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ARTHUR E. LUNDVALL, JR.
VICE PRESIDENT
SUPPLY

September 1, 1983

Director of Nuclear Reactor Regulation
Attention: Mr. J. R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Calvert Cliffs Nuclear Power Plant Unit 1
Dockets Nos. 50-317
Amendment to Operating License DPR 53
Supplement 1 to Seventh Cycle License Application

Reference: (A) A. E. Lundvall, Jr., to R. A. Clark letter dated August 22, 1983, "Unit 1 Seventh Cycle License Application"

(B) A. E. Lundvall, Jr., to R. A. Clark letter dated June 30, 1983, "Unit 1 Planned CEA Inspection Program"

Gentlemen:

Baltimore Gas and Electric Company hereby supplements an earlier request (Reference (A)) for amendment to Operating License DPR 53 for Calvert Cliffs Unit 1. The enclosed Supplement 1 presents a detailed description of the required Technical Specification changes with the supporting safety analysis information to ensure conservative operation at a rated thermal power of 2700 MWTh for Unit 1 Cycle 7.

In Reference (B) we provided the proposed fuel assembly inspection plan to follow End of Cycle 6. Since then, the core loading has been revised. The new loading pattern now places a total of five modified guide tube assemblies under CEAs for a second cycle. The mechanical performance of the five assemblies is discussed in the supplement, Chapter 4.2.

Technical Specification Changes and Justification

The proposed changes to the Standard Technical Specifications (STS) required by this Amendment are described in Chapter 9 in the enclosure to this letter and justified in discussions in Section 1 through 8 of this enclosure. The majority of the Technical Specifications are duplicating the Technical Specifications as they currently exist for Unit 2. Unit 2 Cycle 5 is the reference cycle for Unit 1 Cycle 7.

The changes that are unique to this cycle are STS 3/4.1.1.1, a decrease of the required shutdown margin, and STS 3/4.3.1.1.4, an increase of the allowed moderator temperature coefficient. Both of these changes are justified in the report through the

A001
1/1

reanalysis of the Steam Line Break Design Bases Event. The remaining Technical Specification changes are a result of modifications to the Auxiliary Feedwater Actuation System. The Feed Line Break Analysis justifies these modifications and is provided in the supplement.

Determination of Significant Hazards Consideration

We have determined, based on the analytical information supplied in the supplement, that this amendment request does not involve a significant hazards consideration. This conclusion was derived by applying the Commission's guidance for implementation of 10CFR50.92. The Commission provided this guidance concerning the application of these standards through certain examples in the Federal Register, Volume 48, Number 87, Wednesday, 4/6/83, Rules and Regulations. Example iii of actions involving no significant hazards considerations, on page 14870 of the Federal Register, is quoted below.

> "For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable."

As described in Supplement 1 no fuel assemblies to be loaded into the Cycle 7 core will be of new or different design than those used previously and found to be acceptable to the NRC. No proposed changes to the Technical Specifications for Cycle 7 involve acceptance criteria that are significantly different from those previously found acceptable to the NRC. The analytical methods used to demonstrate conformance with the Technical Specifications and Regulations are consistent with previous NRC approvals and involve no significant changes.

We conclude that the proposed reload license amendment does not involve a significant hazards consideration in that:

1. The probability or consequences of an accident previously evaluated is not significantly increased. The larger part of the reload is enveloped by the previously approved Unit 2 Cycle 5 license amendment. The Steam Line Break Accident was the only Design Bases Event reanalyzed. Its consequences are not significantly increased.

2. This reload application does not create the possibility of a new or different kind of accident from any previously evaluated.

3. This license reload does not involve a significant reduction in the margins to safety. The small changes in the shutdown margin and increase in moderator temperature coefficients are not a significant reduction in margin of safety as the supplement analysis indicates.

The Plant Operation and Safety Review Comittee (POSRC) and the Offsite Safety Review Committee (OSSRC) have reviewed this proposed Amendment and these proposed changes to the Standard Technical Specifications and have concluded that implementation of this change will not result in an undue risk to the health and safety of the public.

Yours very truly,

BALTIMORE GAS AND ELECTRIC COMPANY

Arthur E. Lundvall Jr.

Enclosure (20 copies)

cc: J. A. Biddison, Jr., Esq.
G. F. Trowbridge, Esq.
Mr. D. H. Jaffe, NRC
Mr. R. E. Architzel, NRC
Mr. R. R. Mills, CE
Mr. R. E. Cochran, DHMH

Calvert Cliffs Unit 1 Cycle 7
Refueling License Amendment

Table of Contents

SUPPLEMENT 1

CALVERT CLIFFS

UNIT 1

CYCLE 7

LICENSE APPLICATION

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of design and performance for the operation of Calvert Cliffs Unit 1 during its seventh fuel cycle at full rated power of 2700 MWt. All planned operating conditions remain the same as those for Cycle 6. The core will consist of 113 presently operating F, G, and H assemblies, 64 fresh Batch J assemblies, and 40 B, D, E and F assemblies previously discharged from various cycles of both Calvert Cliffs units.

Plant operating requirements have created a need for flexibility in the Cycle 6 termination point, ranging from 12,700 MWD/T to 13,700 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 7 conditions would be enveloped, provided the Cycle 6 termination points fall within the above burnup range. The analysis presented herein will accommodate a Cycle 7 length of up to 14,000 MWD/T.

The evaluations of the reload core characteristics have been conducted with respect to the Calvert Cliffs Unit 2 Cycle 5 safety analysis described in Reference 1, as supplemented in Reference 2, hereafter referred to as the "reference cycle" in this report unless otherwise noted. This is an appropriate reference cycle because of the similarity in the basic system characteristics of the two reload cores. Reference 1 was the basic Unit 2 Cycle 5 license submittal. Reference 2 was a supplemental report which provided recommended Technical Specification changes to accommodate the installation of the Safety Grade Auxiliary Feedwater Actuation System (SAFAS) and a transient analysis report covering the three transients that were affected by the SAFAS.

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 previous cycle (Unit 1 Cycle 6, References 3 and 4), proposed modifications to the plant Technical Specifications are provided and are justified by the analyses reported herein. These proposed modifications are similar to those proposed for the reference cycle (References 1 and 2).

All Cycle 7 analyses address fuel exposure explicitly. Since the most limiting transient with respect to extended burnup is the steam line break (SLB) event, a generic analysis has been performed to envelope future cycles and thereby reduce the need for licensing review. This generic analysis which entails a more negative MTC and a reduced shutdown margin requirement is presented herein. The operation of the SAFAS was explicitly accounted for in this SLB analysis and in the analysis of feedline break which is presented herein due to the implementation of the SAFAS for Unit 1 Cycle 7.

The performance of Combustion Engineering 14x14 fuel at extended burnup is discussed in Reference 5. Fuel performance for Cycle 7 has been evaluated with the FATES3 computer code (References 21 and 22) as approved by the NRC in Reference 23.

2.0 OPERATING HISTORY OF THE PREVIOUS CYCLE

Calvert Cliffs Unit 1 is presently operating in its sixth fuel cycle utilizing Batch H, G, F, and D fuel assemblies (including eight D assemblies from Unit 2). Calvert Cliffs Unit 1 Cycle 6 began operation on July 5, 1982 and reached full power on July 17, 1982. The Cycle 6 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 having followed the calculated predictions closely.

It is presently estimated that Cycle 6 will terminate on or about September 30, 1983 . The Cycle 6 termination point can vary between 12,700 MWD/T and 13,700 MWD/T to accommodate the plant schedule and still be within the assumptions of the Cycle 7 analyses. As of August 25, 1983, the Cycle 6 burnup had reached 12,245 MWD/T.

3.0 GENERAL DESCRIPTION

The Cycle 7 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 7 is the removal of 52 Batch G/ assemblies, 43 Batch F assemblies, 1 Batch D assembly and 8 Unit 2 Batch D assemblies. These assemblies will be replaced by 48 fresh unshimmed Batch J assemblies (4.05 wt% U-235 enrichment), 16 fresh unshimmed Batch J* assemblies (3.40 wt% U-235 enrichment), 4 Batch F assemblies (3.03 wt% U-235 enrichment) discharged from Unit 1 Cycle 5, 12 Batch E/ assemblies (2.73 wt% U-235 enrichment) discharged from Unit 1 Cycle 4, 12 Batch D/ assemblies (2.73 wt% U-235) discharged from Unit 2 Cycle 3 and 12 Batch B assemblies (2.45 wt% U-235) discharged from Unit 2 Cycle 1.

Figure 3-1 shows the fuel management pattern to be employed in Cycle 7. Figure 3-2 shows the locations of the poison pins within the lattice of once-burned Batch B assemblies and the fuel rod locations in unshimmed assemblies; Figure 3-3 shows the poison pin locations within the lattice of the once-burned Batch H/ fuel. This fuel management pattern will accommodate Cycle 6 termination burnups from 12,700 MWD/T to 13,700 MWD/T.

The Cycle 7 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 3-4 shows the beginning of Cycle 7 assembly burnup distribution for a Cycle 6 termination burnup of 13,700 MWD/T. The initial enrichment of the fuel assemblies is also shown in Figure 3-4. Figure 3-5 shows the end of Cycle 7 assembly burnup distribution. The end of Cycle 7 core average exposure is approximately 27,400 MWD/T and the average discharge exposure is approximately 36,600 MWD/T.

Inconel Irradiation Experiment

The inconel irradiation experiment, described in Reference 7, began in Cycle 5 of Calvert Cliffs Unit 1. At that time three irradiation specimens consisting of inconel tubes were inserted. They did not contain fuel or poison. Before Cycle 6 operation, one of these specimens was discharged and the other two were reinserted for Cycle 6 exposure. These two elements will be reinserted in Cycle 7.

Scout Demonstration Assembly

The Scout demonstration assembly was described in Reference 8. It is a Batch F test assembly which was originally inserted in Cycle 4. Changes, similar to those described in Reference 3, were made to this assembly prior to its third cycle of irradiation in Cycle 6. Further modifications are planned before returning the assembly to the core for its fourth cycle of irradiation in Cycle 7. The present plans are to remove 2 segmented test rods from the assembly and replace them with 2 stainless steel rods.

Prototype Demonstration Assemblies

The Prototype demonstration assemblies were described in Reference 7. These are Batch G assemblies which were originally inserted in Cycle 5. These assemblies will be placed in symmetric locations in the core in Cycle 7 for a third cycle of irradiation. Prior to returning to the core, 2 segmented test rods will be removed from one of these assemblies and replaced with 2 stainless steel rods.

TABLE 3-1

CALVERT CLIFFS UNIT 1 CYCLE 7
CORE LOADING

Assembly Designation	Number of Assemblies	Initial Enrichment wt% U-235	Batch Average Burnup (MWD/T) [EOC6= 13,700]	Poison Rods per Assembly	Initial Poison Loading wt% B_4C	Total Number of Poison and Non-Fuel Rods	Total Number of Fuel Rods
J	48	4.05	0	0	0	0	8448
J*	16	3.40	0	0	0	0	2816
H(1)	40	4.00	11,800	0	0	0	7040
H/(1)	32	3.55	16,400	8	3.03	256	5376
G(1)	40	3.65	25,900	0	0	2	7038
F(2)	1	3.03	33,800	0	0	3	173
F(3)	4	3.03	25,600	0	0	0	704
E/(4)	12	2.73	23,700	0	0	0	2112
D/(5)	12	2.73	23,100	0	0	0	2112
B(6)	12	2.45	17,700	12	3.03	144	1968
TOTAL	217					405	37,787

(1) Carried over from Unit 1 Cycle 6.

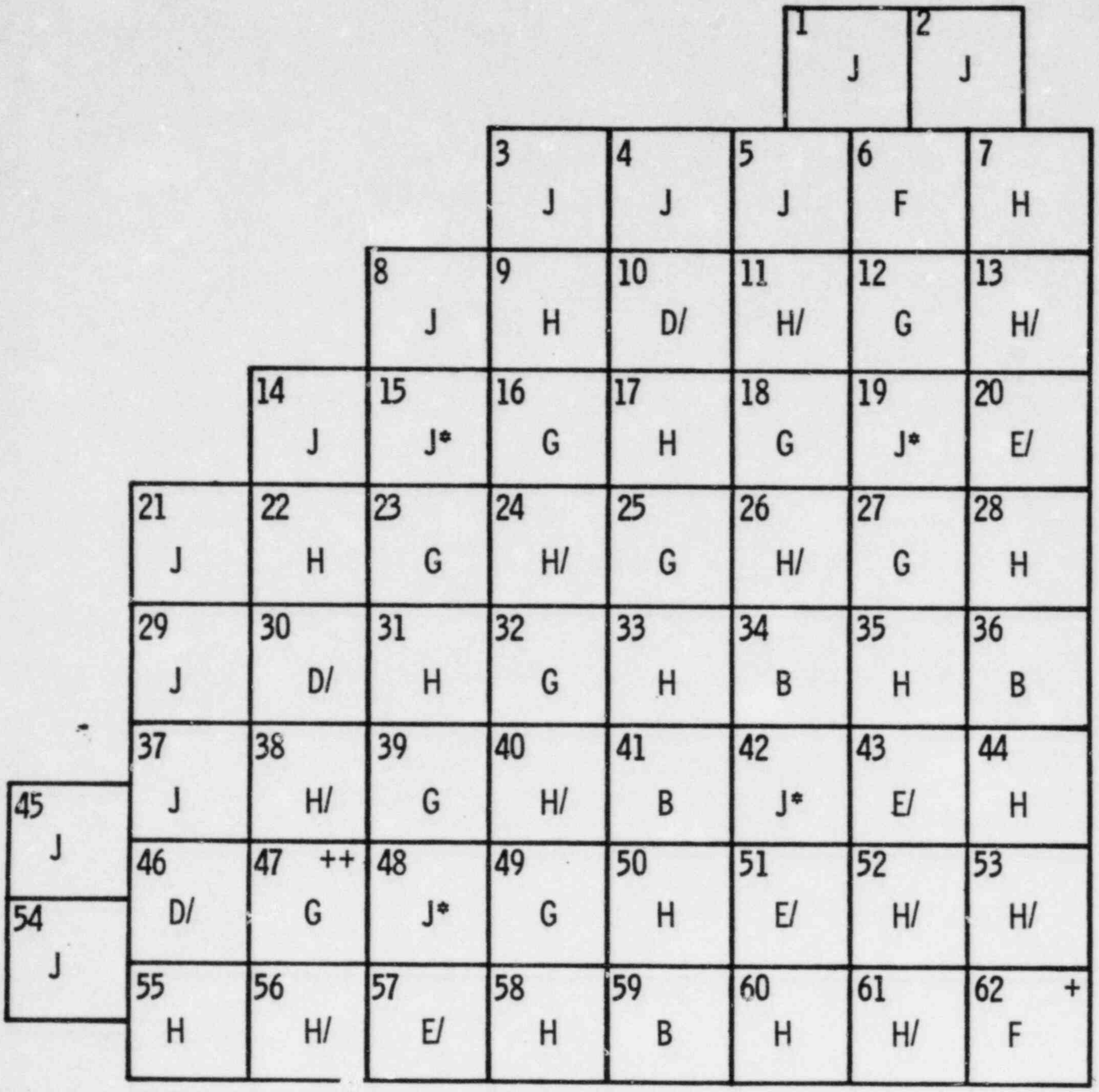
(2) SCOUT assembly carried over from Unit 1 Cycle 6.

(3) Twice burned Batch F fuel discharged from Unit 1 Cycle 5.

(4) Twice burned Batch E/ fuel discharged from Unit 1 Cycle 4.

(5) Twice burned Batch D/ fuel discharged from Unit 2 Cycle 3.

(6) Once burned Batch B fuel discharged from Unit 2 Cycle 1.

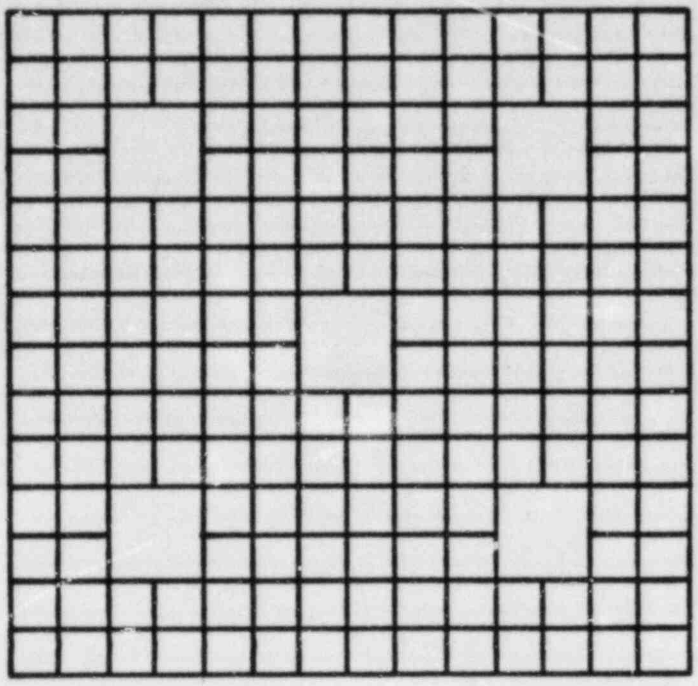


+LOCATION OF DEMONSTRATION ASSEMBLY (SCOUT)

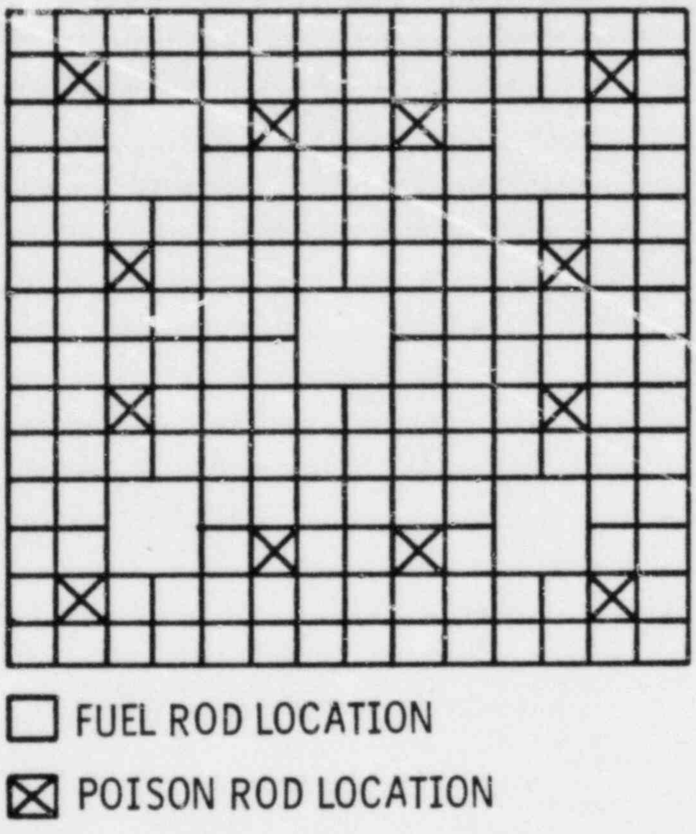
++LOCATION OF PROTOTYPE ASSEMBLIES

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 7 CORE MAP	Figure 3-1

UNSHIMMED ASSEMBLY

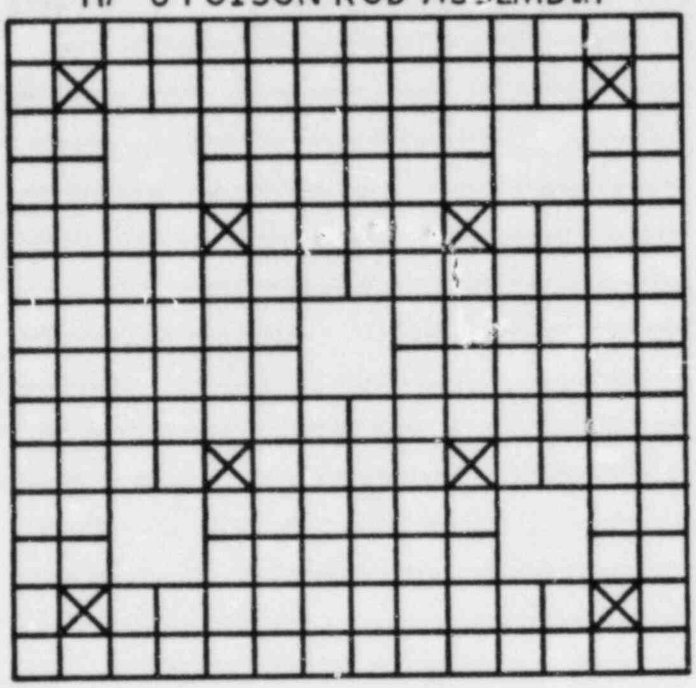


B-12 POISON ROD ASSEMBLY



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 7 ASSEMBLY FUEL AND OTHER ROD LOCATIONS	Figure 3-2

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 7 ASSEMBLY FUEL AND OTHER ROD LOCATIONS	Figure 3-3

INITIAL ENRICHMENT W/O U-235
BOC 7 BURNUP (MWD/T), EOC 6 = 13,700 MWD/T

						1 J 4.05 0	2 J 4.05 0	
				3 J 4.05 0	4 J 4.05 0	5 J 4.05 0	6 F 3.03 25,600	7 H 4.00 13,000
			8 J 4.05 0	9 H 4.00 13,900	10 D/ 2.73 23,400	11 H/ 3.55 17,500	12 G 3.65 27,400	13 H/ 3.55 17,700
		14 J 4.05 0	15 J* 3.40 0	16 G 3.65 26,200	17 H 4.00 10,000	18 G 3.65 25,300	19 J* 3.40 0	20 E/ 2.73 23,800
	21 J 4.05 0	22 H 4.00 13,900	23 G 3.65 26,100	24 H/ 3.55 15,400	25 G 3.65 23,300	26 H/ 3.55 17,000	27 G 3.65 28,000	28 H 4.00 13,000
	29 J 4.05 0	30 D/ 2.73 23,400	31 H 4.00 10,000	32 G 3.65 23,300	33 H 4.00 10,700	34 B 2.45 17,900	35 H 4.00 11,500	36 B 2.45 16,700
	37 J 4.05 0	38 H/ 3.55 17,400	39 G 3.65 25,300	40 H/ 3.55 16,900	41 B 2.45 18,400	42 J* 3.40 0	43 E/ 2.73 23,600	44 H 4.00 10,700
45 J 4.05 0	46 D/ 2.73 22,600	47 G 3.65 26,400	48 J* 3.40 0	49 G 3.65 27,700	50 H 4.00 11,500	51 E/ 2.73 23,700	52 H/ 3.55 14,700	53 H/ 3.55 14,800
54 J 4.05 0	55 H 4.00 13,000	56 H/ 3.55 17,700	57 E/ 2.73 23,800	58 H 4.00 13,000	59 B 2.45 16,700	60 H 4.00 10,700	61 H/ 3.55 14,800	62 F 3.03 33,800

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 7 ASSEMBLY AVERAGE BURNUP AT BOC AND INITIAL ENRICHMENT DISTRIBUTION	Figure 3-4

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

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 7 ASSEMBLY AVERAGE BURNUP AT EOC (MWD/T)	Figure 3-5

4.0 FUEL SYSTEM DESIGN

4.1 Mechanical Design

The mechanical design for the standard Batch J reload fuel is identical to that of the standard Batch G fuel described in the reference cycle submittal (Calvert Cliffs Unit 2 Cycle 5, Reference 1). It is also identical to that of the standard Batch H fuel used in Calvert Cliffs Unit 1 Cycle 6 (Reference 3).

The mechanical designs of the Calvert Cliffs Unit 1 Batch E, F, G, and H fuel assemblies were described in References 9, 10, 7, and 3, respectively. Details of the Calvert Cliffs Unit 2 Batch B and D fuel assemblies that will be used in Cycle 7 can be found in References 11 and 12, respectively.

C-E has performed analytical predictions of cladding creep collapse time for all Calvert Cliffs Units 1 and 2 fuel batches that will be irradiated in Cycle 7 and has concluded that the collapse resistance of all standard fuel rods is sufficient to preclude collapse during their design lifetime. This lifetime will not be exceeded by the Cycle 7 duration (Table 4-1). These analyses utilized the CEPAN computer code (Reference 13) and the analysis procedures described in Reference 14. The analysis procedures described in Reference 14 were approved in Reference 15.

TABLE 4-1

Batch	Minimum Collapse Time	EOC 7 Exposure
B (Calvert Cliffs 2)	30,400 EFPH	22,960 EFPH
D (Calvert Cliffs 2)	>27,000	26,000
E	>27,500	26,320
F	35,100; >39,200*	28,860; 39,200*
G	36,000	30,340
H	>27,500	20,690
J	>27,500	10,340

*SCOUT Assembly

All batches of fuel (standard rods) were also reviewed for dimensional changes using the SIGREEP model described in Reference 16 (approved in Reference 15). Since the licensing of the reference cycle, some of the input correlations to SIGREEP have been refined to include additional extended burnup data. All clearances were found to be adequate during Cycle 7.

The cladding collapse information in Table 4-1 is applicable to the test rods in the SCOUT and PROTOTYPE assemblies. The clearance for fuel rod growth for the test rods will be evaluated during the Cycle 6 outage.

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for the Batch J fuel are identical to those of the other fuel batches to be included in Cycle 7. Thus, the chemical or metallurgical performance of the Batch J fuel will remain unchanged from the performance of the Cycle 6 fuel.

4.2 Hardware Modifications to Mitigate Guide Tube Wear

All standard fuel assemblies which will be placed in CEA locations in Cycle 7 will have stainless steel sleeves installed in the guide tubes to prevent guide tube wear. A detailed discussion of the design of the sleeves and their effect on reactor operation is contained in Reference 17.

Cycle 7 will also utilize five assemblies in CEA locations that were fabricated with modified guide tubes (see References 10 and 18) instead of sleeves to mitigate guide tube wear. Each of these modified assemblies has previously resided in a CEA position for one cycle and has been examined for guide tube wear after that cycle. The test results presented in References 19 and 20 showed no detectable wear.

4.3 Thermal Design

The thermal performance of composite fuel pins that envelope the various fuel assemblies present in Cycle 7 (fuel Batches E, F, G, H and J and Batches B and D from Unit 2) have been evaluated using the FATES3 version of the fuel evaluation model (References 21 and 22), as approved by the NRC (Reference 23). The analyses were performed with histories that modeled the power and burnup levels representative of the peak pins 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 7. In addition, the SCOUT and PROTOTYPE test pins were analyzed and found not limiting with respect to thermal performance over their respective burnup ranges.

The FATES3 power-to-centerline melt limit was determined for Cycle 7 by taking some credit for the decrease in power peaking which is characteristic of highly burned fuel. Since a gradual decrease in the calculated power-to-melt (due to a decrease in the fuel melt temperature) also accompanies burnup, the most limiting power-to-centerline melt has been found to occur within an intermediate burnup range. Using conservative estimates of the burnup point at which the power peaking begins to decrease and the rate at which it decreases for Cycle 7, the most limiting power-to-centerline melt has been determined to be in excess of 22 kw/ft and to occur at a rod average burnup of approximately 33,000 MWD/MTU.

5.0 NUCLEAR DESIGN

5.1 Physics Characteristics

5.1.1 Fuel Management

The Cycle 7 fuel management employs a mixed central region as described in Section 3, Figure 3-1. The fresh Batch J 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 16 assemblies with an enrichment of 3.40 wt% U-235. With this loading, the Cycle 7 burnup capacity for full power operation is expected to be between 12,800 MWD/T and 13,400 MWD/T, depending on the final Cycle 6 termination point. The Cycle 7 core characteristics have been examined for Cycle 6 terminations between 12,700 and 13,700 MWD/T and limiting values established for the safety analyses. The loading pattern (see Section 3) is applicable to any Cycle 6 termination point between the stated extremes.

Physics characteristics including reactivity coefficients for Cycle 7 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 7 zero power steam line break accident and a comparison to reference cycle data. The EOC zero power steam line break was selected since it is the most limiting zero power steam line break accident and, thus, provides the basis for establishing the Technical Specification required shutdown margin. The required shutdown margin for Cycle 7 operation is substantially reduced through the generic steam line break methodology employed (negative reactivity credit due to the local heatup of fluid under the stuck CEA in the hot channel is accounted for).

Table 5-3 shows the reactivity worths of various CEA groups calculated at full power conditions for Cycle 7 and the reference cycle. The CEA group identifications remain 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 Distribution

Figures 5-1 through 5-3 illustrate the all rods out (ARO) planar radial power distributions at BOC7, MOC7 and EOC7, respectively, that are characteristic of the high burnup end of the Cycle 6 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 6 shutdown window tends to increase the power peaking in this axial central region of the core for Cycle 7. The planar radial power distributions for the above region with CEA Group 5 fully inserted at beginning and end of Cycle 7 are shown in Figures 5-4 and 5-5, respectively, for the high burnup end of the Cycle 6 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 7. These conservative values, which are used in Section 7 of this document, establish the allowable limits for power peaking to be observed during operation.

The range of allowable axial peaking is defined by the limiting conditions for operation covering axial shape index (ASI). Within these ASI limits, the necessary DNBR and kw/ft margins are maintained for a wide range of possible axial shapes. The maximum three-dimensional or total peaking factor anticipated in Cycle 7 during normal base load, all rods out operation at full power is 1.92, not including uncertainty allowances and augmentation factors.

5.1.3 Safety Related Data

5.1.3.1 Ejected CEA Data

The maximum reactivity worth and planar power peaks associated with an ejected CEA event are shown in Table 5-4 for Cycle 7 and the reference cycle. These values encompass the worst conditions anticipated during Cycle 7 for any expected Cycle 6 termination point. The values shown for Cycle 7 are the safety analyses values which are conservative with respect to the actual calculated values. This section is included herein only to correct a textual error in Reference 1 (see note in Table 5-4).

5.1.3.2/5.1.3.3 Dropped CEA/Augmentation Factors

The Cycle 7 safety related data for these sections are identical to the safety related data used in the reference cycle.

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 24, 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 with the following exceptions:

a. The HERMITE/TORC methodology discussed in Reference 25 has been applied in the analysis of the steam line break accident, as concurred in by Reference 26.

b. All cross sections used in the Cycle 7 design were generated in accordance with the methods described in Reference 27 and approved for nuclear core design and safety-related neutronics calculations in Reference 28.

5.4 Uncertainties in Measured Power Distributions

The power distribution measurement uncertainties which are applied to Cycle 7 are the same as those applied to the reference cycle, as presented in Section 5.4 of Reference 1.

TABLE 5-1

CALVERT CLIFFS UNIT 1 CYCLE 7
NOMINAL PHYSICS CHARACTERISTICS

	Units	Reference** Cycle	Cycle 7
Dissolved Boron			
Dissolved Boron content for Criticality, CEAs Withdrawn			
Hot Full Power, Equilibrium Xenon, BOC	PPM	1032	1070
Boron Worth			
Hot Full Power BOC	PPM/%Δρ	105	108
Hot Full Power EOC	PPM/%Δρ	85	84
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients, Hot Full power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta\rho/^{o}F$	-0.1	-0.2
End of Cycle	$10^{-4}\Delta\rho/^{o}F$	-2.1	-2.2
Doppler Coefficient			
Hot Zero Power BOC	$10^{-5}\Delta\rho/^{o}F$	-1.48	-1.56
Hot Full Power BOC	$10^{-5}\Delta\rho/^{o}F$	-1.27	-1.28
Hot Full Power EOC	$10^{-5}\Delta\rho/^{o}F$	-1.47	-1.45
Total Delayed Neutron Fraction, βeff			
BOC		0.00609	0.00604
EOC		0.00521	0.00522
Neutron Generation Time, ℓ*			
BOC	10^{-6} sec	24.0	23.4
EOC	10^{-6} sec	30.5	29.8

**Unit 2 Cycle 5.

TABLE 5-2

CALVERT CLIFFS UNIT 1 CYCLE 7 LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCES FOR HOT ZERO POWER STEAM LINE BREAK, $\%\Delta\rho$ END-OF-CYCLE (EOC)

	Reference Cycle*	Cycle 7
1. Worth of all CEA's Inserted	9.4	9.1
2. Stuck CEA Allowance	1.8	2.6
3. Worth of all CEA's Less Worth of CEA Stuck Out**	7.6	6.5
4. Zero Power Dependent Insertion Limit CEA Bite	1.8	1.6
5. Calculated Scram Worth	5.8	4.9
6. Physics Uncertainty plus Bias	0.6	0.6
7. Net Available Scram Worth	5.2	4.3
8. Technical Specification Shutdown Margin	5.2	4.3***
9. Margin in Excess of Technical Specification Shutdown Margin	0.0	0.0

*Unit 2 Cycle 5.

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

***The generic SLB analysis presented herein supports a shutdown margin of $3.7\%\Delta\rho$ (see Table 7-2). Presently, the shutdown margin is only being reduced to $4.3\%\Delta\rho$ to be consistent with other existing safety analyses.

TABLE 5-3

CALVERT CLIFFS UNIT 1 CYCLE 7
REACTIVITY WORTH OF CEA REGULATING
GROUPS AT HOT FULL POWER, %Δρ

	Beginning of Cycle		End of Cycle	
Regulating CEA's	Reference* Cycle	Cycle 7	Reference* Cycle	Cycle 7
Group 5	0.49	0.53	0.63	0.64
Group 4	0.27	0.34	0.36	0.44
Group 3	0.91	0.99	1.16	1.07

Note

Values shown assume sequential group insertion.

*Unit 2 Cycle 5.

TABLE 5-4

CALVERT CLIFFS UNIT 1 CYCLE 7
CEA EJECTION DATA

	Limiting Values	
	Reference Cycle Safety Analysis Value	Unit 1 Cycle 7 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	9.4
Maximum Ejected CEA Worth (%$\Delta\rho$)		
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	0.63

Notes

1. Uncertainties and allowances are included in the above data.

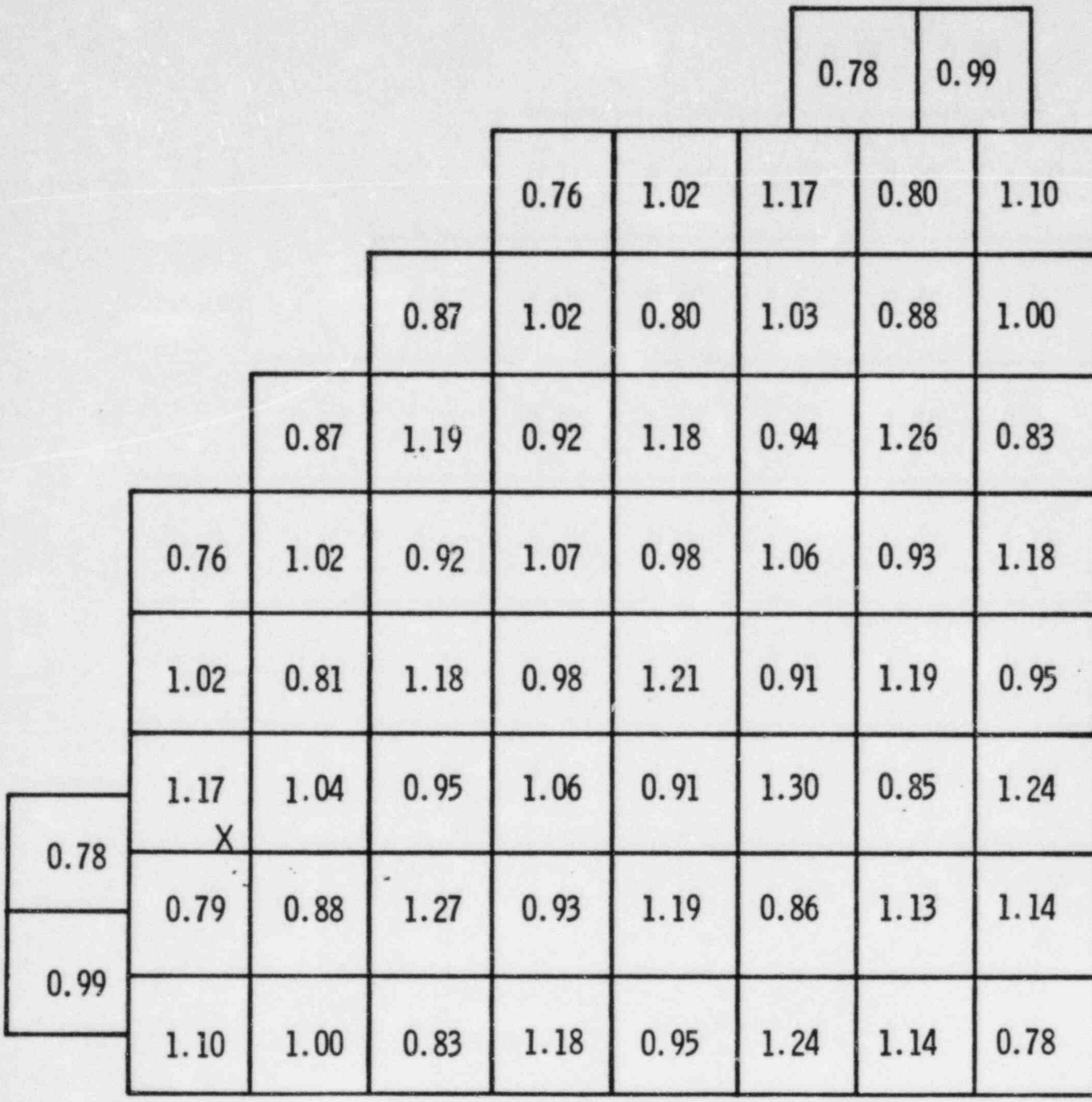
2. The Cycle 7 safety analysis values are conservative with respect to the actual Cycle 7 calculated values.

3. In Reference 1, the correct values used for safety analyses were listed in Table 7.3.1-1. It was erroneously stated in Section 5.1.3 of that reference that the values for the reference cycle were the same as those used in Reference 3. The values listed in the above table were the actual values used in the safety analyses of the reference cycle. This table and the corresponding text are included herein only to correct this textual error.

							0.80	1.05
				0.80	1.09	1.25	0.80	1.14
			0.96	1.10	0.81	1.06	0.87	1.00
		0.96	1.35	0.98	1.23	0.94	1.28	0.80
	0.80	1.11	0.98	1.09	0.97	1.02	0.89	1.14
	1.09	0.81	1.24	0.97	1.19	0.82	1.11	0.84
	1.26 X	1.06	0.95	1.03	0.81	1.22	0.73	1.11
0.80	0.79	0.88	1.30	0.90	1.12	0.75	0.96	0.96
1.05	1.14	1.00	0.80	1.14	0.84	1.11	0.96	0.61

NOTE: X = MAXIMUM 1-PIN PEAK = 1.59

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 7 ASSEMBLY RELATIVE POWER DENSITY AT BOC, EQUILIBRIUM XENON	Figure 5-1



						0.78	0.99	
				0.76	1.02	1.17	0.80	1.10
			0.87	1.02	0.80	1.03	0.88	1.00
		0.87	1.19	0.92	1.18	0.94	1.26	0.83
	0.76	1.02	0.92	1.07	0.98	1.06	0.93	1.18
	1.02	0.81	1.18	0.98	1.21	0.91	1.19	0.95
0.78	1.17 X	1.04	0.95	1.06	0.91	1.30	0.85	1.24
0.99	0.79	0.88	1.27	0.93	1.19	0.86	1.13	1.14
	1.10	1.00	0.83	1.18	0.95	1.24	1.14	0.78

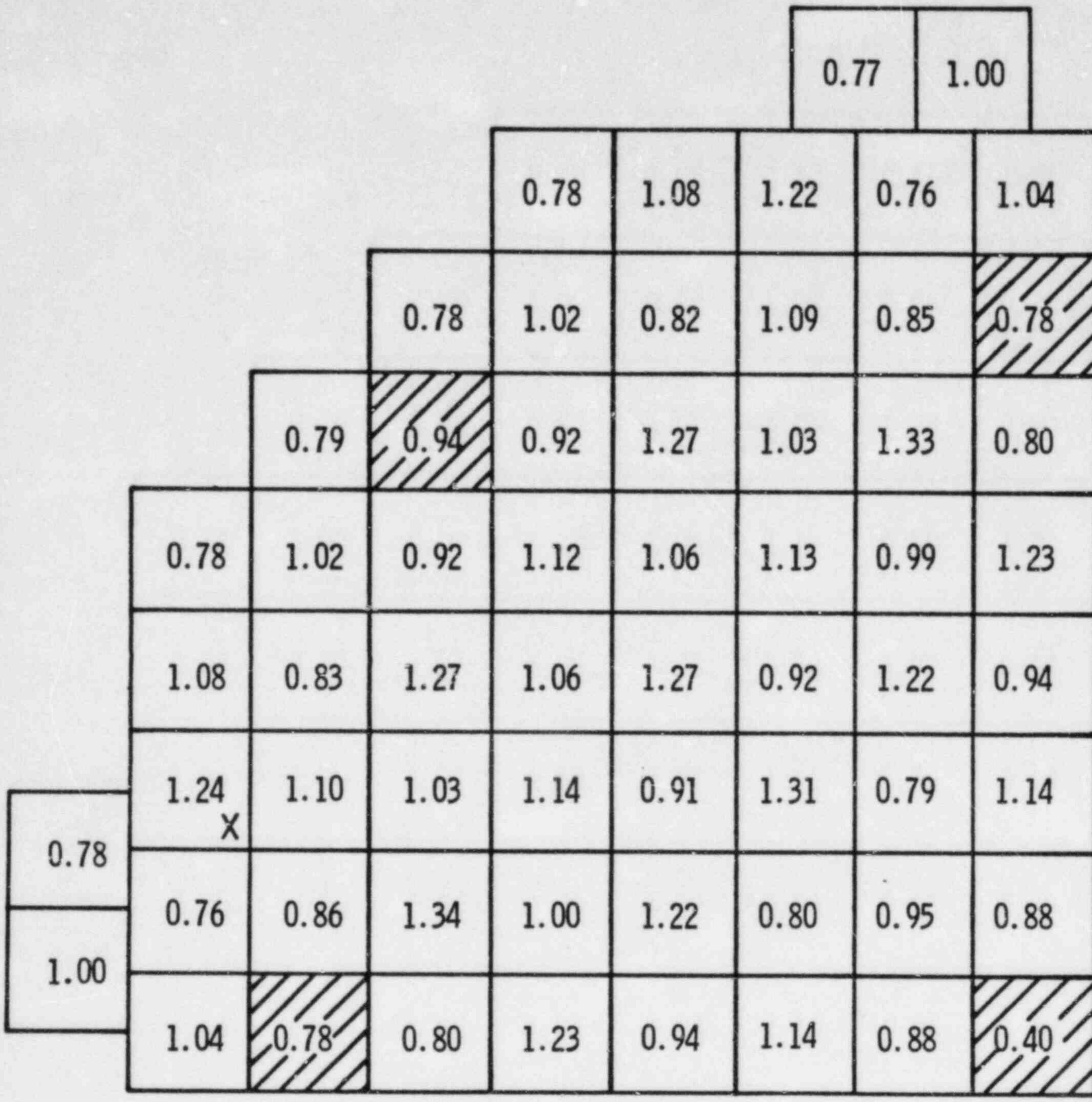
NOTE: X = MAXIMUM 1-PIN PEAK = 1.47

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

							0.82	1.02
				0.80	1.04	1.18	0.84	1.12
			0.89	1.03	0.84	1.04	0.90	1.02
		0.89	1.17	0.93	1.16	0.94	1.23	0.85
	0.80	1.03	0.93	1.05	0.96	1.04	0.92	1.14
	1.04	0.84	1.16	0.96	1.17	0.92	1.15	0.95
	1.18 X	1.04	0.94	1.04	0.92	1.24	0.85	1.18
0.83	0.84	0.90	1.23	0.93	1.15	0.86	1.08	1.08
1.02	1.12	1.02	0.85	1.14	0.95	1.18	1.08	0.78

NOTE: X=MAXIMUM 1-PIN PEAK = 1.45

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 7 ASSEMBLY RELATIVE POWER DENSITY AT EOC, EQUILIBRIUM XENON	Figure 5-3



NOTE: X=MAXIMUM 1-PIN PEAK = 1.56

CEA BANK 5 LOCATIONS

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 7 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, BOC	Figure 5-4

						0.84	1.03	
				0.80	1.06	1.20	0.83	1.04
			0.72	0.96	0.86	1.07	0.86	0.75
		0.72	0.78	0.85	1.18	1.00	1.24	0.83
	0.80	0.96	0.85	1.05	1.02	1.13	1.00	1.20
	1.06	0.86	1.18	1.02	1.25	1.00	1.24	1.02
	1.20	1.07	1.00	1.13	1.00	1.34 X	0.91	1.22
0.84	0.83	0.86	1.24	1.00	1.24	0.92	1.07	1.00
1.03	1.04	0.75	0.83	1.20	1.02	1.22	1.00	0.47

NOTE: X=MAXIMUM 1-PIN PEAK = 1.48

CEA BANK 5 LOCATIONS

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 1 CYCLE 7 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 7 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 for both safety analyses and for generating reactor protective system setpoint information. The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) listed in Table 6-1 have been combined statistically with other uncertainty factors at the 95/95 confidence/probability level (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 7.

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

6.2 Effects of Fuel Bowing on DNBR Margin

The effects of fuel rod bowing on DNB margin for Calvert Cliffs Unit 1 Cycle 7 have been evaluated using the methods described in Reference 8. These methods were approved by NRC in Reference 9.

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 15%, substantially exceeding the corresponding rod bow penalties based upon Reference 8. 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 2, Cycle 5	Cycle 7
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	^oF	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	185,532***	183,000****
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	48,415***	49,100****
Film Coefficient at Average Conditions	BTU/hr-ft^2-^oF	5930	5930

TABLE 6-1
(continued)

General Characteristics	Unit	Reference+ Unit 2, Cycle 5	Cycle 7
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	6.20***	6.12****
Average Core Enthalpy Rise	BTU/lb	66.5	66.5
Maximum Clad Surface Temperature	°F	657	657

Calculational Factors	Reference Unit 2, Cycle 5	Cycle 7
Enginering 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 7.

**Due to the statistical combination of uncertainties described in References 6, 10, and 11, the nominal inlet temperature and nominal primary system pressure were used to calculate some of these parameters.

***Based on a value of 928 shims.

****Based on a Value of 405 shims and non-fuel rods.

+Reference cycle (Unit 2, Cycle 5) analysis is contained in Reference 12.

7.0 TRANSIENT ANALYSIS

This section presents the results of the Baltimore Gas & Electric Calvert Cliffs Unit 1, Cycle 7 Non-LOCA safety analysis at 2700 MWt.

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

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

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

3. Postulated Accidents

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

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

For the event reanalyzed, Table 7-3 shows the reason for the reanalysis, the acceptance criterion to be used in judging the results and a summary of the results obtained. Detailed presentations of the results of the reanalysis are provided in Section 7.3.2.

An appendix is included to formally present an analysis of the Feed Line Break event. This event is included to support the installation of a safety grade auxiliary feedwater actuation system for Calvert Cliffs Unit 1.

TABLE 7-1

CALVERT CLIFFS UNIT 1, CYCLE 7
DESIGN BASIS EVENTS CONSIDERED IN THE NON-LOCA SAFETY ANALYSIS

		Analysis Status
7.1	Anticipated Operational Occurrences for which intervention of the RPS is necessary to prevent exceeding acceptable limits:	
7.1.1	Boron Dilution	Not Reanalyzed
7.1.2	Startup of an Inactive Reactor Coolant Pump[1]	Not Reanalyzed
7.1.3	Loss of Load	Not Reanalyzed
7.1.4	Excess Load	Not Reanalyzed
7.1.5	Loss of Feedwater Flow	Not Reanalyzed
7.1.6	Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed
7.1.7	Reactor Coolant System Depressurization	Not Reanalyzed
7.1.8	Excessive Charging Event	Not Reanalyzed
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]	Not Reanalyzed
7.2.2	Loss of Coolant Flow[3]	Not Reanalyzed
7.2.3	Full Length CEA Drop	Not Reanalyzed
7.2.4	Transients Resulting from the Malfunction of One Steam Generator[4]	Not Reanalyzed
7.2.5	Loss of AC Power[3]	Not Reanalyzed
7.3	Postulated Accidents	
7.3.1	CEA Ejection	Not Reanalyzed
7.3.2	Steam Line Rupture	Reanalyzed
7.3.3	Steam Generator Tube Rupture	Not Reanalyzed
7.3.4	Seized Rotor[3]	Not Reanalyzed

[1]Technical Specifications preclude this event during operation.

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

[3]Requires Low Flow Trip.

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

TABLE 7-2

CALVERT CLIFFS UNIT 1, CYCLE 7
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Physics Parameters	Units	Reference Cycle Values (Unit 2, Cycle 5)	Unit 1, Cycle 7 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	1.055
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^{o}F$	-2.5**→+.5	-2.5→+.5
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)	$\%\Delta\rho$	-5.2	-3.7+
Tilt Allowance	%	3.0	3.0

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

**The effective initial MTC assumed for the Unit II Cycle 5 SLB event was $-2.2\times10^{-4}\Delta\rho/^{o}F$.

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

TABLE 7-2
(continued)

Safety Parameters	Units	Reference Cycle Values (Unit 2, Cycle 5)	Unit 1, Cycle 7 Values
Power Level	MWt	2700*	2700*
Maximum Steady State Temperature	^{o}F	548*	548*
Minimum Steady State RCS Pressure	psia	2200*	2200*
Reactor Coolant Flow	10^{6} lbm/hr	138.5*	138.5*
Negative Axial Shape LCO Extreme Assumed at Full Power (Ex-Cores)	I_p	-.15*	-.15*
Maximum CEA Insertion at Full Power	% Insertion of Bank 5	25	25
Maximum Initial Linear Heat Rate for Transient Other than LOCA	KW/ft	16.0	16.0
Steady State Linear Heat Rate for Fuel CTM Assumed in the Safety Analysis	KW/ft	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 2a, 2b, 2c. These procedures have been approved by NRC for the Calvert Cliffs Units in Reference 3.

TABLE 7-3

DESIGN BASIS EVENT REANALYZED FOR UNIT 1, CYCLE 7

Event	Reason for Reanalysis (changes relative to reference cycle)	Acceptance Criterion	Summary of Results
Steam Line Break	Changes in moderator cooldown curve and available scram worth at trip due to Tech. Spec. MTC limit change, resulting from extended burnup.	Site boundary dose within 10CFR100 limits.	Site boundary doses are a small fraction of 10CFR100 limits; post trip MDNBR does not violate 1.30 (MacBeth) SAFDL for the limiting case. Further details in Section 7.3.2.

7.3.2 Steam Line Rupture Analysis

The steam line rupture (SLB) event has been reanalyzed for Calvert Cliffs Unit I and Unit II. The purpose of the analysis is to provide results based on conservatively enveloping initial conditions and assumptions such that the results will be applicable to a large number of future plant operating cycles. This analysis incorporates the effects of the safety grade auxiliary feedwater actuation system (AFAS). A spectrum of steam line break sizes, both inside and outside containment initiated from hot full power (HFP) and hot zero power (HZP) were analyzed. In addition, the analysis was performed with and without Loss of AC (LOAC) power on turbine trip. For outside containment breaks which credit a high power trip, a study parametric on moderator temperature coefficient (MTC) was also performed. The acceptance criteria for this postulated accident is that site boundary doses will be within the 10CFR100 guidelines. The results of the most limiting steam line break event, inside and outside containment are presented herein.

Analysis Assumptions and Initial Conditions

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 life, since this MTC results in the greatest positive reactivity change during the RCS cooldown caused by the steam line rupture. Since the reactivity change associated with moderator feedback varies significantly over the moderator density covered in the analysis, a curve of reactivity insertion versus density rather than a single value of MTC, is assumed in the analysis. The moderator cooldown curve assumed 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 defect associated with the fuel temperature decreases was also based on an end of life Doppler defect. The Doppler defect based on an end of life fuel temperature coefficient (FTC), in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the steam line rupture event. The Dopper multiplier (uncertainty) 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 life 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 and Low Pressure Safety Injection pumps.

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip at the maximum allowed power level is $5.56\%\Delta\rho$. 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 4 and 5). The actual credit differs from that used in the Calvert Cliffs Unit II Cycle 5 steam line break event (Reference 1) in that the credit in this analysis is Calvert Cliffs specific. The Calvert Cliffs Unit II Cycle 5 analyses used a small fraction of the negative reactivity credit justified for St. Lucie Unit 2 (Flordia Power and Light).

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 Steam Generator Isolation Signal (SGIS) 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.

A safety injection actuation analysis setpoint of 1578.0* psia 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 6) with the Lee non-uniform mixing correlation factor (Reference 7).

Analysis Assumptions and Initial Conditions

SLB Outside Containment

The results of the inside containment break showed that no post-trip Return-To-Power will occur for break sizes less than approximately 3.0 ft^2; consequently, no post-trip return to power will occur for outside containment breaks due to the 2.35 ft^2 in-line flow restrictor. For break sizes greater than 3.0 ft^2 the inside containment analysis presented above is limiting with respect to R-T-P. Therefore, the assumptions for the outside containment break listed below are designed to maximize the power excursion prior to reactor trip, rather than maximizing the post trip return to power.

The limiting SLB event outside containment was initiated from conditions listed in Table 7.3.2-2. The assumptions for the outside containment break which differ from the inside containment event are listed below.

The reactivity defect associated with the fuel temperature change was based on a beginning of life Doppler defect. This Doppler defect based on a beginning of life fuel temperature coefficient (FTC), in conjunction with increasing fuel temperatures, causes the minimum negative reactivity insertion and the maximum power excursion prior to reactor trip. The Doppler Multiplier (uncertainty) on the FTC assumed in the analysis is given in Table 7.3.2-2; a conservatively low value is employed to minimize Doppler feedback during the pre-trip power excursion. The β fraction assumed is the end of cycle minimum absolute value including uncertainties. This too is conservative since it maximizes the power excursion prior to trip by minimizing the contribution of delayed neutrons to the rate-of-change of power.

A spectrum of MTCs was employed to determine the effect of MTC on power range detector response during an outside containment steam line break. Only the High Power Trip and/or the Low Steam Generator Pressure Trip were credited.

*Conservative compared to current Technical Specification and existing uncertainties.

The assumptions made to maximize the site boundary dose are given in Table 7.3.2-3. During the event, two sources of radioactivity contribute to the site boundary dose: (1) the initial activity in the steam generator and (2) the activity associated with primary to secondary leakage. The primary activity includes the maximum initial activity allowed by the Technical Specifications and any activity released to the coolant due to fuel failure. The analysis conservatively assumed that all fuel pins with minimum DNBR below the design limit of 1.23 (CE-1 correlation) failed. The minimum DNBR during the transient was calculated using the thermal-hydraulic code CETOP (Reference 8). In calculating the site boundary dose, the analysis conservatively assumed that all activity is released to the atmosphere with a decontamination factor of 1.0.

Results

SLB Inside Containment

The SLB event with Loss of AC (LOAC) power on turbine trip results in the maximum post trip return-to-power (R-T-P) and, thus, the minimum post trip transient DNBR. This occurs because LOAC power causes the Reactor Coolant Pumps (RCPs) to coast down. The decreasing coolant flow is assumed to result in no flow mixing at the core inlet plenum. Thus, cold edge temperatures are used to calculate the moderator reactivity insertion. This resulted in more positive reactivity being inserted and produced the maximum post trip R-T-P. In addition, the lower core flows resulted in minimizing the transient DNBR.

The results of the parametric analysis in break size indicate that the largest break size results in the maximum post trip R-T-P and, thus, the minimum post trip DNBR. This occurs because the largest break size causes the greatest temperature reduction and, thus, inserts the greatest magnitude of positive reactivity due to moderator reactivity feedback. This results in a higher R-T-P and minimum post trip DNBR. Therefore, the results of the largest inside containment SLB with LOAC on turbine trip are presented herein.

The sequence of events for the 6.305 ft^2 SLB with LOAC on turbine trip initiated from HFP conditions is given in Table 7.3.2-4. 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 7.3.2-3 through 7.3.2-8.

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.

A LOAC power on turbine trip is assumed to occur at 3.4 seconds. At this time, RCPs begin coasting down and the diesel generators start coming on line. At 13.4 seconds, the diesel generators reach full speed and shutdown sequencer is initiated to load emergency systems. At 23.6 seconds the safety injection actuation analysis setpoint is reached and diesel generators switch from shutdown sequencer to LOCA sequencer to load emergency systems. At 28.6 seconds one HPSI pump is loaded on line and at 58.6 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 8.8 seconds. At 28.8 seconds, the AFW block valves associated with the steam generator with 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 104.2 seconds, which terminates the cooldown of the RCS. A peak reactivity of $-.077\%\Delta\rho$ at 147.8 seconds is obtained. A peak R-T-P of 8.1% consisting of 4.7% fission power and 3.4% decay power, is produced at 147.8 seconds. The minimum transient DNBR SAFDL of 1.30 (MacBeth) is not violated during this event.

SLB Outside Containment

The outside containment SLB event initiated from HFP with a LOAC on turbine trip resulted in the maximum site boundary dose. The results of parametric studies in break size and MTC demonstrated that a break size of 0.65 ft^2 coupled with an MTC of $-1.08 \times 10^{-4} \Delta\rho/^{\circ}F$ resulted in the maximum number of predicted fuel failures.

The sequence of events for a 0.65 ft^2 SLB outside containment with LOAC on turbine trip initiated from HFP conditions is given in Table 7.3.2-5. The NSSS response during the transient are given in Figures 7.3.2-9 and 7.3.2-14.

The results of the analysis show that the SLB causes a reactor trip signal to be generated on low steam generator pressure at 33.9 seconds. The trip breakers open at 34.8 seconds, and the CEAs drop into the core at 35.3 seconds, terminating the power and heat flux increases.

A LOAC power on turbine trip is assumed to occur at 34.8 seconds. Reactor power reaches its peak value of 134% at 35.3 seconds

The Steam Generator Isolation Analyses Setpoint is reached at 33.9 seconds. At 34.8 seconds the MSIVs begin to close, and are completely closed at 40.8 seconds.

No return-to-power occurs. This is due to a slower cooldown rate allowing greater beneficial contribution to negative reactivity by decay heat and safety injection.

The 0.65 ft^2 SLB outside containment shows that <1% of the fuel pins experience DNB. To be consistent with the intended enveloping nature of this analysis, 2.0% fuel failure was assumed for site boundary dose calculations. The resultant site boundary dose is:

Thyroid (DEQ I-131) = 81 Rem
Whole Body (DEQ Xe-133) = .3 Rem

Conclusions

The results of the steam line break inside containment shows that the post-trip minimum DNBR is within the design limit of 1.3 (MacBeth). The SLB outside containment results in a site boundary dose which is within 10CFR100 guidelines. Therefore, the results of the inside and outside containment SLB events with LOAC power on turbine trip is acceptable for Unit 1 Cycle 7.

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	$\% \Delta\rho$	-5.56
Doppler Multiplier		1.15
Moderator Cooldown Curve	$\%\Delta\rho$ vs. density	See Figure 7.3.2-1
Inverse Boron Worth	PPM/$\% \Delta\rho$	85.0
Effective MTC	$x10^{-4} \Delta\rho/^{o}F$	-2.5
β Fraction (including uncertainty)		.0060

TABLE 7.3.2-2

KEY PARAMETERS ASSUMED IN THE OUTSIDE 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	2154.0
Initial Steam Generator Pressure	psia	860.0
Low Steam Generator Pressure Trip Setpoint	psia	640.0
Safety Injection Actuation Signal	psia	1578.0
Minimum CEA Worth Available at HFP	$\%\Delta\rho$	-5.56
Doppler Multiplier		0.85
Moderator Cooldown Curve	$\%\Delta\rho$ vs. density	See Figure 7.3.2-2
Effective MTC	$x10^{-4}\Delta\rho/^{o}F$	-2.7 to -.27
β Fraction (including uncertainty)		.0044

TABLE 7.3.2-3

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR
THE STEAM LINE BREAK EVENT

Parameter	Units	Value
Reactor Coolant System Maximum Allowable Concentration (DEQ I-131)[1]	μCi/gm	1.0
Steam Generator Maximum Allowable Concentration (DEQ I-131)[1]	μCi/gm	0.1
Partition Factor Assumed for All Doses	----	1.0
Atmospheric Dispersion Coefficient[2]	sec/M^3	$1.80x10^{-4}$
Breathing Rate	M^3/sec	$3.47x10^{-4}$
Dose Conversion Factor (I-131)	REM/Ci	$1.48x10^6$

[1]Tech Spec limits

[2]0-2 hour accident condition

TABLE 7.3.2-4

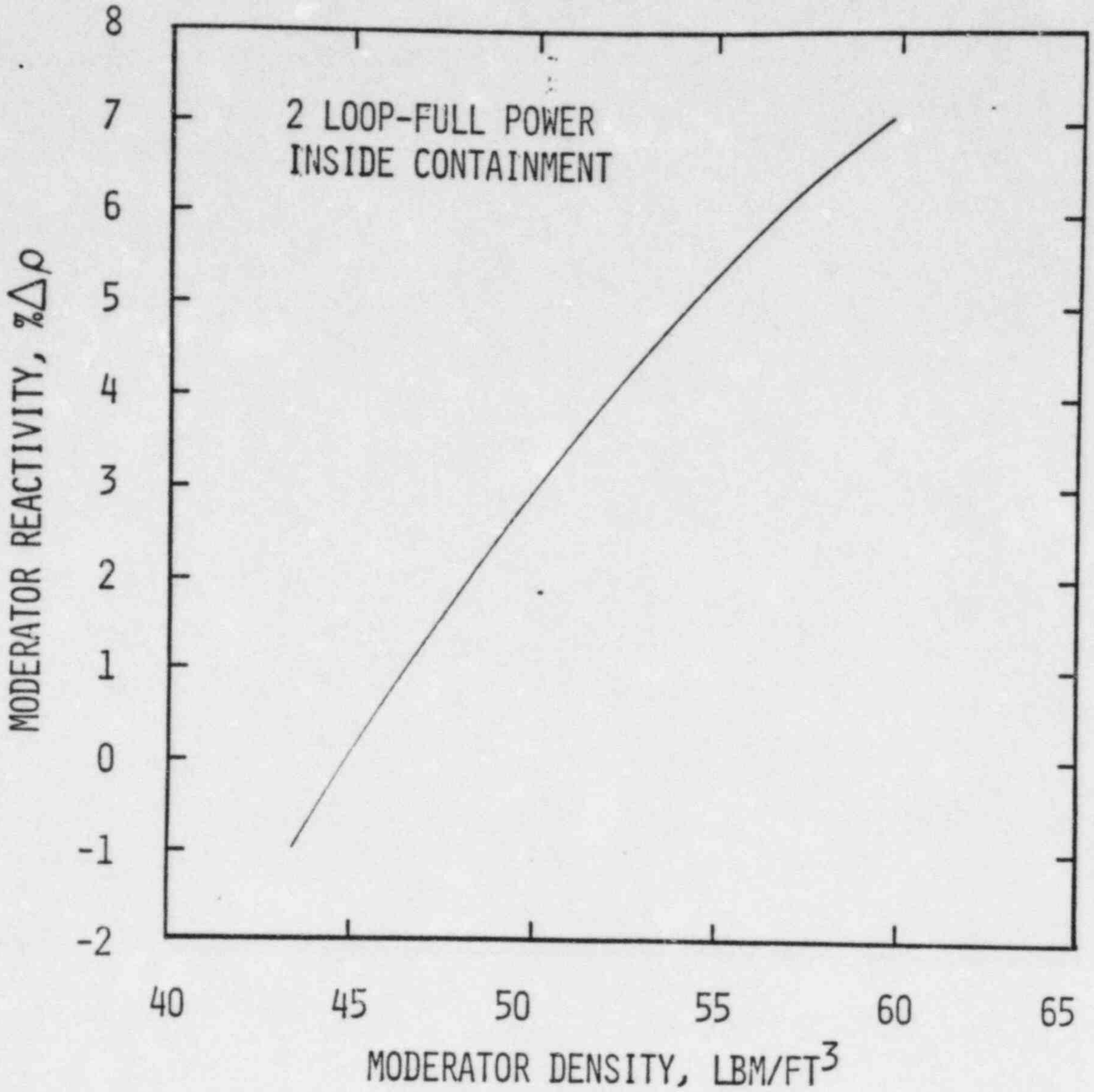
HFP 6.305 FT^2 BREAK
WITH LOAC, INSIDE CONTAINMENT

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; Loss of AC On Turbine Trip; RCPs Coastdown Begins; Main Feed Rampdown Begins; MSIVs Begin to Close; Diesel Starting Sequence Begins	
3.9	CEAs Enter Core	
8.8	Steam Generator Differential Pressure Setpoint Reached	365.0 psid
9.4	Main Steam Isolation Valves Fully Closed	
13.4	Diesel Generator Up to Speed	
23.2	Pressurizer Empties	---
23.6	Safety Injection Setpoint is Reached	1578 psia
28.6	Safety Injection Pumps Loaded on LOCA Sequencer	
28.8	AFW Block Valves Closed Providing Auxiliary Feedwater to Intact S.G. only	
58.6	Safety Injection Pumps up to Full Speed	
83.4	Main Feedwater Isolation Valves Completely Closed	---
104.2	Affected Steam Generator Blows Dry	---
147.5	Peak Power	8.1% of 2700 MWt
147.7	Peak Reactivity	$-.007\%\Delta\rho$

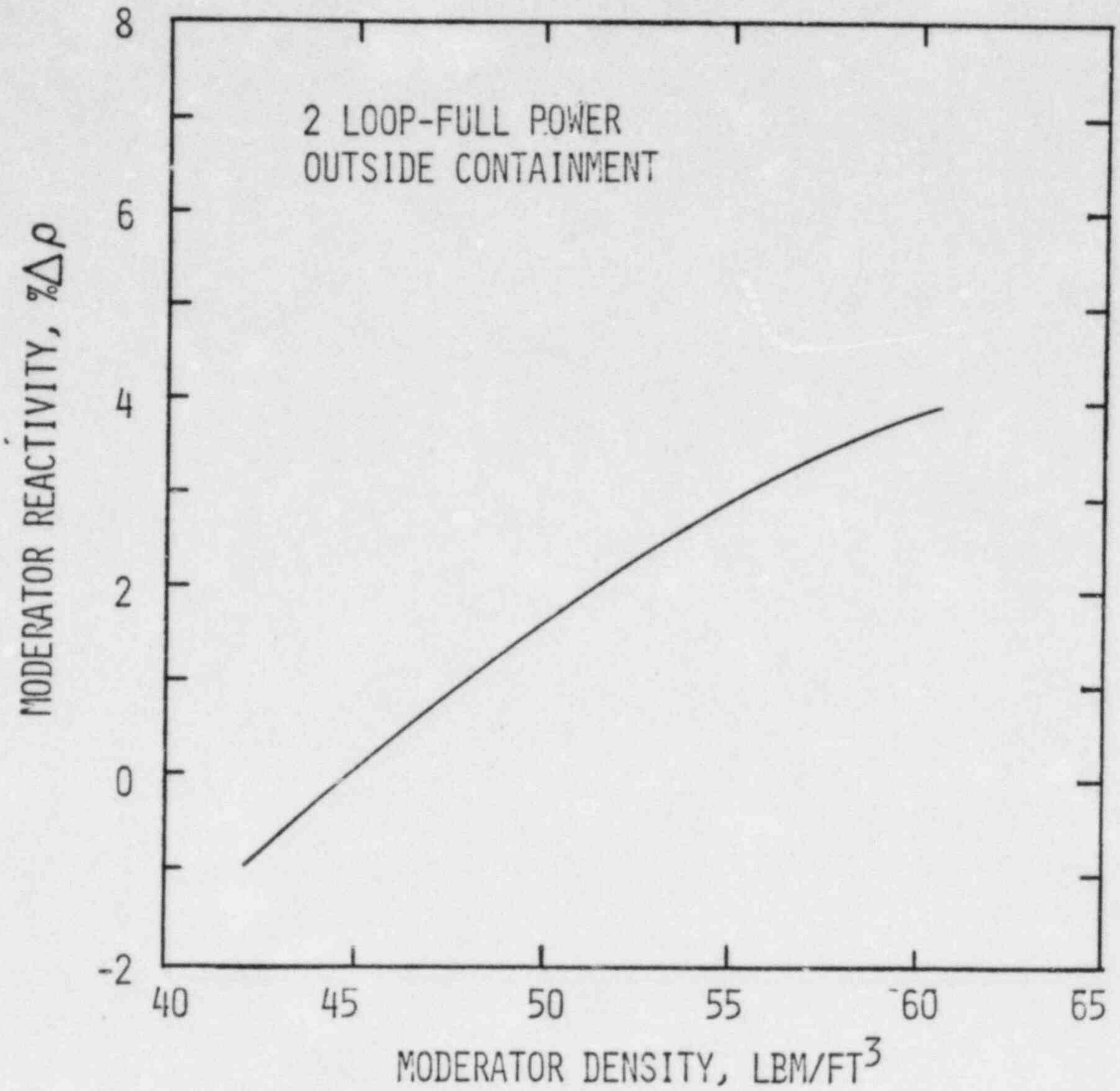
TABLE 7.3.2-5

HFP, 0.65 FT^2 BREAK
OUTSIDE CONTAINMENT

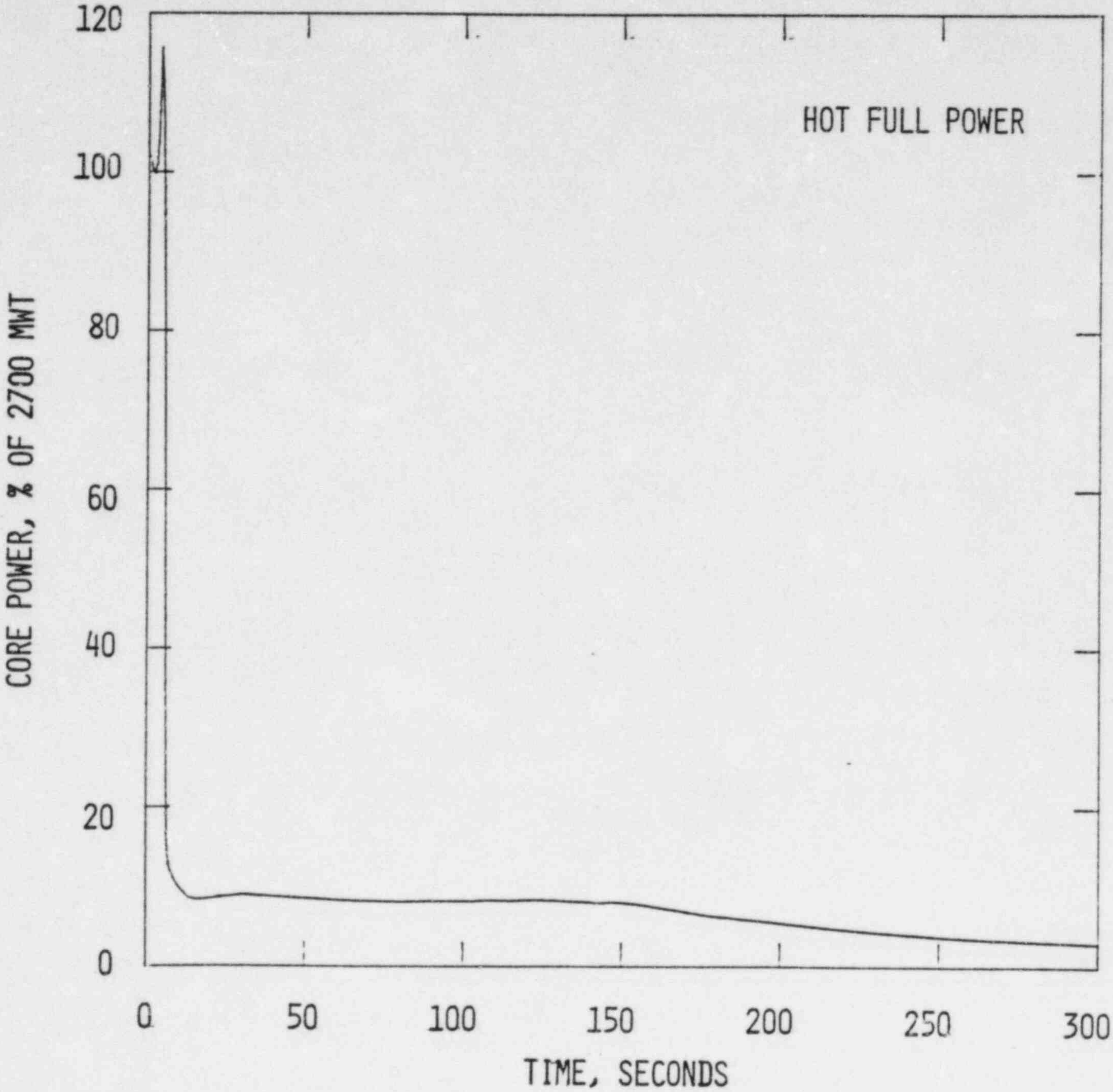
Time (sec)	Event	Setpoint or Value
0.0	Steam Line Break Occurs	0.65 ft^2
33.9	Reactor Trip Signal Generated on Low Steam Generator Pressure	640.0 psia
33.9	SGIS Signal Generated	640.0 psia
34.8	Trip Breakers Open; Loss of AC Power Begins; RCP Coastdown Begins	
34.8	MSIVs Begin to Close	
35.3	Maximum Power Reached	134%
35.3	CEAs Enter Core	
40.8	MSIVs Fully Closed	
	No Return to Power or Criticality Occurs	



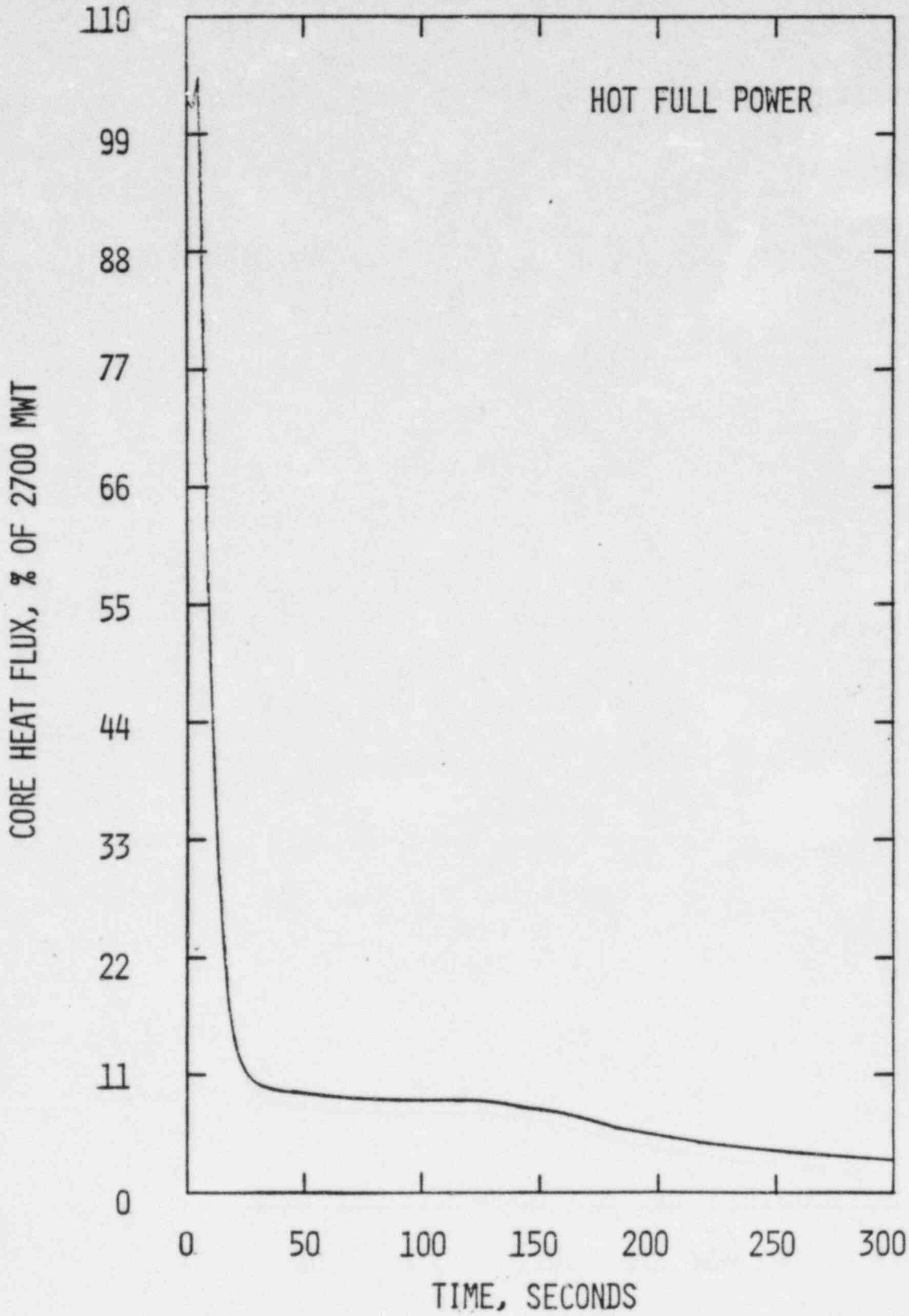
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT MODERATOR REACTIVITY VS MODERATOR DENSITY	FIGURE 7.3.2-1



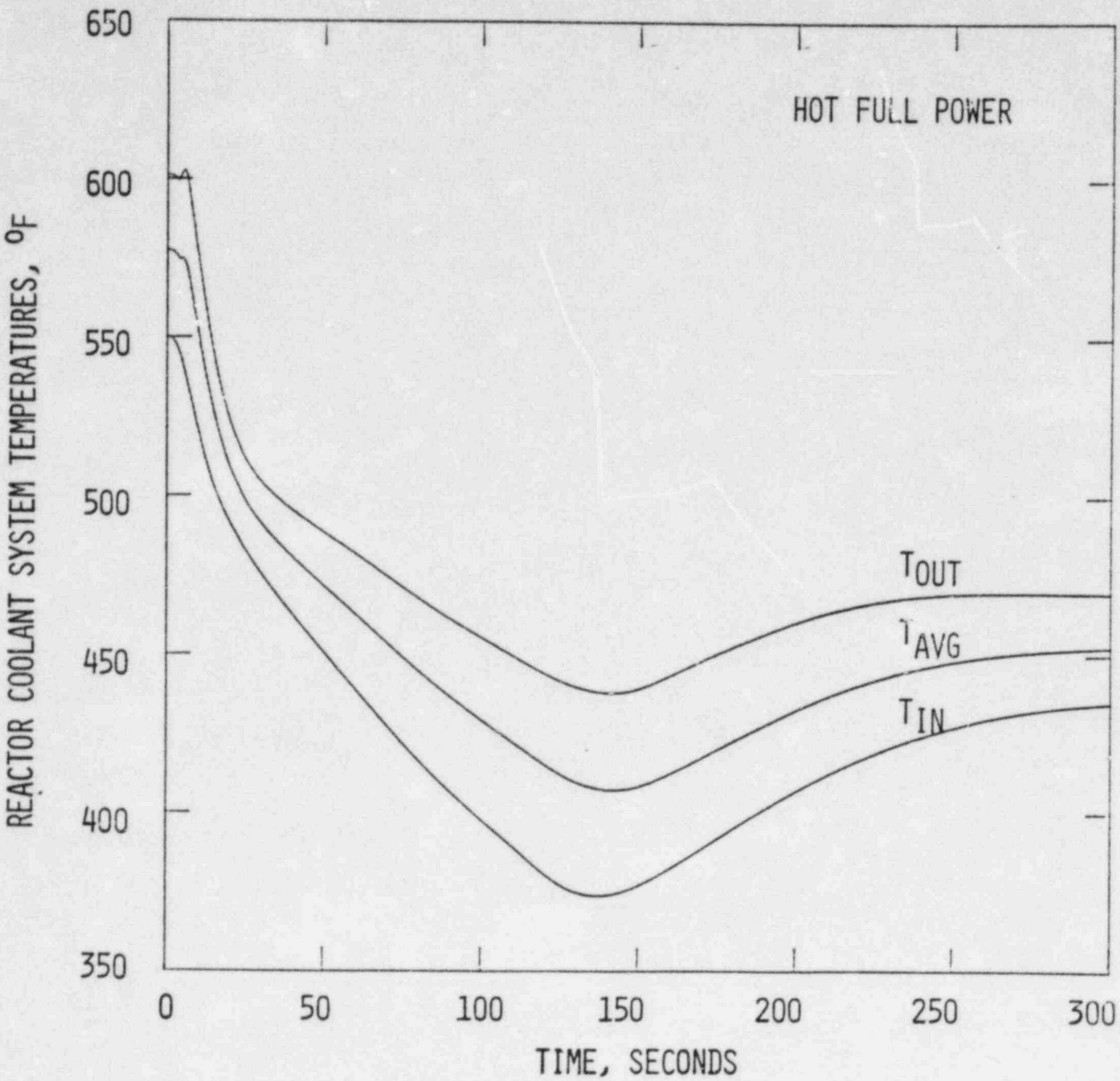
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT OUTSIDE CONTAINMENT MODERATOR REACTIVITY VS MODERATOR DENSITY	FIGURE 7.3.2-2



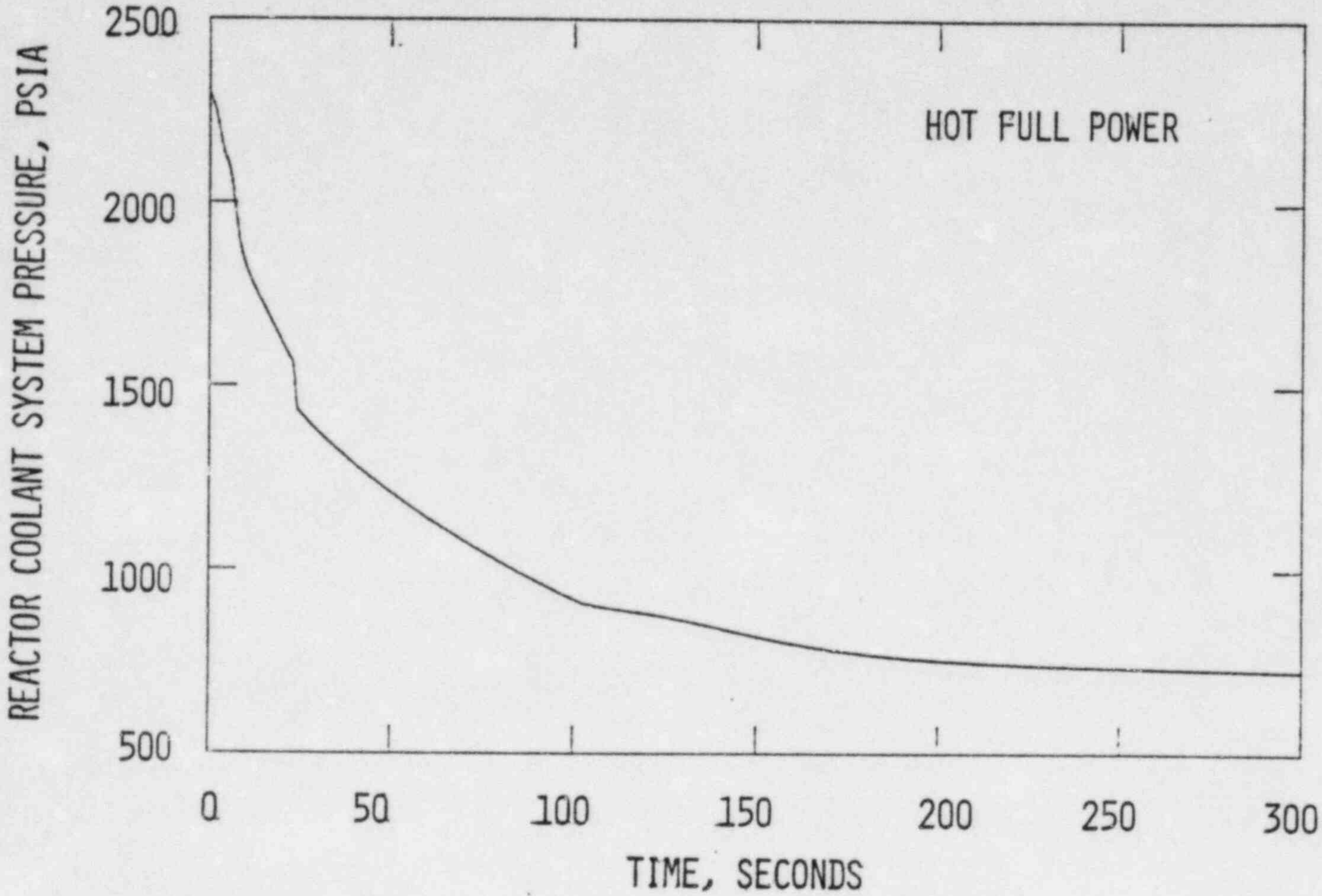
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT CORE POWER VS TIME	FIGURE 7.3.2-3



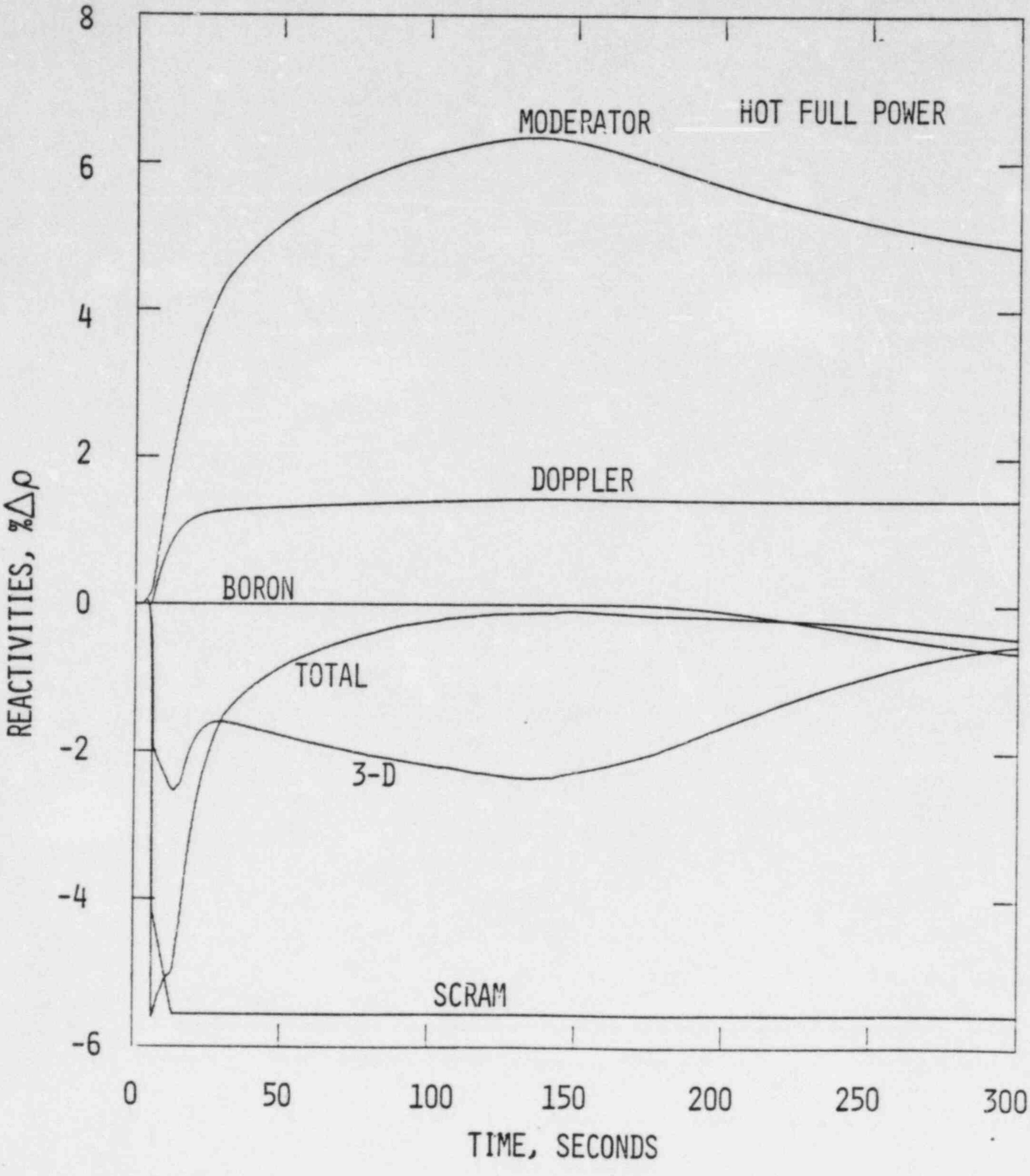
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT CORE HEAT FLUX VS TIME	FIGURE 7.3.2-4



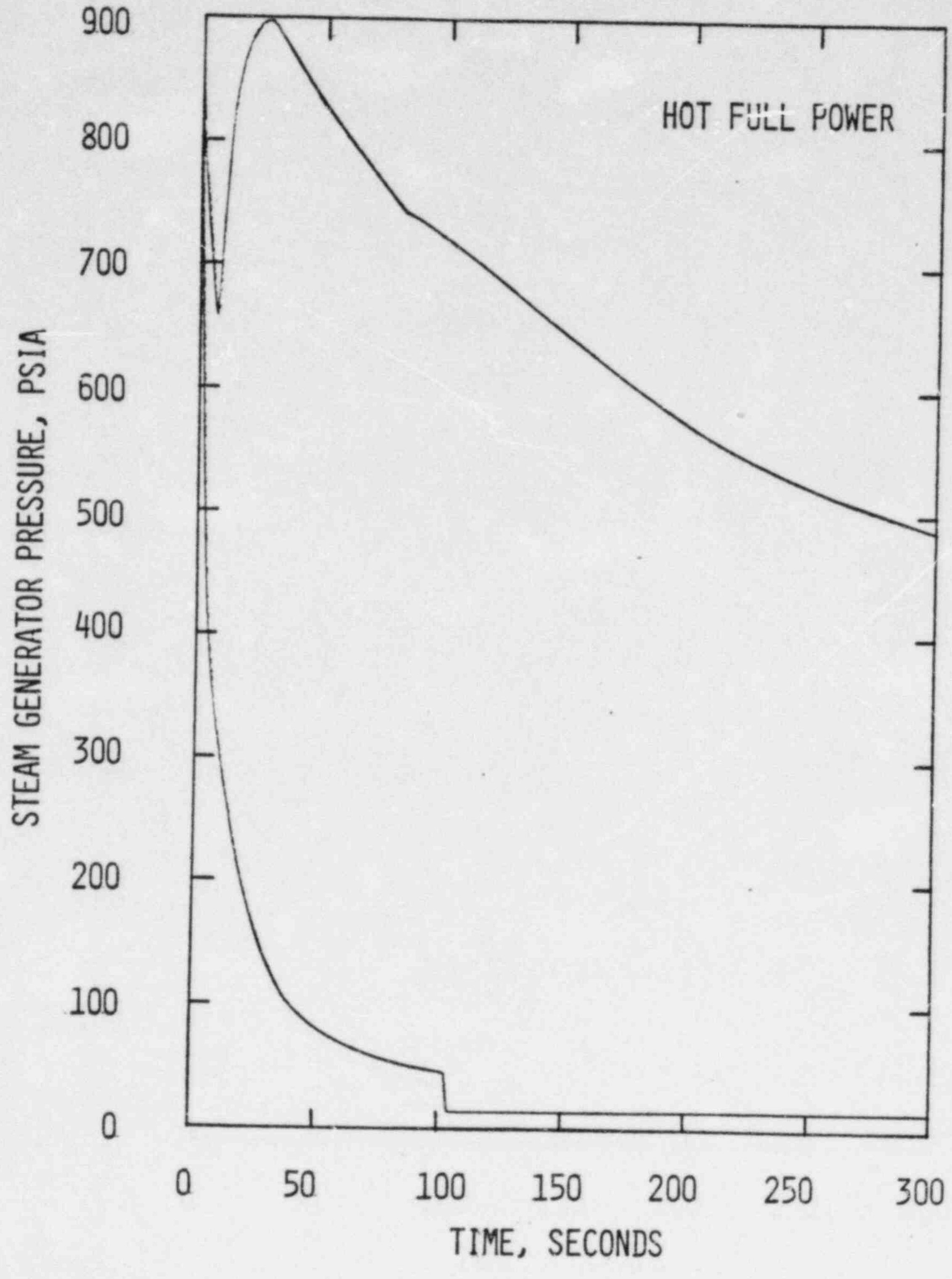
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7.3.2-5



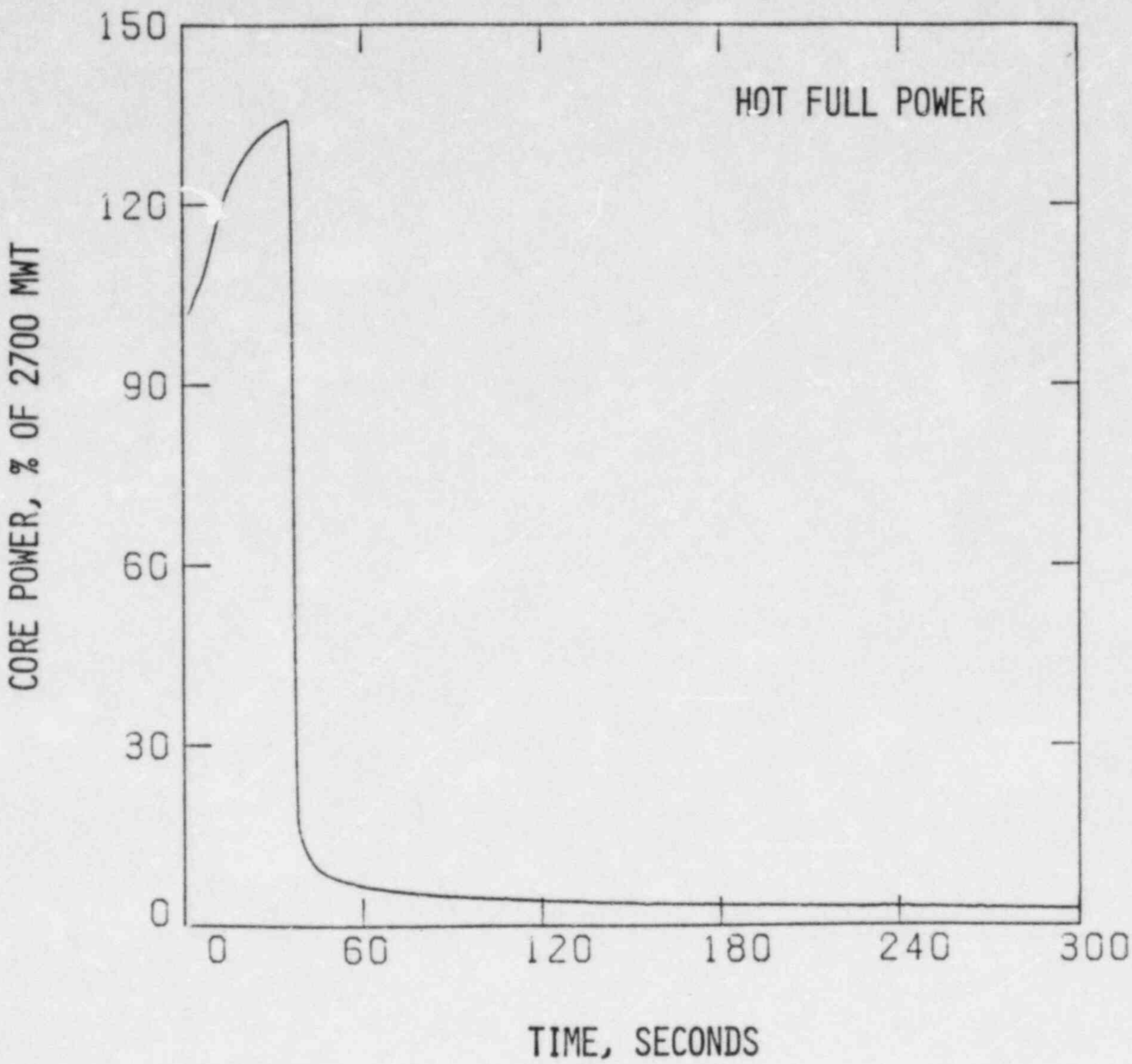
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7.3.2-6



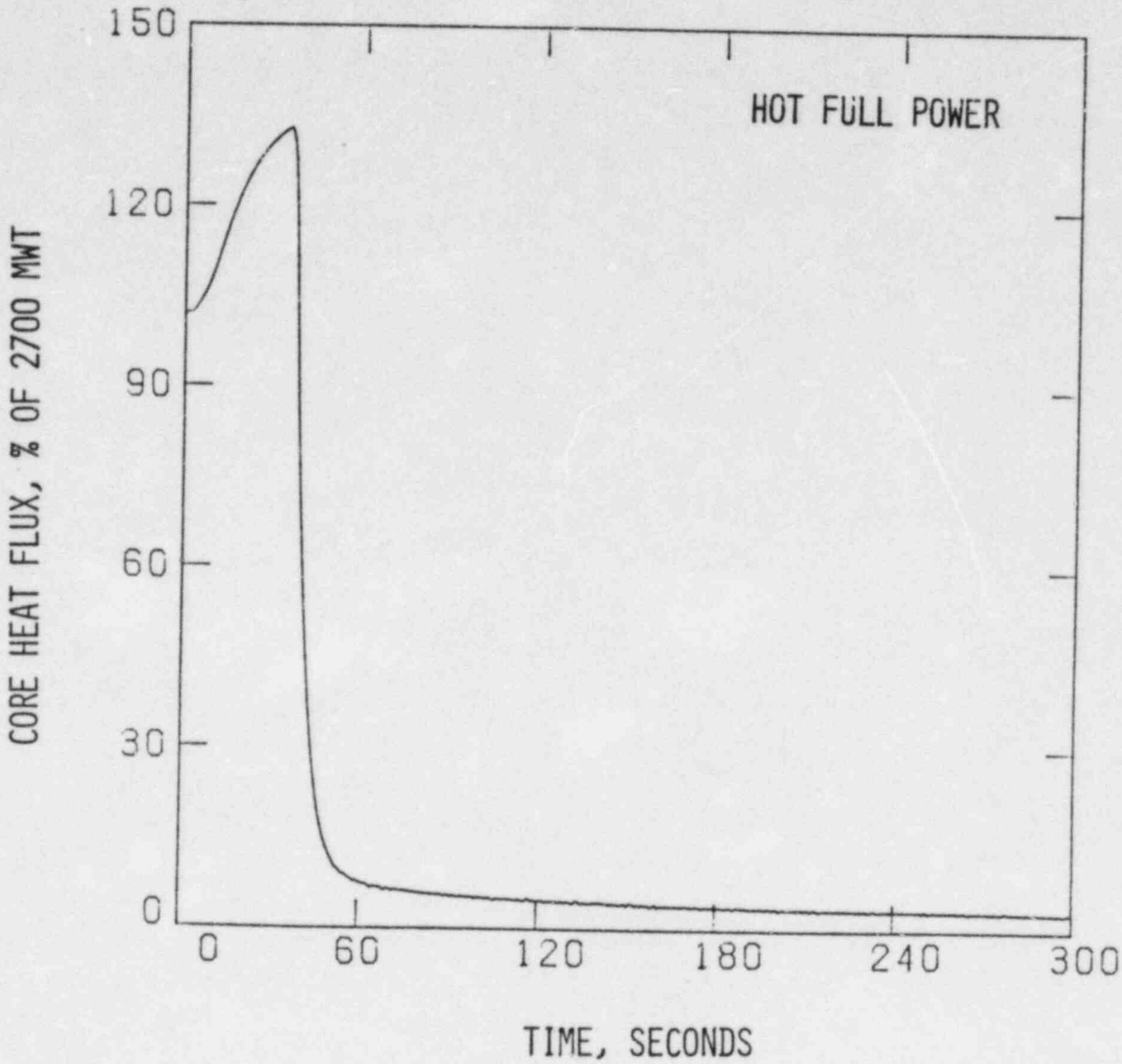
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT REACTIVITIES VS TIME	FIGURE 7.3.2-7



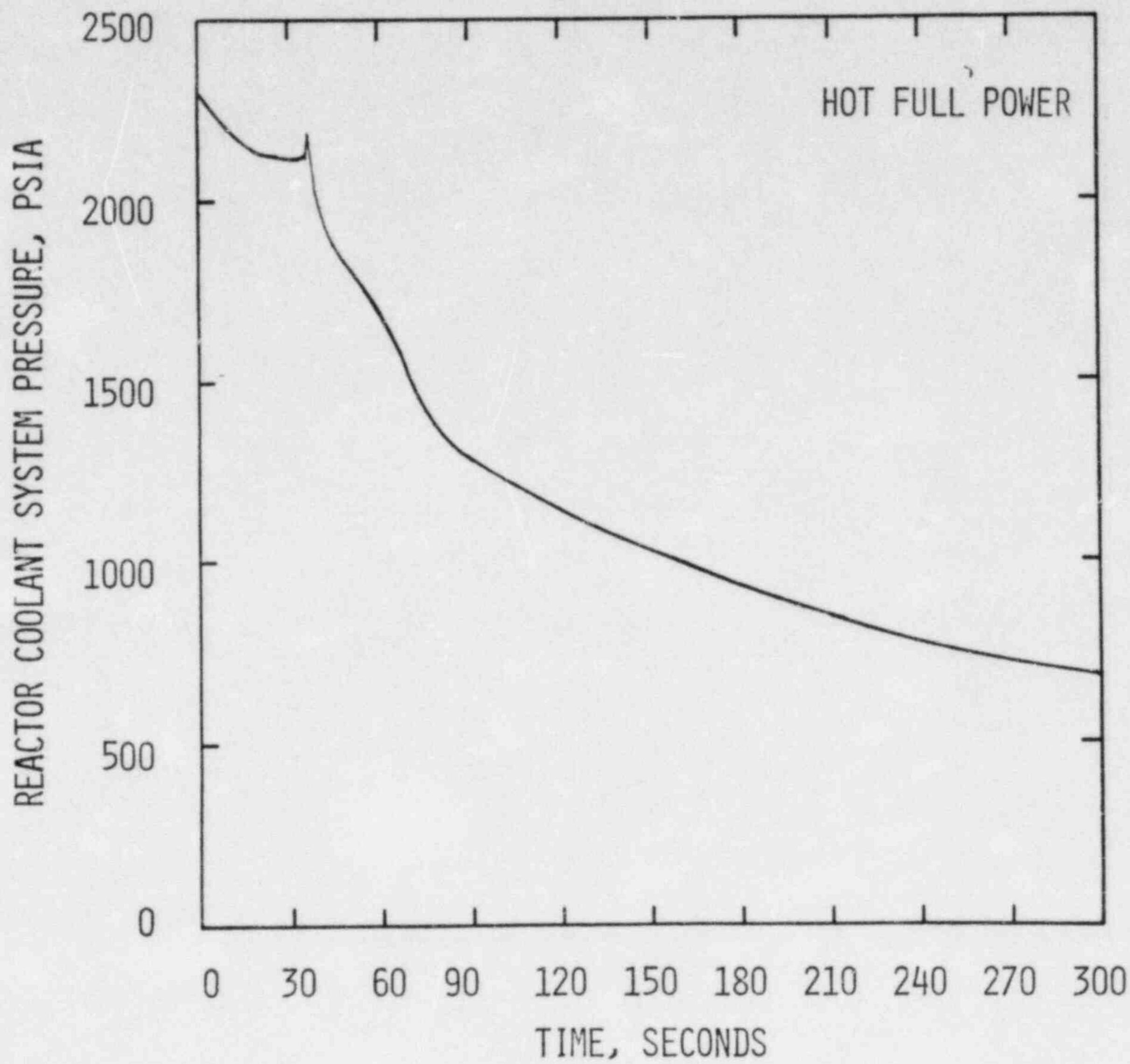
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT INSIDE CONTAINMENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 7.3.2-8



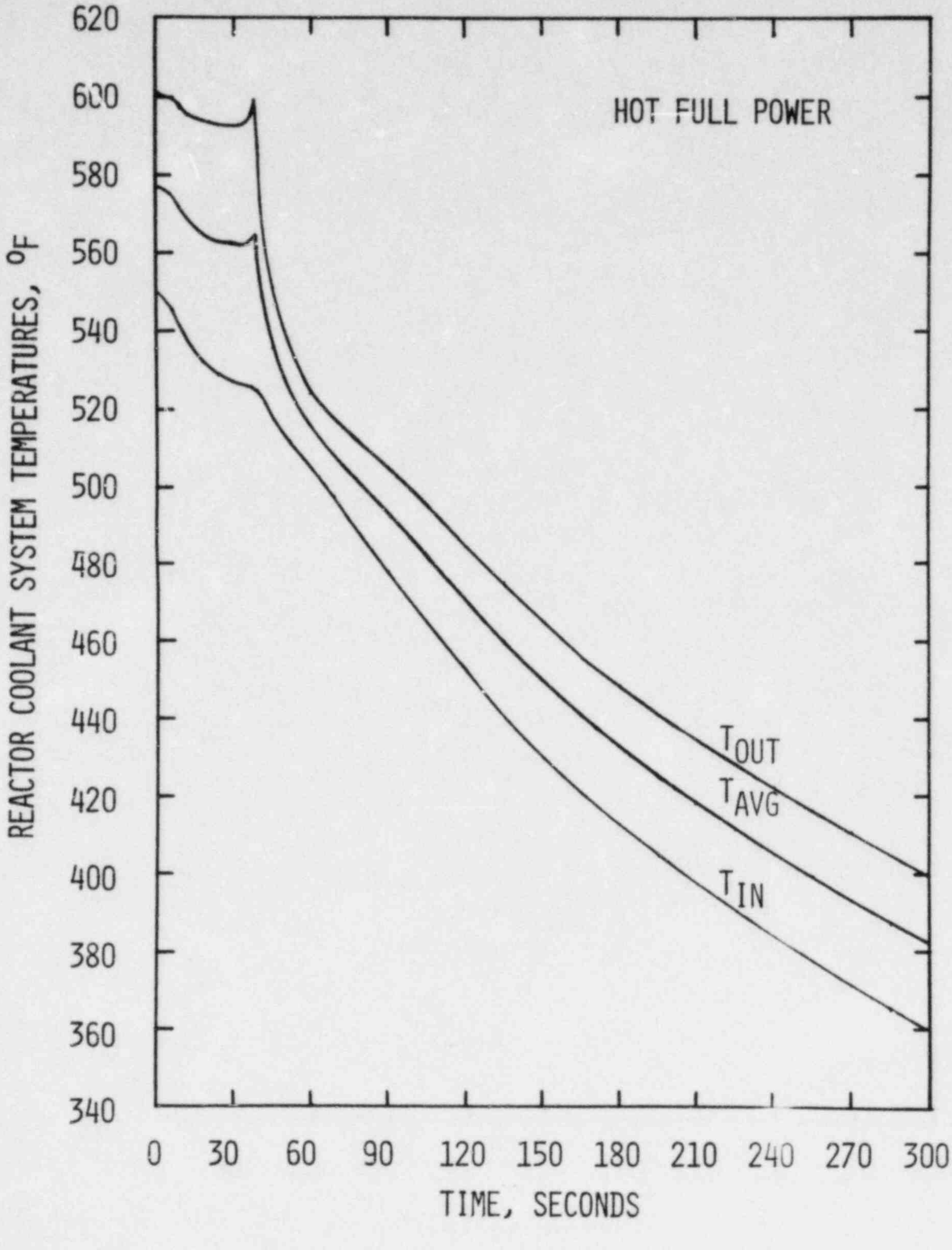
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT OUTSIDE CONTAINMENT CORE POWER VS TIME	FIGURE 7.3.2-9



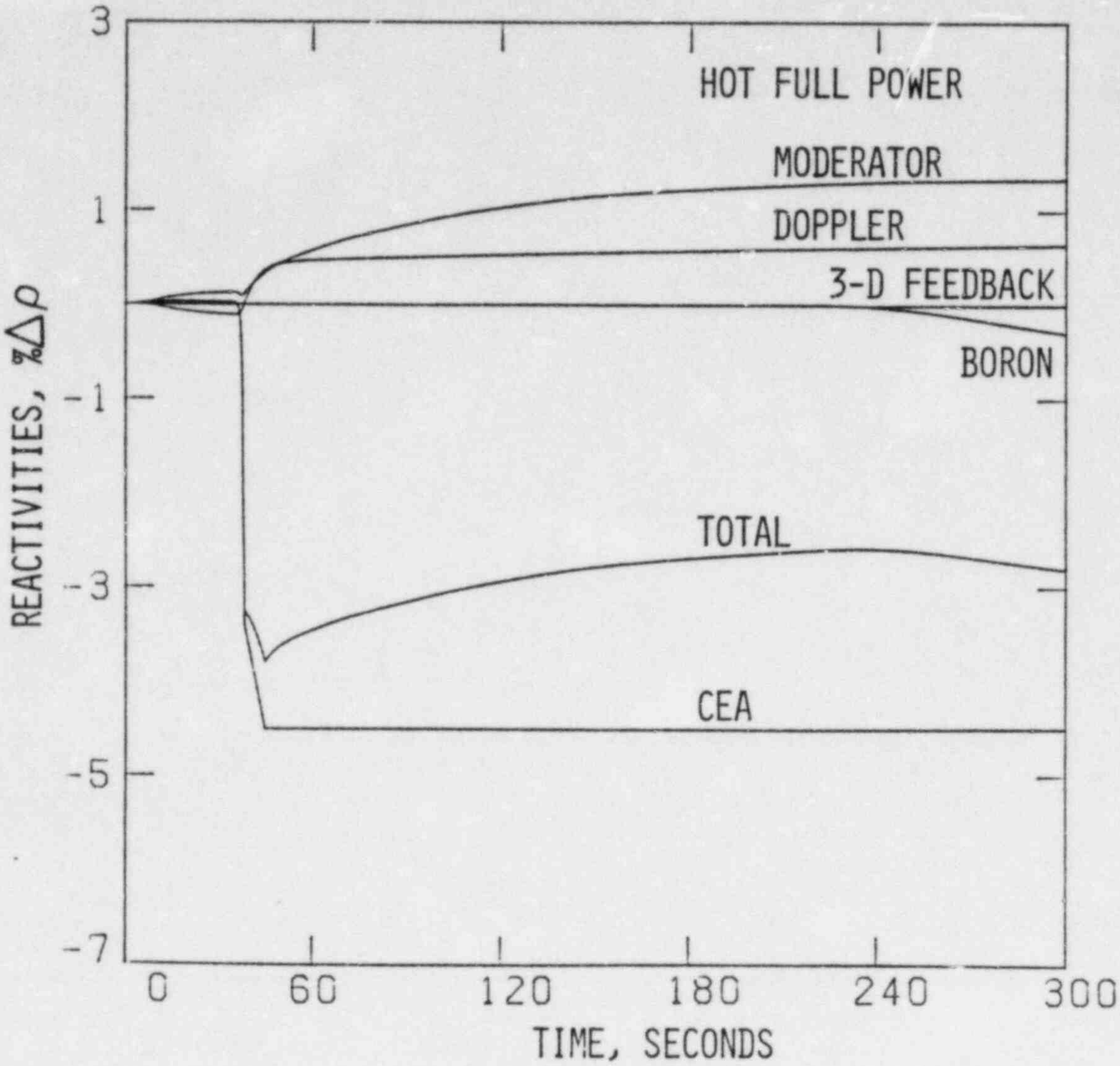
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT OUTSIDE CONTAINMENT CORE HEAT FLUX VS TIME	FIGURE 7.3.2-10



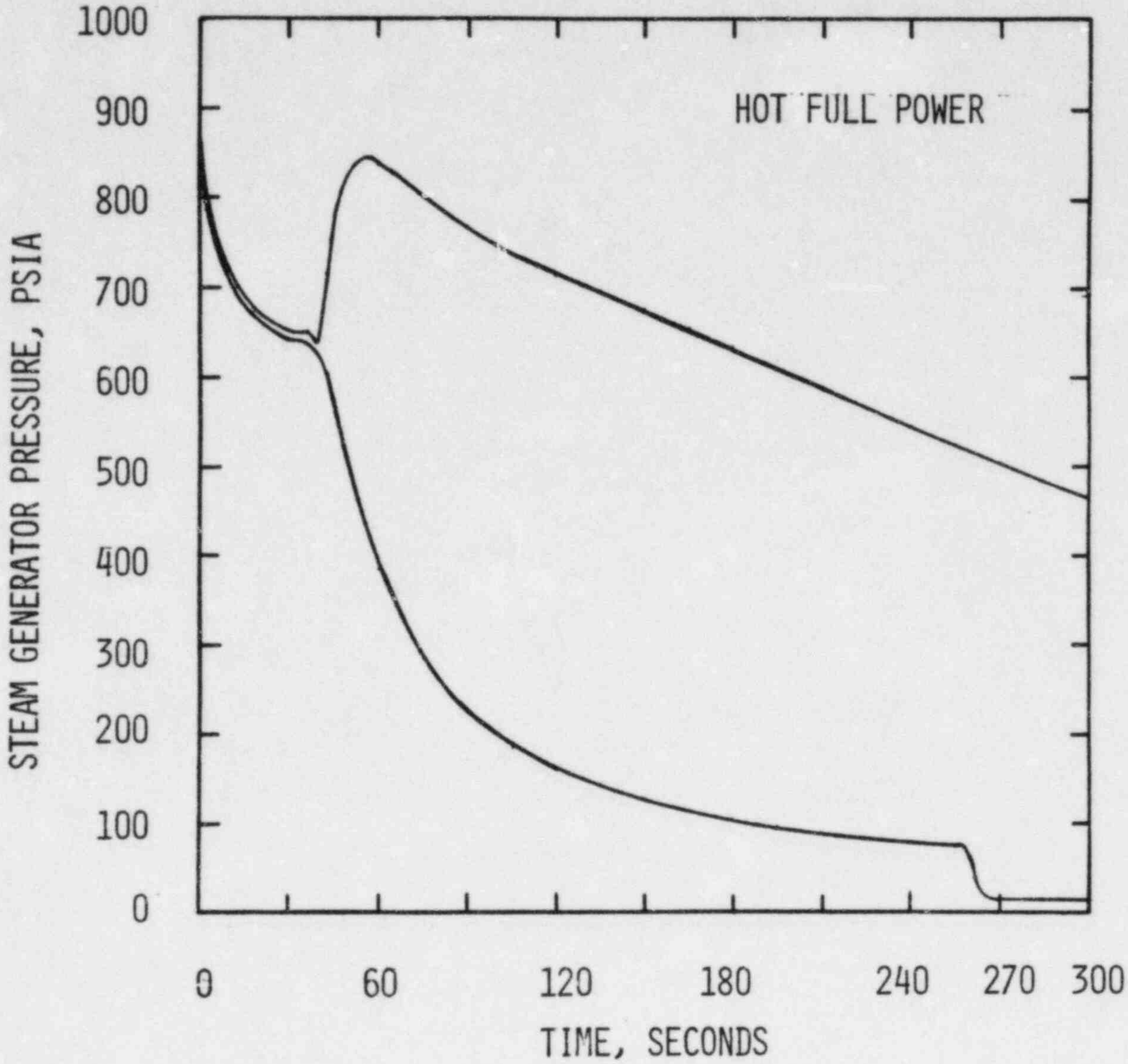
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT OUTSIDE CONTAINMENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7.3.2-11



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT OUTSIDE CONTAINMENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7.3.2-12



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT OUTSIDE CONTAINMENT REACTIVITIES VS TIME	FIGURE 7.3.2-13



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT OUTSIDE CONTAINMENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 7.3.2-14

APPENDIX TO CHAPTER 7

Feedline Break Event

Introduction

The Feedline Break event (FLB) was analyzed for Calvert Cliffs Unit 1 Cycle 7 to demonstrate that the RCS pressure limit of 2750 psia is not exceeded and that the site boundary doses do not exceed 10CFR100 guidelines. The break size and plant conditions that were found to result in the worst pressure excursion in the Calvert Cliffs Unit 2 Cycle 5 analysis were confirmed as worst for Unit 1 Cycle 7 and the results of this limiting break are presented herein.

Discussion

The FLB event is initiated by a break in the main feedwater system (MFS) piping. Depending on the break size and location and the response of the MFS, the effects of a break can vary from a rapid heatup to a rapid cooldown of the Nuclear Steam Supply System (NSSS). In order to discuss the possible effects, breaks are categorized as small, if the associated discharge flow is within the excess capacity of the MFS, and as large, if otherwise. Break locations are identified with respect to the feedwater line reverse flow check valve. The reverse flow check valve of concern is located between the steam generator feedwater nozzle and the containment penetration. Closure of the check valve, to prevent reverse flow from the steam generator, maintains the heat removal capability of that steam generator in the presence of a break upstream of the check valve.

Feedwater line breaks upstream of the reverse flow check valve can initiate one of the following transients. A break of any size, with MFS unavailable, will result in a Loss of Feedwater Flow (LOFW) event. A small pipe break with MFS available will result in no reduction in feedwater flow. Depending on the break size, a large break with MFS available will result in either a partial or a total LOFW event. Since FLBs upstream of the reverse flow check valve result in transients no more severe than a LOFW event, these FLBs were not analyzed.

In addition to the possibility of partial or total LOFW, FLBs downstream of the check valve have the potential to establish reverse flow from the affected steam generator back to the break. Reverse flow occurs whenever the MFS is not operating subsequent to a pipe break, or when the MFS is operating, but without sufficient capacity to maintain pressure at the break above the steam generator pressure. FLBs which develop reverse flow through the break are limiting with respect to primary overpressure. Thus, only these FLBs were considered in the analysis.

FLBs downstream of the check valve with reverse flow may result in either a RCS heatup or a RCS cooldown event, depending on the enthalpy of the reverse flow and the heat transfer characteristics of the affected steam generator. However, excessive heat removal through the feedwater line break is not considered in the analysis because the cooldown potential is less than that for the Steam Line Break (SLB) event. This occurs because SLBs have a greater potential for discharging high enthalpy fluid due to the location of the steam piping which is located above the feedwater piping within a steam generator. In addition, the maximum break area for a FLB is 2.2 ft^2 in comparison to 6.305 ft^2 for a SLB.

Unlike SLBs, FLBs cause a decrease in feedwater flow, resulting in lower steam generator liquid inventory which reduces the heat removal capacity. The reduced heat transfer capability results in a rapid RCS overpressurization and, thus, it is the heatup potential of a FLB which was analyzed.

A general description of the FLB event downstream of the check valves, with the MFS unavailable and with low enthalpy break discharge, is given below. The loss of subcooled feedwater flow to both steam generators causes increasing steam generator temperatures, decreasing liquid inventories and decreasing water levels. The rising secondary temperature reduces the primary-to-secondary heat transfer, which results in a heatup and pressurization of the RCS. The heatup becomes more severe as the affected steam generator experiences a further reduction in its heat transfer capability due to decreasing liquid inventory. The heatup of the RCS and the depletion of liquid inventory in the steam generator will initiate a reactor trip on either High Pressurizer Pressure or Steam Generator Low Water Level. The RCS heatup can continue even after a reactor trip, due to a total loss of heat transfer in the affected steam generator as the liquid inventory is completely depleted. The rise in RCS pressure causes the Pressurizer Safety Valves (PSVs) to open. The rise in secondary pressure is limited by the opening of the Main Steam Safety Valves (MSSVs). The opening of the PSVs and the MSSVs, in conjunction with the reactor trip (which reduces core power to decay level), mitigates the RCS overpressurization.

The reduction of liquid inventory in the unaffected steam generator in conjunction with low level S.G. signal initiates AFW flow to the unaffected steam generator. Automatic initiation of AFW is sufficient to provide a continued heat sink for the removal of decay heat.

Analysis Assumptions and Initial Conditions

The following is a discussion of the conservative assumptions and initial conditions chosen to maximize RCS pressure. Blowdown of the steam generator nearest the feedwater line break is modeled assuming frictionless critical flow as calculated by the Henry-Fauske correlation (Reference 9). The Feedwater Line Break location is conservatively modeled to be near the bottom of the steam generator, even though, in reality, the feedwater line nozzle is at a much higher elevation within the steam generator. The analysis assumes that saturated liquid is discharged through the break until the liquid mass reaches 5000 lbm, at which time saturated steam discharge is assumed. This assumption maximizes the liquid inventory discharge through the break, minimizes the energy removal from the primary by the steam generator, and thereby maximizes the RCS overpressurization.

The analysis also assumes that the effective heat transfer area is decreased linearly as the steam generator liquid mass decreases. The mass interval over which the rampdown is assumed to occur was conservatively chosen to model a rapid loss of heat transfer in the affected steam generator.

To maximize RCS pressure, the analysis conservatively credits only the high pressurizer pressure trip. This assumption maximizes the rate of change of pressure at the time of trip, and thus the peak pressure obtained following the trip. The analysis does not credit either the high containment pressure trip or the steam generator low water level trip.

Table 7A-1 presents the initial conditions chosen to maximize the RCS pressure. A Moderator Temperature Coefficient curve corresponding to beginning of cycle conditions is assumed. This MTC, in conjunction with increasing coolant temperatures, adds positive reactivity, and, thus, maximizes the rate of change of heat flux and pressure at the time of trip. A Fuel Temperature Coefficient (FTC) corresponding to beginning of cycle conditions is used in the analysis. This FTC causes the least amount of negative reactivity feedback, allowing higher increases in both the heat flux and RCS pressure. An uncertainty factor of 15% is used in the analysis.

An initial RCS pressure of 2154 psia is used in the analysis to maximize the rate of change of pressure at time of trip and, thus, the peak pressure obtained following a reactor trip. An initial steam generator pressure of 815 psia is assumed in the analysis. This pressure delays the opening of the Main Steam Safety Valves (MSSVs) and maximizes the peak RCS pressure.

The Steam Dump and Bypass System (SDBS), the Pressurizer Pressure Control System (PPCS), the Pressurizer Level Control System (PLCS) and the Power Operated Relief Valves (PORV) are assumed to be in the manual mode of operation. This assumption enhances the RCS pressure increase, since the automatic operation of these systems mitigates the RCS pressure increase.

This analysis conservatively assumed no automatic initiation of auxiliary feedwater. This assumption increases the RCS heatup and pressurization of concern in the FLB event. Credit was taken for manual initiation of auxiliary feedwater 10 minutes after reactor trip. The steam driven pump's auxiliary feedwater reaches the steam generator 58.0 seconds after the manual initiation. This includes 50 seconds required to open steam admission valves to the pump, 4.5 seconds for the pump to accelerate to speed and 3.5 seconds for the water to flood the piping and reach the steam generator. The flow to the steam generator is 434 gpm/leg.

The assumptions made to maximize the boundary site dose are given in Table 7A-2. During the event, two sources of radioactivity contribute to the site boundary dose: (1) the initial activity in the steam generator and (2) the activity associated with primary to secondary leakage. The leakage through the steam generator tubes is assumed to be the Technical Specification limit of 1.0 GPM. The initial primary and secondary activities are assumed to be at the Technical Specification limits of 1.0 μCi/gm and 0.1 μ Ci/gm, respectively. The analysis assumes that all of the initial activity in the steam generators and the primary activity due to the tube leakage are released to the atmosphere with a decontamination factor of 1.0, resulting in the maximum site boundary dose.

Results

The FLB event with Loss of AC (LOAC) power on reactor trip results in the maximum RCS pressure. This occurs because LOAC power causes the Reactor Coolant Pumps to coastdown. The reduced core flow decreases the rate of heat removal and, thus, maximizes the primary heatup and overpressurization. Thus, only the results of the FLB event with LOAC power on reactor trip are presented herein.

Figure 7A-1 presents the results of the parametric study to determine the break size which leads to the highest RCS peak pressure that was performed in the Calvert Cliffs Unit 2 Cycle 5 FLB event analysis. The trend and results of this parametric study were confirmed to be valid for the Unit 1 Cycle 7 FLB event. Figure 7A-1 shows that, initially, as the break size increases, so does the peak RCS pressure. This is due to faster water drainage out of the ruptured steam generator, which will cause a more rapid primary to secondary heat transfer rampdown. However, as the break size increases further, the greater steam relieving capacity of larger breaks (once the ruptured steam generator feedwater nozzle uncovers) will offset the faster heat transfer rampdown and will result in lower peak pressure. The highest peak pressure was obtained for a break size of 0.275 ft^2.

The sequence of events for a 0.275 ft^2 Feed Line Break downstream of the reverse flow check valve with LOAC on turbine trip is given in Table 7A-3. Figures 7A-2 through 7A-7 present the transient behavior of core power, core average heat flux, RCS temperatures, RCS pressure, steam generator pressure and steam generator liquid inventory for 1800 seconds of transient.

A 0.275 ft^2 break in the main feedwater line is assumed to instantaneously terminate feedwater flow to both steam generators and establish critical flow from the steam generator nearest the break. During the first 24.5 seconds of the event, the absence of subcooled feedwater flow causes the secondary pressure and temperature to increase, which reduces the primary to secondary heat transfer. This causes the primary pressures and temperatures to increase. At 24.5 seconds, the liquid inventory in the ruptured steam generator is sufficiently depleted to cause a further rampdown in the heat transfer rate. This causes the primary pressure and temperature to rapidly increase and at the same time causes the secondary pressure to decrease.

The rapid increase in primary pressure initiates a High Pressurizer Pressure Trip at 27.1 seconds. At 27.9 seconds, the pressure reaches 2525 psia, at which time the Pressurizer Safety Valves (PSVs) open to mitigate the increase in primary pressure. At 28.4 seconds, the turbine stop valves close, increasing the secondary pressure. At 28.5 seconds, the CEAs begin to drop into the core, inserting negative reactivity which mitigates the primary heatup. However, at this time, the Reactor Coolant Pumps (RCPs) are assumed to initiate flow coastdown due to LOAC power on turbine trip. The rapid decrease in core flow slows down the rate of heat removal from the primary. At 28.6 seconds, the feed ring is uncovered and steam is discharged through the break, which mitigates the primary heatup. These competing effects result in a peak RCS pressure of 2749 psia at 31.1 seconds. The increase in secondary pressure is mitigated by the opening of the Main Steam Safety Valves in the undamaged and ruptured steam generator at 35.5 and 36.7 seconds, respectively.

At 185.6 seconds a Steam Generator Isolation Signal is generated. After appropriate delays,* the Main Steam Isolation Valves (MSIVs) are fully closed at 198.5 seconds. This causes the pressure in the undamaged steam generator to increase and the pressure in the damaged steam generator to continue to decrease through blowdown through the rupture.

The water level in the undamaged steam generator continues to decrease as a result of boil-off. At about 229.0 seconds the liquid inventory in the undamaged steam generator is sufficiently depleted that there is no heat transfer from primary to secondary. This causes the primary pressure and temperature to increase again. The increase in primary pressure results in the opening of PSVs at 370.0 seconds.

The analysis conservatively assumes that auxiliary feedwater is initiated manually at 600 seconds rather than automatically by the low steam generator level instrumentation much earlier in the event. This auxiliary feedwater reaches the undamaged steam generator at 658 seconds and at a rate of 434 gpm. This auxiliary feedwater slowly reduces the primary heatup and at 779.5 seconds the primary safety valves are fully closed.

The resultant site boundary dose calculated with the assumptions given in Table 7A-2 is:

Thyroid (DEQ I-131) = 2.2 Rem
Whole Body (DEQ Xe-133) < 0.1 Rem

Conclusion

The results of the FLB event with LOAC power on turbine trip shows that the peak pressure does not exceed the pressure upset limit of 2750 psia and that the site boundary doses are within 10CFR100 guidelines.

*Conservative with respect to current Technical Specifications.

TABLE 7A-1

KEY PARAMETERS ASSUMED IN THE FEEDWATER LINE BREAK ANALYSIS

Parameter	Units	Value
Initial Core Power Level	MWt	2754.0
Initial Core Coolant Inlet Temperature	^{o}F	550.0
Initial RCS Vessel Flow Rate	gpm	370,000.0
Initial Reactor Coolant System Pressure	psia	2154.0
Initial Steam Generator Pressure	psia	815.0
Initial Pressurizer Liquid Volume	ft	975.0
Effective Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^{o}F$	+0.2
Doppler Coefficient Multiplier	----	0.85
High Pressurizer Pressure Trip Setpoint	psia	2470.0
Auxiliary Feedwater Actuation	Manual Initiation	600 sec.
Steam Generator Differential Pressure Setpoint	psid	10.0
CEA Worth at Trip	$\%\Delta\rho$	-5.2
Reactor Regulating System	Operating Mode	Manual**
Steam Dump and Bypass System	Operating Mode	Manual**
Pressurizer Pressure Control System	Operating Mode	Manual**
Pressurizer Level Control System	Operating Mode	Manual**

**These modes of control system operation maximize the peak RCS pressure.

TABLE 7A-2

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE FEED LINE BREAK EVENT

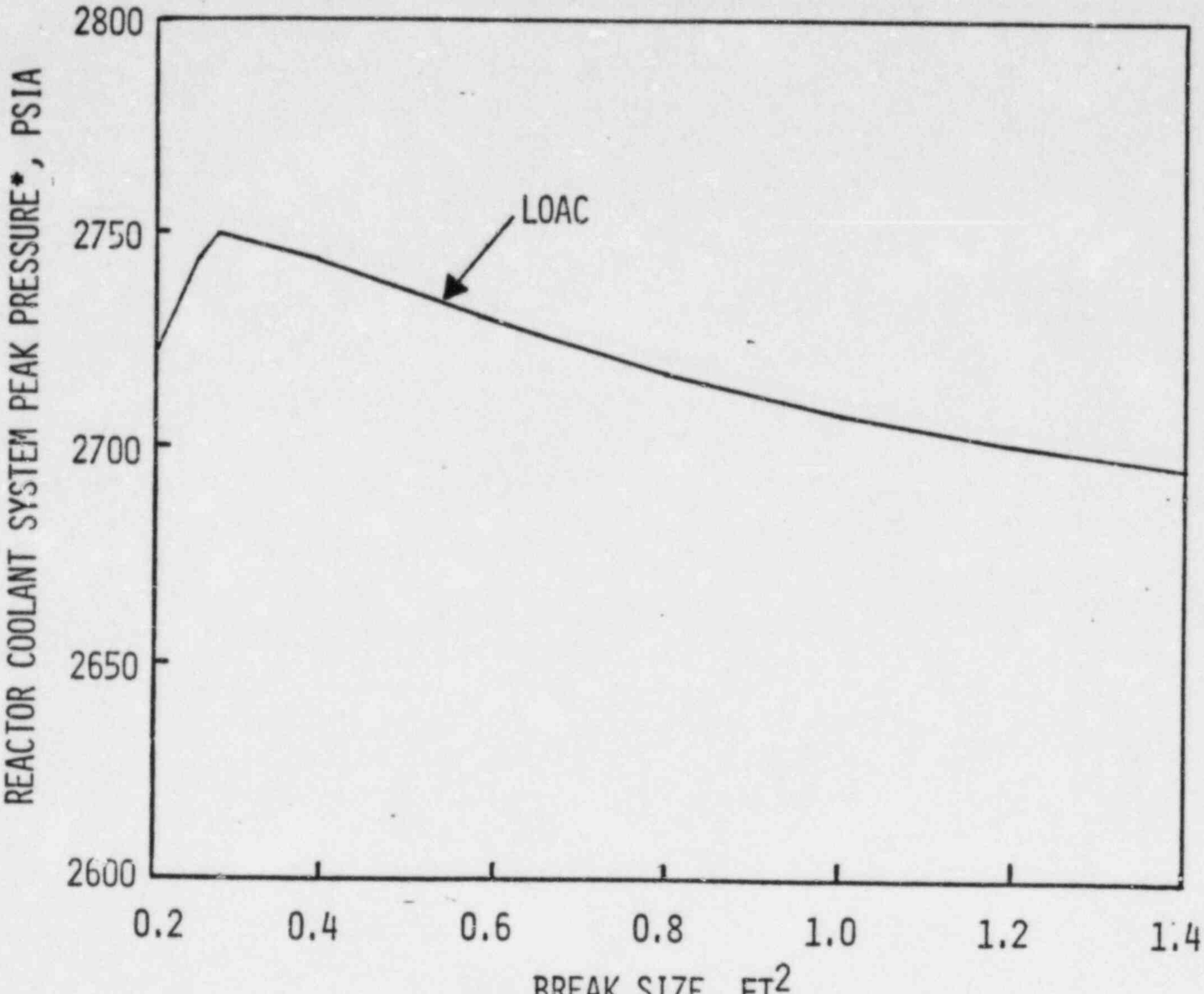
Parameter	Units	Value
Reactor Coolant System Maximum Allowable Concentration (DEQ I-131)[1]	μCi/gm	1.0
Steam Generator Maximum Allowable Concentration (DEQ I-131)[1]	μCi/gm	0.1
Partition Factor Assumed for All Doses	----	1.0
Atmospheric Dispersion Coefficient[2]	sec/M^3	1.80×10^{-4}
Breathing Rate	M^3/sec	3.47×10^{-4}
Dose Conversion Factor (I-131)	REM/Ci	1.48×10^{6}

[1]Tech Spec limits

[2]0-2 hour accident condition

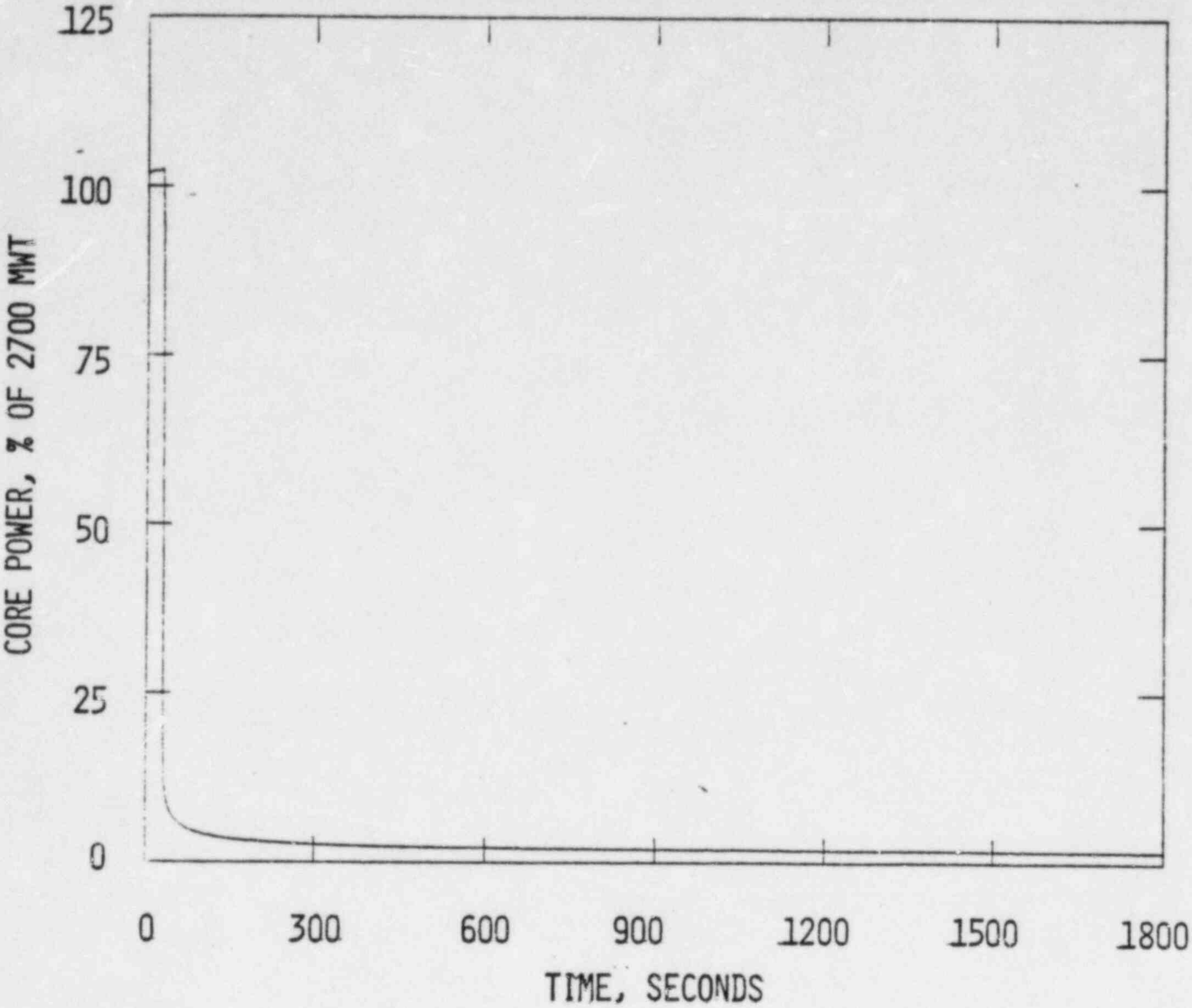
TABLE 7A-3

Time (sec)	Event	Setpoint or Value
0.0	Break in Main Feedwater Line	.275 ft^2
24.5	Heat Transfer Area Rampdown in LHSG Begins	19691 lbm
27.1	High Pressurizer Pressure Trip Setpoint is Reached	2470 psia
27.9	Primary Safety Valves Begin to Open	2525 psia
28.4	CEAs Begin to Enter Core; LOAC on Turbine Trip; RCS Pumps Begin to Coast Down	---
31.1	Peak RCS Pressure	2749 psia
35.5	Undamaged Steam Generator Safety Valves Begin to Open	1000 psia
36.7	Damaged Steam Generator Safety Valves Begin to Open	1000 psia
40.0	Maximum Steam Generator Pressure Undamaged/Damaged	1020.65/1007.46 psia
43.4	Primary Safety Valves are Closed	2424 psia
65.3	Damaged Steam Generator Safety Valves are Closed	960 psia
68.5	Undamaged Steam Generator Safety Valves are Closed	960 psia
185.6	Main Steam Isolation Signal	600 psia
198.5	Main Steam Isolation Valves are Fully Closed	---
370.0	Primary Safety Valves Begin to Open	2525 psia
658.0	Auxiliary Feedwater Flow Established to Undamaged Steam Generator	434 gpm
750.8	Undamaged Steam Generator Safety Valves Begin to Open	1000 psia
779.5	Primary Safety Valves are Fully Closed	2424 psia

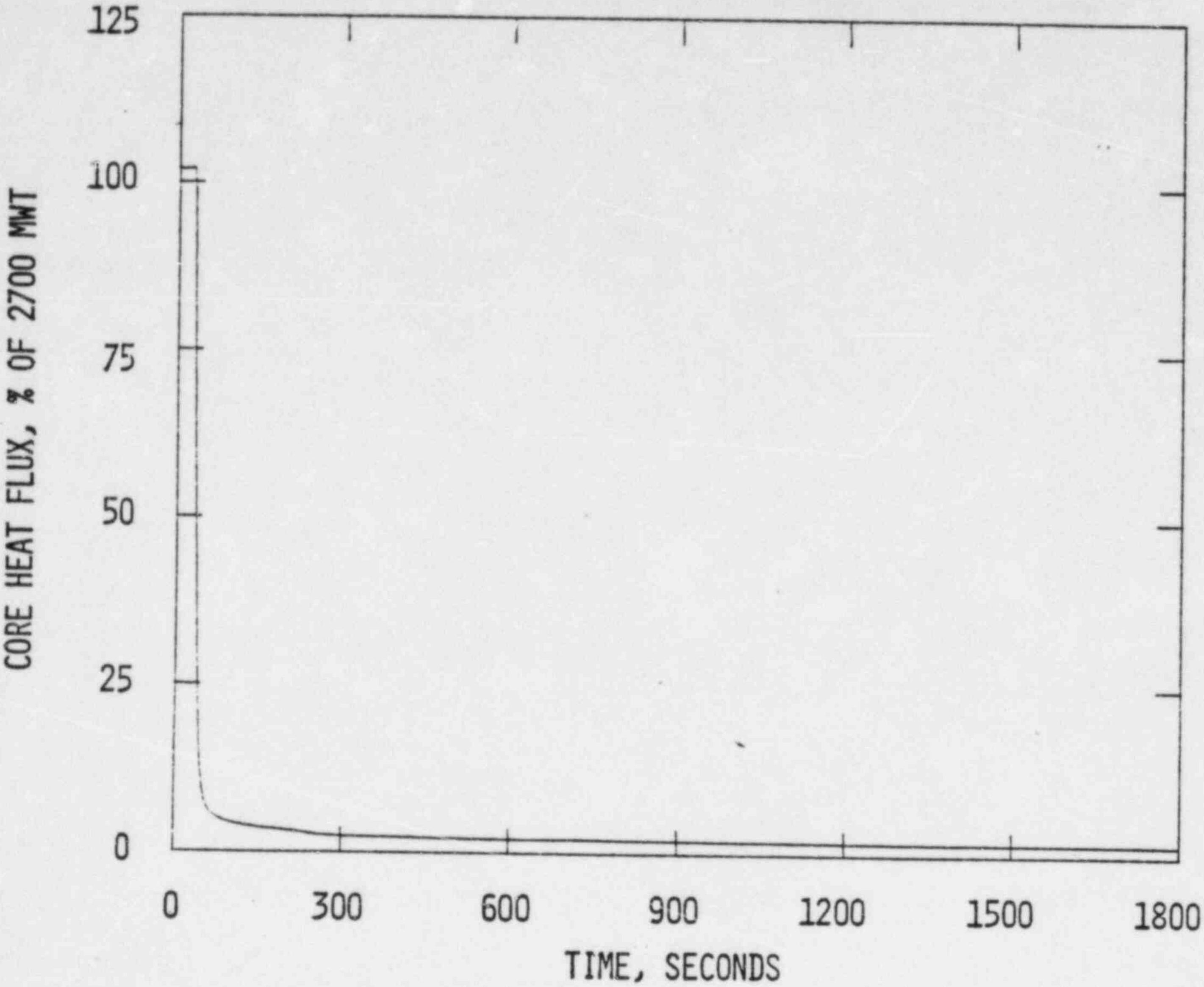


*PRESSURE INCLUDES ELEVATION HEAD

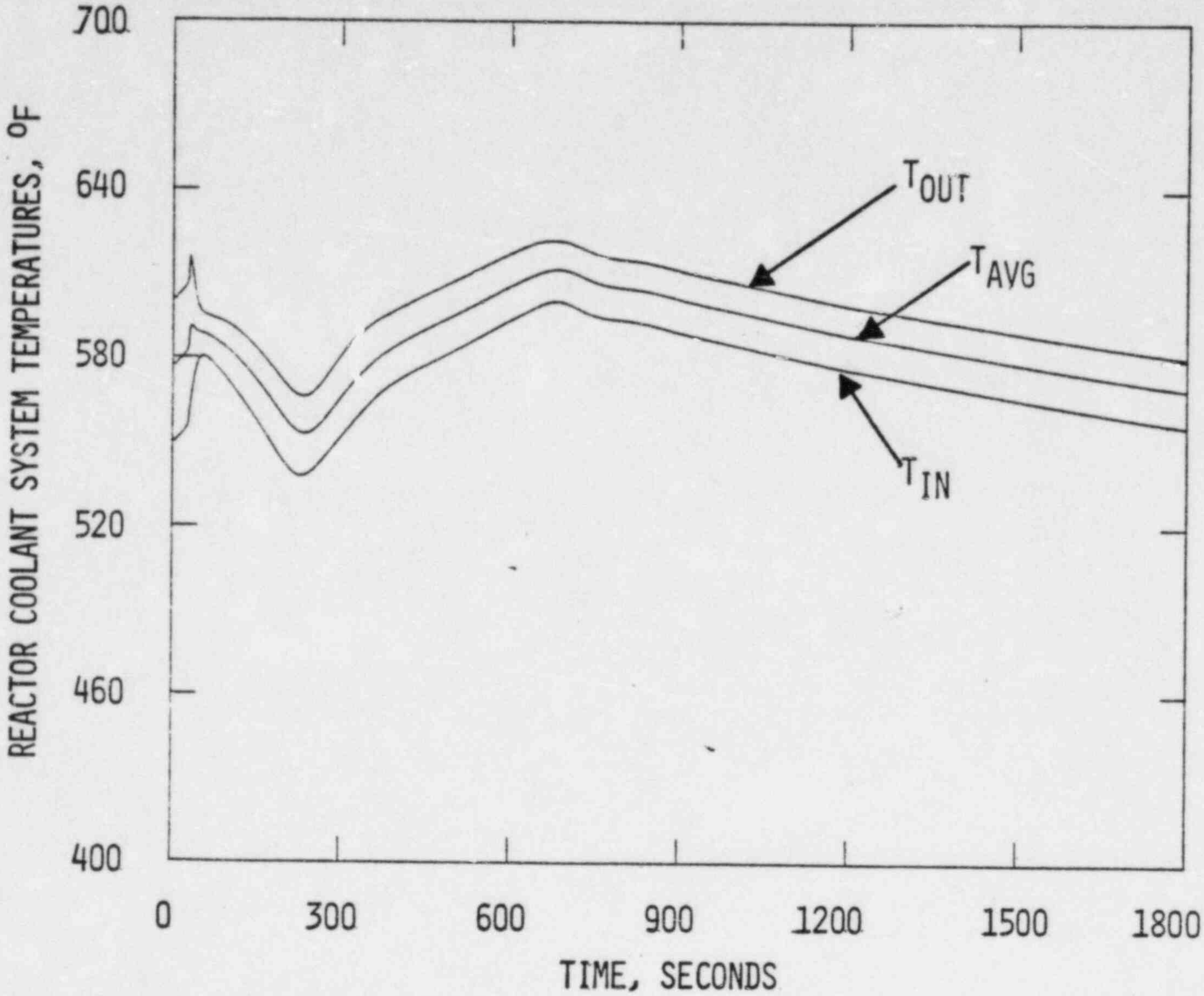
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEED LINE BREAK EVENT REACTOR COOLANT SYSTEM PEAK PRESSURE VS BREAK SIZE	FIGURE 7A-1



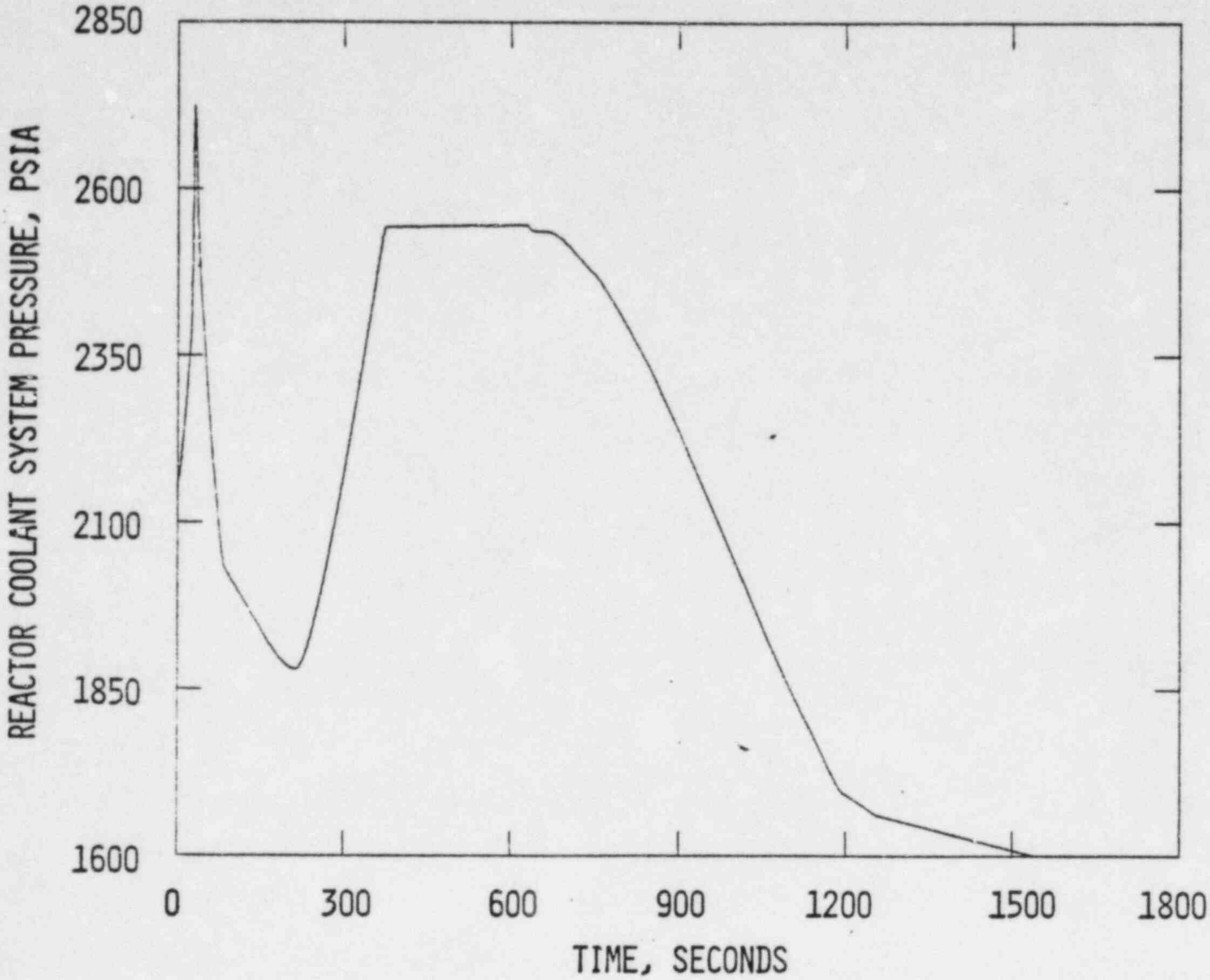
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEEDWATER LINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP CORE POWER VS TIME	FIGURE 7A-2



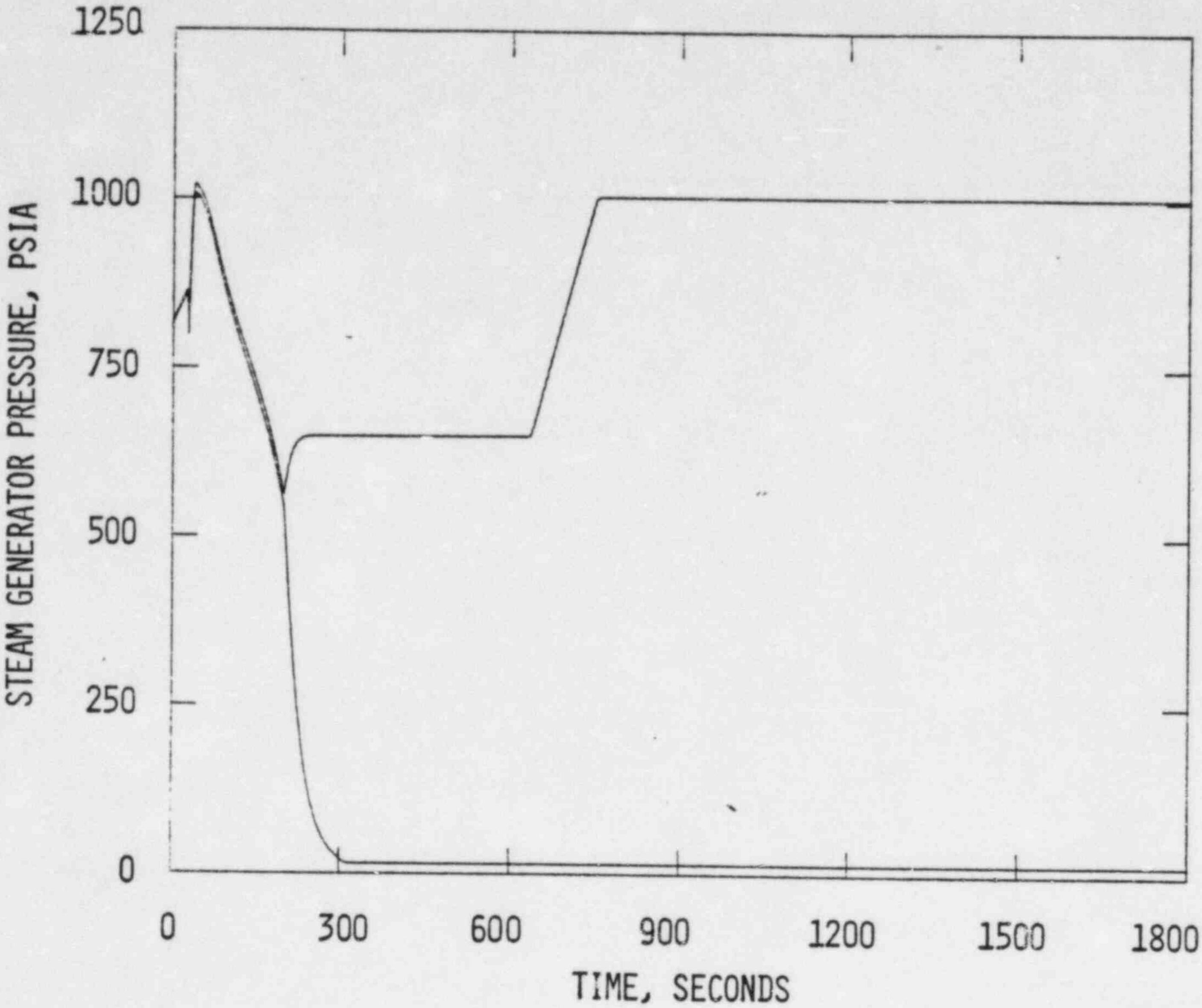
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEEDWATER LINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP CORE HEAT FLUX VS TIME	FIGURE 7A-3



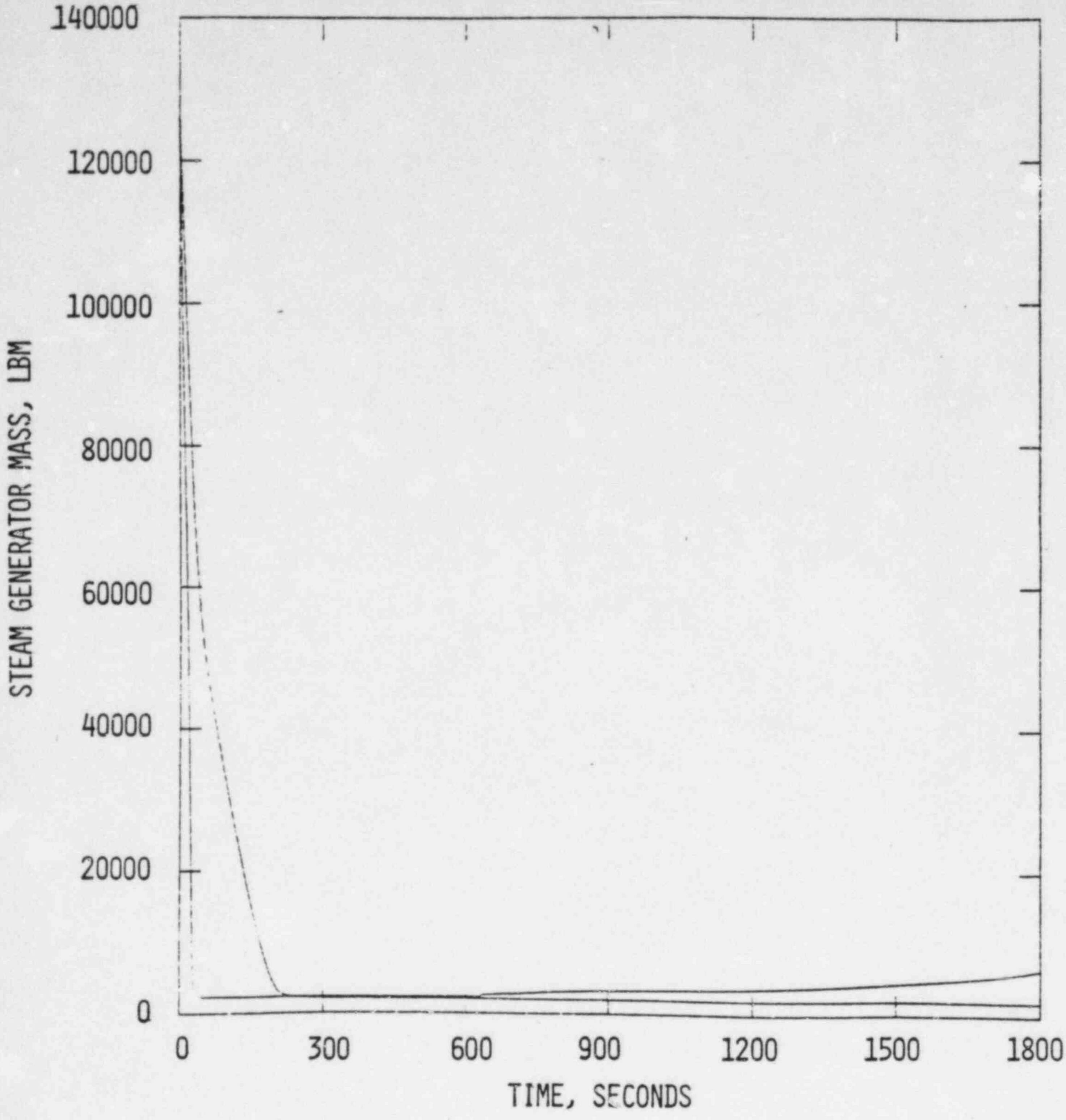
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEEDWATER LINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 7A-4



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEEDWATER LINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 7A-5



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEEDWATER LINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP STEAM GENERATOR PRESSURE VS TIME	FIGURE 7A-6



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FEEDWATER LINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP STEAM GENERATOR MASS VS TIME	FIGURE 7A-7

8.0 ECCS ANALYSIS

8.1 Introduction

An ECCS performance analysis was performed for Calvert Cliffs Unit 1 Cycle 7 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.

8.2 Method of Analysis

The ECCS performance analysis for Calvert Cliffs 1 Cycle 7 consisted of an evaluation of the differences in the fuel rod conditions between Cycle 7 and Cycle 6, the reference cycle analysis. Acceptable ECCS performance was demonstrated for Cycle 6 in Reference 2 and approved by NRC in Reference 3. The blowdown and refill-reflood hydraulic calculations employed in the Cycle 6 evaluation apply to Cycle 7 since the hydraulic parameters of the RCS remain unchanged. Therefore, only the fuel rod operating conditions for Cycle 7 were evaluated. The evaluation was performed using the NRC-approved FATES-3 (Reference 4 and 5) and the NRC-approved STRIKIN-II (Reference 6) computer codes to determine the limiting fuel rod conditions for ECCS performance for Cycle 7 for use in the comparison to the Cycle 6 fuel rod conditions.

8.3 Results

Table 8-1 presents a comparison of the significant parameters for Cycles 6 and 7.

The fuel rod conditions for the limiting case, i.e., maximum initial stored energy, for Cycle 7 are bounded by those of Cycle 6 for the following reasons. First, the initial fuel average stored energy as indicated by the fuel average temperature calculated by STRIKIN-II is greater for Cycle 6 than for Cycle 7 by 18^{o}F. Secondly, the hot assembly average channel PLHGR for Cycle 7 is less than that of Cycle 6. Also, the limiting hot rod radiation heat transfer enclosure for Cycle 7 was found to be less severe than that for Cycle 6.

The fuel rod conditions at extended burnup for Cycle 7 are bounded by those of Cycle 6. The hot rod gas pressure at extended burnup for Cycle 7 is 76 psia lower than the corresponding pressure for Cycle 6 as shown in Table 8-1. The initial fuel average temperature at extended burnup for Cycle 7 is 66^{o}F lower than at the limiting (maximum stored energy) burnup and is nearly identical (2^{o}F higher) to the value at extended burnup in Cycle 6. STRIKIN-II transient calculations confirmed that the 2^{o}F difference is more than compensated for by the less severe radiation enclosure and lower hot assembly average channel PLHGR for Cycle 7.

For these reasons it is concluded that the peak clad temperature and oxidation percentages for Cycle 6 conservatively apply to Cycle 7.

8.4 Conclusions

For the reasons presented in Section 8.3, the results of the Calvert Cliffs Unit 1 Cycle 6 ECCS performance analysis conservatively apply to Cycle 7. In the Cycle 6 analysis the peak clad temperature was calculated to be 2038°F as compared to the acceptance criteria limit of 2200°F. The peak local and core wide clad oxidation percentages were calculated to be 8.5% and $\leq$ 0.51%, respectively, as compared to the acceptance criteria limits of 17% and 1%, respectively. Therefore, operation at a PLHGR of 15.5 kw/ft and a power level of 2754 MWt (102% of 2700 MWt) will result in acceptable ECCS performance for Calvert Cliffs Unit 1 Cycle 7.

TABLE 8-1

CALVERT CLIFFS UNIT 1 CYCLE 7 ECCS ANALYSIS
COMPARISON OF SIGNIFICANT PARAMETERS WITH CYCLE 6

Parameter	Cycle 6	Cycle 7
Power Level (102% of Nominal), MWt	2754	2754
Peak Linear Heat Generation Rate, Hot Assembly, Hot Channel, kw/ft	15.5	15.5
Peak Linear Heat Generation Rate, Hot Assembly, Average Channel, kw/ft	13.14	12.52
Limiting Burnup Case Fuel Conditions		
Fuel Average Temperature at PLHGR, ^{o}F*	2213	2195
Fuel Centerline Temperature at PLHGR, ^{o}F*	3634	3578
Gap Conductance at PLHGR, BTU/hr-ft^2, ^{o}F*	2025	1928
Hot Rod Gas Pressure, psia*	1251	1198
Hot Rod Burnup, MWD/MTU	3000	1000
Extended Burnup Case Fuel Conditions		
Fuel Average Temperature at PLHGR, ^{o}F*	2127	2129
Fuel Centerline Temperature at PLHGR, ^{o}F*	3551	3566
Gap Conductance at PLHGR, BTU/hr-ft^2, ^{o}F*	2470	2441
Hot Rod Gas Pressure, psia*	2191	2115
Hot Rod Burnup, MWD/MTU	34000	52500**

*Values are those at the indicated hot rod burnup as calculated by STRIKIN-II at 15.5 kw/ft.

**The limiting conditions for extended burnup have changed to the end of cycle maximum burnup due to the NRC mandated grain size restriction imposed on the FATES3 code (Reference 5).

9.0 TECHNICAL SPECIFICATIONS

The Technical Specification changes which must be made in order to make the Calvert Cliffs Unit 1 Technical Specifications valid for the operation of Cycle 7 are presented in this section. Table 9-1 presents a summary of the Technical Specification changes. Table 9-2 presents the explanations for the changes summarized in Table 9-1.

The requested Technical Specification modifications for Unit 1 Cycle 7 (Table 9-1) are very similar to those changes requested for the reference cycle (Unit 2 Cycle 5, References 1 and 2). There are two significant differences compared to the requested changes for Unit 2 Cycle 5 both of which are supported by the generic Steam Line Rupture analysis presented in Section 7. These differences are:

1. The shutdown margin is being lowered to 4.3% $\Delta k/k$ to reduce operating requirements with regard to shutdown boron levels. This change is consistent with the generic SLB analysis presented herein (see Table 7-2) and existing safety analyses (see note to Table 5-2).

2. The moderator temperature coefficient negative limit is being increased to accommodate the effects of extended burnup.

Following Table 9-2, for each Technical Specification which must be modified, either:

1. the existing page with the intended modification,
2. the already modified page with a new figure, or
3. a sample page (area to be modified or added identified in enclosed area with "*")

is provided.