ENCLOSURE A

Calvert Cliffs Unit II Cycle 4

Refueling License Amendment

CALVERT CLIFFS UNIT II CYCLE 4

REFUELING LICENSE AMENDMENT

Table of Contents

1. INTRODUCTION AND SUMMARY

This report provides an evaluation of design and performance for the operation of Calvert Cliffs Unit II 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 D and E assemblies and fresh Batch F assemblies.

Plant operating requirements have created a need for flexibility in the Cycle 3 termination point, ranging from 10,000 MWD/T to 11,000 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, provided the Cycle 3 termination point falls within the above burnup range.

The evaluations of the reload core characteristics have been examined with respect to the Calvert Cliffs Unit I Cycle 5 safety analysis described in References 1 and 2, hereafter referred to as the "reference cycle" in this report unless otherwise noted. 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 the revised analyses presented herein continue to show acceptable results. Where dictated by variations from Cycle 3, proposed modifications to the plant Technical Specifications are provided.

2. OPERATING HISTORY OF CALVERT CLIFFS II CYCLE 3

Calvert CLiffs Unit II is presently operating in its third fuel cycle utilizing Batch, B, C, D and E fuel assemblies. Calvert Cliffs Unit II Cycle 3 began operation on December 6, 1979 and reached full power on December 12. The Cycle 3 startup testing was reported to the NRC in Reference 3.

Cycle 3 is presently scheduled to terminate on about January 2, 1981 with a cycle burnup of approximately 10,800 MWD/T. However, flexibility in this endpoint burnup is necessary because of uncertainties in the Unit capacity factor during the remainder of Cycle 3. The Cycle 3 termination point can vary between 10,000 MWD/T and 11,000 MWD/T to accommodate the plant schedule and still be within the assumptions of the Cycle 4 analyses. As of mid-November 1980, the Cycle 3 burnup had reached 9850 MWD/T.

Initial criticality of Cycle 4 is expected to occur on or about February 12, 1981.

3. GENERAL DESCRIPTION

The Cycle 4 core will consist of the number and types of assemblies and fuel batches as described in Table 3-1. The primary change to the core in Cycle 4 is the removal of 1 Batch B assembly, 68 Batch C assemblies, and 59 Batch D assemblies. These assemblies will be replaced by 40 Batch F (3.65 w/o enrichment) and 88 Batch F/ (3.03 w/o enrichment) assemblies. The 88 low enrichment Batch F/ assemblies contain 8 burnable poison pins per assembly. Figure 3-1 shows the fuel management pattern to be employed in Cycle 4. Figure 3-2 shows the locations of the fuel and poison pins within the lattice of the Batch F/ assemblies and the fuel pin locations in the unshimmed Batch F assemblies. This pattern will accommodate Cycle 3 termination burnups from 10,000 MWD/T to 11,000 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.

Figure 3-3 shows the beginning of Cycle 4 assembly burnup distribution for a Cycle 3 termination burnup of 10,500 MWD/T. The initial enrichment of the fuel assemblies is also shown in Figure 3-3.

TABLE 3-1

CALVERT CLIFFS UNIT II CYCLE 4 CORE LOADING

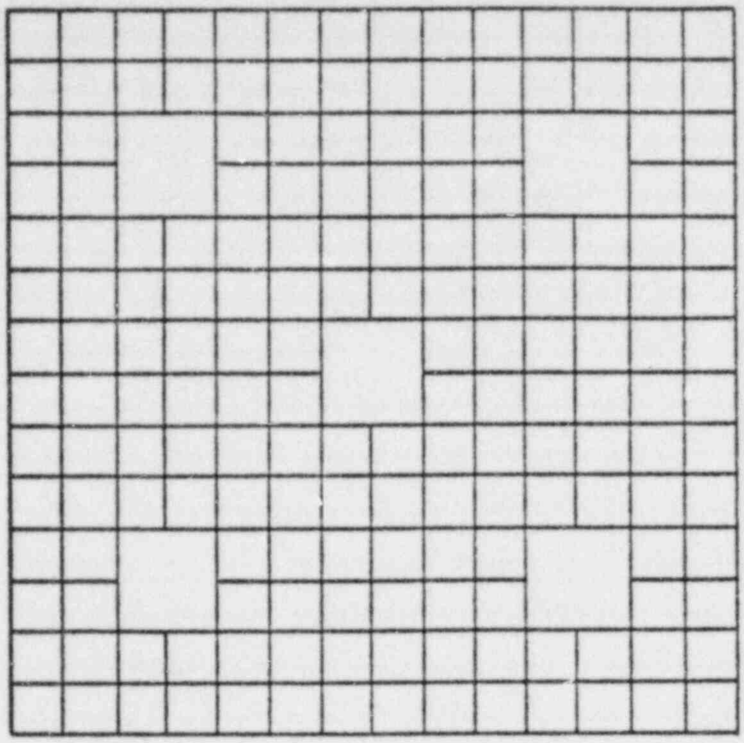
Assembly Designation	Number of Assemblies	Initial Enrichment Wt% U-235	Batch Average Burnup EOC3 = 10,500	Poison Rods per Assembly	Initial Poison Loading Wt% B4C	Total Number Poison Rods	Total Number Fuel Rods
D	25	3.03	20,300	0	0	0	4400
E	48	3.03	9000	0	0	0	8448
E/	16	2.73	12,000	0	0	0	2816
F	40	3.65	0	0	0	0	7040
F/	88	3.03	0	8	3.03	704	14,784
TOTALS	217					704	37,488

Note: Shim B^{10} concentration equals .02685 gms B^{10}/inch

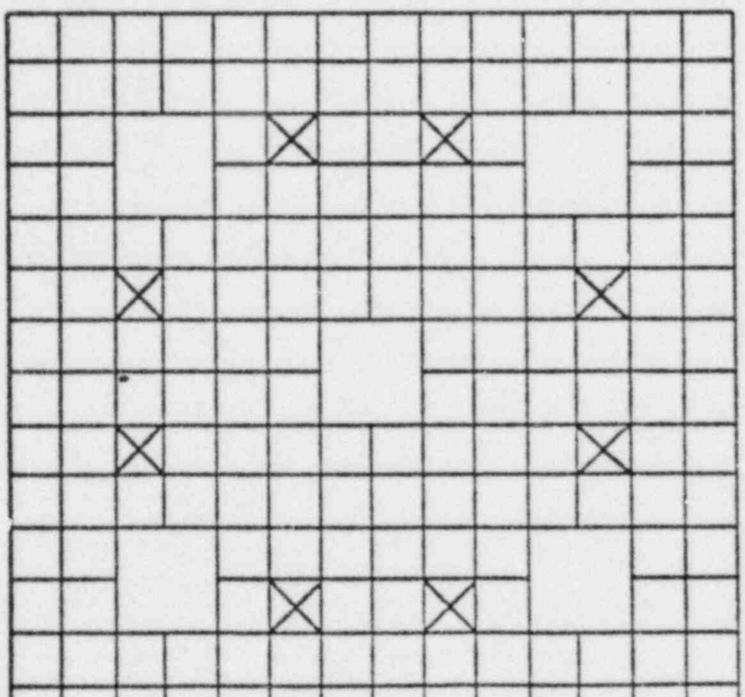
							F	F
				F	F	F/	E	F/
			F	F/	E	F/	E	E/
		F	F/	E	F/	D	F/	E/
	F	F/	E	F/	D	F/	E	F/
	F	E	F/	D	F/	E	F/	E/
	F/	F/	D	F/	E	F/	D	F/
F	E	E	F/	E	F/	D	F/	E/
F	F/	E/	E/	F/	E/	F/	E/	D

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT II CYCLE 4 CORE MAP	Figure 3-1

UNSHIMMED ASSEMBLY



8 POISON ROD ASSEMBLY



☐ FUEL ROD LOCATION

☒ POISON ROD LOCATION

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT II CYCLE 4 ASSEMBLY FUEL AND OTHER ROD LOCATIONS	Figure 3-2

INITIAL ENRICHMENT, w/o U-235
BOC4 BURNUP (MWD/T) EOC3 = 10,500 MWD/T

							3.65 0	3.65 0
				3.65 0	3.65 0	3.03 0	3.03 8,400	3.03 0
			3.65 0	3.03 0	3.03 7,600	3.03 0	3.03 9,000	2.73 12,000
		3.65 0	3.03 0	3.03 7,300	3.03 0	3.03 20,200	3.03 0	2.73 12,000
	3.65 0	3.03 0	3.03 7,300	3.03 0	3.03 20,100	3.03 0	3.03 11,600	3.03 0
	3.65 0	3.03 7,600	3.03 0	3.03 20,100	3.03 0	3.03 10,100	3.03 0	2.73 12,400
3.65 0	3.03 0	3.03 0	3.03 20,200	3.03 0	3.03 10,100	3.03 0	3.03 20,300	3.03 0
3.65 0	3.03 8,400	3.03 9,000	3.03 0	3.03 11,600	3.03 0	3.03 20,300	3.03 0	2.73 11,800
	3.03 0	2.73 12,000	2.73 12,000	3.03 0	2.73 12,400	3.03 0	2.73 11,800	3.03 21,800

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

4.0 FUEL SYSTEM DESIGN

4.1 MECHANICAL DESIGN

The mechanical design for the standard Batch F reload fuel is identical to that of the standard Batch E fuel used in Calvert Cliffs 2 (Reference 9) and of the Calvert Cliffs 1 standard Batch G fuel described in the reference cycle submittal (Reference 1).

Details of the standard Batch D fuel design parameters can be found in Reference 4.

C-E has performed analytical predictions of cladding creep-collapse time for all Calvert Cliffs Unit 2 fuel batches that will be irradiated in Cycle 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 (Table 4-1). 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.

Table 4-1

Batch	Minimum Collapse Time	EOC4 Exposure
D	>29,500 Hours	28,433 Hours
E	>22,200 Hours	21,236 Hours
F	>22,200 Hours	12,965 Hours

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 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.

4.2 HARDWARE MODIFICATIONS TO MITIGATE GUIDE TUBE WEAR

All standard fuel assemblies which will be placed in CEA locations in Cycle 4 will have stainless steel sleeves installed in the guide tubes to prevent guide tube wear. A detailed discussion of the design of the sleeves and their effect on reactor operation is contained in Reference 6.

4.3. THERMAL DESIGN

Using the FATES fuel evaluation model (Reference 7), the thermal performance of the various fuel assemblies (fuel Batches D, E, and F) has been evaluated with respect to prior burnup, the proposed burnup during Cycle 4, their respective fuel characteristics, and expected flux level during Cycle 4. The fresh fuel, Batch F, has been determined to be the limiting fuel batch with respect to stored energy.

5.0 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, Figure 3-1. 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 40 assemblies with an enrichment of 3.65 wt% U-235, 88 assemblies with an enrichment of 3.03 wt% U-235 and 8 poison shims per assembly. With this loading, the Cycle 4 burnup capacity for full power operation is expected to be between 17,100 MWD/T and 17,600 MWD/T, depending on the final Cycle 3 termination point. The Cycle 4 core characteristics have been examined for Cycle 3 terminations between 10,000 and 11,000 MWD/T and limiting values established for the safety analyses. The loading pattern (see Section 3) 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. Please note that the values of parameters actually employed in safety analyses are different from those displayed in Table 5-1 and are typically chosen to conservatively bound predicted values with accommodation for appropriate uncertainties and allowances.

Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for the end of Cycle 4 zero power steam line break accident with a comparison to reference cycle data. The EOC zero power steam line break was selected since it is the most limiting zero power steam line break accident, and thus provides the basis for establishing the Technical Specification shutdown worth.

Table 5-3 shows the reactivity worths of various CEA groups calculated at full power conditions for Cycle 4 and the reference cycle.

5.1.2 Power Distribution

Figures 5-1 through 5-3 illustrate the all rods out (ARO) planar radial power distributions at BOC4, MOC4 and EOC4 that are characteristic of the high burnup end of the Cycle 3 shutdown window. These planar radial power peaks are characteristic of the major portion of the active core length between about 20 and 80 percent of the fuel height. The higher burnup end of Cycle 3 shutdown window tends to increase the power peaking in this axial central region of the core for Cycle 4.

The planar radial power distributions for the above region with CEA Group 5 fully inserted at beginning and end of Cycle 4 are shown in Figures 5-4 and 5-5, respectively, for the high burnup end of the Cycle 3 shutdown window. The maximum planar radial pin peak of 1.48 occurs at beginning of cycle and decreases over the cycle. It is characteristic of both ARO and Bank 5 inserted conditions that the Cycle 4 peaks are highest near BOC.

The radial power distributions described in this sction 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. For both DNB and kw/ft safety and setpoint analyses in either rodded or unrodded configurations, the power peaking values actually used are higher than those expected to occur at any time during Cycle 4. These conservative values, which are used in Section 7 of this document, establish the allowable limits for power peaking to be observed during operation.

The range of allowable axial peaking is defined by the limiting conditions for operation covering axial shape index (ASI). Within these ASI limits, the necessary DNBR and kw/ft margins are maintained for a wide range of possible axial shapes. The maximum three-dimensional or total peaking factor anticipated in Cycle 4 during normal base load, all rods out operation at full power is 1.85, not including uncertainty allowances and augmentation factors.

5.1.3 Safety Related Data

The safety related data for Cycle 4 is identical to the safety related data used in the reference cycle analysis as presented in Section 5.1.3 of Reference 1.

5.2 ANALYTICAL INPUT TO IN-CORE MEASUREMENTS

In-core detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in the manner described in Reference 8, which is the same method used for the reference cycle.

5.3 NUCLEAR DESIGN METHODOLOGY

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

5.4 UNCERTAINTIES IN MEASURED POWER DISTRIBUTIONS

The power distribution measurement uncertainties which are applied to Cycle 4 are the same as those applied to the reference cycle (Reference 1).

TABLE 5-1

CALVERT Cliffs UNIT II CYCLE 4
NOMINAL PHYSICS CHARACTERISTICS

	UNITS	CYCLE 4	REFERENCE CYCLE
Dissolved Boron			
Dissolved Boron Content for Criticality, CEAs Withdrawn			
Hot Full Power, Equilibrium Xenon, BOC	PPM	1150	1010
Boron Worth			
Hot Full Power BOC	PPM/% $\Delta\rho$	105	101
Hot Full Power EOC	PPM/% $\Delta\rho$	83	83
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients, Hot Full Power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta\rho/°F$	0.0	-0.1
End of Cycle	$10^{-4}\Delta\rho/°F$	-1.9	-1.9
Doppler Coefficient			
Hot Zero Power BOC	$10^{-5}\Delta\rho/°F$	-1.55	-1.55
Hot Full Power BOC	$10^{-5}\Delta\rho/°F$	-1.16	-1.21
Hot Full Power EOC	$10^{-5}\Delta\rho/°F$	-1.40	-1.40
Total Delayed Neutron Fraction, βeff			
BOC		.00662	.00628
EOC		.00517	.00521
Neutron Generation Time, ℓ^*			
BOC	10^{-6}sec	23.8	24.4
EOC	10^{-6}sec	29.8	29.7

TABLE 5-2

Calvert Cliffs Unit II Cycle 4 Limiting Values of Reactivity Worths and Allowances for Hot Zero Power Steam Line Break, %$\Delta\rho$ End-of-Cycle (EOC)

	Reference Cycle	Cycle 4
1. Worth of All CEA's Inserted	9.4	9.2
2. Stuck CEA Allowance	2.2	2.0
3. Worth of All CEA's Less Highest Worth CEA Stuck Out	7.2	7.2
4. Zero Power Dependent Insertion Limit CEA Bite	2.0	2.2
5. Calculated Scram Worth	5.2	5.0
6. Physics Uncertainty (10% of Item 5)	.5	.5
7. Net Available Scram Worth (Item 5 minus Item 6)	4.7	4.5
8. Technical Specification Shutdown Worth	4.3	4.3
9. Margin in Excess of Technical Specification Shutdown Worth	+0.4	+0.2

TABLE 5-3

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

	Beginning of Cycle		End of Cycle	
Regulating CEAs	Cycle 4	Reference Cycle	Cycle 4	Reference Cycle
Group 5	0.46	0.49	0.53	0.57
Group 4	0.28	0.32	0.41	0.39
Group 3	0.89	0.97	1.01	0.93

Note:

Values shown assume sequential group insertion.

						0.72	0.98 x	
				0.70	0.98	0.97	1.10	1.20
			0.77	1.04	1.14	1.18	1.14	1.04
		0.77	1.05	1.15	1.16	0.92	1.16	1.02
	0.70	1.04	1.15	1.15	0.90	1.12	1.05	1.14
	0.98	1.14	1.16	0.90	1.10	1.04	1.06	0.89
0.72	0.97	1.18	0.92	1.12	1.04	1.02	0.77	0.93
0.98	1.10	1.14	1.16	1.05	1.06	0.77	0.88	0.74
	1.20	1.04	1.02	1.14	0.89	0.93	0.74	0.63

NOTE: x = MAXIMUM 1 - PIN PEAK = 1.48

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS II CYCLE 4 ASSEMBLY RELATIVE POWER DENSITY AT BOC, EQUILIBRIUM XENON	Figure 5-1

							0.62	0.80
				0.65	0.87	0.92	0.93	1.07
			0.74	1.02	1.03	1.15	1.02	0.92
		0.74	1.05	1.08	1.19	0.91	1.19	0.97
	0.65	1.02	1.08	1.21	0.93	1.22	1.07	1.23
	0.87	1.03	1.19	0.93	1.22 x	1.10	1.23	0.99
0.62	0.92	1.15	0.91	1.22	1.10	1.21	0.91	1.16
	0.93	1.02	1.19	1.07	1.23	0.91	1.13	0.91
	1.07	0.92	0.97	1.23	0.99	1.16	0.91	0.80

NOTE: x = MAXIMUM 1 - PIN PEAK = 1.40

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS II CYCLE 4 ASSEMBLY RELATIVE POWER DENSITY AT 8 GWD/T, EQUILIBRIUM XENON	Figure 5-2

							0.68	0.84
				0.70	0.90	0.99	0.95	1.10
			0.79	1.06	1.02	1.16	1.00	0.92
		0.79	1.08	1.05	1.18	0.91	1.16	0.94
	0.70	1.06	1.05	1.18 x	0.92	1.17	1.01	1.18
	0.90	1.02	1.18	0.92	1.18	1.03	1.18	0.96
	0.99	1.16	0.91	1.17	1.03	1.18	0.91	1.17
0.68	0.95	1.00	1.16	1.01	1.18	0.91	1.17	0.94
0.84	1.10	0.92	0.94	1.18	0.96	1.17	0.94	0.84

NOTE: x = MAXIMUM 1 - PIN PEAK = 1.32

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS II CYCLE 4 ASSEMBLY RELATIVE POWER DENSITY AT EOC EQUILIBRIUM XENON	Figure 5-3

CEA BANK 5 LOCATIONS

							0.71	0.96
				0.70	0.98	0.98	1.06	1.12
			0.66	1.01	1.13	1.19	1.08	0.79
		0.66	0.77	1.07	1.18	0.95	1.17	0.99
	0.70	1.01	1.07	1.15	0.94	1.19	1.11	1.21 x
	0.98	1.13	1.18	0.94	1.19	1.12	1.16	0.99
	0.98	1.19	0.95	1.19	1.12	1.12	0.83	0.99
0.71	1.06	1.08	1.17	1.11	1.16	0.83	0.90	0.70
0.96	1.12	0.79	0.99	1.21	0.99	0.99	0.70	0.38

NOTE: x = MAXIMUM 1 - PIN PEAK = 1.45

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

CEA BANK 5 LOCATIONS

						0.69	0.85	
				0.70	0.92	1.01	0.95	1.05
			0.66	1.02	1.04	1.19	0.97	0.70
		0.66	0.75	0.98	1.19	0.95	1.17	0.93
	0.70	1.02	0.98	1.16	0.95	1.24	1.07	1.23
	0.92	1.04	1.19	0.95	1.24	1.11	1.26 x	1.03
0.69	1.01	1.19	0.95	1.24	1.11	1.25	0.96	1.20
0.85	0.95	0.97	1.17	1.07	1.26	0.96	1.13	0.86
	1.05	0.70	0.93	1.23	1.03	1.20	0.86	0.47

NOTE: x = MAXIMUM 1 - PIN PEAK = 1.40

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

6.0 Thermal Hydraulic Design

6.1 DNBR Analysis

The thermal hydraulic models and pertinent design parameters used for Calvert Cliffs II Cycle 4 are the same as those used in the reference cycle as reported in Reference 2 and corrected in Reference 10.

6.2 Effects of Fuel Rod Bowing on DNBR Margin

The fuel rod bowing effects on DNB margin for Calvert Cliffs Unit II have been evaluated within the guidelines set forth in Reference 11.

A total of 89 fuel assemblies will exceed the NRC-specified DNB penalty threshold burnup of 24,000 MWD/T, as established in Reference 11, during Cycle 4. At the end of Cycle 4, the maximum burnup attained by any of these assemblies will be 37,100 MWD/T. From Reference 11, the corresponding DNB penalty for 37,100 MWD/T is 4.4 percent.

An examination of power distributions for Cycle 4 shows that there exists at least 6.0 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 11 reduction penalty of 4.4 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.

7.0 TRANSIENT ANALYSIS

The purpose of this section is to present the results of the Baltimore Gas & Electric Calvert Cliffs Unit II, Cycle 4 Non-LOCA safety analysis at 2700 MWt. The Design Bases Events (DBEs) considered in the safety analyses are listed in Table 7-1.

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 transient. Each DBE was reviewed by comparing all the current and reference cycle key transient parameters that significantly impact the results of the event. The reference cycle is one for which a DBE in question has been shown to meet required safety criteria. If all the current cycle values of key parameters for a particular event are bounded by (conservative with respect to, or the same as) the reference cycle, no reanalysis is required.

The reference cycle for this analysis is Calvert Cliffs Unit I, Cycle 5 (Reference 1 as amended by Reference 2).

The results of the review were that the key input parameters to all the DBEs for Unit II Cycle 4 operation are the same as or less limiting than the specified reference cycle input parameters (see Table 7-2) except for the Loss of Flow (LOF) event. The Loss of Flow (LOF) event was reanalyzed to account for the fact that the flow coastdown for Unit II is different from, and more adverse than, the coastdown for Unit I. Therefore as indicated in Table 7-1, only the Loss of Flow transient has been reanalyzed for Unit II Cycle 4.

For all DBEs other than the LOF event, the reference cycle safety analyses bound the results that would be obtained for Cycle 4 and demonstrate safe operation of the Calvert Cliffs Unit II Cycle 4 at 2700 MWt.

TABLE 7-1

CALVERT CLIFFS UNIT II, CYCLE 4
INCIDENTS CONSIDERED IN TRANSIENT AND ACCIDENT ANALYSIS

	Analysis Status
Anticipated Operational Occurrences for which the RPS Assures no Violation of SAFDLs:	
Control Element Assembly Withdrawal	Not 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	Not 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
Transients Resulting from Malfunction of One Steam Generator [2]	Not Reanalyzed
Postulated Accidents:	
CEA Ejection	Not Reanalyzed
Steam Line Rupture	Not Reanalyzed
Steam Generator Tube Rupture	Not Reanalyzed
Seized Rotor	Not Reanalyzed

[1] Requires Low Flow Trip.

[2] Requires Asymmetric Steam Generator Protective Trip Function

TABLE 7-2

CALVERT CLIFFS UNIT II CYCLE 4
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Physics Parameters	Units	Reference Cycle Values Unit I Cycle 5	Unit II Cycle 4 Values
Radial Peaking Factors			
For DNB Margin Analyses (F_r^T)			
Unrodded Region		1.62	1.62
Bank 5 Inserted		1.78	1.78
For Planar Radial Component (F_{xy}^T) of 3-D Peak (CTM Limit Analyses)			
Unrodded Region		1.62	1.62
Bank 5 Inserted		1.78	1.78
Maximum Augmentation Factor		1.055	1.055
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	-2.5*→ +.5	-2.5*→ +.5
Shutdown Margin (Value assumed in Limiting EOC Zero Power SLB)	%$\Delta\rho$	-4.3	-4.3
Tilt Allowance	%	3.0	3.0
Safety Parameters			
Power Level	MWt	2754	2754
Maximum Steady State Core Inlet Temperature	°F	550	550
Minimum Steady State RCS Pressure	psia	2200	2200
Reactor Coolant Flow (550°F, 2200 psia)	10^6 lb/hr	133.9	133.9
Negative Axial Shape Index LCO extreme assumed at Full Power	I_p	-.16	-.16
Maximum CEA Insertion at Full Power	% of Insertion of Bank 5	25	25
Maximum Initial Linear Heat Rate for Transient Other Than LOCA	KW/ft	16.0	16.0
Steady State Linear Heat Rate to Fuel Centerline Melt Assumed in the Safety Analyses	KW/ft	21.0	21.0
CEA Drop Time from Removal of Power to Holding Coils to 90% Insertion	sec	3.1	3.1
Minimum DNBR (CE-1)		1.195	1.195

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

POOR ORIGINAL

7.1 Loss of Coolant Flow Event

The Loss of Coolant Flow event was reanalyzed for Cycle 4 to determine the minimum initial margin that must be maintained by the Limiting Conditions for Operation (LCOs) such that in conjunction with the RPS (low flow trip), the DNBR limit will not be exceeded.

The methods used to analyze this event are the same as those used in the reference cycle analysis.

The 4-pump Loss of Coolant Flow produces a rapid approach to the DNBR limit due to the rapid decrease in the core coolant flow. Protection against exceeding the DNBR limit for this transient is provided by the initial steady state thermal margin which is maintained by adhering to the Technical Specifications LCOs on DNB 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 7.1-1. Table 7.1.-1 also lists the key transient parameters used in the present analysis and compares them with comparable reference cycle values.

Table 7.1-2 presents the NSSS and RPS responses during a four pump loss of flow initiated at a 0.0 shape index. The low flow trip setpoint is reached at 0.90 seconds and the scram rods start dropping into the core at 1.9 seconds. A minimum CE-1 DNBR of 1.195 is reached at 3.22 seconds. Figures 7.1-2 to 7.1-6 present the core power, heat flux, RCS pressure, core coolant temperatures and the DNBR as a function of time.

The analysis shows that a Loss of Flow event mitigated by the action of the Low Flow Trip will ensure that DNBR limit will not be exceeded when the initial conditions are no more severe than those permitted by adherence to the Technical Specification LCO's.

TABLE 7.1-1

KEY PARAMETERS ASSUMED IN THE LOSS OF COOLANT FLOW ANALYSIS

Parameter	Units	Reference Cycle*	Cycle 4
Initial Core Power Level	MWt	2754	2754
Initial Core Inlet Coolant Temperature	°F	550	550
Initial Core Mass Flow Rate	10^6 lbm/hr	133.9	133.9
Reactor Coolant System Pressure	psia	2200	2200
Moderator Temperature Coefficient	$10^{-4}\Delta\rho$/F	+.5	+.5
Doppler Coefficient Multiplier	--	1.00	1.00**
LFT Response Time	sec	.5	.5
CEA Holding Coil Delay	sec	.5	.5
CEA Time to 90% Insertion (Including Holding Coil Delay)	sec	3.1	3.1
CEA Worth at Trip (all rods out)	$10^{-2}\Delta\rho$	-5.60	-5.60
Unrodded Radial Peaking Factor (F_r^T)		1.62	1.62
4-Pump RCS Flow Coastdown		Figure 7.2.1-1 of Reference Cycle (see Reference 2)	Figure 7.1-1

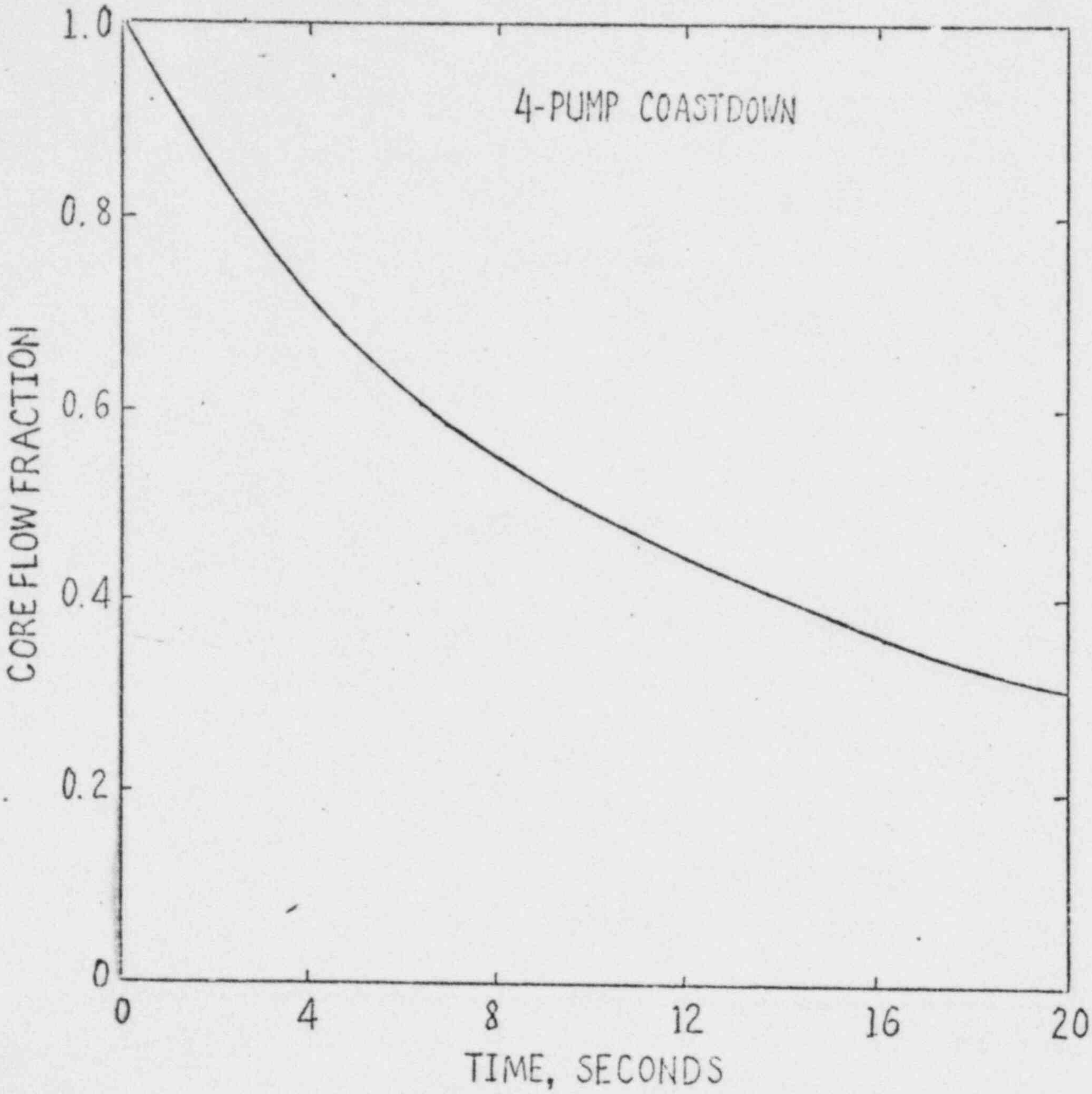
* Unit I Cycle 5

** Since this is a second order effect and the most limiting doppler multiplier varies during the transient, a nominal value is used.

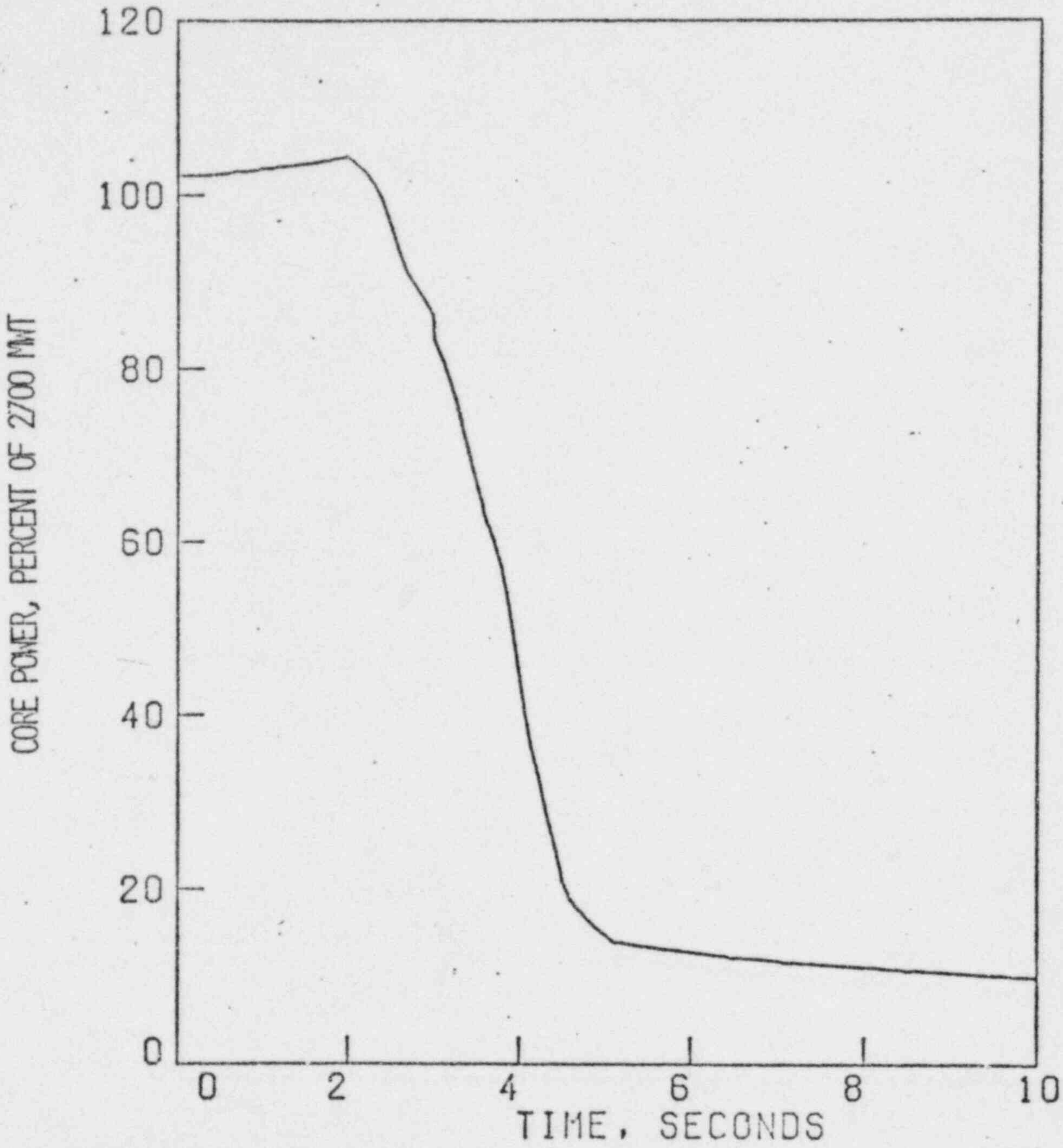
TABLE 7.1-2

SEQUENCE OF EVENTS FOR
LOSS OF FLOW

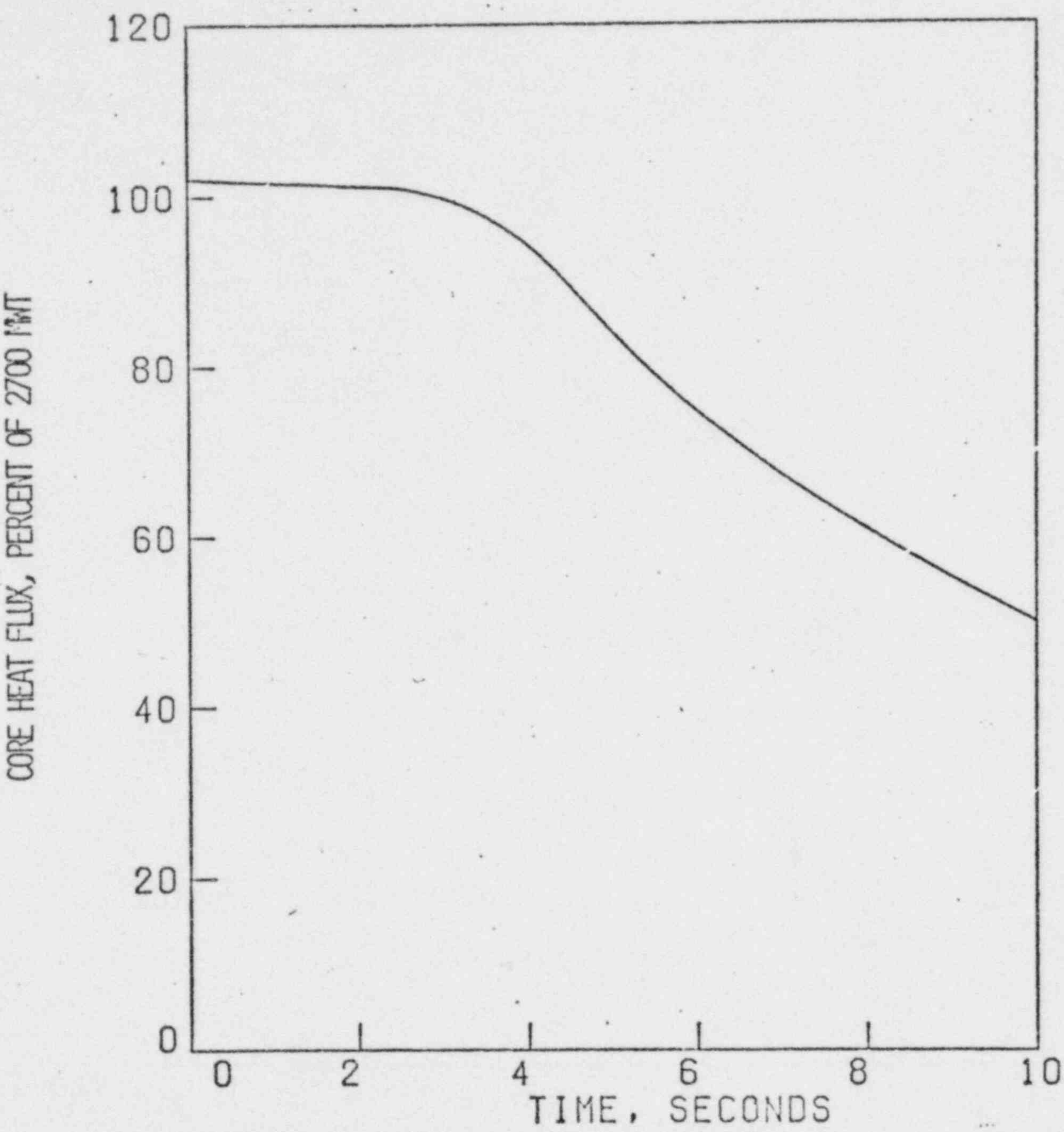
Time(sec)	Event	Setpoint or Value
0.0	Loss of Power to all Four Reactor Coolant Pumps	----
0.90	Low Flow Trip Signal Generated	93% of Initial 4-Pu[illegible] Flo[illegible]
1.40	Trip Breakers Open	----
1.90	Shutdown CEAs Begin to Drop Into Core	----
3.22	Minimum CE-1 DNBR	1.195
6.00	Maximum RCS Pressure, psia	2307



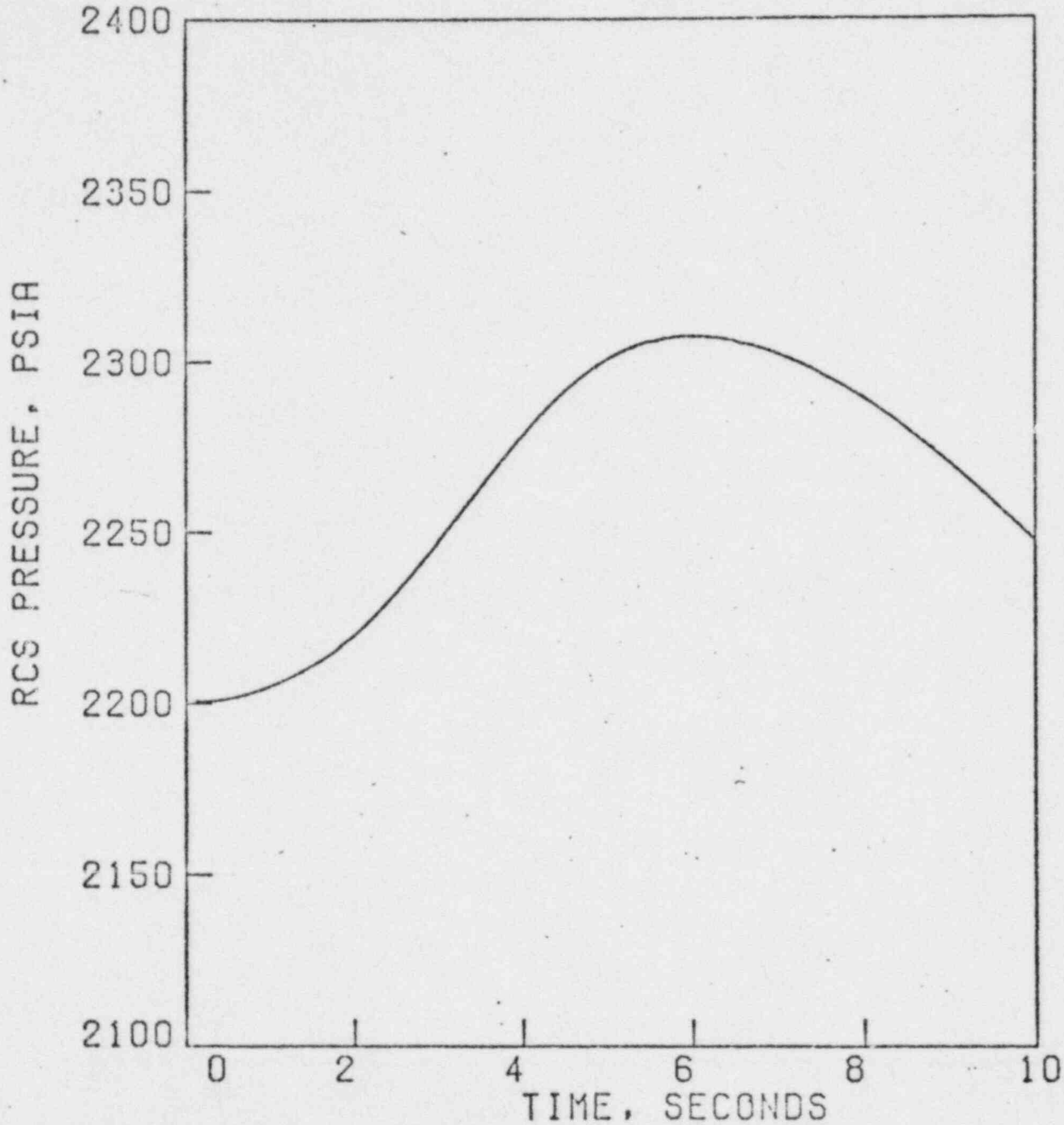
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT CORE FLOW FRACTION VS TIME	7.1-1



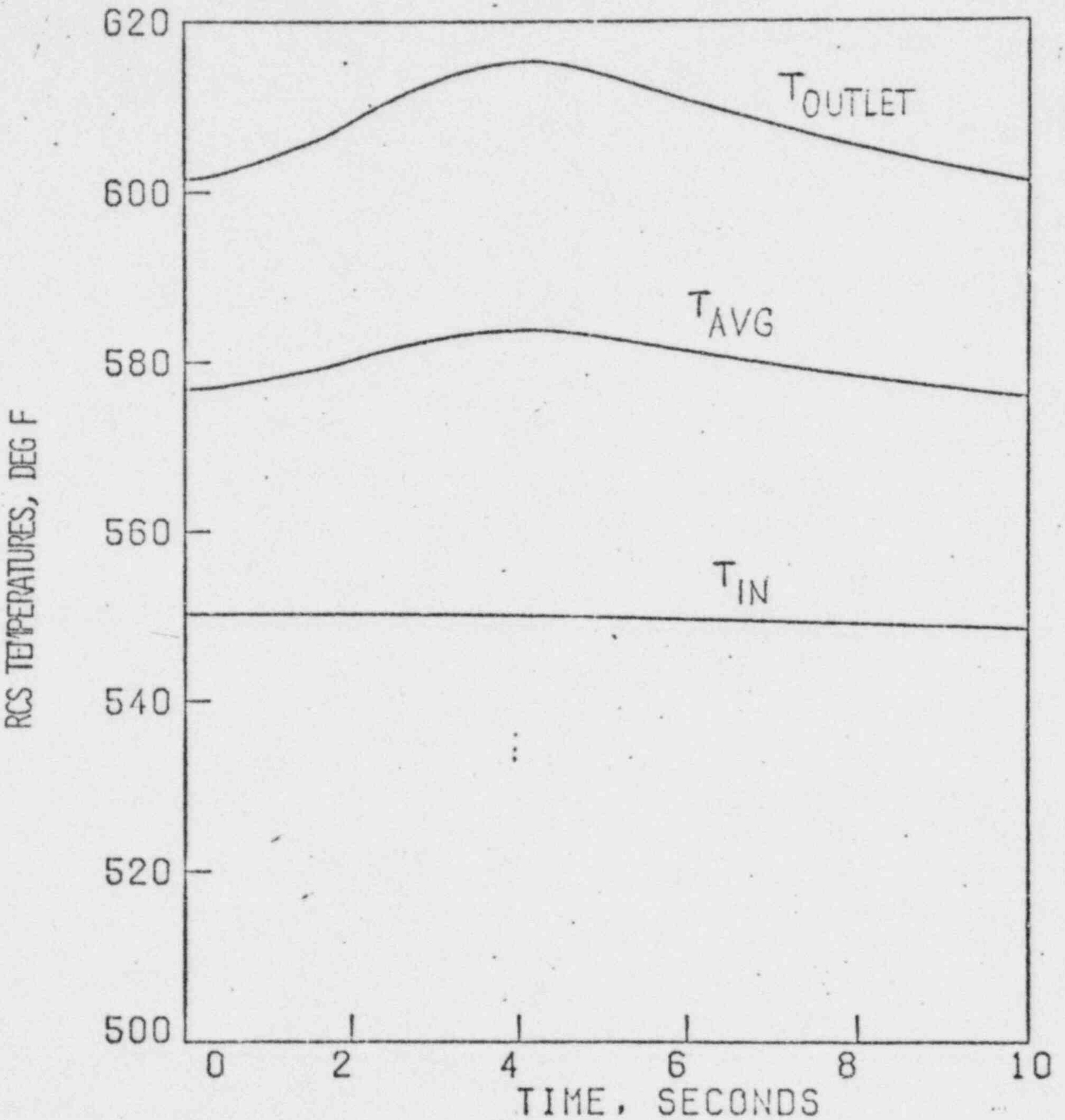
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT CORE POWER vs TIME	7.1-2



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT CORE HEAT FLUX vs TIME	7.1-3

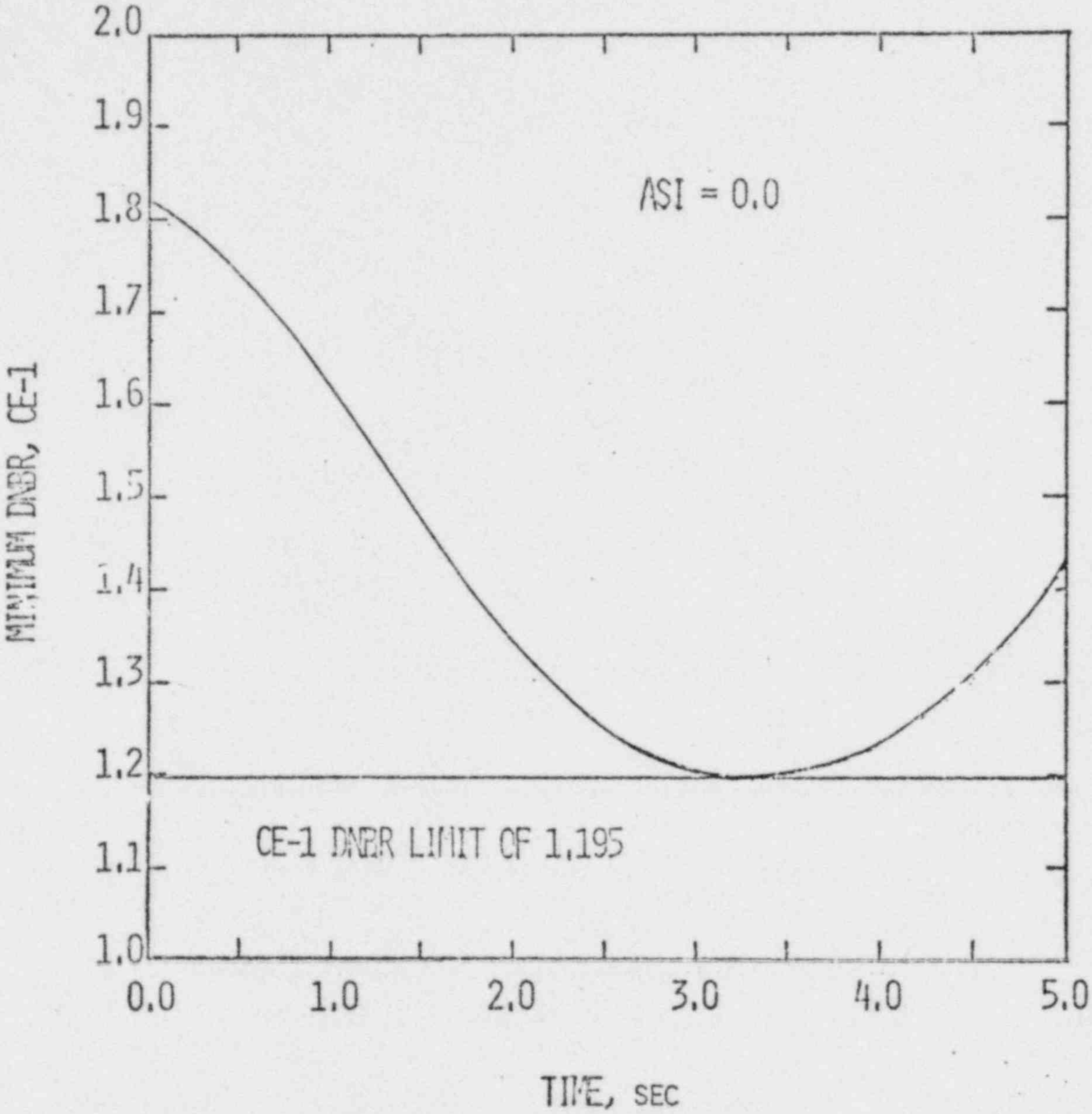


BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT RCS PRESSURE vs TIME	7.1-4



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT RCS TEMPERATURES vs TIME	7.1-5

POOR ORIGINAL



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF COOLANT FLOW EVENT DNBR (CE-1) VS TIME	7.1-6

8.0 ECCS ANALYSIS

An ECCS performance analysis was performed for Calvert Cliffs Unit 2 Cycle 4 to demonstrate compliance with 10CFR50.46 which presents the NRC Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled reactors[12]. The analysis justifies an allowable peak linear heat generation rate (PLHGR) of 15.5 kw/ft which is equal to the existing limit for Unit 2.

The ECCS performance analysis for Calvert Cliffs Unit 1 Cycle 5 operation[13] was used as the reference cycle analysis for the Unit 2 Cycle 4 evaluation. That analysis used fuel performance data which bound both Unit 1, Cycle 5 and Unit 2, Cycle 4. Therefore the results reported in Reference 13 are applicable to Unit 2 Cycle 4.

The results of that analysis identified the peak clad temperature as 1987°F as opposed to the acceptance limit of 2200°F. The peak local clad oxidation was 9.7% versus the acceptance limit of 17% and the peak core wide clad oxidation was less than .51% versus the acceptance limit of 1.0%. Hence, Unit 2 Cycle 4 operation at a peak linear heat generation rate of 15.5 kw/ft and at a power level of 2754 Mw_t (102% of 2700 Mw_t) will result in acceptable ECCS performance.

9.0 Technical Specifications

The Technical Specification changes which must be made in order to make the Calvert Cliffs II Technical Specifications valid for the operation of Cycle 4 are nearly identical to the changes made in the reference cycle as reported in References 1 and 2. Specific differences are:

1. Present Unit II Technical Specifications contain a most negative MTC limit of -2.3×10^{-4} $\Delta k/k/°F$ as compared to the former Unit I limit of -2.5×10^{-4} $\Delta k/k/°F$. For both units the most negative MTC limit is being changed to -2.2×10^{-4} $\Delta k/k/°F$.

2. The present Unit II peak linear heat rate limit is 15.5 kw/ft and, therefore, no change is needed. The Unit I limit was raised from 14.2 kw/ft to 15.5 kw/ft.

3. The present radial peaking factor limits for Unit II are different than the former Unit I limits. As for Unit I, these peaking factor limits must be changed to 1.62.

Table 9-1 presents a summary of the Technical Specification changes required for Unit II. For your convenience, the items in this table are presented in the same order as the changes presented in References 1 and 2.

Specific pages from the Unit II Technical Specifications showing the required modifications are not included since the corresponding Unit I pages can be found in the reference indicated in Table 9-1 for each change. Table 9-2 presents the explanations for the changes summarized in Table 9-1.

POOR ORIGINAL

TABLE 9-1

Calvert Cliffs II Cycle 4
Technical Specification Changes

Change #	Tech Spec #	Action	Reference
1	Figure 2.1-1 page 2-2	Replace Figure 2.1-1	2
2	Table 2.2-1 page 2-9	Change steam generator pressure-low setpoint from $\geq$500 psia to $\geq$570 psia	1
3	Table 2.2-1 page 2-10	Add steam generator pressure difference - high setpoint	1
4	Table 2.2-1 page 2-10	Change steam generator pressure-low trip bypass from below 600 psia to below 685 psia	1
5	Figure 2.2-1 page 2-11	Replace Figure 2.2-1	2
6	Figure 2.2-2 page 2-12	No changes from Cycle 3	--
7	Figure 2.2-3 page 2-13	No changes from Cycle 3.	--
8	B.2.1.1 page B2-1	Remove numerical specification of LHGR to centerline melt	1
9	B.2.1.1 page B2-1	No changes from Cycle 3	--
10	B.2.1, B.2.2 pages B2-1, B2-3, B2-5, B2-6	Change minimum DNBR value from 1.19 to 1.195	2
11	B.2.2.1 page B2-4	No changes from Cycle 3	--
12	B.2.2.1 page B2-5	Change steam generator pressure-low setpoint from 500 psia to 570 psia	1
13	B.2.2.1 page B2-7	Revise description of TM/LP trip and add asymmetric steam generator transient protective trip function description	1

TABLE 9-1 (continued)

Change #	Tech Spec #	Action	Reference
14	3.1.1.1 page 3/4 1-1	Change Shutdown Margin Tavg >200°F from ≥3.4%Δk/k to ≥4.3%Δk/k and change minimum boration concentration from 1720 ppm to 2300 ppm	1
15	3.1.1.2 page 3/4 1-3	Change Shutdown Margin Tavg ≤200°F from ≥1.0%Δk/k to ≥3.0%Δ k/k and change minimum boration concentration from 1720 ppm to 2300 ppm	1
16	3.1.1.4 page 3/4 1-5	Change MTC less negative than -2.3x10^{-4}Δk/k/°F to less negative than -2.2x10^{-4}Δk/k/°F	1
17	3.1.2.2 page 3/4 1-9	Change Shutdown Margin equivalent from at least 1%Δk/k at 200°F to at least 3%Δk/k	1
18	3.1.2.4 page 3/4 1-11	Change Shutdown Margin equivalent from at least 1%Δk/k at 200°F to at least 3%Δk/k	1
19	3.1.2.6 page 3/4 1-13	Change Shutdown Margin equivalent from at least 1%Δk/k at 200°F to at least 3%Δk/k	1
20	3.1.2.7 page 3/4 1-14	Change refueling water tank minimum borated water volume from 9,978 gallons to 9,844 gallons	1
21	3.1.2.7 page 3/4 1-14	Change refueling water tank boron concentration from 1720 ppm to between 2300 and 2800 ppm	1
22	Figure 3.1-1 page 3/4 1- 15	Change minimum boric acid storage tank volume function	1
23	3.1.2.8 page 3/4 1-16	Change refueling water tank boron concentration from between 1720 and 2200 ppm to between 2300 and 2800 ppm and Shutdown Margin equivalent from 1%Δk/k at 200°F to 3%Δk/k at 200°F	1
24	Figure 3.2-1 page 3/4 2-3	No change from Cycle 3.	--
25	Figure 3.2-2 page 3/4 2-4	Replace Figure 3.2-2	2

TABLE 9-1 (continued)

Change #	Tech Spec #	Action	Reference
26	Figure 4.2-1, page 3/4 2-5	Replace Figure 4.2-1	1
27	3.2.2 page 3/4 2-6	Change calculated value of Fxy^T from ≤ 1.610 to ≤ 1.620 and $Fxy^T > 1.610$ to $Fxy > 1.620$	2
28	Figure 3.2-3 page 3/4 2-8	Replace Figure 3.2-3	2
29	3.2.3 page 3/4 2-9	Change calculated value of Fr^T from ≤ 1.540 to ≤ 1.620 and change $Fr^T > 1.540$ to $Fr^T > 1.620$	2
30	Figure 3.2-4 page 3/4 2-11	Replace Figure 3.2-4	2
31	Table 3.3-1, page 3/4 3-2	Add steam generator pressure difference - high description to table	1
32	Table 3.3-1, page 3/4 3-4	Change steam generator pressure-low trip bypass from below 600 psia to below 685 psia	1
33	Table 3.3-2, page 3/4 3-6	Add steam generator pressure difference-high response time	1
34	Table 4.3-1, page 3/4 3-7	Add steam generator pressure difference-high surveillance	1
35	Table 3.3-3, page 3/4 3-15	Change Main Steam Line Isolation steam generator pressure-low trip bypass from below 600 psia to below 685 psia	1
36	Table 3.3-4, page 3/4 3-17	Change Main Steam Line Isolation steam generator pressure-low setpoint from ≥ 478 psia to ≥ 570 psia	1
37	Table 3.3-5, page 3/4 3-20	Change Containment Purge Isolation Valve Response time from ≤ 6 to ≤ 5 sec	1
38	3.5.1 page 3/4 5-1	Change safety injection tank boron concentration from between 1720 and 2200 ppm to between 2300 ppm and 2800 ppm.	2
39	3.5.4 page 3/4 5-7	Change refueling water tank boron concentration from between 1720 and 2200 ppm to to between 2300 and 2800 ppm	1

TABLE 9-1 (continued)

Change #	Tech Spec #	Action	Reference
40	3.9.1 page 3/4 9-1	Change refueling boron concentration of >1720 ppm to >2300 ppm and boration at ≥40 gpm of 1720 ppm to boration at ≥40 gpm of 2300 ppm and shutdown margin from 1%Δk/k to 3%Δk/k	1
41	3.10.1 page 3/4 10-1	Change boration at ≥40 gpm of 1720 ppm to boration at ≥40 gpm of 2300 ppm	1
42	B 3/4.1.1.1 and B 3/4.1.1.2, Page B 3/4 1-1	Change minimum Shutdown Margin with Tavg ≤200°F from 1%Δk/k to 3%Δk/k and revise basis	1
43	B 3/4.1.2, pages B 3/4 1-2, B 3/4 1-3	Change Shutdown Margin of 1.0%Δk/k after xenon decay and cooldown to 200°F to 3.0%Δk/k after xenon decay and cooldown to 200°F and the refueling water tank boron concentration from 1720 ppm to 2300 ppm	1
		Change 3913 gallons of 7.25% boric acid solution to 6500 gallons and 47,204 gallons of borated water to 55,627 gallons.	1
44	B 3/4.1.2, page B 3/4 1-3	Change 9,978 gallons of borated water to 9844 gallons and 439 gallons of 7.25% boric acid to 737 gallons.	1
45	B 3/4.2.5, page B 3/4 2-2	Change minimum DNBR of 1.19 to minimum DNBR of 1.195	2
46	B 3/4.9.1, page B 3/4 9-1	Change minimum boron concentration (1720 ppm) to (2300 ppm)	1
47	3.4.1 page 3/4 4-2	Include specific operation of reactor coolant pumps for Mode 3	1
48	3.1.1.2, page 3/4 1-3, and B 3/4 1.1.1, B 3/4 1.1.2 page B 3/4 1-1	Replace pages 3/4 1-3 and B 3/4 1-1	2
49	4.5.2, e.3 and e.4 pg. 3/4 5-5	Change minimum volume of TSP from 75 cubic feet to 100 cubic feet and change sample volume to 4.0± 0.1 gms in 3.5±.1 liters of RWT water.	2

TABLE 9-2

Explanations for Cycle 4 Tech Spec Changes

Change #	Tech Spec #	Explanation
1	Figure 2.1-1	Thermal limit lines have been changed to reflect different radial peaking factors.
2	Table 2.2-1	The steam generator pressure-low setpoint is being increased to minimize the consequences of a Steam Line Break Event.
3	Table 2.2-1	A trip for Asymmetric Steam Generator pressure has been added to minimize the consequence of the Loss of Load to One Steam Generator Event.
4	Table 2.2-1	The steam generator pressure-low trip bypass has been increased to be consistent with the new trip value.
5	Figure 2.2-1	The LHR LSSS has been changed to reflect different radial peaking factors.
6	Figure 2.2-2	Reanalysis for Cycle 4 has produced no changes in TM/LP trip
7	Figure 2.2-3	Reanalysis for Cycle 4 has produced no changes in TM/LP trip
8	B.2.1.1	The numerical specification of centerline melt limit is being deleted to standardize spec to other C-E plants.
9	B.2.1.1	No changes from Cycle 3.
10	B.2.1, B.2.2	The minimum DNBR has been changed to 1.195.
11	B.2.2.1	No change from Cycle 3.

TABLE 9-2 (continued)

Change #	Tech Spec #	Explanation
12	B.2.2.1	The basis of the steam generator pressure-low trip setpoint has been changed to be consistent with Table 2.2-1.
13	B.2.2.1	The TM/LP basis has been streamlined for clarity and a description of the asymmetric steam generator pressure trip has been added to the bases.
14	3.1.1.1	The shutdown margin has been increased to yield acceptable consequences from a Steam Line Break Event. The new boron concentration is consistent with the new refueling water tank concentration for Cycle 4.
15	3.1.1.2	The shutdown margin has been increased to lengthen the operator action time required in a boron dilution event. The new boron concentration is consistent with the new refueling water tank concentration for Cycle 4.
16	3.1.1.4	The most negative MTC permitted for Cycle 4 has been made less negative to yield acceptable consequences from a Steam Line Break event.
17	3.1.2.2	The required shutdown margin has been increased to be consistent with Tech Spec 3.1.1.2.
18	3.1.2.4	The required shutdown margin has been increased to be consistent with Tech Spec 3.1.1.2.
19	3.1.2.6	The required shutdown margin has been increased to be consistent with Tech Spec 3.1.1.2.
20	3.1.2.7	The volume of borated water has been decreased due to the higher soluble boron concentrations.
21	3.1.2.7	The refueling water tank boron concentration has been changed to be consistent with Tech Spec 3.9.1.
22	Figure 3.1-1	The volume of borated water has been increased to allow a higher shutdown boron insertion due to the higher core average enrichments of future cycles.

Change #	Tech Spec #	Explanation
23	3.1.2.8	The refueling water tank boron concentration has been changed to be consistent with Tech Spec 3.9.1 and the required shutdown margin has been increased to be consistent with Tech Spec 3.1.1.2
24	Figure 3.2-1	No change from Cycle 3.
25	Figure 3.2-2	The LHR LCO is being changed as a result of higher radial peaks.
26	Figure 4.2-1	Augmentation factors have been increased to envelope future cycles
27	3.2.2	Radial peaking factors, both F_{xy}^T and F_r^T, are being raised for Cycle 4.
28	Figure 3.2-3	Radial peaking factors, both F_{xy}^T and F_r^T, are being raised for Cycle 4.
29	3.2.3	Radial peaking factors, both F_{xy}^T and F_r^T, are being raised for Cycle 4.
30	Figure 3.2-4	The DNB LCO limits are changing due to higher radial peaks.
31	Table 3.3-1	The asymmetric steam generator pressure trip has been added to the table.
32	Table 3.3-1	The steam generator pressure-low trip bypass has been increased to be consistent with the new trip value.
33	Table 3.3-2	The asymmetric steam generator pressure trip has been added to the table.
34	Table 4.3-1	The asymmetric steam generator pressure trip has been added to the table.
35	Table 3.3-3	The Main Steam Line Isolation steam generator pressure-low trip bypass has been increased to be consistent with the new trip value.
36	Table 3.3-4	The Main Steam Line Isolation steam generator pressure-low trip setpoint has been increased to be consistent with the reactor trip setpoint.

TABLE 9-2 (continued)

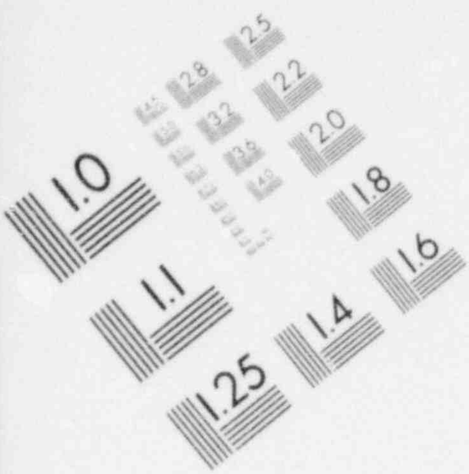
Change #	Tech Spec #	Explanation
37	Table 3.3-5	Containment isolation value response time is being reduced from 6 seconds to 5 seconds to satisfy NRC requirements. (NRC Branch Technical Position CSB 6 4)
38	3.5.1	The safety injection tank boron concentration has been increased to assure a uniform boron concentration in all coolants that have access to the reactor vessel.
39	3.5.4	The refueling water tank boron concentration has been increased to be consistent with Tech Spec 3.9.1
40	3.9.1	The refueling boron concentrations have been increased due to the higher core average enrichment of future cycles and the shutdown margin has increased to be consistent with 3.1.1.2.
41	3.10.1	The boration concentrations have been increased to be consistent with the new boron concentration of the refueling water tank.
42	B 3/4.1.1.1 and B 3/4.1.1.2	The shutdown margins in the bases have been increased to be consistent with those in Tech Specs 3.1.1.1 and 3.1.1.2 and explain applicability of shutdown margin for steam line break accident.
		The number of gallons of PPM boron has increased to accommodate increased boron insertion requirements for future cycles.
43	B 3/4.1.2	The shutdown margin has been increased in the bases to be consistent with Tech Spec 3.1.1.2. The refueling water tank boron concentration in the bases has been increased to be consistent with Tech Spec 3.9.1
44	B 3/4.1.2 page B 3/4 1-3	The volume of borated water in BAST has been decreased due to the higher soluble boron concentration and increased in RWT due to increased boron insertion requirements.

TABLE 9-2 (continued)

Change #	Tech Spec #	Explanation
45	B 3/4.2.5	The minimum DNBR has been changed to 1.195.
46	B 3/4.9.1	The refueling water concentration in the bases has been increased to be consistent with Tech Spec 3.9.1
47	3.4.1	One-loop no load conditions have not been analyzed for Cycle 4
48	3.1.1.2 and B 3/4 1.1.1	Additional requirements to the pressurizer level have been included to increase the time to criticality during a boron dilution event.
49	4.5.2 e.3 and e.4	The minimum volume of TSP needed to raise the PH of the borated water of the ECCS to 7.0 is 100 cubic feet. In order to test the ability of the TSP to raise the PH of the borated water of the ECCS, the ratio of the volume of TSP to the volume of ECCS borated water must be the same in containment as it is in the laboratory.

10.0 Startup Testing

The startup testing program proposed for Calvert Cliffs II Cycle 4 is identical to the program proposed for the reference cycle in References 1 and 14.



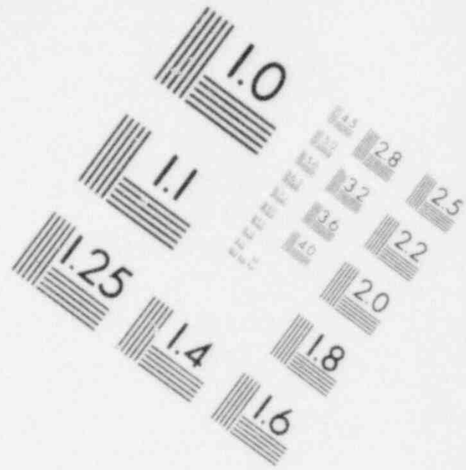
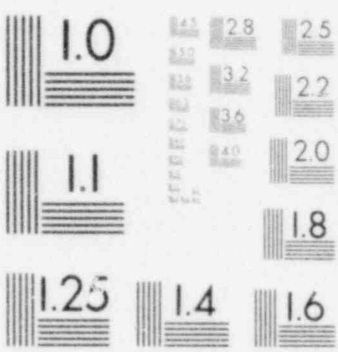
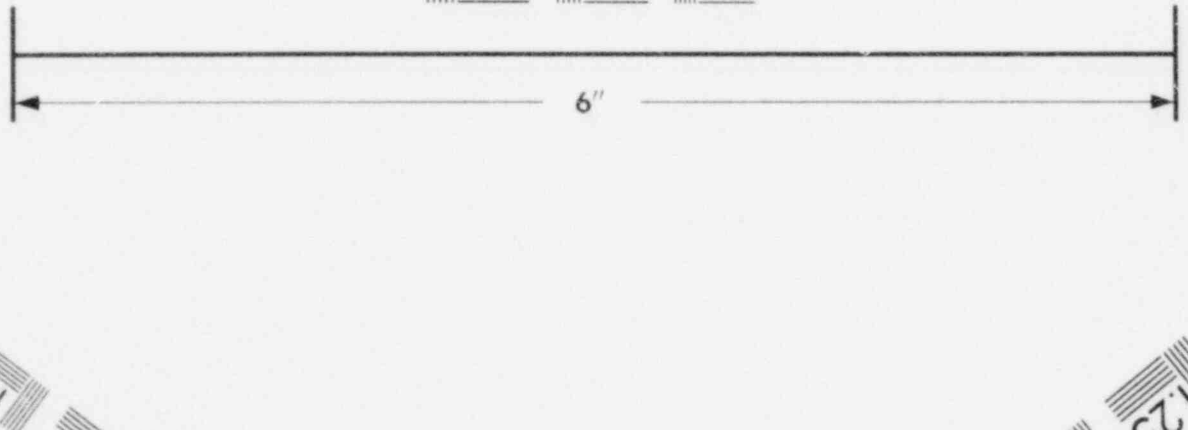
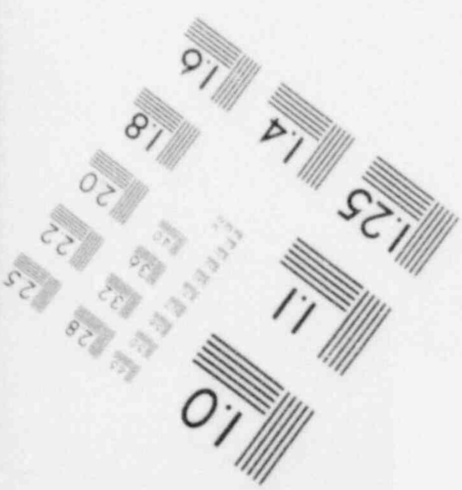
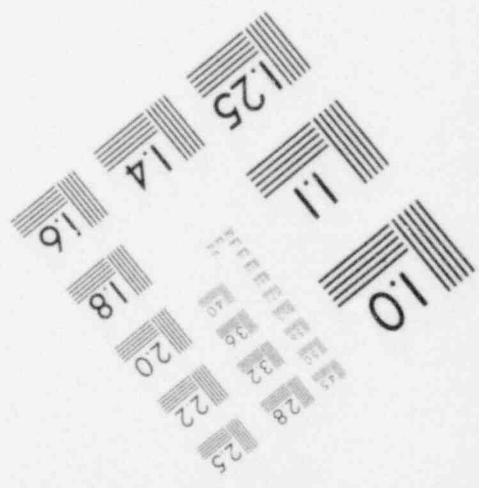


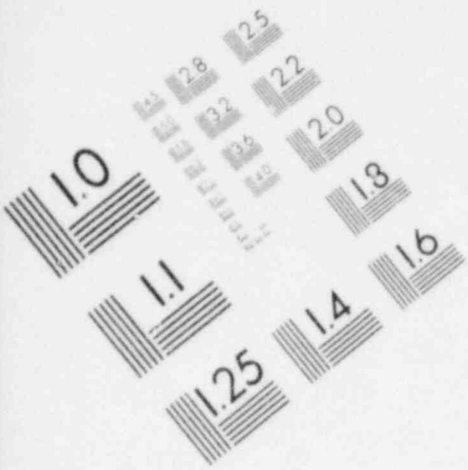
IMAGE EVALUATION TEST TARGET (MT-3)











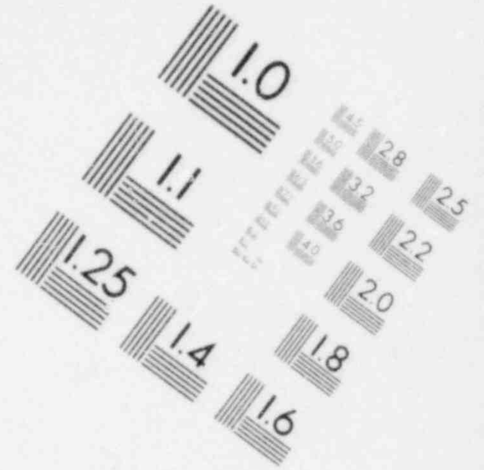
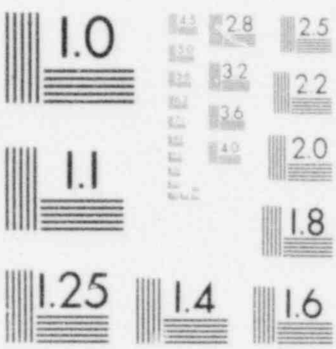
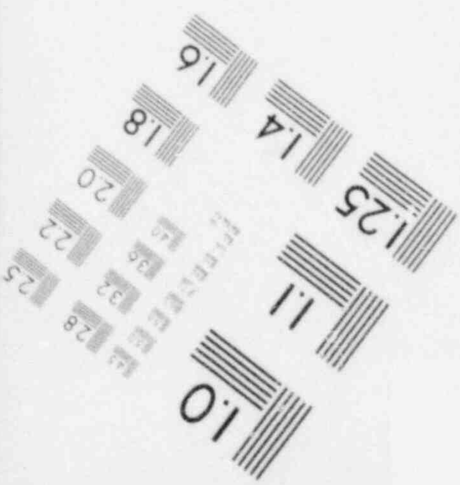
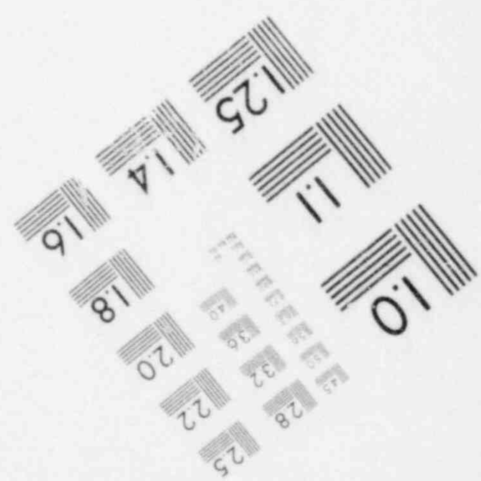


IMAGE EVALUATION TEST TARGET (MT-3)



6″





11.0 References

1. Letter from A. E. Lundvall to R. A. Clark, September 22, 1980, "Calvert Cliffs Nuclear Power Plant Unit I, Docket 50-317 Amendment to Operating License DPR-53 5th Cycle License Application"

2. Letter from A. E. Lundvall to R. A. Clark, November 4, 1980, "Calvert Cliffs Nuclear Power Plant Unit I, Docket 50-317 Amendment to Operating License DPR-53 Supplement 1 to 5th Cycle License Application"

3. Letter from A. E. Lundvall to R. A. Clark, June 12, 1980, "Calvert Cliffs Nuclear Power Plant, Unit No. 2, Docket No. 50-318 Report of Startup Testing for Cycle Three"

4. Letter, A. E. Lundvall to R. W. Reid, "Request for Amendment to Operating License, Unit 2, Cycle 2 License Application", dated July 26, 1978

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

6. CEN-83(B)-P, "Calvert Cliffs Unit 1 Reactor Operation With Modified CEA Guide Tubes", dated February 8, 1979 and letter, A. E. Lundvall, Jr. to V. Stello, Jr., "Reactor Operation With Modified CEA Guide Tubes", dated February 17, 1978

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

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

9. Letter from A. E. Lundvall to R. W. Reid, July 11, 1979, "Proposed Finding of No Unreviewed Safety Questions on Unit 2, Cycle 3 Reload Core Design"

10. Letter A. E. Lundvall to R. A. Clark, December 3, 1980, "Calvert Cliffs Nuclear Power Plant, Unit No. 1, Docket No. 50-317 Amendment to Operating License DPR-53, Fifth Cycle License Application, Responses to NRC Staff Questions"

11. "The Interim SER on Effects of Fuel Rod Bowing in Thermal Margin Calculation for Light Water Reactors," Rev. 1, Feb. 16, 1977.

12. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3, Friday, January 4, 1974.

13. Letter A. E. Lundvall to R. A. Clark, November 25, 1980, "Calvert Cliffs Nuclear Power Plant, Unit No. 1, Docket No. 50-317 Amendment to Operating License DPR-53, Supplement 2 to Fifth Cycle License Application"

14. Letter A. E. Lundvall to R. A. Clark, November 19, 1980, "Calvert Cliffs Nuclear Power Plant, Unit No. 1, Docket No. 50-317 Amendment to Operating License DPR-53, Fifth Cycle License Application, Responses to NRC Staff Questions"