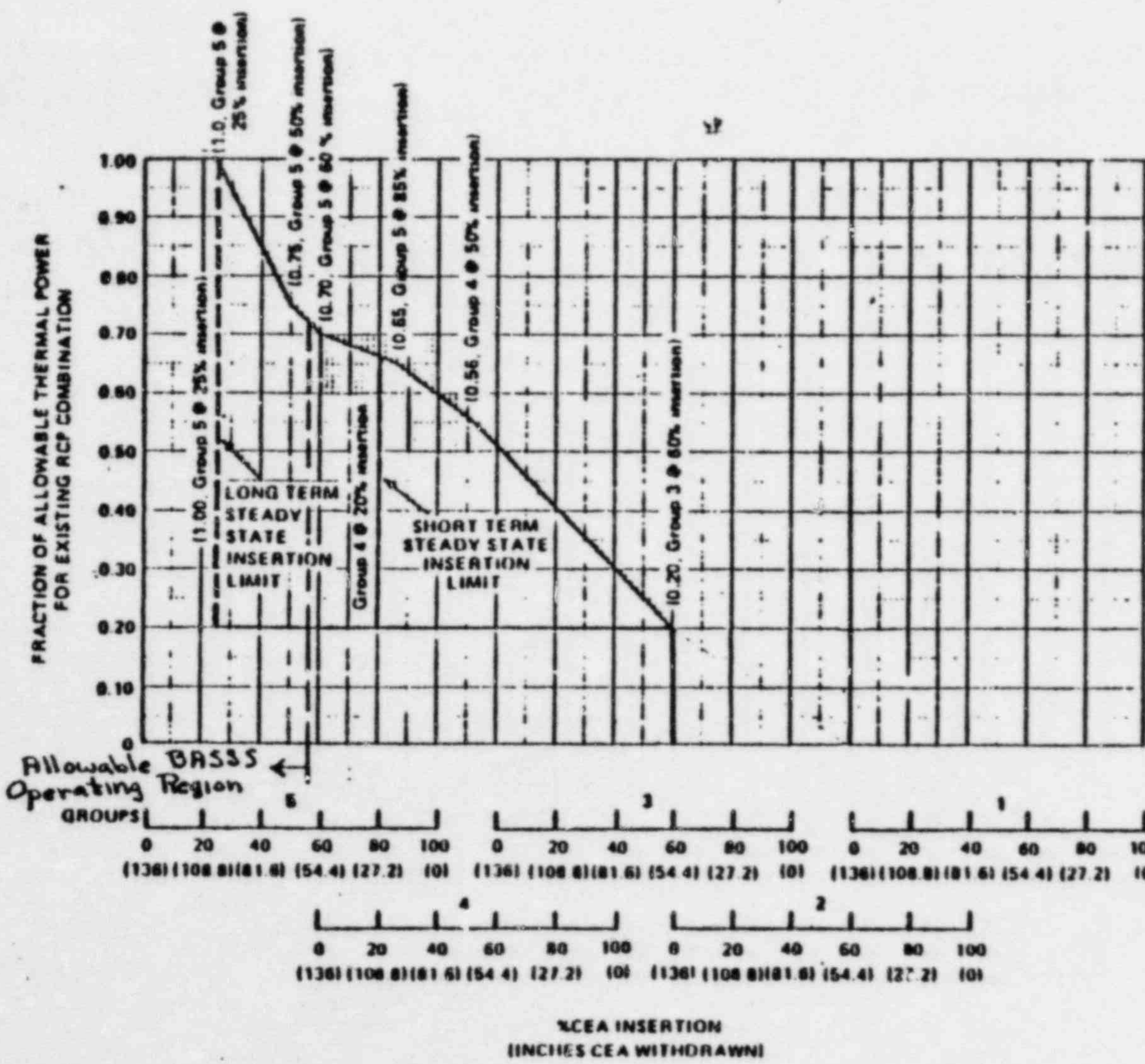


FIGURE 3.1-2 CEA Insertion Limits vs Fraction of Allowable Thermal Power for Existing RCP Combination

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the AXIAL SHAPE INDEX is maintained within the limits of Figure 3.2-2, where 100 percent of the allowable power represents the maximum THERMAL POWER allowed by the following expression:

$$M \times N$$

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F^T_{xy} curve shown up Figure 3.2-3 of Specification 3.2.2.
3.2-3b

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

1. Flux peaking augmentation factors as shown in Figure 4.2-1,

2. A measurement-calculational uncertainty factor of 1.07,

3. An engineering uncertainty factor of 1.03,

4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and

5. A THERMAL POWER measurement uncertainty factor of 1.02.

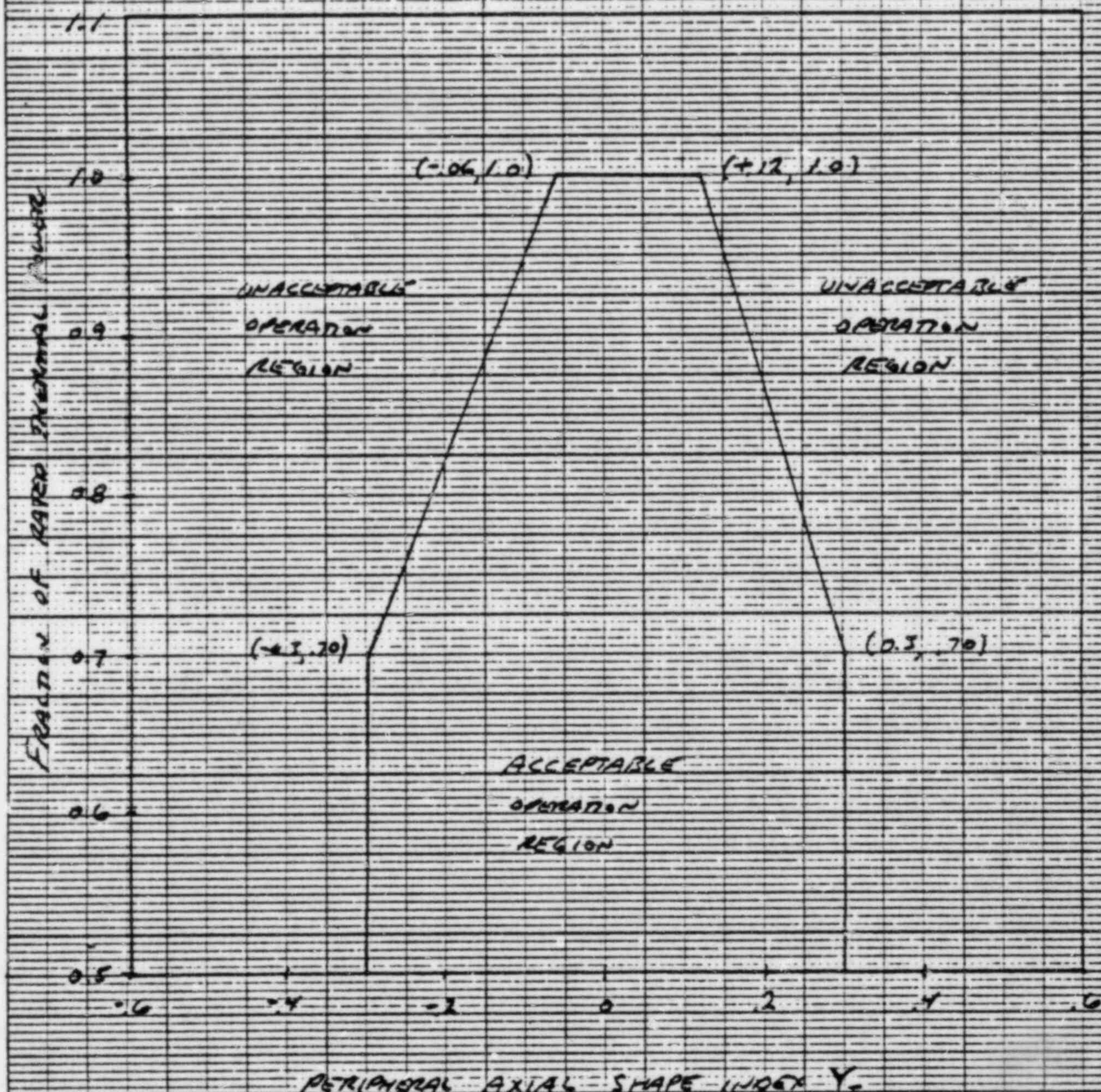
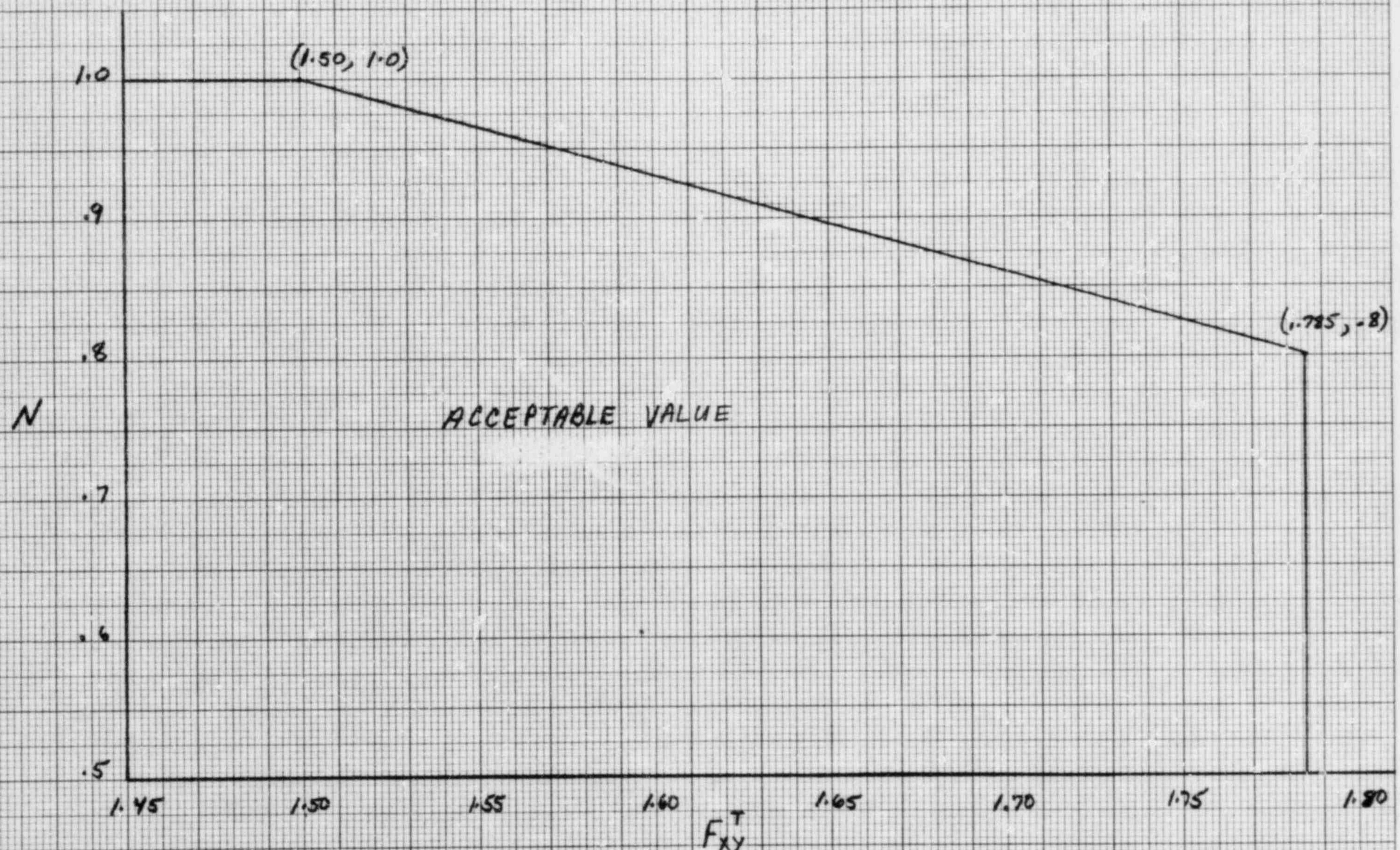


FIGURE 3.2-2

LINEAR HEAT RATE
AXIAL FLUX OFFSET CONTROL LIMITS

Figure 3.2-3b

3/4 2-4a



TOTAL PLANAR RADIAL PEAKING FACTOR VS N

TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2.1 The calculated value of F^T_{xy}, defined as $F^T_{xy} = F_{xy}(1+T_q)$, shall be limited to $\leq$ 1.620 → 1.70

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.620 → 1.70, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits of Figure 3.2-3 → 3.2-3a and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 → 2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 → 2.1.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+T_q)$ and F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

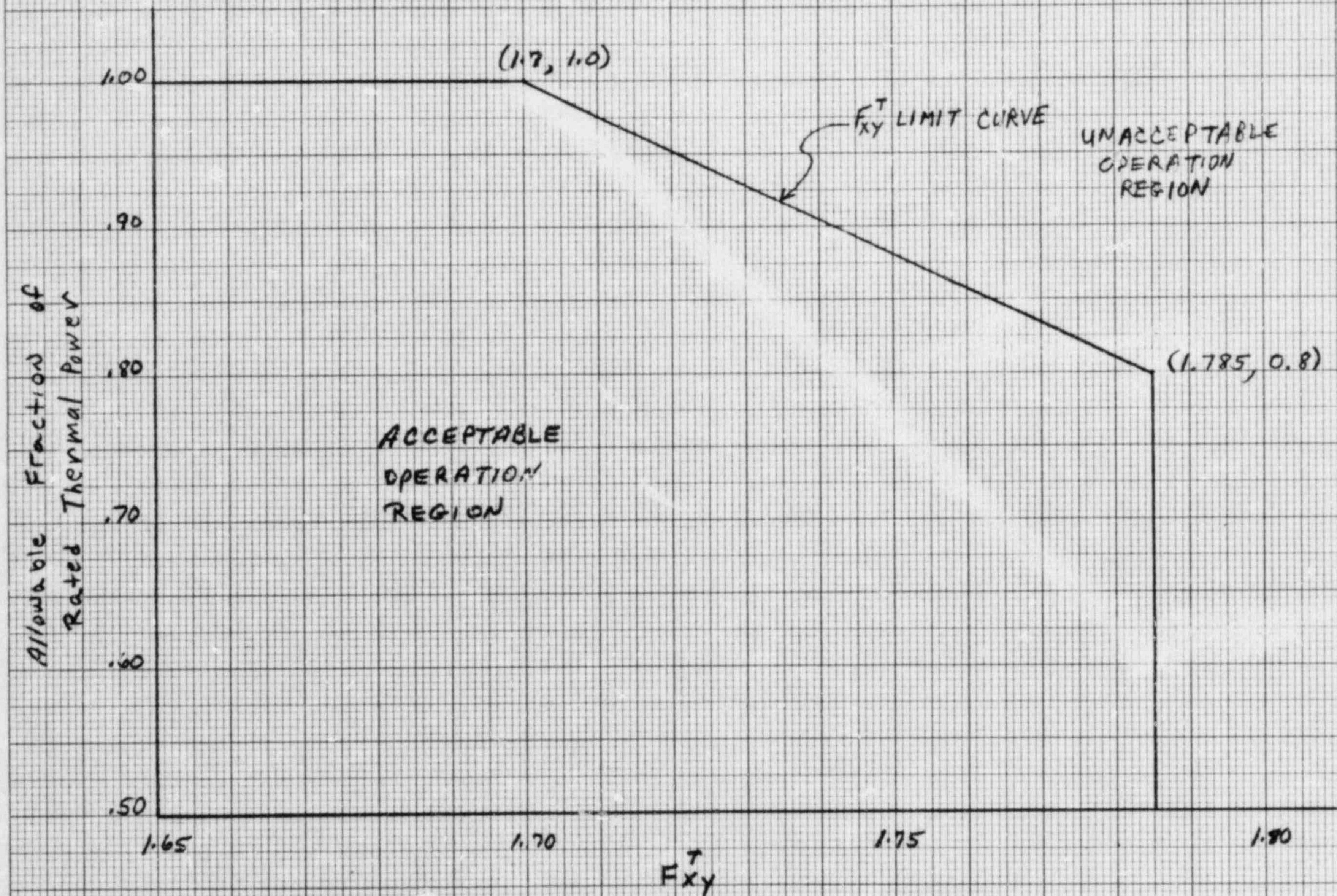
1.3

4.2.2.~~3~~ F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid affects.

1.4

4.2.2.~~4~~ T_q shall be determined each time a calculation of F^T_{xy} is required and the value of T_q used to determine F^T_{xy} shall be the measured value of T_q.

Figure 3.2-3a



TOTAL PLANAR RADIAL PEAKING FACTOR vs ALLOWABLE FRACTION OF RATED THERMAL POWER

TOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T

LIMITING CONDITION FOR OPERATION

3.2.2~~.1~~.2 ~~The calculated value of F_{xy}^T, defined as $F_{xy}^T = F_{xy}(1+T_q)$, shall be limited to ≤ 1.620.~~ The value of N presently used in specification 4.2.1.3 shall be in accordance with Figure 3.2-3b

APPLICABILITY: MODE 1~~*~~X

X When operating in accordance with specification 4.2.1.3

ACTION:

With ~~F_{xy}^T > 1.620~~ the value of N presently used in specification 4.2.1.3 exceeding the limit shown in Figure 3.2-3b, within 6 hours either:

a. ~~Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_{xy}^T to within the limits of Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or~~ Reduce the value of N used in specification 4.2.1.3 to within the limits of figure 3.2-3b; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.~~1~~2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.~~2~~2.2 F_{xy}^T shall be calculated by the expression $F_{xy}^T = F_{xy}(1+T_q)$ and ~~F_{xy}^T~~ N shall be determined to be within its limit by monitoring F_{xy}^T at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per ~~31~~ 3 days of accumulated operation in MODE 1, and

~~c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030.~~

~~*See Special Test Exception 3.10.2.~~

SURVEILLANCE REQUIREMENTS (Continued)

2.3
4.2.2.~~3~~ F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid affects.

2.4
4.2.2.~~4~~ T_q shall be determined each time a calculation of F^T_{xy} is required and the value of T_q used to determine F^T_{xy} shall be the measured value of T_q.

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T, defined as $F_r^T = F_r(1+T_q)$, shall be limited to ≤ 1.620 1.650

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.620 1.650, within 6 hours either:

a. Be in at least HOT STANDBY, or

b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3c, withdraw the full length CEAs to or beyond the Long Term Steady State Limits of Specification 3.1.3.6, and insert new value of F_r^T in BASSS; or

c. ~~b.~~ Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3 3.2-3c and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6. The THERMAL POWER limit determined from Figure 3.2-3 3.2-3c shall then be used to establish a revised upper THERMAL POWER level limit on Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by Figure 3.2-3 3.2-3c) and subsequent operation shall be maintained within the reduced acceptable operation region of Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ and F_r^T shall be determined to be within its limit at the following intervals:

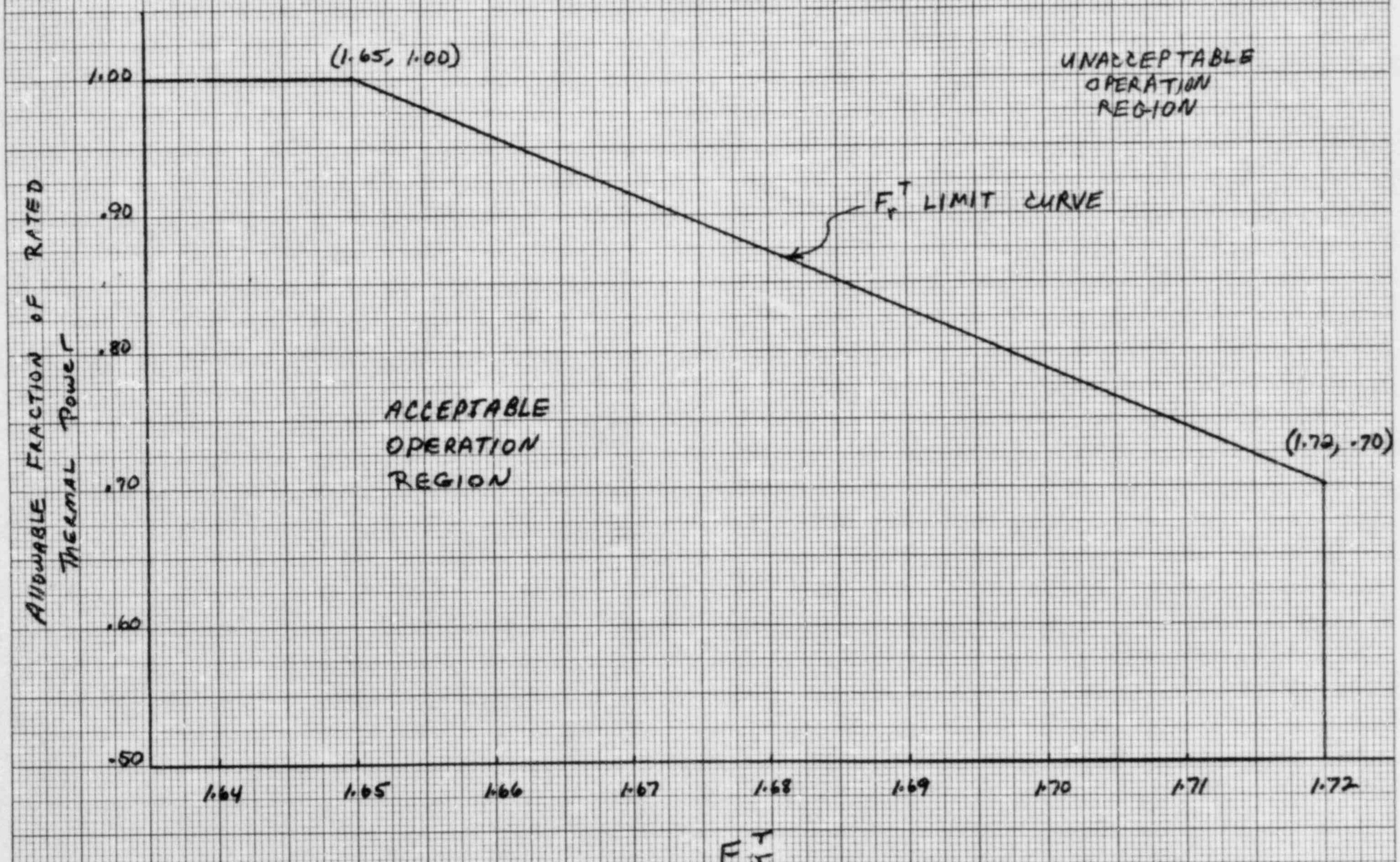
a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030.

*See Special Test Exception 3.10.2.

Figure 3.2-3C



TOTAL INTEGRATED RADIAL PEAKING FACTOR vs ALLOWABLE FRACTION OF RATED THERMAL POWER

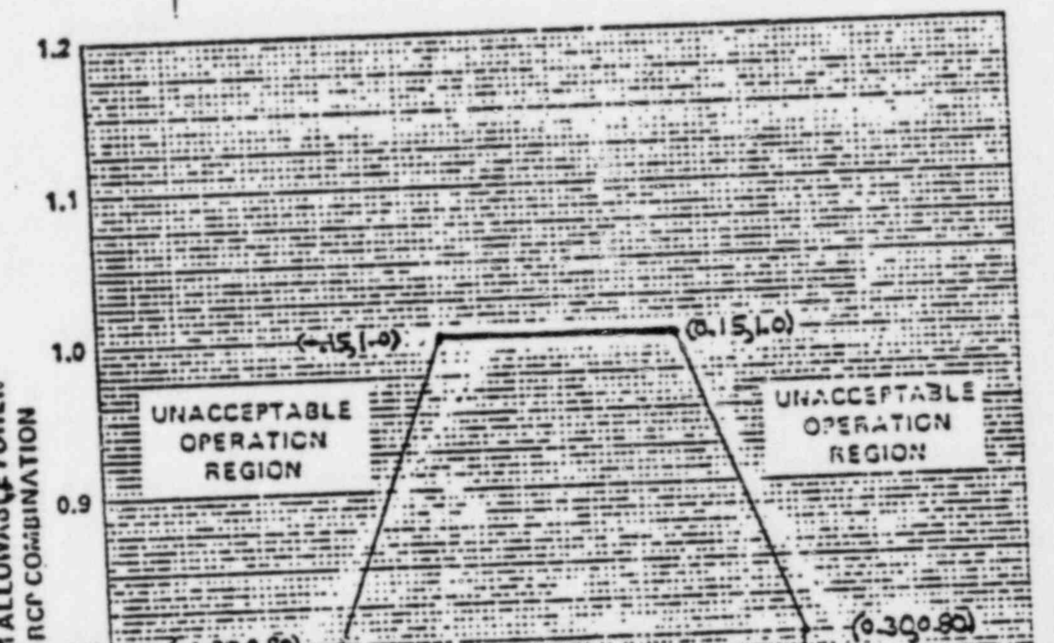


FIGURE 3.2-4

DNB Axial Flux Offset Control Limits

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX, Core Power

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

Parameter	LIMITS			
	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	Two Reactor Coolant Pumps Operating-Same Loop	Two Reactor Coolant Pumps Operating-Opposite Loop
Cold Leg Temperature	≤ 548°F	**	**	**
Pressurizer Pressure	≥ 2225 psia* 2200	**	**	**
Reactor Coolant System Total Flow Rate	≥ 370,000 gpm	**	**	**
AXIAL SHAPE INDEX	Figure 3.2-4 ***	**	**	**

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

**These values left blank pending NRC approval of ECCS analyses for operation with less than four reactor coolant pumps operating.

*** The AXIAL SHAPE INDEX, Core Power shall be maintained within the limits established by the Better Axial Shape Selection System (BASSS) for CEA insertions of the lead bank of ≤55% when BASSS is operable, or within the limits of Figure 3.2-4 for CEA insertions specified by Figure 3.1-2.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be bypassed below 10^{-4} of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$ of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 685 → 710 psia; bypass shall be automatically removed at or above 685 → 710 psia.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 12% of RATED THERMAL POWER.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Level - High	≤ 0.40 seconds*# and ≤ 8.0 seconds## 12.0
3. Reactor Coolant Flow - Low	≤ 0.50 seconds
4. Pressurizer Pressure - High	≤ 0.90 seconds
5. Containment Pressure - High	≤ 0.90 seconds
6. Steam Generator Pressure - Low	≤ 0.90 seconds
7. Steam Generator Water Level - Low	≤ 0.90 seconds
8. Axial Flux Offset	≤ 0.40 seconds*# and ≤ 8.0 seconds## 12.0
9.a. Thermal Margin/Low Pressure	≤ 0.90 seconds*# and ≤ 8.0 seconds ## 12.0
b. Steam Generator Pressure Difference - High	≤ 0.90 seconds
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	Not Applicable
11. Wide Range Logarithmic Neutron Flux Monitor	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portio of the channel shall be measured from detector output or input of first electronic component in channel.

#Response time does not include contribution of RTDs.

##RTD response time only. This value is equivalent to the time interval required for the RTDs output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1700 1800 psia; bypass shall be automatically removed when pressurizer pressure is $\geq$ 1700 1800 psia.

(c) Trip function may be bypassed in this MODE below 685 710 psia; bypass shall be automatically removed at or above 685 710 psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
c. Pressurizer Pressure - Low	$\geq$ 1578 psia 1725	$\geq$ 1578 psia 1725
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
3. CONTAINMENT ISOLATION (CIS) #		
a. Manual CIS (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 4.75 psig	$\leq$ 4.75 psig
4. MAIN STEAM LINE ISOLATION		
a. Manual (MSIV Hand Switches and Feed Head Isolation Hand Switches)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	$\geq$ 570 psia 635	$\geq$ 570 psia 635

Containment isolation of non-essential penetrations is also initiated by SIAS (functional units 1.a and 1.c).

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble and with at least 150 kw of pressurizer heater capacity capable of being supplied by emergency power. ~~The pressurizer level shall be within ± 5 percent of its programmed value.~~

The pressurizer level shall be maintained within an operating band between 133 and 225 inches except when three charging pumps are operating and let down flow is less than 25 GPM. If three charging pumps are operating and letdown flow is less than 25 GPM maximum pressurizer level shall be limited to between 133 and 210 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within ~~± 5 percent of its programmed value~~ at least once per 12 hours.

the above band

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

>5.2%

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The minimum available SHUTDOWN MARGIN for no load operating conditions at beginning of life is 4.1% Δk/k and at end of life is 4.3% Δk/k. The SHUTDOWN MARGIN is based on the safety analyses performed for a steam line rupture event initiated at no load conditions. The most restrictive steam line rupture event occurs at EOC conditions. For the steam line rupture event at beginning of cycle conditions, a minimum SHUTDOWN MARGIN of less than 4.1% Δk/k is required to control the reactivity transient, and end of cycle conditions require 4.3% Δk/k. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 3% Δk/K shutdown margin provides adequate protection. With the pressurizer level less than 90 inches, the sources of non-borated water are restricted to increase the time to criticality during a boron dilution event. 4.5 >5.2% >4.5%

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

BASES

the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_{xy}^T, F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F_{xy}^T = F_{xy}(1 + T_q)$ and $F_r^T = F_r(1+T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}^T, F_r^T and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.195 throughout each analyzed transient.

1.23

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

The operating band for pressurizer level bounds the programmed level and ensures that RCS pressure remains within the bounds of an analyzed condition during the excessive charging event as well as during the limiting depressurization event, Excess Load. The operating band

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer with the level as programmed ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. ~~The programmed level~~ also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valves function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer--Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of off-site power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 276°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 3000 grams uranium. The initial core loading shall have a maximum enrichment of 2.99 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent U-235.
> 4.1

5.3.2 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

CONTROL ELEMENT ASSEMBLIES

5.3.3 The reactor core shall contain 77 full length and no part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

10.0 Startup Testing

The startup testing program proposed for Cycle 5 is identical to the program proposed for the reference cycle in Reference 1.

11.0 References

Chapters 1 through 5

1. Letter, A. E. Lundvall, Jr. (BG&E) to R. A. Clark (NRC), "Sixth Cycle License Application," dated February 17, 1982.

2. Letter, A. E. Lundvall, Jr. (BG&E) to R. A. Clark (NRC), "Supplement 1 to Sixth Cycle License Application," dated April 29, 1982.

3. Letter, A. E. Lundvall, Jr. (BG&E) to R. A. Clark (NRC), "Fourth Cycle License Application," dated December 4, 1980.

4. Letter, A. E. Lundvall, Jr. (BG&E) to R. A. Clark (NRC), "Report of Startup Testing for Cycle Four," dated May 15, 1981.

5. CEN-105(B)-P, "Reconstitutable B_4C Type CEA Design for Use in the BG&E Reactor," dated February 1979.

6. Letter, A. E. Lundvall (BG&E) to R. A. Clark (NRC), "Unit 1 Docket 50-317, Report of Prototype CEA Performance during Cycle 4 (Misnomer, Should have been: 'Report of Prototype CEA Performance during Unit 2 Cycle 3')," dated October 5, 1981.

7. Letter, A. E. Lundvall, Jr. (BG&E) to R. W. Reid (NRC), "Unit 2 Cycle 2 License Application," dated July 26, 1978.

8. CENPD-187, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding," dated June 1975.

9. CEN-182(B)-P, "Statistical Approach to Analyzing Creep Collapse of Oval Fuel Rod Cladding Using CEPAN," dated September 1981.

10. Letter, D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BG&E), "Regarding Unit 1 Cycle 6 License Approval (Amendment #71 to DPR-53 and SER)," dated June 24, 1982.

11. CEN-183(B)-P, "Application of CENPD-198 to Zircalloy Component Dimensional Changes," dated Septemer, 1981.

12. CEN-83(B)-P, "Calvert Cliffs Unit 1 Reactor Operation with Modified CEA Guide Tubes," dated February 8, 1978, and Letter, A. E. Lundvall, Jr. (BG&E) to V. Stello, Jr. (NRC), "Reactor Operation with Modified CEA Guide Tubes," dated February 17, 1978.

13. CENPD-139, "C-E Fuel Evaluation Model Topical Report," dated July, 1974.

14. CEN-161(B)-P, "Improvements to Fuel Evaluation Model," dated July, 1981.

15. CENPD-153-P, Revision 1, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered Fixed In-Core Detector System," dated May 1980.

References (Chapter 6)

1. CENPD-161-P, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core", July 1975

2. CENPD-162-P-A (Proprietary) and CENPD-162-A (Nonproprietary), "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution", April 1975

3. CENPD-206-P, "TORC Code, Verification and Simplified Modeling Methods", January 1977

4. Letter, P. W. Kruse to W. J. Lippold, "Responses to First Round Questions on the SCU Program: CETOP-D Code Structure and Modeling Methods, (CEN-124 (B)-P, Part 2)", May 1981 and letter, P.W. Kruse to W.J. Lippold (above document), BGE-9676-576, May 1, 1981.

5. Letter D.H. Jaffe (NRC) to A.E. Lundvall,Jr. (BG&E), "Regarding Unit 1 Cycle 6 License Approval (Amendment #71 to DPR-53 and SER)," June 24, 1982.

6. CEN-124(B)-P, "Statistical Combination of Uncertainties, Part 2", January 1980

7. CEN-83(B)-P, "Calvert Cliffs Unit 1 Reactor Operation With Modified CEA Guide Tubes", February 8, 1978 and letter, A. E. Lundvall, Jr. to V. Stello, Jr., "Reactor Operation With Modified CEA Guide Tubes", February 17, 1978

8. Letter, D. F. Ross and D. G. Eisenhut (NRC) to D. B. Vassallo and K. R. Goller (NRC), "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculation for Light Water Reactors", February 16, 1977

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References for Chapter 7

1. Letter, A. E. Lundvall, Jr. (BG&E) to R. A. Clark (NRC), "Sixth Cycle License Application," Docket No. 50-317, February 17, 1982.

2a. "Statistical Combination of Uncertainties Methodology; Part 1; C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II," CEN-124(B)-P, December, 1979.

2b. "Statistical Combination of Uncertainties Methodology; Part 2; Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units I and II," CEN-124(B)-P, January 1980.

2c. "Statistical Combination of Uncertainties Methodology; Part 3; C-E Calculated Local Power Density and Departure from Nucleate Boiling Limiting Conditions for Operation for Calvert Cliffs Units I and II," CEN-124(B)-P, March 1980.

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