ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATIONS

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble and with at least 150 kw of pressurizer heater capacity capable of being supplied by emergency power. ~~The pressurizer level shall be within ± 5 percent of its programmed value.~~ * see attached sheet

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water level shall be determined to be within ~~± 5 percent of its programmed value~~ at least once per 12 hours.
the above band

(*) The pressurizer level shall be maintained within an operating band between 133 and 225 inches except when three charging pumps are operating and letdown flow is less than 25 GPM. If three charging pumps are operating and letdown flow is less than 25 GPM maximum pressurizer level shall be limited to ~~less than 210 inches.~~ between 133 and 210 inches.

limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.4 PRESSURIZER

see insert on next page.

A steam bubble in the pressurizer with the level as programmed ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. ~~The programmed level~~ also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valves function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer--Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of off-site power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to

* The operating band for pressurizer level bounds the programmed level and ensures that RCS pressure remains within the bounds of an analyzed condition during the excessive charging event as well as during the limiting depressurization event, Excess Load. The operating band...

ATTACHMENT II

The following study was performed to determine and justify Technical Specification limits for pressurizer level. The approach taken was to reanalyze the Loss of Load event and the Excessive Charging event to determine the maximum pressurizer level and evaluate the Excess Load event to determine the minimum level. The Steam Generator Tube Rupture event was also analyzed to determine site boundary doses, which increase due to higher primary to secondary leak.

The limiting pressurization event, Loss of Load (LOL), was analyzed such that the event initiated from a given pressurizer level does not cause the RCS pressure to exceed the design limit of 2750 psia. The Excessive Charging event initiated from a given pressurizer level was analyzed to assure that the operator has at least fifteen (15) minutes from initiation of a high pressurizer level alarm to take corrective actions and terminate the event prior to filling the pressurizer solid. The maximum pressurizer level is then selected based on the results of the limiting event.

The limiting depressurization event, Excess Load, was analyzed to assure that the draining of the pressurizer and subsequent Reactor Vessel Upper Head (RVUH) voiding do not affect primary coolant circulation and do not result in exceeding fuel design limits. Hence, the Excess Load event was analyzed to determine the Technical Specification limit on the minimum pressurizer level.

Finally, a reanalysis of the Steam Generator Tube Ruputure event was performed to predict the increased site boundary doses, since the increased pressurizer level results in higher primary to secondary leak.

Detailed description and results of each analysis are attached. Based on the results of this study, it is concluded that:

1. Based on the analysis of the Excess Load event, a minimum pressurizer level of 270 ft^3 (this level corresponds to the level required to cover pressurizer heaters) is acceptable.

2. Based on the analysis of the Excess Charging event and the Loss of Load event, a range of maximum pressurizer level (915 ft^3 to 975 ft^3) is acceptable.

1. Loss of Load Event

A reanalysis of the Loss of Load Event was initiated at 975 ft^3 in order to determine that the RCS pressure upset limit of 2750 psia is not exceeded. The transient DNBR was also evaluated to assess that the results are within the design limit of 1.23.

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

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

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

The Loss of Load Event was initiated at the conditions shown in Table 1-1. The combination of parameters shown 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 $+0.5 \times 10^{-4}$ $\Delta\rho$ /^{O}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 1-1. 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 1-1 results in a high pressurizer pressure trip signal at 6.2 seconds. At 10.1 seconds, the primary pressure reaches its maximum value of 2617 psia. The increase in secondary pressure is limited by the opening of the main steam safety valves, which open at 5.8 seconds. the secondary pressure reaches its maximum value of 1047 psia at 10.6 seconds after initiation of the event.

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

The event was also reanalyzed with the initial conditions listed in Table 1-3 to determine that the acceptable DNBR Limit is not exceeded. The minimum transient DNBR calculated for the event is 1.34, compared to the design limit of 1.23.

The results of this analysis demonstrate that during a Loss of Load Event, initiated from a pressurizer level of 975 ft^3, the peak RCS pressure and the minimum DNBR do not exceed their respective design limits.

TABLE 1-1

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

Parameter	Units	Reference* Cycle	This Analysis
Initial Core Power Level	MWt	2754	2754
Initial Core Inlet Coolant Temperature	^{o}F	550	550
Core Coolant Flow	$x10^{6}$ lbm/hr	133.9	133.9
Initial Reactor Coolant System Pressure	psia	2200+	2155++
Initial Steam Generator Pressure	psia	864	864
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^{o}F$	+0.5	+0.5
Doppler Coefficient Multiplier		0.85	0.85
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	Operating Mode	Inoperative	Inoperative

*Unit 1, Cycle 6

+Corresponds to Technical Specification minimum indicated pressure of 2225 psia. The value includes an uncertainty of 25 psia.

++Corresponds to Technical Specification minimum indicated pressure of 2200 psia. The value includes an uncertainty of 45 psia.

TABLE 1-2

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

Parameter	Units	Reference Cycle*	This Analysis
Initial Core Power Level	MWt	2700**	2700**
Initial Core Inlet Coolant Temperature	^{o}F	548**	548**
Core Coolant Flow	$x 10^{6}$ lbm/hr	138.5**	138.5**
Initial Reactor Coolant System Pressure	psia	2225**	2200**
Initial Steam Generator Pressure	psia	864	864
Integrated Radial Peaking Factors, F_r^T (Bank 5 Inserted 25%)		1.75**,+	1.75**,+
Moderator Temperature Coefficient	$x 10^{-4} \Delta\rho / ^{o}F$	+0.5	+0.5
Doppler Coefficient Multiplier		0.85	0.85
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

*Unit 1, Cycle 6

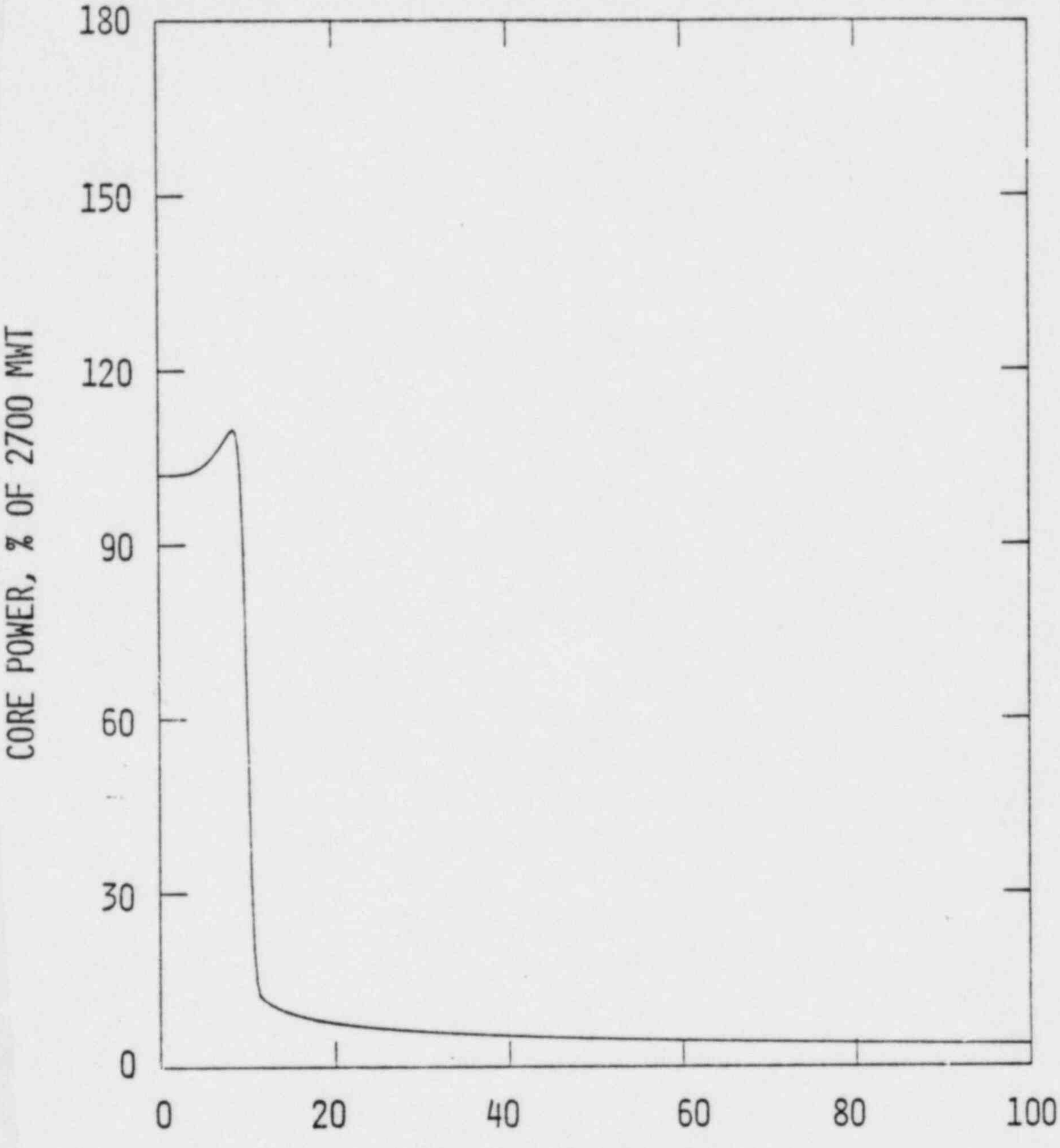
**Effects of uncertainties on these parameters were accounted for statistically.

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

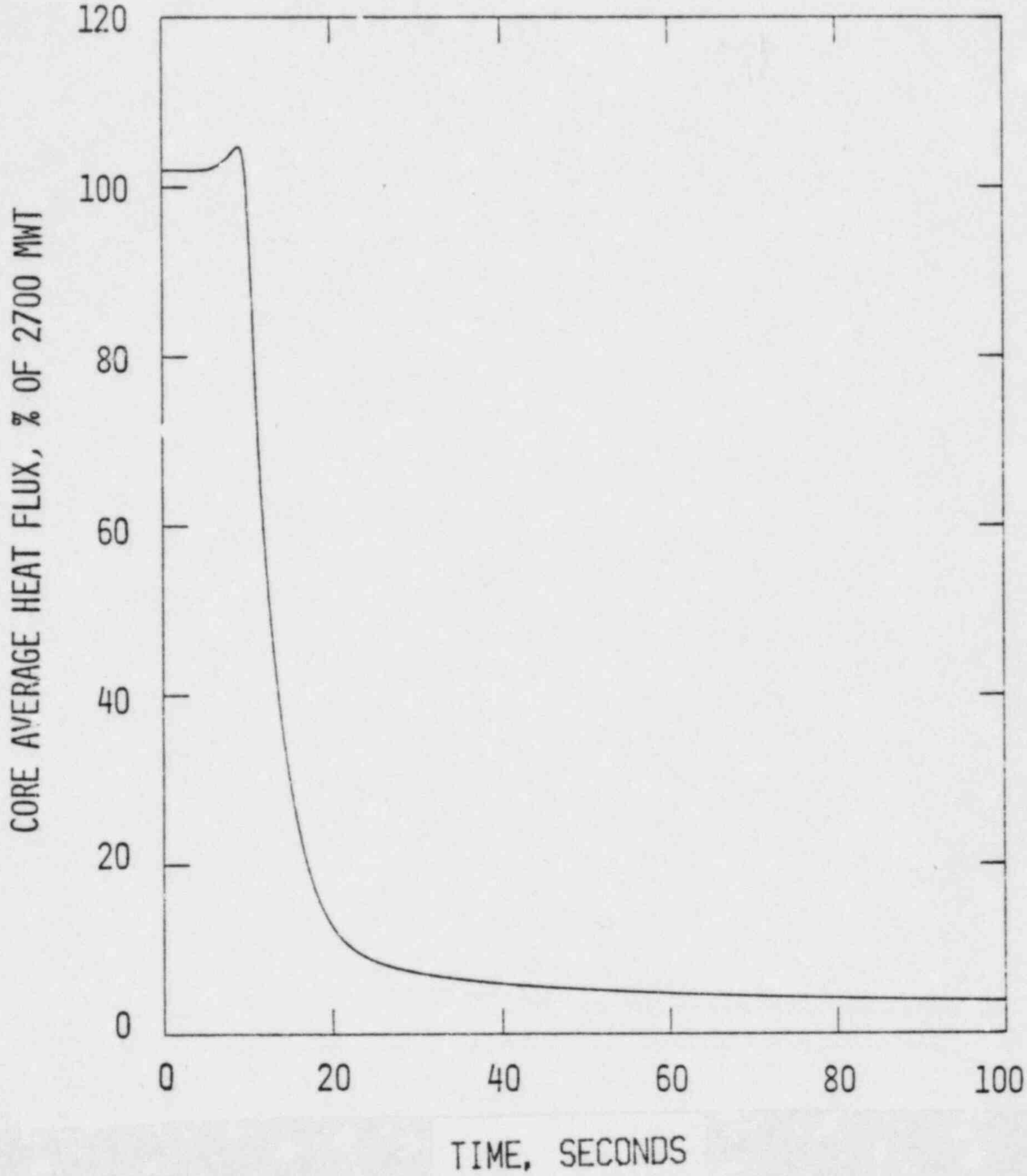
TABLE 1-3

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

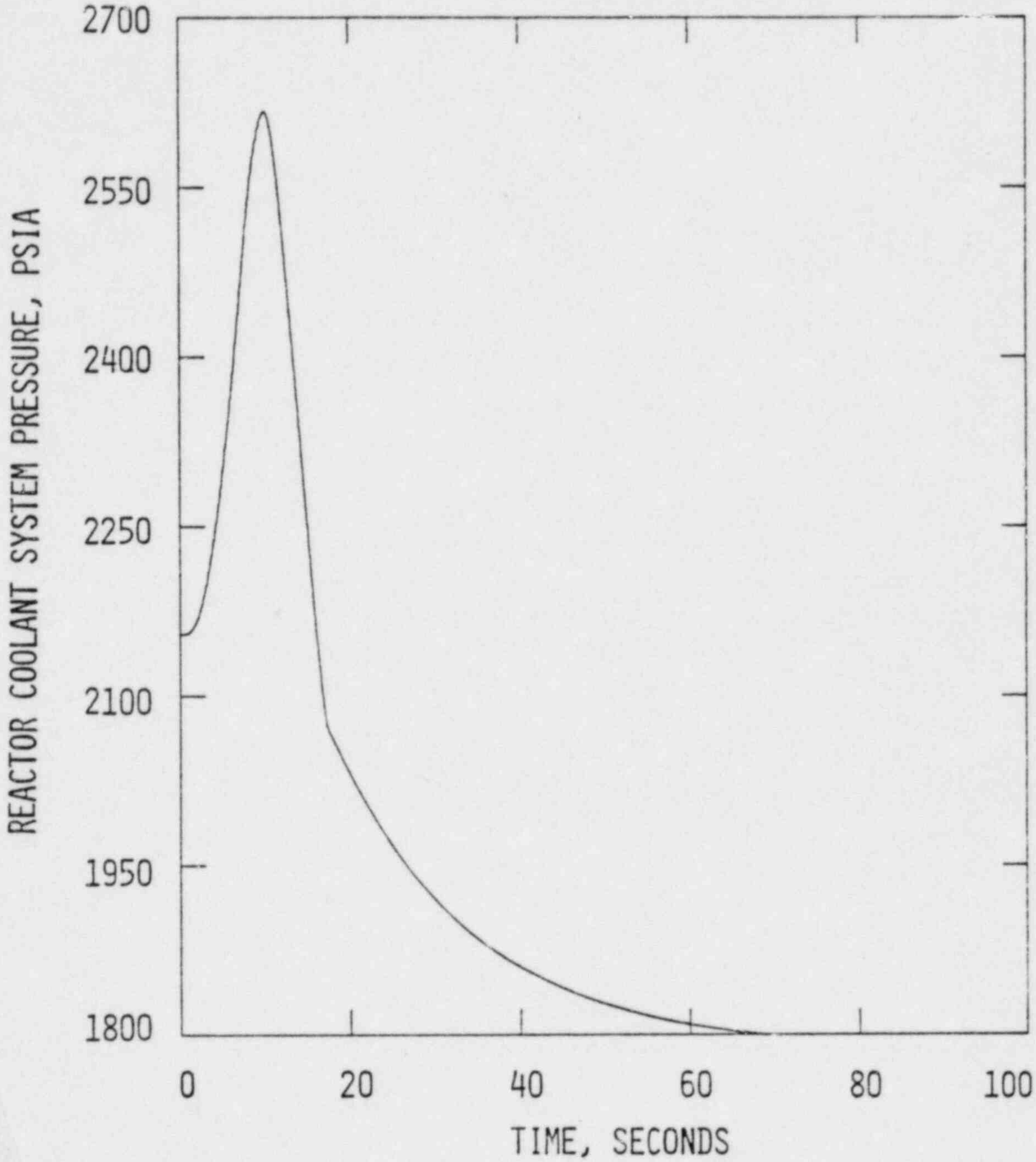
Time (sec)	Event	Setpoint or Value
0.0	Loss of Secondary Load	---
5.8	Steam Generator Safety Valves Open	1000 psia
6.2	High Pressurizer Pressure Trip Signal Generated	2422 psia
7.6	CEAs Begin to Drop Into Core	---
7.9	Pressurizer Safety Valves Open	2500 psia
10.1	Maximum RCS Pressure	2617 psia
10.6	Maximum Steam Generator Pressure	1047 psia
12.3	Pressurizer Safety Valves are Fully Closed	2500 psia



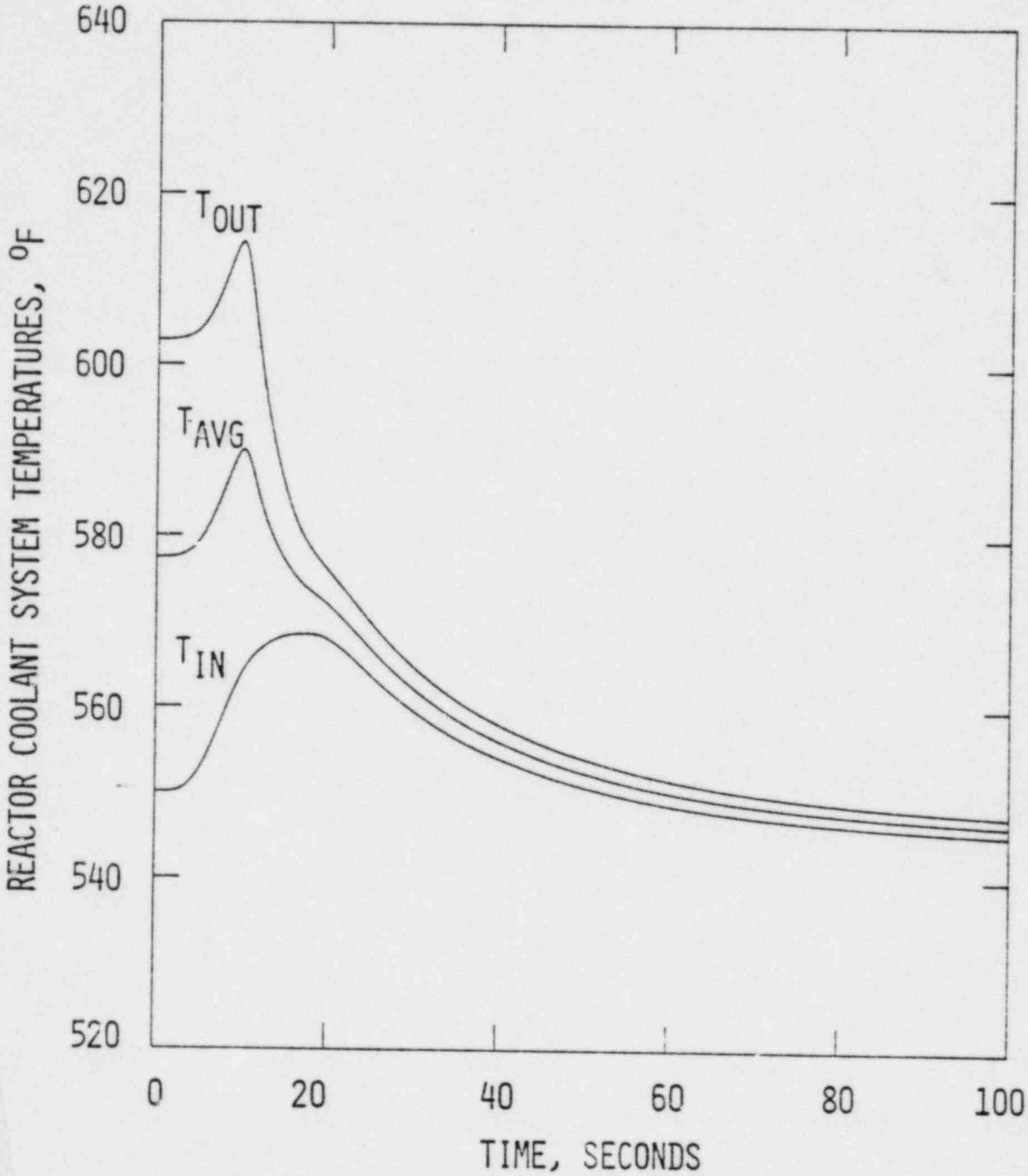
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT CORE POWER VS TIME	Figure 1-1



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



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	Figure 1-3



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	LOSS OF LOAD EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	Figure 1-4

2. Excessive Charging Event

The Excessive Charging event is assumed to occur by inadvertent initiation of charging flow. The event initiated from maximum pressurizer level was analyzed to assure that the operator has at least fifteen (15) minutes from initiation of a high pressurizer level alarm to take corrective action and terminate the event prior to filling the pressurizer solid.

The time required to fill the pressurizer solid was calculated using Equation 2-1.

$$T = \frac{V_S - V_{SL} - V_T}{(F_{CH} - F_{LD})\frac{\nu_2}{\nu_1}} \qquad \text{(Eq. 2-1)}$$

where:

V_S = steam volume in the pressurizer

V_{SL} = equivalent saturated liquid volume of pressurizer steam volume

V_T = volume above the spray nozzles

F_{CH} = charging flow rate

F_{LD} = letdown flow rate

ν_1 = specific volume of liquid at charging and letdown conditions

ν_2 = specific volume of liquid at pressurizer conditions

The analysis was performed for three combinations of charging and letdown flows. Table 2-1 presents the initial conditions assumed in the analysis and the results of the analysis. As seen from the table, all three combinations of charging and letdown flows analyzed provide at least fifteen minutes after initiation of high level alarm for the operator to take corrective actions and terminate the event prior to filling the pressurizer solid.

TABLE 2-1

Volume Control Assumptions		Maximum Initial Pressurizer Level Assumed in Analysis		High Level Alarm Analysis Setpoint		Time to Fill[1] Pressurizer
Charging Flow (GPM)	Letdown Flow (GPM)	Liquid Volume (ft^3)	Level[3] (in)	Liquid Volume (ft^3)	Level[3] (in)	(minutes)
1. 132	0	915	227	920	228	15
2. 132	29	975[2]	242	1040	257	15
3. 88	0	975[2]	242	1100	272	15

[1] From time of initiation of high pressurizer level alarm

[2] Maximum limit based on Loss of Load event

[3] Referenced to the 1" level nozzle at the bottom of the pressurizer

3. Excess Load Event

The Excess Load event initiated at a pressurizer level of 270 ft^3 was analyzed to evaluate the impact of reactor vessel upper head (RVUH) voiding on fuel design limits and on reactor coolant circulation. The analysis included the automatic initiation of auxiliary feedwater three minutes after initiation of reactor trip signal and a manual trip of the reactor coolant pumps (RCPs) following a safety injection actuation signal (SIAS) due to low pressurizer pressure. The RCP coastdown results in a proportionately reduced RVUH flow until natural circulation is established; at that time all flow to the RVUH is assumed to terminate.

The Excess Load event is initiated by the instantaneous opening of steam dump and bypass valves which have a combined capacity of approximately 45% of the nominal full power steam flow. This Excess Load event persists until the steam generators are isolated on steam generator isolation signal (SGIS) due to low secondary pressure. The full power event maximizes primary cooldown, shrinkage and consequently RVUH voiding.

The magnitude of RVUH voiding at full power is significantly greater than at zero power because of the higher RVUH coolant temperatures and the larger primary coolant shrinkage that occurs at the higher system temperatures. Therefore, only the full power excess load transient results are described herein.

The key parameters assumed in the analysis are given in Table 3-1.

The analysis conservatively assumed a Moderator Temperature Coefficient of $+0.5 \times 10^{-4} \Delta\rho/°F$. This MTC, in combination with decreasing coolant temperatures, inserts negative reactivity and causes the core power to decrease. The decreasing core power does not allow either the High Power trip or TM/LP trip to be initiated, and thus the time of reactor trip is delayed until a low pressurizer pressure trip (i.e., floor of the TM/LP trip) is generated. The longer time required to initiate reactor trip causes the pressurizer to drain and thus maximizes RVUH voiding.

The analysis conservatively assumed that all three charging pumps were inoperable and that one High Pressure Safety Injection (HPSI) pump fails to start on SIAS due to low pressurizer pressure.

The effect of auxiliary feedwater was explicitly evaluated by analyzing the event both with and without auxiliary feedwater initiated three minutes after reactor trip signal is generated. An auxiliary feedwater flow of 172 lbm/sec to each steam generator is conservatively assumed (i.e., 10.5% of full power main feedwater flow per generator). The maximum auxiliary feedwater flow causes the fastest primary cooldown and thus enhances the bubble formation in the upper head.

This analysis shows that some RVUH voiding will occur as a result of the RCS depressurization caused by an Excess Load transient. Voiding the RVUH starts when RCS pressure control is lost due to the primary coolant shrinkage which drains the pressurizer. During the period of RVUH voiding, RCS pressure is controlled by the saturation pressure within the upper head. This reduces the RCS depressurization in the latter part of the transient as compared to analyses which do not explicitly model RVUH voids.

RVUH coolant temperatures initially follow the core outlet temperature and therefore decrease immediately following a reactor trip. The decrease in RVUH coolant flow reduces the convective heat transfer and inhibits upper head cooldown. This leads to elevated RVUH coolant temperatures which raise the upper head saturation pressure and therefore increase RCS pressure during the period of voiding. The increased RCS pressure does not adversely impact the approach to any SAFDL; however, safety injection flow is decreased. The reduced safety injection flow does not result in a return to criticality; however, the decreased flow diminishes the mitigating effect of safety injection on coolant shrinkage and therefore enhances voiding. Subsequent RVUH cooldown is accomplished through an exchange of coolant between the RVUH and the core outlet plenum. This exchange of coolant is driven by the expansion and contraction of the steam bubble. Additional RVUH cooling is accomplished through heat conduction across the upper guide structure.

At the time of maximum RVUH voiding approximately 63% of the head is occupied by steam. Since this steam bubble does not expand beyond the upper head, primary coolant circulation is unaffected. Tables 3-2 and 3-3 present the sequence of events for the event initiated without and with auxiliary feedwater flow. Figures 3-1 through 3-14 present the transient behavior of the system variables during the event. The analysis demonstrates that the addition of auxiliary feedwater prolongs the duration of RVUH voiding and delays repressurization of the RCS. However, since auxiliary feedwater is delivered after the time of maximum voiding, the peak void fraction is unchanged.

In conclusion, the potential RVUH voiding associated with an Excess Load transient initiated from a pressurizer level of 270 ft^3 does not extend beyond the upper head and therefore will not affect primary coolant circulation. In addition, approach to SAFDLs are not impacted by RVUH voiding. The results of the analysis also shows that the NSSS achieves stable conditions and that shutdown cooling procedures can be initiated if deemed necessary.

TABLE 3-1

KEY PARAMETERS ASSUMED IN THE EXCESS LOAD EVENT ANALYSIS

Parameter	Units	Value
Initial Core Power Level	MWt	2754
Core Inlet Temperature	°F	550
Reactor Coolant System Pressure	psia	2300
Core Mass Flow Rate	$x10^6$ lbm/hr	133.6
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho$	+.5
CEA Worth Available at Trip	%$\Delta\rho$	-4.7
Auxiliary Feedwater Flow Rate	lbm/sec	172.0/S.G.
Low Pressurizer Pressure Analysis Trip Setpoint	psia	1728
SIAS Analysis Setpoint	psia	1556
SGIS Analysis Setpoint	psia	548

TABLE 3-2

SEQUENCE OF EVENTS FOR THE EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER

Time (sec)	Event	Setpoint or Value
0.0	Steam Dump and Bypass Valves Fully Open	---
24.2	Pressurizer Empties	---
28.9	Pressurizer Pressure Trip Setpoint Reached	1728 psia
29.8	Trip Breakers Open	
30.3	CEAs Begin to Drop Into Core Feedwater Starts Rampdown	---
32.6	SIAS is Initiated Reactor Coolant Pumps Manually Tripped	1556 psia
67.1	SGIS is Generated	548 psia
68.0	Main Steam Isolation Valves Begin to Close	---
80.0	Main Steam Isolation Valves are Closed	---
90.3	Feedwater Rampdown to 5% is Completed	
107.6	Maximum RVUH Void is Reached	63%
210.3*	Main Feedwater Isolated	
521.6	Minimum RCS Pressure	728.50
717.8	Upper Head Void is Zero Pressurizer Starts to Refill	

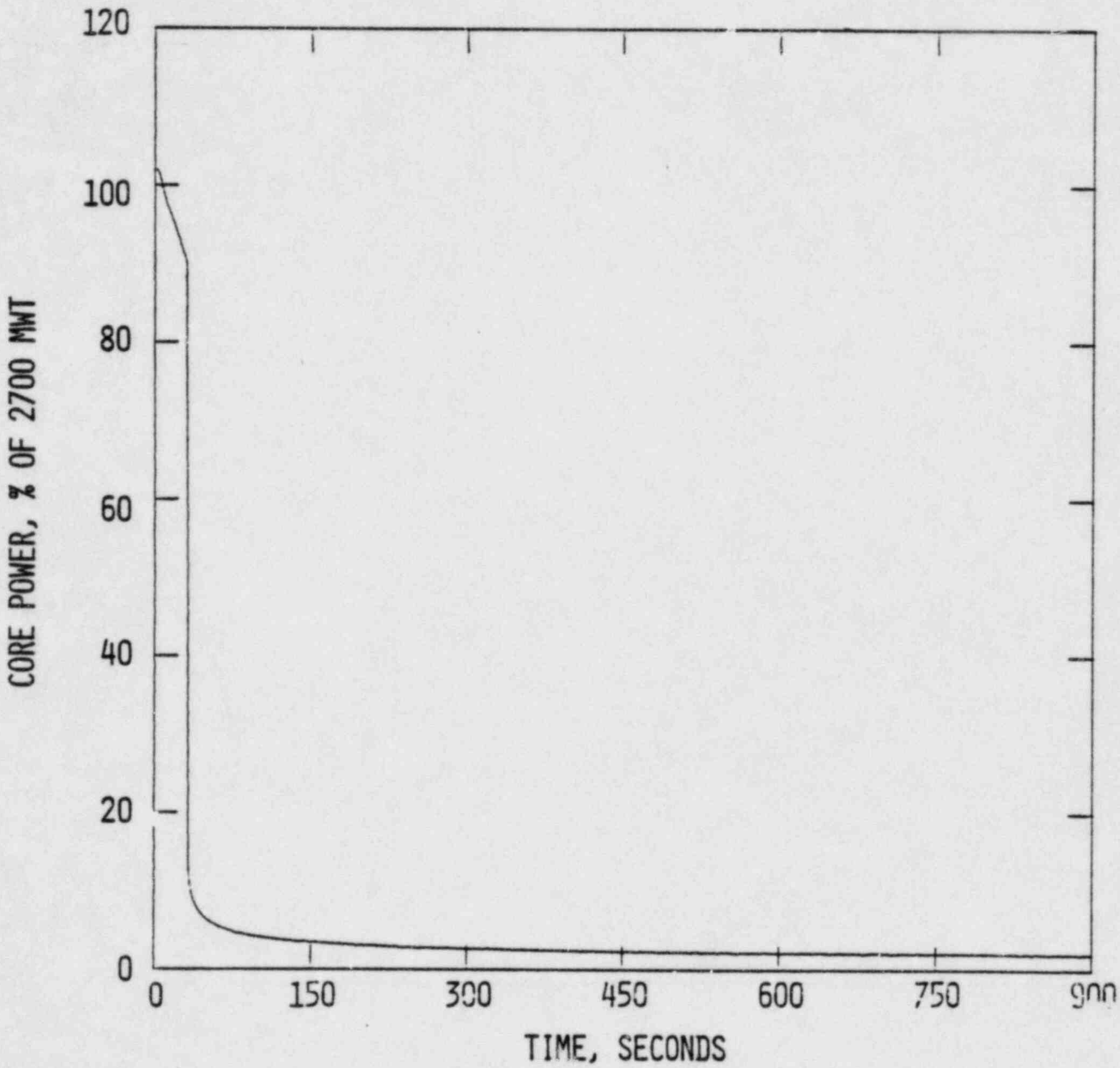
*Main feedwater would have been isolated 80 seconds after SGIS is initiated (i.e., at 147.1 seconds). The analysis conservatively assumed that main feedwater is isolated at 210.3 seconds to prolong the duration of RVUH voiding.

TABLE 3-3

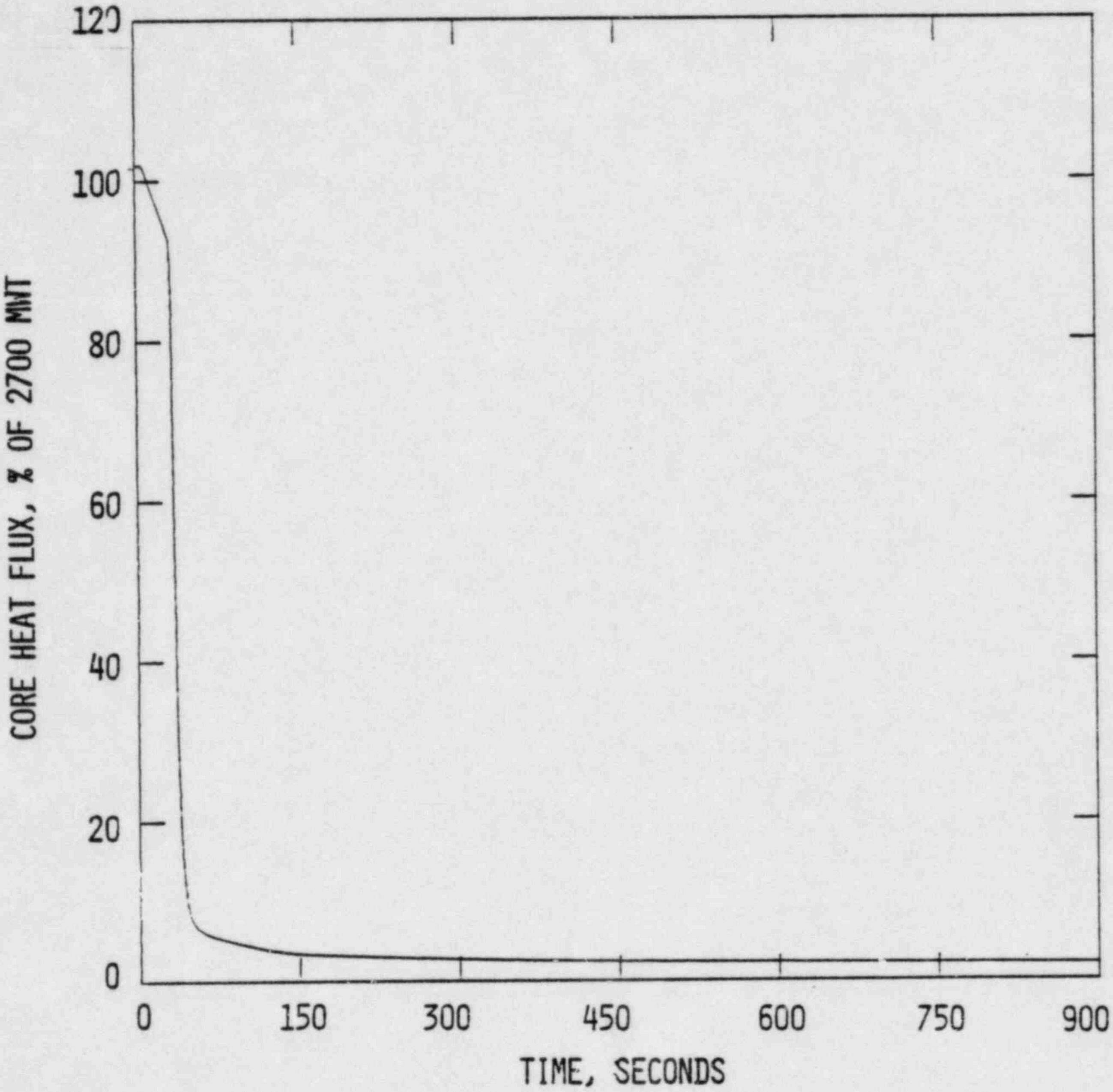
SEQUENCE OF EVENTS FOR THE EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER

Time (sec)	Event	Setpoint or Value
0.0	Steam Dump and Bypass Valves Fully Open	---
24.2	Pressurizer Empties	---
28.9	Low Pressurizer Pressure Trip Setpoint is Reached	1728 psia
29.8	Trip Breakers Open	
30.3	CEAs Begin to Drop Into Core Feedwater Starts Rampdown Turbine Valves Begin to Close	---
32.6	SIAS is Initiated Reactor Coolant Pumps Manually Tripped	1556 psia
67.1	SGIS is Generated	548 psia
68.0	Main Steam Isolation Valves Begin to Close	
80.0	Main Steam Isolation Valves are Closed	
90.3	Feedwater Rampdown to 5% is Completed	
107.6	Maximum RVUH Void is Reached	63%
210.3*	Main Feedwater Isolated Auxiliary Feedwater is Initiated	172 lbm/S.G.
900.0	Operator Action To Isolate Auxiliary Feedwater to Steam Generators	

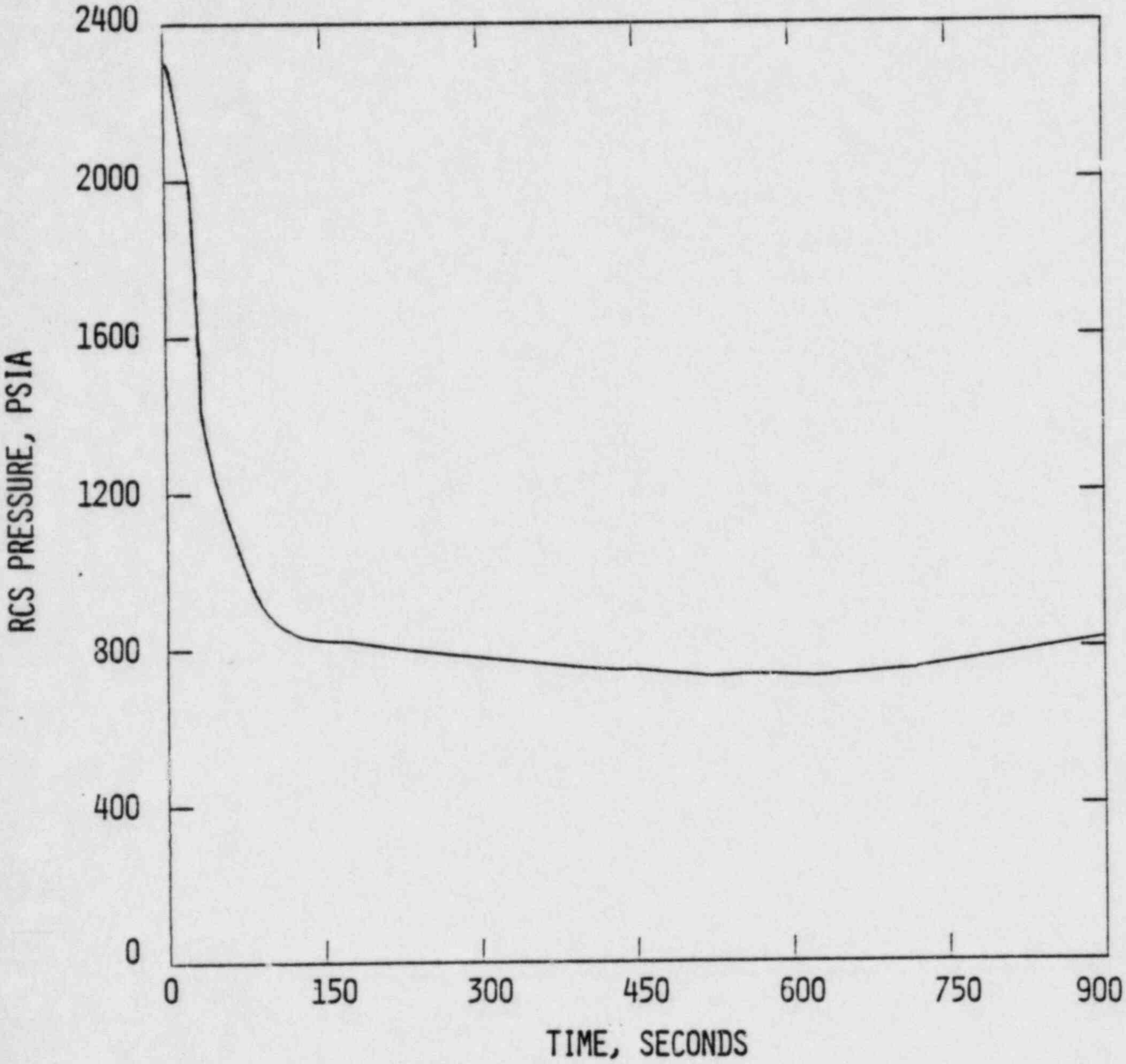
*Main Feedwater would have been isolated 80 seconds after SGIS is initiated (i.e., at 147.1 seconds). The analysis conservatively assumed that main feedwater is isolated at 210.3 seconds to prolong the duration of RVUH voiding.



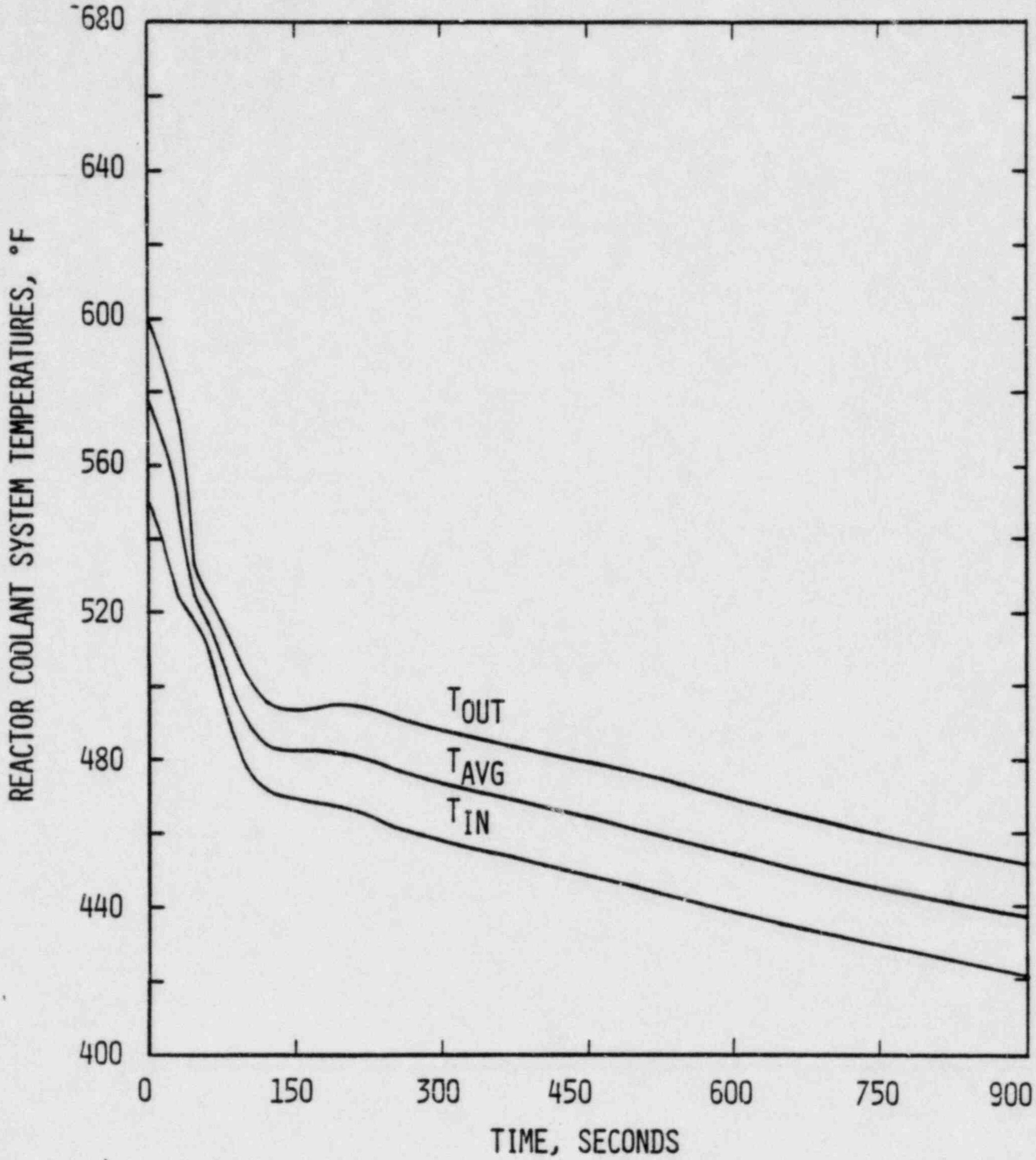
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER CORE POWER VS TIME	FIGURE 3-1



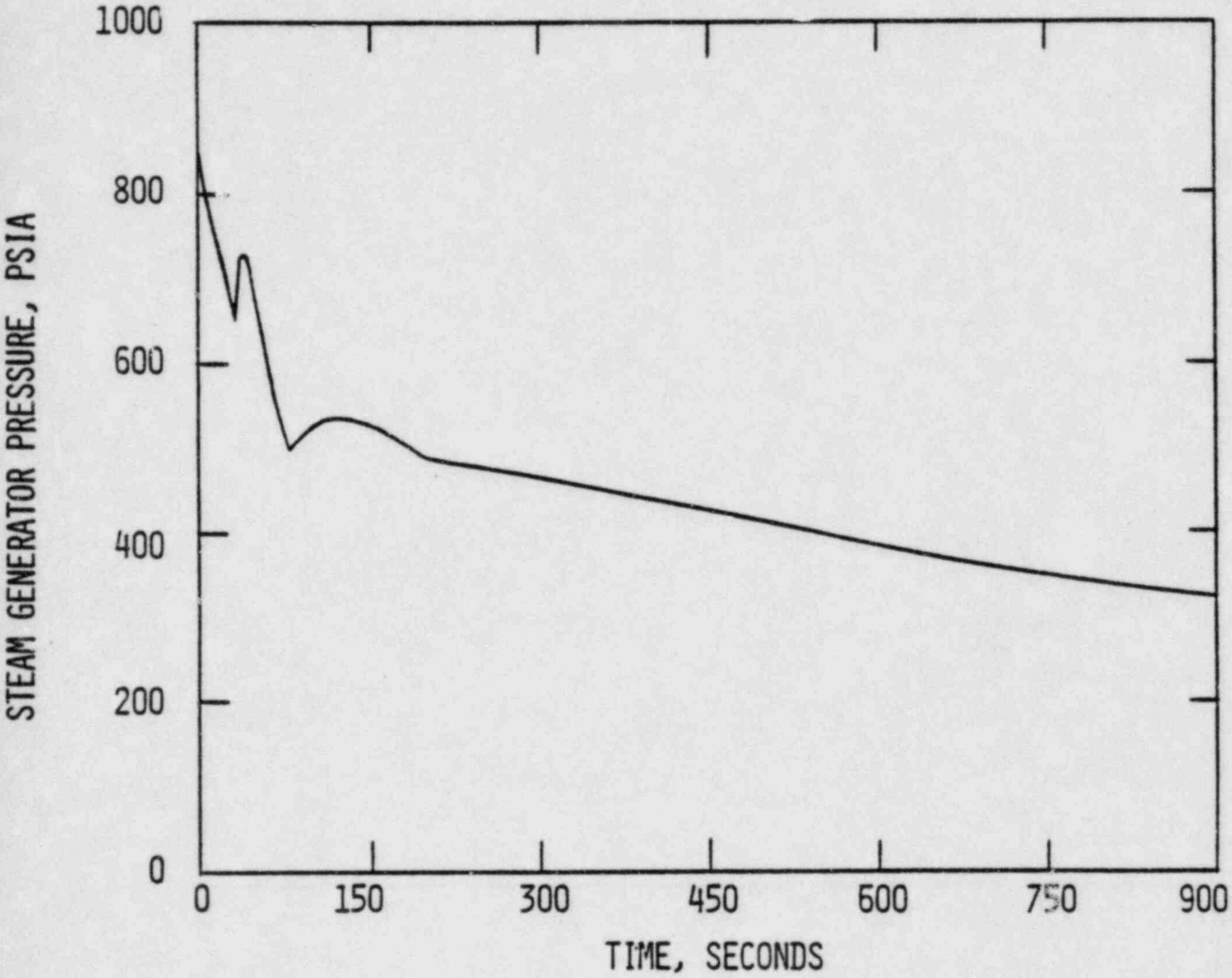
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER CORE HEAT FLUX VS TIME	FIGURE 3-2



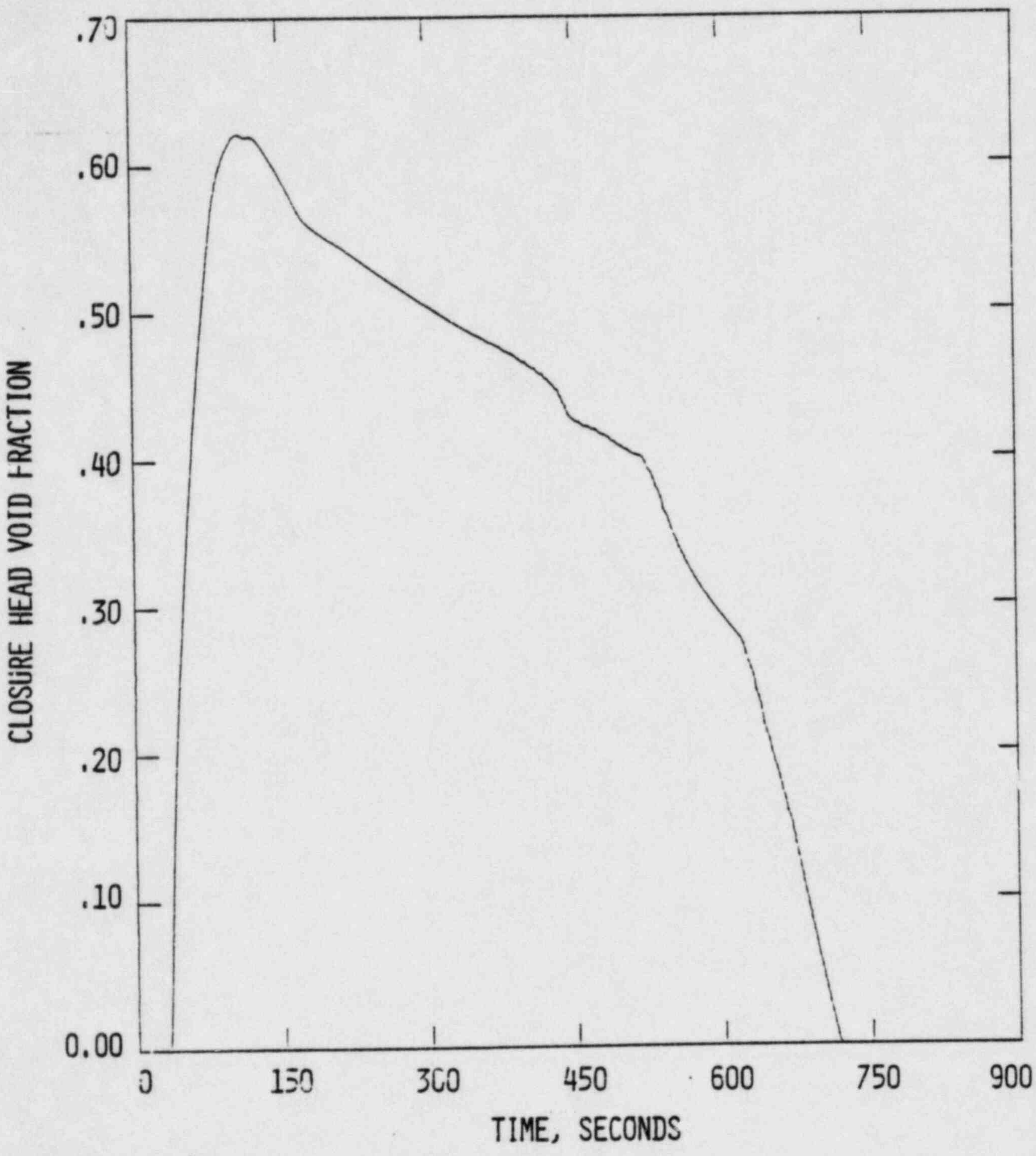
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER RCS PRESSURE VS TIME	FIGURE 3-3



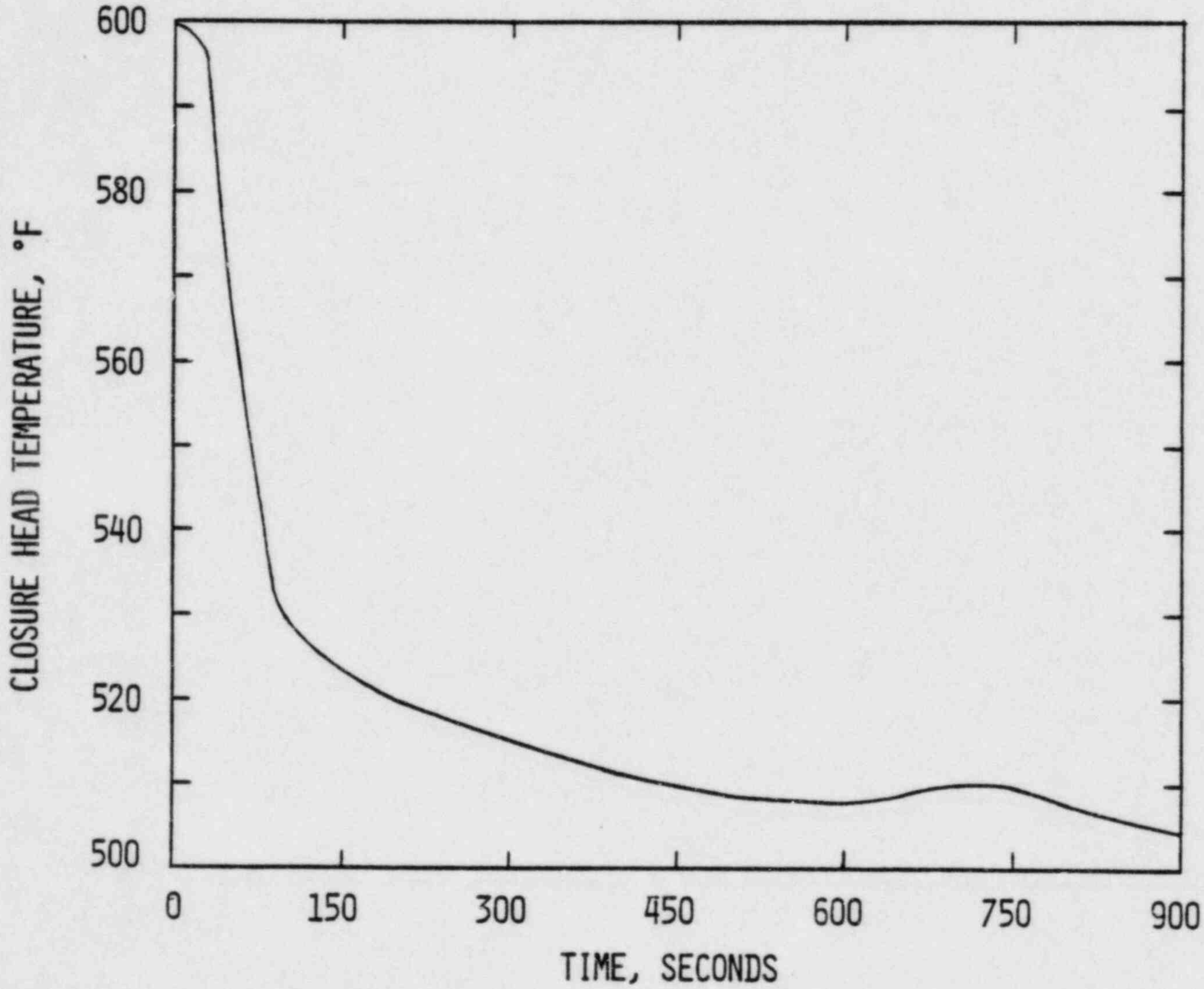
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 3-4



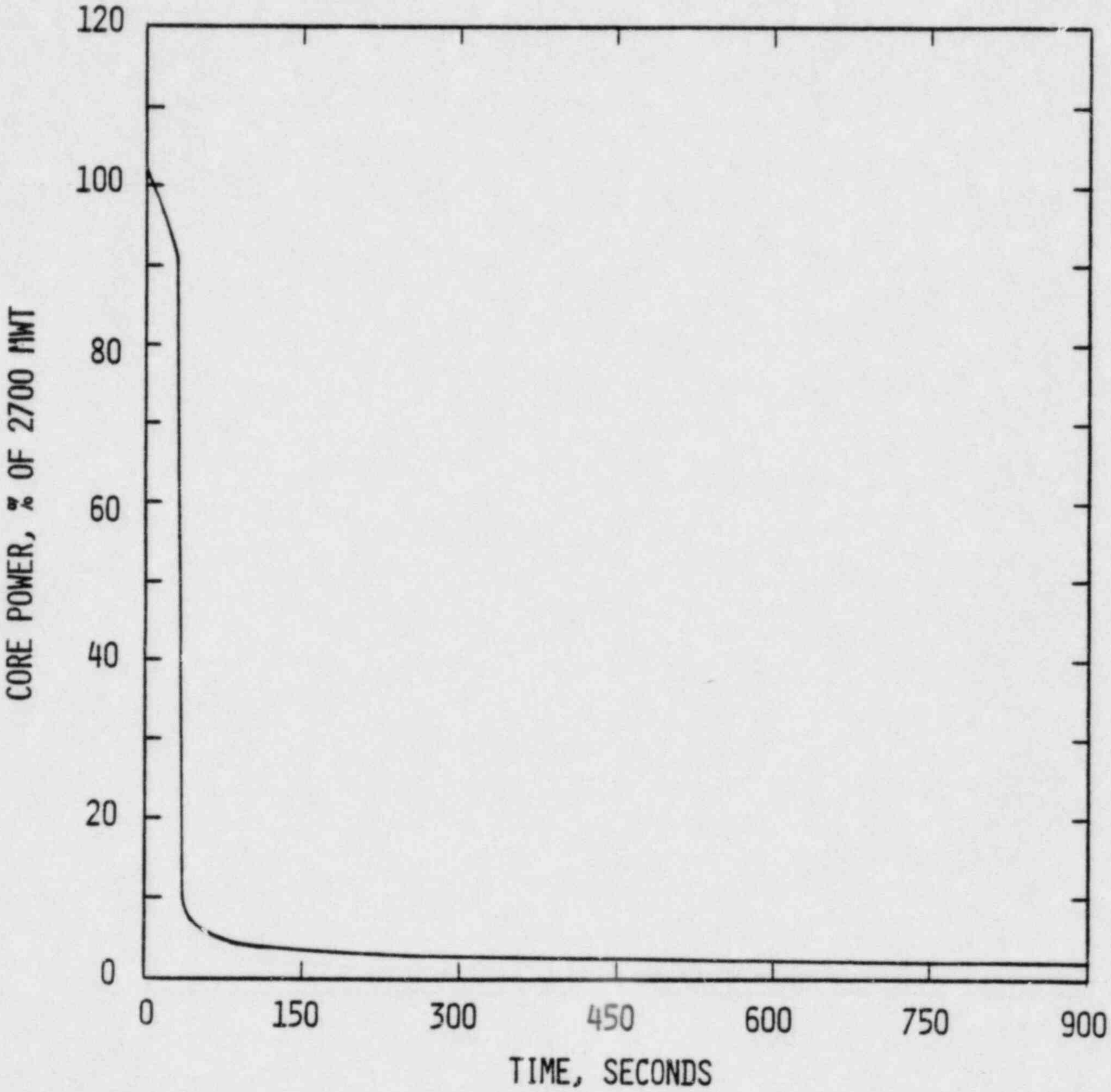
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER STEAM GENERATOR PRESSURE VS TIME	FIGURE 3-5



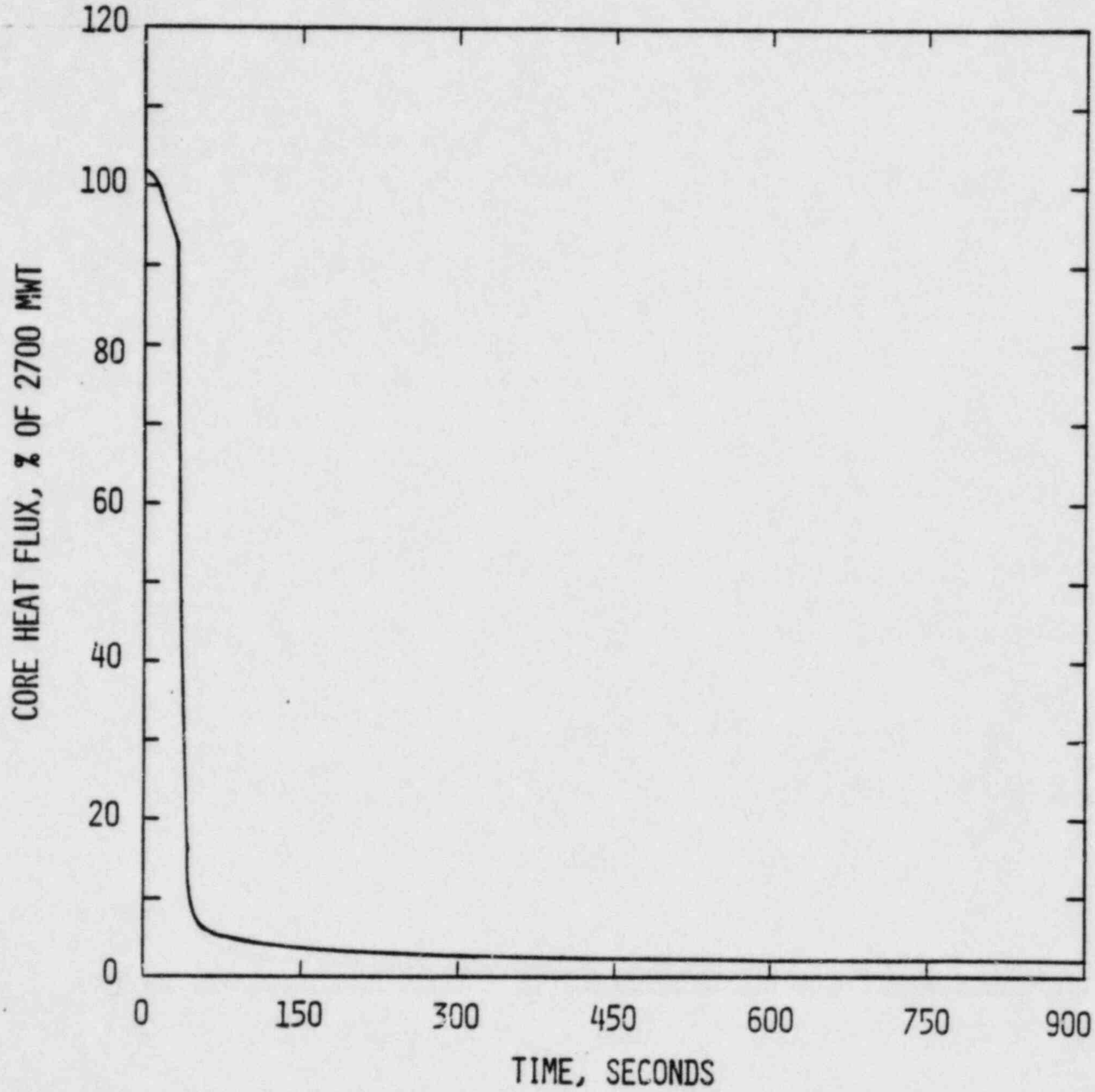
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER CLOSURE HEAD VOID FRACTION VS TIME	FIGURE 3-6



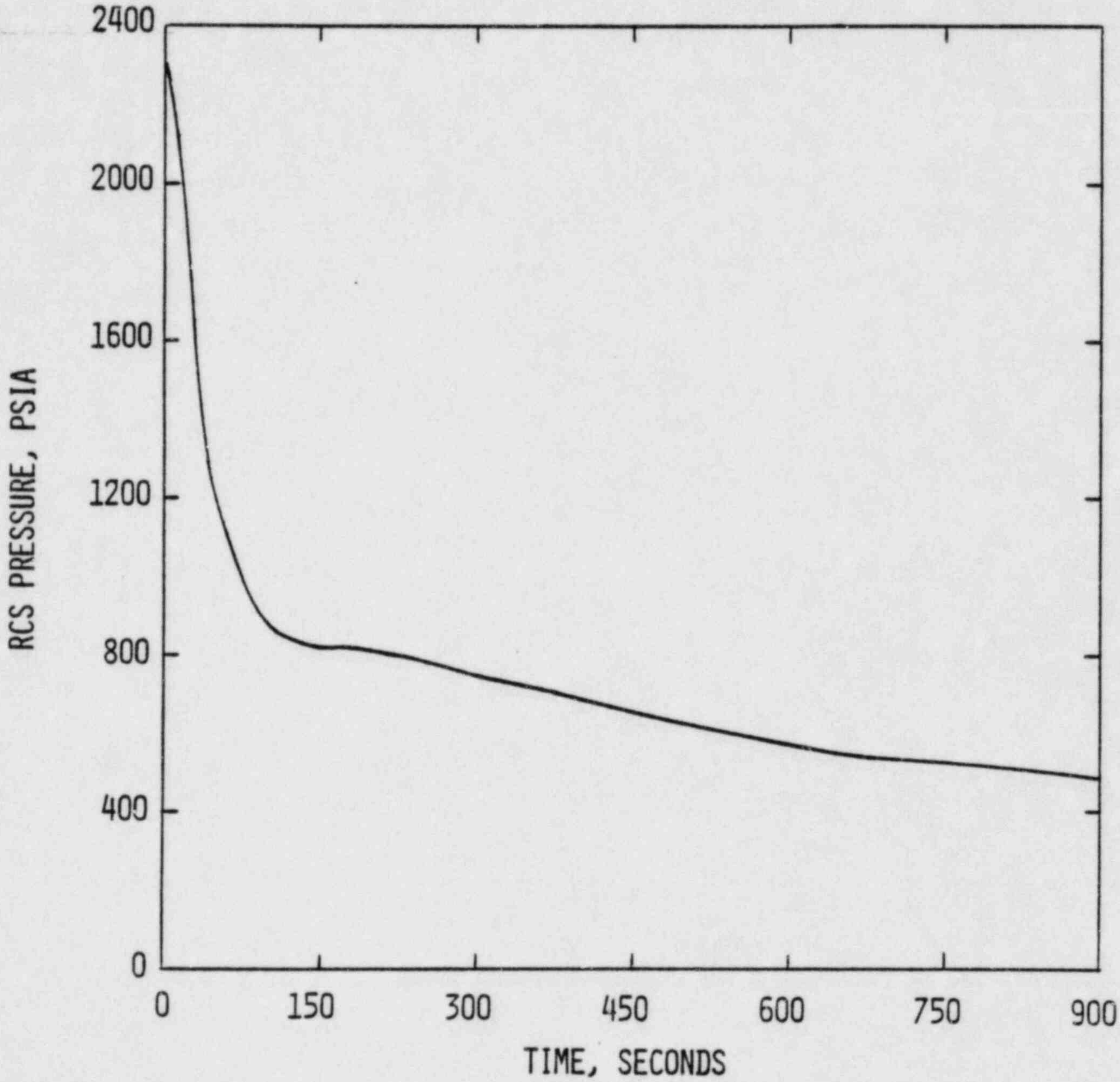
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITHOUT AUXILIARY FEEDWATER CLOSURE HEAD TEMPERATURE VS TIME	FIGURE 3-7



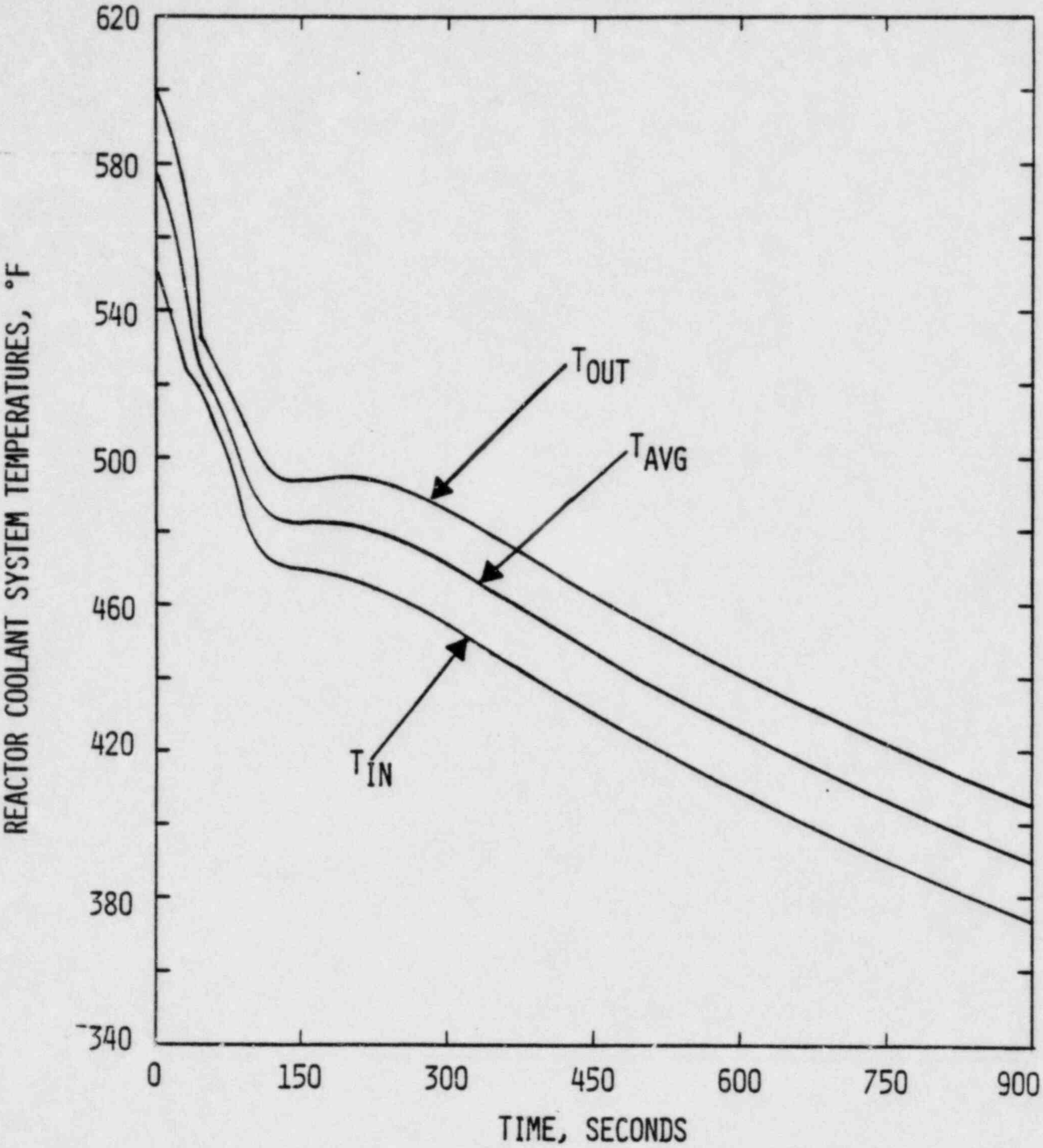
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER CORE POWER VS TIME	FIGURE 3-8



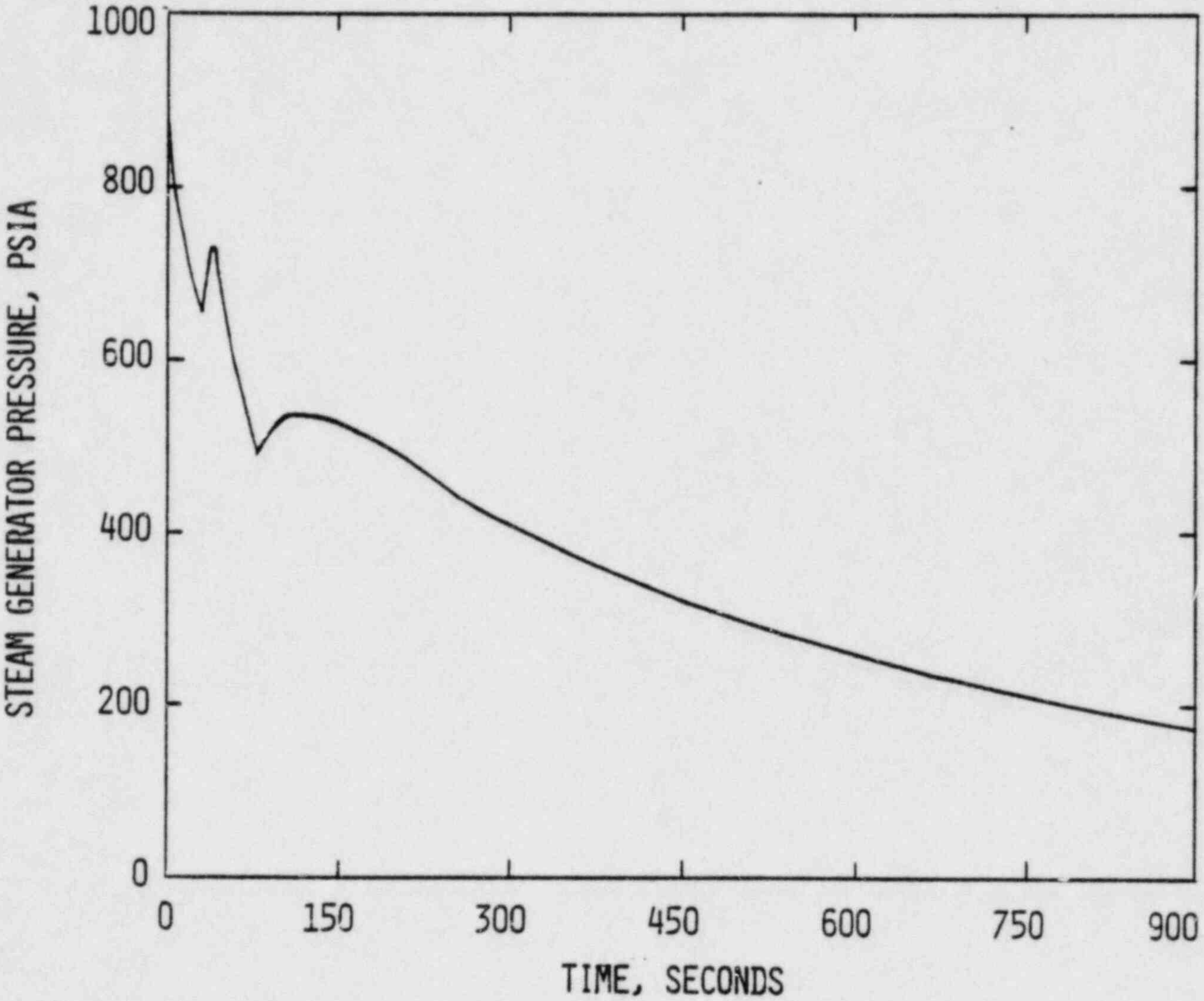
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER CORE HEAT FLUX VS TIME	FIGURE 3-9



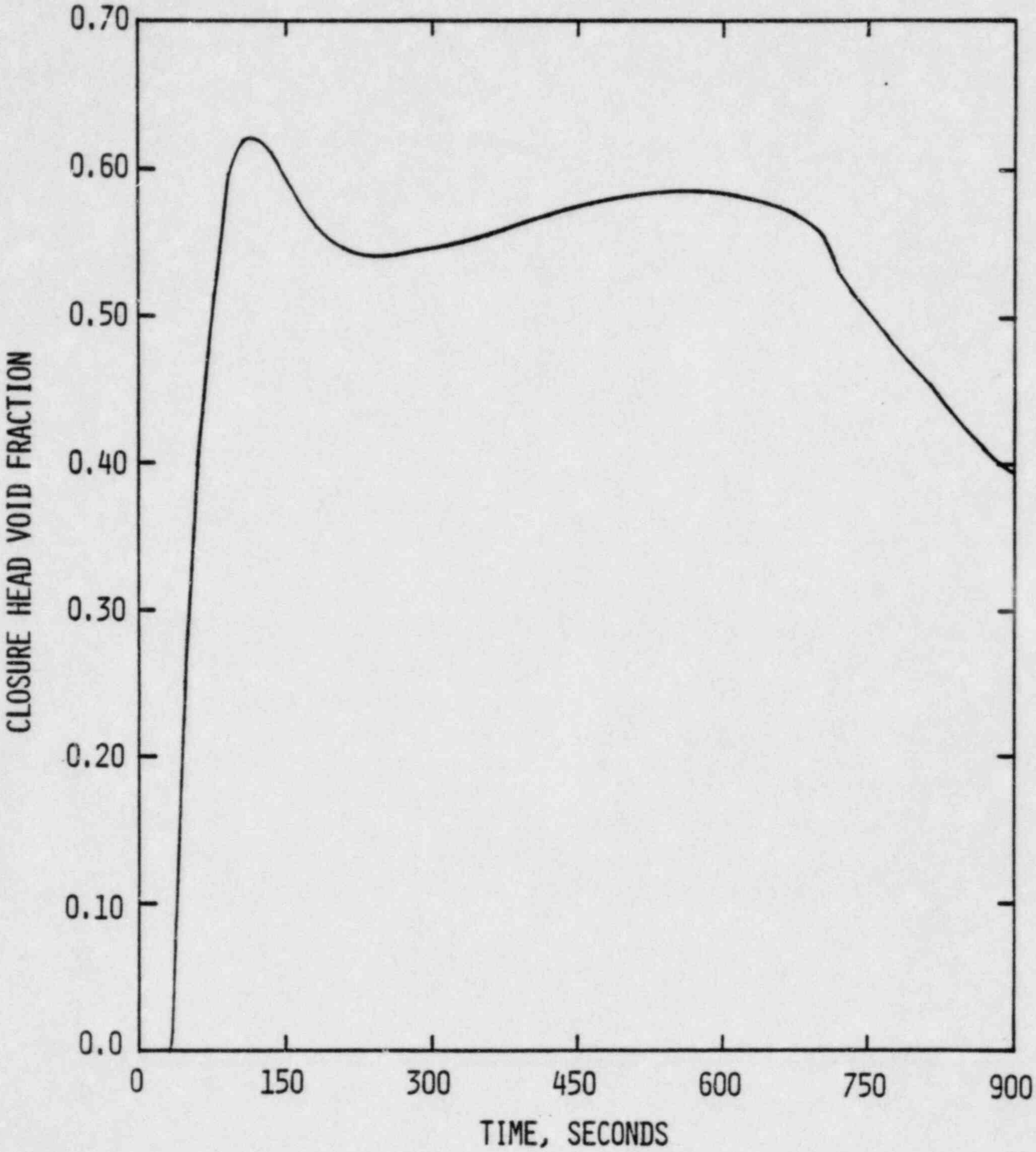
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER RCS PRESSURE VS TIME	FIGURE 3-10



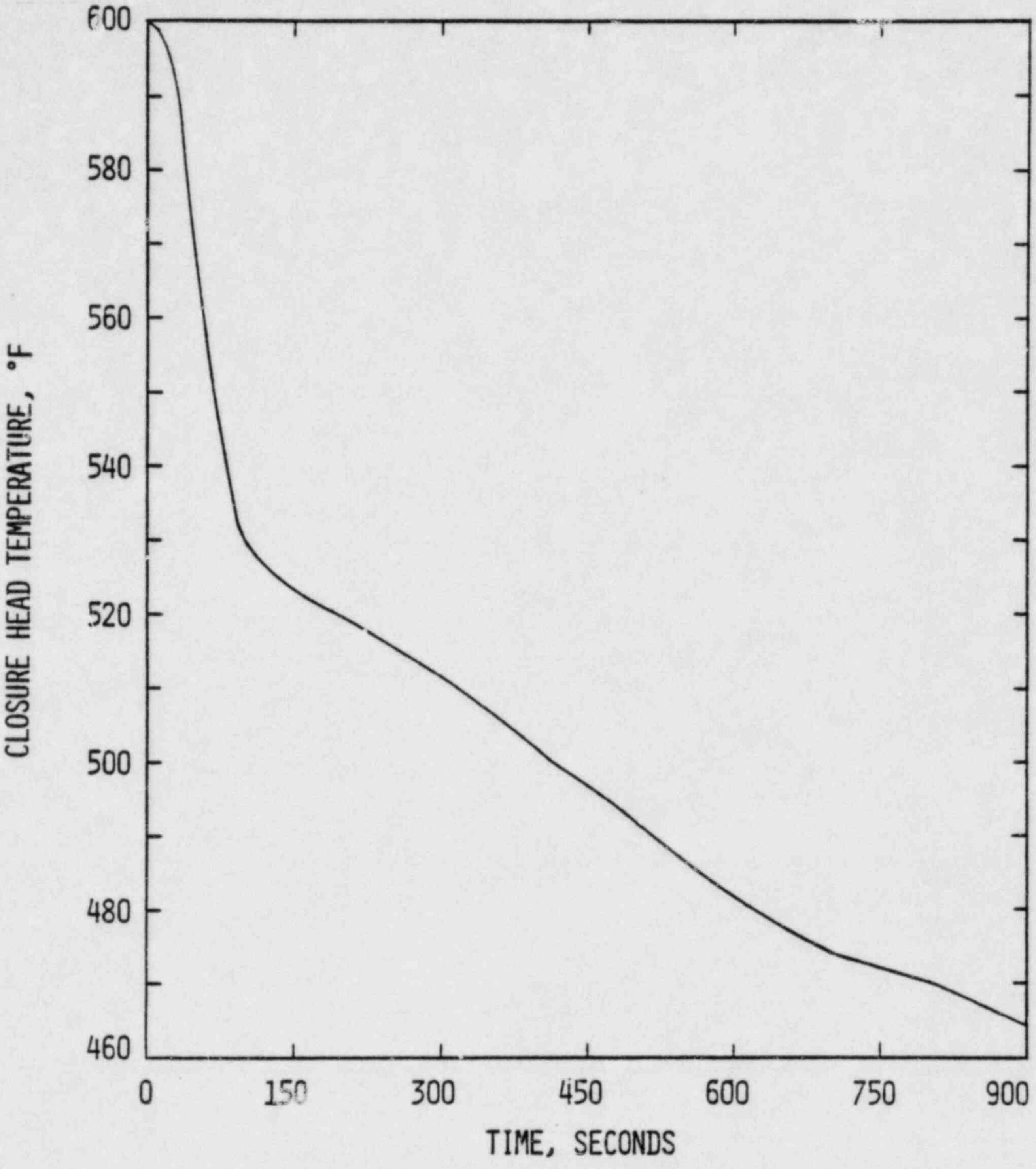
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 3-11



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER STEAM GENERATOR PRESSURE VS TIME	FIGURE 3-12



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER CLOSURE HEAD VOID FRACTION VS TIME	FIGURE 3-13



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	EXCESS LOAD EVENT WITH AUXILIARY FEEDWATER CLOSURE HEAD TEMPERATURE VS TIME	FIGURE 3-14

4. <u>Steam Generator Tube Rupture Event</u>

The Steam Generator Tube Rupture (SGTR) event was analyzed tc verify that the site boundary doses will not exceed the guidelines of 10CFR100 for the event initiated from a pressurizer level of 975 ft^3.

The analysis included the effects of manually tripping the Reactor Coolant Pumps on SIAS due to low pressurizer pressure.

The design basis S TR is a double ended break of one steam generatur U-tube. Table 4-1 lists the key transient related parameters used in this analysis. In the analysis, it is assumed that the initial RCS pressure is as high as 2300 psia. This initial RCS pressure maximizes the amount of primary coolant transported to the secondary steam system since the leak rate is directly proportional to the difference between the primary and secondary pressure. In addition, the higher pressure delays the lower pressurizer pressure trip which prolongs the transient and therefore maximizes the total primary to secondary mass and activities transported.

For this event, the acceptable DNBR limit is not exceeded due to the action of the Thermal Margin/Low Pressure (TM/LP) trip which provides a reactor trip to maintain the DNBR above 1.23. Since the SGTR event does not significantly affect the core power distribution, the PLHGR SAFDL is not approached.

The Thermal Margin/Low Pressure trip, with conservative coefficients which account for the limiting radial and axial peaks, maximum inlet temperature, RCS pressure, core power, and conservative CEA scram characteristics, would be the primary RPS trip intervening during the course of the transient. However, to maximize the coolant transported from the primary to the secondary and thus the radioactive steam releases to the atmosphere, the analysis was performed assuming the reactor does not trip until the minimum setpoint (floor) of the Thermal Margin/Low Pressure trip is reached. This prolongs the steam releases to the atmosphere and thus maximizes the site boundary doses.

The Steam Generator Tube Rupture was analyzed assuming a manual trip of reactor coolant pumps on Safety Injection Actuation Signal (SIAS). The Steam Generator Tube Rupture (SGTR) with RCS trip on SIAS results in higher site boundary doses because: (1) RCP coastdown increases pressure difference between the primary and the secondary, which increases the leak rate, and (2) RCP coastdown decreases the rate of decay heat removal, which increases the steam flow through the atmospheric dump valves.

The Sequence of Events for the SGTR event with manual trip of RPC on SIAS is presented in Table 4-2. Figures 4-1 through 4-5 present the transient behavior of core power, heat flux, RCS pressure, RCS temperatures, and steam generator pressure.

I-131 activity release is based on the primary to secondary leak and on the steam flow required to reach cold shutdown conditions. This release is calculated as the product of steam flow, the time dependent steam activity and the decontamination factors applicable to each release pathway.

The 0 to 2 hour I-131 site boundary does, is calculated from:

$$\text{DOSE(REM)} = A^{I-131} * BR \times \frac{X}{Q} \times CF^{I-131}$$

where:

A^{I-131} = I-131 activity released

BR = breathing rate

χ/Q = dispersion coefficient

CF^{I-131} = I-131 dose conversion factor

In determining the whole body dose, the major assumption made is that all noble gases leaked through the ruptured tube will be released to the atmosphere. Therefore, the whole body dose is proportional to the total primary to secondary leak and is calculated using the following equation.

$$\text{Whole Body Dose} = [.25\ (\overline{E}_\gamma + \frac{.23}{.25}\overline{E}_\beta)] * L * A_{RCS} * \frac{\chi}{Q}$$

where:

$\overline{E}_\gamma$ = average energy release by gamma decay

$\overline{E}_\beta$ = average energy release by beta decay

L = total primary to secondary mass transport

A_{RCS} = noble gas activity of primary coolant

χ/Q = dispersion coefficient

The results of the analysis are that 85616 lbs. of primary coolant are transported to the steam generator secondary side. Based on this mass transport and values in Table 4-3, the site boundary doses calculated are:

Thyroid (DEQ I-131) = 0.34 REM
Whole Body (DEQ Xe-133) = 0.18 REM

The reactor protective system (TM/LP) is adequate to protect the core from exceeding the DNBR limit. The doses resulting from the activity released as a consequence of a double-ended rupture of one steam generator tube, assuming the maximum allowable Tech Spec activity for the primary concentration at a core power of 2754 MWt, are significantly below the guidelines of 10CFR100.

TABLE 4-1

KEY PARAMETERS ASSUMED IN THE STEAM GENERATOR TUBE RUPTURE EVENT

KEY TRANSIENT RELATED PARAMETERS:

Parameter	Units	Value
Power	MWt	2754
MTC	$x10^{-4} \Delta\rho/°F$	-2.5
Doppler Coefficient Multiplier		1.15
Scram Worth	$\%\Delta\rho$	-4.7
T_{in}	°F	550
RCS Pressure	psia	2300
Initial Core Mass Flow Rate	$x10^{6}$ lb/hr	133.9
Initial Secondary Pressure	psia	810
Tube ID	inches	.654
Flow Constant		1.17
ASI (for scram)		+.41

TABLE 4-2

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT WITH RCP COASTDOWN ON SIAS

Time (sec)	Event	Setpoint or Value
0.0	Tube Rupture Occurs	---
786.5	Low Pressurizer Pressure Trip Signal Generated	1728 psia
787.6	Dump Valves Open	---
787.9	CEAs Begin to Drop Into Core	---
791.3	Pressurizer Empties	---
792.6	Safety Injection Actuation Signal Generated, RCP's Manually Tripped	1556 psia
795.5	Bypass Valves Open	---
797.1	Maximum Steam Generator Pressure	906 psia
854.0	Minimum RCS Press[illegible]	1064 psia
1800.0	Operator Isolates Damaged Steam Generator and Begins Cooldown to 300°F	---
12797.0	Operator Initiates Shutdown Cooling (T_{AV} = 300°F)	

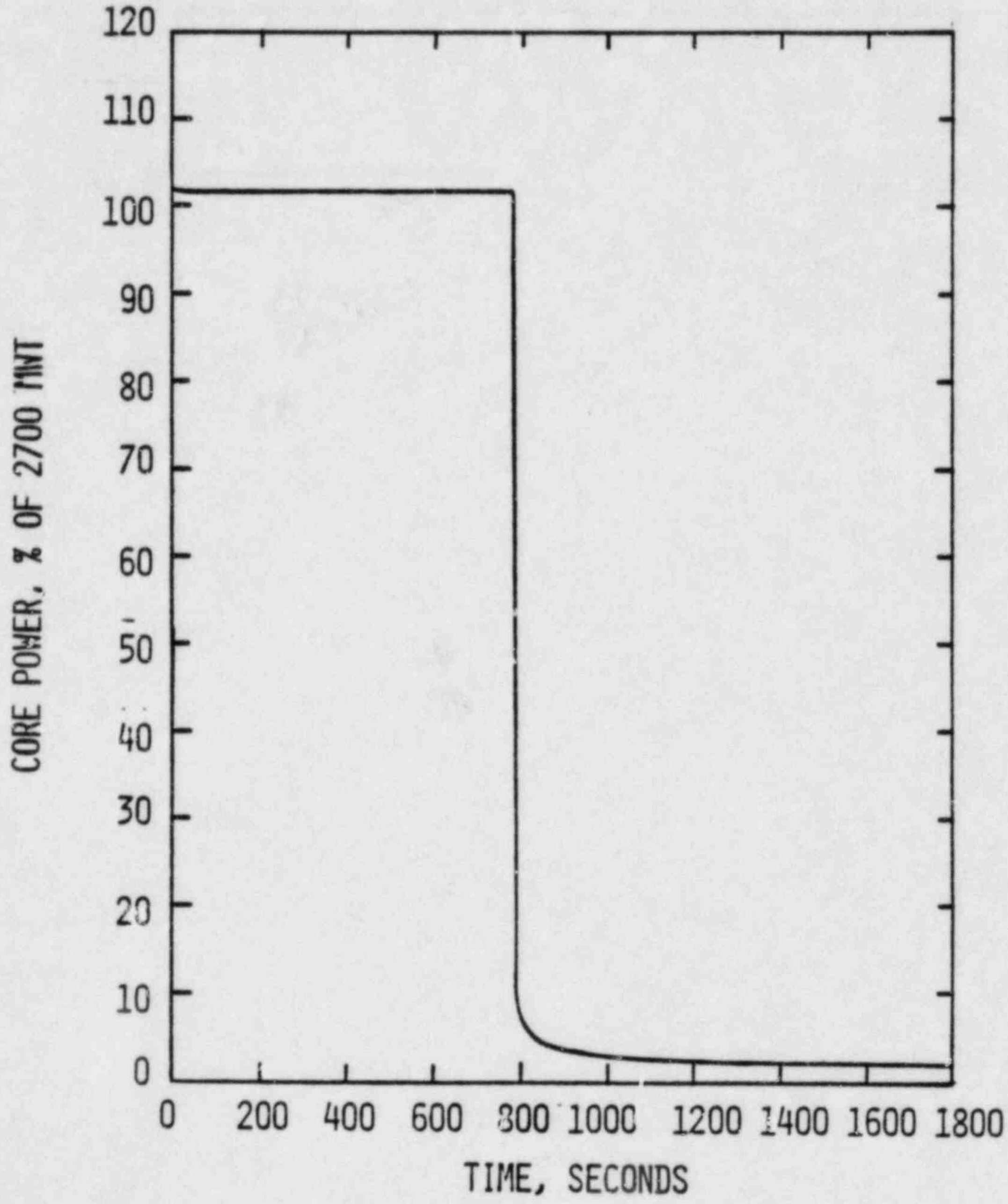
TABLE 4-3

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE STEAM GENERATOR TUBE RUPTURE

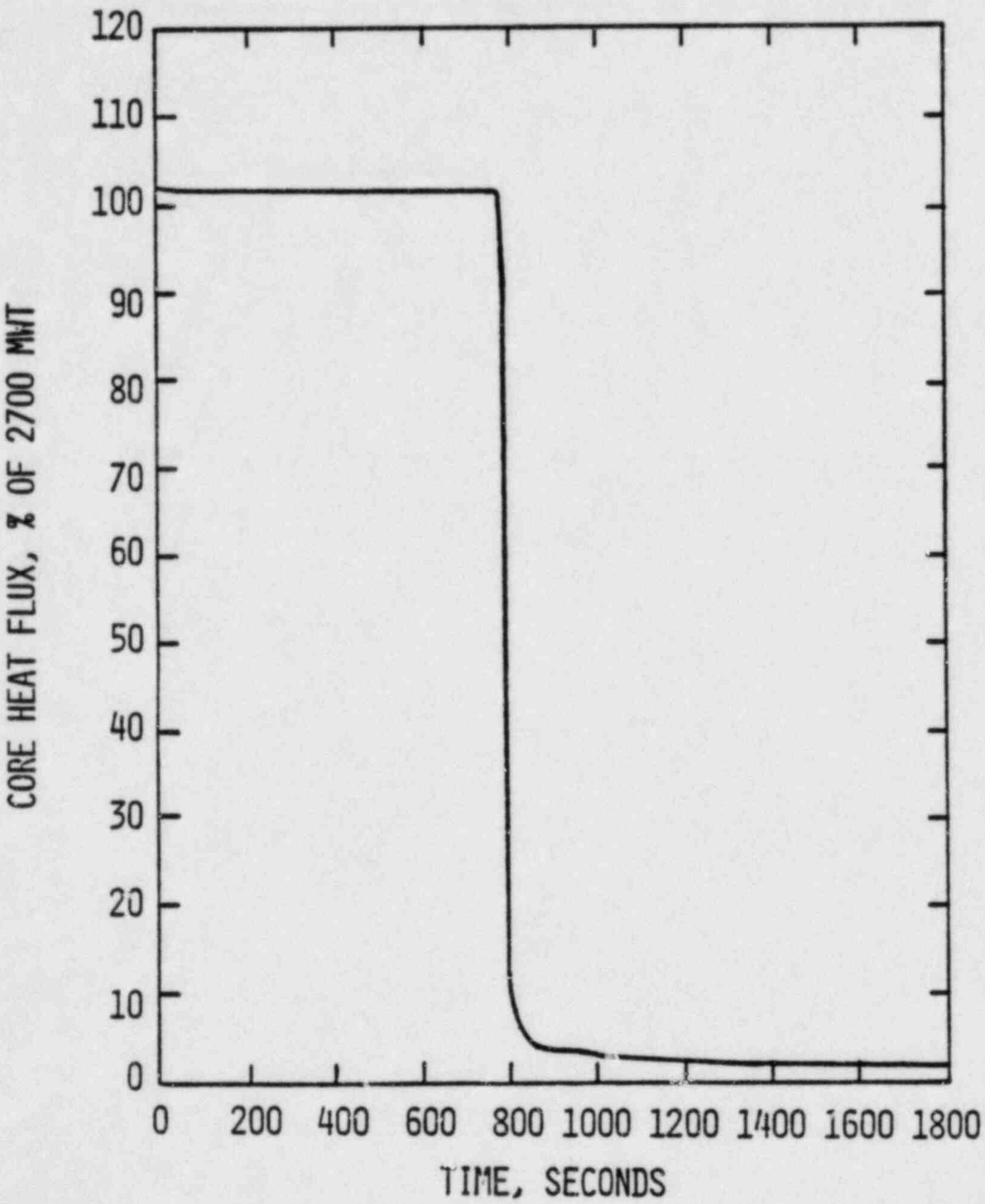
Parameter	Units	Value
Reactor Coolant System Maximum Allowable Concentration (DEQ I-131)[1]	µCi/gm	1.0
Steam Generator Maximum Allowable Concentration (DEQ I-131)[1]	µCi/gm	.1
Reactor Coolant System Maximum Allowable Concentration of Noble Gases (DEQ Xe-133)[1]	µCi/gm	$100/\bar{E}$
Steam Generator Partition Factor	---	.1
Air Ejector Partition Factor	---	.0005
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

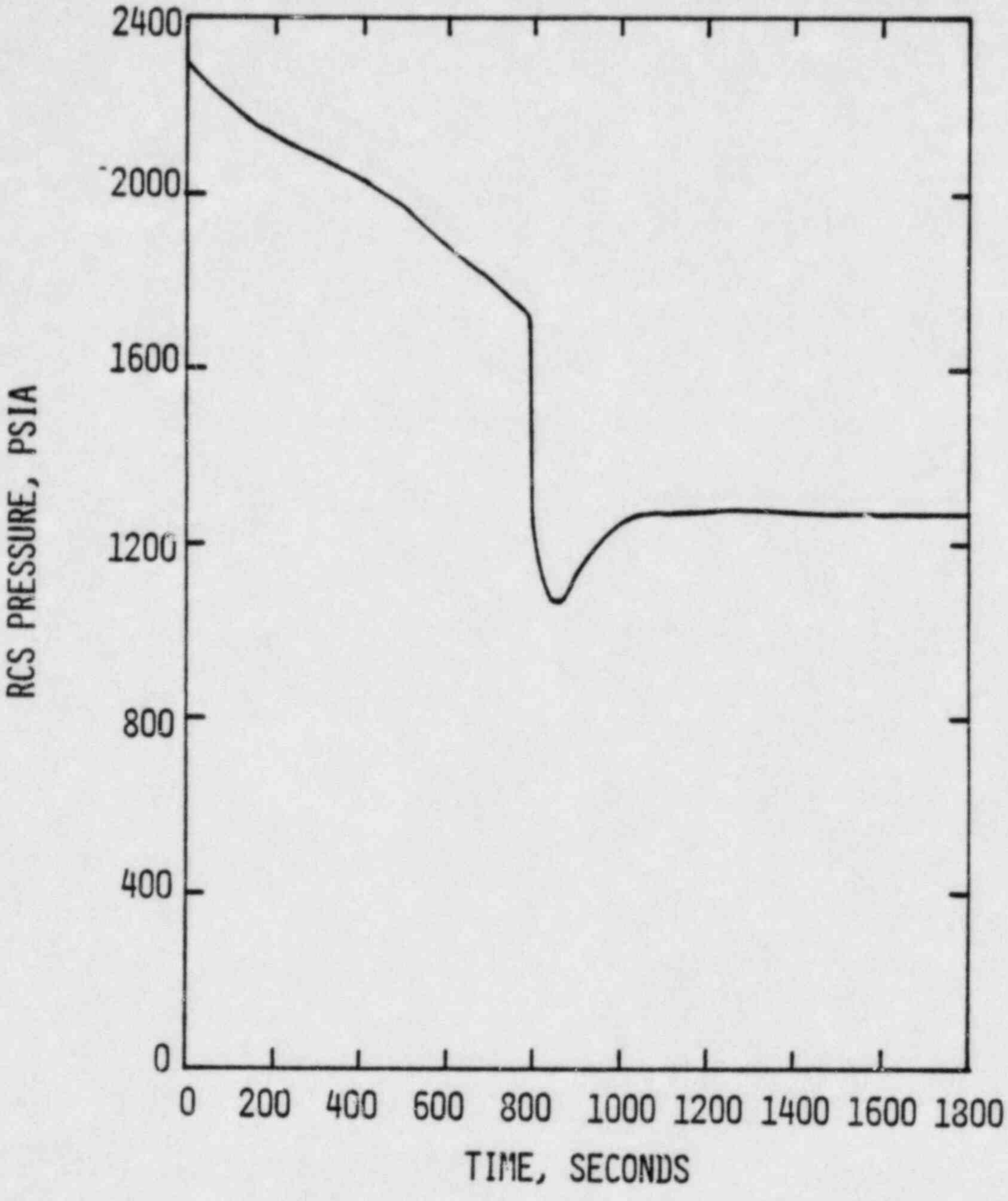
[2]0-2 hour accident condition



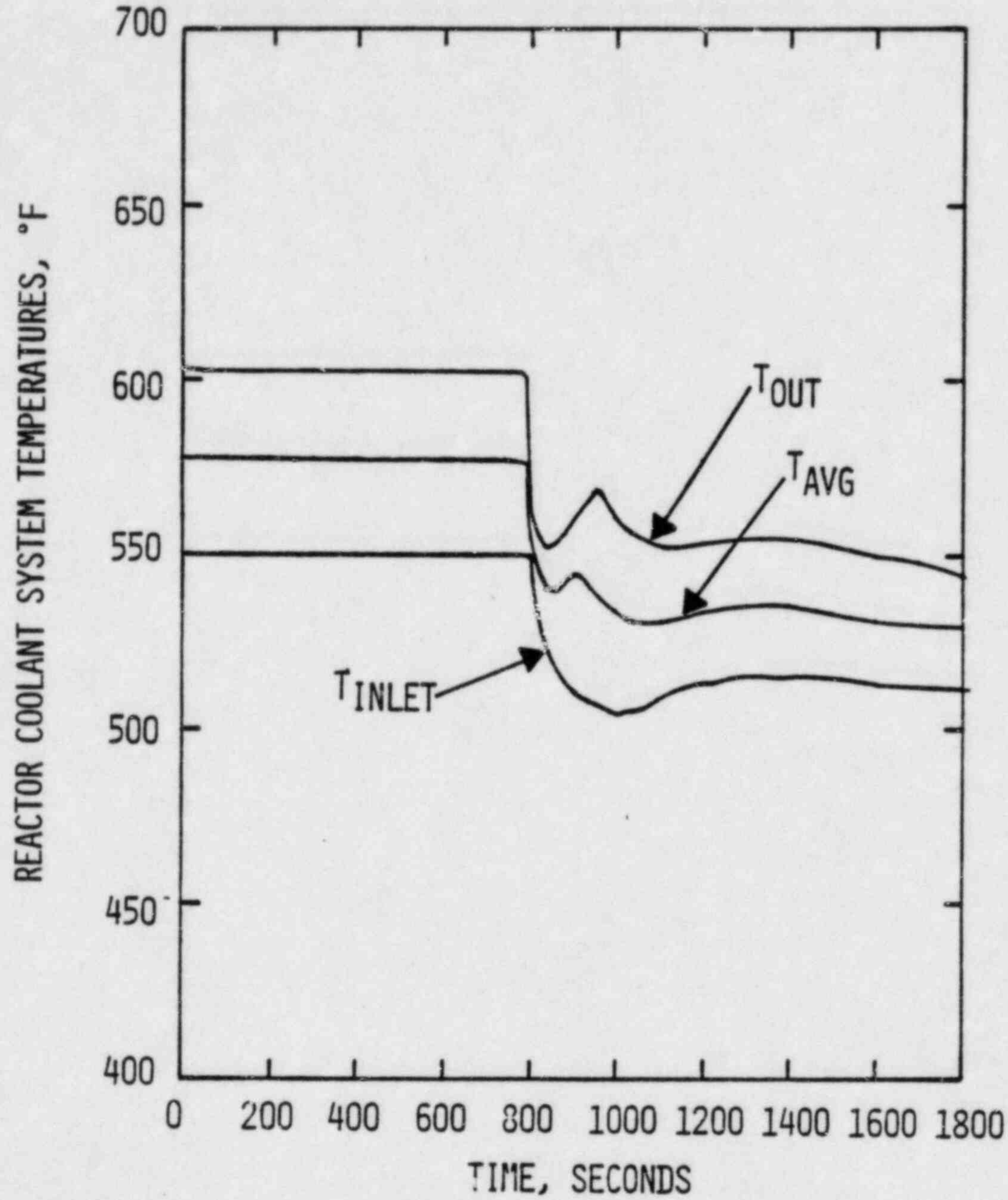
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT CORE POWER VS TIME	FIGURE 4-1



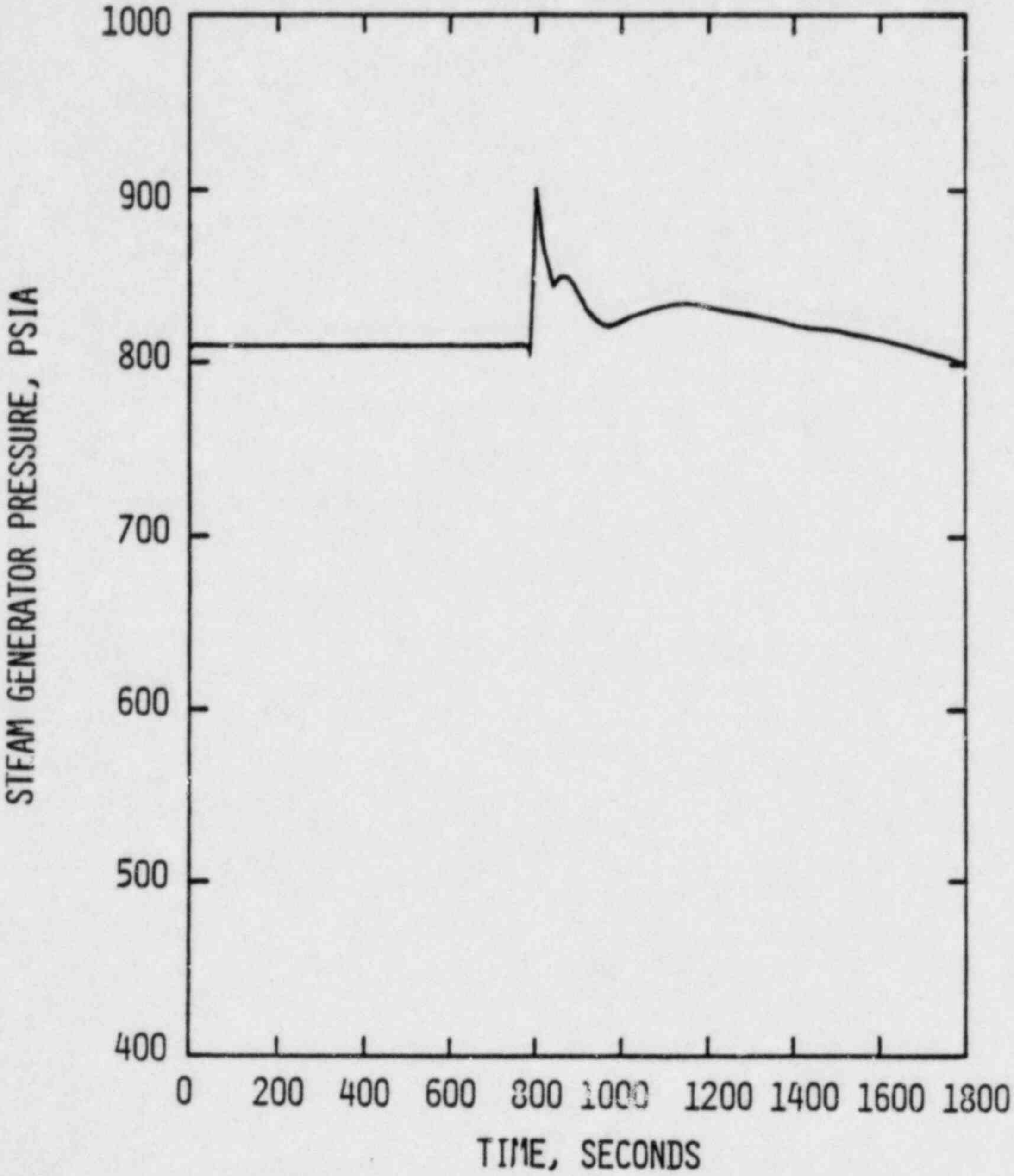
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT CORE AVERAGE HEAT FLUX VS TIME	FIGURE 4-2



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 4-3



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT REACTOR COOLANT SYSTEM TEMPERATURE VS TIME	FIGURE 4-4



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE FAILURE EVENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 4-5