

CALVERT CLIFFS UNIT 1
CYCLE 4 REFUELING LICENSE
AMENDMENT

7903020 357

CALVERT CLIFFS UNIT I
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AMENDMENT

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1. INTRODUCTION AND SUMMARY

This report provides an evaluation of design and performance for the operation of Calvert Cliffs Unit I during its fourth fuel cycle at full rated power of 2700 MWT. All planned operating conditions remain the same as those for Cycle 3. The core will consist of presently operating B, D and E assemblies and fresh Batch F assemblies.

Plant operating requirements have created a need for flexibility in Cycle 3 termination point ranging from 8950 MWD/T to 10,000 MWD/T. In performing analyses of postulated accidents, determining limiting safety settings and establishing limiting conditions for operation, limiting values of key parameters were chosen to assure that expected Cycle 4 conditions are enveloped regardless of the Cycle 3 termination point within the above burnup range.

The evaluations of the reload core characteristics have been examined with respect to the Calvert Cliffs Unit I Cycle 3 safety analysis described in Reference 1, hereafter referred to as the "reference cycle" in this report. This is an appropriate reference cycle because of the similarity in the basic system characteristics of the two reload cores. Specific core differences have been accounted for in the present analysis. In all cases, it has been concluded that either the reference cycle analyses envelope the new conditions, or that the revised analyses presented here continue to show acceptable results. Where dictated by variations from the reference cycle, proposed modifications to the plant Technical Specifications are provided and are justified by the analyses reported herein.

2. OPERATING HISTORY OF THE REFERENCE CYCLE

Calvert Cliffs Unit I is presently operating in its third fuel cycle utilizing Batch A, B, C, D and E fuel assemblies. Cycle 3 operation at full power began on or about April 3, 1978. The Cycle 3 startup testing was reported to the NRC in Reference 2.

It is presently estimated that Cycle 3 will terminate on or about April 21, 1979. However, flexibility in this date is necessary because of uncertainties in the future station capacity factor. The Cycle 3 termination point can vary between 8950 MWD/T and 10,000 MWD/T to accommodate the plant schedule.

As of early February, the Cycle 3 burnup had reached 7100 MWD/T. Initial criticality of Cycle 4 is expected to occur on or about May 22, 1979.

3. GENERAL DESCRIPTION

The Cycle 4 core will consist of the number and types of assemblies from several fuel batches as described in Table 3-1. The primary change to the core in Cycle 4 is the removal of 40 Batch A assemblies, and 32 Batch C assemblies and their replacement by 72 fresh Batch F assemblies. Figure 3-1 shows the fuel management pattern to be employed in Cycle 4. This pattern will accommodate Cycle 3 termination burnups from 8950 MWD/T to 10,000 MWD/T.

The Cycle 4 core loading pattern is 90° rotationally symmetric. That is, if one quadrant of the core were rotated 90° into its neighboring quadrant, each assembly would be aligned with a similar assembly. This similarity includes batch type, number of fuel rods, initial enrichment and burnup. It does not include guide tube sleeves and demonstration fuel rod locations.

Figure 3-2 shows the location of poison rods and stainless steel rods within the lattice of the Batch B irradiation test (EPRI/CE) assembly.

Figure 3-3 shows the beginning of Cycle 4 assembly burnup distribution for both the minimum Cycle 3 termination burnup of 8950 MWD/T and the maximum Cycle 3 termination burnup of 10,000 MWD/T. The initial enrichment of the fuel assemblies is also shown in Figure 3-3. At the beginning of Cycle 4, the residual B-10 content of the burnable poison rods in the irradiation test (EPRI/CE) assembly is negligible.

The Cycle 4 core will contain a high burnup demonstration assembly (SCOUT) and a prototype CEA. The locations of the demonstration assembly and the prototype CEA are shown in Figure 3-1.

TABLE 3-1

CALVERT CLIFFS UNIT 1 CYCLE 4
CORE LOADING

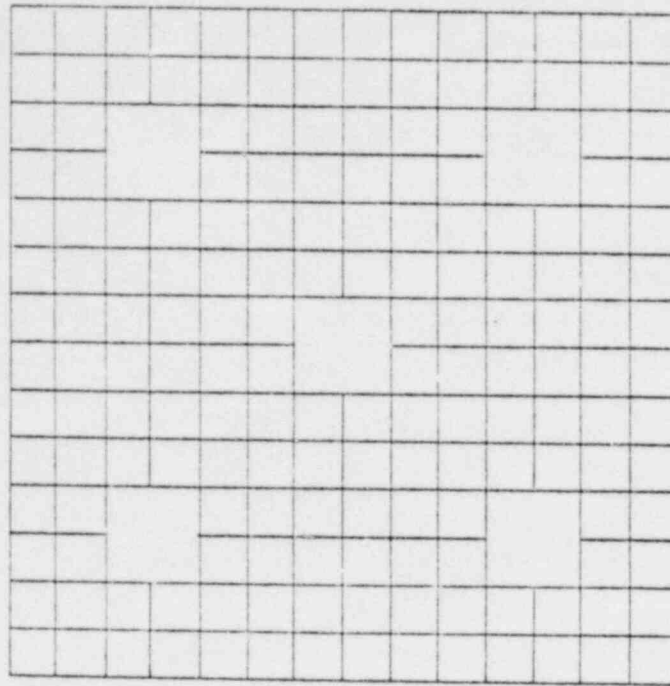
Assembly Designation	Number of Assemblies	Initial Enrichment wt% U-235	Batch-Average Burnup EOC 3 = 8950	Batch-Average Burnup EOC 3 = 10,000	Poison Rods Per Assembly	Initial Poison Loading wt% B ₄ C	Total Number of Poison Rods	Total Number of Fuel Rods
B ¹	1	2.45	34,100	35,000	12	2.9	12	160
D	48	3.03	17,700	18,900	0	0	0	8,448
D/	24	2.73	20,000	21,200	0	0	0	4,224
E	48	3.03	7,400	8,300	0	0	0	8,448
E/	24	2.73	10,900	12,100	0	0	0	4,224
F	48	3.03	0	0	0	0	0	8,448
F/	24	2.73	0	0	0	0	0	4,224
Totals	217						12	38,176

Notes

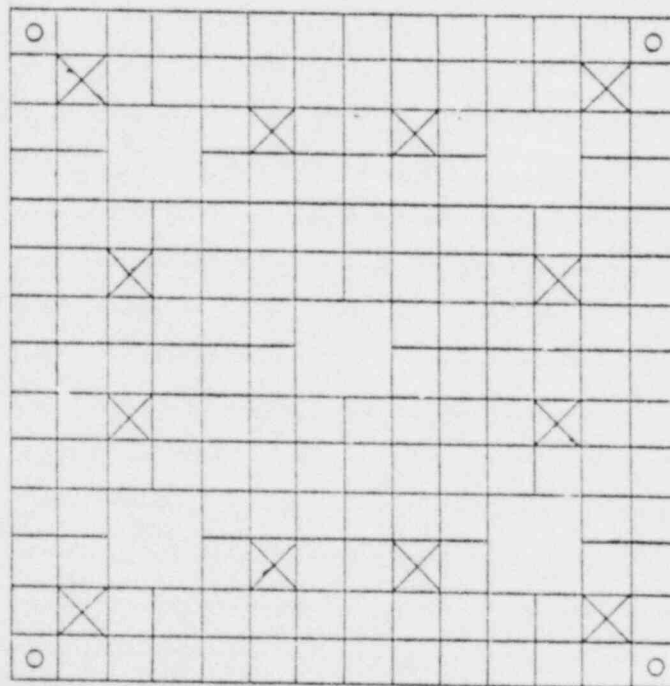
¹This is the irradiation test assembly located in the center of the core. In addition to the twelve poison rods, it contains four stainless steel rods, one in each corner of the assembly.

							F	F
				F*	F	F	E	E/
		F	F/	E	D	D/	F/	
	F	F/ **	E/	D	F/	D	D	
F	F	F/	E/	E/	E/	D	E	D
	F	E	D	E/	D	E	D/	E
	F	D	F/	D	E	D/	E	D/
F	E	D/	D	E	D/	E	D	E
F	E/	F/	D	D	E	D/	E	*** B+

- * Location of Demonstration Assembly (SCOUT)
 ** Location of Prototype CEA
 *** Location of Irradiation Test Assembly (EPRI/CE)



12 POISON ROD ASSEMBLY (CENTER ASSEMBLY ONLY)



- ☐ FUEL ROD LOCATION
- ☒ POISON ROD LOCATION IN TEST ASSEMBLY
- ☐ STAINLESS STEEL ROD LOCATION IN TEST ASSEMBLY

INITIAL ENRICHMENT W/O U-235 ———→ 3.03 3.03
 BOC 4 BURNUP (MWD/T), EOC3=8950 MWD/T ———→ 0 0
 BOC 4 BURNUP (MWD/T), EOC3=10000 MWD/T ———→ 0 0

				3.03 0 0	3.03 0 0	3.03 0 0	3.03 8100 9000	3.03 9400 10400
		3.03 0 0	2.73 0 0	3.03 10100 11300	3.03 17600 18600	2.73 19600 20600	2.73 0 0	
	3.03 0 0	2.73 0 0	2.73 11500 12800	3.03 20300 21500	2.73 0 0	3.03 17700 19000	3.03 17500 18800	
3.03 0 0	2.73 0 0	2.73 11500 12800	2.73 10800 12100	2.73 11100 12400	3.03 14700 15800	3.03 7900 8800	3.03 19300 20600	
3.03 0 0	3.03 10200 11400	3.03 20300 21500	2.73 11200 12400	3.03 18300 19500	3.03 5600 6400	2.73 20200 21400	3.03 6400 7100	
3.03 0 0	3.03 17500 18700	2.73 0 0	3.03 14800 15900	3.03 5700 6400	2.73 20200 21400	3.03 6400 7200	2.73 20200 21400	
3.03 8200 9100	2.73 19500 20700	3.03 17300 19000	3.03 7900 8800	2.73 20300 21500	3.03 6400 7200	3.03 17300 18400	3.03 6300 7100	
2.73 9400 10400	2.73 0 0	3.03 17500 18800	3.03 19200 20500	3.03 6400 7100	2.73 20200 21400	3.03 6300 7100	2.45 34100 35000	

4. FUEL SYSTEM DESIGN

The mechanical design for Batch F reload fuel is essentially identical to that of the Batch E fuel used in Calvert Cliffs Unit I and described in the reference cycle submittal, Reference 1.

Details of the Batch B and D fuel design parameters can be found in References 3 and 4, respectively.

C-E has performed analytical predictions of cladding creep-collapse time for all Calvert Cliffs Unit I fuel batches that will be irradiated in Cycle 4 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 4 duration. These analyses utilized the CEPAN computer code (Reference 5) and included as input conservative values of internal pressure, cladding dimensions, cladding temperature and neutron flux.

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for the Batch F fuel are identical to those of the Batch B, D and E fuel from Cycle 3. Thus, the chemical or metallurgical performance of the Batch F fuel will remain unchanged from the performance of the Cycle 3 fuel.

4.1 HARDWARE MODIFICATIONS TO MITIGATE GUIDE TUBE WEAR

All fuel assemblies presently in Cycle 3 which will be placed in CEA locations in Cycle 4, with the exception of the Batch B test assembly, will have stainless steel sleeves installed in the guide tubes in order to prevent guide tube wear. The sixteen Batch F assemblies which will be placed in dual CEA locations will also have stainless steel sleeves installed in the guide tubes. A detailed discussion of the design of the sleeves and their effect on reactor operation

is contained in Reference 6. The remaining 56 Batch F assemblies will either be modified in the same manner as the sixteen Batch D assemblies in Calvert Cliffs II Cycle 2 or will have stainless steel sleeves. A detailed discussion of the Calvert Cliff II Cycle 2 Batch D modification and its effects on reactor operation is contained in Reference 16.

4.2 BATCH F DEMONSTRATION ASSEMBLY

One Batch F assembly consists of 161 standard fuel rods and 15 demonstration rods. A detailed description of these demonstration rods may be found in Reference 7. The mechanical design of the assembly components other than the 15 demonstration rods in this assembly is identical to the design of the other Batch F assemblies. Figure 3-1 displays the location of the demonstration assembly in Cycle 4. No demonstration fuel rod in the demonstration assembly will have a power level within 10% of the maximum radial power peak in the core during Cycle 4.

Two types of demonstration fuel rods are being introduced in this assembly (Reference 7). Helium fill pressure differences between the demonstration fuel rod designs were introduced for the following reasons:

- a. The location of non-fueled regions at grid contact points results in the possibility of somewhat reduced grid/rod contact forces. To offset this possibility, fill pressure in these rods was increased to raise beginning of life internal pressure, and thus decrease the magnitude of clad creepdown.
- b. A difference in void volume exists between the two types of demonstration rods. A higher initial fill pressure will not appreciably increase end of life internal pressure in the test rod type with greater void volume.

The magnitude of the subject difference in He fill pressure is 65 psi.

CE has performed analytical predictions of cladding creep-collapse time for the demonstration fuel rods that will be irradiated in Cycle 4 and has concluded that the collapse resistance of these demonstration fuel rods is sufficient to preclude collapse during their design lifetime. This lifetime will not be exceeded by the Cycle 4 duration.

4.3 Prototype CEA

Cycle 4 will utilize one prototype CEA as part of regulating Bank 5. The location of this prototype CEA is shown in Figure 3-1. This new CEA design involves a change in cladding material (Inconel to stainless steel) and specially designed reconstitutable poison rods which serve as a lead for all future poison rod designs. The purpose of the poison rod design is to demonstrate that the silver-indium-cadmium which is presently used in the tips of poison rods can be replaced with B_4C . Both the changes in cladding material and the replacement of Ag-In-Cd with B_4C are being made for reasons of economics and to improve material availability. A more detailed description of the design of this prototype CEA is given in Reference 8. This prototype CEA meets the same design criteria and has the same design margin as the CEA's used in the reference cycle. Therefore, it has no adverse affect on mechanical integrity, thermal-hydraulics and safety. A detailed description of the affects of the prototype CEA can be found in Reference 8.

5. NUCLEAR DESIGN

5.1 PHYSICS CHARACTERISTICS

5.1.1 Fuel Management

The Cycle 4 fuel management employs a mixed central region as described in Section 3. The fresh Batch F 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 3.03 wt% U-235 and 24 assemblies with an enrichment of 2.73 wt% U-235. The Cycle 4 burnup capacity for full power operation is expected to be between 10,000 MWD/T and 10,500 MWD/T, depending on the final Cycle 3 termination point. The Cycle 4 performance characteristics have been examined for a Cycle 3 termination between 8950 and 10,000 MWD/MTU and limiting values established. The proposed loading pattern is presented in Figure 3-1. The pattern is applicable to any Cycle 3 termination point between the stated extremes. Physics characteristics including reactivity coefficients for Cycle 4 are listed in Table 5-1 along with the corresponding values from the reference cycle. It is noted that the values of parameters actually employed in safety analyses are typically chosen to conservatively bound predicted values including uncertainties and allowances. Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for Cycle 4 with a comparison to reference cycle data. The power dependent CEA insertion limit and CEA group identification are unchanged from the reference cycle. Table 5-3 shows the reactivity worths of various CEA groups calculated at full power conditions for Cycle 4.

5.1.2 Power Distribution

The radial power distributions described in this section are calculated data without uncertainties or other allowances. However, 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 as described in Reference 9.

Planar radial power distributions for the unrodded core at beginning of cycle, 6 GWD/T and end of cycle are shown in Figures 5-1 through 5-3 for a Cycle 3 endpoint of 10,000 MWD/T. The planar radial peaks described here are characteristic of the major portion of the active core length between approximately 20 percent and 80 percent of the fuel height. The maximum planar radial power peak for the early shutdown of Cycle 3 is less than that for the later shutdown.

The planar radial power distributions for the above regions with CEA Group 5 fully inserted at beginning and end of Cycle 4 are shown in Figures 5-4 and 5-5 for a Cycle 3 endpoint of 10,000 MWD/T. The maximum planar radial pin peak of 1.56 occurs at beginning of cycle and decreases over the cycle.

For both DNB and PLHGR 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 4. These conservative values, which are specified in Section 7 and Section 9 of this document, establish the allowable limits for power peaking to be observed during operation.

The axial peaking is limited by the limiting conditions for operation on the axial shape index (ASI). Within these ASI limits, the necessary DNB and PLHGR margins are maintained for a wide range of possible axial shapes. The maximum three-dimensional or total peaking factor anticipated in Cycle 4 during normal base load

all rods out operation at full power is 1.81, not including uncertainties and augmentation factors.

5.1.3 Safety Related Data

5.1.3.1 Ejected CEA

The maximum reactivity worths and planar radial power peaks associated with an ejected CEA event are shown in Table 5-4 for both beginning of cycle and end of cycle. These values encompass the worst conditions anticipated during Cycle 4 for any expected Cycle 3 termination point. The pointwise Doppler Feedback technique described in Reference 4 was not utilized in the PDQ calculations of ejected CEA worths and associated power peaks.

5.1.3.2 Dropped CEA

Table 5-5 contains data for several dropped CEA configurations for both extremes in core life and covers the range of expected values. These data have been calculated with the pointwise Doppler feedback technique described in Reference 4. This treatment is consistent with the safety analysis since the time to minimum DNBR is on the order of one to two minutes, allowing ample time for fuel temperature redistribution following the CEA drop. The power peaking values used in the safety analysis are higher than those expected to occur at any time in Cycle 4.

5.1.3.3 Augmentation Factors

Augmentation factors have been calculated for the Cycle 4 core using the calculational model described in Reference 10. The input information required for the calculation of augmentation factors that is specific to the core under consideration includes the fuel densification characteristics, the radial pin power distribution and the single gap peaking factors. Augmentation factors

for the Cycle 4 core have been conservatively calculated by combining for input the largest single gap peaking factors with the most conservative (flattest) radial pin power distribution.

The calculations yield non-collapsed clad augmentation factors showing a maximum value of 1.049 at the top of the core. The augmentation factors for Cycle 4 are compared to the reference cycle values calculated with the same model in Table 5-6.

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 11, which is the same method used for the reference cycle.

5.3 NUCLEAR DESIGN METHODOLOGY

The coarse mesh computer program ROCS (Reference 12) has been used along with the standard fine mesh design program PDQ (Reference 13) in the Cycle 4 safety analysis.

- a. ROCS was used to survey a variety of core configurations to determine limiting conditions.
- b. ROCS was used to obtain axial power shapes, to weight the relative importance of fine mesh PDQ planar power and burnup distributions in the determination of three-dimensional effects and to determine the impact of the three-dimensional gross power distributions on reactivity parameters.
- c. ROCS was used to compute selected safety parameters. The calculation of those limiting parameters which require knowledge of 1-pin peaking factors continues to be based on the fine mesh PDQ program.
- d. Two- and three-dimensional ROCS calculations were used in conjunction with two-dimensional PDQ calculations to obtain best estimate core parameters such as those shown in Table 5-3.

5.4 UNCERTAINTIES IN MEASURED POWER DISTRIBUTIONS

The power distribution measurement biases and uncertainties which are applied to reload cycles are:

	<u>Base Load Operation</u>	<u>Load Follow Operation</u>
Fq: $\bar{\mu} + \sigma$	7.0%	10.0%
Fr: $\bar{\mu} + \sigma$	6.0%	8.0%

These numbers are to be used when the INCA model described in Reference 11 is used for monitoring power distribution parameters during operation.

In the development of LCOs and LSSS data for Cycle 4, allowances for power distribution uncertainties have been applied which are consistent with the measurement uncertainties and biases given above. The allowances are presented in Table 5-7.

CALVERT CLIFFS UNIT 1 CYCLE 4
PHYSICS CHARACTERISTICS

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	<u>Units</u>	<u>Reference Cycle</u>	<u>Reload Cycle Calculated Values</u>
<u>Dissolved Boron</u>			
<u>Dissolved boron content for criticality, CEAs withdrawn</u>			
hot (565°F), full power, equilibrium xenon, BOC	PPM	913 ¹ 854 ²	1030 ¹ 970 ²
Boron worth			
hot (565°F) BOC	PPM/%Δρ	93	95
hot (565°F) EOC	PPM/%Δρ	81	83
<u>Reactivity Coefficients (CEAs Withdrawn)</u>			
<u>Moderator temperature coefficients, hot operating</u>			
hot (565°F), equilibrium xenon, BOC	10 ⁻⁴ Δρ/°F	-0.4	-0.4
hot (565°F), EOC	10 ⁻⁴ Δρ/°F	-2.2	-2.1
Doppler coefficient			
hot (532°F) zero power, BOC	10 ⁻⁵ Δρ/°F	-1.50	-1.50
hot (565°F), full power, BOC	10 ⁻⁵ Δρ/°F	-1.14	-1.20
hot (565°F), full power, EOC	10 ⁻⁵ Δρ/°F	-1.24	-1.37
<u>Total Delayed Neutron Fraction, β_{eff}</u>			
BOC		.00610	.00616
EOC		.00530	.00522
<u>Neutron Generation Time, λ*</u>			
BOC	10 ⁻⁶ sec	25.4	24.8
EOC	10 ⁻⁶ sec	29.2	29.1

Notes¹Early previous cycle shutdown.²Late previous cycle shutdown.

TABLE 5-2

CALVERT CLIFFS UNIT 1 CYCLE 4
 LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCES, $\% \Delta \rho$

	BOC		EOC	
	<u>Reference Cycle</u>	<u>Reload Cycle</u>	<u>Reference Cycle</u>	<u>Reload Cycle</u>
Worth Available				
Worth of all CEAs inserted ¹	8.1	8.9	8.9	9.4
Stuck CEA allowance	1.1	1.8	1.6	1.7
Worth of all CEAs less highest worth CEA stuck out	7.0	7.1	7.3	7.7
Worth Required (Allowances)				
Power defect, HFP to HZP	2.0	1.6	2.7	2.4
Moderator voids	0.0	0.0	0.1	0.1
CEA bite, boron deadband and maneuvering band	0.5	0.5	0.5	0.5
Shutdown margin and safeguards allowance	3.4 ²	3.4 ²	3.4	3.4
Total reactivity required	5.9	5.3	6.7	6.2
Available Worth Less Allowances				
Margin available	>1.1	>1.8	0.7	1.5

Notes¹PLRs not included.²Less than 3.4% $\Delta \rho$ is required at BOC.

TABLE 5-3

CALVERT CLIFFS UNIT I CYCLE 4
REACTIVITY WORTH OF CEA REGULATING GROUPS AT HOT FULL POWER, $\% \Delta \rho$

<u>Regulating CEAs</u>	<u>Beginning of Cycle</u>	<u>End of Cycle</u>
Group 5	.64	.68
Group 4	.29	.39
Group 3	.77	.88

Notes

Values shown assume sequential group insertion

TABLE 5-4
 CALVERT CLIFFS UNIT 1 CYCLE 4
 LIMITING VALUES OF CEA EJECTION DATA

	<u>Reference Cycle</u>	<u>Reload Cycle</u>
<u>Post-Ejection Maximum Radial Power Peak</u>		
Full power with Bank 5 inserted; worst CEA ejected	3.20	3.36
Zero power with Banks 5+4+3+2 inserted; worst CEA ejected	8.05	9.83
<u>Maximum Ejected CEA Worth (%$\Delta\rho$)</u>		
Full power with Bank 5 inserted; worst CEA ejected	0.38	0.32
Zero power with Banks 5+4+3+2 inserted; worst CEA ejected	0.52	0.60

Notes

¹ All values are the limiting values used in the transient analyses.

² Locations of CEA types are shown in Figure 5-2 of Reference 4.

³ Uncertainties and allowances are included in the above data.

TABLE 5-5
 CALVERT CLIFFS UNIT 1 CYCLE 4
 FULL LENGTH CEA DROP DATA

<u>BOC</u>	<u>Worth, %Δp</u>	<u>Maximum Percent Increase in Pin Peak</u>
ARO, Drop 12	.11	11.6
ARO, Drop 11	.13	13.3
ARO, Drop 10	.12	8.7
Bank 5 In, Drop 12	.11	11.3
Bank 5 In, Drop 11	.12	12.4
Bank 5 In, Drop 10	.14	9.8
<u>EOC</u>		
ARO, Drop 12	.11	11.5
ARO, Drop 11	.12	13.1
ARO, Drop 10	.16	11.5
Bank 5 In, Drop 12	.11	10.7
Bank 5 In, Drop 11	.12	11.9
Bank 5 In, Drop 10	.17	12.1

Notes

¹No uncertainties are included in the data.

²Location of CEA types can be found in Figure 5-2 of Reference 4.

³CEAs are either fully inserted or fully withdrawn in radial calculations.

⁴ARO = All Rods Out.

TABLE 5-6

CALVERT CLIFFS UNIT 1 CYCLE 4
AUGMENTATION FACTORS AND GAP SIZES

Core Height (Percent)	Core Height (inches)	<u>Reference Cycle</u>		<u>Reload Cycle</u>	
		Noncollapsed Clad Augmentation Factor	Gap Size (Inches)	Noncollapsed Clad Augmentation Factor	Gap Size (Inches)
98.5	134.7	1.069	2.94	1.049	2.94
86.8	118.6	1.065	2.59	1.045	2.59
77.9	106.5	1.061	2.33	1.042	2.33
66.2	90.5	1.056	1.98	1.037	1.98
54.4	74.4	1.049	1.64	1.031	1.64
45.6	62.3	1.043	1.38	1.027	1.38
33.8	46.2	1.034	1.04	1.021	1.04
22.1	30.2	1.024	0.69	1.015	0.69
13.2	18.1	1.017	0.43	1.010	0.43
1.5	2.0	1.003	0.086	1.001	0.086

Notes

Values are based on approved model described in Reference 10.

The conservative reference cycle peak augmentation factor of 1.069 was used in the Cycle 4 safety analyses.

TABLE 5-7CALVERT CLIFFS UNIT I CYCLE 4
ALLOWANCES FOR BASE LOAD OPERATION

	<u>Cycle 4</u>	<u>Reference Cycle</u>
kw/ft LCO	7.0%	5.8%
kw/ft LSSS	7.0%	5.8%
DNBR LCO	6.0%	5.1%
DNBR LSSS	6.0%	5.1%

							0.72	0.93	
					0.73	0.99	1.16	1.15	1.10
				0.85	1.21	1.15	1.02	0.91	1.27
			0.85	1.24	1.08	0.98	1.28 X	0.98	0.97
		0.73	1.21	1.08	1.02	0.99	1.05	1.08	0.89
		0.99	1.15	0.97	0.99	0.93	1.11	0.82	1.05
0.73		1.16	1.02	1.27	1.04	1.10	0.81	1.01	0.78
		1.16	0.91	0.95	1.06	0.80	1.00	0.85	0.95
0.95		1.13	1.26	0.91	0.85	0.98	0.76	0.98	0.61

NOTE: X=MAXIMUM 1-PIN PEAK = 1.48

							0.71	0.90	
					0.71	0.94	1.09	1.08	1.05
				0.81	1.12	1.08	0.98	0.90	1.22
			0.81	1.14	1.03	0.96	1.24 _x	0.98	0.98
		0.71	1.12	1.03	1.02	1.01	1.06	1.12	0.95
		0.94	1.08	0.96	1.01	0.98	1.17	0.90	1.14
		1.09	0.98	1.24	1.06	1.16	0.91	1.13	0.90
0.71		1.09	0.90	0.97	1.11	0.89	1.13	0.98	1.10
0.90		1.07	1.22	0.94	0.92	1.09	0.89	1.13	0.75

NOTE: X=MAXIMUM 1-PIN PEAK = 1.39

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT I CYCLE 4 ASSEMBLY RELATIVE POWER DENSITY AT 6 GWD/T, EQUILIBRIUM XENON	Figure 5-2
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						0.72	0.90	
				0.72	0.93	1.07	1.07	1.03
			0.81	1.09	1.06	0.97	0.91	1.20
		0.81	1.11	1.01	0.96	1.21 _X	0.98	0.98
	0.72	1.10	1.01	1.01	1.01	1.05	1.12	0.96
	0.93	1.06	0.95	1.01	0.98	1.16	0.93	1.15
0.72	1.07	0.97	1.21	1.05	1.16	0.94	1.15	0.93
	1.07	0.91	0.98	1.11	0.92	1.15	1.02	1.13
	0.90							
	1.05	1.20	0.96	0.95	1.12	0.93	1.16	0.81

NOTE: X=MAXIMUM 1-PIN PEAK = 1.35

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT I CYCLE 4 ASSEMBLY RELATIVE POWER DENSITY AT EOC, EQUILIBRIUM XENON	Figure 5-3
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							0.72	0.91
				0.72	1.01	1.17	1.10	1.00
			0.72	1.14	1.16	1.04	0.85	0.90
		0.72	0.86	0.99	1.00	1.34	0.98	0.91
	0.73	1.14	1.00	1.01	1.05	1.14	1.16	0.95
	1.01	1.16	1.00	1.05	1.02	1.23	0.91	1.16
	1.18	1.04	1.33	1.12	1.22	0.91	1.12	0.85
0.73	1.11	0.84	0.96	1.14	0.88	1.10	0.89	0.93
0.93	1.03	0.89	0.86	0.90	1.09	0.83	0.96	0.39

NOTE: X=MAXIMUM 1-PIN PEAK=1.56

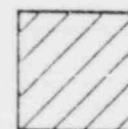


CEA BANK 5
LOCATIONS

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT I CYCLE 4 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, BOC	Figure 5-4
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						0.72	0.88	
				0.71	0.95	1.08	1.02	0.93
			0.67	1.03	1.07	0.99	0.84	0.82
		0.67	0.75	0.93	0.98	1.27	0.98	0.93
	0.71	1.03	0.93	1.01	1.08	1.14	1.20	1.03
	0.95	1.07	0.98	1.08	1.08	1.29	1.02	1.27
	1.08	0.99	1.27	1.14	1.29 _x	1.04	1.25	1.00
0.72	1.02	0.84	0.98	1.20	1.02	1.25	1.03	1.06
0.88	0.94	0.82	0.90	1.01	1.23	0.99	1.09	0.47

NOTE: X = MAXIMUM 1-PIN PEAK=1.47



CEA BANK 5
LOCATIONS

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT I CYCLE 4 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, EOC	Figure 5-5
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6. THERMAL-HYDRAULIC DESIGN

6.1 DNBR ANALYSIS

Steady-state DNBR analyses of Cycle 4 at the rated power of 2700 MWT have been performed using TORC/CE-1. Appropriate adjustments were made to the input of these codes to reflect the Cycle 4 power distribution. Table 6-1 contains a list of pertinent thermal-hydraulic design parameters used for both the safety analysis and the generation of reactor protection system setpoint information.

The analyses were performed in the same manner as for the reference cycle and include the following two considerations.

- (1) As described in Reference 9, TORC/CE-1 was used in the generation of limiting conditions for operation on DNBR margin in the Technical Specifications. TORC was used for all AOOs and postulated accidents which were reanalyzed for Cycle 4.
- (2) The engineering factor on local heat flux, the nuclear uncertainty on F_r and an additional factor designated as an augmentation factor for fuel rod bowing (although a "bowing augmentation factor" has been used here, the bowing penalty on DNBR is still evaluated within NRC's interim guidelines and presented in Section 6.2) were combined statistically for these analyses. (The method by which these factors are combined is described in Section 4 of Reference 14.) Because the local heat flux factor was applied in this way, credit for its inclusion in the TORC deck was taken in determining the setpoints. This is identical to the procedure used in the revised analysis for Calvert Cliffs Unit I Cycle 3 (Reference 1).

Investigations have been made to ascertain the effect of the CEA guide tube wear problem and the sleeving repair on DNBR margins as established by this type of analysis. The findings were reported to the NRC in References 6 and 15 which conclude that the wear problem and the sleeving repair do not adversely affect DNBR margin.

6.2 EFFECTS OF FUEL ROD BOWING ON DNB MARGIN

The fuel rod bowing effects on DNB margin for Calvert Cliffs Unit I have been evaluated within the guidelines set forth in Reference 14, as used in the reference cycle analysis (Reference 1).

A total of 81 fuel assemblies will exceed the NRC-specified DNB penalty threshold burnup of 24,000 MWD/T, as established in Reference 14 during Cycle 4. At the end of Cycle 4, the maximum burnup attained by any of these assemblies will be 42,800 MWD/T. From Reference 14, the corresponding DNB penalty for 42,800 MWD/T is 6.30 percent.

An examination of power distributions for Cycle 4 shows that the maximum radial peak at hot full power in any of the assemblies that eventually exceed 24,000 MWD/T is at least 10.30 percent less than the maximum radial peak in the entire core. Since the percent increase in DNBR has been confirmed to be never less than the percent decrease in radial peak, there exists at least 10.30 percent DNB margin for assemblies exceeding 24,000 MWD/T relative to the DNB limits established by other assemblies in the core. This margin is considerably greater than the Reference 14 reduction penalty of 6.30 percent imposed upon fuel assemblies exceeding 24,000 MWD/T in Cycle 4. Therefore, no power penalty for fuel rod bowing is required in Cycle 4.

TABLE 6-1
 CALVERT CLIFFS UNIT I CYCLE 4
 THERMAL-HYDRAULIC PARAMETERS AT FULL POWER

<u>General Characteristics</u>	<u>Unit</u>	<u>Reference Cycle</u>	<u>Reload Cycle</u>
Total Heat Output (Core Only)	MWT	2700	2700
	10^6 BTU/hour	9215	9215
Fraction of Heat Generated in Fuel Rod		0.975	0.975
Primary System Pressure			
Nominal	PSIA	2250	2250
Minimum in steady state	PSIA	2200	2200
Maximum in steady state	PSIA	2300	2300
Design Inlet Temperature	°F	549	550
Total Reactor Coolant Flow (Minimum Steady State)	GPM	370,000	370,000
	10^6 lb/hour	139.0	139.0
Coolant Flow Through Core (at 2250 psia, 550°F)	10^6 lb/hour	133.9	135.3 **
Hydraulic Diameter (Nominal Channel)	ft	0.044	0.044
Core Average Mass Velocity (at 2250 psia, 550°F)	10^6 lb/hour-ft ²	2.51	2.53 **
Pressure Drop Across Core (Minimum Steady State Flow Irreversible Δp Over Entire Fuel Assembly)	PSI	10.2	10.6 **
Pressure Drop Across Vessel (Based on Nominal Dimensions and Minimum Steady Flow)	PSI	32.2	32.6 **

(CONTINUED)

TABLE 6-1

(CONTINUED)

<u>General Characteristics</u>	<u>Unit</u>	<u>Reference Cycle</u>	<u>Reload Cycle</u>
Number of Fuel Rods in Core		38,176	38,176
Core Average Heat Flux (Accounts for Above Fraction of Heat Generated in Rod and Axial Densification Factor)	BTU/hour-ft ²	181,200	181,200
Total Heat Transfer Areas (Accounts for Axial Densification Factor)	ft ²	49,600	49,600
Average Linear Heat Rate of Fuel Rod (Cold, Undensified, Includes Above Fraction of Heat Generated in Rod)	kw/ft	6.05	6.05
Film Coefficient at Average Conditions	BTU/hour-ft ² , °F	5850	5850
Maximum Clad Surface Temperature	°F	657	657
Average Film Temperature Difference	°F	33	33
Average Core Enthalpy Rise	BTU/lb	69	68**
Calculational Factors			
Engineering heat flux factor		1.03	1.03
Engineering factor on hot channel heat input		1.03	1.03
Flow factors			
inlet plenum nonuniform distribution		1.05	*
rod pitch, bowing and clad diameter		1.065	1.065
Fuel densification factor (axial shrinkage)		1.01	1.01

*Not used in TORC; inlet flow distribution is input

**These parameters have changed due to the decrease in bypass flow

7.0 ACCIDENT AND TRANSIENT ANALYSIS OTHER THAN LOCA

The purpose of this section is to present the results of the safety analysis (other than LOCA) for Calvert Cliffs Unit 1, Cycle 4 at 2700 MWt. The events considered for this analysis are listed in Table 7-1. These are the design basis events for the plant. These events can be categorized into the following groups:

1. Anticipated Operational Occurrences for which the Reactor Protection System prevents the Specified Acceptable Fuel Design Limits (SAFDLs) from being exceeded;
2. Anticipated Operational Occurrences for which the initial steady state overpower margin must be maintained in order to prevent the SAFDLs from being exceeded;
3. Postulated Accidents.

Each of the events listed in Table 7-1 has been reviewed for Cycle 4 to determine if an explicit reanalysis was required. Table 7-1 indicates the analysis status of each event. Table 7-2 presents the safety parameters used in the cycle 4 analysis in comparison to the reference cycle. The review of each design basis event (DBE) entails a comparison between all the current and reference cycle key transient parameters that significantly impact the results of the event. The reference analysis for each event is the analysis upon which the licensing of Calvert Cliffs Unit 1, Cycle 3 was based except where noted differently. If all the current cycle values of key parameters for a particular event are bounded by (conservative with respect to) the reference cycle, no reanalysis is required. In some instances, a reanalysis is performed if it is deemed beneficial from the standpoint of enhanced operating flexibility or if it is desired to bound parameters which are expected to become more adverse in future cycles.

The results of the review show that the key parameters to all the DBEs for Cycle 4 operation are the same as the specified reference cycle input parameters, except for the following:

1. CEA drop time to 90% inserted
2. Integrated Radial Peaking Factor (F_r)
3. Seized Rotor Pin Census
4. Core Bypass Flow Fraction
5. RTD Response Time

For all DBEs other than those reanalyzed, the Calvert Cliffs Unit 1 safety analysis submitted either in the FSAR or in previous reload cycle license submittals bound the results that would be obtained for Unit 1, Cycle 4 and demonstrate safe operation of Calvert Cliffs Unit 1, Cycle 4 at 2700 MWt.

Since the CEA drop time to 90% insertion has increased for Cycle 4, the Loss of flow event, CEA ejection event, RCS depressurization event, Seized Rotor event and the CEA withdrawal event were reanalyzed. These events are adversely impacted by the CEA drop time, since a reactor trip is necessary to terminate the event.

For Cycle 4 the Integrated Radial Peaking Factor (F_r) has decreased in comparison to Cycle 3. In addition, the net core mass flow has increased for Cycle 4 in comparison to Cycle 3 due to the decrease in the calculated value of the core bypass flow. The decreased peaking factor and the increased core flow will not have any adverse impact on the events listed in Table 7-1 for Cycle 4.

TABLE 7-1

CALVERT CLIFFS UNIT 1, CYCLE 4
EVENTS CONSIDERED IN TRANSIENT AND ACCIDENT ANALYSIS

	<u>Analysis Status</u>
Anticipated Operational Occurrences for which the RPS Assures no Violation of SAFDLs:	
Control Element Assembly Withdrawal	Reanalyzed
Boron Dilution	Not Reanalyzed
Startup of an Inactive Reactor Coolant Pump	Not Reanalyzed
Excess Load	Not Reanalyzed
Loss of Load	Not Reanalyzed
Loss of Feedwater Flow	Not Reanalyzed
Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed
Reactor Coolant System Depressurization	Reanalyzed
Loss of Coolant Flow ¹	Reanalyzed
Loss of AC Power	Not Reanalyzed
Anticipated Operational Occurrences which are Dependent on Initial Overpower Margin for Protection Against Violation of SAFDLs:	
Loss of Coolant Flow ¹	Reanalyzed
Loss of AC Power	Not Reanalyzed
Full Length CEA Drop	Not Reanalyzed
Part Length CEA Drop	Not Reanalyzed
Part Length CEA Malpositioning	Not Reanalyzed
Transients Resulting from Malfunction of One Steam Generator	Not Reanalyzed
Postulated Accidents:	
CEA Ejection	Reanalyzed
Steam Line Rupture	Not Reanalyzed
Steam Generator Tube Rupture	Not Renanalyzed
Seized Rotor	Reanalyzed

¹Requires Low Flow Trip

TABLE 7-2

CALVERT CLIFFS UNIT 1, CYCLE 4
CORE PARAMETERS ASSUMED IN THE SAFETY ANALYSES

<u>Physics Parameters</u>	<u>Units</u>	<u>Unit 1 Cycle 3 Values</u>	<u>Unit 1 Cycle 4 Values</u>
Planar Radial Peaking Factors			
For DNB Margin Analyses			
Unrodded Region		1.65*	1.58
Bank 5 Inserted		1.78*	1.71
For kw/ft Limit Analyses (F_{xy})			
Unrodded Region		1.66*	1.66
Bank 5 Inserted		1.79*	1.79
Peak Augmentation Factor		1.069	1.069
Moderator Temperature Coefficient	$10^{-4} \Delta\rho/^{\circ}\text{F}$	-2.5 \rightarrow +.5	-2.5 \rightarrow +.5
Shutdown Margin (Value used in Zero Power (SLB))		-3.4	-3.4
<u>Safety Parameters</u>			
Power Level	% of 2700 Mwt	102	102
Maximum Steady State Core Inlet Temperature	$^{\circ}\text{F}$	550	550
Minimum Steady State RCS Pressure	psia	2200	2200
Minimum Reactor Coolant Core Flow (2200 psia, 550 $^{\circ}\text{F}$)	lb/hr	134.22	135.24
Full Power	lp	-.12*	-.14
Maximum CEA Insertion at Full Power	% Insertion of Group 5	25	25
Maximum Allowable Initial Peak Linear Heat Rate for DBEs other than LOCA	kw/ft	16.0	16.0
Steady State Linear Heat Rate to Fuel Centerline Melt	kw/ft	21.0	21.0

*These values are revised limits quoted in the revised Unit 1 Cycle 3 license submittal (Reference 2C), not the values quoted in the original Cycle 3 license submittal (Reference 2A).

7.1 CEA WITHDRAWAL EVENT

The CEA withdrawal event was reanalyzed for Cycle 4 due to the increase in the Resistance Temperature Detector (RTD) response time to envelope future cycles and the increase in the CEA drop time to 90% insertion in comparison to the reference cycle. The reference cycle for this event is the analysis upon which the licensing of Calvert Cliffs Unit 2, Cycle 2 (see Reference 12) was based.

As stated in CENPD-199-P (Reference 1), the CEA Withdrawal event initiated at rated thermal power is one of the DBEs analyzed to determine a bias factor used in establishing the TM/LP setpoints. This bias factor, along with conservative temperature, pressure, and power readings assures that the TM/LP trip prevents the DNBR from dropping below the SAFDL limits (DNBR = 1.19 based on CE-1 correlation) for a CEA Withdrawal event. Hence, this event was analyzed for Cycle 4 to generate the bias term input to the TM/LP trip.

The CEA Withdrawal transient may require protection against exceeding both the DNBR and fuel centerline melt (kw/ft) SAFDLs. Depending on the initial conditions and the reactivity insertion rate associated with the CEA withdrawal, either the Variable High Power Level or Thermal Margin/Low Pressure (TM/LP) trip reacts to prevent exceeding the DNBR SAFDL. An approach to the kw/ft limit is terminated by either the Variable High Power Level trip or the Axial Flux Offset trip.

The zero power case was analyzed to demonstrate that SAFDLs are not exceeded. For the zero power case, a reactor trip, initiated by the variable high power trip at 40% of rated thermal power, is assumed in the analysis.

The key parameters for the cases analyzed are reactivity insertion rate due to rod motion and moderator temperature feedback effects, and initial axial power distribution. The Resistance Temperature Detector (RTD) response time is also important in determining the pressure bias factor.

The range of reactivity insertion rates considered in the analysis is given in Table 7.1-1, along with the values of other key parameters used in the analysis of this event.

The maximum reactivity insertion rate for cycle 4 is 1.3×10^{-4} $\Delta\rho/\text{sec}$ at all power levels. This reactivity withdrawal rate was calculated by combining the maximum CEA differential worth of 2.6×10^{-4} $\Delta\rho/\text{inch}$ and the maximum CEA Withdrawal speed of 30 inches per minute.

The initial axial power shape and the corresponding scram worth versus insertion used in the analysis of both cases is a bottom peaked shape. This power distribution maximizes the time required to terminate the decrease in DNBR following a trip.

The CEA Withdrawal transient initiated at rated thermal power results in the maximum pressure bias factor of 62.0 psia. This bias factor accounts for measurement system processing delays during the CEA Withdrawal event. The pressure bias factor for this cycle has increased from the reference cycle due to the increase in the RTD time constant and the increase in the CEA drop time to 90% insertion. This pressure bias factor is used in generating TM/LP trip setpoints to prevent the SAFDLs from being exceeded during a CEA Withdrawal event.

The zero power case initiated at the limiting conditions of operation results in a minimum DNBR of 1.71. Also, the analysis shows that the fuel centerline temperatures are well below those corresponding to the fuel centerline melt SAFDL.

The Sequence of events for the zero power case is presented in Table 7.1-2. Figures 7.1-1 to 7.1-4 presents the transient behavior of core power, core average heat flux, the RCS pressure and the RCS temperatures.

The analysis of the CEA Withdrawal event presented herein, shows that the DNBR and fuel centerline melt SAFDLs will not be exceeded during a CEA Withdrawal transient.

TABLE 7.1-1

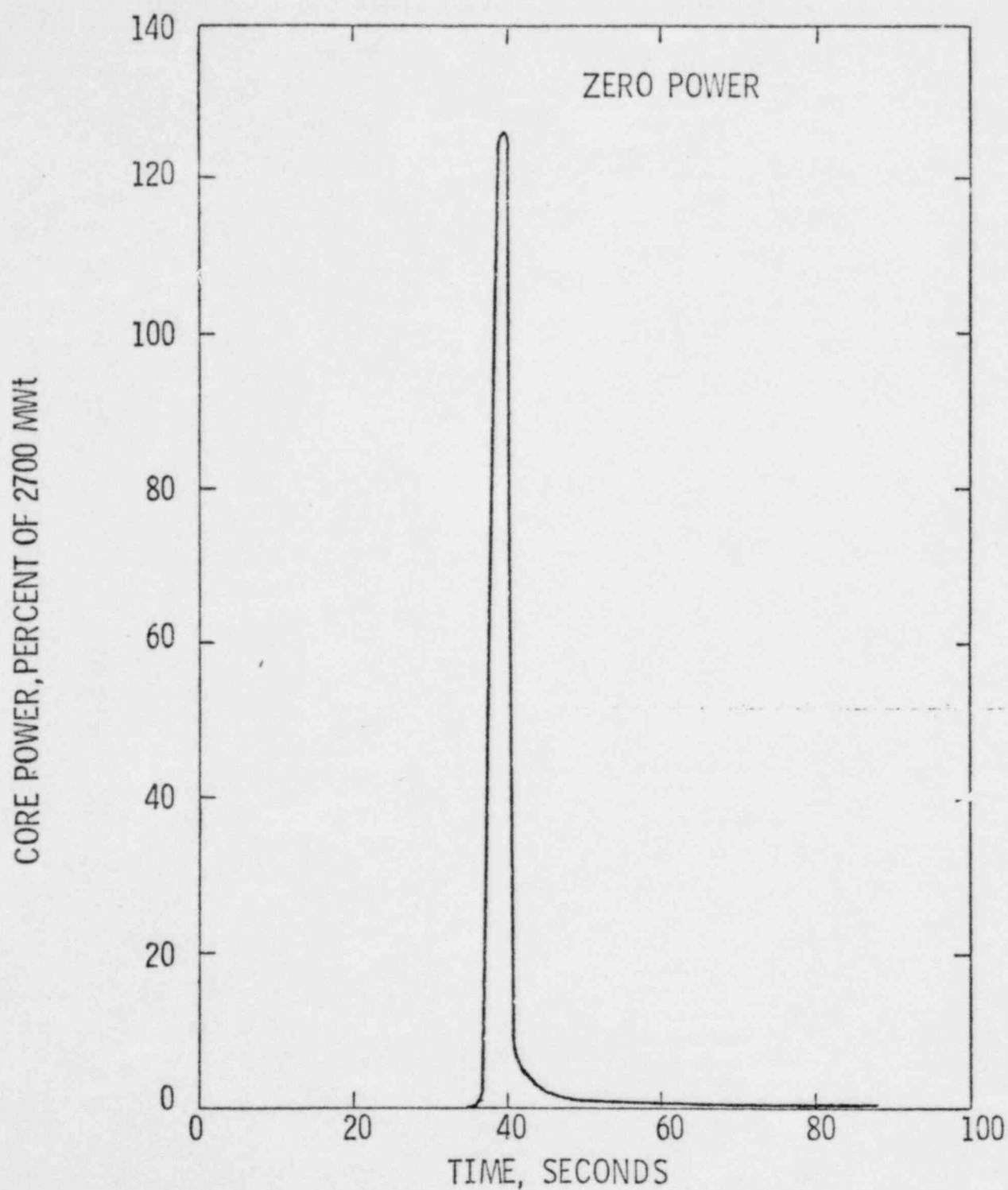
KEY PARAMETERS ASSUMED IN THE CEA WITHDRAWAL ANALYSIS

<u>Parameter</u>	<u>Units</u>	<u>Unit 2 Cycle 2</u>	<u>Unit 1 Cycle 4</u>
Initial Core Power Level (H2P, HFP)	MWt	0,102% of 2700	0,102% of 2700
Core Inlet Coolant Temperature (H2P, HFP)	$^{\circ}\text{F}$	532,550	532,550
Reactor Coolant System Pressure	psia	2200	2200
Moderator Temperature Coefficient	$10^{-4} \Delta\rho/^{\circ}\text{F}$	+5	+5
Doppler Coefficient Multiplier		.85	.85
CEA Worth at Trip - FP	$10^{-2} \Delta\rho$	-5.14	-5.14
CEA Worth at Trip - ZP	$10^{-2} \Delta\rho$	-3.4	-3.4
Reactivity Insertion Rate	$\times 10^{-4} \Delta\rho/\text{sec}$	2.0	1.3
Holding Coil Delay Time	sec	0.5	0.5
CEA Time to 90 Percent Insertion (Including Holding Coil Delay)	sec	2.5*	3.1
Resistance Temperature Detector Response Time (T)	sec	5.0	8.0
Rod Group Withdrawal Speed	in/min	30.0	30.0
Maximum CEA Differential Worth	$\times 10^{-4} \Delta\rho/\text{inch}$	4.0	2.6

*The reference cycle analysis assumed a CEA drop time to 90% insertion value of 2.5 seconds (see Reference 12a) but in a subsequent submittal (see Reference 12b) a CEA drop time to 90% insertion value of 3.0 seconds was justified for Unit 2, Cycle 2.

TABLE 7.1-2SEQUENCE OF EVENTS FOR
CEA WITHDRAWAL FROM ZERO POWER

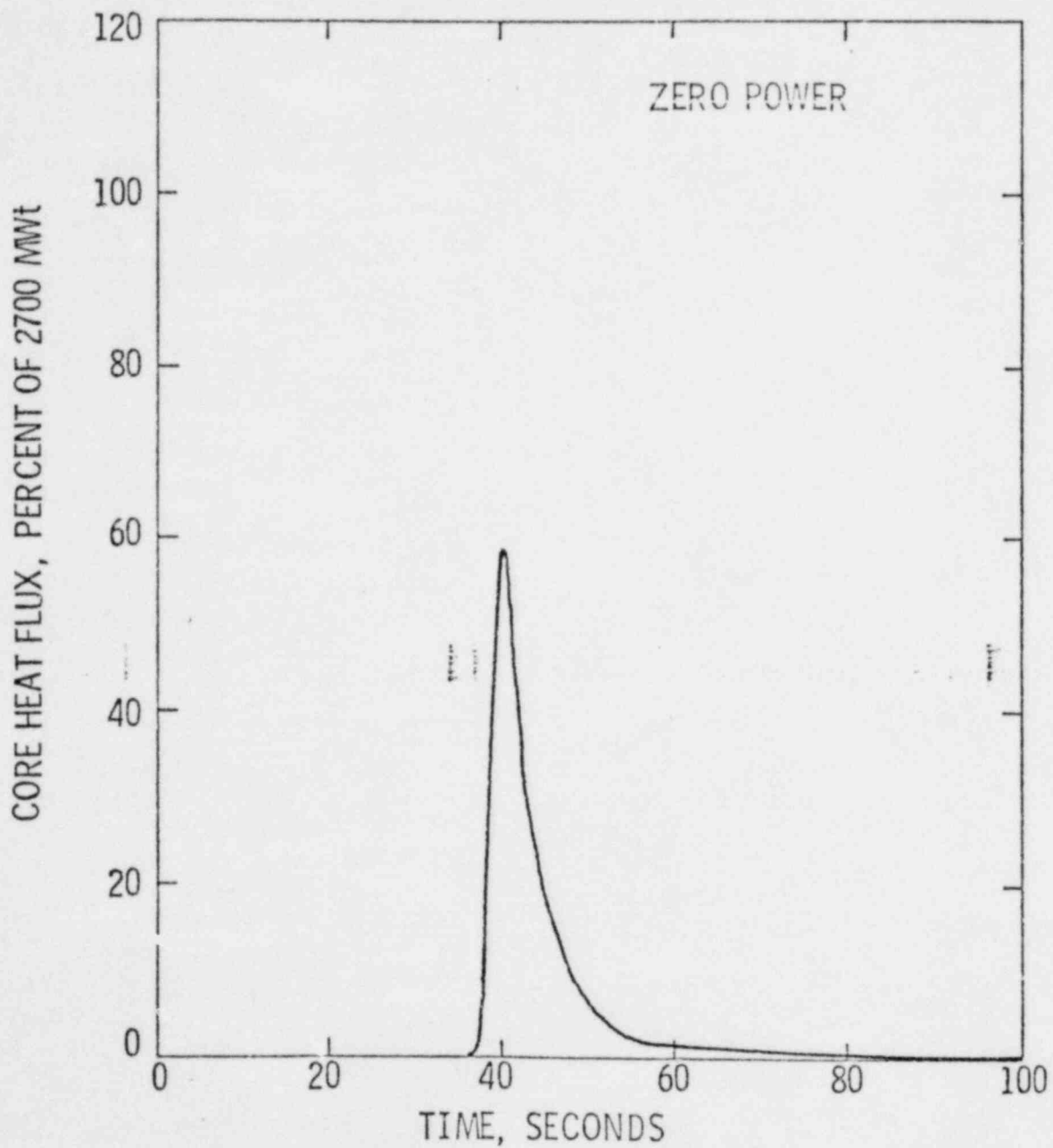
<u>Time (sec)</u>	<u>Event</u>	<u>Setpoint or Value</u>
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	---
37.7	High Power Trip Signal Generated	40% of 2700 MWt
38.1	Trip Breakers Open	----
38.6	CEAs Begin to Drop Into Core	---
39.0	Maximum Power Reached	126.4 of 2700 MWt
40.15	Maximum Heat Flux Reached	58.7 of 2700 MWt
40.15	Minimum DNBR Occurs	1.71
42.3	Maximum Pressurizer Pressure Reached	2353 psia



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CEA WITHDRAWAL EVENT
CORE POWER VS TIME

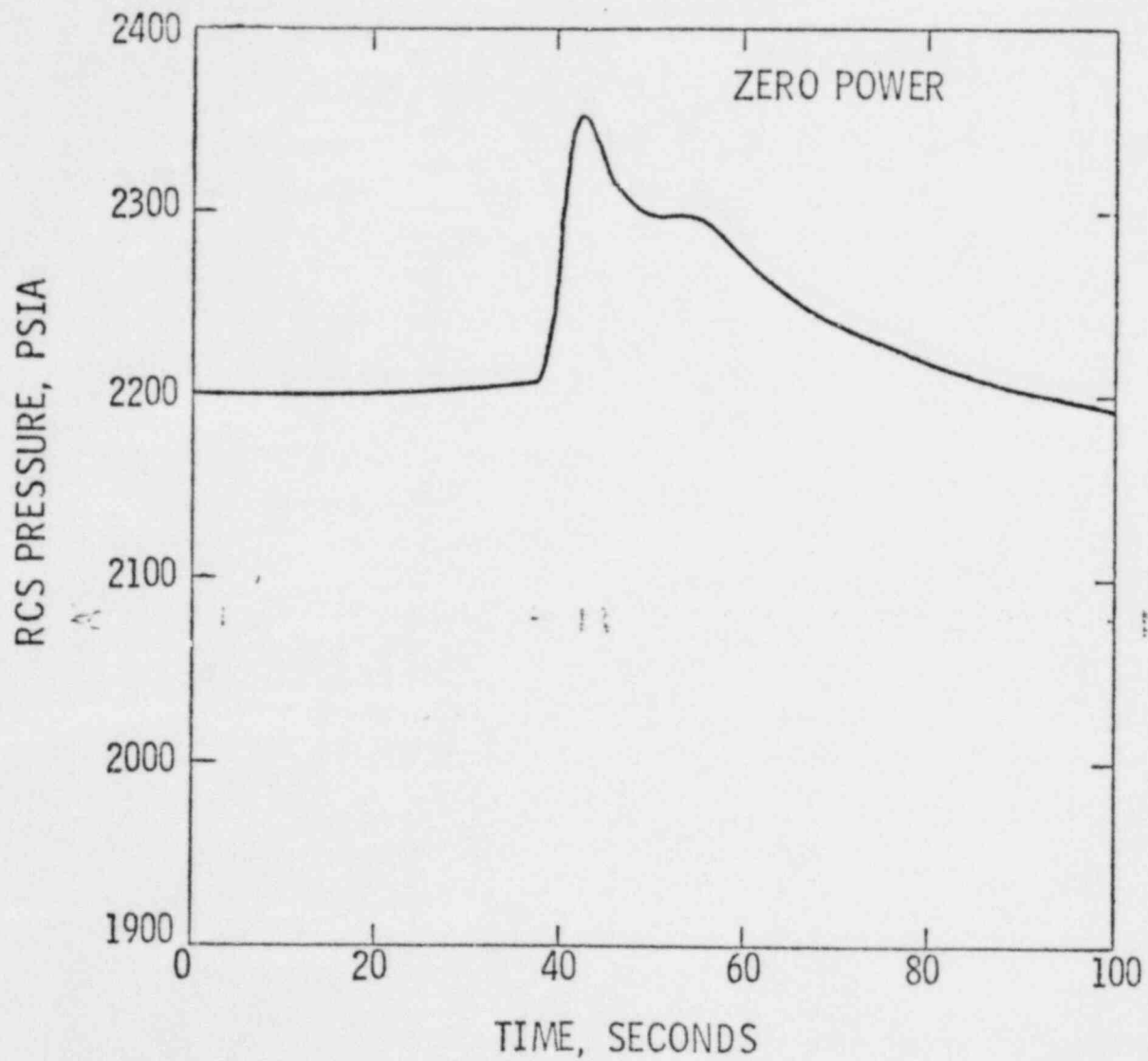
Figure
7.1-1



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CEA WITHDRAWAL EVENT
CORE HEAT FLUX VS TIME

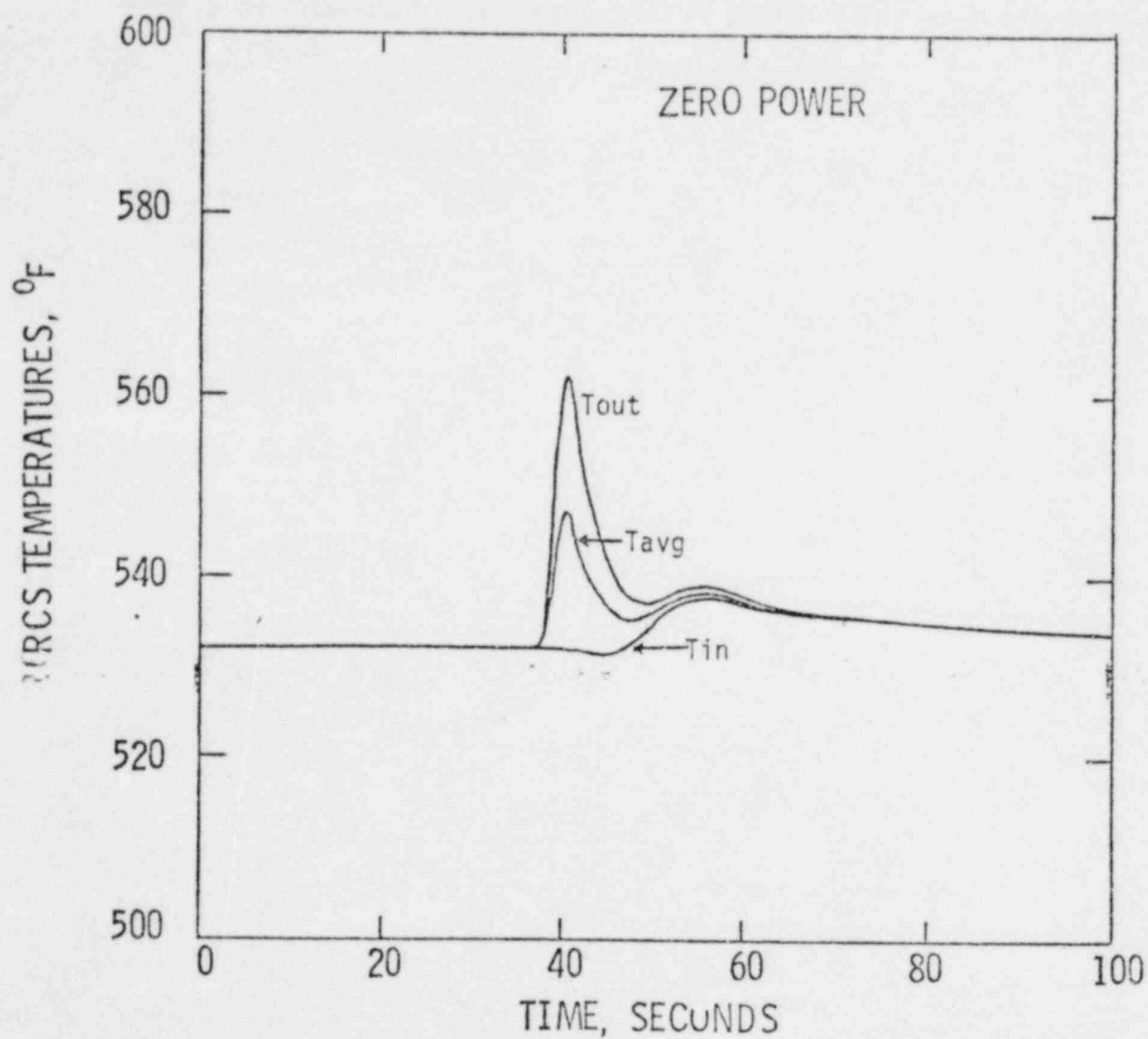
Figure
7.1-2



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CEA WITHDRAWAL EVENT
RCS PRESSURE VS TIME

Figure
7.1-3



7.1-2. RCS DEPRESSURIZATION EVENT

The RCS Depressurization event was reanalyzed for Cycle 4 to assess the impact of increasing the CEA drop time to 90% insertion from 2.5 seconds for Cycle 3 to 3.1 seconds for Cycle 4. As stated in CENPD-199-P (Reference 1), this event is one of the DBEs analyzed to determine a bias term input to the TM/LP trip. Hence, this event was analyzed for Cycle 4 to obtain a pressure bias factor. This bias factor accounts for measurement system processing delays during this event. The trip setpoints incorporating a bias factor at least this large will provide adequate protection to prevent the DNBR SAFDL from being exceeded during this event.

The assumptions used to maximize the rate of pressure decrease and consequently the fastest approach to DNBR SAFDL's are:

- 1) The event is assumed to occur due to an inadvertent opening of both pressurizer relief valves while operating at rated thermal power. This results in a rapid drop in the RCS pressure and consequently a rapid decrease in DNBR.
- 2) The initial axial power shape and the corresponding scram worth versus insertion used in the analysis is a bottom peaked shape. This power distribution maximizes the time required to terminate the decrease in DNBR following a trip.
- 3) The charging pumps, the pressurizer heaters and the pressurizer backup heaters are assumed to be inoperable. This maximizes the rate of pressure decrease and consequently maximizes the rate of approach to DNBR SAFDL.

The analysis of this event shows that the pressure bias factor is 35 psia which is less than that required by the CEA Withdrawal event. Hence, the use of the pressure bias factor determined by the CEA Withdrawal event will prevent exceeding the SAFDLs during an RCS Depressurization event.

7.3 LOSS OF COOLANT FLOW EVENT

The Loss of Coolant Flow event was reanalyzed for Cycle 4 to determine the impact on margin requirements that must be built into the Limiting Conditions for Operations (LCOs) due to the increase in the CEA drop time to 90% insertion.

The methodology used to evaluate this event is identical to that employed in the Unit 1, Cycle 3 license submittal (Reference 2). The methodology utilizes the computer code STRIKIN II (Reference 3) to determine the time dependent hot channel and core average heat fluxes distributions during the transient. For conservatism credit for the heat flux decay was taken only for axial power distributions for which the initial minimum DNBR was located in an axial region of the core where the scram rods have passed the axial node of minimum DNBR before the time at which minimum DNBR is reached. For those axial power distributions analyzed that did not meet the above criterion, the methodology utilized is consistent with CENPD-199-P (Reference 1).

The computer code TORC (Reference 4) was used for all DNBR calculations. This is consistent with the methods used by C-E to calculate the DNB margin requirements. The TORC code and its application to the analyses were reviewed and approved by NRC in response to the revised Unit 1, Cycle 3 license submittal (Reference 2).

The 4-Pump Loss of Coolant Flow produces a rapid approach to the DNBR SAFDL due to the rapid decrease in the core coolant flow. Protection against exceeding the DNBR SAFDL for this transient is provided by the initial steady state thermal margin which is assured by maintaining the technical specifications' LCOs on DNBR margin and by the response of the RPS which provides an automatic reactor trip on low reactor coolant flow as measured by the steam generator differential pressure transmitters.

The transient is characterized by the flow coastdown curve given in Figure 7.3-1. Table 7.3-1 lists the key transient parameters used in the present analysis.

Table 7.3-2 presents the NSSS and RPS responses during a four pump loss of flow initiated at the most negative shape index (I_p) allowed by the LCOs. The low flow trip setpoint is reached at 1.0 seconds and the scram rods start dropping into the core one second later. A minimum CE-1 DNBR of 1.25 is reached at 2.3 seconds. Figures 7.3-2 to 7.3-5 presents the core power, heat flux, RCS pressure, and core coolant temperatures as a function of time. Figure 7.3-6 presents a trace of hot channel DNBR vs time for the limiting case that is characterized by an $I_p = -.15$.

The low flow trip, in conjunction with the Initial Overpower Margin maintained by the LCOs in the Technical Specifications assure that the minimum DNBR will be greater than or equal to 1.19 for the Loss of Coolant Flow event.

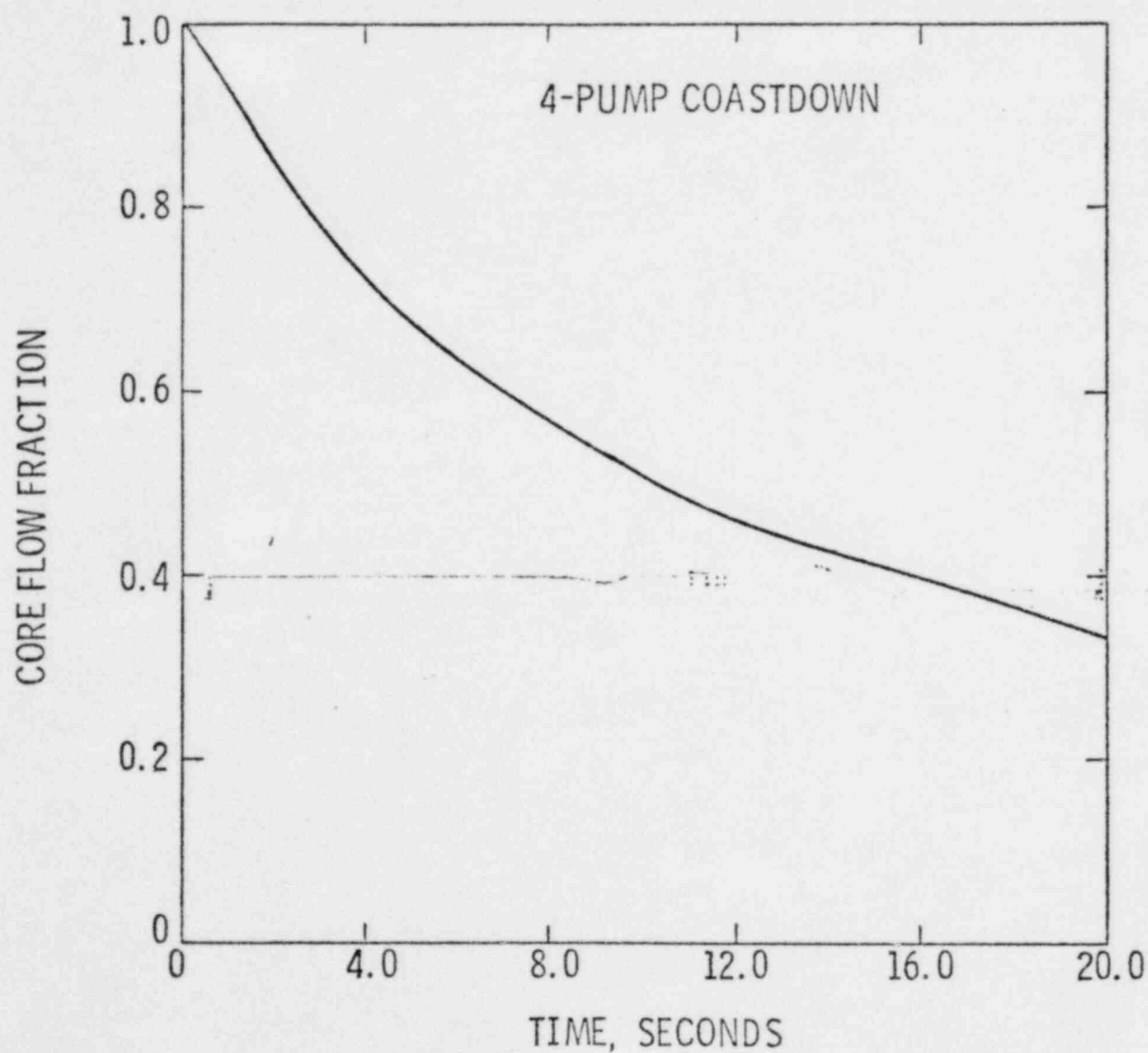
TABLE 7.3-1

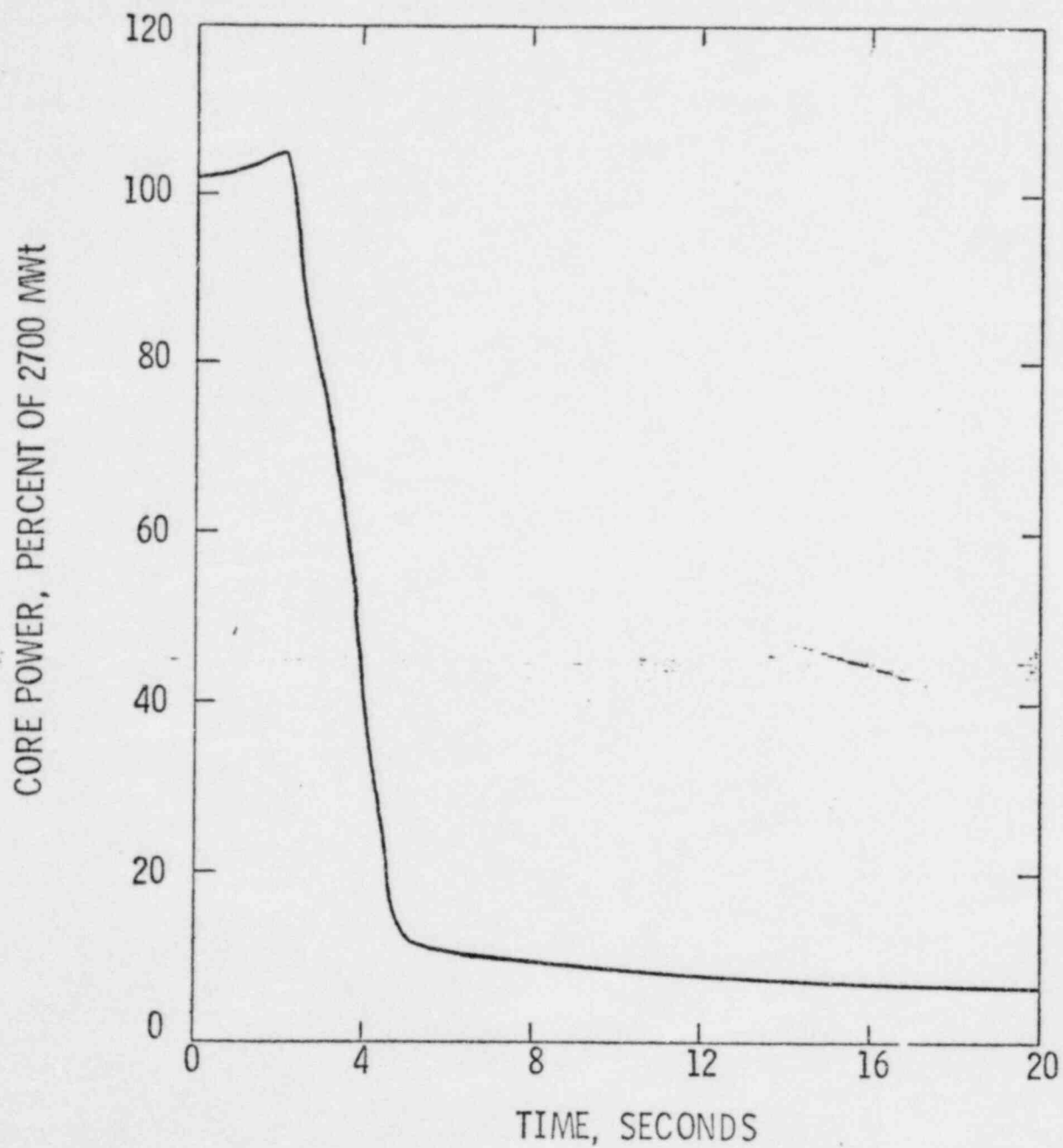
KEY PARAMETERS ASSUMED IN THE LOSS OF COOLANT FLOW ANALYSIS

<u>Parameter</u>	<u>Units</u>	<u>Unit 1, Cycle 3</u>	<u>Unit 1, Cycle 4</u>
Initial Core Power Level	(MWt)	102% of 2700	102% of 2700
Initial Core Inlet Coolant Temperature	(°F)	550	550
Initial Core Mass Flow Rate	(10 ⁶ lbm/hr)	134.22	135.24
Reactor Coolant System Pressure	(psia)	2200	2200
Initial Steam Generator Pressure	(psia)	861	861
Moderator Temperature Coefficient	(10 ⁻⁴ Δρ/F)	+1.5	+1.5
Doppler Coefficient Multiplier	--	.85	.85
LFT Response Time	sec	0.5	0.5
CEA Holding Coil Delay	sec	0.5	0.5
CEA Time to 90% Insertion (Including Holding Coil Delay)	sec	2.5	3.1
CEA Worth at Trip	(10 ⁻² Δρ)	-5.7	-5.7
Total Radial Peaking Factor (F_r^T)		1.65	1.58
4-Pump RCS Flow Coastdown		Figure 7.1-1 of Reference 2	Figure 7.3-1

TABLE 7.3-2SEQUENCE OF EVENTS FOR
LOSS OF FLOW

<u>Time (sec)</u>	<u>Event</u>	<u>Setpoint or Value</u>
0.0	Loss of Power to all Four Reactor Coolant Pumps	----
1.0	Low Flow Trip	38% of 4-Pump Flow
1.5	Trip Breakers Open	----
2.0	Shutdown, CEAs begin to Drop into Core	----
2.3	Minimum CE-1 DNBR	1.25
5.5	Maximum RCS Pressure, psia	2276

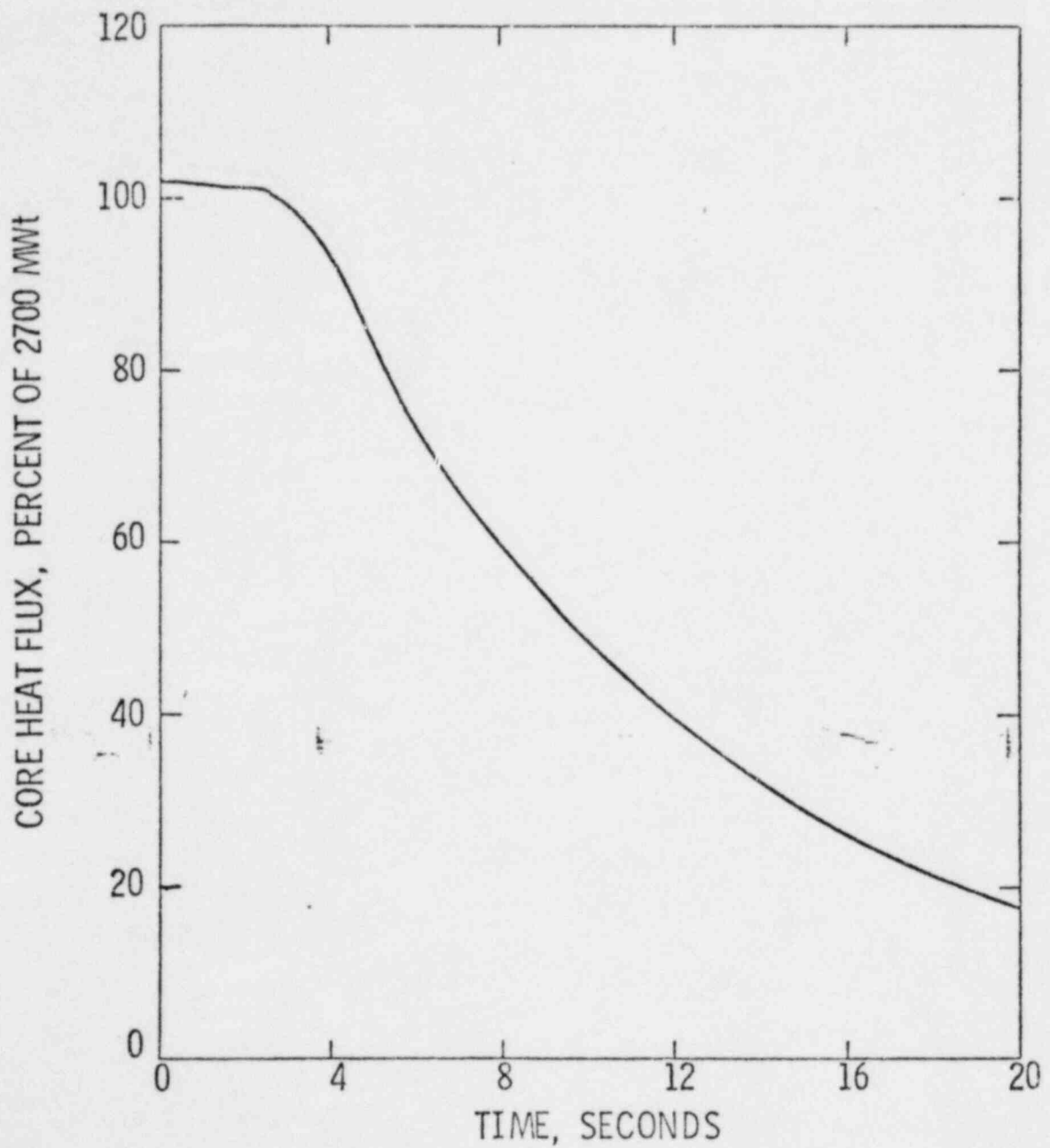




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LOSS OF COOLANT FLOW EVENT
CORE POWER VS TIME

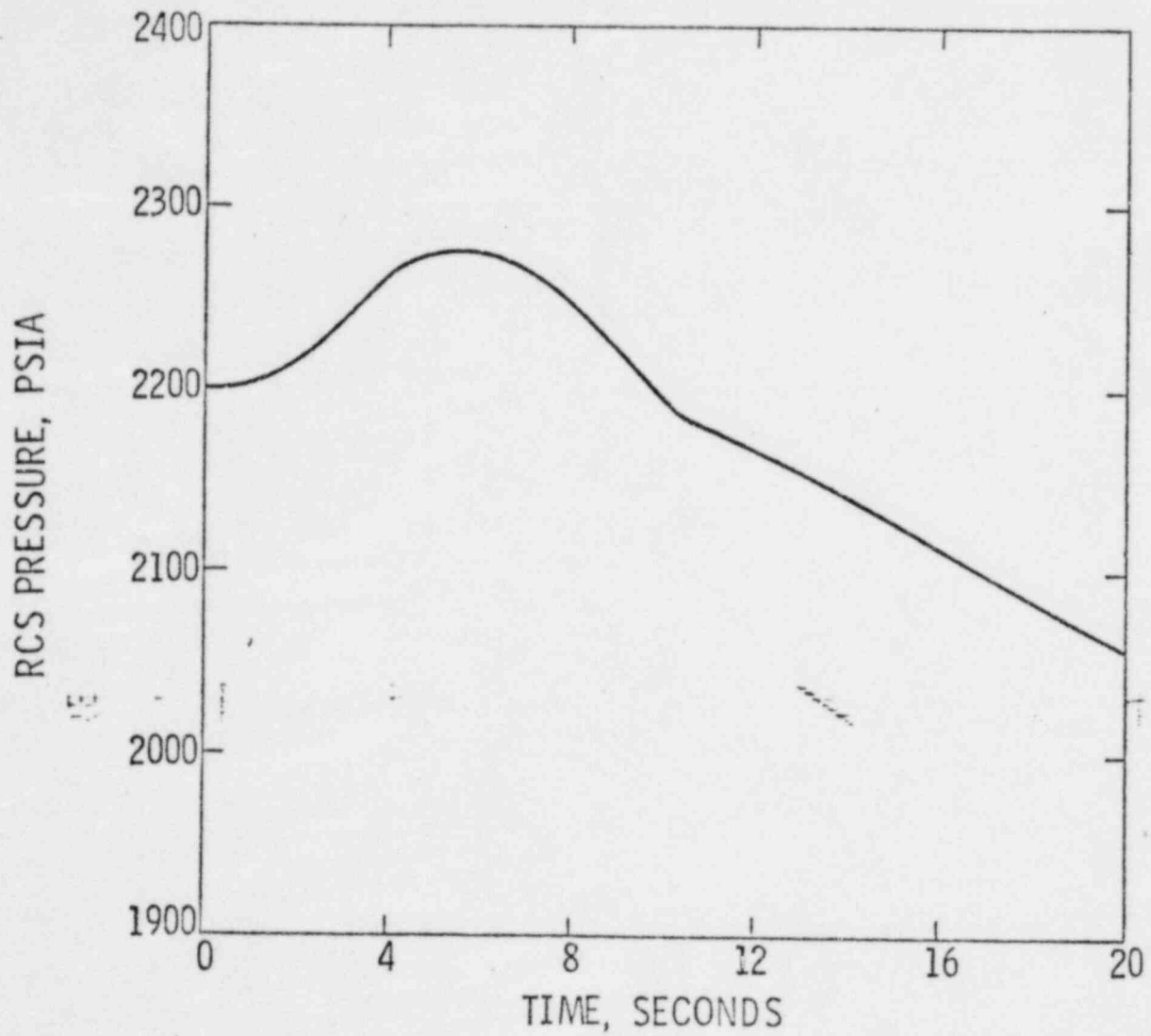
Figure
7.3-2



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LOSS OF COOLANT FLOW
CORE HEAT FLUX VS TIME

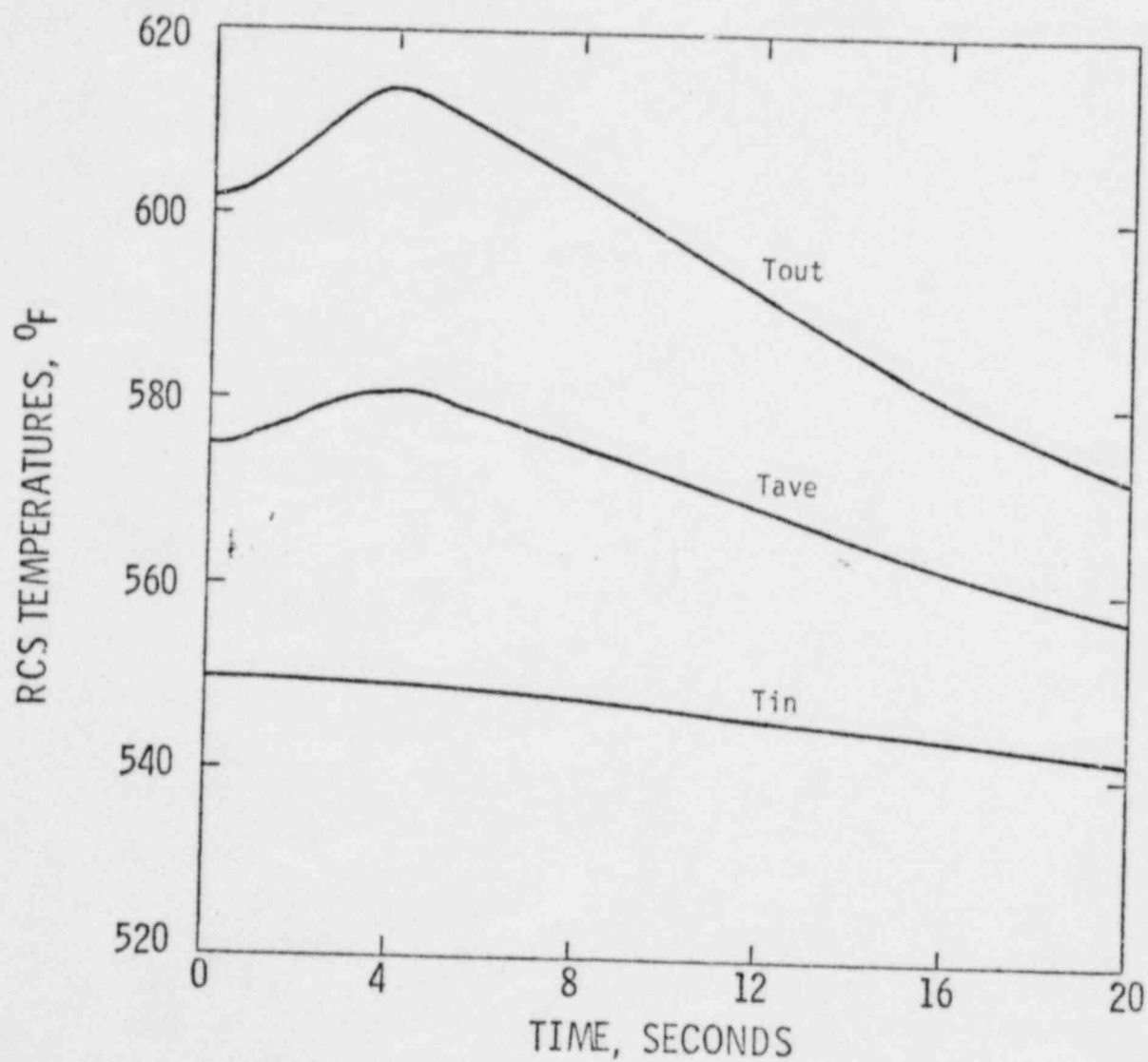
Figure
7.3-3



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LOSS OF COOLANT FLOW EVENT
RCS PRESSURE VS TIME

Figure
7.3-4

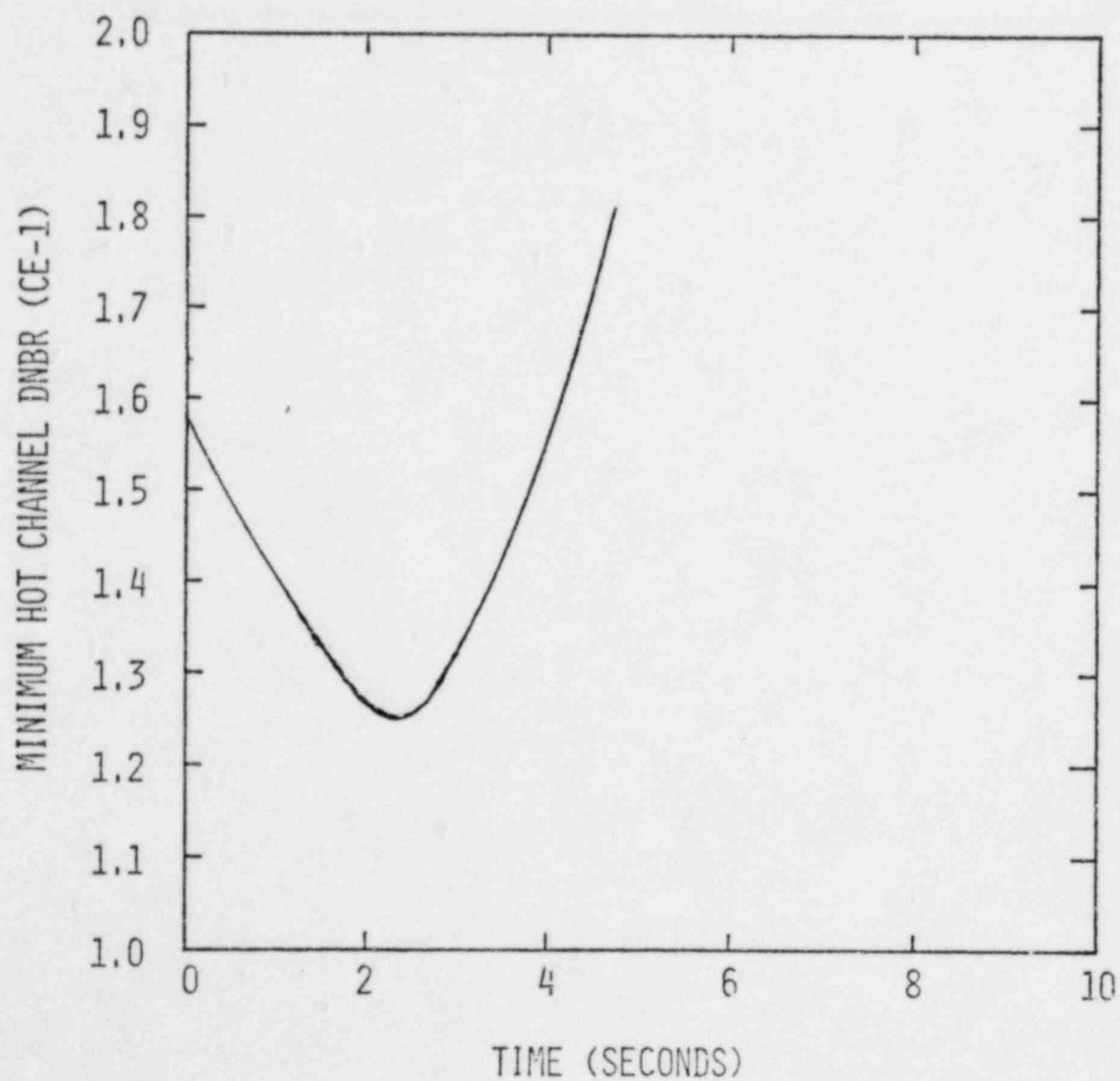


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LOSS OF COOLANT FLOW EVENT
RCS TEMPERATURES VS TIME

Figure

7.3-5



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LOSS OF COOLANT FLOW EVENT
MINIMUM HOT CHANNEL DNBR (CE-1) VS TIME

Figure
7.3-6

7.4 CEA EJECTION EVENT

The CEA Ejection event was reanalyzed for Cycle 4 to assess the impact of increasing the CEA drop time to 90% insertion and the increase in the augmentation factor in comparison to the reference cycle. In addition, the zero power case was analyzed due to the decrease in axial peak in comparison to the reference cycle. The reference cycle for this event is the analysis upon which the licensing of Calvert Cliffs Unit 2, Cycle 2 (see Reference 12) was based on. Hence, this event was reanalyzed to demonstrate that the criterion for clad damage is not exceeded during Cycle 4 operation.

To bound the most adverse conditions during the cycle, the most limiting of either the Beginning of Cycle (BOC) or End of Cycle (EOC) value was 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 $+0.5 \times 10^{-4} \Delta p / ^\circ F$ was used which results in positive reactivity feedback with increasing coolant temperatures. A EOC Beta Fraction was used in the analysis to produce the highest power rise during the event. For the full power and zero power cases, the axial power distributions were selected to yield conservative results. The corresponding shape indices were -0.21 for the full power case and -0.40 for the zero power case. These shapes are conservative with respect to the most negative shape indices allowed by the DNBR monitoring band. This is consistent since the power shifts to the top of the core after the CEA ejects.

The reactivity-forced power transient was simulated by a digital computer program, CHIC-KIN (Reference 5), which simultaneously solves the one group neutron point kinetics equations together with the time and space dependent thermal and hydraulics equations for heat generation and transport within a single channel. The kinetics model incorporates the standard six-delay group representation along with explicit reactivity contributions from: (a) CEA motion, (b) Doppler effect, and (c) moderator density variations. By simulating the core average channel, the CHIC-KIN code computes the core average integrated energy output during the course of the transient.

In the CEA Ejection event, the principal reactivity feedback mechanism affecting the power transient is the Doppler feedback. In the point kinetics approach, utilized in CHIC-KIN, a spatial Doppler weighting factor (k) accounts for the fact that the Doppler feedback effect is a function of the spatial flux distribution. In order to represent the radial Doppler effect in a conservative manner, a space-time analysis was performed in which point kinetics calculations for various radial slices were compared with time-dependent, two-dimensional diffusion theory results obtained with a C-E modified version of the TWIGL code (Reference 6). The results of the space-time analysis have demonstrated that the use of the static (non-Doppler flattened) radial fuel rod peaking factor, as obtained from two-dimensional diffusion theory calculations, in conjunction with the average hot spot energy releases, yield energy increases that are conservatively large. Radial Doppler weighting factors obtained as a function of the ejected CEA worth are defined such that CHIC-KIN and TWIGL results give the same total core energy release.

The average energy rise in the hottest fuel pellet is obtained from the following relationship:

$$\Delta E_H = (P/A)_H \times \Delta E_{Ave} \times K - E_{HT} \quad (7.4-1)$$

Where ΔE_{Ave} is the average core energy rise obtained from CHIC-KIN; $(P/A)_H$ (the three-dimensional fuel rod peaking factor) is the ratio of the hot spot power density to the core average power density obtained from static, non-Doppler flattened diffusion theory calculations; K is the reduction factor defined above. For the zero power case, it is conservatively assumed that E_{HT} , which accounts for heat transferred out of the fuel rod during the transient, is zero.

The average energy in the hottest fuel pellet at the beginning of the transient is added to the net average energy rise in the hottest fuel pellet as obtained from Equation (7.4-1) to determine the total average enthalpy in the hottest fuel spot in the core. A similar procedure is used to compute the total centerline enthalpy in the hottest spot. The initial energy is obtained by correlating the initial local fuel temperature with an empirical temperature-enthalpy relationship (see Reference 7).

The spatial variation of the core local-to-average power ratio results from the convolution of the axial power distribution with radial pin power census distributions for the post-ejection condition, which are based on static core physics calculations. Combining these results with the total average and centerline enthalpies in the hottest fuel spot yields the fractional number of fuel rods with specific total average and centerline enthalpies. The calculated enthalpy values are compared to threshold enthalpy values to determine the amount of fuel experiencing the various degrees of fuel damage. These threshold enthalpy values are (References 8, 9, and 10).

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

The criterion for determining the fraction of fuel rods that will release their radioactive fission products during a CEA ejection is the same as the one quoted above for determining clad damage. Thus, it is assumed that any fuel rod that exceeds a total average enthalpy of 200 cal/gm releases all of its gap activity. The gap activity corresponding to the hottest fuel rod during the core cycle is conservatively assumed for each rod that suffers clad damage.

The zero power CEA ejection event was analyzed assuming the core is initially operating at 1 MWt for conservatism. At zero power, a Variable Overpower trip is conservatively assumed to initiate at 40% (30% + 10% uncertainty) of 2754 MWt and terminates the event.

The full and zero power cases were analyzed, assuming a value of 0.05 seconds for the total ejection time, which is consistent with the FSAR. Table 7.4-1 lists all the key parameters used in this analysis.

Table 7.4-2 presents the results of the two ejection cases analyzed for Cycle 4 in comparison to the reference cycle. As seen from Table 7.4-2, the average energy deposited for both the full and zero power cases has increased. The increase for the full power case is due to the increase in CEA drop time to 90% insertion in comparison to the reference cycle analyses; while for the zero power case, the decrease in the axial peak offsets the increase in the CEA drop time to 90% insertion.

The power transient produced by a CEA ejection initiated at the maximum allowed power is shown in Figure 7.4-1, and at zero power is shown in 7.4-2.

Since the criterion of clad damage (i.e., less than 200 cal/gm) is not exceeded for either the full or zero power CEA ejection, no fuel pins are predicted to fail.

KEY PARAMETERS ASSUMED IN THE CEA EJECTION ANALYSES

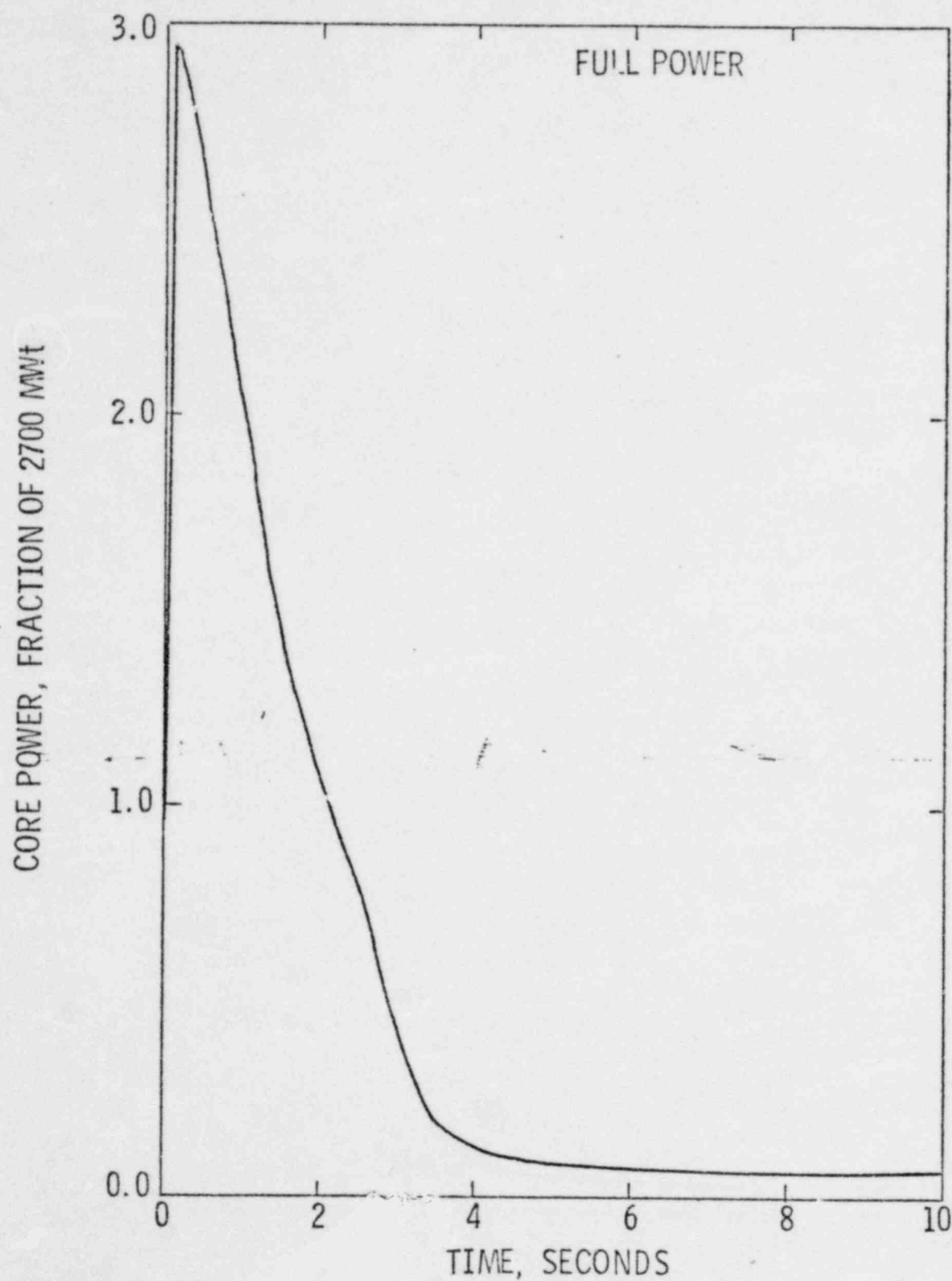
<u>Parameter</u>	<u>UNITS</u>	<u>Unit 2 Cycle 2</u>	<u>Unit 1 Cycle 4</u>
<u>Full Power</u>			
Core Power Level	MWt	2754	2754
Core Average Linear Heat Rate of Fuel Rod	kw/ft	6.23	6.12
Moderator Temperature Coefficient	$10^{-4} \Delta p/^{\circ}F$	+5	+5
Ejected CEA Worth	% Δp	.32	+32
Delayed Neutron Fraction, β		.0047	.0047
Post-Ejected Radial Power Peak		3.36	3.36
Axial Power Peak		1.39	1.39
CEA Bank Worth at Trip	% Δp	3.88	3.88
Augmentation Factor		1.060	1.069
K-Factor		.92	.92
Tilt Allowance		1.03	1.03
CEA Drop Time to 90% Inserted	sec	2.5*	3.1
<u>Zero Power</u>			
Core Power Level	MWt	1.0	1.0
K-factor		.89	.89
Ejected CEA Worth	% Δp	.60	.60
Post-Ejected Radial Power Peak		9.83	9.83
Axial Power Peak		2.0	1.6
CEA Bank Worth at Trip	% Δp	2.58	2.58
Tilt Allowance		1.10	1.10
CEA Drop Time to 90% Inserted		2.5*	3.1

*The reference cycle analysis assumed a CEA drop time to 90% insertion value of 2.5 seconds (see Reference 12a) but in a subsequent submittal (see Reference 12b) a CEA drop time to 90% insertion value of 3.0 seconds was justified for Unit 2, Cycle 2.

TABLE 7.4-2

CEA EJECTION RESULTS

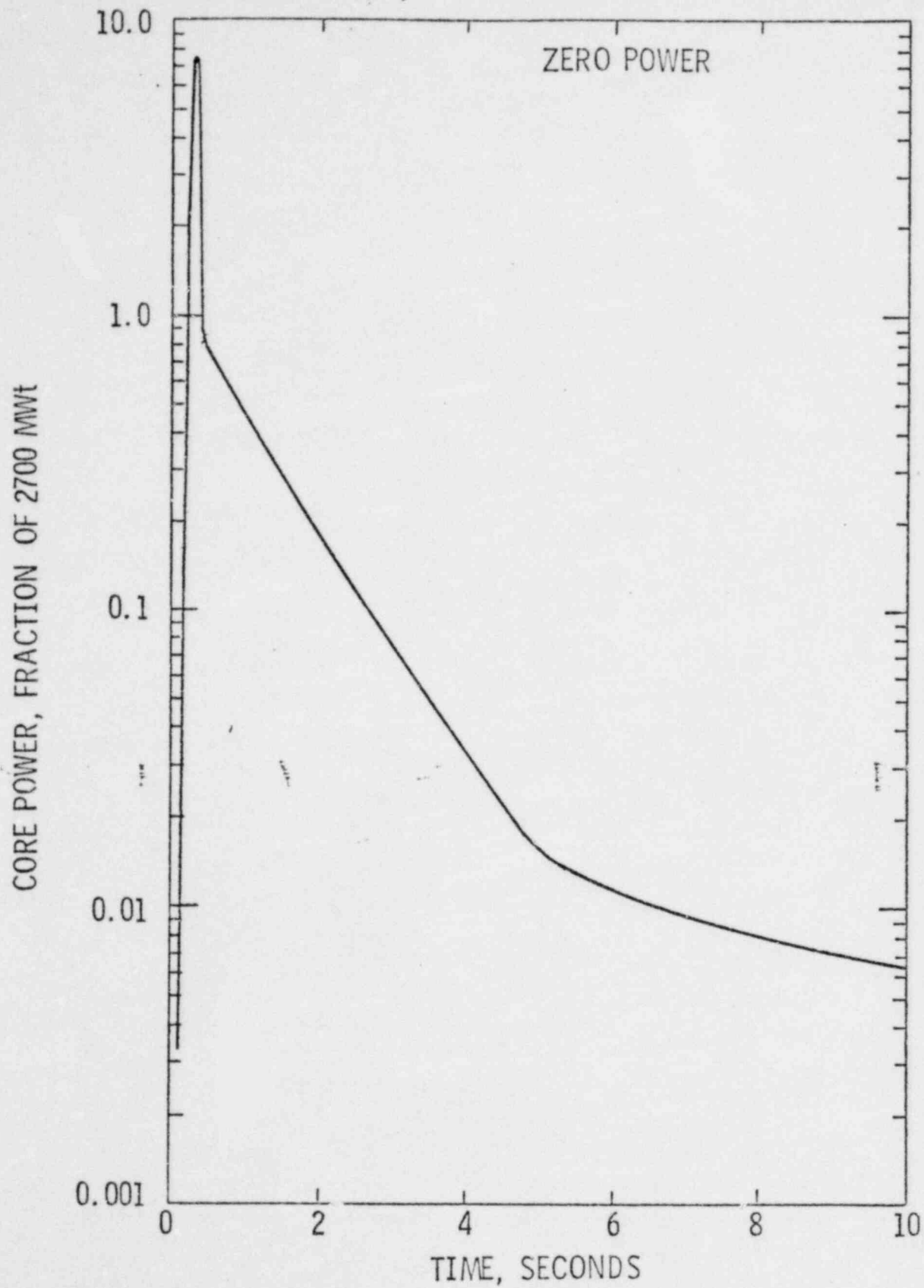
<u>Full Power</u>	<u>Unit 2 Cycle 2</u>	<u>Unit 1 Cycle 4</u>
Total Average Enthalpy of Hottest Fuel Pellet (cal/gm)	193	198
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	263	268
Fraction of Rods that Suffer Clad Damage (Average Enthalpy ≥ 200 cal/gm)	0.0	0.0
Fraction of Fuel Having at Least Incipient Centerline Melting (Centerline Enthalpy ≥ 250 cal/gm)	0.01	.01
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy ≥ 310 cal/gm)	0.0	0.0
<u>Zero Power</u>		
Total Average Enthalpy of Hottest Fuel Pellet (cal/gm)	173	177
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	173	177
Fraction of Rods that Suffer Clad Damage (Average Enthalpy ≥ 200 cal/gm)	0.0	0.0
Fraction of Fuel Having at least Incipient Centerline Melting (Centerline Enthalpy ≥ 250 cal/gm)	0.0	0.0
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy ≥ 310 cal/gm)	0.0	0.0



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CEA EJECTION EVENT
CORE POWER VS TIME

Figure
7.4-1



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CEA EJECTION EVENT
CORE POWER VS TIME

Figure
7.4-2

7.5 SEIZED ROTOR EVENT

The Seized Rotor event was reanalyzed for Cycle 4 due to the changes in the following key parameters.

- 1) The increase in the CEA drop time to 90% insertion
- 2) The decrease in core bypass flow, which increases the net core flow
- 3) The decrease in the Radial Peaking Factor
- 4) A more adverse (flatter) pin census.

The increase in the CEA drop time and the flatter pin census adversely impact the consequences of this event. Increasing the net core flow and decreasing the Radial Peaking Factor will succor the consequences of this event. Hence, a reanalysis was performed for Cycle 4 to ensure that only a small fraction of fuel pins are predicted to fail during a Seized Rotor event.

The Seized Rotor event is assumed to be initiated by the complete seizure of a shaft in one of the reactor coolant pumps. This reduces the core coolant flow rapidly to the 3 pump flow. In the analysis of the event, it is conservatively assumed that the core flow is instantaneously reduced to 3 pump flow. This initiates a reactor trip on low primary coolant flow (93% of 4 pump flow) which terminates the decrease in the DNBR.

The methods used to analyze this event are identical to those reported in the Unit 1, Cycle 2 Stretch Power license submittal (Reference 11). The techniques and methods used to calculate the number of fuel pins experiencing DNB is discussed in detail in Section 7.1 of the Calvert Cliffs Unit 1 Cycle 3 revised license submittal (Reference 2) except TORC/CE-1 instead of COSMO/W-3 was used to calculate the DNBR. The key transient parameters used in this analysis are compared to the reference cycle analysis in Table 7.5-1.

The NSSS and RPS response for the Seized Rotor event initiated at an $I_p = -.15$ is shown in Table 7.5-2. Figures 7.5-1 through 7.5-4 show the core power, core average heat flux, RCS Pressure, and RCS temperatures as a function of time during this event.

A conservatively "flat" pin census distribution (a histogram of the number of pins with radial peaks in intervals of 0.1 in radial peak normalized to the maximum peak) was used to determine the number of pins that experience DNB.

The results indicate that increasing the core flow and decreasing the radial peaking factor offset the increase in the CEA drop time to 90% insertion. It was calculated that for Cycle 4, less than .5 percent of fuel pins will experience DNB for even a short period of time.

For the case of the loss of coolant flow arising from a seized rotor shaft, it is assumed that there is an instantaneous reduction to 3 pump flow. The low flow trip assures that less than .5% of fuel pins experience DNB. This is the same as that calculated for the Reference cycle. Hence, the conclusions reached for reference cycle remain valid for cycle 4.

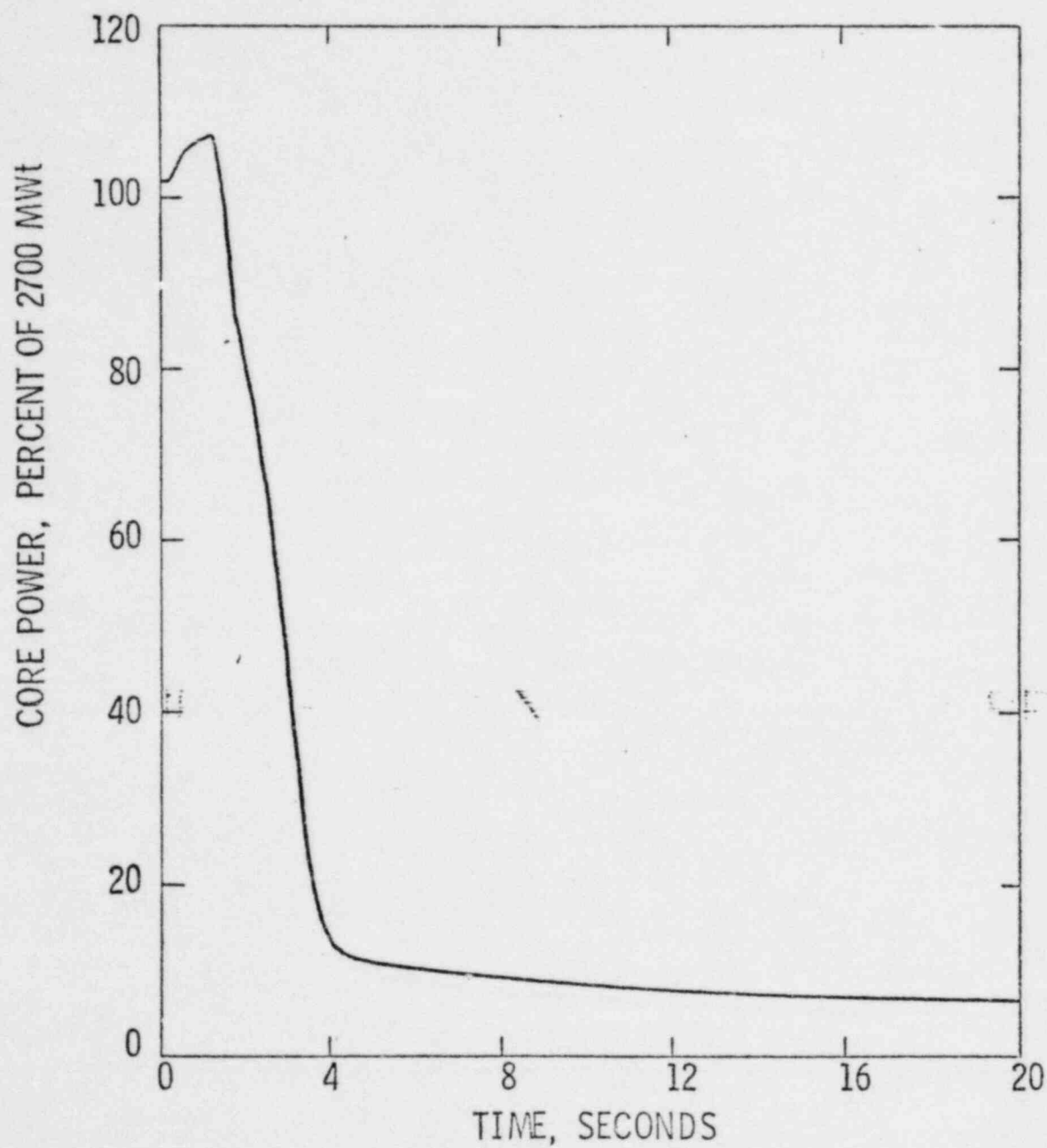
ASSUMPTIONS FOR SEIZED ROTOR INCIDENT

<u>Parameter</u>	<u>Units</u>	<u>Unit 1, Cycle 3</u>	<u>Unit 1, Cycle 4</u>
Initial Core Power Level	MWt	102% of 2700	102% of 2700
Core Inlet Coolant Temperature	$^{\circ}\text{F}$	550	550
Four Pump Core Mass Flow Rate (2200 psia, 550 $^{\circ}\text{F}$)	10^6 lbm/hr	134.22	135.24
Three Pump Core Mass Flow Rate (2200 psia, 550 $^{\circ}\text{F}$)	10^6 lbm/hr	103.4	104.44
Reactor Coolant System Pressure	psia	2200	2200
Steam Generator Pressure	psia	861	861
Moderator Temperature Coefficient	$10^{-4} \Delta\rho/^{\circ}\text{F}$	+5	+5
Doppler Coefficient Multiplier	---	.85	.85
CEA Worth on Trip	$10^{-2} \Delta\rho$	-5.7	-5.7
CEA Drop time to 90% insertion	sec	2.5	3.1

TABLE 7.5-2

SEQUENCE OF EVENTS FOR SEIZED ROTOR

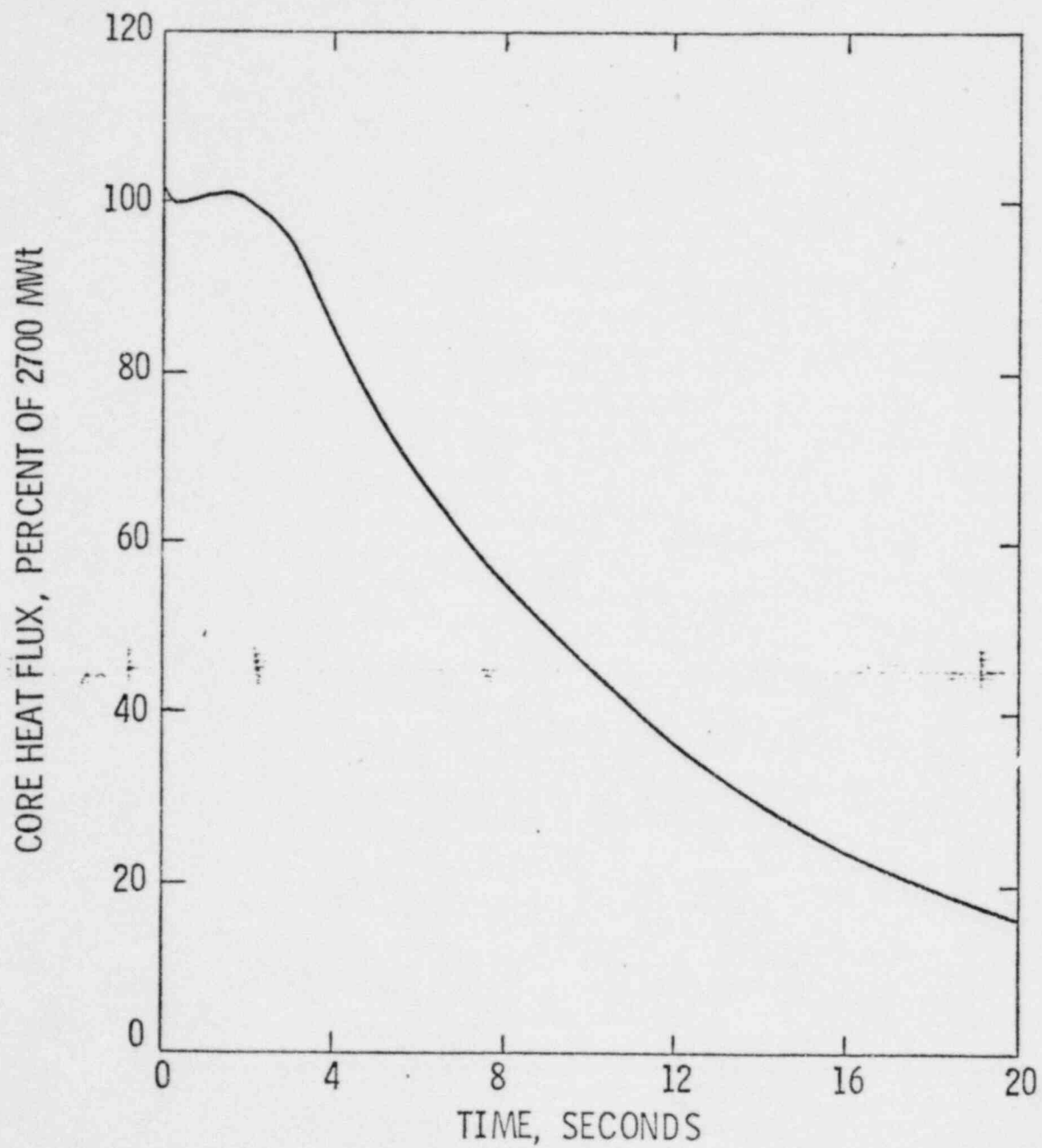
<u>Time (sec)</u>	<u>Event</u>	<u>Setpoint or Value</u>
0.0	Seizure of One Reactor Coolant Pump	---
0.0	Low Coolant Flow Trip	38% of 4-Pump Flow
0.1	Dump Valve Opens	---
0.5	Trip Breakers Open	---
1.0	Shutdown CEAs Begin Dropping into Core	---
3.5	Maximum RCS Pressure	2277



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SEIZED ROTOR EVENT
CORE POWER VS TIME

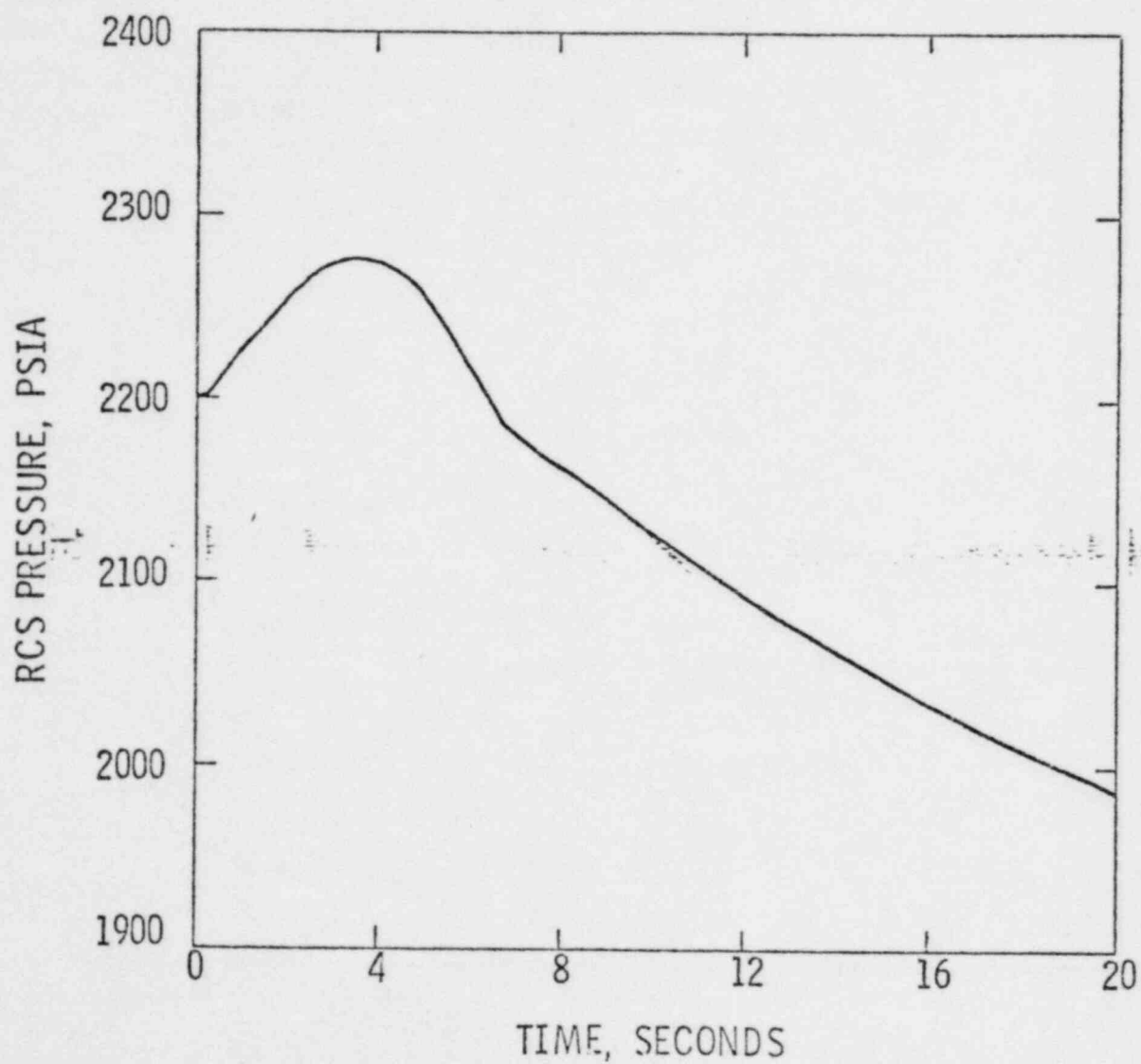
Figure
7.5-1



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SEIZED ROTOR EVENT
CORE HEAT FLUX VS TIME

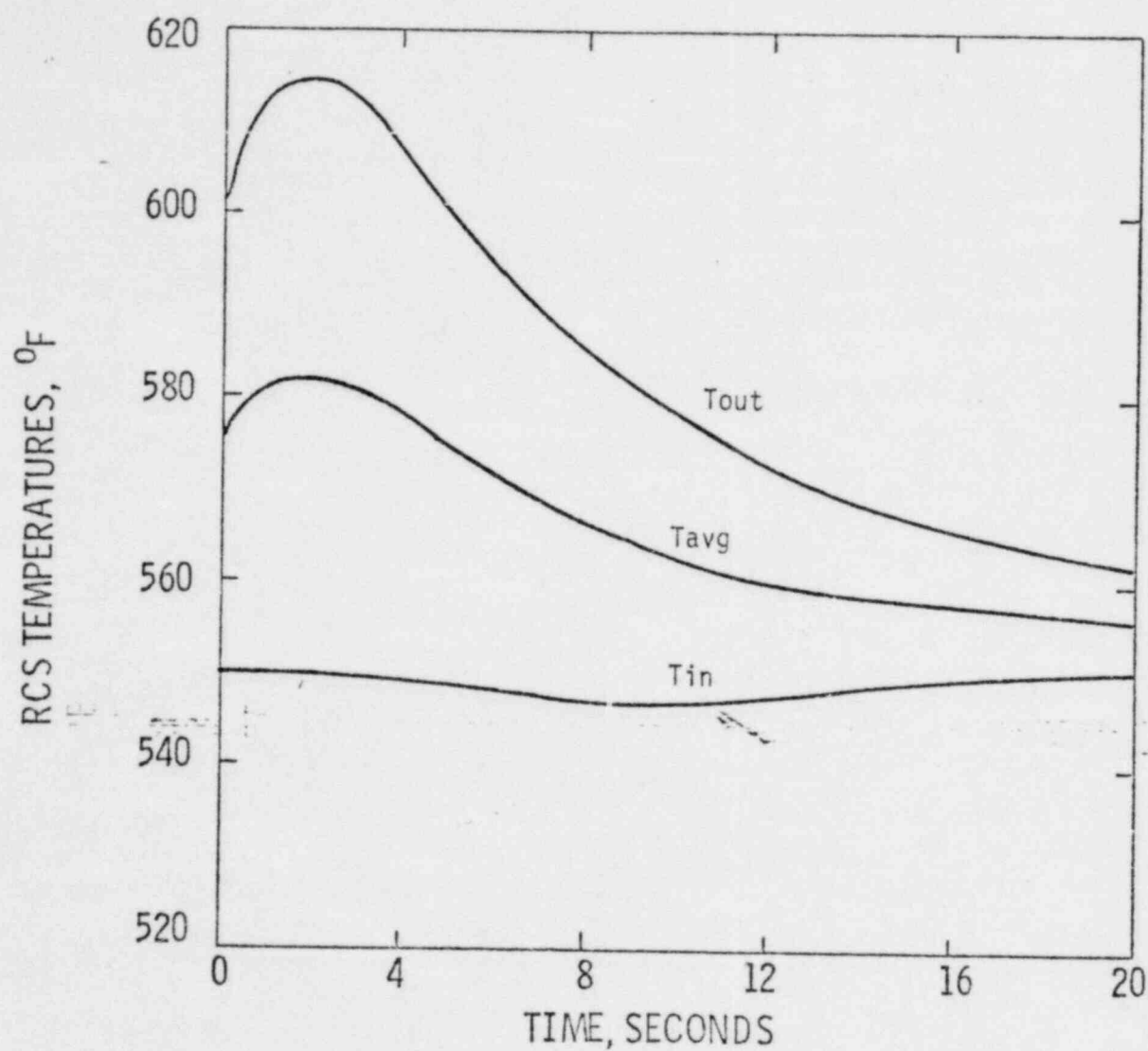
Figure
7.5-2



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SEIZED ROTOR EVENT
RCS PRESSURE VS TIME

Figure
7.5-3



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SEIZED ROTOR EVENT
RCS TEMPERATURES VS TIME

Figure
7.5-4

8.0 ECCS Analysis

8.1 Introduction and Summary

The ECCS performance evaluation for Calvert Cliffs Unit I, Cycle 4, presented herein demonstrates appropriate conformance with the Acceptance Criteria for Light-Water-Cooled Reactors as presented in 10CFR50.46⁽¹⁾. The evaluation demonstrates acceptable ECCS performance for Calvert Cliffs Unit I, during cycle 4, at a peak linear heat generation rate of 14.2 kw/ft. The method of analysis and results are presented in the following sections.

8.2 Method of Analysis

The method of analysis consisted of a comparison of the fuel specific parameters for the limiting fuels in cycles 3 and 4. The comparison demonstrates that the limiting fuel during cycle 4 has a much lower stored energy (i.e. higher gap conductance) than the limiting fuel identified in the cycle 3 analysis of Reference 8. As a result, the peak clad temperature and local clad oxidation (STRIKIN-II)^(4,5) calculations performed for cycle 3⁽⁸⁾ are conservative and applicable for cycle 4. In addition, the blowdown (CEFLASH-4A)⁽²⁾, refill (COMPERC-II)⁽³⁾, and core wide oxidation (COMZIRC)^(3, sup. 1) analyses from the cycle 2 analysis⁽⁷⁾ remain valid for cycle 4. Therefore, the ECCS performance results reported in Reference 8 for cycle 3 are also applicable to cycle 4. The comparison of the fuel specific parameters supporting this method of analysis for cycle 4 is presented below.

8.3 Results

The cycle 4 core contains 216 high density fuel assemblies and one low density Batch B assembly. The highest power pin in the low density Batch B

assembly will not achieve a power level greater than 75% of the highest power pin in the core. Therefore, a Batch B fuel pin will not be limiting in cycle 4.

The remaining 216 high density fuel assemblies contain 72 partially depleted Batch D assemblies, 72 partially depleted Batch E assemblies and 72 fresh Batch F assemblies. Burnup dependent calculations were performed for the high density fuel assemblies with the FATES⁽⁶⁾ and STRIKIN-II^(4,5) codes. The results demonstrate that the most limiting fuel pin during cycle 4 is located in one of the partially depleted Batch E assemblies.

Table 8.1 compares the fuel specific parameters which correspond to the limiting fuels in cycle 3 and cycle 4. As shown in the table, the limiting high density fuel in cycle 4 has a stored energy 268⁰F lower than the limiting fuel in cycle 3. Consequently, the ECCS performance results reported for cycle 3 conservatively bound the performance for cycle 4.

8.4 Conclusion

The comparison between the fuel specific parameters for the limiting fuels in cycles 3 and 4 demonstrates that the cycle 3 ECCS performance analysis conservatively bounds the performance for cycle 4. Therefore, the peak linear heat generation rate of 14.2 kw/ft which was demonstrated to be acceptable for cycle 3 is also an acceptable limit for cycle 4 operation. Conformance of this evaluation is the same as stated in Reference 8.

The statements in Reference 8 demonstrating compliance with Criterion 4 (coolable geometry) and Criterion 5 (long term cooling) remain unchanged. As presented in References 8 and 9, the small breaks are not limiting.

8.5 Computer Code Version Identification

Version 77063 of the STRIKIN-II code of Combustion Engineering's ECCS Evaluation Model was used to perform the burnup dependent calculations in evaluating the fuel data.

Table 8.1

Calvert Cliffs I Cycle IV Core Parameters

<u>Quantity</u>	<u>Value</u>		
	<u>Cycle III</u> <u>Batch B</u>	<u>Cycle IV</u> <u>Batch E</u>	
Gap Conductance at PLHGR	859	1426	BTU/hr-ft ² -°F
Fuel Centerline Temperature at PLHGR	3692	3405	°F
Fuel Average Temperature at PLHGR	2419	2151	°F
Hot Rod Gas Pressure	1221	1026	psia
Hot Rod Burnup (Minimum HGAP)	9170	1522	MWD/MTU

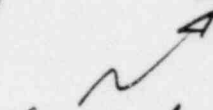
9. TECHNICAL SPECIFICATIONS

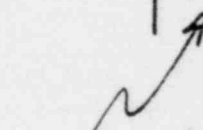
In this section all changes that must be made to the Technical Specifications are provided in order to make the Technical Specifications valid for operation of Cycle 4.

Each page from the Technical Specifications which must be modified is shown with the modification included.

EXAMPLE:

3/4 X.X The existing value is 1.06.

*Current
value* 

 1.10
*Amended
value*

INDEXLIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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DEFINITIONSCHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
- b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor subcritical is or would be subcritical from its present condition assuming:

- a. ^a All full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn, and
- b. ~~No change in part length control element assembly position.~~

DEFINITIONSREACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.26 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.28 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

LOAD FOLLOW OPERATION

1.29 LOAD FOLLOW OPERATION shall involve daily power level changes of more than 10% RATED THERMAL POWER or daily insertion of control rods below the long term insertion limits.

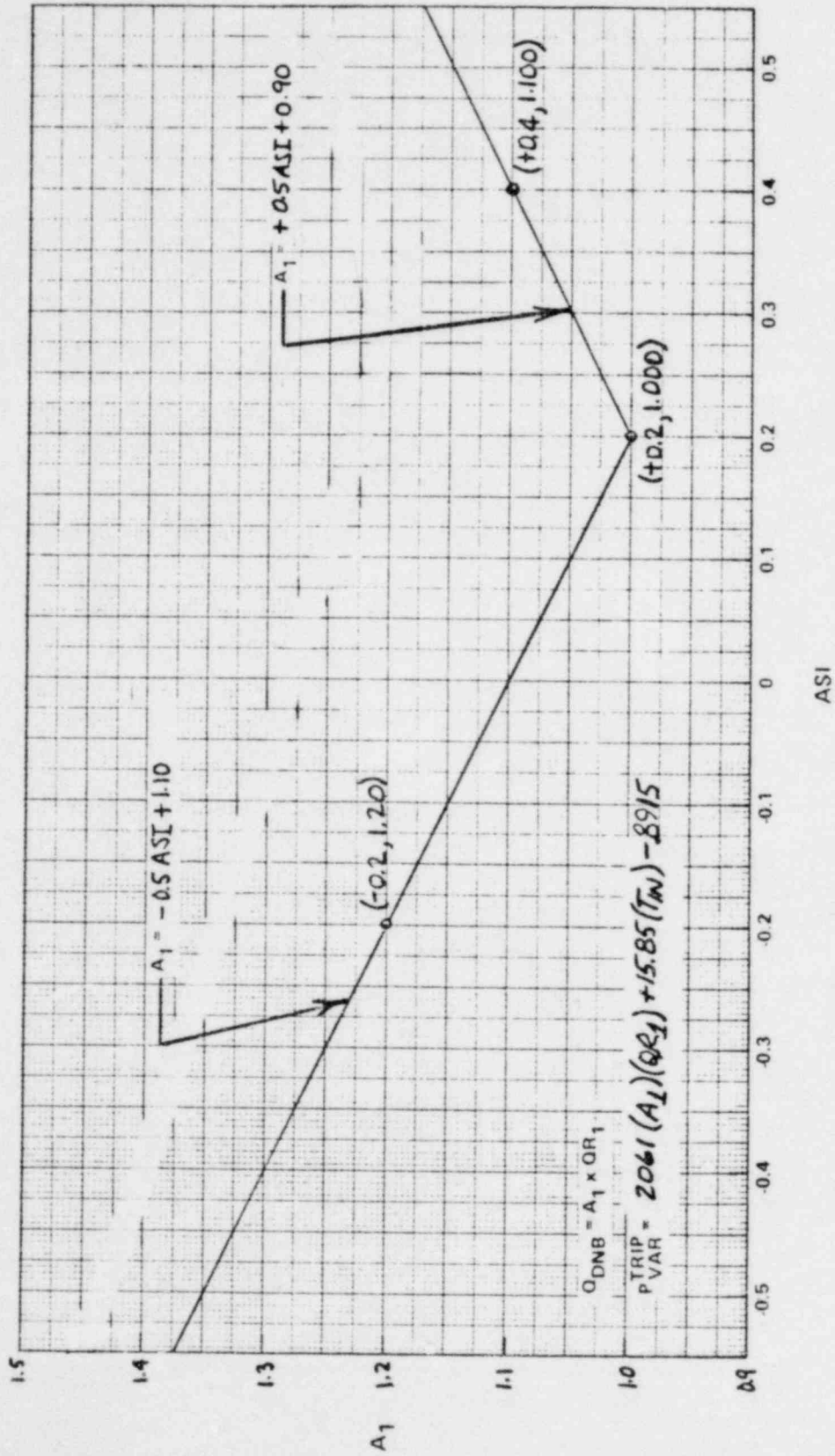


FIGURE 2.2.2
Thermal Margin/Low Pressure Trip Setpoint
Part 1 (ASI Versus A_1)

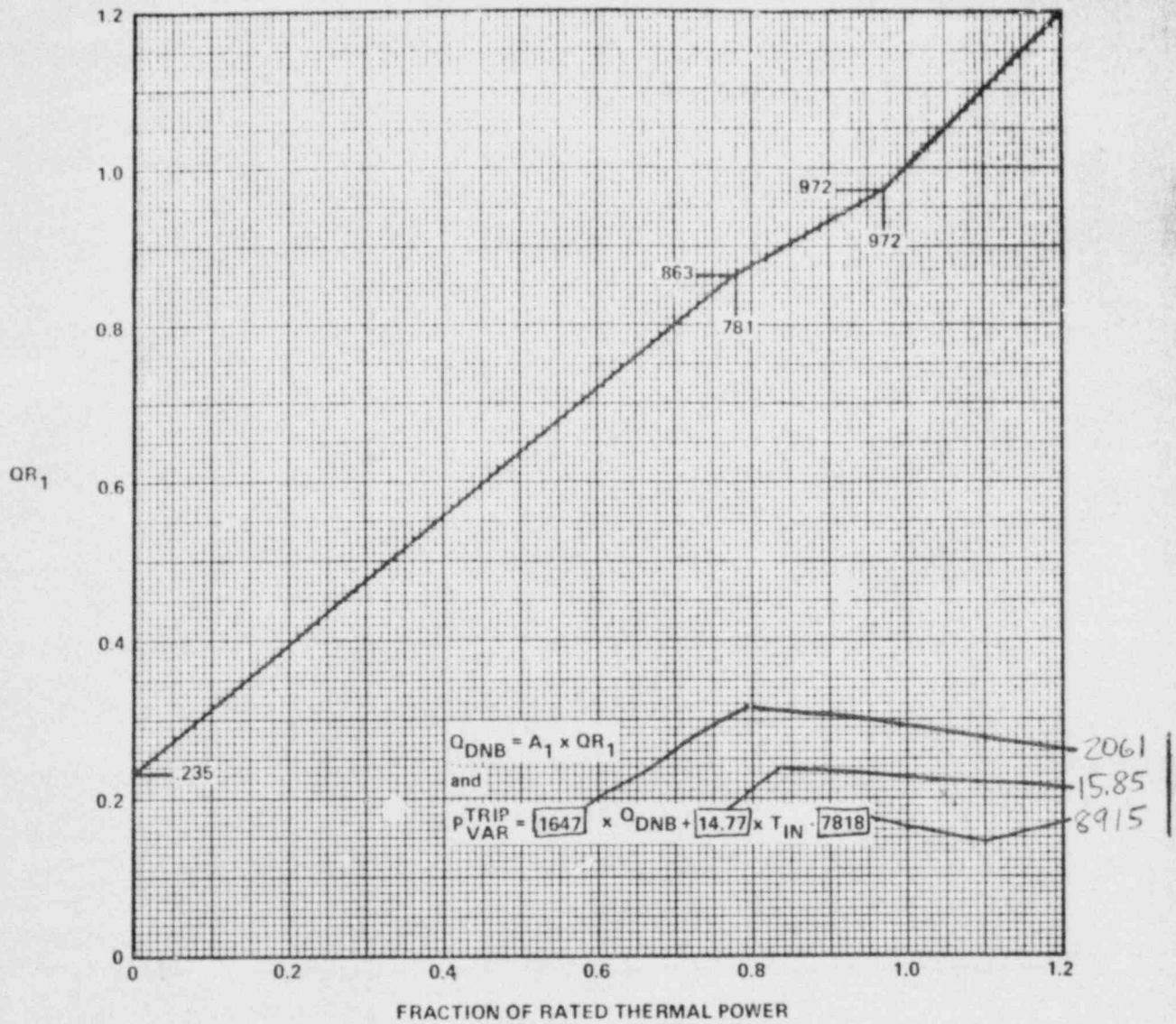


FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint
 Part 2 (Fraction of RATED THERMAL POWER versus QR_1)

SAFETY LIMITSBASES

Table 2.1-1. The area of safe operation is below and to the left of these lines. ~~For both 2 pump configurations, the limiting condition is void fraction rather than DNBR. The void fraction limits assure stable flow and maintenance of DNBR greater than 1.19.~~

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 1.19 and preclude the existence of flow instabilities.

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 1, 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.

the difference between the trip setpoint and the

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 increases. The trip setpoint has a maximum value of 106.5% 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.

Maximum
Low
Power

high
power

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

LIMITING SAFETY SYSTEM SETTINGSBASESSteam 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 ~~design~~ pressure of the reactor coolant system will not be exceeded. 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.

safety
limit

Axial Flux Offset

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 1.19 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

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.19, ~~or when a void fraction limit is exceeded which could result in local flow instability.~~

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 SETTINGSBASES

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 52 psia to compensate for pressure measurement error, trip system 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 52 psia allowance is made up of a 22 psia pressure measurement allowance and a 30 psia time delay allowance.

84

84

62

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 valve 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.

REACTIVITY CONTROL SYSTEMSCEA DROP TIMELIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be ≤ 2.5 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

3.1

- a. $T_{avg} \geq 515^{\circ}\text{F}$, and
- b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

- a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.
- b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

- a. For all CEAs following each removal of the reactor vessel head,
- b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and
- c. At least once per 18 months.

POWER DISTRIBUTION LIMITSSURVEILLANCE 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_{xy} curve of Figure 3.2-3.

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 $\boxed{1.058}^* | 1.070$
 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.

** An uncertainty factor of 1.10 applies when in LOAD FOLLOW OPERATION*

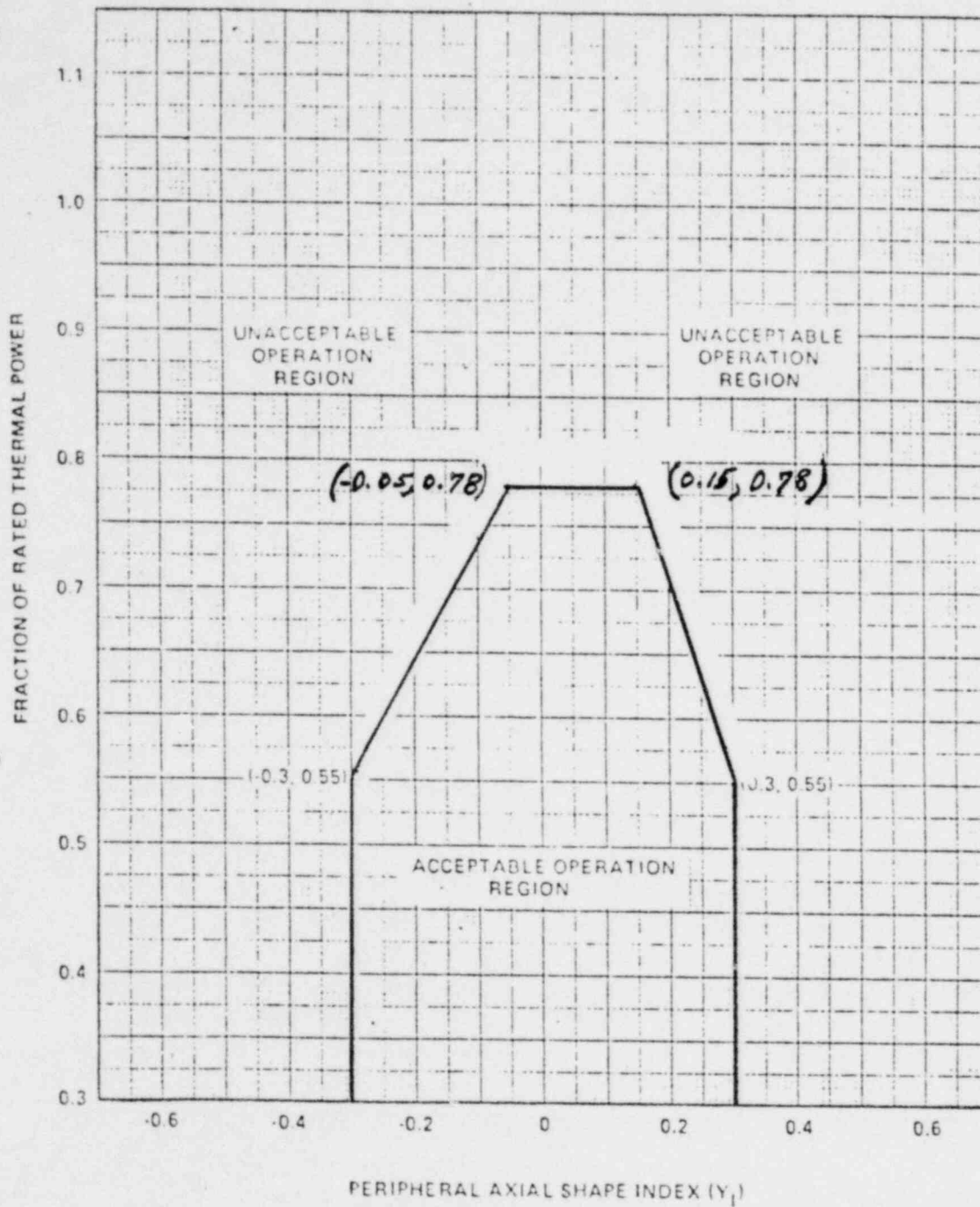
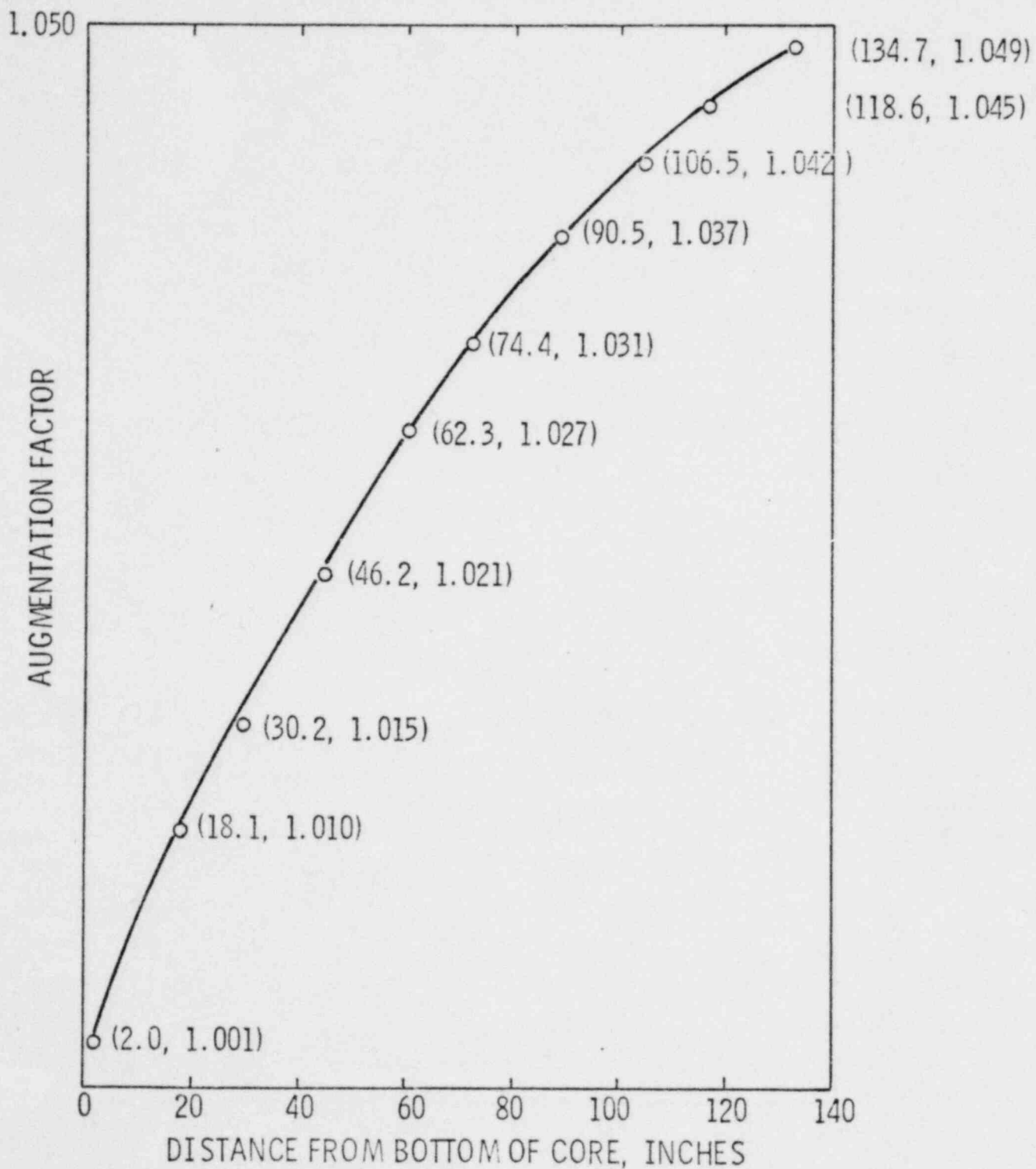


FIGURE 3.2.2
Linear Heat Rate
Axial Flux Offset Control Limits



BALTIMORE
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Calvert Cliffs
Nuclear Power Plant

AUGMENTATION FACTORS vs DISTANCE
FROM BOTTOM OF CORE

Figure
4.2-1

POWER DISTRIBUTION LIMITSTOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F_{xy}^T , defined as $F_{xy}^T = F_{xy}(1+T_q)$, shall be limited to ≤ 1.660 .

APPLICABILITY: MODE 1*.

ACTION:

With $F_{xy}^T > 1.660$, within 6 hours either:

- 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
- Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

- Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
- At least once per 31 days of accumulated operation in MODE 1, and
- Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030 .

*See Special Test Exception 3.10.2.

when in non-LOAD FOLLOW OPERATION and by the expression $F_{xy} = 1.03 F_{xy}(1 + T_q)$ when in LOAD FOLLOW OPERATION.

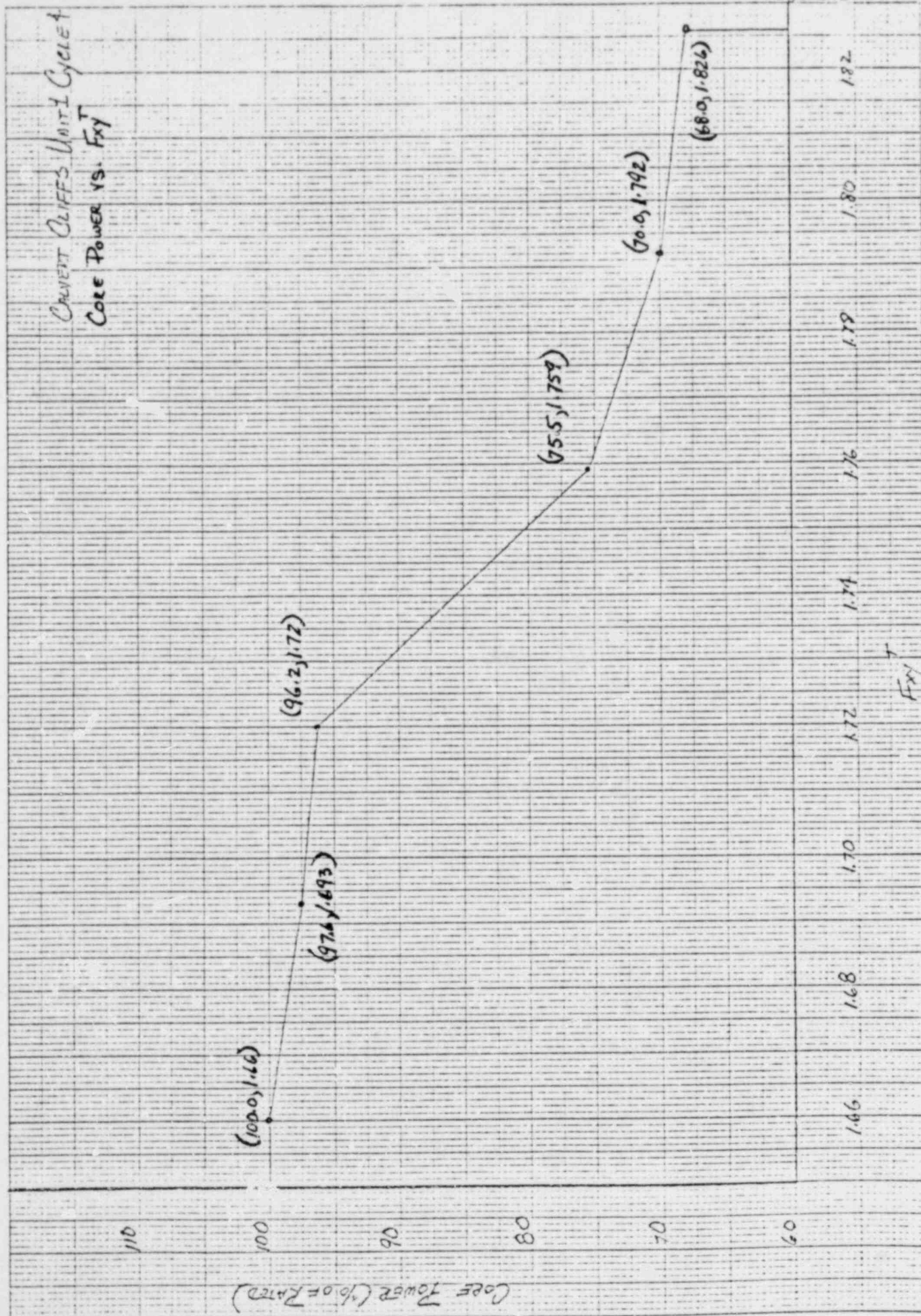


Figure 3.2-3a

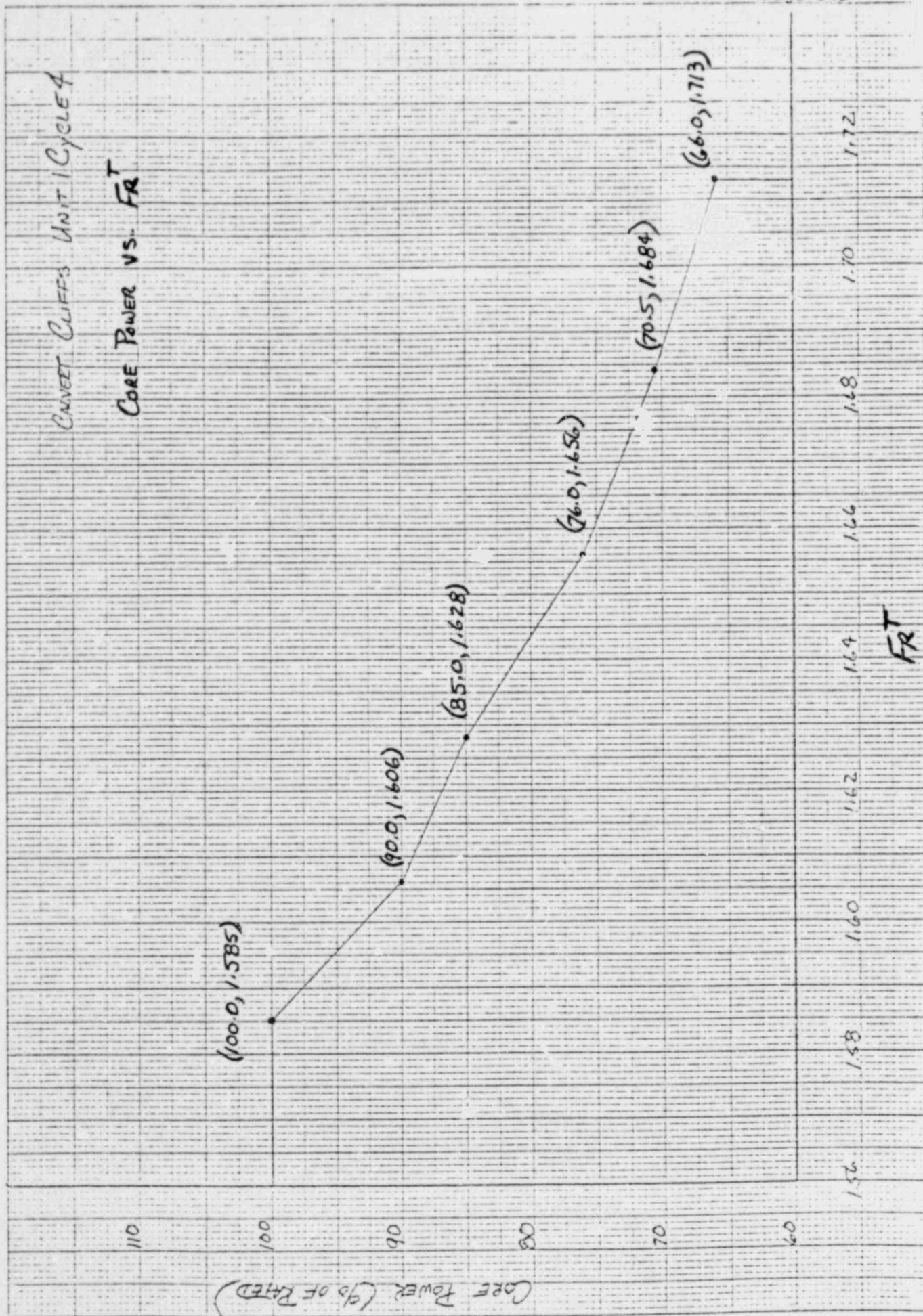


Figure 3.2-36

POWER DISTRIBUTION LIMITSTOTAL 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 ≤ 1.650 .

1.571

APPLICABILITY: MODE 1*.

ACTION:

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

1.571

- Be in at least HOT STANDBY, or
- Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^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. The THERMAL POWER limit determined from Figure 3.2-3 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) 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:

- Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
- At least once per 31 days of accumulated operation in MODE 1, and
- Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030 .

when in non-LOAD FOLLOW OPERATION, and
by the expression $F_r^T = 1.02 F_r (1 + T_q)$
when in LOAD FOLLOW OPERATION.

*See Special Test Exception 3.10.2.

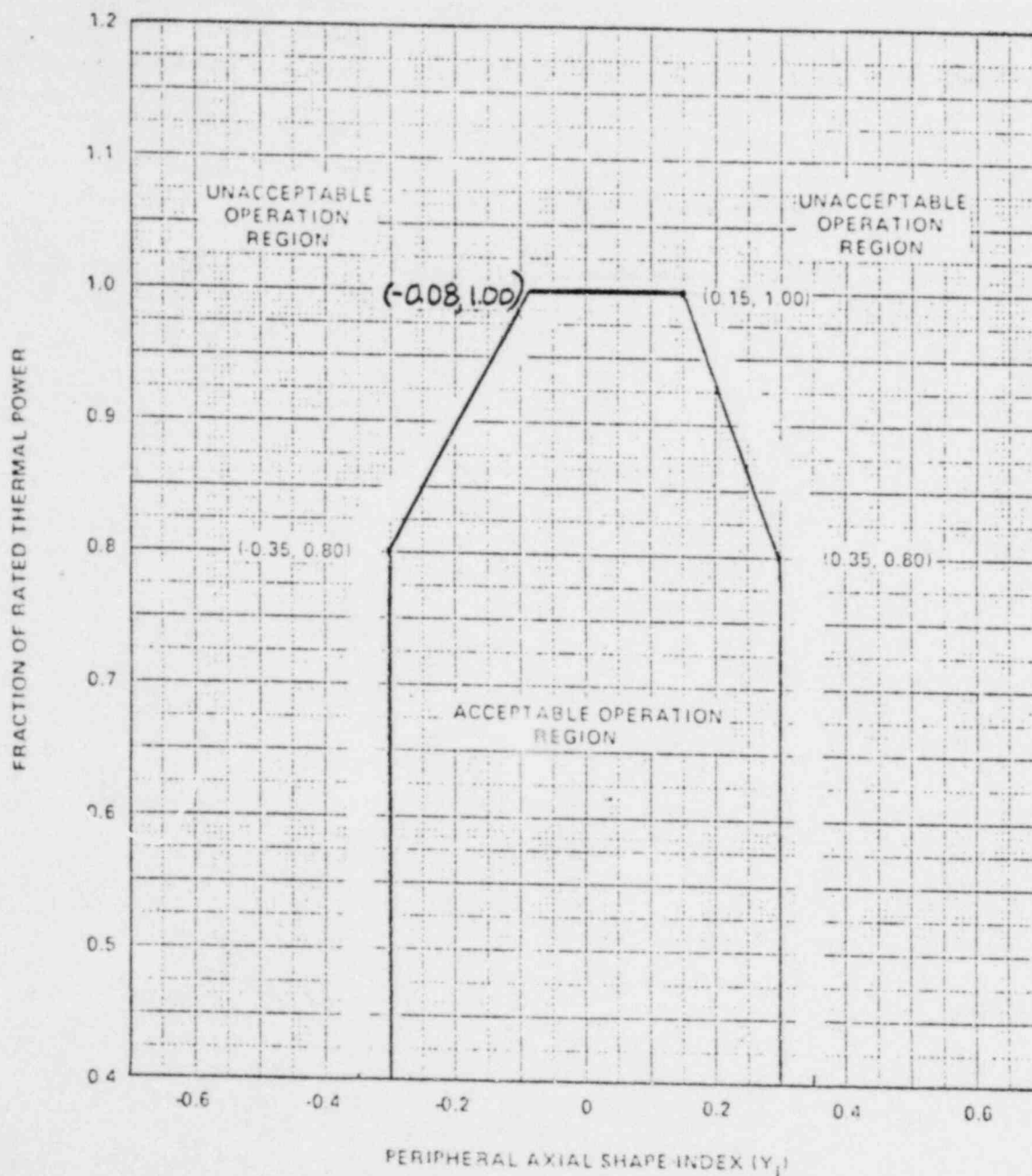


FIGURE 3.2-4
DNB Axial Flux Offset Control Limits

POWER DISTRIBUTION LIMITSFUEL RESIDENCE TIMELIMITING CONDITION FOR OPERATION

3.2.5 The additional core average fuel burnup accumulated during fuel cycle 3 shall be limited to ≤ 310 Effective Full Power Days.

APPLICABILITY: MODE 1.

ACTION:

With the additional core average fuel burnup accumulated during fuel cycle 3 determined to exceed 310 Effective Full Power Days, be in at least HOT STANDBY within the next 6 hours.

DELETE

SURVEILLANCE REQUIREMENTS

4.2.5 The core average fuel burnup, based on gross thermal energy generation, shall be determined by calculation at least once per 31 days.

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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	$\leq 541^{\circ}\text{F}$ ⁸	** 547°F	** 547°F	** 547°F
Pressurizer Pressure	≥ 2225 psia*	** 2225 psia*	** 2225 psia*	** 2225 psia*
Reactor Coolant System Total Flow Rate	$\geq 370,000$ gpm	** Figure 3.2-1	** Figure 3.2-1	** Figure 3.2-1
AXIAL SHAPE INDEX	Figure 3.2-4	** Figure 3.2-1	** Figure 3.2-1	** Figure 3.2-1

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 1% 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.

CALVERT CLIFFS - UNIT 1

3/4 2-15

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excure Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excure Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excure neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excure monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.10×3 , 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02. 1.070

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F_{xy}^T AND F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_{xy}^T and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F_r^T and T_q are provided to ensure that the assumptions used in

* An uncertainty factor of 1.10 applies when in LOAD FOLLOW OPERATION

POWER DISTRIBUTION LIMITS

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.

3/4.2.4 FUEL RESIDENCE TIME

~~The limitation on fuel burnup during the third fuel cycle insures that fuel cladding collapse will not occur. Performance data from similar fuel rods and analyses of the installed fuel rods show that cladding collapse will not occur in the limiting batch until well beyond the proposed third cycle of operation. However, operation beyond the specified third cycle fuel burnup limitation will require further analyses.~~

DELETE

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 1.19 throughout each analyzed transient.

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.

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 ≤ 5.0 seconds## 8.0
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 ≤ 5.0 seconds## 8.0
9. Thermal Margin/Low Pressure	≤ 0.90 seconds*# and ≤ 5.0 seconds## 8.0
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 portion 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.

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10.0 Startup Testing

The following discussions represent the major startup tests proposed for Calvert Cliffs 1, Cycle 4. Sufficient data is obtained to verify that the plant operates in a safe condition within the bounds of the applicable acceptance criteria and therefore, the safety analysis.

Hot Functional Testing

CEDM Performance Testing

During this testing, the proper functioning of the CEAs, CEDMs, and CEA position indication will be verified through the insertion and withdrawal of the CEAs. Rod drop times will be measured and evaluated. Any irregularities shall be analyzed.

RCS Flow Verification

RCS flow rates will be verified based on differential pressure measurements obtained across the RCPs and RV. These values will be compared to those obtained during previous testing for consistency.

Initial Criticality

Approach to criticality will commence with the withdrawal of the Shutdown CEA Groups, followed by the withdrawal, in sequence, of the Regulating CEA Groups resulting with Group 5 at mid-core. Criticality will be established through boron dilution. The plant will be allowed to stabilize following Critical Boron Concentration, and then proceed to the Low Power Physics Tests to verify physics design parameters.

Low Power Physics Testing

CEA Symmetry Check

CEAs will be inserted into the core, and withdrawn from the core to confirm proper latching to their respective CEA extension shafts. A qualitative reactivity change will be apparent for single CEAs; and a quantitative reactivity, for dual CEAs.

Critical Boron Concentration

Critical Boron Concentrations will be determined for ARO, and Groups 1 through 5 inserted.

Isothermal Temperature Coefficient

By varying the RCS temperature, the Isothermal Temperature Coefficient will be determined. CEA Regulating Group 5 will be used to control and maintain flux and reactivity within a defined operating band.

CEA Group Worth Measurements

The RCS will be diluted/borated while the CEAs are inserted/withdrawn to compensate for a change in reactivity. These changes will be monitored via the reactivity computer. Non-overlapped worths will be determined.

Power Ascension Test

Two major plateaus for testing will be the 50% plateau and 100% plateau. The following specific tests will be performed as shown in order to compare and verify as-built characteristics of the core with their respective predictions. In addition to the 50% and 100% plateaus, the core power distributions will be determined and verified with predictions.

50% and 100% Plateau Testing

Upon reaching the 50% power level, Xenon Equilibrium will be established with all rods out. Preparation for the Variable T_{avg} test will commence by diluting CEA Group 5 to approximately 105 inches. Following Xenon Equilibrium, T_c will be varied, thereby yielding data for the isothermal temperature coefficient determination. The Power Coefficient will be determined by maintaining T_{avg} constant and varying the power level. In both cases, CEA 5-1 will be used for reactivity control and maintaining power.

Acceptance Criteria

Acceptance criteria for the above startup testing will be developed consistent with those presented during previous startups.

Acceptance Limits:

CEA Groups Worth $\pm 15\%$ on each group

$\pm 10\%$ on sum of all groups measured.

Critical Boron Measurements $\pm 10\%$

Temperature Coefficient $\pm .3 \times 10^{-4} \quad \Delta \rho / \Delta F$

Power Coefficient $\pm .2 \times 10^{-4} \quad \Delta \rho / \%$

Rod Drops ≤ 3.1 seconds

Power Distribution $F_r T, F_{xy} T, \text{ and } T_q$ within Technical Specification Limits.

If any acceptance criteria limits listed above are exceeded, an evaluation shall be made to determine first, the applicability of the prediction to the precise plant conditions under which the test was performed; second, the accuracy of the measurement; finally, the validity of the physics data input to the safety analysis for the entire cycle. Specifically, if any regulating bank worth measurement falls outside of its acceptance criteria or if the total worth of the regulating banks falls outside of its acceptance criteria, shutdown bank C shall be measured and compared with its acceptance criteria. If shutdown Bank C worth fails outside of its acceptance criteria or if the accumulated total worth of all the banks measured falls below their total worth acceptance criterion (after appropriate corrections and adjustments) then an evaluation shall be made of the validity of the safety analyses for the entire cycle.

A summary report of the results of these tests will be submitted to the NRC within 45 Days of completion of the startup program.

11. REFERENCESA. Chapters 1 through 6

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4. Letter, A. E. Lundvall, Jr. to B. C. Rusche, "Second Cycle License Application", dated October 1, 1976
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E) Letter from A. E. Lundvall to R. W. Reid, March 29, 1978.
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