ENCLOSURE TO

CALVERT CLIFFS UNIT 1 CYCLE 8

RELOAD LICENSE SUBMITTAL

Calvert Cliffs Unit 1 Cycle 8
Reload License Submittal

Table of Contents

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of design and performance for the operation of Calvert Cliffs Unit 1 during its eighth fuel cycle, at full rated power of 2700 MWt. All planned operating conditions remain the same as those for Cycle 7. The core will consist of 141 presently operating Batch F, G, H, and J assemblies, 72 fresh Batch K assemblies, and 4 Batch E assemblies previously discharged from Cycle 4 of Calvert Cliffs Unit 2.

Plant operating requirements have created a need for flexibility in the Cycle 7 termination point, ranging from 12,900 MWD/T to 13,900 MWD/T. In performing analyses of design basis events, determining limiting safety settings and establishing limiting conditions for operation, limiting values of key parameters were chosen to assure that expected Cycle 8 conditions would be enveloped, provided the Cycle 7 termination point falls within the above cycle burnup range. The analysis presented herein will accommodate a Cycle 8 length of up to 14,400 MWD/T.

The evaluations of the reload core characteristics have been conducted with respect to the Calvert Cliffs Unit 1 Cycle 7 safety analysis described in Reference 1, 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 the reference cycle, proposed modifications to the plant Technical Specifications are provided and are justified by the analyses reported herein.

All Cycle 8 analyses address fuel exposure explicitly. The performance of Combustion Engineering 14x14 fuel at extended burnup is discussed in Reference 2. Fuel performance for Cycle 8 has been evaluated with the FATES3 computer code (References 3 and 4) as approved by the NRC in Reference 5.

2.0 OPERATING HISTORY OF THE PREVIOUS CYCLE

Calvert Cliffs Unit 1 is presently operating in its seventh fuel cycle utilizing Batch J, H, G, F, E, D and B fuel assemblies (including twenty-four Batch D and B assemblies from Unit 2). Calvert Cliffs Unit 1 Cycle 7 began operation on November 30, 1983 and reached full power on December 22, 1983. The Cycle 7 startup testing was reported to the NRC in Reference 6. The reactor has operated up to the present time with the core reactivity, power distributions and peaking factors closely following the calculated predictions.

It is presently estimated that Cycle 7 will terminate on or about April 5, 1985. The Cycle 7 termination point can vary between 12,900 MWD/T and 13,900 MWD/T to accommodate the plant schedule and still be within the assumptions of the Cycle 8 analyses. As of February 12, 1985, the Cycle 7 burnup had reached 11,554 MWD/T.

3.0 GENERAL DESCRIPTION

The Cycle 8 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 8 is the removal of 76 assemblies (52 Unit 1 assemblies: 36 Batch G, 4 Batch F, 12 Batch E/; 24 Unit 2 assemblies: 12 Batch D/ and 12 Batch B). These assemblies will be replaced by 48 fresh unshimmed Batch K assemblies (4.05 wt% U-235 enrichment), 24 fresh unshimmed Batch K* assemblies (3.43 wt% U-235 enrichment) and 4 Batch E assemblies (3.03 wt% U-235) discharged from Unit 2 Cycle 4.

Figure 3-1 shows the fuel management pattern to be employed in Cycle 8. Figure 3-2 shows the locations of the poison pins within the lattice of twice-burned Batch H/ assemblies and the fuel rod locations in unshimmed assemblies. This fuel management pattern will accommodate Cycle 7 termination burnups from 12,900 MWD/T to 13,900 MWD/T.

The Cycle 8 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 8 assembly burnup distribution for a Cycle 7 termination burnup of 13,900 MWD/T. The initial enrichment of the fuel assemblies is also shown in Figure 3-3. Figure 3-4 shows the end of Cycle 8 assembly burnup distribution. The end of Cycle 8 core average exposure is approximately 29,400 MWD/T and the average discharge exposure is approximately 42,200 MWD/T. The end of cycle burnups are based on Cycle 7 and Cycle 8 lengths of 13,900 MWD/T and 14,400 MWD/T, respectively.

3.1 SCOUT Demonstration Assembly

The original configuration of the SCOUT demonstration assembly was described in Reference 7. It is a Batch F test assembly which was initially inserted in Cycle 4. Changes, similar to those described in Reference 8, were made to this assembly prior to its third cycle of irradiation in Cycle 6. Before returning the assembly to the core for its fourth cycle of irradiation in Cycle 7, 2 segmented test rods were removed from the assembly and replaced with 2 stainless steel rods. The Scout assembly will be reinserted in the core for its fifth cycle of irradiation in Cycle 8 without further replacement of fuel rods.

3.2 PROTOTYPE Demonstration Assemblies

The original configuration of the PROTOTYPE demonstration assemblies was described in Reference 9. These are Batch G demonstration assemblies which were initially inserted in Cycle 5. Before returning the assemblies to the core for their third cycle of irradiation in Cycle 7, 2 segmented test rods were removed from one of these assemblies and replaced with 2 stainless steel rods. The Prototype assemblies will be reinserted in the core for their fourth cycle of irradiation in Cycle 8 without further replacement of fuel rods.

3.3 CEA Patterns

The composition of nine CEAs, the configurations of two CEA banks and, consequently, the overall CEA bank pattern are being changed for Cycle 8. These changes are being made to support the expansion of the negative MTC Tech Spec. (Section 9.0). This support comes in the form of increased net available scram worth which is being used in the Steam Line Rupture Analysis (Section 7.3.2) to compensate for the more adverse reactivity cooldown data that results from the MTC change.

This increase in scram worth is being brought about by fully strengthening the weak CEA in the vicinity of the worst stuck CEA. In addition to changing the strength of this particular CEA, the compositions of other CEAs and the configurations of both the lead bank and another CEA bank are being altered. Such additional changes are being made because the weak CEA in the vicinity of the worst stuck CEA is presently part of the lead bank and strengthening this CEA without changing the composition of the lead bank and, consequently, other banks would lead to an undersirable increase in rodded peaking factors for the lead bank.

The specific changes that will be made to the compositions of individual CEAs, to the configurations of CEA banks and to the overall CEA bank pattern during the Cycle 7 to Cycle 8 outage are summarized below:

1. All eight weak CEAs which are presently part of the lead bank, Bank 5, (Figure 3-5) will be converted to full strength CEAs.

2. The center CEA which is presently part of Bank 5 and is presently a full strength CEA (Figure 3-5) will be converted to a very weak CEA, i.e., it will be composed of only Al_2O_3 CEA fingers.

3. The present Bank 5 configuration of 9 CEAs, eight weak and one full strength (Figure 3-5), will be changed to 5 CEAs, four full strength CEAs and one very weak CEA (Figure 3-8).

4. The present Bank 4 configuration of four full strength CEAs (Figure 3-6) will be changed to eight full strength CEAs (Figure 3-9) by the addition of the four CEAs which will be removed from Bank 5.

5. The configuration of all other CEA banks will remain unchanged. The present and future overall CEA bank patterns are shown in Figures 3-7 and 3-10, respectively.

TABLE 3-1
CALVERT CLIFFS UNIT 1 CYCLE 8
CORE LOADING

Assembly Designation	Number of Assemblies	Initial Enrichment (wt% U-235)	Batch Burnup(MWD/T) BOC8[3]	EOC8[3,4]	Poison Rods Per Assembly	Initial Poison Loading (wt% B_4C)	Total Number of Poison and Non-Fuel Rods	Total Number of Fuel Rods
K	48	4.05	0	13,300	0	0	0	8448
K*	24	3.40	0	17,800	0	0	0	4224
J[1]	48	4.05	12,700	29,000	0	0	0	8448
J*[1]	16	3.40	17,300	31,800	0	0	0	2816
H[1]	40	4.00	27,400	41,100	0	0	0	7040
H/[1]	32	3.55	30,700	43,300	8	3.03	256	5376
G[1]	4	3.65	38,700	50,400	0	0	2	702
F[1]	1	3.03	44,200	52,100	0	0	3	173
E[2]	4	3.03	25,600	36,800	0	0	0	704
TOTAL	217		15,000	29,400			261	37,931

(1) Carried over from Cycle 7 to Cycle 8 of Unit 1

(2) Twice burned Batch E fuel discharged from unit 2 Cycle 4.

(3) Cycle 7 burnup of 13,900 MWD/T

(4) Cycle 8 burnup of 14,400 MWD/T

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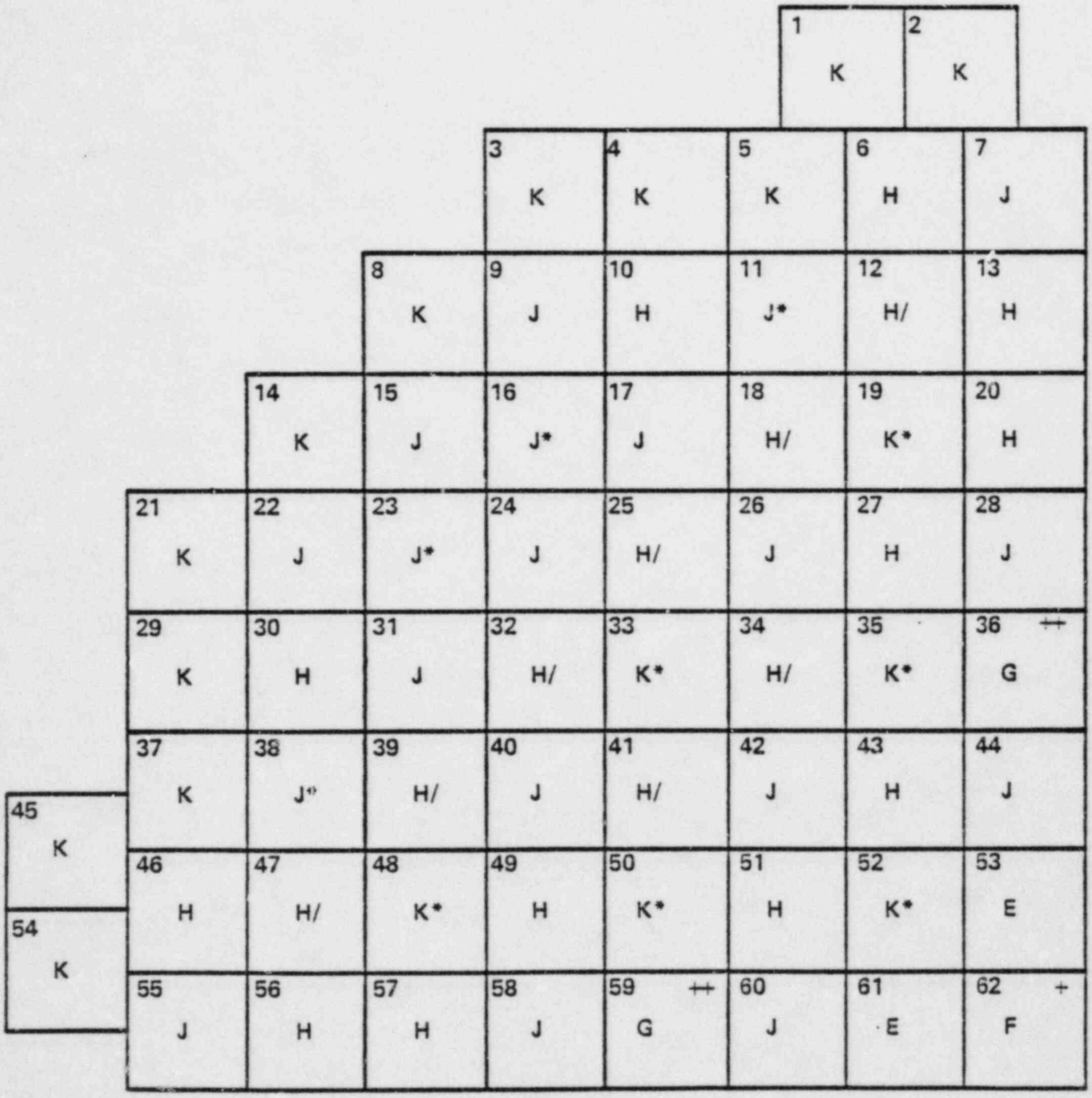
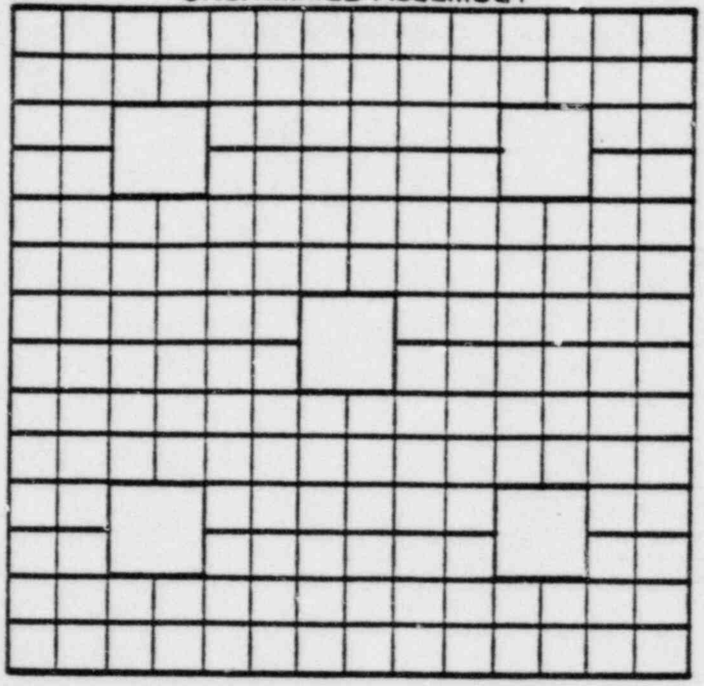


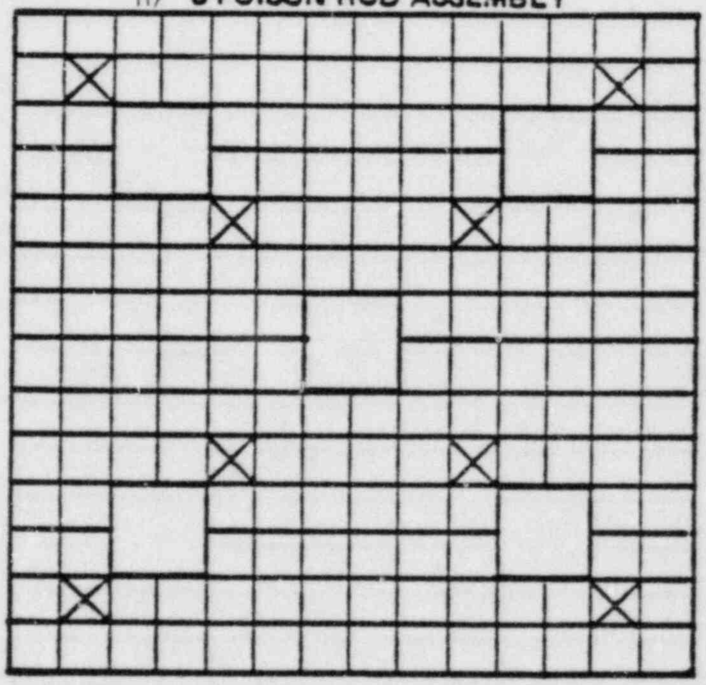
+LOCATION OF DEMONSTRATION ASSEMBLY (SCOUT)
++LOCA..ON OF PROTOTYPE ASSEMBLIES

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 8 CORE MAP	Figure 3-1

UNSHIMMED ASSEMBLY



H/ 8 POISON ROD ASSEMBLY



FUEL ROD LOCATION

POISON ROD LOCATION

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 8 ASSEMBLY FUEL AND OTHER ROD LOCATIONS	Figure 3-2

INITIAL ENRICHMENT W/O U-235
BOC 8 BURNUP (MWD/T), EOC 7 = 13,900 MWD/T

						1 K 4.05 0	2 K 4.05 0	
				3 K 4.05 0	4 K 4.05 0	5 K 4.05 0	6 H 4.00 27,500	7 J 4.05 13,400
			8 K 4.05 0	9 J 4.05 10,500	10 H 4.00 26,300	11 J* 3.40 17,500	12 H/ 3.55 29,600	13 H 4.00 26,900
		14 K 4.05 0	15 J 4.05 13,900	16 J* 3.40 16,700	17 J 4.05 10,500	18 H/ 3.55 31,300	19 K* 3.40 0	20 H 4.00 27,000
	21 K 4.05 0	22 J 4.05 10,500	23 J* 3.40 17,600	24 J 4.05 13,900	25 H/ 3.55 30,300	26 J 4.05 12,200	27 H 4.00 27,700	28 J 4.05 13,400
	29 K 4.05 0	30 H 4.00 26,400	31 J 4.05 10,500	32 H/ 3.55 31,100	33 K* 3.40 0	34 H/ 3.55 31,400	35 K* 3.40 0	36 G 3.65 38,700
45 K 4.05 0	37 K 4.05 0	38 J* 3.40 17,500	39 H/ 3.55 31,300	40 J 4.05 12,200	41 H/ 3.55 31,400	42 J 4.05 15,800	43 H 4.00 27,700	44 J 4.05 15,900
54 K 4.05 0	46 H 4.00 27,500	47 H/ 3.55 29,500	48 K* 3.40 0	49 H 4.00 27,700	50 K* 3.40 0	51 H 4.00 28,900	52 K* 3.40 0	53 E 3.03 25,600
	55 J 4.05 13,400	56 H 4.00 26,900	57 H 4.00 27,000	58 J 4.05 13,400	59 G 3.65 38,700	60 J 4.05 15,900	61 E 3.03 25,600	62 F 3.03 44,200

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 8 ASSEMBLY AVERAGE BURNUP AT BOC AND INITIAL ENRICHMENT DISTRIBUTION	Figure 3-3

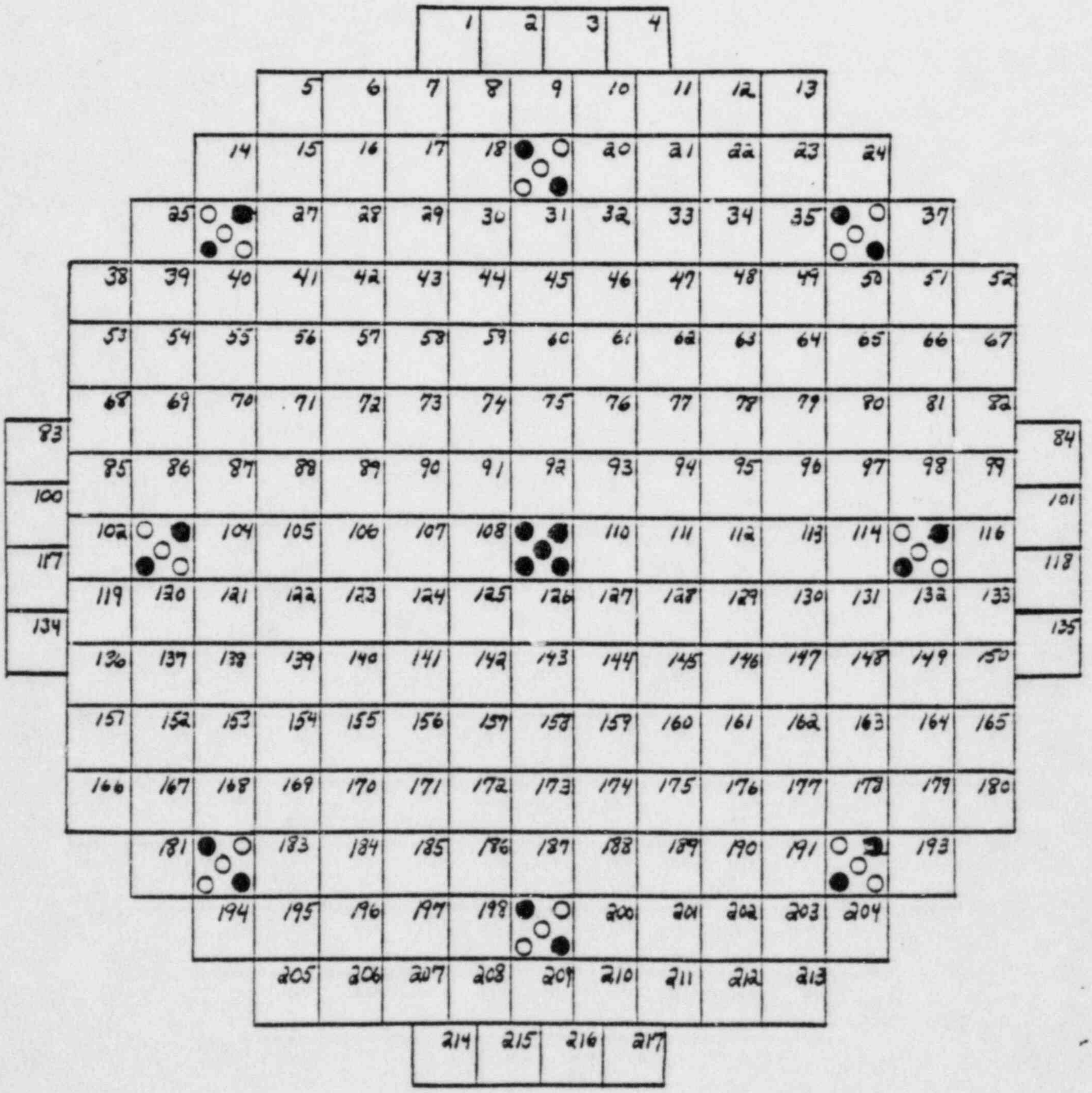
							1 K 11,000	2 K 14,000
				3 K 11,300	4 K 14,800	5 K 16,400	6 H 40,200	7 J 28,800
			8 K 12,500	9 J 26,500	10 H 40,500	11 J* 31,300	12 H/ 41,500	13 H 39,600
		14 K 12,500	15 J 29,600	16 J* 32,000	17 J 27,600	18 H/ 43,900	19 K* 17,900	20 H 41,200
	21 K 11,300	22 J 26,500	23 J* 32,600	24 J 30,600	25 H/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

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 8 ASSEMBLY AVERAGE BURNUP AT EOC (MWD/T)	Figure 3-4

Figure 3-5

CALVERT CLIFFS UNIT 1 CYCLE 7

BANK-5 CONFIGURATION



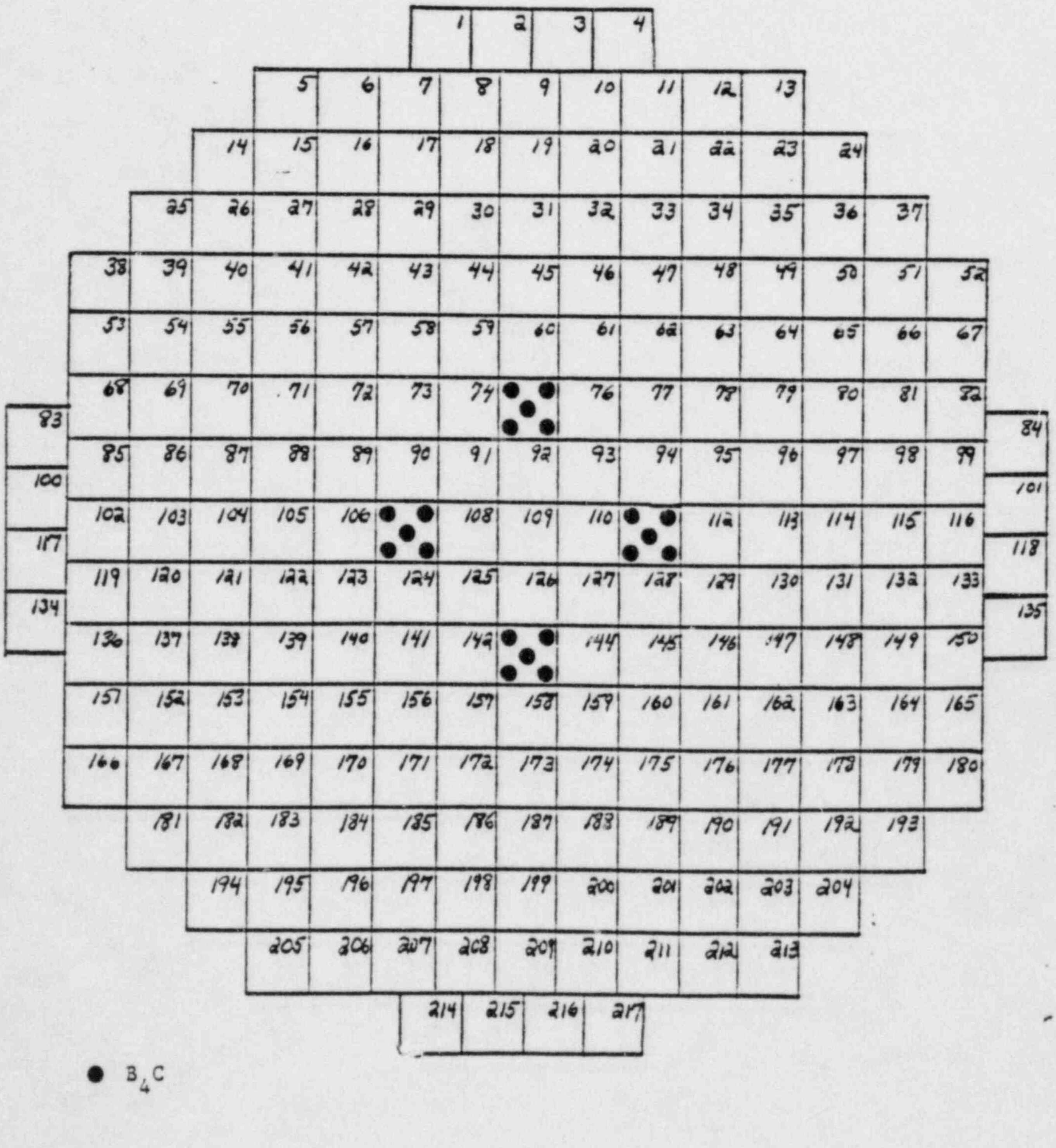
● B_4C

○ Al_2O_3

Figure 3-6

CALVERT CLIFFS UNIT 1 CYCLE 7

BANK-4 CONFIGURATION



<u>Figure 3-7</u>

CALVERT CLIFFS UNIT 1 CYCLE 7

CEA BANK PATTERN

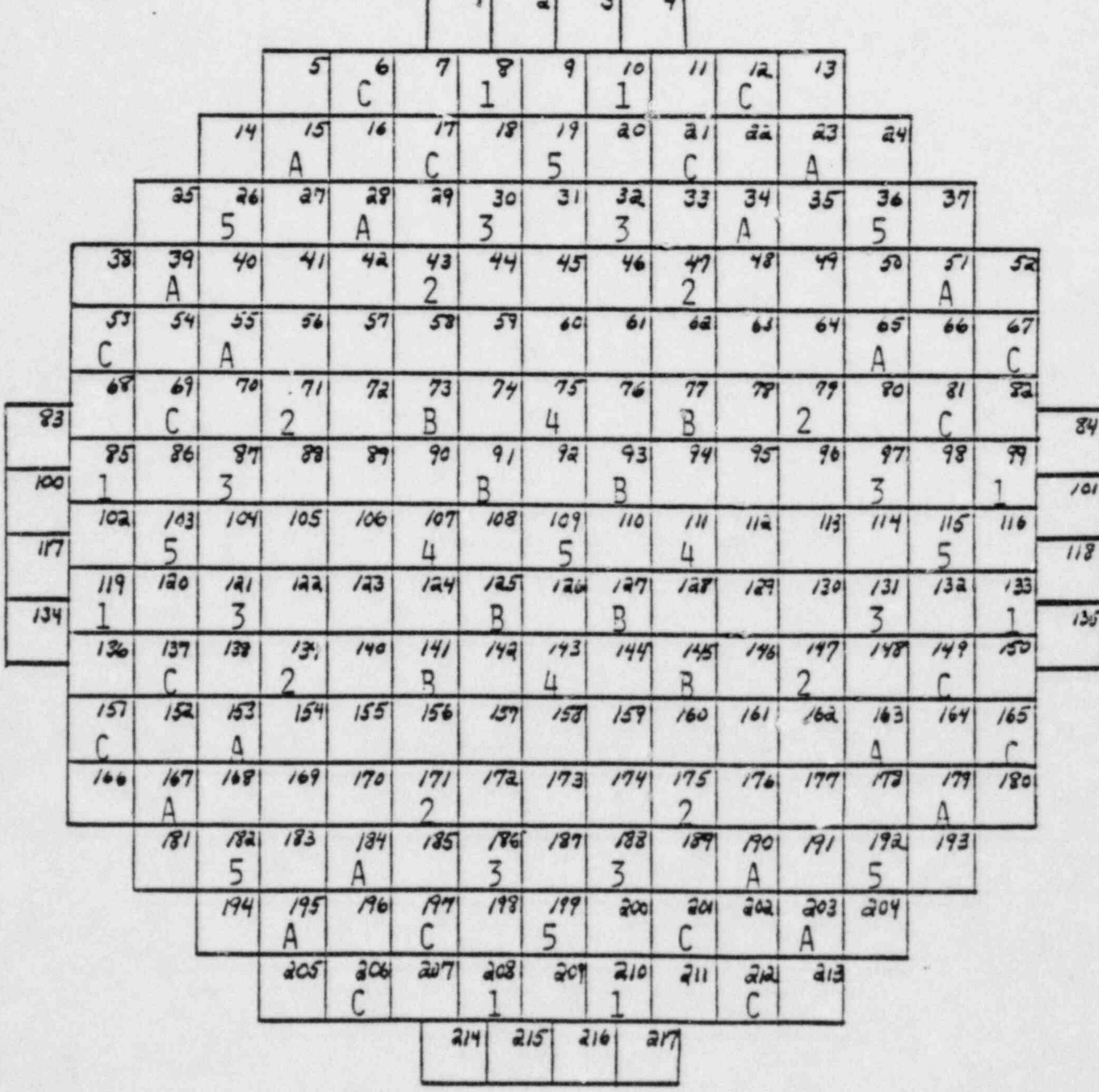
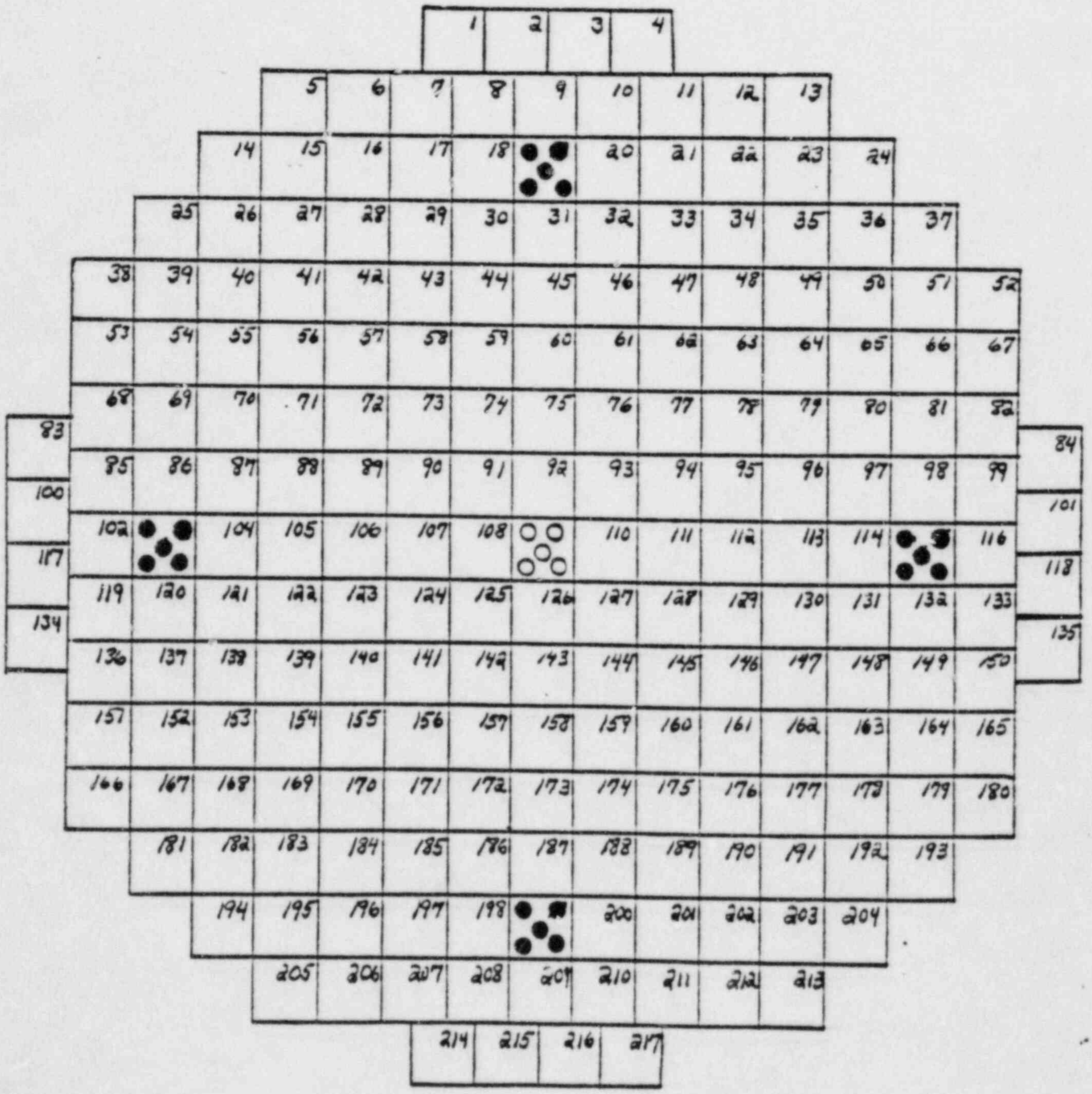


Figure 3-8

CALVERT CLIFFS UNIT 1 CYCLE 8

BANK-5 CONFIGURATION



● B_4C

○ Al_2O_3

Figure 3-9

CALVERT CLIFFS UNIT 1 CYCLE 8

BANK-4 CONFIGURATION

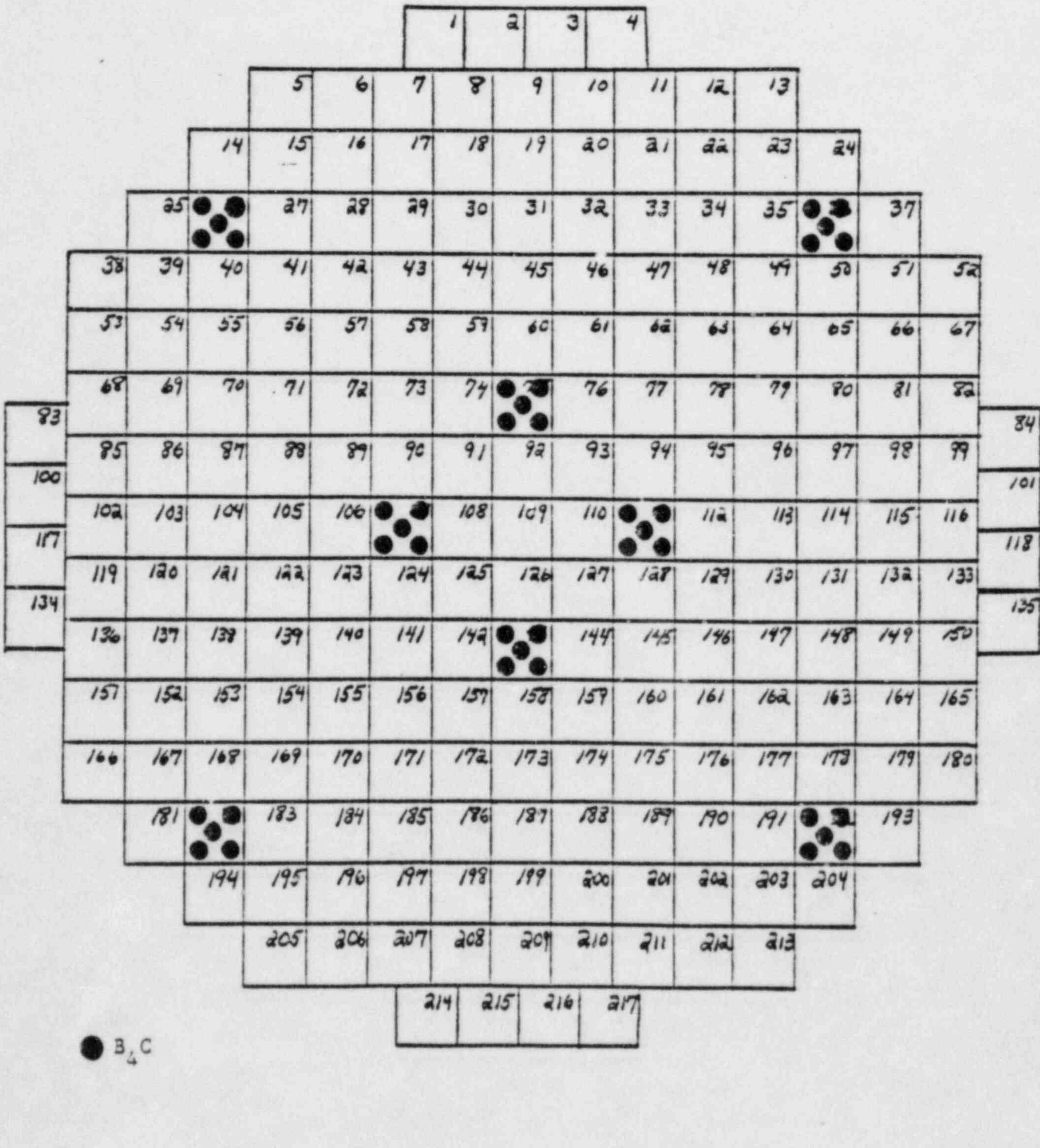
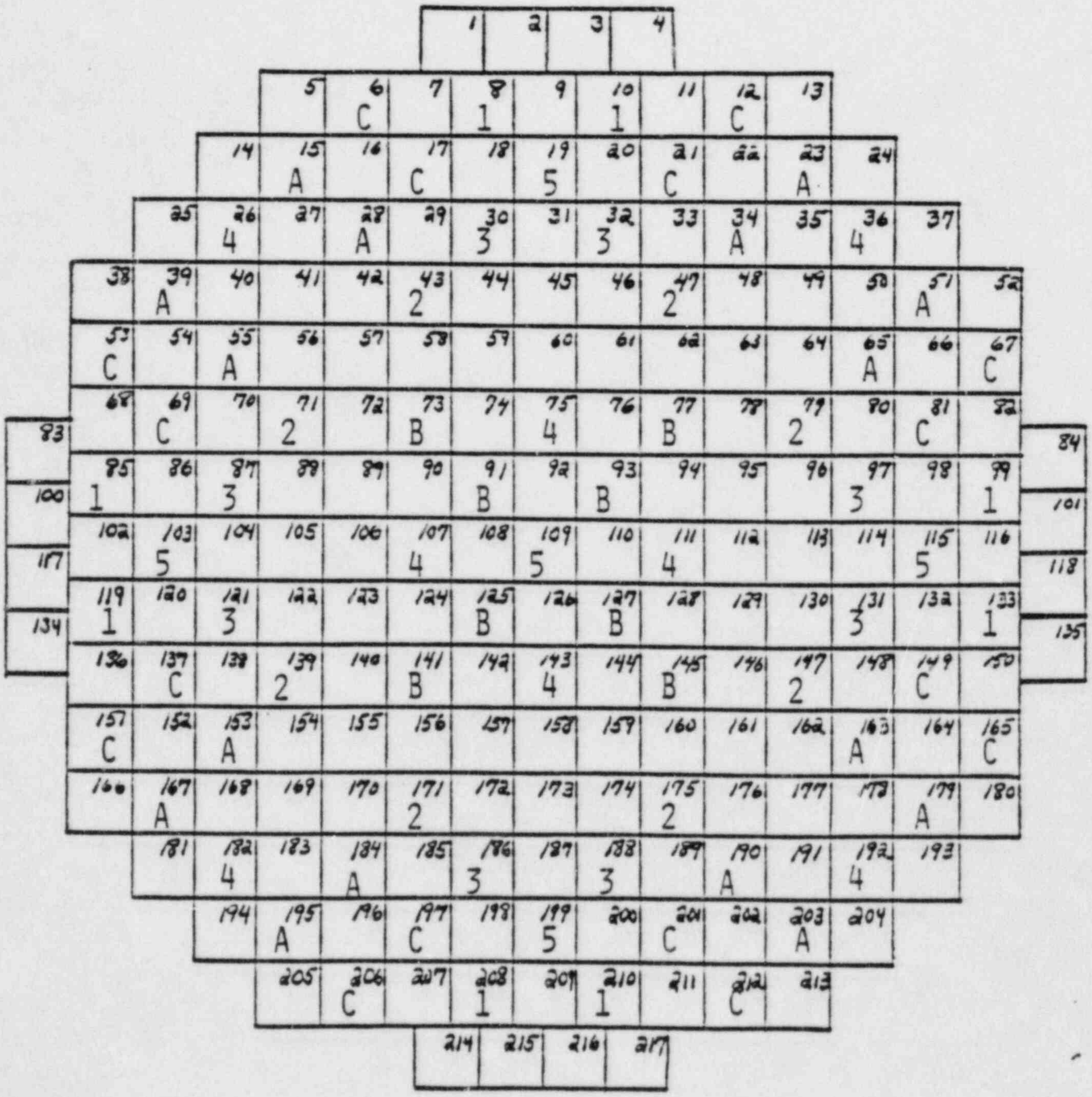


Figure 3-10

CALVERT CLIFFS UNIT 1 CYCLE 8

CEA BANK PATTERN



4.0 FUEL SYSTEM DESIGN

4.1 Mechanical Design

4.1.1 Fuel Design

The mechanical design for the Batch K reload fuel is identical to that of the Batch J fuel described in the reference cycle submittal (Calvert Cliffs Unit 1 Cycle 7, Reference 1), with the exception of the design features listed below.

a. The height of the lower end fitting is shorter. This reduction is achieved by shortening the legs of the lower end fitting assembly.

b. The overall lengths of the guide tubes are increased to compensate for the shorter lower end fitting described in a. This increase is achieved by increasing the length of the buffer region, i.e., tapered region. The combination of this shorter lower end fitting and the longer guide tubes maintains the same overall assembly length as that of the Batch J fuel.

c. The elevations of the Inconel grid and the uppermost Zircaloy grid are changed to maintain their same relative elevations with respect to fuel rods as those of the reference cycle fuel design.

The changes described above were analyzed and found to have no significant adverse effect on the performance of the Batch K fuel relative to that of the Batch J fuel. These changes will result in improved performance by increasing the shoulder gap from 1.400 inches to 1.775 inches.

The mechanical designs of the Calvert Cliffs Unit 1 Batch F, G, and H fuel assemblies were described in References 2, 3, and 4, respectively. Details of the Calvert Cliffs Unit 2 Batch E/ fuel assemblies that will be used in Cycle 8 can be found in Reference 5.

4.1.2 Clad Collapse

C-E recently completed an EPRI-sponsored reassessment of the phenomena of interpellet gap formation and clad collapse in modern PWR fuel rods (i.e., nondensifying fuel in prepressurized tubes). The report concluded that the collapse time for modern fuel is significantly larger than its expected useful life. This conclusion was based upon both empirical data covering several vendors' fuel and an analytical evaluation of the propensity for clad collapse into a postulated gap of finite length. A draft copy of this report was submitted to the NRC for evaluation as Attachment 5 of Reference 6. A synopsis of this report focusing on C-E manufactured modern fuel was submitted along with this draft copy as Attachment 4 of Reference 6. Based upon the conclusion and recommendation of Attachment 4 (Reference 6) that cycle specific clad collapse analyses are not necessary for modern C-E manufactured fuel, a cycle specific calculation has not been prepared for Unit 1 Cycle 8.

There will be four test rods in the SCOUT assembly in Cycle 8 which have gap regions in the active core of sufficient length to require evaluation. Such an evaluation has been completed which shows that the minimum collapse time for these rods (52,000 EFPH) exceeds the cumulative exposure at the end of Cycle 8 (49,600 EFPH). The calculations for these test rods utilized the finite gap version of the CEPAN computer described in Attachment 5 of Reference 6.

4.1.3 Dimensional Changes

All standard fuel assemblies in Cycle 8 were reviewed for shoulder gap clearance using the SIGREEP model described in Reference 7 (approved in Reference 8) and for fuel assembly length clearance using the refined correlation discussed in References 1 and 9. All clearances were found to be adequate for Cycle 8. The clearance for fuel rod growth in the SCOUT and PROTOTYPE assemblies will be evaluated during the cycle 7 outage and modified if necessary.

4.1.4 CEA Design

The replacement CEAs to be utilized for the changes described in Section 3.3 have essentially the same design as the original components (Reference FSAR) with the exception that all will include reconstitutable features which are similar to those used in a Calvert Cliffs Unit 2 demonstration CEA (Reference 10). This reconstitutable design will also be used for the replacement of discharged CEAs.

The full strength replacement CEA will use Ag-In-Cd pellets at the tip of all five control rods; the previous design used Ag-In-Cd pellets at the tip of just the four outer control rods. The very weak replacement CEA for the core center location will contain only Al_2O_3 and Zircaloy pellets in lieu of B_4C and Ag-In-Cd pellets.

4.1.5 Removal of CEA Plugs

Unit 1 is presently operating with CEA plugs installed in the locations originally occupied by Part Length Rods (PLRs). These CEA plugs will be removed for Cycle 8 to facilitate the installation of the Reactor Vessel Level Monitoring System and to expedite refueling outage operations. An assessment of the effects that removing these CEA plugs would produce has been completed. This assessment concluded that the removal of CEA plugs from all eight PLR positions can be effected safely for both Calvert Cliffs units.

4.1.6 Metallurgical Requirements

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for the Batch K fuel are identical to those of the other fuel batches to be included in Cycle 8. Thus, the chemical or metallurgical performance of the Batch K fuel will remain unchanged from that the Unit 1 Cycle 7 fuel.

4.2 <u>Hardware Modifications to Mitigate Guide Tube Wear</u>

All standard fuel assemblies which will be placed in CEA locations in Cycle 8 will have stainless steel sleeves installed in the guide tubes to prevent guide tube wear. A detailed discussion of the design of the sleeves in irradiated fuel assemblies and their effect on reactor operation is contained in Reference 11. A modified short sleeve design will be used in Batch K fuel assemblies. This will allow for reconstitution of the Batch K fuel assemblies without having to remove and, consequently, reinstall the guide tube sleeves. A discussion of the short sleeve design is contained in Reference 12.

Cycle 8 will also utilize one fuel assembly (Batch F, SCOUT) in a CEA location that was fabricated with modified guide tubes (see Reference 2) instead of sleeves to mitigate guide tube wear. This modified assembly has previously resided in a CEA position for two cycles. An examination for guide tube wear was conducted after one cycle of residence in a CEA position. The test results presented in Reference 13 showed no detectable wear.

4.3 Thermal Design

The thermal performance of a composite fuel pin which envelopes the various fuel assemblies present in Cycle 8 (fuel Batches F, G, H, J, and K and Batch E from Unit 2) has been evaluated using the FATES3 version of the fuel evaluation model (References 14 and 15), as approved by the NRC (Reference 16). The analysis was performed with a history that modeled the power and burnup levels representative of the peak pin at each burnup interval, from beginning of cycle to end of cycle burnups. The burnup range analyzed is in excess of that expected at end of Cycle 8. In addition, the SCOUT and PROTOTYPE test pins were analyzed and found to be bounded in both temperature and pressure by the standard fuel batches in Cycle 8. Consequently, the test pins are not limiting with respect to thermal performance.

The augmentation factor is being removed from the Tech. Specs. and the values of several items used in the Incore Monitoring System, i.e., measurement-calculational uncertainty and axial fuel densification and thermal expansion factor, are being lowered (See Section 9.0). These changes when coupled with an unchanging LOCA limit present the potential for core operation at a higher steady state local linear heat rate. This higher local power level which can result in more adverse fuel performance was included in the Cycle 8 analysis.

5.0 NUCLEAR DESIGN

5.1 Physics Characteristics

5.1.1 Fuel Management

The Cycle 8 fuel management employs a mixed central region as described in Section 3, Figure 3-1. The fresh Batch K fuel is comprised of two sets of assemblies, each having a unique enrichment in order to minimize radial power peaking. There are 48 assemblies with an enrichment of 4.05 wt% U-235 and 24 assemblies with an enrichment of 3.40 wt% U-235. With this loading, the Cycle 8 burnup capacity for full power operation is expected to be between 13,700 MWD/T and 14,400 MWD/T, depending on the final Cycle 7 termination point. The Cycle 8 core characteristics have been examined for Cycle 7 terminations between 12,900 and 13,900 MWD/T and limiting values established for the safety analyses. The loading pattern (see Section 3) is applicable to any Cycle 7 termination point between the stated extremes.

Physics characteristics including reactivity coefficients for Cycle 8 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 8 zero power steam line break accident and a comparison to reference cycle data. The EOC zero power steam line accident was selected since it is the most limiting zero power transient with respect to reactivity requirements and, thus, provides the basis for verifying the Technical Specification required shutdown margin.

Table 5-3 shows the reactivity worths of the three CEA groups which are allowed in the core during critical/power conditions. These reactivity worths were calculated at full power conditions for Cycle 8 and the reference cycle. The configurations of CEA Groups 5 and 4 have been changed as described in Section 3; the configuration of Group 3 remains the same as in the reference cycle. The power dependent insertion limit (PDIL) curve is the same as for the reference cycle.

5.1.2 Power Distributions

Figures 5-1 through 5-3 illustrate the all rods out (ARO) planar radial power distributions at BOC8, MOC8 and EOC8, respectively, that are characteristic of the high burnup end of the Cycle 7 shutdown window. These planar radial power peaks are characteristic of the major portion of the active core length between about 20 and 80 percent of the fuel height. The high burnup end of the Cycle 7 shutdown window tends to increase the power peaking in this axial central region of the core for Cycle 8. The planar radial power distributions for the above region with CEA Group 5 fully inserted at beginning and end of Cycle 8 are shown in Figures 5-4 and 5-5, respectively, for the high burnup end of the Cycle 7 shutdown window.

The radial power distributions described in this section are calculated data without uncertainties or other allowances. However, the single rod power peaking values do include the increased peaking that is characteristic of fuel rods adjoining the water holes in the fuel assembly lattice. For both DNB and kw/ft safety and setpoint analyses in either

rodded or unrodded configurations, the power peaking values actually used are higher than those expected to occur at any time during Cycle 8. 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 (LCOs) 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 8 during normal base load, all rods out operation at full power is 1.92, not including uncertainty allowances.

5.1.3 Safety Related Data

5.1.3.1 Ejected CEA Data

The maximum reactivity worths and planar power peaks associated with an Ejected CEA Event are shown in Table 5-4 for Cycle 8 and the reference cycle. These values encompass the worst conditions anticipated during Cycle 8 for any expected Cycle 7 termination point. The values shown for Cycle 8 are the safety analysis values which are conservative with respect to the actual calculated values. The data for the full power condition remained unchanged relative to the reference cycle; however, the data for the zero power condition was revised due to the change in CEA configuration discussed in Section 3.

5.1.3.2 Dropped CEA Data

The Cycle 8 safety related data for this section are identical to the safety related data used in the reference cycle.

5.1.3.3 Augmentation Factors

Recently completed analyses (Reference 1) have demonstrated that the increased power peaking associated with the small interpellet gaps found in modern, i.e, pre-pressurized and non-densifying, fuel is insignificant compared to the uncertainties in the safety analyses and Tech. Specs. Consequently, augmentation factors are being eliminated from reload analyses (See Sections 4.3, 7.0 and 8.1) and the Tech. Specs. (See Section 9.0).

5.1.3.4 Fuel Temperature Coefficient Bias

A negative bias of 15% is being added to the Fuel Temperature Coefficient (FTC) data used in the safety analyses to estalish consistency with the bias on Power Coefficient presented in the ROCS/DIT Topical (Reference 2). This negative bias is being used conservatively by selective application, i.e., FTC data which is bounding in the more negative direction is being adjusted to be even more negative while the FTC data which is bounding in the less negative direction remains unadjusted. This application procedure when combined with the standard ± 15% uncertainty results in a negative adjustment of 30% for the more negatively bounding data and a positive adjustment of 15% for the less negatively bounding data.

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

5.3 Nuclear Design Methodology

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

5.4 Uncertainties in Measured Power Distributions

The power distribution measurement uncertainties which are applied to Cycle 8 are the same as those applied to the reference cycle.

TABLE 5-1

CALVERT CLIFFS UNIT 1 CYCLE 8
NOMINAL PHYSICS CHARACTERISTICS

	Units	Reference Cycle (Unit 1 Cycle 7)	Cycle 8
Dissolved Boron			
Hot Full Power, All Rods Out Equilibrium Xenon Boron Content for Criticality at BOC	PPM	1090+	1200
Boron Worth			
Hot Full Power BOC	PPM/%$\Delta\rho$	105+	107
Hot Full Power EOC	PPM/%$\Delta\rho$	84	88
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients, Hot Full power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta\rho/^{o}F$	-0.2	-0.1
End of Cycle	$10^{-4}\Delta\rho/^{o}F$	-2.2	-2.3
Doppler Coefficient			
Hot Zero Power BOC	$10^{-5}\Delta\rho/^{o}F$	-1.56	-1.56
Hot Full Power BOC	$10^{-5}\Delta\rho/^{o}F$	-1.28	-1.26
Hot Full Power EOC	$10^{-5}\Delta\rho/^{o}F$	-1.45	-1.44
Total Delayed Neutron Fraction, βeff			
BOC		0.00604	0.00604
EOC		0.00522	0.00516
Neutron Generation Time, ℓ*			
BOC	10^{-6} sec	23.4	23.1
EOC	10^{-6} sec	29.8	28.2

+ A slightly incorrect value was reported in Reference 4.

TABLE 5-2

CALVERT CLIFFS UNIT 1 CYCLE 8
LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCES
FOR THE END-OF-CYCLE (EOC) HOT ZERO POWER (HZP)
STEAM LINE RUPTURE ACCIDENT, $\%\Delta\rho$

	Reference Cycle*	Cycle 8
1. Worth of all CEA's Inserted	9.1	9.3
2. Stuck CEA Allowance	2.6	2.6
3. Worth of all CEA's less Worth of CEA Stuck Out**	6.5	6.7
4. Power Dependent Insertion Limit CEA Bite at Zero Power	1.6	1.8
5. Calculated Scram Worth	4.9	4.9
6. Physics Uncertainty plus Bias	0.6	0.6
7. Net Available Scram Worth	4.3	4.3
8. Technical Specification Shutdown Margin	4.3	3.5
9. Margin in Excess of Technical Specification Shutdown Margin	0.0	0.8

*Unit 1 Cycle 7.

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

TABLE 5-3

CALVERT CLIFFS UNIT 1 CYCLE 8
REACTIVITY WORTH OF CEA REGULATING
GROUPS AT HOT FULL POWER, %$\Delta\rho$

Regulating CEA's	Beginning of Cycle: Reference* Cycle	Beginning of Cycle: Cycle 8**	End of Cycle: Reference* Cycle	End of Cycle: Cycle 8**
Group 5	0.53	0.28	0.64	0.36
Group 4	0.34	0.82	0.44	0.91
Group 3	0.99	0.94	1.07	1.02

Note

Values shown assume sequential group insertion.

* Unit 1 Cycle 7.
** CEA configurations of Groups 5 and 4 have been modified for Cycle 8 as described in Chapter 3.

TABLE 5-4
CALVERT CLIFFS UNIT 1 CYCLE 8
CEA EJECTION DATA

	Limiting Values	
	Reference Cycle Safety Analysis Value*	Unit 1 Cycle 8 Safety Analysis Value
Maximum Radial Power Peak		
Full power with Bank 5 inserted; worst CEA ejected	3.6	3.6
Zero power with Banks 5+4+3 inserted; worst CEA ejected	9.4	9.5
Maximum Ejected CEA Worth (%Δρ)		
Full power with Bank 5 inserted; worst CEA ejected	0.28	0.28
Zero power with Banks 5+4+3 inserted; worst CEA ejected	0.63	0.86

* Unit 1 Cycle 7

Notes

1. Uncertainties and allowances are included in the above data.
2. The Cycle 8 safety analysis values are conservative with respect to the actual Cycle 8 calculated values.

						1 0.76	2 1.00	
				3 0.81	4 1.09	5 1.20 X	6 0.87	7 1.08
			8 0.91	9 1.20	10 1.00	11 0.96	12 0.79	13 0.82
		14 0.91	15 1.16	16 1.11	17 1.26	18 0.86	19 1.26	20 0.95
	21 0.80	22 1.19	23 1.06	24 1.20	25 0.89	26 1.17	27 0.98	28 1.16
	29 1.07	30 0.98	31 1.23	32 0.86	33 1.24	34 0.85	35 1.23	36 0.72
	37 1.18	38 0.94	39 0.84	40 1.15	41 0.84	42 1.06	43 0.92	44 1.01
45 0.75	46 0.86	47 0.78	48 1.25	49 0.97	50 1.21	51 0.89	52 1.15	53 0.69
54 0.99	55 1.08	56 0.82	57 0.95	58 1.16	59 0.72	60 1.01	61 0.69	62 0.46

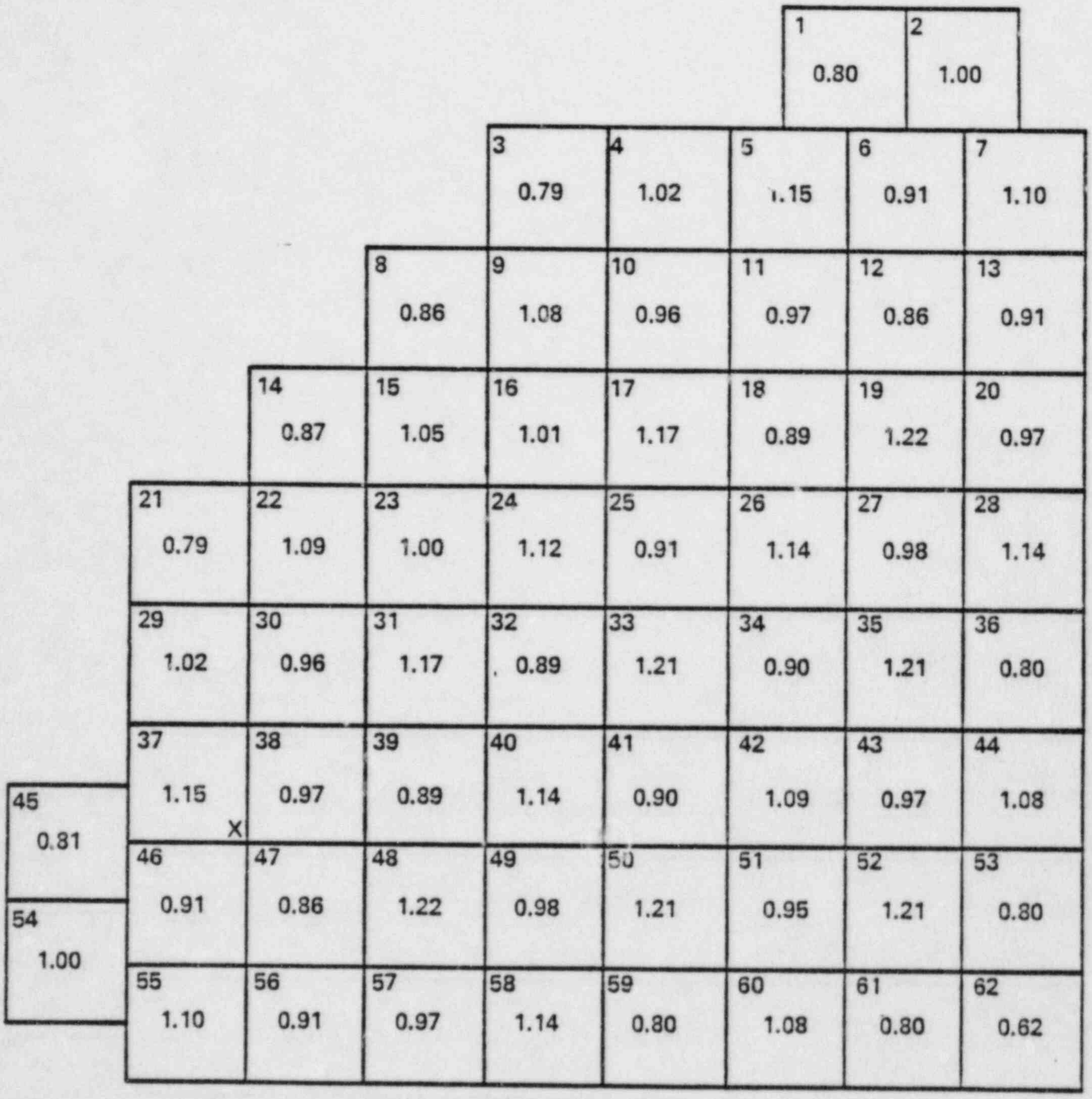
NOTE: X = MAXIMUM 1-PIN PEAK = 1.54

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 8 ASSEMBLY RELATIVE POWER DENSITY AT BOC, EQUILIBRIUM XENON	Figure 5-1

						1 0.77	2 0.99	
				3 0.78	4 1.03	5 1.15	6 0.89	7 1.10
			8 0.86	9 1.12	10 0.97	11 0.96	12 0.83	13 0.87
		14 0.87	15 1.08	16 1.05	17 1.21	18 0.87	19 1.24	20 0.97
	21 0.78	22 1.12	23 1.02	24 1.15	25 0.90	26 1.16	27 0.98	28 1.17
	29 1.03	30 0.96	31 1.19	32 0.88	33 1.24	34 0.88	35 1.24	36 0.78
45 0.78	37 1.16 X	38 0.96	39 0.87	40 1.16	41 0.88	42 1.10	43 0.96	44 1.07
54 1.00	46 0.89	47 0.82	48 1.25	49 0.99	50 1.24	51 0.94	52 1.21	53 0.76
	55 1.10	56 0.87	57 0.97	58 1.17	59 0.78	60 1.07	61 0.76	62 0.55

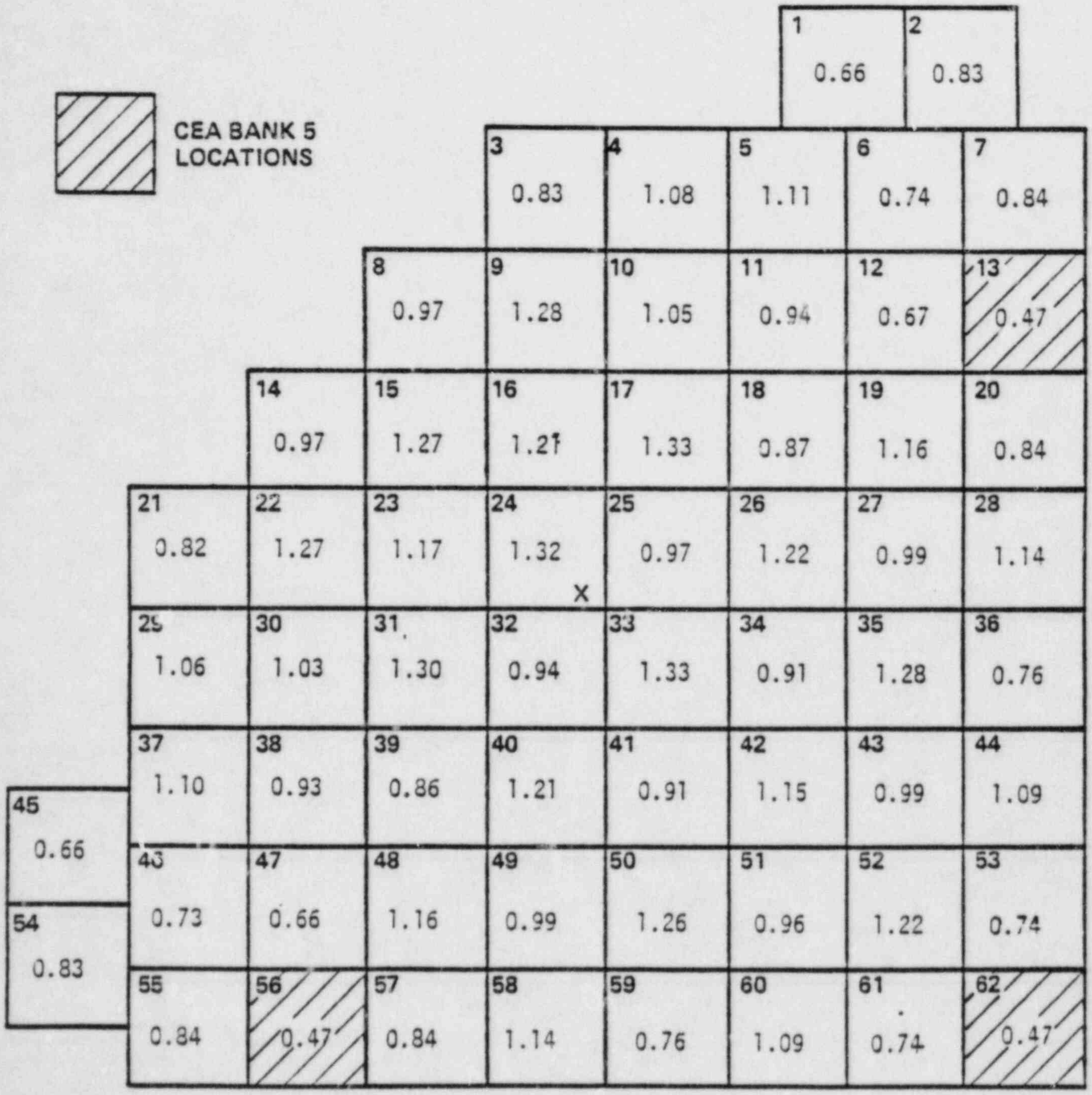
NOTE: X = MAXIMUM 1-PIN PEAK = 1.45

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 8 ASSEMBLY RELATIVE POWER DENSITY AT 7 GWD/T EQUILIBRIUM XENON	Figure 5-2



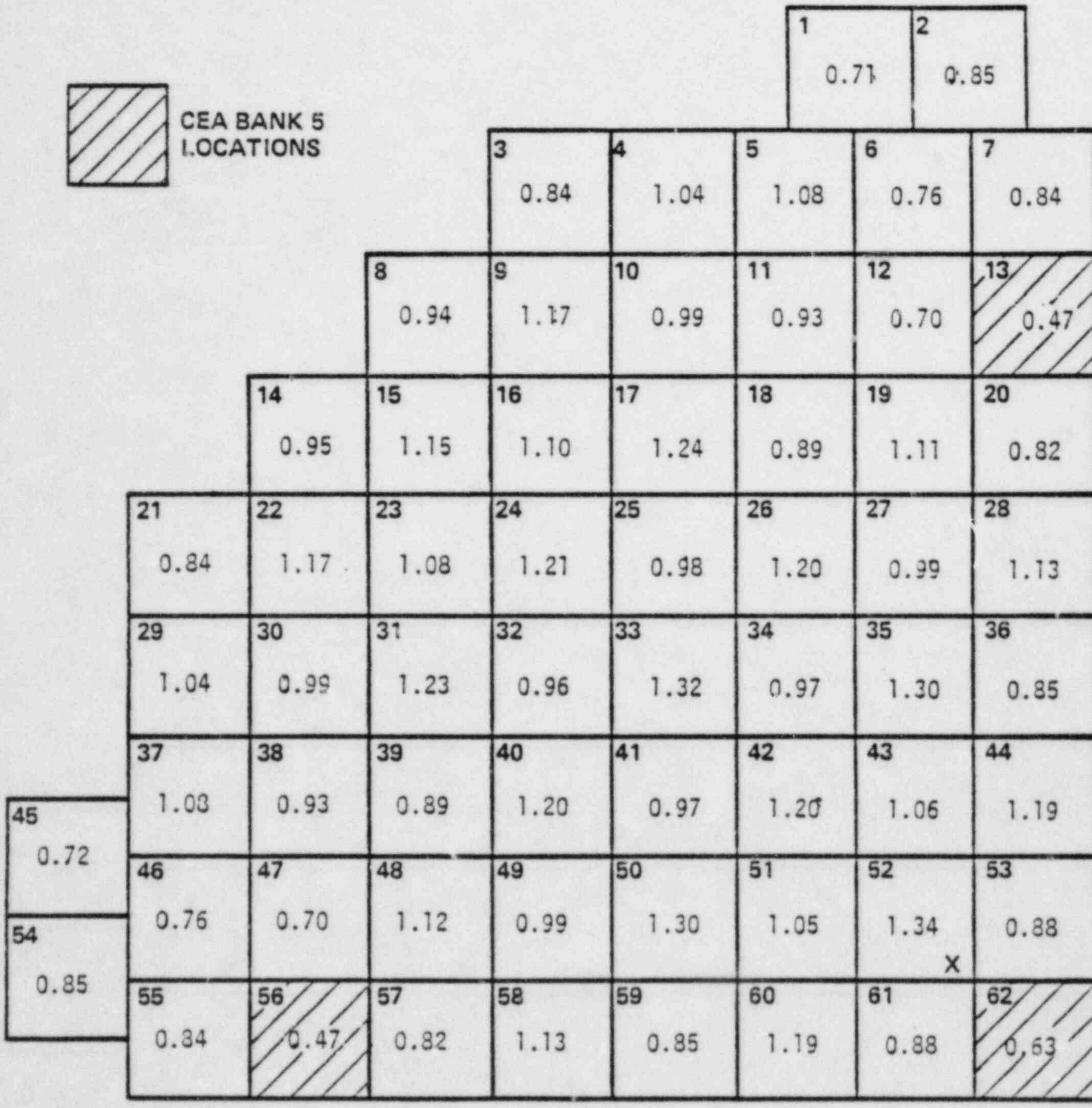
NOTE: X = MAXIMUM 1-PIN PEAK = 1.40

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 8 ASSEMBLY RELATIVE POWER DENSITY AT EOC, EQUILIBRIUM XENON	Figure 5-3



NOTE: X = MAXIMUM 1-PIN PEAK = 1.61

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT I CYCLE 8 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, BOC	Figure 5-4.



NOTE: X = MAXIMUM 1-PIN PEAK = 1.50

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 8 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, EOC	Figure 5-5

6.0 THERMAL HYDRAULIC DESIGN

6.1 DNBR Analysis

Steady state DNBR analyses of Cycle 8 at the rated power level of 2700 MWt have been performed using the TORC computer code described in Reference 1, the CE-1 critical heat flux correlation described in Reference 2, and the simplified modeling methods described in Reference 3.

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

Table 6-1 contains a list of pertinent thermal-hydraulic design parameters applicable to both safety analyses and the generation of reactor protective system setpoint information. The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) listed in Table 6-1 have been combined statistically with other uncertainty factors at the 95/95 confidence/probability level (Reference 6) to define a design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6 and approved by the NRC in Reference 5. The applicability of this minimum DNBR limit was verified for Cycle 8.

6.2 Effects of Fuel Bowing on DNBR Margin

The effects of fuel rod bowing on DNB margin for Calvert Cliffs Unit 1 Cycle 8 have been evaluated using the methods described in Reference 7. These methods were approved by NRC in Reference 8.

Based upon these methods, a penalty of 0.3% DNBR is required to account for the adverse T-H effects of rod bow at an assembly average burnup of 30 GWD/T. An equivalent penalty of 0.4% in radial peak was applied in the determination of the Tech. Spec. limit on radial peak. A conservative (i.e., maximum over all operating ranges) conversion factor of -1.2% radial peak / % DNBR was used to determine the equivalent radial peak penalty.

For those assemblies with an assembly average burnup in excess of 30 GWD/T, the minimum best estimate margin available relative to more limiting peaking values present in other assemblies is greater than 5%, exceeding the corresponding rod bow penalties based upon Reference 7. Hence, sufficient available margin exists to offset rod bow penalties for assemblies with burnup greater than 30 GWD/T.

TABLE 6-1

CALVERT CLIFFS UNIT 1
THERMAL-HYDRAULIC PARAMETERS AT FULL POWER**

General Characteristics	Unit	Reference+ Unit 1, Cycle 7	Cycle 8
Total Heat Output (core only)	MWt 10^6 BTU/hr	2700 9215	2700 9215
Fraction of Heat Generated In Fuel Rod		.975	.975
Primary System Pressure (Nominal)	psia	2250	2250
Inlet Temperature	°F	548	548
Total Reactor Coolant Flow (steady state)	gpm 10^6 lb/hr	381,600 143.8	381,600 143.8
Coolant Flow Through Core	10^6 lb/hr	138.5	138.5
Hydraulic Diameter (nominal channel)	ft	0.044	0.044
Average Mass Velocity	10^6 lb/hr-ft^2	2.59	2.59
Pressure Drop Across Core (steady state flow irreversible ΔP over entire fuel assembly)	psi	11.1	11.1
Total Pressure Drop Across Vessel (based on steady state flow and nominal dimensions)	psi	34.7	34.7
Core Average Heat Flux (Accounts for above fraction of heat generated in fuel rod and axial densification factor)	BTU/hr-ft^2	183,000***	182,300****
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	49,100***	49,300****
Film Coefficient at Average Conditions	BTU/hr-ft^2-°F	5930	5930

TABLE 6-1
(continued)

General Characteristics	Unit	Reference+ Unit 1, Cycle 7	Cycle 8
Average Film Temperature Difference	°F	31	31
Average Linear Heat Rate of Undensified Fuel Rod (accounts for above fraction of heat generated in fuel rod)	kw/ft	6.12***	6.09****
Average Core Enthalpy Rise	BTU/lb	66.5	66.5
Maximum Clad Surface Temperature	°F	657	657

Calculational Factors	Reference+ Unit 1, Cycle 7	Cycle 8
Engineering Heat Flux on Hot Channel	1.03*	1.03*
Engineering Factor on Hot Channel Heat Input	1.02*	1.02*
Rod Pitch and Clad Diameter Factor	1.065*	1.065*
Fuel Densification Factor (axial)	1.01++	1.01++

Notes

*These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level (Reference 6) to define a design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6 and approved by the NRC in Reference 5. This limit was verified to be applicable to Cycle 8.

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

***Based on a value of 400 shims and 5 non-fuel rods.

****Based on a value of 256 shims and 5 non-fuel rods.

+Reference cycle (Unit 1, Cycle 7) analysis is contained in Reference 11.

++This value is conservative with respect to existing calculations.

7.0 Transient Analysis

This section presents the results of the Baltimore Gas & Electric Calvert Cliffs Unit 1, Cycle 8 non-LOCA safety analysis.

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

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

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

3. Postulated Accidents

A re-evaluation of all DBEs was performed to determine the impact of the following changes.

a. Fuel temperature coefficient (FTC) multiplier changed from ±15% to -15%, +30% (see Section 5.1.3.4).

b. Opening pressure setpoint for SG safety valves assumed for the non-LOCA safety analysis (see Chapter 9)

Bank 1	Changes from 1000 psia to 1050 psia*
Bank 2	Changes from 1015 psia to 1050 psia
Bank 3	Changes from 1030 psia to 1080 psia
Bank 4	Changes from 1050 psia to 1080 psia

c. Moderator Temperature Coefficient (MTC) Range

Positive MTC range was changed from +0.5 to $+0.7 \times 15^{-4} \Delta\rho / ^{o}F$
Negative MTC range was changed from -2.5 to $-2.7 \times 10^{-4} \Delta\rho / ^{o}F$

d. Shutdown margin was reduced from 4.3 to 3.5%$\Delta\rho$.

e. Proposed Tech. Specs. change to HPSI pump flow rate (see Chapter 9).

Table 7-2 summarizes the core parameters assumed in the Unit 1 Cycle 8 transient analysis and compares them to the values used in the Reference Cycle. Specific initial conditions for each event are tabulated in that event's section of the report. For some DBEs presented, certain initial core parameters were assumed to be more limiting than the actual calculated Cycle 8 values (e.g., CEA worth at trip). This was done to bound future cycles.

For all DBEs that have results bounded by the Reference Cycle, the margin of safety has not degraded from that of the Reference Cycle. Those events whose results were not bounded by the Reference Cycle, are presented herein.

*The value assumed is conservative with respect to the Technical Specification limit.

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

TABLE 7-1

CALVERT CLIFFS UNIT 1, CYCLE 8
DESIGN BASIS EVENTS CONSIDERED IN THE NON-LOCA SAFETY ANALYSIS

	Results
7.1 Anticipated Operational Occurrences for which intervention of the RPS is necessary to prevent exceeding acceptable limits:	
7.1.1 Boron Dilution	Presented
7.1.2 Startup of an Inactive Reactor Coolant Pump[1]	Bounded by Reference Cycle
7.1.3 Loss of Load	Presented
7.1.4 Excess Load	Presented
7.1.5 Loss of Feedwater Flow	Bounded by Reference Cycle
7.1.6 Excess Heat Removal Due to Feedwater Malfunction	Bounded by Reference Cycle
7.1.7 Reactor Coolant System Depressurization	Bounded by Reference Cycle
7.1.8 Excessive Charging Event	Bounded by Reference Cycle
7.2 Anticipated Operational Occurrences for which RPS trips and/or sufficient initial steady state thermal margin, maintained by the LCOs, are necessary to prevent exceeding the acceptable limits:	
7.2.1 Sequential CEA Group Withdrawal[2]	Bounded by Reference Cycle
7.2.2 Loss of Coolant Flow[3]	Bounded by Reference Cycle
7.2.3 Full Length CEA Drop	Bounded by Reference Cycle
7.2.4 Transients Resulting from the Malfunction of One Steam Generator[4]	Presented
7.2.5 Loss of AC Power[3]	Bounded by Reference Cycle
7.3 Postulated Accidents	
7.3.1 CEA Ejection	Presented
7.3.2 Steam Line Rupture	Presented
7.3.3 Steam Generator Tube Rupture	Bounded by Reference Cycle
7.3.4 Seized Rotor[3]	Bounded by Reference Cycle

[1]Technical Specifications preclude this event during operation.

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

[3]Requires Low Flow Trip.

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

TABLE 7-2

CALVERT CLIFFS UNIT 1, CYCLE 8
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Physics Parameters	Units	Reference Cycle Values** (Unit 1, Cycle 7)	Unit 1, Cycle 8 Values
Radial Peaking Factors			
For DNB Margin Analyses (F_r^T)			
Unrodded Region		1.70+,*	1.70+,*
Bank 5 Inserted		1.87+,*	1.87+,*
For Planar Radial Component (F_{xy}^T) of 3-D Peak (CTM Limit Analyses)			
Unrodded Region		1.70*	1.70*
Bank 5 Inserted		1.87*	1.87*
Maximum Augmentation Factor		1.055	1.0
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^{\circ}F$	-2.5 → +.5	-2.7 → +.7
Shutdown Margin	%$\Delta\rho$	-4.3	-3.5
Tilt Allowance	%	3.0	3.0

*For DNBR and CTM calculations, effects of uncertainties on these parameters were accounted for statistically. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2, 3, and 4. These procedures have been approved by NRC for the Calvert Cliffs Units in Reference 5.

**Reference 1.

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

TABLE 7-2
(continued)

Physics Parameters	Units	Reference Cycle Values (Unit 1, Cycle 7)	Unit 1, Cycle 8 Values
Power Level	MWt	2700*	2700*
Maximum Steady State Temperature (T_{in})	^oF	548*	548*
Minimum Steady State RCS Pressure	psia	2200*	2200*
Reactor Coolant Flow	10^6lbm/hr	138.5*	138.5*
Negative Axial Shape LCO Extreme Assumed at Full Power (Ex-Cores)	I_p	-.15*	-.15*+
Maximum CEA Insertion at Full Power	% Insertion of Bank 5	25	25
Maximum Initial Linear Heat Rate for Transients Other than LOCA	KW/ft	16.0	16.0
Steady State Linear Heat Rate for Fuel CTM Assumed in the Safety Analysis	KW/ft	22.0	22.0
CEA Drop Time from Removal of Power to Holding Coils to 90% Insertion	sec	3.1	3.1
Minimum DNBR (CE-1)		1.23*	1.23*

*For DNBR and CTM calculations, effects of uncertainties on these parameters were accounted for statistically. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2, 3, and 4. These procedures have been approved by NRC for the Calvert Cliffs Units in Reference 5.

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

Table 7-3

DESIGN BASIS EVENT PRESENTED FOR UNIT 1 CYCLE 8

Event	Reason for Reanalysis *(changes relative to reference cycle)	Acceptance Criterion	Summary of Results
Boron Dilution	Decrease in Shutdown Margin	Time to Criticality no less than 15 minutes for Modes 2, 3 and 4.	Results acceptable. Further details in Section 7.1.1.
Loss of Load	Increase in Moderator Temperature Coefficient (MTC) and increase in opening pressure setpoints of SG safety valves.	Peak RCS pressure less than 2750 psia	Peak RCS pressure calculated to be <2750 psia. Further details in Section 7.1.3.
Excess Load	Decrease in negative MTC and change in HPSI pump flow.	DNBR and CTM SAFDL's not exceeded.	Results acceptable. Further details in Section 7.1.4
Transients Resulting from the Malfunction of One Steam Generator	Decrease in negative MTC and increase in opening pressure setpoints of SG safety valves.	DNBR and CTM SAFDL's not exceeded.	Results acceptable. Further details in Section 7.2.4
CEA Ejection	Increase in post ejected 3-D peak and ejected CEA worth for hot zero power case	Total average enthalpy < 200 cal/gm. Total centerline enthalpy < 310 cal/gm.	Results show no pin experiences clad damage or incipient centerline melting. Further details in Section 7.3.1
Steam Line Rupture (Inside Containment)	Changes in moderator cooldown curve and available scram worth at trip, and change in HPSI pump flow.	Radiological dose is less than 10CFR100.	Site boundary doses are bounded by the outside containment dose calculated for Calvert Cliffs Unit 1 Cycle 7.

*Reference Cycle is specified for each event in the subsequent sections.

7.1.1 BORON DILUTION EVENT

The Boron Dilution event is analyzed for Cycle 8 to demonstrate that sufficient time is available for an operator to identify the cause and to terminate an approach to criticality for subcritical Modes 2, 3, and 4 of operation. This event was reanalyzed on the basis of a reduction in shutdown margin for operational Modes 2, 3, and 4 as shown in Table 7.1.1-1.

An inadvertent boron dilution adds positive reactivity, produces power and temperature increases, and during operation at power (for Mode 1 and 2) can cause an approach to both the DNBR and CTM limits. Since the TM/LP trip system monitors the transient behavior of core power level and core inlet temperature at power, the TM/LP trip will intervene, if necessary, to prevent the DNBR limit from being exceeded for power increase within the setting of the Variable High Power Level trip. For more rapid power excursions the Variable High Power Level trip initiates a reactor trip. The approach to the CTM limit is terminated by either the Local Power Density (ASI) trip, Variable High Power Level trip, or the DNBR related trip discussed above. The trip which is actuated depends on the rate of reactivity increase resulting from the dilution event. For a boron dilution initiated from hot zero power, critical, the power transient resulting from the slow reactivity insertion rate is terminated by the Variable High Power Level trip prior to approaching the limits.

Table 7.1.1-1 compares the values of the key transient parameters assumed in each mode of operation for Cycle 8 and the reference cycle. The conservative input data chosen consists of high critical boron concentrations and low inverse boron worths. These choices produce the most adverse effects by reducing the calculated time to criticality. The time to criticality was determined by using the same expression as in the reference cycle (Unit 1, Cycle 5).

Table 7.1.1-2 compares the results of the analysis for Cycle 8 with those for the Reference Cycle. The key results are the minimum times required to lose prescribed negative reactivity in each operational mode. Modes 2, 3, and 4 results are more limiting than the reference cycle due to a lower shutdown margin. As seen from Table 7.1.1-2, sufficient time exists for the operator to initiate appropriate action to mitigate the consequences of this event.

TABLE 7.1.1-1

KEY PARAMETERS ASSUMED IN THE BORON DILUTION ANALYSIS

Parameter	Unit 1 Cycle 5*	Unit 1 Cycle 8
Critical Boron Concentration, PPM (All Rods Out, Zero Xenon)		
Startup (Mode 2)	1900	1900
Hot Standby (Mode 3)	1900	1900
Hot Shutdown (Mode 4)	1900	1900
Inverse Boron Worth, PPM/%$\Delta\rho$		
Startup	65	65
Hot Standby	55	55
Hot Shutdown	55	55
Minimum Shutdown Margin Assumed, %$\Delta\rho$		
Startup	-4.0	-3.5
Hot Standby	-4.0	-3.5
Hot Shutdown	-4.0	-3.5

*Reference 7.

TABLE 7.1.1-2

RESULTS OF THE BORON DILUTION EVENT

Mode	Time to Lose Prescribed Shutdown Margin (Min)		Criterion for Minimum Time to Lose Prescribed Shutdown Margin (Min)
	Unit 1 Cycle 5	Unit 1 Cycle 8	
Startup	69.8	60	15
Hot Standby	59.6	50	15
Hot Shutdown	59.6	50	15

7.1.3 LOSS OF LOAD EVENT

The Loss of Load event is analyzed to demonstrate that the DNBR limit and the RCS pressure upset limit are not exceeded during Cycle 8. This event was reanalyzed due to an increase in the positive moderator temperature coefficient Technical Specifications limit and to an increase in the opening setpoint of the main steam safety valves.

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

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

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

The Loss of Load event was initiated at the conditions shown in Table 7.1.3-1. The combination of parameters shown in Table 7.1.3-1 maximizes the calculated peak RCS pressure. The methods used to analyze this event are identical to those applied in the reference cycle.

The initial core average axial power distribution for this analysis was assumed to be a bottom peaked shape. This distribution is assumed because it minimizes the negative reactivity inserted during the initial portion of the scram following a reactor trip and maximizes the time required to mitigate the pressure and heat flux increases. A Moderator Temperature Coefficient (MTC) of $+.7 \times 10^{-4}\ \Delta\rho\ /^{o}F$ was conservatively assumed in this analysis at full power despite a Technical Specification limit of $+0.2 \times 10^{-4}\ \Delta\rho\ /^{o}F$. An MTC of $+0.7 \times 10^{-4}\Delta\rho\ /^{o}F$ is allowed by the Technical Specification to exist at or below 70% power. This conservative assumption of full power with the maximum positive MTC was used to bound the event. This MTC in conjunction with the increasing coolant temperatures, enchances the rate of change of heat flux and the pressure at the time of reactor trip. A Fuel Temperature Coefficient (FTC) corresponding to beginning of cycle conditions was used in the analysis. This FTC causes the least amount of negative reactivity feedback to mitigate the transient increases in both the core heat flux and the pressure. The multiplier on the FTC used in the analyses is shown in Table 7.1.3-1. The lower limit on initial RCS pressure less uncertainties is used to maximize the rate of change of pressure, and thus peak pressure, following trip.

The Loss of Load event, initiated from the conditions given in Table 7.1.3-1, results in a high pressurizer pressure trip signal at 5.8 seconds. At 8.8 seconds, the primary pressure reaches its maximum value which is less than 2750 psia. The increase in secondary pressure is limited by the opening of the main steam safety valves, which open at 6.1 seconds. The minimum main steam safety valve opening setpoint is increased from 1000 psia to 1050 psia. This setpoint change results in a maximum secondary pressure of 1095 psia at 12.1 seconds after the initiation of the event.

The event was also analyzed to demonstrate that the acceptable DNBR limit is not violated. The minimum transient DNBR calculated for the event is greater than the DNBR SAFDL.

Table 7.1.3-2 presents the sequence of events for this event. Figures 7.1.3-1 to 7.1.3-5 show the transient behavior of core power, core heat flux, RCS coolant temperatures, the RCS pressure, and the steam generator pressure.

The results of this analysis demonstrate that the Loss of Load event will not result in a DNBR that violates the DNBR SAFDL and that the peak RCS pressures will not exceed the upset pressure limit of 2750 psia.

TABLE 7.1.3-1

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

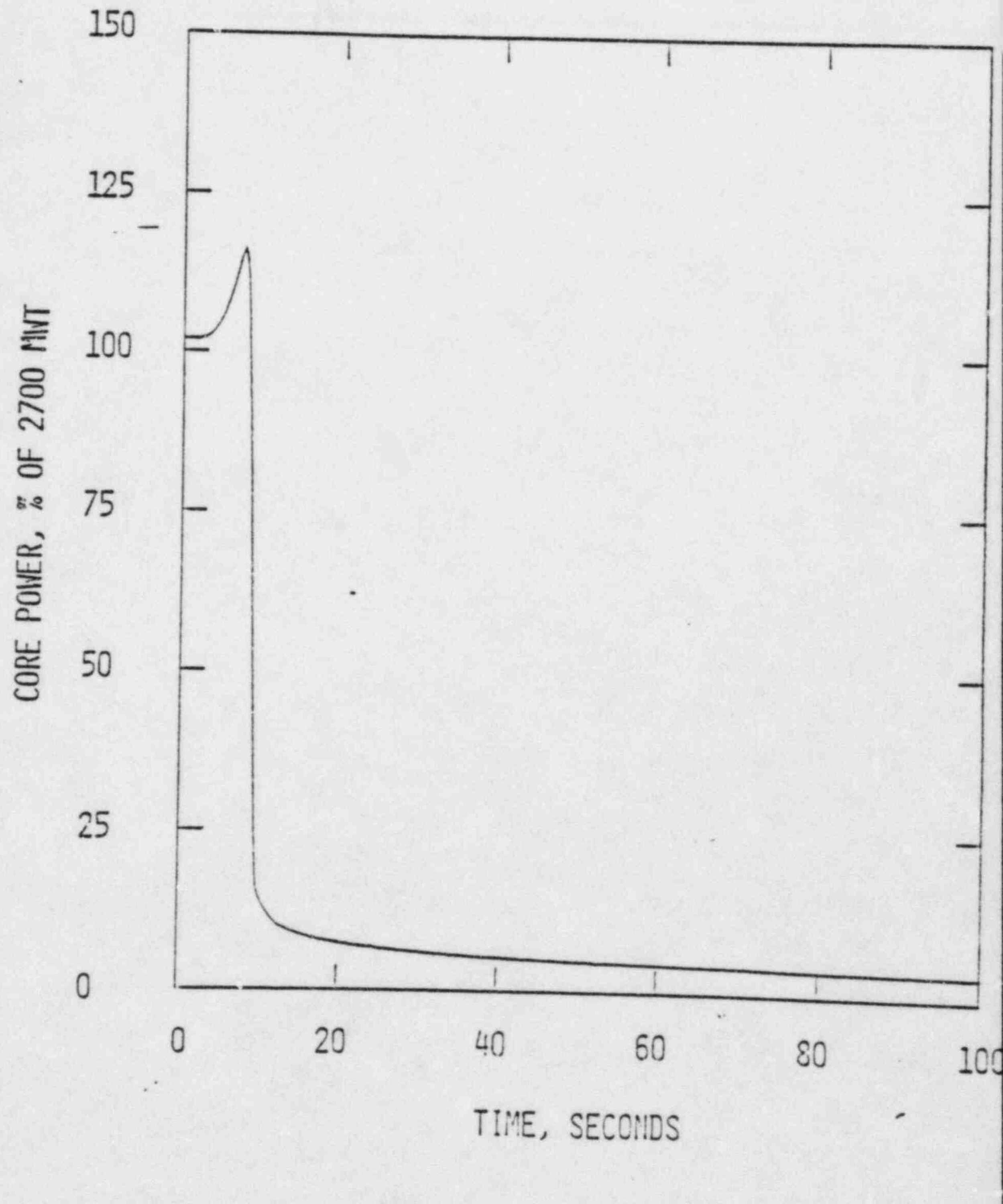
Parameter	Units	Unit 2* Cycle 5	Unit 1 Cycle 8
Initial Core Power Level	MWt	2754	2754
Initial Core Inlet Coolant Temperature	^oF	550	550
Core Coolant Flow	$X10^6$lbm/hr	133.9	133.9
Initial Reactor Coolant System Pressure	psia	2154	2154
Initial Steam Generator Pressure	psia	864	843
Moderator Temperature Coefficient	$X10^{-4}\Delta\rho/^oF$	+.5	+.7
Doppler Coefficient Multiplier	-	.85	.85
CEA Worth at Trip	$\%\Delta\rho$	-4.7	-4.7
Minimum Main Steam Safety Valve Opening Setpoint	psia	1000	1050

*Reference 9.

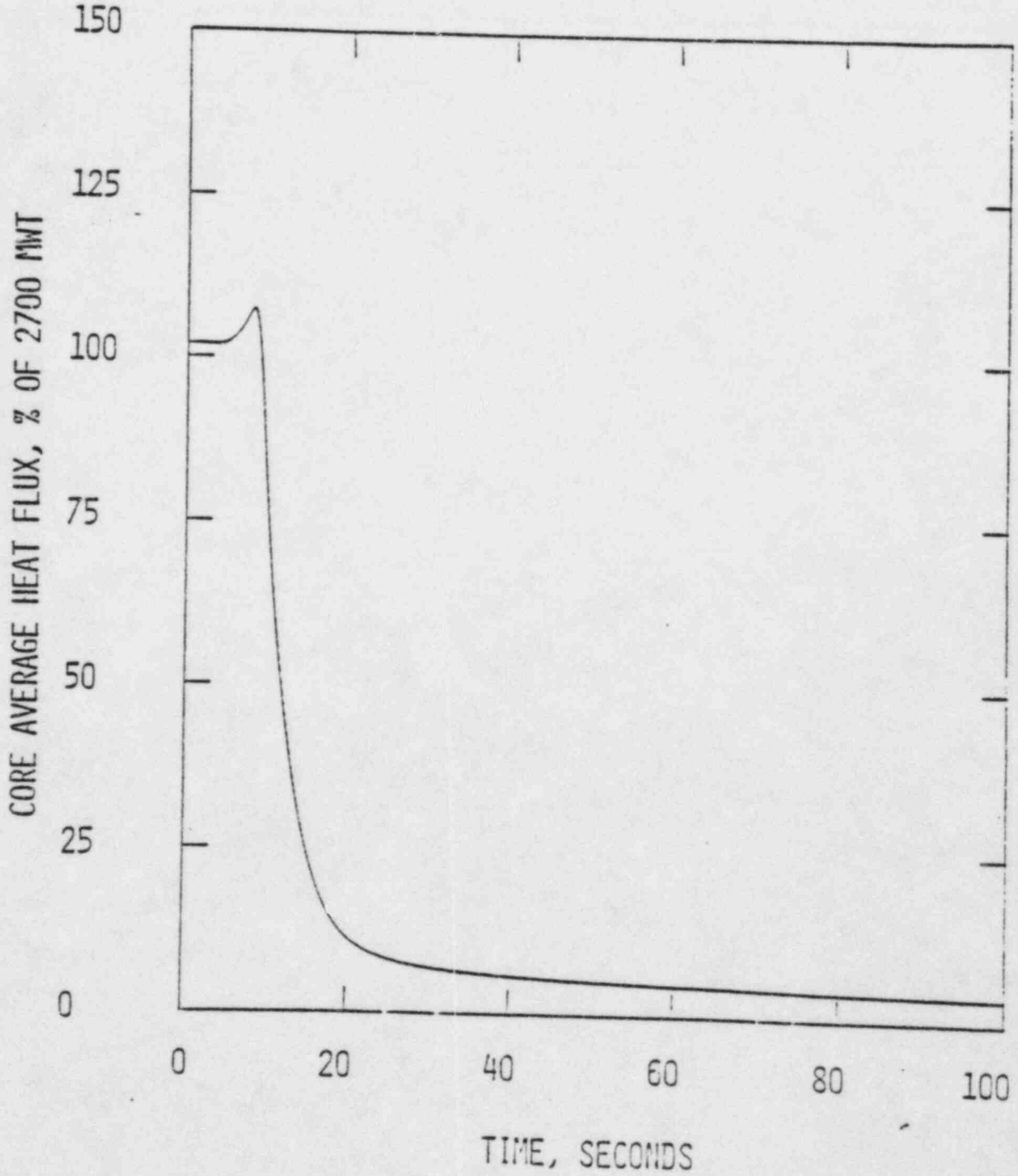
TABLE 7.1.3-2

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

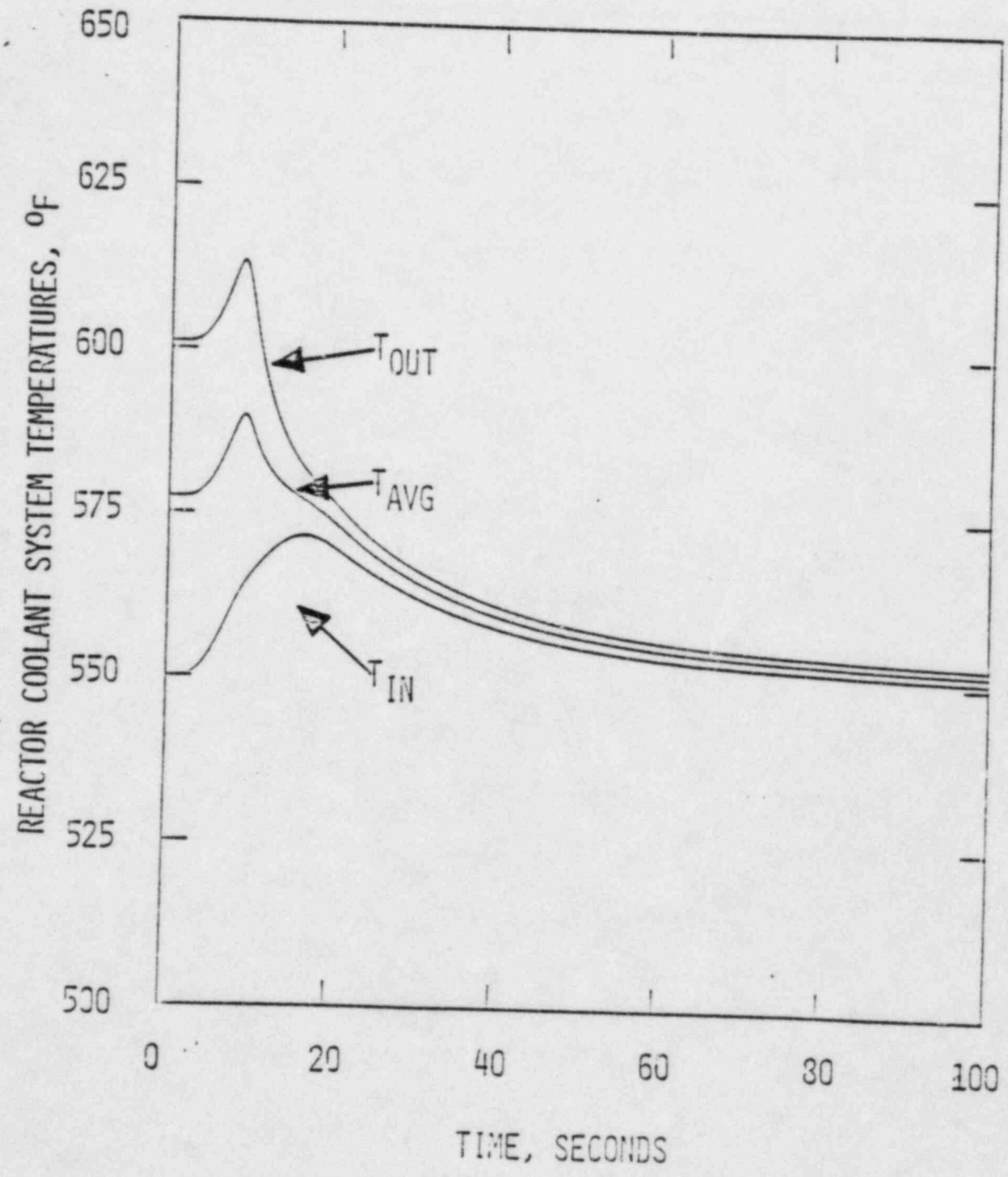
TIME (SEC)	EVENT	SETPOINT or VALUE
0.0	Loss of Secondary Load	-
5.8	High Pressurizer Pressure Trip Signal Generated	2435 psia
6.1	Steam Generator Safety Valves Open	1050 psia
6.4	Pressurizer Safety Valves Open	2500 psia
7.2	CEA's begin to drop into core	-
8.8	Maximum RCS Pressure	<2750 psia
12.1	Maximum Steam Generator Pressure	1095 psia
12.4	Pressurizer Safety Valves are fully closed	2400 psia



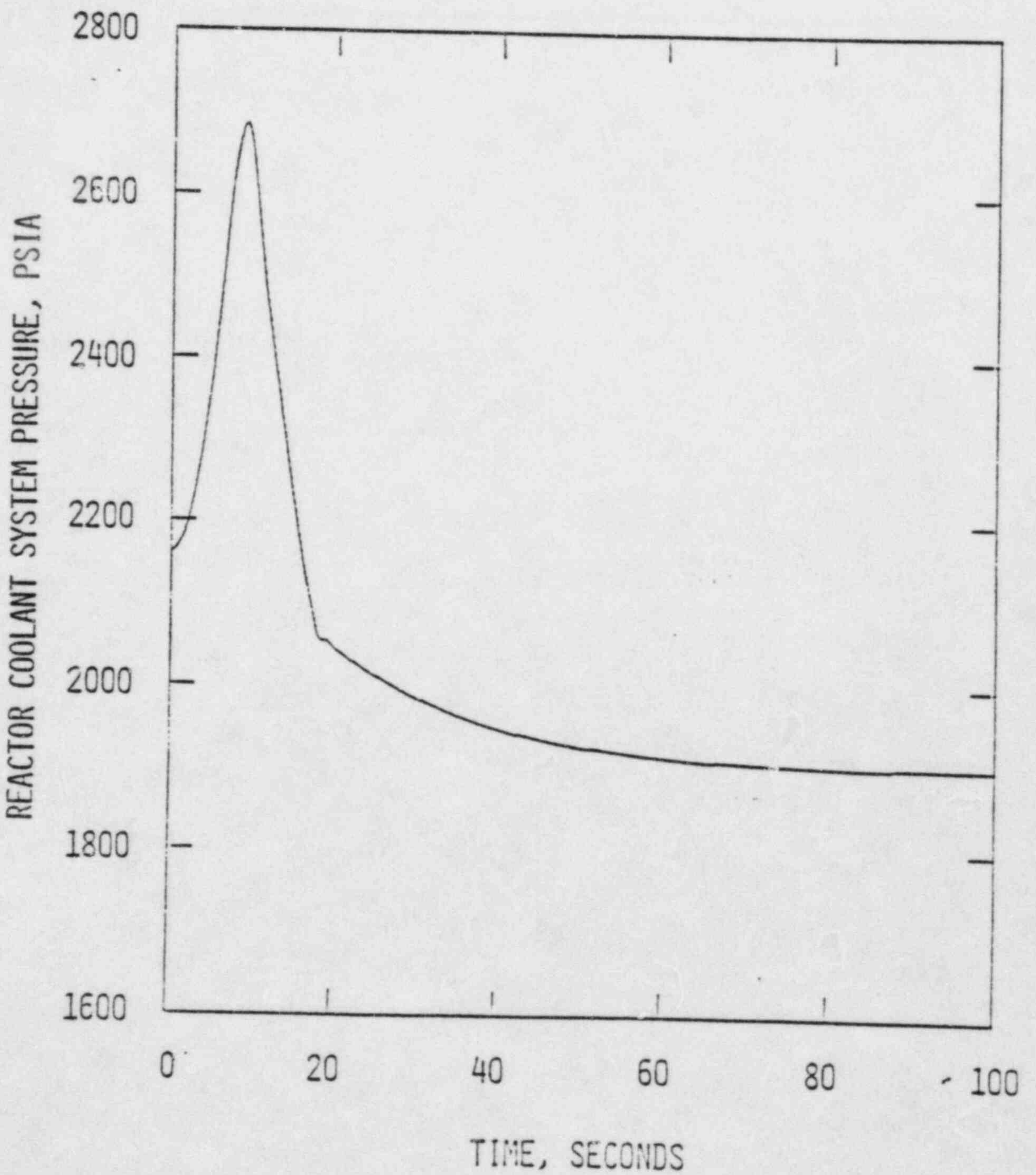
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT CORE POWER VS TIME	FIGU 7.1.



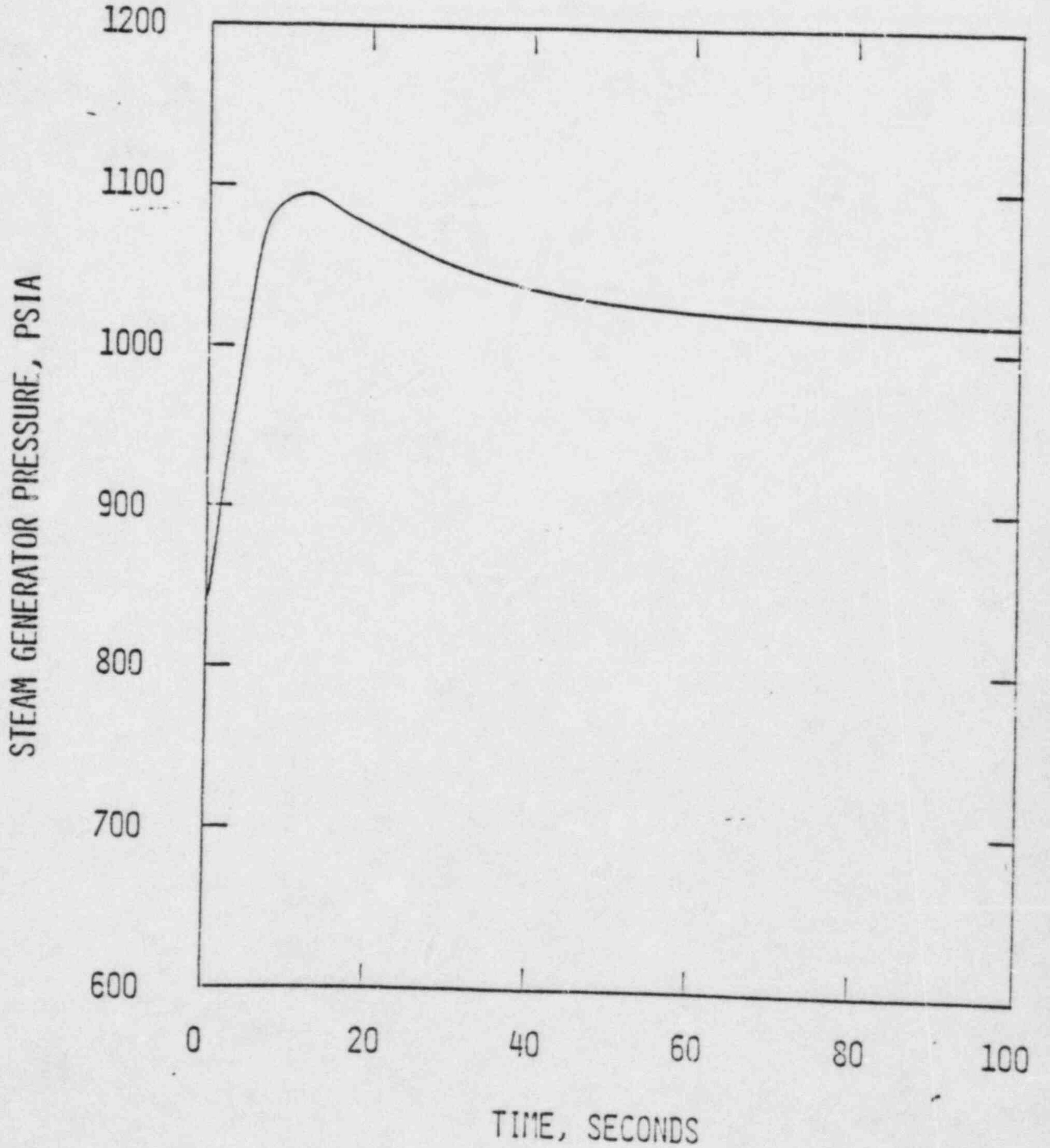
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT CORE AVERAGE HEAT FLUX VS TIME	FIGURE 7.1.3-2



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



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



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT STEAM GENERATOR PRESSURE VS TIME	FIGUR 7.1.3-

7.1.4 EXCESS LOAD EVENT

The Excess Load Event was reanalyzed to demonstrate that the SAFDL'S are not violated during Cycle 8. The reanalyses was necessary to include the effects of a more negative MTC, a lower CEA worth available at trip and changes in the HPSI flow characteristics.

The High Power Level and Thermal Margin/Low Pressure (TM/LP) trips provide primary protection to prevent exceeding the DNBR SAFDL during this event. Additional protection is provided by other trip signals including high rate of change of power, low steam generator water level, and low steam generator pressure. The approach to CTM SAFDL is terminated by either the Local Power Density trip, Variable High Power Level trip or the DNB related trip discussed above. In this analysis, credit is taken only for the action of the High Power trip in the determination of the minimum transient DNBR.

The most limiting load increase events at HFP (Hot Full Power) and HZP (Hot Zero Power) conditions, for approach to the SAFDL'S, are due to the complete opening of the steam dump and bypass valves and the complete opening of the turbine control valves, respectively.

The Excess Load Event at HFP was initiated at the conditions given in Table 7.1.4-1. An MTC of $-2.7 \times 10^{-4}\ \Delta\rho/^{O}F$ was assumed in this analysis. This MTC, in conjunction with the decreasing coolant inlet temperature, enhances the rate of increase of heat flux at the time of reactor trip. An FTC corresponding to beginning-of-cycle conditions with an multiplier of 0.85 was used in the analysis since this FTC causes the least amount of negative reactivity change for mitigating the transient increase in core heat flux. The minimum CEA worth assumed to be available for shutdown at the time of reactor trip for full power operation is $4.3\%\Delta\rho$. The analysis conservatively assumed that the worth of boron injected from the safety injection tank is $-1.00\%\Delta\rho$ per 105 PPM. The pressurizer pressure control system was assumed to be inoperable because this minimizes the RCS pressure during the event and therefore reduces the calculated DNBR.

The HFP Excess Load Event results in a High Power trip at 7.1 seconds. The minimum DNBR calculated for the event at the conditions specified in Table 7.1.4-1 is greater than the design limit of 1.23.

For the Excess Load Event initiated from HFP conditions, SIAS is generated at 50.3 seconds at which time the RCP's are manually tripped by the operator. The coastdown of the pumps decreases the rate of decay heat removal and therefore keeps the RCS coolant temperatures and pressure at higher values.

Auxiliary feedwater flow is delivered to both steam generators at 187.5 seconds. The feedwater flow causes additional cooldown of the RCS. The decreasing temperatures in combination with a negative MTC inserts positive reactivity which enables the core to approach criticality. The negative reactivity inserted due to the CEAs and boron injected via the High Pressure Safety Injection (HPSI) pumps however is sufficient to maintain the core subcritical up to 600 seconds when the operator is assumed to terminate the auxiliary feedwater flow to both steam generators.

Table 7.1.4-2 presents the sequence of events for an Excess Load Event initiated at HFP conditions. Figures 7.1.4-1 to 7.1.4-6 show the NSSS transient response for power, heat flux, RCS temperatures, RCS pressure, steam generator pressures and reactivities.

The Excess Load Event at HZP was initiated at the conditions given in Table 7.1.4-3. The MTC value assumed in the analysis is the same as for the full power case for the reasons previously given. However, the FTC corresponding to beginning-of-cycle conditions with an multiplier of 0.85 was used in the zero power cases since this FTC cause the least amount of negative reactivity change for mitigating the transient increase in core heat flux. The minimum CEA shutdown worth available is conservatively assumed to be -3.5% $\Delta\rho$.

The results of the analysis show that a variable high power trip occurs at 13.0 seconds. The minimum DNBR calculated during the event is greater than 1.23 and the CTM limit is not exceeded.

The sequence of events for the HZP case is presented in Table 7.1.4-4. Figures 7.1.4-7 to 7.1.4-12 show the NSSS transient response for core power, core heat flux, RCS temperature, RCS pressure, steam generator pressures and reactivities. Note that the core remains subcritical at all times after trip for an Excess Load Event initiated from HZP conditions.

For both the full and zero power Excess Load Events the DNBR and CTM SAFDL's are not violated.

TABLE 7.1.4-1

KEY PARAMETERS ASSUMED FOR HOT FULL POWER EXCESS LOAD EVENT ANALYSIS

Parameter	Units	Unit 2* Cycle 5	Unit 1 Cycle 8
Initial Core Power Level+	MWt	2700	2700
Core Inlet Temperature+	^{o}F	548	548
Reactor Coolant System Pressure+	psia	2200	2200
Core Mass Flow Rate+	$X10^{6}$ lbm/hr	138.4	138.4
Moderator Temperature Coefficient	$X10^{-4}\ \Delta\rho/^{o}F$	-2.5	-2.7
CEA Worth Available at Trip	$\%\Delta\rho$	-4.3	-4.3
Doppler Coefficient Multiplier	-	.85	.85
Inverse Boron Worth	PPM/$\%\Delta\rho$	105	105
Auxiliary Feedwater Flow Rate	lbm/sec	175	175
High Power Level Trip Setpoint	% of Full Power	110	107++
Low S. G. Water Level Trip Setpoint	ft	30.9	30.9

*Reference 9.

+For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

++Temperature decalibration is being included explicitly.

TABLE 7.1.4-2

SEQUENCE OF EVENTS FOR THE EXCESS LOAD EVENT AT HOT FULL POWER TO CALCULATE MINIMUM DNBR

Time (sec)	Event	Setpoint or Value
0.0	Complete Opening of Steam Dump and Bypass Valves at Full Power	-
7.1	High Power Trip Signal Generated	107% of full power
7.5	Trip Breakers Open	-
8.0	CEA's Begin to Drop Into Core	-
8.4	Maximum Power	114% of full power
8.9	Minimum DNBR (CE-1)	>1.23
9.2	Low Steam Generator Level Trip Setpoint Reached	30.9 ft.
28.0	Rampdown of main Feedwater Flow Completed	8% of full power main feedwater flow
49.4	Pressurizer Empties	-
50.3	Safety Injection Actuation Signal Initiated; Manual Trip of RCP's	1578 psia
50.4	RCS Pump Coastdown Begins	-
76.2	Main Steam Isolation Signal	548 psia
170.1	Isolation of Main Feedwater Flow to Both Steam Generators	-
187.5	Auxiliary Feedwater Flow Delivered to both Steam Generators	175 lbm/sec to each steam generator
484.2	Pressurizer Begins to Refill	-
600.0	Operator Terminates Auxiliary Feedwater Flow to Both Steam Generators; Total Reactivity	$-2.00\% \Delta\rho$

TABLE 7.1.4-3

KEY PARAMETERS ASSUMED FOR HOT ZERO POWER EXCESS LOAD EVENT ANALYSIS

Parameter	Units	Unit 1* Cycle 6	Unit 1 Cycle 8
Initial Core Power Level+	MWt	1	1
Core Inlet Temperature+	^{o}F	532	532
Reactor Coolant System Pressure+	psia	2225	2200
Core mass Flow Rate+	$X10^{6}$ lbm/hr	141.3	141.3
Moderator Temperature Coefficient	$X10^{-4}\Delta\rho\ /^{o}F$	-2.5	-2.7
CEA Worth Available at Trip	$\%\ \Delta\rho$	-4.0	-3.5
Doppler Coefficient Multiplier	-	.85	.85
Inverse Boron Worth	PPM/$\%\Delta\rho$	100	105
Variable High Power Trip Setpoint	% of full power	40	40

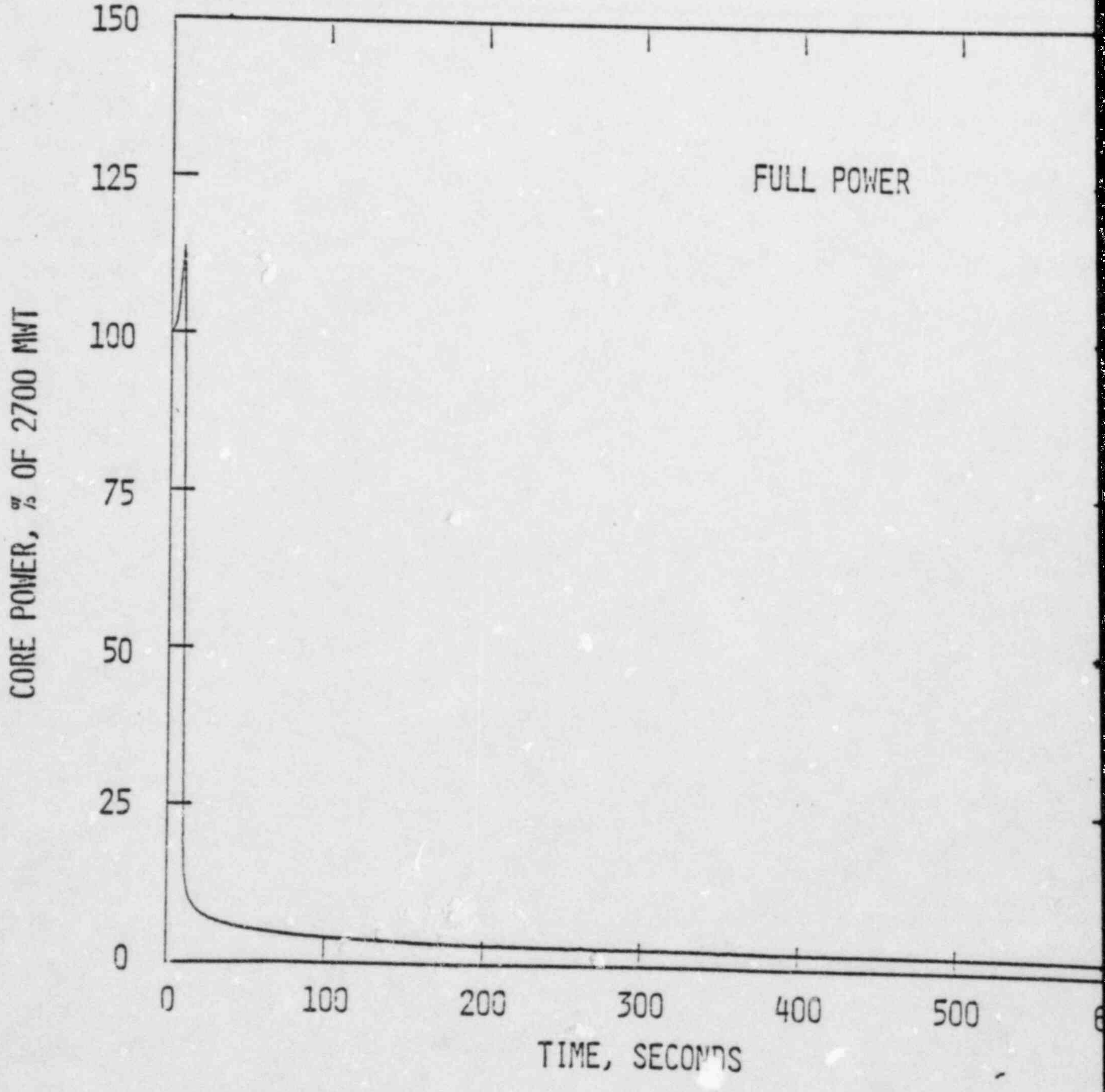
*Reference 8.

+For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

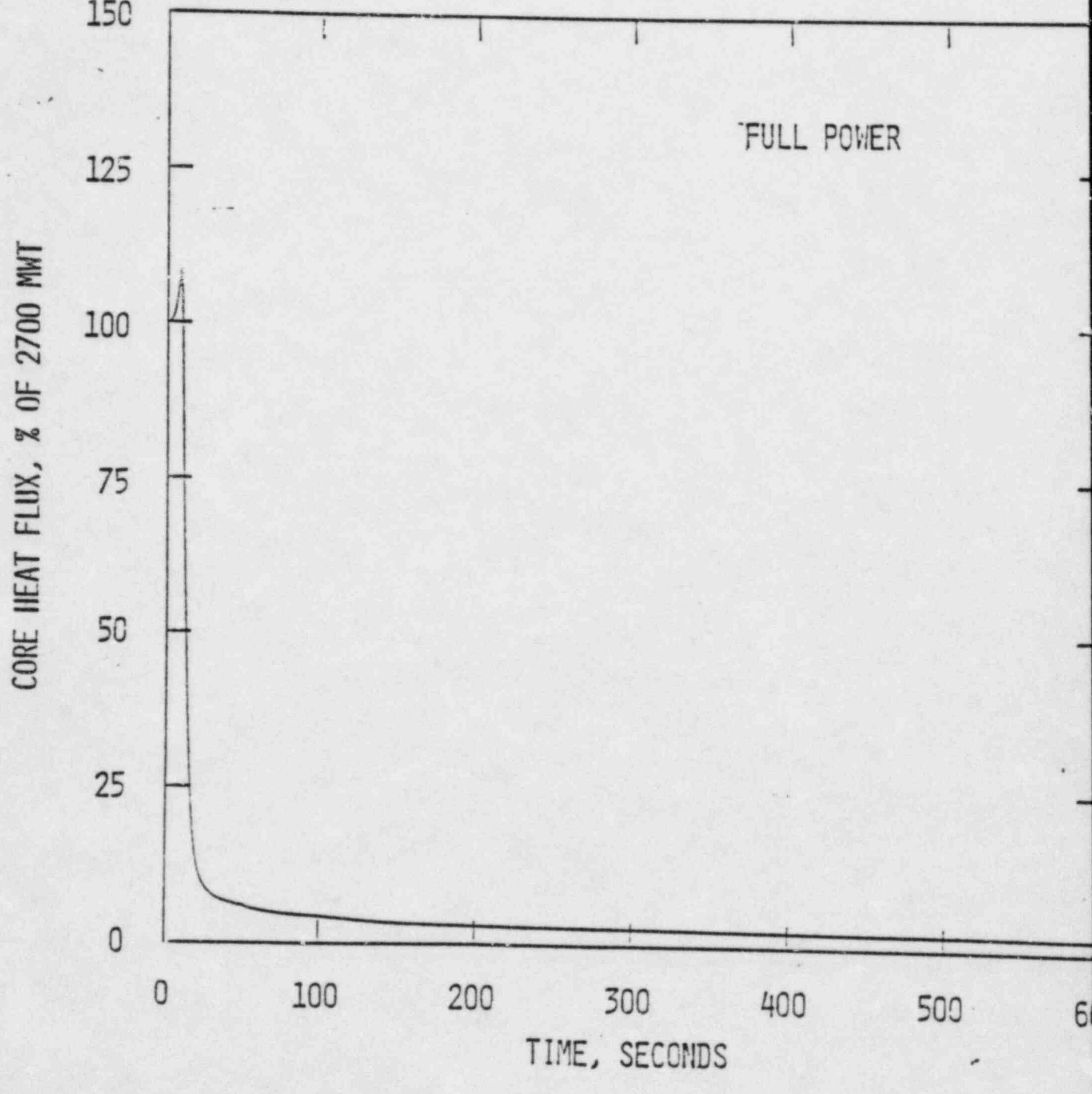
TABLE 7.1.4-4

SEQUENCE OF EVENTS FOR EXCESS LOAD EVENT AT HOT ZERO POWER CONDITIONS TO CALCULATE MAXIMUM LHR

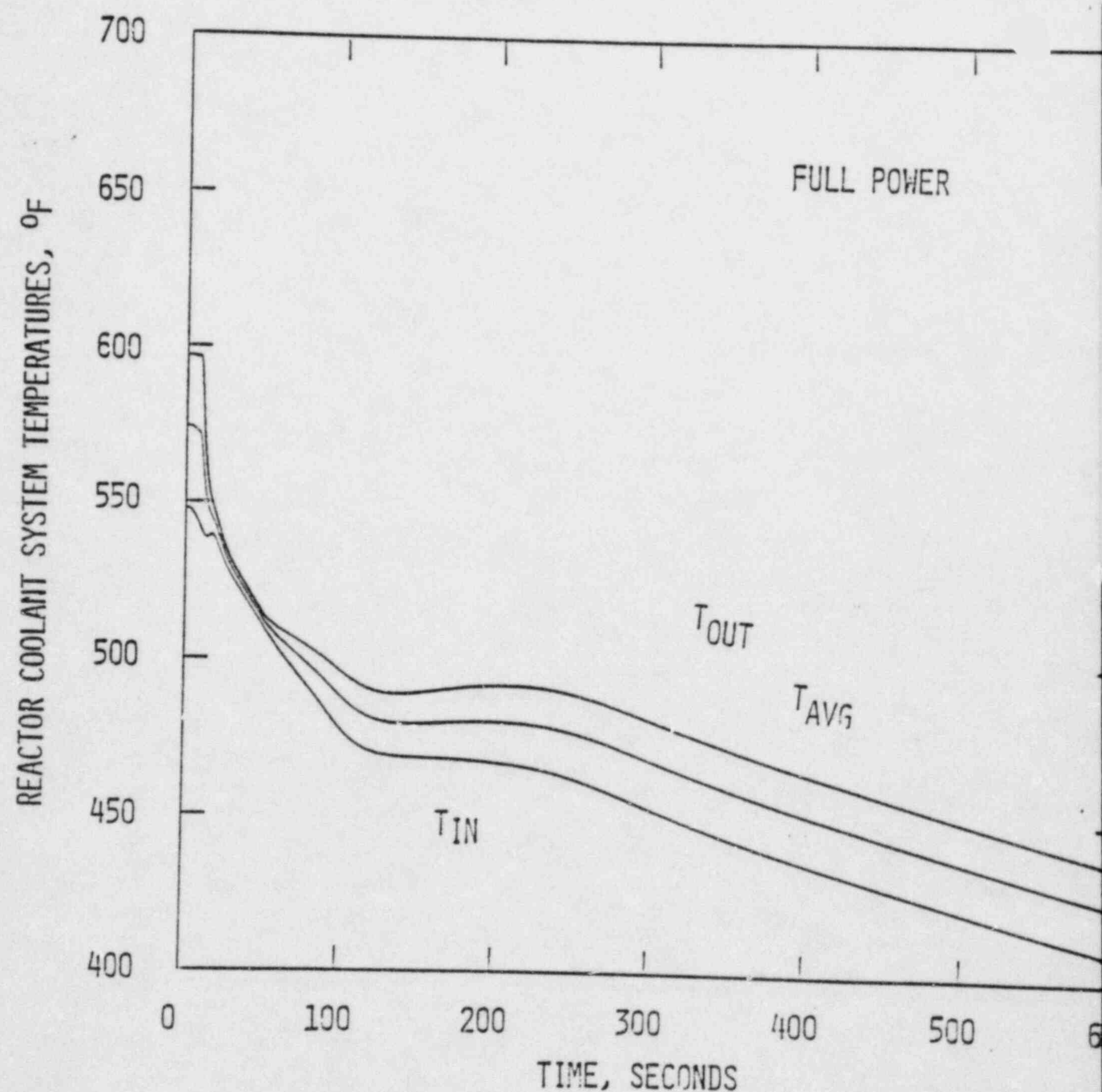
Time(sec)	Event	Setpoint or Value
0.0	Turbine Admission Valve Opens	120% Steam Flow at Full Power
13.0	Variable High Power Trip Signal Generated	40% of full power
13.4	Trip Breakers Open	-
13.5	Core Power Reaches Maximum;	75.1 % of full power
13.9	CEA's begin to drop into Reactor Core	-



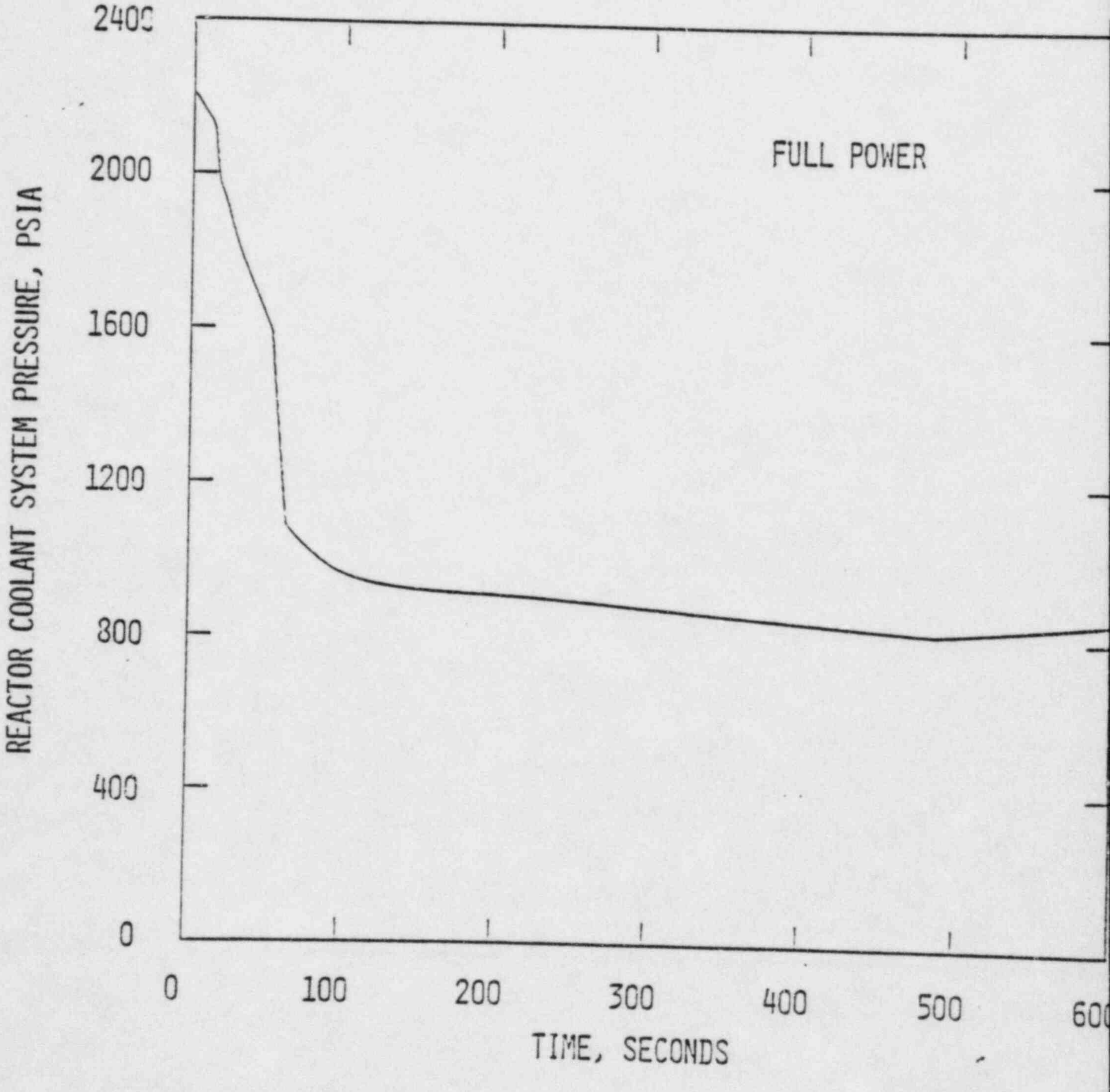
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT CORE POWER VS TIME	FIGURE 7.1.4-



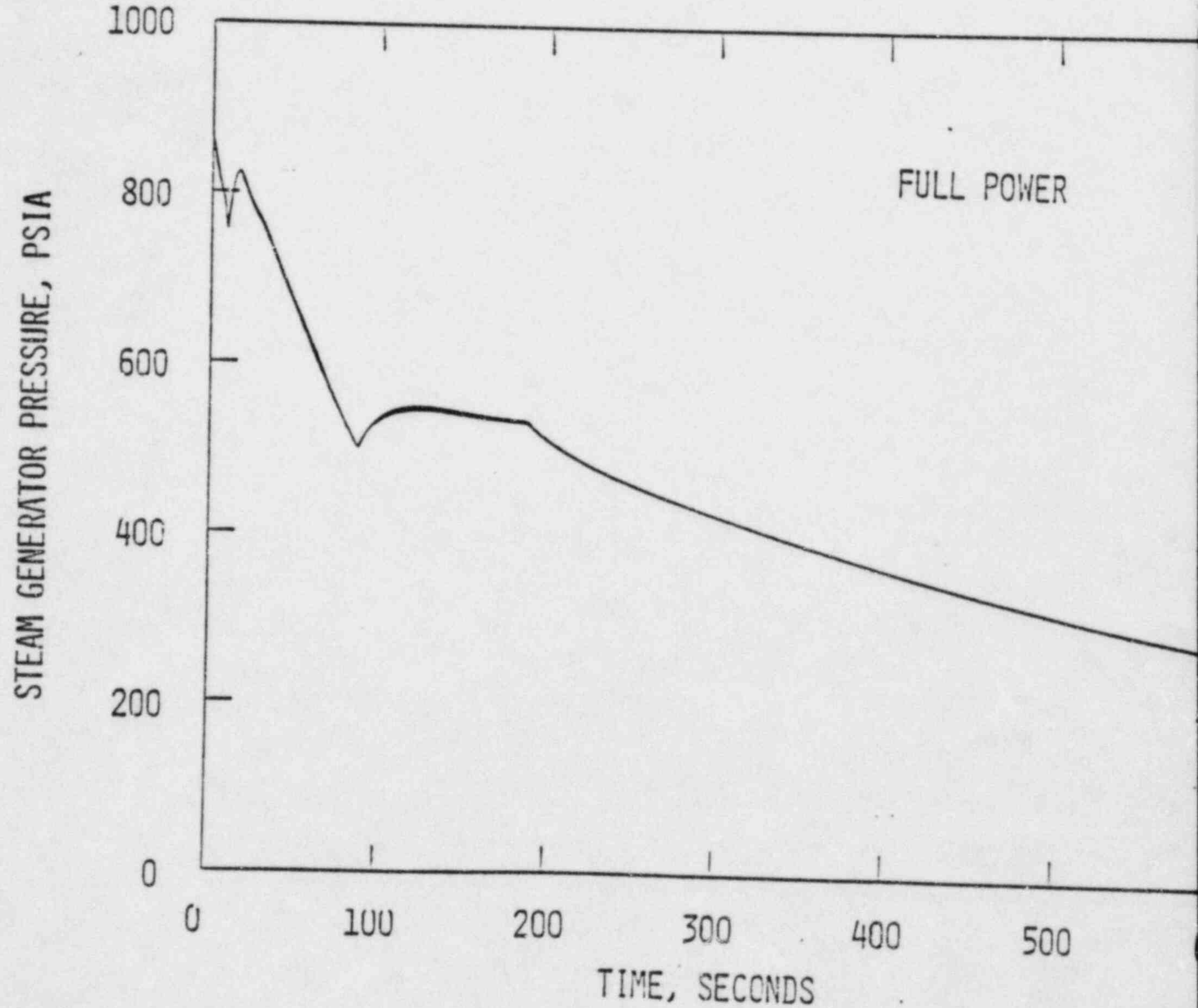
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT CORE HEAT FLUX VS TIME	FIGURE 7.1.4-2



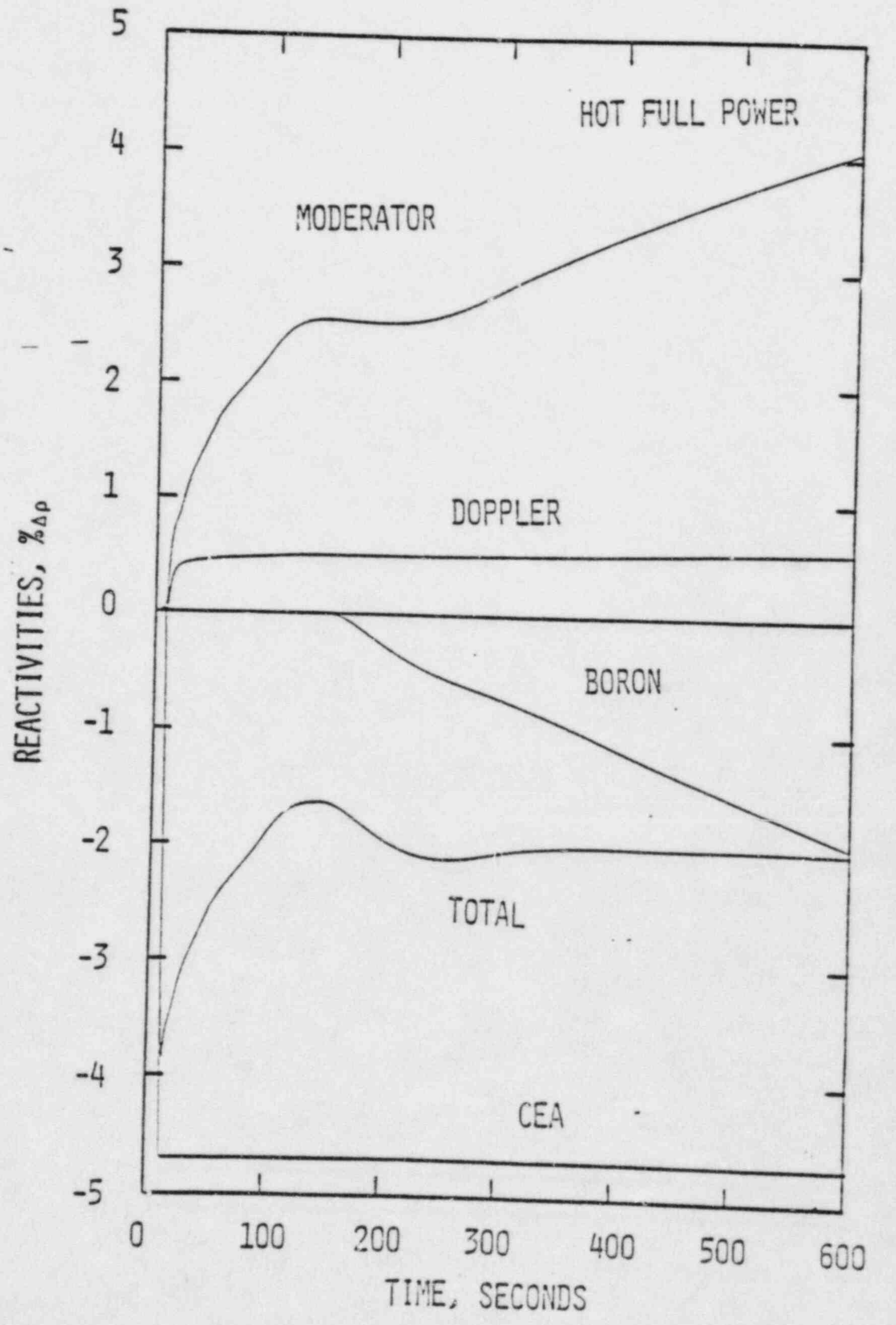
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7.1.4-



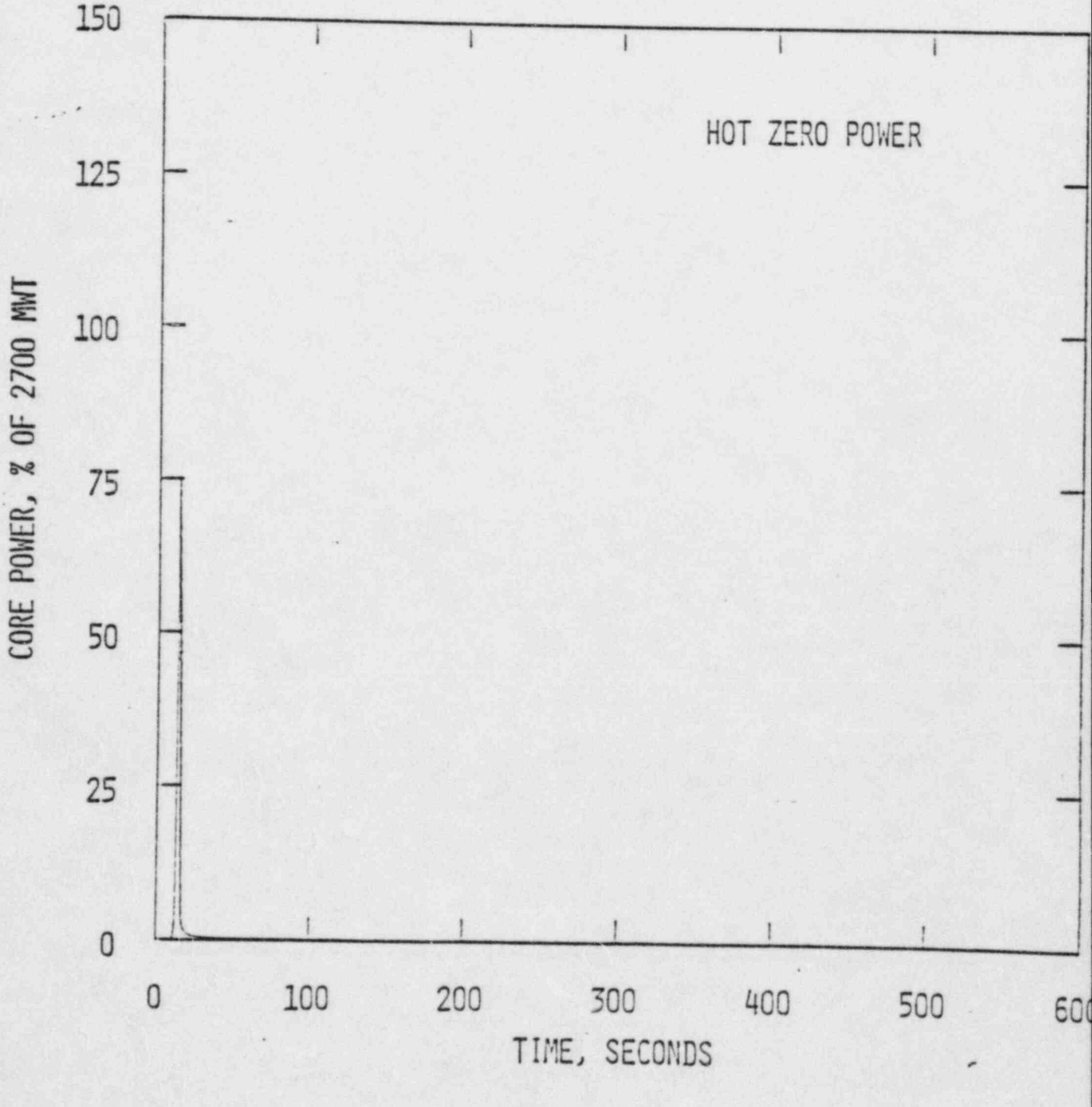
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7.1.4-4



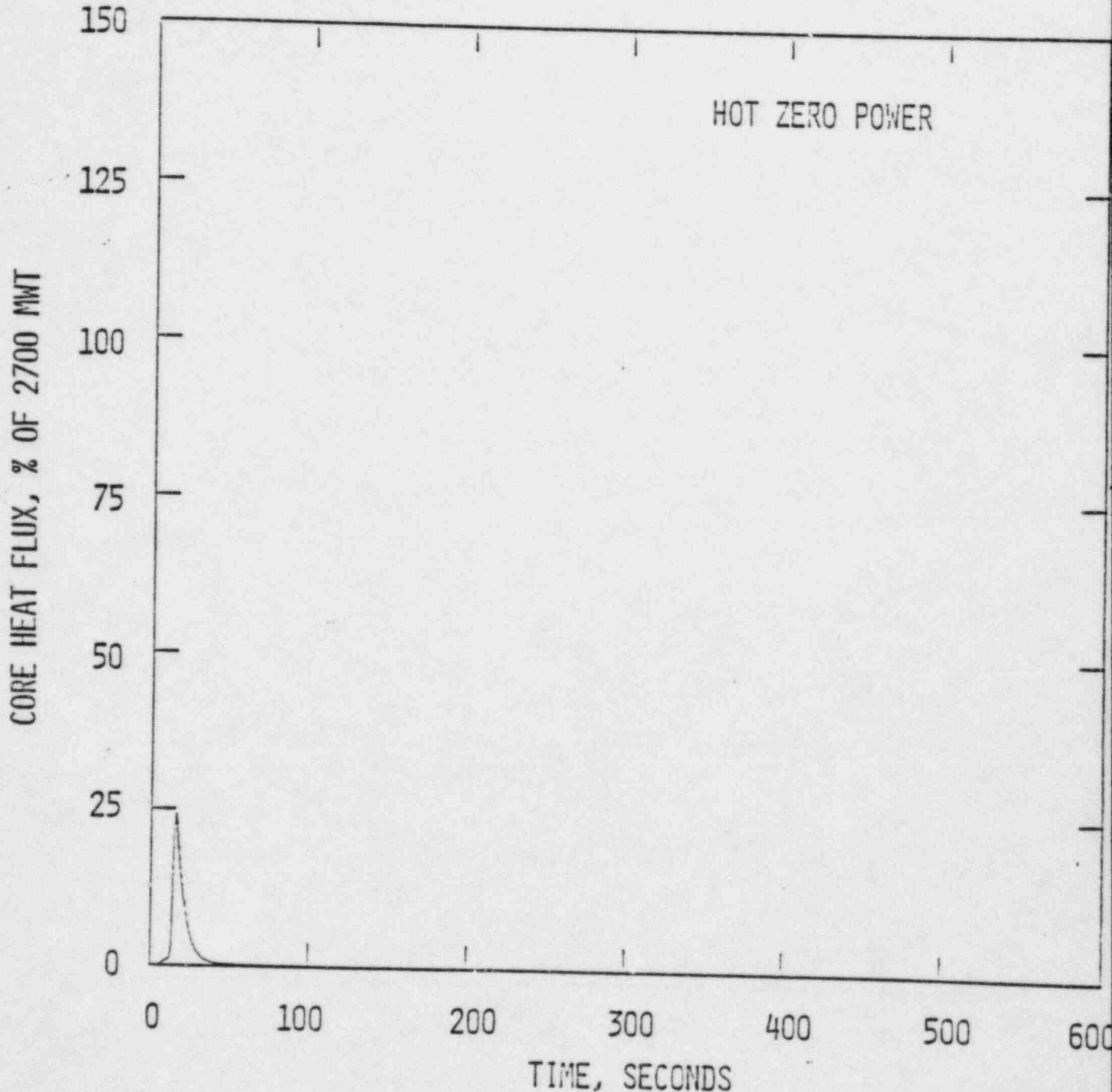
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 7.1.4-



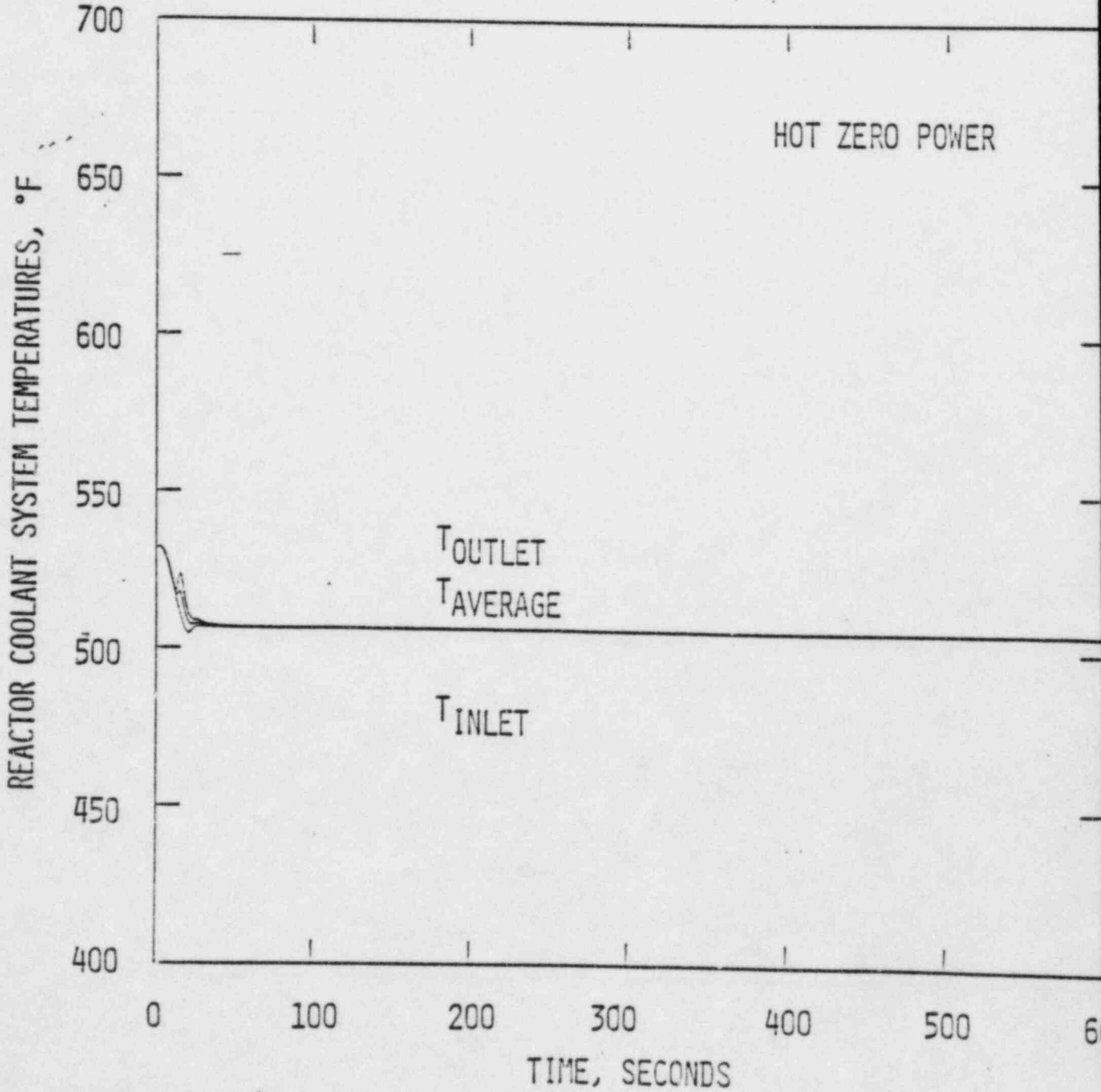
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT REACTIVITIES VS TIME	FIGU 7.1.4-



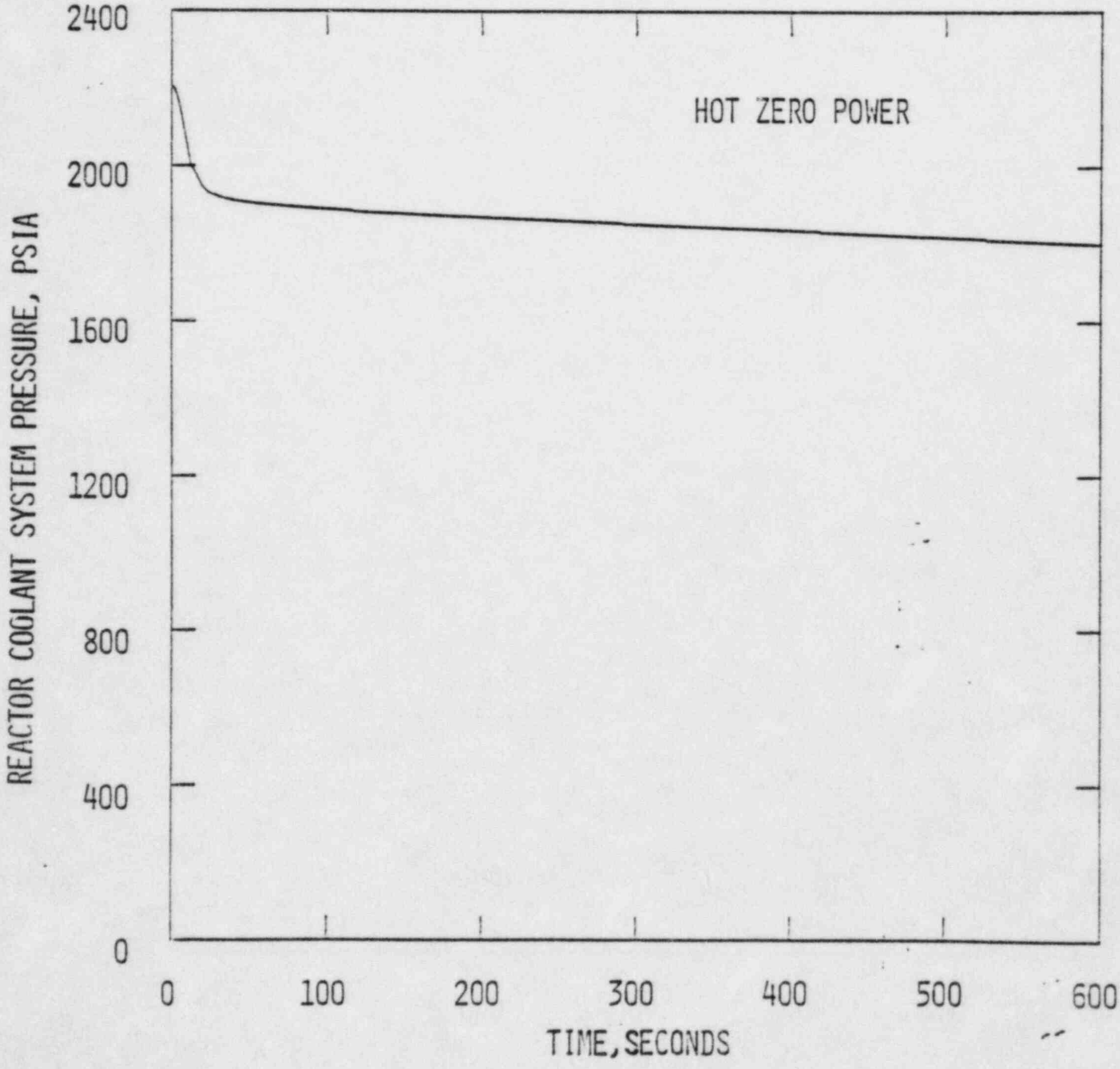
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT CORE POWER VS TIME	FIGURE 7.1.4-7



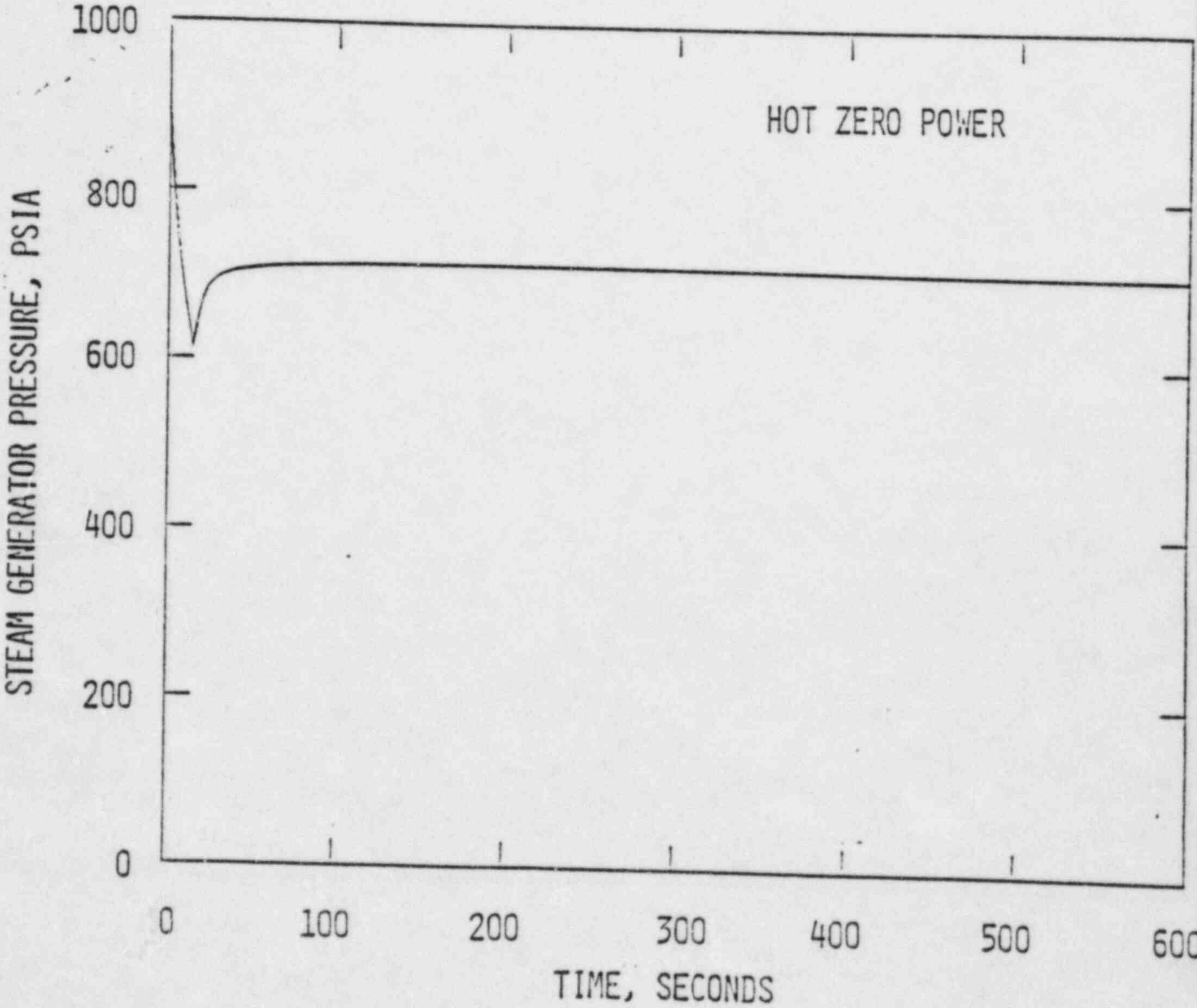
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT CORE HEAT FLUX VS TIME	FIGURE 7.1.4-8



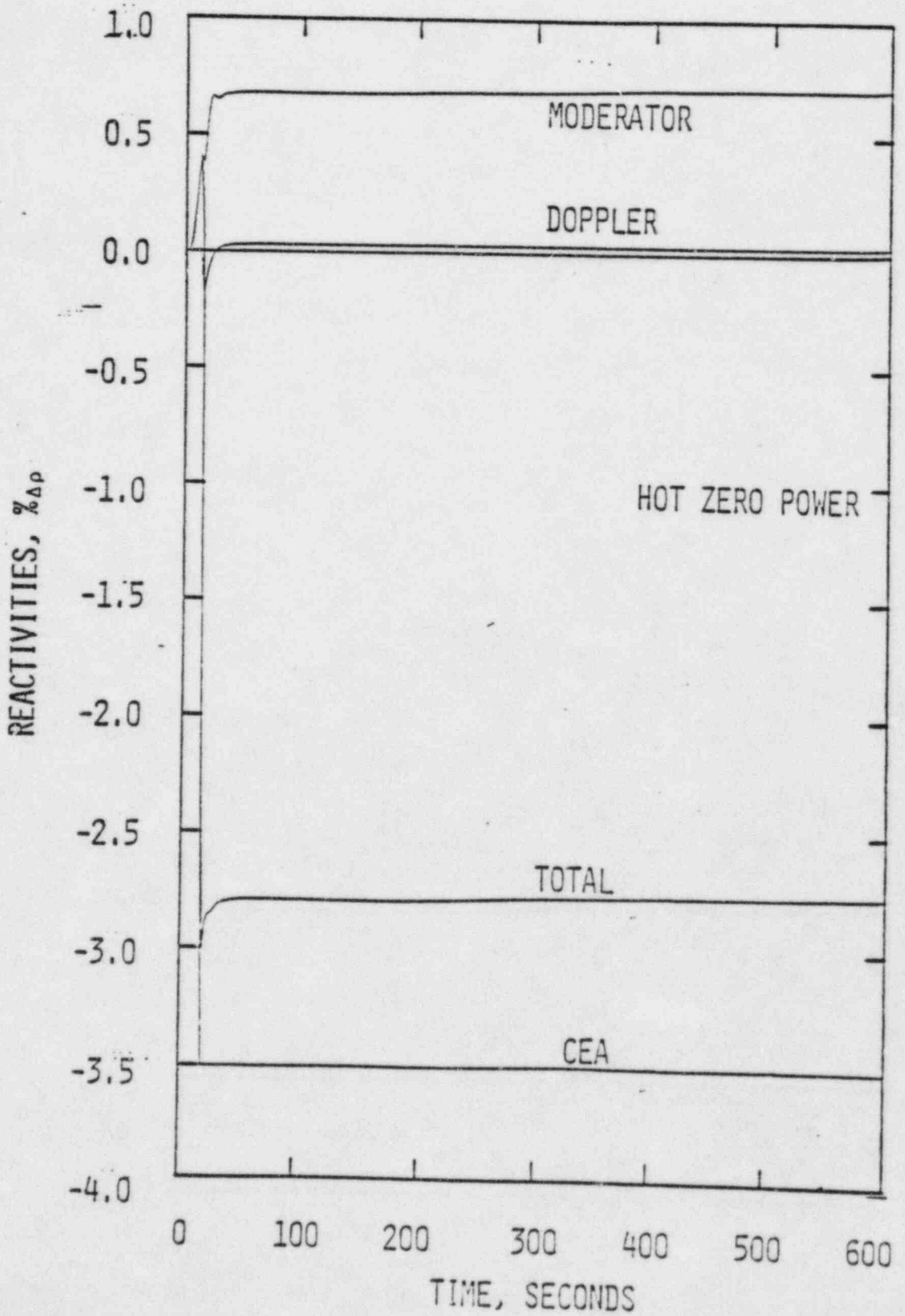
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7.1.4-9



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7.1.4-10



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 7.1.4-11



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT REACTIVITIES VS TIME	FIGUR 7.1.4-

7.2.4 AOO's RESULTING FROM THE MALFUNCTION OF ONE STEAM GENERATOR

The transient resulting from the malfunction of one steam generator is analyzed for cycle 8 to determine the initial margin that must be maintained by the LCO's such that in conjunction with the RPS (asymmetric steam generator protective trip ASGPT), the DNBR and fuel centerline melt design limits are not exceeded. Reanalysis of this event was required due to changes in the Moderator Temperature Coefficient (MTC) and the Main Steam Safety Valve Opening Setpoints.

The methods used to analyze these events are consistent with those used in the reference cycle (Unit 1 Cycle 6).

The four events which affect a single generator are identified below:

1. Loss of Load to One Steam Generator

2. Excess Load to One Steam Generator

3. Loss of Feedwater to One Steam Generator

4. Excess Feedwater to One Steam Generator

Of the four events described above, it has been determined that the Loss of Load to One Steam Generator (LL/1SG) transient is the limiting asymmetric event. Hence, only the results of this transient are reported.

The event is initiated by the inadvertent closure of a single main steam isolation valve. Upon the loss of load to the steam generator, its pressure and temperature increase to the opening pressure of the secondary safety valves. This analysis assumed the new opening setpoint pressure of 1050 psia for the Main Steam Safety Valves. The intact steam generator "picks up" the lost load, which causes its temperature and pressure to decrease, thus causing the core average inlet temperature to decrease and enhancing the asymmetry in the reactor inlet temperature. In the presence of a negative MTC this causes an increase in core power and radial peaking on the "cold-side" of the core. The most most negative MTC value of $-2.7X10^{-4}\,\Delta\rho\,/^{o}F$ is used in the analysis. The LL/1SG event results in the greatest asymmetry in core inlet temperature distribution and the most limiting DNBR for the transients resulting from the malfunction of one steam generator.

The LL/1SG was initiated at the initial conditions given in Table 7.2.4-1. A reactor trip is generated by the Asymmetric Steam Generator Trip at 3.0 seconds based on high differential pressure between the steam generators. A new differential pressure trip value of 186.0 psia was used to include maximum measurement uncertainty.

Table 7.2.4-2 presents the sequence of events for the Loss of Load to One Steam Generator. The transient behavior of key NSSS parameters are presented in Figures 7.2.4-1 to 7.2.4-5. The minimum transient DNBR calculated for this event is greater than the DNBR limit of 1.23.

An allowable initial linear heat generation rate of 18.5 KW/ft could exist as an initial condition without exceeding the acceptable fuel to centerline melt of 22 KW/ft during this transient. The actual initial linear heat generation rate will be less since the Linear Heat Rate LCO is based on the more limiting linear heat rate for LOCA (e.g., 15.5 KW/ft).

The event initiated from the extremes of the LCO in conjunction with the ASGPT protective trip will not lead to DNBR or a centerline fuel temperature which exceed the DNBR and centeline to melt design limits.

TABLE 7.2.4-1

KEY PARAMETERS ASSUMED IN
THE ANALYSIS OF LOSS OF LOAD TO ONE STEAM GENERATOR

	Units	Unit 1* Cycle 6	Unit 1 Cycle 8
Initial Core Power+	MWt	2700	2700
Initial Core Inlet Temperature+	^oF	548	548
Initial Reactor Coolant System Pressure+	psia	2225	2200
Moderator Temperature Coefficient	$X10^{-4}\Delta\rho/^oF$	-2.5	-2.7
Doppler Coefficient Multiplier	-	.85	.85
ASGPT Setpoint	psid	175	186

*Reference 8.

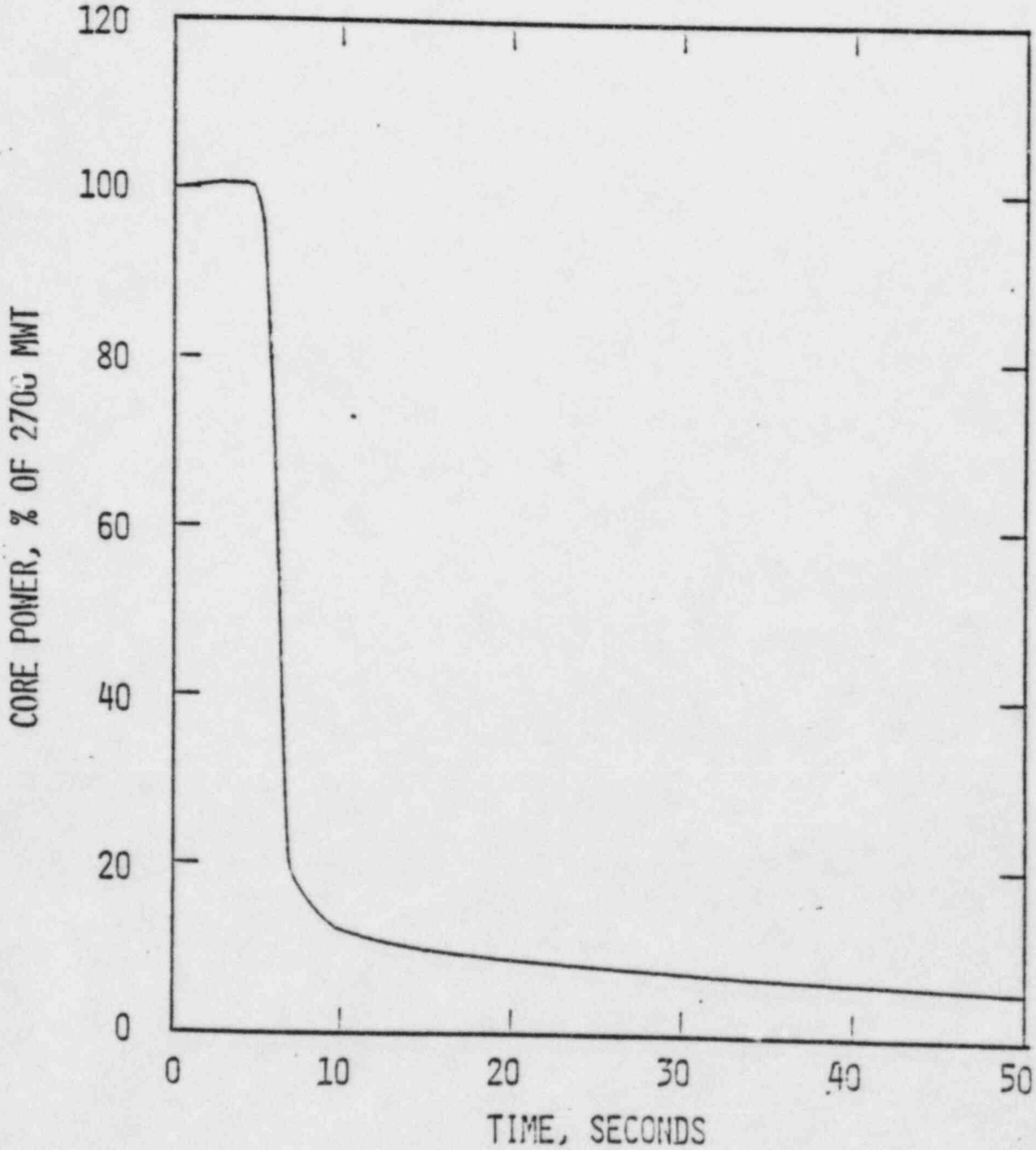
+For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

TABLE 7.2.4-2

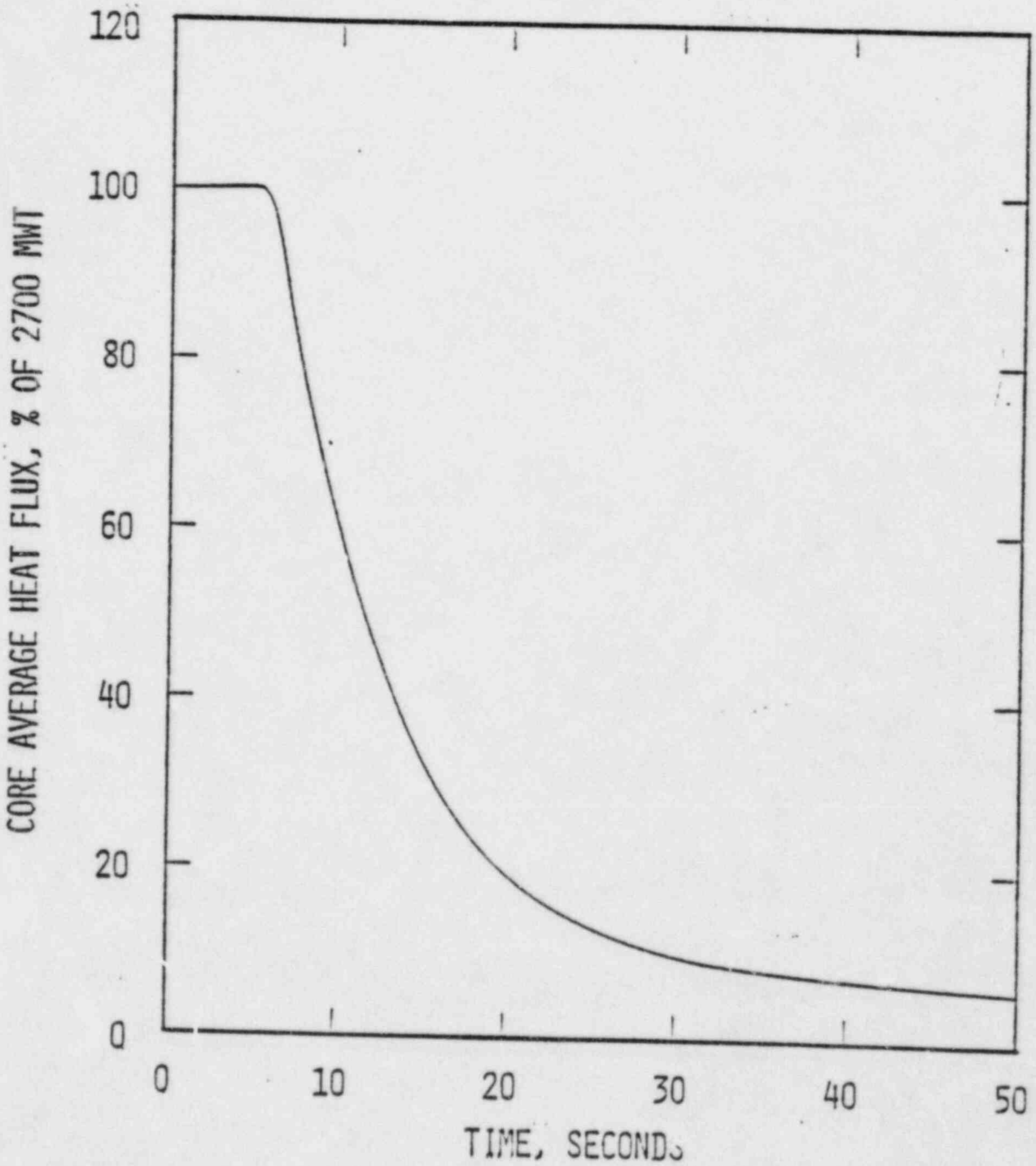
SEQUENCE OF EVENTS FOR LOSS OF LOAD TO ONE STEAM GENERATOR

TIME (sec)	EVENT	SETPOINT or VALUE
0.0	Spurious closure of a single main steam isolation valve	-
0.0	Steam Flow from unaffected steam operator increases to maintain turbine power	-
3.0	ASGPT* setpoint reached (differential pressure)	186 psid
3.9	Trip breakers open	-
4.4	CEA's begin to insert	
4.4	Atmospheric Dump and Byp :s valves open	-
6.0	Minimum DNBR occurs	>1.23
6.6	Main Steam Safety Valves Open on isolated steam generator	1050 psia
10.2	Maximum steam generator pressure	1074 psia

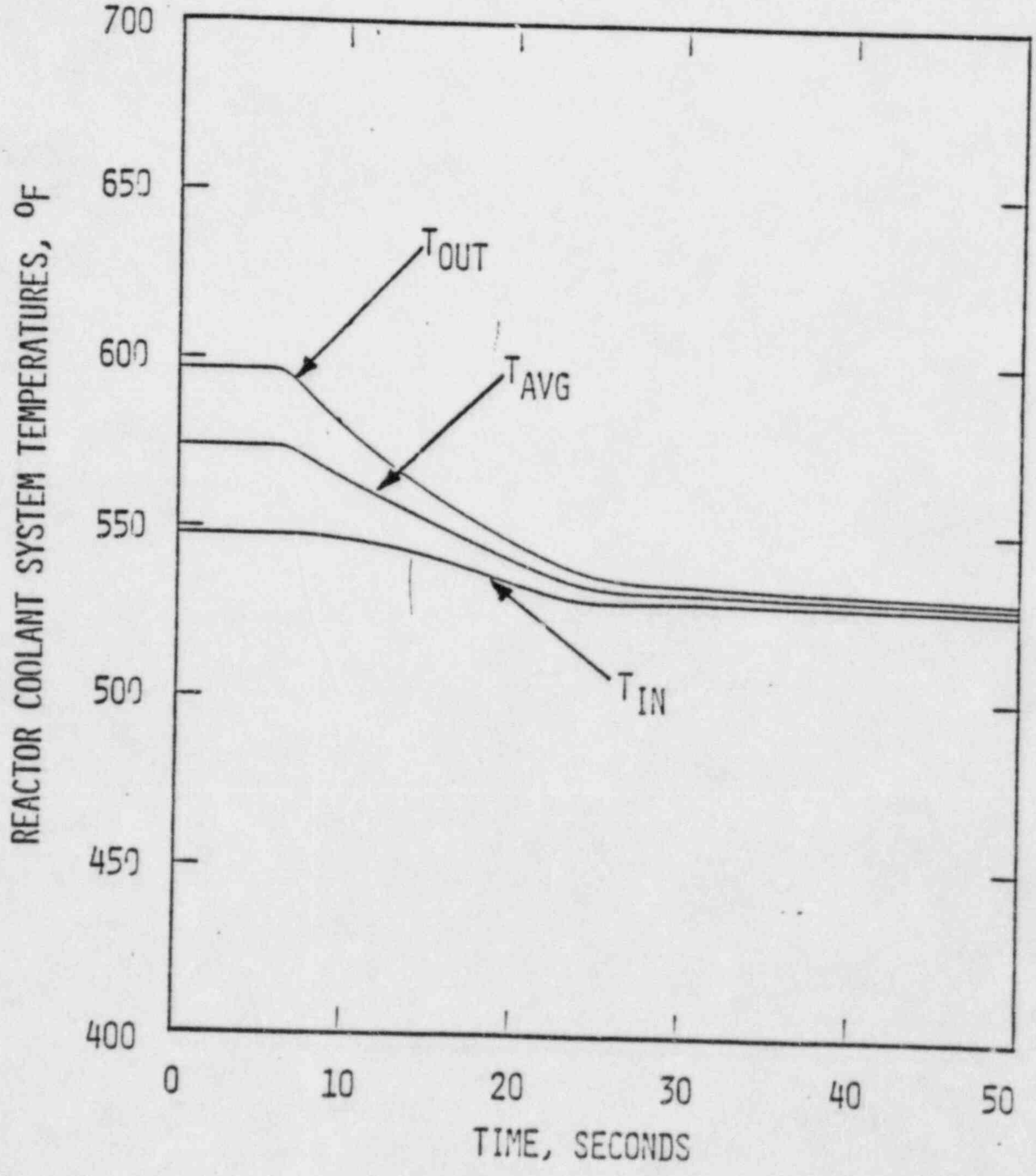
*ASGPT-Asymmetric Steam Generator Protection Trip



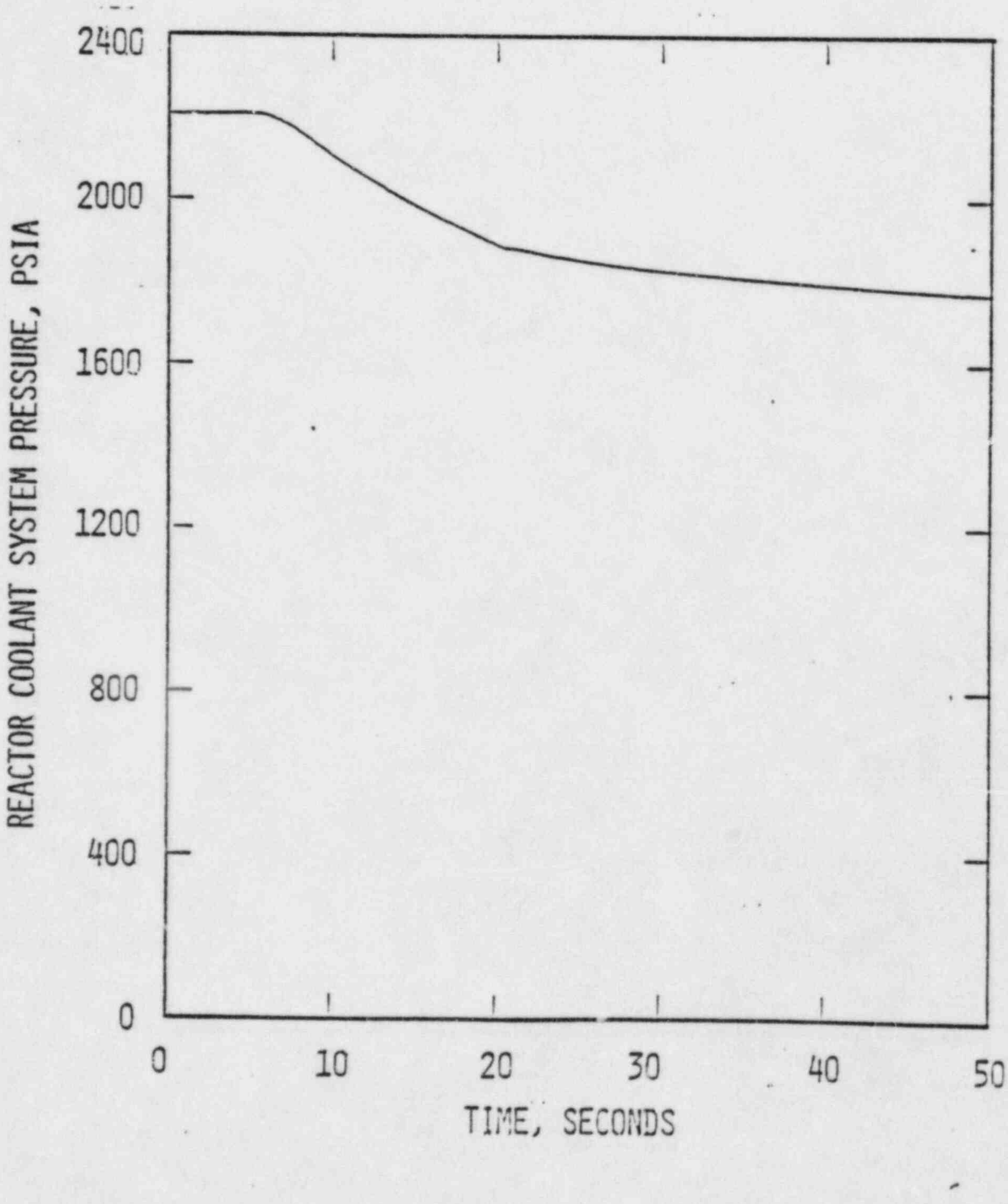
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT CORE POWER VS TIME	FIGU 7.2.4



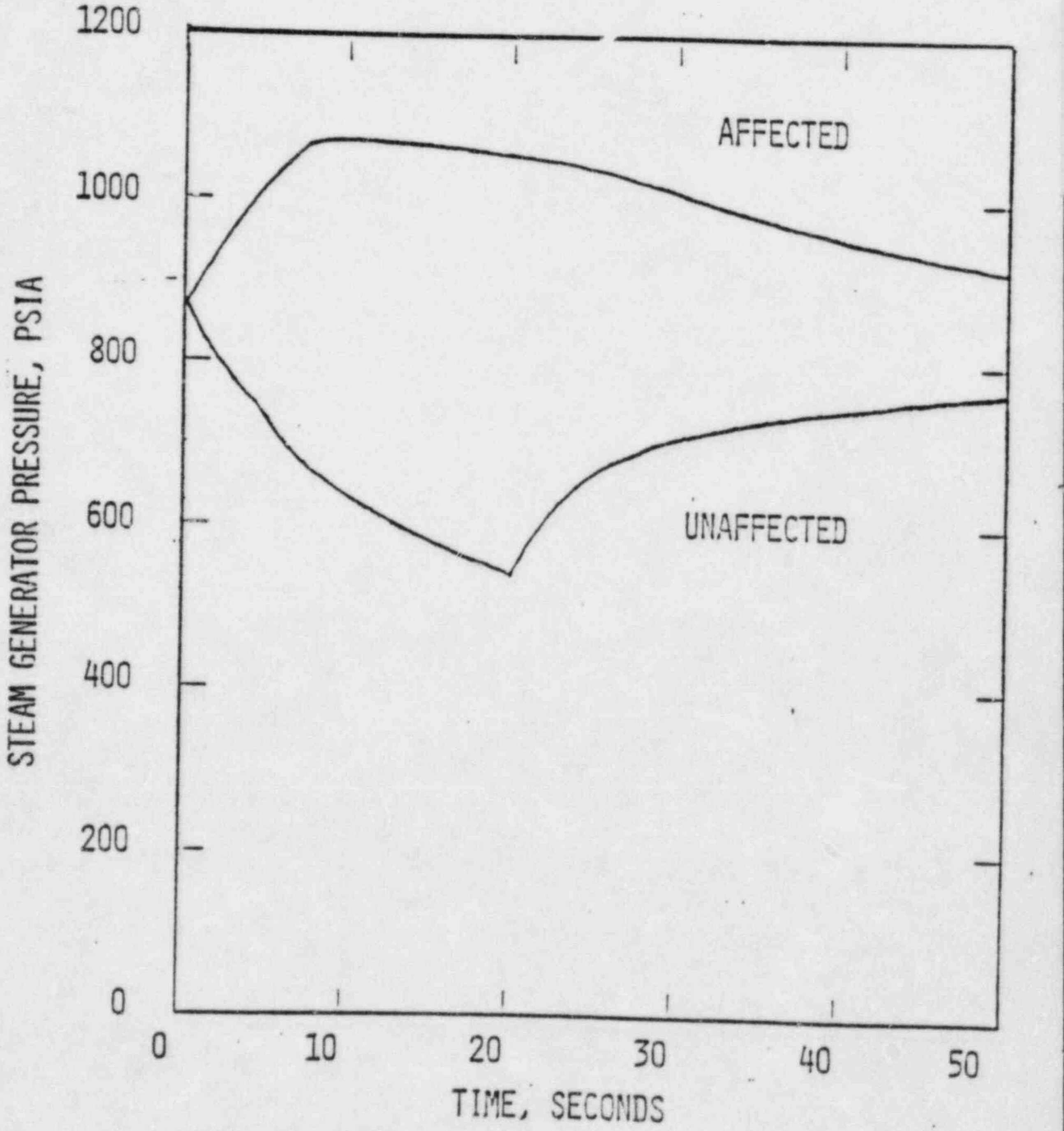
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT CORE AVERAGE HEAT FLUX VS TIME	FIG 7.2



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGU 7.2.4-



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGU 7.2.4



BALTIMORE AS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT STEAM GENERATOR PRESSURE VS TIME	FIGU 7.2.4-

7.3.1 CEA EJECTION EVENT

The zero power case of the CEA Ejection event is analyzed for Cycle 8 to determine the fraction of fuel pins that exceed criteria for clad damage. Reanalysis is required due to increases i the CEA ejected worth and the post-ejected radial power peak for the zero power case.

The analytical method employed in the analysis of this event is the NRC approved Combustion Engineering CEA Ejection method which is described in CENPD-190-A, (Reference 6).

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

Clad Damage Threshold:
Total Average Enthalpy = 200 cal/gm

Incipient Centerline Melting Threshold:
Total Centerline Enthalpy = 250 cal/gm

Fully Molten Centerline Threshold:
Total Centerline Enthalpy = 310 cal/gm

To bound the most adverse conditions during the cycle, the most limiting of either the Beginning of Cycle (BOC) or End of Cycle (EOC) parameter values were used in the analysis. A BOC Doppler defect was used since it produces the least amount of negative reactivity feedback to mitigate the transient. A BOC moderator temperature coefficient of $+0.7\times10^{-4}\ \Delta\rho\ /^{o}F$ was used because a positive MTC results in positive reactivity feedback and thus increases coolant temperatures. Although the MTC value is also an increase from the previous cycle values of $+0.5\times10^{-4}\ \Delta\rho/^{o}F$, it has only a second order effect on the results of the analysis. EOC delayed neutron fractions and neutron kinetics were used in the analysis to produce the highest power rise during the event.

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

The zero power case was analyzed, assuming the value of 0.05 seconds for the total ejection time, which is consistent with the Reference Cycle.

The power transient produced by a CEA ejection initiated for the zero power case is shown in Figure 7.3.1-1.

The results of the zero power CEA ejection case analyzed (Table 7.3.1-2) show that the maximum total energy deposited during the event in the pin is less than both the criterion for clad damage (i.e., 200 cal/gm) and the incipient centerline melt threshold of 250 cal/gm. Consequently, no fuel pin failures are calculated to occur.

TABLE 7.3.1-1

KEY PARAMETERS ASSUMED IN THE CEA EJECTION ANALYSIS
ZERO POWER CASE

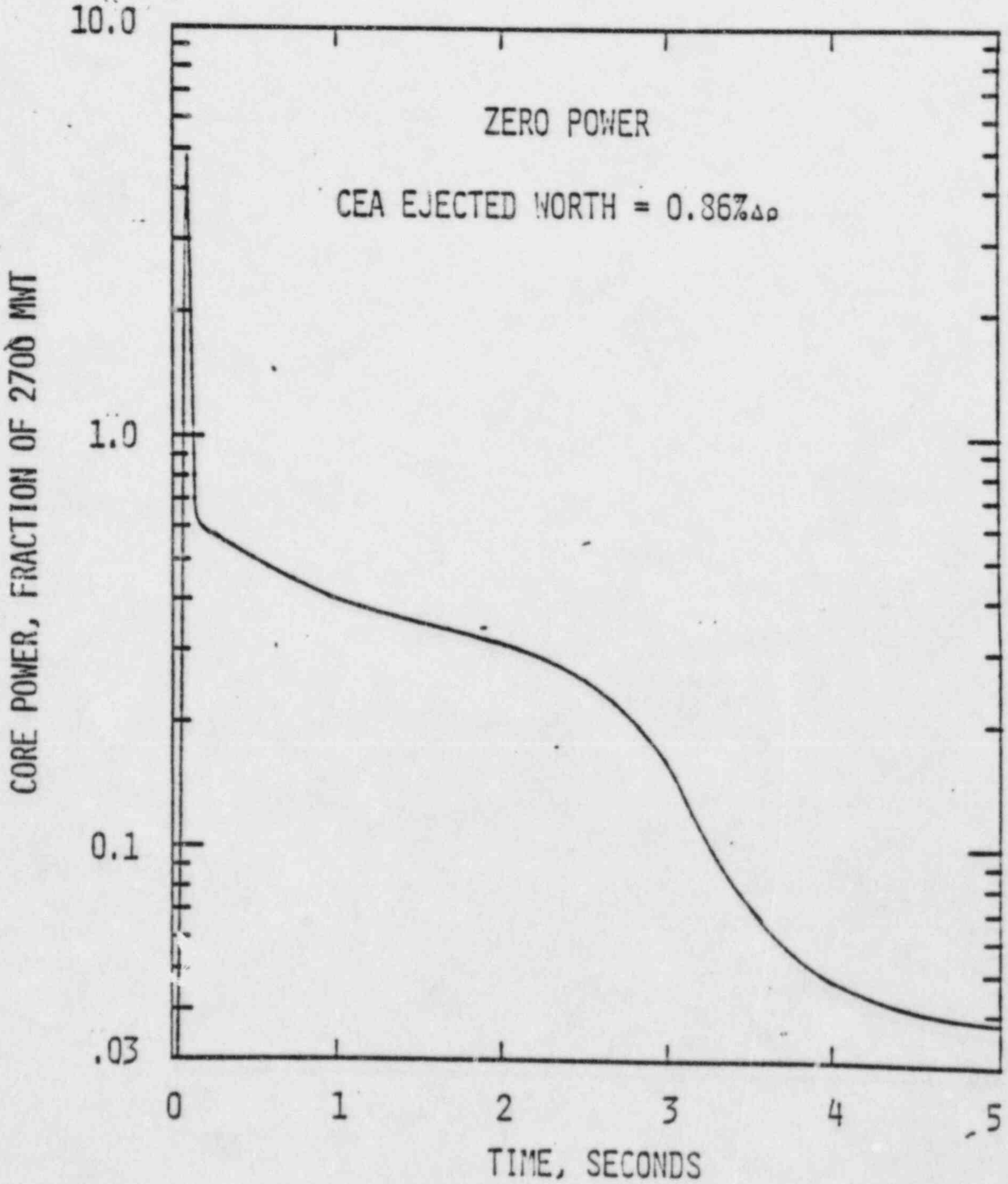
Parameter	Units	Unit 2* Cycle 5	Unit 1 Cycle 8
Core Power Level	MWt	1.	1.
Ejected CEA Worth	%	.63	.86
Post-Ejected Radial Power Peak	-	9.4	9.5
Axial Power Peak	-	1.60	1.64
CEA Bank Worth at Trip	$\%\Delta\rho$	-1.5	-1.5
Doppler Coefficient Multiplier	-	.85	.85
Moderator Temperature Coefficient	$X10^{-4}\Delta\rho/^{o}F$	.5	.7
Delayed Neutron Fraction	-	.0044	.0044

*Reference 9.

TABLE 7.3.1-2

CEA EJECTION EVENT RESULTS

	Unit 2 Cycle 5	Unit 1 Cycle 8
Zero Power		
Total Average Enthalpy of Hottest Pellet (cal/gm)	145.	<200
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	199.	<250
Fraction of Rods that Suffer Clad Damage (Average Enthalpy ≥ 200 cal/gm)	0	0
Fraction of Fuel Having at Least Incipient Centerline Melting (Centerline Enthalpy ≥ 250 cal/gm)	0	0
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy ≥ 310 cal/gm)	0	0



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

7.3.2 Steam Line Rupture Analysis

The steam line rupture (SLB) event has been reanalyzed for Calvert Cliffs Unit 1 Cycle 8. The purpose of this reanalysis is to extend the conservatively enveloping analysis performed in support of Unit 1 Cycle 7 to represent Unit 1 Cycle 8. There have been no changes in plant parameters or core characteristics which would impact the pre-trip SLB events; therefore, the results of the Unit 1 Cycle 7 analysis remain valid for this class of breaks. Changes did, however, occur in core characteristics which necessitated the reanalysis of the post trip steam line break. Previous analysis has shown that the consequences of an inside containment post-trip SLB are more adverse than those of the outside containment breaks due to the action of the flow restrictors. The post-trip inside containment breaks were initiated from both Hot Full Power (HFP) and Hot Zero Power (HZP) and each was performed with and without a Loss of AC (LOAC) power on turbine trip. The acceptance criteria for this postulated accident is that the site boundary doses will be within the 10CFR100 guidelines and that coolable geometry will be maintained.

Analysis Assumptions and Initial Conditions for SLB Inside Containment

The SLB event was initiated from the conditions listed in Table 7.3.2-1. The moderator temperature coefficient (MTC) of reactivity assumed in the analysis corresponds to end of cycle, since this MTC results in the greatest positive reactivity change during the RCS cooldown caused by the steam line rupture. Since the reactivity change associated with moderator feedback varies significantly over the moderator density covered in the analysis, a curve of reactivity insertion versus density rather than a single value of MTC, is assumed in the analysis. The moderator cooldown curve assumed in the analysis is given in Figure 7.3.2-1. This moderator cooldown curve was conservatively calculated assuming that on reactor trip the control element assembly which yields the most severe combination of scram worth and reactivity insertion is stuck in the fully withdrawn position.

The reactivity change associated with the fuel temperature decrease was also based on an end of cycle Doppler coefficient because this fuel temperature coefficient (FTC), in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the steam line rupture event. The Doppler multiplier on the FTC assumed in the analysis is given in Table 7.3.2-1. The β fraction assumed was the maximum absolute value including uncertainties for end of cycle conditions. This too is conservative since it maximizes subcritical multiplication and, thus, enhances the potential for Return-To-Power (R-T-P). The analysis also assumed a conservatively low value of boron reactivity worth of $-1.0\%\Delta\rho$ per 85 PPM for safety injection flow from the High Pressure Safety Injection pumps.

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip is $6.33\%\Delta\rho$ at the maximum allowed power level and $3.5\%\Delta\rho$ at zero power. This available scram worth was calculated for the stuck rod which produced the moderator cooldown curve in Figure 7.3.2-1.

During a return-to-power, negative reactivity credit was assumed in the analysis. This negative reactivity credit is due to the local heatup of the inlet fluid in the hot channel, which occurs near the location of the stuck CEA. This credit is based on three-dimensional coupled neutronic-

thermal-hydraulic calculations performed with the HERMITE/TORC code (References 10 and 11). The magnitude of the credit is similar to that used in the Calvert Cliffs Unit 1 Cycle 7 steam line break event (Reference 1). 3-D power distribution peaks (Fq) were also determined by the HERMITE/TORC methodology. The limiting SLB case in this analysis was found to be the Hot Full Power, non-LOAC case. The Fq used for this case corresponded to operation at the extremes of the HFP ASI LCO limits. The actual value of Fq is a function of both fission power and core flow.

The analysis only credited the low steam generator pressure trip. An analysis trip setpoint of 600.0 psia was assumed in the analysis. This represents the Technical Specification setpoint of 685.0 psia and an uncertainty of 85.0 psia. The analysis also assumed that a Main Steam Isolation Signal (MSIS) is generated when secondary pressure reaches 600.0 psia. This represents the Technical Specification setpoint of 685.0 psia and an uncertainty of 85.0 psia. A Main Steam Isolation Valve (MSIV) closure time of 6.9 seconds (includes valve closure time and signal processing delay time) was conservatively assumed in the analysis.

The analysis conservatively assumed that following reactor trip, the main feedwater flow is ramped down to 8% of full power feedwater flow in 20 seconds and that the main feedwater isolation valves are completely closed in 80 seconds after a low steam generator pressure or a main steam isolation signal. These assumptions are consistent with Technical Specification limits.

The analysis assumptions regarding the auxiliary feedwater actuation setpoint, the associated time delays, and the AFW flow through each leg are given below. They were conservatively chosen to initiate AFW flow sooner and deliver the maximum AFW flow to the ruptured steam generator, which maximizes the primary cooldown and enhances the potential R-T-P.

The auxiliary feedwater Technical Specification actuation setpoint is 45% of steam generator level wide range indication with an uncertainty of ±18%. Auxiliary feedwater (AFW) was conservatively assumed to initiate at time of reactor trip, which in all cases resulted in AFW initiation at a level far above the Technical Specification actuation setpoint plus uncertainties. This was done to ensure the analysis results would remain bounding in the event of any future revision of the Technical Specification or uncertainty. Time delays associated with the AFW pumps were conservatively set to zero resulting in instantaneous flow, even in loss of AC cases. This is consistent with the enveloping nature of the analysis.

All flow from the AFW pumps is conservatively directed to the damaged steam generator until automatic isolation of that steam generator. AFW pump flow is assumed to be at a runout value of 1300 gpm.

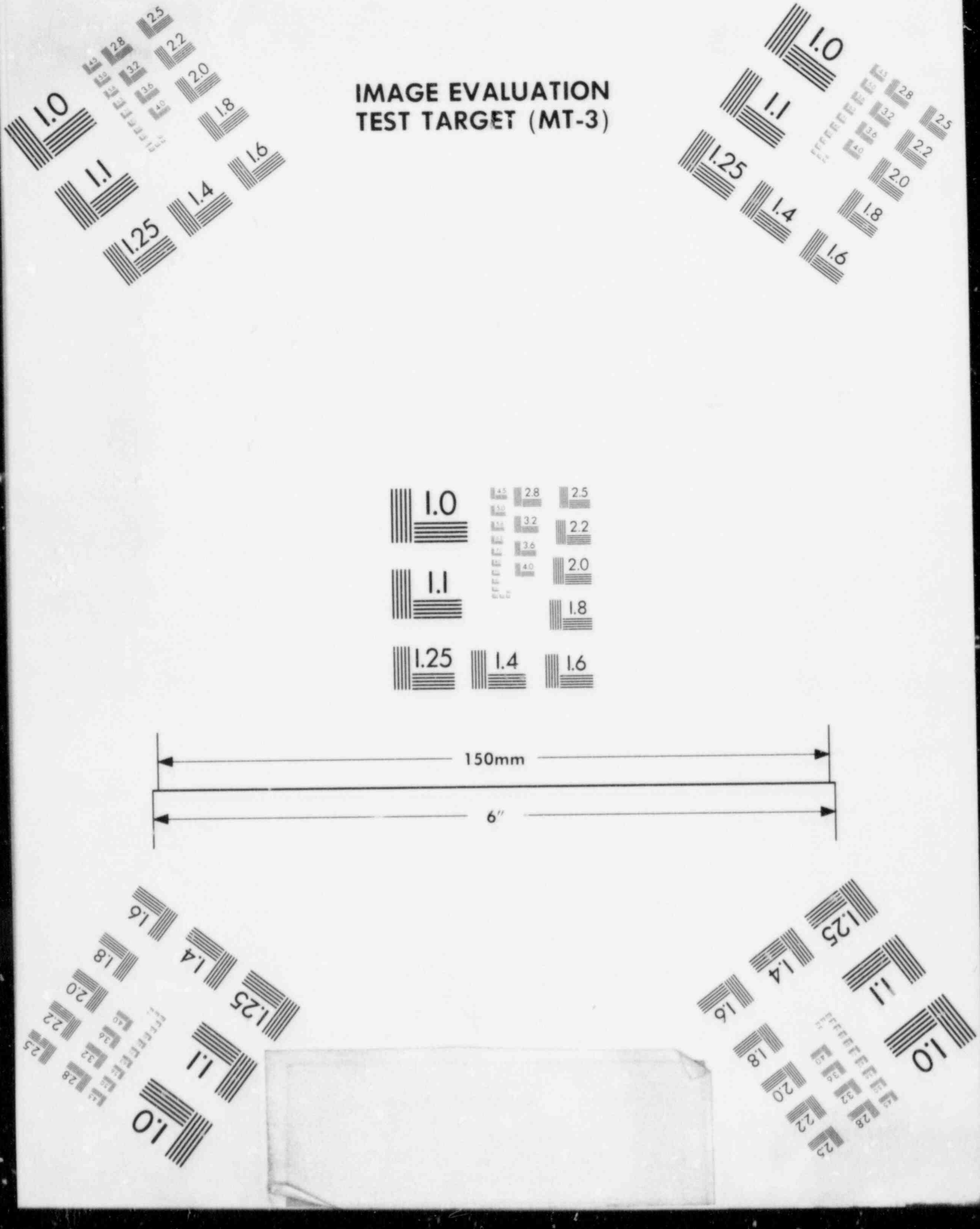
The analysis also included isolation of the ruptured steam generator when the steam generator differential pressure reached the analysis setpoint of 365.0 psid. This represents a Technical Specification setpoint of 135.0 psid and an uncertainty of 230.0 psid. In addition, a 20.0 second time delay was assumed in the analysis to close the AFW isolation (i.e., block) valves. These assumptions are conservative since it delays the isolation of AFW to the ruptured steam generator.

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

IMAGE EVALUATION
TEST TARGET (MT-3)



A safety injection actuation analysis setpoint of 1578.0 psia, which is conservative compared to current Technical Specifications and existing uncertainties, was assumed in the analysis. The analysis conservatively assumed that on a Safety Injection Actuation Signal (SIAS), only one High Pressure Safety Injection (HPSI) pump starts. In addition, a maximum time delay of 30 seconds for HPSI pumps to accelerate to full speed was assumed in the analysis. In case of LOAC power, additional time delays were included in the analysis. It included 10.0 seconds for the diesel generators to start and reach speed following the LOAC and 5.0 seconds for the HPSI pump to be loaded on line regardless of which sequencer (i.e., shutdown or LOCA) was initiated.

The post-trip minimum DNBRs were calculated using the MacBeth correlation (Reference 12) with the Lee non-uniform mixing correlation factor (Reference 13).

Results for SLB Inside Containment

It was found that the SLB events with Loss of AC (LOAC) power on turbine trip, initiated from either Hot Full Power or Hot Zero Power conditions, do not approach the SAFDL on DNBR as closely as did the LOAC cases in the Unit 1 Cycle 7 analysis; consequently, the Unit 1 Cycle 7 results remain valid. This is due to improved modeling of the coolant flow in the hot channel under natural circulation conditions based upon the results of the HERMITE/TORC methodology. This improved flow is used together with the MacBeth correlation in the calculation of the minimum DNBR value.

The SLBs without a LOAC do not approach the SAFDL on DNBR. They do, however, present a challenge to the Linear Heat Generation Rate (LHGR) SAFDL. This occurs because the higher core flow reduces the negative reactivity inserted by the local heatup of the moderator in the hot channel. This results in a maximum post trip LHGR. Therefore, the results of the largest inside containment SLB without LOAC on turbine trip are presented herein.

The sequence of events for the 6.305 ft^2 SLB without LOAC on turbine trip initiated from HFP conditions is given in Table 7.3.2-2. The break size of 6.305 ft^2 was determined in Reference 1 to be the most severe. The reactivity insertion as a function of time is presented in Figure 7.3.2-1. The NSSS responses during the transient are given in Figures 7.3.2-2 through 7.3.2-7.

The results of the analysis show that the HFP SLB causes the secondary pressure to rapidly decrease until a reactor trip on low steam generator pressure is initiated at 2.5 seconds. The CEAs drop into the core at 3.9 seconds and terminate the power and heat flux increases. Auxiliary feedwater is initiated at runout flow to the damaged side steam generator at time of trip. At 21.8 seconds one HPSI pump is loaded on line and at 51.8 seconds the HPSI pump reaches full speed.

The Steam Generator Isolation Analysis Setpoint is reached at 2.5 seconds. At 3.4 seconds, the MSIVs begin to close and are completely closed at 9.4 seconds. The blowdown from the intact steam generator is terminated at this time.

An AFW isolation signal based on steam generator differential pressure is initiated at 8.5 seconds. At 27.9 seconds, the AFW block valves associated with the steam generator with the lowest pressure (i.e., ruptured steam generator) are completely closed.

The continued blowdown from the ruptured steam generator causes the core reactivity to approach criticality. The ruptured steam generator blows dry at 67.6 seconds, which terminates the cooldown of the RCS. A peak reactivity of -.42%$\Delta\rho$ at 70.0 seconds is obtained. A peak R-T-P of 9.29% consisting of 5.36% fission power and 3.93% decay power, is produced at 66.3 seconds. Less than one percent of the fuel exceeds the centerline melt limit; the DNBR limit is not approached for this event.

Conclusions

Site boundary doses are bounded by the outside containment doses calculated for Calvert Cliffs Unit 1 Cycle 7 which remain valid. The DNBR results of the steam line break, inside containment of the Unit 1 Cycle 7 analysis continue to envelope the Unit 1 Cycle 8 results. The LHGR results of the steam line break, inside containment predict that less than one percent of the fuel would exceed the centerline melt limit, thus, ensuring a coolable geometry. Therefore, the results of the outside and inside containment SLB events are acceptable for Calvert Cliffs Unit 1 Cycle 8.

TABLE 7.3.2-1

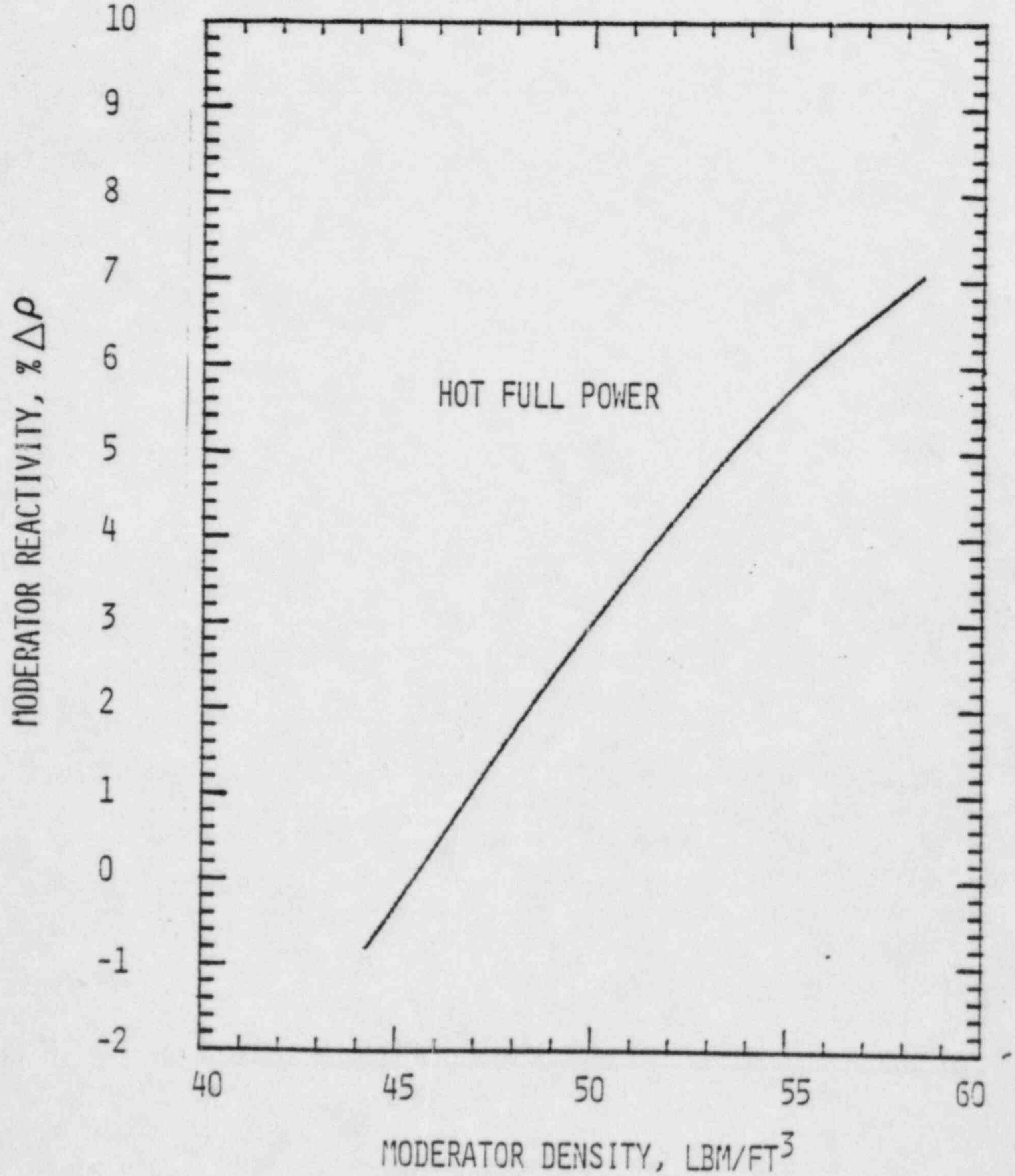
KEY PARAMETERS ASSUMED IN THE INSIDE CONTAINMENT
STEAM LINE BREAK EVENT INITIATED FROM HFP

Parameter	Units	Value
Initial Core Power	MWt	2754.0
Initial Core Inlet Temperature	^{o}F	550.0
Initial RCS Pressure	psia	2300.0
Initial Steam Generator Pressure	psia	860.0
Low Steam Generator Pressure Trip Setpoint	psia	600.0
Steam Generator Differential Pressure Setpoint	psid	365.0
Safety Injection Actuation Signal	psia	1578.0
Minimum CEA Worth Available at Trip	$\%\Delta\rho$	-6.33
Doppler Multiplier		1.30
Moderator Cooldown Curve	$\%\Delta\rho$ vs. density	See Figure 7.3.2-1
Inverse Boron Worth	PPM/$\%\Delta\rho$	85.0
Effective MTC	$x10^{-4}\Delta\rho/^{o}F$	-2.7
β Fraction (including uncertainty)		.0060

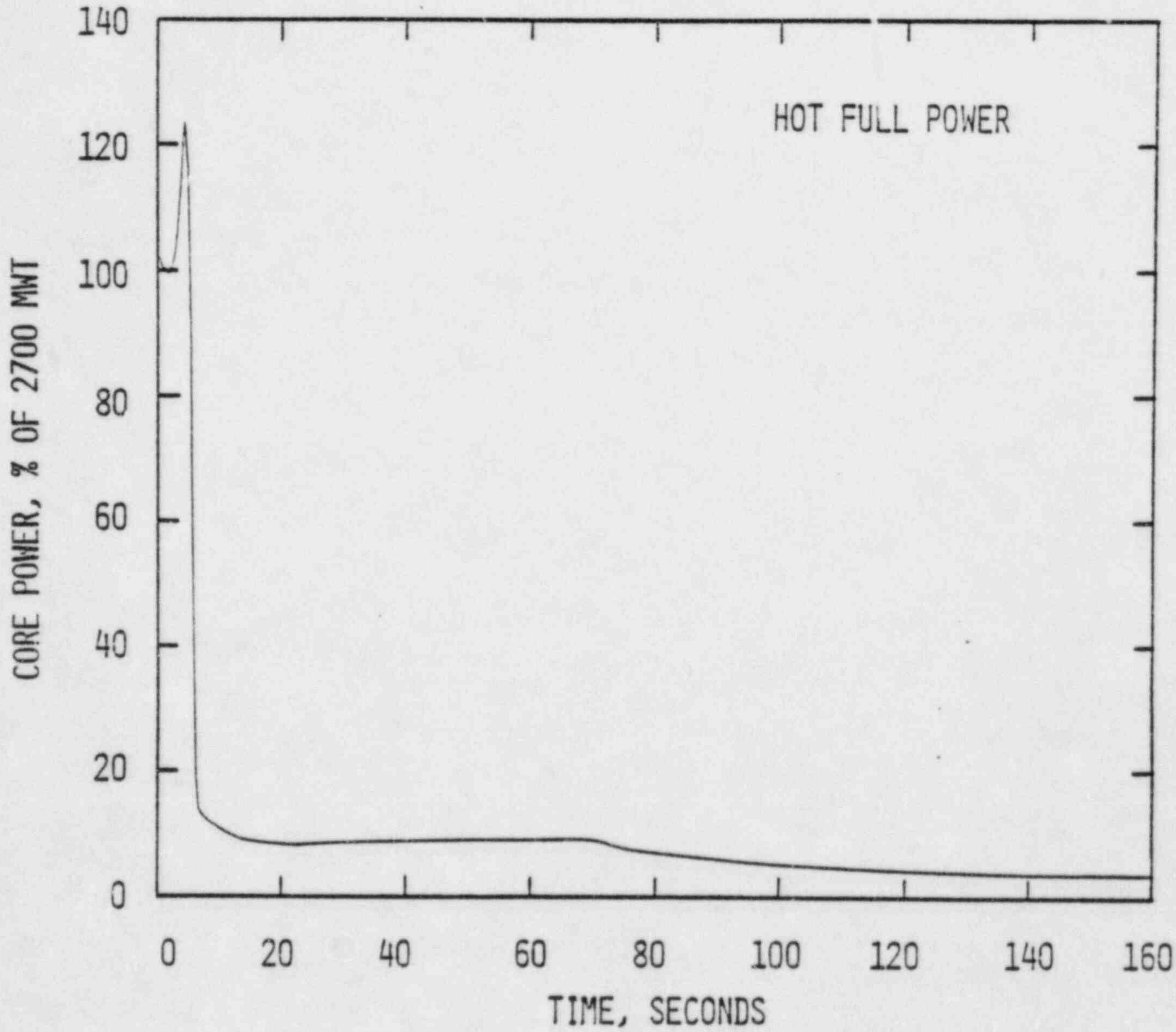
TABLE 7.3.2-2

HFP 6.305 FT^2 BREAK
WITHOUT LOAC, INSIDE CONTAINMENT

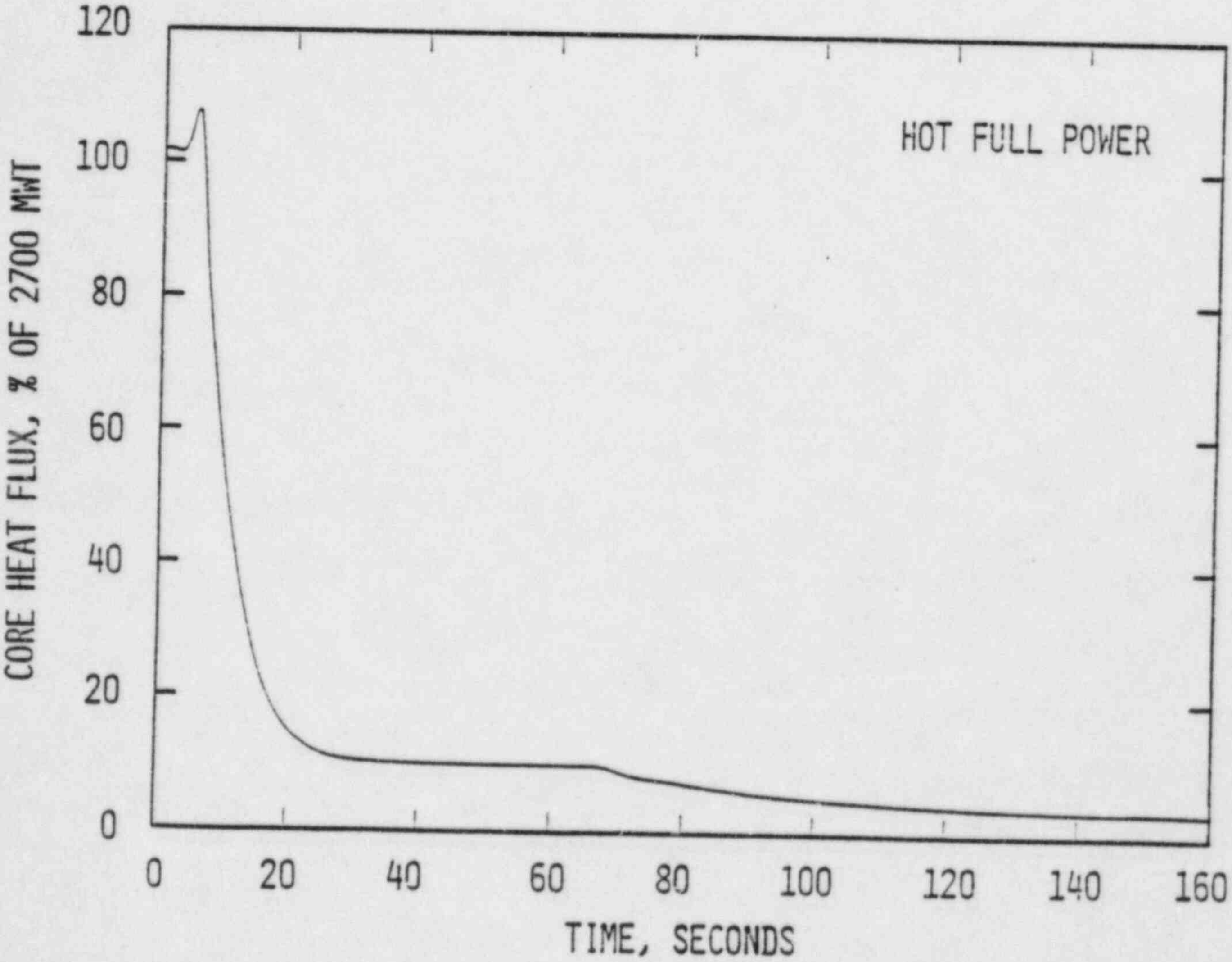
Time (sec)	Event	Setpoint or Value
0.0	Steam Line Break Occurs	6.305 ft^2
2.5	Low Steam Generator Pressure Analysis Trip Setpoint is Reached; Steam Generator Isolation Analysis Setpoint is Reached;	600.0 psia
3.4	Trip Breakers Open; Main Feed Rampdown Begins; MSIVs Begin to Close	
3.9	CEAs Enter Core	
8.5	Steam Generator Differential Pressure Setpoint Reached	365.0 psid
9.4	Main Steam Isolation Valves Fully Closed	
21.4	Pressurizer Empties	---
21.8	Safety Injection Setpoint is Reached	1578 psia
27.9	AFW Block Valves Closed Providing Auxiliary Feedwater to Intact S.G. only	
51.8	Safety Injection Pumps up to Full Speed	
66.3	Peak Power	9.29% of 2700 MWt
67.6	Affected Steam Generator Blows Dry, RCS Cooldown Stopes	---
70.0	Peak Reactivity	-.42% $\Delta\rho$
83.4	Main Feedwater Isolation Valves Completely Closed	---
115.0	Safety Injection Boron Begins Appearing in the Core in Increasing Amounts Ensuring Shutdown	---



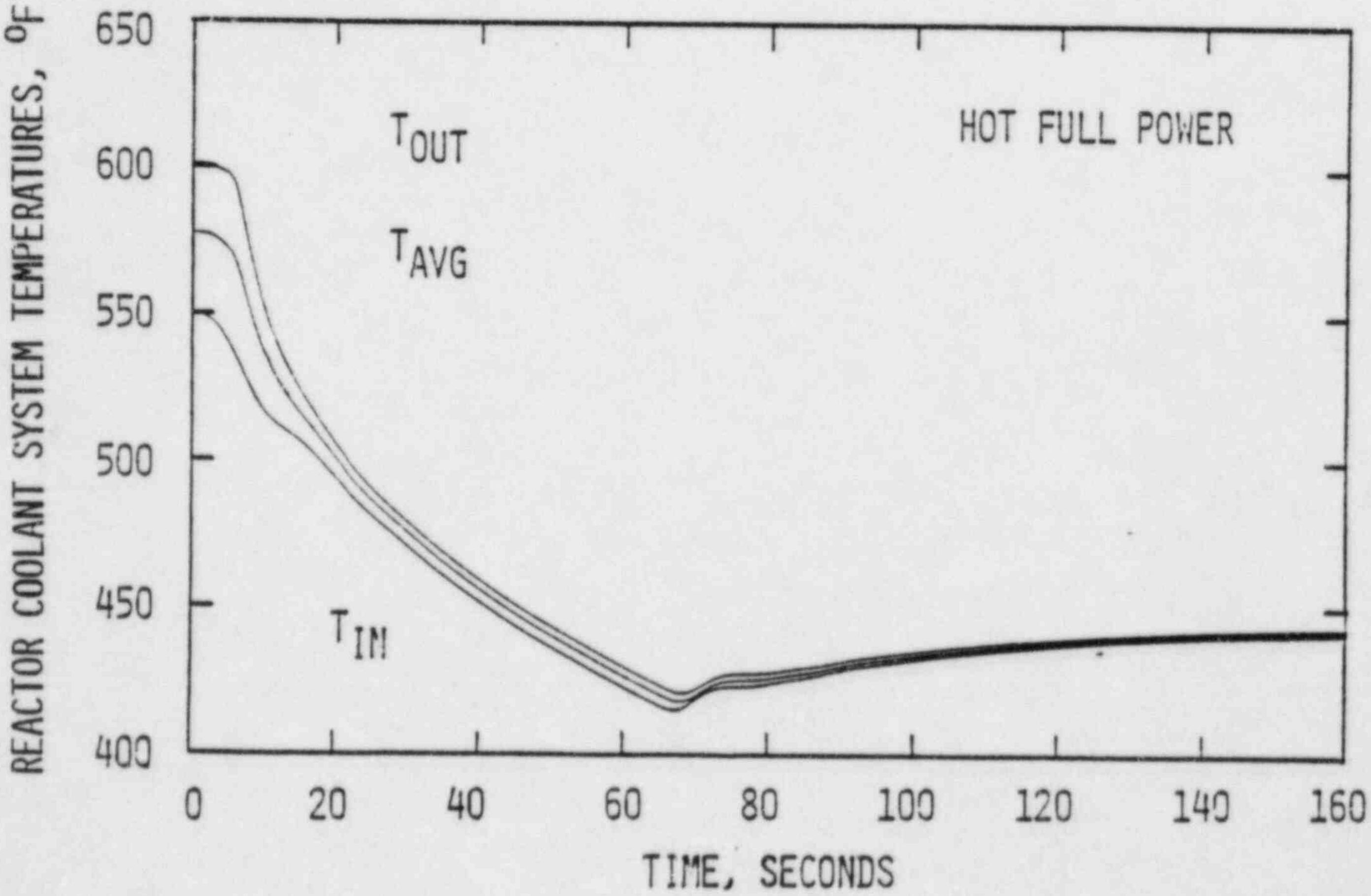
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT MODERATOR REACTIVITY VS MODERATOR DENSITY	FIGURE 7.3.2-1



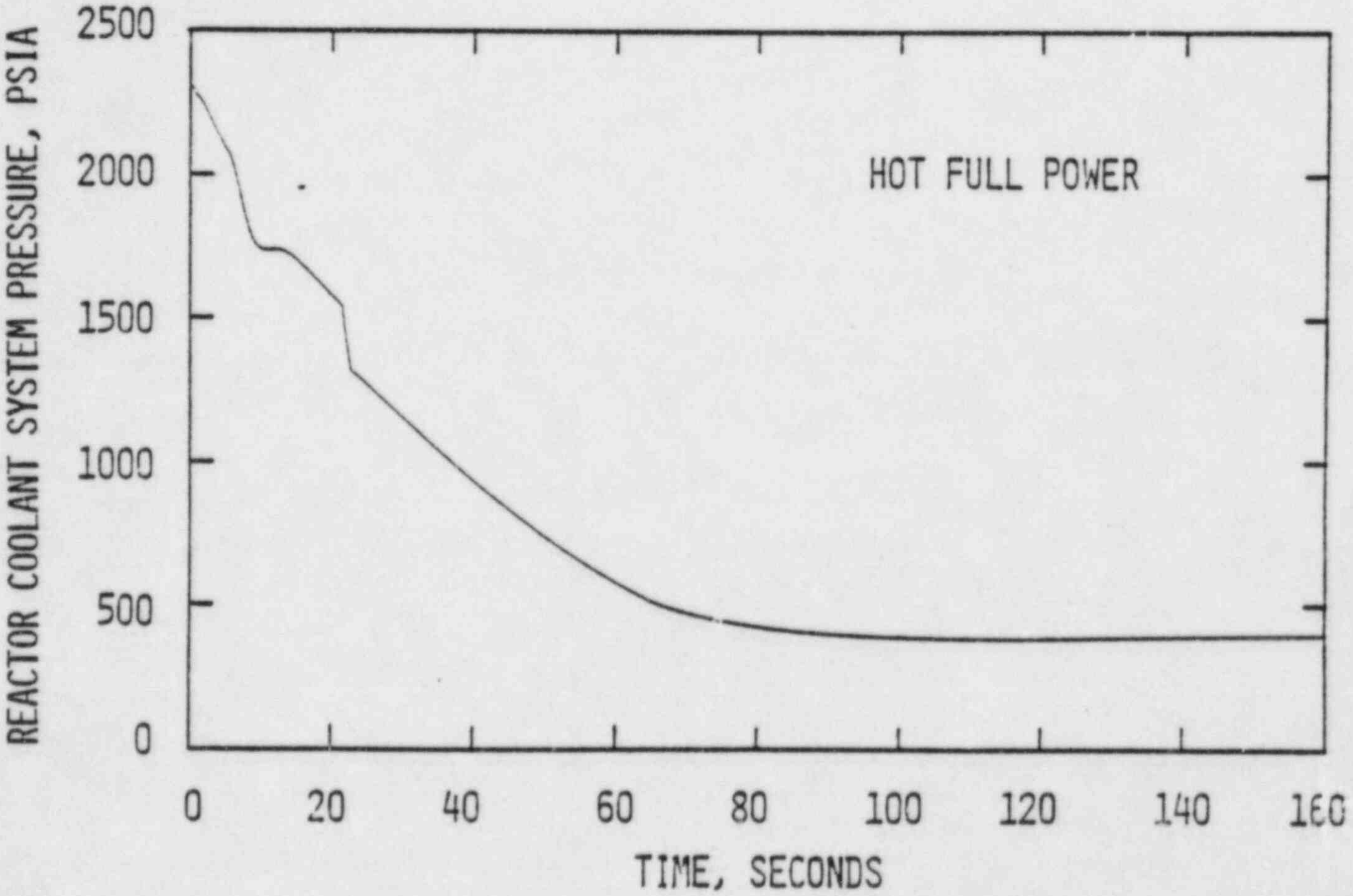
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT CORE POWER VS TIME	FIGURE 7.3.2-2



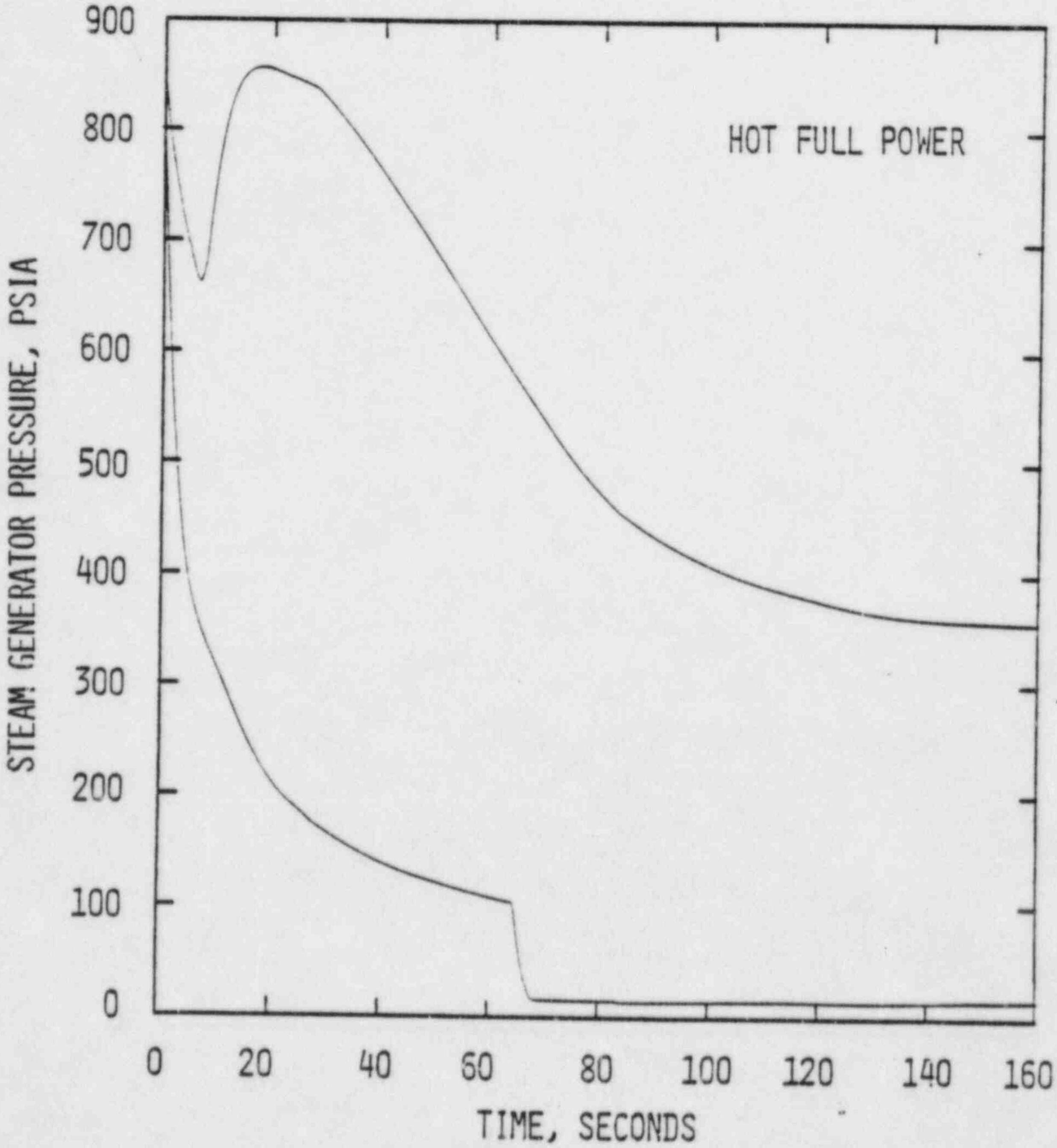
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT CORE HEAT FLUX VS TIME	FIGURE 7.3.2-3



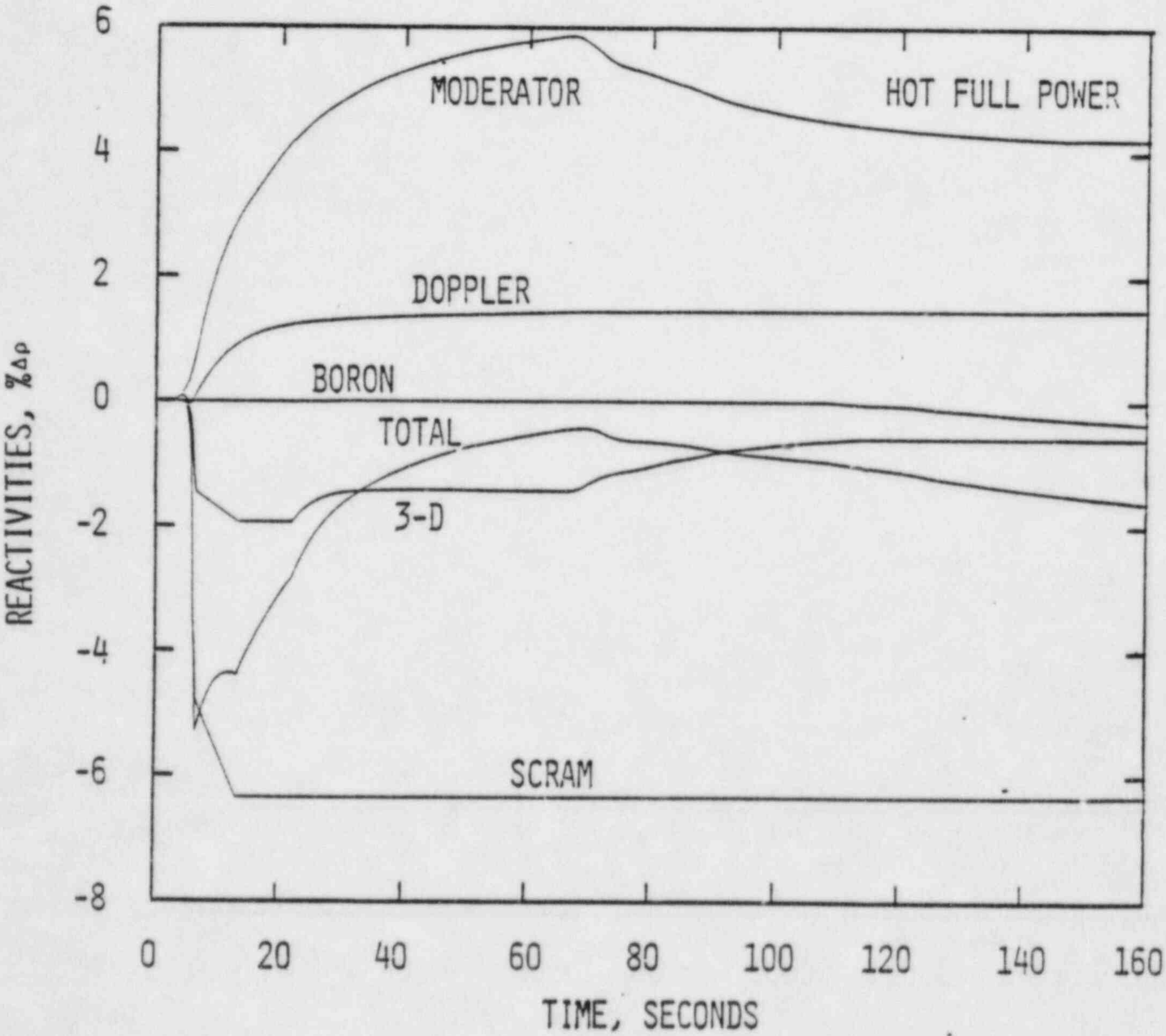
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7.3.2-4



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7.3.2-5



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 7.3.2-6



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT REACTIVITIES VS TIME	FIGURE 7.3.2-7

8.0 ECCS ANALYSIS

8.1 Large Break Loss-of-Coolant Accident

8.1.1 Introduction and Summary

An ECCS performance analysis was performed for Calvert Cliffs Unit 1 Cycle 8 to demonstrate compliance with 10CFR50.46 which presents the NRC Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors (Reference 1). The analysis justifies an allowable Peak Linear Heat Generation Rate (PLHGR) of 15.5 kw/ft. This PLHGR is equal to the existing limit for Calvert Cliffs Unit 1. The method of analysis and detailed results which support this value are presented in the following sections.

8.1.2 Method of Analysis

The NRC approved C-E large break evaluation model[2] was used to re-evaluate ECCS performance for the limiting large break LOCA. The blowdown hydraulic calculations employed in the cycle 6 evaluation[3], the reference cycle, apply to Cycle 8 since there have been no significant changes to RCS hardware characteristics. Refill/reflood hydraulic calculations were performed for Cycle 8 to account for the lowering of the minimum containment pressure from 14.7 psia to 13.7 psia and for the reduction in the minimum HPSI flow. These computations were performed using the NRC approved COMPERC-II[4] code. The hot rod clad temperature and oxidation calculations were performed to account for removal of the hot rod augmentation factors, the fuel rod conditions specific to Cycle 8 and the revised refill/reflood hydraulics. The NRC approved STRIKIN-II[5] code was used for this purpose.

Burnup dependent calculations were performed to determine the limiting conditions for the ECCS performance analysis. The nuclear and fuel thermal performance (FATES3)[6] data used as input to the ECCS analysis considered high burnup effects specific to the Cycle 8 reload. Two STRIKIN-II temperature calculations were performed: One at a low rod average burnup which yields the maximum initial fuel stored energy and one at a high burnup which yields the highest initial rod pressure. The low burnup maximum fuel stored energy case was demonstrated to be limiting.

The temperature calculations for both cases were performed for the 1.0 DES/PD* break. The break spectrum analysis performed for Unit 1 Cycle 2[7] determined that the 1.0 DES/PD is the limiting break since it yields the highest residual fuel stored energy at the end of the blowdown period and, therefore, yields the highest peak cladding temperature during the reflood period.

The NRC approved PARCH[8] code was utilized in the cycle 8 evaluation to produce the steam cooling heat transfer coefficients during the late reflood period when reflood rates are below one inch per second. Previous reload cycle analyses used a more simplified but unnecessarily conservative constant minimum steam heat transfer value. As a result of this analysis improvement, the peak clad temperatures reported herein are significantly reduced.

* DES/PD - Double Ended Slot at Pump Discharge

8.1.3 Results

Table 8.1-1 summarizes the results calculated for the two rod average burnup cases selected. A summary of the fuel parameter input values is shown in Table 8.1-2. For comparison purposes, the corresponding values of the reference cycle analysis (Unit 1 Cycle 6) are also presented in Tables 8.1-1 and 8.1-2. A list of the significant parameters displayed graphically for the limiting case (Figures 8.1-1 through 8.1-9) is presented in Table 8.1-3.

The decrease in intial containment pressure in the refill/reflood hydraulic analysis resulted in a lower transient minimum containment pressure as shown on Figure 8.1-7. Consequently, slightly lower reflood rates (Figure 8.1-8) and slightly lower heat transfer coefficients (Figure 8.1-5) were calculated in comparison to the reference analysis.

The results of the evaluation confirm that 15.5 kw/ft is an acceptable value for the PLHGR in Cycle 8. As shown in Table 8.1-1, the peak clad temperature, maximum local oxidation and core wide clad oxidation values of 1896°F, 5.2% and <.51%, respectively, are well below the 10CFR50.46 acceptance criteria limits of 2200°F, 17% and 1%, respectively.

As shown in Table 8.1-1, the burnup with the maximum initial stored energy in the fuel (943 MWD/MTU) resulted in the highest peak clad temperature. The transient results for the limiting case are presented in Figures 8.1-1 to 8.1-9. The fuel cladding is predicted to rupture, as well as achieve its peak temperature, during the reflood period (at 37 seconds and 240 seconds, respectively).

The high burnup (50,000 MWD/MTU) case resulted in a peak clad temperature of 1887°F, 9°F lower than that for the maximum initial fuel stored energy case. This case assumed that the rod operates at a PLHGR of 15.5 kw/ft. This assumption is very conservative since a rod at such a high burnup would actually be at a power level significantly below 15.5 kw/ft. At 15.5 kw/ft the pin pressure was sufficient to cause an earlier hot rod rupture time. However, the lower initial stored energy at that burnup causes a lower peak clad temperature than the case at 943 MWD/MTU, as shown in Table 8.1-1.

8.1.4 Conclusions

As discussed above, conformance to the ECCS criteria is summarized by the analysis results presented in Table 8.1-1. The most limiting case results in a peak clad temperature of 1896°F, which is well below the acceptance limit of 2200°F. The maximum local and core wide values for zirconium oxidation percentages, as shown in Table 8.1-1, remain well below the acceptance limit values of 17% and 1%, respectively. Therefore, operation of Unit 1 Cycle 8 at a PLHGR of 15.5 kw/ft and a power level of 2754 MWT (102% of 2700 MWT) results in compliance with the 10CFR50.46 acceptance criteria.

Table 8.1-1

SUMMARY OF ECCS PERFORMANCE RESULTS FOR CALVERT CLIFFS 1 CYCLE 8 FOR THE LIMITING BREAK SIZE (1.0 DES/PD)

Parameter	Limiting Case (Maximum Initial Fuel Stored Energy)		High Burnup Case (Maximum Initial Rod Pressure)	
	Unit 1 Cycle-6	Unit 1 Cycle-8	Unit 1 Cycle-6	Unit 1 Cycle-8
Rod Average Burnup, MWD/MTU	3000	943	34,000	50,000
Peak Clad Temperature (PCT), °F	2038	1896	2024	1887
Time of PCT, seconds	249	240	248	237
Time of clad rupture, seconds	31.9	36.5	10.6	10.3
Peak local oxidation, %	8.5	5.20	8.4	5.31
Core wide oxidation, %	< 0.51	< 0.51	< 0.51	< 0.51

Table 8.1-2

CALVERT CLIFFS 1 CYCLE 8
FUEL PARAMETERS

Quantity	Unit 1 Cycle-6 Value	Unit 1 Cycle-8 Value
Reactor Power Level (102% of Nominal) (Mwt)	2754	2754
Average Linear Heat Rate (102% of Nominal) (kw/ft)	6.44	6.37
Hot Channel Peak Linear Heat Generation Rate (kw/ft)	15.5	15.5
Hot Assembly Peak Linear Heat Generation Rate (kw/ft)	13.14	12.52
*Gap Conductance at PLHGR (Btu/hr-ft^2-°F)	2025	2447
*Fuel Centerline Temperature at PLHGR (°F)	3634	3586
*Fuel Average Temperature at PLHGR (°F)	2213	2160
*Hot Rod Gas Pressure (Psia)	1251	1216
*Hot Rod Burnup (MWD/MTU)	3000	943
Hot Rod Augmentation Factor (Maximum)	1.04	1.00

*Initial fuel rod parameters, in STRIKIN-II, which yield the limiting ECCS performance results (maximum peak cladding temperature).

Table 8.1-3

CALVERT CLIFFS 1 CYCLE 8
ANALYSIS PLOTS FOR LIMITING CASE

Variable	Figure Number
Peak Clad Temperature	8.1-1
Hot Spot Gap Conductance	8.1-2
Peak Local Clad Oxidation	8.1-3
Temperature of Fuel Centerline, Fuel Average, Clad and Coolant at Hottest Node	8.1-4
Hot Spot Heat Transfer Coefficient	8.1-5
Hot Rod Internal Gas Pressure	8.1-6
Containment Pressure	8.1-7
Mass Added to Core During Reflood	8.1-8
Water Level in Downcomer	8.1-9

FIGURE 8.1-1

CALVERT CLIFFS I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

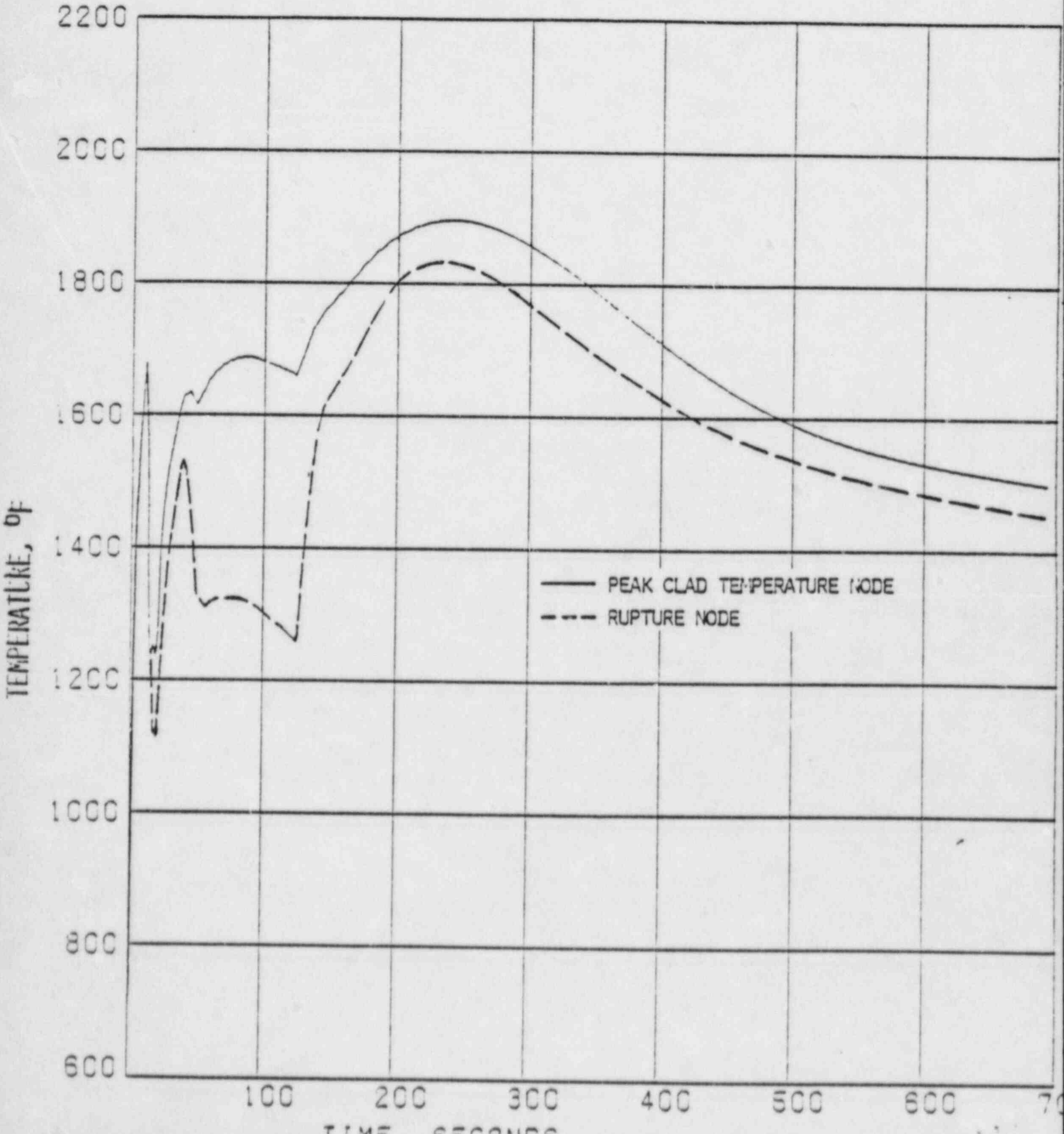


FIGURE 8.1-2

CALVERT CLIFFS I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT SPOT GAP CONDUCTANCE

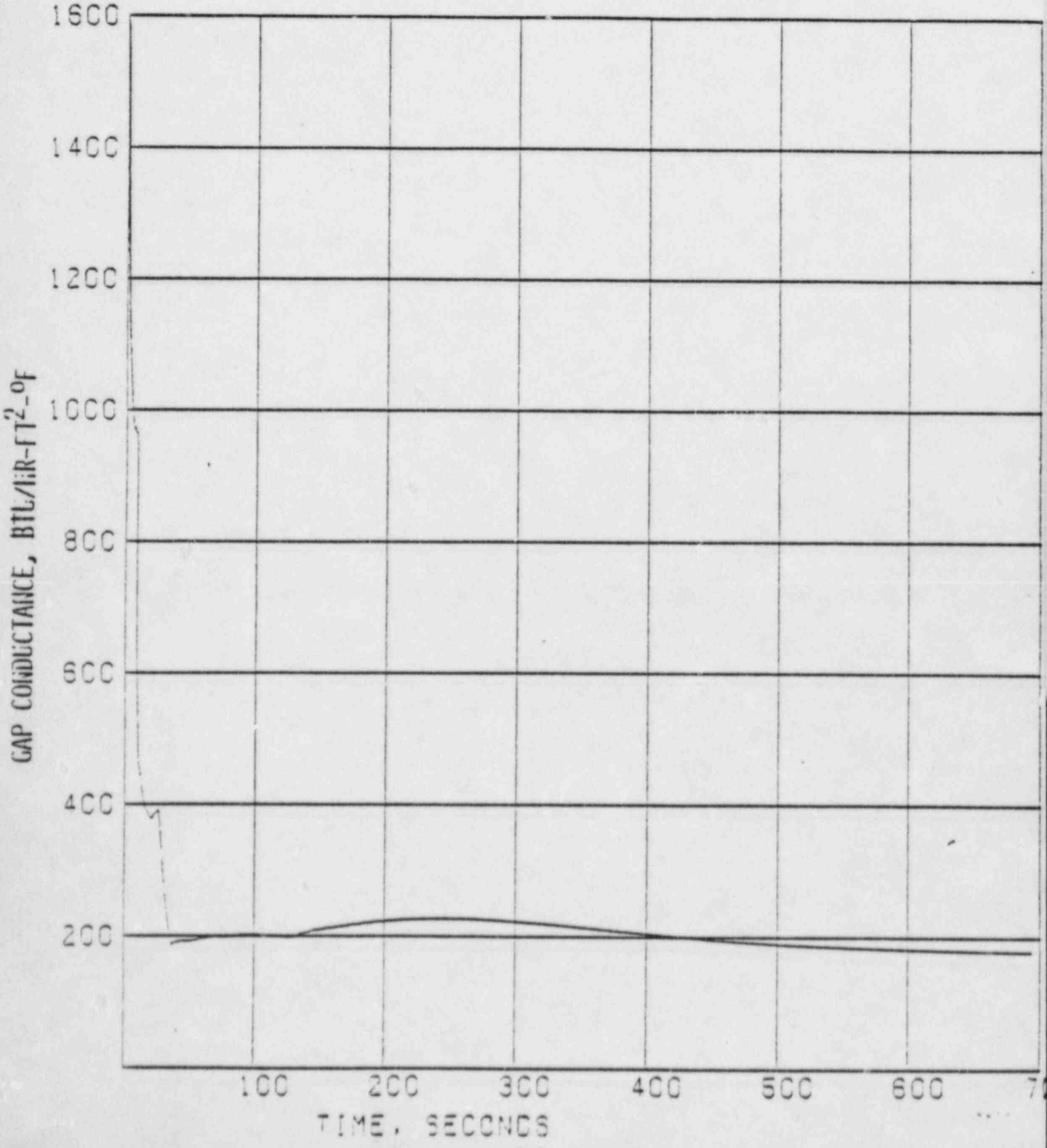


FIGURE 8.1-3

CALVERT CLIFFS I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK LOCAL CLAD OXIDATION

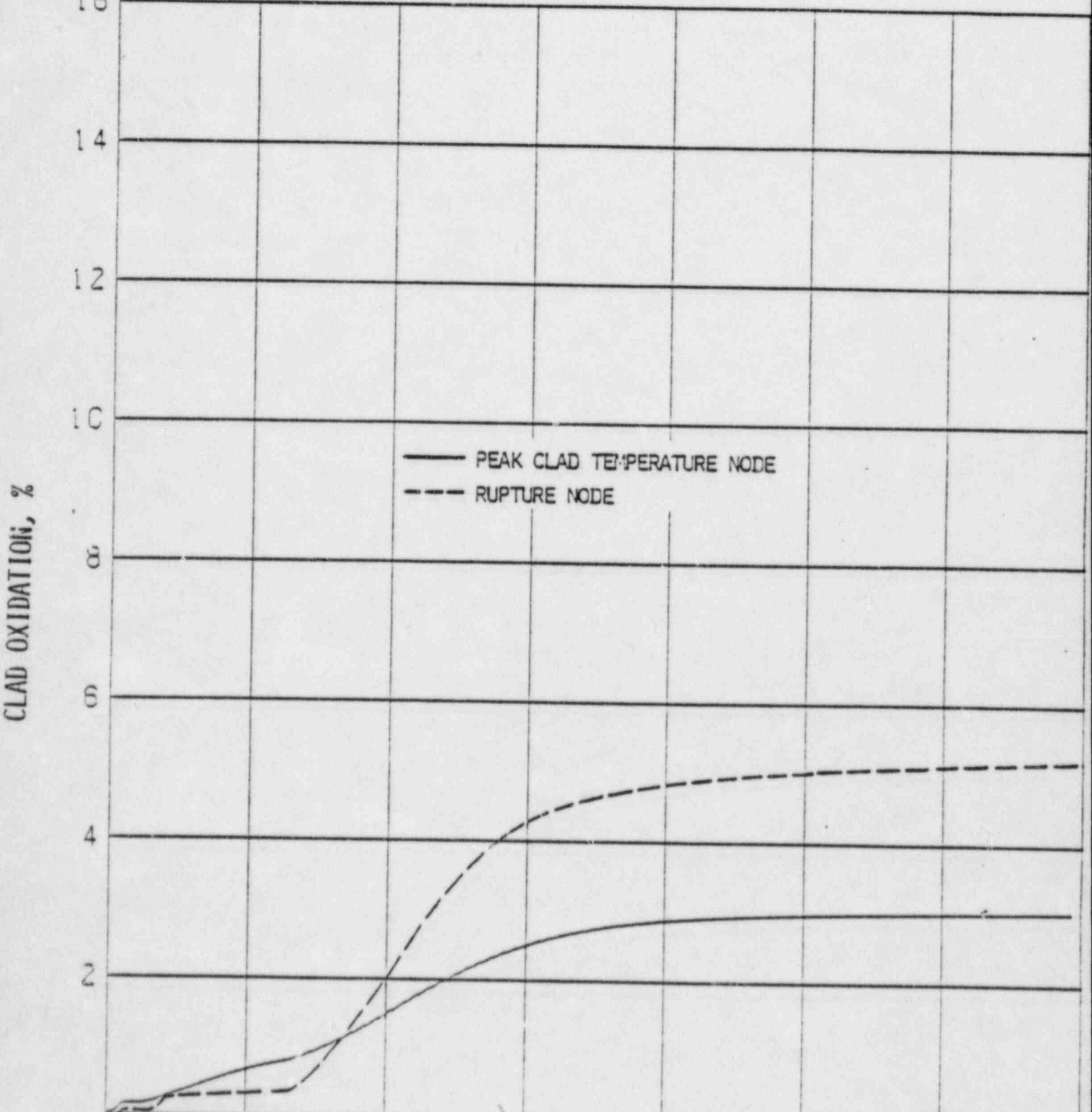


FIGURE 8.1-4

CALVERT CLIFFS I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
TEMPERATURE OF FUEL CENTERLINE, FUEL AVERAGE,
CLAD AND COOLANT OF HOTTEST NODE

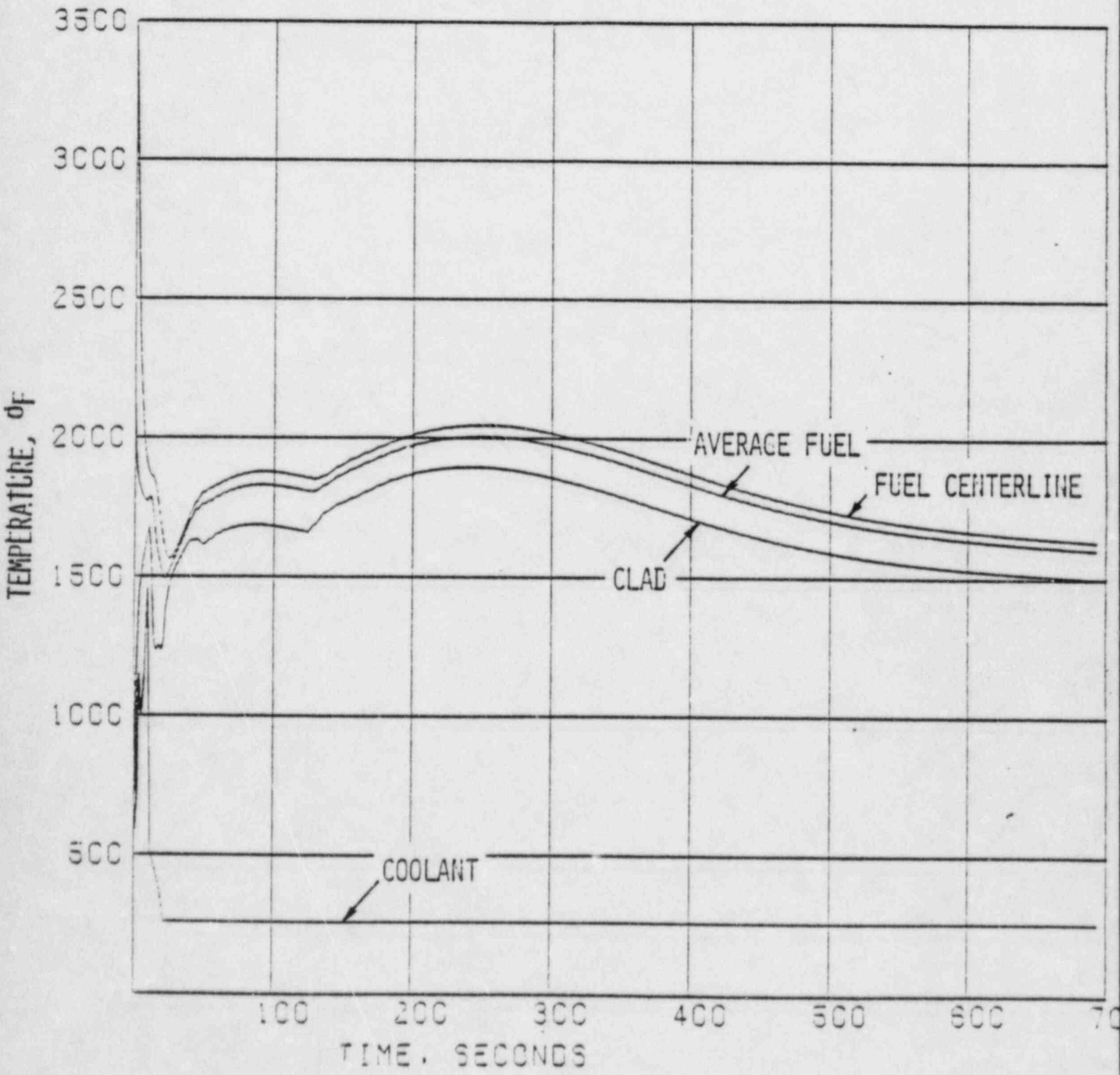


FIGURE 8.1-5

CALVERT CLIFFS I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT SPOT HEAT TRANSFER COEFFICIENT

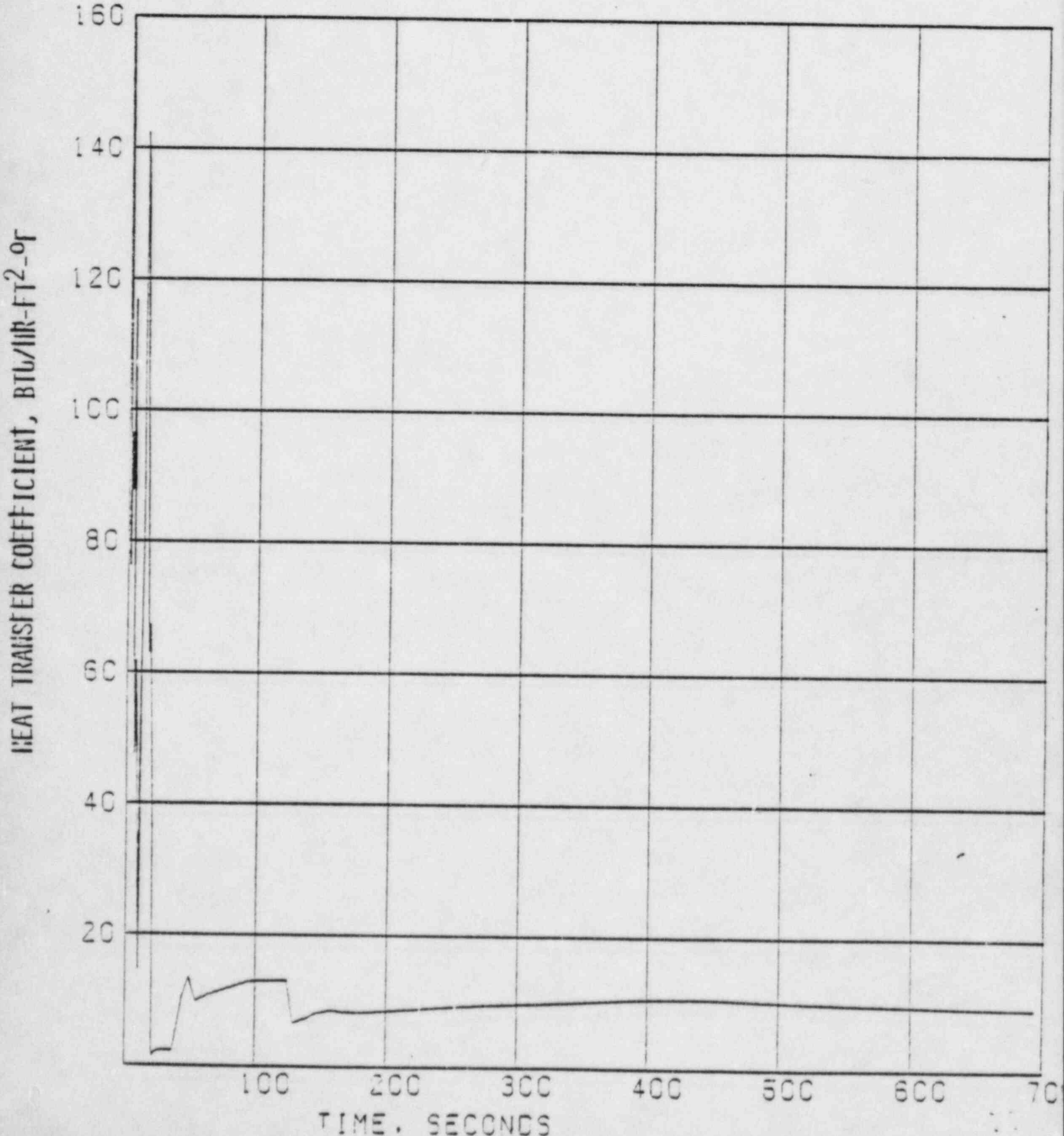


FIGURE 8.1-6

CALVERT CLIFFS UNIT I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT ROD INTERNAL GAS PRESSURE

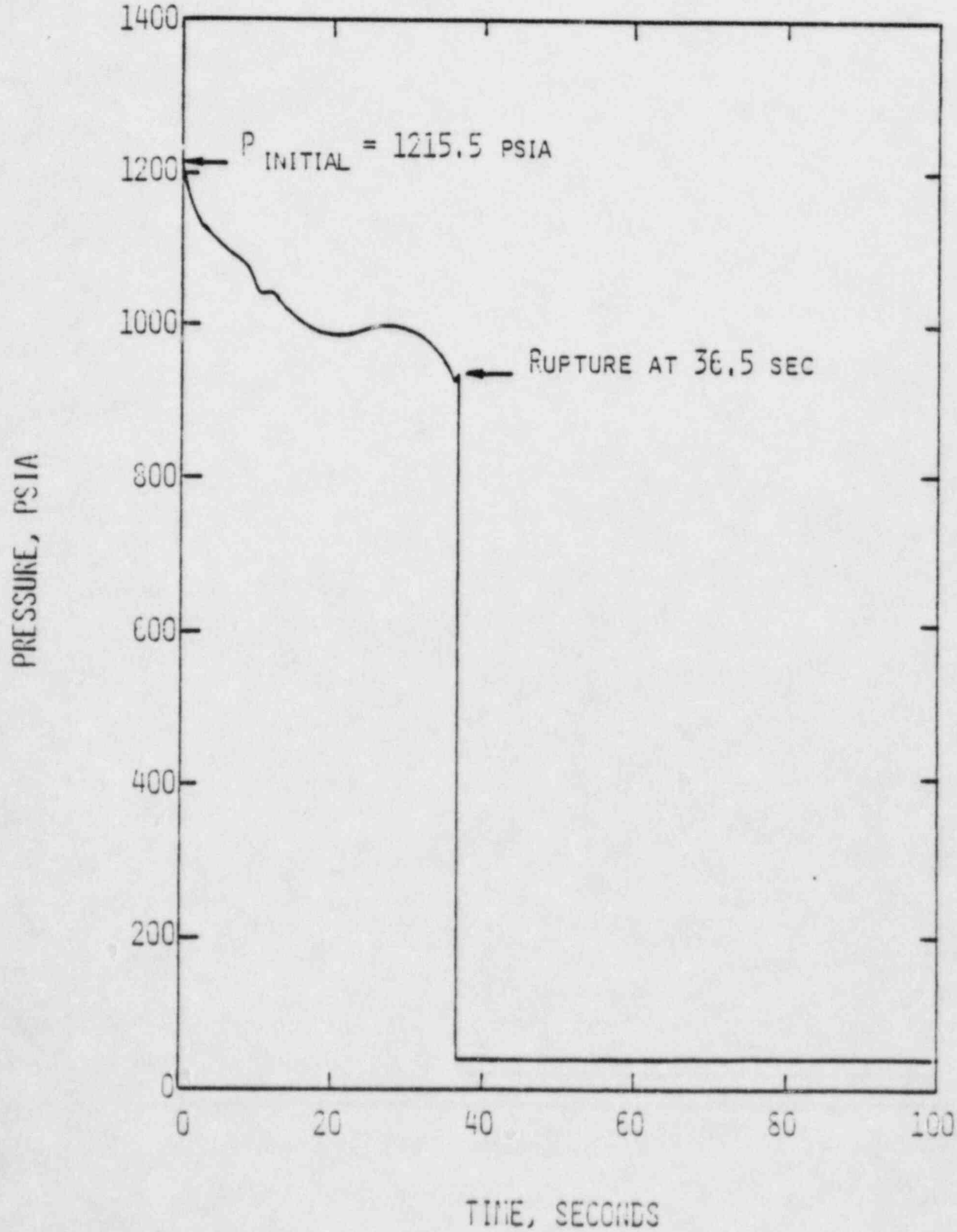


FIGURE 8.1-7

CALVERT CLIFFS UNIT I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

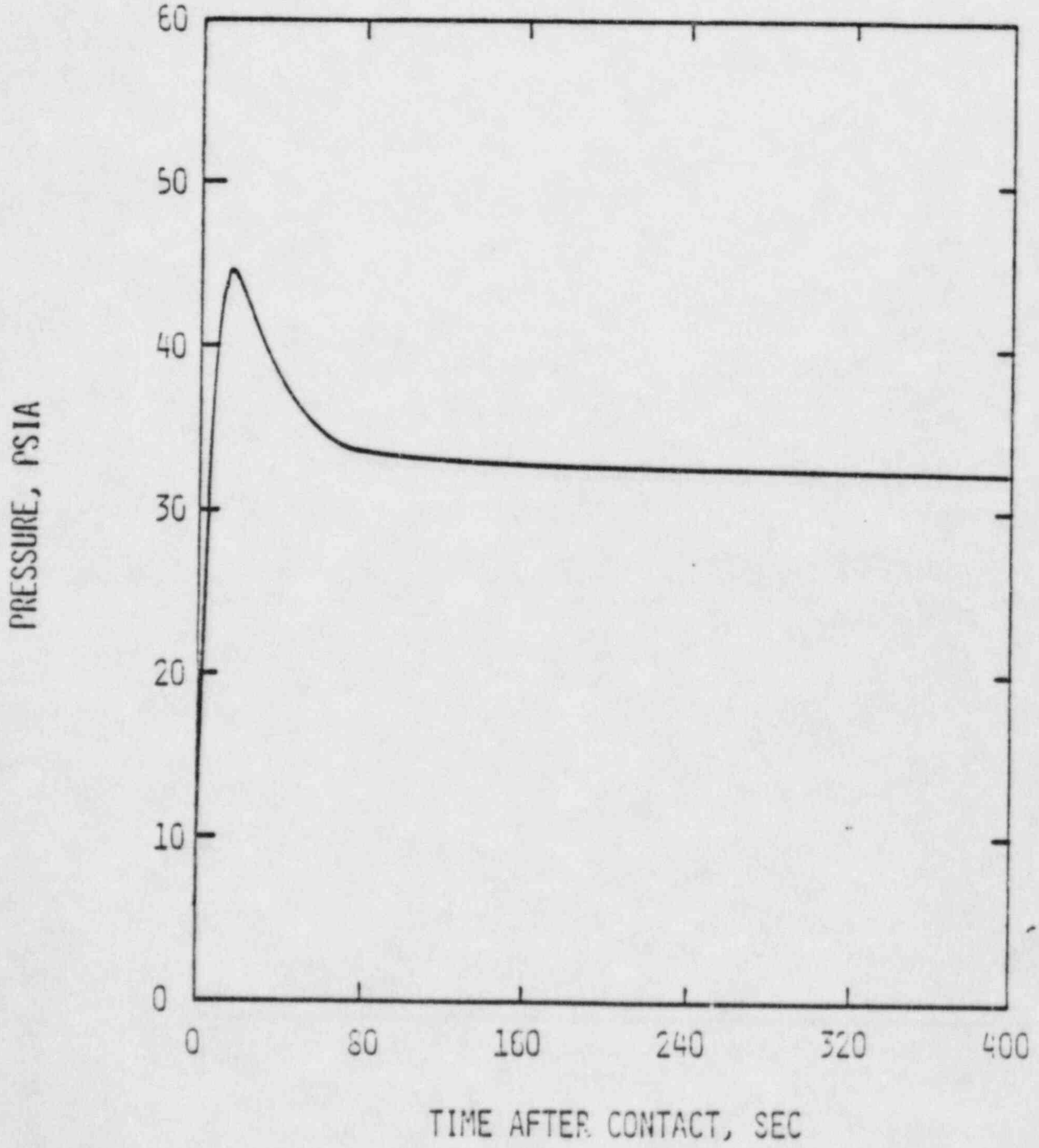


FIGURE 8.1-8

CALVERT CLIFFS I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

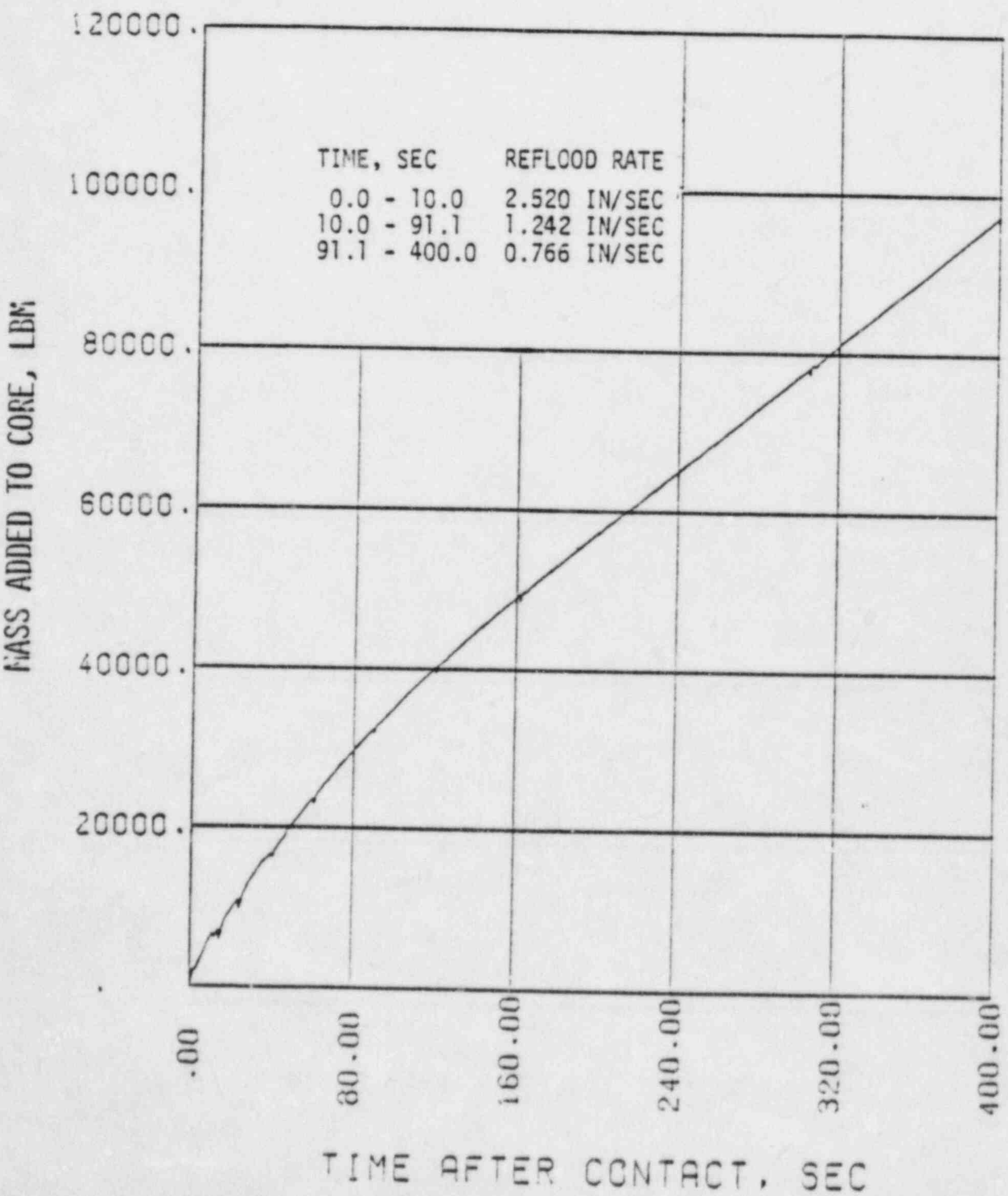
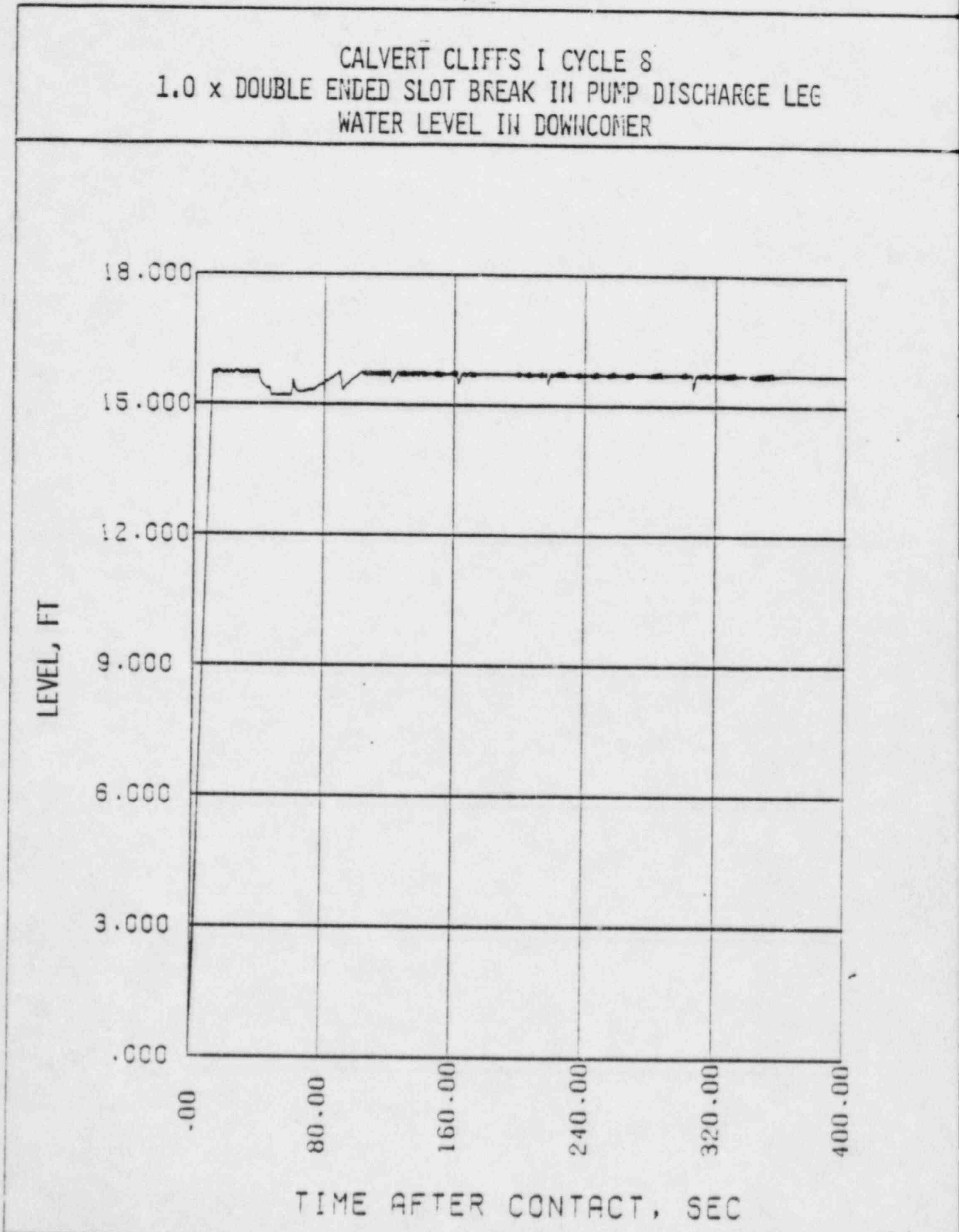


FIGURE 8.1-9


CALVERT CLIFFS I CYCLE 8
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
WATER LEVEL IN DOWNCOMER
18.000
15.000
12.000
9.000
6.000
3.000
.000
LEVEL, FT
.00
80.00
160.00
240.00
320.00
400.00
TIME AFTER CONTACT, SEC

8.2 Small Break Loss-of-Coolant Accident

8.2.1 Introduction and Summary

The ECCS performance evaluation for the small break loss-of-coolant accident (LOCA) for Calvert Cliffs Unit 1 Cycle 8, presented herein, demonstrates conformance with the acceptance criteria of 10CFR50.46[1]. The evaluation demonstrates acceptable small break LOCA ECCS performance at a peak linear heat generation rate (PLHGR) of 15.5 kw/ft and a reactor power level of 2754 Mwt (102% of 2700 Mwt) with an assumed small reduction in high pressure safety injection flow capacity. Injection flow from one charging pump was assumed in this analysis. A revised axial power distribution which is intended to envelope present and future cycles was used. It is expected that the analysis presented herein will apply directly to other cycles of both Calvert Cliffs units.

8.2.2 Method of Analysis

Reference 9 presented the Calvert Cliffs small break LOCA ECCS performance at a reactor power level of 2611 Mwt (102% of 2560 Mwt) and a PLHGR of 15.8 kw/ft. In reference 9, the 0.1 ft^2 break in the reactor coolant pump discharge leg was identified as the worst small break size. In Reference 10, the 0.1 ft^2 break was reanalyzed at a reactor power level of 2754 Mwt (102% of 2700 Mwt) and a PLGHR of 16.0 kw/ft, hereafter referred to as the reference analysis. This latter analysis resulted in a peak clad temperature and a peak local clad oxidation percentage of 1940°F and 7.96%, respectively. The evaluation presented herein is a re-analysis of the 0.1 ft^2 break at a reactor power level of 2754 Mwt with an assumed reduction in high pressure safety injection flow capacity.

The calculation was performed using Combustion Engineering's NRC approved Small Break Evaluation Model as described in References 9 and 11. Evaluation of small break transients involves the use of the following computer codes. Blowdown hydraulics are calculated using the CEFLASH-4AS[12] code. Fuel rod temperatures and clad oxidation percentages are calculated using the STRIKIN-II[5] and PARCH[8] codes. Details of the interfacing of these codes are discussed in Reference 11.

The significant core and system parameters which changed from the reference analysis are listed in Table 8.2-1. The peak linear heat rate assumed was 15.5 kw/ft which is the current maximum allowed by the Technical Specification. In addition to the assumed high pressure safety injection flow, the flow from one charging pump is credited. The charging pump delivers flow to two cold legs. For breaks in a reactor coolant pump discharge leg, it is assumed that all injection flow delivered to the broken leg spills out the break. After accounting for instrument error and applying a conservatively determined flow split, credit is taken for a minimum of 13 gpm delivered to the intact cold leg. To permit the assumed reduction in high pressure safety injection flow, the allowable axial shape index is being reduced in the Tech. Specs. from -.15 to -.10 ASIU (See Section 9.0). Increased low pressurizer pressure setpoints for reactor trip and safety injection actuation relative to the reference analysis were credited. This latter change is based on the present Technical Specification values which were increased after the reference analysis submittal. (The values shown in the table account for measurement uncertianty under accident conditions). The Main Steam Safety Valve Setpoint Technical Specification change discussed in Table 9-1 has been incorporated in this analysis.

The total flow from one high pressure safety injection pump assumed in this analysis is given in Table 8.2-2. The reduction in high pressure safety injection pump flow will allow greater flexibility in the surveillance testing of these pumps.

8.2.3 Results

The results of the ECCS performance analysis for the 0.1 ft^2 break are summarized in Table 8.2-3. The peak clad temperature is 1877°F with a peak local clad oxidation percentage of 4.91% and a core-wide oxidation percentage less than 0.632%. The results are comparable or slightly lower than those of the reference analysis.

The important transient parameters which have been plotted as a function of time (Figure 8.2-1 through 8.2-8) are listed in Table 8.2-4.

8.2.4 Conclusion

A reanalysis of the ECCS performance at an assumed reduction in high pressure safety injection pump flow capacity for the worst small break LOCA has been performed for Calvert Cliffs Unit 1 Cycle 8 at a reactor power level of 2754 Mwt and a PLHGR of 15.5 kw/ft. The analysis demonstrated a peak clad temperature of 1877°F and a peak local clad oxidation percentage of 4.91%, thereby demonstrating appreciable margin relative to the Acceptance Criteria(1) for the worst small break LOCA. Therefore, it can be concluded that operation of Calvert Cliffs Units 1 Cycle 8 at the reactor power level of 2754 Mwt is acceptable. It is expected that this conclusion will apply directly to other cycles of both Calvert Cliffs units.

Table 8.2-1

System Parameters and Initial Conditions

Calvert Cliffs

Parameters	Value: Present Analysis	Value: Reference Analysis
Peak Linear Heat Rate	15.5 kw/ft	16.0 kw/ft
Charging Pump Minimum Flow	13 gpm delivered to intact leg	none
Axial Shape Index (Most Negative Value Including Uncertainty)	-.16 ASIU	-.21 ASIU
Low Pressurizer Pressure Setpoints		
Reactor Trip	1741 psia	1728 psia
Safety Injection Actuation	1591 psia	1578 psia

Table 8.2-2

Calvert Cliffs Units 1 and 2
HPSI Pump Flow for Small Break Analysis

RCS Pressure, psig	Present Analysis Flow, GPM	Reference Analysis Flow, GPM
1276.0	0.0	0.0
1266.3	0.0	35.0
1250.0	69.50	104.5
1200.0	145.50	180.5
1150.0	188.25	223.25
1100.0	226.25	261.25
1050.0	254.75	289.75
1000.0	278.50	313.50
900.0	321.25	356.25
800.0	368.75	403.75
700.0	402.0	437.0
600.0	435.25	470.25
500.0	473.25	508.25
300.0	535.0	570.0
0.0	606.25	641.25

Table 8.2-3

Times of Interest and Fuel Rod
Performance Summary for 0.1 ft^2 Break

Time for HPSI Pump On	56 sec.
Time for LPSI Pump and SI Tanks On	a
Time Hot Spot Peak Clad Temperature Occurs	1650 sec.
Maximum Clad Surface Temperature	1877°F
Elevation of Hot Spot (from Bottom of Core)	10.8 ft
Peak Local Clad Zirconium Oxidation	4.92%
Core Wide Zirconium Oxidation	<0.632%

[a] Calculation terminated before LPSI pump or SI tank actuation.

Table 8.2-4

Variables Plotted as a Function of Time
Calvert Cliffs Units 1&2 0.1 ft^2/PD

Variable	Figure No.
Normalized Total Core Power	8.2-1
Inner Vessel Pressure	8.2-2
Break Flow Rate	8.2-3
Inner Vessel Inlet Flow Rate	8.2-4
Inner Vessel Two-Phase Mixture Volume	8.2-5
Heat Transfer Coefficient at Hot Spot	8.2-6
Coolant Temperature at Hot Spot	8.2-7
Clad Surface Temperature at Hot Spot	8.2-8

FIGURE 8.2-1

CALVERT CLIFFS UNITS I AND II

0.1 FT^2 COLD LEG BREAK AT PUMP DISCHARGE

NORMALIZED TOTAL CORE POWER

(SMALL BREAK ANALYSIS)

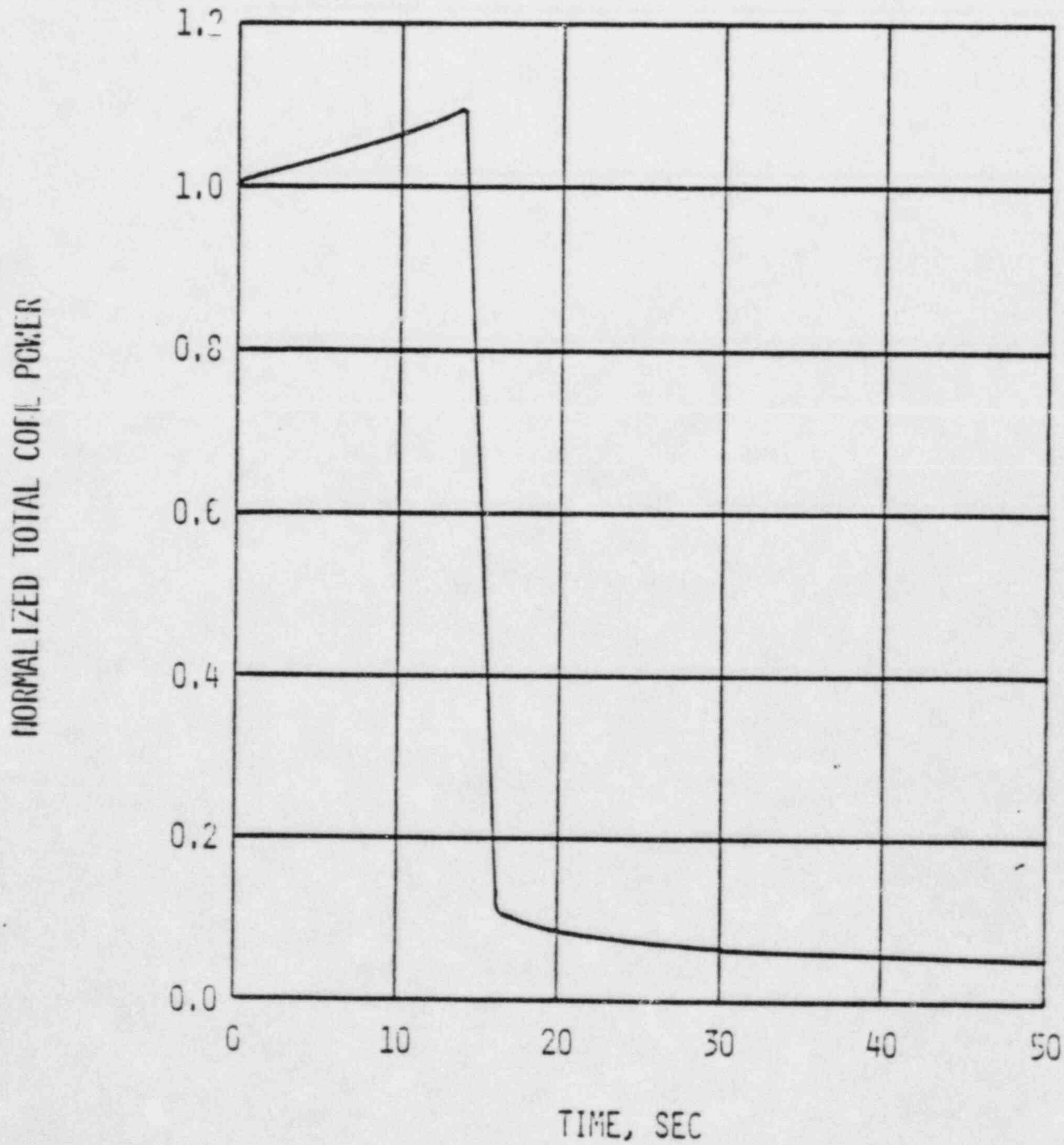


FIGURE 8.2-2

CALVERT CLIFFS UNITS I AND II
0.1 FT^2 COLD LEG BREAK AT PUMP DISCHARGE
INNER VESSEL PRESSURE
(SMALL BREAK ANALYSIS)

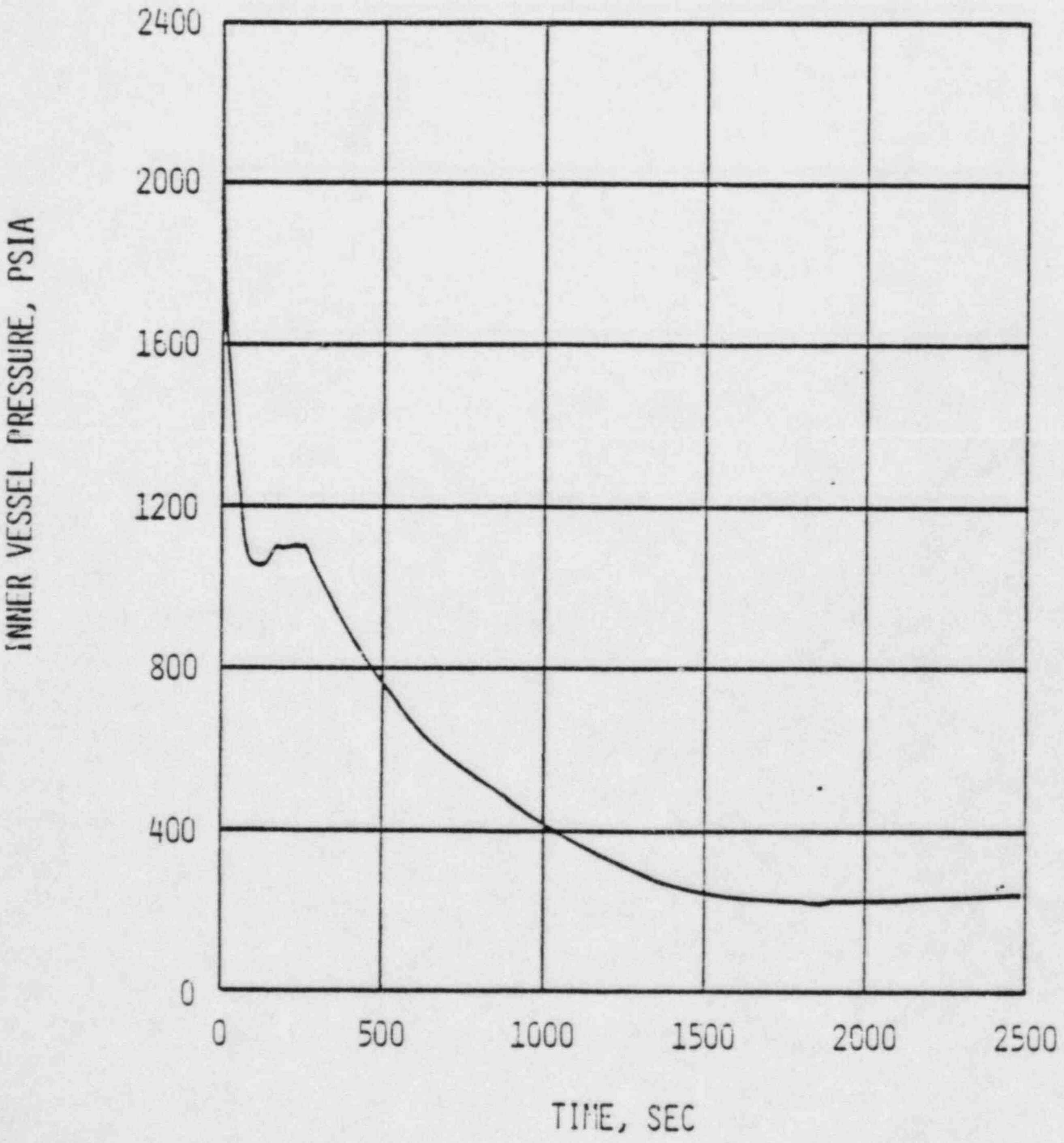


FIGURE 8.2-3

CALVERT CLIFFS UNITS I AND II
0.1 FT^2 COLD LEG BREAK AT PUMP DISCHARGE
BREAK FLOW RATE
(SMALL BREAK ANALYSIS)

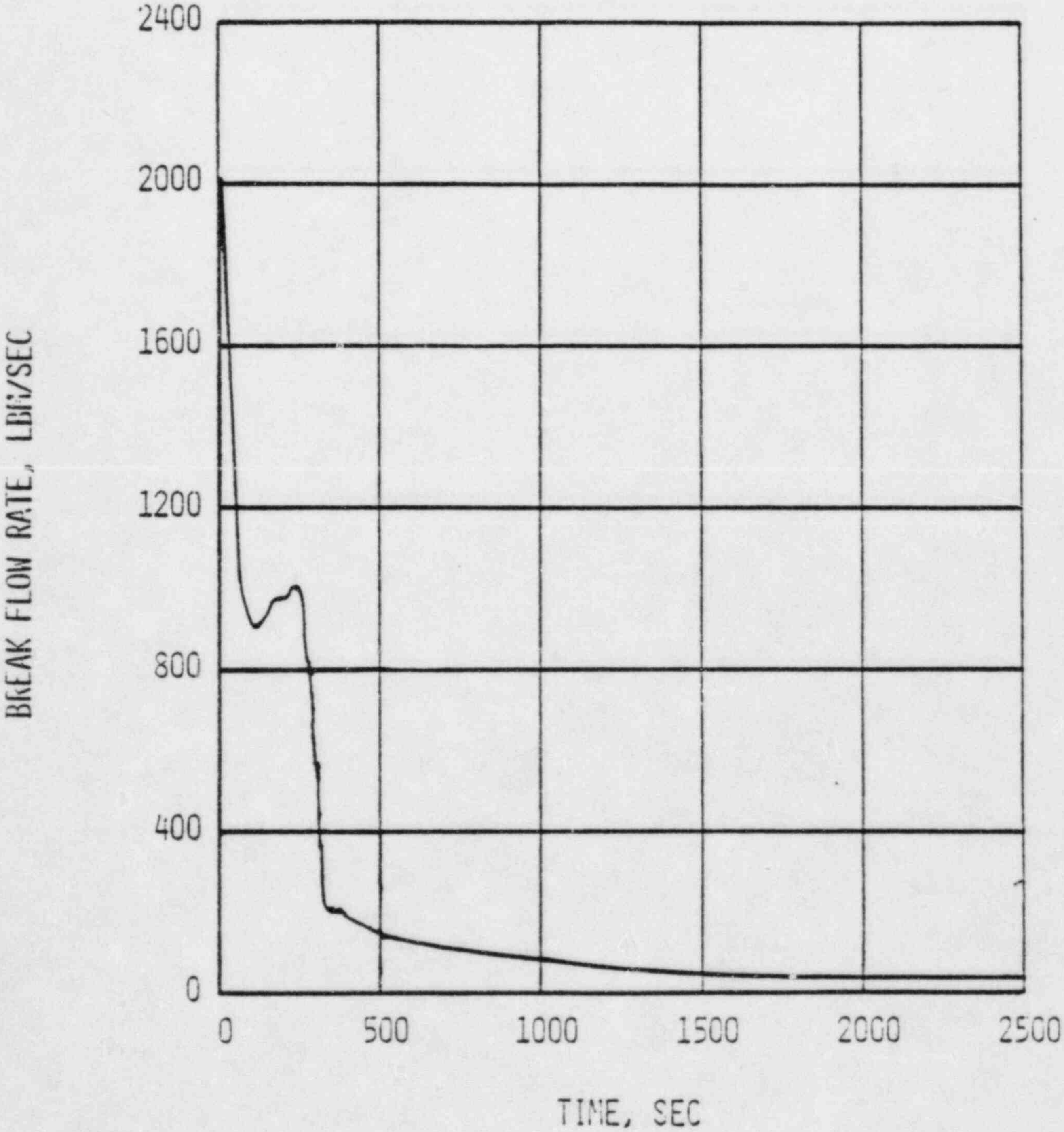


Figure 8.2-4

CALVERT CLIFFS UNITS I AND II
0.1 FT^2 COLD LEG BREAK AT PUMP DISCHARGE
INNER VESSEL INLET FLOW RATE
(SMALL BREAK ANALYSIS)

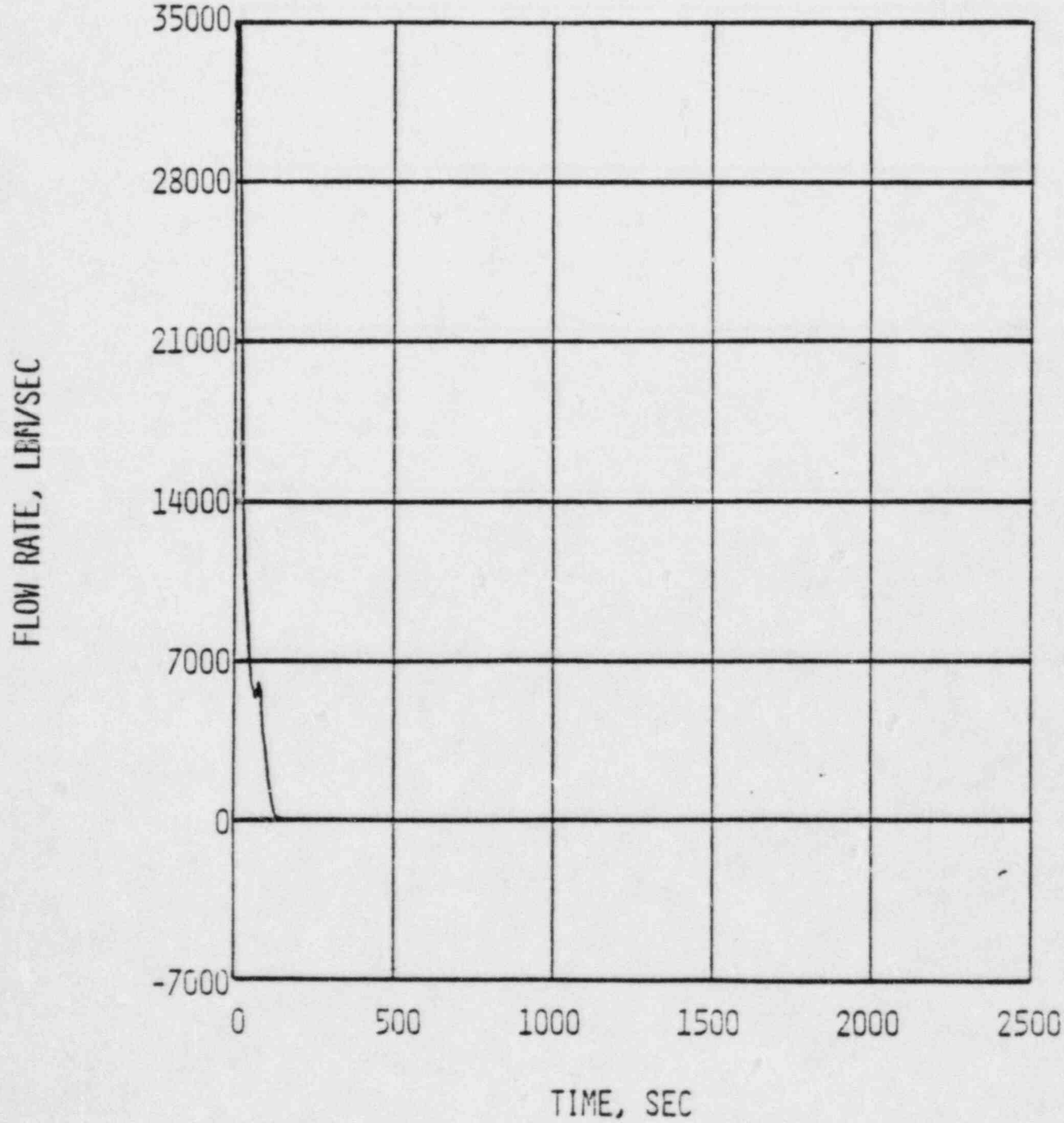


FIGURE 8.2-5

CALVERT CLIFFS UNITS I AND II
0.1 FT^2 COLD LEG BREAK AT PUMP DISCHARGE
INNER VESSEL TWO-PHASE MIXTURE VOLUME
(SMALL BREAK ANALYSIS)

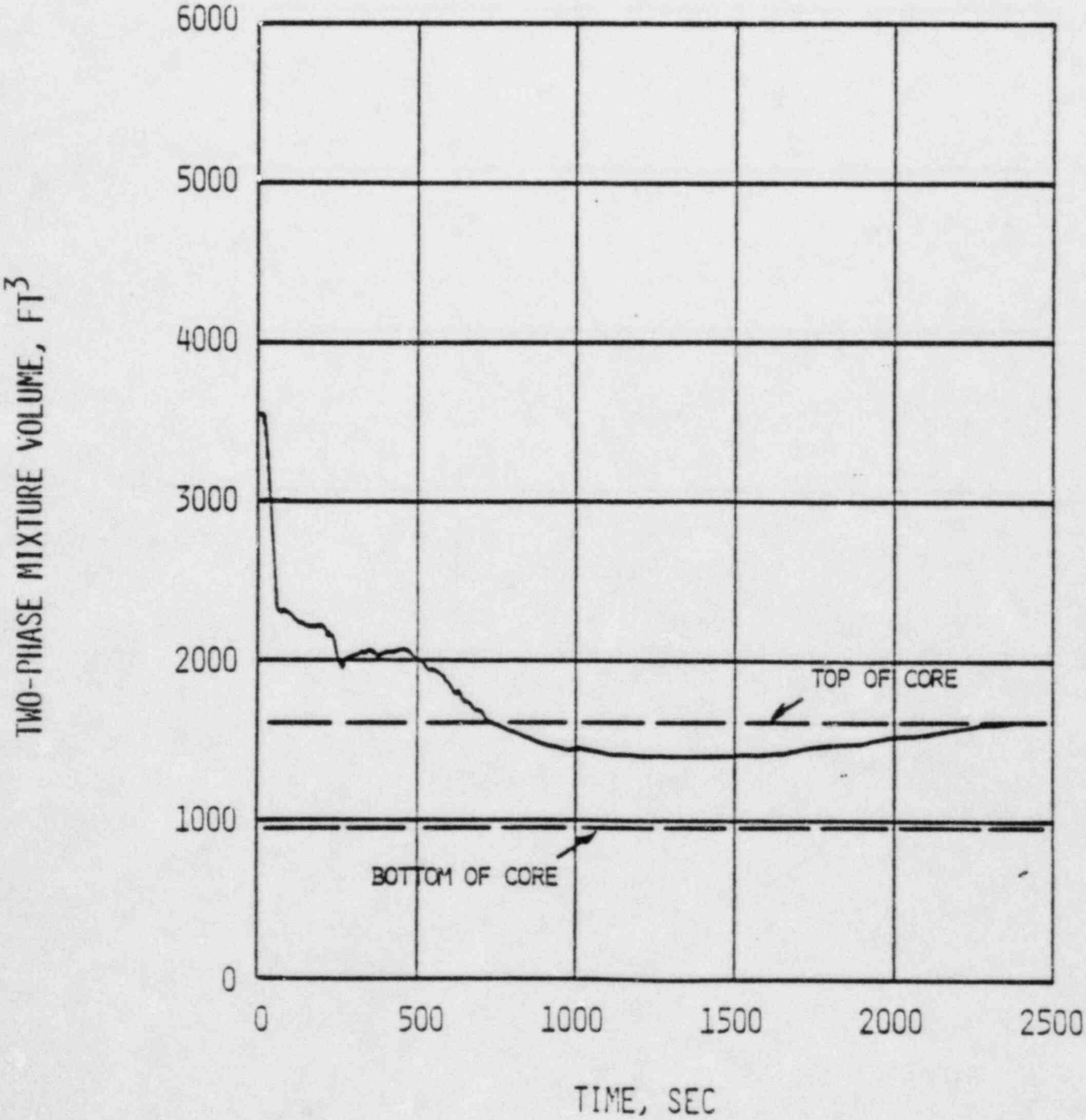


FIGURE 8.2-6

CALVERT CLIFFS UNITS I AND II
0.1 FT^2 COLD LEG BREAK AT PUMP DISCHARGE
HEAT TRANSFER COEFFICIENT AT HOT SPOT
(SMALL BREAK ANALYSIS)

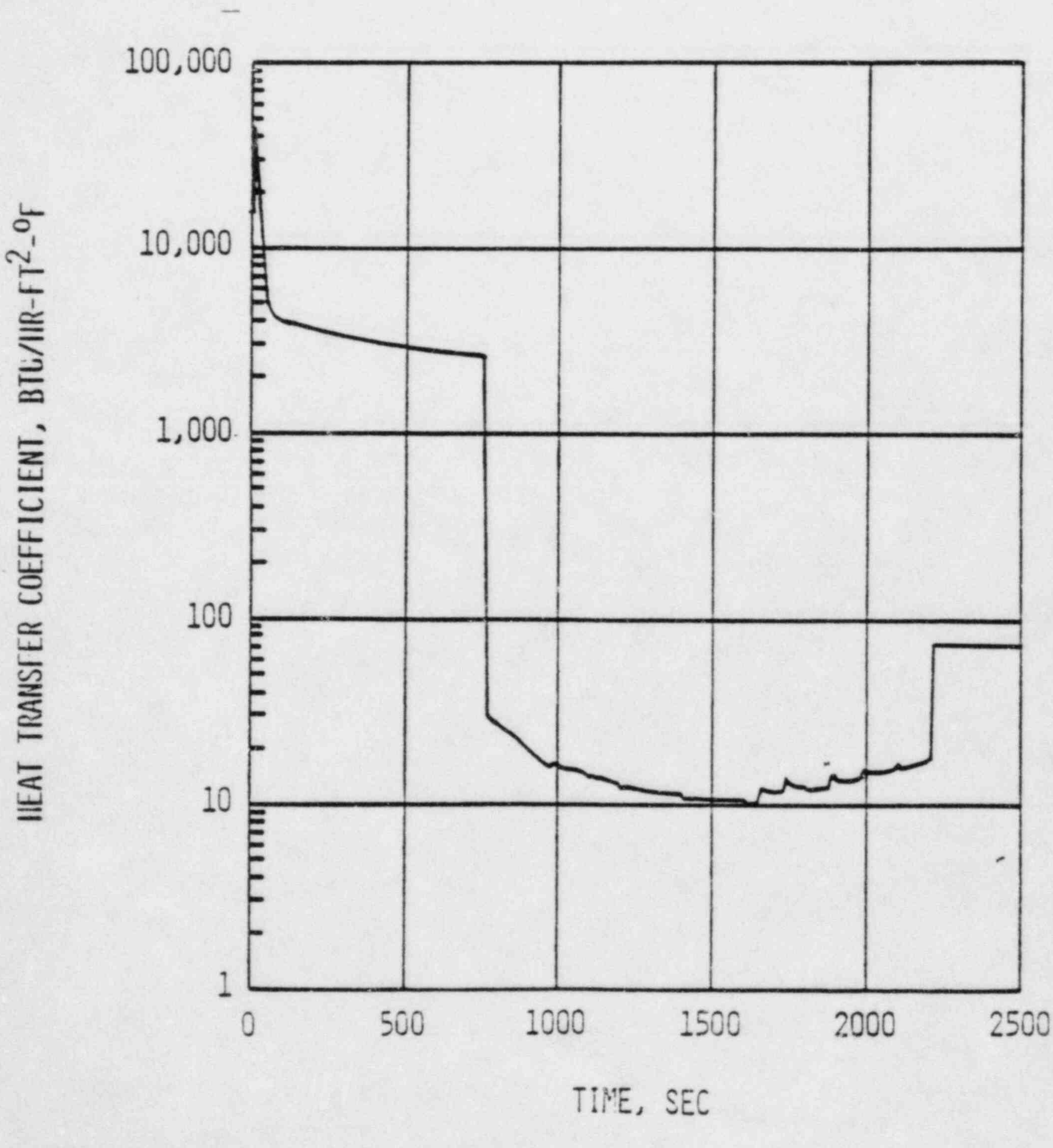


FIGURE 8.2-7

CALVERT CLIFFS UNITS I AND II
0.1 FT^2 COLD LEG BREAK AT PUMP DISCHARGE
COOLANT TEMPERATURE AT HOT SPOT
(SMALL BREAK ANALYSIS)

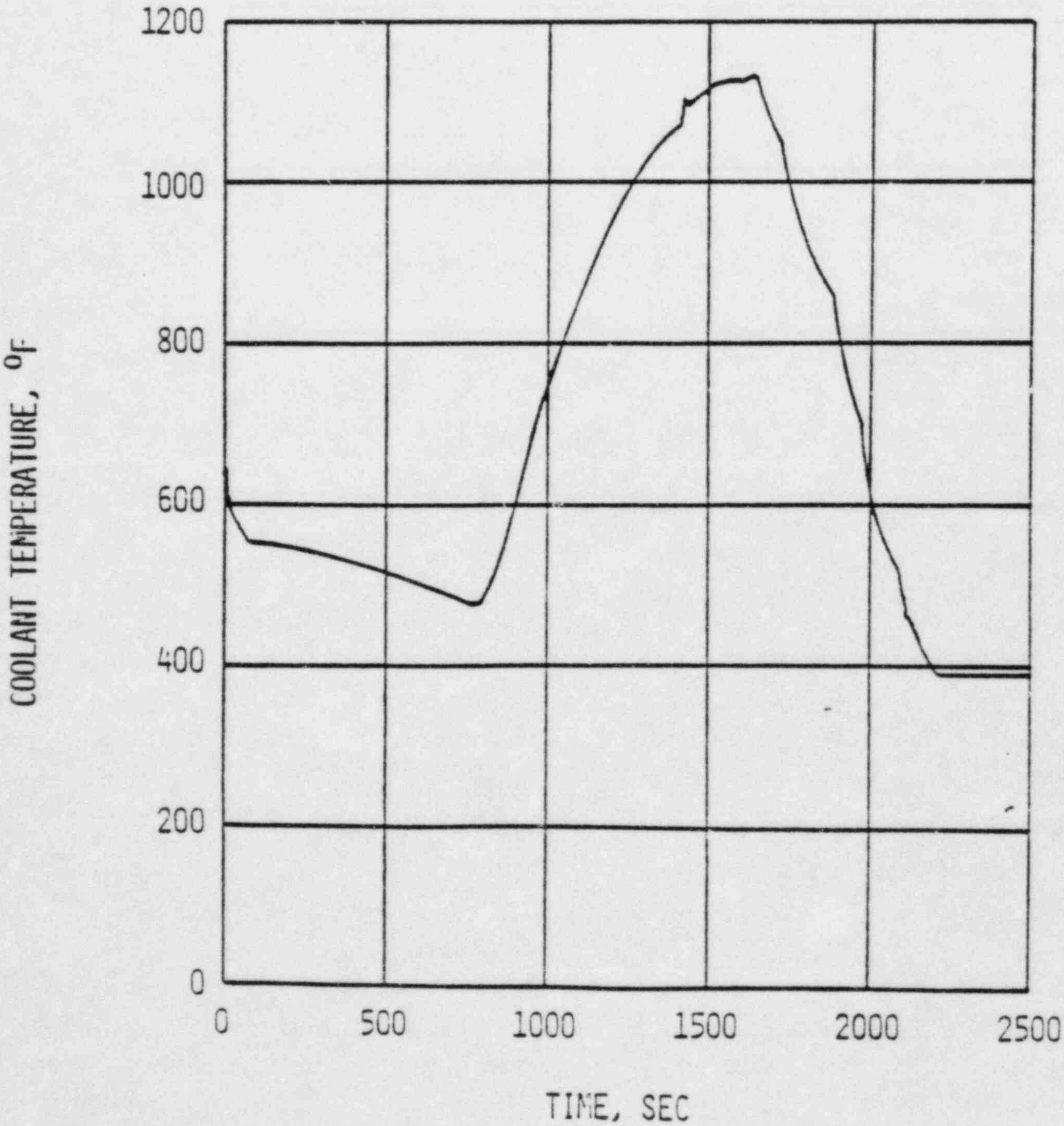


FIGURE 8.2-8

CALVERT CLIFFS UNITS I AND II

0.1 FT^2 COLD LEG BREAK AT PUMP DISCHARGE

CLAD SURFACE TEMPERATURE AT HOT SPOT

(SMALL BREAK ANALYSIS)