CHARLES H. CRUSE
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BGE

November 20, 1996

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Request for Additional Information - Change to the Moderator Temperature Coefficient

REFERENCE: (a) Letter from Mr. C. H. Cruse to NRC Document Control Desk, dated March 28, 1996, License Amendment Request: Change to the Moderator Temperature Coefficient

In response to verbal NRC Staff questions Baltimore Gas and Electric Company is providing the additional information requested. The information is provided in Attachment (1). The additional information does not alter the Determination of No Significant Hazards submitted in the referenced letter.

A001

Should you have additional questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

Charles H. Cruse

STATE OF MARYLAND :
: TO WIT:
COUNTY OF CALVERT :

I hereby certify that on the 20th day of November, 1996 before me, the subscriber, a Notary Public of the State of Maryland in and for Calvert County, personally appeared Charles H. Cruse, being duly sworn, and states that he is Vice President of the Baltimore Gas and Electric Company, a corporation of the State of Maryland; that he provides the foregoing response for the purposes therein set forth; that the statements made are true and correct to the best of his knowledge, information, and belief; and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal: Denise D. Saulis
Notary Public

My Commission Expires: 2/2/98
Date

CHC/PSF/dlm

Attachment (1) Baltimore Gas and Electric Company's Response to NRC's Request for Additional Information Concerning License Amendment Request: Moderator Temperature Coefficient Change

Appendix A - Transient Data

cc: A. W. Dromerick, NRC

(Without Appendix A)
D. A. Brune, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
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ATTACHMENT (1)

BALTIMORE GAS AND ELECTRIC COMPANY'S RESPONSE

TO

NRC'S REQUEST FOR ADDITIONAL INFORMATION CONCERNING LICENSE AMENDMENT REQUEST: MODERATOR TEMPERATURE COEFFICIENT CHANGE

Calvert Cliffs Nuclear Power Plant
November 20, 1996

BALTIMORE GAS AND ELECTRIC COMPANY'S RESPONSE TO NRC'S REQUEST FOR ADDITIONAL INFORMATION CONCERNING LICENSE AMENDMENT REQUEST: MODERATOR TEMPERATURE COEFFICIENT CHANGE

Question 1:

Identify the transients and accidents that are <u>not affected</u> by increasing the number of plugged steam generator (SG) U-tubes. Provide justification as to why these events are not affected.

Response:

The limiting safety analyses for the following transients and accidents are not affected by increasing the number of plugged SG U-tubes:

- Excess Load
- Loss of Feedwater Flow (SG Dryout)
- Excess Feedwater Heat Removal
- Reactor Coolant System (RCS) Depressurization
- Loss of Non-Emergency AC Power
- Control Element Assembly (CEA) Drop
- CEA Eject
- Main Steam Line Break
- Steam Generator Tube Rupture
- Seized Reactor Coolant Pump (RCP) Rotor
- Excess Charging

The limiting safety analyses for these events are not affected since plugging SG U-tubes either would have no effect on the analysis results, or improve the results.

The number of plugged SG U-tubes will be controlled by our reload design process such that the minimum RCS flow assumed in the limiting safety analyses for these events is maintained at 370,000 gpm. The steady state full power RCS temperatures (T_{HOT} and T_{AVE}) are therefore not affected. As a result, the primary effect of increasing the number of plugged SG U-tubes, with respect to these events, is to reduce the SG heat transfer area.

<u>Excess Load, Excess Feedwater Heat Removal, and Main Steam Line Break</u>

All of these events are initiated by an increase in RCS heat removal by the SGs. The effect of plugging SG U-tubes, while maintaining constant RCS flow, is beneficial for these events. The net effect of plugged tubes is to reduce the available heat transfer area of a SG. This would result in a slower rate of RCS heat removal than assumed in these safety analyses. Therefore, the limiting safety analyses for these events are not affected.

<u>Loss of Feedwater Flow (SG Dryout)</u>

The time to SG dryout is dependent on the available SG water inventory at the SG low level trip setpoint and core decay heat. Plugging SG U-tubes does not affect the available SG inventory or decay heat assumed in the limiting safety analysis for this event, and this analysis is therefore not affected.

<u>RCS Depressurization, Loss of Non-Emergency AC Power, CEA Drop, and Seized RCP Rotor</u>

These events involve approach to departure from nucleate boiling (DNB) and/or linear heat rate (LHR) limits as a result of changing RCS pressure, flow, or core power distribution. Since plugging SG U-tubes within the limits established and controlled by our reload design process will not affect the limiting RCS parameters (flow, temperature, and pressure) or the limiting power distribution, and other core

characteristics assumed in the limiting safety analyses, these analyses are not affected. The seized RCP rotor analysis assumes a step change in RCS flow rate when the RCP rotor seizes, and so the flow transient assumed for this event is not affected by the increased RCS flow resistance due to plugging SG U-tubes.

CEA Eject Accident

This event is a rapid reactivity insertion coupled with a rapid RCS depressurization. Again, plugging SG U-tubes will not affect the RCS or core parameters assumed in the limiting safety analysis. This analysis is therefore not affected.

Steam Generator Tube Rupture

This event involves offsite dose due to the release of activity from the affected SG. The amount of activity released is a function of the ruptured tube leak rate and subsequent steam release rates offsite. The number of plugged SG U-tubes allowed by internal controls will not cause SG steam pressure to decrease below that assumed in the limiting safety analysis. Therefore, the ruptured tube leak rate and steam release rates are not affected, and neither is this analysis.

Excess Charging

This event involves increasing RCS pressure as a result of inadvertent charging system operation. The time available for operators to respond to the event is determined by the available pressurizer bubble volume and the charging system capacity. Plugging SG U-tubes will not affect these parameters, and the limiting safety analysis for this event is not affected.

Containment Response to Loss-of-Coolant Accident and Main Steam Line Break

These events result in rapid pressurization of the containment due to the mass and energy blowdown from the RCS and SGs. Plugging SG U-tubes slows the energy transfer between the RCS and SGs, which has a beneficial effect on the results of the limiting analyses for these events. In addition, plugging SG U-tubes reduces RCS inventory slightly which has an additional beneficial effect on the loss-of-coolant accident (LOCA) analysis. Again, since RCS and SG parameters for initial flow, temperature, and pressure assumed in the limiting analyses for these events are not affected by plugging SG U-tubes within the limits established and controlled by the reload design process, these limiting analyses are not affected.

Question 2:

Identify the transients and accidents that are not significantly affected by increasing the number of plugged SG U-tubes.

Response:

The following transients are not significantly affected by plugging SG U-tubes. Evaluations have been performed for these events to confirm that the results of the limiting safety analyses for these events still meet appropriate NRC acceptance criteria.

Boron Dilution
Loss of Flow
Asymmetric SG

Boron Dilution

The analyses for the Modes 1 through 4 Boron Dilution events assume an RCS volume corresponding to zero plugged SG U-tubes. Plugging SG U-tubes reduces the available RCS volume. Reducing the RCS volume will exacerbate the effect of the diluting water from the charging system assumed for the event. An evaluation was performed to verify the adverse effect of reducing the RCS volume is not significant.

The Mode 1 Boron Dilution event involves an approach to the DNB and LHR specified acceptable fuel design limits (SAFDLs). The current Mode 1 Boron Dilution safety analysis demonstrates that the maximum rate of reactivity addition is several orders of magnitude slower than a CEA withdrawal event. Therefore, the reduced RCS volume due to plugging SG U-tubes would not have a significant effect on the maximum reactivity addition rate in the Mode 1 scenario. The CEA withdrawal event remains limiting. Analysis of the CEA withdrawal event demonstrates that the DNB and LHR SAFDLs are not exceeded.

The Modes 2 through 4 Boron Dilution events involve an erosion of the available shutdown margin. The NRC acceptance criteria for Calvert Cliffs for these events is that shutdown margin will not be lost in less than 15 minutes. The current analyses for the Modes 2 through 4 Boron Dilution events demonstrate at least 45 minutes is available prior to loss of shutdown margin. The maximum number of plugged SG U-tubes allowed under internal controls represents less than 5% of the total available RCS volume. Therefore, the change in RCS volume associated with plugging SG U-tubes will not significantly reduce the time to loss of shutdown margin. At least 15 minutes will remain available. Therefore, plugging SG U-tubes will not significantly affect the limiting safety analysis for the Boron dilution event.

Loss of Flow

The limiting Loss of Flow event for Calvert Cliffs is a concurrent loss of power to all four RCPs. Total RCS flow rapidly decreases until the RCS low flow trip causes a reactor trip. The rate of RCS flow decrease is determined by the momentum of the RCP rotating element and the flow resistance of the RCS. The flow "coastdown" rate assumed in the limiting analysis was established based on test data taken in 1981 for an essentially zero plugged tube condition. The NRC acceptance criteria for this event is that the DNB and LHR fuel SAFDLs are not exceeded.

Plugging SG U-tubes will increase the RCS flow resistance seen by the RCPs. As previously discussed, Calvert Cliffs will ensure that steady state RCS flow remains above 370,000 gpm, the minimum RCS flow rate assumed in the limiting Loss of Flow safety analysis. The increased RCS flow resistance will increase the rate of RCS flow coastdown slightly during the Loss of Flow event. An evaluation was performed to determine the impact of a slightly faster RCS flow coastdown on the results of the limiting analysis for this event.

The effect of plugging SG U-tubes on the four-pump-flow coastdown was quantified using ABB/CE's COAST code. It was determined that the more rapid flow coastdown results in 0.4% less flow during the time of minimum departure from nuclear boiling ratio (DNBR) for this event. Review of the available margin established by the Reactor Protective System (RPS) setpoints, in conjunction with the Technical Specification LCOs, confirmed that the fuel SAFDLs are not exceeded given the change in RCS flow coastdown. Therefore, plugging SG U-tubes while maintaining constant RCS flow does not significantly affect the results of the limiting safety analysis for Loss of Flow.

Asymmetric SG

The limiting Asymmetric SG event at Calvert Cliffs is a Loss of Load to one SG caused by closure of one Main Steam Isolation Valve. During this event the non-uniform core inlet temperature distribution in conjunction with the moderator temperature reactivity feedback causes an increase in local core power peaking and an approach to the fuel SAFDLs. The reactor trip during this event is initiated as a result of increasing SG differential pressure.

An evaluation was performed to address the effect of a larger SG tube plugging asymmetry on the limiting safety analysis for this event. As more SG tubes are plugged, the potential exists to have a larger differential in the number of plugged tubes between the SGs. A maximum difference between the number of plugged tubes in each SG was established and will be controlled by our reload design process. This maximum difference was evaluated for this event. Additionally, a calculation was performed to confirm that the RCS flow splits caused by the tube plugging asymmetry would not affect the core inlet flow distribution. The effect of the tube plugging asymmetry on SG differential pressure was also considered. The concern was that a larger SG tube plugging asymmetry may cause a pressure difference between the SGs prior to event initiation that would affect the results of the limiting asymmetric event safety analysis.

The thermal margin degradation occurs during this event as a result of a temperature difference across the core caused by the asymmetric SG loading. It was determined that the inlet temperature difference across the core at the time of the reactor trip on high SG differential pressure would not be significantly affected by the initial pressure difference between SGs. Therefore, the evaluation concluded that the established maximum SG tube plugging asymmetry does not significantly affect the results of the limiting safety analysis for the Asymmetric SG event.

Question 3:

Identify the transients and accidents that are significantly affected by increasing the number of plugged SG U-tubes. Provide detailed results of your reanalyses for these events with the proposed Technical Specification change for the value of the MTC (moderator coefficient temperature). Include the following:

1. *Confirm that the methodologies used in the reanalyses are NRC approved.*
2. *Confirm that all assumptions used in the reanalyses are consistent with the analyses of record.*
3. *Provide transient data.*
4. *Confirm that the acceptance criteria are met for each event analyzed.*

Identify the transients and accidents that are significantly affected by increasing the number of plugged SG U-tubes.

Response:

The limiting safety analyses for the following transients and accidents are significantly affected by increasing the number of plugged SG U-tubes:

CEA Withdrawal (Hot Full Power, Hot Zero Power, Over-Pressure)
Loss of Load
Loss of Feedwater (Over-Pressure)
Feed Line Break
LOCA (Small & Large Breaks)

The first four events above are events during which the RCS heats up as a result of increasing reactor power or a degradation of the secondary side heat sink. During these four events, plugging SG U-tubes causes further degradation of heat transfer to the secondary system, which exacerbates the RCS heatup. Therefore, the limiting safety analyses for these events are significantly affected by plugging SG U-tubes. Reanalyses of these events have been performed to quantify the effect of plugging SG U-tubes on these events. Credit for a less positive (more restrictive) MTC than required by Technical Specifications was taken in order to mitigate the effect of the exacerbated RCS heatup.

For Small Break LOCA the number of plugged SG U-tubes is significant for two reasons: RCS inventory is reduced, and heat transfer to the secondary system is reduced. The reduced RCS inventory results in a more rapid core uncovery during the course of the accident. The reduced heat transfer to the secondary system results in elevated RCS pressure, which reduces flow from the high pressure safety pumps and prolongs the core uncovery period. Therefore, the limiting Small Break LOCA analysis is significantly affected, and the event has been reanalyzed to quantify the effect of plugging additional SG U-tubes.

For Large Break LOCA, the number of plugged SG U-tubes is significant primarily since it affects the available flow area through the SGs. This flow area is important for cold leg breaks (the limiting break location at Calvert Cliffs) since it affects the pressure drop caused by steam flow from the core, through

the SG, and out the break. A smaller flow area results in a larger pressure drop, which depresses the two-phase water level in the core. This effect will elevate the peak clad temperature for the event. Therefore, the limiting safety analysis for Large Break LOCA is significantly affected, and the event has been reanalyzed to quantify the effect of plugging additional SG U-tubes.

1. *Confirm that the methodologies used for the reanalyses are consistent with the analyses of record.*

Response:

The methodologies used in the reanalyses are consistent with those previously used in the most recent analyses reviewed by NRC.

For the non-LOCA transients, CESEC was used as the general plant transient code to calculate the time dependent core power, flow, coolant temperatures, and pressure. CETOP, based on a simplified TORC model, is the code used to calculate the DNBR during the course of the transient using input from CESEC.

For the Small Break LOCA analysis, CEFLASH-4AS is used to calculate the RCS blowdown. PARCH is used to calculate the hot rod heatup.

For the Large Break LOCA analysis, CEFLASH-4A is used to calculate the RCS blowdown. COMPERC-II is used to calculate the RCS refill and reflood, PARCH is used to calculate the steam cooling heat transfer coefficients, and STRIKIN-II is used to calculate the hot rod heatup.

2. *Confirm that all assumptions used in the reanalyses are consistent with the analyses of record.*

Response:

The assumptions used in the reanalyses are consistent with the assumptions previously used in the most recent analyses reviewed by the NRC. The significant assumptions used for each reanalysis are provided in the "Initial Conditions and Input Parameters" tables of Appendix A. The only significant change from previous assumptions is the more restrictive MTC ($+0.15 \times 10^{-4}$ $\Delta\rho/°F$ in place of $+ 0.3 \times 10^{-4}$ $\Delta\rho/°F$) supported by the subject change to the Technical Specifications. In addition, a reduced RCS flow rate (358,900 gpm in place of 370,000 gpm) is conservatively used in anticipation of the need to reduce the required minimum RCS flow rate in the future. Other assumptions are changed in an insignificant manner to match the physical effects of tube plugging, to update core and physics parameters, to incorporate RPS and Engineered Safety Features Actuation Signal analytical setpoints based on updated uncertainty calculations, and to update the main steam safety valve model and pressurizer safety valve model based on vendor test data.

3. *Provide the transient data associated with the reanalyses.*

Response:

Transient data are provided in Appendix A in the form of plots of time-dependent values for the significant parameters for each event.

4. *Confirm that the NRC acceptance criteria are met for each event reanalyzed.*

Response:

CEA Withdrawal

The reanalysis of the CEA Withdrawal event demonstrates that the action of the RPS prevents exceeding the fuel SAFDLs and the RCS pressure safety limit of 2750 psia.

Loss of Load, Loss of Feedwater (over pressure) and Feed Line Break

The reanalyses of these events demonstrates that the action of the RPS, the pressurizer safety valves, and the main steam safety valves is sufficient to ensure that the RCS pressure safety limit of 2750 psia, and the secondary system pressure limit of 1100 psia, are not exceeded.

Small Break LOCA and Large Break LOCA

The reanalyses of these events demonstrates that the acceptance criteria of 10 CFR 50.46(b) are met.

APPENDIX (A)

TRANSIENT DATA

TABLE 14.2-1

INITIAL CONDITIONS AND INPUT PARAMETERS
FOR THE CEA WITHDRAWAL EVENT

PARAMETER	UNITS	UNIT 1	UNIT 2
Initial Core Power Level			
HZP	MWt	0	0
HFP		2700[(a)]/2754[(b)]	2700[(a)]/2754[(b)]
Core Inlet Coolant Temperature			
HZP	^{o}F	532	532
HFP		548[(a)]/550[(b)]	548[(a)]/550[(b)]
RCS Pressure	psia	2200[(a)] 2165[(b)]	2200[(a)] 2165[(b)]
MTC	$10^{-4}\ \Delta\rho/^{o}F$	+0.7 to -3.0	+0.7 to -3.0
Doppler Coefficient Multiplier	---	.85	.85
CEA Worth at Trip - (HFP)	$10^{-2}\ \Delta\rho$	-5.0	-5.0
CEA Worth at Trip - (HZP)	$10^{-2}\ \Delta\rho$	-3.5	-3.5
Reactivity Insertion Rate	$X10^{-4}\ \Delta\rho/sec$	0 to 1.6	0 to 1.6
Holding Coil Delay Time	sec	0.5	0.5
CEA Time to 90% Insertion (Including Holding Coil Delay)	sec	3.1	3.1

TABLE 14.2-1 (Continued)

INITIAL CONDITIONS AND INPUT PARAMETERS
FOR THE CEA WITHDRAWAL EVENT

PARAMETER	UNITS	UNIT 1	UNIT 2
Rod Group Withdrawal Speed	in/min	30.0	30.0
CEA Differential Worth	$X10^{-4}$ $\Delta\rho$/inch	0.0 to 3.2	0.0 to 3.2

[a] For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

[b] For the peak RCS pressure case, the effects of uncertainties on these parameters were combined deterministically.

TABLE 14.2-2

SEQUENCE OF EVENTS FOR ZERO POWER CEA WITHDRAWAL EVENT

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	---
26.9	VHPT Signal Generated	40% of 2700 MWt
27.3	Reactor Trip Breakers Open	---
27.7	Core Power Reaches Maximum	131% of 2700 MWt
27.7	Maximum Peak LHR and Maximum Fuel Centerline Temperature	43.6 kW/ft 3,260°F
27.8	CEAs Begin to Drop into Core	---
28.5	Core Heat Flux Reaches Maximum	58.7% of 2700 MWt
28.5	Minimum DNBR Occurs	1.22
29.9	RCS Pressure Reaches Maximum	2384 psia

TABLE 14.2-3

SEQUENCE OF EVENTS FOR FULL POWER CEA WITHDRAWAL EVENT

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	---
3.95	HPT Signal Generated	110.2% of 2700 MWt
4.35	Reactor Trip Breakers Open	---
4.55	Core Power Reaches Maximum	118.3% of 2700 MWt
4.65	Minimum DNBR Occurs	> 1.21
4.85	CEAs Begin to Drop Into Core	---
5.15	Core Heat Flux Reaches Maximum	110.2% of 2700 MWt
6.1	RCS Pressure Reaches Maximum	2263 psia

TABLE 14.2-4

SEQUENCE OF EVENTS FOR
FULL POWER CEA WITHDRAWAL EVENT
WITH RESPECT TO PEAK PRESSURE

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	---
56.3	High Pressurizer Pressure Trip Signal Generated	2420 psia
56.9	VHPT Signal Generated	112.2% of 2700 MWt
57.2	Trip Breakers Open	---
57.7	CEAs Begin to Drop Into Core	---
57.8	Core Power Reaches Maximum	112.4% of 2700 MWt
57.9	Core Heat Flux Reaches Maximum	111.5% of 2700 MWt
59.4	Pressurizer Pressure Reaches Maximum	2496 psia[a]
60.1	Steam Generator Safety Valves Begin to Open	1010 psia

[a] Maximum RCS pressure includes elevation head.

FIGURE 14.2-1

CEA Withdrawal Event

Core Power Versus Time

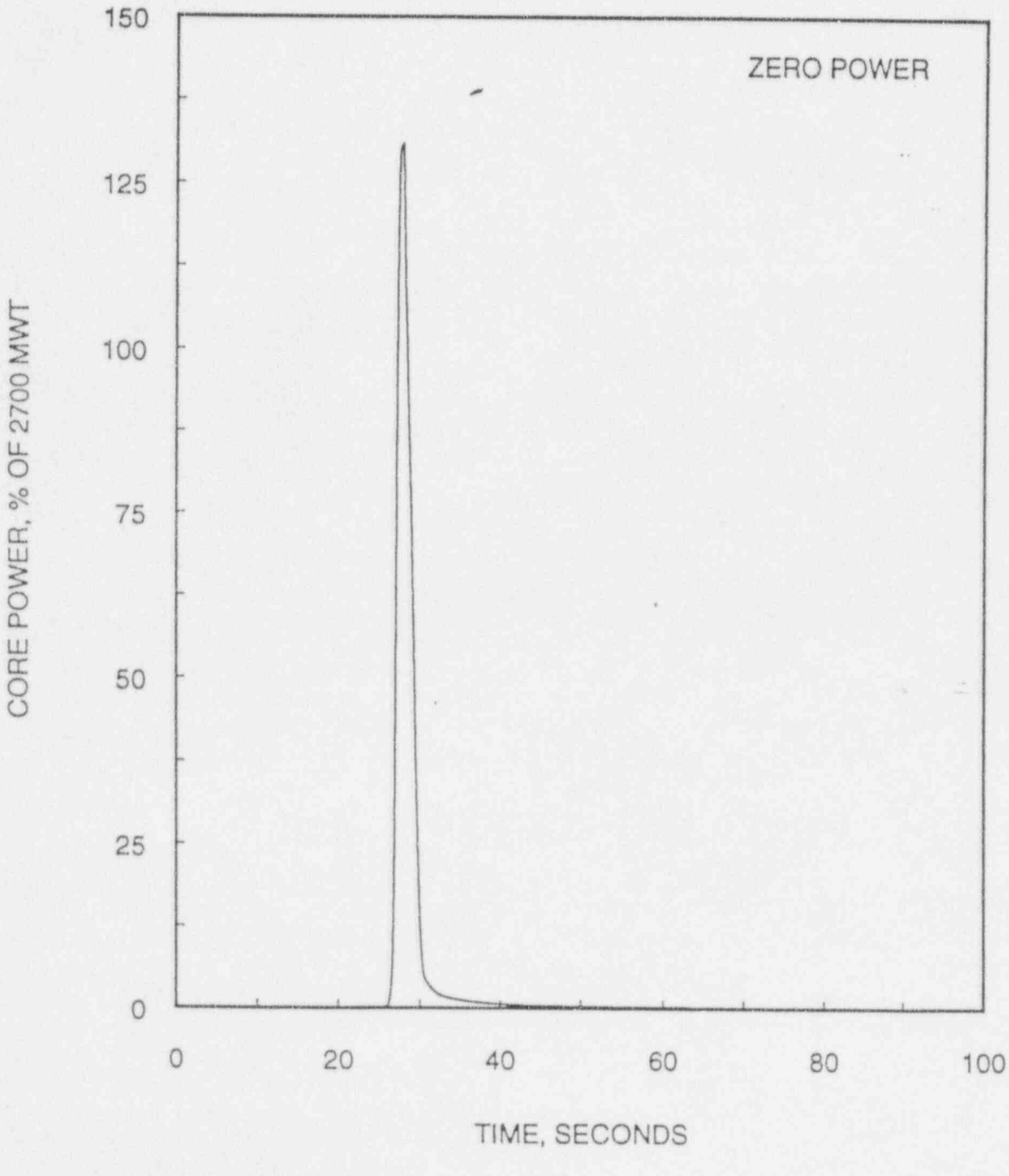


FIGURE 14.2-2
CEA Withdrawal Event
Core Heat Flux Versus Time

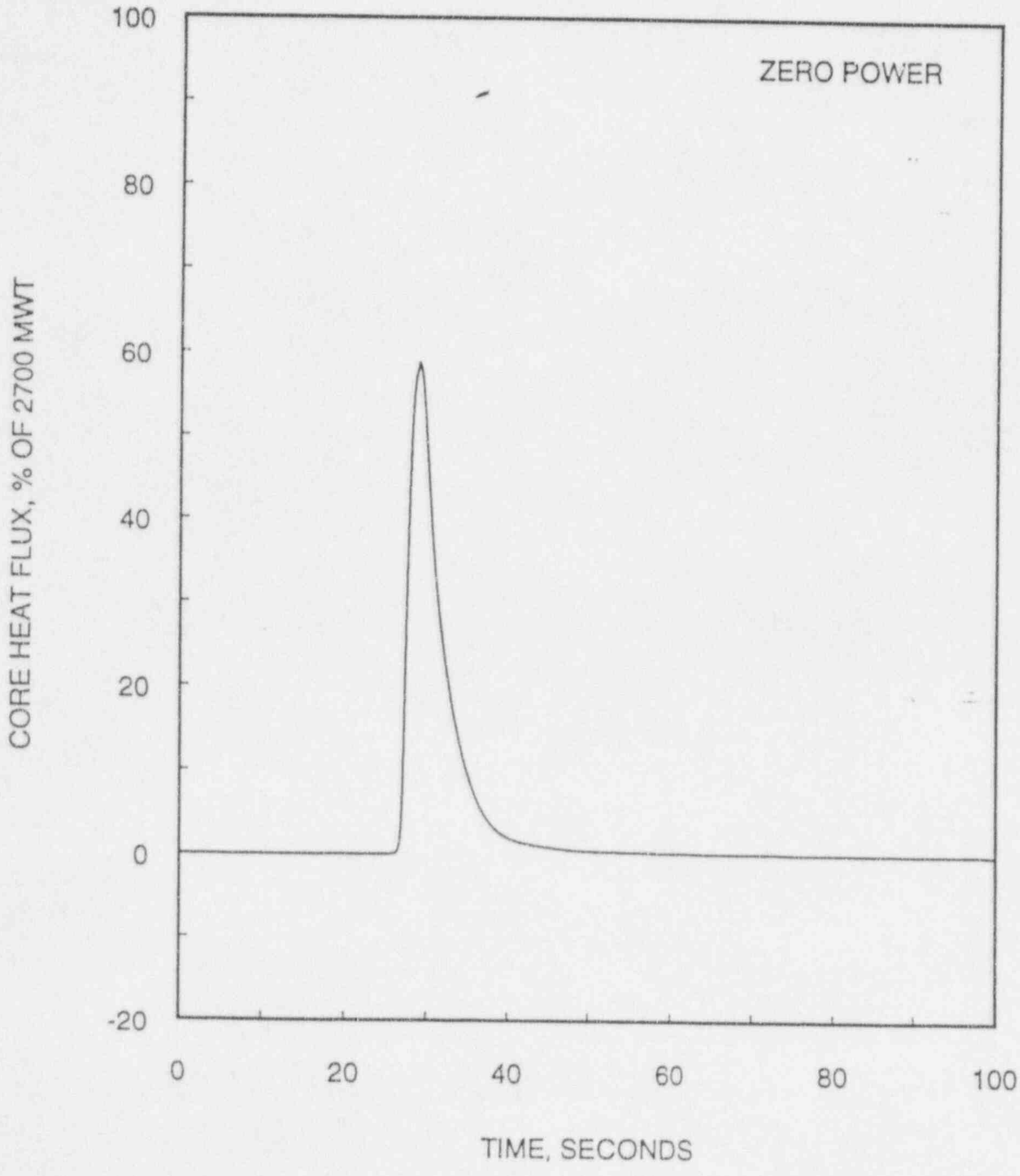


FIGURE 14.2-3

CEA Withdrawal Event

RCS Temperatures Versus Time

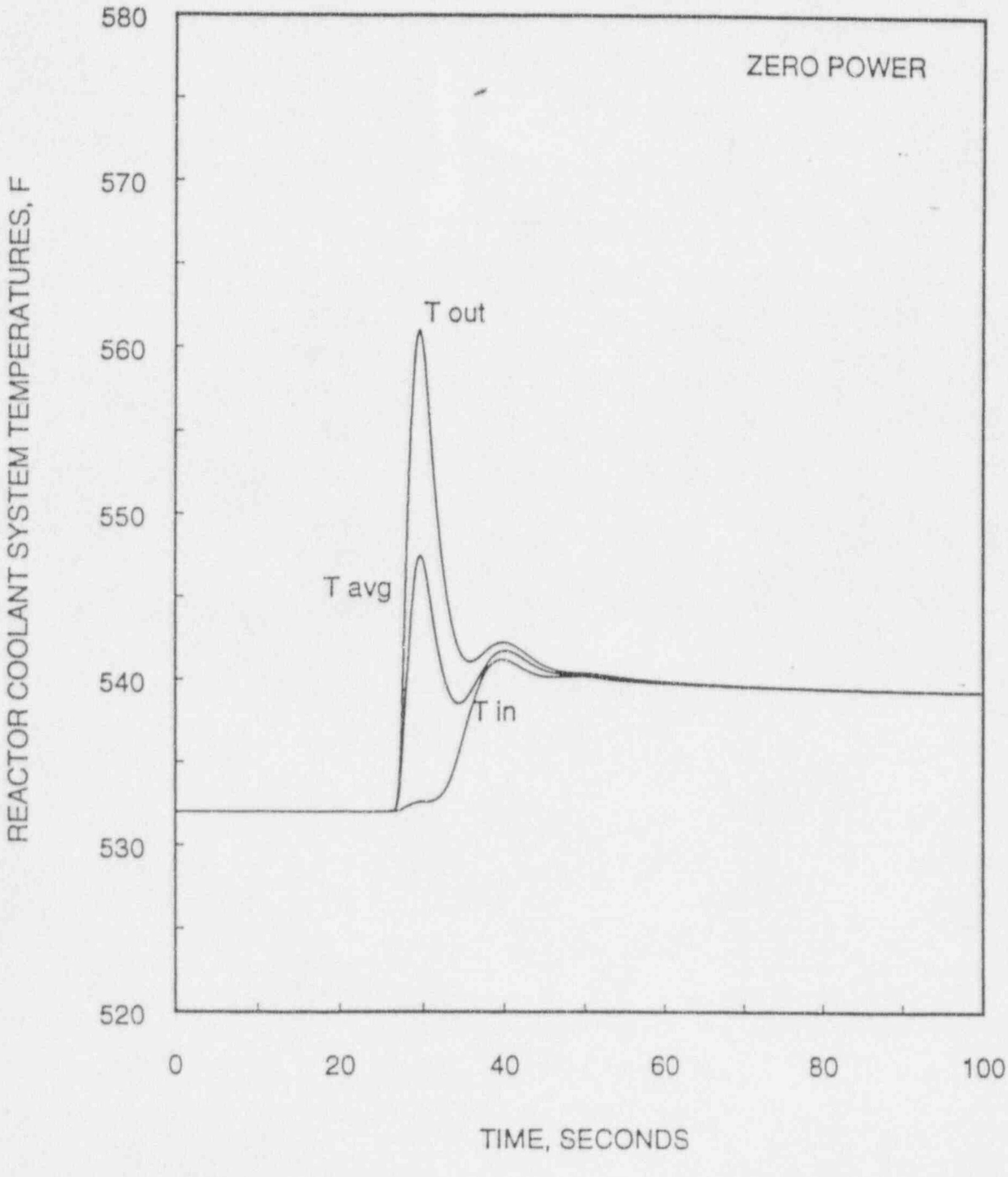


FIGURE 14.2-4
CEA Withdrawal Event
RCS Pressure Versus Time

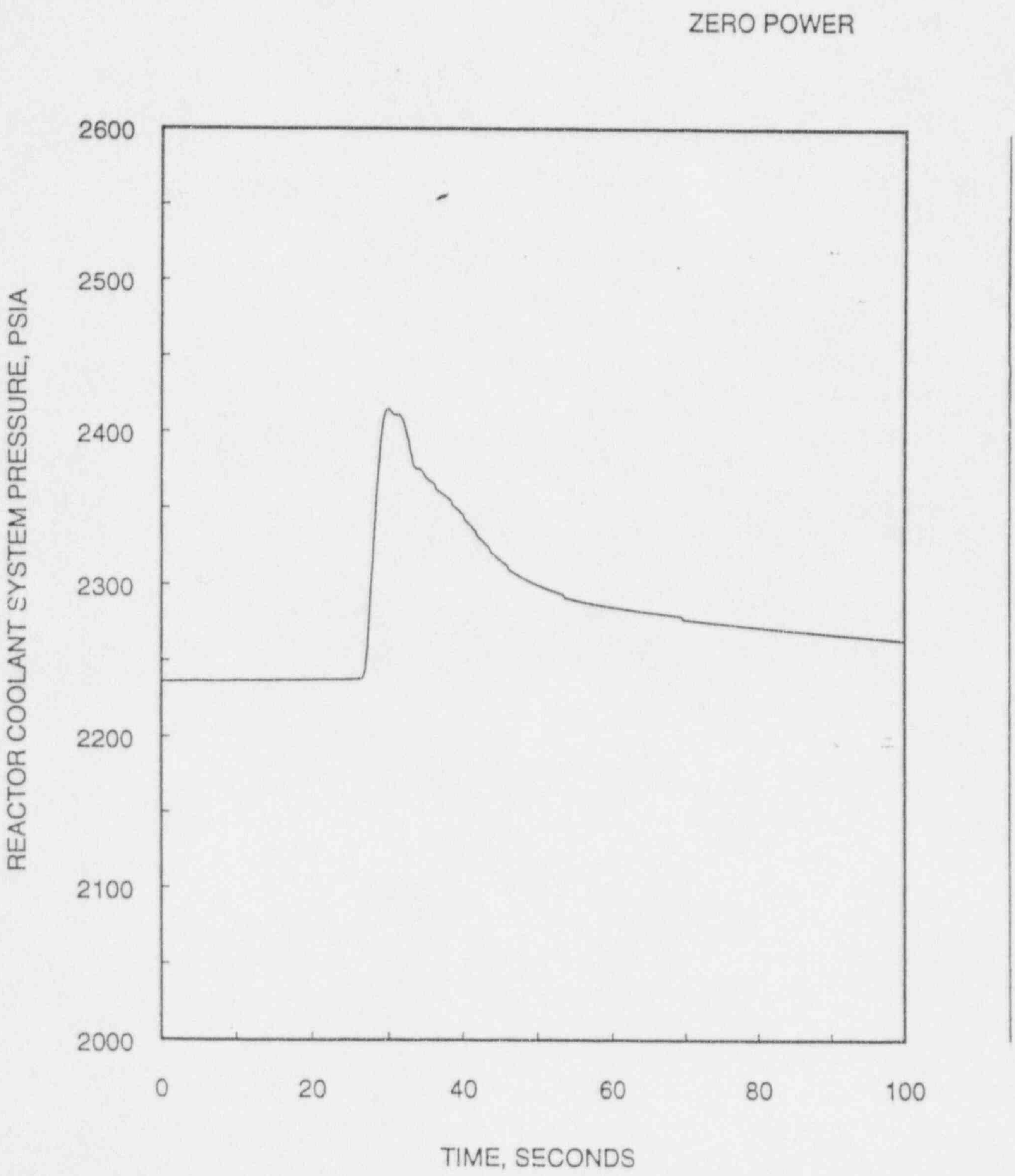


FIGURE 14.2-5

CEA Withdrawal Event

Core Power Versus Time

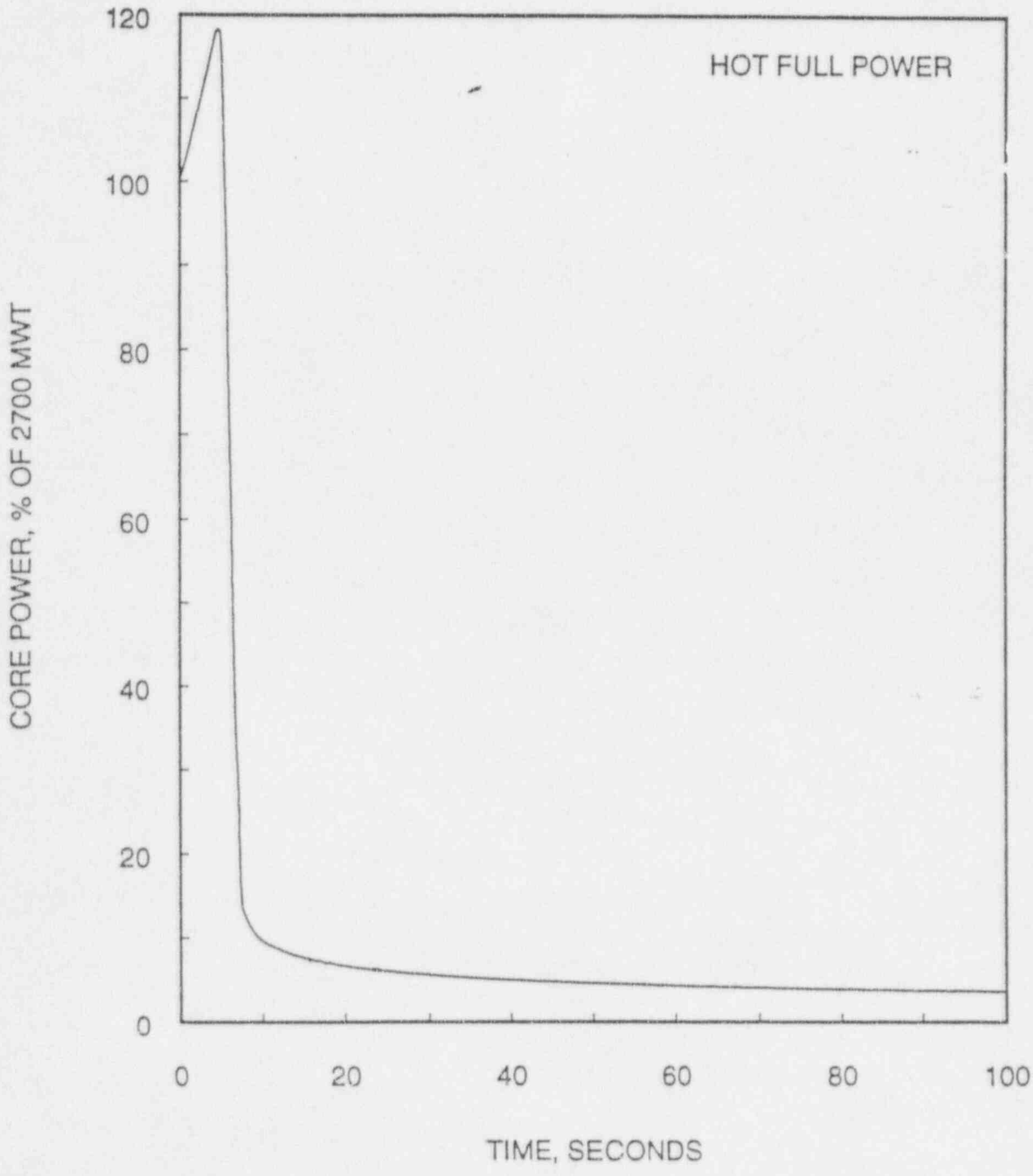


FIGURE 14.2-6

CEA Withdrawal Event

Core Heat Flux Versus Time

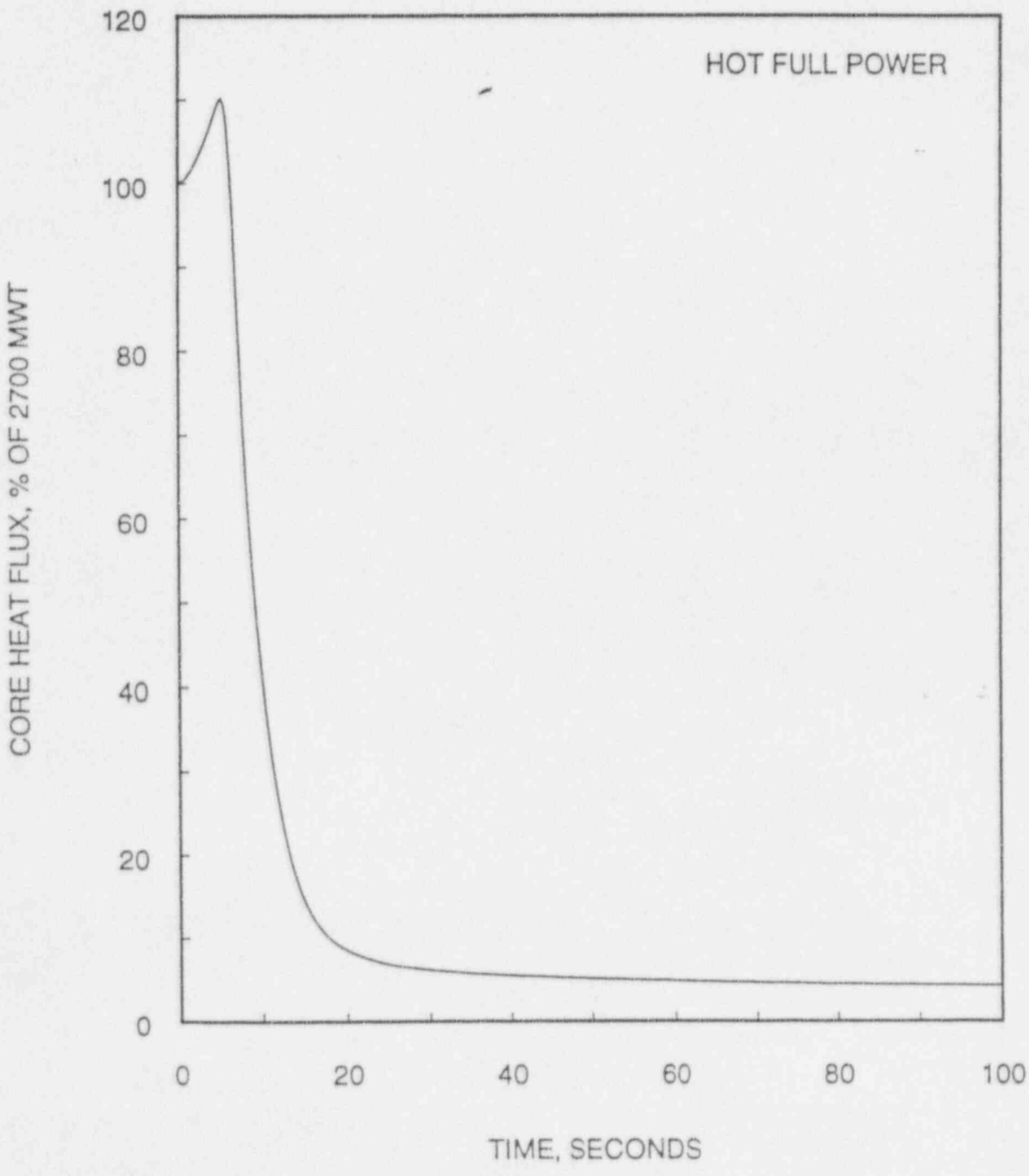


FIGURE 14.2-7

CEA Withdrawal Event

RCS Temperatures Versus Time

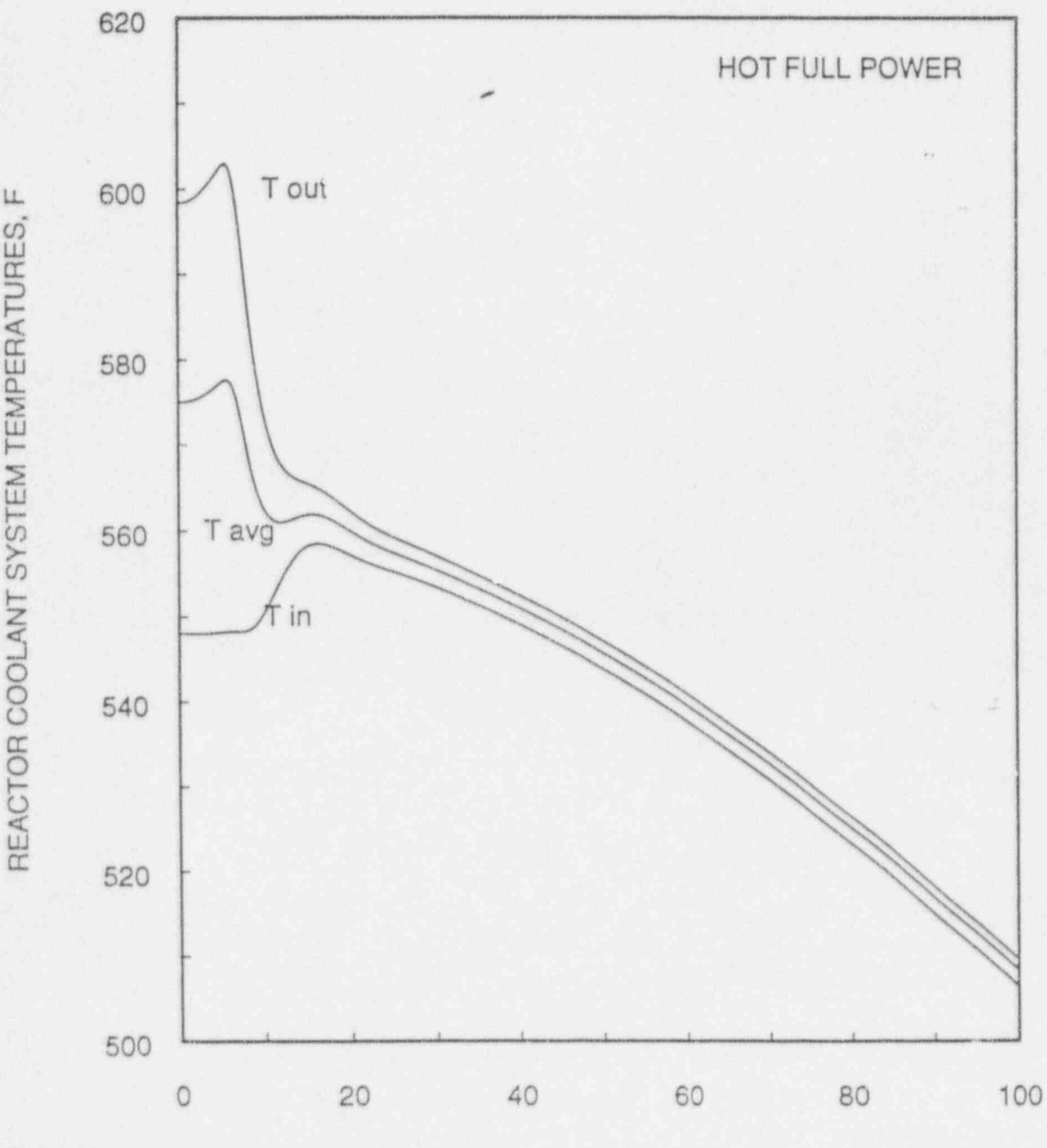


FIGURE 14.2-8

CEA Withdrawal Event

RCS Pressure Versus Time

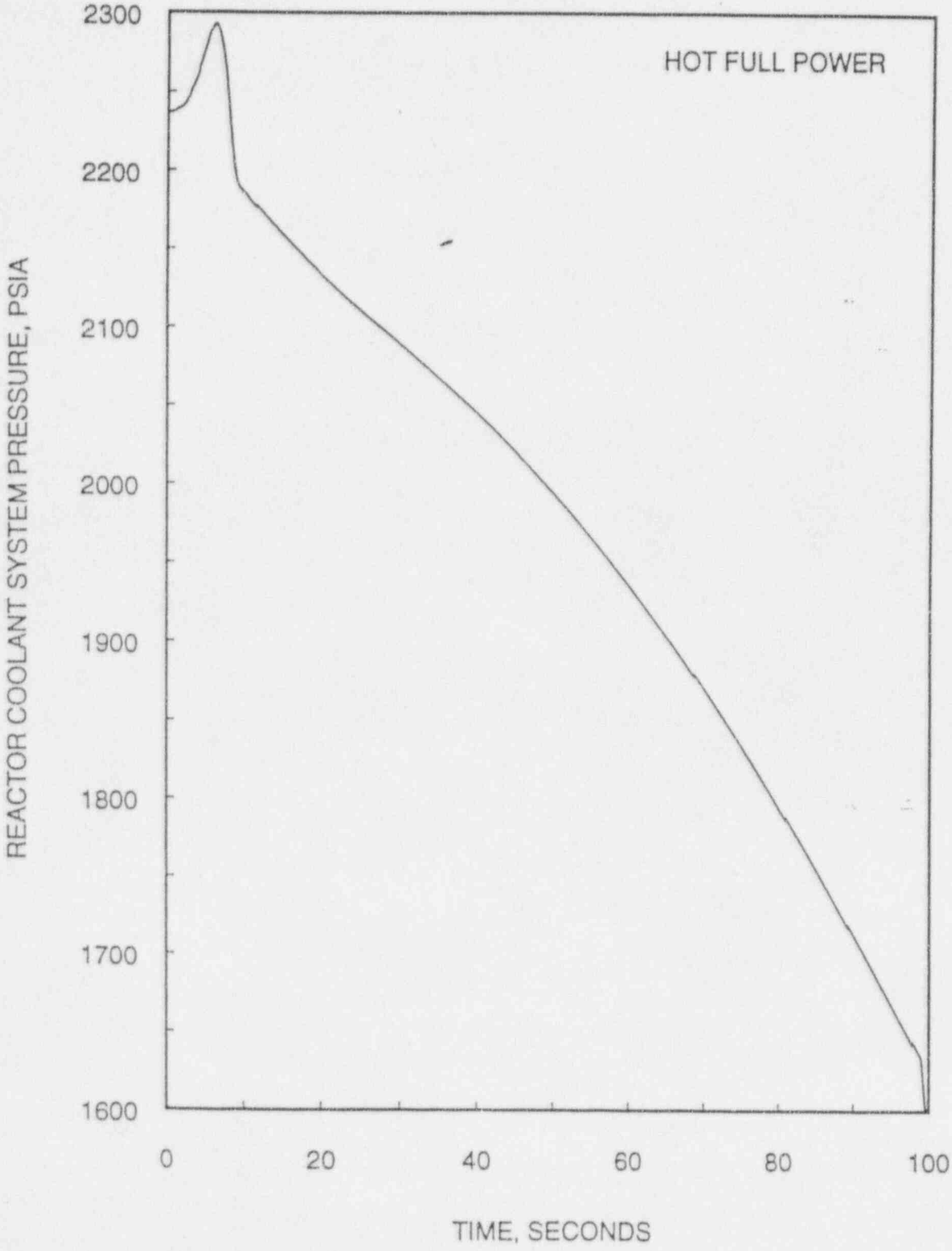


FIGURE 14.2-10

CEA Withdrawal Event

Core Power Versus Time

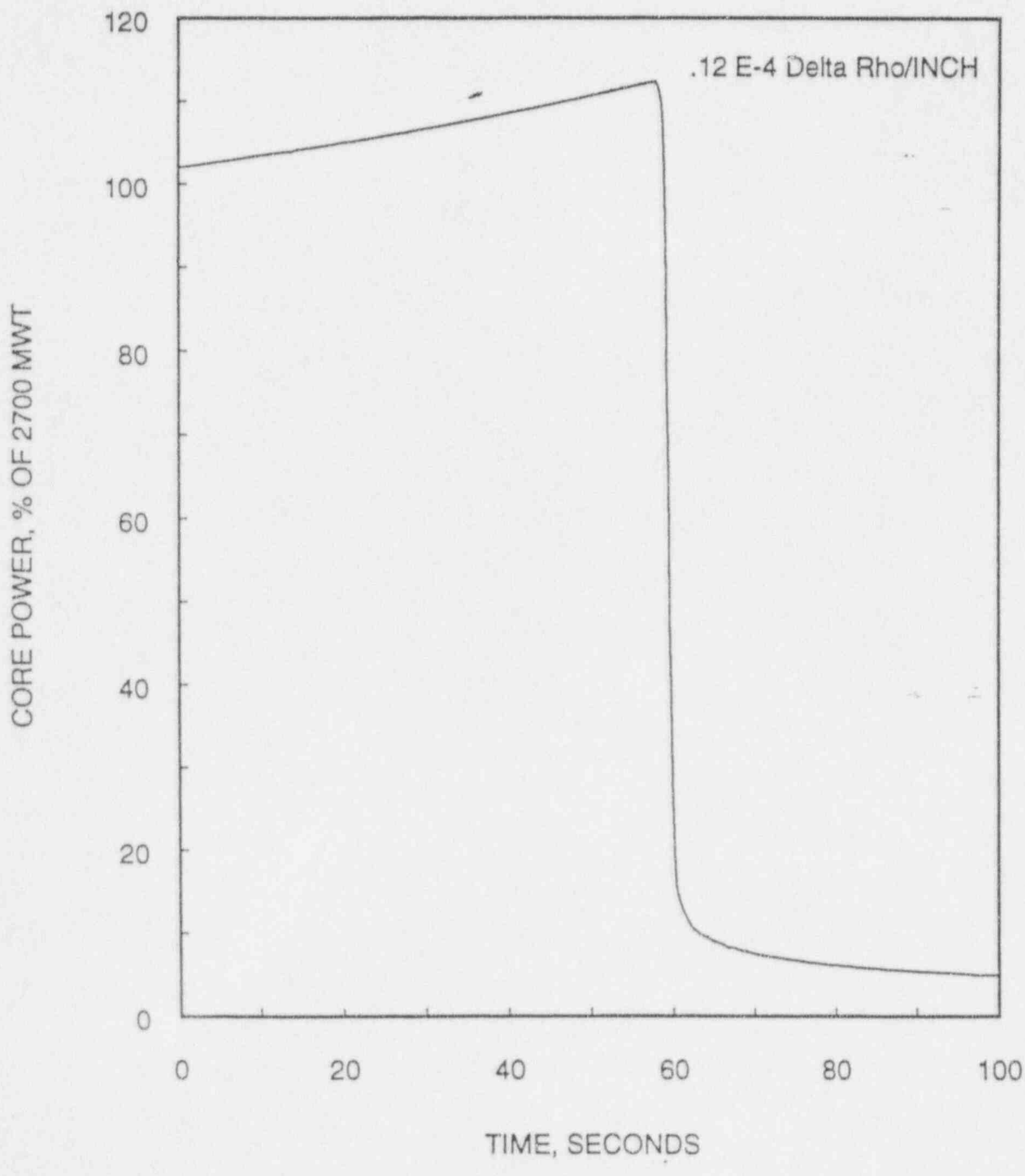


FIGURE 14.2-11

CEA Withdrawal Event

Core Heat Flux Versus Time

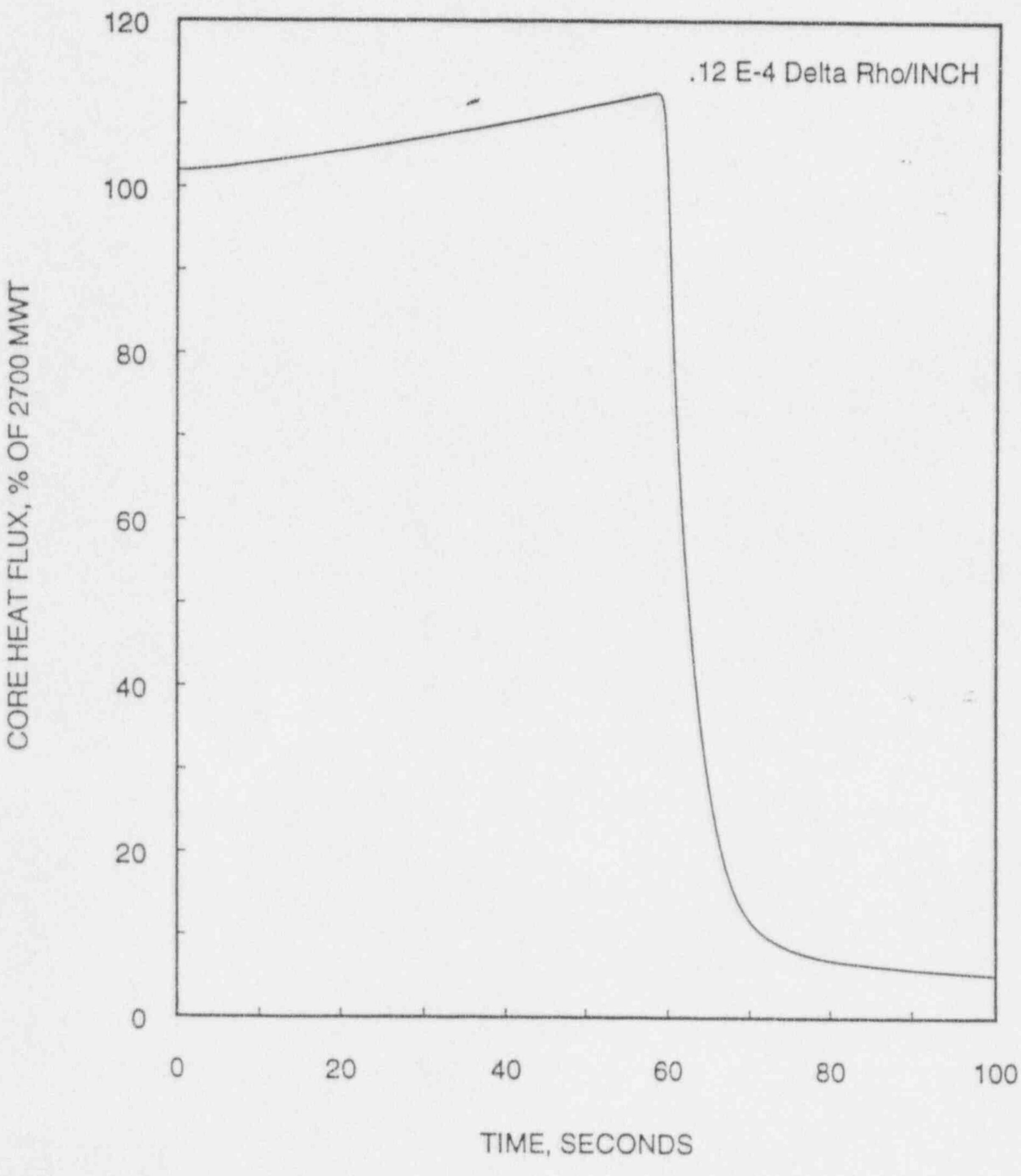


FIGURE 14.2-12

CEA Withdrawal Event

RCS Pressure Versus Time

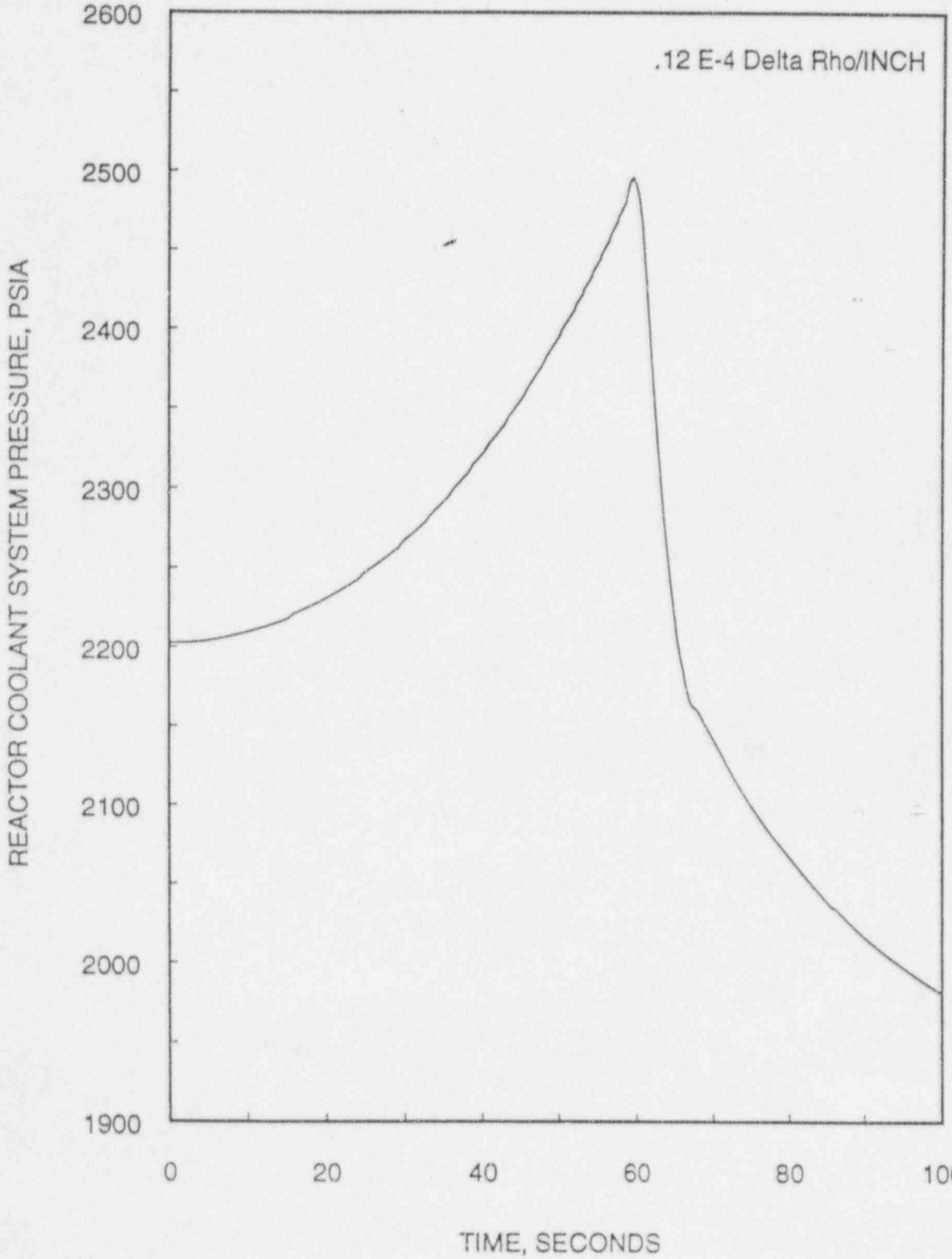


FIGURE 14.2-13
CEA Withdrawal Event
RCS Temperatures Versus Time

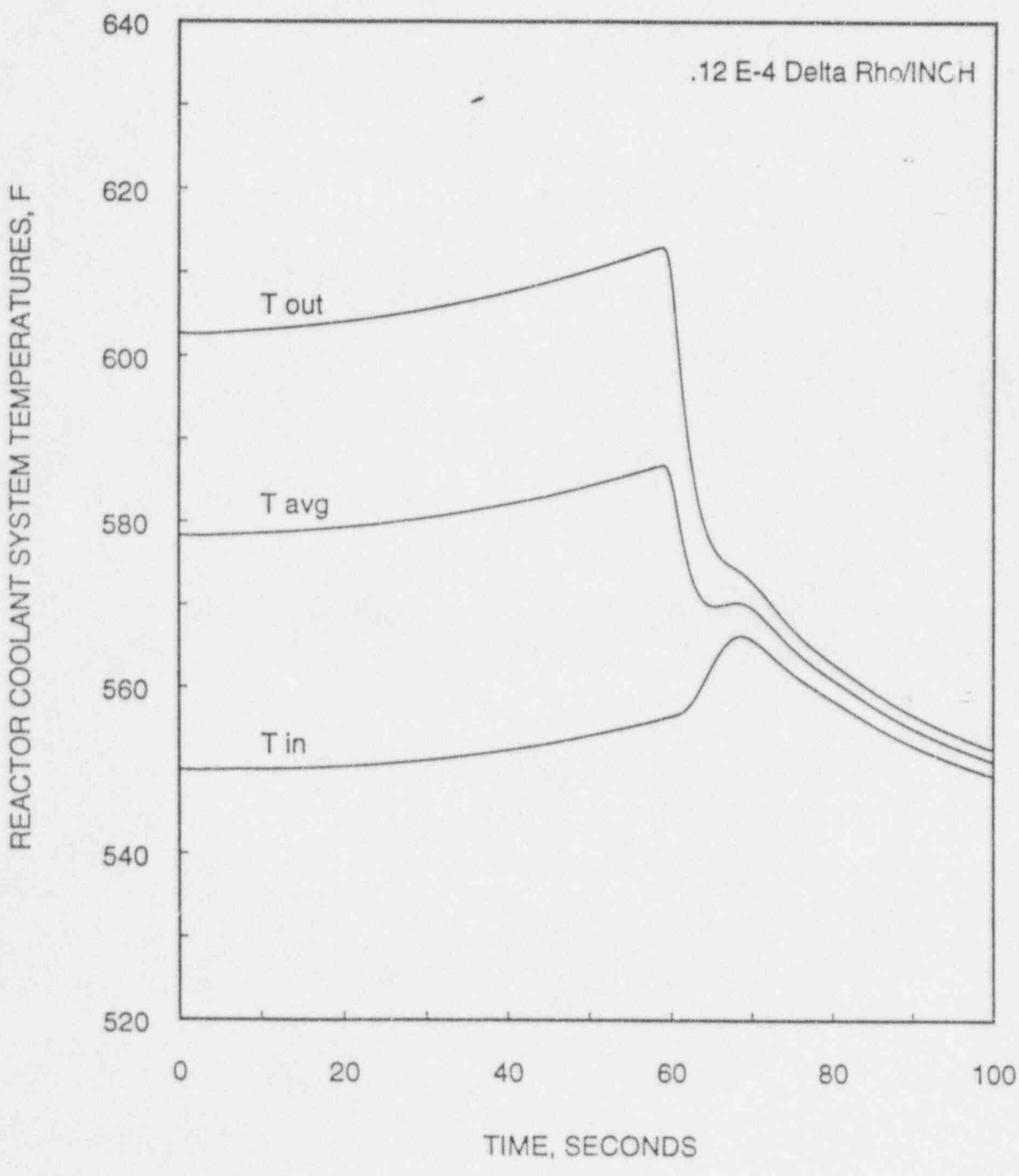


FIGURE 14.2-14

CEA Withdrawal Event

Steam Generator Pressure Versus Time

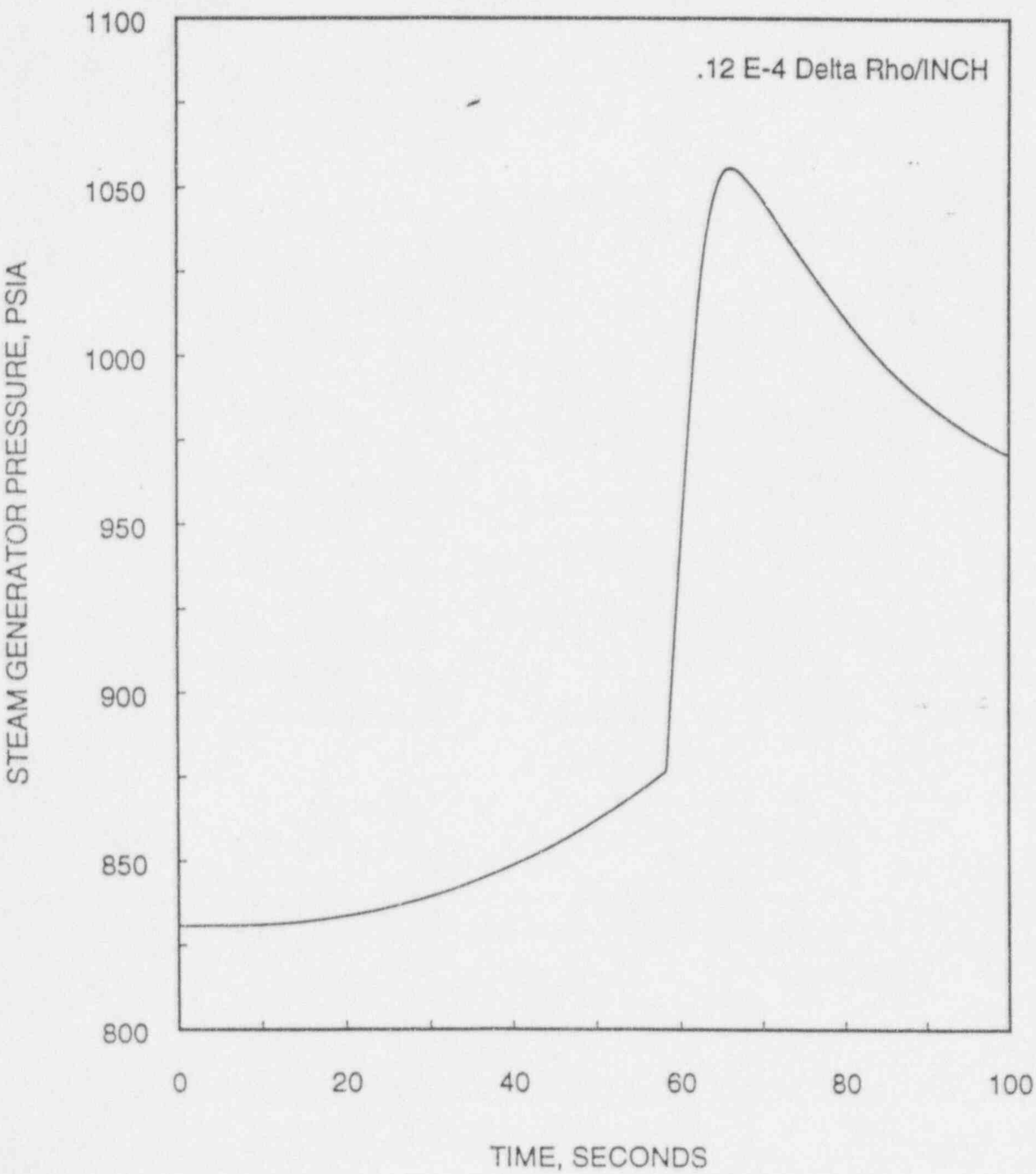


TABLE 14.5-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOSS OF LOAD EVENT TO CALCULATE MAXIMUM RCS PRESSURE

PRESSURE	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2754[b]	2754[b]
Initial Core Inlet Coolant Temperature	°F	550	550
Core Mass Flow Rate	X 10^6 lbm/hr	129.8	129.8
Initial RCS Pressure	psia	2165[a]	2165[a]
Initial Pressurizer Liquid Level at Full Power	ft^3	975	975
Initial SG Pressure	psia	831	831
MTC	X 10^{-4} $\Delta\rho$/°F	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
Number of Plugged SG Tubes	---	1500	1500
Axial Shape Index	---	+0.6	+0.6
CEA Worth at Trip	% $\Delta\rho$	-5.0	-5.0
Time to 90% Insertion of SCRAM Rods	sec	3.1	3.1
RRS	Operating Mode	Manual	Manual
SDBS	Operating Mode	Inoperative	Inoperative
Minimum MSSV Opening	psia	1010	1010
Pressurizer Pressure Control System	Operating Mode	Manual	Manual
Pressurizer Level Control System	Operating Mode	Manual	Manual

(a) Corresponds to Technical Specification minimum indicated pressure of 2200 psia. The value includes an uncertainty of 35 psia.

(b) Value does not include 17 MWt of pump heat added to core power in CESEC.

TABLE 14.5-2

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

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Loss of Secondary Load	---
5.85	Steam Generator Safety Valves Begin to Open	1010 psia
7.44	High Pressurizer Pressure Trip Signal Generated	2420 psia
7.94	CEAs Begin to Drop Into the Core	---
8.39	Pressurizer Safety Valves Begin to Open	2550 psia
10.06	Maximum RCS Pressure	2658 psia[a]
11.24	Maximum SG Pressure	1092 psia[b]
14.5	PSVs are Fully Closed	2448 psia

(a) RCS pressure includes elevation head.
(b) Steam Generator pressure includes downcomer liquid head.

TABLE 14.5-4

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOSS OF LOAD EVENT TO CALCULATE MAXIMUM SECONDARY PRESSURE

PRESSURE	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2754(b)	2754(b)
Initial Core Inlet Coolant Temperature	^{o}F	550	550
Core Mass Flow Rate	X 10^6 lbm/hr	133.9	133.9
Initial RCS Pressure	psia	2165(a)	2165(a)
Initial Pressurizer Liquid Level at Full Power	ft^3	800	800
Initial SG Pressure	psia	865	865
MTC	X 10^{-4} $\Delta\rho/^{o}F$	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
CEA Worth at Trip	% $\Delta\rho$	-5.0	-5.0
Number of Plugged SG Tubes	---	0	0
Axial Shape Index	---	+0.6	+0.6
Time to 90% Insertion of SCRAM Rods	sec	3.1	3.1
RRS	Operating Mode	Manual	Manual
SDBS	Operating Mode	Inoperative	Inoperative
Minimum MSSV Opening	psia	1010	1010
Pressurizer Pressure Control System	Operating Mode	Auto	Auto
Pressurizer Level Control System	Operating Mode	Auto	Auto

(a) Corresponds to Technical Specification minimum indicated pressure of 2200 psia. The value includes an uncertainty of 35 psia.

(b) Value does not include 17 MWt of pump heat added to core power in CESEC.

FIGURE 14.5-1

LOSS OF LOAD EVENT

CORE POWER VERSUS TIME

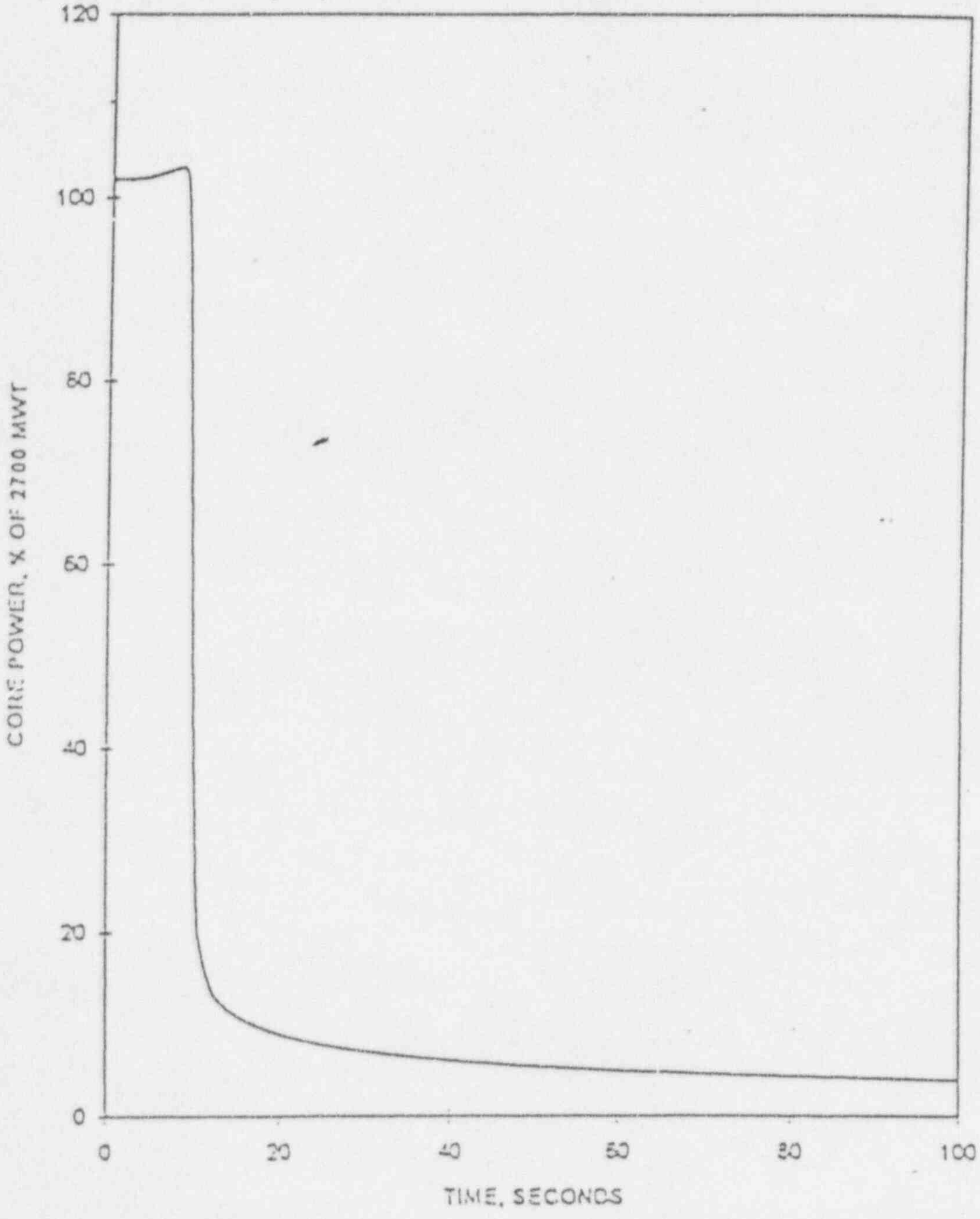


FIGURE 14.5-2

LOSS OF LOAD EVENT

CORE HEAT FLUX VERSUS TIME

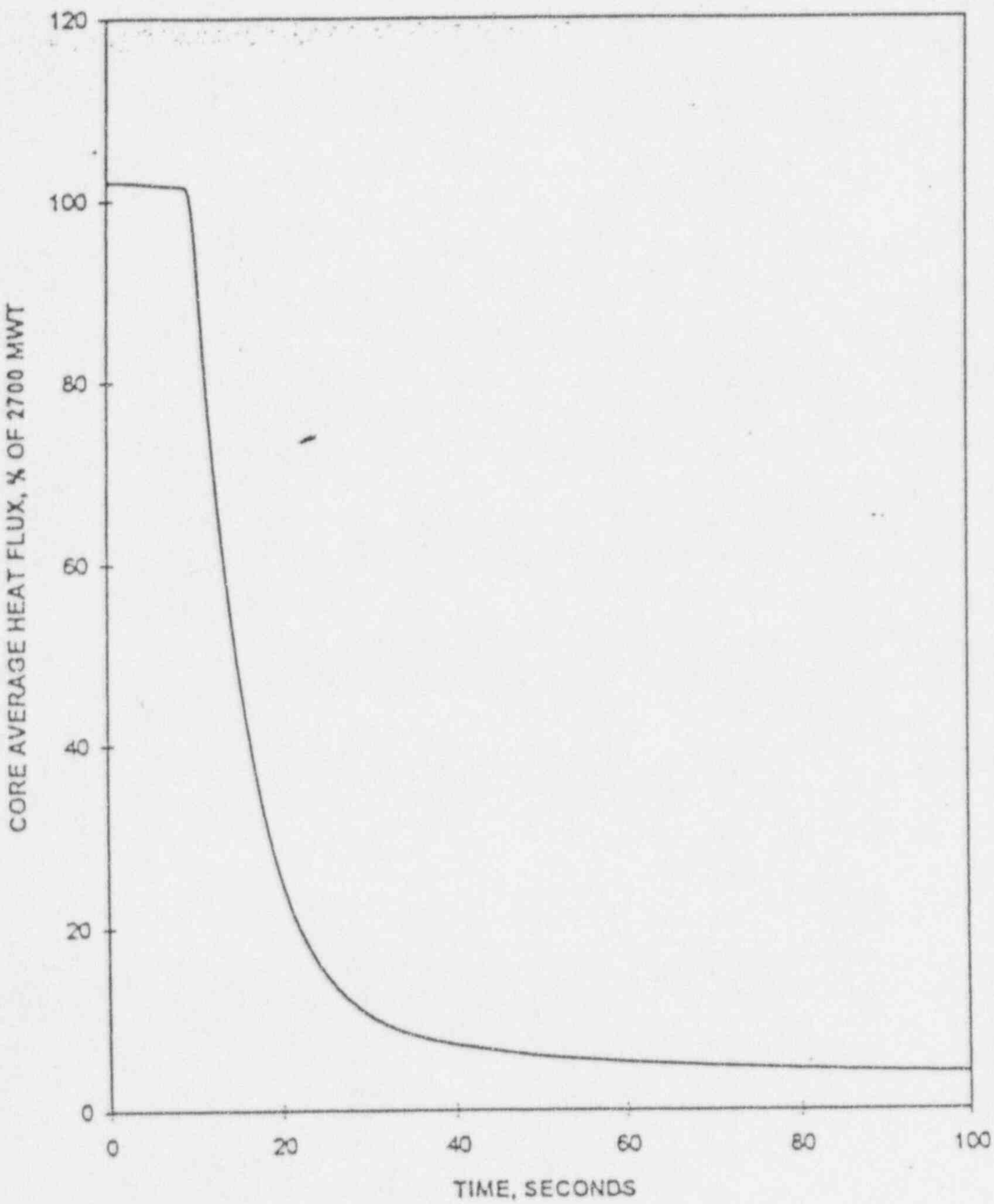


FIGURE 14.5-3

LOSS OF LOAD EVENT

RCS PRESSURE VERSUS TIME

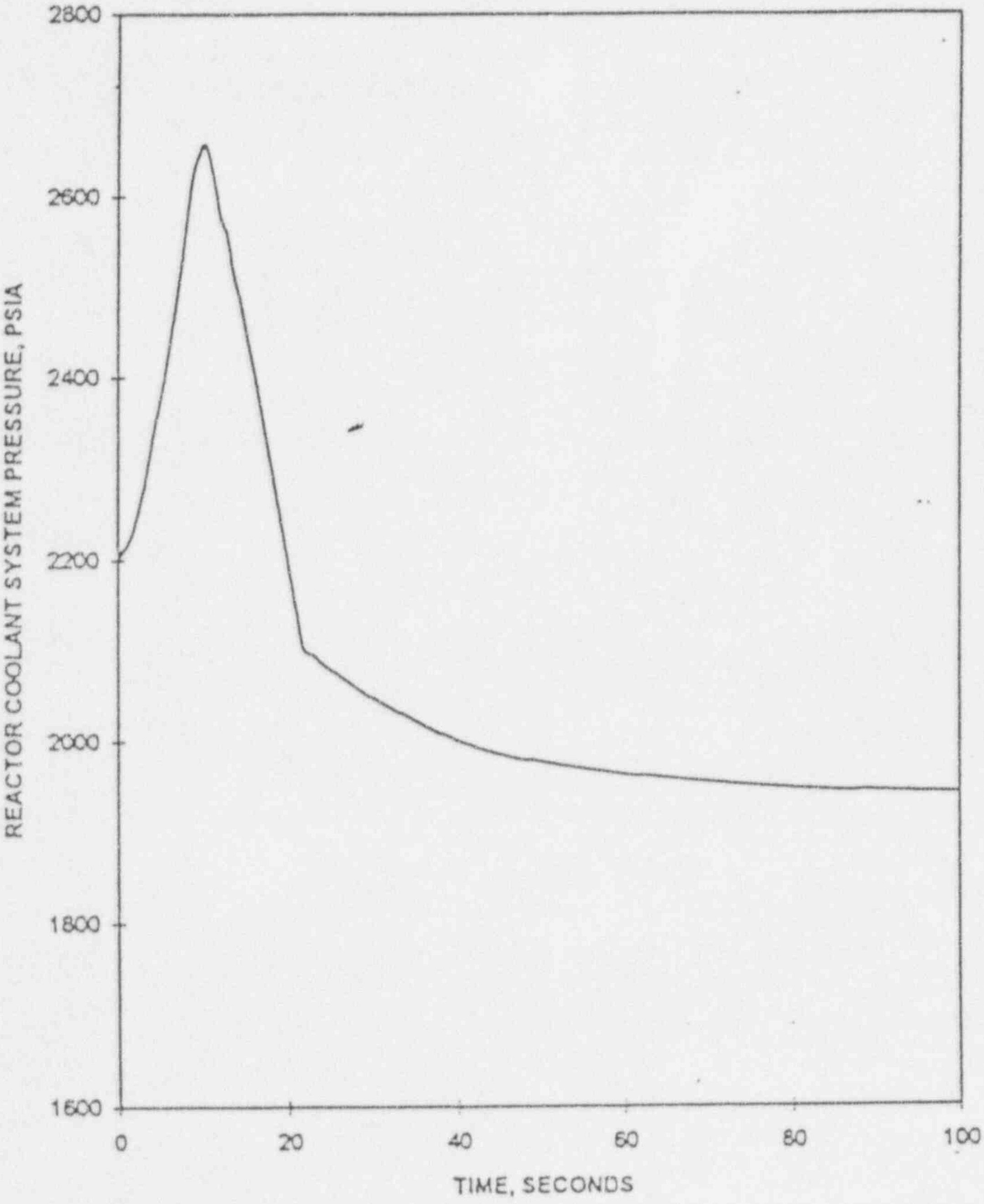


FIGURE 14.5-4

LOSS OF LOAD EVENT

RCS TEMPERATURES VERSUS TIME

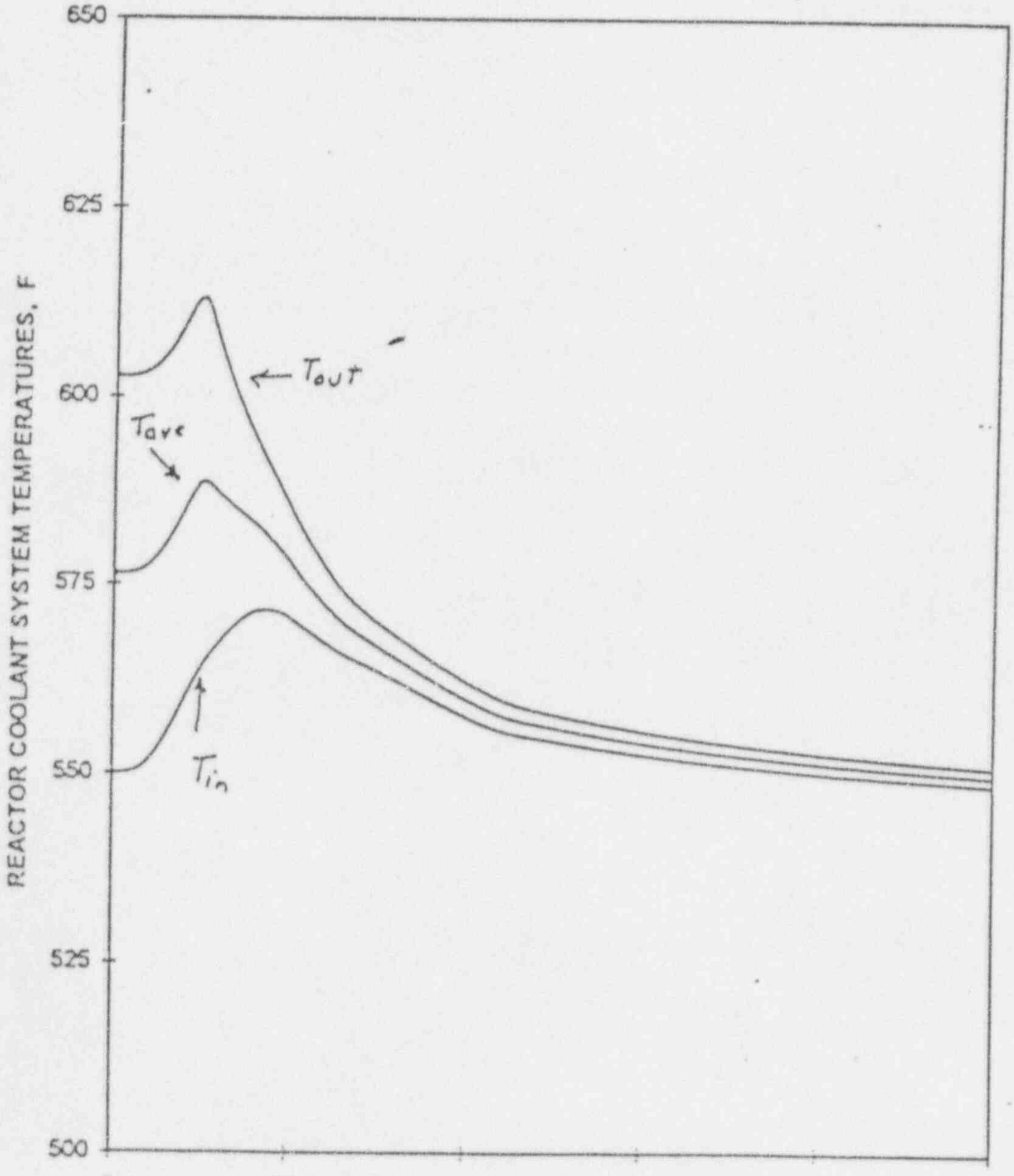
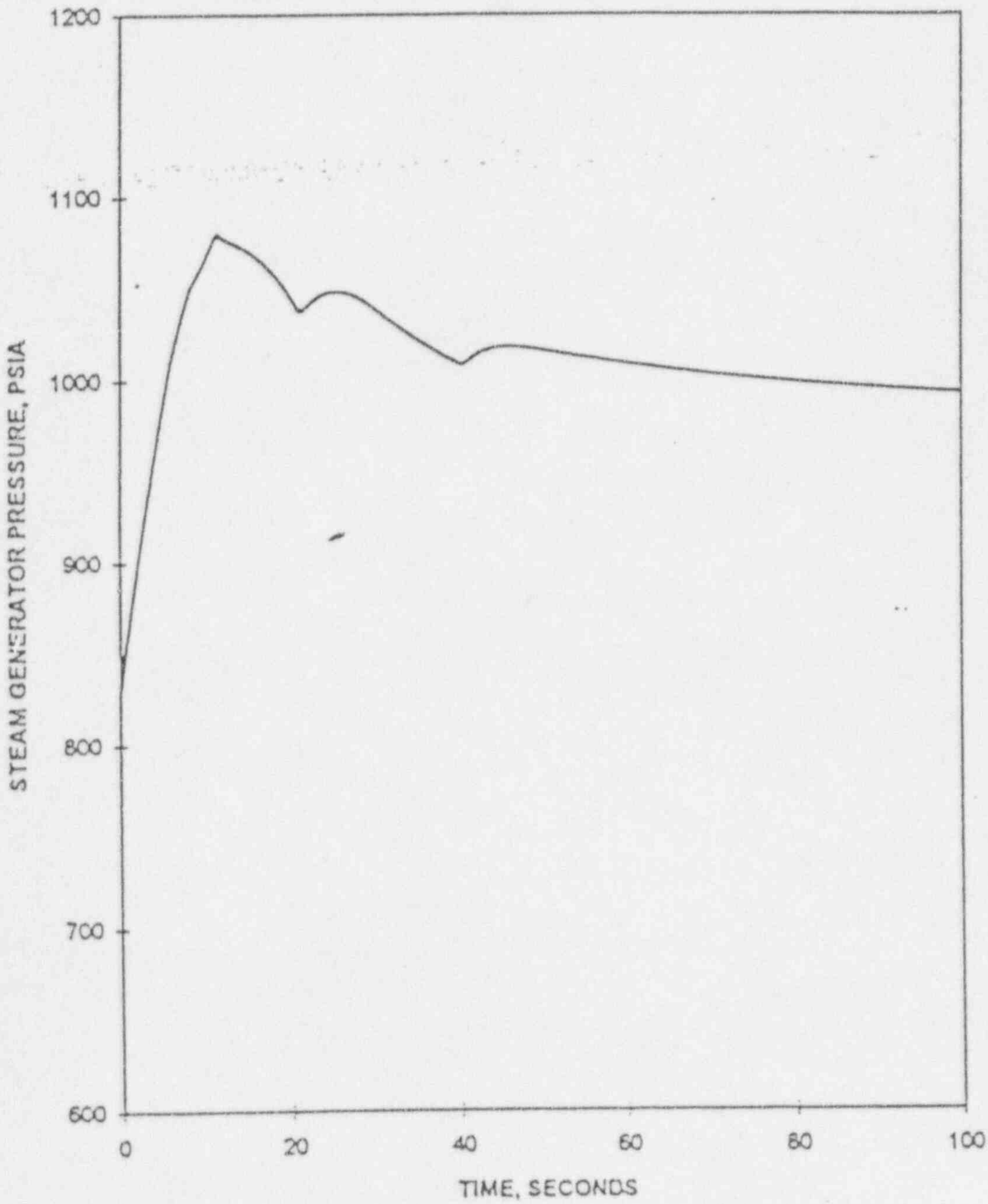


FIGURE 14.5-5

LOSS OF LOAD EVENT

STEAM GENERATOR PRESSURE VERSUS TIME



* Does not include downcomer liquid head.

TABLE 14.6-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE LOFW EVENT TO MAXIMIZE CALCULATED RCS PEAK PRESSURE WITH LOAC

PARAMETER	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2754	2754
Initial Core Coolant Inlet Temperature	$^{\circ}F$	550	550
Initial RCS Vessel Flow Rate	gpm	358,900	358,900
Initial RCS Pressure	psia	2165	2165
Initial SG Pressure	psia	865	865
Initial Pressurizer Liquid Volume	ft^3	975	975
MTC	$X\ 10^{-4}\ \Delta\rho/^{\circ}F$	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
High Pressurizer Pressure Analysis Trip Setpoint	psia	2420	2420
RRS	Operating Mode	Manual[(a)]	Manual[(a)]
SDBS	Operating Mode	Manual[(a)]	Manual[(a)]

TABLE 14.6-1 (Continued)

INITIAL CONDITIONS AND INPUT PARAMETERS
FOR THE LOFW EVENT TO
MAXIMIZE CALCULATED RCS PEAK PRESSURE WITH LOAC

PARAMETER	UNITS	UNIT 1	UNIT 2
PPCS	Operating Mode	Manual[a]	Manual[a]
PLCS	Operating Mode	Manual[a]	Manual[a]

[a] These modes of control system operation maximize the peak RCS pressure.

TABLE 14.6-3

SEQUENCE OF EVENTS FOR LOFW EVENT TO MAXIMIZE CALCULATED RCS PRESSURE WITH LOAC

TIME (sec)	EVENT	SETPOINT OR VALUE
0.0	Loss of MFW	---
21.2	High Pressurizer Pressure Trip Setpoint Reached	2420 psia
22.1	Trip Breakers Open	---
22.4	Turbine Stop Valves Close	---
22.6	CEAs Begin to Drop into Core, Loss of AC Power	---
24.1	SG Safety Valves Begin to Open	1010 psia
24.5	Primary Safety Valves Begin to Open	2550 psia
26.3	Maximum RCS Pressure	2629 psia[a]
29.8	Maximum SG Pressure	1099 psia[b]
31.2	Primary Safety Valves Close	2448 psia

[a] Pressure includes elevation head.
[b] Steam Generator pressure includes downcomer liquid head.

FIGURE 14.6-1

LOFW FLOW EVENT/MAXIMIZE RCS PEAK PRESSURE
WITH LOAC FOLLOWING TRIP
CORE POWER VERSUS TIME

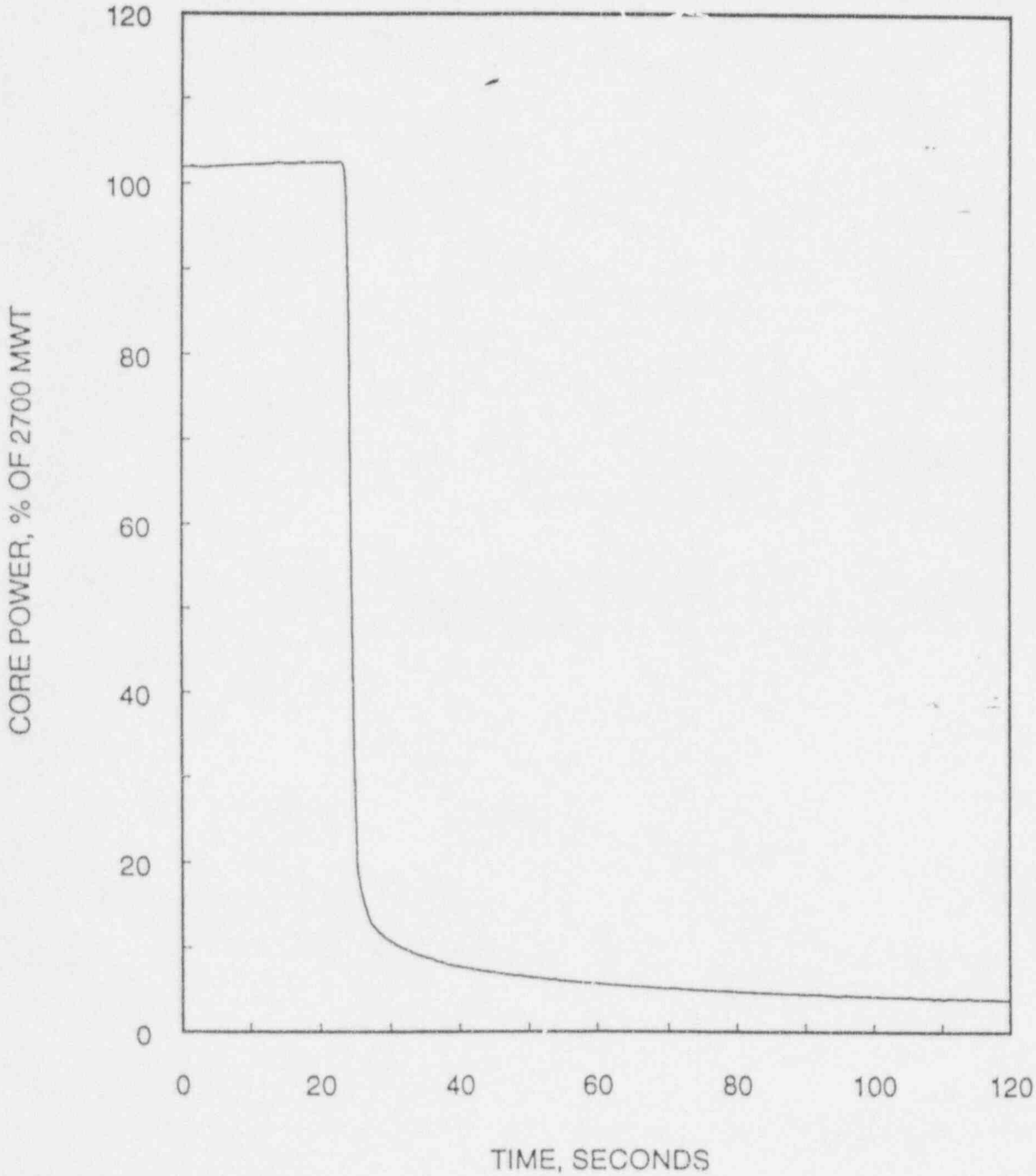


FIGURE 14.6-2

LOFW FLOW EVENT/MAXIMIZE RCS PEAK PRESSURE
WITH LOAC FOLLOWING TRIP
CORE HEAT FLUX VERSUS TIME

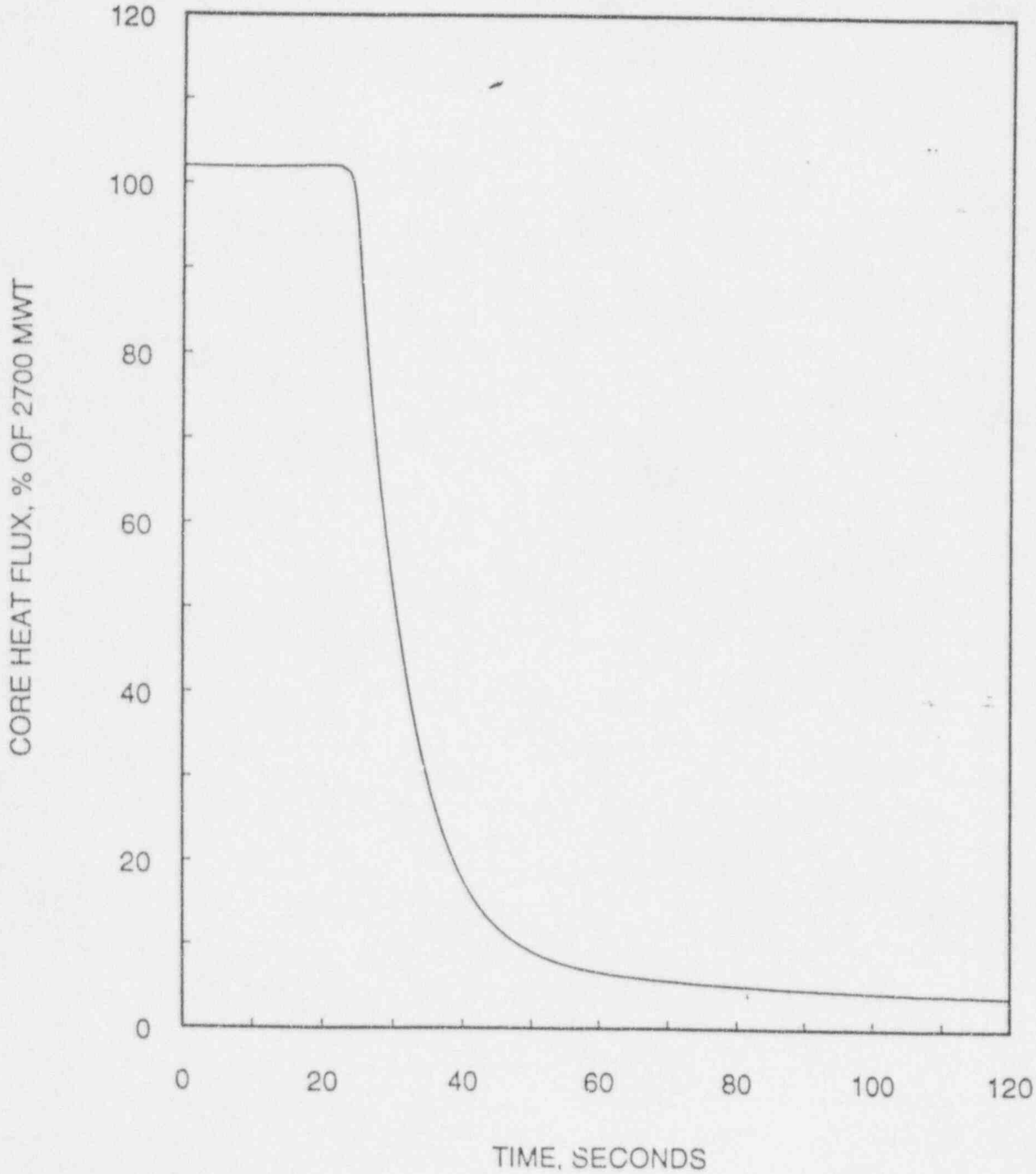


FIGURE 14.6-3

LOFW FLOW EVENT/MAXIMIZE RCS PEAK PRESSURE
WITH LOAC FOLLOWING TRIP
RCS TEMPERATURES VERSUS TIME

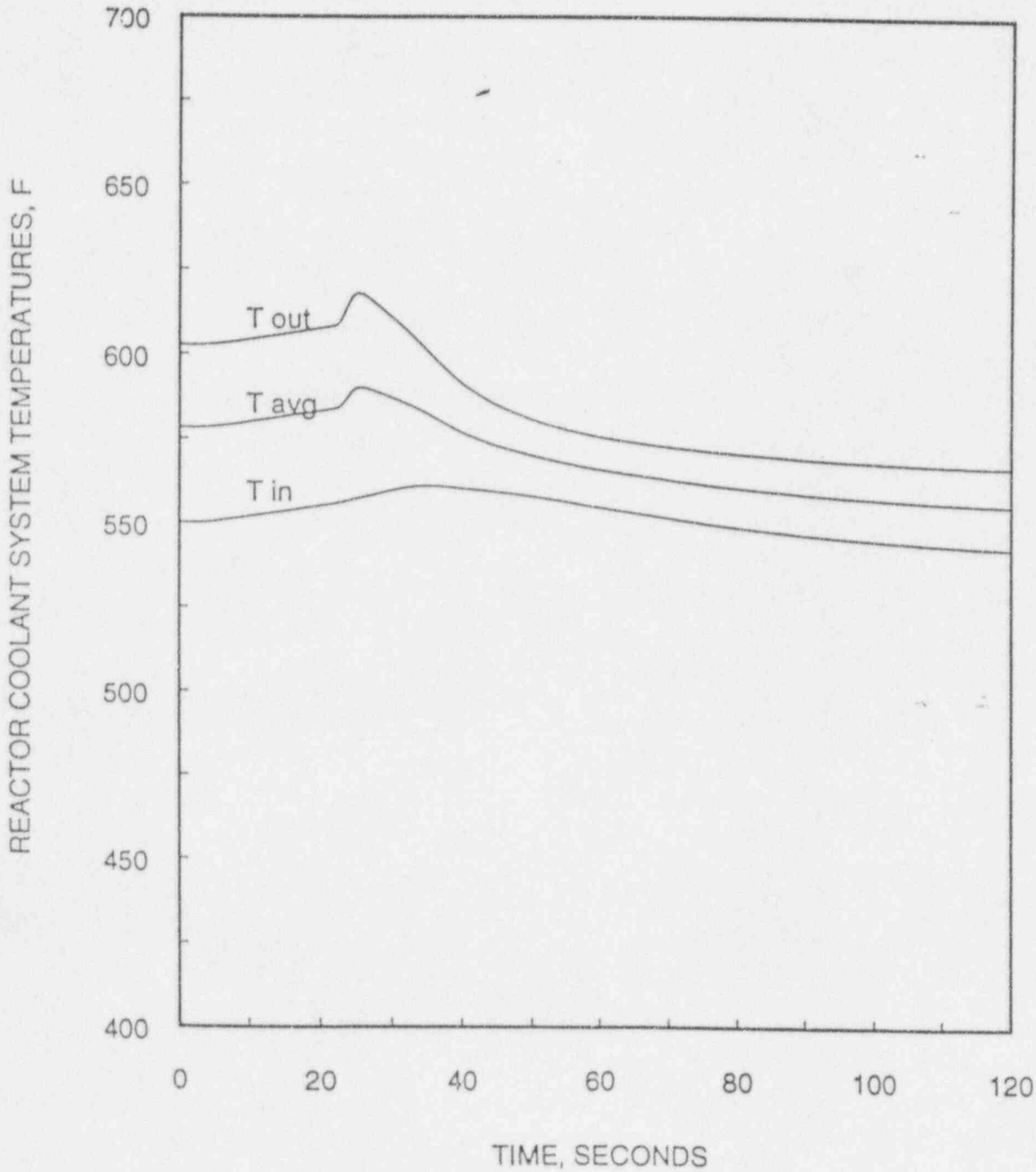


FIGURE 14.6-4

LOFW FLOW EVENT/MAXIMIZE RCS PEAK PRESSURE
WITH LOAC FOLLOWING TRIP
RCS PRESSURE VERSUS TIME

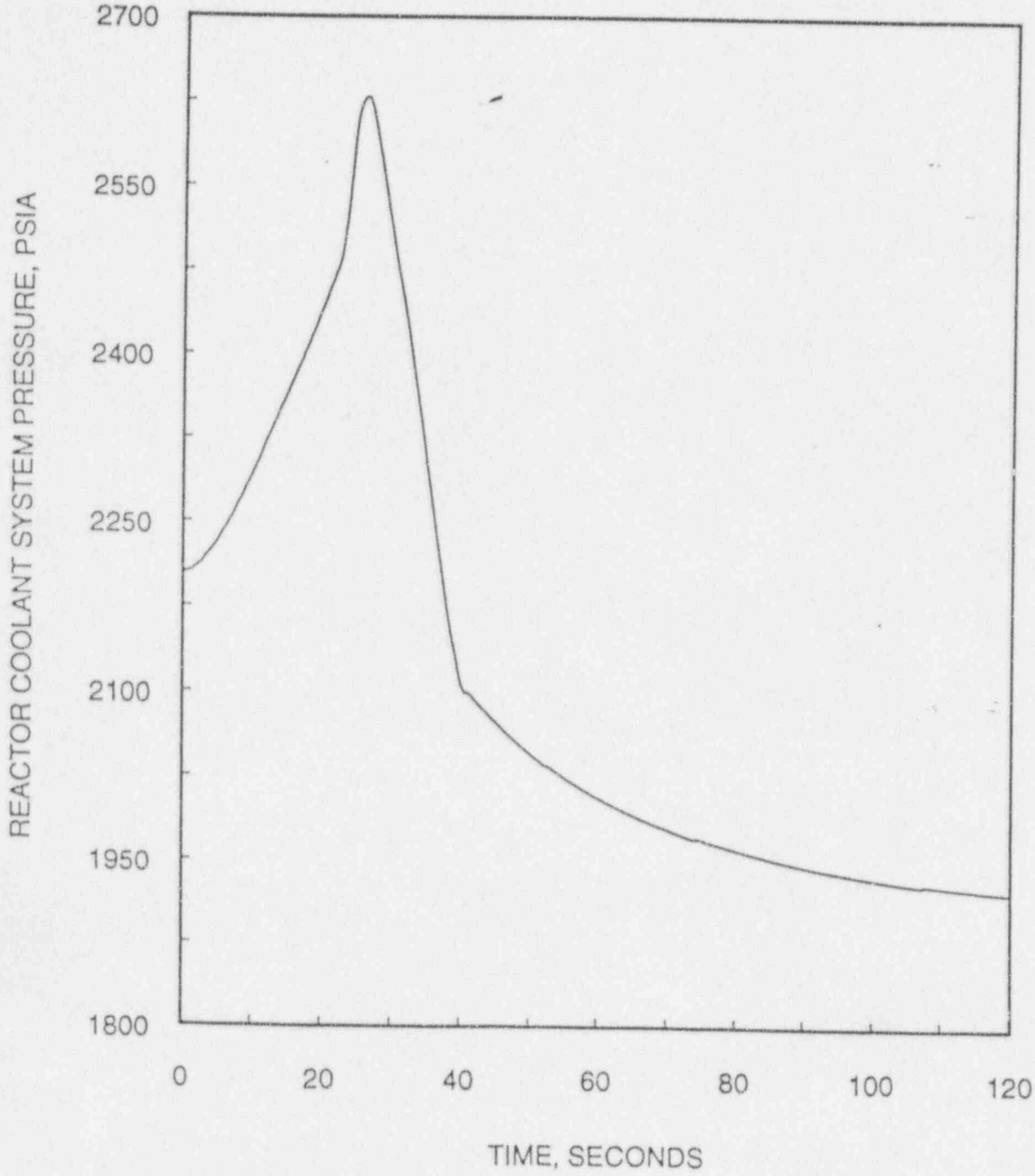


FIGURE 14.6-5

LOFW FLOW EVENT/MAXIMIZE RCS PEAK PRESSURE
WITH LOAC FOLLOWING TRIP
STEAM GENERATOR PRESSURES VERSUS TIME

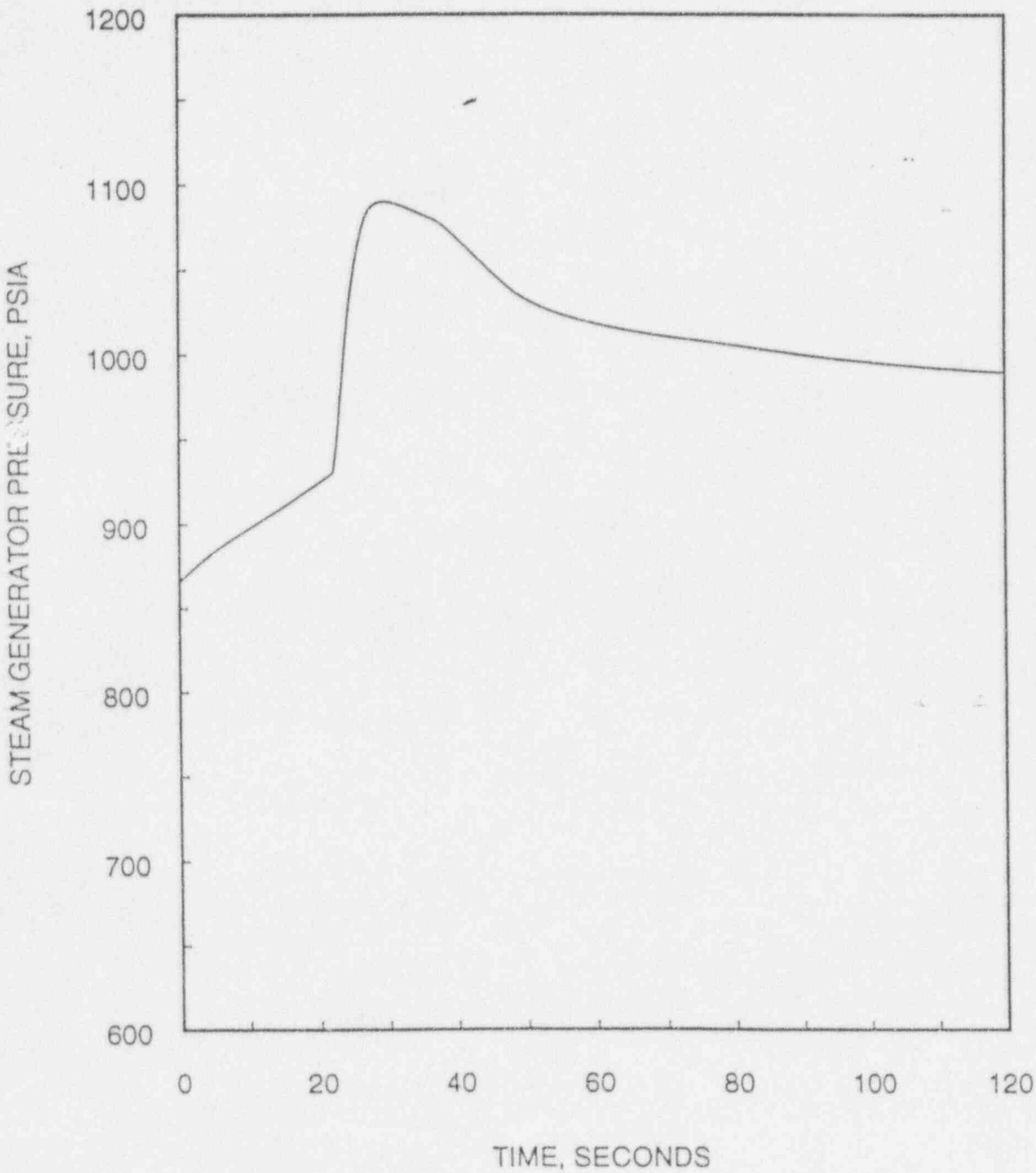


TABLE 14.17-1

CALVERT CLIFFS UNIT 1
COMPARISON OF SIGNIFICANT SYSTEM PARAMETERS

PARAMETERS	VALUES CYCLE 2	VALUES CYCLE 13
Reactor Power Level (102% of Rated), MWth	2754	2754
Average LHR (102% of Rated), kw/ft	6.5474	6.36
MTC at Initial Density, x 10^{-4} $\Delta\rho/^{o}F$	+0.3	+0.3
System Flow Rate (Total), x 10^{6} lbs/hr	139.08	134.68
Core Flow Rate, x 10^{6} lbs/hr	134.21	129.69
Initial System Pressure, psia	2250	2250
Core Inlet Temperature, ^{o}F	550	550
Core Outlet Temperature, ^{o}F	600.6	602
Active Core Height, ft	11.39	11.39
Fuel Rod Outside Diameter, in	0.44	0.44
Number of Cold Legs	4	4
Number of Hot Legs	2	2
Cold Leg Diameter, in	30	30
Hot Leg Diameter, in	42	42

TABLE 14.17-1 (Continued)

CALVERT CLIFFS UNIT 1
COMPARISON OF SIGNIFICANT SYSTEM PARAMETERS

PARAMETERS	VALUES CYCLE 2	CYCLE 13
SIT Pressure, psia	215	195
SI Response Time, sec	30	40
Number of Tubes Plugged per SG		
Large Breaks	0	≤ 1500
Small Breaks	0	≤ 2130
SIT Gas/Water Volume, ft^3	855/1145	910/1090
PLHGR, kw/ft	14.2[a] 16.5	14.5
Gap Conductance at PLHGR, Btu/hr-ft^2-^oF[b]	683.4[a] 2000.0	1851
Fuel Centerline Temperature at PLHGR, ^oF[b]	3894.8[a] 3788.3	3492
Fuel Average Temperature at PLHGR, ^oF[b]	2609.3[a] 2303.6	2160
Hot Rod Gas Pressure, psia[b]	1198.9[a]	1208

TABLE 14.17-1 (Continued)

CALVERT CLIFFS UNIT 1
COMPARISON OF SIGNIFICANT SYSTEM PARAMETERS

PARAMETERS	VALUES CYCLE 2	CYCLE 13
Hot Average Rod Burnup (Min Hgap), MWD/T[b]	3402[a] 680	1000

[a] For low density fuel, when gap conductance is minimum.

[b] Fuel rod values given are those which yield the limiting ECCS performance results.

TABLE 14.17-2

CALVERT CLIFFS UNIT 2
COMPARISON OF SIGNIFICANT SYSTEM PARAMETERS

PARAMETERS	VALUES	
	CYCLE 2	CYCLE 11[a]
Reactor Power Level (102% of Nominal), MWth	2754	2754
Average LHR (102% of Nominal), kw/ft	6.5205	6.36
MTC at Initial Density, x 10^{-4} $\Delta\rho/^{o}F$	+0.3	+0.3
System Flow Rate (Total), x 10^{6} lbs/hr	139.08	134.68
Core Flow Rate, x 10^{6} lbs/hr	134.21	129.69
Initial System Pressure, psia	2250	2250
Core Inlet Temperature, ^{o}F	550	550
Core Outlet Temperature, ^{o}F	600.6	602
Active Core Height, ft	11.39	11.39
Fuel Rod Outside Diameter, in	0.44	0.44
Number of Cold Legs	4	4
Number of Hot Legs	2	2
Cold Leg Diameter, in	30	30
Hot Leg Diameter, in	42	42

TABLE 14.17-2 (Continued)

CALVERT CLIFFS UNIT 2
COMPARISON OF SIGNIFICANT SYSTEM PARAMETERS

PARAMETERS	VALUES CYCLE 2	CYCLE 11[a]
SIT Pressure, psia	215	195
SI Response Time, sec	30	40
Number of Tubes Plugged per Generator		
Large Break	0	≤ 1500
Small Break	0	≤ 2130
SIT Gas/Water Volume, ft^3	855/1145	910/1090
PLHGR, kw/ft	15.5	14.5
Gap Conductance at PLHGR, Btu/hr-ft^2-^oF[b]	2000.0	1851
Fuel Centerline Temperature at PLHGR, ^oF[b]	3528.4	3492
Fuel Average Temperature at PLHGR, ^oF[b]	2126.3	2160
Hot Rod Gas Pressure, psia[b]	1636.9	1208
Hot Average Rod Burnup (Min h_{gap}), MWD/T[b]	27506	1000

(a) The Unit 1 Cycle 13 data conservatively apply for Unit 2 Cycle 11.

(b) Fuel rod values given are those which yield the limiting ECCS performance results.

TABLE 14.17-2 (Continued)

CALVERT CLIFFS UNIT 2
COMPARISON OF SIGNIFICANT SYSTEM PARAMETERS

PARAMETERS	VALUES CYCLE 2	CYCLE 11[a]
SIT Pressure, psia	215	195
SI Response Time, sec	30	40
Number of Tubes Plugged per Generator		
Large Break	0	≤ 1500
Small Break	0	≤ 2130
SIT Gas/Water Volume, ft^3	855/1145	910/1090
PLHGR, kw/ft	15.5	14.5
Gap Conductance at PLHGR, Btu/hr-ft^2-°F[b]	2000.0	1851
Fuel Centerline Temperature at PLHGR, °F[b]	3528.4	3492
Fuel Average Temperature at PLHGR, °F[b]	2126.3	2160
Hot Rod Gas Pressure, psia[b]	1636.9	1208
Hot Average Rod Burnup (Min h_{gap}), MWD/T[b]	27506	1000

[a] The Unit 1 Cycle 13 data conservatively apply for Unit 2 Cycle 11.
[b] Fuel rod values given are those which yield the limiting ECCS performance results.

TABLE 14.17-6

TIME OF INTEREST AND FUEL ROD
PERFORMANCE SUMMARY FOR 0.1 FT^2 BREAK
FOR UNIT 1 CYCLE 13

Time for HPSI pump on	68 sec
Time for Low Pressure Safety Injection (LPSI) pump and SI tanks on	[a]
Time for SI H_2O level to reach bottom of fuel	[b]
Hot spot PCT occurs	1473 sec
Maximum clad surface temperature	2031°F
Elevation of hot spot (from bottom of core)	10.82 ft
Core-wide zirconium oxidation[c]	< 0.95%
Peak local clad zirconium oxidation	6.40%

(a) Calculation terminated before LPSI pump or SI tank actuation.
(b) Core never totally uncovers.
(c) The average oxidation over the length of the hot rod is used as a conservative representation of the core-wide zirconium oxidation.

TABLE 14.17-14

CALVERT CLIFFS UNIT 1 AND 2
COMPARISON TO NRC ACCEPTANCE CRITERIA

CYCLE	LIMITING BREAK	MAXIMUM ALLOWABLE PLHGR, kw/ft	CRITERION(1) PCT, °F	CRITERION(2) MAX. CLAD OXIDATION, %	CRITERION(3) MAX. HYDROGEN GENERATION, %
			UNIT 1		
2	0.8xDES/PD	14.2	2145	7.2	less than 0.541
13	0.6xDEG/PD	14.5	2143	7.27	less than 0.51
			UNIT 2[a]		
11	0.6xDEG/PD	14.5	2143	7.27	less than 0.51

Criterion (1) Peak Clad Temperature. "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

Criterion (2) Maximum Cladding Oxidation. "The calculated total oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation."

Criterion (3) Maximum Hydrogen Generation. "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

[a] The Unit 1 Cycle 13 results conservatively apply for Unit 2 Cycle 11.

TABLE 14.17-15

SUMMARY OF ECCS PERFORMANCE RESULTS FOR THE LIMITING BREAK SIZE (0.6xDEG/PD) (for Unit 1 Cycle 13 and Unit 2 Cycle 11)

PARAMETERS	LIMITING CASE (MAXIMUM INITIAL FUEL STORED ENERGY) UNIT 1 CYCLE 13	UNIT 2[a] CYCLE 11
Rod Average Burnup MWD/MTU	1000	1000
PCT, °F	2143	2143
Time of PCT, seconds	266	266
Time of Clad Rupture, seconds	26.0	26.0
Peak Clad Oxidation, %	7.27	7.27
Core Wide Oxidation, %	<.51	<.51
Maximum Allowable Peak Linear Heat Generation Rate, kw/ft	14.5	14.5

[a] The Unit 1 Cycle 13 results conservatively apply for Unit 2 Cycle 11.

FIGURE 14.17-260

0.1 SQ FT BREAK IN PUMP DISCHARGE LEG,

NORMALIZED TOTAL CORE POWER VS TIME

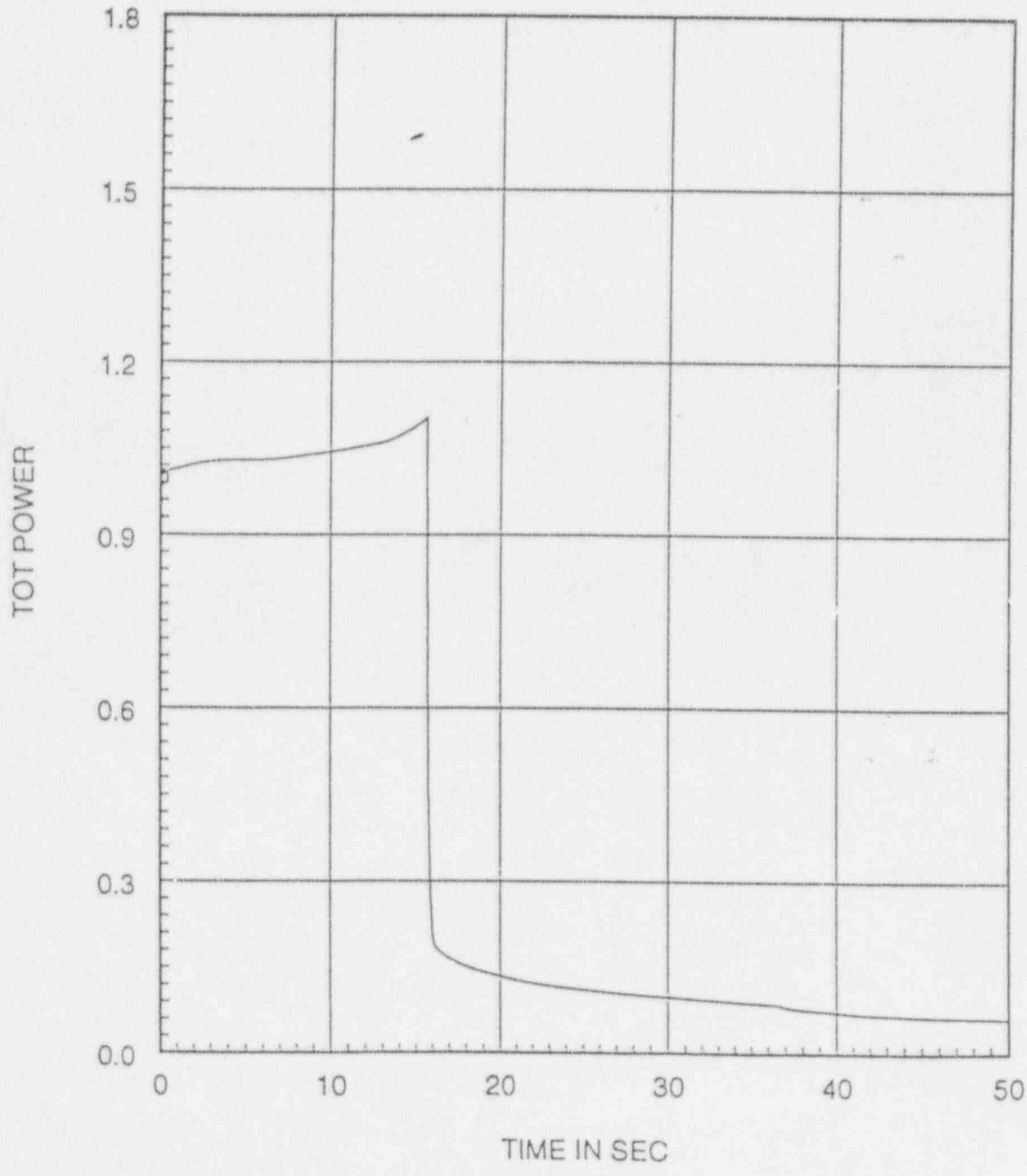


FIGURE 14.17-261

0.1 SQ FT BREAK IN PUMP DISCHARGE LEG,

INNER VESSEL PRESSURE VS TIME

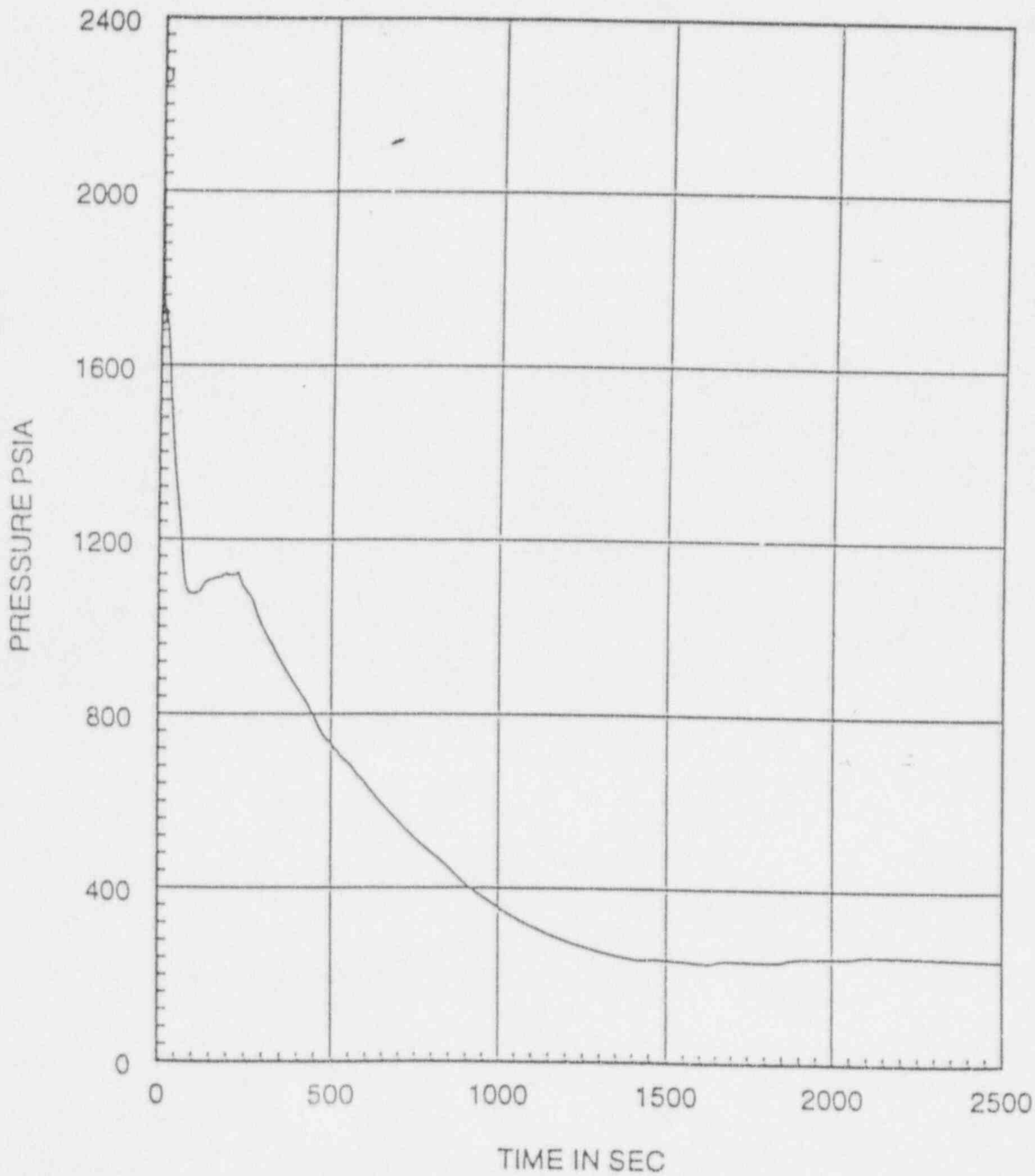


FIGURE 14.17-262

0.1 SQ FT BREAK IN PUMP DISCHARGE LEG,

LEAK FLOW RATE VS TIME

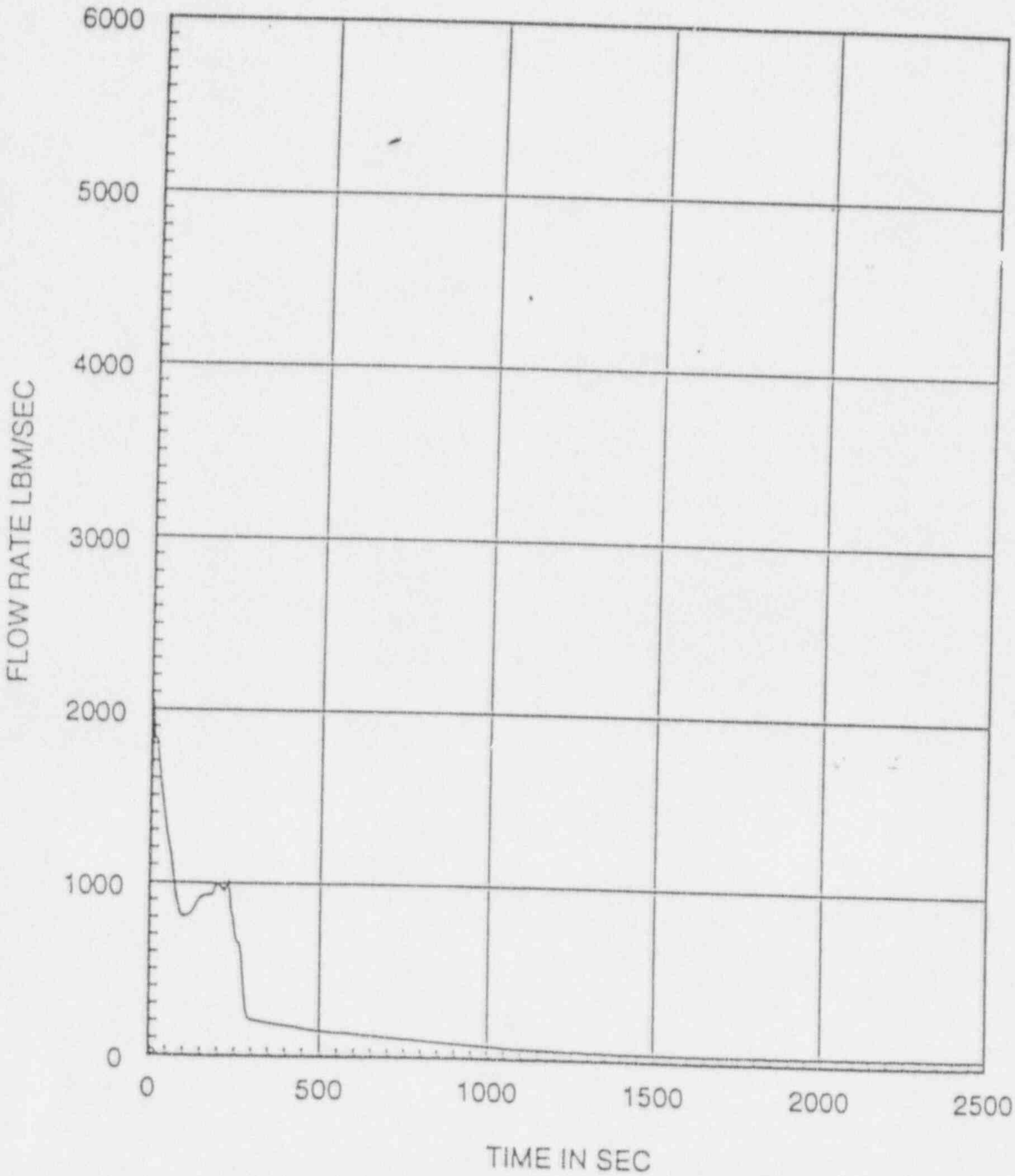


FIGURE 14.17-263

0.1 SQ FT BREAK IN PUMP DISCHARGE LEG,

INNER VESSEL INLET FLOW RATE VS TIME

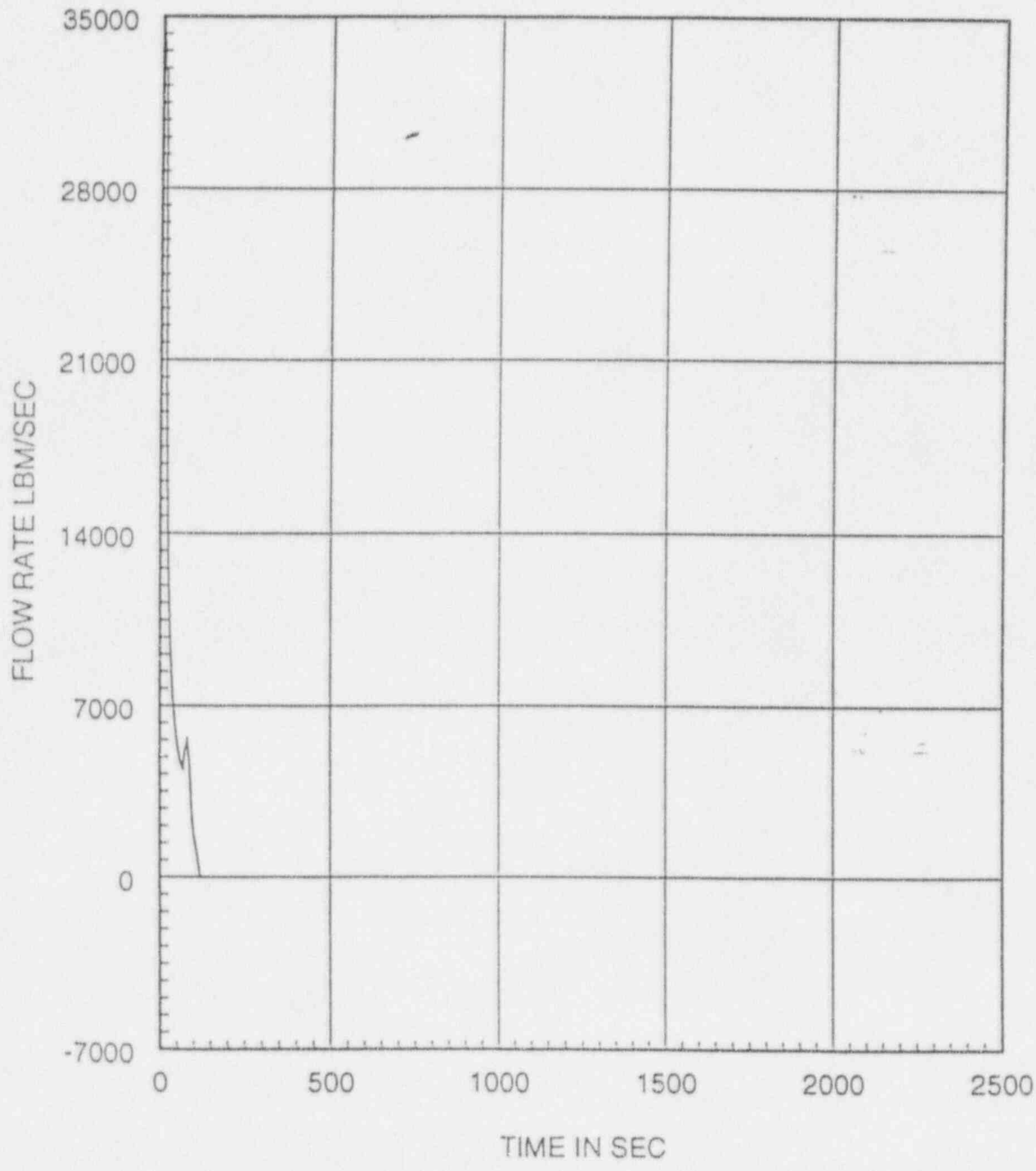


FIGURE 14.17-264

0.1 SQ FT BREAK IN PUMP DISCHARGE LEG,

INNER VESSEL TWO PHASE MIXTURE HEIGHT

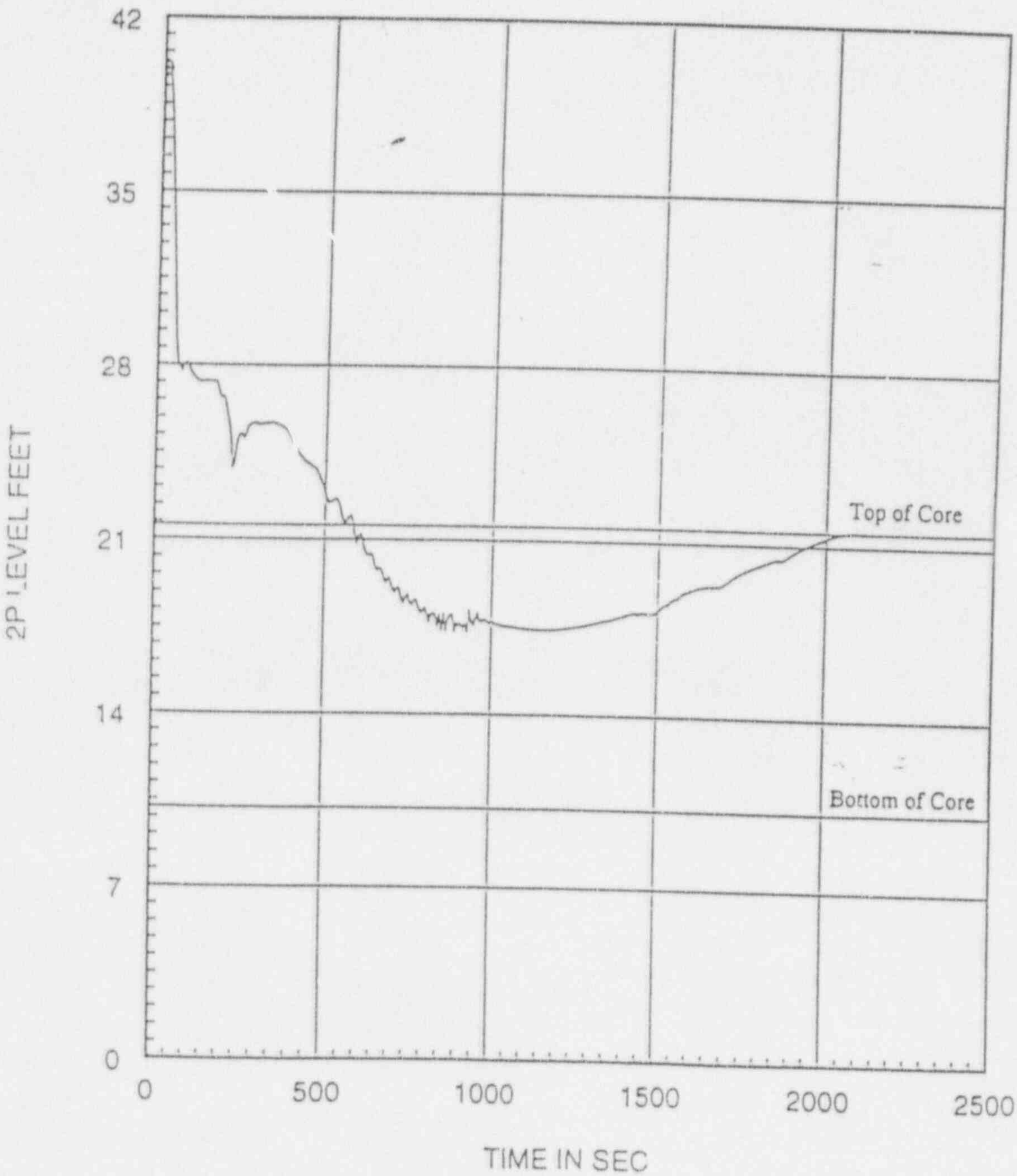


FIGURE 14.17-265

0.1 SQ FT BREAK IN PUMP DISCHARGE LEG,

HEAT TRANSFER COEFFICIENT AT LOCATION OF PCT

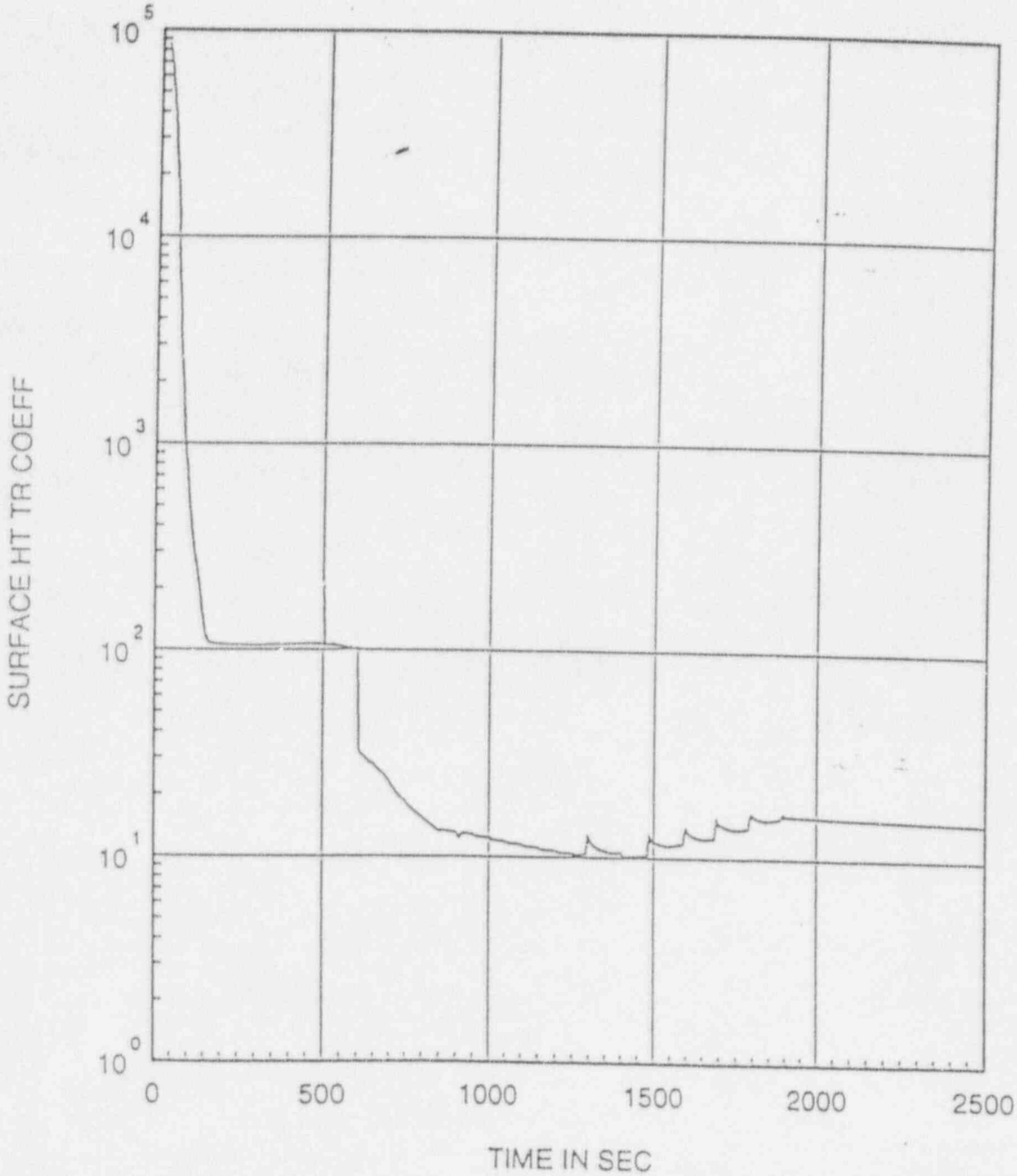


FIGURE 14.17-266

0.1 SQ FT BREAK IN PUMP DISCHARGE LEG,

COOLANT TEMPERATURE AT LOCATION OF PCT

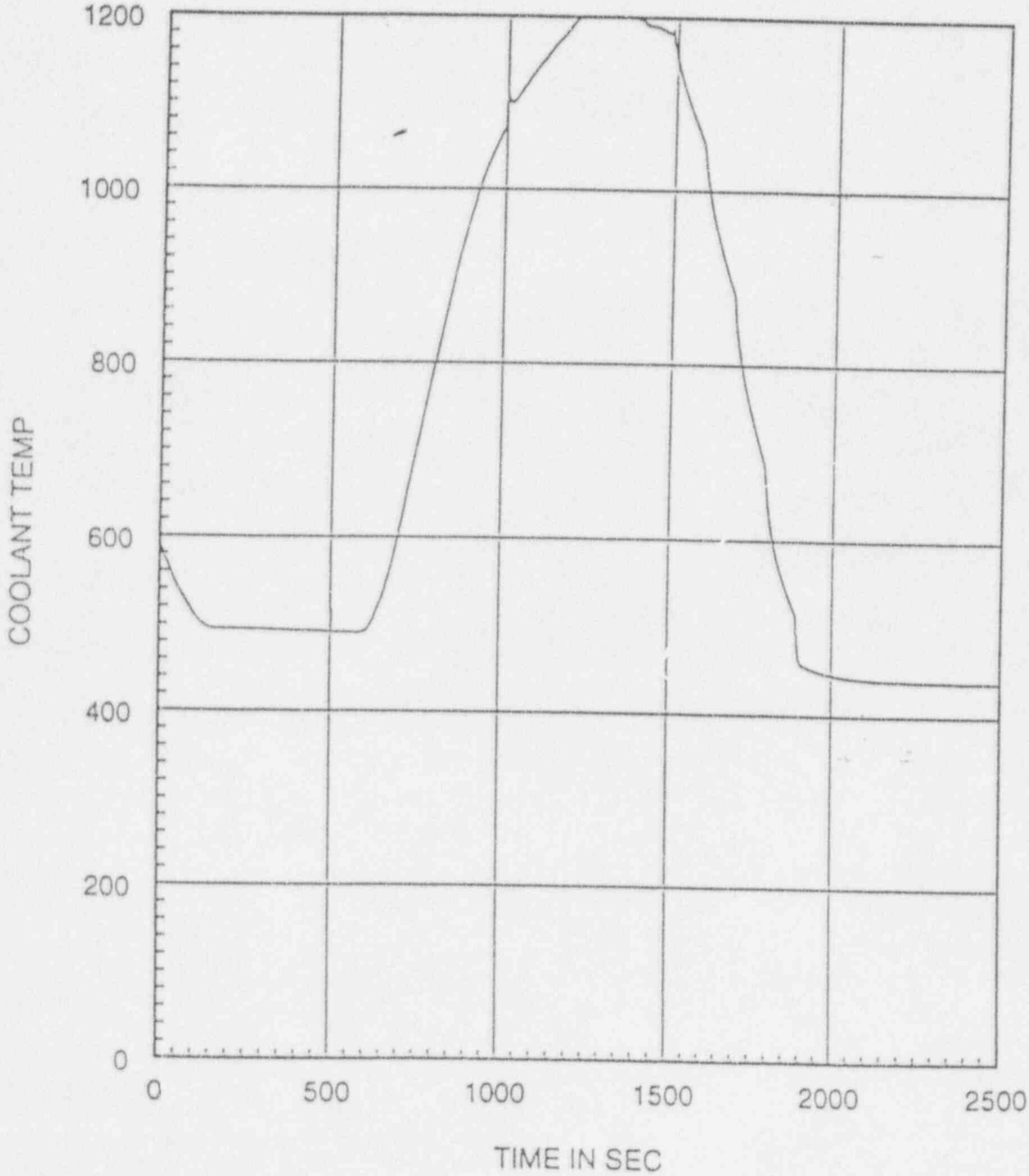


FIGURE 14.17-267

0.1 SQ FT BREAK IN PUMP DISCHARGE LEG,

PEAK CLADDING TEMPERATURE VS TIME

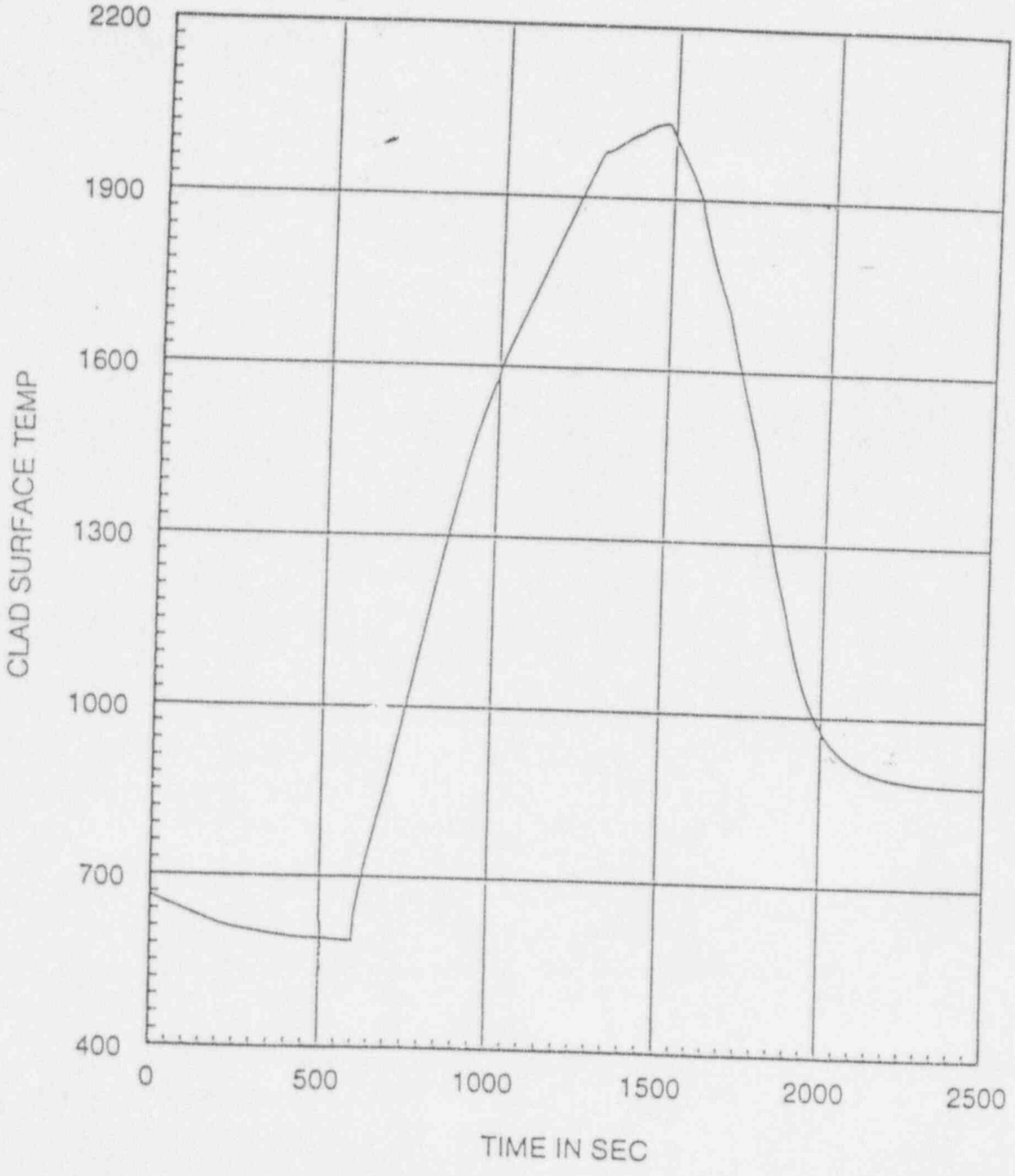


FIGURE 14.17-268

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

CORE POWER VS TIME

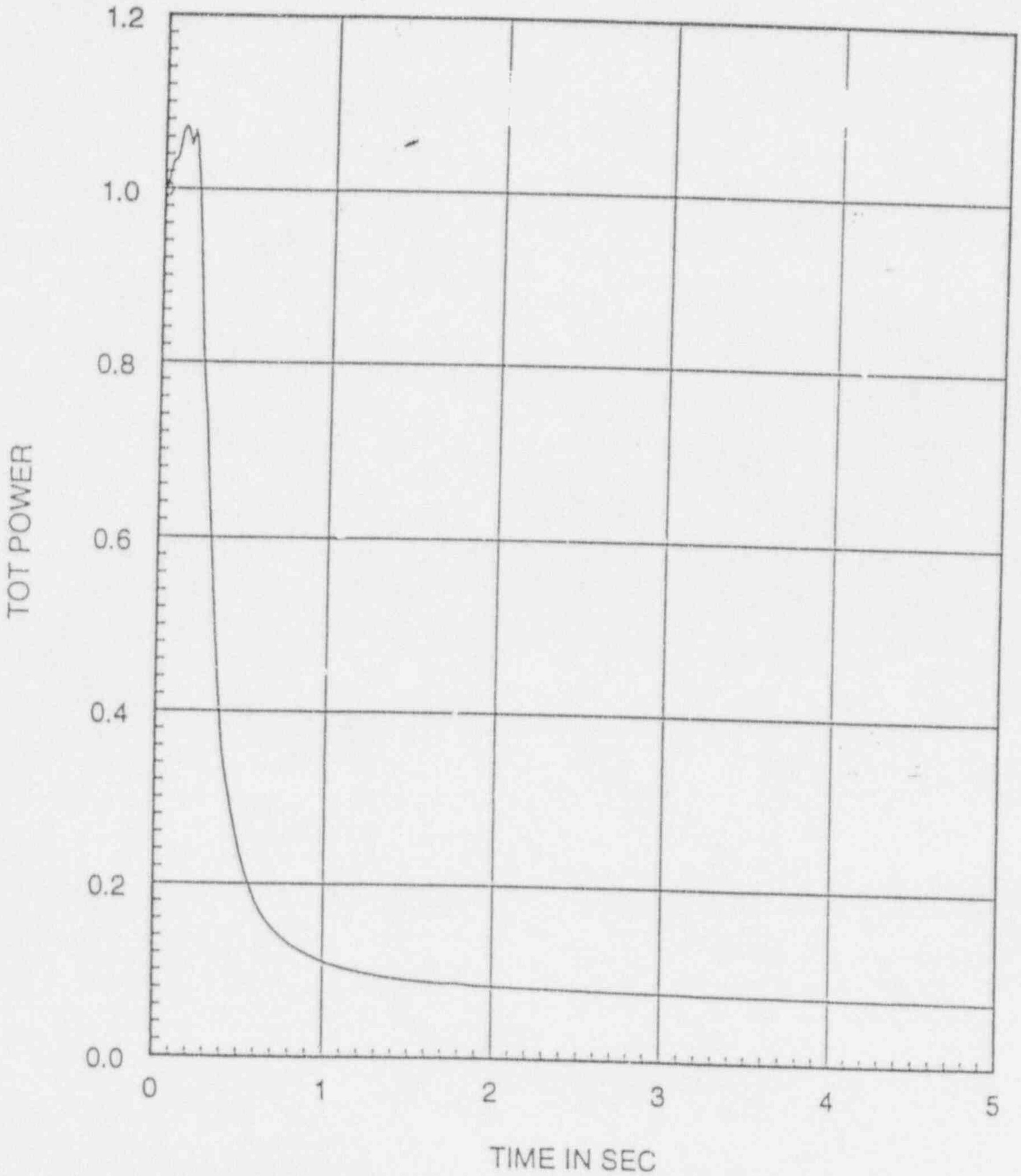


FIGURE 14.17-269

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

PRESSURE IN CENTER HOT ASSEMBLY NODE VS TIME

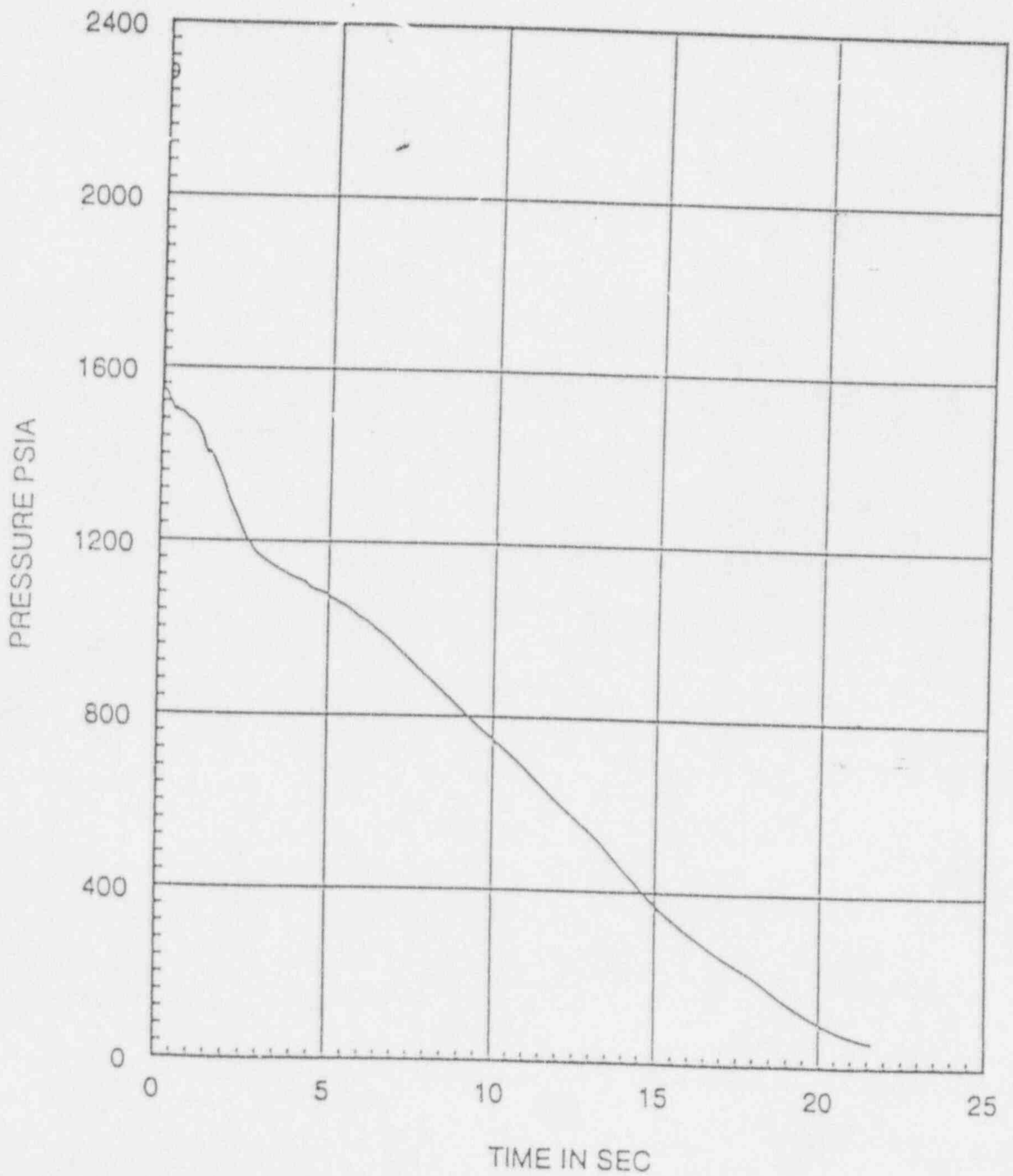


FIGURE 14.17-270

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

REACTOR COOLANT PUMP SIDE LEAK FLOW RATE VS TIME

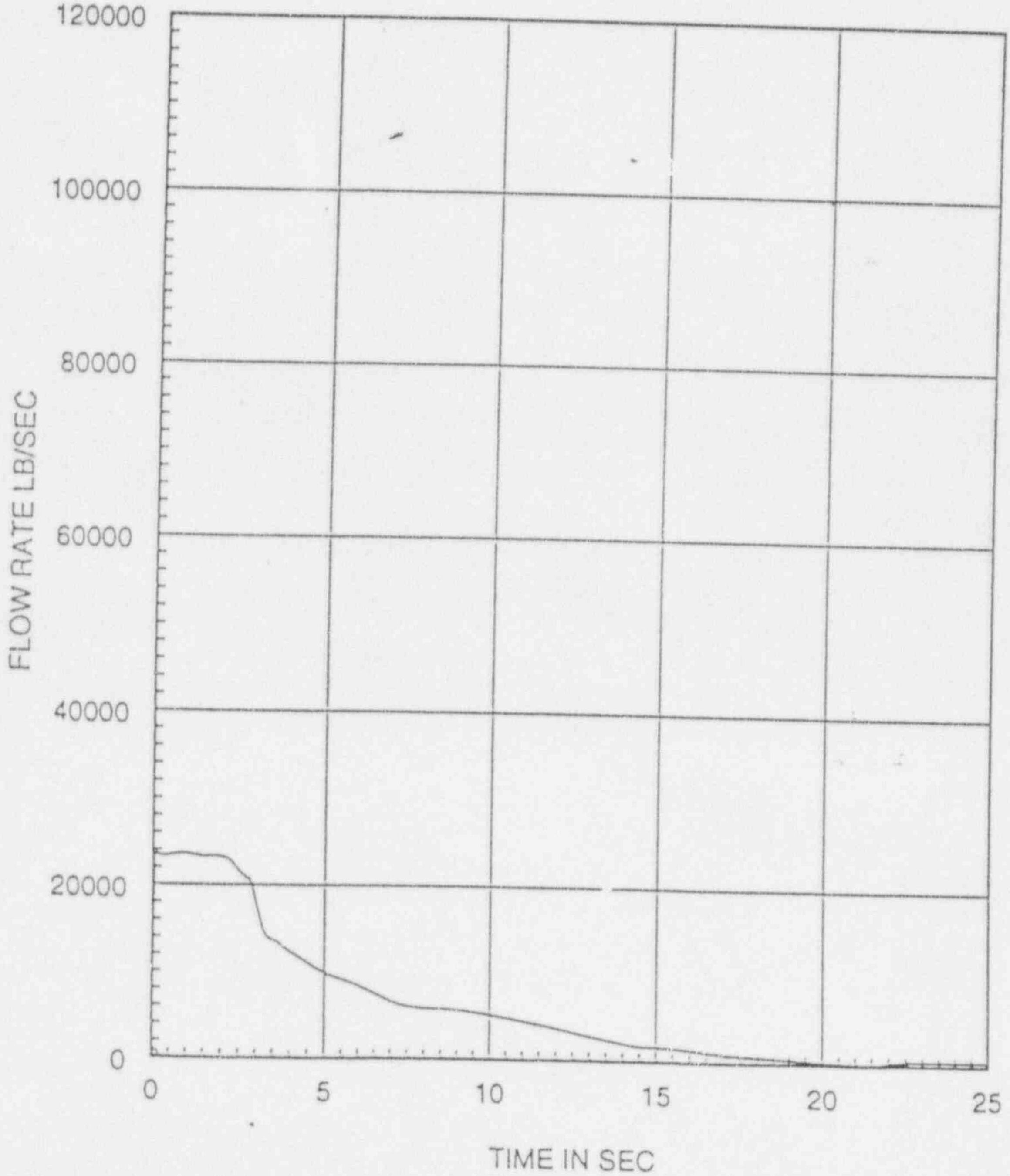


FIGURE 14.17-271

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

REACTOR VESSEL SIDE LEAK FLOW RATE VS TIME

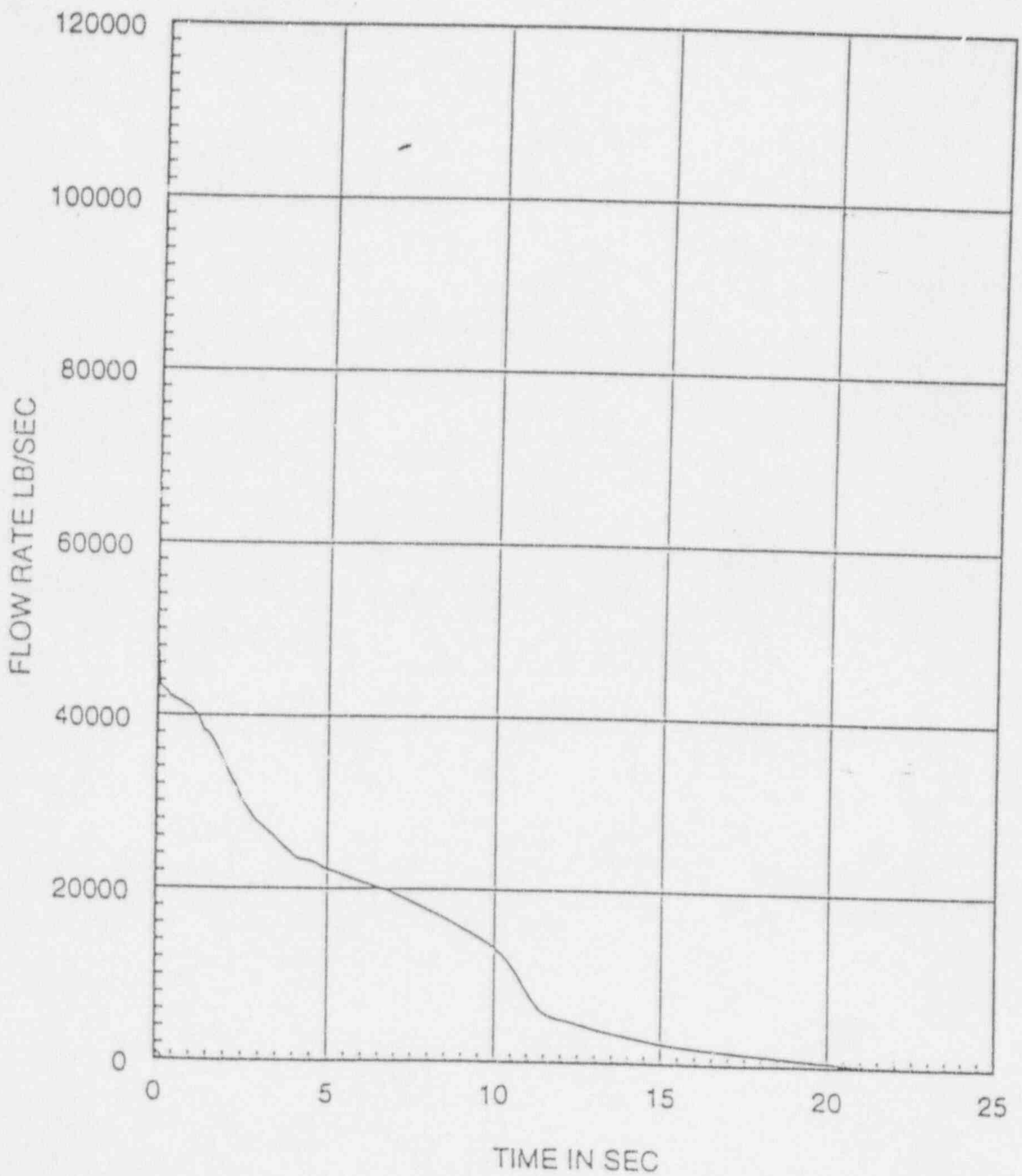


FIGURE 14.17-272

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

HOT ASSEMBLY FLOW RATE (BELOW HOT SPOT) VS TIME

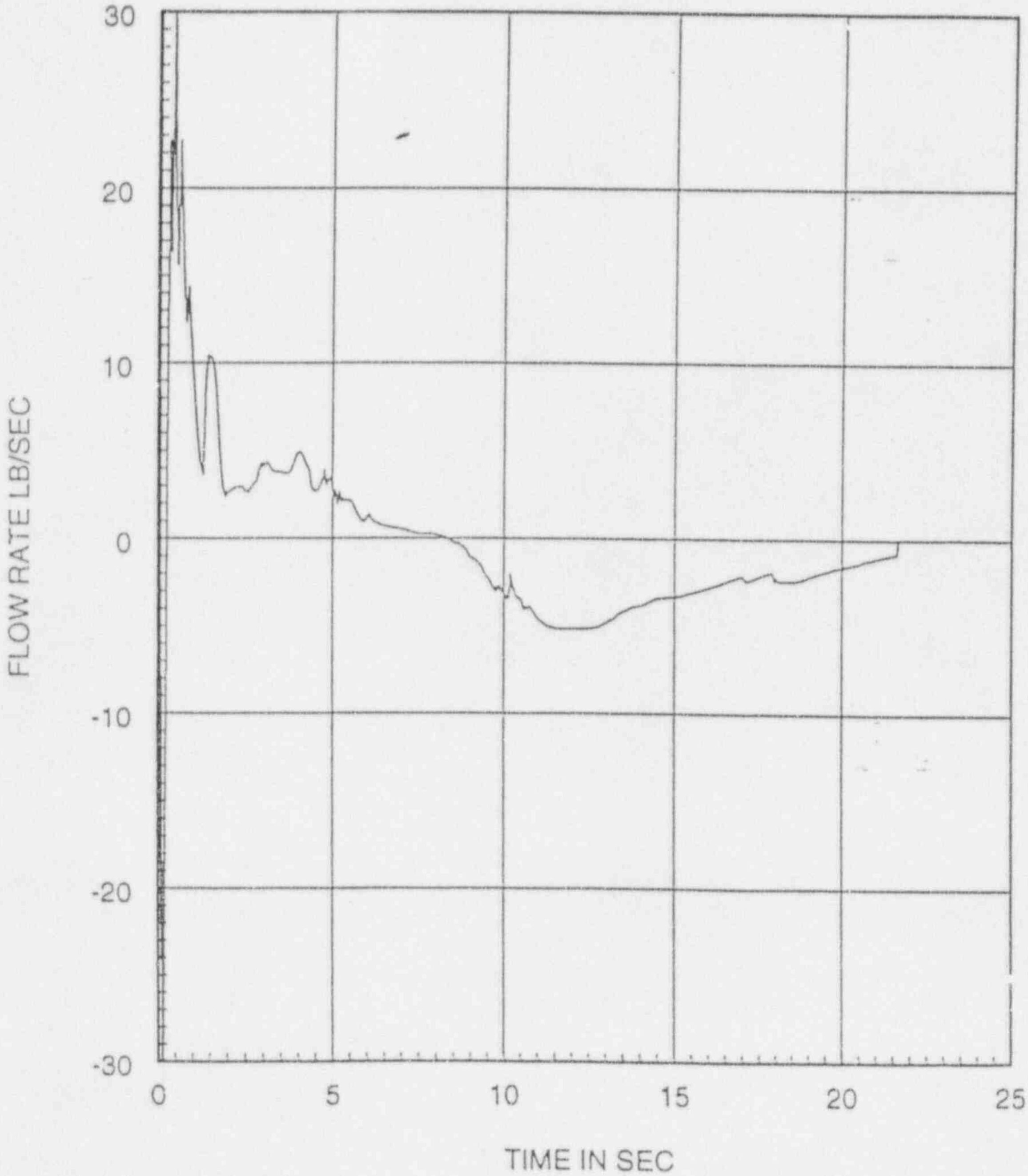


FIGURE 14.17-273

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

HOT ASSEMBLY FLOW RATE (ABOVE HOT SPOT) VS TIME

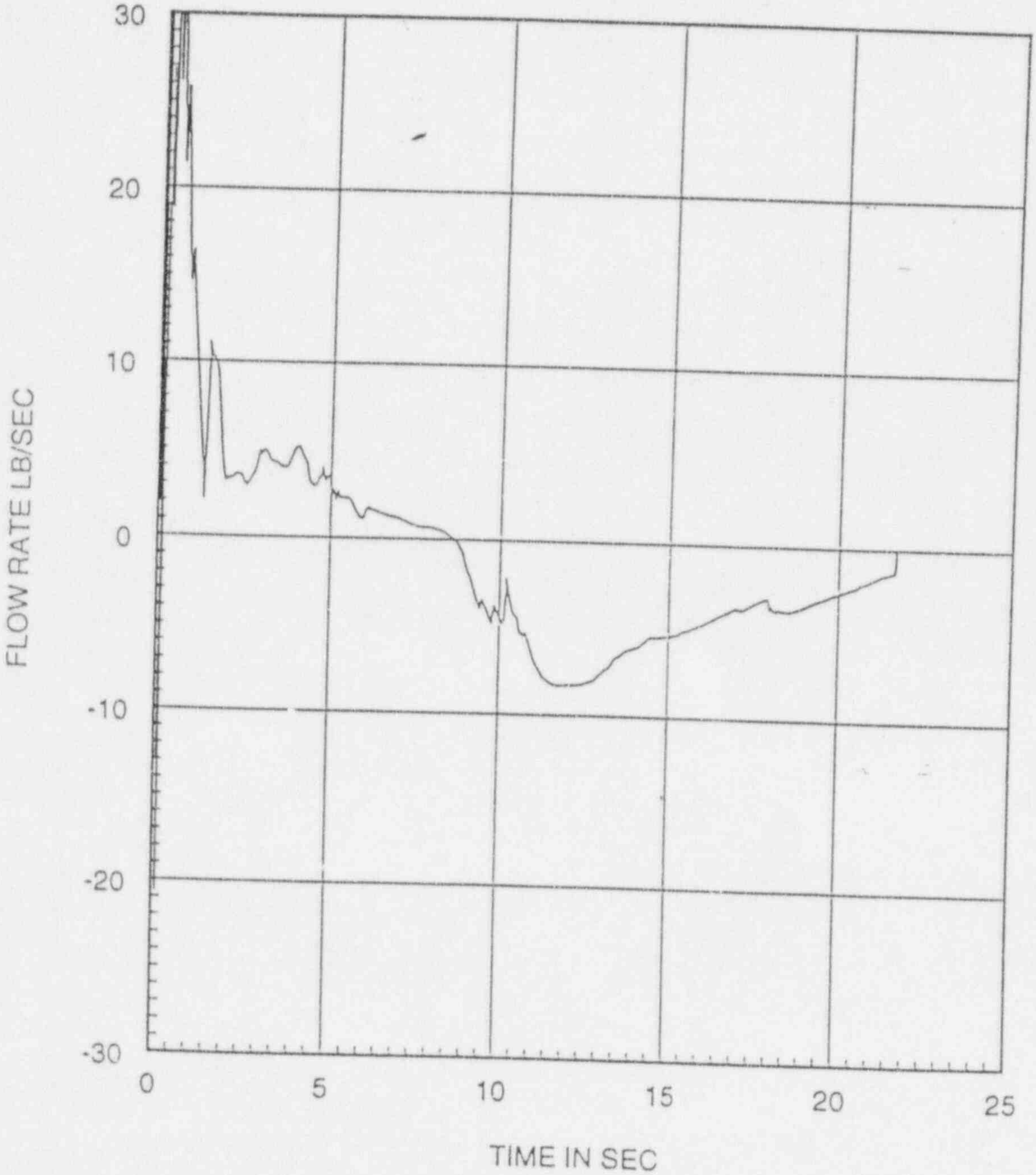


FIGURE 14.17-274

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

HOT ASSEMBLY QUALITY (BELOW HOTTEST REGION) VS TIME

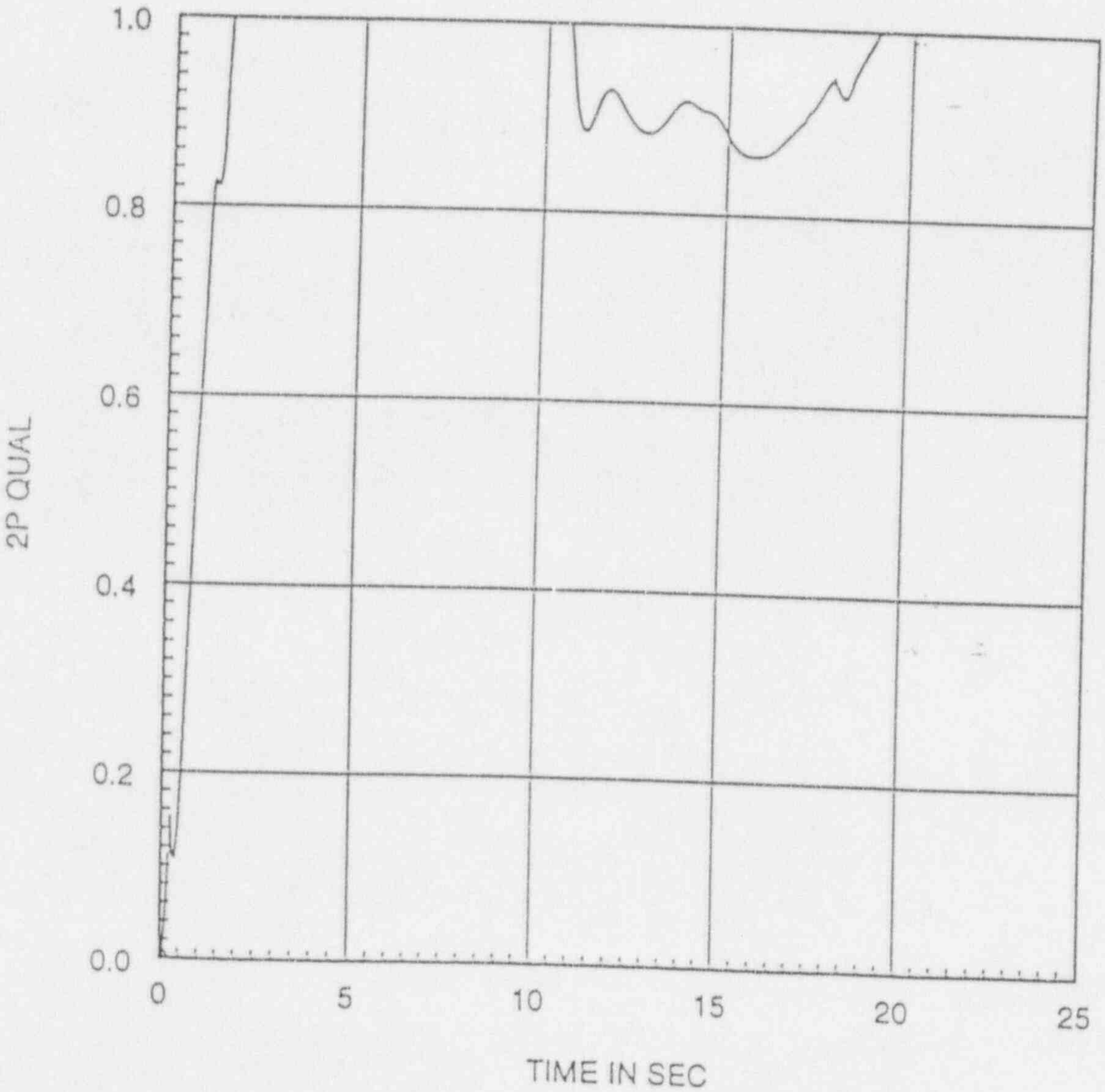


FIGURE 14.17-275

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

HOT ASSEMBLY QUALITY (AT HOTTEST REGION) VS TIME

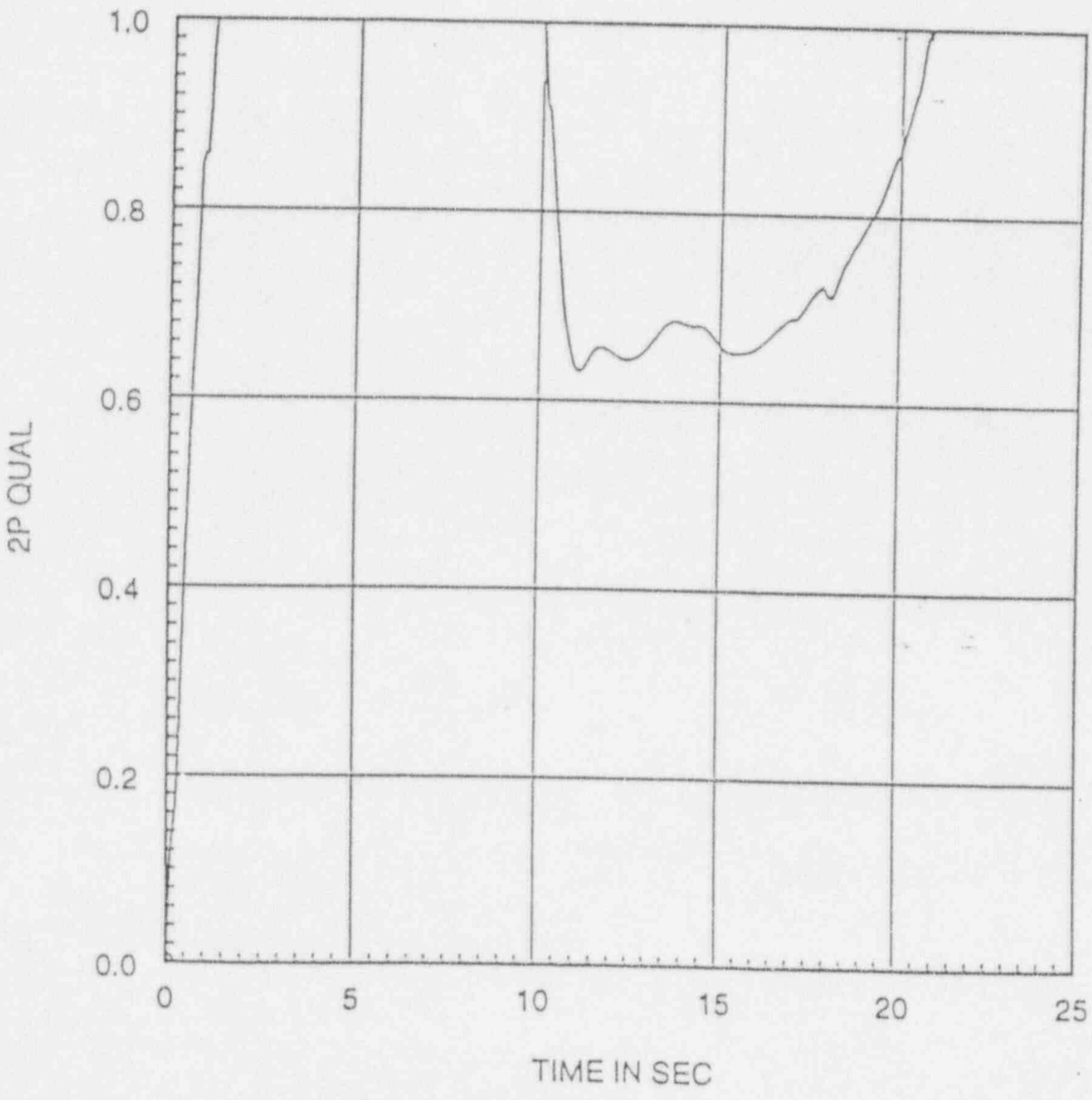


FIGURE 14.17-276

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

HOT ASSEMBLY QUALITY (ABOVE HOTTEST REGION) VS TIME

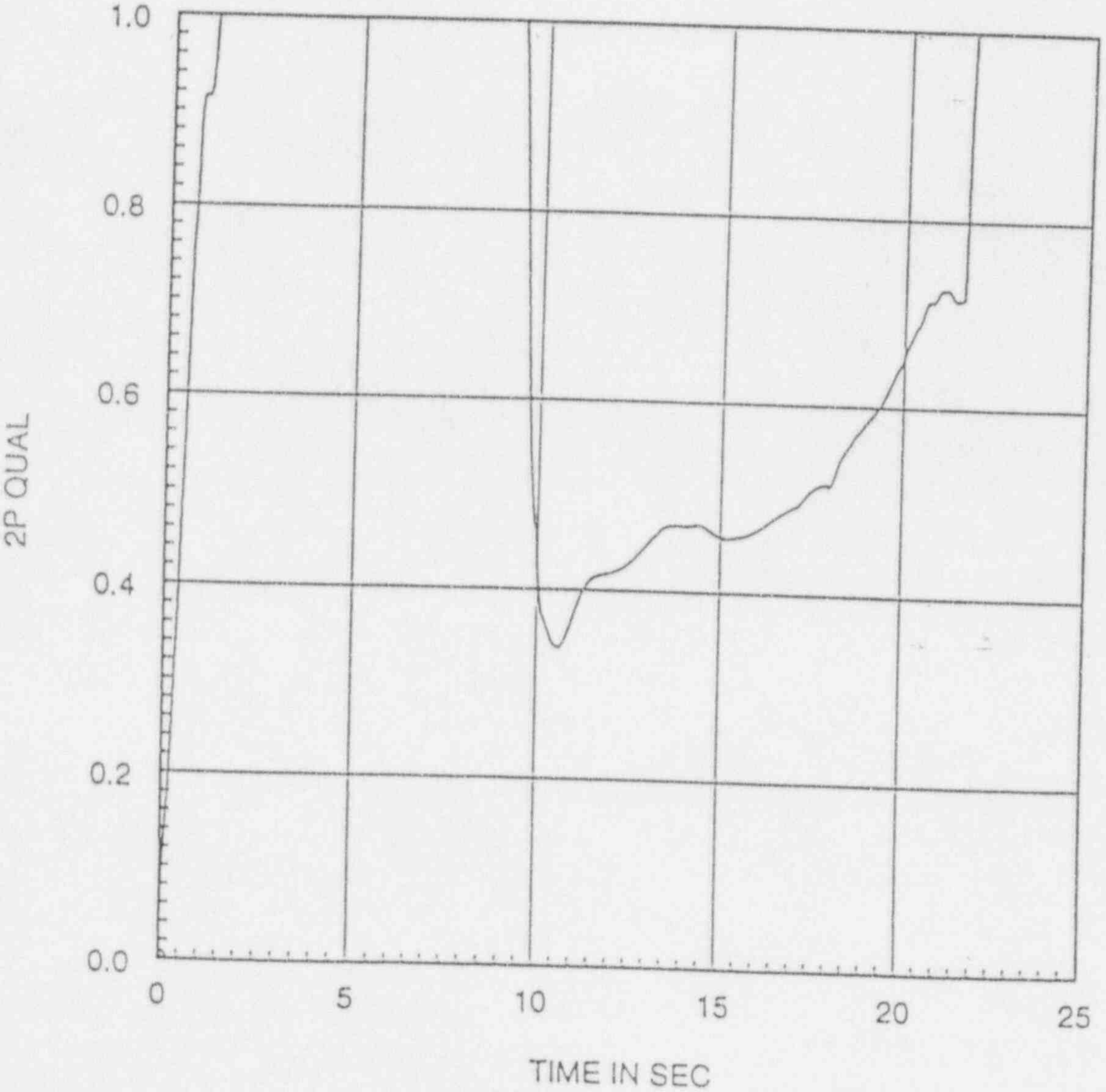


FIGURE 14.17-277

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

CONTAINMENT PRESSURE VS TIME

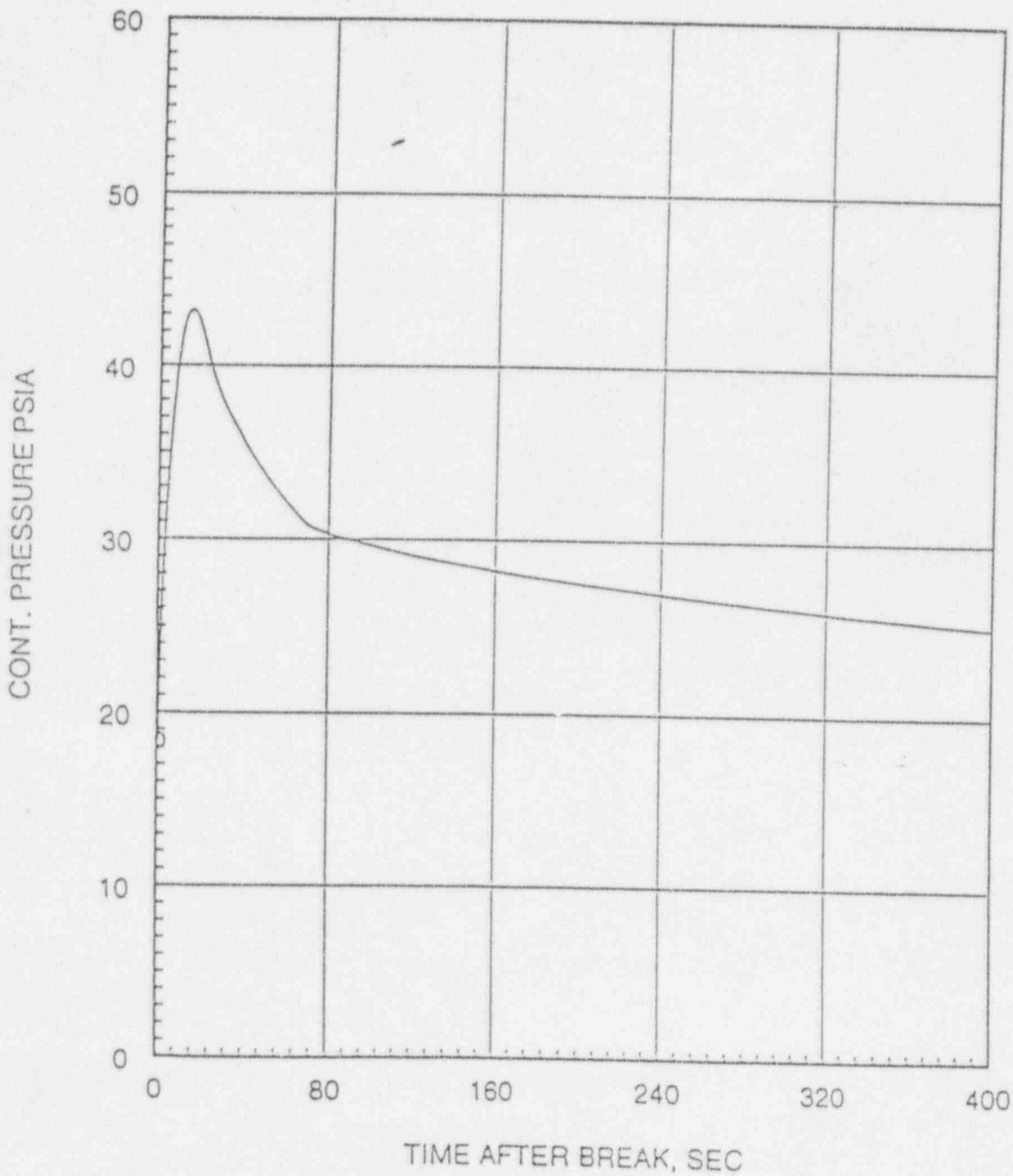


FIGURE 14.17-278

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

MASS ADDED TO CORE DURING REFLOOD

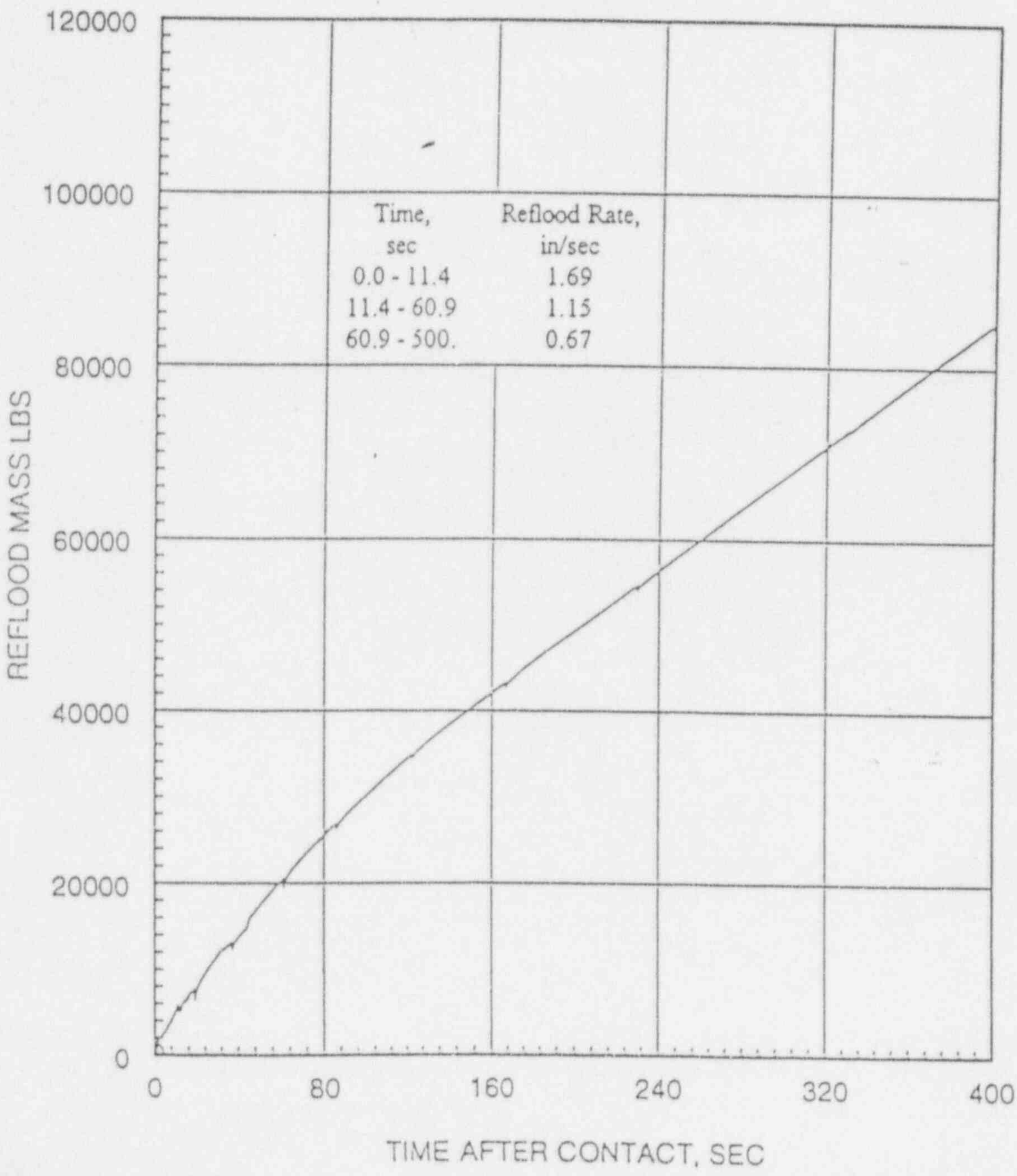


FIGURE 14.17-279

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

PEAK CLADDING TEMPERATURE VS TIME

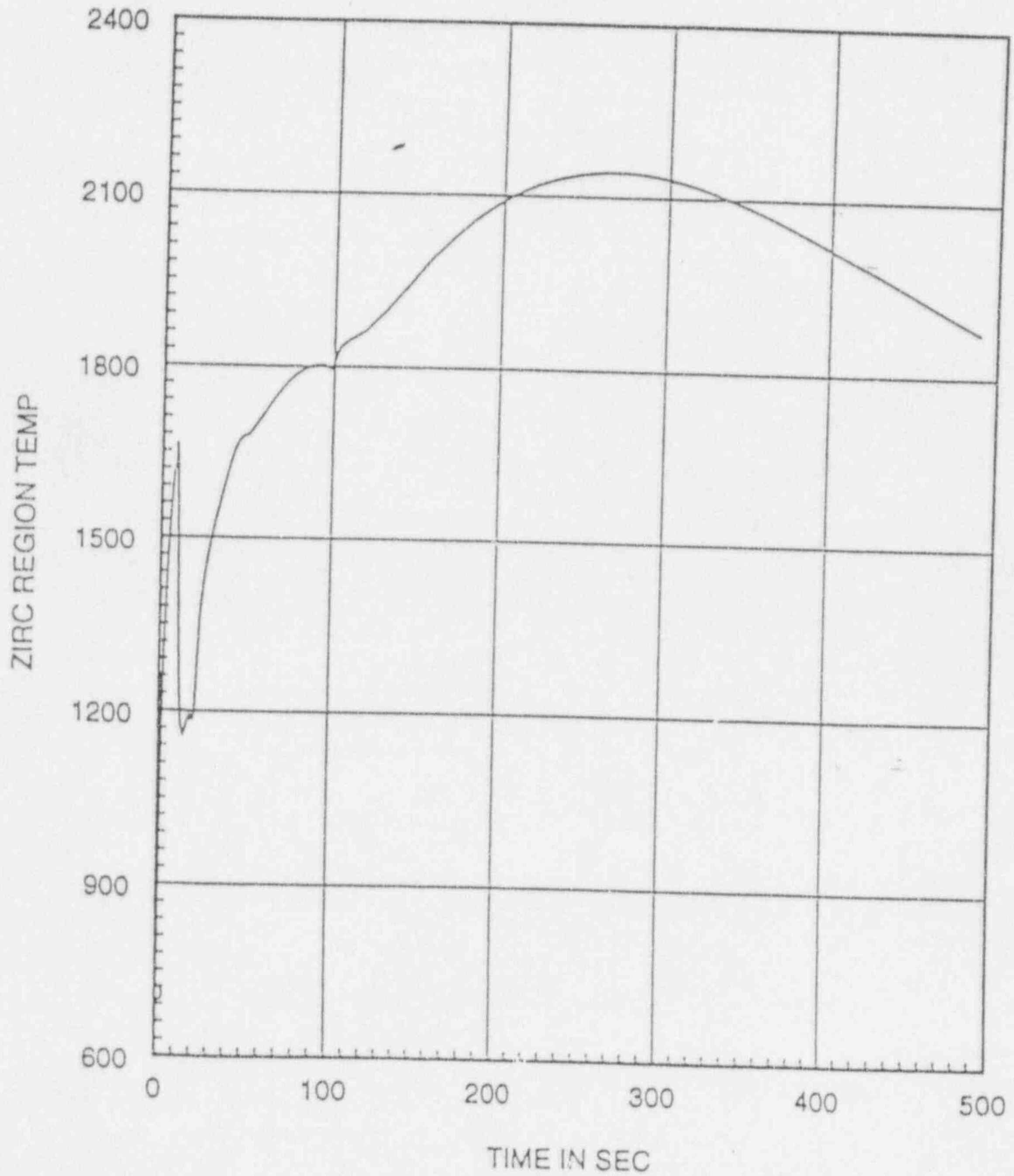


FIGURE 14.17-280

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

GAP CONDUCTANCE AT LOCATION OF PCT

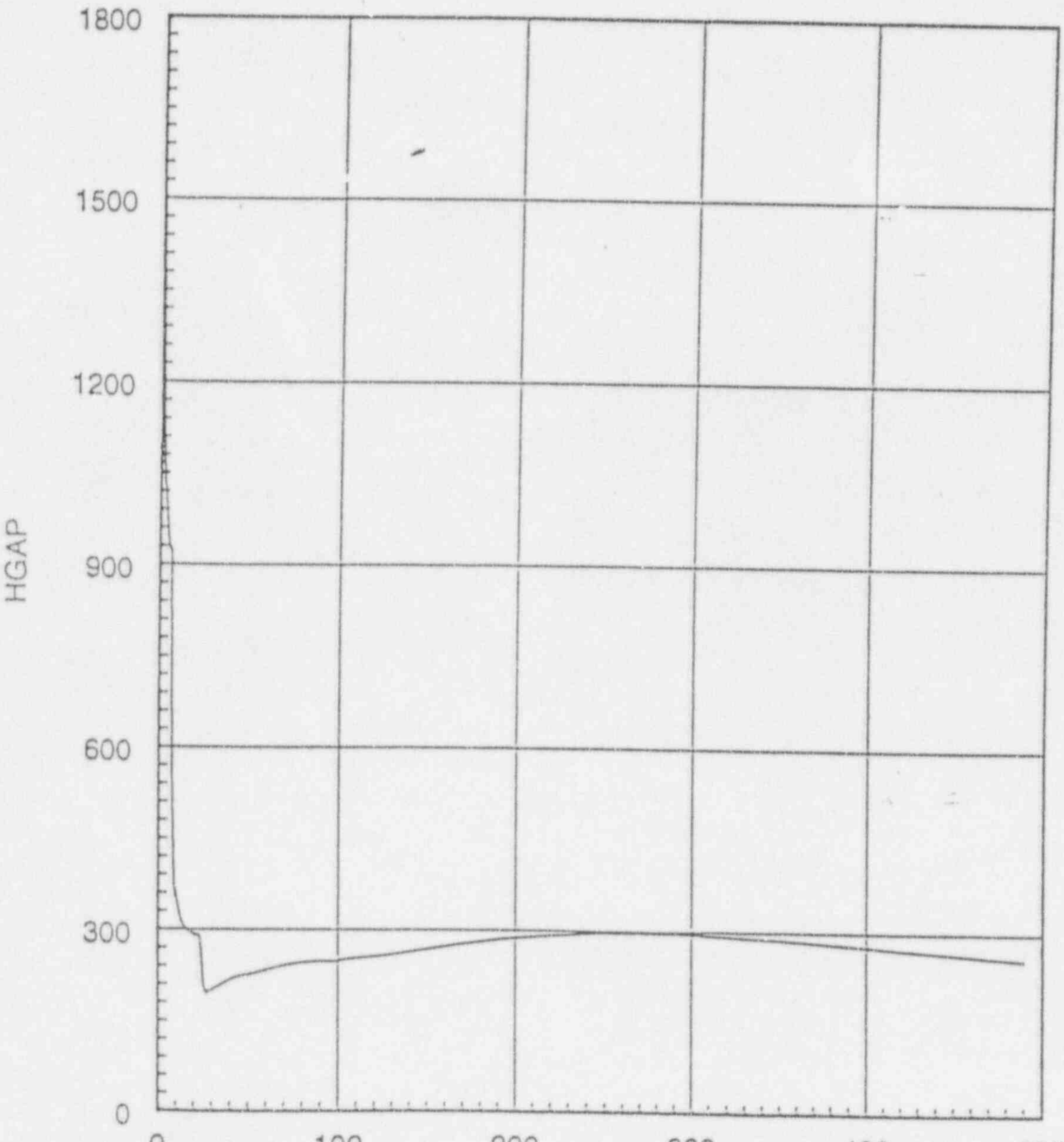


FIGURE 14.17-281

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

MAXIMUM LOCAL CLADDING OXIDATION VS TIME

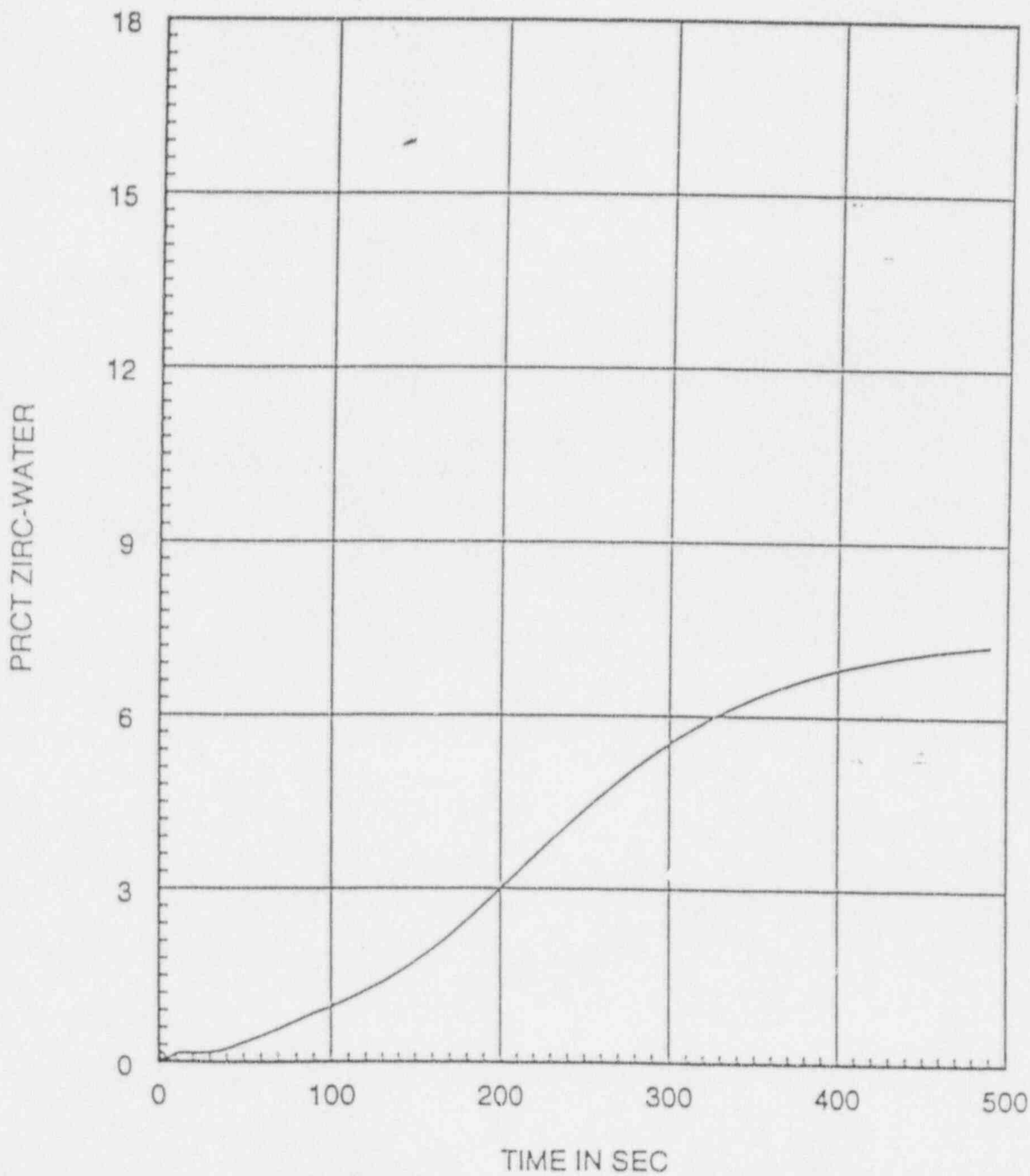


FIGURE 14.17-282

0.6 DEG/PD, TEMPERATURE OF FUEL CENTERLINE, FUEL AVERAGE,

CLADDING AND COOLANT AT LOCATION OF PCT VS TIME

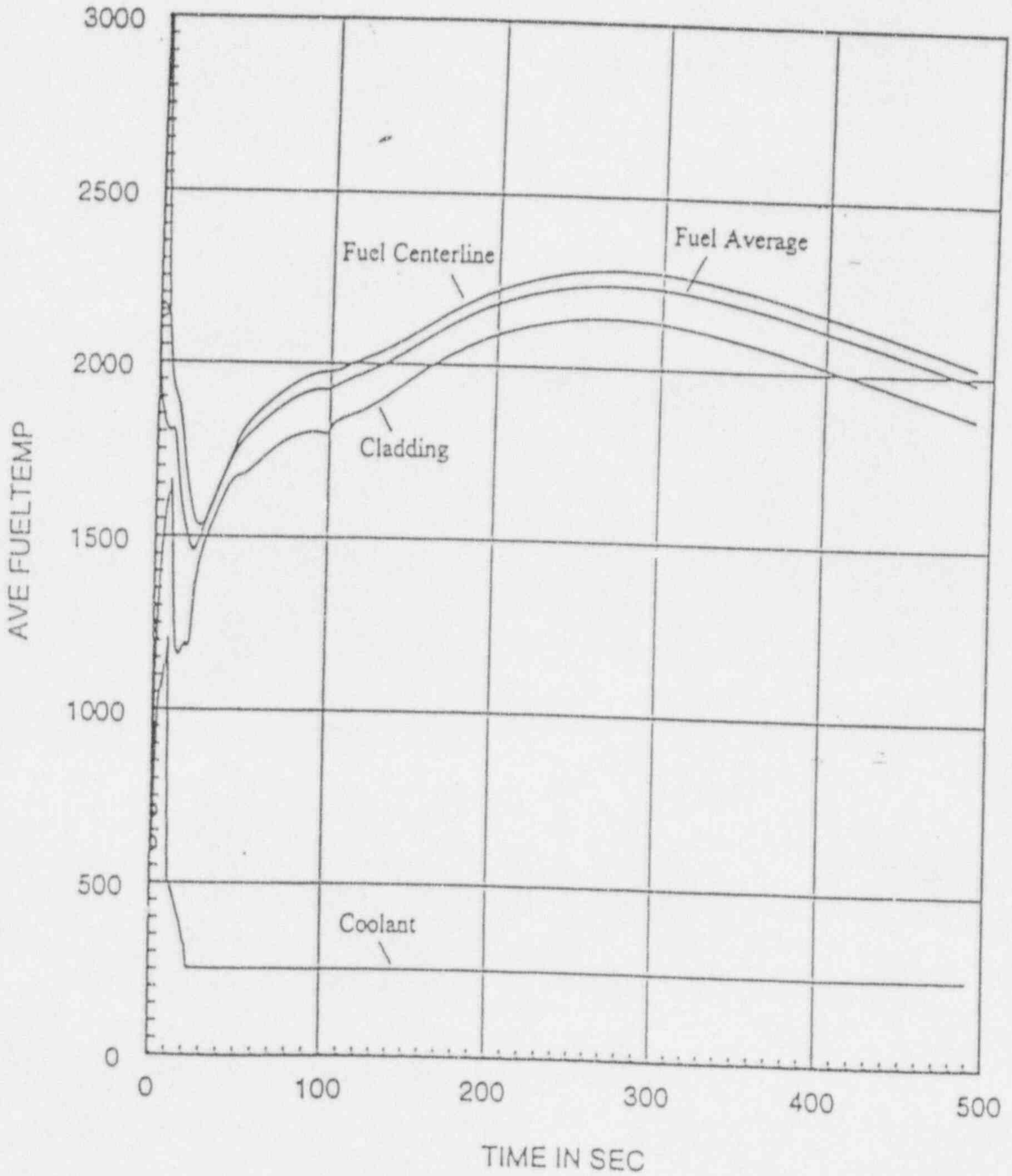


FIGURE 14.17-283

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

HEAT TRANSFER COEFFICIENT AT LOCATION OF PCT VS TIME

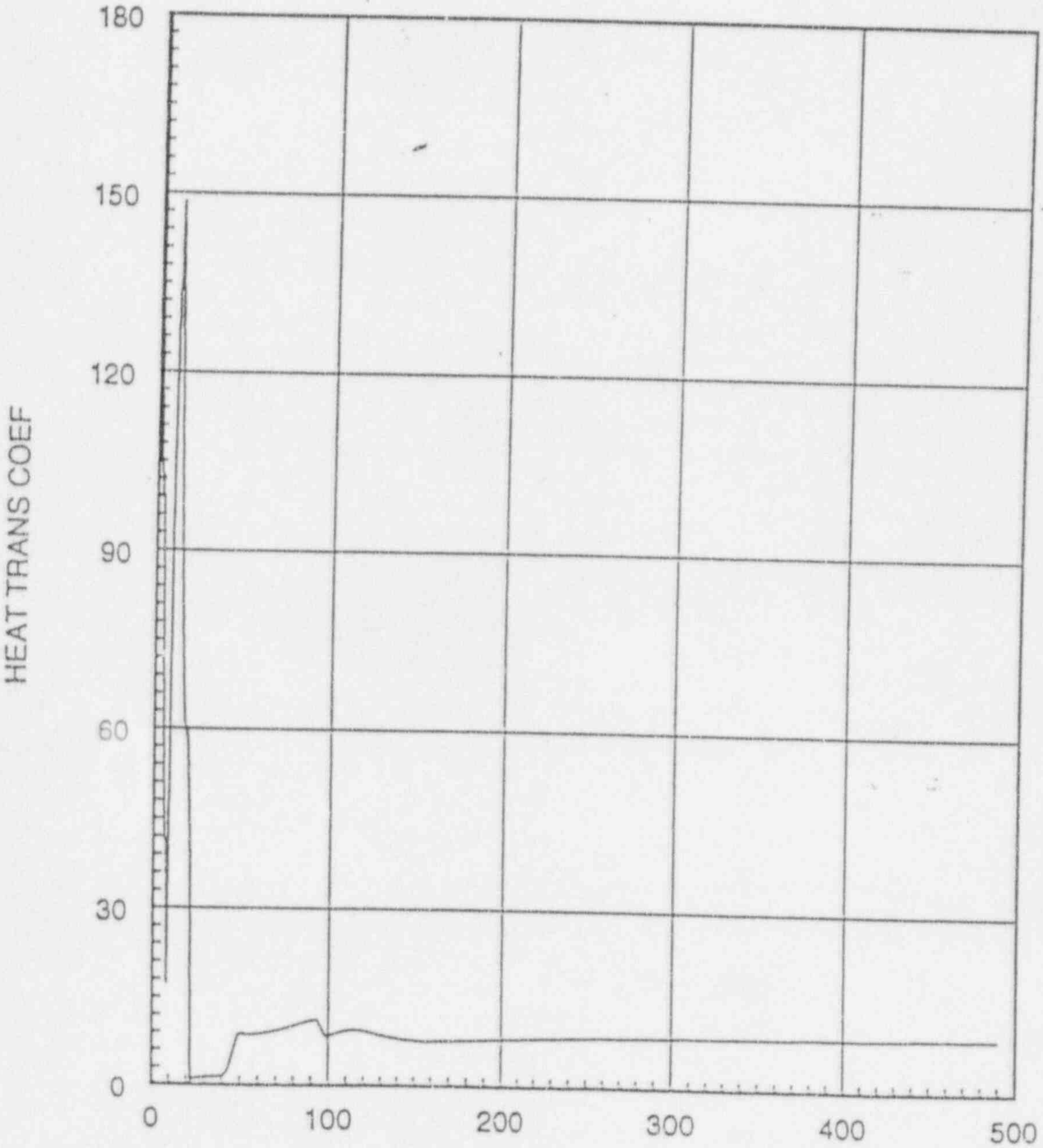


FIGURE 14.17-284

0.6 DOUBLE-ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG,

HOT ROD INTERNAL GAS PRESSURE VS TIME

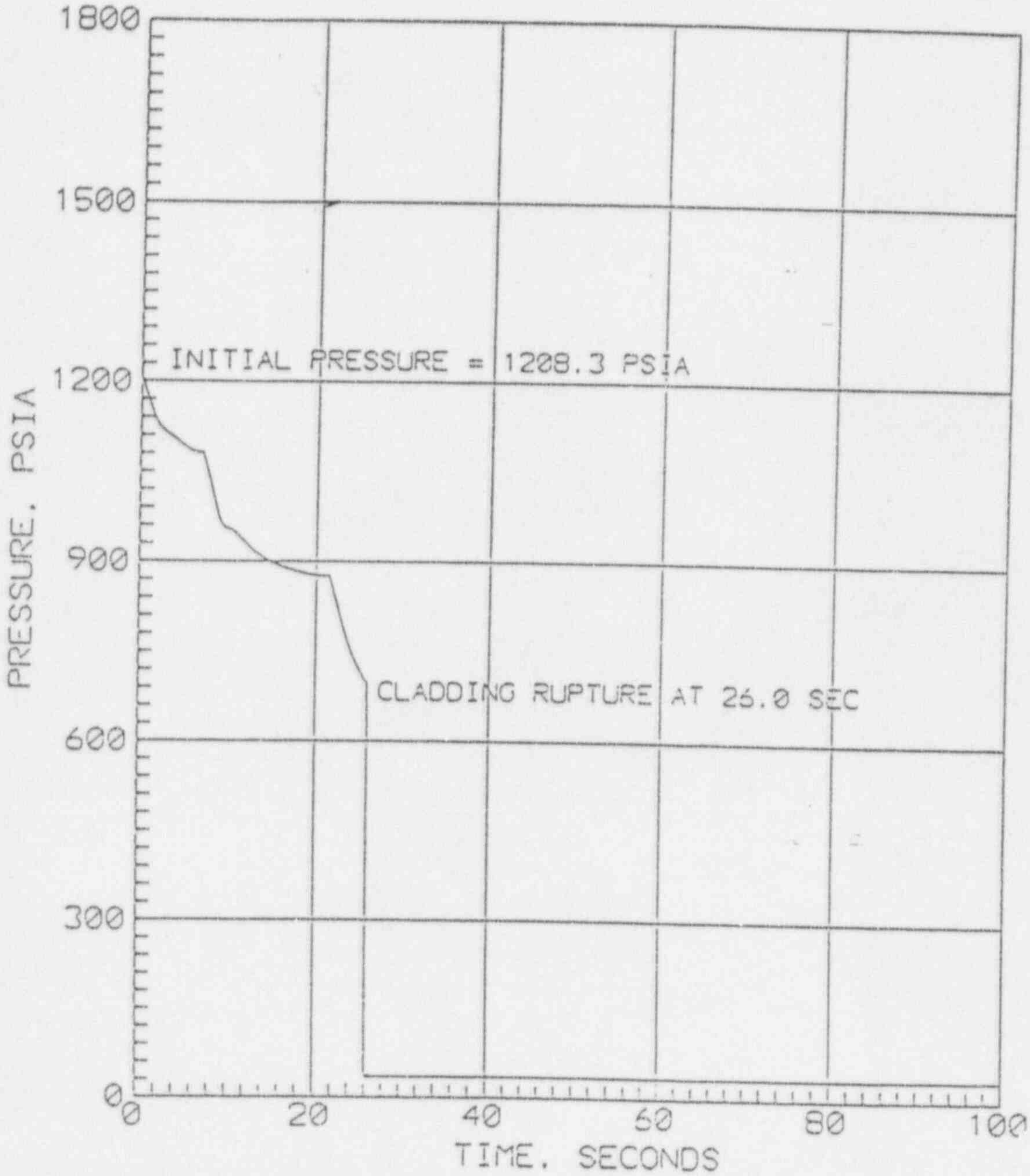


TABLE 14.26-1

INITIAL CONDITIONS AND INPUT PARAMETERS ASSUMED IN THE FEEDWATER LINE BkEAK EVENT

PARAMETER	UNITS	UNIT 1	UNIT 2
Initial Core Power Level	MWt	2754.0	2754.0
Initial Core Coolant Inlet Temperature	$^{\circ}F$	550.0	550.0
Initial RCS Vessel Flow Rate	gpm	358,900	358,900
Initial RCS Pressure	psia	2165.0	2165.0
Initial SG Pressure	psia	828.8	828.8
Initial Pressurizer Liquid Volume	ft	975.0	975.0
Effective MTC	$x10^{-4}\ \Delta\rho/^{\circ}F$	+0.15	+0.15
Doppler Coefficient Multiplier	---	0.85	0.85
High Pressurizer Pressure Analysis Trip Setpoint	psia	2470	2470
AFW Actuation	% WR Tap Span	29.1	29.1
SG Differential Pressure Analysis Setpoint	psid	10.0	10.0
CEA Worth at Trip	% $\Delta\rho$	-5.3	-5.3
RRS	Operating Mode	Manual[a]	Manual[a]

TABLE 14.26-1 (Continued)

INITIAL CONDITIONS AND INPUT PARAMETERS ASSUMED IN THE FEEDWATER LINE BREAK EVENT

PARAMETER	UNITS	UNIT 1	UNIT 2
SDBS	Operating Mode	Manual[a]	Manual[a]
PPCS	Operating Mode	Manual[a]	Manual[a]
PLCS	Operating Mode	Manual[a]	Manual[a]

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[a] These modes of control system operation maximize the peak RCS pressure.

TABLE 14.26-3

SEQUENCE OF EVENTS FOR FEED LINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	Break in Main Feedwater Line	0.325 ft^2
22.1	Heat Transfer Area Rampdown in LHSG Begins	19691 lbm
24.9	High Pressurizer Pressure Trip Analysis Setpoint is Reached	2470 psia
25.7	Level in the Ruptured SG decreases below the assumed nozzle level; steam will be blown out of the break	5000 lbm
25.9	First Primary Safety Valve Begins to Open	2550 psia
26.3	Trip Breakers Open	---
26.8	CEAs Begin to Enter Core; LOAC on Turbine Trip; RCS Pumps Begin to Coastdown	---
28.9	Peak RCS Pressure	2747 psia[(b)]
35.0	Undamaged SG Safety Valves Begin to Open	1010 psia
37.7	Damaged SG Safety Valves Begin to Open	1010 psia

TABLE 14.26-3 (Continued)

SEQUENCE OF EVENTS FOR FEED LINE BREAK EVENT WITH LOAC FOLLOWING REACTOR TRIP

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
38.7	Maximum SG Pressure, Undamaged	1040.6 psia[a]
39.0	Maximum SG Pressure, Damaged	1011.5 psia
41.7	Damaged SG Safety Valves are Closed	1010 psia
42.1	Primary Safety Valves are Closed	2448 psia
48.5	Undamaged SG Safety Valves are Closed	1010 psia
63.0	AFW Analysis Setpoint is reached in Undamaged SG	29.1% of WR Tap Span
163.0	Main Steam Isolation Signal	600 psia
163.9	MSIVs Begin to Close	---
169.9	MSIVs are Fully Closed	---
247.6	AFW Flow Established to Undamaged SG	180 gpm
375.5	Undamaged SG Safety Valves Begin to Open	1010 psia
453.5	First Primary Safety Valve Begins to Open	2550 psia

TABLE 14.26-3 (Continued)

SEQUENCE OF EVENTS FOR FEED LINE BREAK EVENT
WITH LOAC FOLLOWING REACTOR TRIP

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
626.3	Operator Increases AFW Flow	---
630.4	Primary Safety Valves are Fully Closed	2448 psia

[a] SG pressure includes downcomer liquid head.
[b] Peak RCS pressure includes elevation head.