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ARTHUR E. LUNDVALL, JR.
VICE PRESIDENT
SUPPLY

November 19, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

ATTENTION: Mr. R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Subject: Calvert Cliffs Nuclear Power Plant
Units No. 1 and 2
Docket Nos. 50-317 and 50-318
Phase I Cycle 6 Reload Application for
Amendment to Operating License

Reference (A): A. E. Lundvall, Jr. to R. A. Clark letter, dated 9/22/81, Fifth Cycle License Application

Gentlemen:

At an October 15, 1981 meeting with NRC staff in Bethesda we agreed to make an early submittal of a portion of the Cycle 6 reload application. That portion is titled 'Phase I' and is attached hereto. Specifically, Phase I consists of the following sections of a standard reload application:

-- Chapter 6.0 Thermal Hydraulics Design

-- Chapter 7.0 Transient Analyses

* 7.1.4 Excess Load Event

* 7.1.5 Loss of Load Event

* 7.2.3 Full Length CEA Drop Event

* 7.2.4 AOO's Resulting from the Malfunction of One Steam Generator

-- Chapter 9.0 Technical Specifications

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* (Nine (9) of the anticipated total of twelve (12) modifications to Technical Specifications are included in Phase I.)

The sections of Phase I are not significantly different from those submitted in Reference (A). Phase II of the Cycle 6 application will include Phase I as well as the rest of the sections which constitute a standard reload application. Phase II will be submitted on or about February 15, 1982.

Very truly yours,

BALTIMORE GAS AND ELECTRIC COMPANY

A. E. Lundvall, Jr.
Vice President-Supply

Attachment

Copies to: J. A. Biddison, Esquire (w/o encl.)
G. F. Trowbridge, Esquire (w/o encl.)
Mr. D. H. Jaffe - NRC
Mr. P. W. Kruse - CE

ATTACHMENT

PHASE I

6.0 THERMAL HYDRAULIC DESIGN

6.0 THERMAL HYDRAULIC DESIGN

6.1 DNBR Analysis

Steady state DNBR analyses of Cycle 6 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; a similar discussion of CETOP methodology was submitted on the Arkansas Nuclear One Unit 2 (ANO-2) docket in Reference 5. CETOP was approved for use on ANO-2 in Reference 6. In general, this code differs from earlier versions of TORC only in that enthalpy transport coefficients are used to improve modeling of coolant conditions in the vicinity of the hot subchannel and in that more rapid equation-solving routines are used. Direct comparisons show that CETOP models tend to be slightly more conservative than TORC design models in computing minimum DNBR for limiting cases. (Note that application of the methods of Reference 3 assures that design models set up with either TORC or CETOP are always conservative relative to detailed TORC analyses.) CETOP is used only because it reduces computer costs significantly; no margin gain is expected or taken credit for.

Table 6-1 contains a list of pertinent thermal-hydraulic design parameters used for both safety analyses and for generating reactor protective system setpoint information. Also note that 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 7) to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 7.

Investigations have been made to ascertain the effect of the CEA guide tube wear problem and the sleeving repair on DNBR margins as established by this type of analysis. The findings were reported to the NRC in Reference 8 which concluded that the wear problem and the sleeving repair do not adversely affect DNBR margin.

6.2 Effects of Fuel Rod Bowing on DNBR Margin

The fuel rod bowing effects on DNB margin for Calvert Cliffs-1 Cycle 6 have been evaluated according to the guidelines set forth in Reference 9.

A total of 137 fuel assemblies will exceed the NRC-specified DNB penalty threshold burnup of 24,000 MWD/T during Cycle 6, as established by the guidelines in Reference 9. At the end of Cycle 6, the maximum burnup attained by any of these assemblies will be 42,800 MWD/T. Based upon an extrapolation of the formula contained in Reference 6, the corresponding DNB penalty for 42,800 MWD/T has been determined to be 6.3 percent.

An examination of power distributions for Cycle 6 shows that DNB margin exists for assemblies exceeding 24,000 MWD/T relative to the DNB limits estaolished by other assemblies in the core. This margin is greater than the Reference 9 reduction penalty of 6.3 percent imposed upon fuel asemblies exceeding 24,000 MWD/T in Cycle 6. Therefore, no power penalty for fuel rod bowing is required in Cycle 6.

TABLE 6-1

Calvert Cliffs Unit 1

Thermal-Hydraulic Parameters at Full Power

General Characteristics	Unit	Reference Cycle 5*	Cycle 6**
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
Minimum in steady state	psia	2200	
Maximum in steady state	psia	2300	
Inlet Temperature	°F	550	548
Total Reactor Coolant Flow (steady state)	gpm 10^6 lb/hr	370,000 139.0	381,600 143.8
Coolant Flow Through Core	10^6 lb/hr	133.9	138.5
Hydraulic Diameter (nominal channel)	ft	0.044	0.044
Average Mass Velocity	10^6 lb/hr-ft^2	2.51	2.61
Pressure Drop Across Core (minimum steady state flow irreversible Δp over entire fuel assembly)	psi	10.4	11.1
Total Pressure Drop Across Vessel (based on nominal dimensions and minimum steady state flow)	psi	32.4	34.4
Core Average Heat Flux (accounts for above fraction of heat generated in fuel rod and axial densification factor)	Btu/hr-ft^2	186,435***	186,435***
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	48,192***	48,192***
Film Coefficient at Average Conditions	Btu/hr-ft^2-°F	5765	5930
Average Film Temperature Difference	°F	32	31
Average Linear Heat Rate of Undensified Fuel Rod (accounts for above fraction of heat generated in fuel rod)	kw/ft	6.23***	6.23***
Average Core Enthalpy Rise	Btu/lb	68.8	66.5
Maximum Clad Surface Temperature	°F	657	657

TABLE 6-1 (cont'd)

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

NOTES

*Design inlet temperature and nominal primary system pressure were used to calculate these parameters.

**Due to the statistical combination of uncertainties described in References 7, 10 and 11, the nominal inlet temperature and nominal primary system pressure were used to calculate some of these parameters.

***Based on a generic value of 1100 shims.

****These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level (Reference 7) to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 7.

References (Chapter 6)

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

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

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

4. Letter, A. E. Lundvall, Jr. to R. A. Clark, "Response to Questions on SCU, CEN-124(B)," June 2, 1981

5. Letter, D. C. Trimble (AP&L) to Director, NRR, "CETOP-D Code Structure and Modeling Methods, Response to First Round Questions on the Statistical Combination of Uncertainties Program (CEN-139(A)-P)", July 15, 1981

6. Final Safety Evaluation Report Supporting Facility Operating License Amendment No. 26 on Docket No. 50-368 and Operation of ANO-2 During Cycle 2, July 21, 1981

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

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

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

10. CEN-124(B)-P, "Statistical Combination of Uncertainties, Part 1," January 1980

11. CEN-124(B)-P, "Statistical Combination of Uncertainties, Part 3," March 1980

7.0 DESIGN BASIS EVENTS

7.1.4 EXCESS LOAD EVENT

7.1.5 LOSS OF LOAD EVENT

7.2.3 FULL LENGTH CEA DROP EVENT

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

7.1.4 EXCESS LOAD EVENT

The Excess Load Event was reanalyzed to determine that the DNBR and CTM design limits are not exceeded during Cycle 6.

The analyses included the effects of manually tripping the RCP's on SIAS due to low pressurizer pressure and the automatic initiation of auxiliary feedwater flow on low steam generator level trip signal.

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

The most limiting load increase events at full power and at hot standby conditions, for approach to the DNBR limit of 1.23 (CE-1), are due to the complete opening of the steam dump and bypass valves.

For conservatism in the analyses, auxiliary feedwater flow rate corresponding to 21% of full power main feedwater flow was assumed (i.e., 10.5% of full power main feedwater flow per generator). Also, the addition of the auxiliary feedwater to each steam generator was conservatively assumed to occur 180 seconds after reactor trip. The addition of the auxiliary feedwater flow to both steam generators results in an additional cooldown of the RCS and a potential for a return-to-power (R-T-P) or criticality arising from reactivity feedback mechanisms.

The Excess Load event at full power was initiated at the conditions given in Table 7.1.4-1. A Moderator Temperature Coefficient of $-2.5\times10^{-4}\Delta\rho$/F was assumed in this analysis. This MTC, in conjunction with the decreasing coolant inlet temperature, enhances the rate of increase of heat flux at the time of reactor trip. A Fuel Temperature Coefficient (FTC) corresponding to beginning of cycle conditions with an uncertainty of 15% was used in the analysis since this FTC causes the least amount of negative reactivity change for mitigating the transient increase in core heat flux. The minimum CEA worth assumed to be available for shutdown at the time of reactor trip for full power operation is 4.3%$\Delta\rho$. The analysis conservatively assumes that the worth of boron injected from the safety injection tank is -1.00%$\Delta\rho$ per 105 PPM. The pressurizer pressure control system was assumed to be inoperable because this minimizes the RCS pressure during the event and therefore reduces the calculated DNBR. All other control systems were assumed to be in manual mode of operation and have no impact on the results of this event.

The Full Power Excess Load event results in a High Power trip at 7.2 seconds. The minimum DNBR calculated for the event at the conditions specified in Table 7.1.4-1 is 1.48 compared to the design limit of 1.23. The maximum local linear heat generation rate for the event is 18.4 KW/ft.

For the Excess Load event initiated from HFP conditions, SIAS is generated at 34.3 seconds at which t ·· the RCP's are manually tripped by the operator. The coastdown of the pumps decreases the rate of decay heat removal and therefore keeps the RCS coolant temperatures and pressure at higher values.

Auxiliary feedwater flow is delivered to both steam generators at 187.2 seconds. The feedwater flow causes additional cooldown of the RCS. The decreasing temperatures in combination with a negative MTC inserts positive reactivity which enables the core to approach criticality. The negative reactivity inserted due to the CEAs and Boron injected via the High Pressure Safety Injection (HPSI) pumps however is sufficient to maintain the core subcritical at all times.

Table 7.1.4-2 presents the sequence of events for an Excess Load event initiated at HFP conditions. Figures 7.1.4-1 to 7.1.4-5 show the NSSS response for power, heat flux, RCS temperatures, RCS pressure, and steam generator pressure during this event.

The Zero Power Excess Load event was initiated at the conditions given in Table 7.1.4-3. The MTC and FTC values assumed in the analysis are the same as for the full power case for the reasons previously given. The minimum CEA shutdown worth available is conservatively assumed to be -4.0%Δρ.

The results of the analysis show that a variable high power trip occurs at 35.9 seconds. The minimum DNBR calculated during the event is 2.92 and the peak linear heat generation rate is 14.4 KW/ft.

As with the HFP Excess Load event, an SIAS signal on low pressurizer pressure is generated at 76.6 seconds for the zero power excess load event. At 215.9 seconds auxiliary feedwater flow is delivered to both steam generators. The additional positive reactivity due to the cooldown of the RCS is mitigated by the negative reactivity inserted due to CEA's and the boron injected via the HPSI pumps. The core remains subcritical at all times during an Excess Load event initiated from HZP conditions.

The sequence of events for the zero power case is presented in Table 7.1.4-4. Figures 7.1.4-6 to 7.1.4-10 show the NSSS response for core power, core heat flux, RCS temperature, RCS pressure and steam generator pressure.

For the full and zero power Excess Load events initiated by a full opening of the steam dump and bypass valves the DNBR and CTM limits are not exceeded. In addition the core remains subcritical even after automatic initiation of the auxiliary feedwater flow and following manual trip of the RCP's on SIAS due to low pressurizer pressure. The reactivity transient during a HFP and HZP Excess Load event is less limiting than the corresponding Steam Line Rupture events (See Section 7.3.2).

TABLE 7.1.4-1

KEY PARAMETERS ASSUMED FOR FULL POWER EXCESS LOAD EVENT ANALYSIS

Parameter	Units	Reference Cycle	Cycle 6
Initial Core Power Level	MWt	2754	2700 +
Core Inlet Temperature	°F	550	548 +
Reactor Coolant System Pressure	psia	2200	2225 +
Core Mass Flow Rate	$X10^{6}$ lbm/hr	133.9	138.5 +
Moderator Temperature Coefficient	$X10^{-4}\Delta\rho/°F$	-2.5	-2.5
CEA Worth Available at Trip	%Δρ	-4.3	-4.3
Doppler Multiplier		.85	.85
Inverse Boron Worth	PPM/%Δρ	105	105
Auxiliary Feedwater Flow Rate	lbm/sec	175.0/S.G.	175.0/S. G.
High Power Level Trip Setpoint	% of Full Power	112	110
Low S. G. Water Level Trip Setpoint	ft.	30.9	30.9

Reference cycle is Cycle 5, Reference 2.

+For DNBR calculations, effects of uncertainties on these parameters were combined statistically (see Reference 1)

TABLE 7.1.4-2

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

Time(sec)	Event	Setpoint or Value
0.0	Complete Opening of Steam Dump and Bypass Valves at Full Power	--
7.2	High Power Trip Signal Generated	110% of full power
7.6	Trip Breakers Open	--
8.1	CEA's Begin to Drop Into Core	--
8.6	Maximum Power;	113.2% of full power
	Maximum Local Linear Heat Rate Occurs, KW/ft	18.4
9.0	Minimum DNBR Occurs	1.48
10.6	Low Steam Generator Level Trip Setpoint Reached	30.9 ft
34.1	Pressurizer Empties	--
34.3	Safety Injection Actuation Signal Initiated; Manual Trip of RCP's	1578 psia
52.5	Main Steam Isolation Signal	548 psia
68.1	Rampdown of Main Feedwater Flow Completed	5% of full power main feedwater flow
96.5	Pressurizer Begins to Refill	--
132.5	Isolation of Main Feedwater Flow to Both Steam Generators	--
187.2	Auxiliary Feedwater Flow Delivered to Both Steam Generators	175.0 lbm/sec to each steam generator
600.0	Operator Terminates Auxiliary Feedwater Flow to Both Steam Generators	--

TABLE 7.1.4-3

KEY PARAMETERS ASSUMED FOR HOT STANDBY EXCESS LOAD EVENT ANALYSIS

Parameter	Units	Reference* Cycle	Cycle 6
Initial Core Power Level	MWt	1	1+
Core Inlet Temperature	°F	532	532+
Reactor Coolant System Pressure	psia	2200	2225+
Core Mass Flow Rate	$X10^6$ lbm/hr	137.1	141.35+
Moderator Temperature Coefficient	$X10^{-4} \Delta\rho/°F$	-2.5	-2.5
CEA Worth Available at Trip	$\%\Delta\rho$	-4.0	-4.0
Doppler Multiplier		.85	.85
Inverse Boron Worth	$PPM/\%\Delta\rho$	100	100
Variable High Power Trip Setpoint	% of full power	40	40
Low S. G. Water Level Trip Setpoint	ft.	30.9	3[illegible].9
Auxiliary Feedwater Flow Rate	lbm/sec	175.0/S.G.	175.0/S. G.

* Reference Cycle is Cycle 5 in Reference 2.

\+ For DNBR calculations, effects of uncertainties on these parameters were combined statistically (see Reference 1).

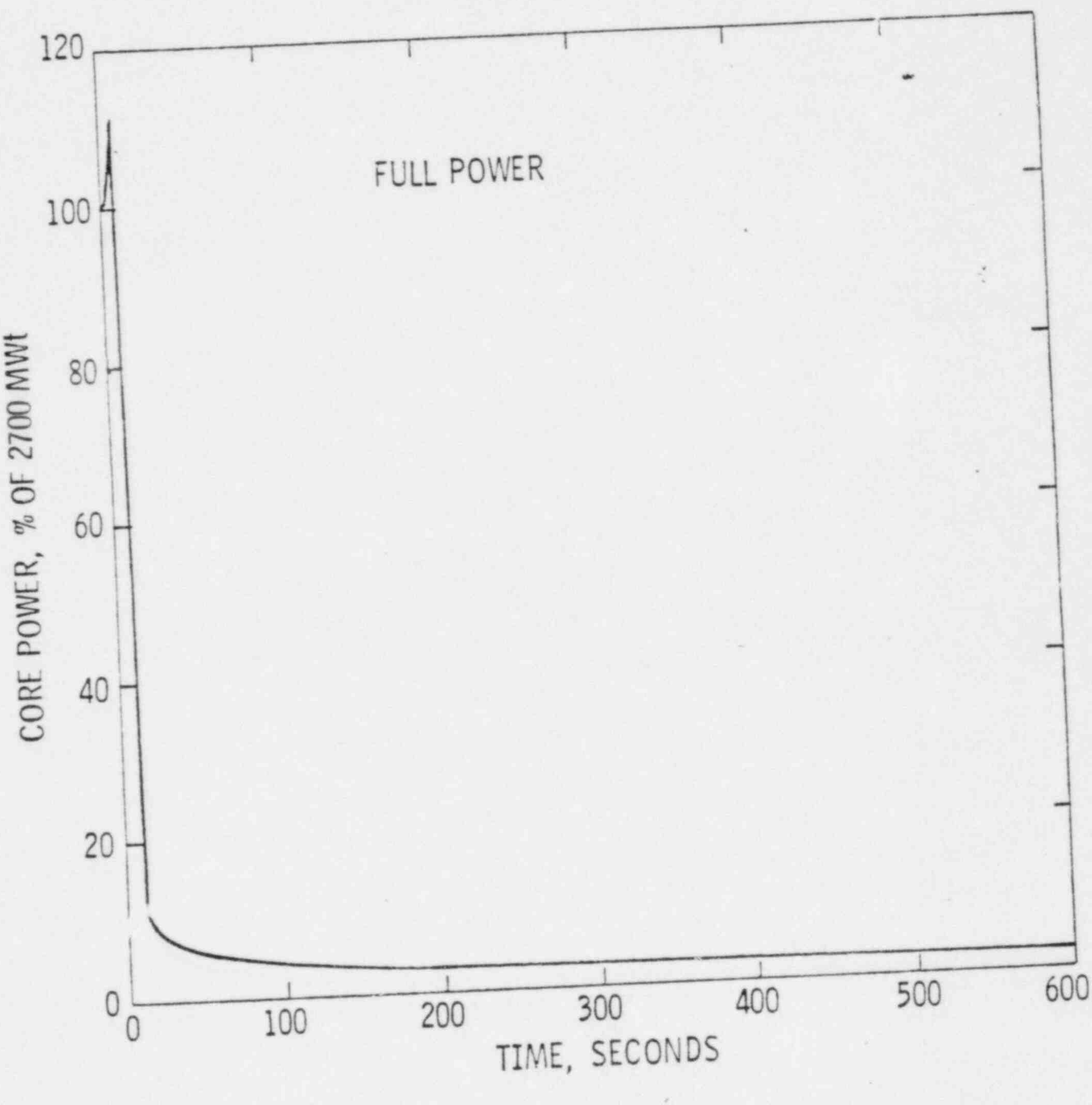
TABLE 7.1.4-4

SEQUENCE OF EVENTS FOR EXCESS LOAD EVENT AT HOT STANDBY CONDITIONS TO CALCULATE MINIMUM DNBR

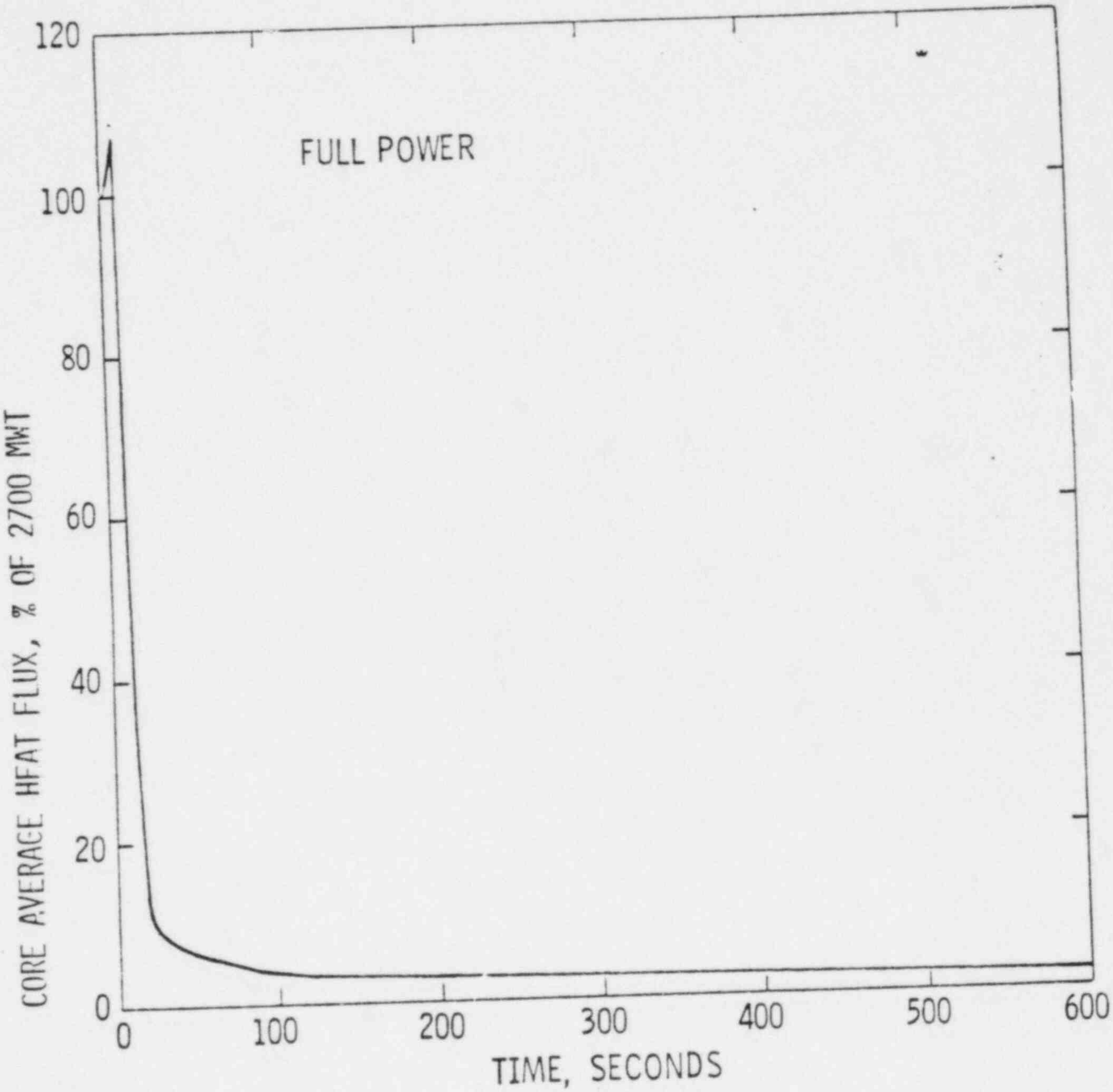
Time(sec)	Event	Setpoint or Value
0.0	Steam Dump and Bypass Valves Open to Maximum Flow Capacity	--
35.9	Variable High Power Trip Signal Generated	40% of full power
36.3	Trip Breakers Open	--
36.9	Core Power Reaches Maximum	40.4% of full
37.6	Minimum DNBR (CE-1)	2.92
72.3	Pressurizer Empties	--
76.6	Safety Injection Actuation Signal Generated; Manual Trip of RCS Coolant Pumps	1578 psia --
82.6	Main Steam Isolation Signal Generated	548 psia
88.7	Low Steam Generator Water Level Trip Setpoint Reached	30.9 ft
106.8	Pressurizer Begins to Refill	--
162.6	Isolation of Main Feedwater Flow to Both Steam Generators	--
215.9	Auxiliary Feedwater Flow Delivered to Both Steam Generators	175.0 lbm/sec to each steam generator

TABLE 7.1 4-4 (CONTINUED)

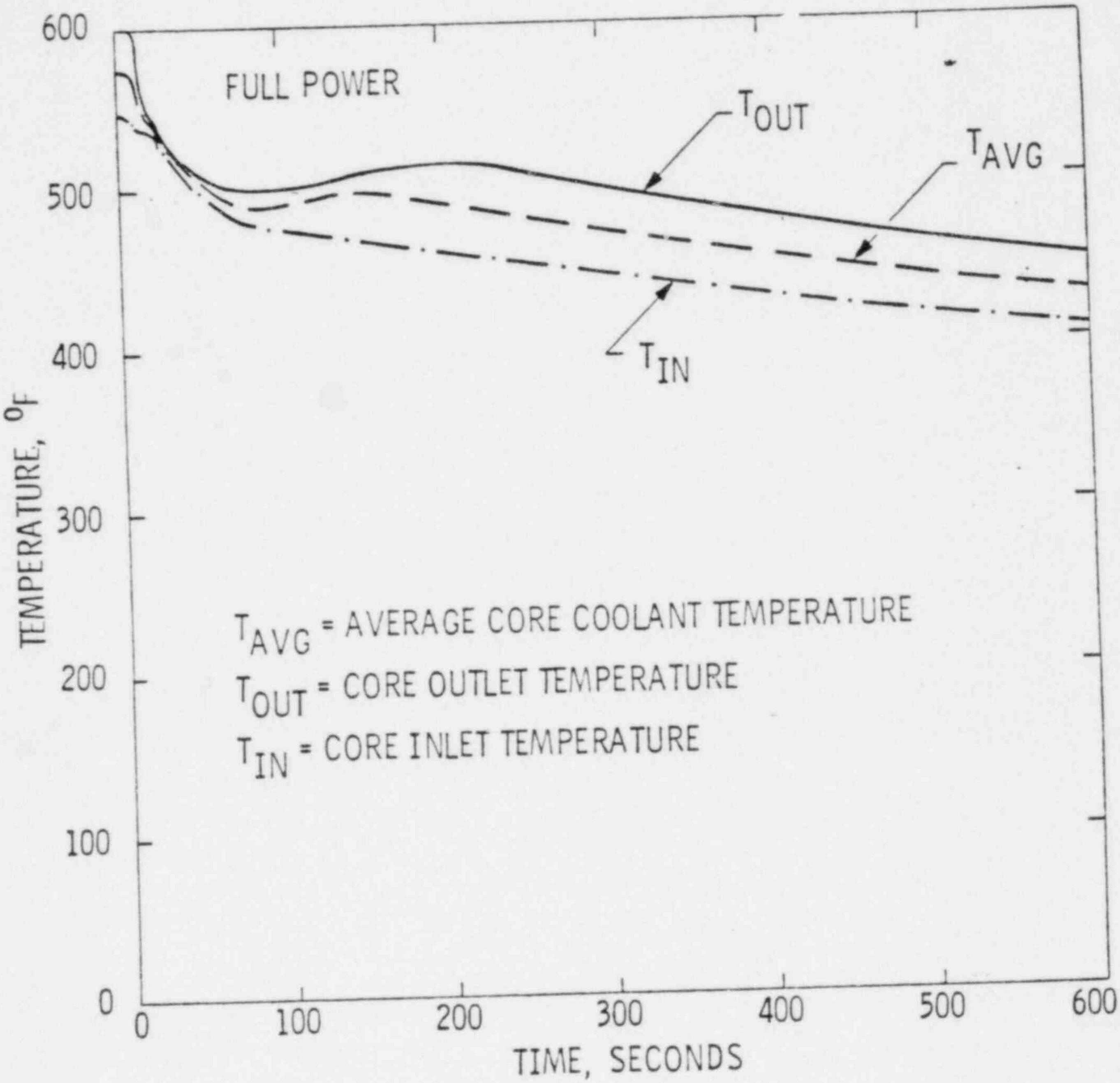
Time(sec)	Event	Setpoint or Value
600.0	Operator Terminates Auxiliary Feedwater Flow to Both Steam Generators	--



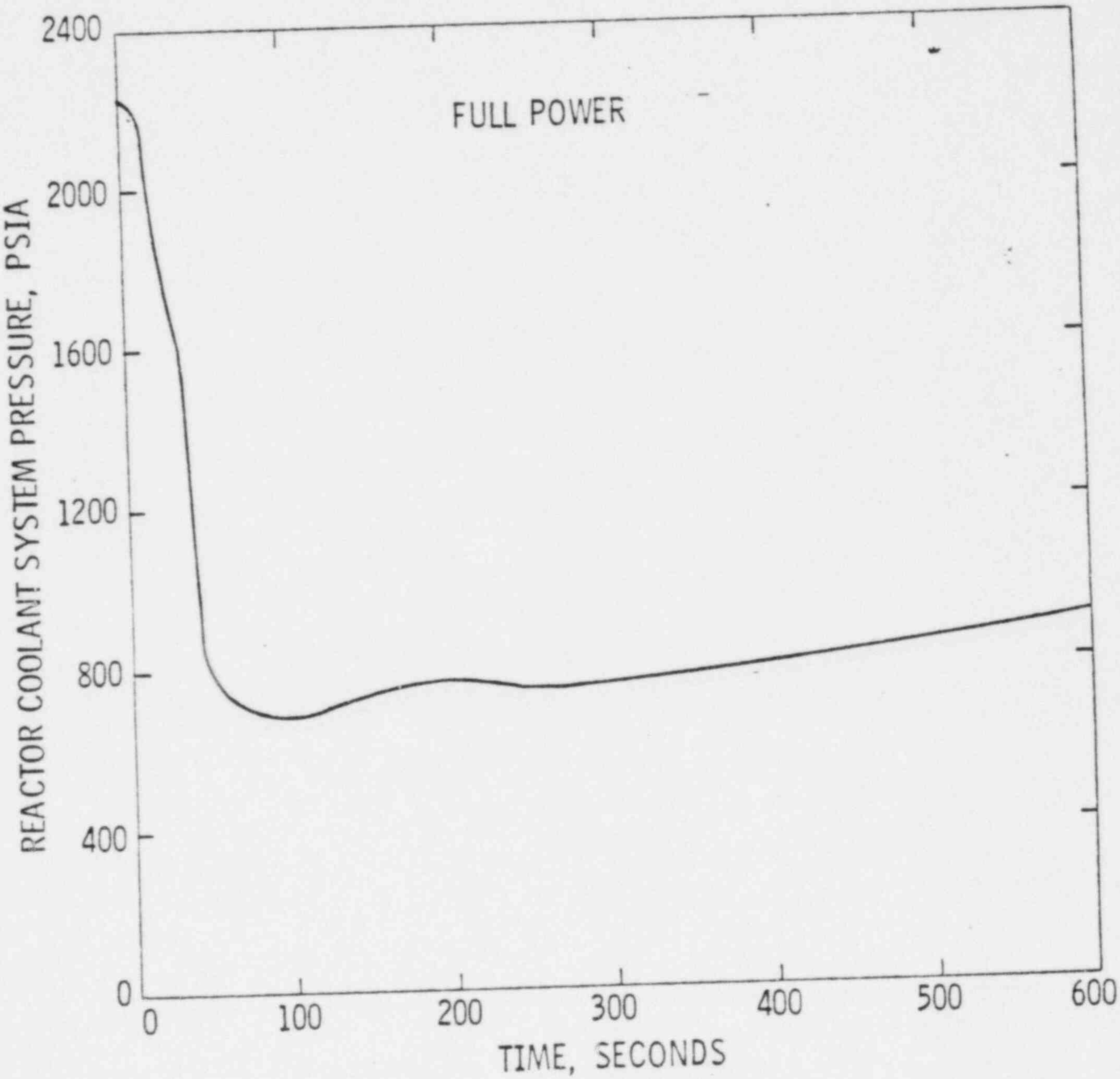
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT CORE POWER vs TIME	Figure 7.1.4-1



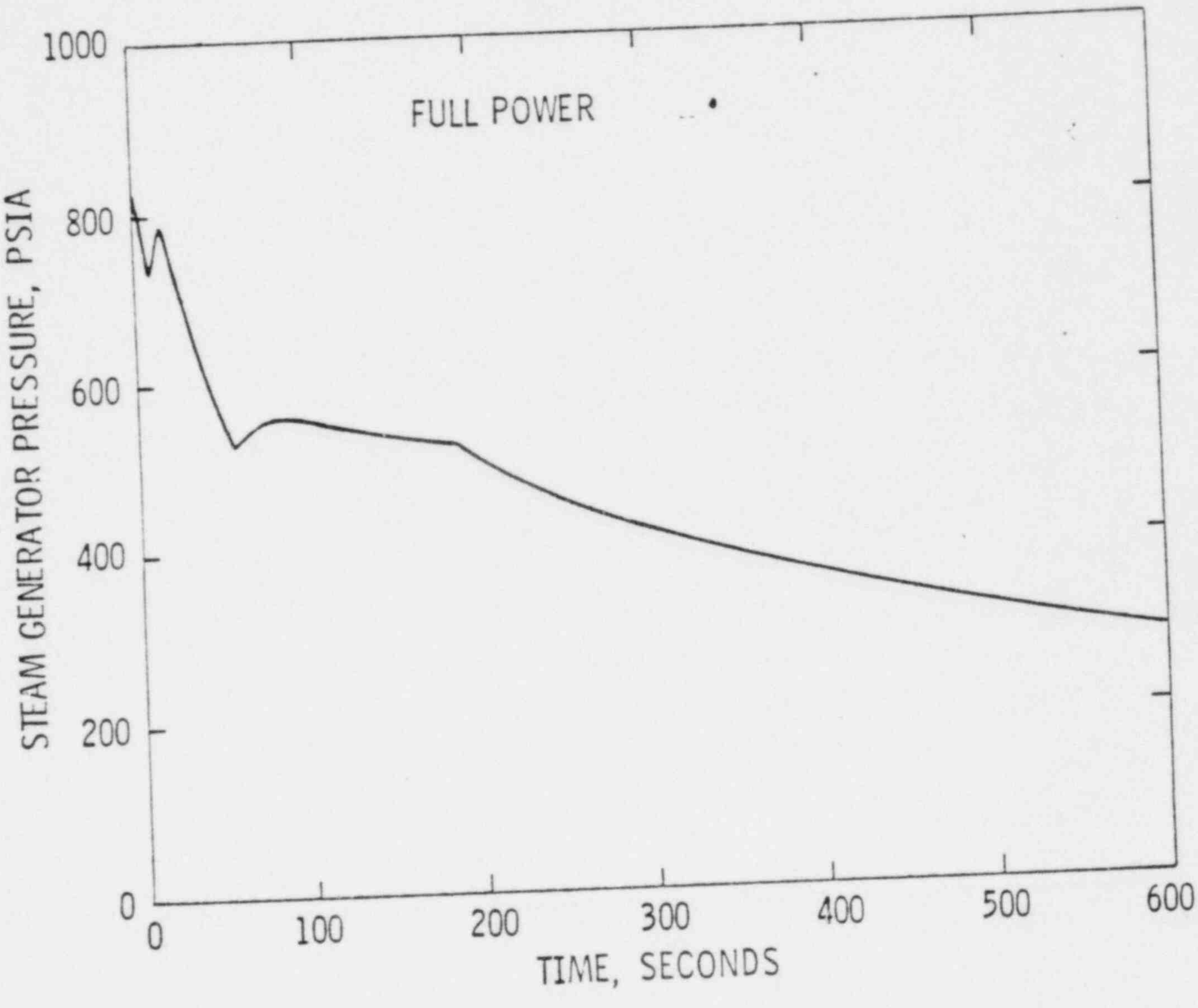
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT HEAT FLUX vs TIME	Figure 7.1.4-2



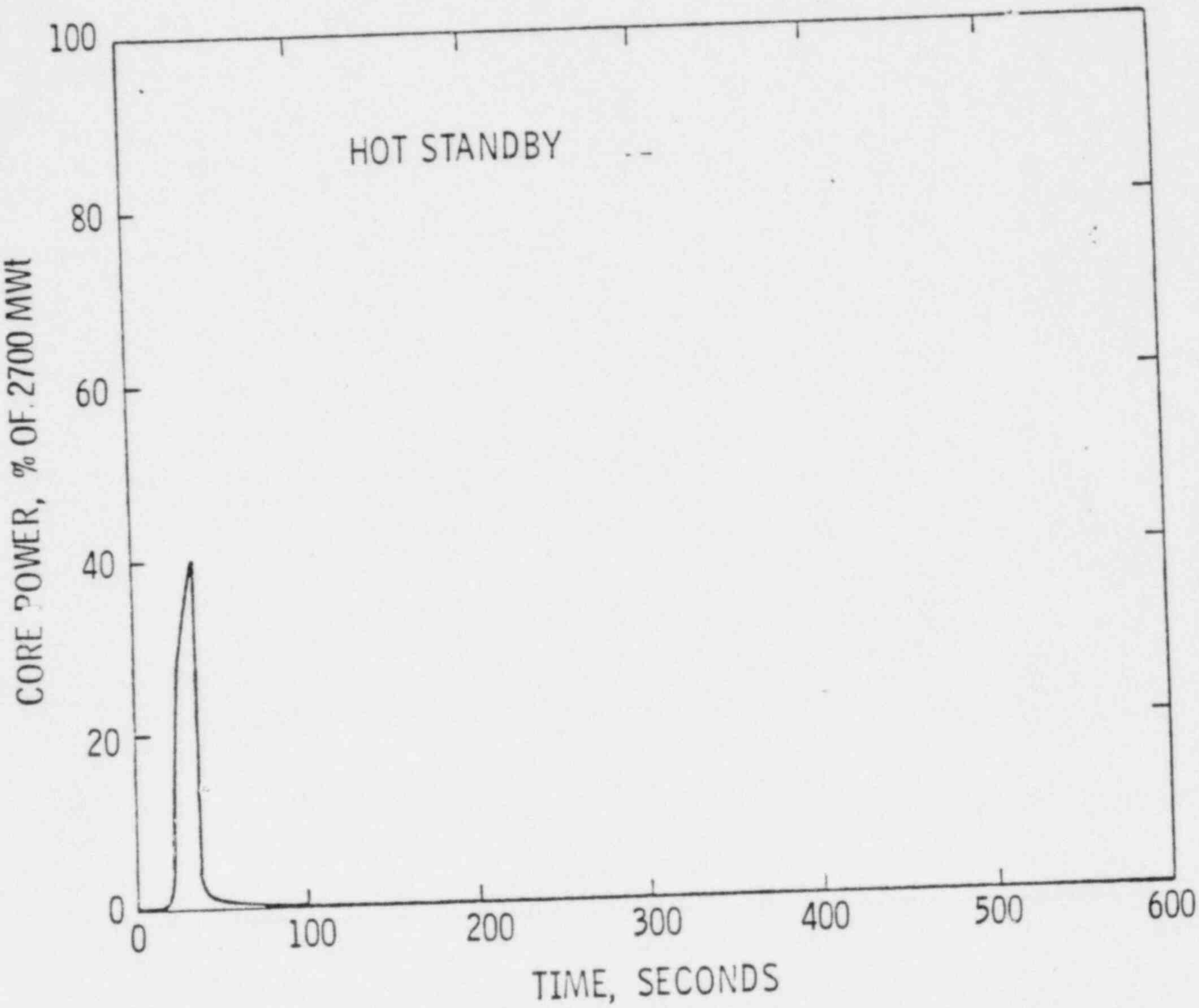
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT TEMPERATURE vs TIME	Figure 7.1.4-3



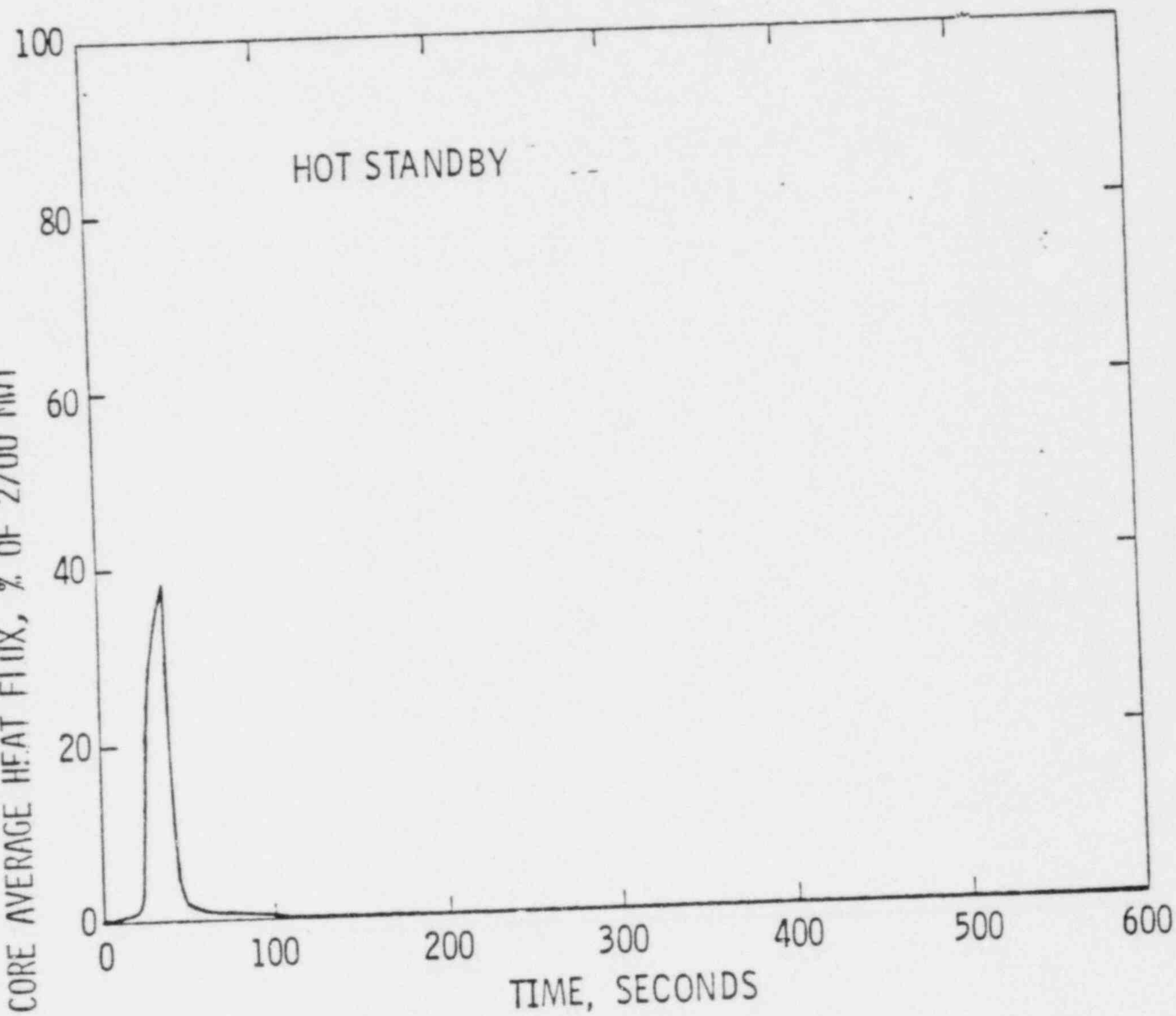
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT REACTOR COOLANT SYSTEM PRESSURE vs TIME	Figure 7.1.4-4



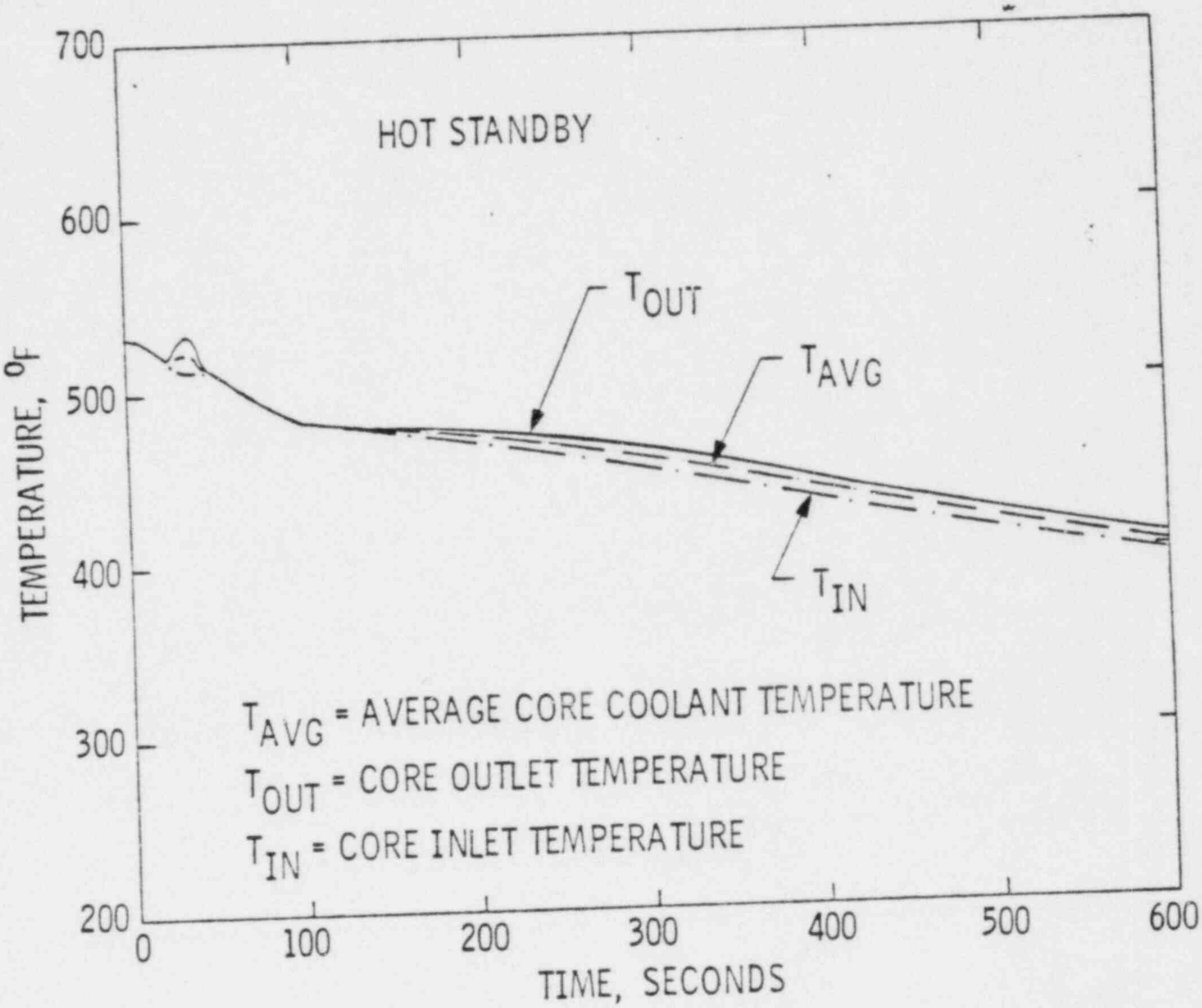
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT MAIN STEAM PRESSURE vs TIME	Figure 7.1.4-5



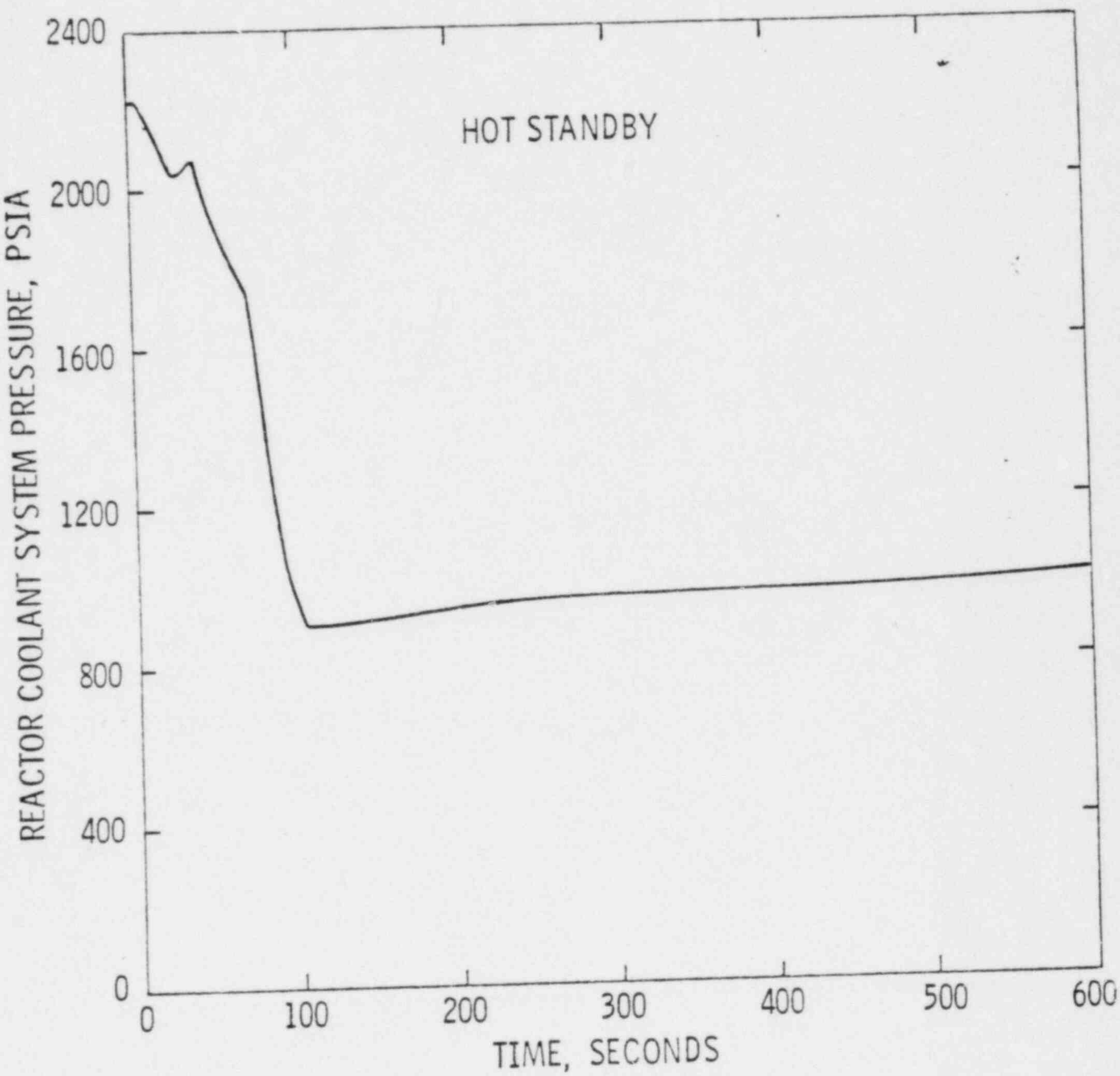
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT CORE POWER vs TIME	Figure 7.1.4-6



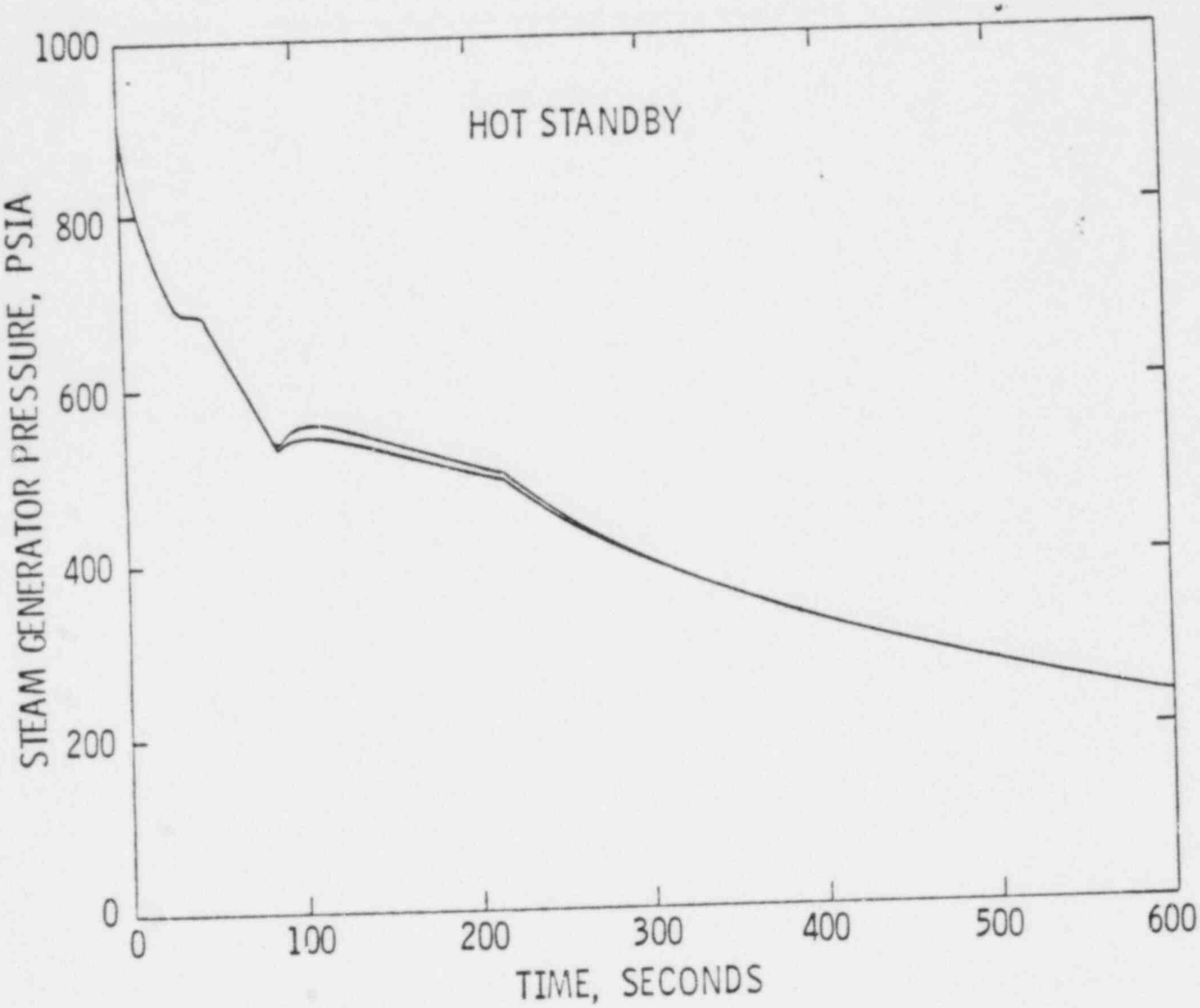
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT HEAT FLUX vs TIME	Figure 7.1.4-7



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT TEMPERATURE vs TIME	Figure 7.1.4-8



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT REACTOR COOLANT SYSTEM PRESSURE vs TIME	Figure 7.1.4-9



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD INCIDENT MAIN STEAM PRESSURE vs TIME	Figure 7.1.4-10

References for Section 7 (Non-LOCA Transient Analysis)

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

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

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

2. A. E. Lundvall to R. A. Clark, Calvert Cliffs Nuclear Power Plant - I Docket No. 50-317, "Amendment to Operating License DPR-53 Supplement 1 5th Cycle License Application," November 4, 1980.

7.1.5 LOSS OF LOAD EVENT

The Loss of Load event was reanalyzed for Cycle 6 to determine that the transient DNBR does not exceed the new design limit and that the RCS pressure upset limit of 2750 psia is not exceeded.

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

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

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

The Loss of Load event was initiated at the conditions shown in Table 7.1.5-1. The combination of parameters shown in Table 7.1.5-1 maximizes the calculated peak RCS pressure. As can be inferred from the table, the key parameters for this event are the initial primary and secondary pressures and the moderator and fuel temperature coefficients of reactivity.

The initial core average axial power distribution for this analysis was assumed to be a bottom peaked shape. This distribution is assumed because it minimizes the negative reactivity inserted during the initial portion of the scram following a reactor trip and maximizes the time required to mitigate the pressure and heat flux increases. The Moderator Temperature Coefficient (MTC) of $+.5 \times 10^{-4} \Delta\rho/°F$ was assumed in this analysis. This MTC in conjunction with the increasing coolant temperatures, maximizes the rate of change of heat flux and the pressure at the time of reactor trip. A Fuel Temperature Coefficient (FTC) corresponding to beginning of cycle conditions was used in the analysis. This FTC causes the least amount of negative reactivity feedback to mitigate the transient increases in both the core heat flux and the pressure. The uncertainty on the FTC used in the analyses is shown in Table 7.1.5-1. The lower limit on initial RCS pressure is used to maximize the rate of change of pressure, and thus peak pressure, following trip.

The Loss of Load event, initiated from the conditions given in Table 7.1.5-1, results in a high pressurizer pressure trip signal at 8.3 seconds. At 11.5 seconds, the primary pressure reaches its maximum value of 2550.0 psia. The increase in secondary pressure is limited by the opening of the main steam safety valves, which open at 3.7 seconds. The secondary pressure reaches its maximum value of 1050.0 psia at 11.4 seconds after initiation of the event.

Table 7.1.5-2 presents the sequence of events for this event. Figures 7.1.5-1 to 7.1.5-4 show the transient behavior of power, heat flux, RCS coolant temperatures, and RCS pressure.

The event was also reanalyzed with the initial conditions listed in Table 7.1.5-3 to determine that the acceptable DNBR limit is not exceeded. The minimum transient DNBR calculated for the event is 1.38 as compared to the design limit of 1.23.

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

TABLE 7.1.5-1

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

Parameter	Units	Reference* Cycle	Cycle 6
Initial Core Power Level	MWt	2754	2754
Initial Core Inlet Coolant Temperature	°F	550	550
Core Coolant Flow	$X10^6$ lbm/hr	133.9	133.9
Initial Reactor Coolant System Pressure	psia	2200	2200
Initial Steam Generator Pressure	psia	864.0	864.0
Moderator Temperature Coefficient	$X10^{-4} \Delta\rho/°F$	+.5	+.5
Doppler Coefficient Multiplier		.85	.85
CEA Worth at Trip	$\%\Delta\rho$	-4.7	-4.7
Time to 90% Insertion of Scram Rods	sec	3.1	3.1
Reactor Regulating System	Operating Mode	Manual	Manual
Steam Dump and Bypass System	Operating Mode	Inoperative	Inoperative

* Cycle 5 (Reference 2)

TABLE [illegible].1.5-2

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

Time(sec)	Event	Setpoint or Value
0.0	Loss of Secondary Load	--
3.7	Steam Generator Safety Valves Open	1000 psia
8.3	High Pressurizer Pressure Trip Signal Generated	2422 psia
9.7	CEAs Begin to Drop Into Core	--
9.8	Pressurizer Safety Valves Open	2500 psia
11.4	Maximum Steam Generator Pressure	1050 psia
11.5	Maximum RCS Pressure	2550 psia
13.4	Pressurizer Safety Valves are Fully Closed	2500 psia

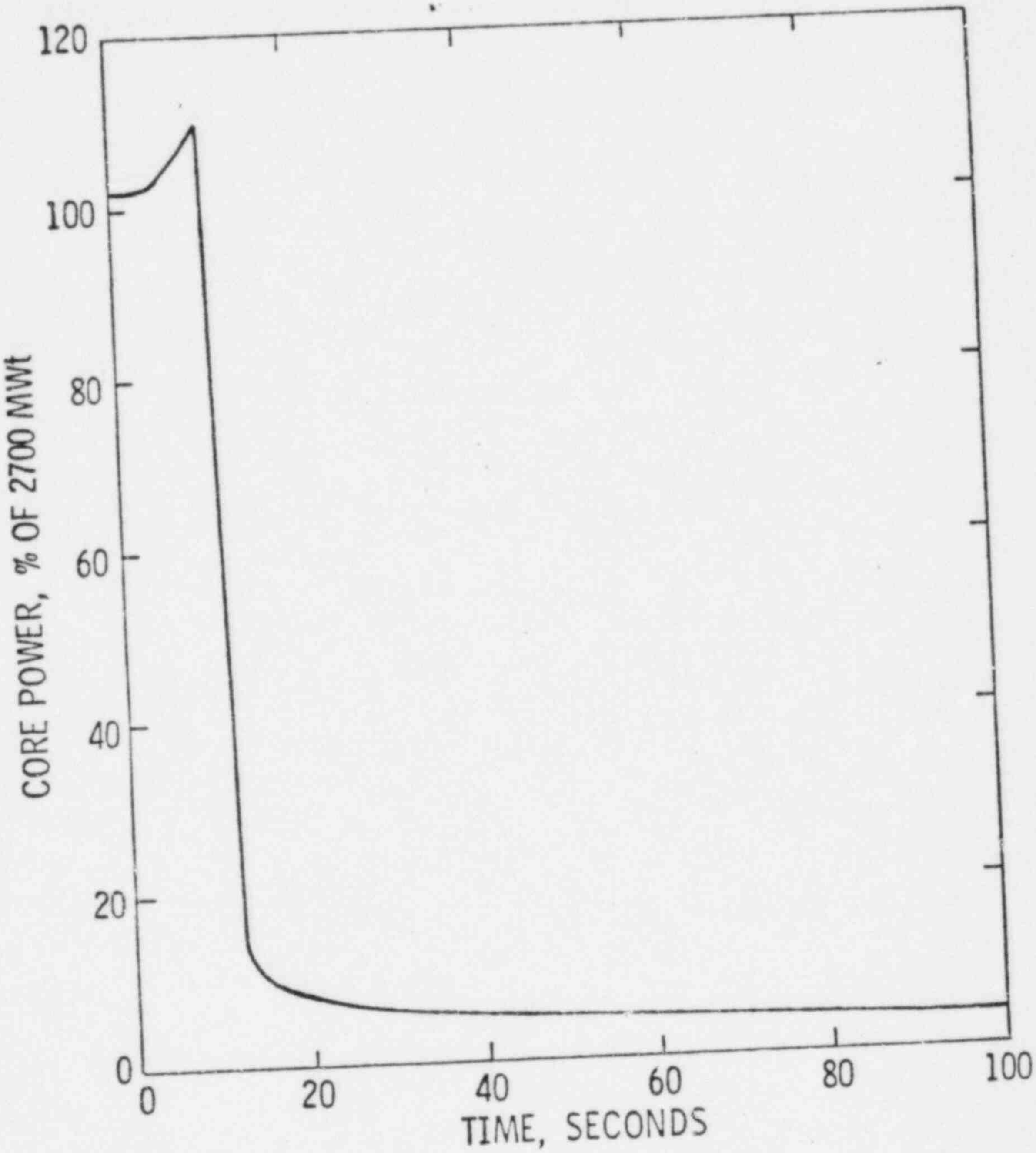
TABLE 7.1.5-3

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

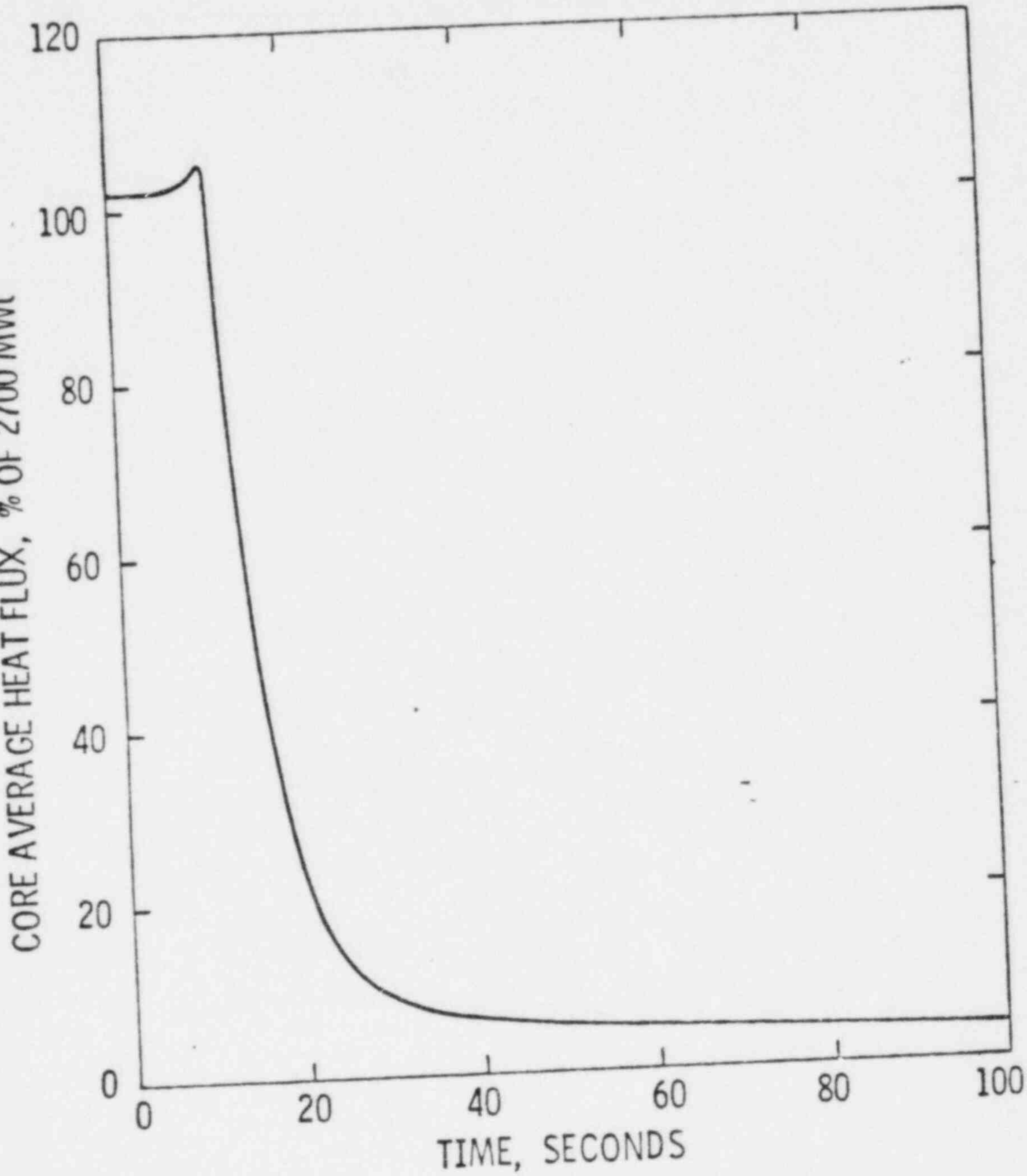
Parameter	Units	Reference* Cycle	Cycle 6
Initial Core Power Level	MWt	2754	2700 **
Initial Core Inlet Coolant Temperature	°F	550	548 **
Core Coolant Flow	$X10^6$ lbm/hr	133.9	138.5 **
Initial Reactor Coolant System Pressure	psia	2200	2225 **
Initial Steam Generator Pressure	psia	864.0	864.0
Integrated Radial Peaking Factors, F_r^t (Bank 5 inserted 25%)		1.71	1.75 **
Moderator Temperature Coefficient	$X10^{-4}\Delta\rho/°F$	+.5	+.5
Doppler Coefficient Multiplier		.85	.85
CEA Worth at Trip	$\%\Delta\rho$	-4.7	-4.7
Time to 90% Insertion of Scram Rods	sec	3.1	3.1
Reactor Regulating System	Operating Mode	Manual	Manual
Steam Dump and Bypass System	Operating Mode	Inoperative	Inoperative

* Cycle 5 (Reference 2)

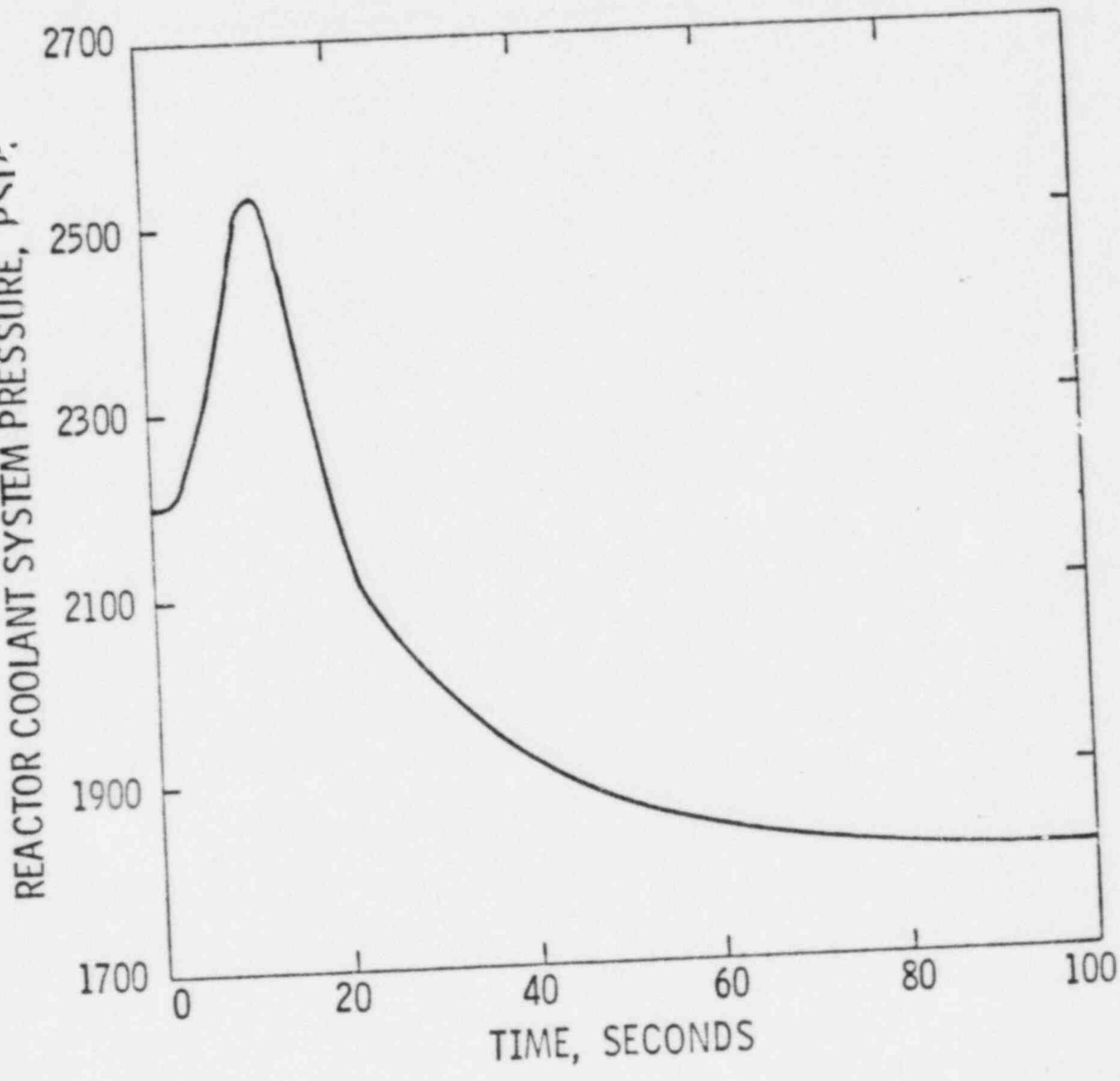
** Effects of uncertainties on these parameters were accounted for statistically. (See Reference 1)



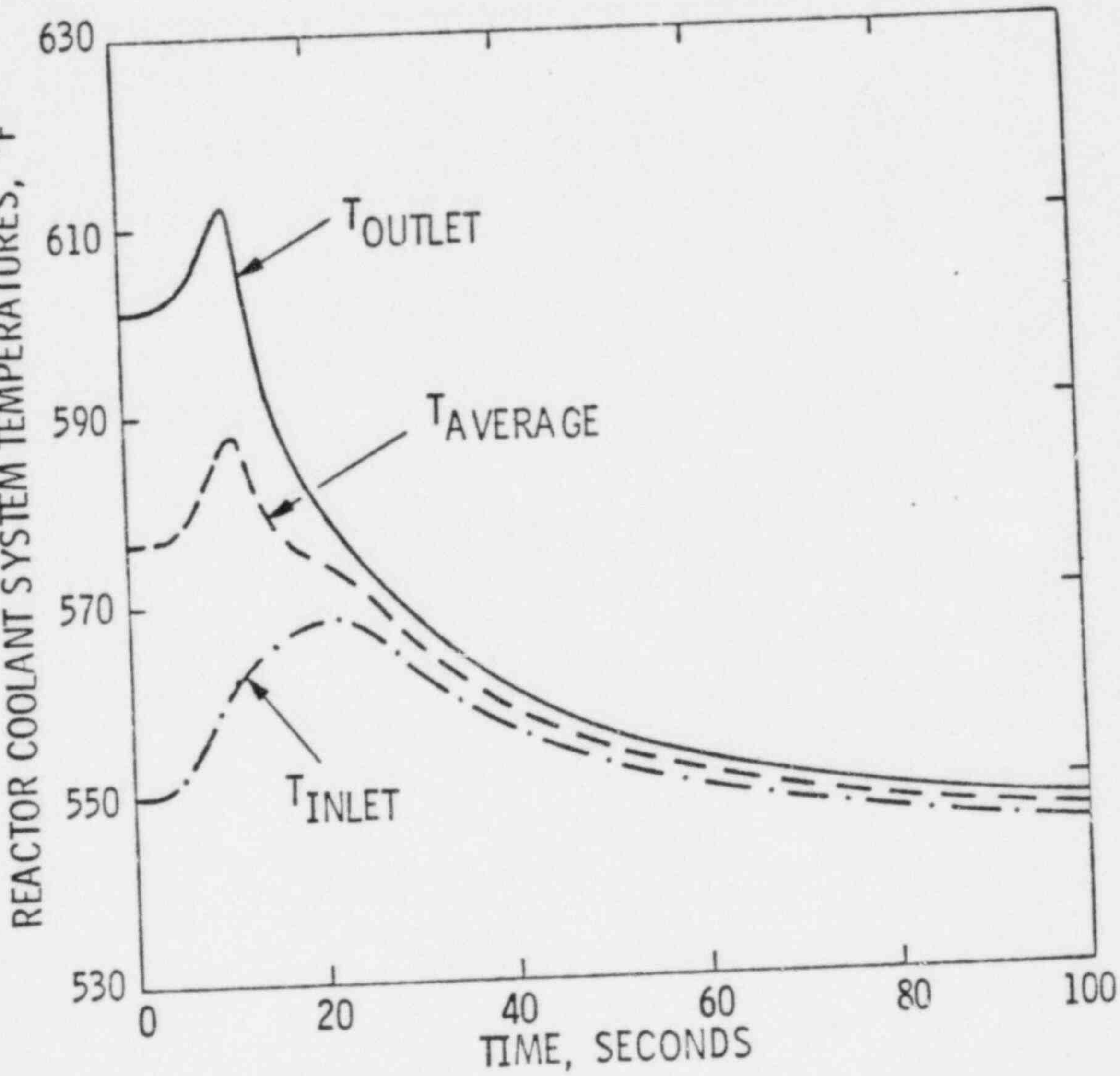
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD CORE POWER vs TIME	Figure 7.1.5-1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD CORE AVERAGE HEAT FLUX vs TIME	Figure 7.1.5-2



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD REACTOR COOLANT SYSTEM PRESSURE vs TIME	Figure 7.1.5-3



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD REACTOR COOLANT SYSTEM TEMPERATURES vs TIME	Figure 7.1.5-4

7.2.3 FULL LENGTH CEA DROP EVENT

The Full Length CEA Drop event was reanalyzed for Cycle 6 to determine the initial thermal margins that must be maintained by the Limiting Conditions for Operation (LCOs) such that the DNBR and fuel centerline melt design limit will not be exceeded.

The methods used to analyze this event are consistent with those discussed in Reference 1 except CETOP/CE-1 was used instead of TORC/CE-1 to calculate DNBR.

Table 7.2.3-1 lists the key input parameters used for Cycle 6 and compares them to the reference cycle values. Conservative assumptions used in the analysis include:

1. The most negative moderator and fuel temperature coefficients of reactivity (including uncertainties), because these coefficients produce the minimum RCS coolant temperature decrease upon return to 100% power level and lead to the minimum DNBR.

2. Charging pumps and proportional heater systems are assumed to be inoperable during the transient. This maximizes the pressure drop during the event.

3. All other systems are assumed to be in manual mode of operation and have no impact on this event.

The event is initiated by dropping a full length CEA over a period of 1.0 second. The maximum increases in (integrated and planar) radial peaking factors in either rodded or unrodded planes were used in all axial regions of the core once the power returns to the initial level. Values of 16% were assumed for these peak increases at full power. The axial power shape in the hot channel is assumed to remain unchanged and hence the increase in the 3-D peak is proportional to the maximum increase in radial peaking factor of 16%. Since there is no trip assumed, the peaks will stabilize at these asymptotic values after a few minutes since the secondary side continues to demand 100% power.

Table 7.2.3-2 presents the sequence of events for the Full Length CEA Drop event initiated at the conditions described in Table 7.2.3-1. The transient behavior of key NSSS parameters are presented in Figures 7.2.3-1 to 7.2.3-4.

The transient initiated at the most negative shape index LCO (-.15) and at the maximum power level allowed by the LCO, results in a minimum CE-1 DNBR of 1.23. A maximum allowable initial linear heat generation rate of 18.2 KW/ft could exist as an initial condition without exceeding 21.3 KW/ft during this transient. This amount of margin is assured by setting the Linear Heat Rate related LCO's based on the more limiting allowable linear heat rate for LOCA.

Consequently, it is concluded that the Full Length CEA Drop event initiated from the Tech Spec LCOs will not exceed the DNBR and centerline to melt design limits.

TABLE 7.2.3-1

KEY PARAMETERS ASSUMED IN THE FULL LENGTH CEA DROP ANALYSIS

Parameter	Units	Reference Cycle*	Cycle 6
Initial Core Power Level	MWt	2754	2700 +
Core Inlet Temperature	°F	550	548 +
Reactor Coolant System Pressure	psia	2200	2225 +
Core Mass Flow Rate	$X10^6$ lbm/hr	133.9	138.5 +
Moderator Temperature Coefficient	$X10^{-4}$ $\Delta\rho$/°F	-2.5	-2.5
Doppler Coefficient Multiplier	--	1.15	1.15
Maximum CEA Insertion at Allowed Power	% Insertion of Bank 5	25	25
Dropped CEA Worth	%$\Delta\rho$ unrodded PDIL	-.04 -.04	-.04 -.04
Most Negative Axial Shape Index Allowed at Full Power (LCO)		-.16	-.15+
Integrated and Planar Radial Peaking Distortion Factor (Full Power)	Unrodded Region Bank Inserted Region	1.16 1.16	1.16 1.16

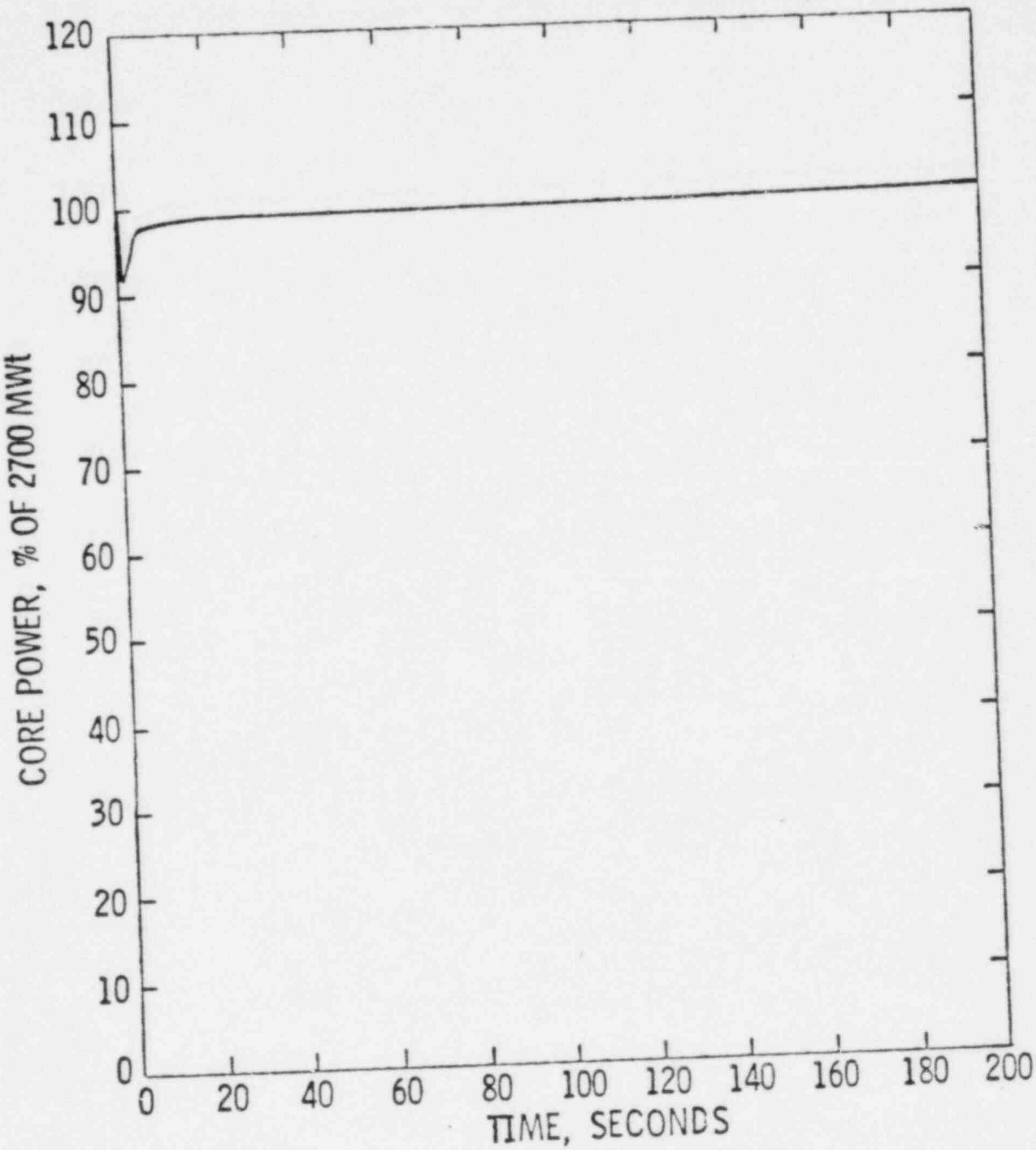
* Cycle 5 (Reference 2)

\+ For DNBR calculations, effects of uncertainties on these parameters were combined statistically. (See Reference 1)

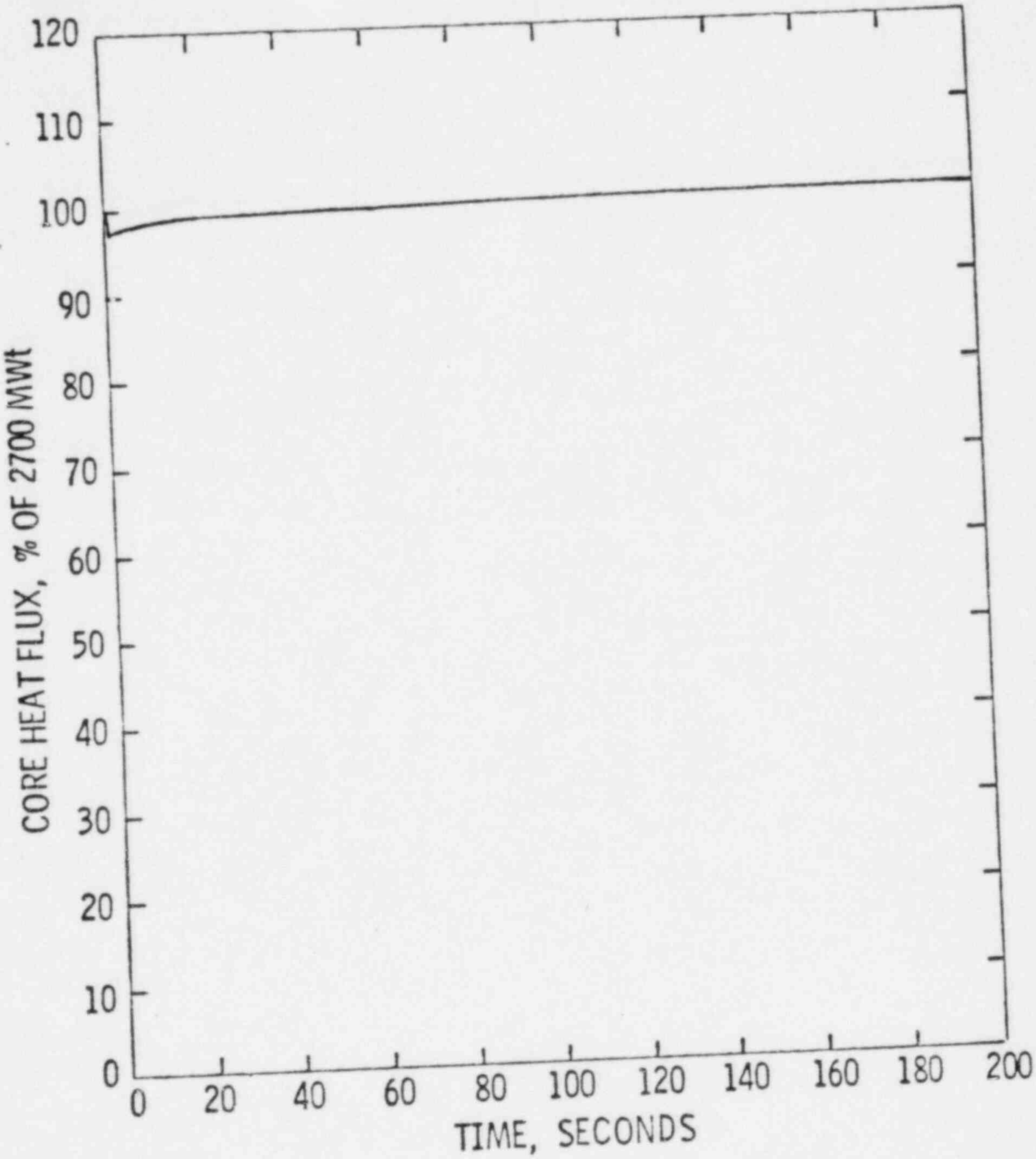
TABLE 7.2.3-2

SEQUENCE OF EVENTS FOR CEA DROP

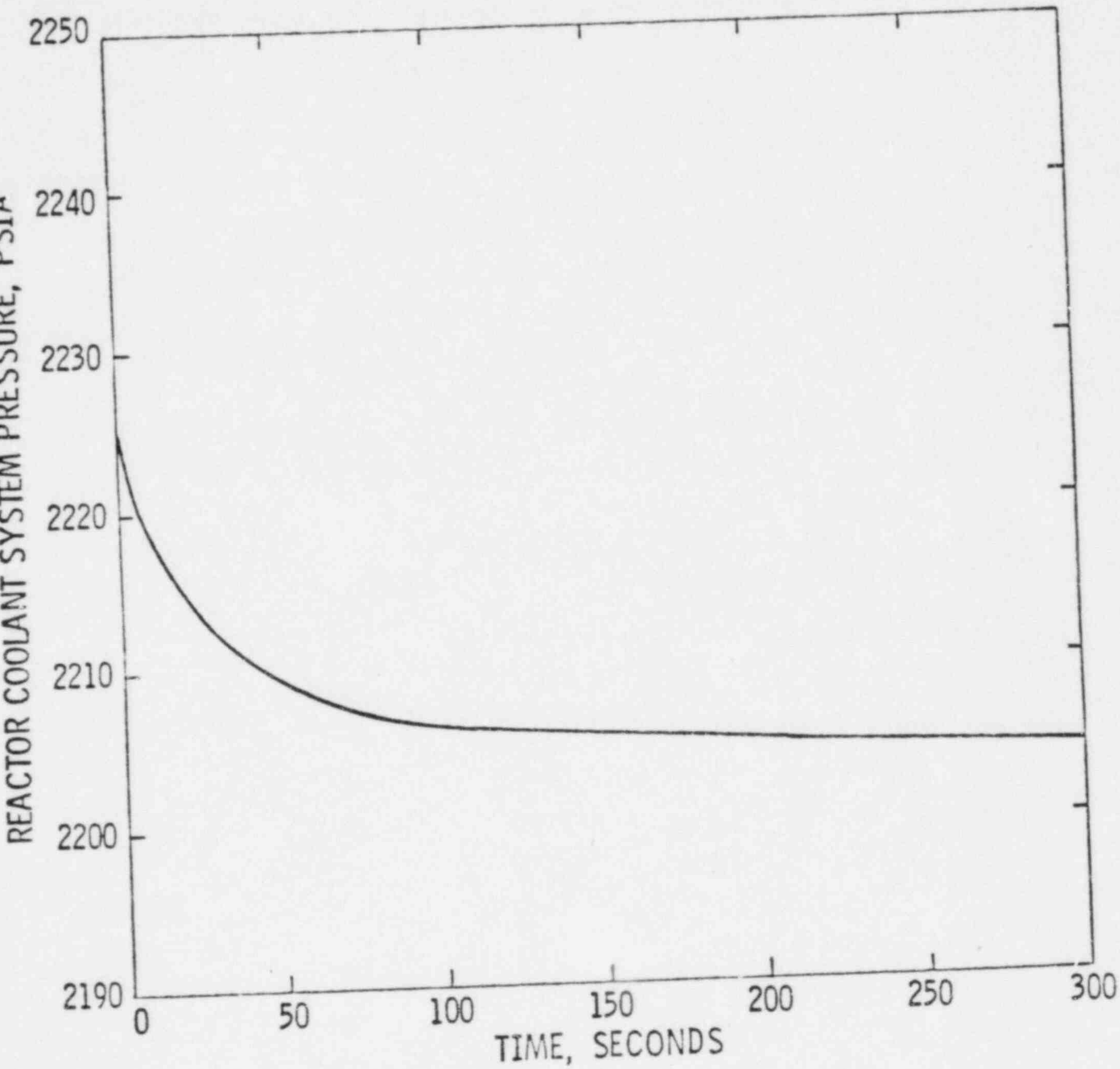
Time(sec)	Event	Setpoint Value
0.0	CEA Begins to Drop	----
1.0	CEA Fully Dropped	-0.04%Δρ
1.1	Core Power Reaches Minimum	92.2%
4.2	Core Heat Flux Reaches Minimum	98.1%
300.	Heat Flux Reaches Final Value	100%
300.	Core Inlet Temperature Reaches Minimum	546.5°F
300.	RCS Pressure Reaches Minimum	2204.3 psia
300.	Minimum DNBR Reached	1.23



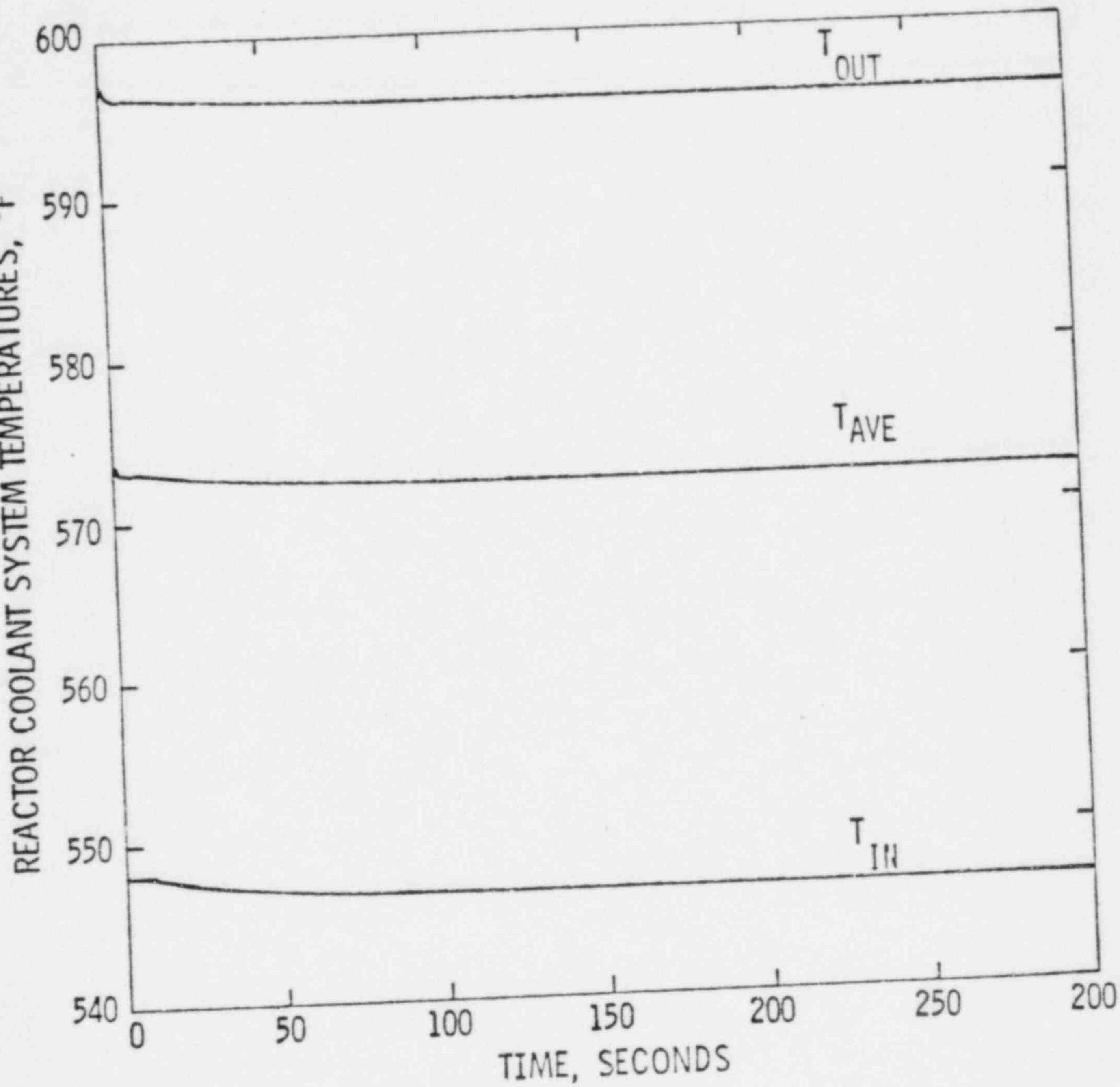
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FULL LENGTH CEA DROP CORE POWER vs TIME	Figure 7.2.3-1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FULL LENGTH CEA DROP CORE AVERAGE HEAT FLUX vs TIME	Figure 7.2.3-2



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FULL LENGTH CEA DROP REACTOR COOLANT SYSTEM PRESSURE vs TIME	Figure 7.2.3-3



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	FULL LENGTH CEA DROP REACTOR COOLANT SYSTEM TEMPERATURES vs TIME	Figure 7.2.3-4

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

The transients resulting from the malfunction of one steam generator were analyzed for Cycle 6 to determine the initial margins that must be maintained by the LCO's such that in conjunction with the RPS (Asymmetric Steam Generator Protective trip), the DNBR and fuel centerline melt design limits are not exceeded.

The methods used to analyze these events are consistent with those reported in the reference cycle, except that CETOP/CE-1 was used instead of TORC/CE-1 to calculate the DNBR.

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

1. Loss of Load to One Steam Generator
2. Excess Load to One Steam Generator
3. Loss of Feedwater to One Steam Generator
4. Excess Feedwater to One Steam Generator

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

The event is initiated by the inadvertent closure of a single main steam isolation valve. Upon the loss of load to the single steam generator, its pressure and temperature increase to the opening pressure of the secondary safety valves. The intact steam generator "picks up" the lost load, which causes its temperature and pressure to decrease. The cold leg asymmetry causes an inlet temperature tilt which results in an azimuthal power tilt, increased PLHGR and a degraded DNBR.

The LL/1SG was initiated at the conditions given in Table 7.2.4-1. A reactor trip is generated by the Asymmetric Steam Generator Protection Trip at 2.6 seconds based on high differential pressure between the steam generators.

Table 7.2.4-2 presents the sequence of events for the Loss of Load to One Steam Generator. The transient behavior of key NSSS parameters are presented in Figures 7.2.4-1 to 7.2.4-5.

A maximum allowable initial linear heat generation rate of 19.3 KW/ft could exist as an initial condition without exceeding 21.3 KW/ft during this transient. This amount of margin is assured by setting the Linear Heat Rate LCO based on the more limiting allowable linear heat rate for LOCA.

The event initiated from the extremes of the LCO in conjunction with the ASGP trip will not lead to DNBR or centerline fuel temperatures which exceed the DNBR and centerline to melt design limits.

The minimum transient DNBR calculated for the LL/1SG event is 1.43 as compared to the minimum acceptable DNBR of 1.23.

TABLE 7.2.4-1

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

Parameter	Units	Reference Cycle*	Cycle 6
Initial Core Power	MWt	2754	2700 +
Initial Core Inlet Temperature	°F	550	548+
Initial Reactor Coolant System Pressure	psia	2200	2225+
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	-2.5	-2.5
Doppler Coefficient Multiplier	--	0.85	0.85

* Cycle 5 (Reference 2)

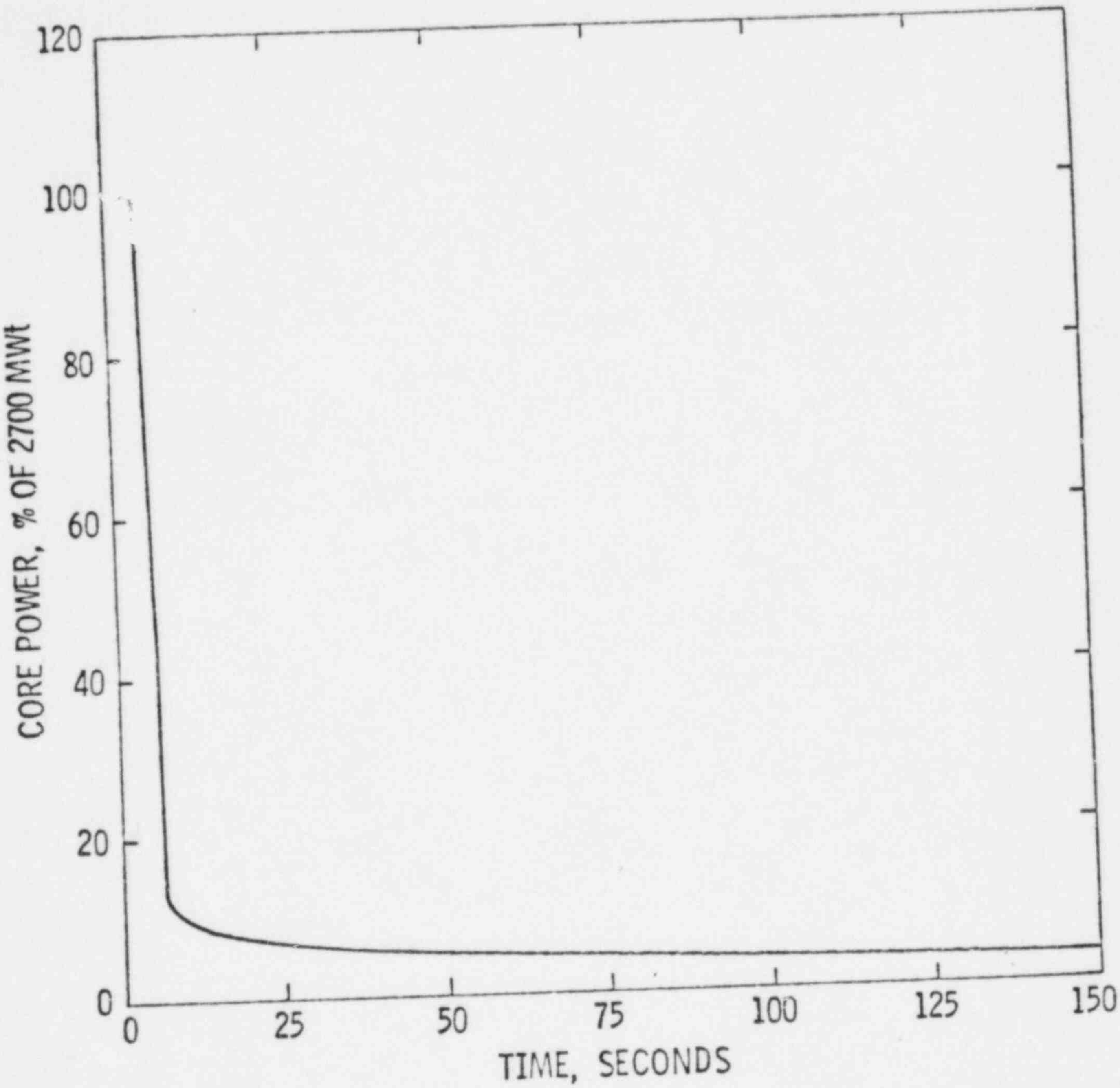
\+ For DNBR calculations, effects of uncertainties on these parameters were combined statistically. (See Reference 1)

TABLE 7.2.4-2

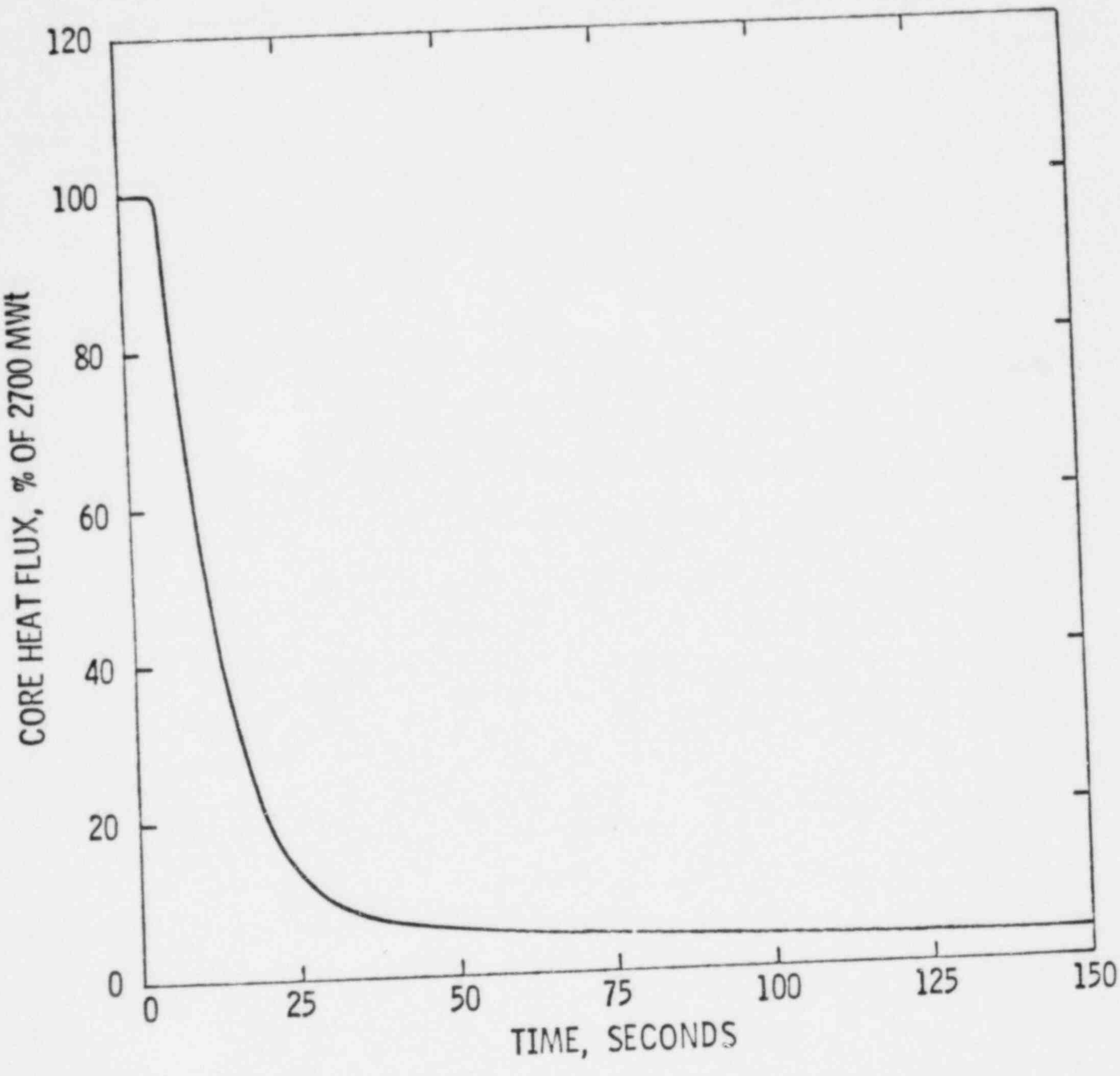
SEQUENCE OF EVENTS FOR LOSS OF LOAD TO ONE STEAM GENERATOR

Time(sec)	Event	Setpoint or Value
0.0	Spurious closure of a single main steam isolation valve	----
0.0	Steam flow from unaffected steam generator increases to maintain turbine power	----
2.6	ASGPT* setpoint reached (differential pressure)	175 psid
3.2	Dump and Bypass valves are open	----
3.5	Trip breakers open	----
4.0	CEAs begin to insert	----
4.0	Safety valves open on isolated steam generator	1000 psia
5.5	Minimum DNBR occurs	1.43
10.1	Maximum steam generator pressure	1050 psia

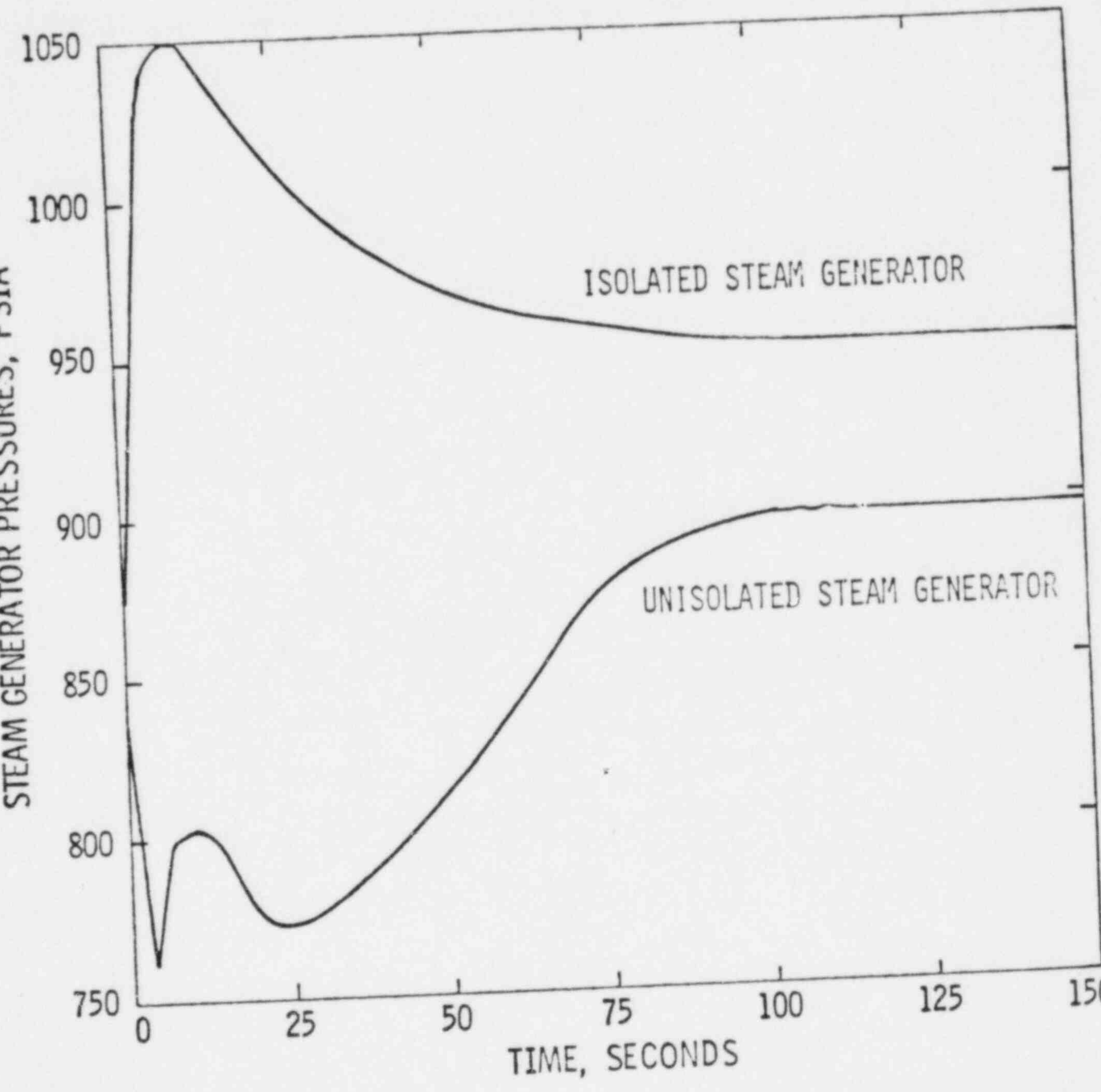
* ASGPT - Asymmetric Steam Generator Protection Trip



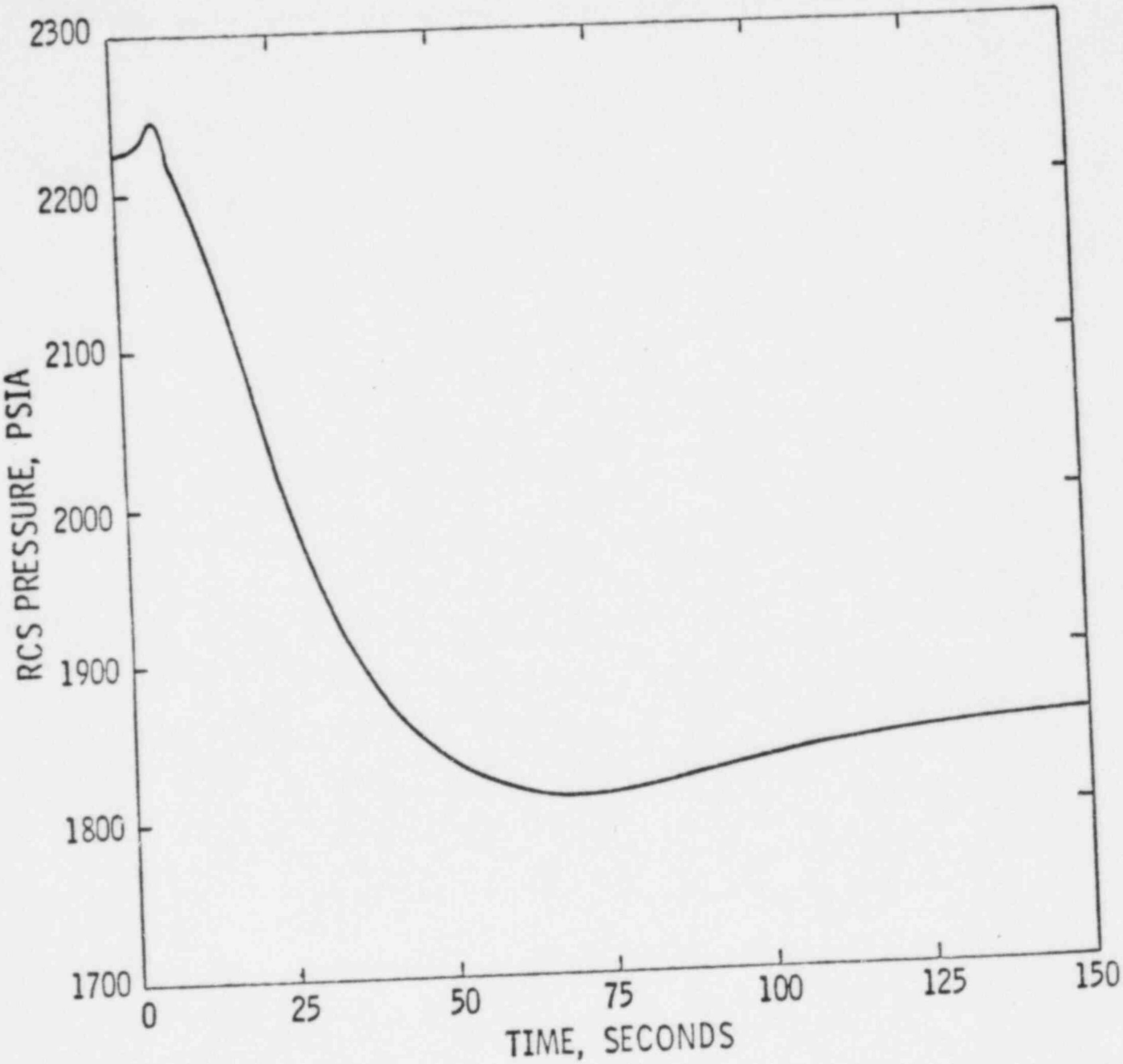
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT CORE POWER vs TIME	Figure 7.2.4-1



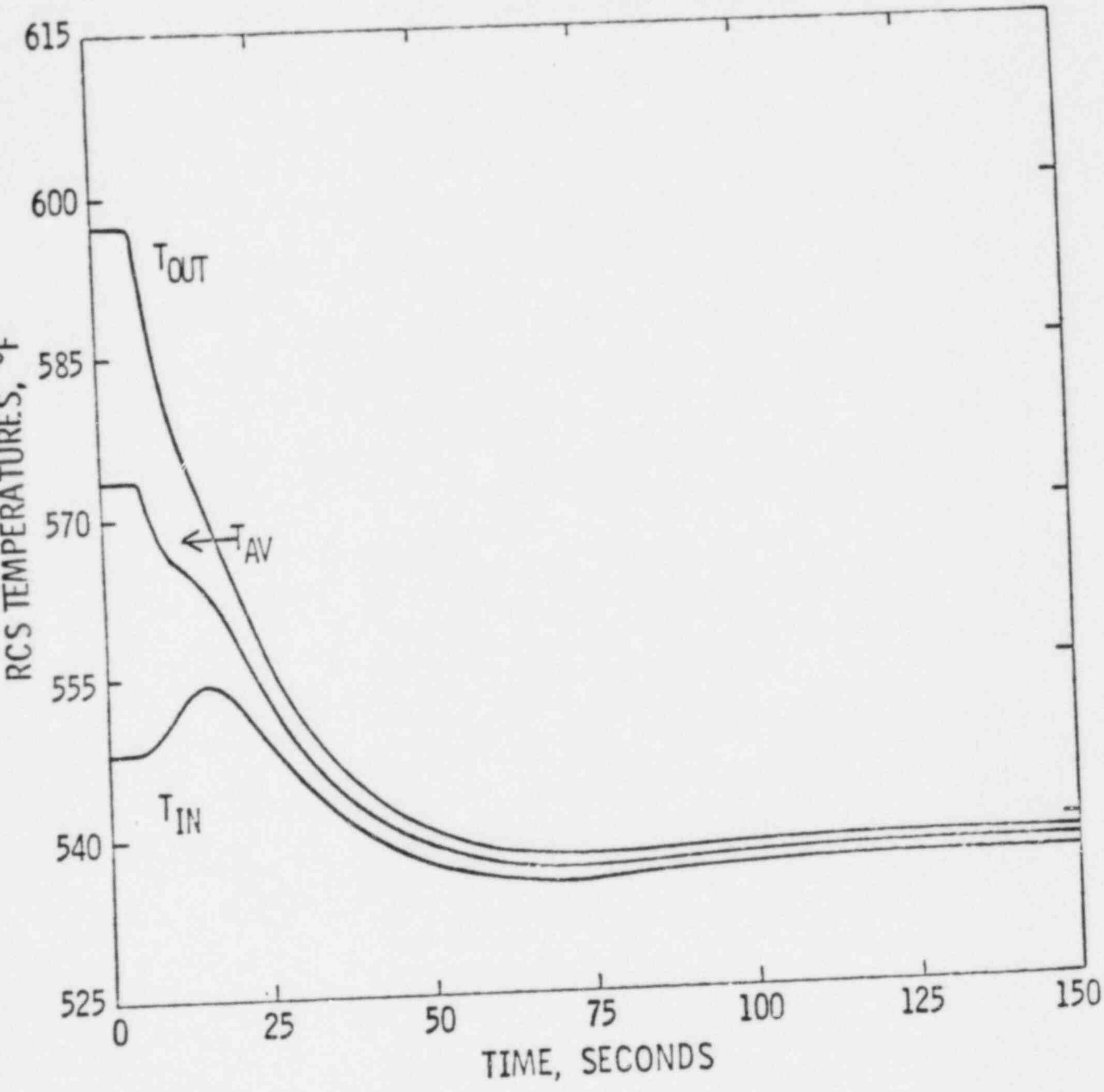
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT CORE AVERAGE HEAT FLUX vs TIME	Figure 7.2.4-2



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT STEAM GENERATOR PRESSURE vs TIME	Figure 7.2.4-3



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT REACTOR COOLANT SYSTEM PRESSURE vs TIME	Figure 7.2.4-4



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD/1 STEAM GENERATOR EVENT REACTOR COOLANT SYSTEM TEMPERATURE vs TIME	Figure 7.2.4-5

References for Section 7 (Non-LOCA Transient Analysis)

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

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

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

2. A. E. Lundvall to R. A. Clark, Calvert Cliffs Nuclear Power Plant - I Docket No. 50-317, "Amendment to Operating License DPR-53 Supplement 1 5th Cycle License Application," November 4, 1980.