ANF-87-150 (NP)

VOLUME 2

ADVANCED NUCLEAR FUELS CORPORATION

PALISADES MODIFIED REACTOR PROTECTION SYSTEM REPORT: ANALYSIS OF CHAPTER 15 EVENTS

JUNE 1988

ADVANCED NUCLEAR FUELS CORPORATION

ANF-87-150(NP)
Volume 2

Issue Date: 6/13/88

PALISADES MODIFIED REACTOR PROTECTION SYSTEM REPORT: ANALYSIS OF CHAPTER 15 EVENTS

Prepared by:
T. P. Lindquist, Project Engineer
PWR Safety Analysis
Licensing & Safety Engineering
Fuel Engineering & Technical Services

June 1988

gf

NUCLEAR REGULATORY COMMISSION REPORT DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

This technical report was derived through research and development programs sponsored by Advanced Nuclear Fuels Corporation. It is being submitted by Advanced Nuclear Fuels Corporation to the U.S. Nuclear Regulatory Commission as part of a technical contribution to facilitate safety analyses by licensees of the U.S. Nuclear Regulatory Commission which utilize Advanced Nuclear Fuels Corporation-fabricated reload fuel or other technical services provided by Advanced Nuclear Fuels Corporation for light water power reactors and it is true and correct to the best of Advanced Nuclear Fuels Corporation's knowledge, information, and belief. The information contained herein may be used by the U.S. Nuclear Regulatory Commission in its review of this report, and under the terms of the respective agreements, by licensees or applicants before the U.S. Nuclear Regulatory Commission which are customers of Advanced Nuclear Fuels Corporation in their demonstration of compliance with the U.S. Nuclear Regulatory Commission's regulations.

Advanced Nuclear Fuels Corporation's warranties and representations concerning the subject matter of this document are those set forth in the agreement between Advanced Nuclear Fuels Corporation and the customer to which this document is issued. Accordingly, except as otherwise expressly provided in such agreement, neither Advanced Nuclear Fuels Corporation nor any person acting on its behalf:

A. Makes any warranty, or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights, or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this document.

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

LIST OF TABLES

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF FIGURES

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

LIST OF FIGURES (Cont.)

1.0 INTRODUCTION

This report documents the results of Standard Review Plan (SRP)[1] Chapter 15 event analyses performed in support of Palisades operation with up to 29.3% steam generator tube plugging and the modified reactor protection system (RPS). The modified reactor protection system includes a variable-overpower trip and an improved thermal margin/low pressure (TM/LP) trip with axial monitoring. Cycle 7 is the reference cycle for this analysis. The Chapter 15 events were selected in accordance with Advanced Nuclear Fuels Corporation methodology.[2] The basis for event selection is documented in Reference 3, the Dispoistion of Events report. References for LOCA/ECCS analyses in support of the reference cycle are also documented in Reference 3.

Section 2.0 presents a summary of the results and review of SRP Chapter 15 events. Section 3.0 presents the conditions employed in the event analyses and the results of these event analyses. Events are numbered in accordance with the SRP to facilitate review. It includes a tabular list of the disposition of events and analysis of record for Palisades, Chapter 15 events, with a cross reference between SRP event numbers and the Palisades Updated FSAR.[5]

2.0 SUMMARY AND CONCLUSIONS

A summary Disposition of Events for the Palisades reference cycle is given in Table 2-1. This table lists each SRP Chapter 15 event, indicates whether that event is reanalyzed for this submittal, and provides a reference to the bounding event or analysis of record for events not reanalyzed. The Disposition of Events is reported in greater detail in Reference 3.

The results of Anticipated Operational Occurrences and Postulated Accidents reanalyzed for this submittal are listed in Table 2-2. Acceptance criteria are met for each event.

The results reported herein confirm that event acceptance criteria, defined in Section 15.0.1.1 of this document, are met for reference cycle operation as defined by the operating parameter ranges in Sections 15.0.1 - 15.0.8 of this report. These results support operation with up to 29.3% average steam generator tube plugging at a rated thermal power of 2530 MWt.

Table 2-1 Disposition of Events Summary for Palisades

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.1	INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM				
	15.1.1	Decrease in Feedwater Temperature	Bounded	15.1.3	14.9.4
	15.1.2	Increase in Feedwater Flow			
		1) Power	Bounded	15.1.3	14.9.6
		2) Startup	Bounded	15.1.3	14.9.5
	15.1.3	Increase in Steam Flow	Analyze		14.10
	15.1.4	Inadvertent Opening of a Steam Generator Relief of Safety Valve			
		1) Power	Bounded	15.1.3	
		2) Scram Shutdown Margin	Bounded	15.1.3	
	15.1.5	Steam System Piping Failures Inside and Outside of Containment	Bounded	Ref.14,15&24	14.14
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY STEAM				
	15.2.1	Loss of External Load	Analyze		14.12
	15.2.2	Turbine Trip	Bounded	15.2.1	
	15.2.3	Loss of Condenser Vacuum	Bounded	15.2.1	
	15.2.4	Closure of the Main Steam Isolation Valves (MSIVs)	Bounded	15.2.1	
	15.2.5	Steam Pressure Regulator Failure	Not applicable; BWR Event		

Table 2-1 Disposition of Events Summary for Palisades (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
	15.2.6	Loss of Nonemergency A.C. Power to the Station Auxiliaries	Short term bounded Long term bounded	15.3.1 15.2.7	
	15.2.7	Loss of Normal Feedwater Flow	Analyze		14.13
	15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Cooldown Bounded Heatup Bounded	15.1.5 15.2.7	
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW				
	15.3.1	Loss of Forced Reactor Coolant Flow	Analyze		14.7
	15.3.2	Flow Controller Malfunction	Not Applicable		14.7
	15.3.3	Reactor Coolant Pump Rotor Seizure	Analyze		14.7
	15.3.4	Reactor Coolant Pump Shaft Break	Bounded	15.3.3	14.7
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES				
	15.4.1	Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Condition	Analyze		14.2.2.2
	15.4.2	Uncontrolled Control Rod Bank Withdrawal at Power Operation Conditions	Analyze		14.2.2.3
	15.4.3	Control Rod Misoperation			
		1) Dropped Control Bank/Rod	Analyze		14.4
		2) Dropped Part-Length Control Rod	Bounded	15.4.3(1)	14.6

Table 2-1 Disposition of Events Summary for Palisades (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
		3) Malpositioning of the Part-Length Control Group	Not Applicable		14.6
		4) Statically Misaligned Control Rod/Bank	Analyze		
		5) Single Control Rod Withdrawal	Analyze	Ref. 5	14.2.2.4
		6) Core Barrel Failure	Bounded	Ref. 5	14.5
	15.4.4	Startup of an Inactive Loop	Bounded		14.8
	15.4.5	Flow Controller Malfunction	Not applicable; No Flow Controller		
	15.4.6	CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant			
		1) Rated and Power Operation Conditions	Analyze		14.3
		2) Reactor Critical, Hot Standby and Hot Shutdown	Analyze		14.3
		3) Refueling Shutdown Condition, Cold Shutdown Condition and Refueling Operation	Analyze		14.3
	15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Administrative Procedures Preclude this Event		

Table 2-1 Disposition of Events Summary for Palisades (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
	15.4.8	Spectrum of Control Rod Ejection Accidents	Analyze		14.16
	15.4.9	Spectrum of Rod Drop Accidents (BWR)	Not applicable; BWR Event		
15.5	INCREASES IN REACTOR COOLANT INVENTORY				
	15.5.1	Inadvertent Operation of the ECCS that Increases Reactor Coolant Inventory	Overpressure Bounded Reactivity Bounded	15.2.1 15.4.6	
	15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory	Overpressure Bounded Reactivity Bounded	15.2.1 15.4.6	
15.6	DECREASES IN REACTOR COOLANT INVENTORY				
	15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	Bounded	15.6.5	
	15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside of Containment	Bounded	15.6.5	
	15.6.3	Radiological Consequences of Steam Generator Tube Failure	Bounded	Ref. 5	14.15
	15.6.4	Radiological Consequences of a Main Steamline Failure Outside Containment	Not applicable; BWR Event		

Table 2-1 Disposition of Events Summary for Palisades (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
	15.6.5	Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	Bounded	Ref. 16-21 Ref. 5	14.17 14.18 14.22
15.7	RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT				
	15.7.1	Waste Gas System Failure	Deleted*		14.21
	15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	Deleted*		
	15.7.3	Postulated Radioactive Releases due to Liquid-Containing Tank Failures	Bounded	Ref. 5	14.20
	15.7.4	Radiological Consequences of Fuel Handling Accidents	Bounded	Ref. 5	14.19
	15.7.5	Spent Fuel Cask Drop Accidents	Bounded	Ref. 5	14.11

*This section of the Standard Review Plan has been deleted.

Table 2-2 Summary of Results

Anticipated Operational Occurrence	Maximum Power Level (MWt)	Maximum Core Average Heat Flux (Btu/hr-ft^2)	Maximum Pressurizer Pressure (psia)	MDNBR (XNB)
15.1.3 Increase in Steam Flow[++]	2867.6	185326	2110	1.497
15.2.1 Loss of External Load	2626.7[*]	167000	2584.7[**]	1.776[*]
15.2.7 Loss of Normal Feedwater	2580.6	167000	2271.9	-
15.3.1 Loss of Forced Reactor Coolant Flow	2620.3	167000	2160.9	1.455
15.4.2 Uncontrolled Control Bank Withdrawal at Power[++]	2888.5	185136	2154.2	1.304
15.4.3 Control Rod Misoperation[##]				
o Dropped Rod or Bank	2580.6	167000	2010	1.301
o Statically Misaligned Rod or Bank[++]	2580.6	167000	2010	#
Single Rod Withdrawal[++]	2684.9	175382	2071.7	1.273
15.4.6 CVCS Malfunction resulting in Decreased Boron Concentration	(Adequacy of Shutdown Margin is Demonstrated.)			

* MDNBR case
** Peak pressure case
\+ Maximum pressure difference case
++ 100% power case
\# Bounded by the Dropped Control Rod Event
\#\# Results are based on conservative assumptions pertaining to control rod/bank configurations.

Table 2-2 Summary of Results (Cont.)

Anticipated Operational Occurrence	Maximum Power Level (MWt)	Maximum Core Average Heat Flux (Btu/hr-ft^2)	Maximum Pressurizer Pressure (psia)	MDNBR (XNB)
15.3.3 Reactor Coolant Pump Rotor Seizure	2580.6	167000	2010	1.409
15.4.1 Uncontrolled Control Bank Withdrawal at Subcritical or Low Power	6643.1	153170	2426	1.036*
15.4.8 Control Rod Ejection	5348.1	189877	2452.1+	<1.17**

* <2.3% of the core is calculated to experience DNB

** 12.2% of the core is calculated to experience DNB

\+ Peak pressure case

3.0 ANALYSIS OF PLANT TRANSIENTS

This section provides the results of event analyses performed to support the Palisades operation with the modified RPS. Event numbering and nomenclature are consistent with the SRP to facilitate review.

This section also provides information on the plant licensing basis as it affects the event analyses, including the classification of plant conditions, event acceptance criteria, and single failure criteria. Plant operating mode and analysis initial conditions are listed. Neutronics data and core and fuel design parameters are provided. Listings of systems and components available for accident mitigation, trip setpoints, time delays and component capacities are also included. These data, together with the design parameters[6] and the event specific input data given in each event subsection, represent a comprehensive summary of analysis inputs.

15.0 ACCIDENT ANALYSES

15.0.1 CLASSIFICATION OF PLANT CONDITIONS

Plant operations are placed in one of four categories. These categories are those adopted by the American Nuclear Society (ANS). The categories are:

NORMAL OPERATION AND OPERATIONAL TRANSIENT

- Events which are expected to occur frequently in the course of power operation, refueling, maintenance, or plant maneuvering.

FAULTS OF MODERATE FREQUENCY

- Events which are expected to occur on a frequency of once per year during plant operation.

INFREQUENT FAULTS

- Events which are expected to occur once during the lifetime of the plant.

LIMITING FAULTS

- Events which are not expected to occur but which are evaluated to demonstrate the adequacy of the design.

15.0.1.1 Acceptance Criteria

Operational Events

This condition describes the normal operational modes of the reactor. As such, occurrences in this category must maintain margin between operating conditions and the plant setpoints. The setpoints are established to assure maintenance of margin to design limits. The set of operating conditions, together with conservative allowances for uncertainties establish the set of initial conditions for the other event categories.

Moderate Frequency Events

1. The pressures in reactor coolant and main steam systems should be less than 110% of design values.

2. The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. That is, the minimum calculated departure from nucleate boiling ratio is not less than the applicable limits of the DNBR correlation being used (e.g., DNBR < 1.17 for the XNB correlation).

3. The radiological consequences should be less than 10 CFR 20 guidelines.

4. The event should not generate a more serious plant condition without other faults occurring independently.

Infrequent Events

1. The pressures in reactor coolant and main steam systems should be less than 110% of design values.

2. A small fraction of fuel failures may occur, but these failures should not hinder the core coolability.

3. The radiological consequences should be a small fraction of 10 CFR 100 guidelines (generally <10%).

4. The event should not generate a limiting fault or result in the consequential loss of the reactor coolant or containment barriers.

Limiting Fault Events

1. Radiological consequences should not exceed 10 CFR 100 guidelines.

2. The event should not cause a consequential loss of the required functions of systems needed to cope with the reactor coolant and containment systems transients.

3. Additional criteria to be satisfied by specific events are:

   a. LOCA - 10 CFR 50.46 and Appendix K.

   b. Rod Ejection - Radially averaged fuel enthalpy <280 cal/gm.

15.0.1.2 Classification Of Accident Events By Category

Table 15.0.1-1 lists the accident category used for each event analyzed in this report. This classification is used in evaluating the acceptability of the results obtained from the analysis.

Table 15.0.1-1 Accident Category Used for Each Analyzed Event

Event		Accident Category
15.1.3	Increase in Steam Flow	Moderate
15.2.1	Loss of External Load	Moderate
15.2.7	Loss of Normal Feedwater Flow	Moderate
15.3.1	Loss of Forced Reactor Coolant Flow	Moderate
15.3.3	Reactor Coolant Pump Rotor Seizure	Infrequent
15.4.1	Uncontrolled Bank Withdrawal at Subcritical or Low Power	Infrequent
15.4.2	Uncontrolled Bank Withdrawal at Power	Moderate
15.4.3	Control Rod Misoperation	Moderate
15.4.6	CVCS Malfunction Resulting in Decreased Boron Concentration	Moderate
15.4.8	Control Rod Ejection	Limiting Fault

15.0.2 PLANT CHARACTERISTICS AND INITIAL CONDITIONS

Eight operational modes have been considered in the analysis and are characterized as follows:

Mode	Reactivity	Power*	Average Core Temp.
Rated Power (1)	Critical	2530 Mwt	≥525°F
Power Operation (2)	Critical	≥2%	≥525°F
Reactor Critical (3)	Critical	$\geq 10^{-4}$%	≥525°F
Hot Standby (4)	Any Withdrawn Rod	10^{-4}% to 2%	>525°F
Hot Shutdown (5)	Shutdown Margin ≥2%$\Delta\rho$	$<10^{-4}$%	>525°F
Refueling Shutdown Condition (6)	Shutdown margin of at least 5%$\Delta\rho$ with all control rods withdrawn	0	≤210°F
Cold Shutdown Condition (7)	k_{eff} ≤.98 with all control rods in the core and the highest worth control rod fully withdrawn		≤210°F
Refueling Operation (8)	Any operation involving movement of core components when the vessel head is unbolted or removed		

Mode numbers are given in parenthesis. These operational modes have been considered in establishing the subevents associated with each event initiator.

---

* Does not include decay heat.

A set of initial conditions is established for the events analyzed that is consistent with the conditions for each mode of operation.

The nominal plant rated operating conditions are presented in Table 15.0.2-1 and principal fuel design characteristics in Table 15.0.2-2. The uncertainties used in the accident analysis applicable to the operating conditions are:

Core Power	± 2%
Primary Coolant Temperature	± 5°F
Primary Coolant Pressure	± 50 psi
Primary Coolant Flow	± 3%

Table 15.0.2-1 Nominal Plant Operating Conditions

Core Thermal Power	2530 MWt
Pump Thermal Power (total)	15 MWt
System Pressure	2060 psia
Vessel Coolant Flow Rate*	120.3 Mlbm/hr
Core Coolant Flow Rate**	116.7 Mlbm/hr
Average Coolant Temperature	570.58°F
Core Inlet Coolant Temperature	543.65°F
Steam Generator Pressure	730 psia
Steam Flow Rate	10.97 Mlbm/hr
Feedwater Temperature	435°F
Number of Active Steam Generator Tubes* (per steam generator)	6023

* Reflects 29.3% average steam generator tube plugging.

** Reflects a 3% bypass flow.

Table 15.0.2-2 Nominal Fuel Design Parameters

Total Number of Fuel Assemblies	204
Fuel Assembly Design Type	15x15
Fueled Rods per Gadolinia Assembly	216
Fueled Rods per Nongadolinia Assembly	208
Instrument Tubes per Assembly	1
Guide Bars per Assembly	8
Plugged Tubes or B_4C Rods per Nongadolinia Assembly	8
Assembly Pitch	8.485 in.
Rod Pitch	0.550 in.
Fuel Pellet Outside Diameter	0.350 in.
Clad Inside Diameter	0.358 in.
Clad Outside Diameter	0.417 in.
Active Fuel Length	131.8 in.
Number of Spacers	10

15.0.3 POWER DISTRIBUTION

The radial and axial power peaking factors used in the analysis are presented in Table 15.0.3-1. Figures 15.0.3-1 and 15.0.3-2 show the limiting axial shapes for 100% power and 50% power, respectively. These axial shapes have ASIs of -0.127 for 100% power and -0.342 for 50% power. In this context, ASI is defined as:

$$\frac{P_{Lower} - P_{Upper}}{P_{Lower} + P_{Upper}}$$

P_{Lower} corresponds to the power generated in the lower half of the core and P_{Upper} corresponds to the power generated in the upper half of the core.

The limiting DNBR occurs on an interior pin of an assembly with 208 rods. The Technical Specification[7] Limiting Conditions of Operation assure that the power distribution is maintained within these limits during normal operation. However, some events analyzed result in transient redistribution of the radial power peaking factors. Transient radial power redistribution is treated as described in Section 15.4.3.2 of Reference 2.

The analyses for the inlet temperature LCO and for the TM/LP trip utilize axial power distributions and associated axial shape indices (ASI). These axial power distributions are generated from a one-dimensional core physics model and thus represent core average axial power shapes. The appropriate axial power shapes to use in conjunction with an integrated rod power radial peaking factor, F_r, would be an assembly axial power distribution. The assembly axial power distribution would have a somewhat higher axial peaking factor than a core average axial power distribution. To account for the DNBR effect of using a core average axial power distribution rather than an assembly axial power distribution, the F_r used in the analysis has been increased by 3% over the value in the Technical Specifications.

Table 15.0.3-1 Core Power Distribution

Radial Peaking Factor:	
- Interior rod of 208 rod bundle/ave. bundle	1.64
- Factor to account for using core average axials rather than assembly average axials	1.03
- Engineering Uncertainty	1.03
Total Radial	1.74*
Axial Peaking Factor:	
- 100% power	1.30
- 50% power	1.67
Fraction of Power Deposited in the Fuel	.974

* For power operation at less than rated, the radial peaking is $1.74[1+0.3(1-f)]$ for $0.5 \leq f \leq 1$ and 2.00 for $f<0.5$, where f is the fractional power of 2530 MWt.

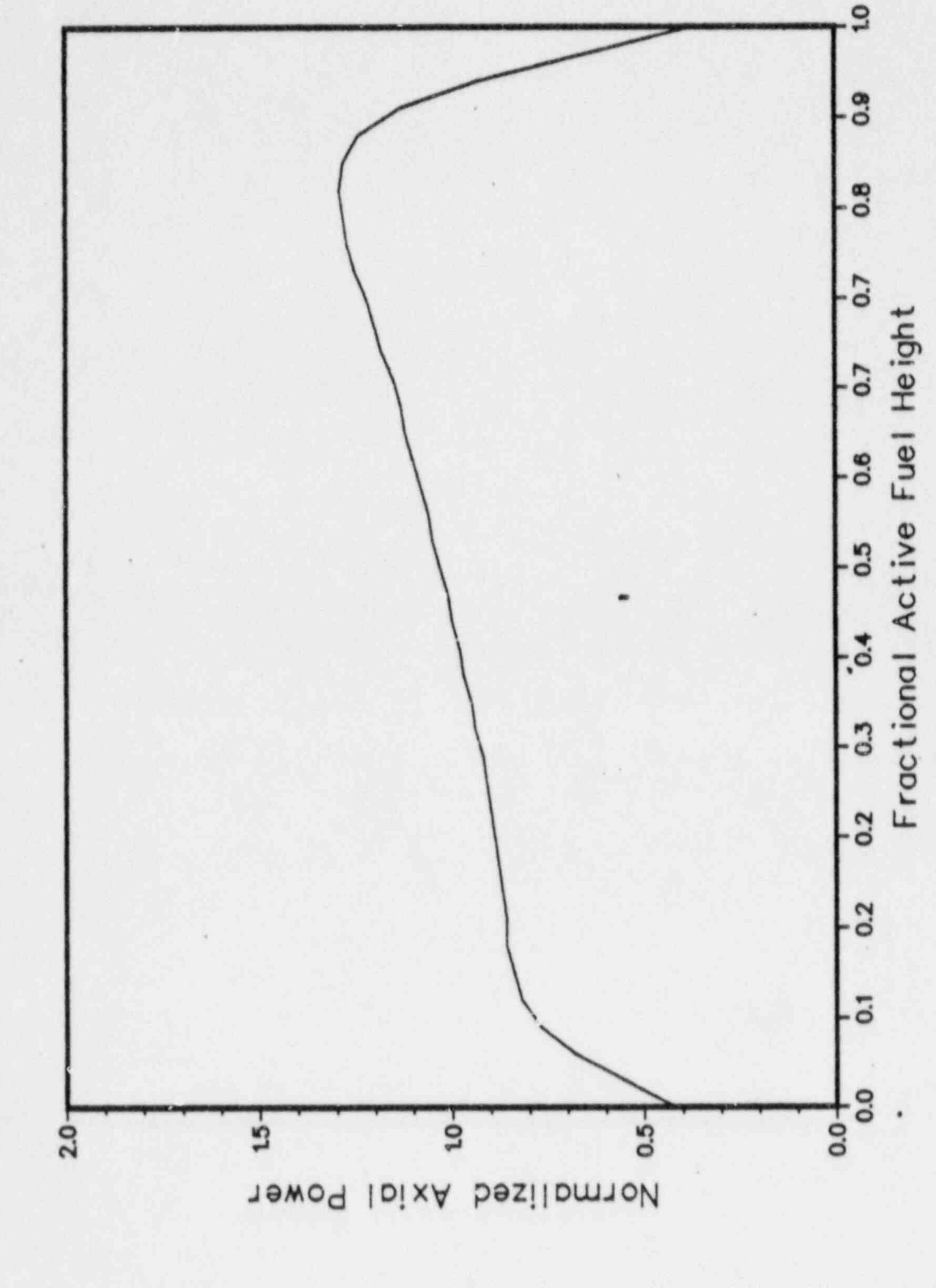


Figure 15.0.3-1 Axial Power Distribution at 2530 MWt
(ASI = -0.127)

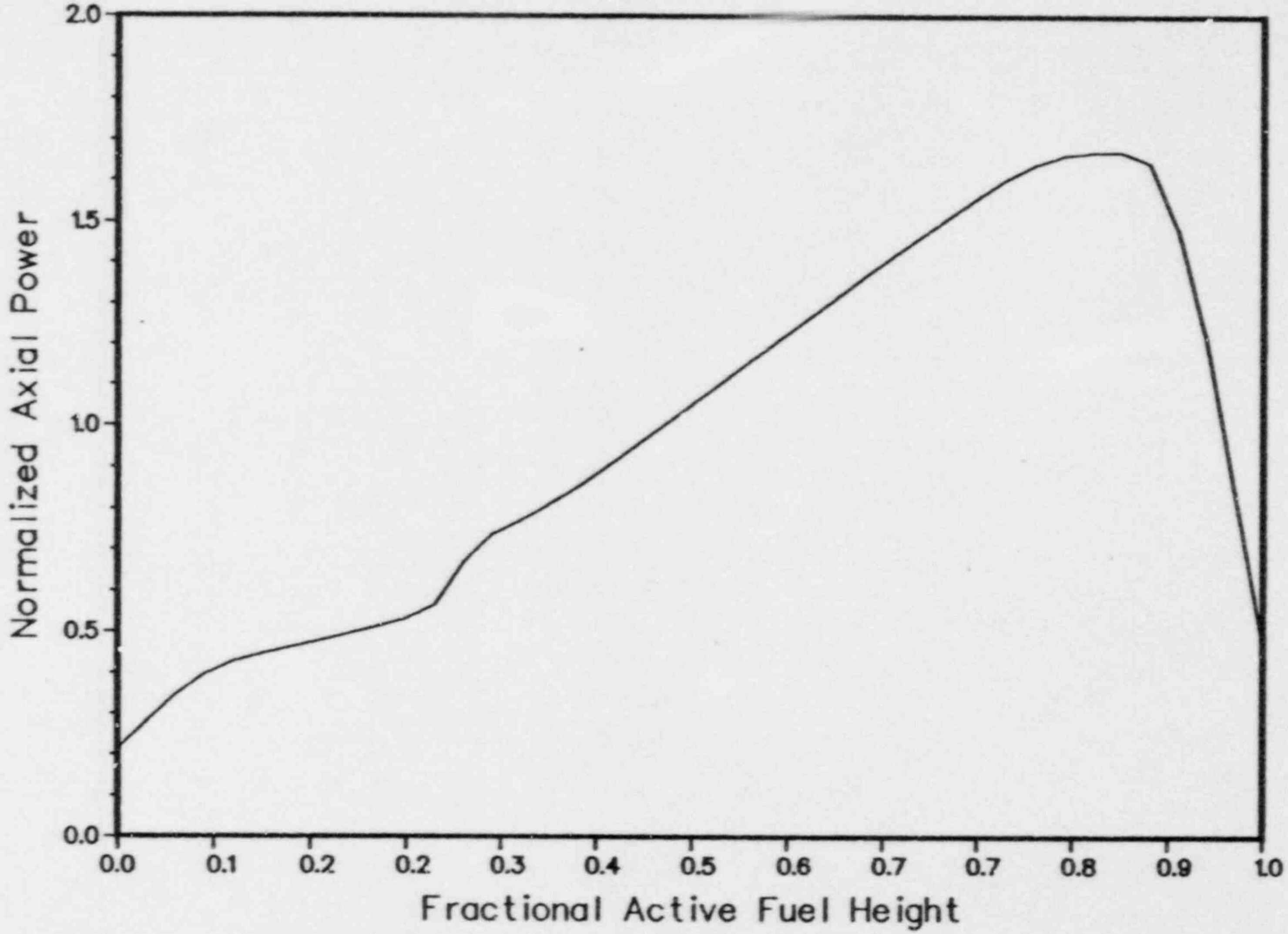


Figure 15.0.3-2 Axial Power Distribution at 1265 MWt
(ASI = -0.342)

15.0.4 RANGE OF PLANT OPERATING PARAMETERS AND STATES

Table 15.0.4-1 presents the range of key plant operating parameters considered in the analysis. A broader range of power, core inlet temperature, and primary pressure is considered in establishing the trip setpoints verified by the analysis results presented in this document. The broader range is consistent with that indicated in Figure 15.0.7.2-2, core protection boundaries for four-loop operation.

The range of operating states of the reactor is also considered in the analysis. The effect of exposure on fuel thermal performance and neutronics parameters is considered. State values are selected for the event analyzed to provide the greatest challenge to the acceptance criteria for an event. Several calculations may be required to bound the range of the state variable. For example, a range of neutronic parameters is used in the analysis of rod withdrawal events in order to verify the range of protection of the challenged trip setpoints.

The range of initiating events is also considered in formulating the analysis conditions for an event. The initiating conditions are examined to identify the set which most challenge the acceptance criteria. Where not obvious, sensitivity studies are performed. For example, analyses are performed for uncontrolled rod withdrawal events throughout the range of reactivity insertion rate possible from boron dilution to maximum withdrawal rate of the highest worth control banks. Since the most challenging initial power level is not obvious, the range of power levels permitted by the reactor protection system is also analyzed.

Table 15.0.4-1 Range of Key Initial Condition Operating Parameters

Core thermal power	Subcritical to 2580.6 MWt*
Average coolant temperature (Power operation)	Programmed ± 5°F**
Reactor coolant system pressure	2060 psia ± 50 psi
Pressurizer water level	Programmed ± 5% of level span
Feedwater flow and temperature	Range consistent with power level

* 1.02*2530.

** Program linear with load between (zero load, 532°F) and (rated load, 570.58).

15.0.5 REACTIVITY COEFFICIENTS USED IN THE SAFETY ANALYSIS

Table 15.0.5-1 presents the reactivity coefficients used in the analysis. As discussed in 15.0.4, the set of these parameters which most challenges the event acceptance criteria is used in each analysis and is listed in the appropriate Section for that event. A 20% conservatism factor above those shown in Table 15.0.5-1 is applied to the normal Doppler and moderator coefficients in order to bound the estimates. The conservatism factor is applied in a sense to most challenge the event acceptance criteria.

Table 15.0.5-1 shows that positive moderator temperature coefficient was assumed in the analysis as appropriate. The assumption demonstrates safety of the system under an extreme set of initial conditions, and serves to bound lower power operation with a positive moderator temperature coefficient by a single analysis. The analysis conservatively supports the Technical Specification moderator temperature coefficient of $\leq +0.5 \times 10^{-4}\ \Delta\rho/°F$.

Table 15.0.5-1 Reactivity Parameters

Item	BOC		EOC	
	Nominal	Bounding	Nominal	Bounding
Moderator Temp Coef, 10^{-4} $\Delta\rho$/°F	0.19	0.5	-2.31	-3.5
Doppler Temp Coef, 10^{-5} $\Delta\rho$/°F	-1.32	-1.09	-1.39	-1.76
Moderator Pres Coef, 10^{-6} $\Delta\rho$/psi	0.18	-1.0	2.63	7.0
Delayed Neutron Fraction	0.006	0.0075	0.0053	0.0045
Effective Neutron Lifetime, 10^{-6} seconds	21.7	41.9	24.7	19.9
U^{238} Atoms Consumed per Total Atoms Fissioned	.656	.54	.685	.70

15.0.6 SCRAM INSERTION CHARACTERISTICS

Figure 15.0.6-1 presents the negative insertion used in the analysis for reactor trip. The insertion worth includes the most reactive rod stuck out. The insertion worth of 2.0% $\Delta\rho$ and a control rod drop time of 2.5 seconds (to 90% insertion) have been supported by the transient analysis for the reference cycle.

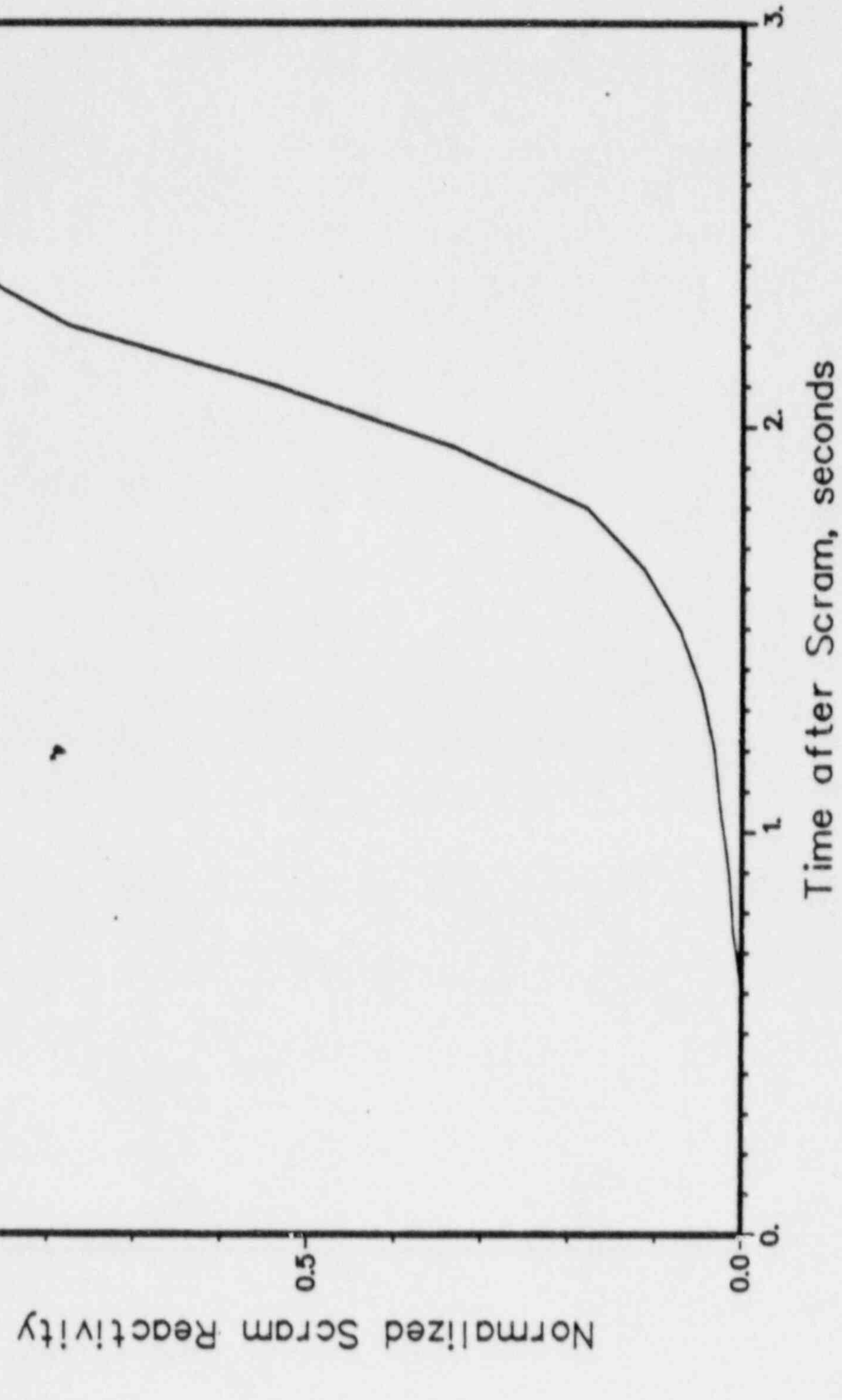


Figure 15.0.6-1 Reactor Trip Reactivity Insertion Curve with Most Reactive Rod Stuck Out

15.0.7 TRIP SETPOINTS AND TIME DELAYS

Table 15.0.7-1 presents the trip setpoints, biases, and time delays used in the analysis. The actual trip setpoints used in each transient analysis were biased such that a minimum DNBR would be calculated. The setpoints used are discussed in the section describing each transient.

A new Variable High Power trip (VHPT) is also accounted for in the analysis. This trip is set to trip the plant at a power level 10 percent above the power at which the plant is currently operating. Two exceptions at which the 10 percent margin does not exist are: 100 percent power; and, powers less than 20 percent of rated. At 100 percent power the trip is set at 106.5 percent of rated. For power levels less than 100%, the trip is designed not to go below a power level of 30 percent of rated. The trip is designed to follow the plant power as the power is reduced maintaining the 10 percent margin to the trip point. When the plant power is being increased the trip point must be manually adjusted upward to avoid tripping the plant. This trip provides additional plant protection for all events initiated from part power conditions.

A new inlet temperature limiting condition of operation (T_{inlet} LCO) and thermal margin/low pressure (TM/LP) trip have also been developed. Their development and the results are briefly presented in the following two sections. This new T_{inlet} LCO was used to develop the initial conditions used in the transient analyses and the new TM/LP trip was included in all transient analyses.

15.0.7.1 Inlet Temperature Limiting Condition Of Operation

The inlet temperature limiting condition of operation (T_{inlet} LCO) provides protection against penetrating DNB during limiting anticipated operational occurance (AOO) transients from full power operation. As shown in Table 2-2, the most limiting AOO transient that does not produce a reactor trip is the inadvertent drop of a full length control assembly. Therefore, the T_{inlet} LCO must provide DNB protection for this transient assuming a return to full power with enhanced peaking due to the anomalous control assembly insertion pattern.

The T_{inlet} LCO was set using the XCOBRA-IIIC computer code(4,11) with a peaking augmentation factor of 1.16. This augmentation factor compares to a maximum calculated value of 1.15. The XCOBRA-IIIC calculations were run to determine the inlet temperature which resulted in a DNB of 1.17 for a range of pressurizer pressures and primary coolant system flow rates. These calculations were performed at 102 percent of rated power, i.e. 2530 MWt, and an axial shape with an axial shape index (ASI) of -.127. Based on an analysis of axial shapes within the range of -.14 to +.544, this was the shape with the minimum DNBR. The derived T_{inlet} LCO will support operation at 100 percent of rated power as long as the measured plant ASI does not become less than -.08 or greater than +.484. This allows for a plant ASI measurement uncertainty of $\pm$.06.

The results of the above analysis will correspond to plant measured values of pressurizer pressure, primary coolant system flow rate and the inlet temperature and will include only the 2 percent power and $\pm$.06 ASI measurement uncertainties. These results must, therefore, be biased to account for both measurement uncertainty and variations due to the control assembly drop transient. The uncertainties which were applied are $\pm$50 psia to the pressurizer pressure, $\pm$7°F to the inlet temperature (5°F tilt allowance + 2°F measurement uncertainty), and $\pm$6 percent to the flow rate (3% bypass flow + 3% measurement uncertainty). The biases resulting from the dropped control assembly were taken from Section 15.4.3 as a 65 psia decrease in the pressurizer pressure, a 4.7°F decrease in the inlet temperature, and an increase in the flow rate of 0.42 Mlb/hr. Applying these biases to the results and fitting gives a T_{inlet} LCO equation of:

$$T_{inlet} \leq 543.35 + .0575*(P-2060) + 5.0 \times 10^{-5}*(P-2060)^2 + 1.173*(W-120) - .0102*(W-120)^2$$

$$1800 \leq P \leq 2200 \text{ psia}$$
$$100 \leq W \leq 130 \text{ Mlb/hr.}$$

For primary loop flow rates greater than 130 Mlbm/hr, the inlet temperature should be limited to the T_{inlet} LCO value at 130 Mlbm/hr.

As a result of the manner in which this T_{inlet} LCO has been developed, it is applicable at 100 percent power for all measured ASI in the range from -.08 to +.484 and can be compared to an average cold leg temperature for each of the four loops. In order that the plant can still operate should the measured ASI become less than -.08 the applicability of the T_{inlet} LCO equation has been extended to a measured ASI of -.30 at 70 percent of rated power. The applicable range of the T_{inlet} LCO is shown in Figure 15.0.7.1-1.

Table 15.0.7-1 Trip Setpoints for Operation of Palisades Reactor at 2530 MWt

	Setpoint	Uncertainty	Delay Time
Low Reactor Coolant Flow	95%	± 2.0%	0.6 sec
High Pressurizer Pressure	2255 psia	± 22 psi	0.6 sec
Low Pressurizer Pressure	1750 psia	± 22 psi	0.6 sec
Low Steam Generator Pressure	500 psia	± 22 psi	0.6 sec
Low Steam Generator Level*	6 feet	± 10 in	0.6 sec
Thermal Margin**	$P = f(T_H, T_C)$	±165 psi	0.6 sec
Variable High Power	106.5% max 30.0% min 10.0% above thermal power	±5.5%	0.4 sec

* Below operating level.

** The thermal margin trip setpoint is a functional pressurizer pressure (P) setpoint, varying as function of the maximum cold leg temperature (T_C), the measured power, and the measured axial shape index.

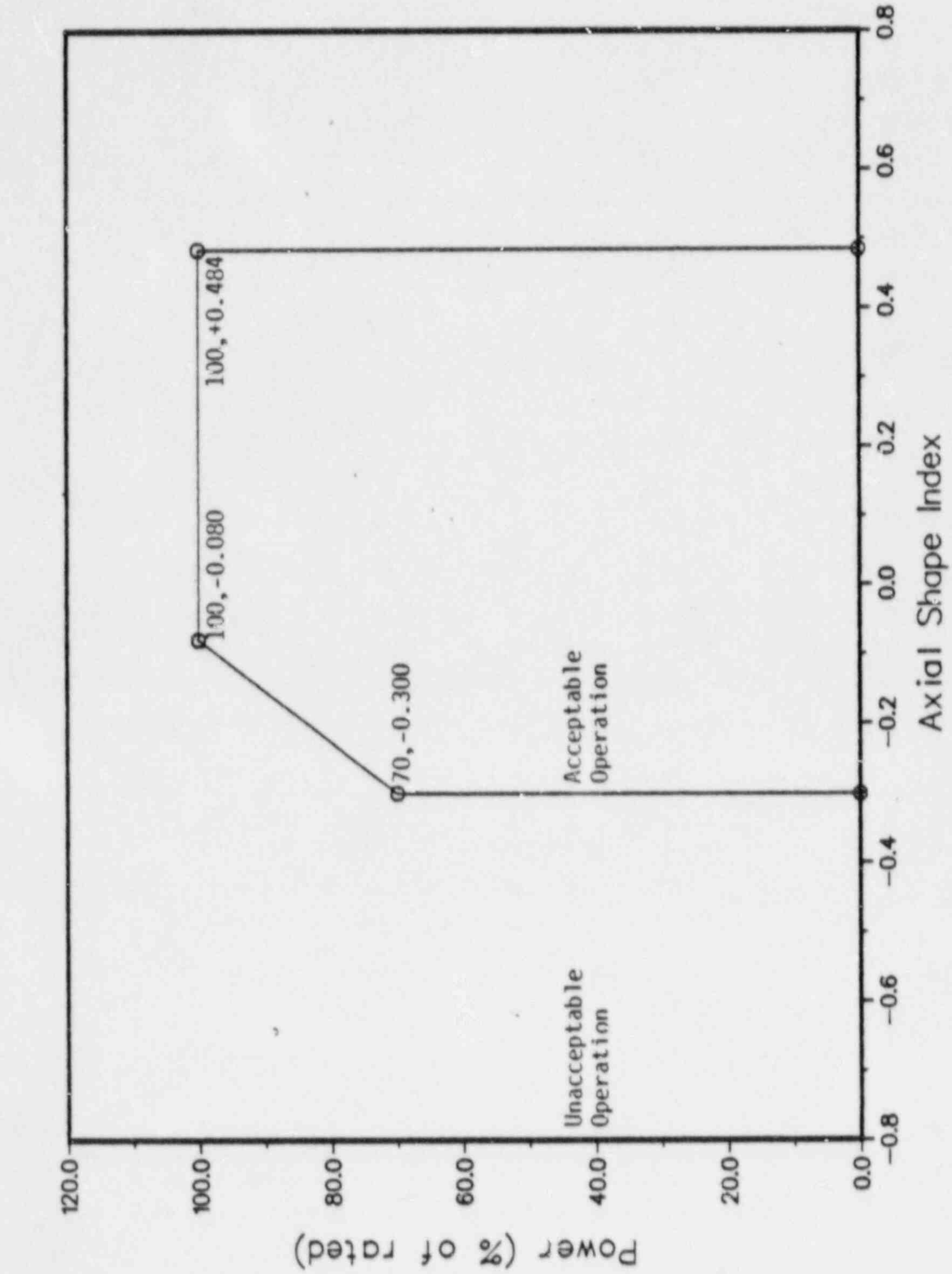


Figure 15.0.7.1-1 Applicable Range of the Tinlet LCO

15.0.7.2 Thermal Margin/Low Pressure (TM/LP) Trip

New hardware for the TM/LP trip is to be installed at the Palisades reactor. This new TM/LP is an improvement over the previous trip in that it allows monitoring of the core axial shape index which provides additional margin in the TM/LP. In the previous Palisades TM/LP, the axial shape used in its development had to be conservative to insure that all probable axial shapes were bounded. With the new trip, the TM/LP can be developed for the optimum axial shape and the axial shape function will adjust the trip as the axial shap aries from the optimum shape.

function of the TM/LP trip is to protect against slow heatup and depressurization transient events. In order to perform this function, the TM/LP trip must initiate a scram signal prior to exceeding the specified acceptable fuel design limits (SAFDLs) on departure from nucleate boiling (DNB) or before the average core exit temperature exceeds the saturation temperature. The SAFDL ins s that there is no damage to the fuel rods and the limit on core exit saturation is imposed to assure meaningful thermal power measurements.

The NSSS is protected against penetrating DNB during rapid power, flow, and pressure transient events by the Variable High Power Trip (VHPT), the low flow trip, and the high pressure trip. For extremely slow transient events, however, it is possible that either the SAFDL on DNB or hot leg saturation could be achieved prior to reaching these trip setpoints. These slow transients generally involve a slow heatup of the primary system caused by a power mismatch between the primary and secondary systems or a slow depressurization of the primary system with or without slow power ramps. Transient events that exhibit these characteristics and must, therefore, be protected by the TM/LP are: rod withdrawals; boron dilution; excess load; loss of feedwater; and, reactor coolant system (RCS) depressurization.

The TM/LP trip works in conjunction with the other trips and the limiting conditions of operation (LCO) on control rod group position, radial peaking, and reactor coolant flow. The VHPT is factored into the TM/LP development by limiting the maximum possible power that can be achieved at a particular radial peaking to 10% above the power corresponding to that radial peaking. The LCO on the control rod group position is included in the TM/LP through monitoring of the axial shapes and the LCO on radial peaking is factored in by including its variation with power level in the TM/LP development. Finally, the LCO on reactor coolant flow is built into the TM/LP through the use of conservative flows throughout its development.

The functional form for the new Palisades TM/LP trip is:

$$P_{var} = \alpha Q_{DNB} + \beta T_{cal} + \gamma \qquad (3.1)$$

where: P_{var} is the low pressure trip limit; α, β, and γ are constants to be determined; T_{cal} is the highest measured cold leg temperature adjusted for possible coolant stratification in the cold leg; and, Q_{DNB} is a function representing axial and radial power peaking effects. The adjusted cold leg temperature T_{cal} is calculated from:

$$T_{cal} = T_{in} + K_c B \qquad (3.2)$$

where B is the measured ΔT power, K_c is a flow stratification factor, and T_{in} is the highest measured cold leg temperature. For NSSS like Palisades, which have the cold leg temperature sensors located downstream of the reactor coolant pumps, there is sufficient mixing so that K_c = 0.0 (Ref. 29) and T_{cal} equals T_{in}. The Q_{DNB} function is represented as:

$$Q_{DNB} = (QA)\ (QR_1)$$

where QA is a function representing the variation in power versus axial shape at constant DNB, and QR_1 is a function representing the variation in power with radial peaking and/or hot leg saturation.

15.0.7.2.1 TM/LP Uncertainties

In setting the TM/LP trip it is necessary to conservatively account for the uncertainties in the measured parameters used to determine the trip. These uncertainties result not only from the inability of the instrumentation to exactly measure the value of a parameter, but also from the fact that the changes in the parameters being measured may actually lag behind the event of interest. Therefore, in any transient uncertainty analysis, both static and transient effects must be considered.

The input parameters for the TM/LP, for which uncertainties must be determined and accounted for in the TM/LP development, are: inlet temperature; power; pressure; and, axial shape index. Three other uncertainties are also included in the analysis. The first is a 3% uncertainty applied to the radial peaking of the peak bundle to account for a power tilt across the bundle. The second is another 3% uncertainty to account for manufacturing tolerances and is applied to the radial peaking of the peak pin in the peak bundle. Finally, the third uncertainty is a 6% decrease in reactor core flow to account for 3% core bypass flow and a 3% measurement uncertainty.

The uncertainty applied to the pressure in this analysis will be 165 psi.[28] This uncertainty was developed to account for most of the uncertainties in the TM/LP. Included in this 165 psi are: instrument drift in both power and inlet temperature; calorimetric power measurement; inlet temperature measurement; and, primary pressure measurement. The uncertainties in these parameters will, consequently, not be treated separately in this analysis.

An additional uncertainty, not accounted for in the 165 psi, is associated with the inlet temperature. This uncertainty accounts for the lag time in the

RTDs and reactor coolant system transit times. The RTD lag time used in the analysis was 12 seconds and the transit time was determined to be 3 seconds. These times were converted to temperature uncertainties through the use of a typical temperature ramp for a slow rod withdrawal. The inlet temperature uncertainty for these time delays was found to be bounded by 1.5°F.

The final uncertainty is in the measured axial shape index (ASI) used in the TM/LP. This uncertainty was taken to be ±.06 consistent with the ASI uncertainty assumed for other CE plants. The uncertainties applied in the development of the TM/LP are summarized in Table 15.0.7.2-1.

15.0.7.2.2 TM/LP Development

In the actual development of the TM/LP, a definite step-by-step procedure is followed. First, the axial shape function QA is developed. This is followed by the determination of the radial peaking function QR_1. Finally, the coefficients α, β, and γ are derived. Throughout this development the various uncertainties are applied to assure that the final TM/LP function is conservative.

The first function derived, QA, is the axial shape function and corrects the TM/LP for the variation in the power at constant DNB with axial shape index (ASI). This function was generated by first finding 21 limiting axial shapes from 504 axial shapes. The limiting axial shapes were determined in .057 ASI increments covering the ASI range from -.653 to +.544. The 21 limiting axial shapes were then used in the XCOBRA-IIIC model to determine the power level required to reach a DNB of 1.17 using the XNB correlation[8]. These powers are then plotted versus the ASI and the data conservatively fitted with three straight lines maximizing the slopes. The QA function is then derived by normalizing the straight line functions to the peak power and inverting them. The derived QA function is plotted in Figure 15.0.7.2-1 and is given in equation form as:

$QA = +.226\ (ASI) + .964 \qquad +.162 \leq ASI \leq +.544$

$QA = -.521\ (ASI) + 1.085 \qquad -.156 \leq ASI \leq +.162$

$QA = -.691\ (ASI) + 1.058 \qquad -.653 \leq ASI \leq -.156$

Note that this QA function has been developed to cover the full range of the possible ASIs.

The radial peaking function, QR_1, accounts for the changes in slope of the safety limit lines (SLL) with radial peaking and hot leg saturation. The SLLs are parallel equally spaced lines representing the variation in the maximum allowed inlet temperature with power and pressure. These lines are composed of two limiting portions: the first, which dominates at low powers, is the inlet temperature which produces saturation in the hot leg; and, the second is the inlet temperature which produces the minimum allowed DNBR.

These SLLs are generated by first determining the hot leg saturation curves which can be easily derived from the ASME steam tables. The DNB limiting portion of the lines are calculated using XCOBRA-IIIC and the XNB correlation until it intersects with the hot leg saturation curve. The final SLLs are derived by fitting these curves with parallel equally spaced line segments. Uncertainties in power, pressure, and inlet temperature are also applied in the generation of these final SLLs.

In order to insure that the QR_1 function derived for the TM/LP is conservative for all ASIs, it must be derived using the axial shape which corresponds to the QA = 1 point of the ASI function. The QR_1 will be conservative because the SLLs will be dominated by the hot leg saturation and will give the largest slope change between hot leg saturation and DNB limited portions of the SLLs. For the QA function derived alone, the axial shape to be used is, therefore, the shape with an ASI = +.162. The SLLs generated with this axial shape are shown in Figure 15.0.7.2-2.

With the completion of the SLLs, the QR_1 function can now be derived. Since there is only one slope change in the final SLLs, the QR_1 function will be composed of two straight lines. The QR_1 function is therefore:

$$QR_1 = 0.412\ Q + 0.588 \qquad Q \leq 1.0$$

and

$$QR_1 = Q \qquad Q > 1.0$$

where Q is the fraction of 100% power, i.e., 2530 MWt.

With the completion of the QR_1 function, it is now possible to derive the coefficients in the TM/LP equation. These coefficients are derived using the SLLs for an ASI of +.162, i.e. QA=1.0, and adjusting the coefficients for an ASI uncertainty of ±.06. This yields coefficients of:

$$\alpha = 1563.7$$
$$\beta = 12.3$$
$$\gamma = -6503.4$$

Thus, the TM/LP trip function is

$$P_{var} = 1563.7\ (QA)\ (QR_1) + 12.3\ (T_{in}) - 6503.4$$

where:

$$QR_1 = 0.412\ (Q) + 0.588 \qquad Q \leq 1.0$$
$$QR_1 = Q \qquad Q \geq 1.0$$

and,

QA = +.226 (ASI) + .964	$+.162 \le ASI \le +.544$
QA = -.521 (ASI) + 1.085	$-.156 \le ASI \le +.162$
QA = -.691 (ASI) + 1.058	$-.657 \le ASI \le -.156$

This TM/LP is applicable over a pressure range from 1700 psia to 2300 psia and to a minimum measured HZP primary coolant flow rate of 124.3 Mlb/hr.

Table 15.0.7.2-1 TM/LP Uncertainties

Instrument Drift (Power, T_{inlet})	165 psi
Calorimetric Power	
T_{inlet} measurement	
Pressure Measurement	
RTD Measurement	
Engineering Tolerances	3%
Reactor Coolant Flow	6%
Inlet Temperature Time Delay	1.5°F
Axial Shape Index	±.06

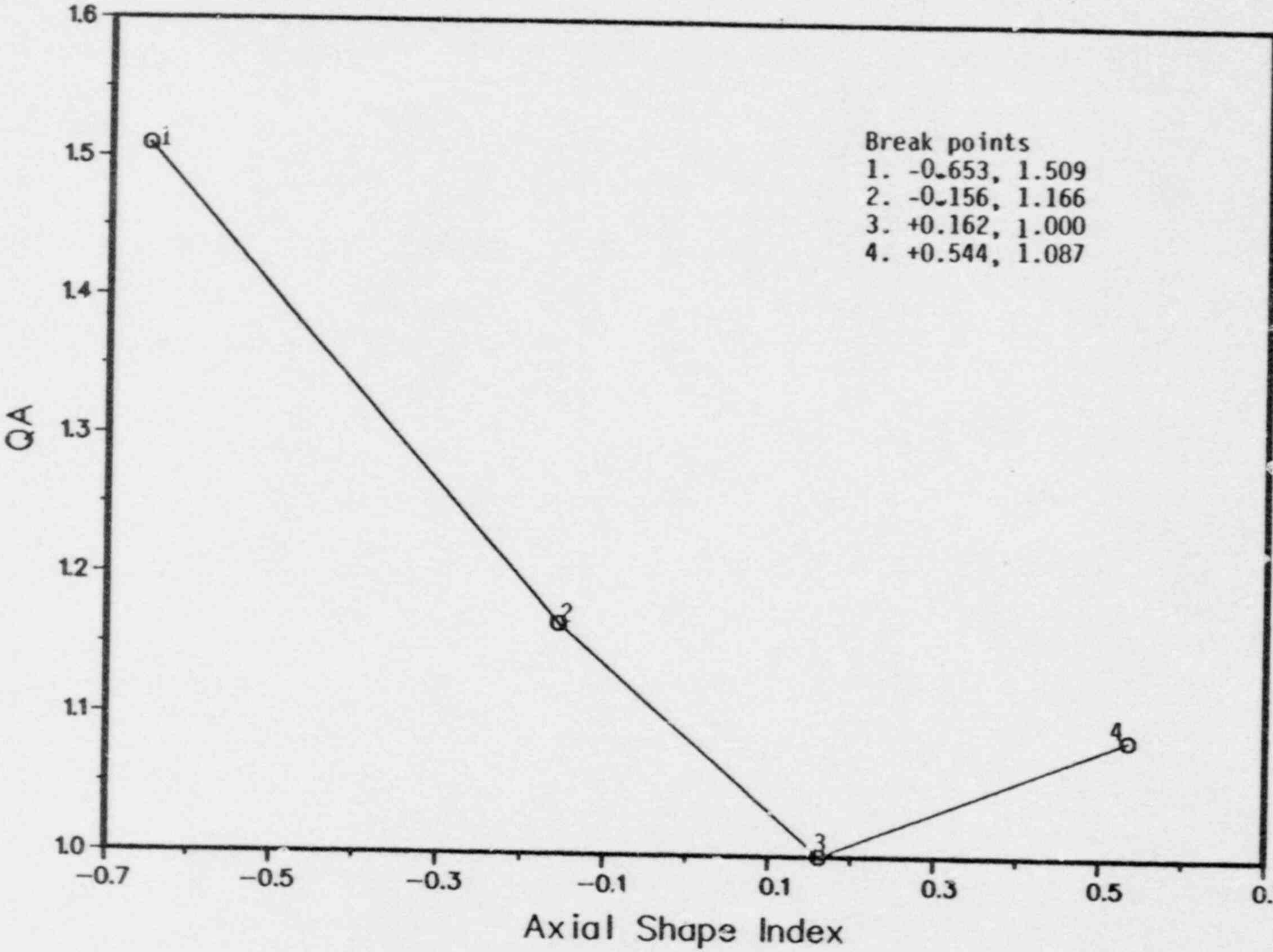


Figure 15.0.7.2-1 Axial Shape Function with no ASI Limits

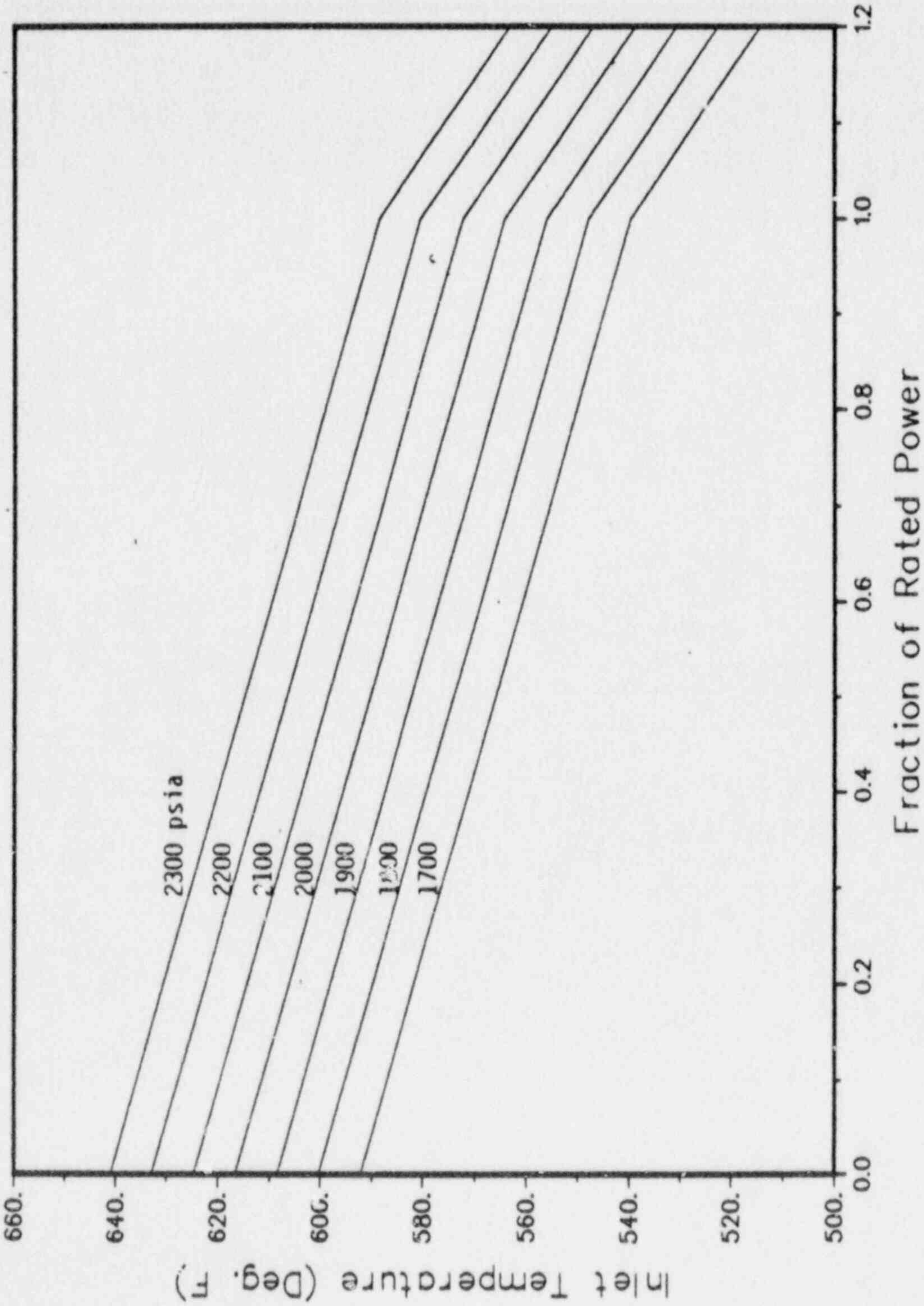


Figure 15.0.7.2-2 Biased Safety Limit Lines
(ASI = +0.162)

15.0.8 COMPONENT CAPACITIES AND SETPOINTS

Table 15.0.8-1 presents the component setpoints and capacities used in the analysis. Refer to Reference 6 for a more complete compilation of the plant components.

Table 15.0.8-1 Component Capacities and Setpoints

	Response Time (sec)	Setpoint Nominal	Setpoint Uncertainty	Total Capacity
Pressurizer safety valves (3)		2500 psia 2540 psia 2580 psia	25 psia	191.7 lbm/s
Pressurizer relief valves		Blocked closed		
Steamline relief valves (24)		Group A at 1000 psia Group B at 1020 psia Group C at 1040 psia	3%	3244 lbm/s at 1000 psia
Turbine stop and centered valves	0.1			
Steam dump valves and turbine bypass	3.0	Turbine trip then T_{ave} program		1173 lbm/s at 770 psia
Pressurizer backup heaters		Always On		1350 kW
Pressurizer proportional heaters		Full On-1960 psia Full Off-2010 psia	50 psia 50 psia	150 kW
Pressurizers sprays		Full On-2110 psia Full Off-2060 psia	50 psia 50 psia	29.4 lbm/s (1.5 gpm continuous flow)
Letdown orifice valves		Level controller		12.6 lbm/s
CVCS Makeup system		Level controller		18.5 lbm/s
Normal Feedwater system	20.5	Feedwater controller		3321.4 lbm/s

15.0.9 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS

Table 15.0.9-1 is a summary of trip functions, engineered safety features, and other equipment available for mitigation of accident effects. These are listed for all Chapter 15 SRP events. A more detailed listing of available reactor protection for each event in each operating mode is given in Reference 3.

Table 15.0.9-1 Overview of Plant Systems and Equipment Available for Transient and Accident Conditions

	Event	Reactor Trip Functions	Other Signals and Equipment
15.1	Increase in Heat Removal by the Secondary System		
	Feedwater System Malfunctions	Variable High Power Trip Lower Pressurizer Pressure Trip Thermal Margin/Low Pressure Trip Low Steam Generator Pressure Trip Safety Injection Actuation Signal	Steam Generator Water Level Signals Feedwater Isolation Valves Main Steamline Isolation Valves Turbine Trip on Reactor Trip Chemical and Volume Control System (CVCS)
	Increase in Steam Flow	Low Steam Generator Pressure Trip Thermal Margin/Low Pressure Trip Variable High Power Trip Low Pressurizer Pressure Trip Safety Injection Actuation Signal	Steam Generator Water Level Signals Main Steamline Isolation Valves Turbine Trip on Reactor Trip Atmospheric Steam Dump Controller Steam Bypass to Condenser Controller Auxiliary Feedwater System CVCS
	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Low Steam Generator Pressure Trip Thermal Margin/Low Pressure Trip Variable High Power Trip Low Pressurizer Pressure Trip Safety Injection Actuation Signal	Steam Generator Water Level Signals Main Steamline Isolation Valves Turbine Trip on Reactor Trip Atmospheric Steam Dump Controller Steam Bypass to Condenser Controller Auxiliary Feedwater System CVCS

Table 15.0.9-1 Overview of Plant Systems and Equipment Available for Transient and Accident Conditions, (Cont.)

	Event	Reactor Trip Functions	Other Signals and Equipment
	Steam System Piping Failure	Low Steam Generator Pressure Trip Thermal Margin/Low Pressure Trip Variable High Power Trip Low Pressurizer Pressure Trip Safety Injection Actuation Signal High Containment Pressure	Steam Generator Water Level Signals Main Steamline Isolation Valves Turbine Trip on Reactor Trip Atmospheric Steam Dump Controller Steam Bypass to Condenser Controller Auxiliary Feedwater System Containment Spray Containment Isolation Containment Air Coolers CVCS
15.2	Decrease in Heat Removal by the Secondary System		
	Loss of External Load/Turbine Trip/Loss of Condenser Vacuum	High Pressurizer Pressure Trip Variable High Power Trip Thermal Margin/Low Pressure Trip Low Steam Generator Water Level Trip	Steam Generator Water Level Signals Turbine Trip on Reactor Trip Atmospheric Steam Dump Controller Steam Bypass to Condenser Controller Steam Generator Safety Valves Pressurizer Safety Valves Pressurizer Sprays
	Loss of Nonemergency AC Power to the Station Auxiliaries	Low Reactor Coolant Flow Trip High Pressurizer Pressure Trip Thermal Margin/Low Pressure Trip Low Steam Generator Water Level Trip	Steam Generator Water Level Signals Steam Generator Safety Valves Pressurizer Safety Valves Auxiliary Feedwater System
	Loss of Normal Feedwater Flow	Low Steam Generator Water Level Trip High Pressurizer Pressure Trip Thermal Margin/Low Pressure Trip	Steam Generator Water Level Signals Steam Generator Safety Valves Pressurizer Safety Valves Auxiliary Feedwater System Pressurizer Sprays and Level Control

Table 15.0.9-1 Overview of Plant Systems and Equipment Available for Transient and Accident Conditions, (Cont.)

	Event	Reactor Trip Functions	Other Signals and Equipment
	Feedwater System Pipe Break	High Pressurizer Pressure Trip Thermal Margin/Low Pressure Trip Low Steam Generator Water Level Trip Low Steam Generator Pressure Trip	Steam Generator Water Level Signals Steam Generator Safety Valves Pressurizer Safety Valves Auxiliary Feedwater System Pressurizer Sprays and Level Control
15.3	Decrease in Reactor Coolant System Flow Rate		
	Loss of Forced Reactor Coolant Flow	Low Reactor Coolant Flow Trip Thermal Margin/Low Pressure Trip High Pressurizer Pressure Trip	Atmospheric Steam Dump Controller Steam Bypass to Condenser Controller Steam Generator Safety Valves Pressurizer Safety Valves
	Reactor Coolant Pump Rotor Seizure/Shaft Break	Low Reactor Coolant Flow Trip High Pressurizer Pressure Trip	Atmospheric Steam Dump Controller Steam Bypass to Condenser Controller Steam Generator Safety Valves Pressurizer Safety Valves
15.4	Reactivity and Power Distribution Anomalies		
	Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Startup Condition	Thermal Margin/Low Pressure Trip Variable High Power Trip High Pressurizer Pressure Trip	Nonsafety Grade High Rate-of-Change of Power Trip High Rate-of-Change of Power Alarms, which initiate Rod Withdrawal Prohibit Action
	Uncontrolled Control Rod Bank Withdrawal at Power Operation Conditions	Variable High Power Trip Thermal Margin/Low Pressure Trip High Pressurizer Pressure Trip	Pressurizer Safety Valves Steam Generator Safety Valves Pressurizer Spray and Level Control Control Rod and Bank Deviation Alarms

Table 15.0.9-1 Overview of Plant Systems and Equipment Available for Transient and Accident Conditions, (Cont.)

Event	Reactor Trip Functions	Other Signals and Equipment
Control Rod Misoperation	Low Pressurizer Pressure Trip Thermal Margin/Low Pressure Trip Low Steam Generator Water Level Trip Safety Injection Actuation Signal	Pressurizer Safety Valves Steam Generator Safety Valves Pressurizer Spray and Level Control Control Rod and Bank Deviation Alarms CVCS
Startup of an Inactive Loop	Variable High Power Trip Thermal Margin/Low Pressure Trip	Administrative Procedures for Startup of an Idle Pump. Plant Operation with less than all four primary coolant pumps is not permitted by Technical Specifications except for very short periods of time and at reduced power levels (Tech Spec Table 2.3.1).
Chemical Volume and Control System (CVCS) Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	Variable High Power Trip Thermal Margin/Low Pressure Trip High Pressurizer Pressure Trip	Nonsafety Grade High Rate-of-Change of Power Trip Administrative Procedures Sufficient Operator Response Time
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	(Technical Specification Measurement Requirement and Administrative Procedures preclude Occurrence)	
Spectrum of Control Rod Ejection Accidents	Variable High Power Trip Thermal Margin/Low Pressure Trip Long Term, Low Pressurizer Pressure Trip Long Term, Safety Injection Actuation Signal	Nonsafety Grade High Rate-of-Change of Power Trip CVCS

Table 15.0.9-1 Overview of Plant Systems and Equipment Available for Transient and Accident Conditions, (Cont.)

	Event	Reactor Trip Functions	Other Signals and Equipment
15.5	Increase in Reactor Coolant Inventory		
	Inadvertent Operation of the ECCS/CVCS Malfunction that Increases Reactor Coolant Inventory	Variable High Power Trip Thermal Margin/Low Pressure Trip High Pressurizer Pressure Trip	Nonsafety Grade High Rate-of-Change of Power Trip Pressurizer Safety Valves Overpressurization Mitigation System (Modes 6-8)
15.6	Decrease in Reactor Coolant Inventory		
	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	Low Pressurizer Pressure Trip Thermal Margin/Low Pressure Trip Safety Injection Actuation Signal	Safety Injection System Pressurizer Heaters CVCS
	Steam Generator Tube Failure	Thermal Margin/Low Pressure Trip Low Pressurizer Pressure Trip Safety Injection Actuation Signal	Steam Generator Safety Valves Main Steamline Isolation Valves (MSIVs) Atmospheric Steam Dump Controller Steam Bypass to Condenser Controller Auxiliary Feedwater System CVCS
	Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	No credit taken for a reactor trip by the Reactor Protection System (RPS) due to the rapid depletion of the moderator which shuts down the reactor core almost immediately, followed by ECCS injection which contains sufficient boron to maintain the reactor core in a subcritical configuration.	Emergency Core Cooling System (ECCS) Auxiliary Feedwater System Containment Isolation Containment Spray and Air Cooler

15.0.10 EFFECTS OF MIXED ASSEMBLY TYPES AND FUEL ROD BOWING

The Palisades reference cycle core contains only ANF assemblies and the thermal hydraulic designs of these assemblies are identical. Therefore, there is no need to apply a mixed core DNBR penalty.

The effects of rod bow on limiting DNB and heat flux peaking were considered. Reference 9 concludes that due to the short distances between spacers the 15x15 design does not exhibit fuel rod bow of any significance to plant operating margins. Therefore, no penalty is applied due to rod bow effects.

15.0.11 PLANT LICENSING BASIS AND SINGLE FAILURE CRITERIA

The licensing basis for Palisades is as stated in the Final Safety Analysis Report[5]. The event scenarios depend on single failure criteria established by the plant licensing basis. Examination of the Palisades licensing basis yields the following single failure criteria:

(1) The Reactor Protection System (RPS) is designed with redundancy and independence to assure that no single failure or removal from service of any component or channel of a system will result in the loss of the protection function.

(2) Each Engineered Safety Feature (ESF) is designed to perform its intended safety function assuming a failure of a single active component.

(3) The onsite power system and the offsite power system are designed such that each shall independently be capable of providing power for the ESF assuming a failure of a single active component in either power system.

The safety analysis is structured to demonstrate that the plant systems design satisfies these single failure criteria. The following assumptions result:

(1) The ESF required to function in an event are assumed to suffer a worst single failure of an active component.

(2) Reactor trips occur at the specified setpoint within the specified delay time assuming a worst single active failure.

(3) The following postulated accidents are considered assuming a concurrent loss of offsite power: main steamline break, control rod ejection, steam generator tube rupture, and LOCA.

(4) The loss of normal feedwater, an anticipated operational occurrence, is analyzed assuming a concurrent loss of offsite power.

The requirements of 10 CFR 50, Appendix A, Criteria 10, 20, 25 and 29 require that the design and operation of the plant and the reactor protective system assure that the Specified Acceptable Fuel Design Limits (SAFDLs) not be exceeded during Anticipated Operational Occurrences (AOOs). As per the definition of AOO in 10 CFR 50, Appendix A, "Anticipated Operational Occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the plant and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power". The Specified Acceptable Fuel Design Limits (SAFDLs) are that: (1) the fuel shall not experience centerline melt (21 kW/ft); and (2) the departure from nucleate boiling ratio (DNBR) shall have a minimum allowable limit such that there is a 95% probability with a 95% confidence interval that departure from nucleate boiling (DNB) has not occurred (XNB DNBR of 1.17).

15.0.12 PLOT VARIABLE NOMENCLATURE

Plotted results presented in this report employ PTSPWR2 output variable nomenclature. Specific variables plotted are listed and defined in Table 15.0.12-1.

Table 15.0.12-1 Nomenclature Used in Plotted Results

Variable Name	Definition
DK	Total Reactivity
DKDOP	Doppler Reactivity
DKMOD	Moderator Temperature Reactivity
LEVPR	Pressurizer Liquid Level
LEVSG1	Steam Generator Liquid Level, Loop 1
PDO1	Steam Generator Dome Pressure, Loop 1
PL	Core Power Level
PPR	Pressurizer Pressure
QOA	Core Average Heat Flux
TAVG1	Average Coolant Temperature, Loop 1
TCIO	Core Inlet Coolant Temperature
TCLAD	Average Clad Temperature
TCL1	Cold Leg Temperature, Loop 1
TFAVG	Average Fuel Temperature
THL1	Hot Leg Temperature, Loop 1
WDOSLT	Total Steamline Steam Flow Rate
WFWT	Total Feedwater Flow Rate
WLPCR	Vessel Flow Rate

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.1.3 INCREASE IN STEAM FLOW (EXCESS LOAD)

15.1.3.1 Identification of Causes and Event Description

The increase in steam flow event is initiated by an increase in steam demand. The increased steam demand may be initiated by the operator or by regulating valve malfunction. The step increase in steam flow results from a rapid opening of the turbine control valves, atmospheric dump valves, or the turbine bypass valve to condenser.

The following increased steam flow events were considered and represent the limiting increased steam demand events. First, an increased steam flow from 102% of rated power caused by the sudden opening of the turbine control valves, the atmospheric dump valves and the turbine bypass valves. Second, an increased steam demand from hot shutdown conditions caused by the rapid opening of the atmospheric dump valves and the turbine bypass valves.

The event initiator is a step increase in steam flow. The feedwater regulating valves open to increase the feedwater flow in an attempt to match the increased steam demand and maintain steam generator water level. In response to the increased steam flow, the secondary system pressure decreases, resulting in an increase in the primary-to-secondary heat transfer rate. The primary side steam generator outlet temperature decreases due to the enhanced heat removal. As a consequence, the primary system core average temperature decreases and the primary system fluid contracts, resulting in an outsurge of fluid from the pressurizer. The pressurizer level and pressure decrease as fluid is expelled from the pressurizer. If the moderator temperature coefficient is negative, the reactor core power increases as the moderator temperature decreases due to the mismatch between the power being removed by the steam generators and the power being generated in the core.

Thermal margin/low pressure and variable overpower trips are available to prevent the violation of the acceptance criteria. Depending on the magnitude of the increase in steam demand, a reactor trip may not be activated. Instead, the reactor system will reach a new steady-state condition at a power level greater than the initial power level which is consistent with the increased heat removal rate. The final steady-state conditions which are achieved will depend upon the magnitude of the moderator temperature coefficient. If the moderator temperature coefficient is positive, the reactor power would decrease as the core average coolant temperature decreased, and this event would not produce a challenge to the acceptance criteria.

This event is classified as a Moderate frequency event (Table 15.0.1.2-1). The relevant acceptance criteria are described in 15.0.1.1. Single failure criteria for Palisades are given in 15.0.11. Single failures in the reactor protection system (RPS) or in the engineered safety features (ESF) will not affect the outcome of this event because neither PPS nor ESF is required to function to assure that acceptance criteria are met.

15.1.3.2 <u>Analysis Method</u>

The transient response of the reactor system is calculated using the PTSPWR2 computer program[10]. The core thermal hydraulic boundary conditions from the PTSPWR2 calculation are used as input to the XCOBRA-IIIC code[11] to predict the minimum DNBR for the event initiated at 102% power. Due to the low pressure conditions experienced in the event initiated from hot shutdown, the Modified Barnett correlation was used in place of XNB.

15.1 3.3 Definition of Events Analyzed and Bounding Input

This event is predominantly a depressurization event, so the primary concern for this event is the challenge to the specified acceptable fuel design limits (SAFDLs). Therefore, the cases identified for analysis for this event are selected on the basis of bounding the largest challenge to the SAFDLs.

Two cases are analyzed for this event. The first case is initiated from full power rated conditions. At full power, the margin to the SAFDLs is the smallest. Thus, full power conditions bound operation at lower power levels. The second case is initiated from hot shutdown conditions. This case bounds the event consequences for transients initiated from refueling shutdown, cold shutdown and refueling operation initial conditions.

In both cases, end of cycle moderator and Doppler feedback coefficients were selected to maximize the challenge to the SAFDLs. The time in the cycle will determine the value of the moderator reactivity temperature coefficient. The moderator temperature coefficient becomes more negative as the cycle burnup increases. If the moderator reactivity temperature coefficient is negative, there will be a positive reactivity insertion dependent on the magnitude of the moderator reactivity temperature coefficient. If it is positive, then negative reactivity will be inserted as the coolant temperature decreases, causing the power to decrease with less challenge. The reactor control rod system at Palisades is disabled so that the control rods will not withdraw automatically in response to the decrease in core average temperature. Therefore, the consequences of this event are bounded by the event consequences at end of cycle conditions when the moderator temperature coefficient is at its maximum negative value.

The two cases were analyzed using the following assumptions:

Table 15.1.3-0 Conservative Assumptions Used in the Increase in Steam Flow (Excess Load)

	Zero Power Case	Full Power Case
Control	Manual	Manual
Core Power	1×10^{-10} MWt	Rated +2%
Core Inlet temperature	Nom. +5°F	Nom. +5°F
Primary Pressure	Nom. +50 psi	Nom. -50 psi
Moderate temperature coefficient	1.2 EOC	1.2 EOC
Doppler coefficient	.8 EOC	.8 EOC
Pellet-to-clad heat transfer coef.	Nom. +20%	Nom. +20%
Pressurizer Level	Nom. +5% of Level Span	Nom. +5% of Level Span
Pressurizer heater	Unavailable	Unavailable
Pressurizer level control	Constant charging flow	Disable

The above conditions conservatively bound operating uncertainties.

15.1.3.4 <u>Analysis Results</u>

For the full power case, the event is initiated by a rapid opening of the turbine control valves, the atmospheric dump valves and/or the turbine bypass valves resulting an increase in steam flow. The maximum increased steam flow rate at full power is 130% of rated, or 3961.4 lbm/sec, assuming the simultaneous opening of each of the secondary-side valves. A bounding value for the negative moderator temperature coefficient (EOC conditions) is assumed.

To bound the potential consequences of an increase in steam flow event from full power initial conditions, several cases were examined in which the steam flow rate was varied between 110% and 130%. The minimum DNBR for this event occurred for a steam flow increase of about 112%. At this steam flow rate, the TM/LP and the variable high power trips coincide producing nearly simultaneous trip signals. The junction of these two trips represents the worst possible DNB conditions, that is, maximum core power is attained combined with a low pressurizer pressure. For steam flow rates less than 112%, the primary system heat generation is balanced by the heat extraction rate by the secondary side at less limiting steady-state conditions within the setpoints of both the variable high power and TM/LP trips. For steam flow rates greater than 112%, either the variable high power or the TM/LP trip will terminate the event with less limiting DNB conditions.

The above results for this event initiated from full power were obtained assuming an initial pressurizer pressure bias of -50 psia and a steam flow ramp rate based on a 0.1 second valve opening time. Since the minimum DNBR occurs at the junction of both the variable high power and TM/LP trips, the MDNBR result is independent of the value of the initial pressurizer pressure bias and the steam flow ramp rate. This is true because the core conditions at the point of MDNBR are determined by the setpoints at the intersection of the variable high power and TM/LP trips.

For the hot shutdown case, the event was initiated by a rapid opening of the atmospheric dump valves and the turbine bypass valves resulting in a steam flow increase of 28% of the nominal full power steam flow. A bounding value for the negative moderator temperature coefficient (EOC conditions) was assumed. Due to the cooldown of the primary coolant, coupled with a negative moderator temperature coefficient, the reactor becomes critical resulting in a significant return-to-power.

Control rods were inserted at 0.0 seconds with a reactivity worth representative of the required shutdown margin for four primary coolant pump operation. No subsequent scram reactivity was assumed to be available, hence the rapid increase in reactor power due to the positive reactivity insertion was terminated by the effects of Doppler feedback only. Figure 15.1.3-6 shows the change in reactivity during this event. Insertion of 2% shutdown reactivity is shown at 0.0 seconds. As the fuel temperature decreases, as a result of the increased heat extraction rate from the primary system, a negative Doppler coefficient inserts positive reactivity. This erodes the available shutdown margin and causes a return-to-power. The Doppler coefficient eventually terminates the reactivity increase as the core power becomes equilibrated with the heat removal capacity. No credit was taken for boron addition to the primary system from either the charging or high pressure safety injection (HPSI) systems.

The primary system cooldown results in the emptying of the pressurizer during the transient. When the pressurizer is determined to empty, PTSPWR2 forces the pressurizer level to be a small non-zero value. This level is maintained as long as the pressurizer is "empty". The pressure used in XCOBRA-IIIC to evaluate the margin to the SAFDLs was taken as the saturation pressure at the hot leg temperature. Since this pressure is below the valid range of the XNB correlation, the Modified Barnett correlation was used to predict the critical heat flux ratio. This approach is acceptable since the primary system cooldown is not sufficient enough to uncover the core and the pressure used in the DNBR calculations is taken to be the value at saturated conditions corresponding to the hot leg temperature.

Initial conditions used in the analyses are given in Table 15.1.3-1.

The transient response for the EOC zero-power case is shown in Figures 15.1.3-1 to 15.1.3-[illegible]. The event sequence is summarized in Table 15.1.3-2. The minimum DNBR [illegible] case, using the Modified Barnett correlation, is 2.06.

The transient response to the EOC full-power case is shown in Figures 15.1.3-12 to [illegible] 22. The event sequence is summarized in Table 15.1.3-2. Reactor trip occurred on a variable high power signal at 71.41 seconds. The minimum DNBR computed for this case is 1.497.

Plotted variables are defined in Table 15.0.12-1.

15.1.3.5 Conclusion

The results of the analysis demonstrate that the event acceptance criteria are met since the minimum DNBR predicted for the full power case is greater than the XNB correlation safety limit of 1.17 and the minimum CHFR predicted for the hot shutdown case is greater than the Modified Barnett Safety limit of 1.135. The correlation limit assures that with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The event results in depressurization or small pressure increase, and therefore does not challenge the vessel pressurization criterion of 2750 psia. The fuel centerline melt threshold of 21 kW/ft is not approached in this event. Peak pellet LHGR does not exceed 13.2 kW/ft for the full power case.

Table 15.1.3-1 Summary of Initial Operating Conditions for Increase in Steam Flow (Excess Load)

	Zero Power Case	Full Power Case
Power (MWt)	$1. \times 10^{-10}$	2580.6
Core Inlet Temperature (°F)	537.0	548.65
Pressurizer Pressure (psia)	2110	2010
Reactor Coolant System Flow Rate (lbm/hr)	116.6×10^{6}	116.6×10^{6}
Steam Dome Pressure (psia)	939.0	769.8

Table 15.1.3-2 Event Summary for Increase in Steam Flow (Excess Load)

Zero Power Case Event Summary

Event	Time (sec)
28% Step Increase in Steam Flow	0.00
Peak Pressurizer Pressure	0.13
Peak Core Average Temperature	2.34
Peak Power	72.95
Peak Core Average Heat Flux	81.73

102% Rated Power Case Event Summary

Event	Time (sec)
12% Step Increase in Steam Flow	0.00
Peak Core Average Temperature	.74
Reactor trip (VHP trip)	71.41
Peak Power	71.98
Minimum DNBR	72.02
Peak Pressurizer Pressure	73.73

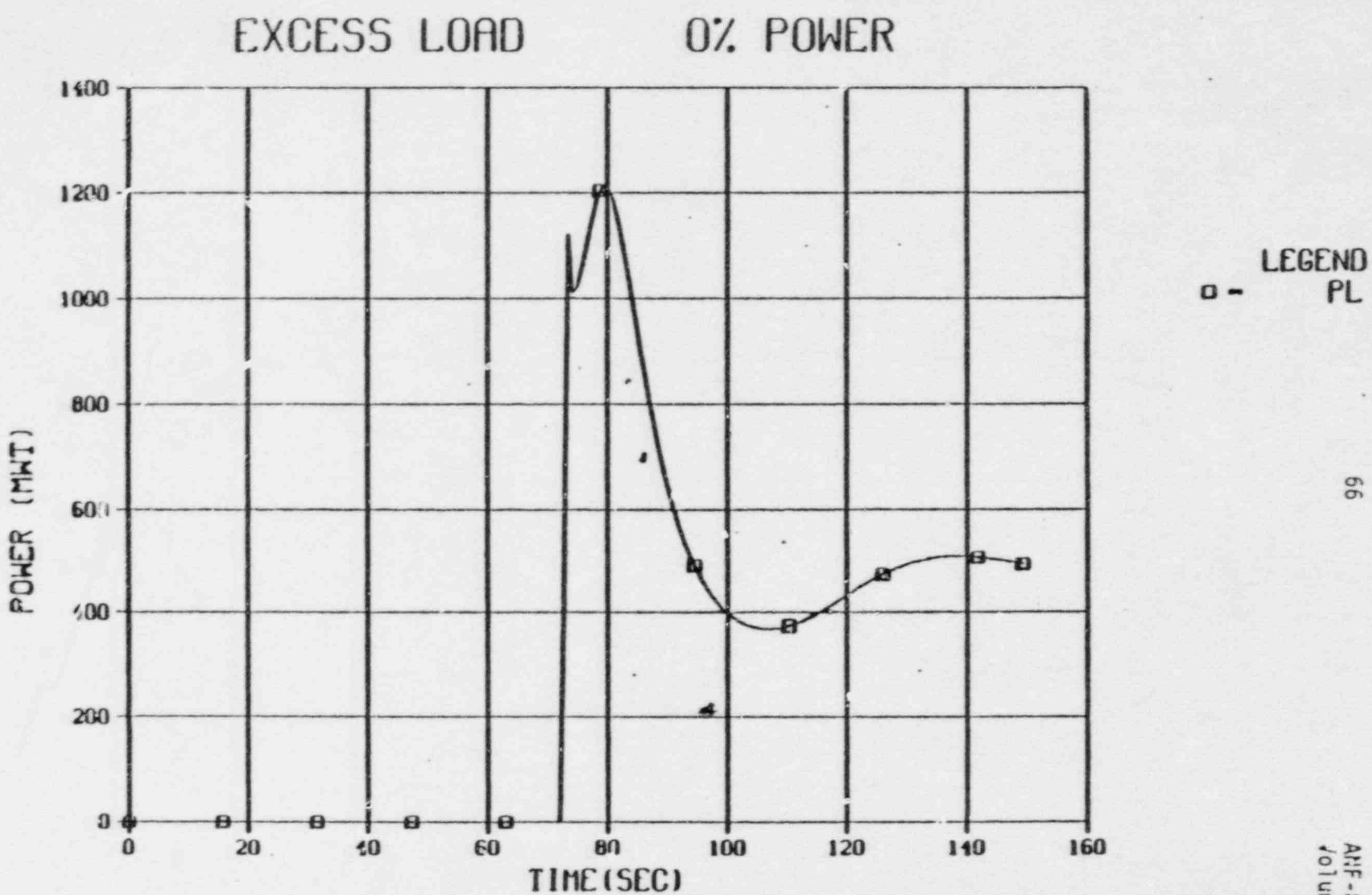


Figure 15.1.3-1 Reactor Power Level for Excess Load from Zero Power

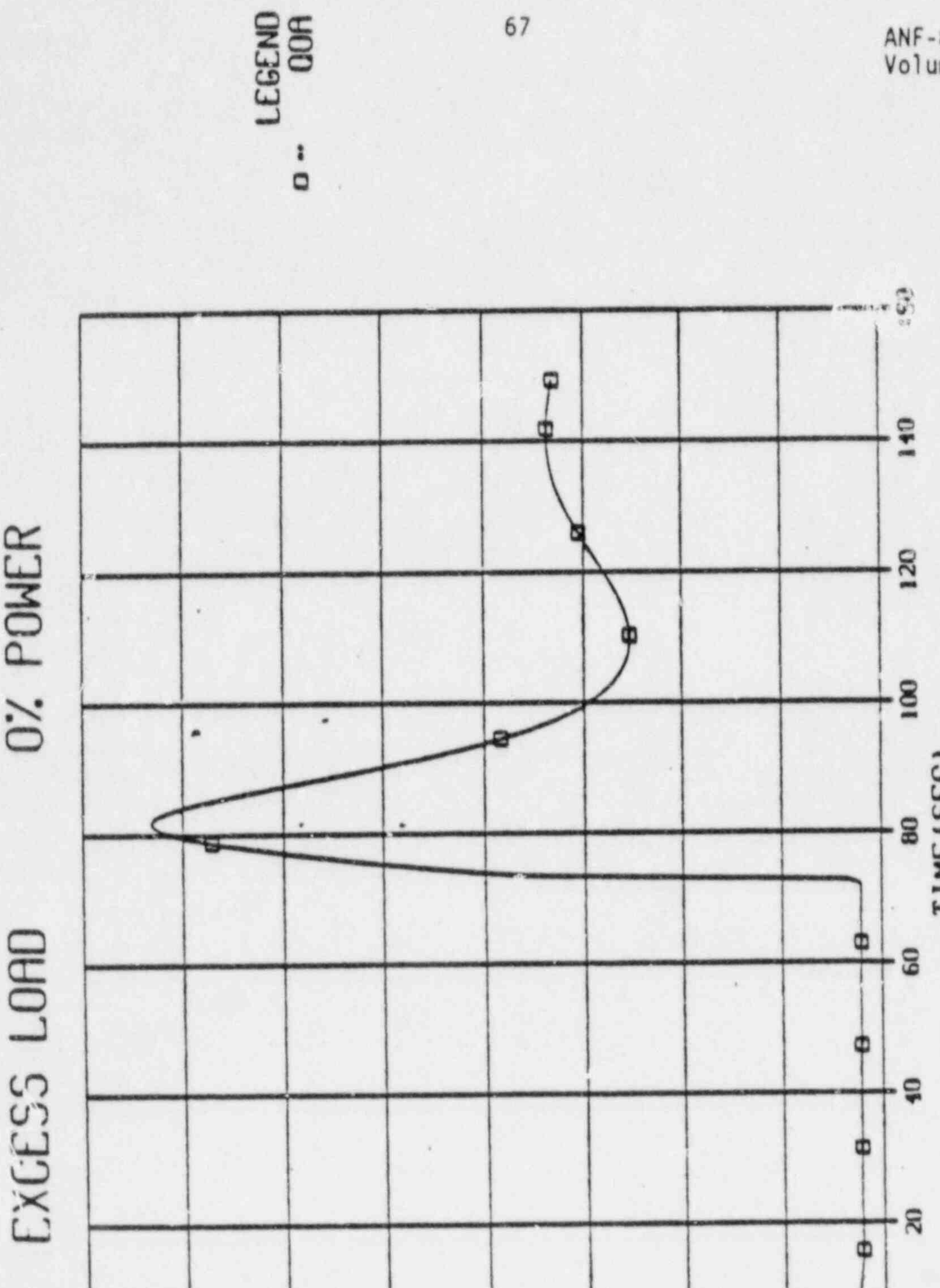


Figure 15.1.3-2 Core Average Heat Flux for Excess Load from Zero Power

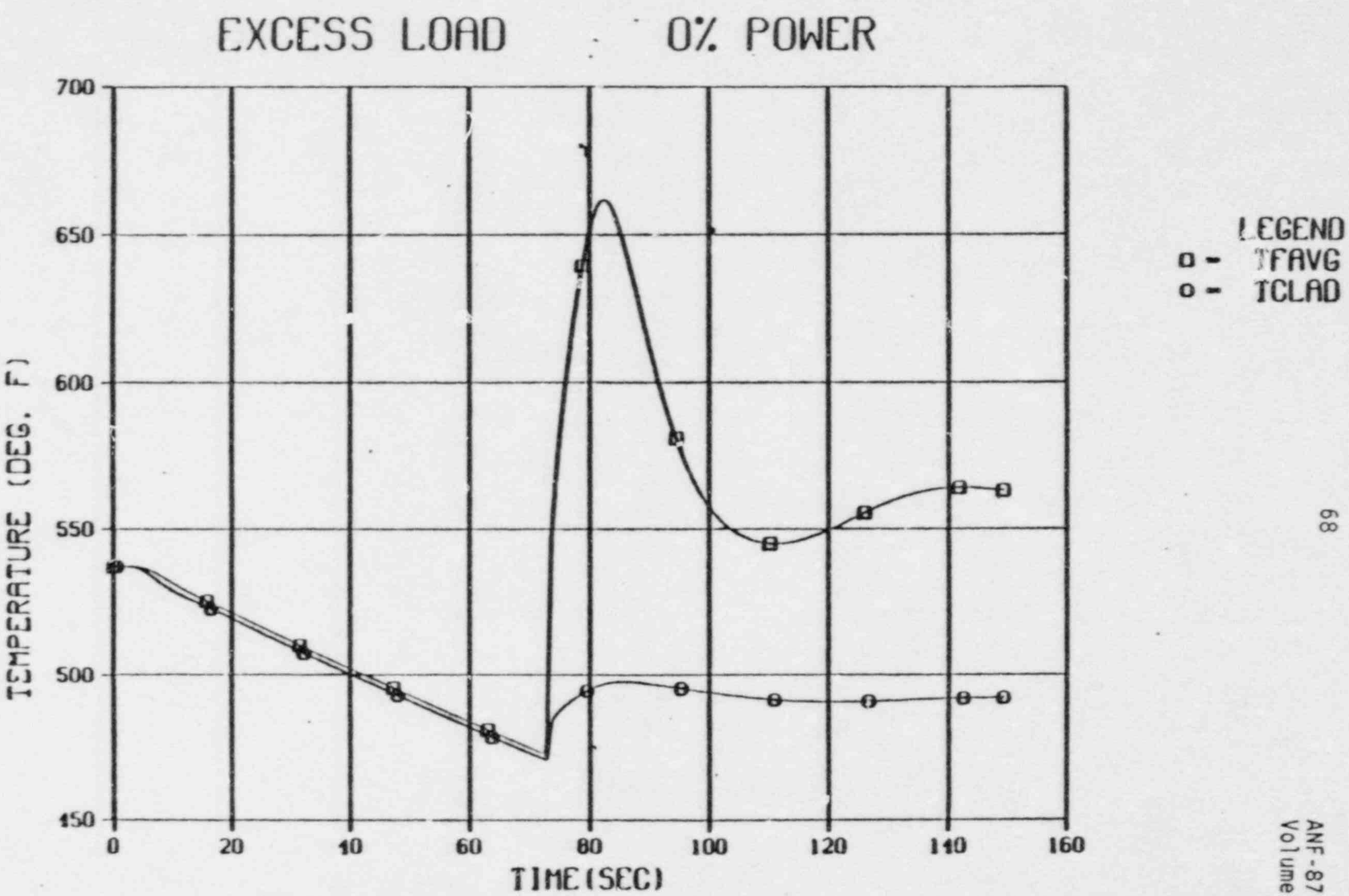


Figure 15.1.3-3 Average Fuel and Clad Temperatures for Excess Load from Zero Power

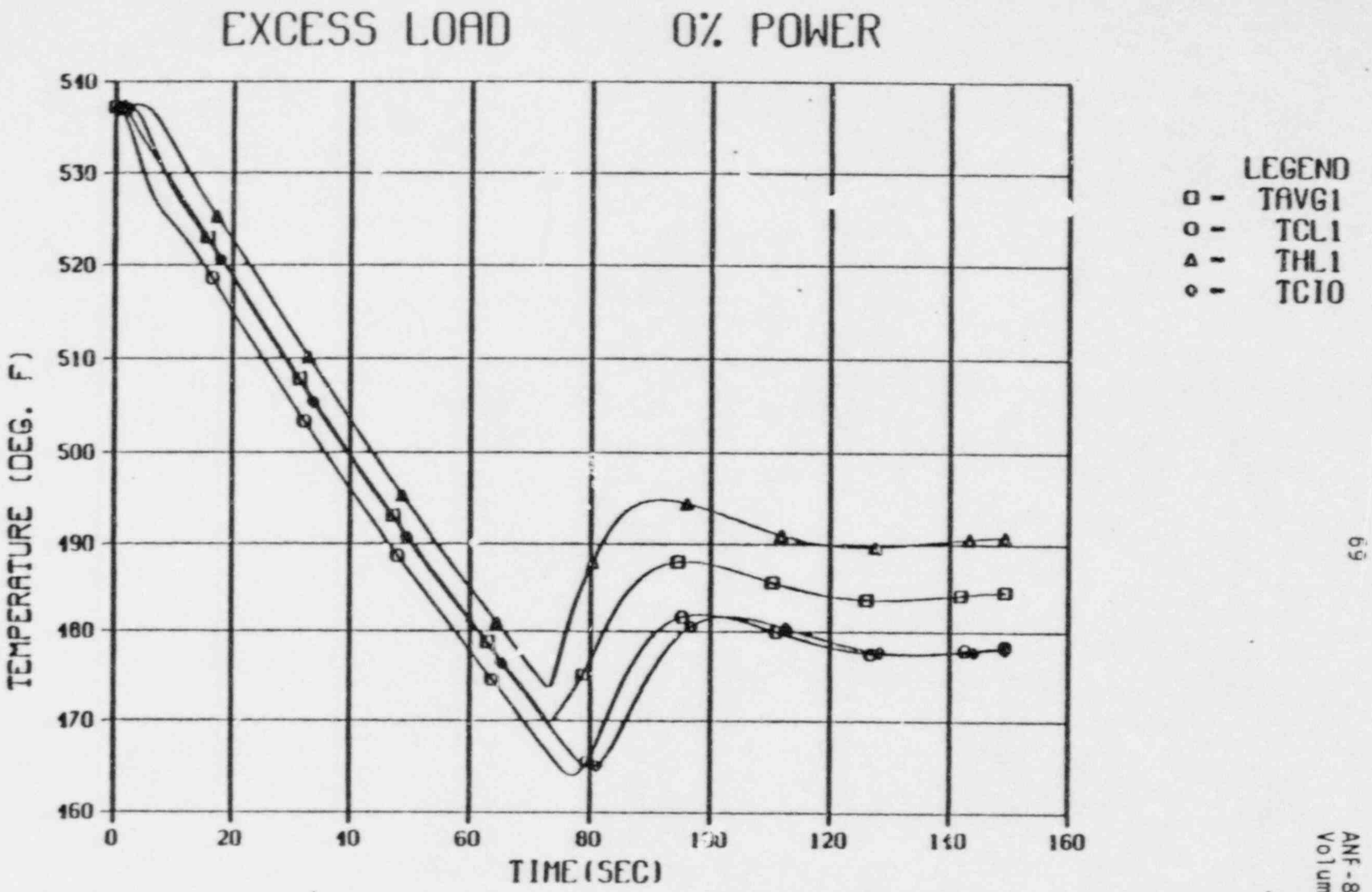


Figure 15.1.3-4 Reactor Coolant System Temperatures for Excess Load from Zero Power

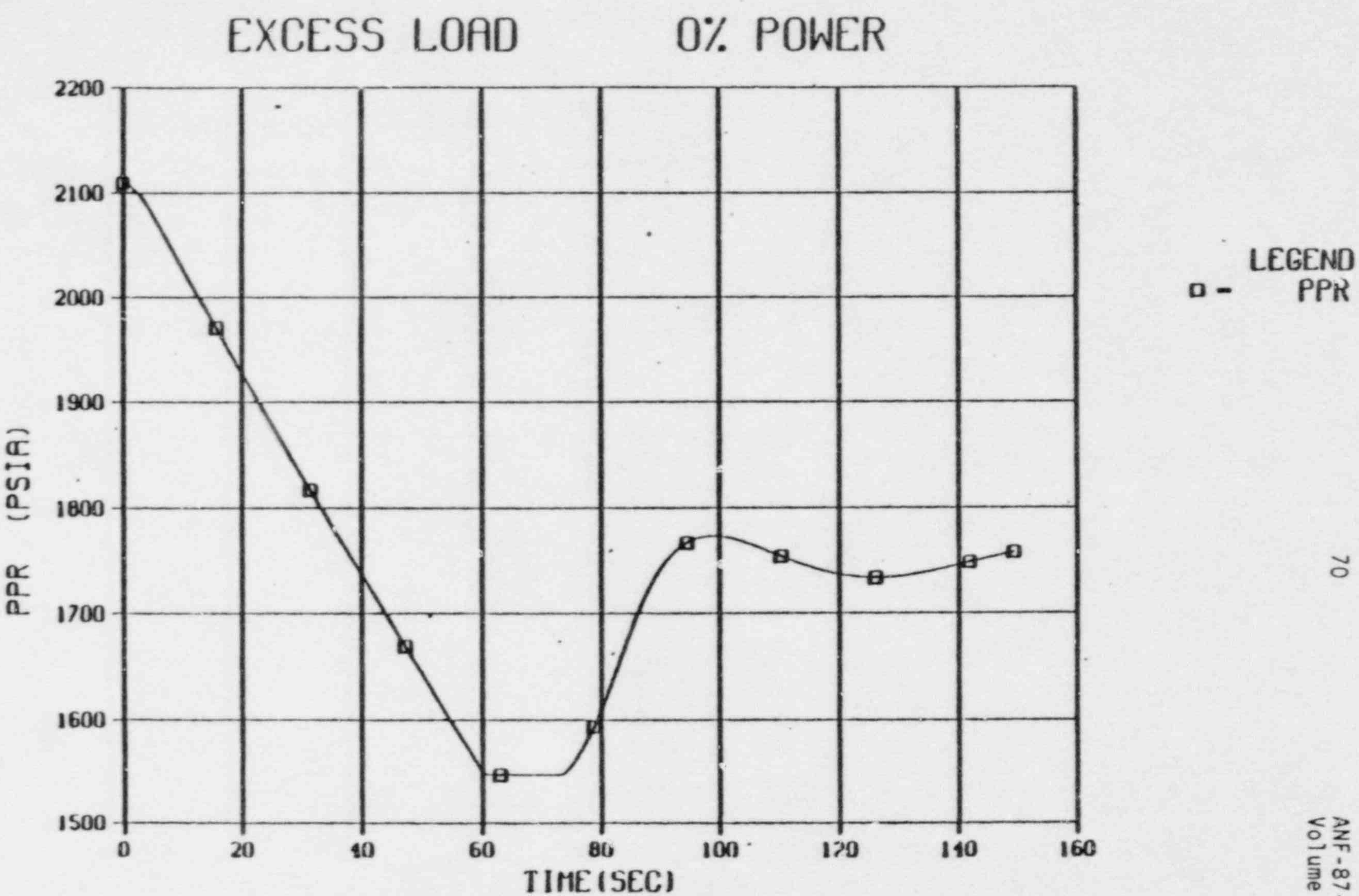


Figure 15.1.3-5 Pressurizer Pressure for Excess Load from Zero Power

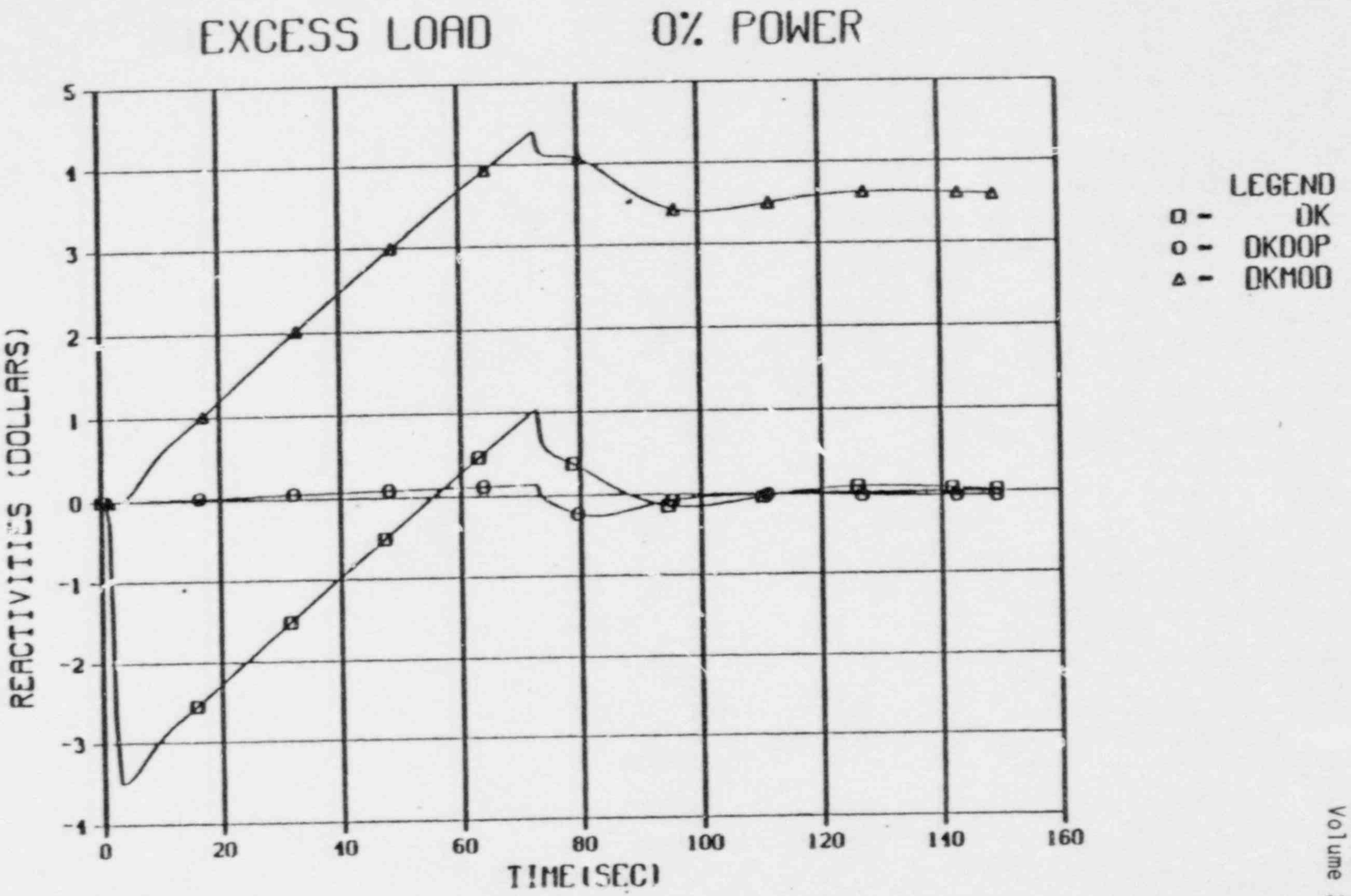


Figure 15.1.3-6 Reactivities for Excess Load from Zero Power

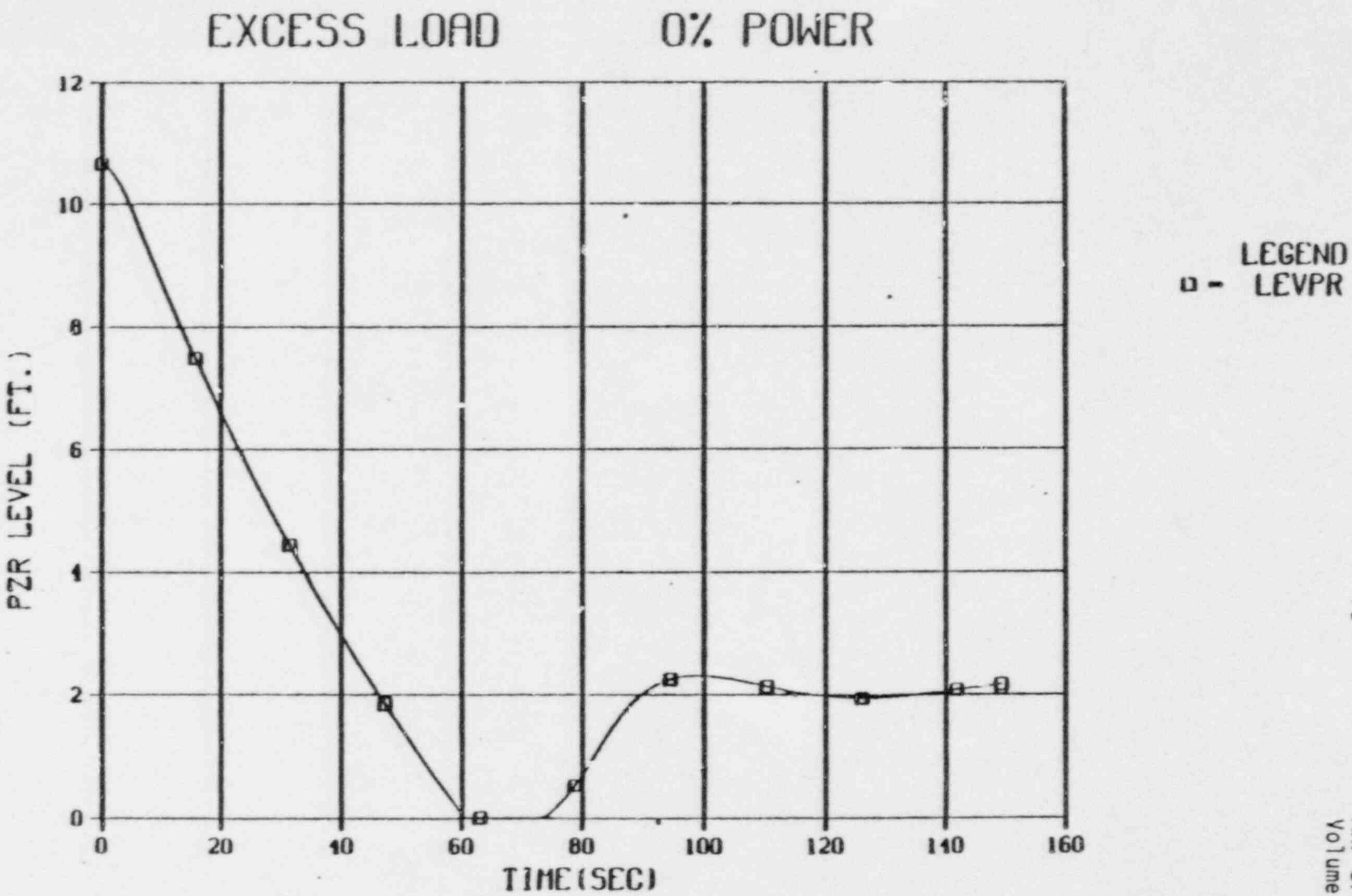


Figure 15.1.3-7 Pressurizer Level for Excess Load from Zero Power

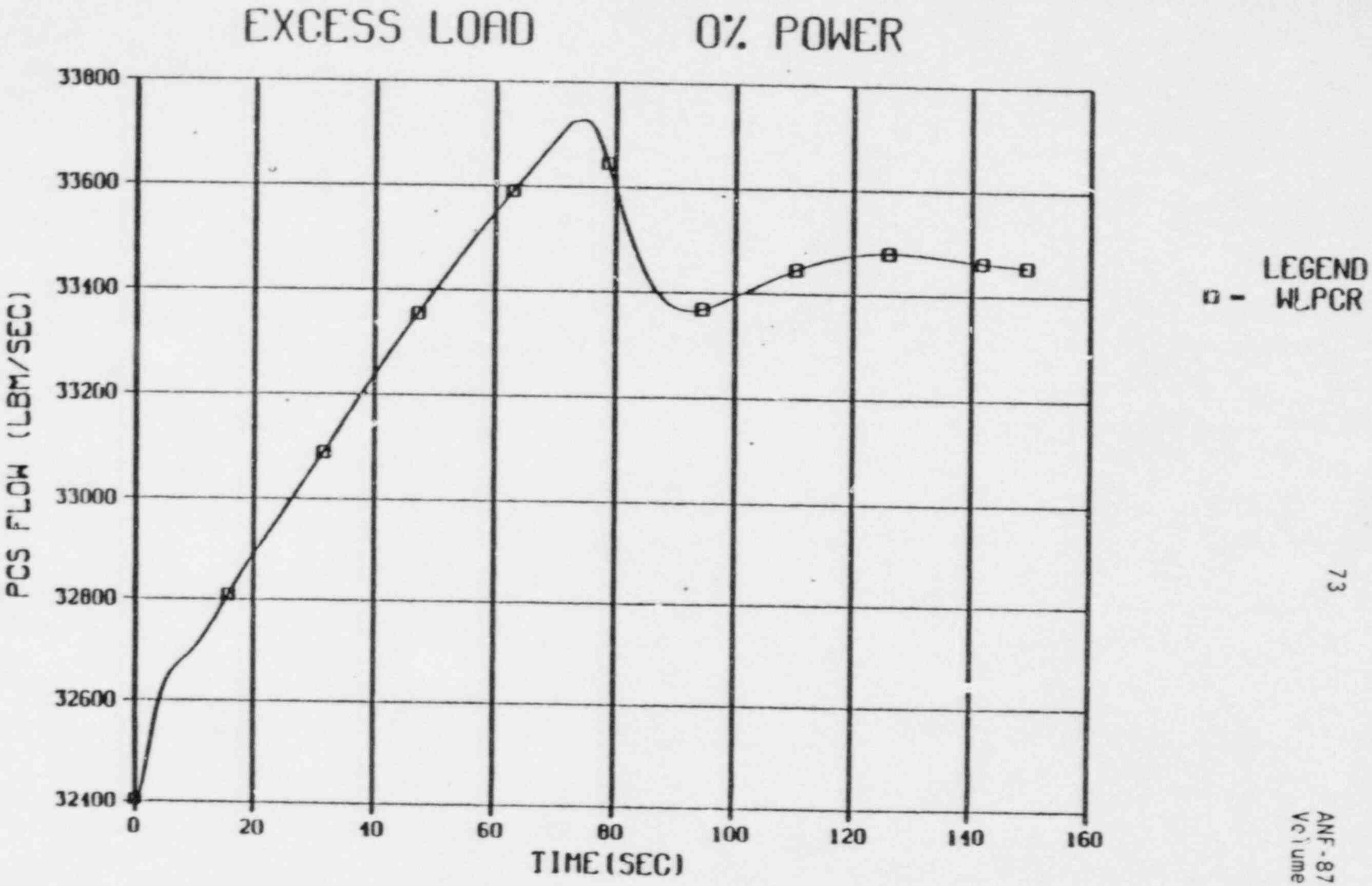


Figure 15.1.3-8 Primary Coolant Flow Rate for Excess Load from Zero Power

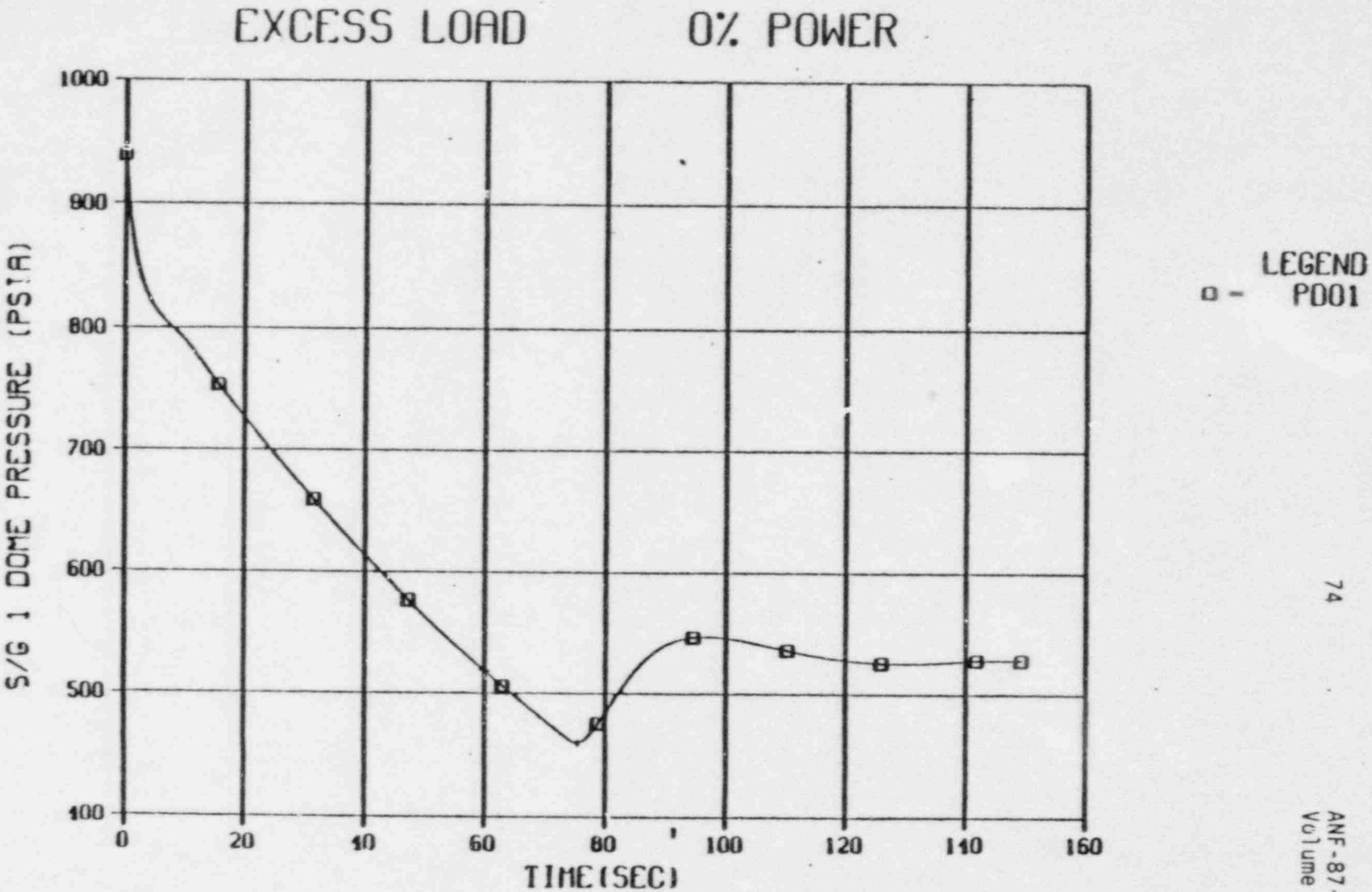


Figure 15.1.3-9 Secondary Pressure for Excess Load from Zero Power

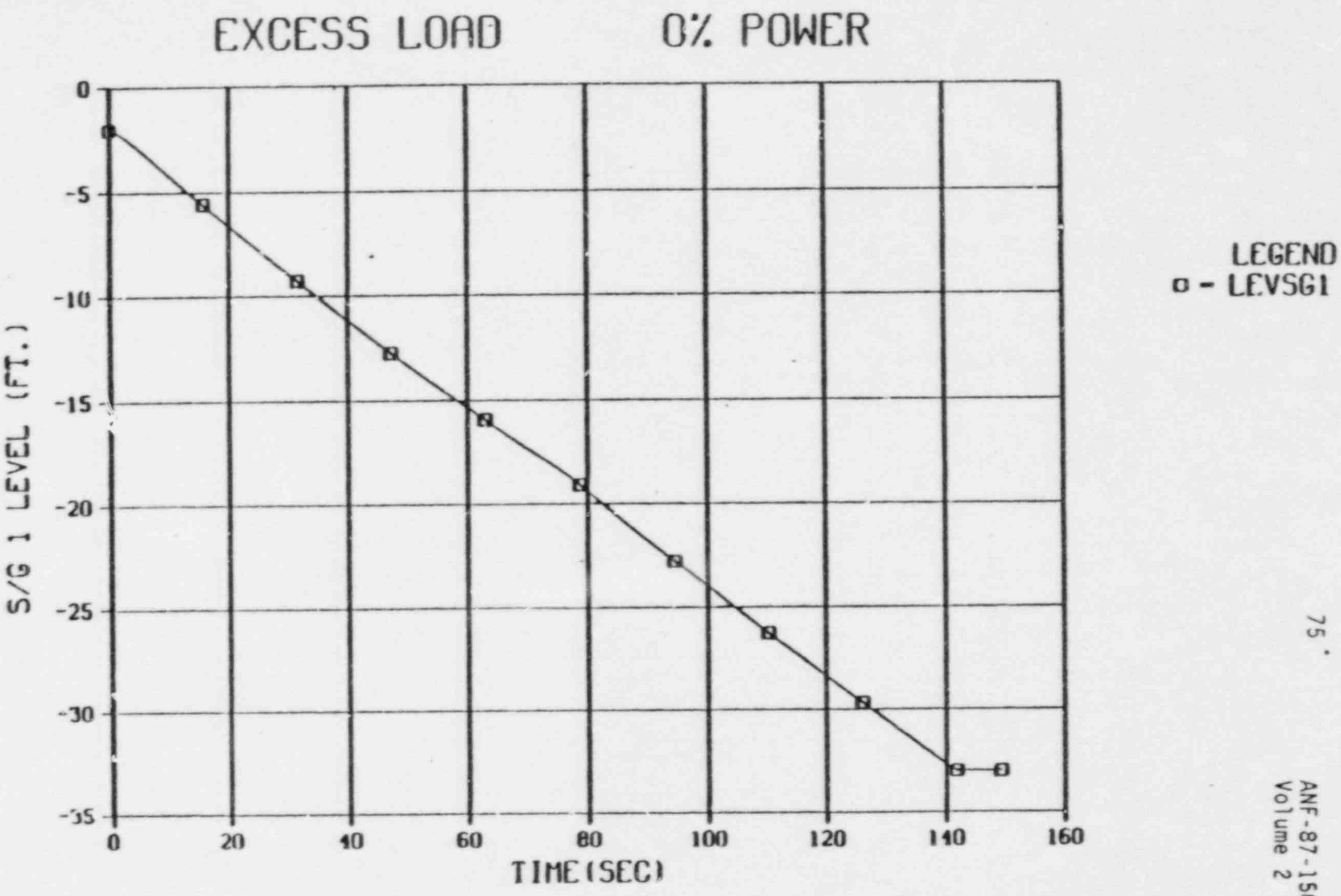


Figure 15.1.3-10 Steam Generator Liquid Level for Excess Load from Zero Power

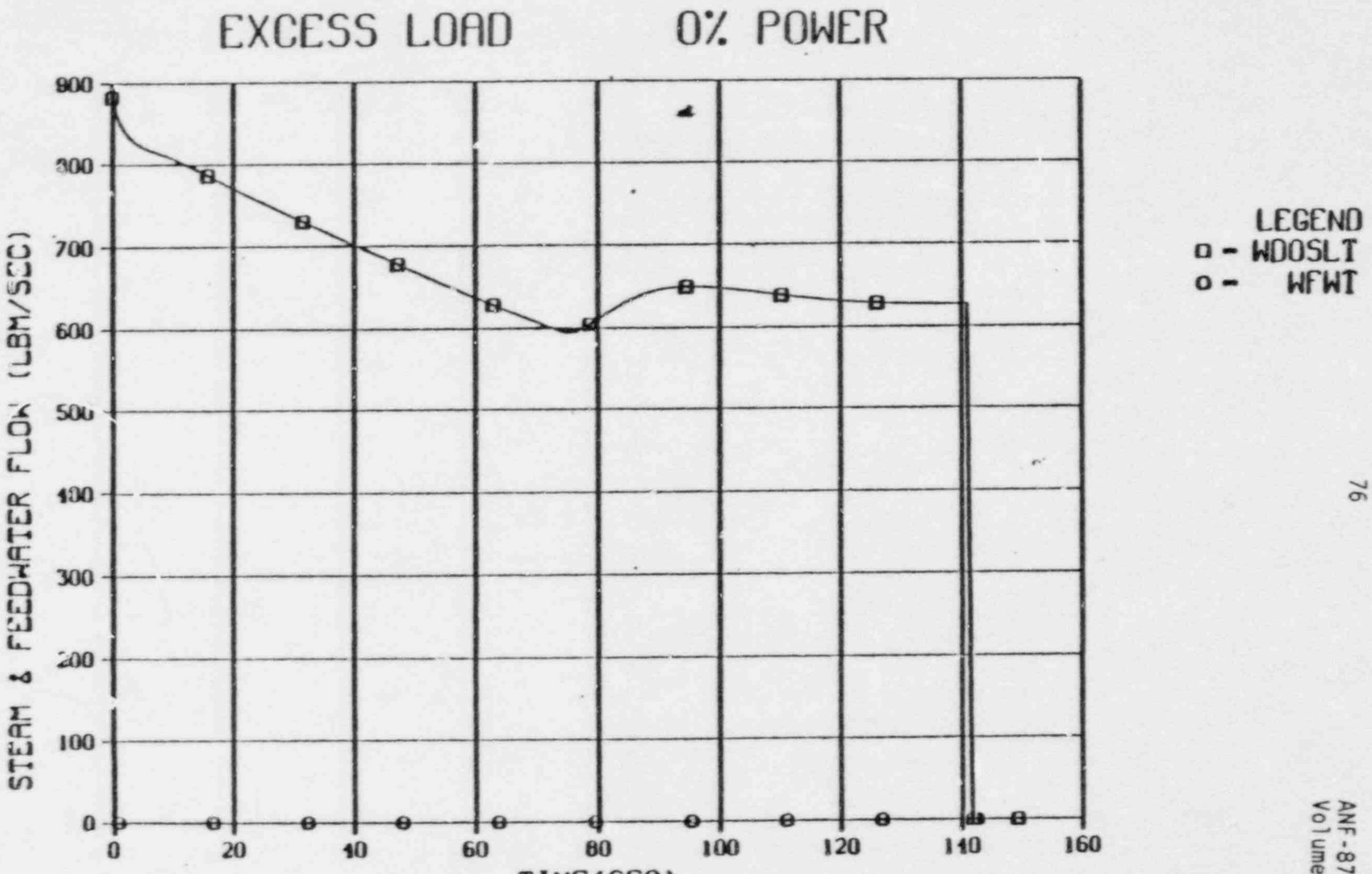


Figure 15.1.3-11 Secondary Steam and Feedwater Flow Rates for Excess Load from Zero Power

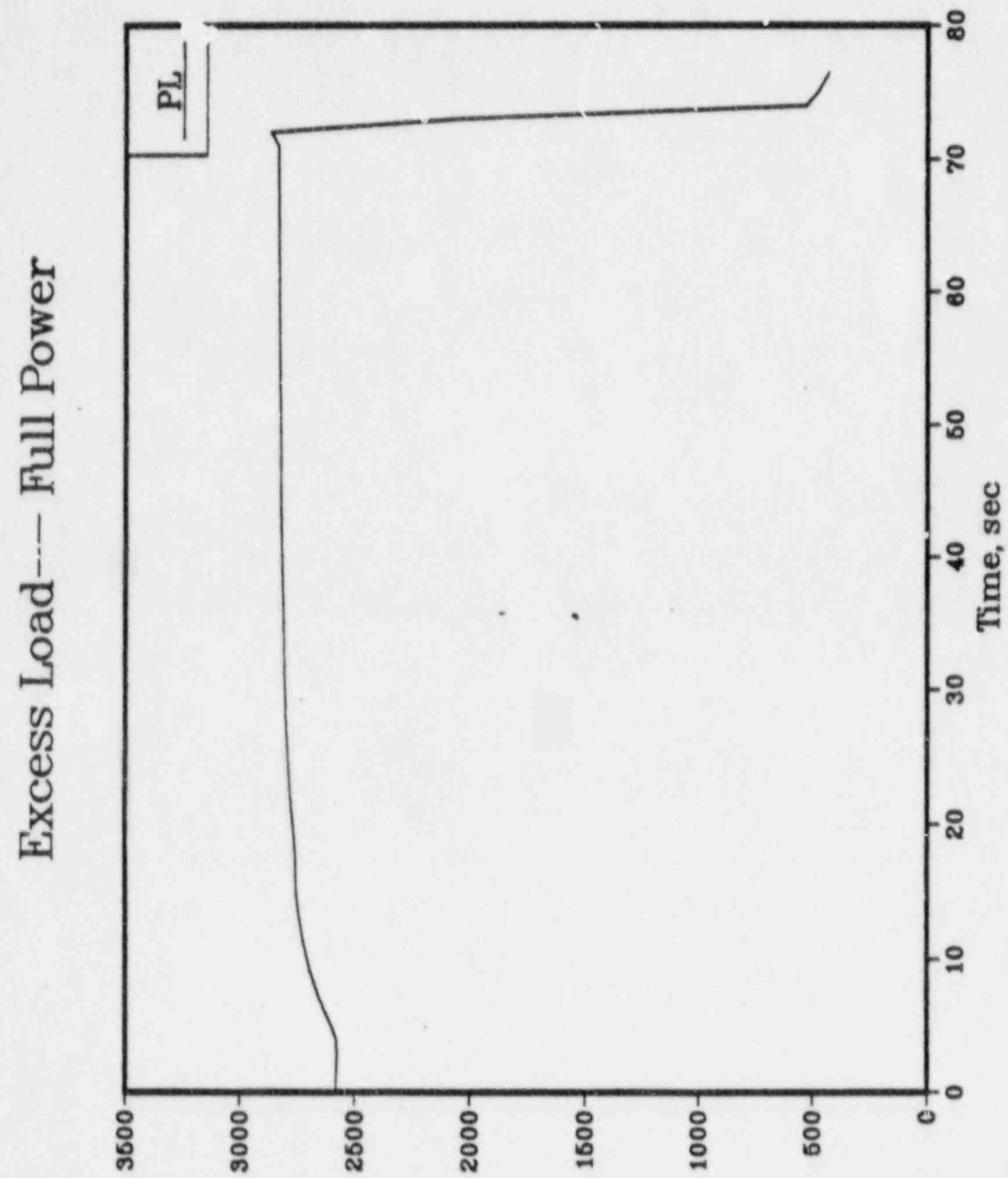


Figure 15.1.3-12 Reactor Power Level for Excess Load from Full Power

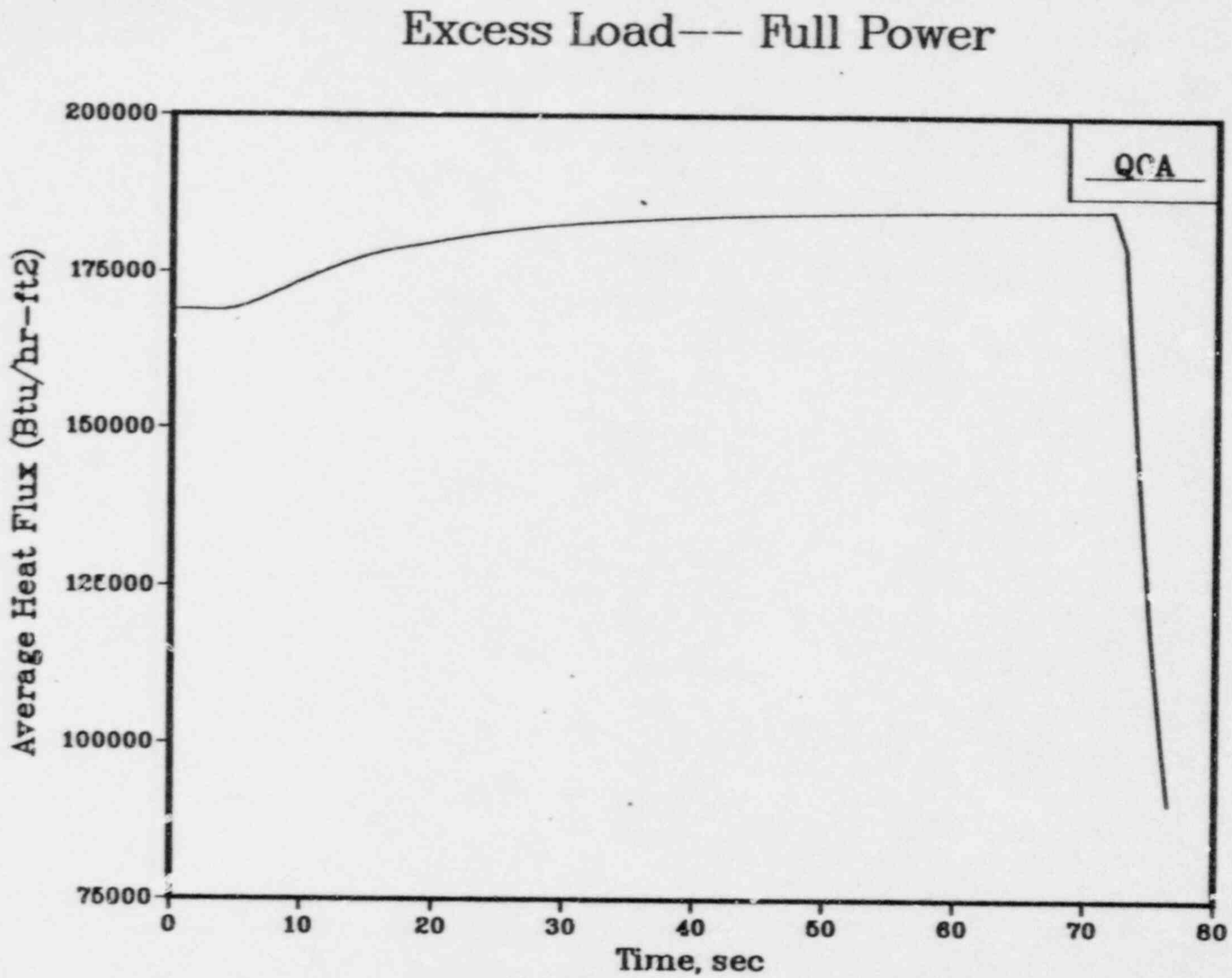


Figure 15.1.3-13 Core Average Heat Flux for Excess Load from Full Power

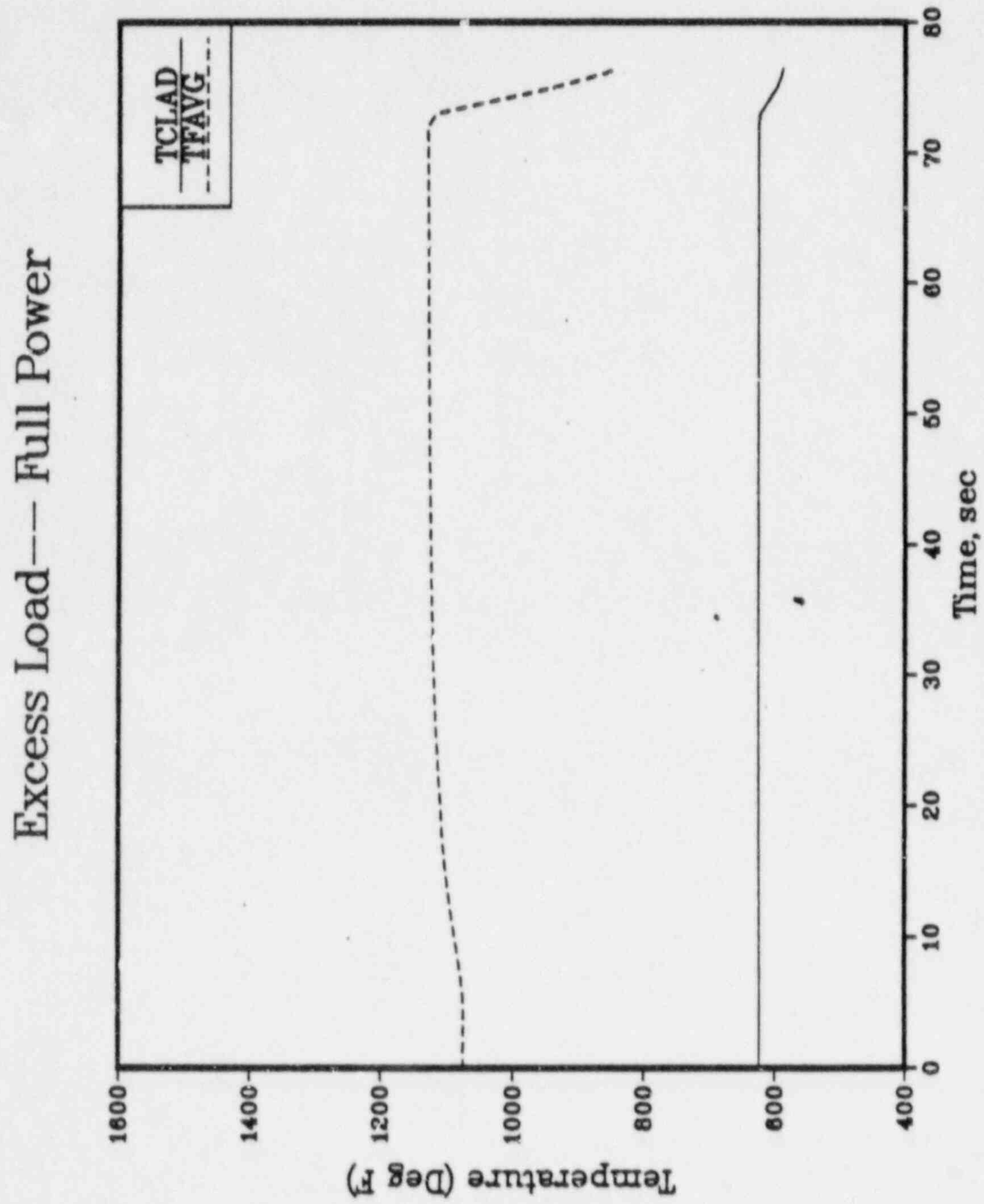


Figure 15.1.3-14 Average Fuel and Clad Temperature for Excess Load from Full Power

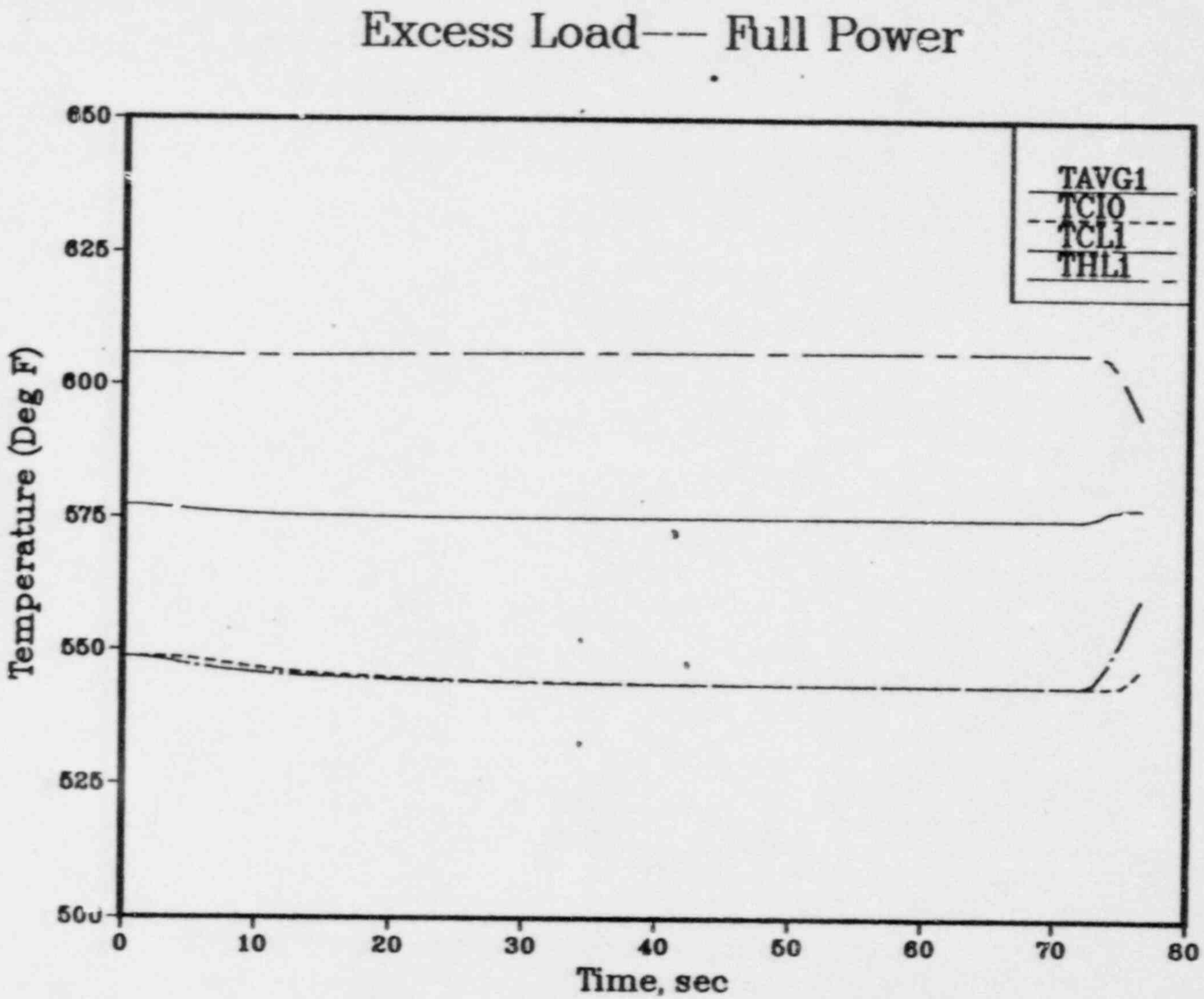


Figure 15.1.3-15 Reactor Coolant Temperatures for Excess Load from Full Power

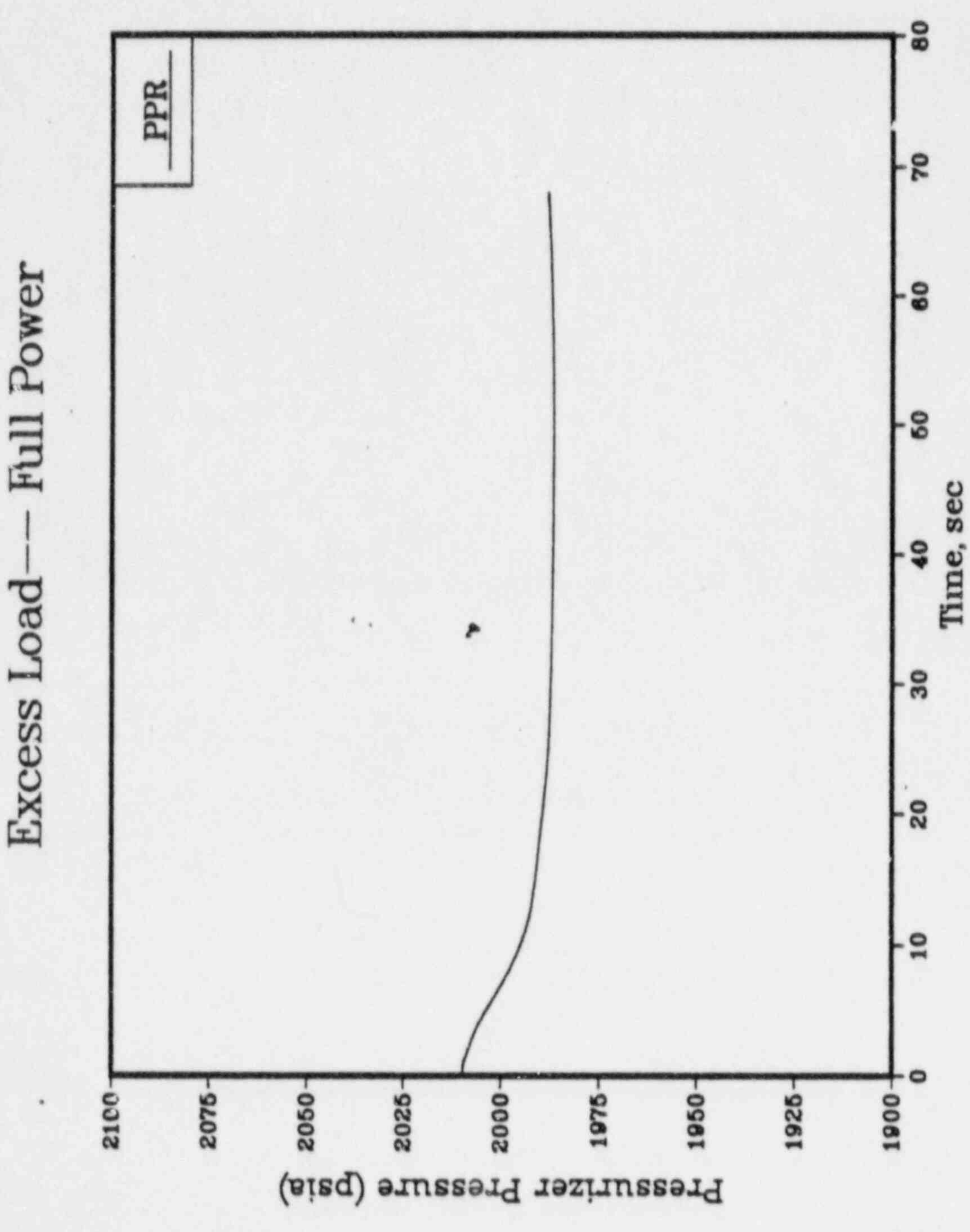


Figure 15.1.3-16 Pressurizer Pressure for Excess Load from Full Power

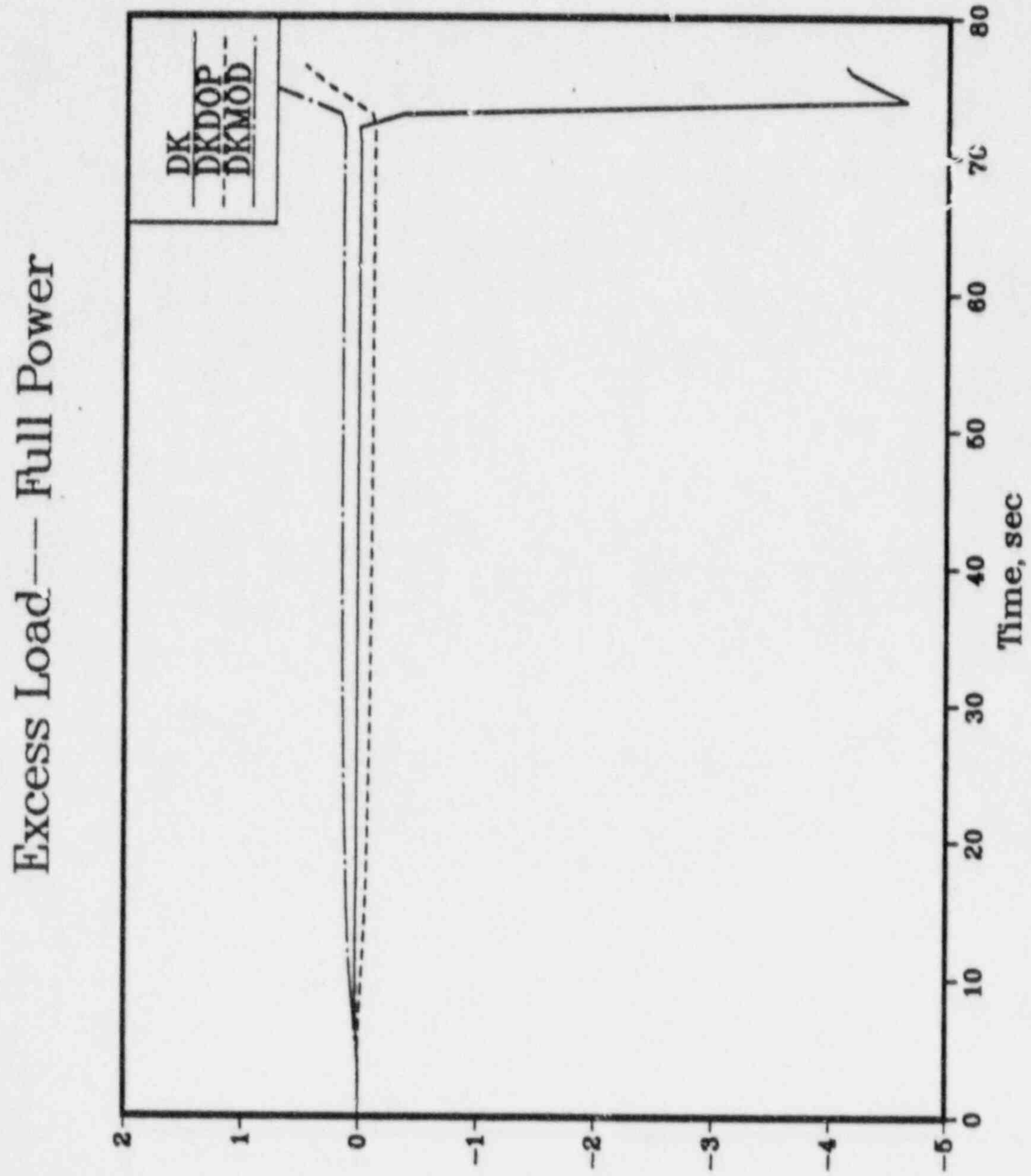


Figure 15.1.3-17 Reactivities for Excess Load from Full Power

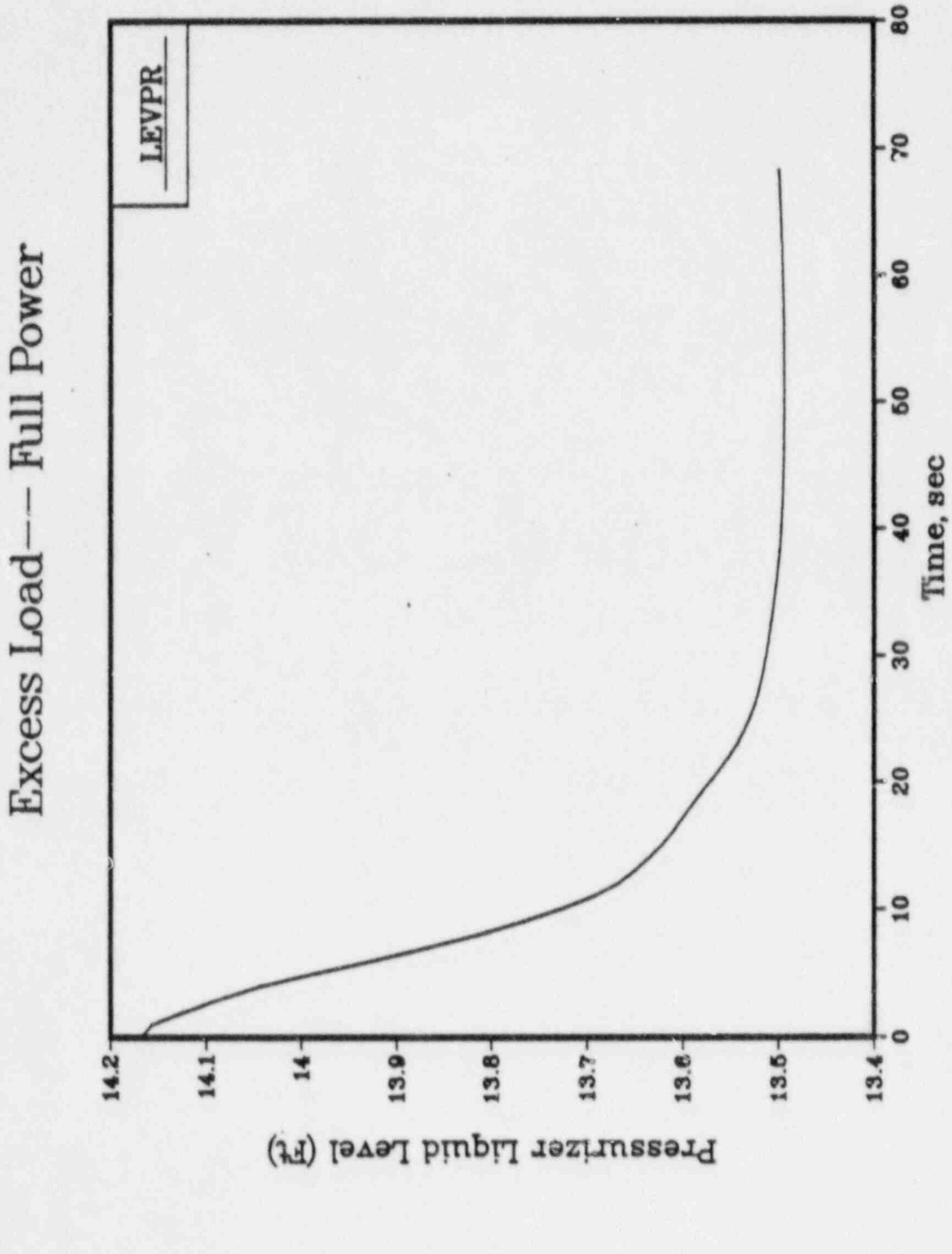


Figure 15.1.3-18 Pressurizer Liquid Level for Excess Load from Full Power

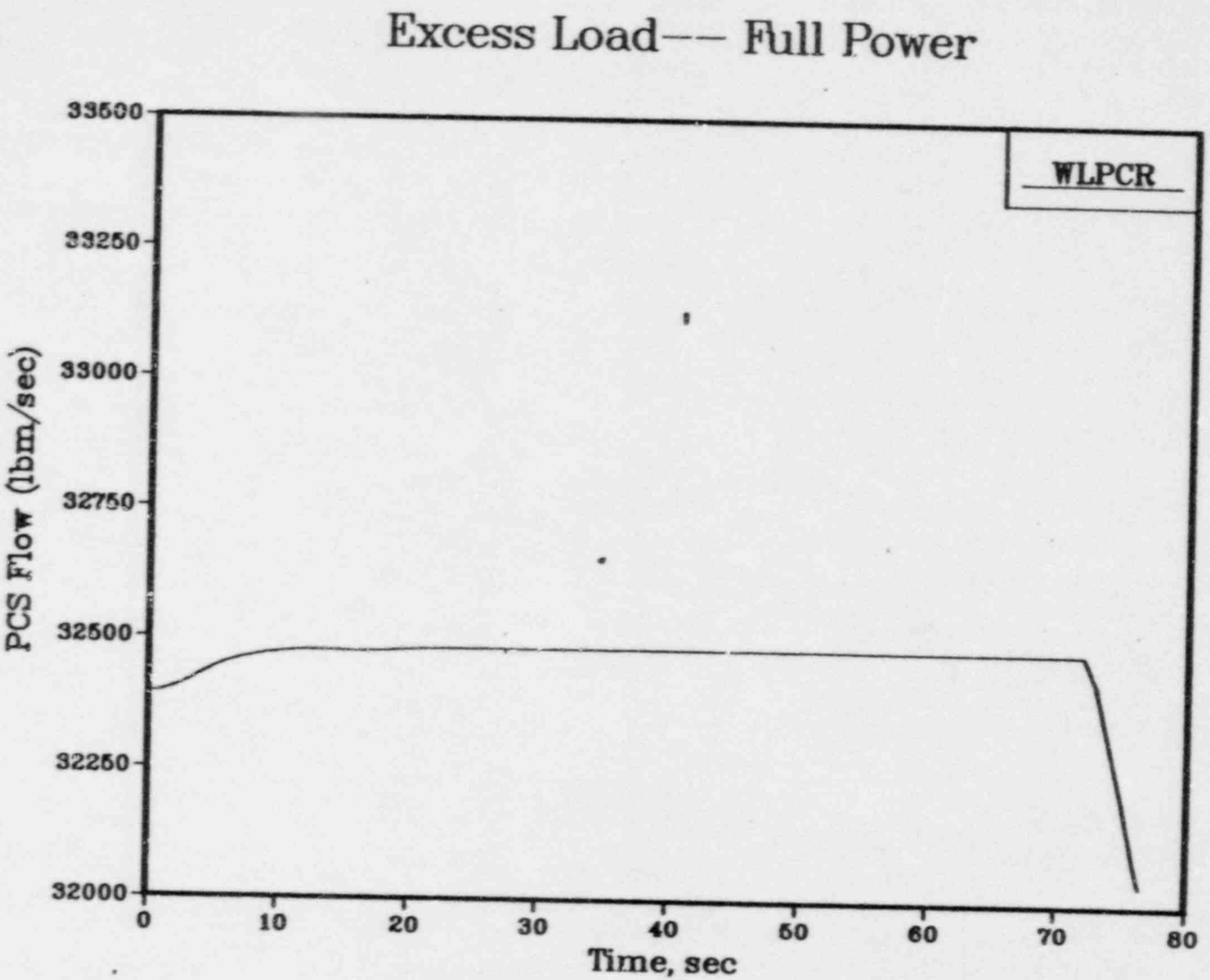


Figure 15.1.3-19 Primary Coolant Flow Rate for Excess Load from Full Power

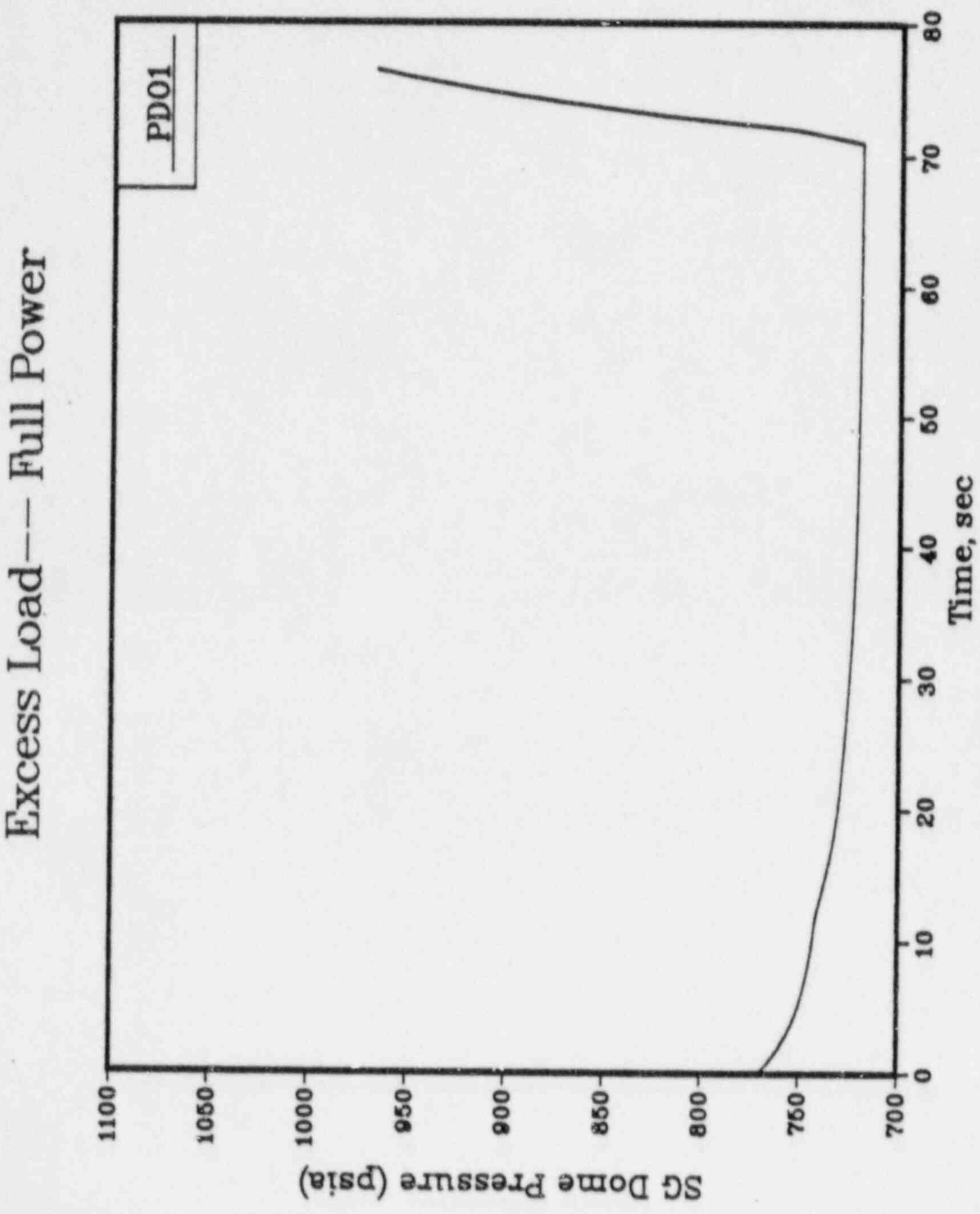


Figure 15.1.3-20 Secondary Pressure for Excess Load from Full Power

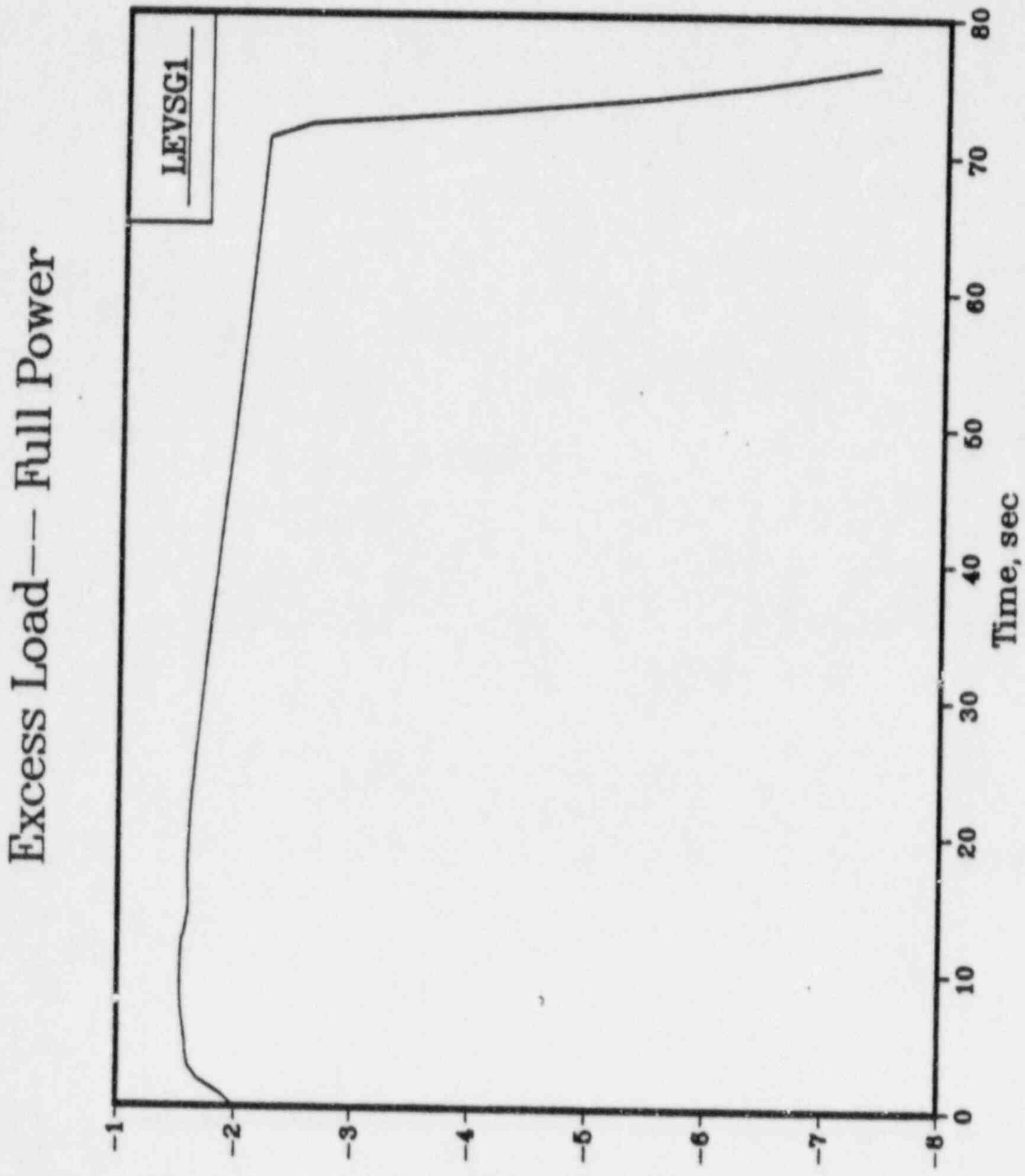


Figure 15.1.3-21 Steam Generator Liquid Level for Excess Load from Full Power

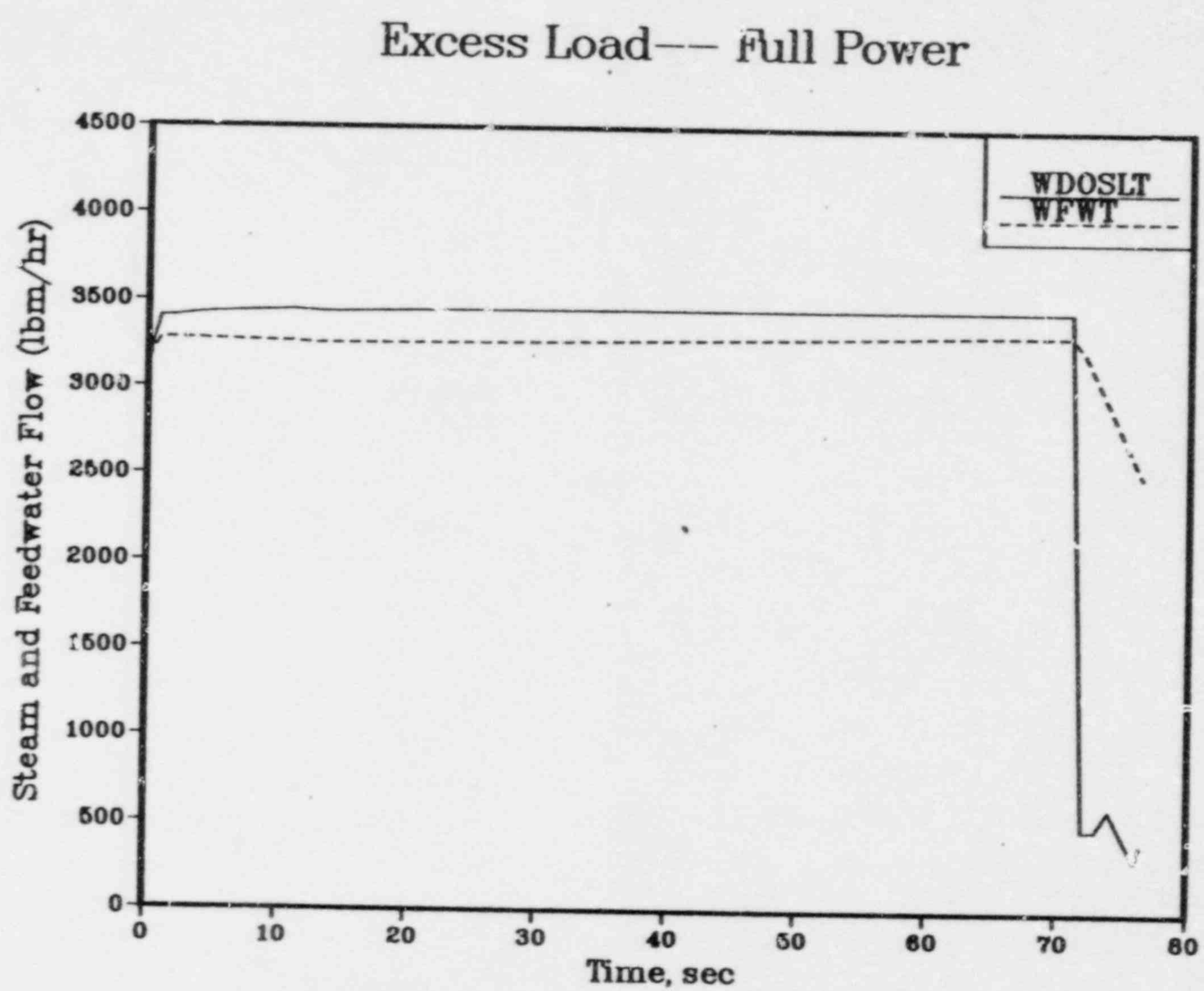


Figure 15.1.3-22 Secondary Steam and Feedwater Flow Rates for Excess Load from Full Power

15.2 DECREASED HEAT REMOVAL BY THE SECONDARY SYSTEM

15.2.1 LOSS OF EXTERNAL LOAD

15.2.1.1 Identification of Causes and Event Description

A major load loss on the generator can result from the loss of external load due to an electrical system disturbance. Offsite electrical power is available to operate the reactor coolant system pumps and other station auxiliaries. Following the loss of generator load, the turbine stop valve closes, terminating the steam flow and causing the secondary system temperature and pressure to increase. The primary-to-secondary heat transfer decreases as the secondary system temperature increases.

If the reactor is not tripped when the turbine is tripped, the primary system temperature continues to rise. The primary liquid expands and the pressurizer steam space is compressed, causing the pressurizer pressure to rise. If this continues, the reactor will trip on high pressurizer pressure, reducing the primary heat source. As the heat load into the primary system decreases, the primary system pressure will also begin to decrease.

The pressure increase on the primary side is mitigated by the pressurizer safety valves. The pressure increase on the secondary side is mitigated by the steamline relief valves, the atmospheric dump valves and the turbine bypass to the condenser.

This event challenges two of the acceptance criteria. It challenges limits on: (1) Primary pressure; and, (2) MDNBR SAFDL due to the increasing core inlet temperature and the potential for an increase in reactor power prior to scram. Automatic Reactor Control is disabled at Palisades, so the reactor power is not reduced with the increase in average primary system temperature.

This event is a moderate frequency event (Table 15.0.1-1). The acceptance criteria for this event are listed in 15.0.1.1. Single failure criteria for Palisades are given in 15.0.11. For this analysis, the systems challenged in this event are redundant; no single active failure in the RPS or ESF will adversely affect the consequences of the event.

15.2.1.2 Analysis Method

This event is analyzed with the PTSPWR2 computer program[10]. The core thermal hydraulic boundary conditions from the PTSPWR2 calculation are used as input to the XCOBRA-IIIC methodology[11] to predict the minimum DNBR for the event.

15.2.1.3 Definition of Events Analyzed and Bounding Input

The objectives in analyzing this event are to: (1) Demonstrate that the primary pressure relief capacity is sufficient to limit the pressure to less than 110% (2750 psia) of the design pressure; (2) Evaluate the maximum primary-to-secondary side pressure differential; and, (3) Demonstrate that the MDNBR remains above the XNB correlations safety limit of 1.17.

Three cases are analyzed for this event. They correspond to each of the above objectives. Case specific biases on the input parameters are selected to maximize the probability for exceeding the appropriate operating limit. The bounding operating mode for this event is full power initial conditions.

Conditions used in the analysis of each case are as shown in Table 15.2.1-0:

Table 15.2.1-0 Conservative Assumptions Used in the Loss of External Load Event

	Maximum Primary Pressure	Maximum Pressure Difference	Minimum DNBR
Case	1	2	3
Rod Control	Manual	Manual	Manual
Power	Nom. +2%	Nom. +2%	Nom. +2%
Pressure	Nom. -50 psi	Nom. +50 psi	Nom. -50 psi
Core Inlet Temperature	Nom. +5°F	Nom. +5°F	Nom. +5°F
Primary Flow Rate	Nom. -3%	Nom. -3%	Nom. -3%
Kinetics	BOC	BOC	BOC
Pressurizer Spray	Disabled	Disabled	Full on
Pressurizer Heaters	Full on	Full on	Disabled
Pressurizer Safety Valve Setpoint	Nom. +1%	Nom. +1%	Nom. -1%
Steam Bypass (and ADVs)	Disabled	Available	Disabled
Secondary Relief Valves	*	*	*
Scram On Turbine Trip	Disabled	Disabled	Disabled
Low S/G Level Trip	Disabled	Disabled	Disabled
CVCS (Max makeup or letdown flow)	Makeup	Makeup	Letdown

* Supports a 3% tolerance on the secondary side safety/relief valve setpoint.

15.2.1.4 Analysis of Results

The maximum pressurization case (Case 1) initiates with a ramp closure of the turbine control valve in 0.1 seconds. Steam line pressure increases until the relief valves open at 7.26 seconds. The maximum pressure in the steam dome of the steam generators of 1036.6 psi is achieved at 9.93 seconds. The maximum required steam line relief valve flow capacity to control the secondary-side pressure is about 3.8 Mlbm/hr. The pressurization of the secondary side results in decreased primary to secondary heat transfer, and a substantial rise in primary system temperature.

A primary coolant temperature increase of about 12.2°F has occurred by 8.75 seconds. This results in a large insurge into the pressurizer, compressing the steam space and pressurizing the primary system. The reactor trips on high pressure with rods beginning to insert at 5.07 seconds, and the pressurizer safety valves open at 6.60 seconds. The capacity of two valves** is enough to contain the pressure transient within the vessel pressure criterion of 2750 psia. The increase in coolant temperature also causes the core power to rise to about 103.1% due to positive moderator feedback. The transient is terminated shortly after reactor scram due to decreasing primary coolant temperature and pressure.

The maximum tube pressure difference case (Case 2) also initiates on the closure of the turbine control valve. The capacity of the atmospheric dump valves and turbine bypass valves is used to relieve and minimize the secondary pressure. This capacity was sufficient to prevent the steam line relief valve opening since the pressure never reached the setpoint. Steam dome pressure rose to a maximum of 921.4 psia at 15.0 seconds.

The lower secondary pressure than in Case 1 allows greater heat transfer to the primary system. As a result, the increase in the average coolant temperature is smaller which leads to a lower peak primary pressure. The pressurizer safety valves are not actuated. The reactor scrammed at 4.67 on the high pressurizer pressure signal.

The minimum DNBR case is initiated in the same manner. The transient proceeds in a similar fashion except that the primary side pressure is limited by the pressurizer safety valve setpoint. The setting of the pressurizer safety valve setpoints were biased downward to 2475 psia with a flow capacity corresponding to all three valves. This maximizes the challenge to DNBR

**Although the plant has three safety valves, only two were modeled in the PTSPWR2 calculation.

pressurizer safety valve setpoint. The setting of the pressurizer safety valve setpoints were biased downward to 2475 psia with a flow capacity corresponding to all three valves. This maximizes the challenge to the DNBR limit. The safety valves open at 6.95 seconds. The peak pressure is 2477 psia at 7.02 seconds.

The increased secondary pressure causes a core average temperature rise of 9.4°F at 7.91 seconds. The DNBR challenge results from the core power and primary coolant temperature increase. The challenge is further exacerbated by the limitation on pressure rise.

All transients were analyzed from 102% initial power. Pressurizer pressure control system parameters were appropriately biased to either maximize or minimize pressure. A conservatively small value for the turbine control valve closing time was used. Plant initial operating conditions assumed in the analyses are summarized in Table 15.2.1-1.

An event summary for each case is shown in Table 15.2.1-2. The transient response to the maximum pressurization case is shown in Figures 15.2.1-1 to 15.2.1-9. The maximum reactor coolant system boundary pressure computed for this case is 2584.7 psia. This is below the 110% of design limit criterion of 2750 psia. The results for the maximum tube pressure difference case are plotted in Figures 15.2.1-10 to 15.2.1-20. The maximum pressure difference of 1604.4 psi occurred at 7.5 seconds. The response to the minimum DNBR case is given in Figures 15.2.1-21 to 15.2.1-30. The minimum DNBR is computed to be 1.776. This is above the DNBR limit of 1.17 for the XNB correlation.

Plotted variables are defined in Table 15.0.12-1.

15.2.1.5 Conclusion

The maximum pressure is less than the acceptance limit of 110% of design pressure, or 2750 psia. Calculated MDNBR for the event is above the XNB critical heat flux correlation safety limit, so the DNB SAFDL is not penetrated in this event. Peak pellet LHGR for the event is about 12.7 kW/ft, well below the fuel centerline melt criterion of 21 kW/ft. Applicable acceptance criteria for the event are therefore met.

Table 15.2.1-1 Summary of Initial Conditions for the Loss of External Load Event

	Maximum Pressure Case	Maximum Pressure Differential Case	Minimum DNBR Case
Power (MWt)	2580.6	2580.6	2580.6
Core Inlet	548.65	548.65	548.65
Pressurizer Pressure (psia)	2010	2110	2010
Reactor Coolant System Flow Rate (Mlbm/hr)	116.7	116.7	116.7
Steam Dome Pressure (psia)	730.	730.	730.

Table 15.2.1-2 Event Summary for the Loss of External Load Event

Maximum Pressurization Event Summary (Case 1)

Event	Time (sec)
Turbine Trip	0.00
Reactor Scram (Begin Rod Insertion) on High Pressurizer Pressure	5.07
Peak Power	5.62
Pressurizer Safety Valves Open	6.60
Peak Core Average Temperature	7.01
Steam Line Safety Valves Open	7.26
Peak Pressure	7.32
Peak Steam Dome Pressure	9.93

Maximum Pressure Differential Event Summary (Case 2)

Event	Time (sec)
Turbine Trip	0.00
ADV and turbine bypass valves open (on 3 sec ramp)	0.00
Reactor Scram (Begin Rod Insertion) on High Pressurizer Pressure	4.67
Peak Power	5.22
Peak Core Average Temperature	6.42
Peak S/G Tube ΔP	7.50
Peak Pressure	7.59
Peak Steam Dome Pressure	14.99

Minimum DNBR Event Summary (Case 3)

Event	Time (sec)
Turbine Trip	0.00
Minimum DNBR	0.53
Reactor Scram (Begin Rod Insertion)	6.02
Steam Line Relief Valves Open	6.53
Peak Power	6.57
Pressurizer Safety Valves Open	6.95
Peak Core Average Temperature	7.91

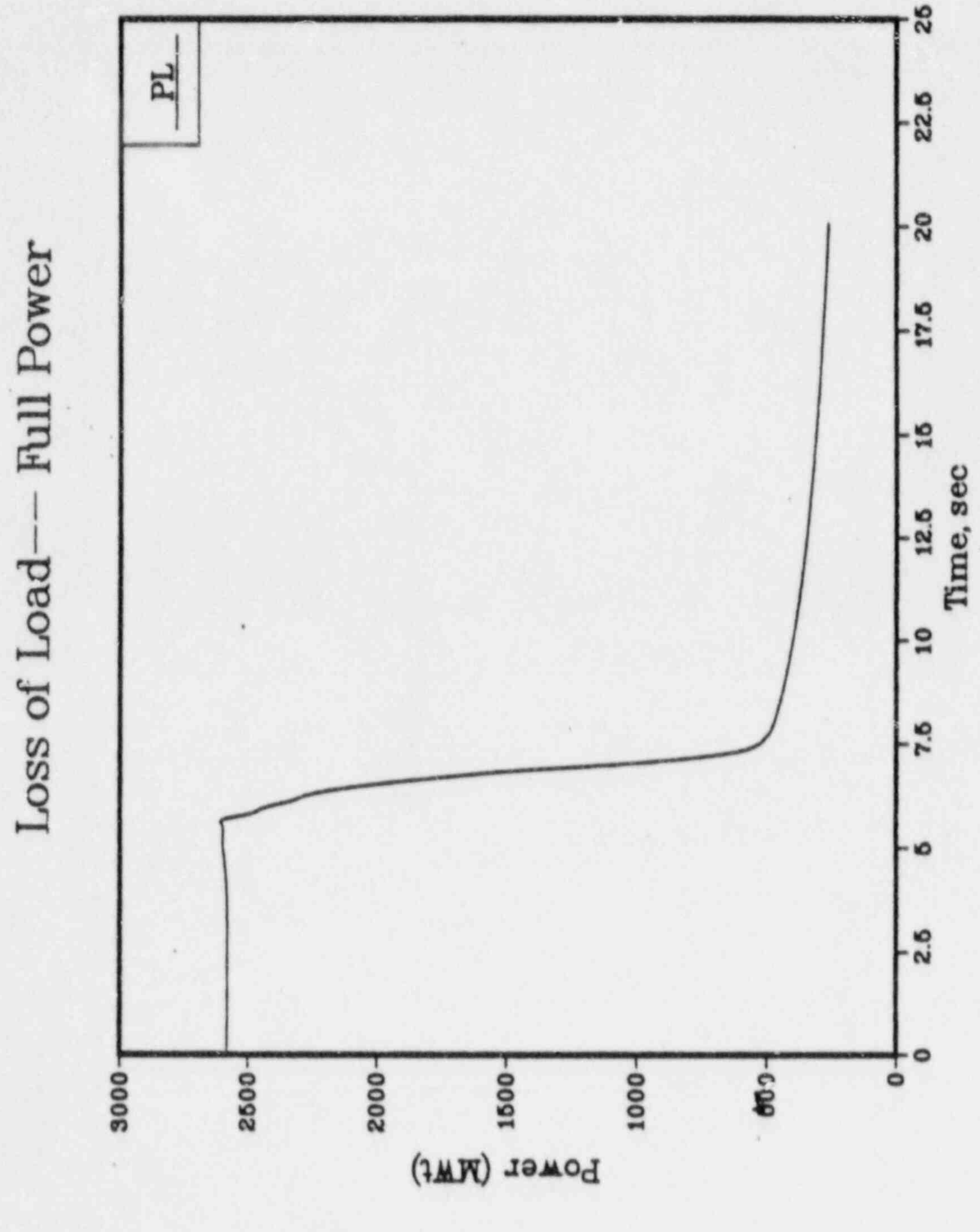


Figure 15.2.1-1 Reactor Power Level for Loss of External Load (Pressurization Case)

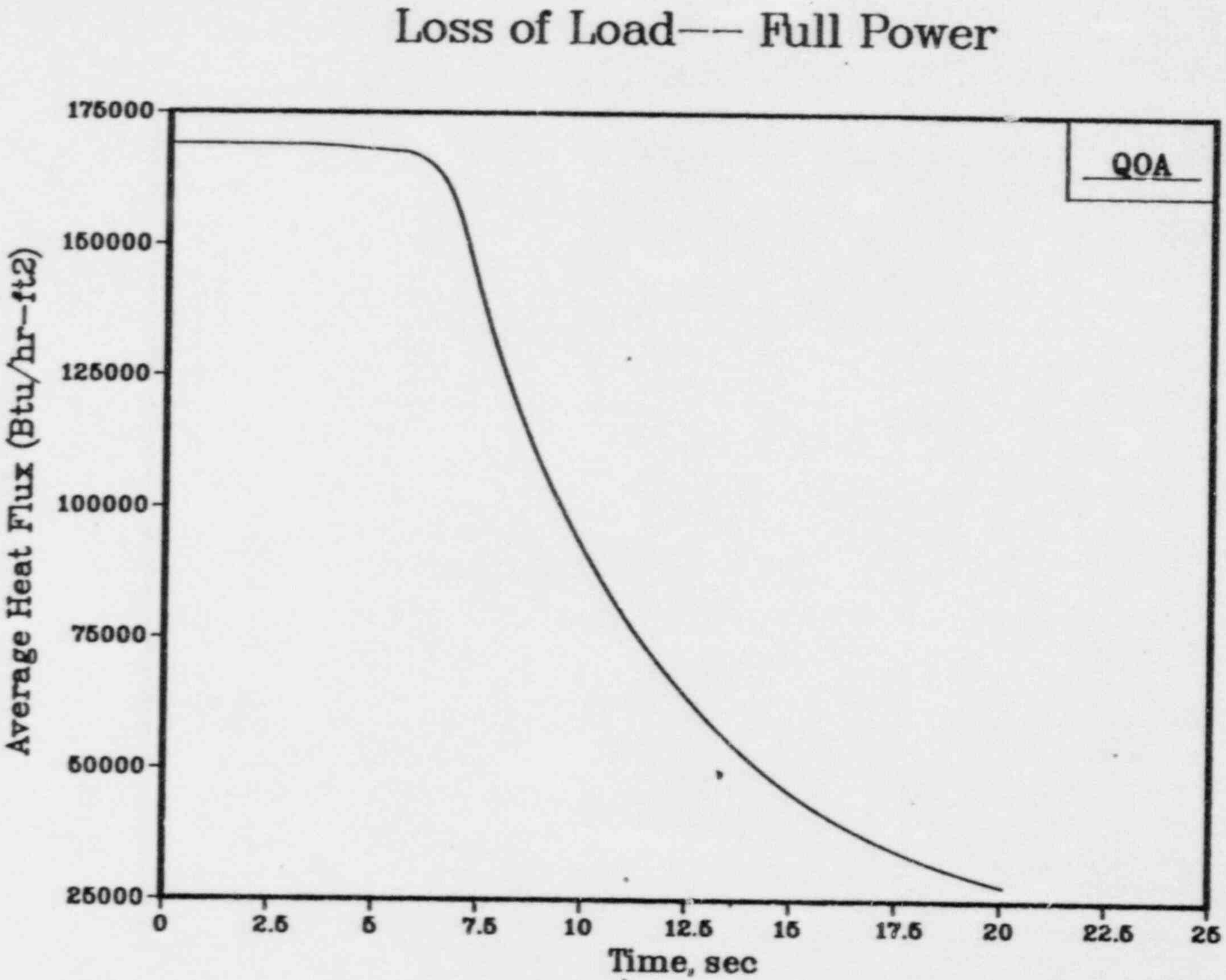


Figure 15.2.1-2 Core Average Heat Flux for Loss of External Load (Pressurization Case)

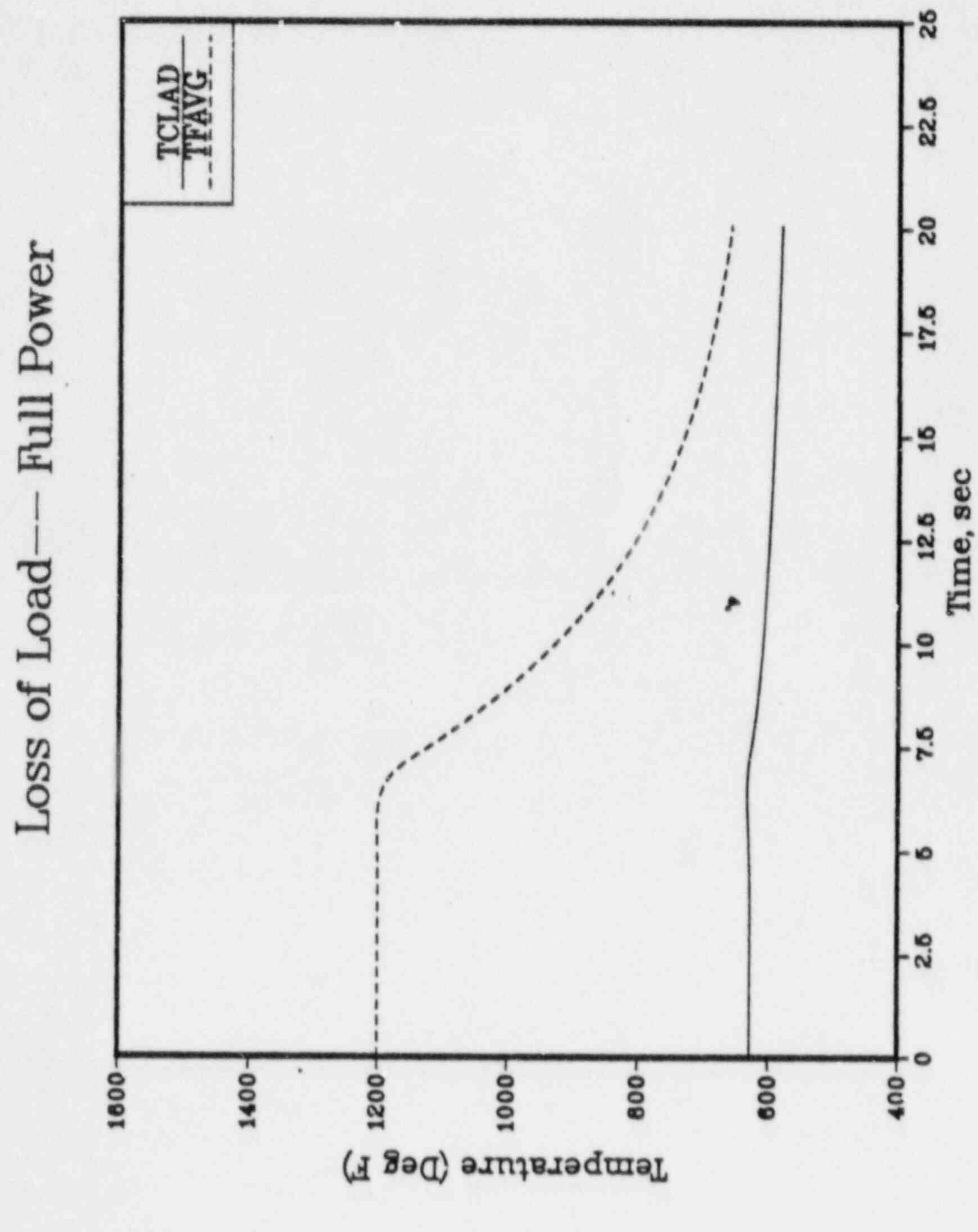


Figure 15.2.1-3 Average Fuel and Clad Temperatures for Loss of Load (Pressurization Case)

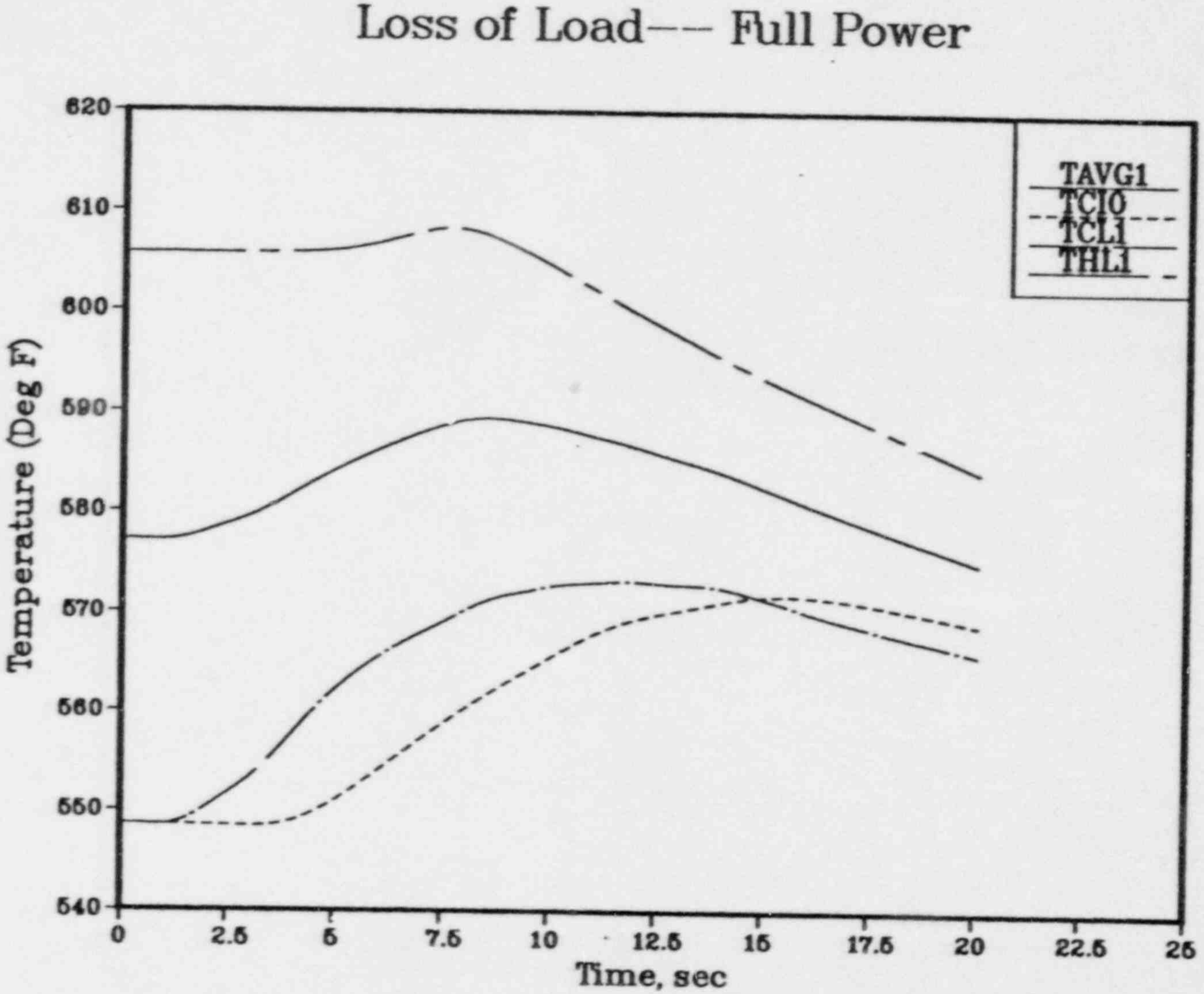


Figure 15.2.1-4 Reactor Coolant System Temperatures for Loss of External Load (Pressurization Case)

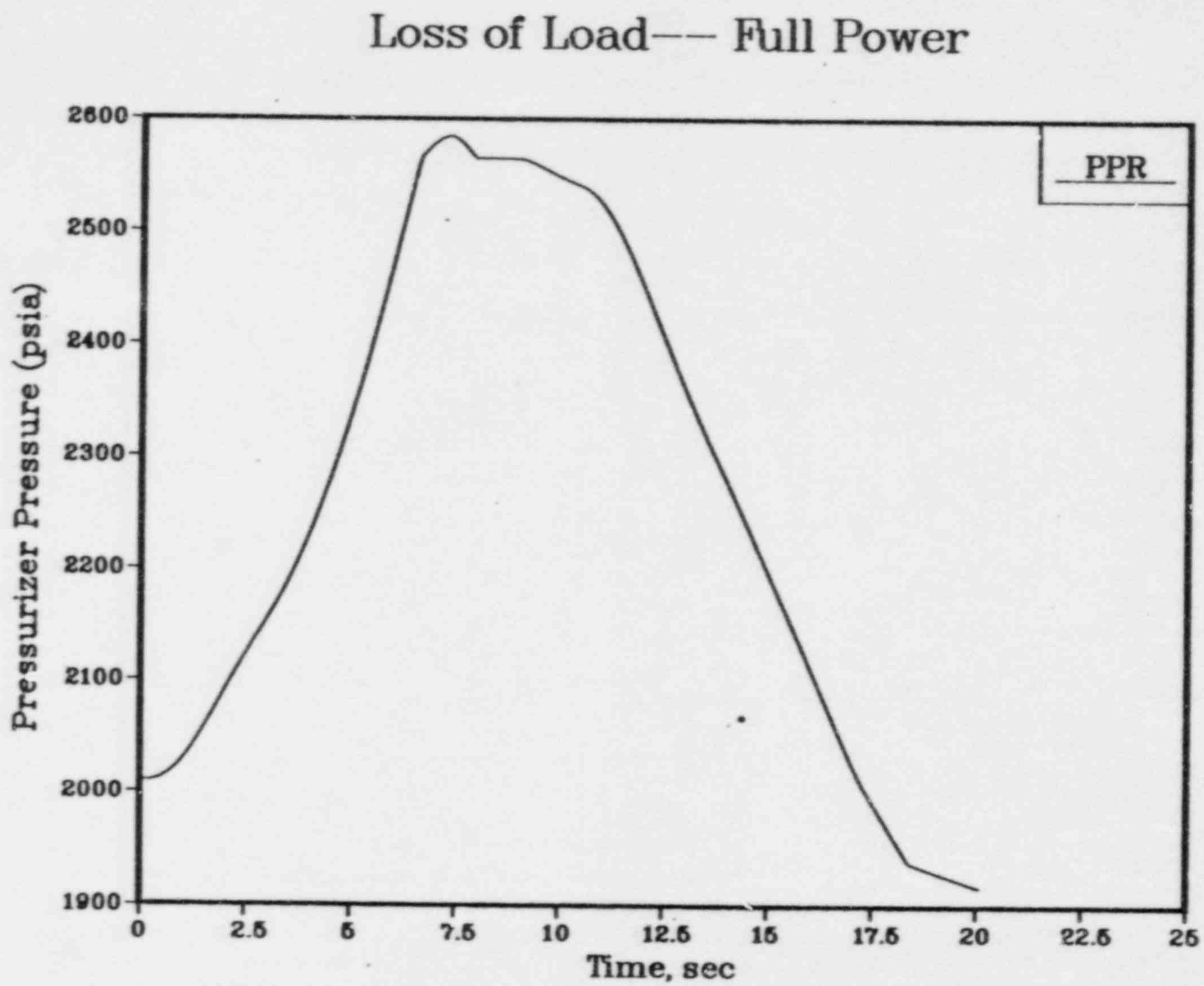


Figure 15.2.1-5 Pressurizer Pressure for Loss of External Load (Pressurization Case)

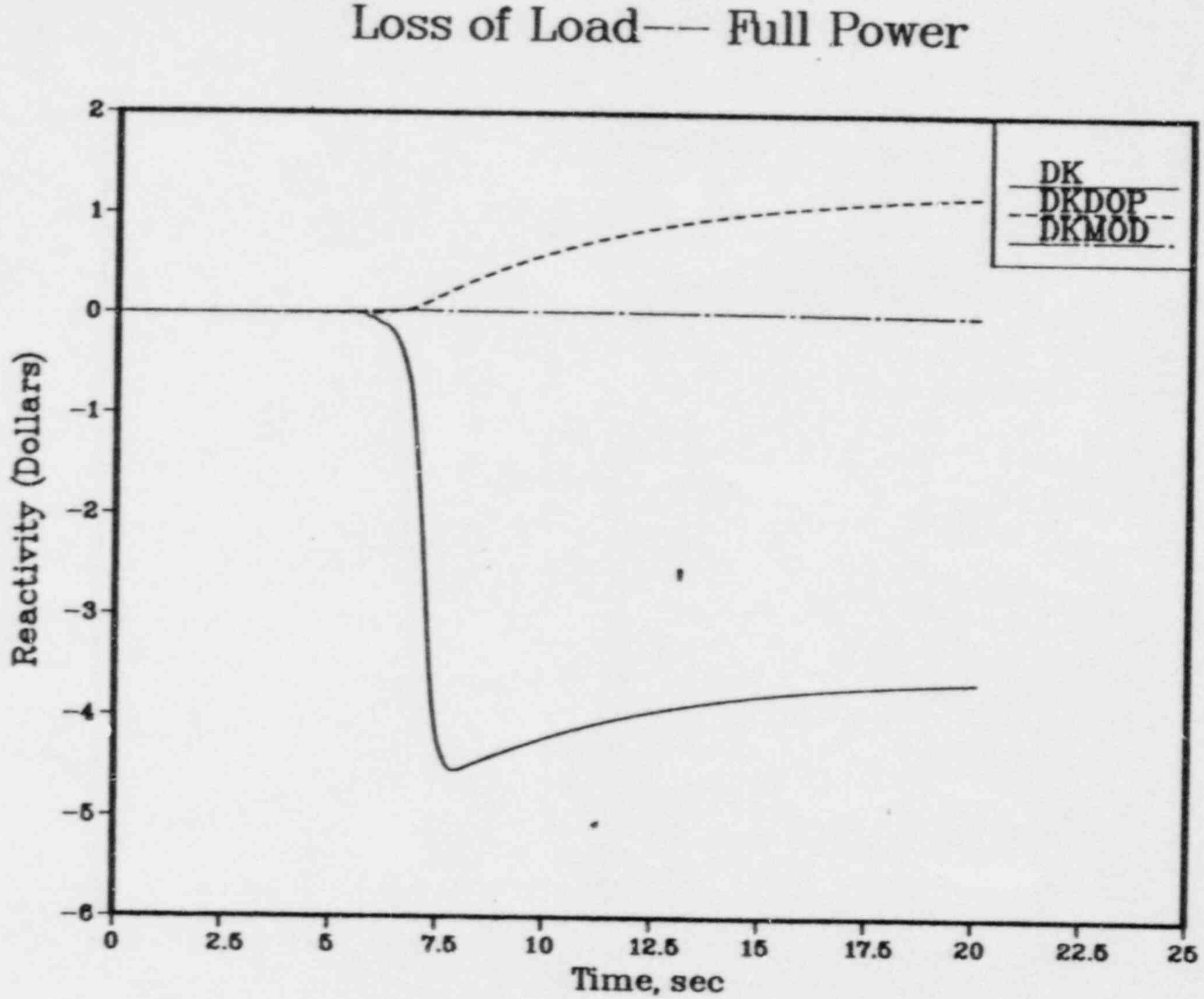


Figure 15.2.1-6 Reactivities for Loss of External Load (Pressurization Case)

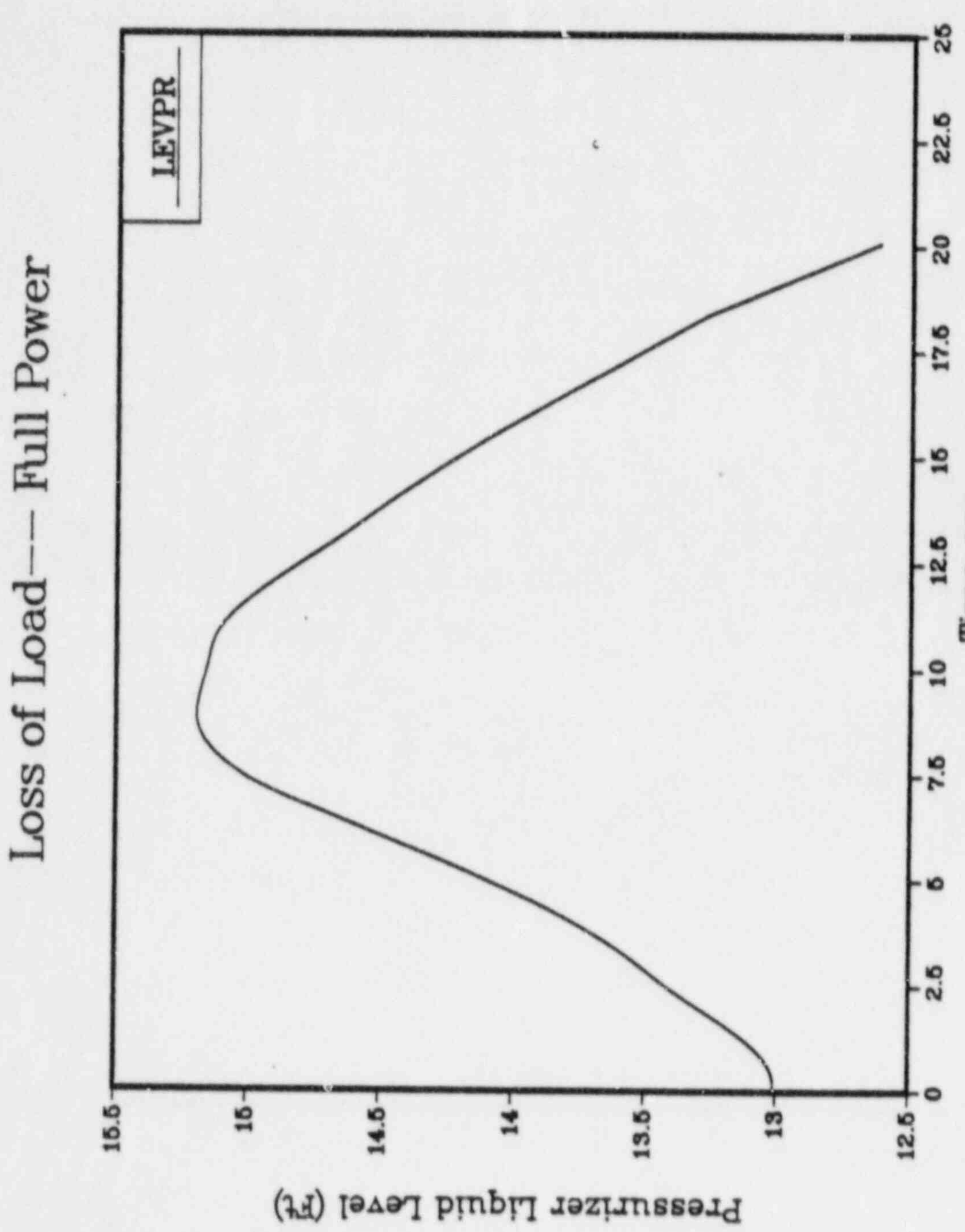


Figure 15.2.1-7 Pressurizer Liquid Level for Loss of External Load (Pressurization Case)

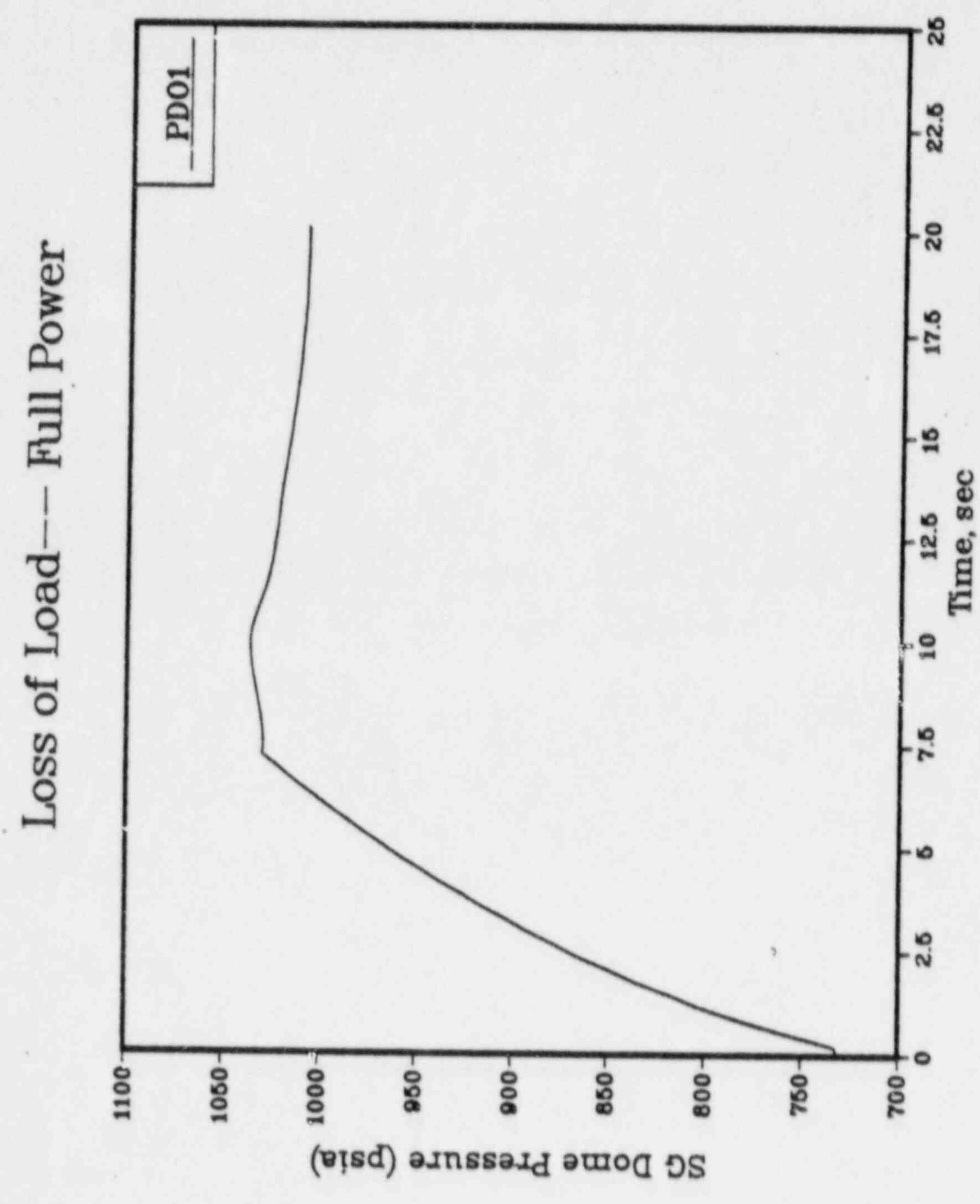


Figure 15.2.1-8 Secondary Pressure for Loss of External Load (Pressurization Case)

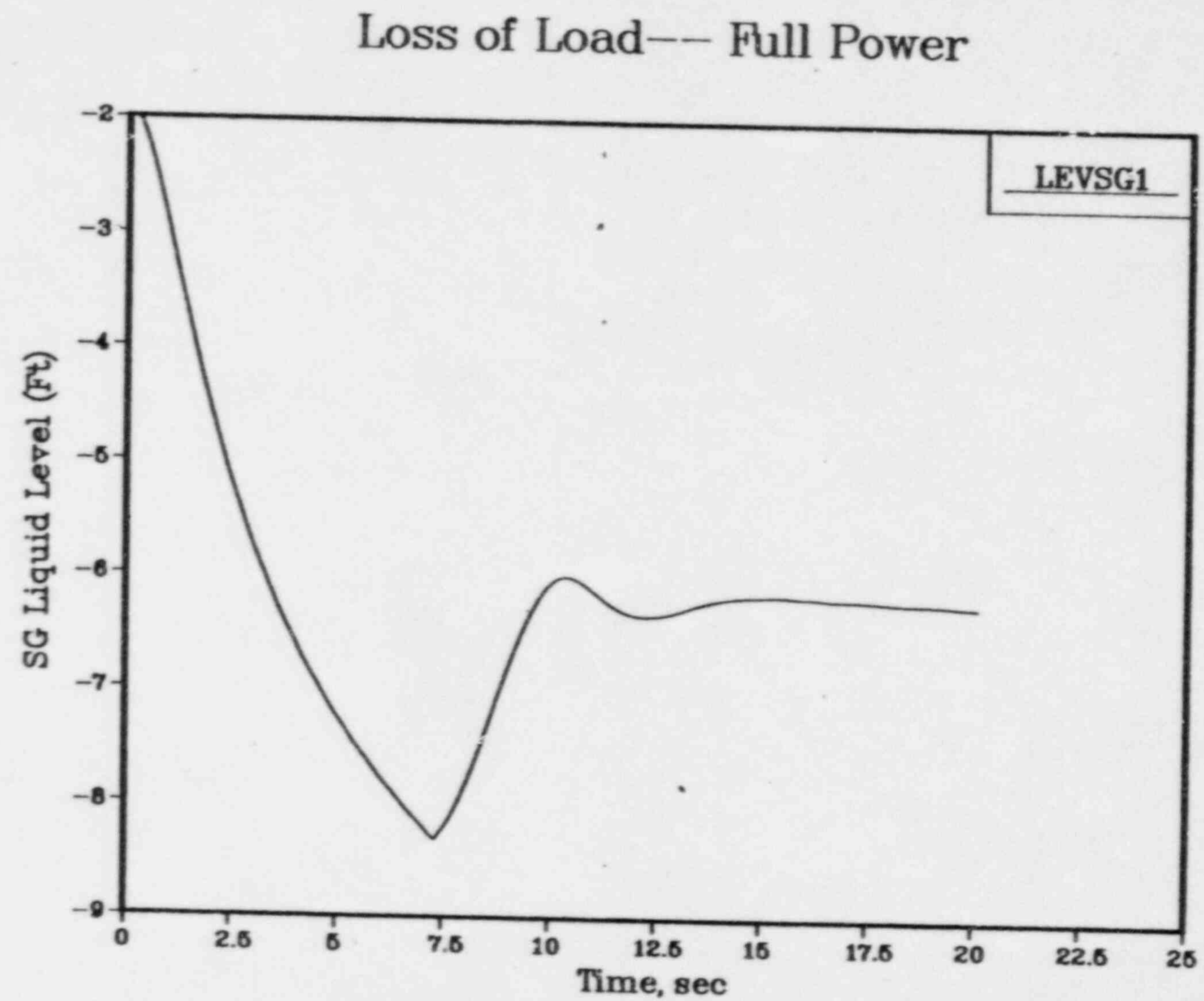


Figure 15.2.1-9 Steam Generator Liquid Level for Loss of External Load (Pressurization Case)

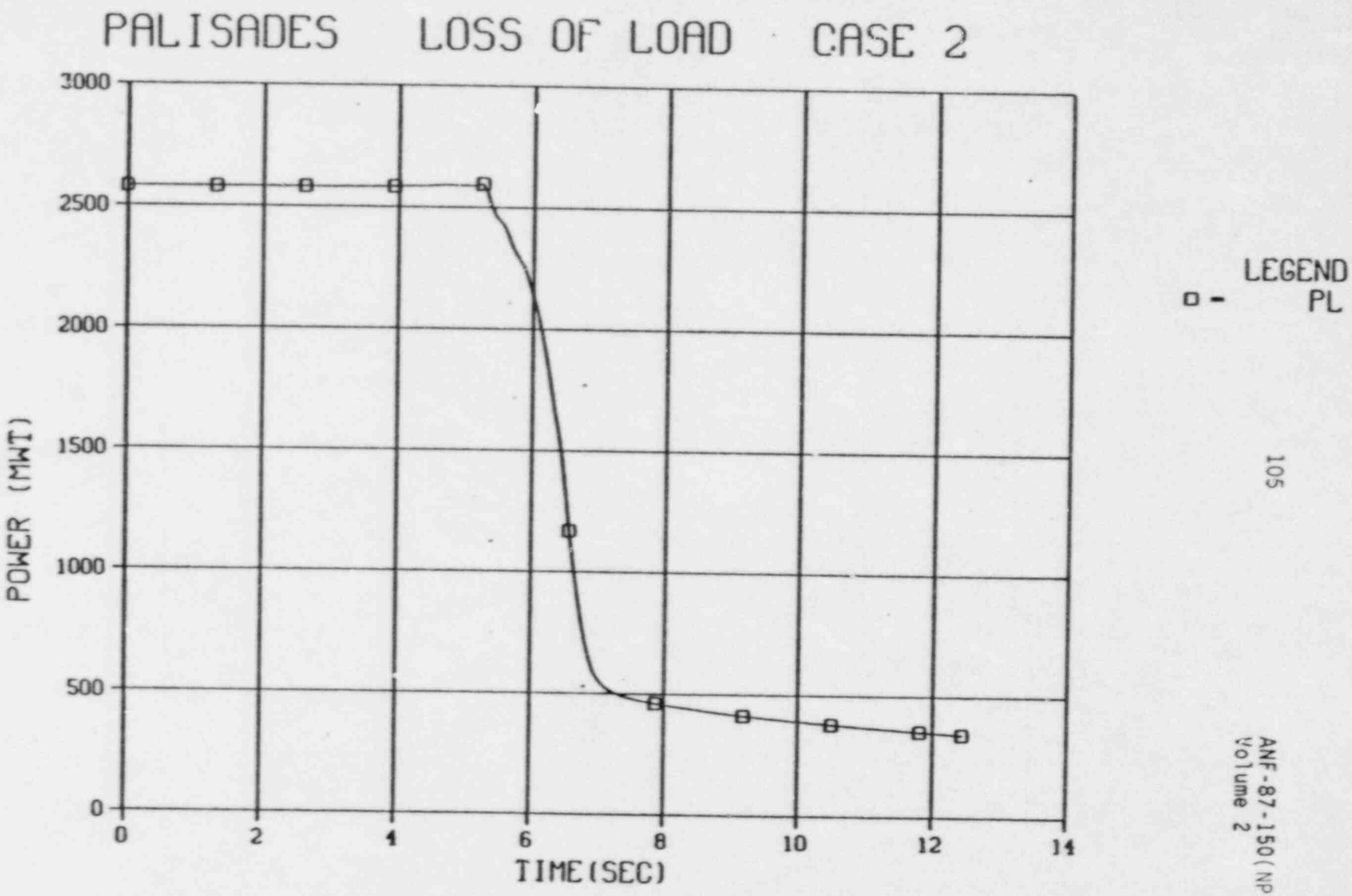


Figure 15.2.1-10 Reactor Power Level for Loss of External Load (Maximum delta P Case)

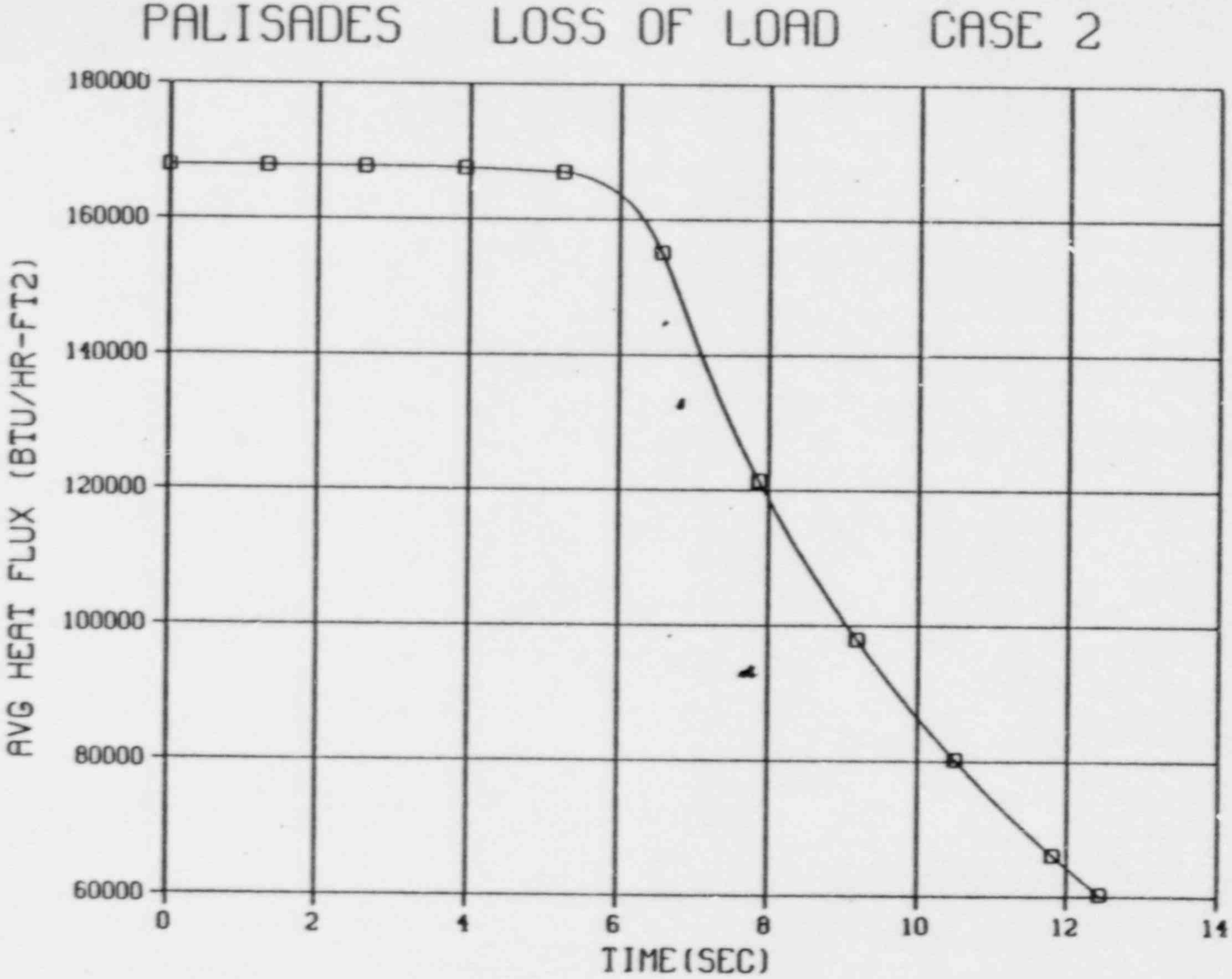


Figure 15.2.1-11 Core Average Heat Flux for Loss of External Load (Maximum delta P Case)

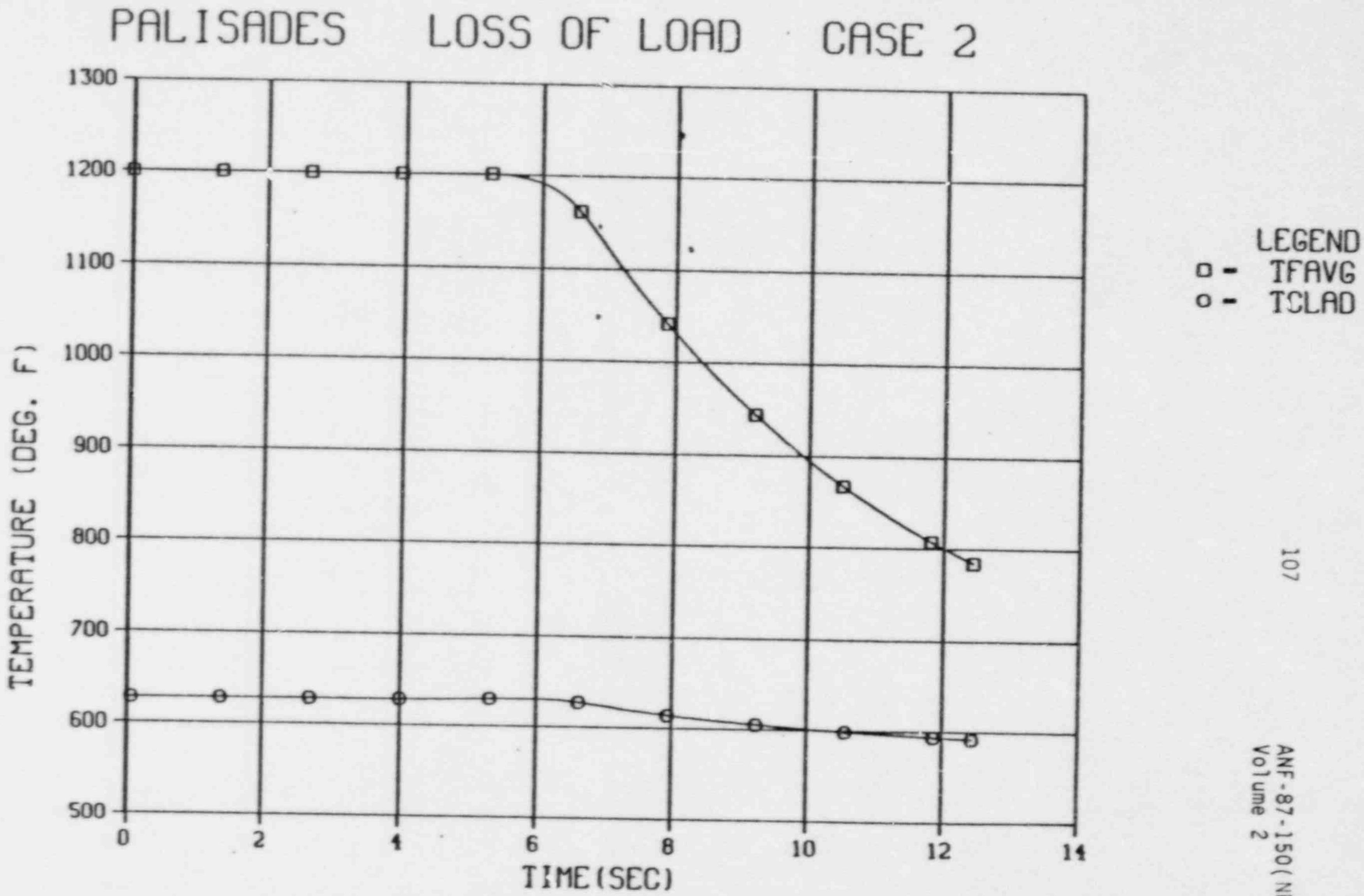


Figure 15.2.1-12 Average Fuel and Clad Temperatures for Loss of External Load (Maximum delta P Case)

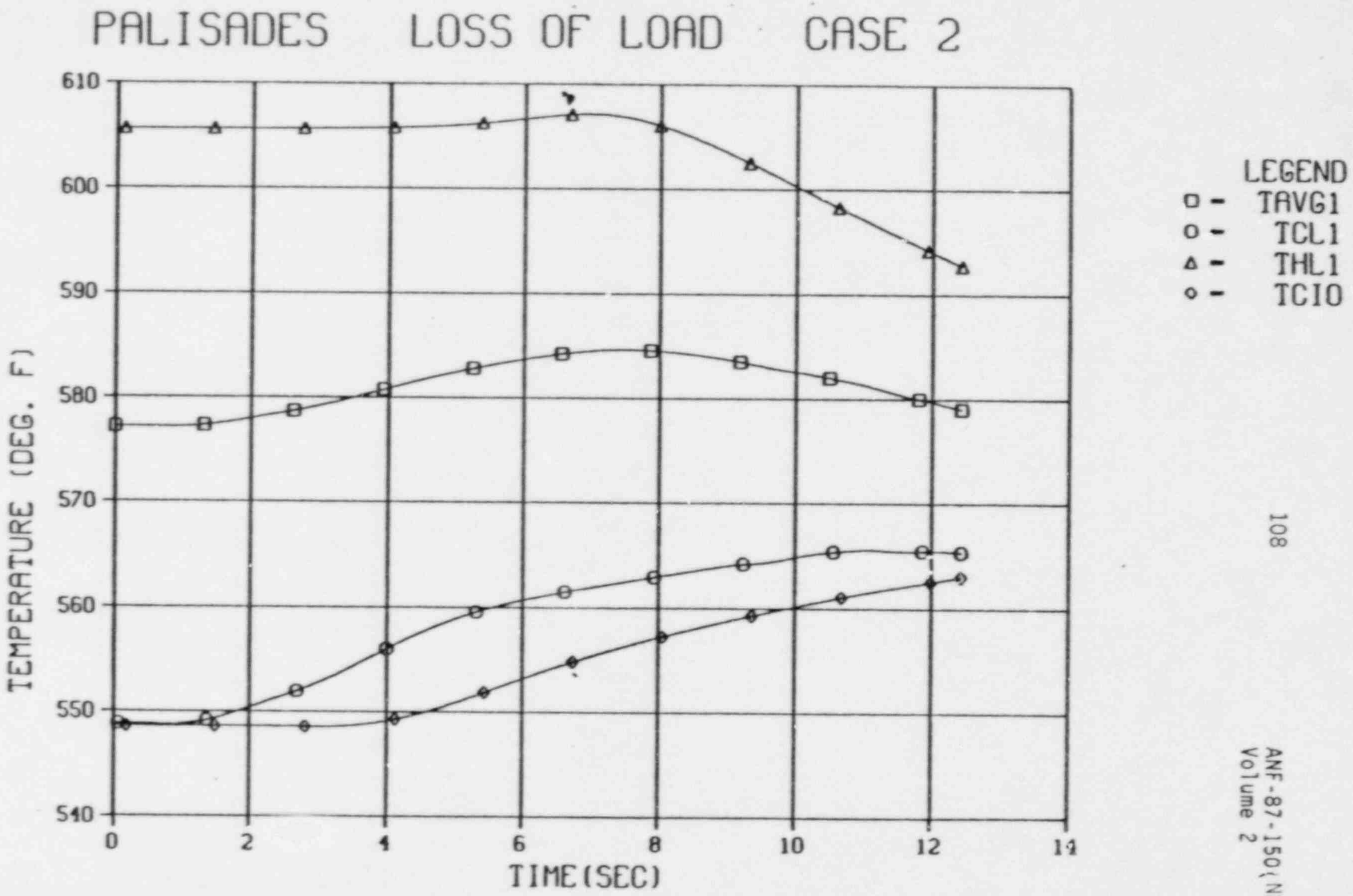


Figure 15.2.1-13 Reactor Coolant System Temperatures for Loss of External Load (Maximum delta P Case)

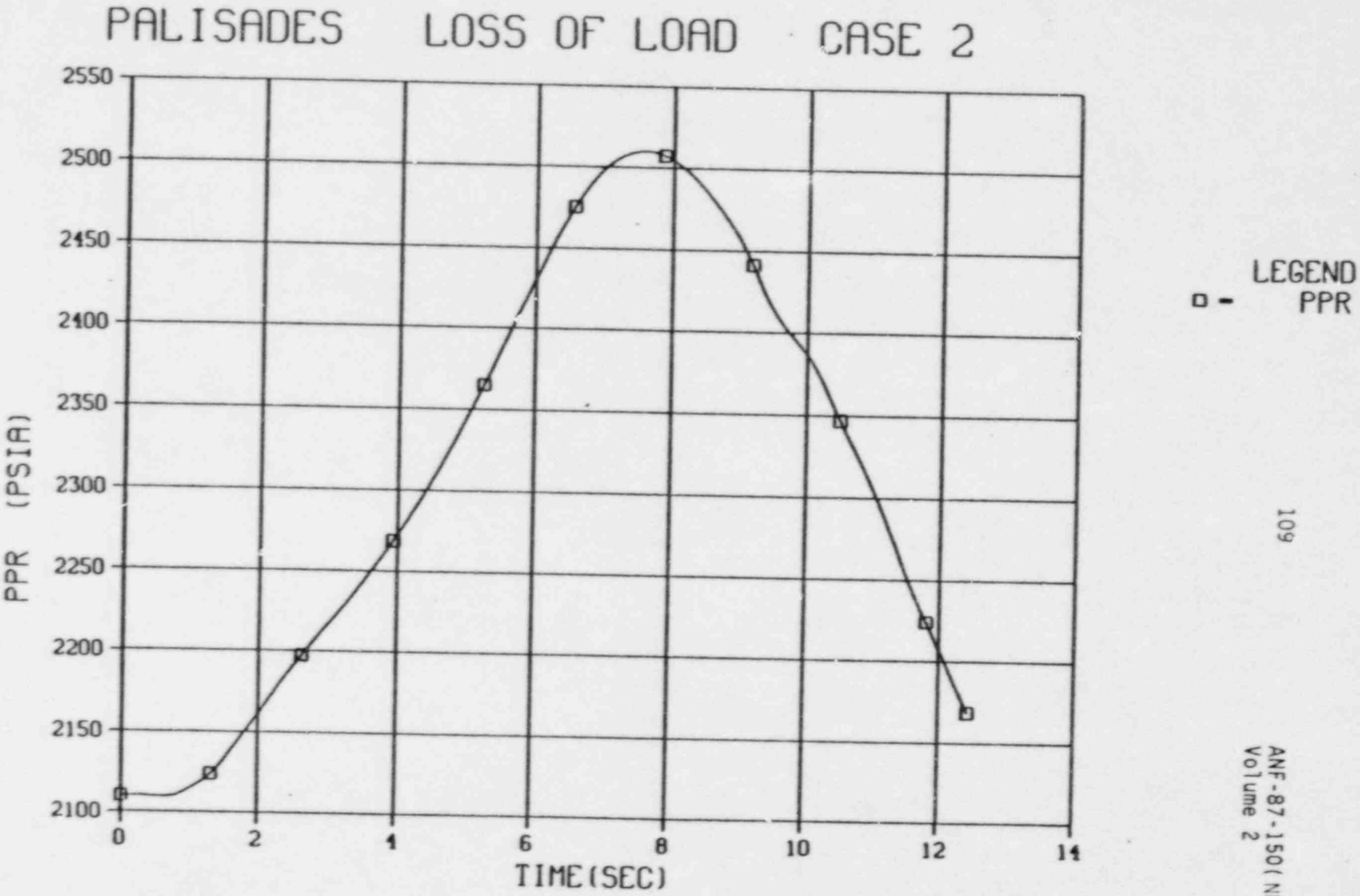


Figure 15.2.1-14 Pressurizer Pressure for Loss of External Load (Maximum delta P Case)

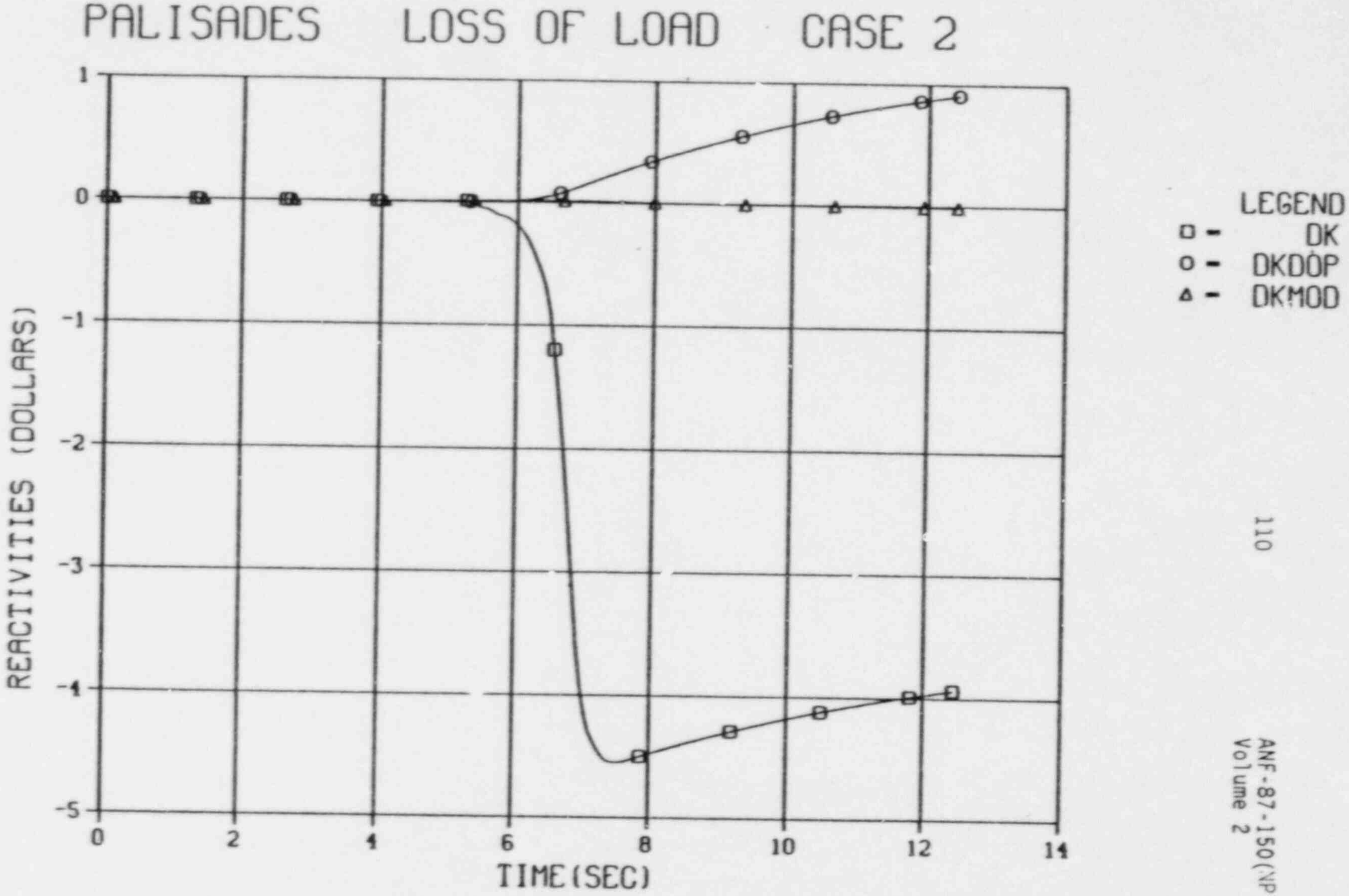


Figure 15.2.1-15 Reactivities for Loss of External Load
(Maximum delta P Case)

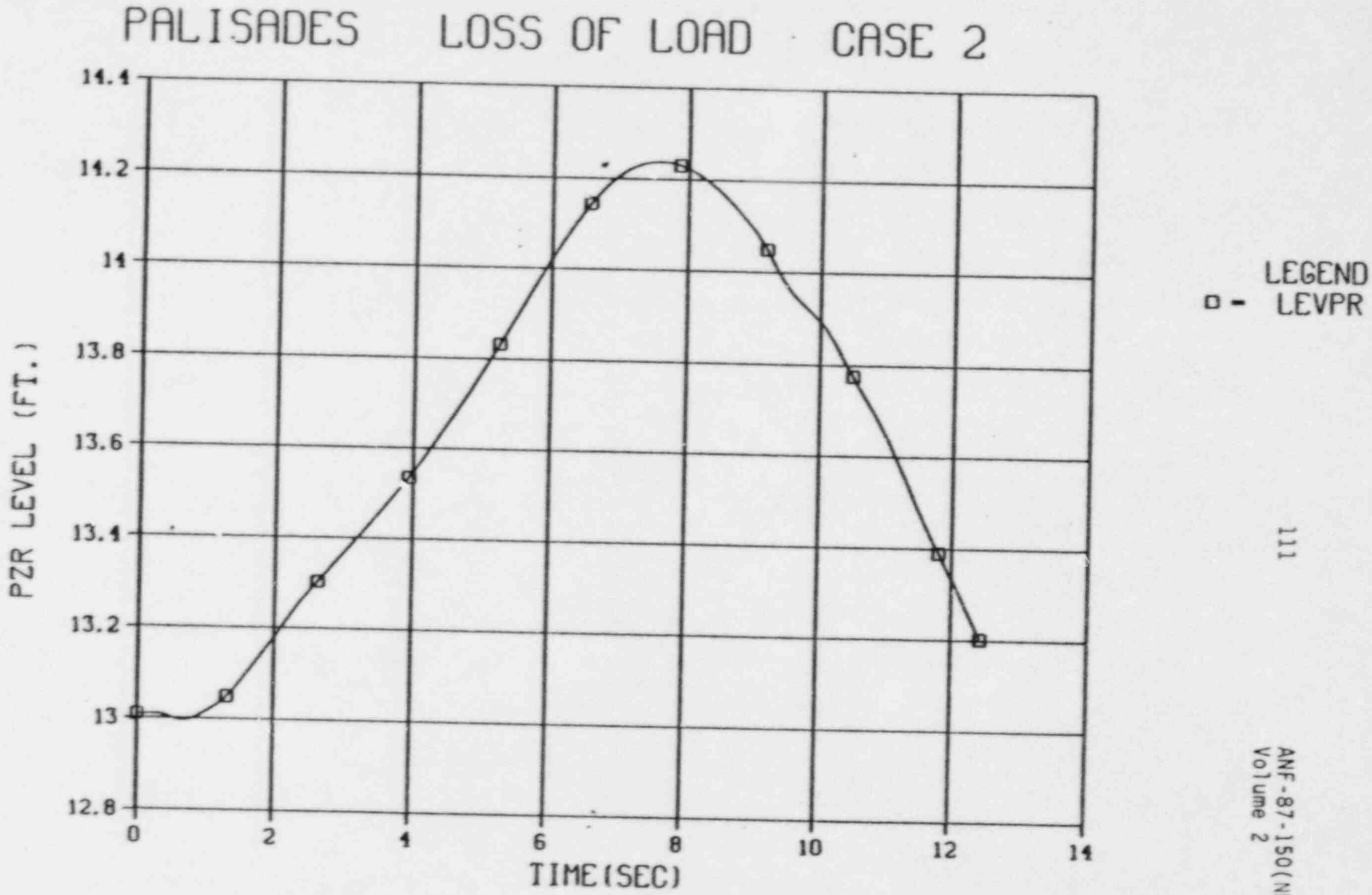


Figure 15.2.1-16 Pressurizer Liquid Level for Loss of External Load (Maximum delta P Case)

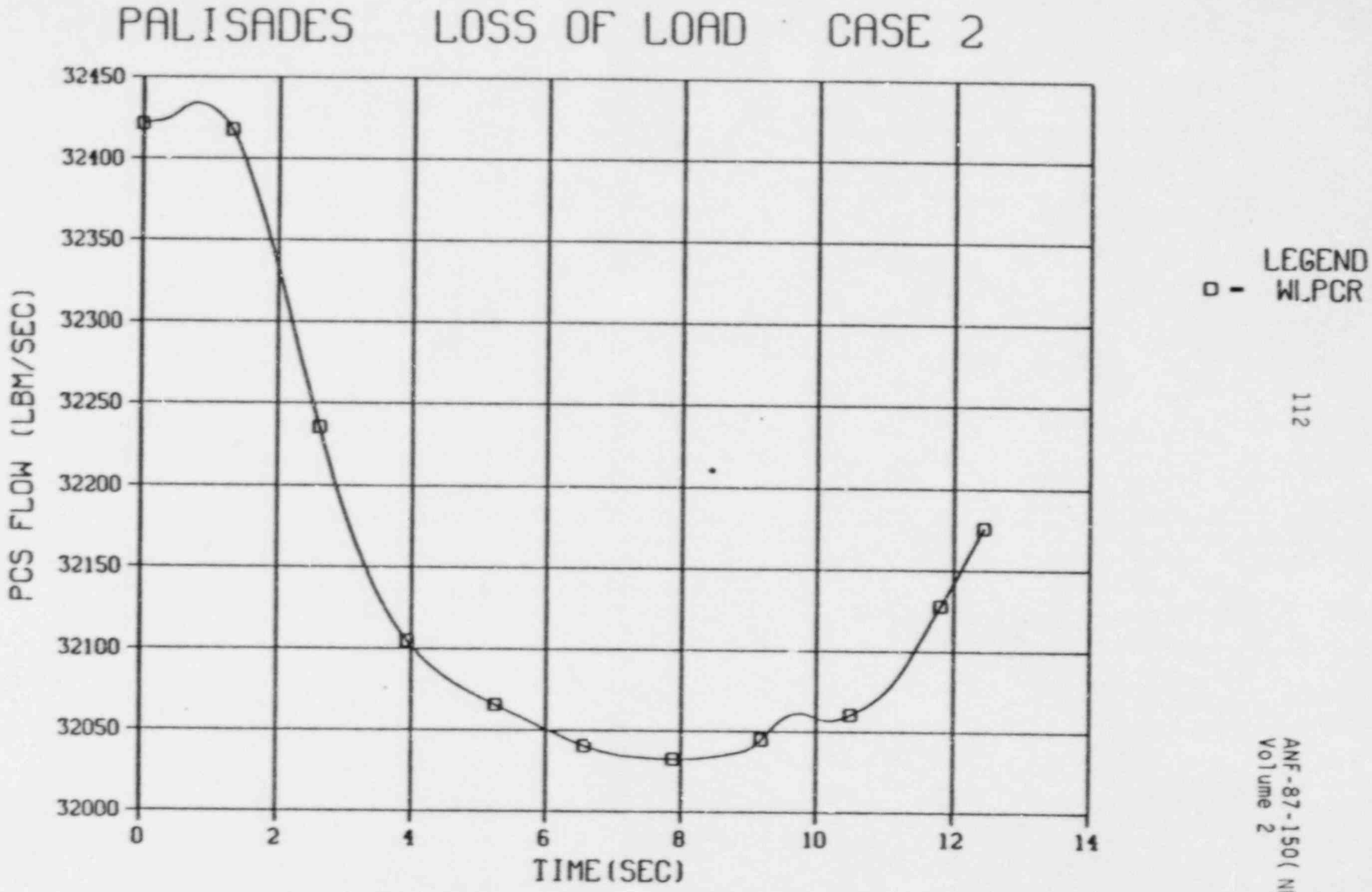


Figure 15.2.1-17 Primary Coolant Flow Rate for Loss of External Load (Maximum delta P Case)

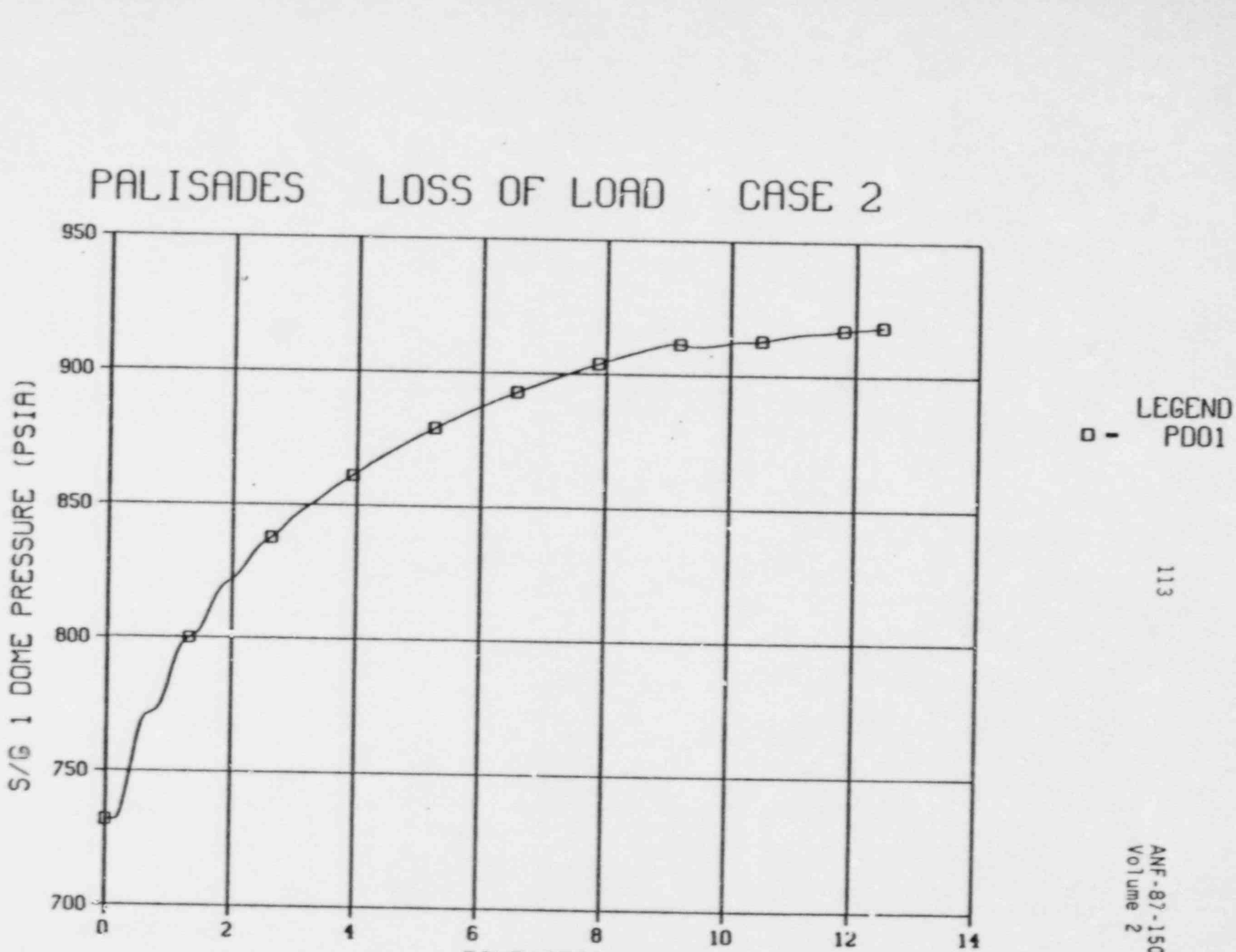


Figure 15.2.1-18 Secon[illegible] [illegible]essure for Loss of External Lo[illegible] [illegible]m delta P Case)

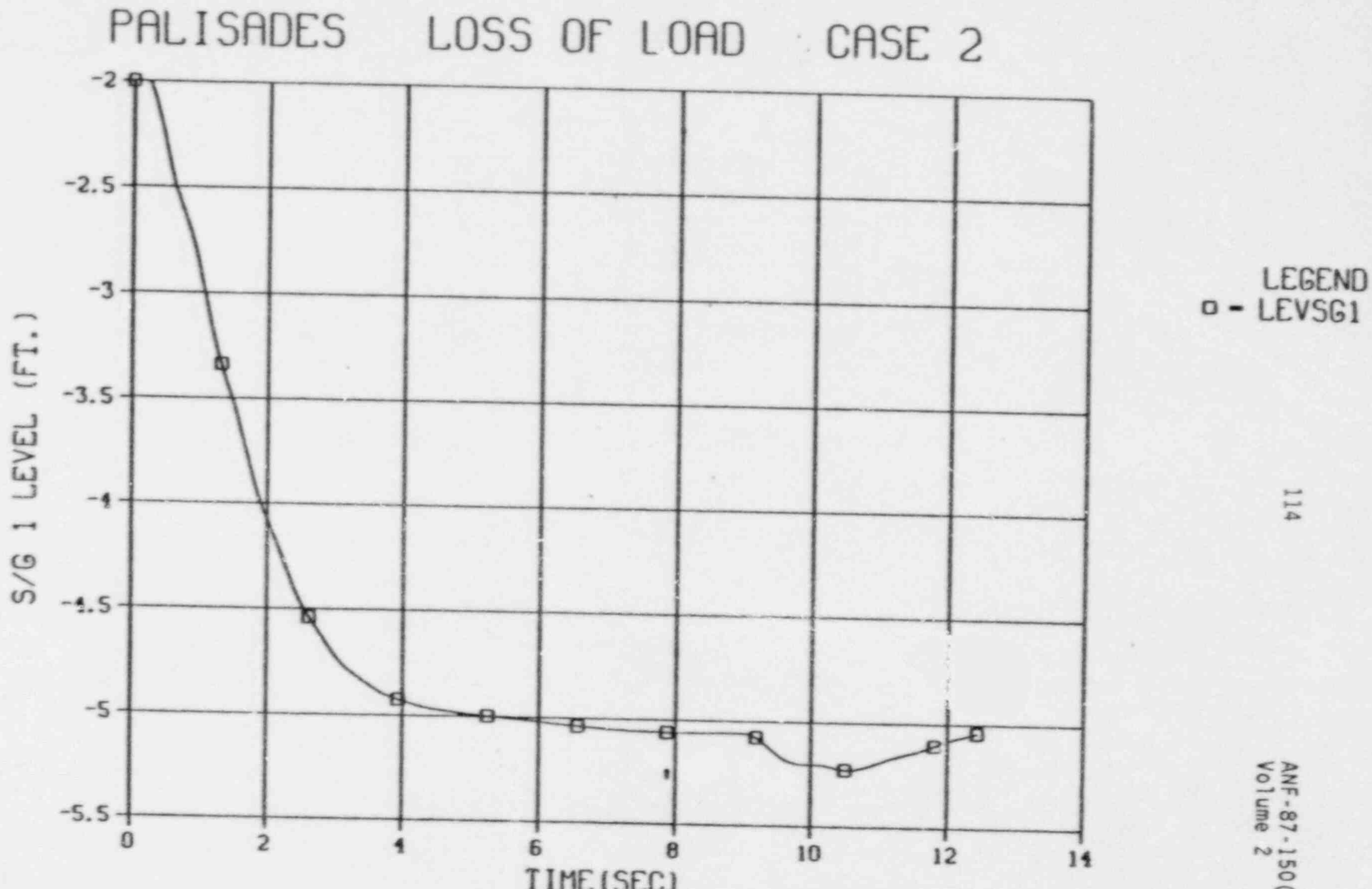


Figure 15.2.1-19 Steam Generator Liquid Level for Loss of External Load (Maximum [illegible] P Case)

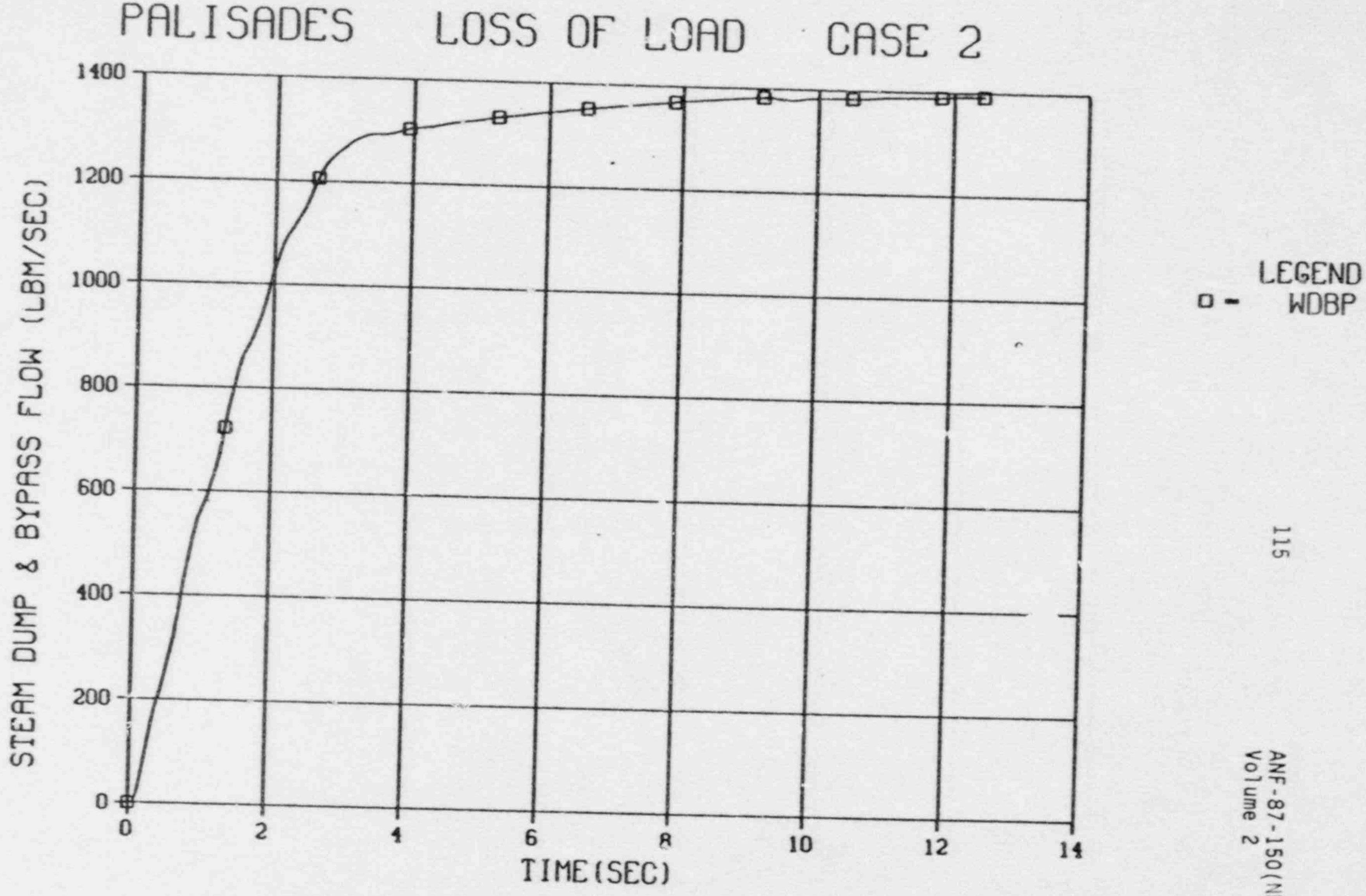


Figure 15.2.1-20 Steam Dump and Turbine Bypass Flow Rate for Loss of External Load (Maximum delta P Case)

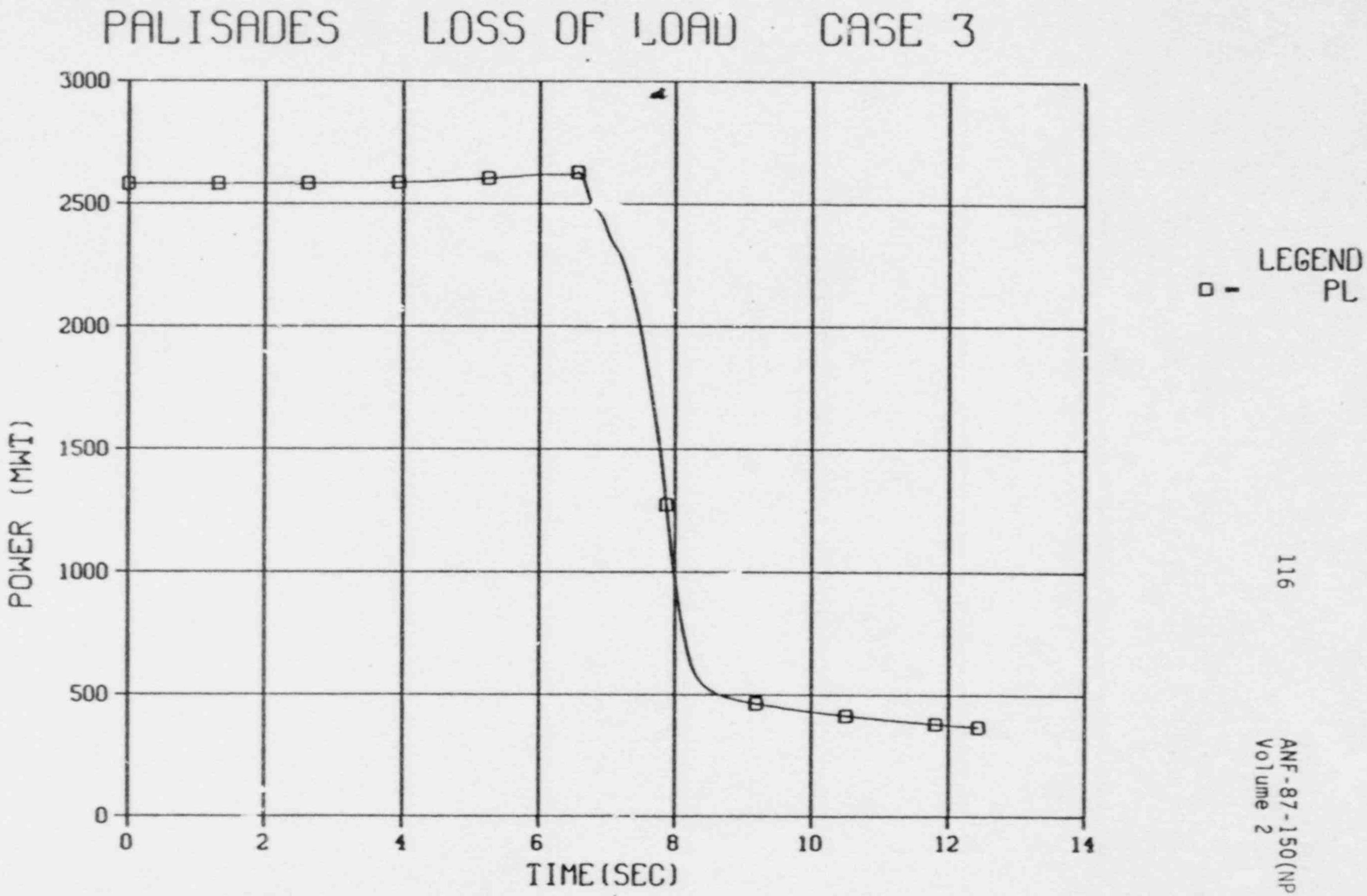


Figure 15.2.1-21 Reactor Power Level for Loss of External Load (MDNBR Case)

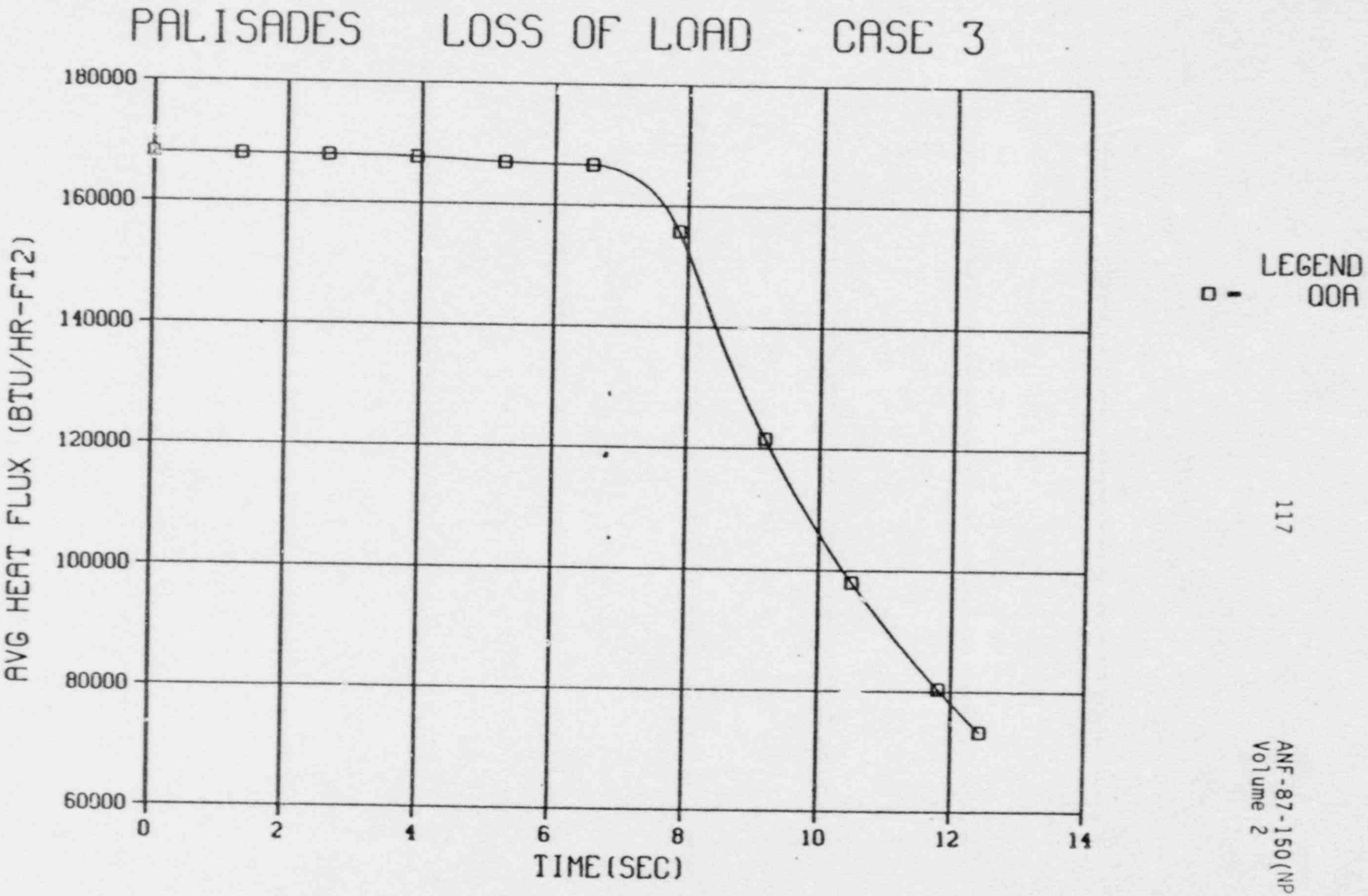


Figure 15.2.1-22 Core Average Heat Flux for Loss of External Load (MDNBR Case)

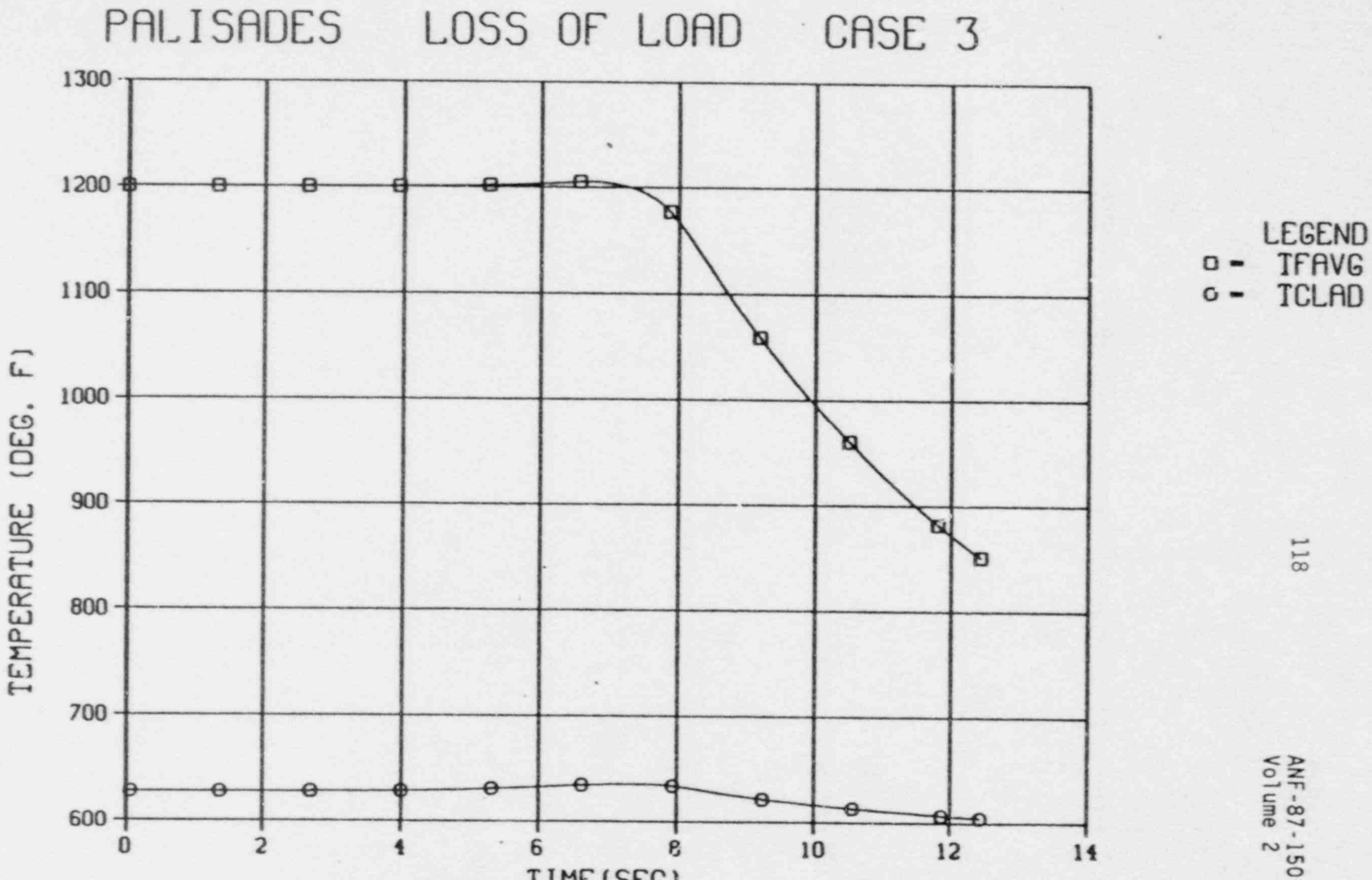


Figure 15.2.1-23 Average Fuel and Clad Temperatures for Loss of External Load (MDNBR Case)

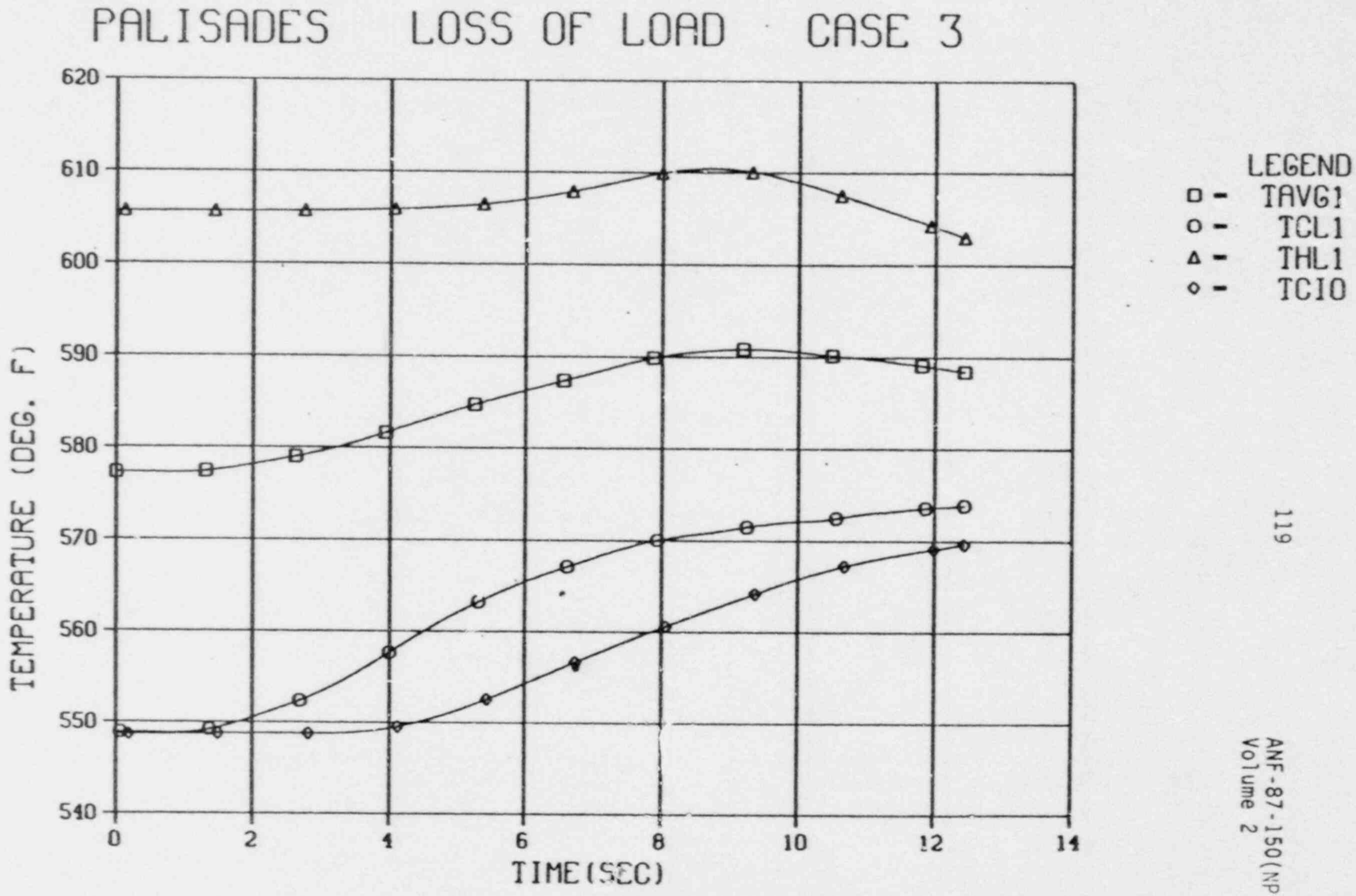


Figure 15.2.1-24 Reactor Coolant System Temperatures for Loss of External Load (MDNBR Case)

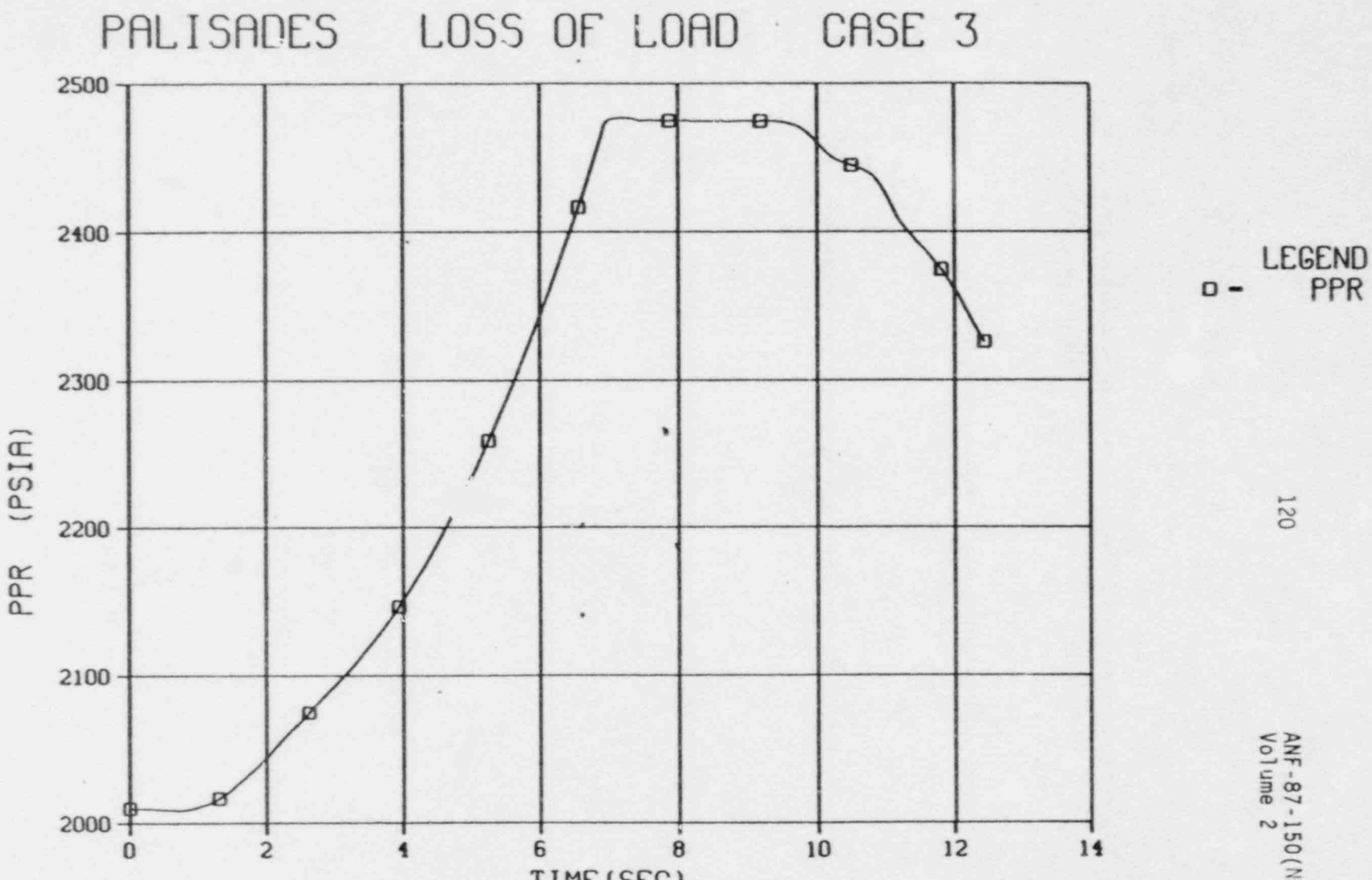


Figure 15.2.1-25 Pressurizer Pressure for Loss of External Load (MDNBR Case)

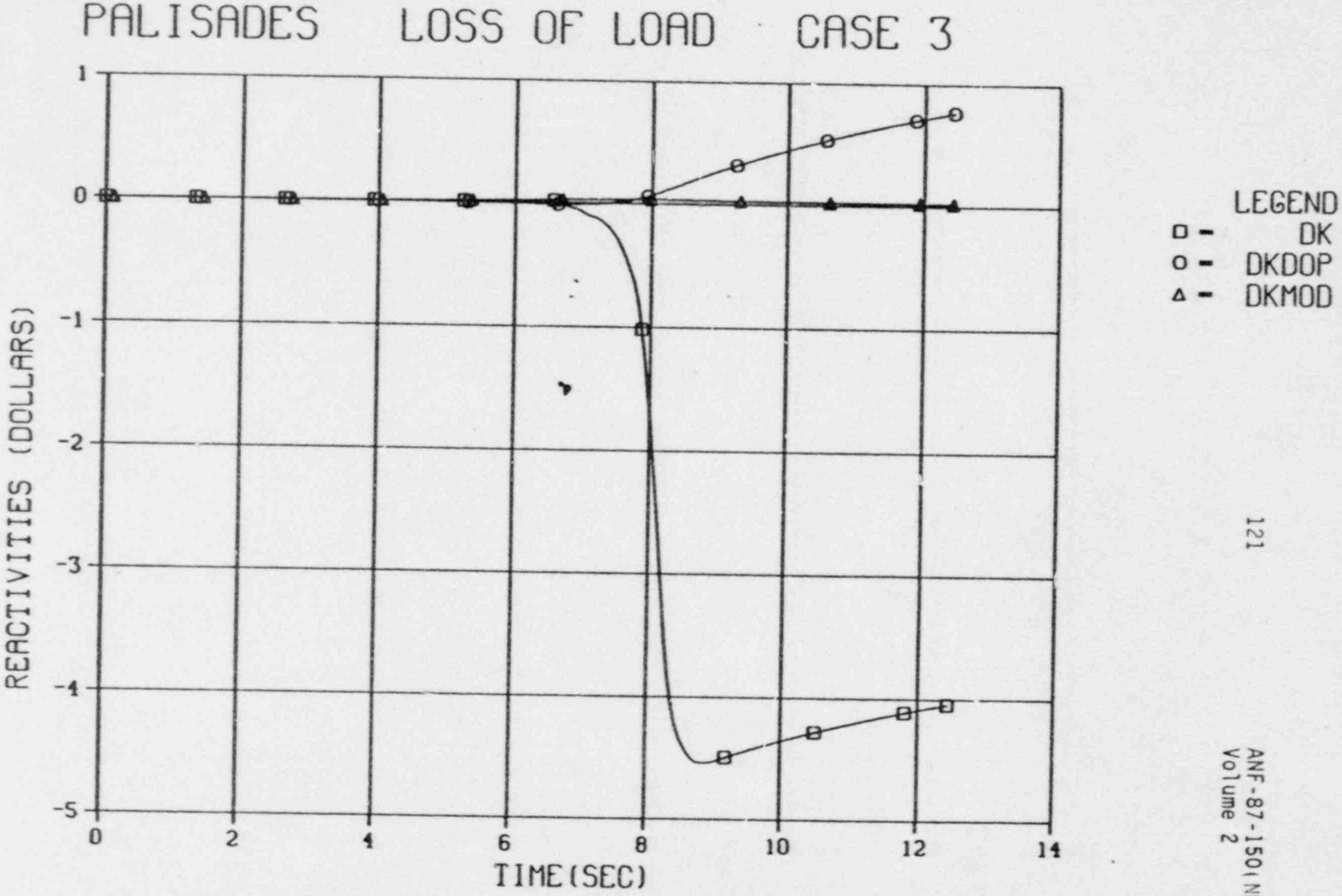


Figure 15.2.1-26 Reactivities for Loss of External Load (MDNBR Case)

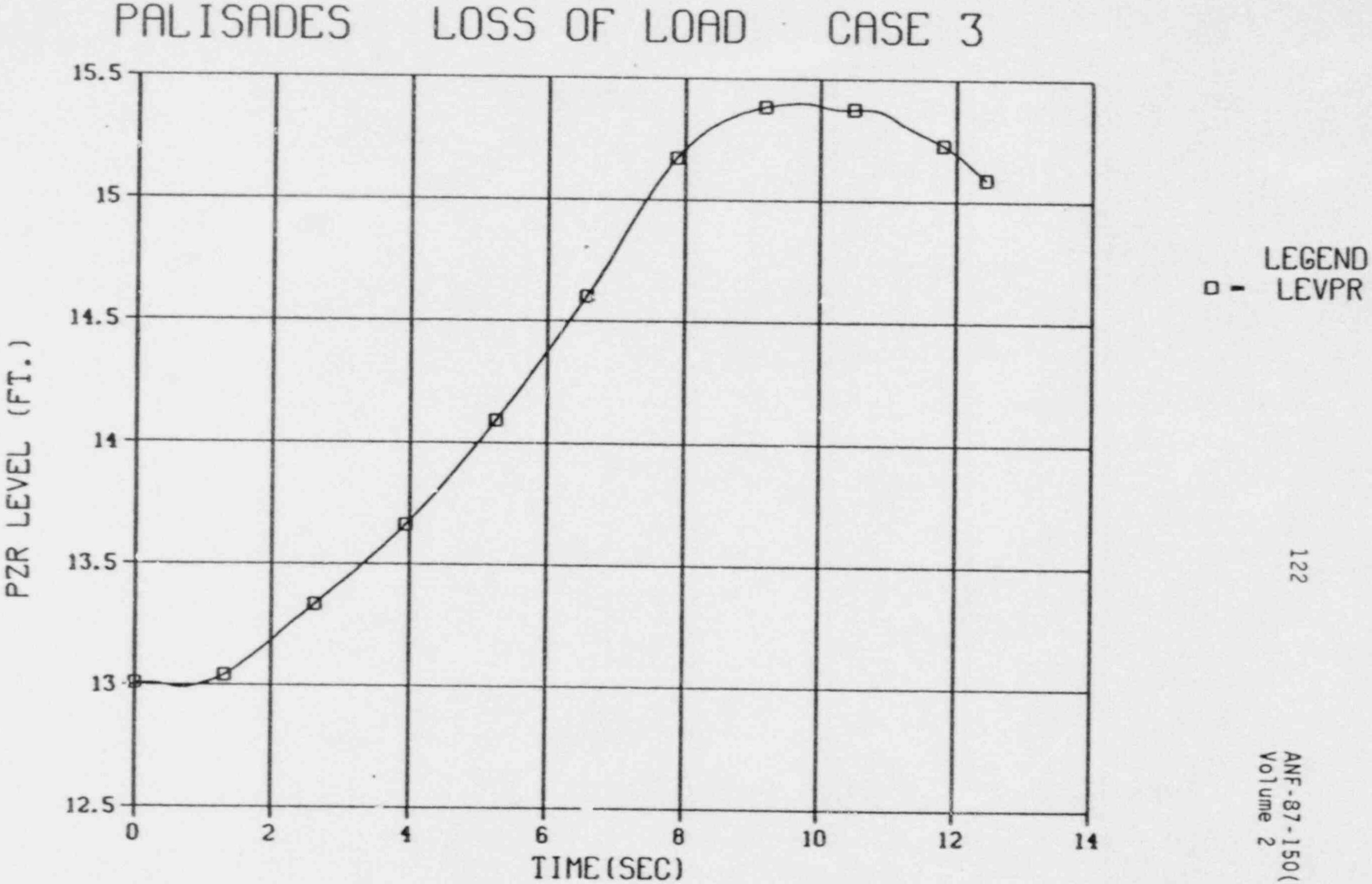


Figure 15.2.1-27 Pressurizer Liquid Level for Loss of External Load (MDNBR Case)

PLOT 5 13.17.09 THUR 31 OCT, 1985 JOB-OUFTIAS , U C C DISPLA VER 8.2

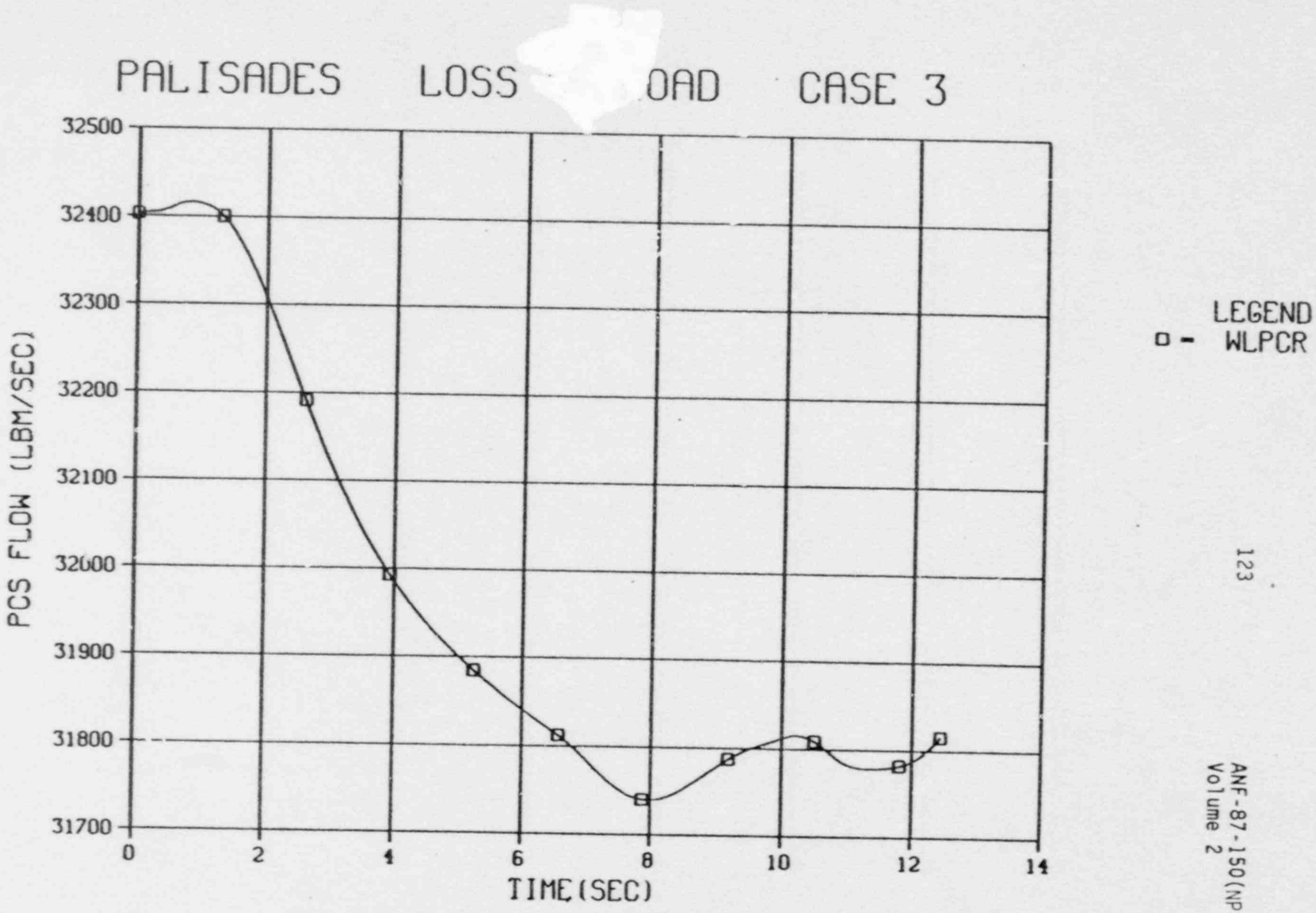


Figure 15.2.1-28 Primary Coolant Flow Rate for Loss of External Load (MDNBR Case)

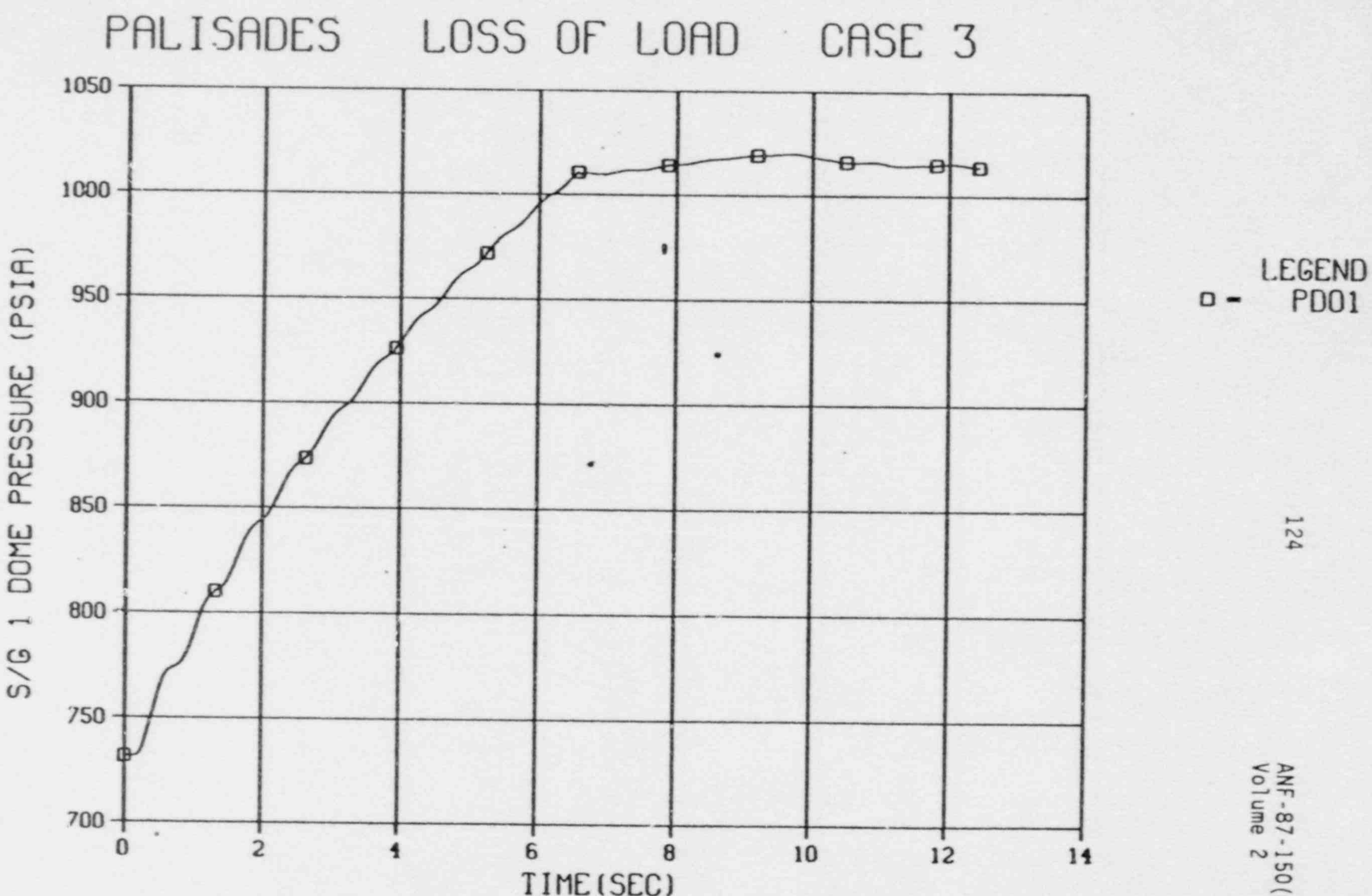


Figure 15.2.1-29 Secondary Pressure for Loss of External Load (MDNBR Case)

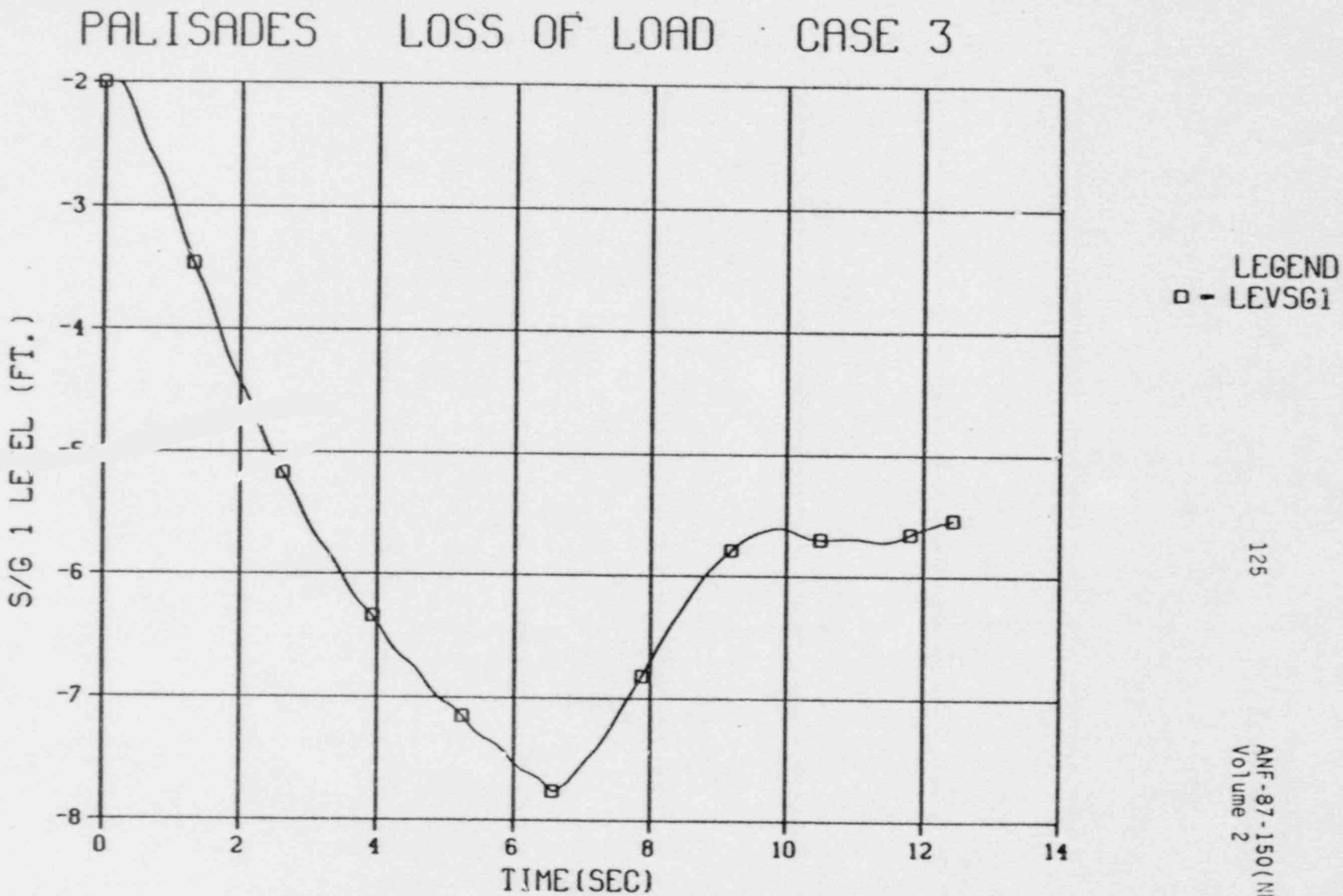


Figure 15.2.1-30 Steam Generator Liquid Level for Loss of External Load (MDNBR Case)

13.17.10 THUR 31 OCT, 1985 JOB=OLFTTAS , U C C DISSPLA VER 8.2

15.2.7 LOSS OF NORMAL FEEDWATER

15.2.7.1 Identification of Causes and Event Description

The Loss of Normal Feedwater Flow transient could result from a trip of the main feedwater pumps or a malfunction in the feedwater control valves. Consequently, there is a total loss of all main feedwater flow to the steam generators. Because the main feedwater system is supplying subcooled water to the steam generators, the loss of main feedwater flow will result in a reduction of the secondary system heat removal capability. The decrease in energy removal rate from the primary system causes the primary system fluid temperature to increase. The resulting primary system fluid expansion results in an insurge into the pressurizer, compressing the steam space and causing the primary system pressure to increase.

Long-term cooling capability through the steam generators is assured by the feedwater supplied to the steam generators through the Auxiliary Feedwater System. The motor driven auxiliary feedwater pumps (two) and the turbine driven auxiliary feedwater pump are automatically started on the following signal: low level in any one steam generator. The motor driven auxiliary feedwater pumps are powered by the emergency diesels if a loss of offsite power occurs. The steam for the turbine driven pump is provided through the plant secondary system.

15.2.7.2 Analysis Method

The analysis is performed using the SLOTRAX code(13), supplemented with separate calculations to assure bounding results for pressurizer liquid volume and maximum reactor coolant system pressurization. The SLOTRAX code includes relevant aspects of the primary and secondary systems. The following events are assumed to occur at event initiation:

a) Reactor trips on steam generator low level with specified time delay.

b) Turbine conservatively trips with simultaneous closure of turbine stop valve at initiation of event.

c) Main feedwater valves are closed with specified delivery.

d) Backup heaters in the pressurizer are assumed to be full on throughout the transient.

e) Start sequence for emergency diesel generators is initiated with specified time delay for delivery of auxiliary feedwater.

The analyses considered asymmetric and symmetric steam generator tube plugging with and without primary coolant pump coastdown where auxiliary feedwater was available to both steam generators or only one steam generator.

The SLOTRAX pressurizer model is an equilibrium model, i.e., pressurizer pressure is predicted solely as a function of pressurizer liquid temperature. Pressurizer pressure is thus predicted by SLOTRAX to decrease throughout the event due to the continual insurge of hot leg fluid that is at a lower temperature than the pressurizer liquid. The pressurizer safety valves and PORVs are thus not actuated in the SLOTRAX simulation, and no steam is vented. During an insurge into the pressurizer, the steam volume is partially condensed and added to the pressurizer liquid mass, leading to an overly conservative prediction of pressurizer liquid volume.

The compression of the steam bubble during this event will actually result in a pressure increase rather than a decrease as the SLOTRAX simulation predicts. The pressurizer pressure response and liquid volume swell can be obtained by transferring, time-dependent pressurizer surge line flow and temperature from the SLOTRAX simulation of the loss of normal feedwater event to a non-equilibrium pressurizer model. The pressurizer model used for these calculations is the model found in the PTSPWR2[10] plant simulation transient code. The PTSPWR2 pressurizer model is a two-region, adiabatic non-equilibrium model. The governing mass and energy equations, along with the thermodynamic state equations, are solved simultaneously for each of the two phases. Pressurizer relief and safety valve actuation may be modeled when the appropriate setpoints are reached. The resulting pressure response and liquid volumetric swell are conservatively predicted due to conservatism in the SLOTRAX boundary conditions and conservative assumptions governing the operation of the pressurizer heaters, sprays, PORVs, safety valves, and the steam generator heat transfer. In the present analysis, there is sufficient margin in the pressurizer liquid level to not predict filling the pressurizer liquid solid. There is also sufficient excess capacity in the pressurizer safety valve discharge to accommodate the steam region compression and maintain the peak pressurizer pressure at the safety valve setpoint. Thus, the time dependent pressurizer pressure calculation has not been included.

15.2.7.3 Definition of Events Analyzed and Bounding Input

Three cases were analyzed at rated thermal power:

1) Nominal initial conditions

2) Worse case initial conditions - biased to minimize steam generator inventory during the transient

3) Worse case initial conditions - biased to maximize pressurizer liquid level, i.e. pressure, during the transient

Conservative conditions are established for analysis of each event, as noted in Table 15.2.7-1. For each of these three cases, both symmetric and asymmetric steam generator tube plugging was considered, in addition to reactor coolant pumps on and reactor pumps off. The SLOTRAX code does not have a kinetics model, therefore, a power decay curve incorporating the power response to scram and the long-term affects of decay heat is input. The curve used for the short-term initial decrease in power due to reactor trip is referenced to that for Event 15.2.1. For the long-term, decay heat from fission products is used.

15.2.7.4 Analysis of Results

The event is initiated by reactor trip on the steam generator low level trip setpoint. The steam generators are conservatively assumed to isolate at event initiation. The primary coolant pumps are also tripped at initiation for the pumps off case.

The initial operating conditions are presented in Table 15.2.7-2. An event summary is presented in Table 15.2.7-3. The transient response is presented in Figures 15.2.7-1 to 15.2.7-3 for the nominal initial condition case, in Figures 15.2.7-4 to 15.2.7-6 for the worse inventory case, and Figures 15.2.7-7 to 15.2.7-13 for the worse pressurizer swell/pressure case. These limiting cases are all symmetric steam generator tube plugging cases which were found to bound the asymmetric plugging cases.

For both pumps on and pumps off cases, core power decreases to decay levels within a few seconds of event initiation, due to reactor scram on low steam generator level. Primary coolant average temperature peaks rapidly during this time due to the short term loss of heat sink caused by the turbine trip, with subsequent pressurizer level and pressure increase due to primary coolant insurge. Primary system heatup is reversed by the effects of reactor scram, the opening of the main steam system safety valves, and delivery of auxiliary feedwater. The auxiliary feedwater flow rate used in this analysis is 300 gpm at 120 °F. This flow rate corresponds to the capacity of one motor driven auxiliary feedwater pump. For all pumps on cases, the reactor coolant temperature stabilizes at about 565°F, then begins to decrease as the auxiliary feedwater heat removal capacity exceeds the decay heat production. For all of the pumps off cases (following the initial coolant temperature increase as the pumps coast down), the reactor coolant temperature decreases throughout the transient.

The pumps off cases are the most limiting with respect to pressurizer liquid volume and pressure. The pressure is calculated to reach the pressurizer safety valve setpoint in about 6 seconds (pumps on) and about 5 seconds (pumps off). No credit is taken for the operation of the power operated relief valves (PORVs) since they are normally blocked closed. The discharge capacity of the pressurizer safety valves is sufficient to maintain peak pressurizer pressure at or near the maximum safety valve setpoint pressure (2580 psia). Maximum liquid volume is calculated to be about 1206 ft^3 as compared to a total pressurizer capacity of 1500 ft^3. Sufficient steam volume remains to preclude the expulsion of liquid from the pressurizer safety valves.

15.2.7.5 Conclusion

A loss of normal feedwater event does not result in the violation of SAFDLs, peak pressurizer pressure does not exceed 110% of the design rating, and primary liquid is not expelled through the pressurizer safety valves. Adequate cooling water is supplied by the auxiliary feedwater system to allow a safe and orderly plant shutdown and to prevent steam generator dryout, assuming minimum auxiliary feedwater capacity.

Table 15.2.7-1 Conservative Assumptions Used in the Loss of Normal Feedwater Event

	Nominal Initial Condition	Worse Case Inventory	Worse Case Pressurizer Swell/Pressure
Control	Manual	Manual	Manual
Power	Rated +2%	Rated +2%	Rated +2%
Power Decay	Ref. Event 15.2.1	Ref. Event 15.2.1	Ref. Event 15.2.1
Long Term	ANS 5.1 + Actinides	ANS 5.1 + Actinides	ANS 5.1 + Actinides
Core Coolant Average Temperature	Nom.	Nom. -4°F	Nom. -4°F
Primary Pressure	Nom. -50 psi	Nom. +50 psi	Nom. +50 psi
Pressurizer Level	TS* Max.	TS* Max. +2.5%	TS* Max. +5%
Steam Generator Pressure	Nom.	Nom.	Nom. -50 psi
Steam Generator Safety Valve Setpoints	Nom.	Nom. -3%	Nom. +3%
Condensate Storage Tank Temperature	Max.	Max.	Max.
Auxiliary Feedwater	Min.	Min.	Min.
Availability of Auxiliary Feedwater	120 sec.	120 sec.	120 sec.
Pressurizer Heaters	Available	Available	Available
Pressurizer Spray	Disable	Disable	Disable
Pressurizer PORVs	Disable	Disable	Disable
Pressurizer Safety Valve Setpoints	2580 psia	2580 psia	2580 psia
Steam Bypass	Disable	Disable	Disable
Motor Driven Aux. Feedwater Pumps	1 Available	1 Available	1 Available
Steam Driven Aux. Feedwater Pumps	Unavailable	Unavailable	Unavailable
Reactor Trip	S.G. Low Level	S.G. Low Level	S.G. Low Level

* Technical Specifications

Table 15.2.7-2 Summary of Initial Operating Conditions for Loss of Normal Feedwater

	Nominal Initial Condition	Worse Case Inventory	Worse Case Pressurizer Swell/Pressure
Total Steam Generator Heat Load (MWt)	2596.9	2596.9	2596.6
Core Coolant Average Temperature (°F)	569.4	566.6	566.6
Pressurizer Pressure (psia)	2010.	2110.	2110.
Pressurizer Level (ft)	12.94	13.6	14.08
Pressurizer Liquid Volume (ft^3)	891.	935.	967.
Steam Generator Water Inventory (lb)	78931.	78931.	78931.
Steam Dome Pressure (psia)	730.	730.	680.
Steam Generator Safety Valve Setpoints (psia)	1000., 1025., 1040.	970., 993., 1009.	1029., 1054., 1070.
Pressurizer Safety Valve Setpoint (psia)	2580.	2580.	2580.

Table 15.2.7-3 Event Summary for Loss of Normal Feedwater

	Nominal Initial Condition	Worse Case Inventory	Worse Case Pressurizer Swell/Pressure
Reactor Trip on Low Steam Generator Level; Auxiliary Feedwater Initiates (sec)	0.0	0.0	0.0
Primary Coolant Pumps Trip (sec)	0.0	N/A*	0.0
Pressurizer Valves open (sec)	5.66	5.96	4.66
Max. Pressurizer Liquid Volume (sec)	13.0	~5500	15.0
Max. Pressurizer Liquid Volume (ft^3)	1107.	1076.	1206.
Auxiliary Feedwater available to S/G (sec)	120.	120.	120.
Minimum S/G Level (sec)	-2000	-5500	-2000
Mass per S/G at time of min. level (lb)	33571.	-2828	-37885

*Pumps on case.

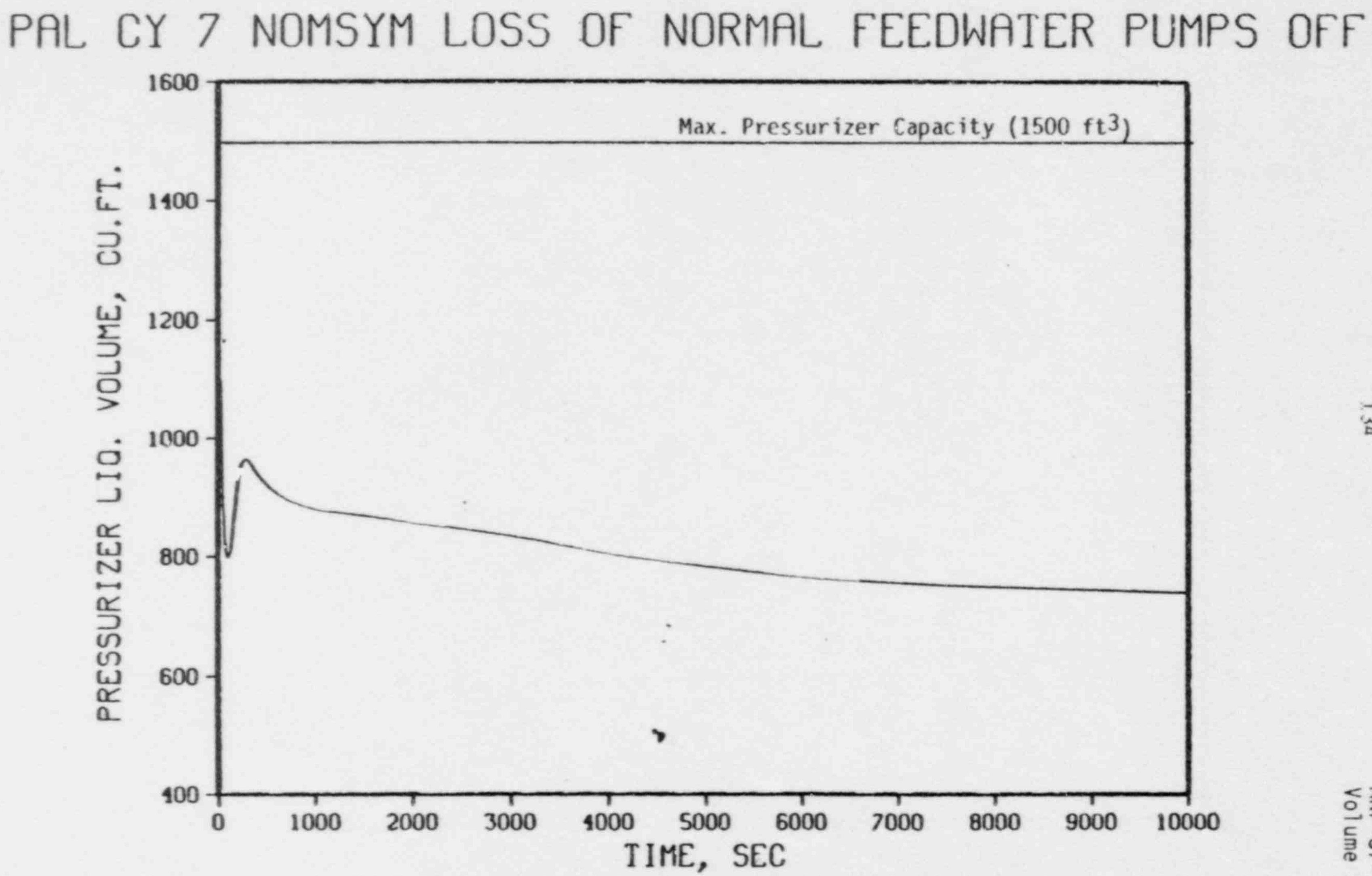


Figure 15.2.7-1 Nominal Conditions, Symmetric Steam Generator Tube Plugging, Pumps off- Pressurizer Liquid Volume vs. Time- Loss of Normal Feedwater

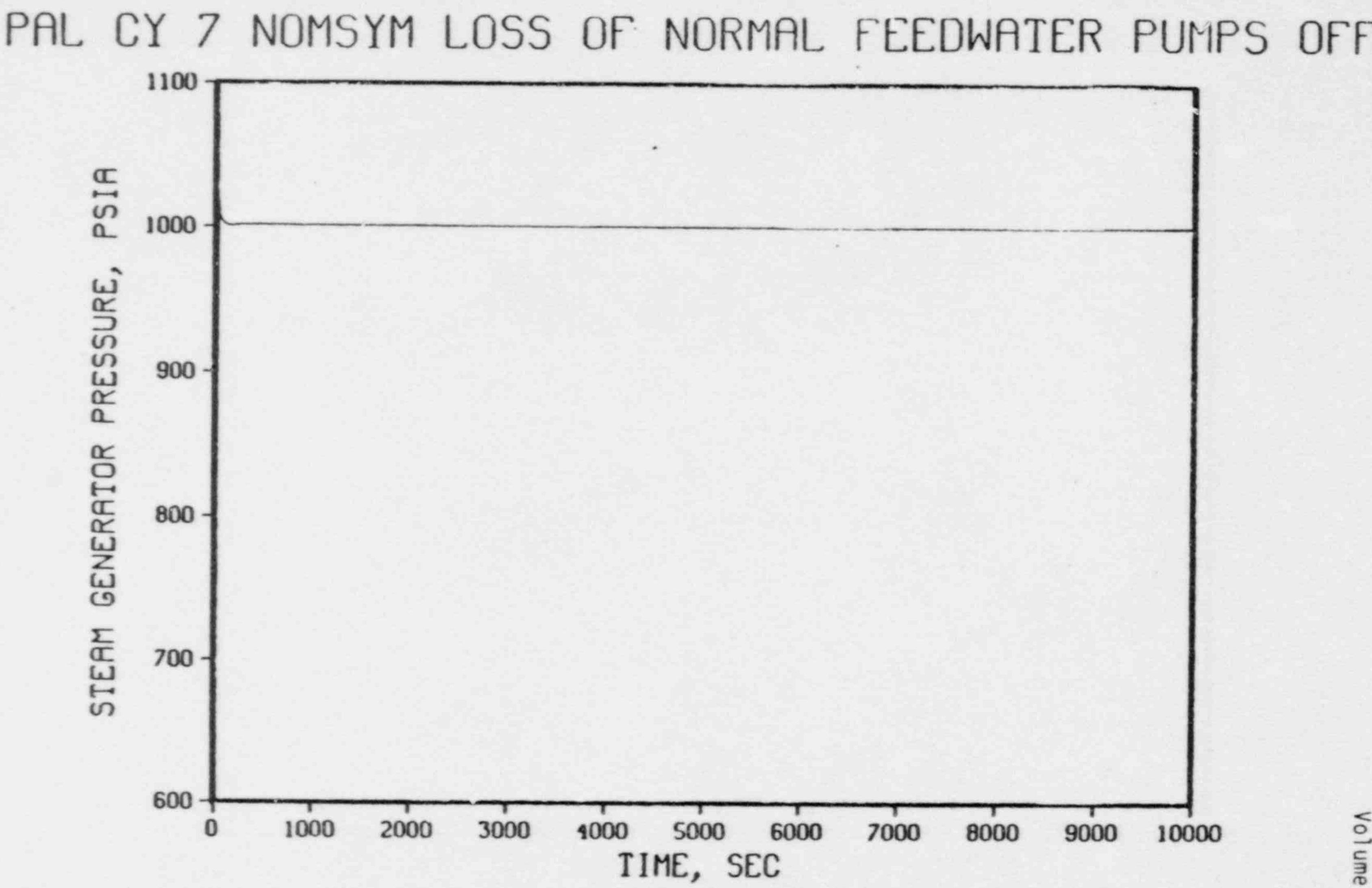


Figure 15.2.7-2 Nominal Conditions, Symmetric Steam Generator Tube Plugging, Pumps off- Steam Generator Pressure vs. Time- Loss of Normal Feedwater

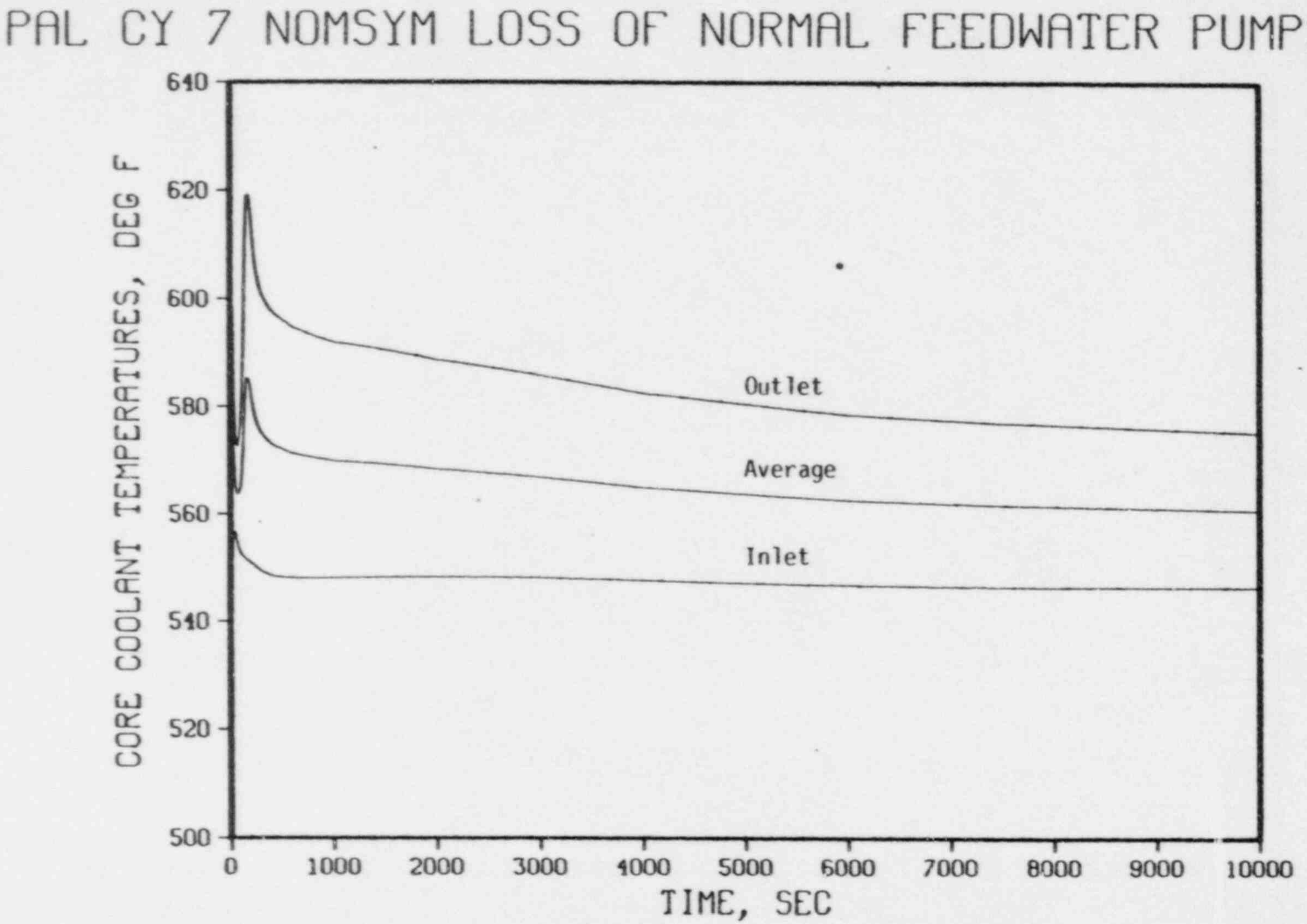


Figure 15.2.7.3 Normal Conditions, Symmetric Steam Generator Tube Plugging, Pumps off- Reactor Coolant Temperatures vs. Time- Loss of Normal Feedwater

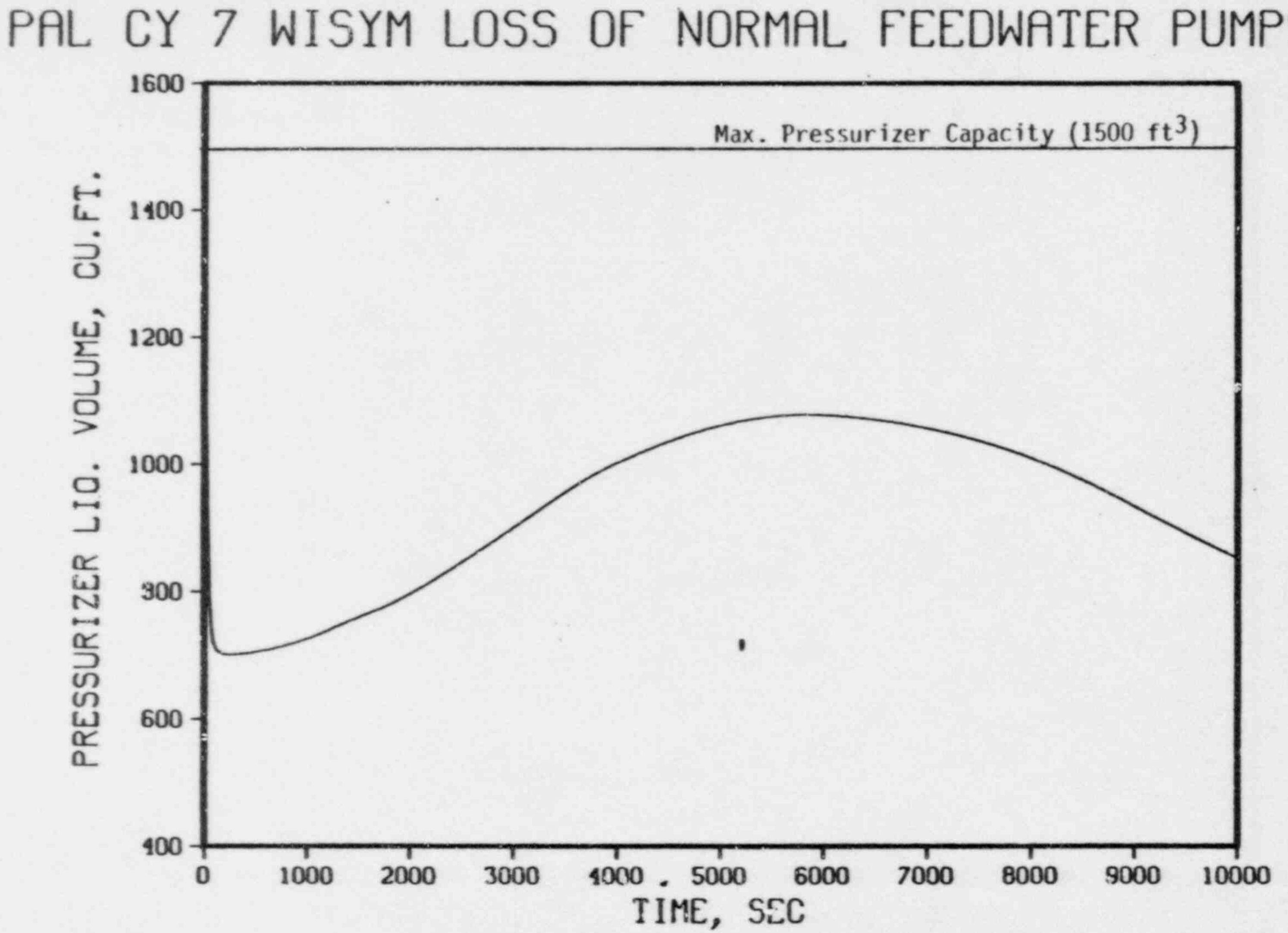


Figure 15.2.7-4 Minimum Transient Steam Generator Inventory, Symmetric Steam Generator Tube Plugging, Pumps on, Pressurizer Liquid Volume vs. Time-Loss of Normal Feedwater

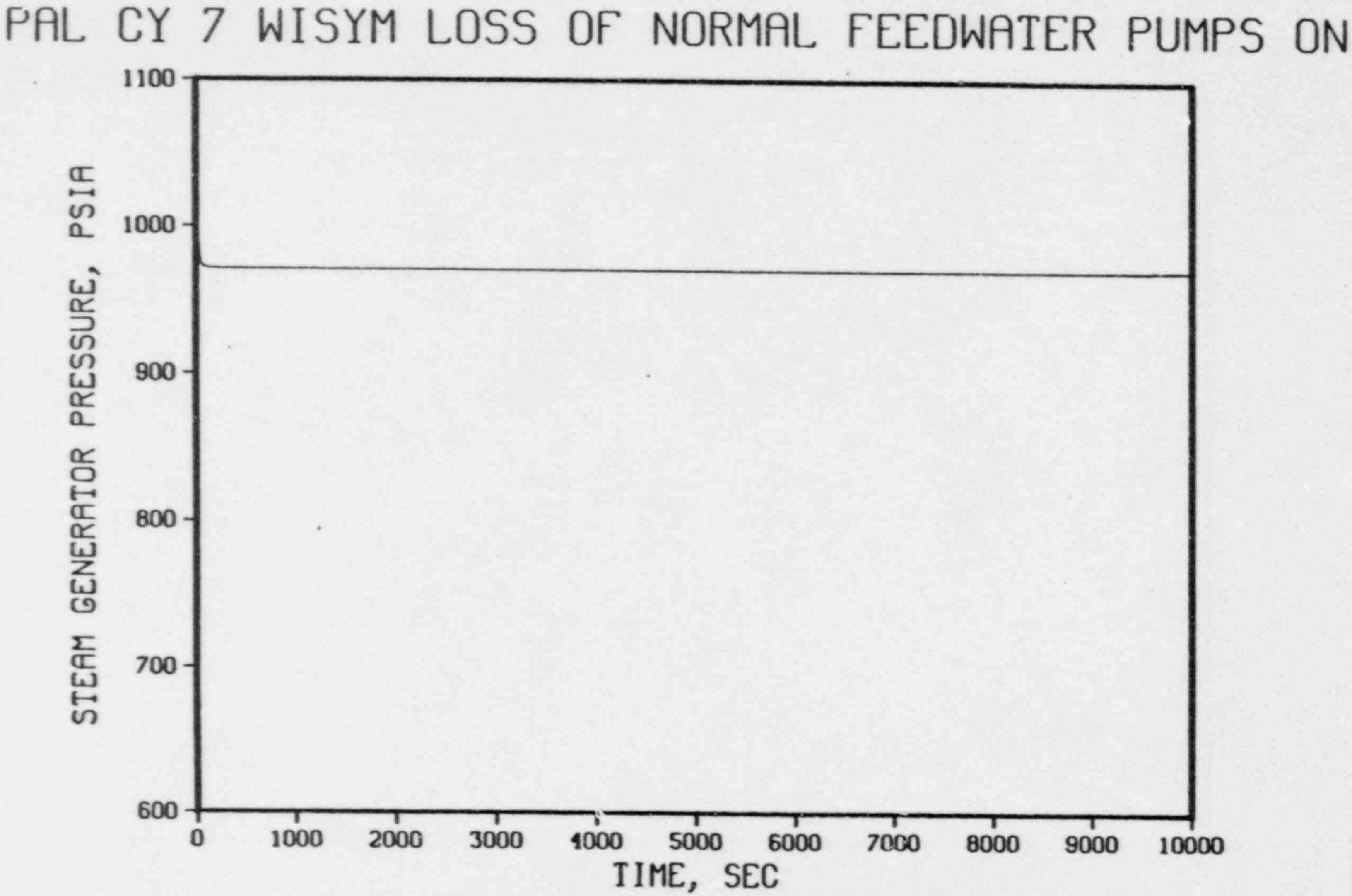


Figure 15.2.7-5 Minimum Transient Steam Generator Inventory, Symmetric Steam Generator Tube Plugging, Pumps on- Steam Generator Pressure vs. Time-Loss of Normal Feedwater

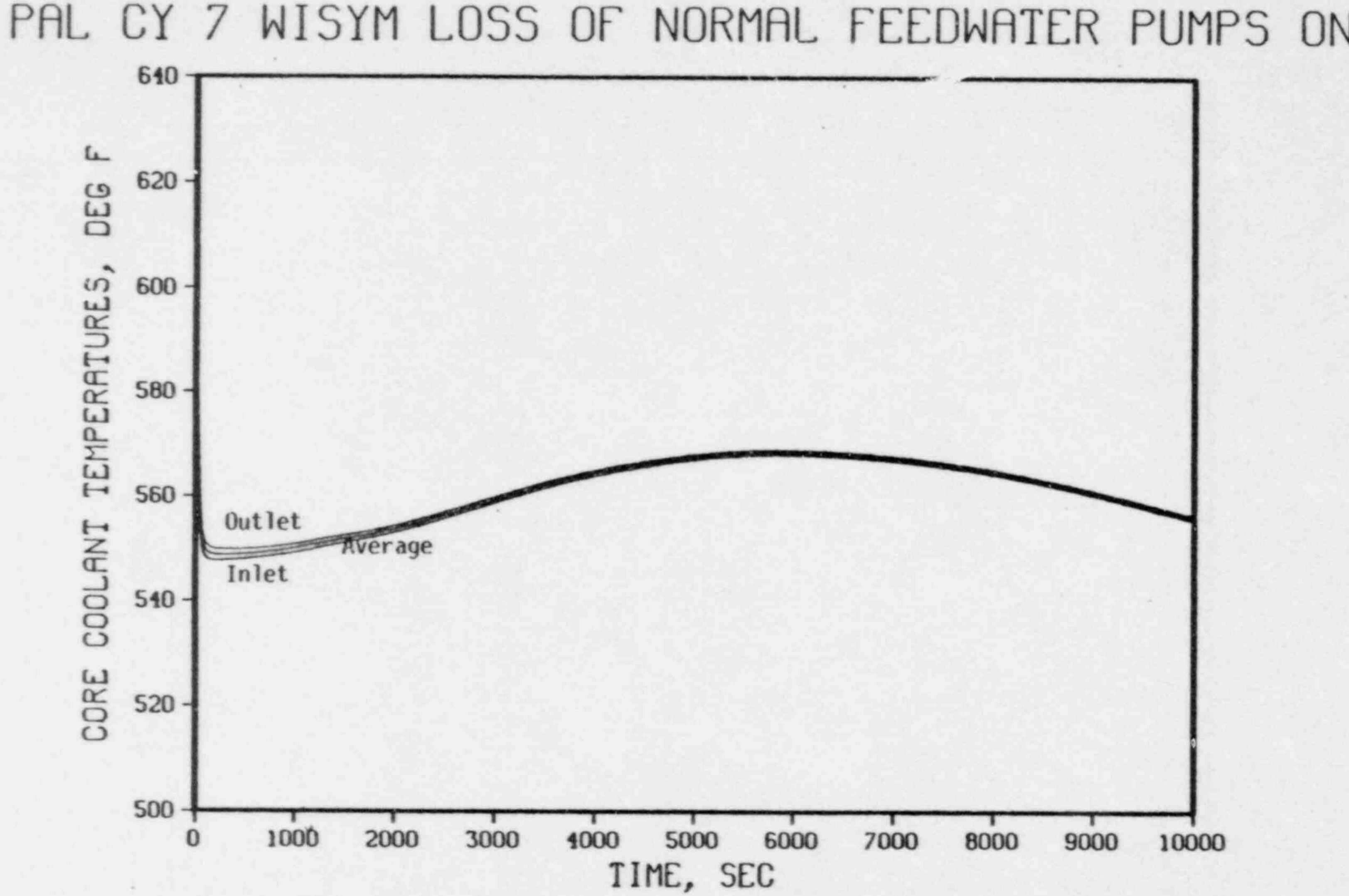


Figure 15.2.7-6 Minimum Transient Steam Generator Inventory, Symmetric Steam Generator Tube Plugging, Pumps on- Reactor Coolant Temperatures vs. Time-Loss of Normal Feedwater

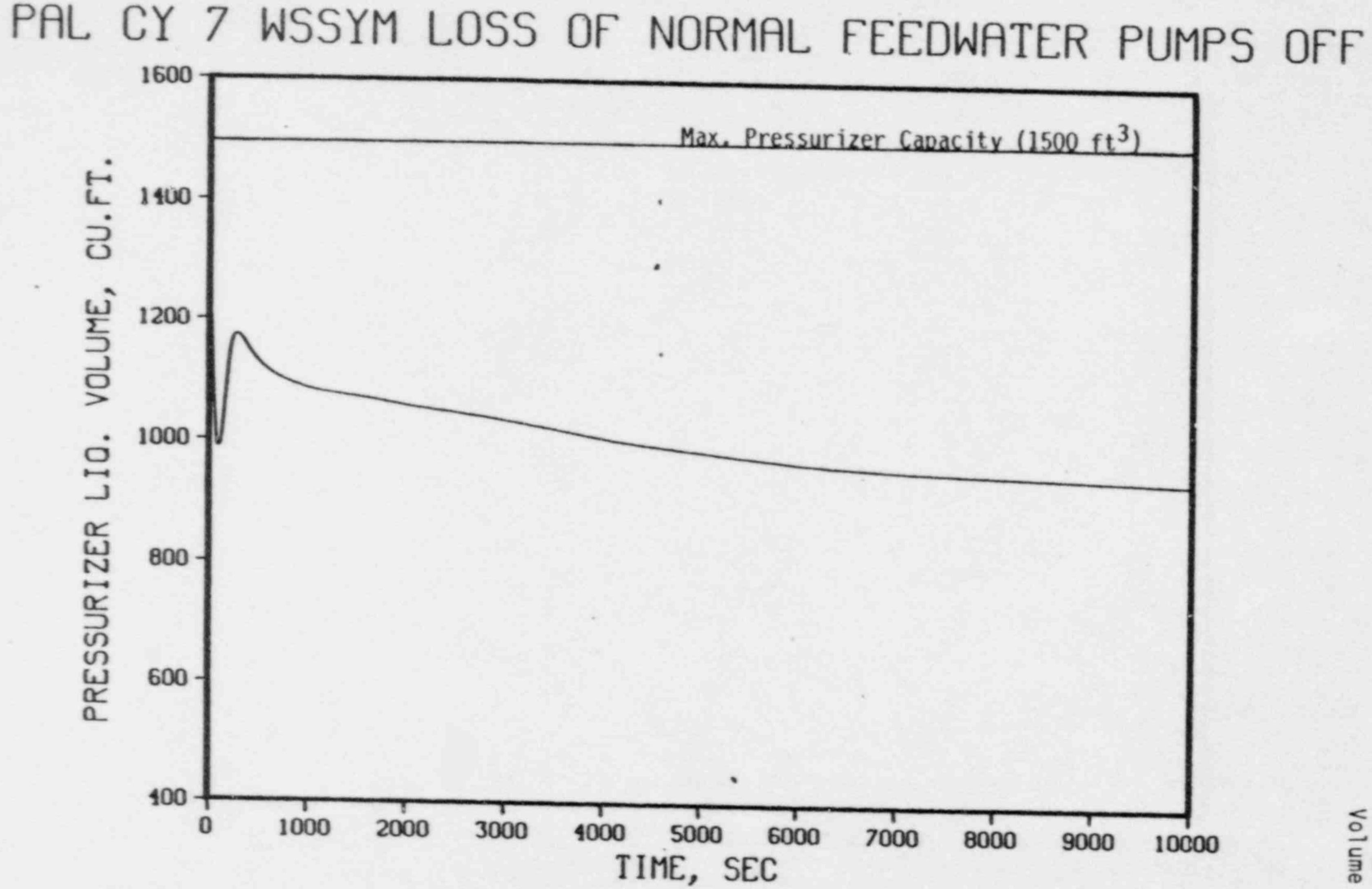


Figure 15.2.7-7 Maximum Pressurizer Pressure/Swell- Symmetric Steam Generator Tube Plugging, Pumps off- Pressurizer Liquid Volume vs. Time (0- 10,000 seconds) - Loss of Normal Feedwater

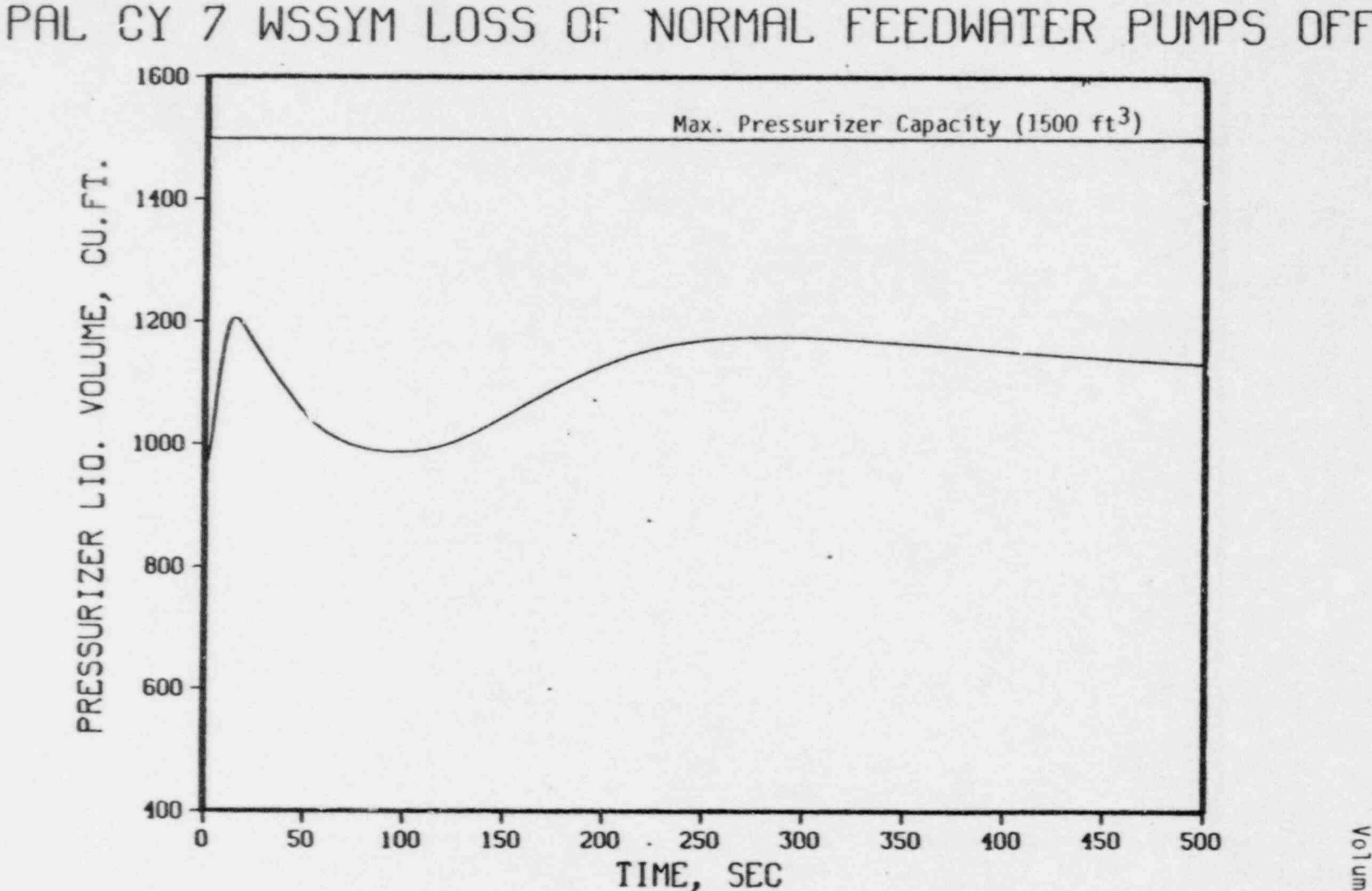


Figure 15.2.7-8 Maximum Pressurizer Pressure/Swell- Symmetric Steam Generator Tube Plugging, Pumps off- Pressurizer Liquid Volume vs. Time (0-500 seconds) - Loss of Normal Feedwater

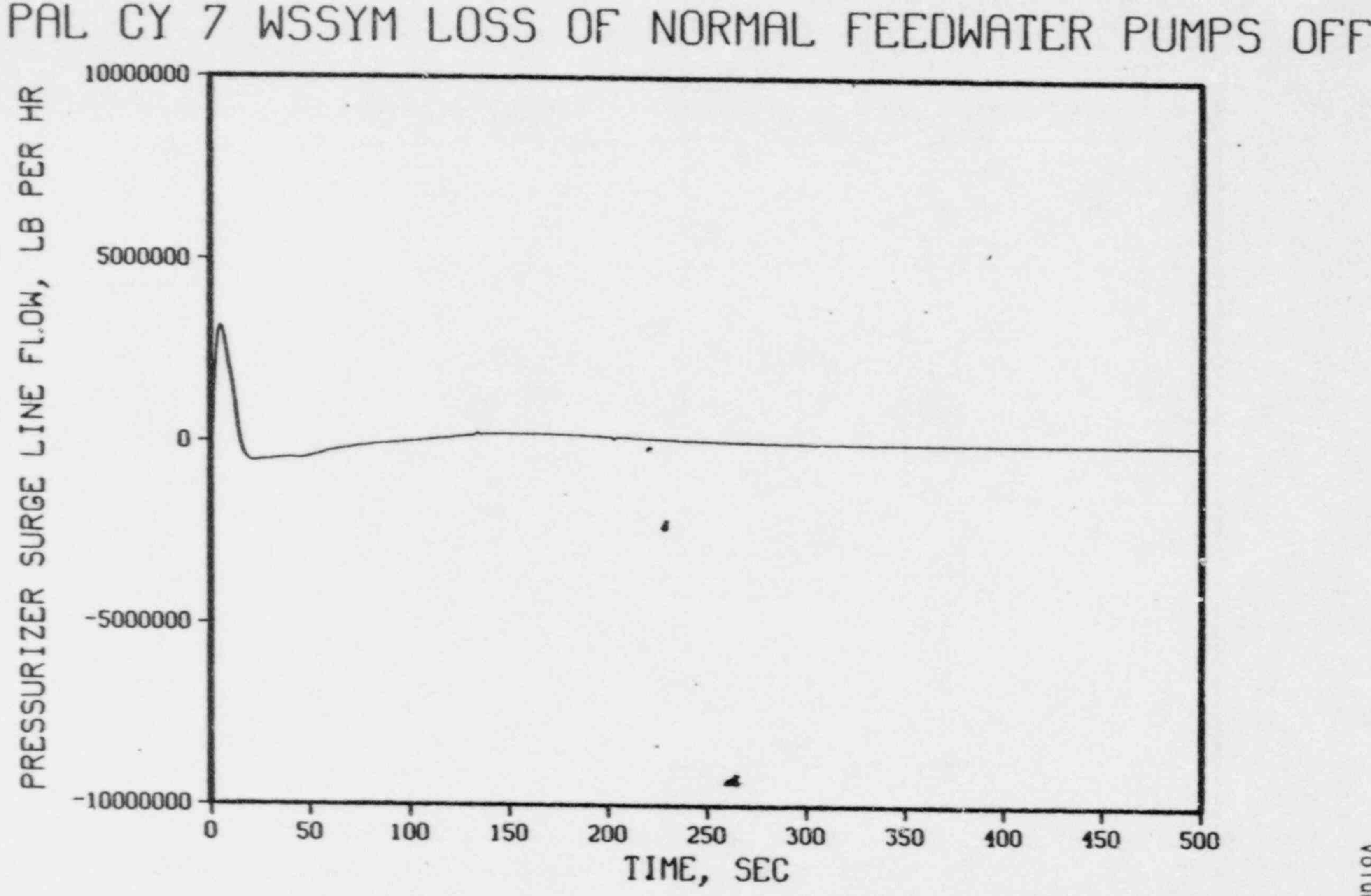


Figure 15.2.7-[illegible] Maximum Pressurizer Pressure/Swell-Symmetric Steam Generator Tube Plugging, Pumps off-Pressurizer Surge Line Flow Rate vs. Time (0-500 seconds)- Loss of Normal Feedwater

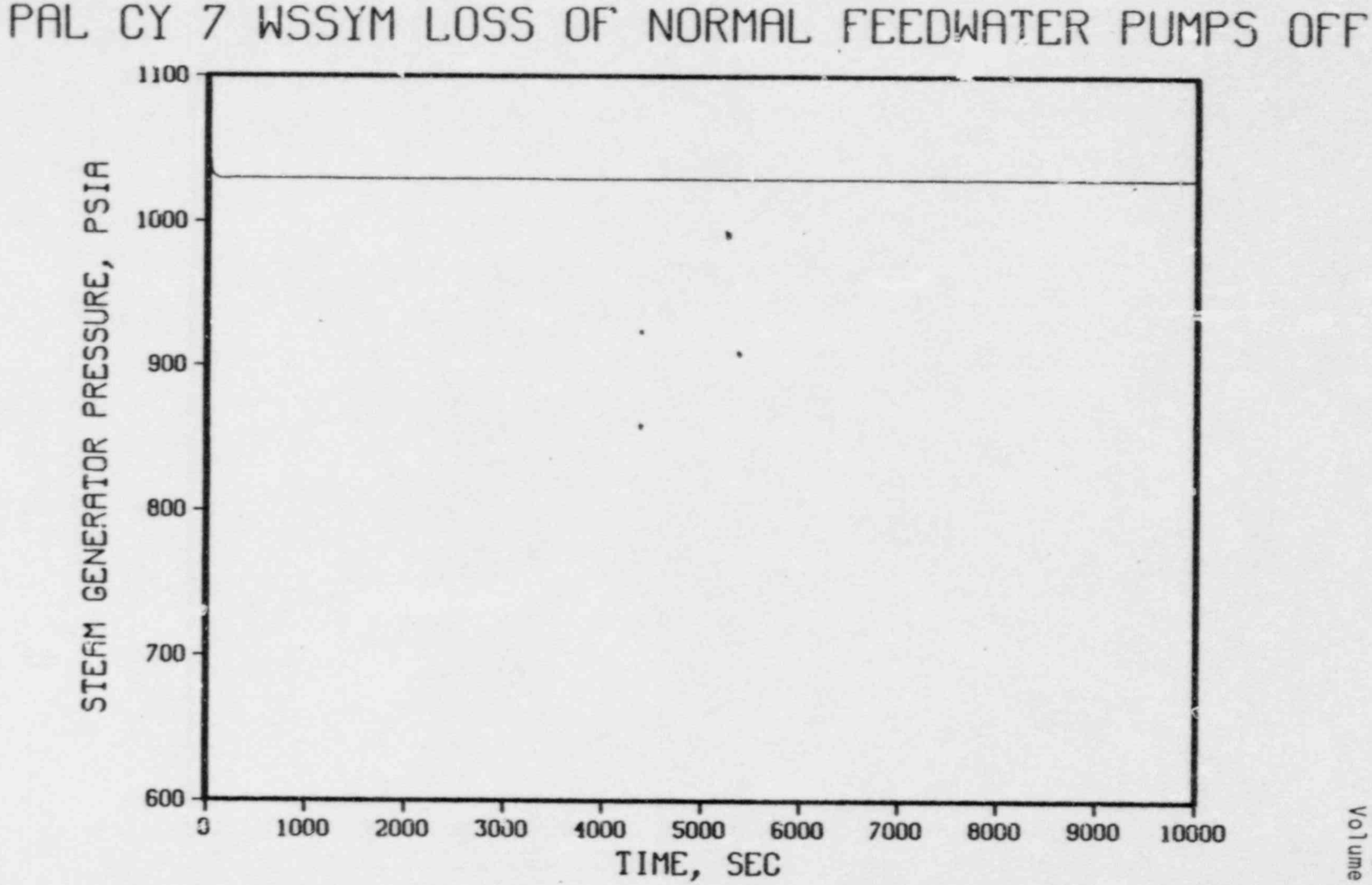


Figure 15.2.7-10 Maximum Pressurizer Pressure/Swell-Symmetric Steam Generator Tube Plugging, Pumps off-Steam Generator Pressure vs. Time (0-10,000 seconds)- Loss of Normal Feedwater

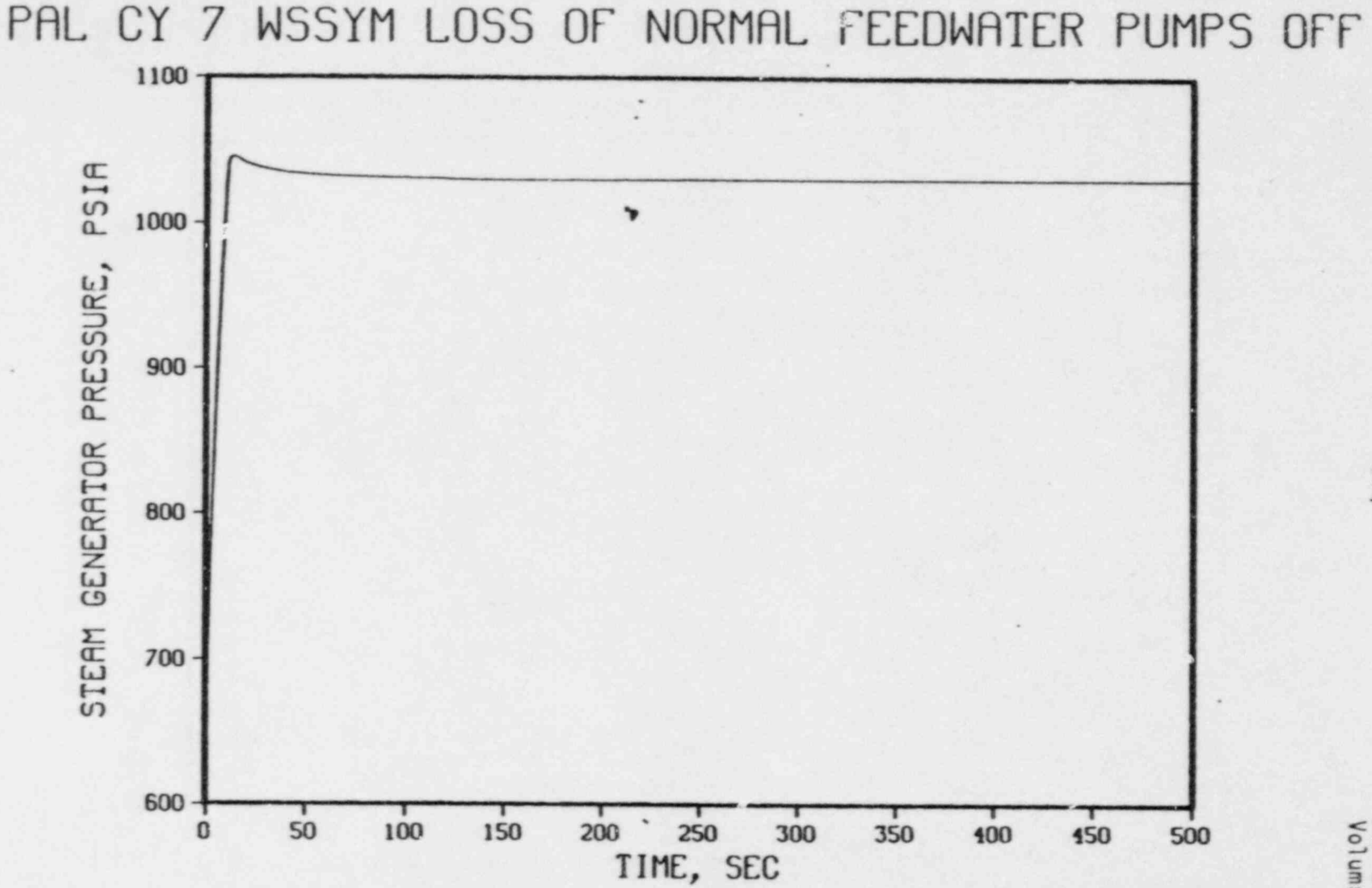


Figure 15.2.7-11 Maximum Pressurizer Pressure/Swell-Symmetric Steam Generator Tube Plugging, Pumps off- Steam Generator Pressure vs. Time (0-500 seconds) - Loss of Normal Feedwater

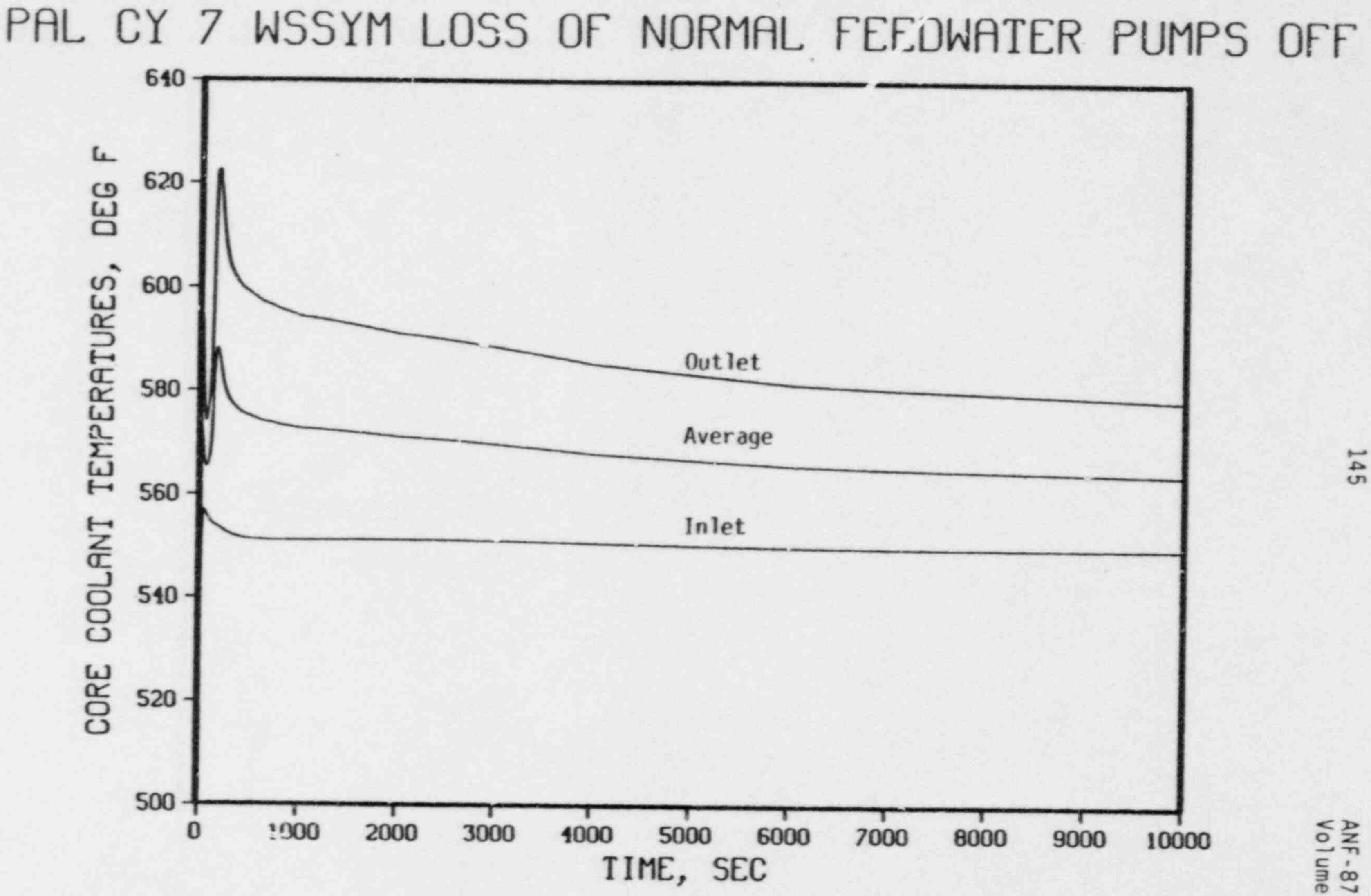


Figure 15.2.7-12 Maximum Pressurizer Pressure/Swell -Symmetric Steam Generator Tube Plugging, Pumps off- Reactor Coolant Temperatures vs. Time (0-10,000 seconds)- Loss of Normal Feedwater

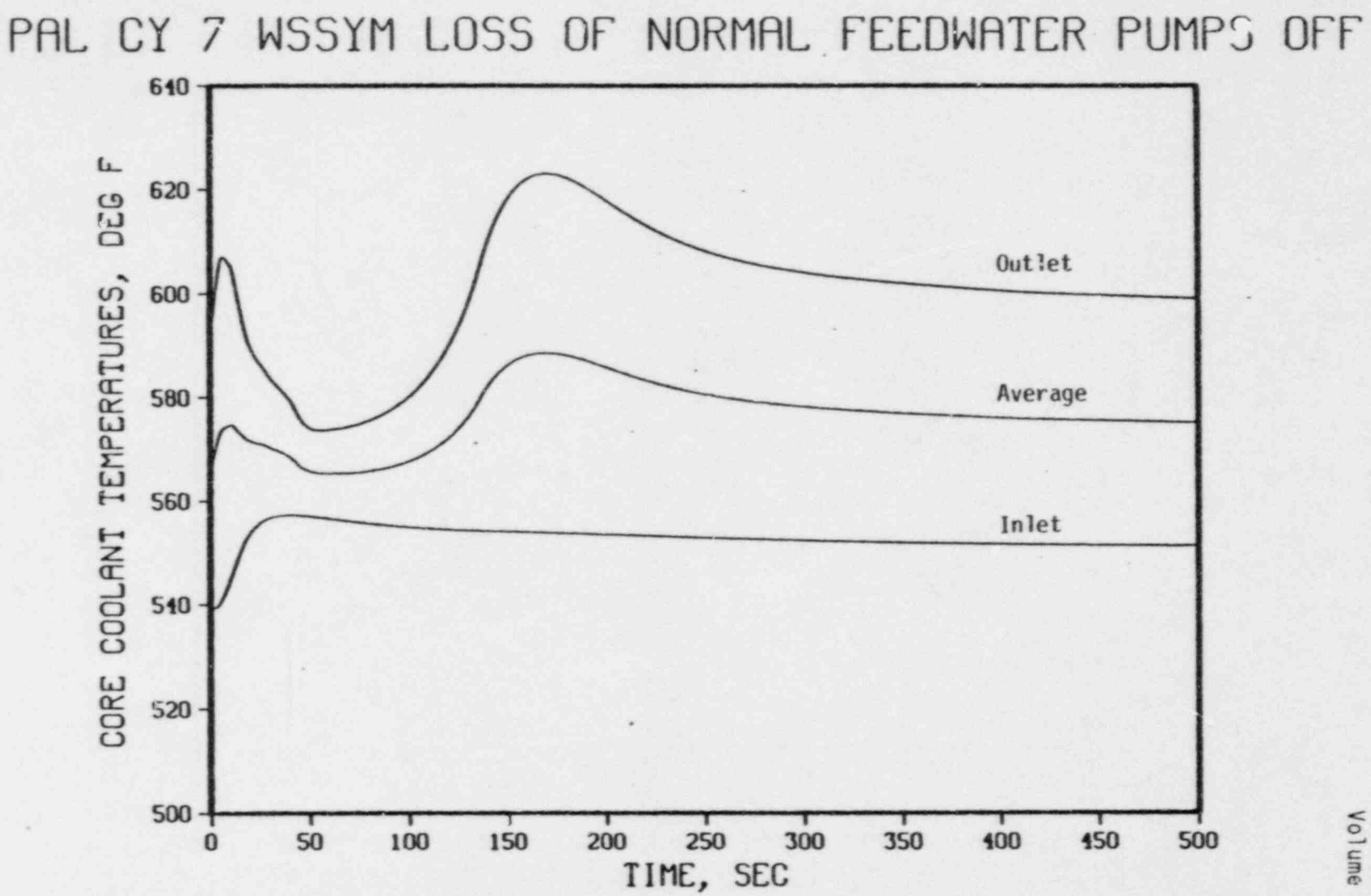


Figure 15.2.7-13 Maximum Pressurizer Pressure/Swell- Symmetric Steam Generator Tube Plugging, Pumps off-Reactor Coolant Temperatures vs. Time (0-500 seconds)- Loss of Normal Feedwater

15.3 DECREASED REACTOR COOLANT FLOW

15.3.1 LOSS OF FORCED REACTOR COOLANT FLOW

15.3.1.1 Identification of Causes and Event Description

This event is characterized by a total loss of forced reactor coolant flow which is caused by the simultaneous loss of electric power to all of the reactor primary coolant pumps. Following the loss of electrical power, the reactor coolant pumps begin to coast down.

If the reactor is at power when the event occurs, the loss of forced coolant flow causes the reactor coolant temperatures to rise rapidly. This results in a rapid reduction in DNB margin, and could result in DNB if the reactor is not tripped promptly. Also, as the reactor coolant temperatures rise the primary coolant expands, which causes an insurge into the pressurizer, a compression of the pressurizer steam space, and a rapid increase in reactor coolant system pressure. The primary system overpressurization will be mitigated by the action of the primary system safety valves and the reduction in core power following reactor trip.

Reactor trip signals are provided based on the reactor coolant pumps low reactor coolant loop flow.

The minimum DNBR is controlled by the interaction of the primary coolant flow decay and the core power decrease following reactor trip. The power to flow ratio initially increases, peaks, and then declines as the challenge to the SAFDLs is mitigated by the decline in core power due to the reactor trip. If a reactor trip can be obtained promptly, the power to flow ratio will first peak and then decrease during the transient such that the SAFDLs will be no longer challenged.

The pump coastdown characteristics and the timing of the reactor trip, trip delays and scram rod insertion characteristics are key parameters.

Natural circulation flow is developed in the primary system and the steam generators are available to remove the decay power. Therefore, long term cooling of the core can be achieved.

The primary concern with this event is the challenge to the SAFDLs. The event is analyzed to verify that the reactor protection system can respond fast enough to prevent penetration of the DNBR SAFDL.

This event is classified as a moderate frequency event (Table 15.0.1-1). The acceptance criteria are as described in 15.0.1.1. For this analysis, the systems challenged in this event are redundant; no single active failure in the RPS or ESF will adversely affect the consequences of the event. Long term recovery is provided by the auxiliary feedwater system, as demonstrated in the analysis of event 15.2.7.

15.3.1.2 Analysis Method

The overall response of the primary and secondary systems for this event is calculated by the PTSPWR2 computer code[10]. The MDNBR for the event is calculated using the thermal hydraulic conditions from the PTSPWR2 calculation as input to XCOBRA-IIIC[11].

The event is initiated by simultaneously tripping of all of the reactor coolant pumps. The pump coastdown is governed by a conservative estimate of the pump flywheel inertia, the homologous pump curves and the loop hydraulics. Reactor trip is delayed until the low reactor coolant loop flow signal is obtained. This trip setpoint is conservatively reduced to account for uncertainties in flow measurement.

15.3.1.3 Definition of Events Analyzed and Bounding Input

This event is analyzed from full power initial conditions with the reactor control rod system in manual. The core thermal margins are at a minimum at full power conditions. This is the bounding mode of operation for this event.

The following conservative conditions are established which minimize DNB for this event:

Table 15.3.1-0 Conservative Assumptions Used in the Loss of Forced Reactor Coolant Flow

Item	Biased Condition
Power	Rated +2%
Core Inlet temperature	Nom. +5°F
Primary pressure	Nom. -50 psi
Pressurizer level	Nom. -5% of pzr height
Pressurizer PORVs	Not Available
Pressurizer spray	Available with pressure setpoints biased downward by 50 psi
Pressurizer heaters	Backup heaters available
Pump flywheel inertia	Nom. -10%
Initial primary coolant flow rate	Nom. -3%
Reactor flow rate trip	Low flow -7%
Moderator temperature coefficient	1.2 BOC Nom.
Doppler	0.8 BOC Nom.
Pellet to clad heat transfer coef.	Nom. +20%
Clad to coolant heat transfer coef.	Nom. +20%

15.3.1.4 <u>Analysis of Results</u>

The transient is initiated by tripping all four primary coolant pumps. As the pumps coast down, the core flow is reduced, causing a reactor scram on low flow with rod insertion beginning at 1.74 seconds. The core flow is reduced to about 43% of initial in 10 seconds.

As the flow coasts down, primary temperatures increase. The average core temperature increases about 8°F before being turned around due to the power decrease following reactor scram. This increase in temperature causes a subsequent power rise due to moderator reactivity feedback. The power peaks at about 103.6% of rated.

The temperature increase also causes an insurge into the pressurizer and resultant pressurization of the reactor coolant system. The peak pressure was 2160.9 psi at 5.2 seconds. The primary challenge to DNB is from the decreasing flow rate and resulting increase in coolant temperatures.

The transient was analyzed from 102% power. Pressurizer pressure control system parameters were biased to minimize the pressure rise. A conservative value was used for the pump moment of inertia. Initial plant operating conditions assumed in the analysis are summarized on Table 15.3.1-1.

The transient response is shown in Figures 15.3.1-1 to 15.3.1-10, with an associated event summary given in Table 15.3.1-2. The minimum DNBR for this case is computed as 1.455.

Plotted variables are defined in Table 15.0.12-1.

15.3.1.5 Conclusion

The XNB critical heat flux safety correlation limit of 1.17 is not penetrated, so event results are acceptable with respect to the DNBR SAFDL. Peak pressurization is bounded by that calculated for the Loss of External Load (Pressurization Case), Event 15.2.1. Maximum peak pellet LHGR for this event is about 12.7 kW/ft, well below the incipient fuel centerline melt criterion of 21 kW/ft. Applicable acceptance criteria for the event are therefore met.

Table 15.3.1-1 Summary of Initial Conditions for Loss of Forced Reactor Coolant Flow

	Minimum Pressure Case
Power (MWt)	2580.6
Core Inlet Temperature (°F)	548.65
Pressurizer Pressure (psia)	2010
Pressurizer Level	Programmed Full Power Level Minus 5% of Height
Reactor Coolant System Flow Rate (lbm/hr)	116.6×10^6
Steam Dome Pressure (psia)	731.5

Table 15.3.1-2 Event Summary for the Loss of Forced Reactor Coolant Flow

Event	Time (sec)
Initiate Four-Pump Coastdown	0.00
Reactor Scram (Begin Rod Insertion)	1.74
Peak Power	2.29
Minimum DNBR	2.65
Peak Core Average Temperature	3.49

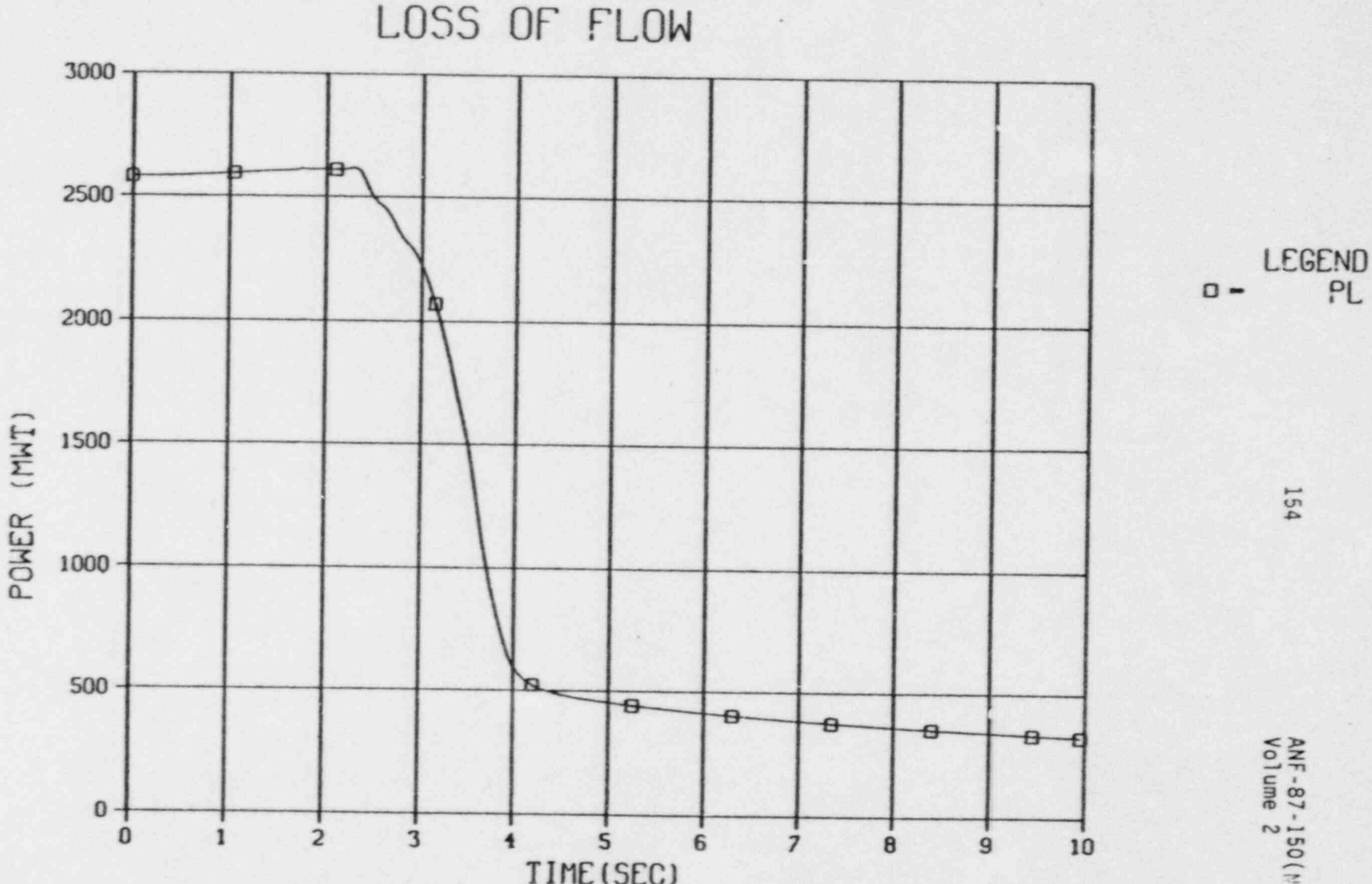


Figure 15.3.1-1 Reactor Power Level for Loss of Forced Flow

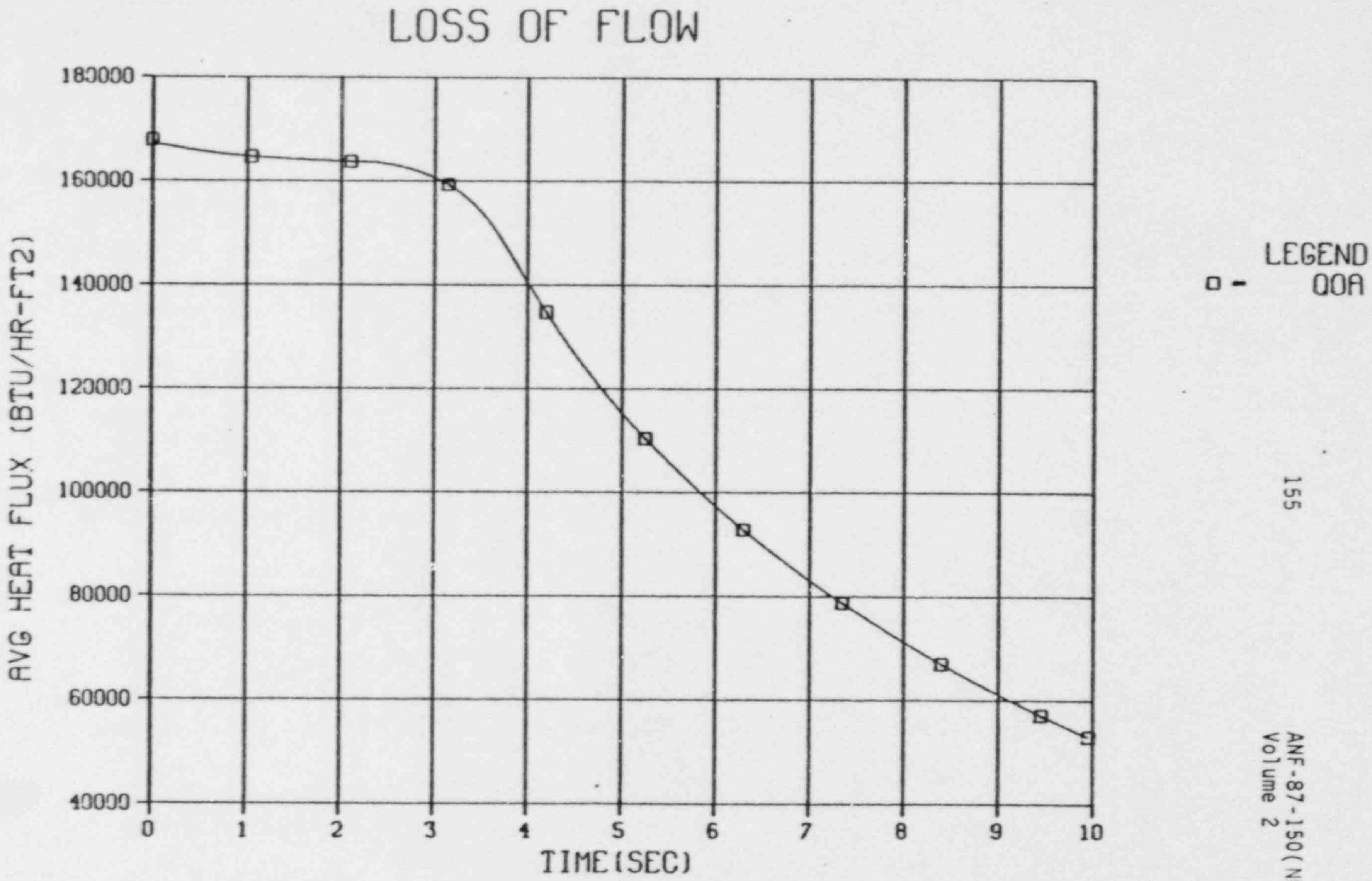


Figure 15.3.1-2 Core Average Heat Flux for Loss of Forced Flow

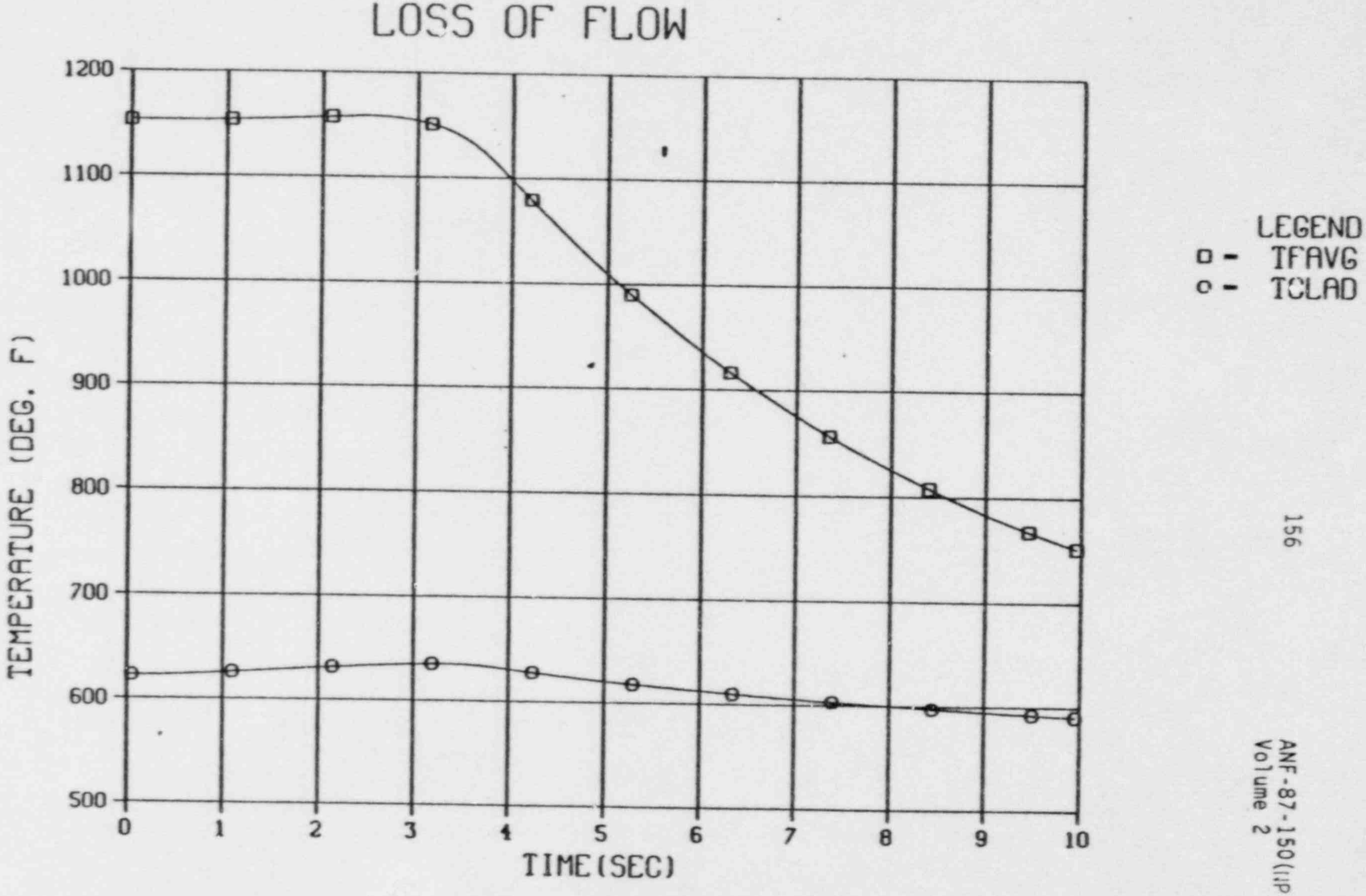


Fig. [illegible] 3.1-3 Average Fuel and Clad Temperatures for Loss [illegible] Forced Flow

11.41.27 THUR 12 DEC, 1985 JOB-TGPMLGZ , U C C DISSPLA VER 8.2

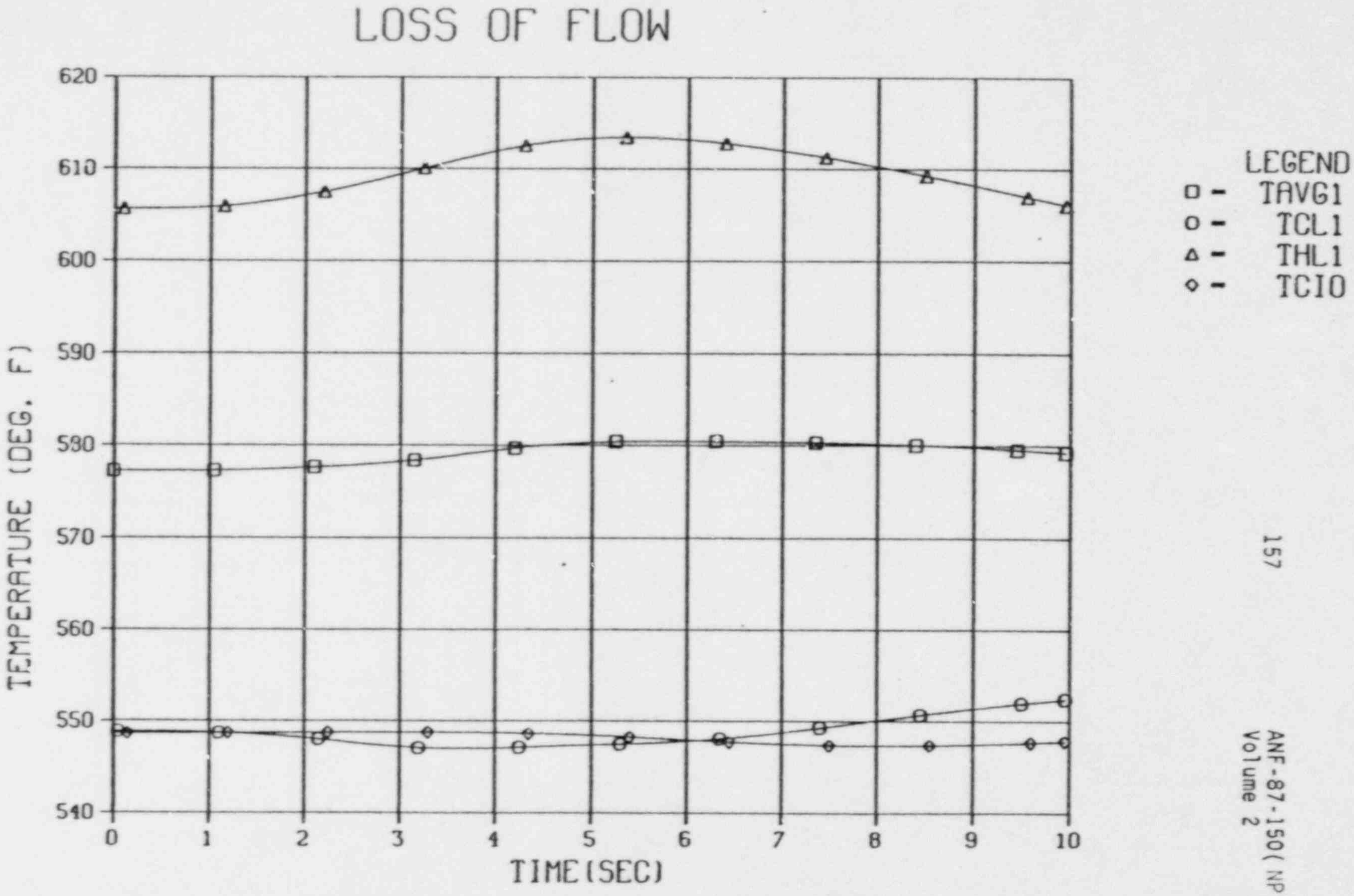


Figure 15.3.1-4 Reactor Coolant System Temperatures for Loss of Forced Flow

PLOT 1 16.17.19 MON 2 DEC, 1985 JOB=MOGRMC3 , U G C DISSPLA VER 8.2

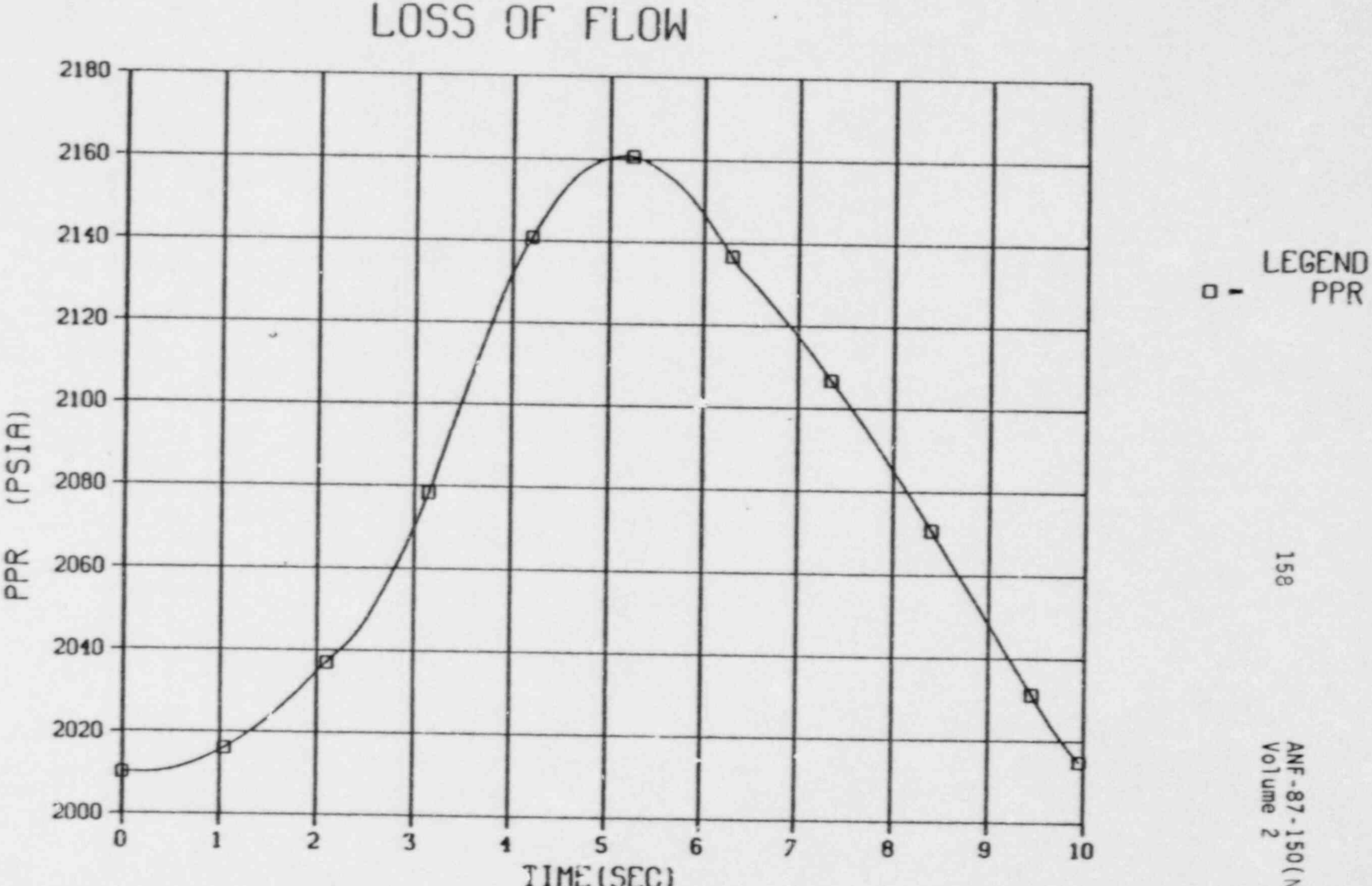


Figure 15.3.1-5 Pressurizer Pressure for Loss of Forced Flow

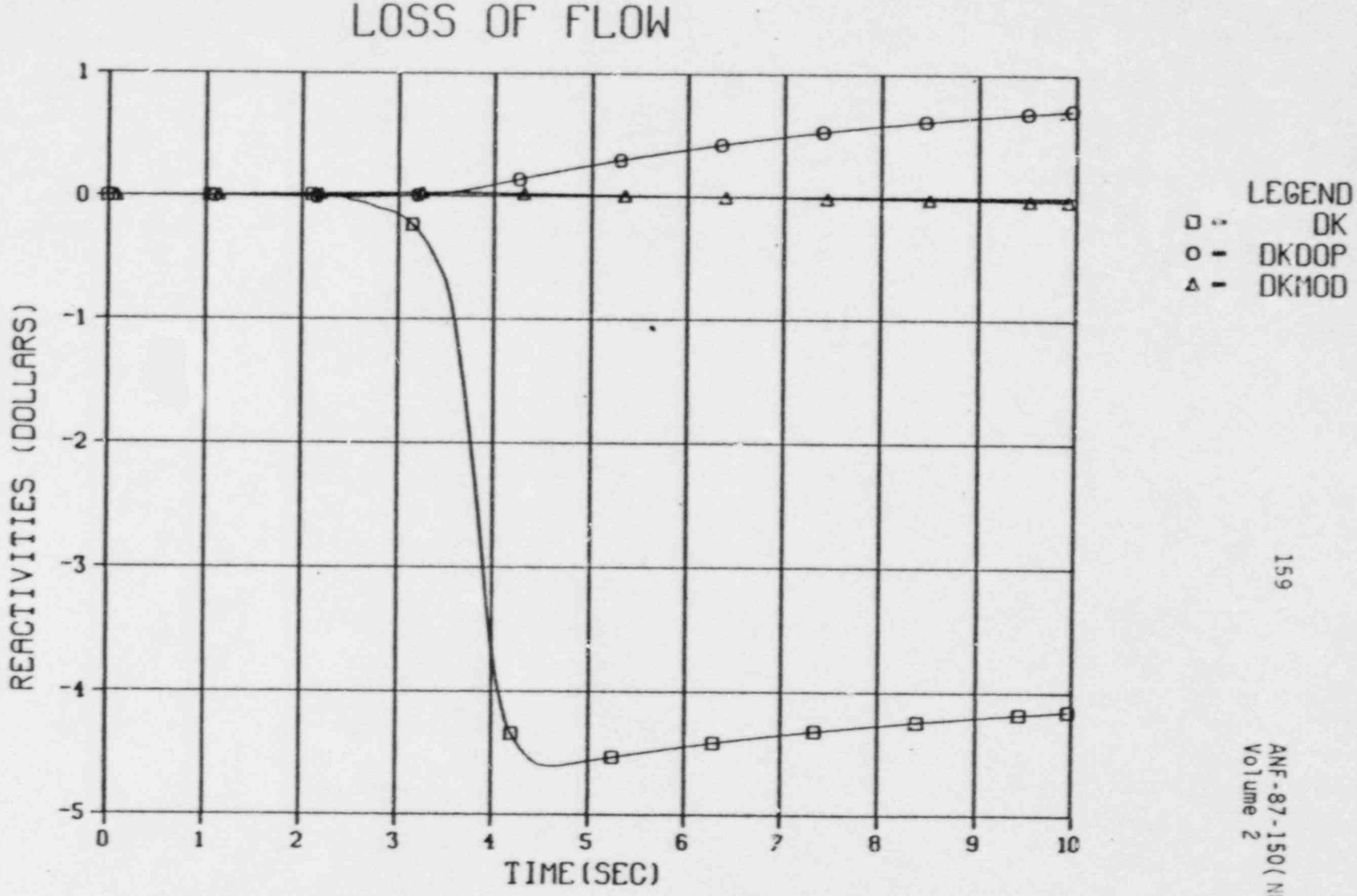


Figure 15.3.1-6 Reactivities for Loss of Forced Flow

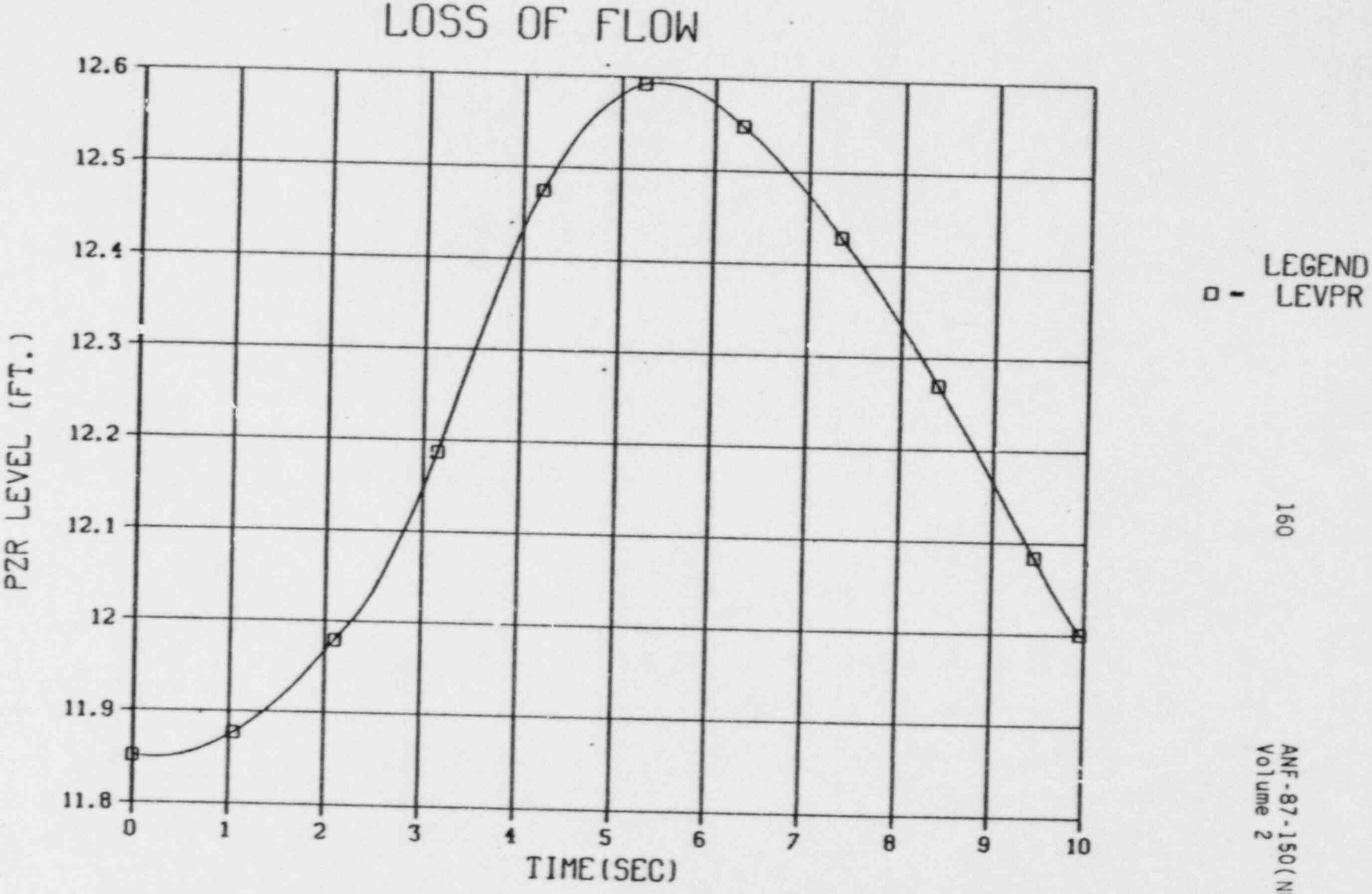


Figure 15.3.1-7 Pressurizer Liquid Level for Loss of Forced Flow

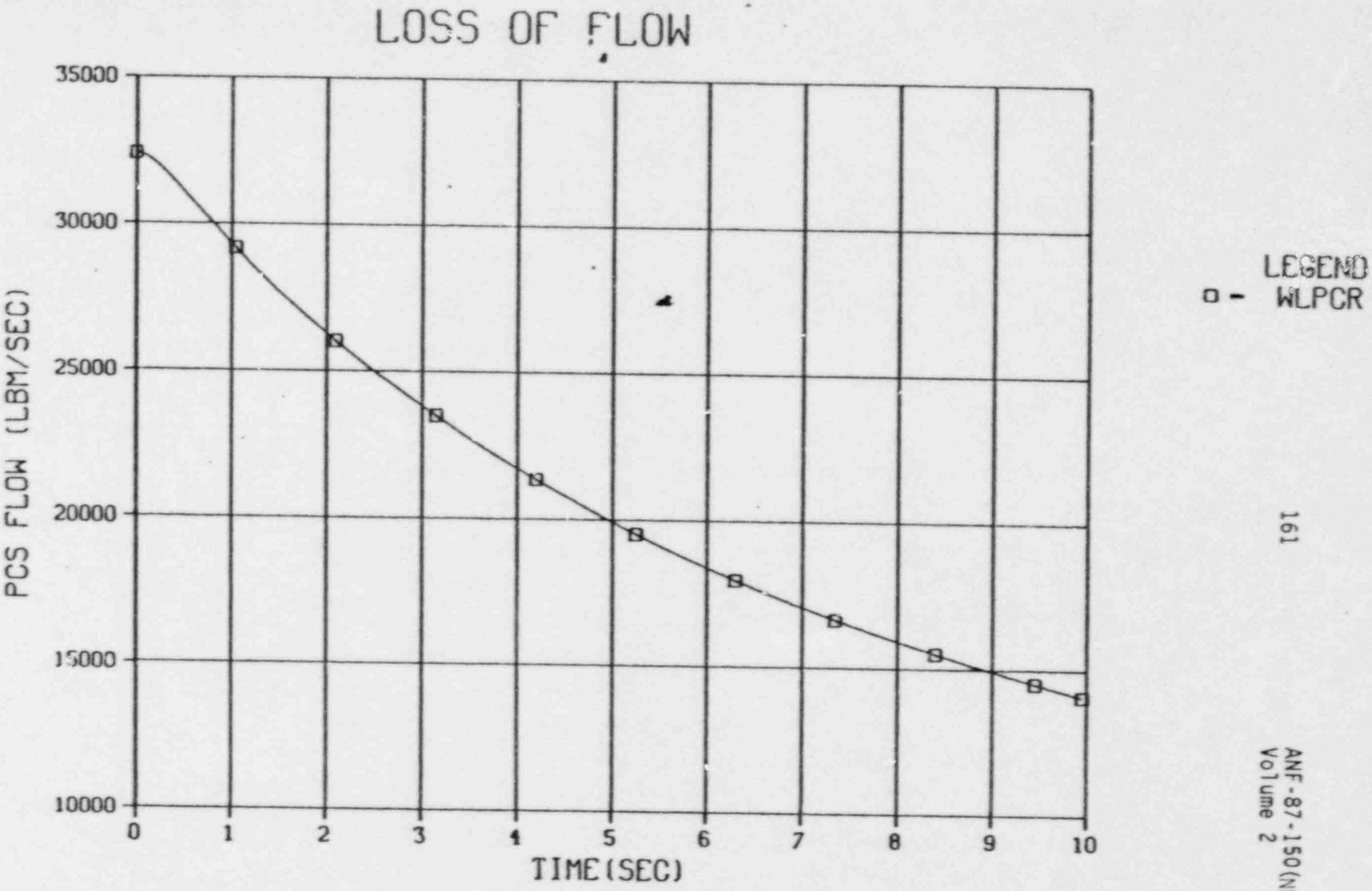


Figure 15.3.1-8 Primary Coolant Flow Rate for Loss of Forced Flow

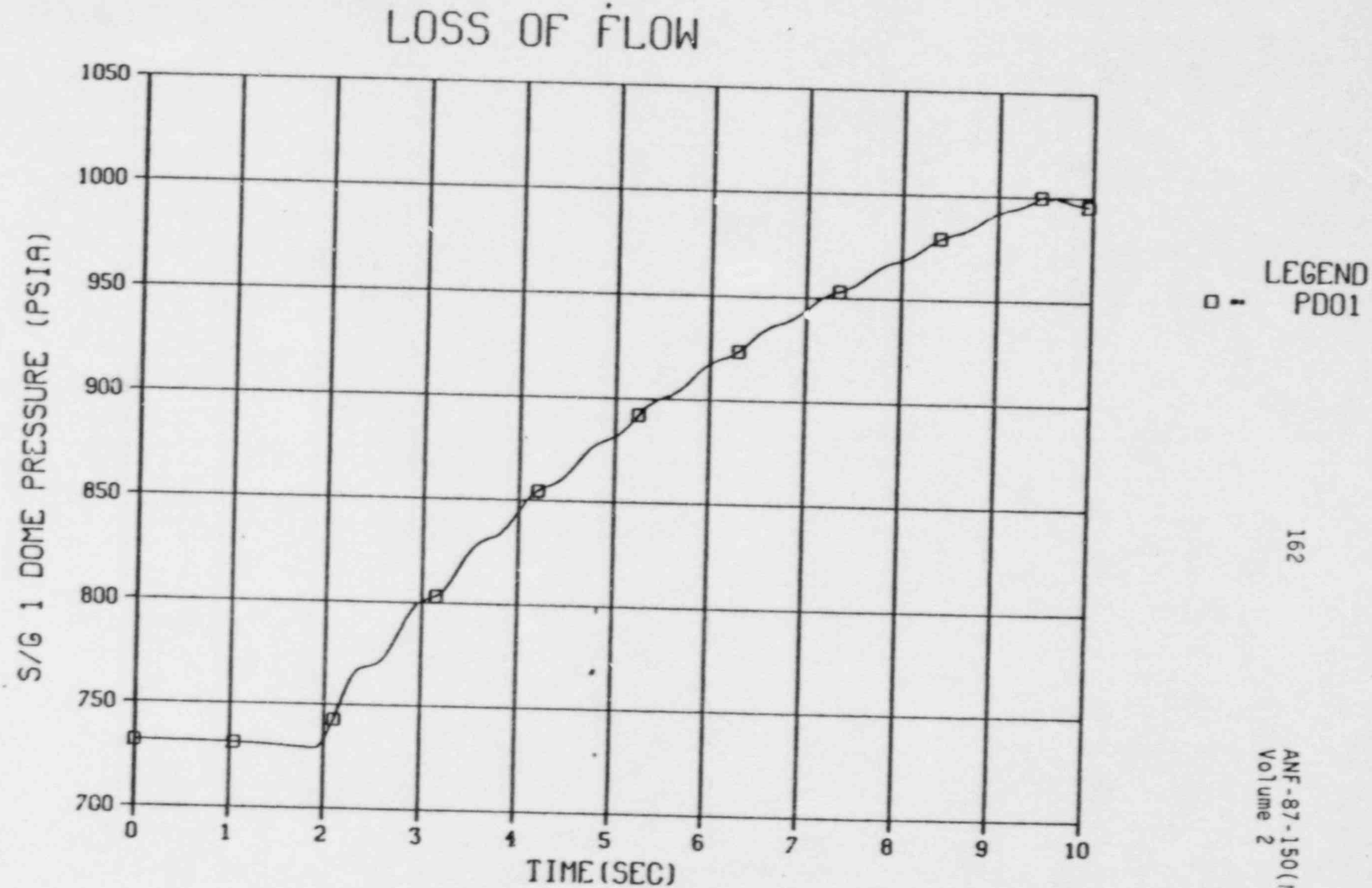


Figure 15.3.1-9 Secondary Pressure for Loss of Forced Flow

PLOT 7 12.34.10 MON 2 DEC, 1985 U C C DISSPLA VER 8.2

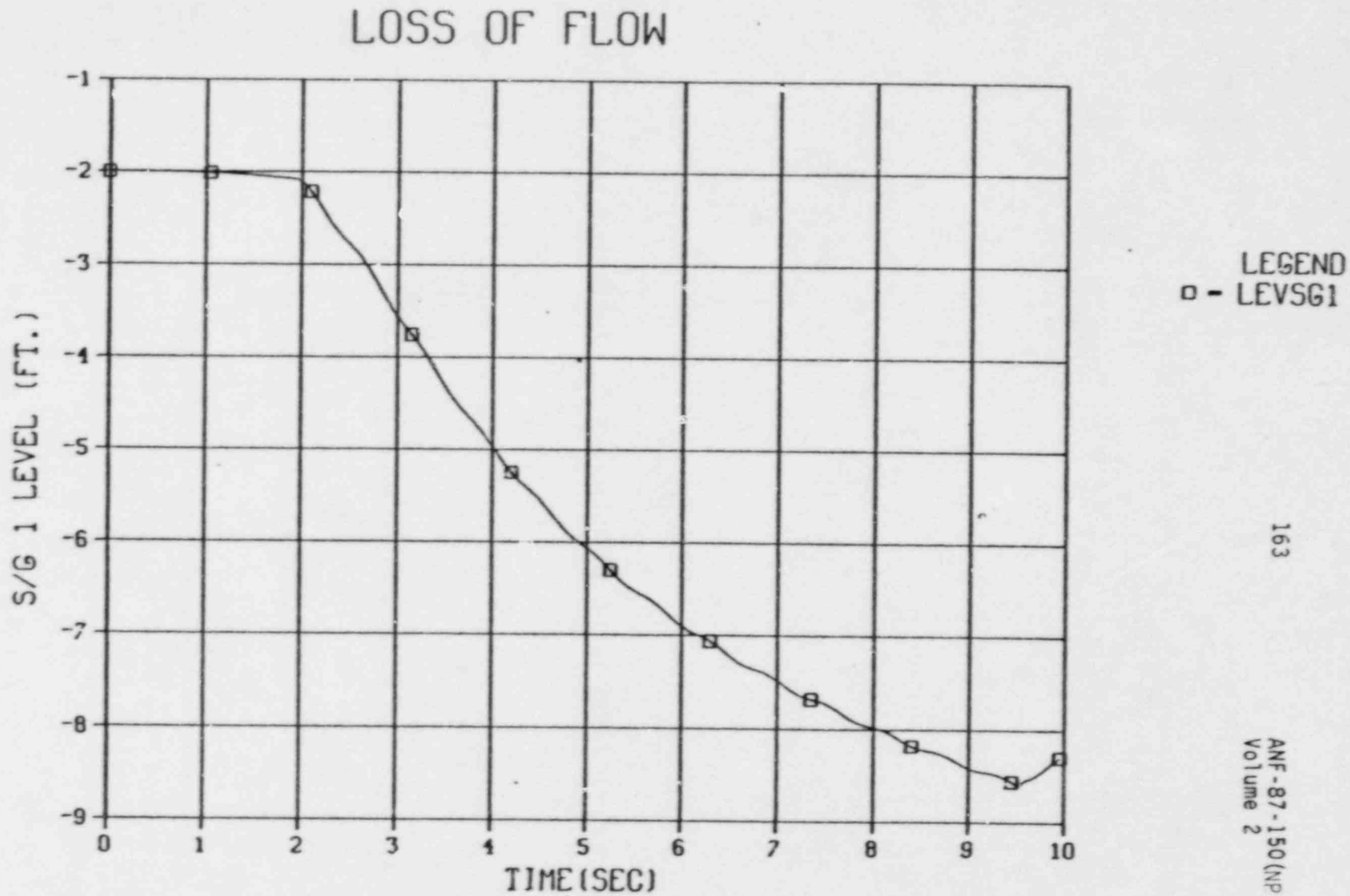


Figure 15.3.1-10 Steam Generator Liquid Level for Loss of Forced Flow

15.3.3 REACTOR COOLANT PUMP ROTOR SEIZURE

15.3.3.1 Identification of Causes and Event Description

The locked rotor event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal.

Following the reactor trip, the heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, the coolant temperatures will begin to rise, causing expansion of the primary coolant and consequent pressurizer insurge flow and RCS pressurization. As the pressure increases, pressurizer sprays and safety valves would act to mitigate the pressure transient.

The rapid reduction in core flow and the increase in coolant temperature may seriously challenge or penetrate the DNBR SAFDL. The event is thus evaluated to assess the DNBR challenge. The fuel centerline melt SAFDL is not seriously challenged by the small power increase typical of this event. RCS pressurization criteria have not been approached in ANF analyses of this event; no case addressing pressurization is therefore performed.

The event as simulated is structured to provide a bounding determination of MDNBR for both the locked rotor and broken shaft (15.3.4) events.

The reactor pump rotor seizure is an infrequent event (Table 15.0.1-1). The acceptance criteria for this event are presented in Section 15.0.1.1. For this analysis, the systems challenged in this event are redundant; no single active failure in the RPS or ESF will adversely affect the consequences of the event. The auxiliary feedwater pumps will provide cooling capability after scram, as demonstrated in event 15.2.7, Loss of Normal Feedwater.

15.3.3.2 Analysis Method

The MDNBR is calculated using biased input to the XCOBRA-IIIC code(11). The boundary condition input for inlet flow rate, temperature and pressure are chosen to minimize DNBR. The pressure rises during this event, so the minimum initial pressure is chosen. The core inlet temperature remains constant during the short time interval before the scram on low reactor coolant flow. Therefore, the maximum initial core inlet temperature is used. The core inlet flow rate is determined by calculating the steady-state flows in each of the loops with one pump locked. This calculation uses the system loss coefficients and homologous pump curves(6) from the PTSPWR2 analyses.

15.3.3.3 Definition of Events Analyzed and Bounding Input

Two cases are analyzed for this event. The first case uses bounding input and the calculated steady-state, locked rotor flow rate. The second case uses bounding input and the three-pump flow rate specified in the Technical Specifications.

The bounding operating mode for this event is full power initial conditions. The conservative conditions used in these analyses are:

Table 15.3.3-0 Conservative Assumptions Used in the Reactor Coolant Pump Rotor Seizure Event

Item	Biased Condition
Power	Rated +2%
Core Inlet Temperature	Nominal +5°F
Pressure	Nominal -50 psi

15.3.3.4 Analysis of Results

The first single locked rotor case is analyzed using the calculated value of core flow. Assuming the locked pump loss coefficient given by the homologous curves at zero pump speed, the core flow is 78% of the nominal full-power, four-pump operation value. The second case is analyzed at 74.7% flow as specified in the Technical Specifications (Reference 7, page 2-7). Plant conditions used are summarized in Table 15.3.3-1.

The XCOBRA-IIIC calculated MDNBRs are 1.409 and 1.341 for Case 1 and Case 2, respectively.

15.3.3.5 Conclusion

The XNB critical heat flux correlation safety limit of 1.17 is not penetrated. Therefore, no fuel failures are expected for this infrequent event. System pressurization is less than calculated for Event 15.2.1, Loss-of-Load. Thus, applicable acceptance criteria for this event are met.

Table 15.3.3-1 Summary of Conditions Used in MDNBR Calculations for the Reactor Coolant Pump Rotor Seizure Event

Condition	Case 1	Case 2
Power, MWt	2580.6	2580.6
Core Inlet Temperature, °F	548.65	548.65
Pressurizer Pressure, psia	2010	2010
Reactor Vessel Flow Rate, lbm/hr	90.9×10^6	87.1×10^6

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 UNCONTROLLED CONTROL ROD ASSEMBLY (CRA) WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

15.4.1.1 Identification of Causes and Event Description

This event is initiated by the uncontrolled withdrawal of a control rod bank, which results in the insertion of positive reactivity and consequently a power excursion. It could be caused by a malfunction in the reactor control or rod control systems. The consequences of a single bank withdrawal from reactor critical, hot standby, and hot shutdown (subcritical) operating conditions are considered in this event category; the consequences at rated power and initial operating conditions are considered in Event 15.4.2.

The control rods are wired together into preselected bank configurations. These circuits prevent the control rods from being withdrawn in other than their respective banks. Power is supplied to the banks in such a way that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence.

The reactivity insertion rate is rapid enough that very high neutron powers are calculated, but of short enough duration that excessive energy deposition does not occur. Rod surface heat flux lags the neutron power but still approaches a significant fraction of full power. Because the event is very rapid, primary coolant temperature lags behind power. The reactivity insertion rate is initially countered by the fuel temperature reactivity (Doppler) coefficient followed by trip and rod insertion.

The power transient (as well as the control rod withdrawal) is eventually terminated by the reactor protection system on one of the following signals:

(1) Nonsafety grade high rate-of-change of power trip, .0001% to 15% power

(no credit taken);

(2) Variable overpower trip;

(3) Thermal margin/low pressure trip;

(4) High pressurizer pressure trip; or

(5) High rate-of-change of power alarms, which initiate Rod Withdrawal Prohibit Action (no credit taken).

Further protection is provided by the Doppler reactivity feedback in the fuel and by available DNBR margin between the initial operating condition and the XNB correlation thermal limit.

For reasons described in the disposition-of-events[3] this transient is classified as an Infrequent event (Table 15.0.11). The acceptance criteria are as described in Section 15.0.1.1 with the addition of fuel centerline melt criterion. For this analysis, the systems challenged are redundant; no single active failure in the RPS or ESF will adversely affect the consequences of the event.

15.4.1.2 Analysis Method

The analysis is performed using the PTSPWR2[10] and XCOBRA-IIIC[11] codes. The PTSPWR2 code models the salient system components and calculates reactor power, fuel thermal response, surface heat transport and fluid conditions, including coolant flow rate, temperature and primary pressure. The core boundary conditions are then input into XCOBRA-IIIC to obtain the MDNBR.

15.4.1.3 Definition of Events Analyzed and Bounding Input

One case is analyzed for three pump operation. The case input and initial conditions bound reactor critical, hot standby and hot shutdown modes. The lowest initial power yields the maximum margin to trip, and hence maximum time for withdrawal to trip. This yields the largest prompt multiplication and maximizes overshoot past trip. The power used conservatively bounds the possible initial power in critical and hot shutdown operation. Maximum coolant temperature is used, since it minimizes DNBR. The biases for core age and the pellet-to-cladding heat transfer coefficient are selected to minimize Doppler feedback. Consistent beginning of cycle parameters are used.

To reduce DNBR, maximum radial peaking and minimum core flow rate are chosen. The results for three-pump operation bound those with four-pump operations.

Conservative conditions are established for the analysis:

Table 15.4.1-0 Conservative Assumptions for the Uncontrolled Bank Withdrawal from Low Power Event

Control	Manual
Core power	10^{-4}% of rated
Core Inlet temperature	Nom. +5°F
Primary Pressure	Nom. -50 psi
Primary Coolant Pumps Operating	3
Reactivity Insertion Rate	6.22×10^{-4} $\Delta\rho/s$
Reactor Trip	35.5% of rated (30% VHP setting + 5.5% uncer.)
Moderator Coefficient	1.2 BOC
Doppler Coefficient	0.8 BOC
Pellet-to-Clad HTC	Nom. +20%

15.4.1.4 Analysis of Results

The event is initiated with control bank withdrawal. At approximately 12 seconds reactor power increases. The peak nuclear power of 6648.1 MWt is reached at 12.8 seconds. The rapid power increase results in a fuel temperature increase and negative Doppler reactivity feedback which limits the peak power. The trip signal occurs at 12.3 seconds on the high neutron flux trip with rod insertion beginning at 12.8 seconds. A peak surface heat flux equivalent to 92% of rated power occurs at 13.9 seconds. The MDNBR calculated for the event is 1.036, which is below the 1.17 95/95 DNB safety limit for the XNB critical heat flux correlation. The percent of the core experiencing boiling transition is less than 2.3%. The radiological offsite doses for this event are about 20% of the doses calculated for a control rod ejection accident in which 12.2% fuel failure was predicted to occur (Event 15.4.8). The offsite radiological doses for the uncontrolled bank withdrawal from low power are less than 10% of the 10CFR100 limits.

Initial conditions employed in the simulation are listed in Table 15.4.1-1 and conditions used in the MDNBR calculation are given in Table 15.4.1-2. An event summary is presented in Table 15.4.1-3. Transient results are plotted in Figures 15.4.1-1 through 15.4.1-10. Figures 15.4.1-1 through 15.4.1-10 are plotted to 15 seconds, even though PTSPWR2 was run for 30 seconds. The limiting MDNBR occurs within the initial 15 second period.

Plotted variables are defined in Table 15.0.12-1.

15.4.1.5 Conclusions

In this infrequent event, only a small fraction of the core is calculated to experience boiling transition. Possible radiological releases are less than 10% of the 10 CFR 100 guidelines. For four primary coolant pump operations no fuel failures would be predicted to occur. Therefore, this event meets the applicable acceptance criteria.

Table 15.4.1-1 Initial Conditions for the Uncontrolled Bank Withdrawal from Low Power Event

Power, MWt	2.530×10^{-3}
Core Inlet Temperature, °F	537
Vessel Flow Rate***, lb/hr	87.1×10^{6}
Pressurizer Pressure, psia	2010

***3 pumps operating.

Table 15.4.1-2 Summary of Conditions used in MDNBR Calculations for the Uncontrolled Bank Withdrawal from Low Power Event

Peak Heat Flux in the Core, Btu/hr-ft^2	513,000
Core Inlet Temperature, °F	537.2
Vessel Flow Rate, lbm/hr	86.7×10^6
Pressurizer Pressure, psia	2010

Table 15.4.1-3 Event Summary for the Uncontrolled Bank Withdrawal from Low Power Event

Event	Time (sec)
Bank Withdrawal begins	0.00
High Power Trip Setpoint reached	12.28
Peak Nuclear Power occurs	12.82
Scram Reactivity Insertion begins	12.88
Peak Core Heat Flux occurs	13.90
MDNBR occurs	13.78

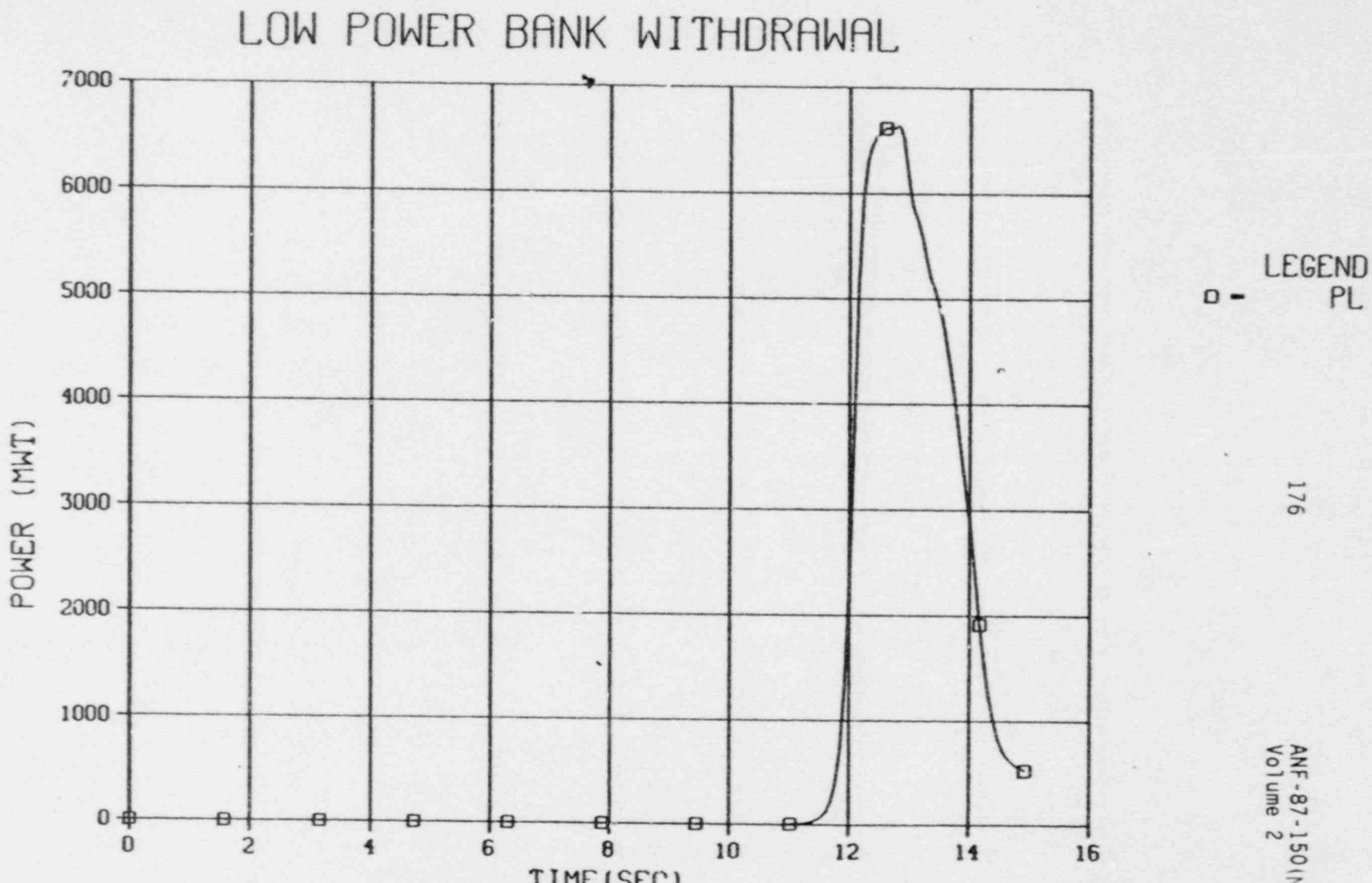


Figure 15.4.1-1 Reactor Power Level for Low Power Bank Withdrawal

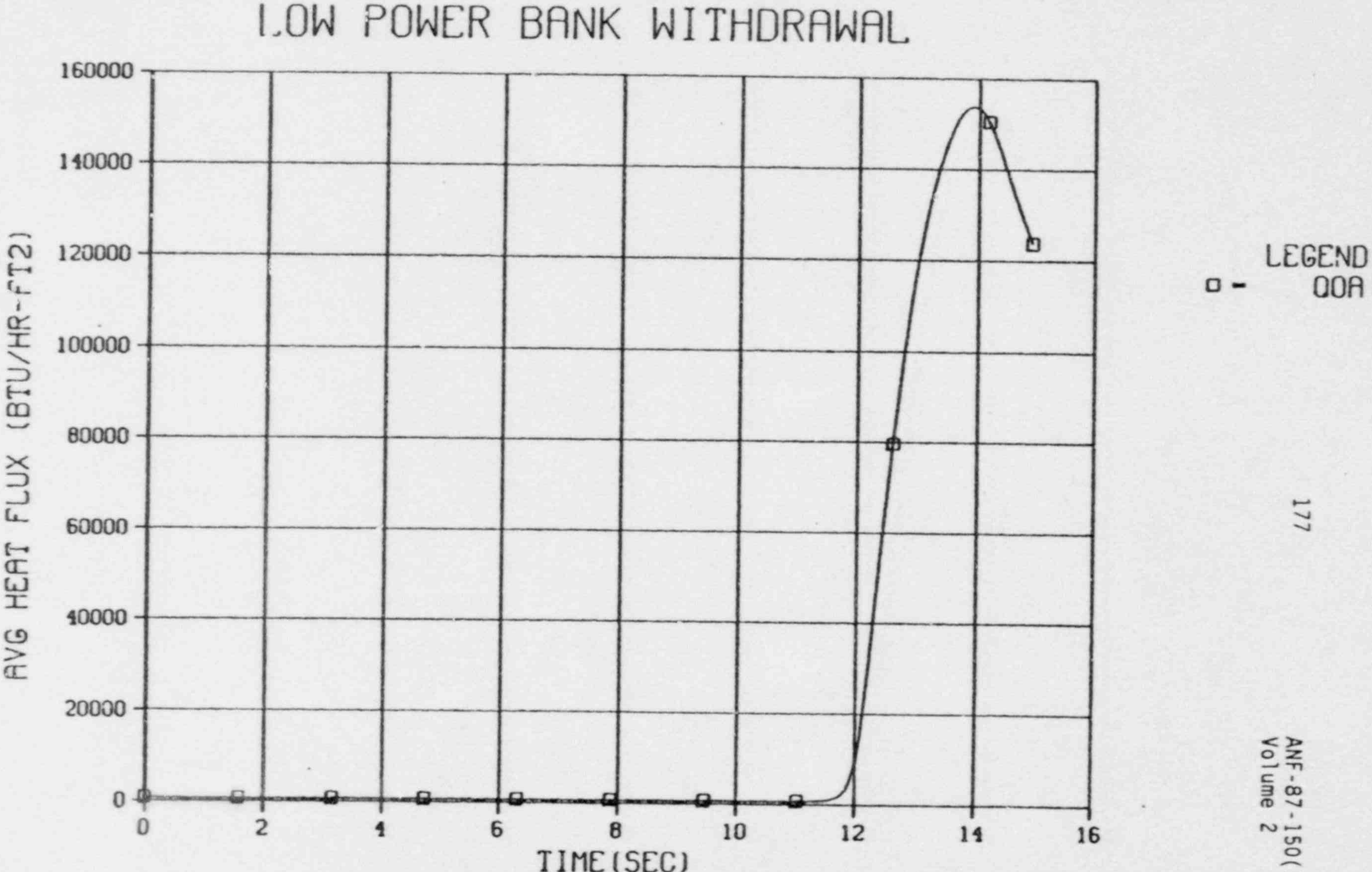


Figure 15.4.1-2 Core Average Heat Flux for Low Power Bank Withdrawal

OT 3 18.47.30 MON 2 DEC, 1985 JOB=HDGRH12 , U C C DISSPLA VER 8.2

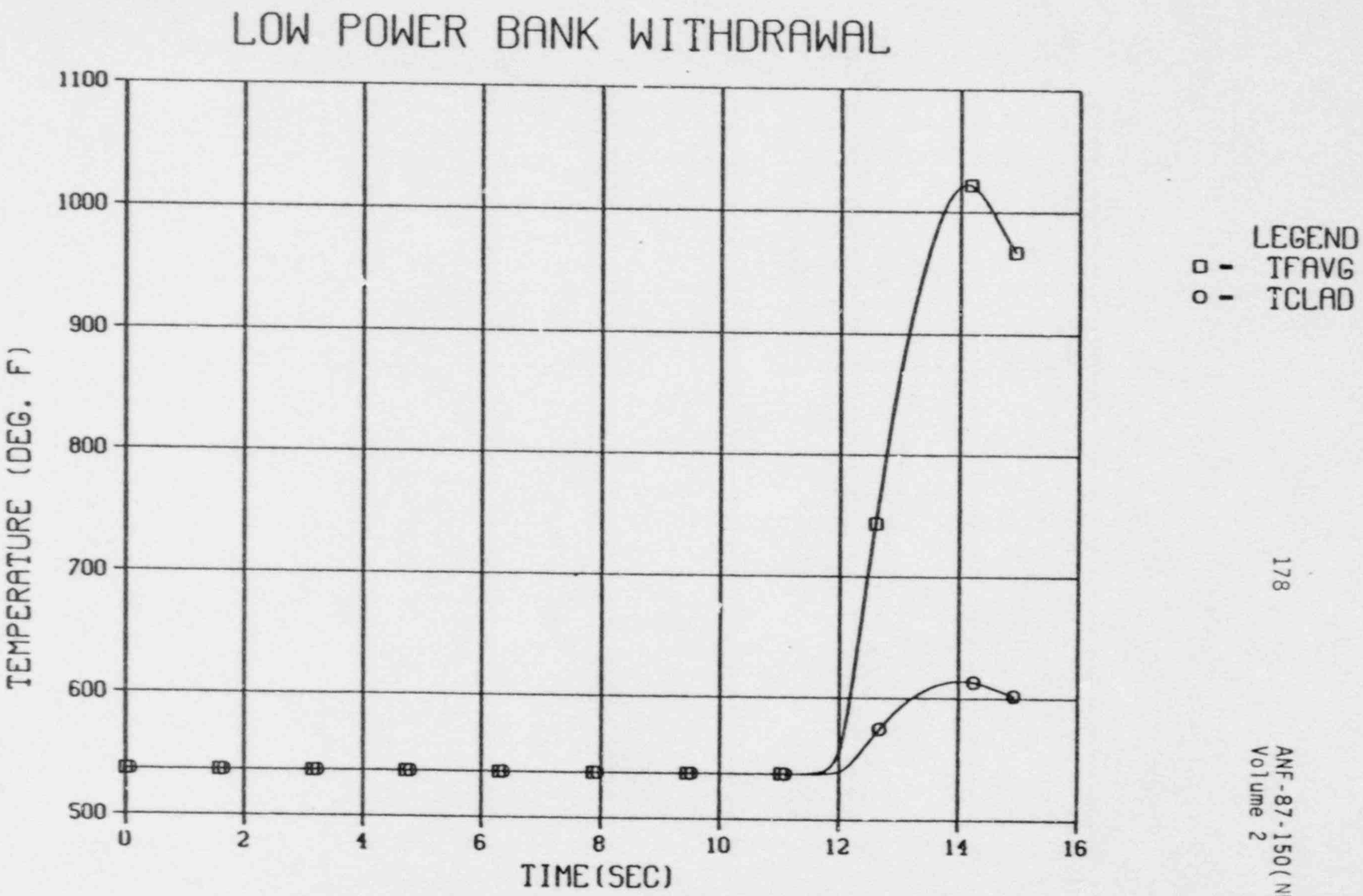


Figure 15.4.1-3 Average Fuel and Clad Temperatures for Low Power Bank Withdrawal

PLOT 1 11.42.05 THUR 12 DEC 1985 JOB-TGPNL52 , U C C DISSPLA VER 8.2

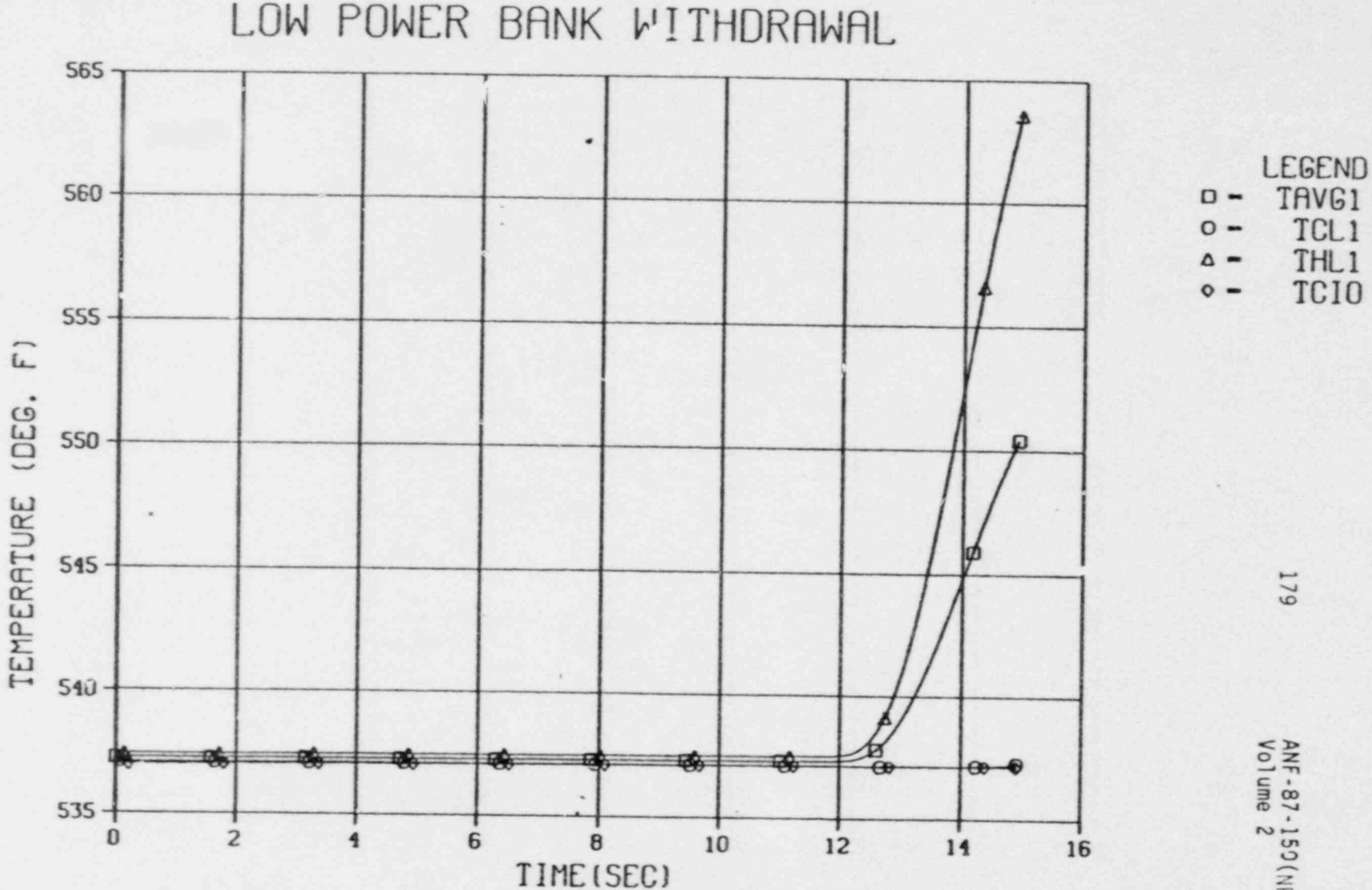


Figure 15.4.1-4 Reactor Coolant System Temperatures for Low Power Bank Withdrawal

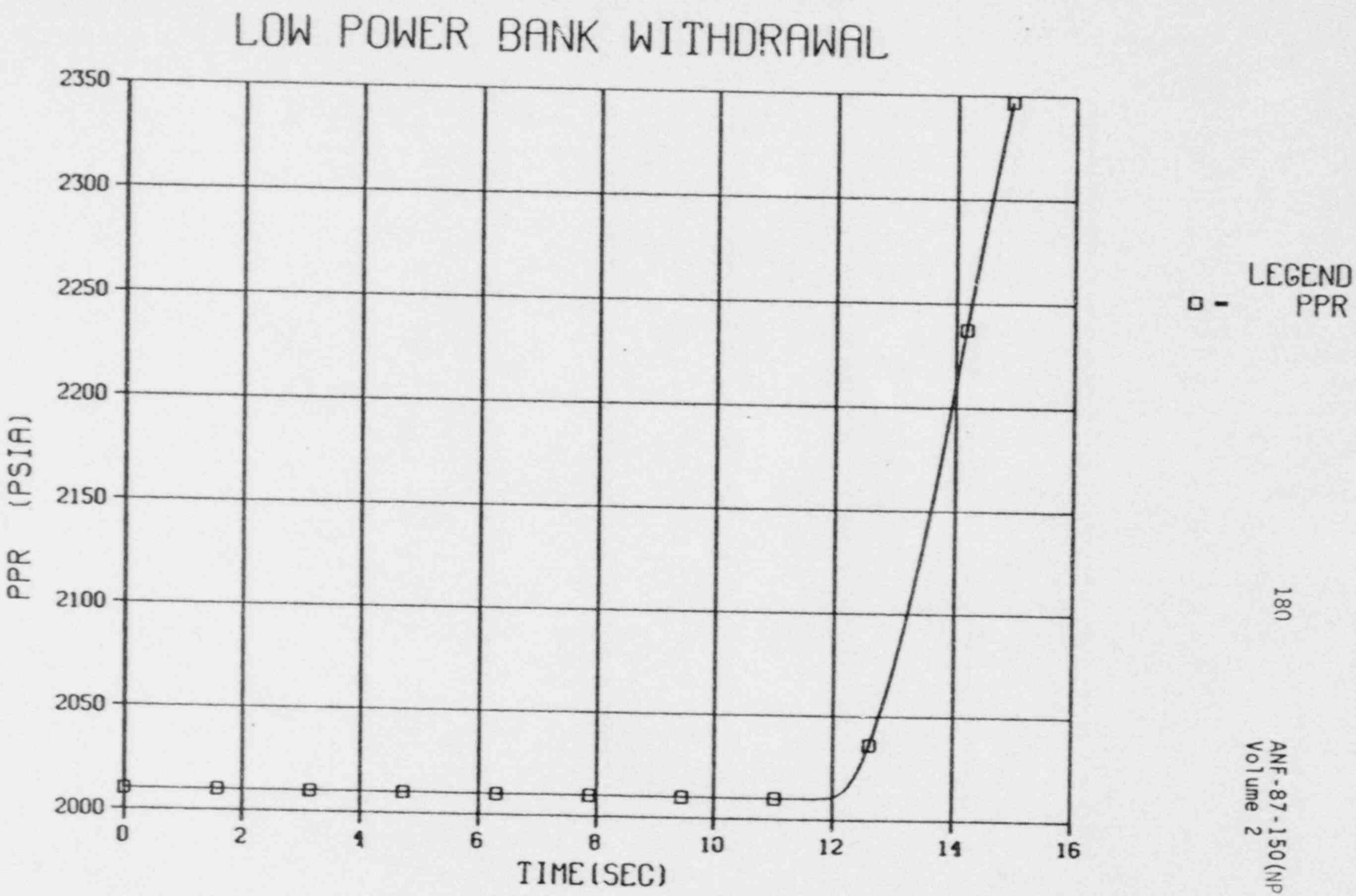


Figure 15.4.1-5 Pressurizer Pressure for Low Power Bank Withdrawal

PLOT 4 12.32.00 MON 2 DEC, 1985 JOB-MOGRH38 , U C C DISSPLA VER 8.2

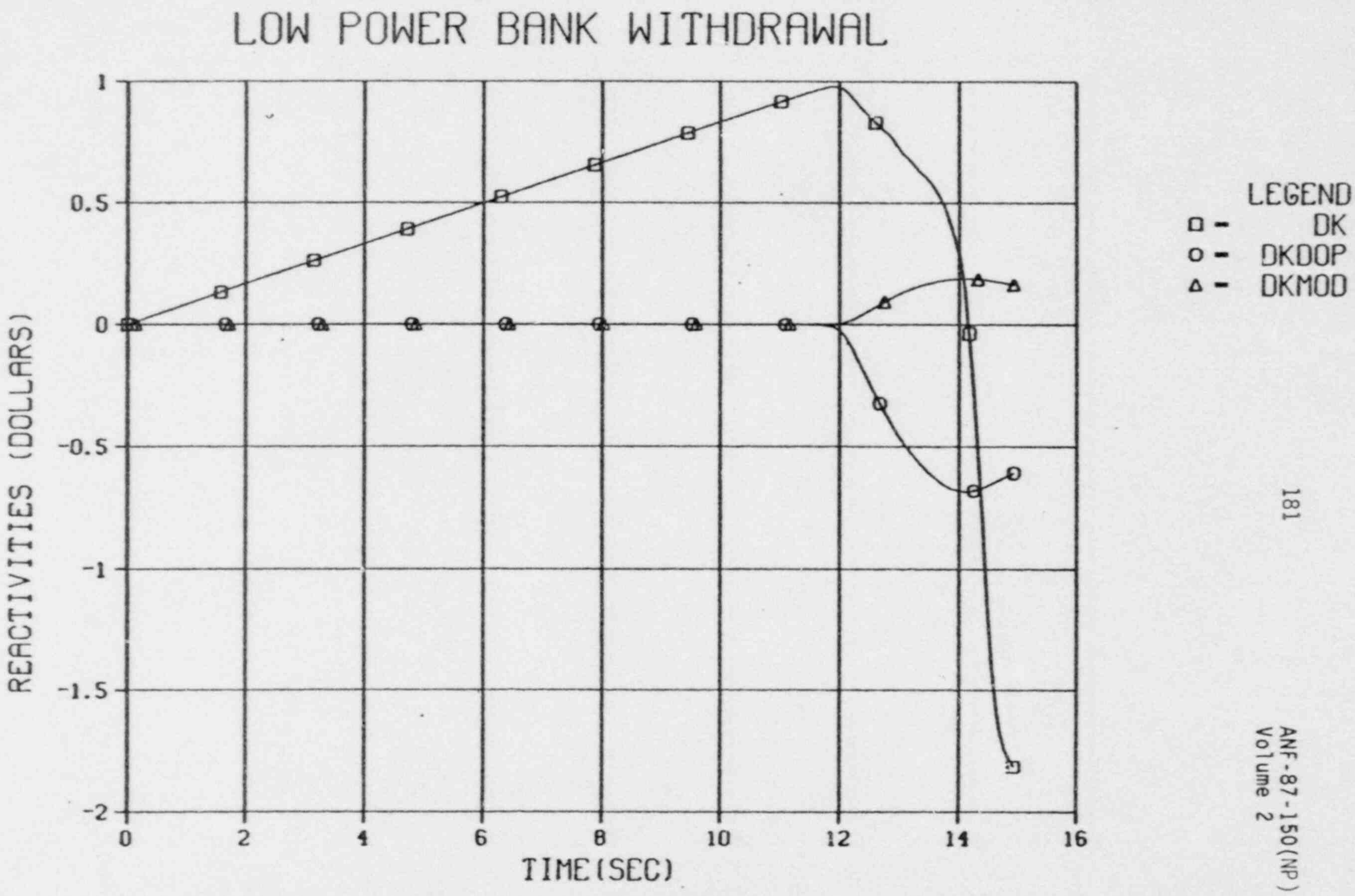


Figure 15.4.1-6 Reactivities for Low Power Bank Withdrawal

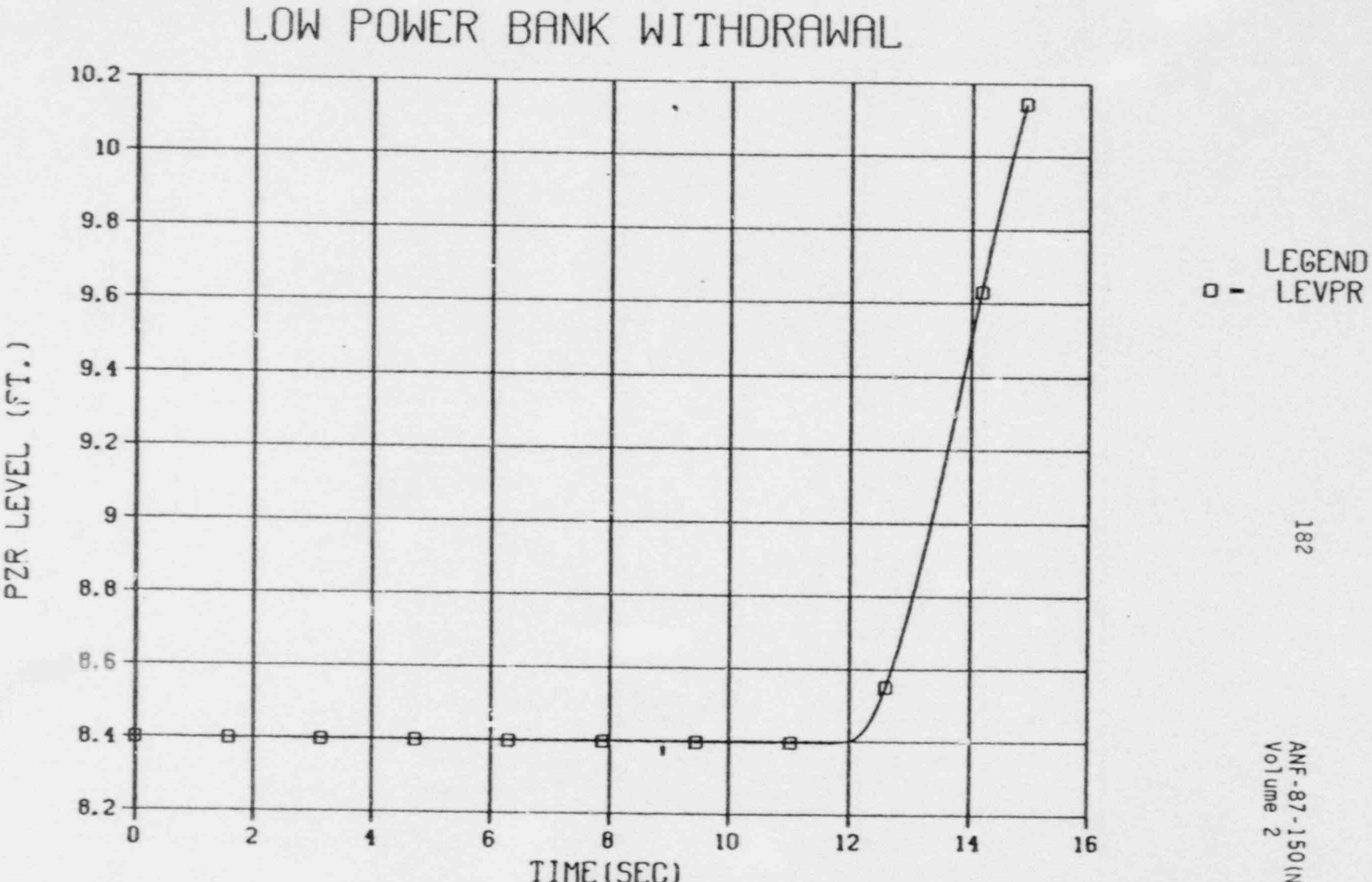


Figure 15.4.1-7 Pressurizer Liquid Level for Low Power Bank Withdrawal

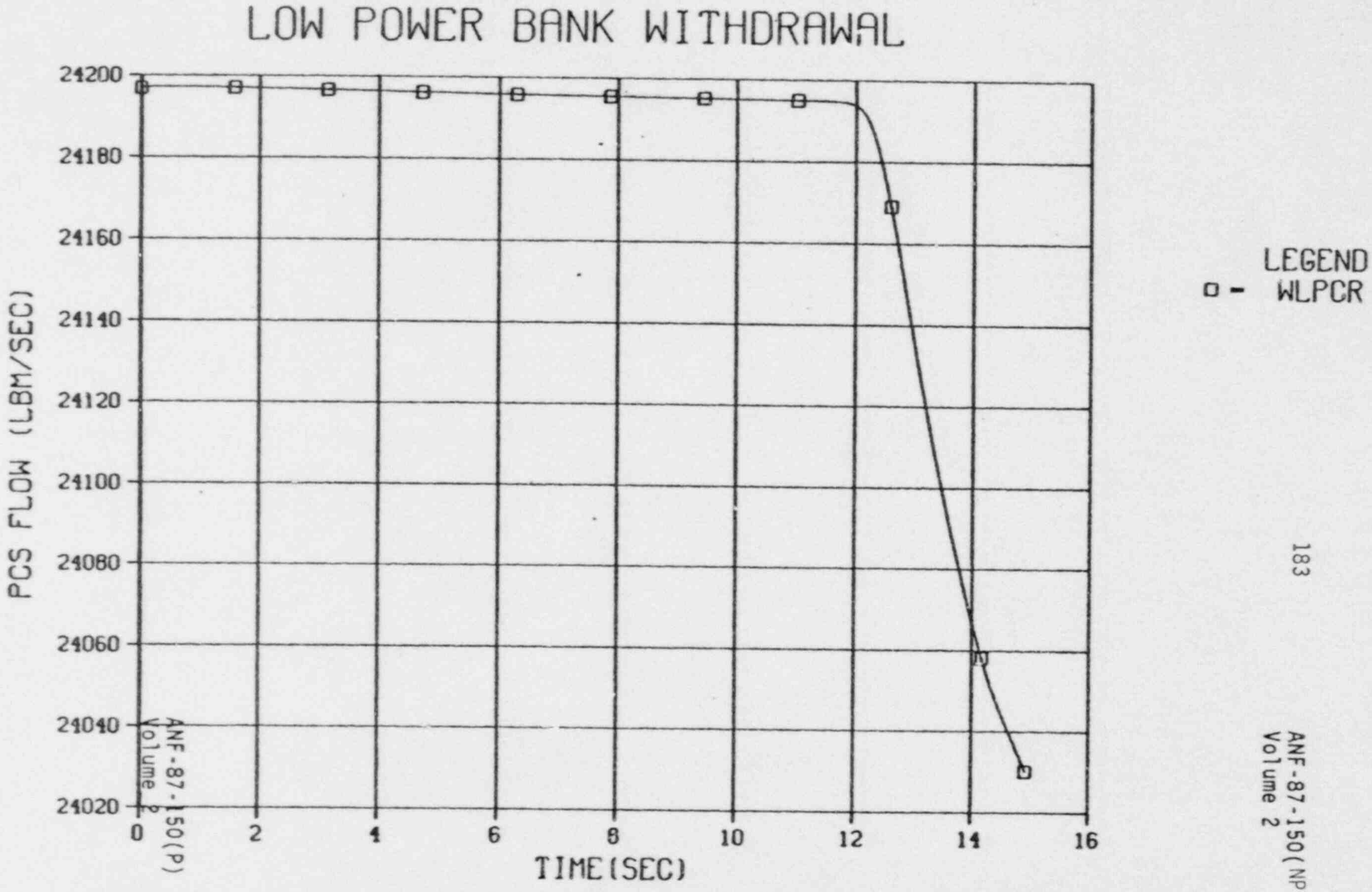


Figure 15.4.1-8 Primary Coolant Flow Rate for Low Power Bank Withdrawal

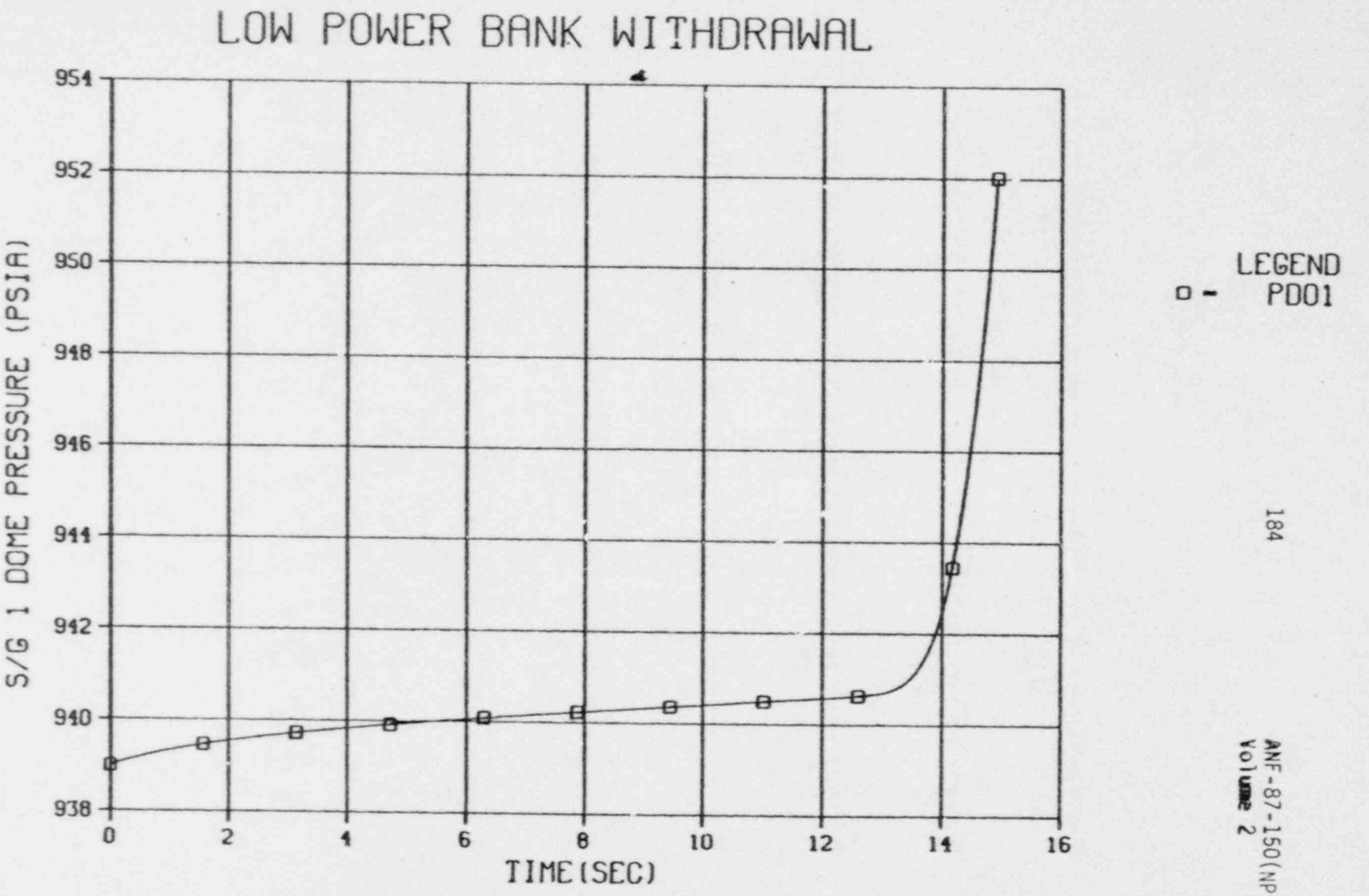


Figure 15.4.1-9 Secondary Pressure for Low Power Bank Withdrawal

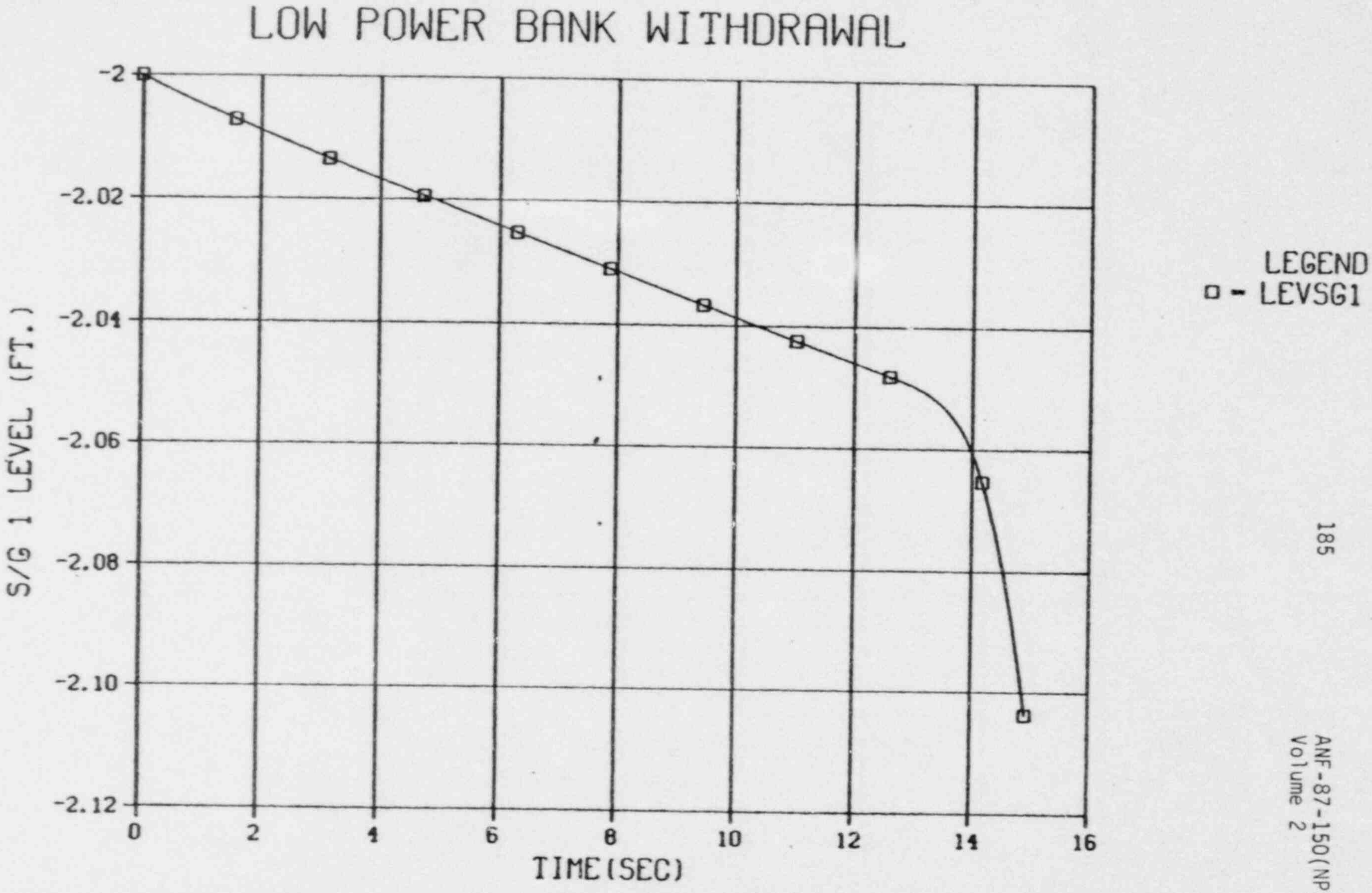


Figure 15.4.1-10 Steam Generator Liquid Level for Low Power Bank Withdrawal

15.4.2 UNCONTROLLED CONTROL ROD BANK WITHDRAWAL AT POWER

15.4.2.1 Identification of Causes and Event Description

This event is defined to result from an uncontrolled control rod bank withdrawal at power. The two cases considered are the 50% and 100% initial power cases. The 50% power case represents the highest power that is allowable with the most limited assembly peaking.

The reactor protection trip system is designed and set to preclude penetration of the SAFDLs. Because of the design of this analysis, the thermal margin/low pressure and variable overpower trips are principally challenged.

The thermal margin/low pressure trip function is designed and set to protect against DNB. Principal DNB parameters such as power (the highest auctioned value of either calorimetric or neutronic power), core inlet temperature, and core power distribution are measured. The function decreases margin to trip setpoint when process variables indicate a decrease in operating margin. This function is based on the core protection boundaries. Operation within these boundaries assures protection of the SAFDLs.

A broad range of reactivity insertion rates and initial operating conditions are possible. The range of reactivity insertion is from very slow, as would be associated with a gradual boron dilution, and bounded on the fast end of the range by bank withdrawal.

The objective of the analysis is to demonstrate the adequacy of the trip setpoints to assure meeting the acceptance criteria. To assure this objective, the analysis is performed for a spectrum of reactivity insertion rates and initial power levels. Since neutronic feedback as a function of cycle exposure and design also influences results, these effects are also included in the analysis.

This event is classified as a moderate frequency event (Table 15.0.1-1). The acceptance criteria are as described in 15.0.1.1. The single failure criteria are given in 15.0.11. The safety systems challenged in this event are redundant and no single active failure will adversely affect the consequences of the event.

15.4.2.2 Analysis Method

The analysis is performed using the PTSPWR2 code[10] and XCOBRA-IIIC[11]. The PTSPWR2 code models the salient system components and calculates neutron power, fuel thermal response, and fluid conditions. The fluid conditions and rod surface heat transport at the time of MDNBR are input to the XCOBRA-IIIC code for calculation of the MDNBR. Systems which minimize DNBR are enabled in the analysis.

The sequence of events is generally the same throughout the event spectrum, differing only in which trip is challenged, i.e.,

(a) Reactivity is inserted

(b) Nuclear power increases

(c) Thermal power increases

(d) Primary temperature increases

(e) Reactor trips on thermal margin/low pressure or variable overpower. No engineered safeguard features are challenged.

15.4.2.3 Definition of Events Analyzed and Bounding Input

The analysis bounds power operation. Two case series are analyzed: one for negative and the other for positive neutronics feedback****.

No cases were run for power levels less than 50% because the allowable peaking distributions remain constant for lower powers. Thus 50% power represents the worst combination of initial power and peaking.

Case Series	Nominal Initial Power	Reactivity Rate	Neutronics
1	Rated	Low to high	Neg. Feedback
	50% Rated	Low to high	Neg. Feedback
2	Rated	Low to high	Pos. Feedback
	50% Rated	Low to high	Pos. Feedback

A summary of the initial operating conditions for these transients is provided in Table 15.4.2-1. Conservative conditions were established for these transients using the methodology as follows:

****The descriptions "negative" and "positive" are in accord with the sign of the moderator temperature coefficient and do not indicate the sign of the overall power coefficient.

Table 15.4.2-0 Conservative Assumptions for the Uncontrolled Rod Bank Withdrawal at Power Event

Control	Manual	
Core power	Nom. +2% Rated	
Core inlet temperature	Nom. +5°F	
Primary pressure	Nom. ±50 psi	
Pressurizer spray	Available	
Pressurizer PORVs	Unavailable (Blocked off for power operation)	
Pressurizer level	Nom. -5% of span	
Turbine bypass valves	Disable	
Atmospheric dump valves	Disable	
Reactor Trips	Thermal margin/low pressure, Variable overpower, High pressure	
Pellet-to-clad heat transfer coef.	Nom. +20%	
Reactivity insertion rate	Maximum to very low	
	Max.Pos.	Max.Neg.
Moderator temperature coefficient	1.2 BOC	1.2 EOC
Doppler coefficient	.8 BOC	1.2 EOC

The maximum reactivity insertion rate used bounds the most reactive banks which may be withdrawn together moving at the maximum rate. Based on BOC and EOC neutronics rod withdrawal simulations, a bounding insertion rate of 6 x 10^{-4} $\Delta\rho$/s was determined for both mid and full power cases. These simulations modelled the withdrawal of the control rods from an initial state at their PDILs to a fully withdrawn condition. The minimum reactivity insertion rate used was 10^{-5} $\Delta\rho$/s. This lower limit is typical of the insertion rate resulting from a gradual boron dilution. At these low rates, system response is quasi-static and the transient evolves over very long time periods. Section 15.4.3 considers insertion rates down to 10^{-6} $\Delta\rho$/s to bound single rod withdrawals.

The initial pressurizer pressure was biased low for the 100% power cases. For all but the slow BOC reactivity insertions, the events tripped on a TM/LP signal. The slow rod withdrawals from BOC tripped on the variable high power trip. For the slow withdrawals, the time to trip is long enough that the primary system power and pressure are allowed to increase sufficiently to activate a variable high power trip before a TM/LP trip. For fast rod withdrawals, a TM/LP trip is initiated due to the rapid increase in core power coupled with a slower responding pressurizer pressure.

Several 100% power cases, both BOC and EOC, were rerun using an initial pressure biased high rather than low. Each of the cases terminated on a variable high power trip. For these cases the MDNBRs are less limiting, as compared to those for an initially low biased pressure, because the higher pressure is maintained throughout the event. Therefore, the limiting MDNBRs resulted from full power cases with the initial pressurizer pressure biased low.

The initial conditions for the rod withdrawal from mid-power were established based on a heat balance of the primary and secondary systems at 52% of rated power.

The limiting axial shapes used in this analysis are shown in Figures 15.0.3-1 and 15.0.3-2 for 50% and 100% power, respectively. The TM/LP trip used in this analysis is described in Section 15.0.7.2. This trip function includes measurement uncertainties and allowances on the relevant variables. The TM/LP trip also incorporates the respective axial power profiles for both 50% and 100% power biased by an ASI uncertainty of -0.06.

15.4.2.4 Analysis of Results

The uncontrolled rod withdrawal transients were analyzed for full power (100% of rated) and mid power (50% of rated). The calculated MDNBR for the event is 1.304, and occurred in a rod withdrawal from 100% of rated thermal power. The mid power case series was in general less limiting than the full power cases. The effects of local voiding on the power distribution are conservatively neglected in these calculations.

Figures 15.4.2-1 and 15.4.2-2 present MDNBR versus Reactivity Insertion Rate for the mid and full power transients, respectively. MDNBR versus insertion rates are shown for both positive (BOC) and negative (EOC) feedback. MOC kinetics are bounded in the analysis by considering conservatively bounding BOC and EOC kinetics, along with a comprehensive range of reactivity insertion rates. The minimum reactivity insertion rate is 1×10^{-5} $\Delta\rho$/sec, and the maximum rate is 6×10^{-4} $\Delta\rho$/sec. The range of insertion rates was conservatively calculated based on control rod worth and withdrawal speed.

The limiting rod withdrawal at 50% power and EOC kinetics occurred at an insertion rate of 3×10^{-5} $\Delta\rho$/sec. The MDNBR was calculated as 2.373. This transient did not scram, but was ended when the rods were fully withdrawn. The limiting case run under BOC kinetics was not as severe. This case occurred for an insertion rate of 1×10^{-5} $\Delta\rho$/sec, for which the MDNBR was 2.45. This transient was terminated by a reactor scram on high power.

Figures 15.4.2-3 through 15.4.2-13 show the plant responses for the limiting uncontrolled control rod bank withdrawal transient from 50% power. Table 15.4.2-2 presents the sequence of events for this transient.

The limiting uncontrolled control rod bank withdrawal at 100% power and EOC kinetics occurred at an insertion rate of 17.0×10^{-5} $\Delta\rho$/sec. The MDNBR was calculated at 1.304. This transient tripped on a thermal margin/low pressure signal. The limiting case under BOC kinetics was not as severe. This case occurred for an insertion rate of 2.117 x 10^{-5} $\Delta\rho$/sec, for which the MDNBR was 1.522. This transient was terminated by a variable high power trip.

Figures 15.4.2-14 through 15.4.2-24 show the plant responses for the limiting uncontrolled control rod bank transient from 100% power. Table 15.4.2-3 presents the sequence of events for this transient.

Plotted variables are defined in Table 15.0.12-1.

15.4.2.5 Conclusion

Reactivity insertion transient calculations demonstrate that the XNB DNB correlation limit of 1.17 will not be penetrated during any credible reactivity insertion transient at full power or mid power. The maximum peak pellet linear heat rate for these events is 15.3 kw/ft, well below the incipient fuel centerline melt criterion of 21 kw/ft. Pressure is bounded by that reported in event 15.2.1, Loss of External Load. Applicable acceptance criteria are therefore met, and the adequate functioning of the thermal margin/low pressure trip demonstrated.

Table 15.4.2-1 Summary of Initial Conditions for the Uncontrolled Rod Bank Withdrawal at Power Event

Variable	Case Designation	
	Mid Power	Rated Power
Power (MWt)*	1315.6	2580.6
Core Inlet Temperature (°F)	542.2	548.65
Pressurizer Pressure (psia)	2010	2010
Pressurizer Level	Nom. -5% of span	Nom. -5% of span
Reactor Coolant System Flow Rate (lbm/hr)	118.7×10^6	116.6×10^6
Steam Dome Pressure (psia)	853.7	731.6

*Nominal power levels of 50% and 100% were augmented by 2%.

Table 15.4.2-2 Event Summary for the Uncontrolled Rod Bank Withdrawal Event for the Limiting 50% Power Case (EOC Kinetics)

Event	Time (sec)
Start Rod Withdrawal	0.
Steam Line Safety Valves Open	296.
Peak Power Level	310.48
Reactivity Insertion Ends	310.48
Peak Pressurizer Pressure	311.37
Peak Heat Flux	311.44
Peak Core Average Temperature	328.62
Minimum DNBR	482.

Table 15.4.2-3 Event Summary for the Uncontrolled Rod Bank Withdrawal Event for the Limiting 100% Power Case (EOC Kinetics)

Event	Time (sec)
Start Rod Withdrawal	0.
Peak Power Level	13.58
Peak Heat Flux	13.88
Peak Pressurizer Pressure	25.44
Peak Core Average Temperature	101.95
TM/LP Trip Signal	221.43
Minimum DNBR	221.43

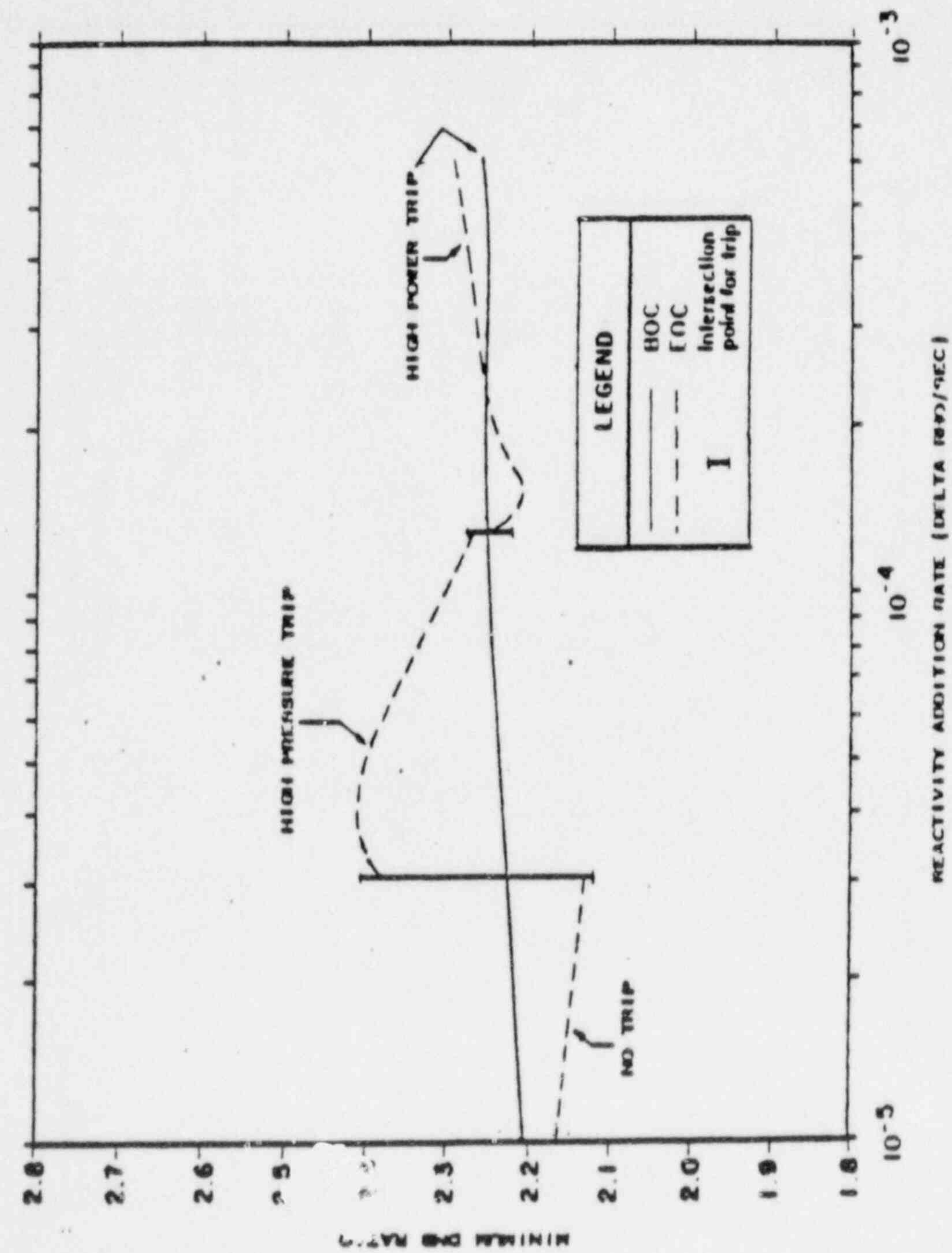


Figure 15.4.2-1 MDNBR Values for Uncontrolled Bank Withdrawal Initiated from 52% of Rated Power

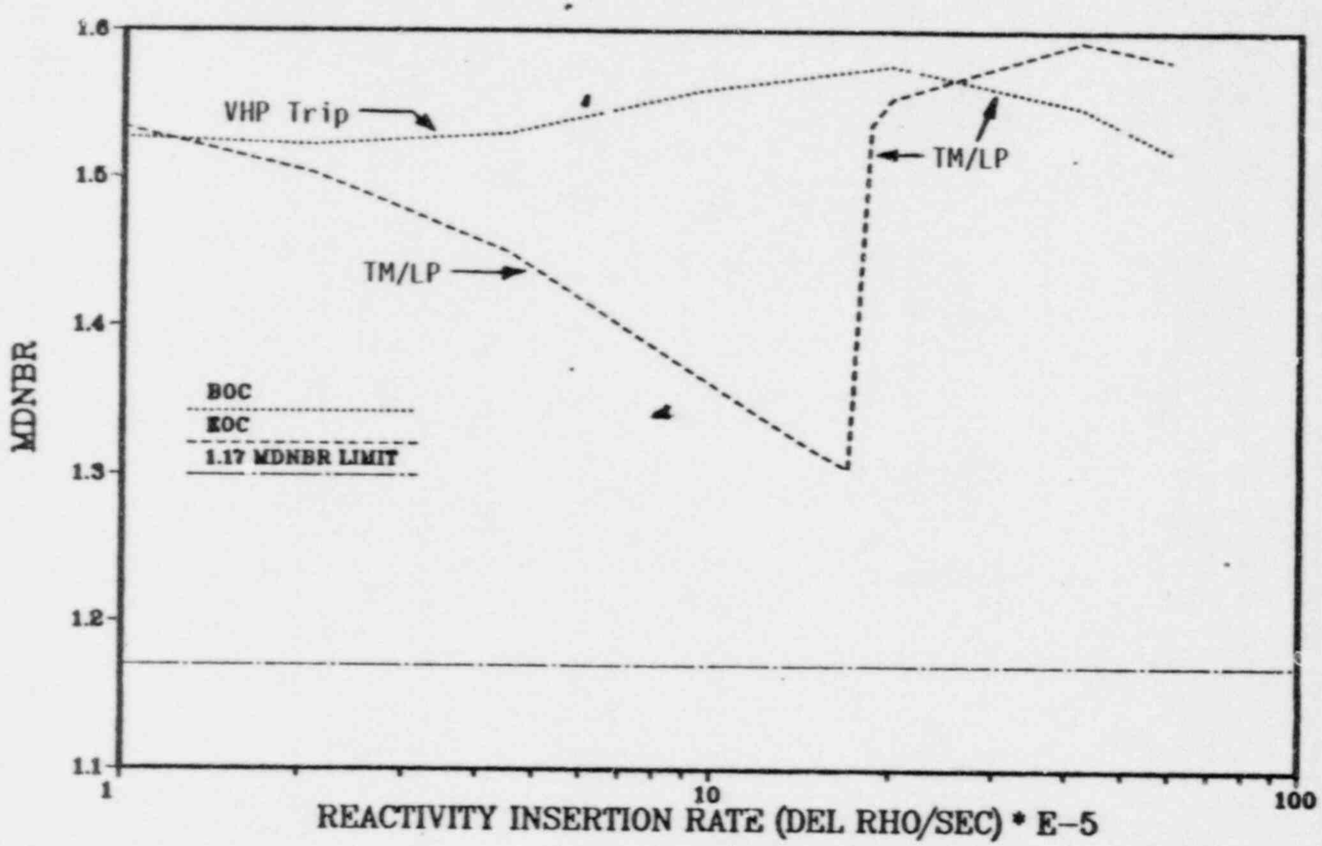


Figure 15.4.2-2 MDNBR Values for Uncontrolled Bank Withdrawal Initiated from 102% of Rated Power

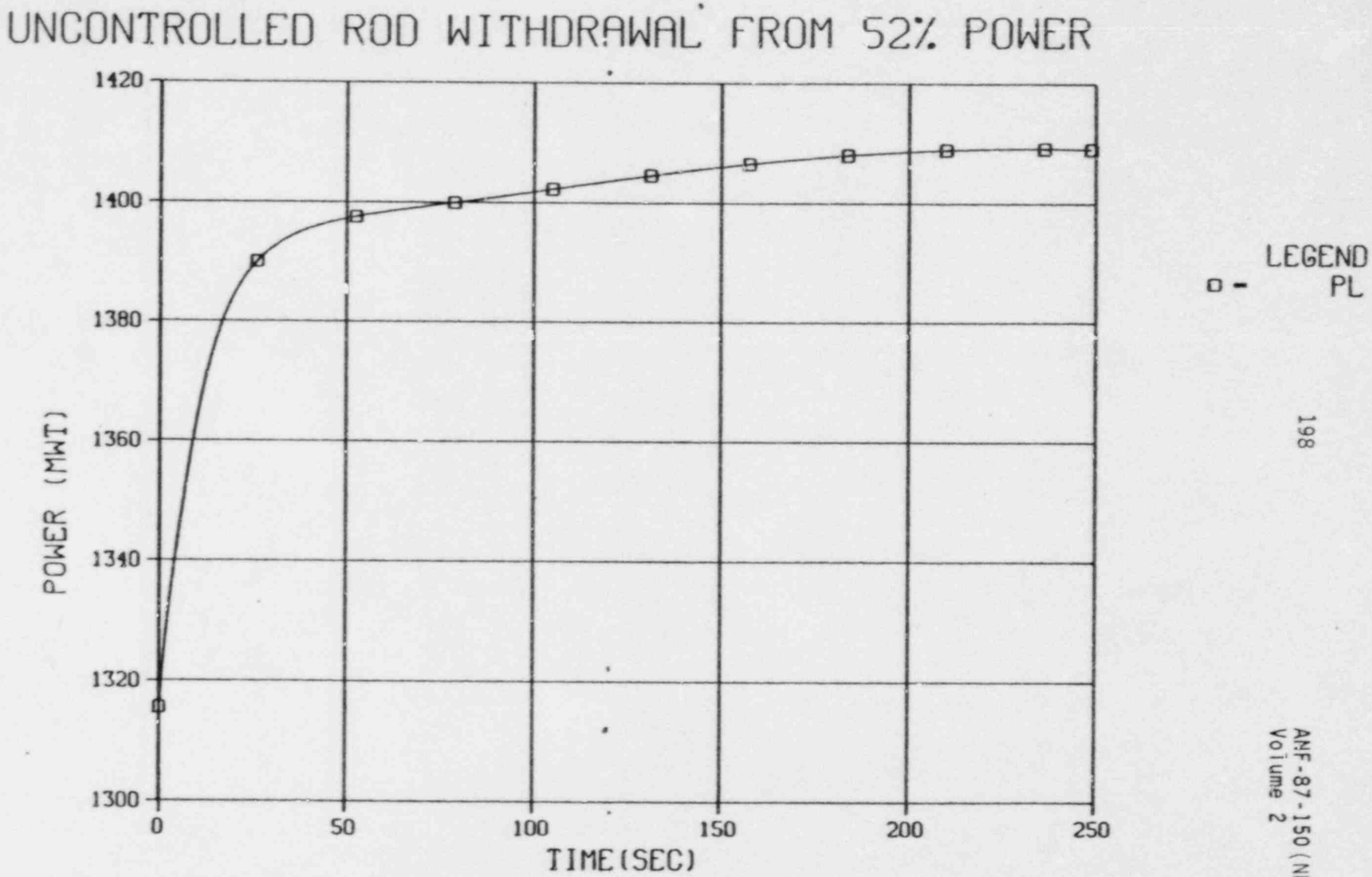


Figure 15.4.2-3 Reactor Power Level for Uncontrolled Bank Withdrawal at Mid-Power

PLOT 3 16.13.18 TUES NOV, 1985 U C C DISSPLA VER 8.2

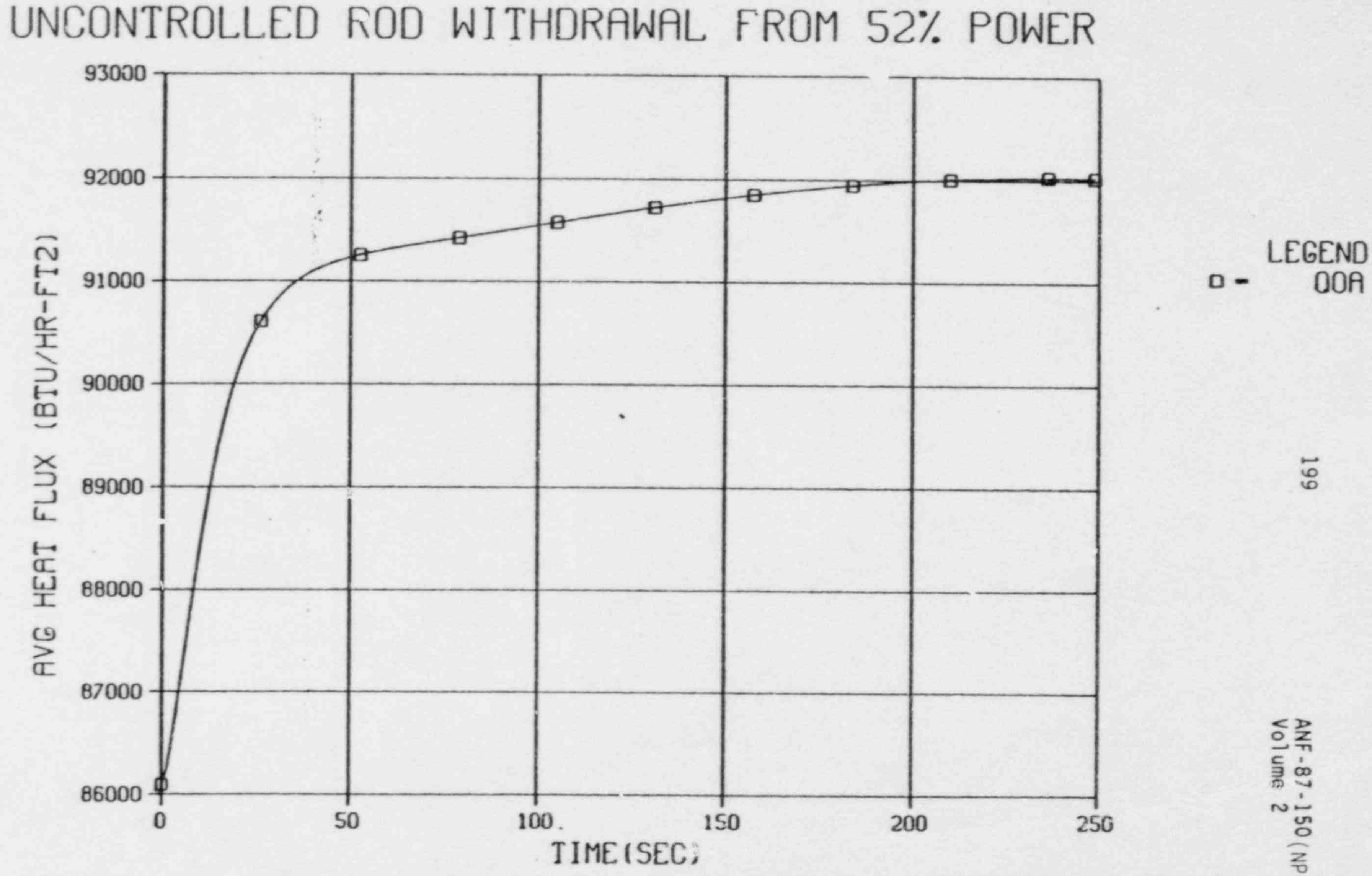


Figure 15.4.2-4 Core Average Heat Flux for Uncontrolled Bank Withdrawal at Mid-Power

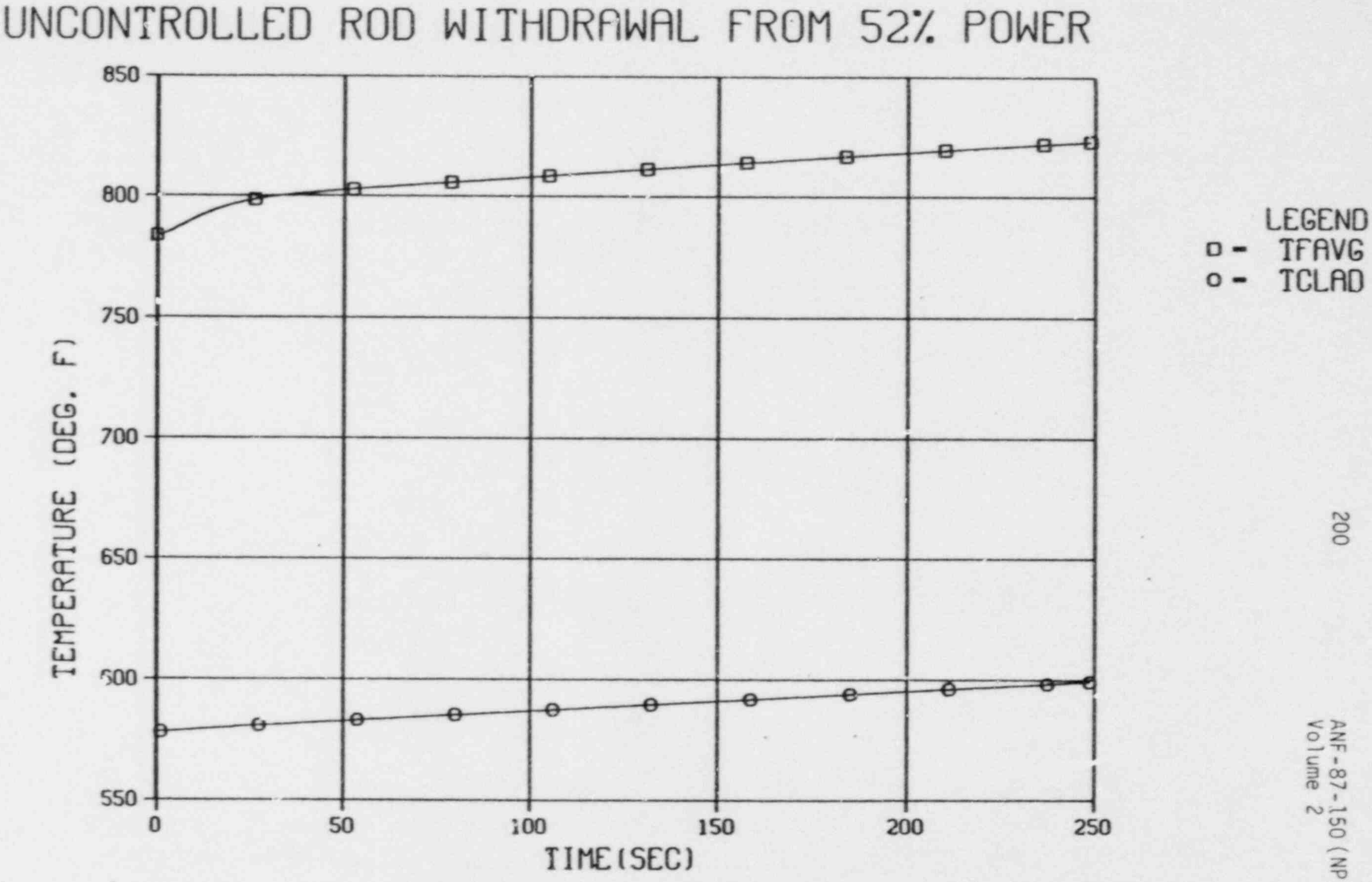


Figure 15.4.2-5 Average Fuel and Clad Temperatures for Uncontrolled Bank Withdrawal at Mid-Power

PLOT 1 18.59.57 THUR 12 DEC, 1985 JOB-TOPALS1 , U C C DISSPLA VER 8.2

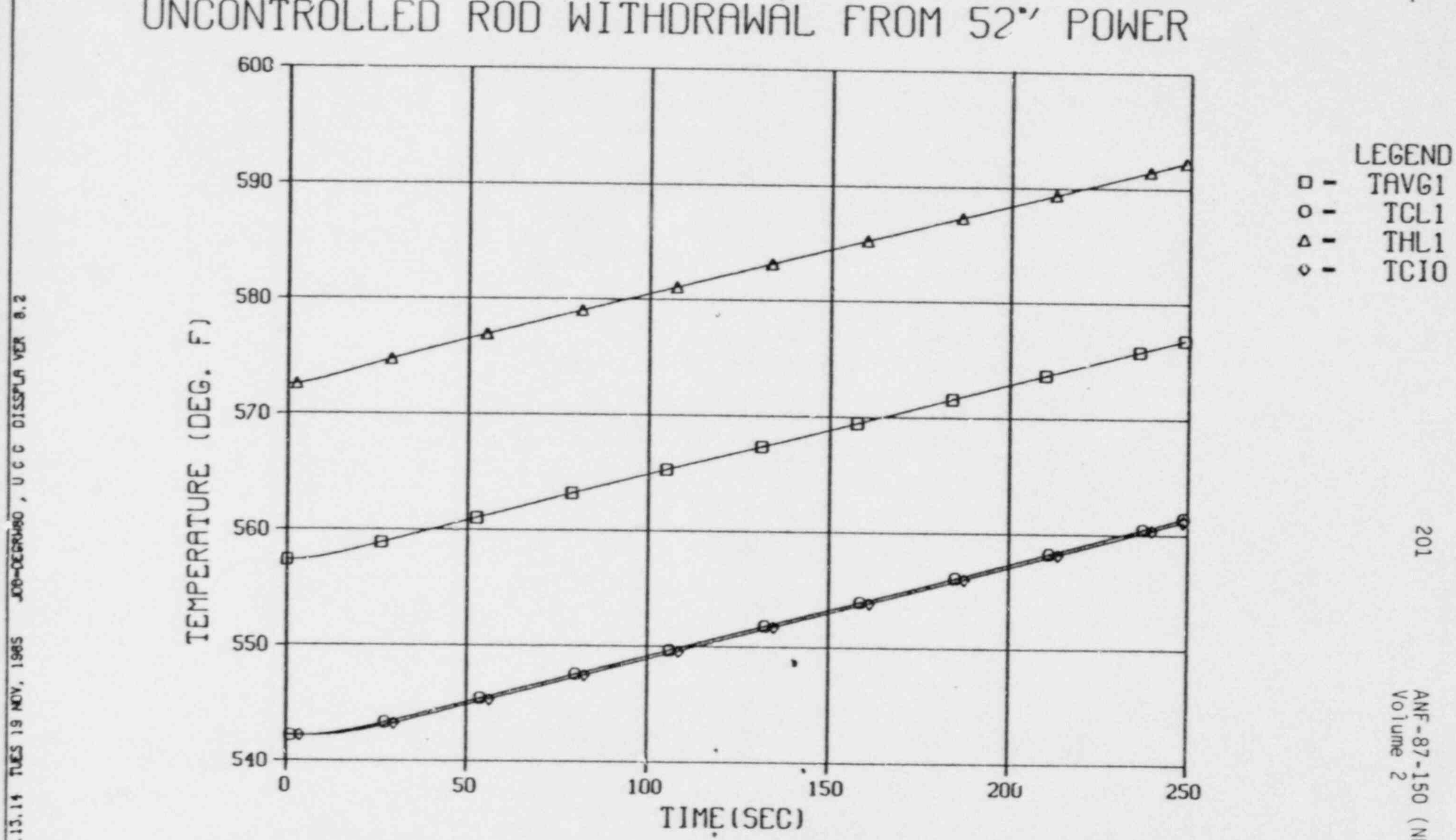


Figure 15.4.2-6 Reactor Coolant System Temperatures for Uncontrolled Bank Withdrawal at Mid-Power

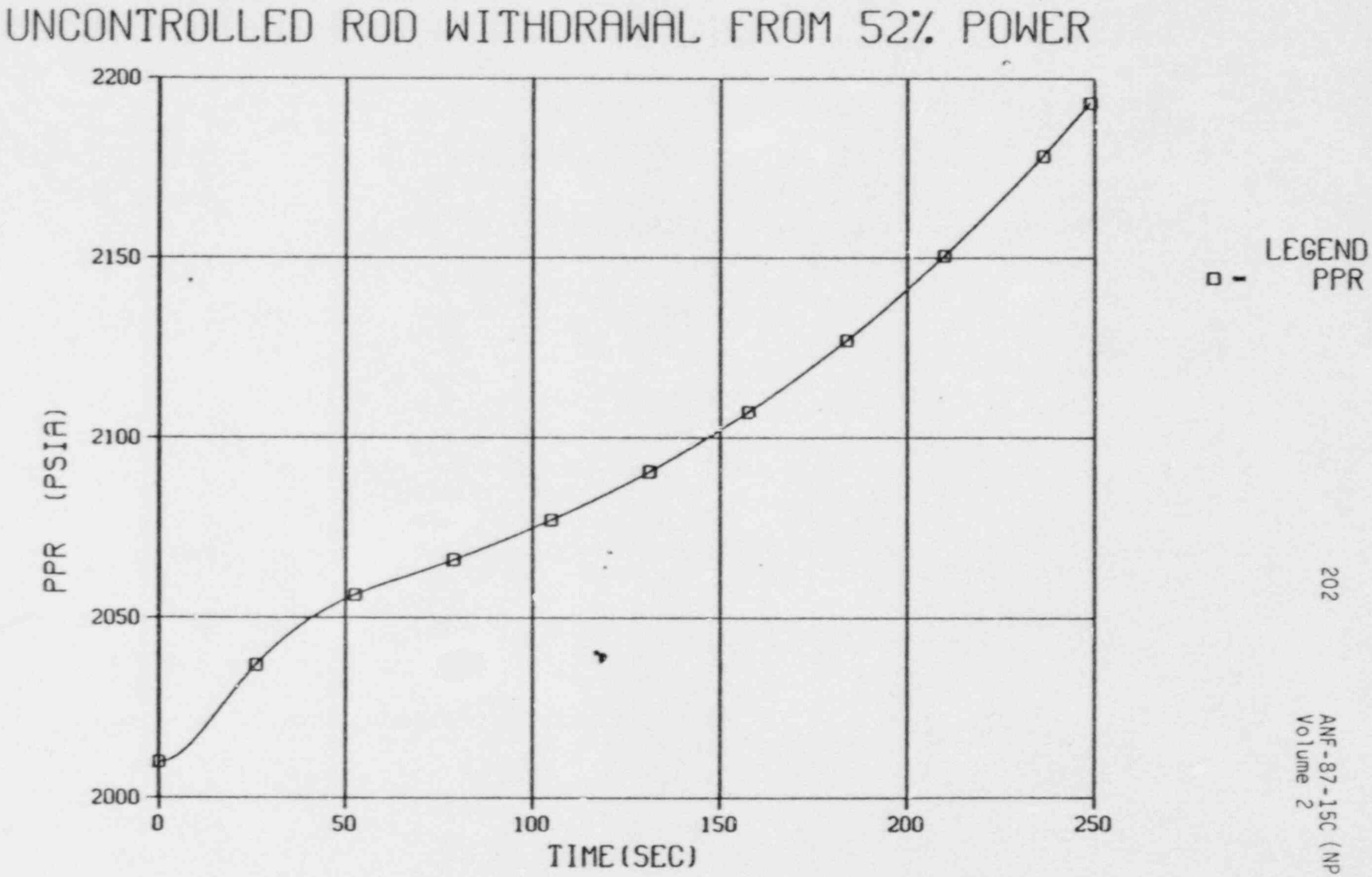


Figure 15.4.2-7 Pressurizer Pressure for Uncontrolled Bahk Withdrawal at Mid-Power

PLOT 1 16.13.20 TUES 19 NOV, 1985 JOB-DEGRW80 , U C C DISSPLA VER 8.2

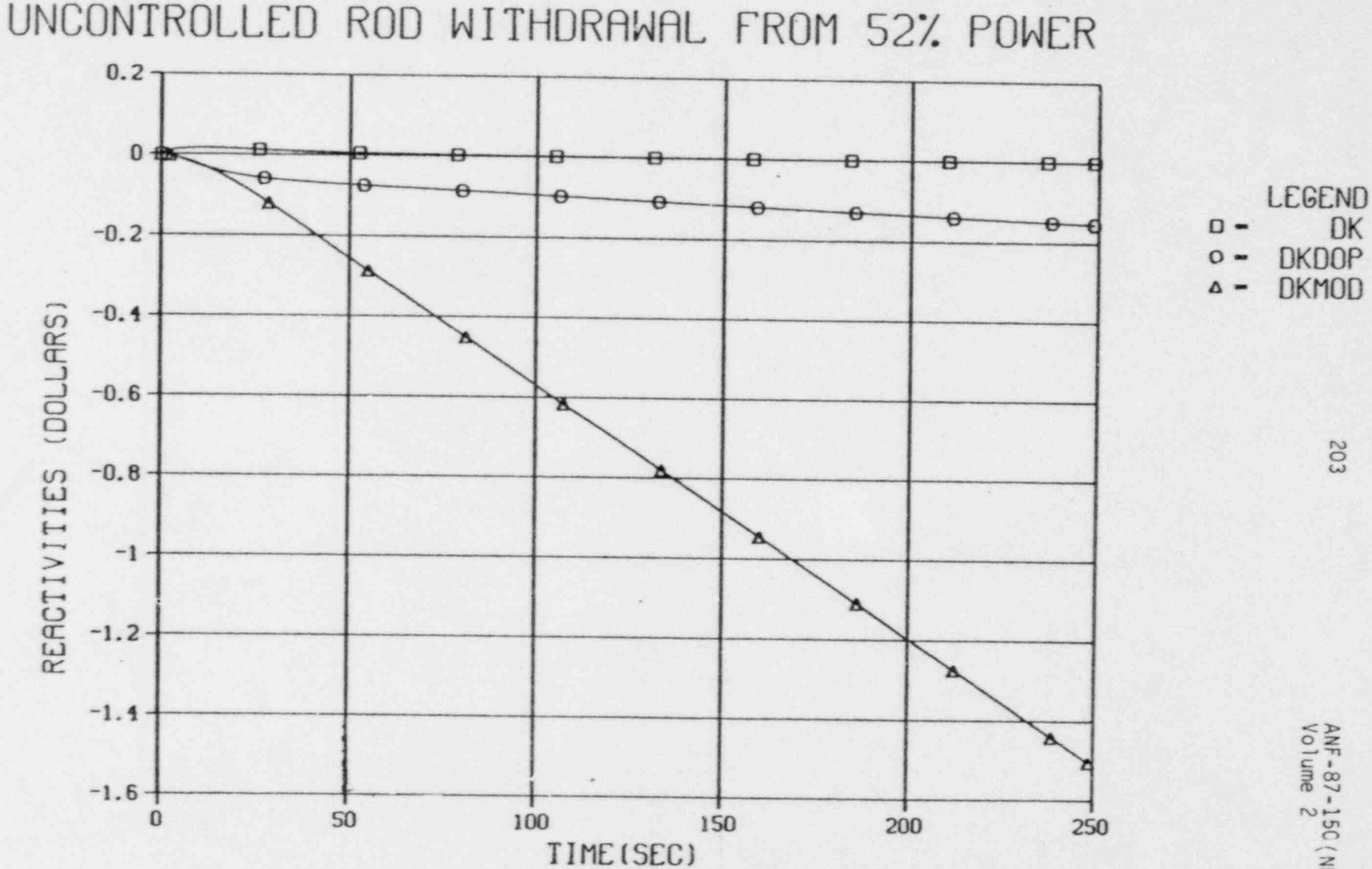


Figure 15.4.2-8 Reactivities for Uncontrolled Bank Withdrawal at Mid-Power

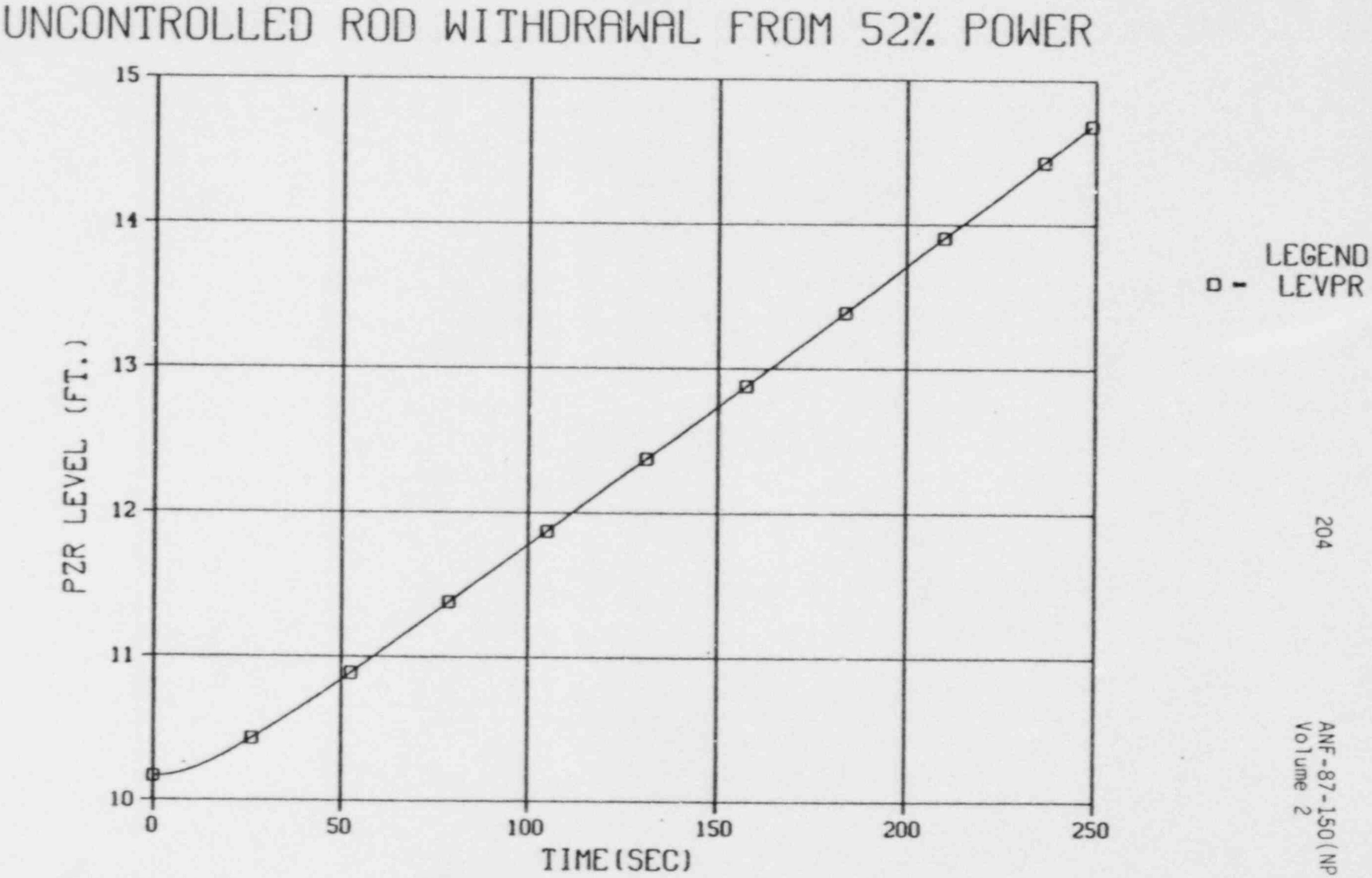


Figure 15.4.2-9 Pressurizer Liquid Level for Uncontrolled Bank Withdrawal at Mid-Power

PLOT 5 16.13.22 TUES 19 NOV, 1985 JOB=DEGRW80 , U C C DISSPLA VER 8.2

UNCONTROLLED ROD WITHDRAWAL FROM 52% POWER

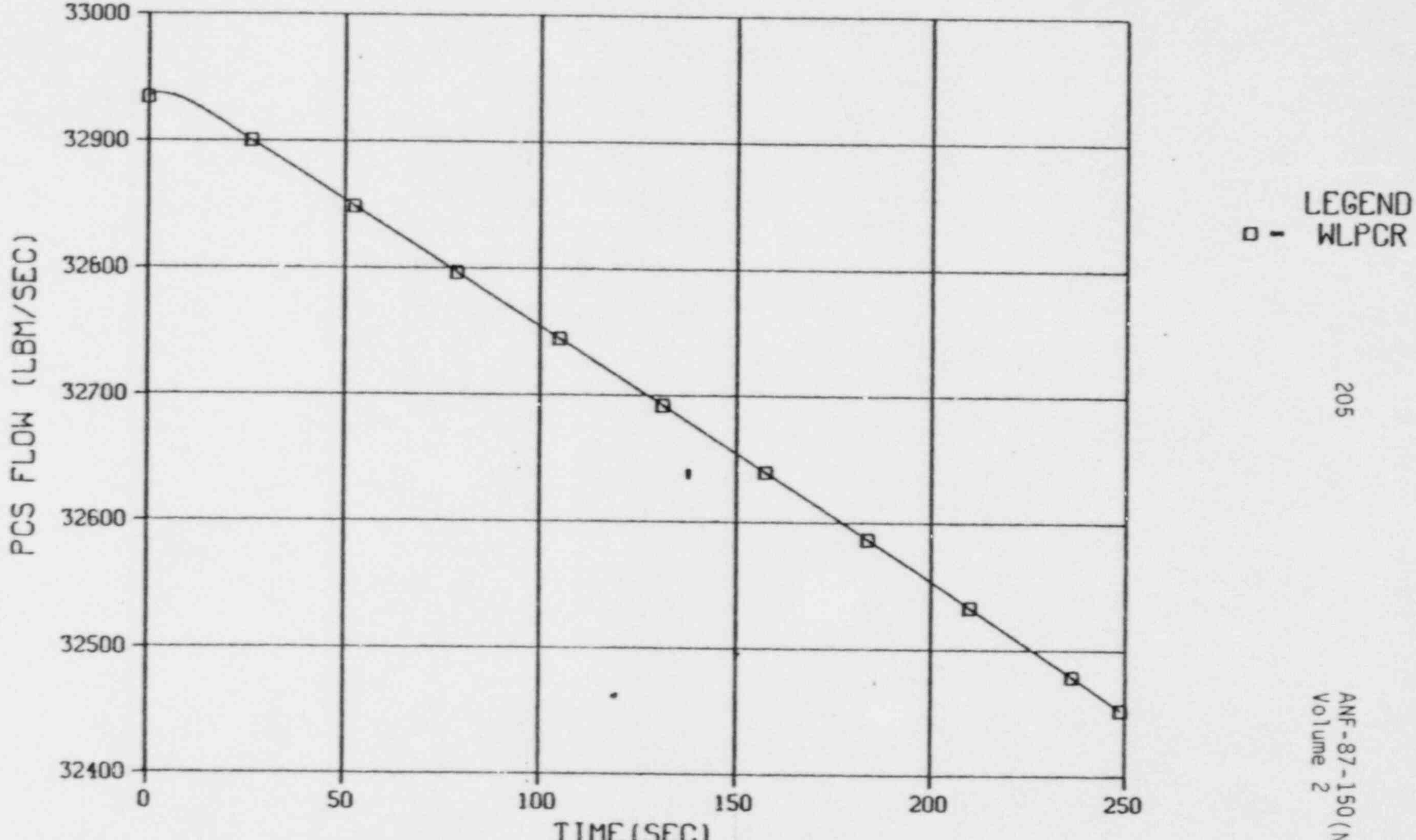
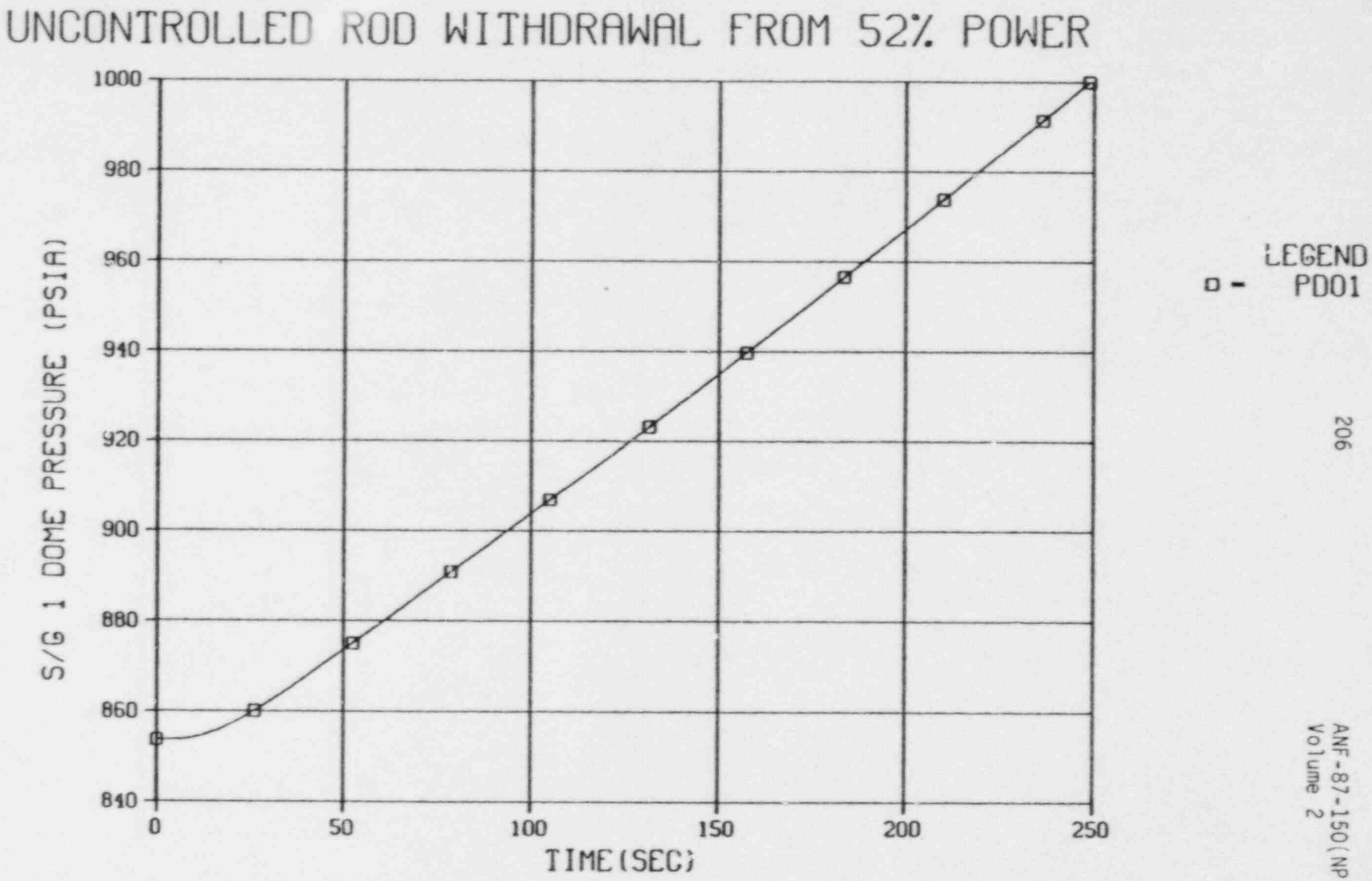


Figure 15.4.2-10 Primary Coolant Flow Rate for Uncontrolled Bank Withdrawal at Mid-Power

16.13.30 TUES 19 NOV, 1985 JOB=OEGRWBO , U C C DISSPLA VER 8.2



PLOT 7 16.13.26 TUES 19 NOV, 1985 JOB-DEGRABO , U C C DISSPLA VER 8.2

Figure 15.4.2-11 Secondary Pressure for Uncontrolled Bank Withdrawal at Mid-Power

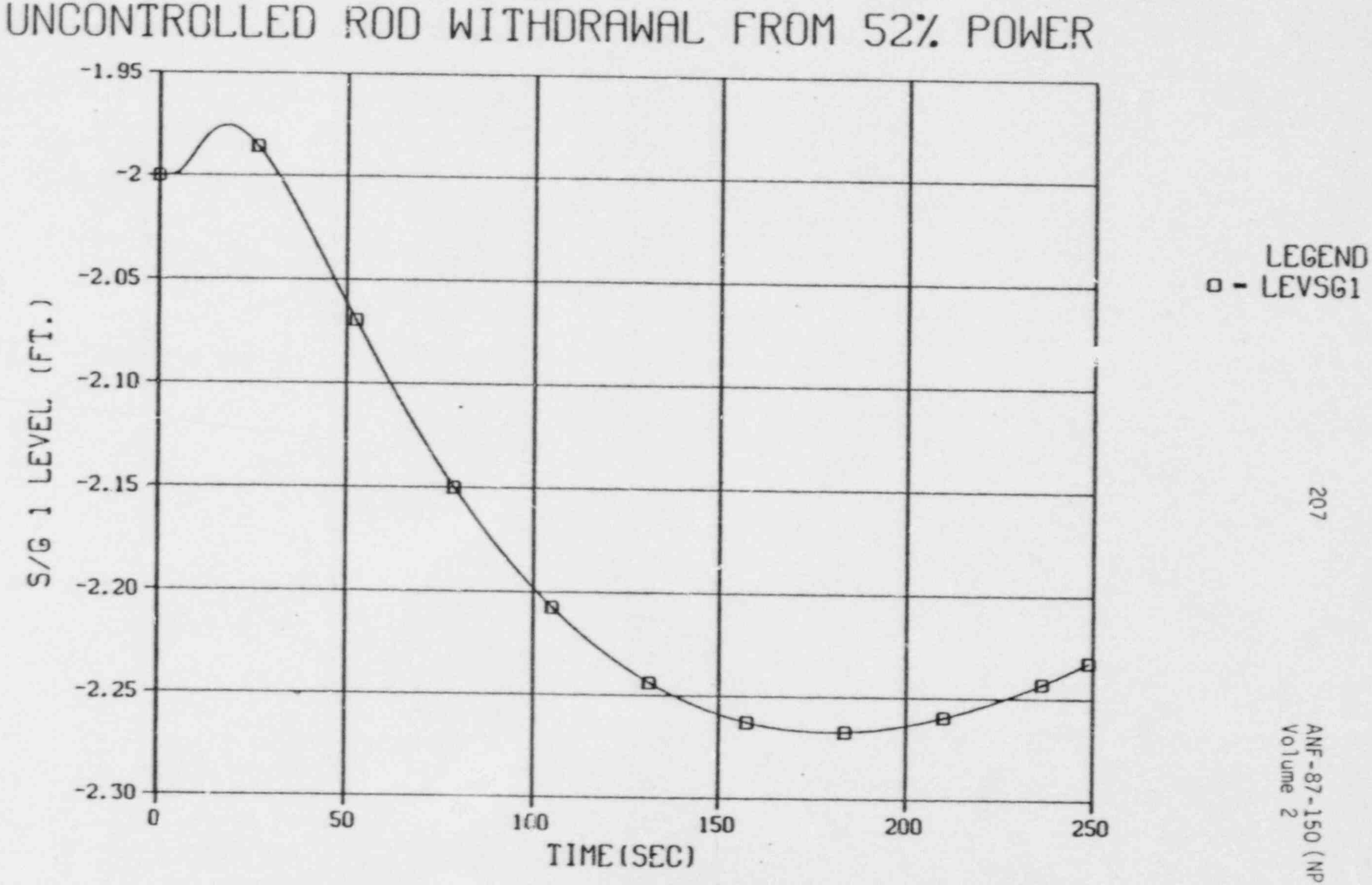
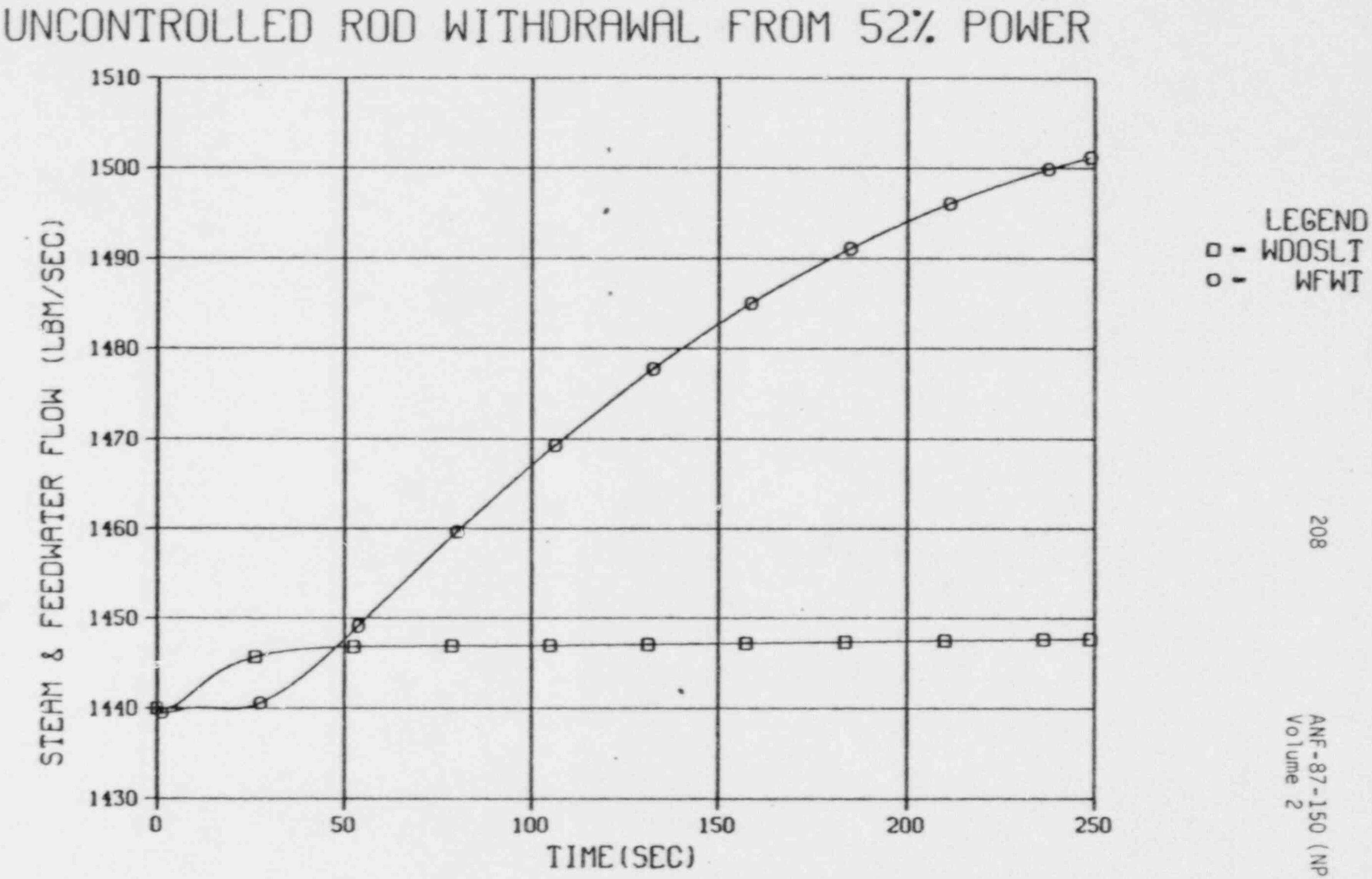


Figure 15.4.2-12 Steam Generator Liquid Level for Uncontrolled Bank Withdrawal at Mid-Power

16.13.24 TUES 19 NOV, 1985 JOB-DCGRWBO , U C C DISSPLA VER 8.2



PLOT 10 16.13.31 TUES 19 NOV, 1985 JOB-DEGRABO , U C C DISSPLA VER 8.2

Figure 15.4.2-13 Secondary Steam and Feedwater Flow Rates for Uncontrolled Bank Withdrawal at Mid-Power

Uncontrolled Rod Withdrawal— Full Power

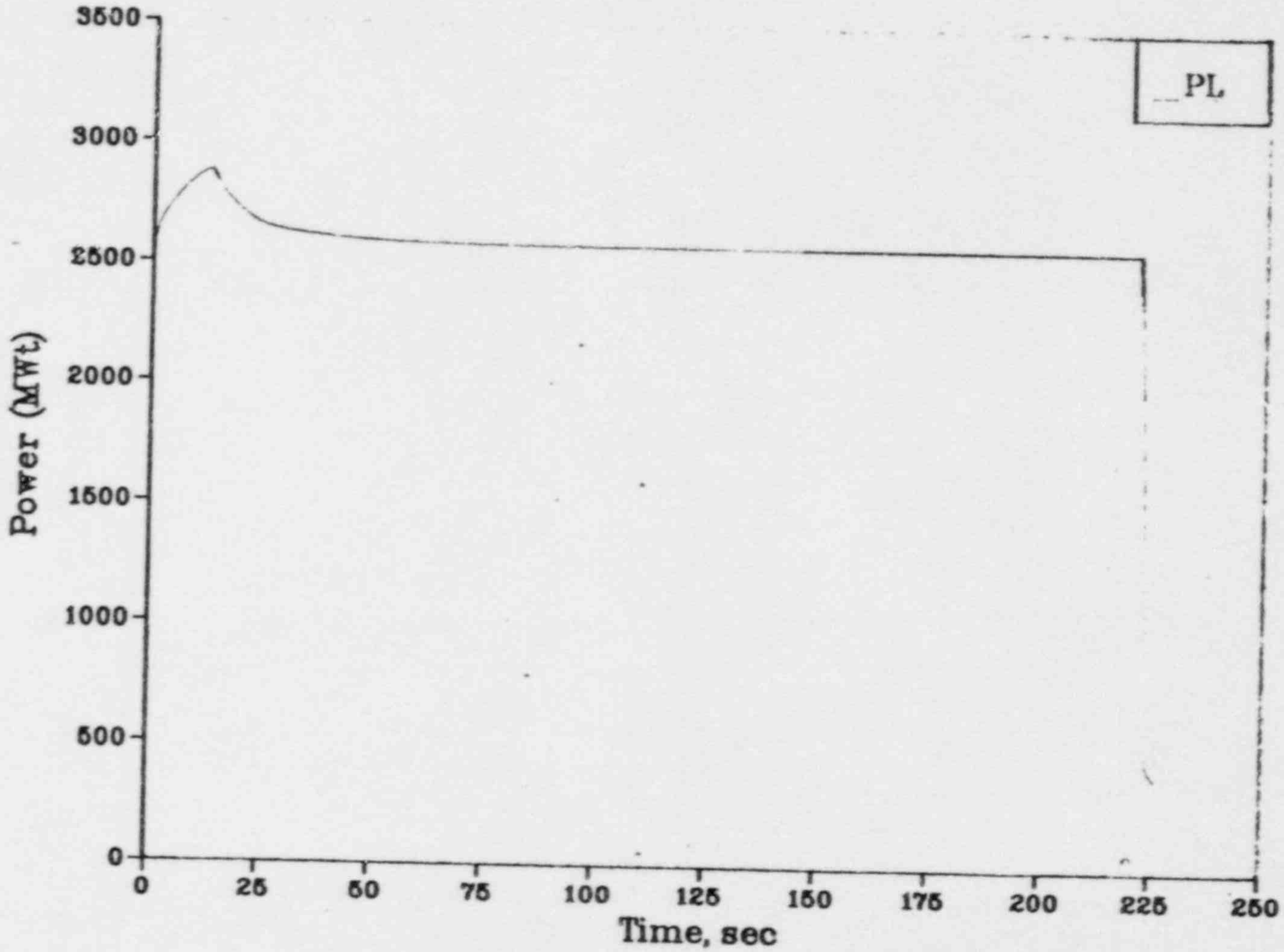


Figure 15.4.2-14 Reactor Power Level for Uncontrolled Withdrawal from Full Power Bank

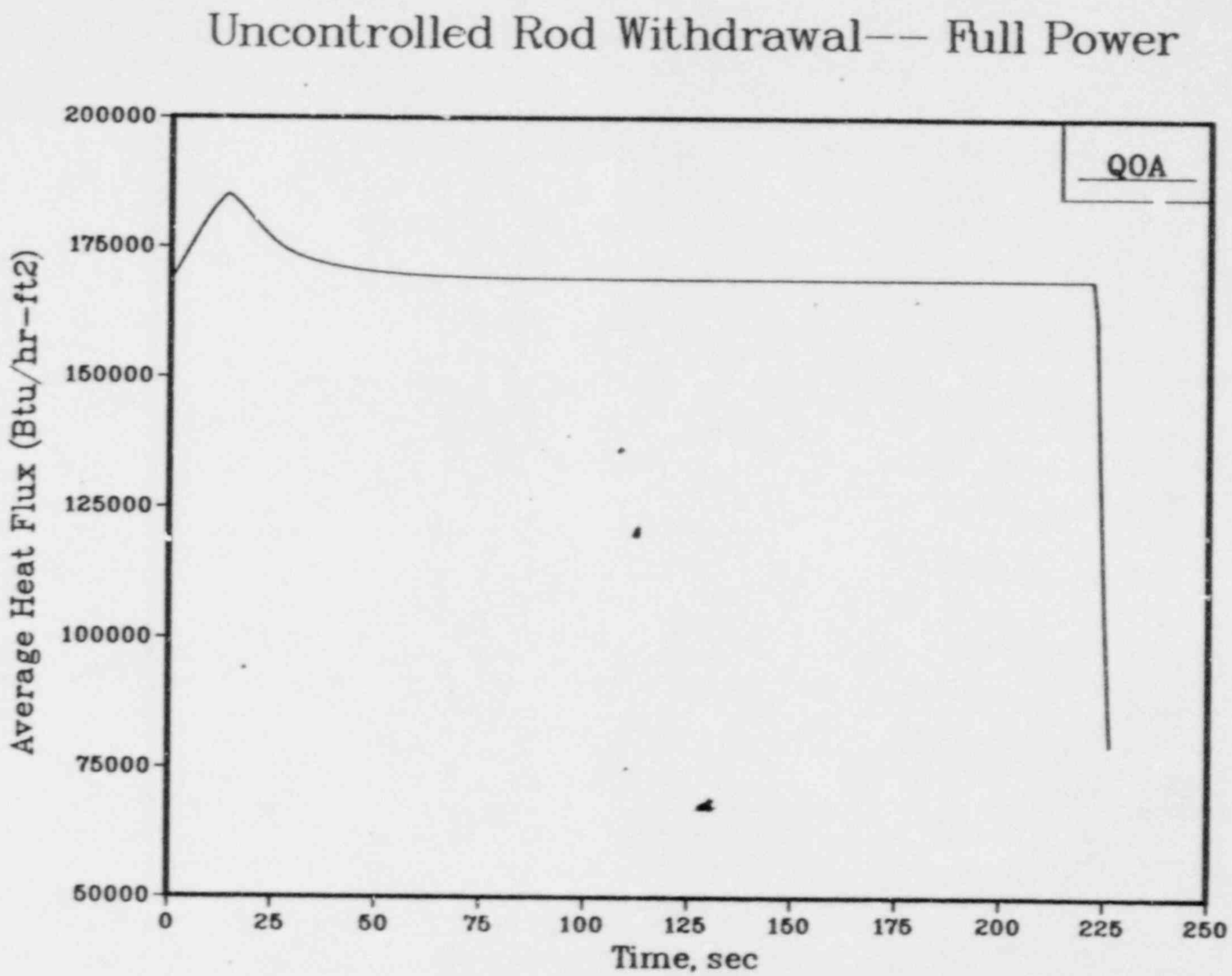


Figure 15.4.2-15 Core Average Heat Flux for Uncontrolled Bank Withdrawal from Full Power

Uncontrolled Rod Withdrawal-- Full Power

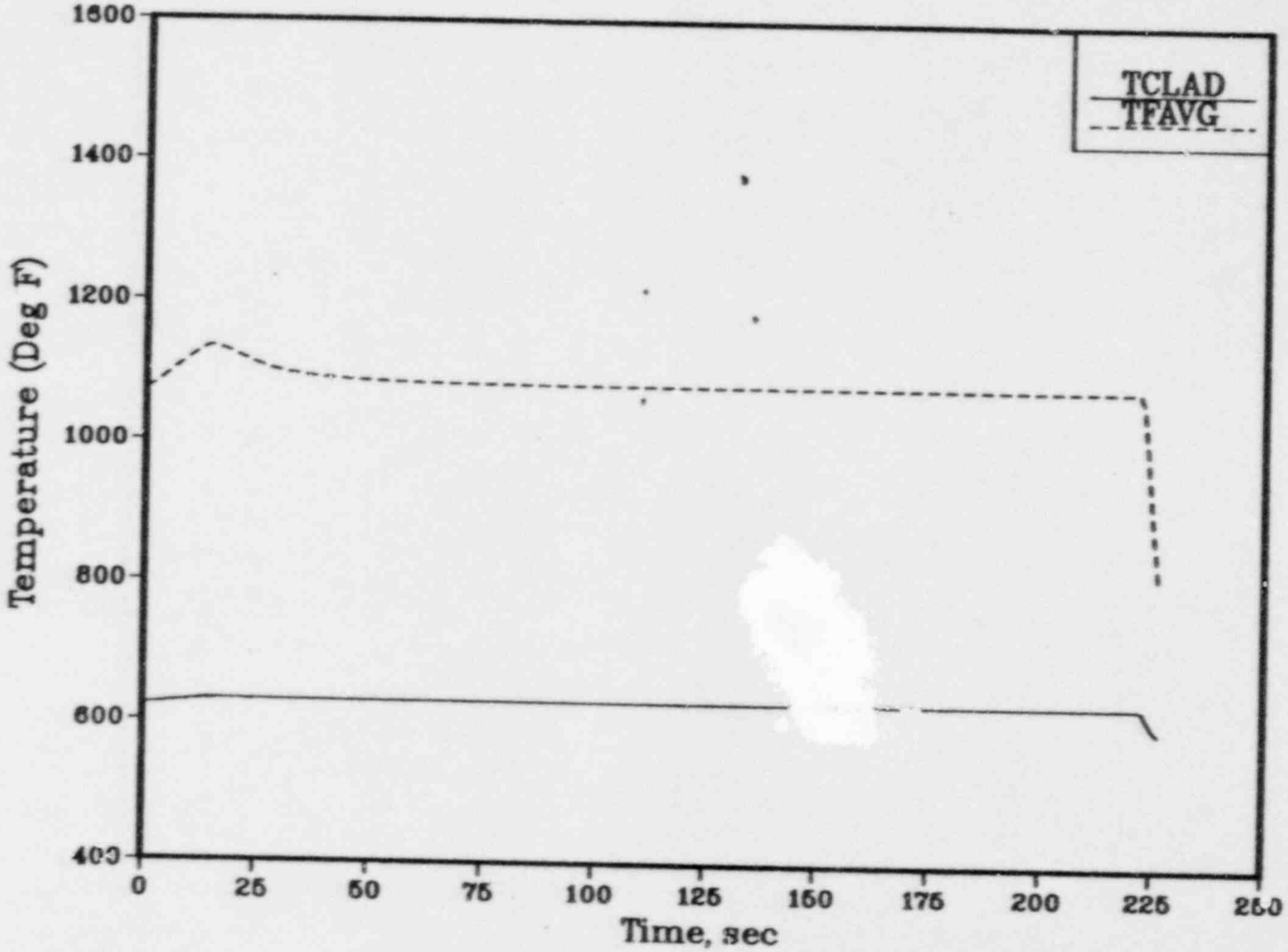


Figure 15.4.2-16 Average Fuel and Clad Temperatures for Uncontrolled Bank Withdrawal from Full Power

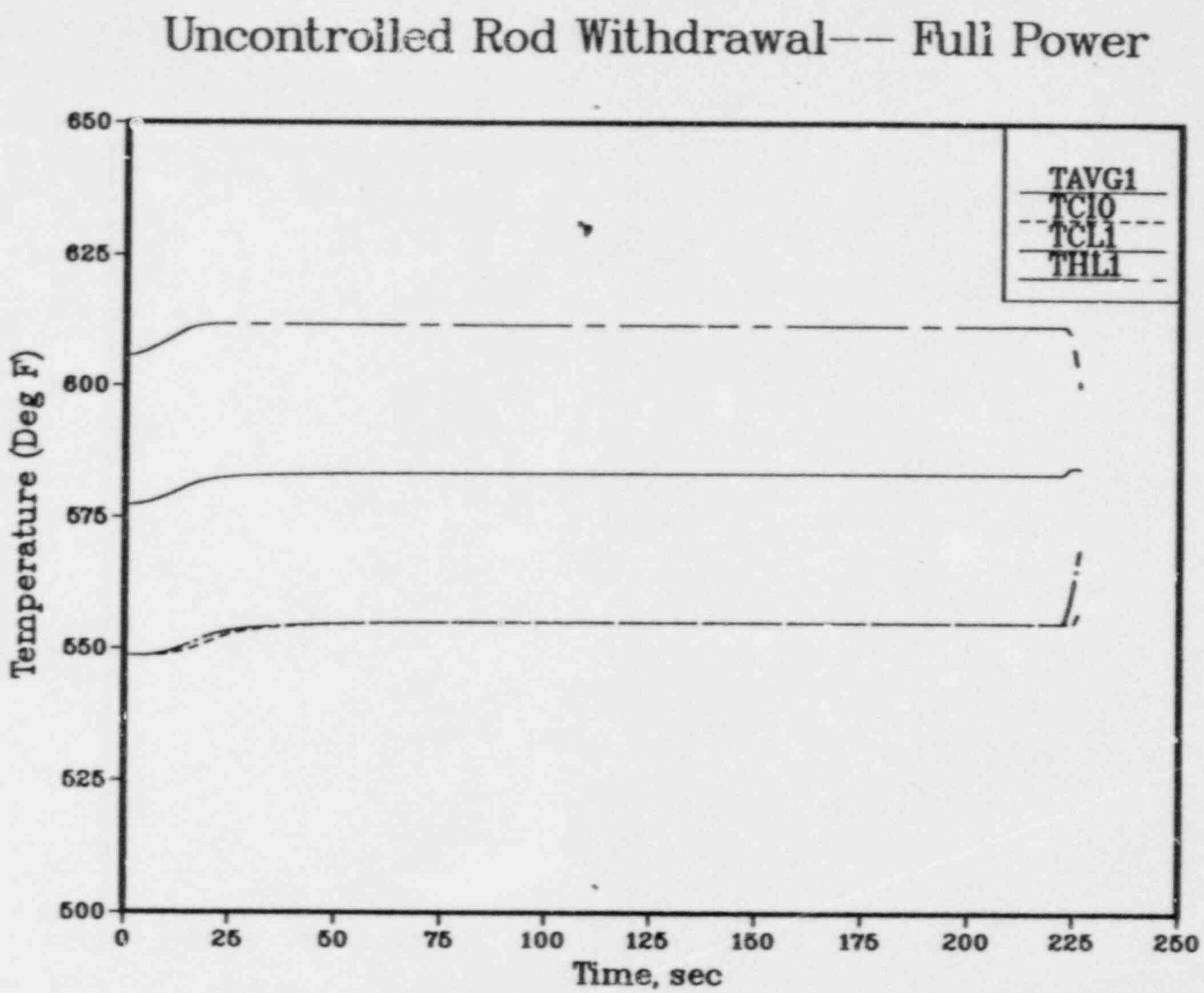


Figure 15.4.2-17 Reactor Coolant Temperatures for Uncontrolled Bank Withdrawal from Full Power

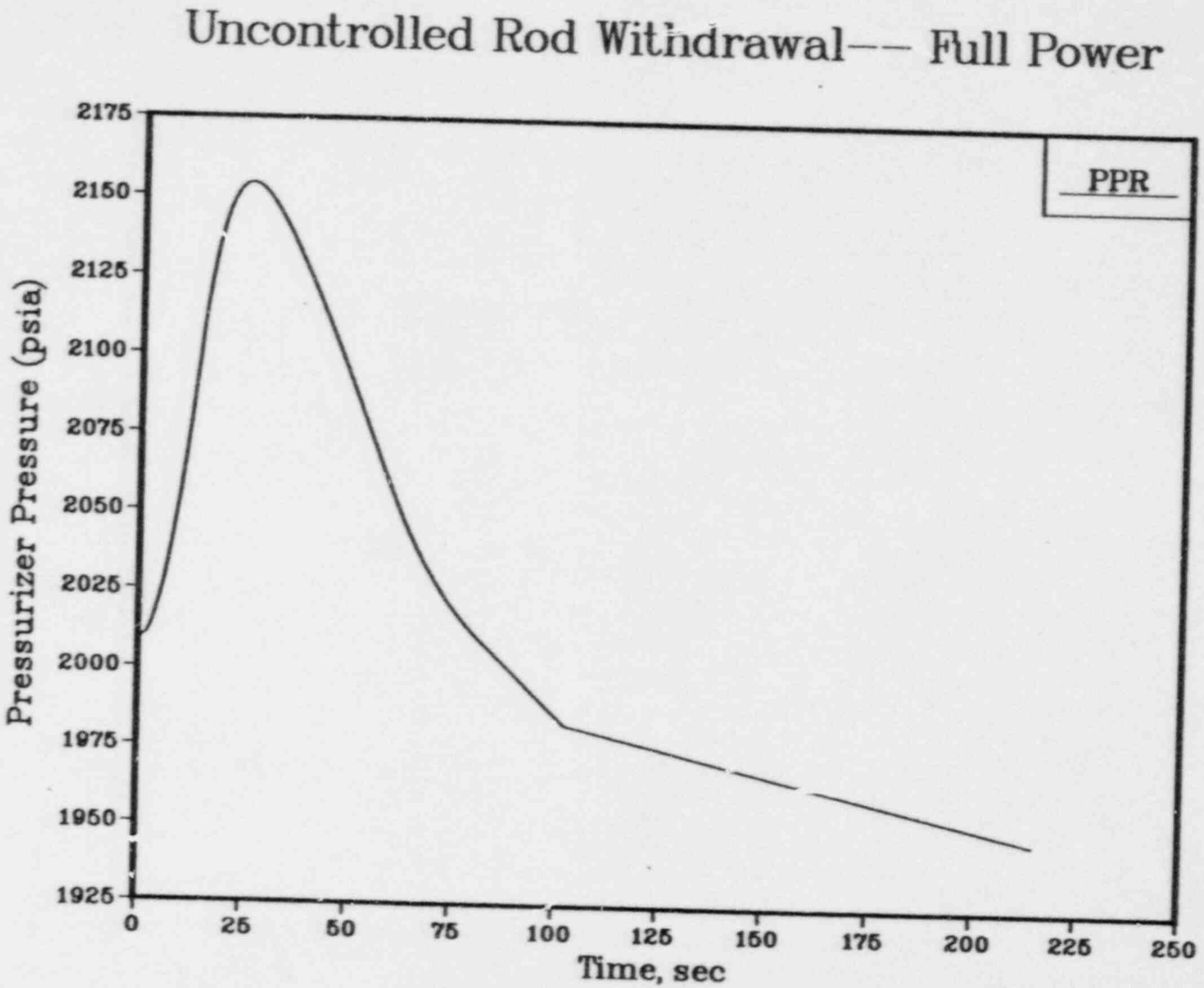


Figure 15.4.2-18 Pressurizer Pressure for Uncontrolled Bank Withdrawal from Full Power

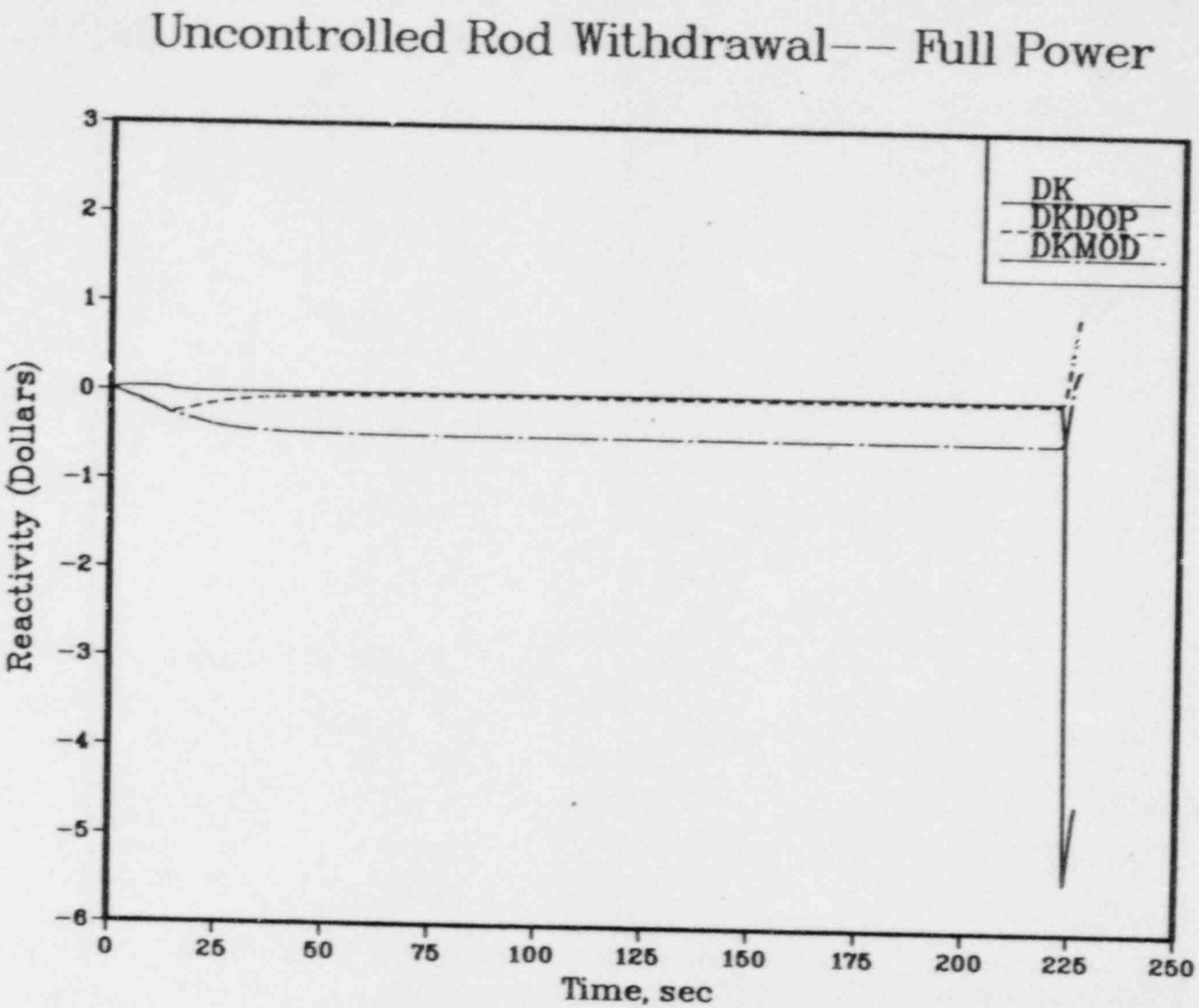


Figure 15.4.2-19 Reactivities for Uncontrolled Bank Withdrawal from Full Power

Uncontrolled Rod Withdrawal-- Full Power

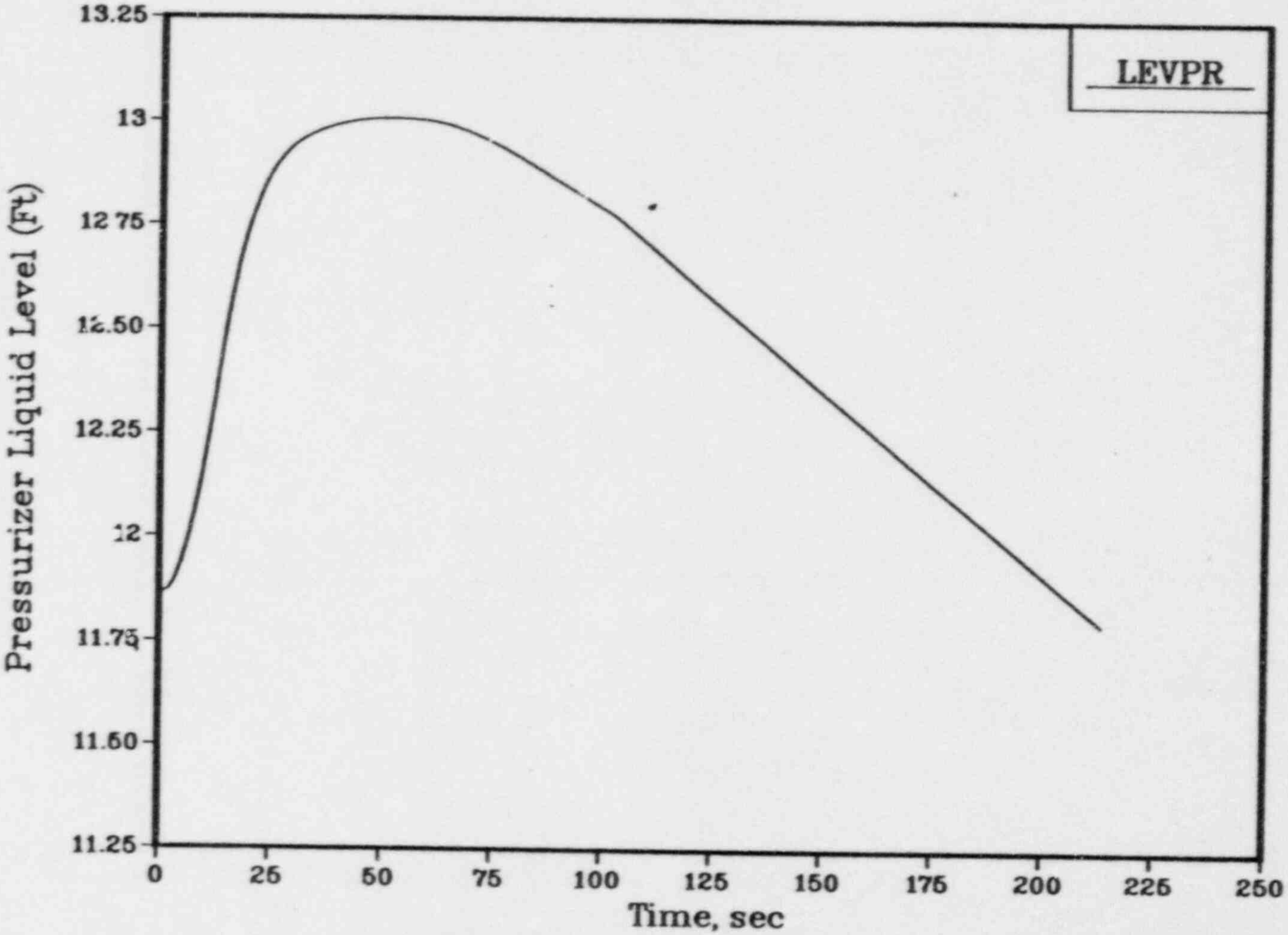


Figure 15.4.2-20 Pressurizer Liquid Level for Uncontrolled Bank Withdrawal from Full Power

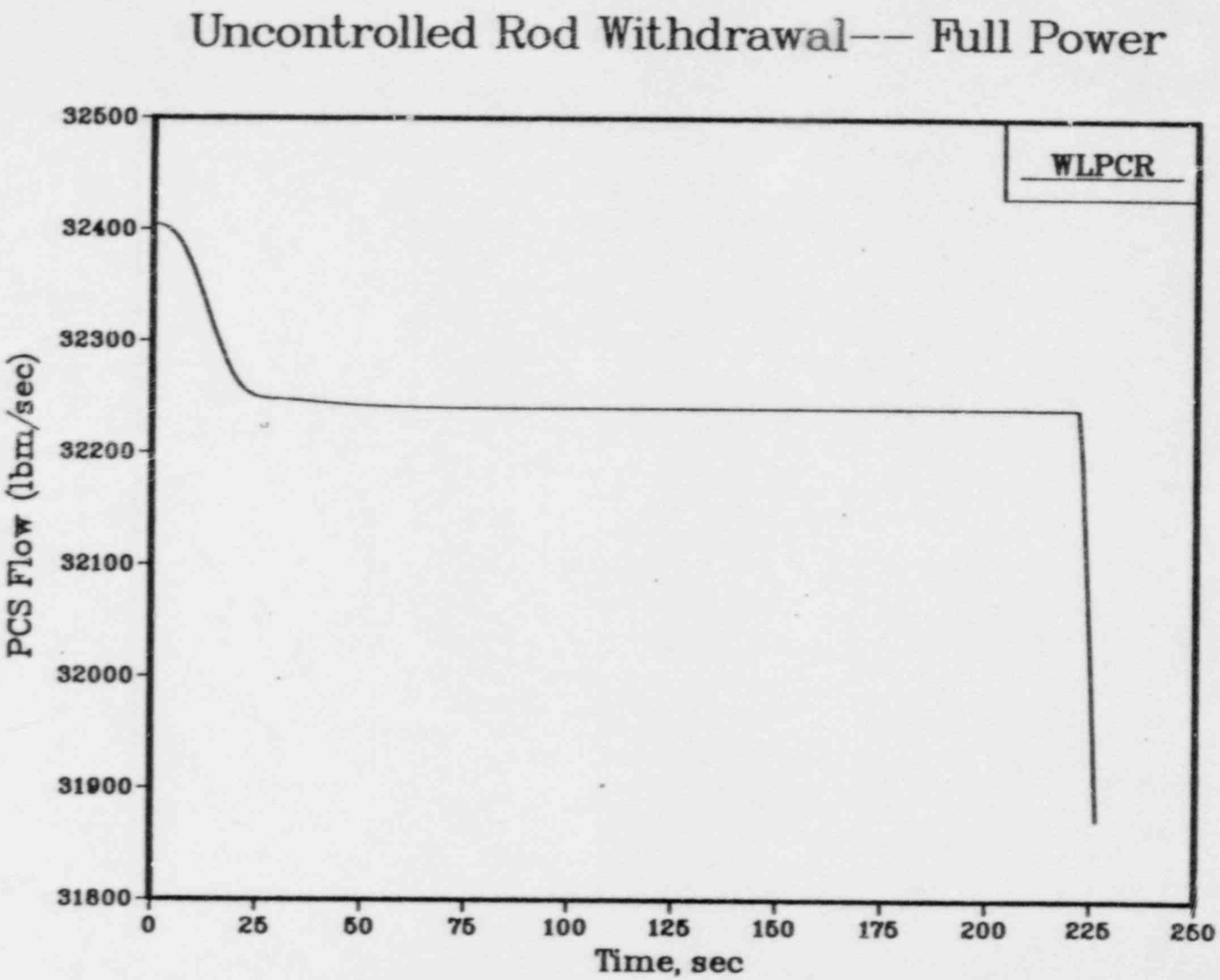


Figure 15.4.2-21 Primary Coolant Flow Rate for Uncontrolled Bank Withdrawal from Full Power

UNCONTROLLED ROD WITHDRAWAL--FULL POWER

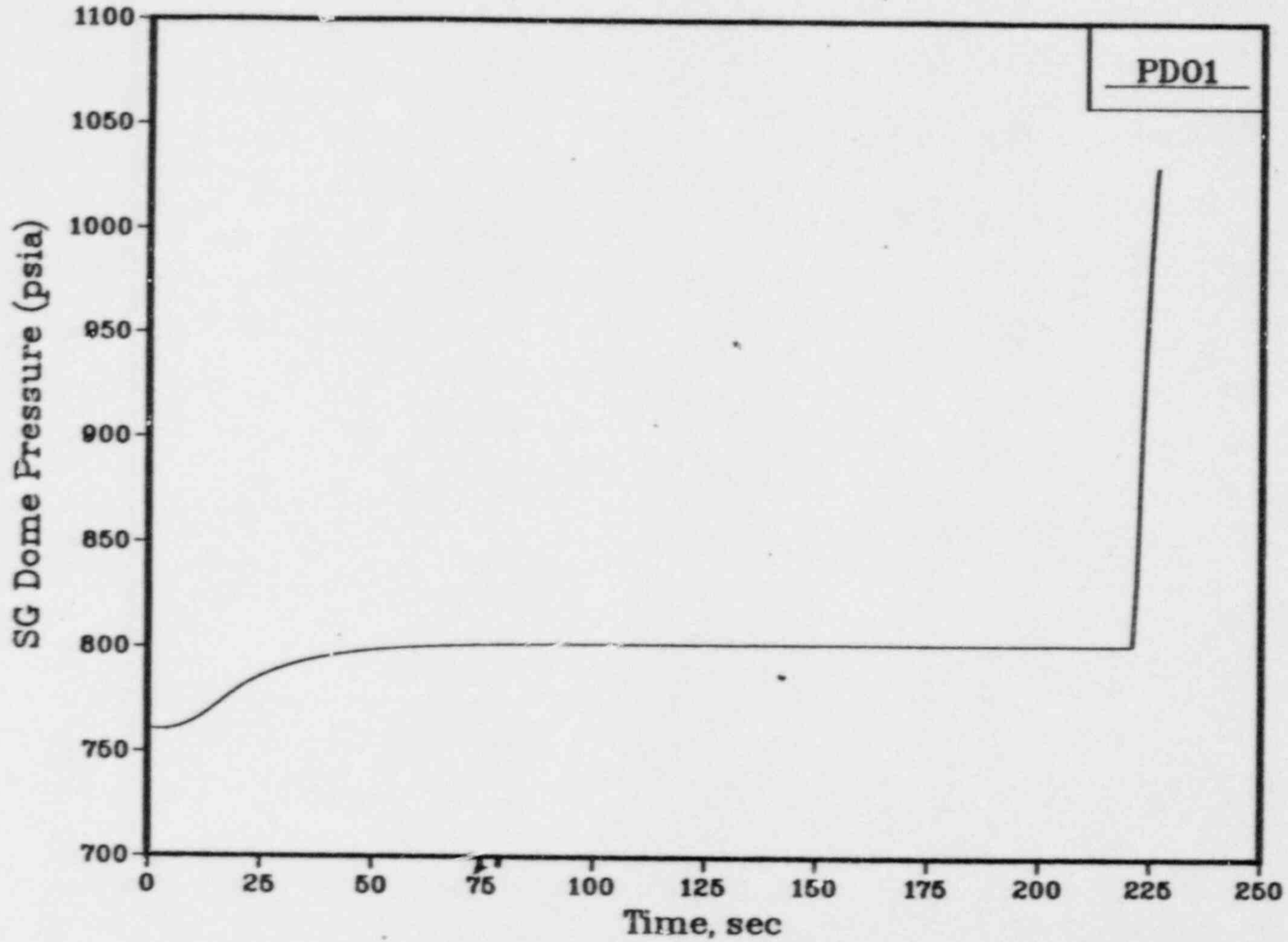


Figure 15.4.2-22 Secondary Pressure for Uncontrolled Bank Withdrawal from Full Power

Uncontrolled Rod Withdrawal-- Full Power

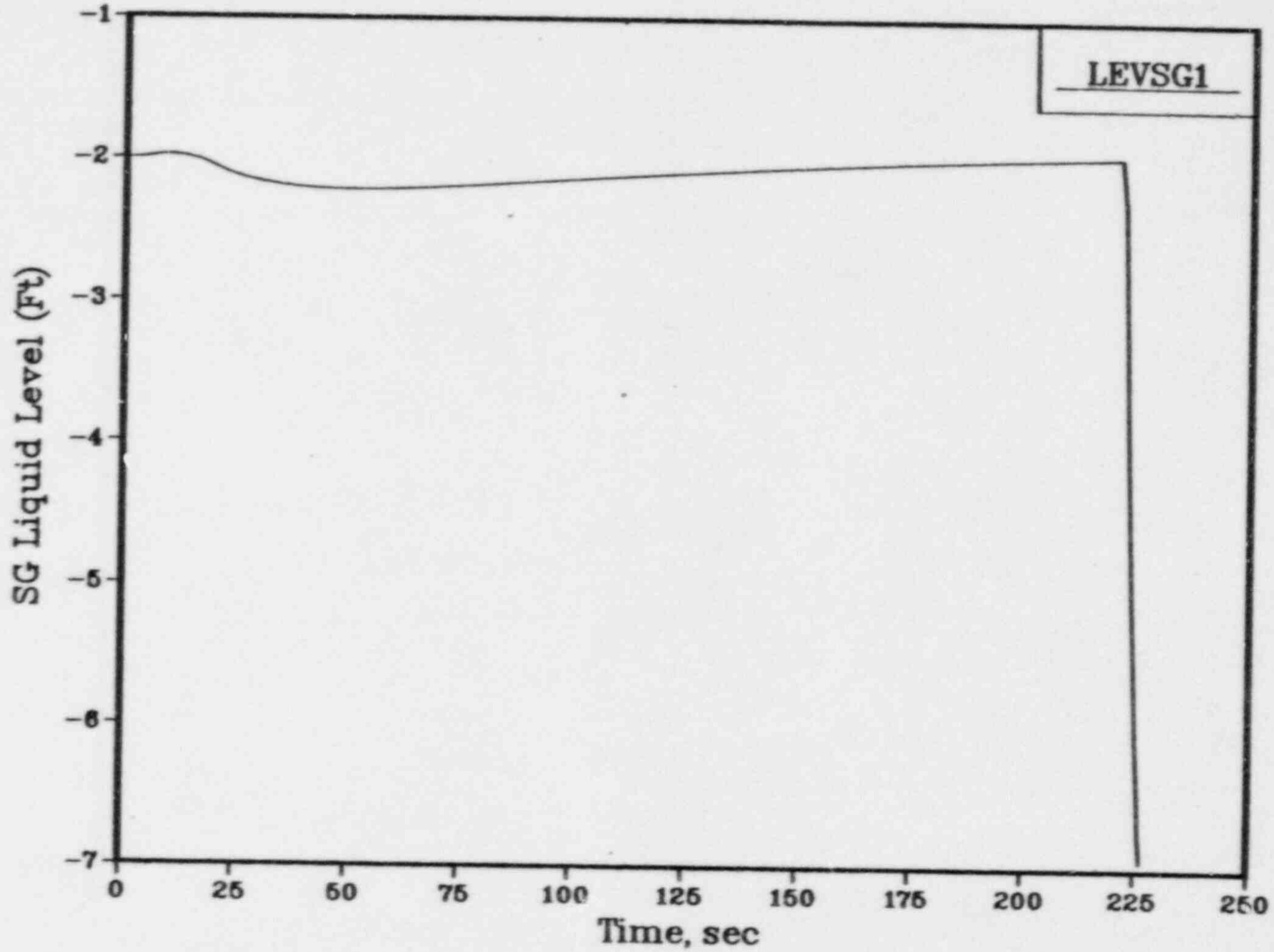


Figure 15.4.2-23 Steam Generator Liquid Level for Uncontrolled Bank Withdrawal from Full Power

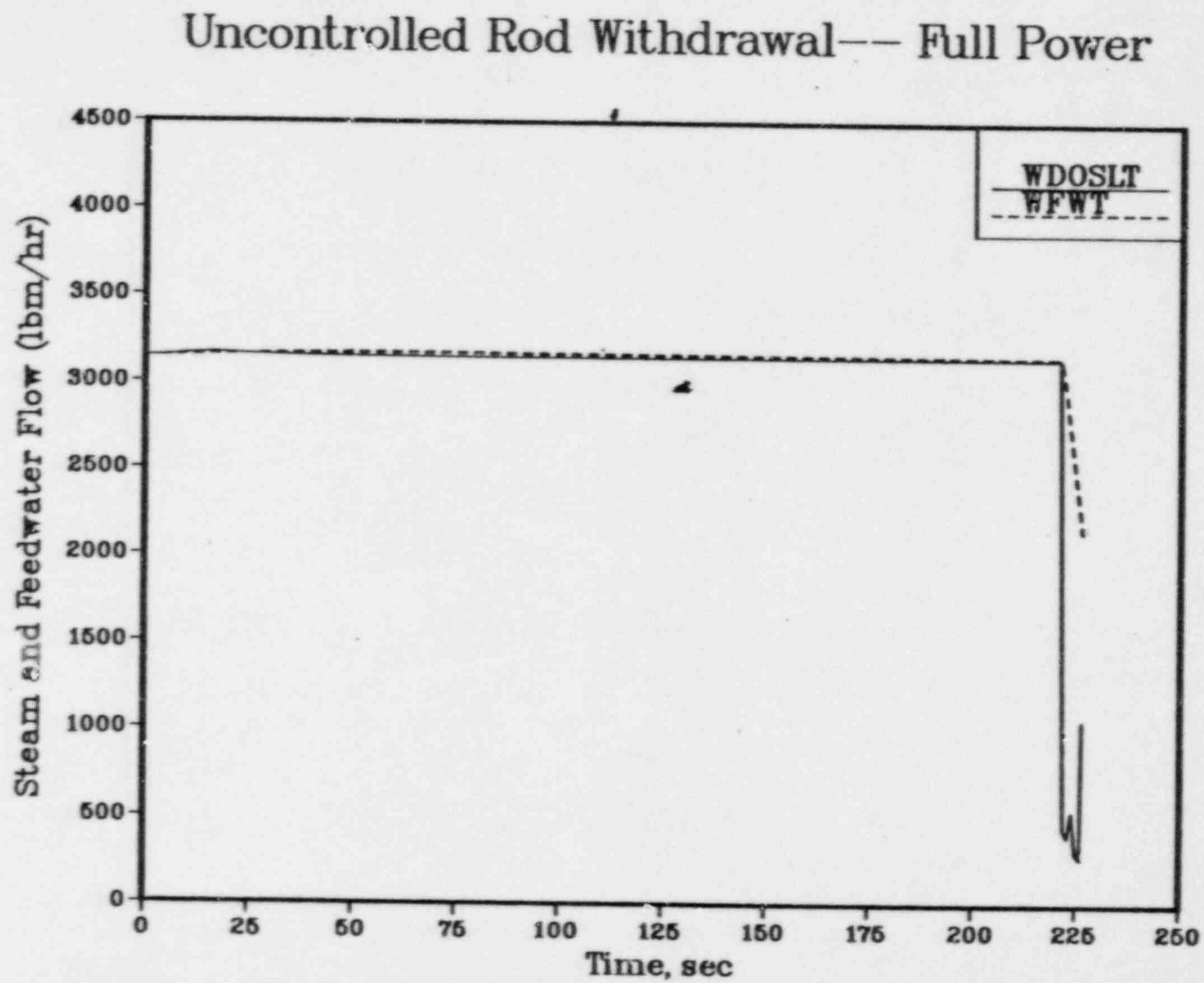


Figure 15.4.2-24 Secondary Steam and Feedwater Flow Rates for Uncontrolled Bank Withdrawal from Full Power

15.4.3 CONTROL ROD MISOPERATION

The control rod misoperation events encompass transient and steady state configurations resulting from different event initiators. The specific events analyzed under this event category are:

- Dropped control rod or bank;
- Statically misaligned control rod or bank;
- Single control rod withdrawal.

The rod drop events challenge the acceptance criteria only in Mode 1 (Chapter 15.0.2) operation. The static misalignment events challenge the acceptance criteria in Mode 1 and 2 operation, while the rod withdrawal events challenge them in operating Modes 1 through 5.

15.4.3.1 Identification of Causes and Event Description

The dropped rod and dropped bank events are initiated by a de-energized control rod drive mechanism or by a malfunction associated with a control rod bank. The dropped rod events are classified as Moderate Frequency events. Acceptance criteria are given in 15.0.1.1.

In the dropped rod or dropped bank events, the reactor power initially drops in response to the insertion of negative reactivity. This results in reduction of the moderator temperature due to a mismatch between core power being generated and secondary system load demand. The core power redistributes due to the local power effect of the dropped assembly or bank. The reactor power will return to the initial level due to the combined effects of a negative moderator temperature coefficient and reduced moderator temperature. The moderator temperature will not decrease below the temperature necessary to return the core to initial power because at that temperature, the core power and secondary system load demand are equalized, removing the driving force for further moderator cooldown. The rod and bank drop events challenge the DNBR SAFDL because of the increased radial peaking and the potential return to initial power.

The static misalignment events occur when a malfunction of the Control Rod Drive mechanism causes a control rod to be out of alignment with its bank, i.e., either higher or lower than any of the other control rods in the same bank or when a bank(s) is out of alignment with the Power Dependent Insertion Limit (PDIL). The reactor is at steady state, rated full power (Mode 1) or part-power (Mode 2) conditions with enhanced power peaking. This event is classified as a Moderate Frequency occurrence. Acceptance criteria are given in 15.0.1.1.

In the static rod misalignment event, a control bank is inserted but one of the rods remains in a withdrawn state. This results in a local increase of the radial power peaking factor and a corresponding reduction in the DNB margin. The most severe misalignment occurs at full power operation, with one bank inserted beyond its control rod insertion limit and one of the bank control assemblies fully withdrawn. The radial power redistribution consequences of a reverse misalignment, wherein one rod is inserted while the bank remains withdrawn, are essentially the same as the dropped rod event. The bank misalignment event occurs when one bank is inserted or withdrawn beyond the PDIL. The situation of concern is the power interval between 35% to 65% where control rod banks 3 and 4 are used.

The rod withdrawal event is initiated by an electrical or mechanical failure in the Rod Control System that causes the inadvertent withdrawal of a single control rod. A rod is withdrawn from the reactor core causing an insertion of positive reactivity which results in a power excursion transient. The movement of a single rod out of sequence from the rest of the bank results in a local increase in the radial power peaking factor.

The combination of these two factors results in a challenge to DNB margin. The system response is essentially the same as that occurring in the Uncontrolled Bank Withdrawal event at power (15.4.2).

Acceptable outcomes for these events rely only on the Reactor Protection System (RPS) or on the Technical Specifications limiting the conditions of operation. The elements of the RPS challenged are redundant and have been designed to provide their function in the event of a single failure in the RPS. Single failures in the RPS or Engineered Safety Features thus do not affect event results. Single failure criteria for the Palisades plant are given in 15.0.11

15.4.3.2 <u>Analysis Method</u>

The events considered have in common the radial redistribution of power in the core, and can result in radial peaking factors in excess of Technical Specification limits. The analyses are performed by coupling a conservative power peak to transient response and DNB calculations. The power peak associated with each event is characterized through an augmentation factor which relates the maximum power peak to the steady state power peak. The steady state power distributions and augmentation factors are calculated with the XTGPWR[25] reactor simulator. DNB calculations are performed using the XCOBRA-IIIC code[11].

The analysis of rod and bank drop events is performed using XTGPWR, XCOBRA-IIIC and PTSPWR. The XTGPWR code is used to calculate neutronic parameters such as rod/bank worth and power peaking augmentation factors. A radial power peaking augmentation factor on $F^N_{\Delta H}$ is included in the XCOBRA-IIIC MDNBR calculation to account for radial power redistribution effects typical of the event. The dropped bank event is distinguished from the dropped rod event by the greater magnitude of augmentation factors. However, PTSPWR2 is used to confirm that even the minimum worth bank drop would cause a trip on variable overpower. The variable high power trip is set at a power level 10% higher than actual power for decreasing power, but remains constant for increasing power. Therefore, bank drop augmentation factors are not considered in the XCOBRA-IIIC MDNBR calculations.

Simulation of the system transient for rod drops is not performed. Because the secondary system load demand remains constant through the event, the moderator will continue to cool down until moderator feedback is sufficient to restore the initial power level. At that point, the moderator temperature stabilizes because no mismatch between core power production and secondary system load demand exists. The transient thus results in a new steady state condition characterized by a power level equal to the initial power and a core coolant temperature substantially reduced from the initial condition value. During the event, a reduction in primary system pressure also occurs; the impact of this reduction on calculated DNBR is more than compensated by the reduction in coolant temperature which occurs in the transient. The transient DNBR is therefore evaluated with an XCOBRA-IIIC calculation using the initial condition power, coolant temperature, pressure, and flow. The redistribution of the radial peaking factor is incorporated as noted above. Conditions employed in the analysis are given in Table 15.4.3-1.

In the analysis of the statically misaligned rod, primary system pressure, core inlet temperature, and coolant flow rate at the rated full power operating point are input into the XCOBRA-IIIC code to calculate MDNBR. The rated full power core average clad surface heat flux is input to the MDNBR calculation after having been adjusted to include the design radial and axial power peaking distribution factors and a radial peaking augmentation factor calculated to bound the radial power redistribution of a misaligned rod. Since the augmentation factor for the rod drop event was greater than for the rod misalignment event, the rod misalignment MDNBR is bounded by the rod drop

MDNBR. Conditions employed in the analysis are given in Table 15.4.3-1. The radial peaking augmentation factor represents, conservatively, the most limiting static misalignment, i.e., bank 4 fully inserted with one rod fully withdrawn (bank 4 is 99 inches out of alignment with rated power PDIL). By determining the radial peaking augmentation factor in this manner, MDNBRs for this event are conservatively calculated.

Augmentation factors for bank misalignments are calculated by XTGPWR using a 16 inch misalignment from PDIL at 50% and 65% power levels. The 50% power level is chosen because the Technical Specifications allow the largest radial peaking at that level (15% higher than rated power peaking). The 65% power level is chosen because, according to the rated power PDIL, bank 3 begins inserting at that level. The positions of banks 3 and 4 were individually withdrawn and inserted 16 inches beyond PDILs to determine the highest augmentation factor for each power level. Then XCOBRA-IIIC calculations are made using these augmentation factors at their respective power levels to determine MDNBR. Conditions used in the analyses are given in Table 15.4.3-1.

In the analysis of the single rod withdrawal event, the core boundary conditions of average heat flux, temperature, pressure, and flow are selected to conservatively bound rated and power operation (Mode 1 and Mode 2). The bank withdrawal analysis (15.4.2) considers reactivity insertion rates down to 10^{-5} $\Delta\rho/s$. Because of the lower worth of a single rod, this range was extended down to 10^{-6} $\Delta\rho/s$ for this analysis. The boundary conditions used in the XCOBRA-IIIC calculation of MDNBR are obtained from the limiting transient response from Event 15.4.2, ($\Delta\rho/s > 10^{-5}$) and from further analyses with $\Delta\rho/s < 10^{-5}$. Those conservatively biased core boundary conditions are then combined in an XCOBRA-IIIC calculation with a radial augmentation peaking factor calculated to bound the possible single rod withdrawal radial power redistribution. Conservative conditions employed in the analysis are given in Table 15.4.3-1.

Operating Modes 3, 4 and 5 (Table 2.1) are dispositioned to be analyzed for three pump operation. However, consequences of a single rod withdrawal from these modes are either bounded or the event does not challenge the acceptance criteria. Mode 3 operation (Reactor Critical) is defined as having a power greater than 10^{-4}% and T_{ave} greater than 525°F. Since the peak power obtained during a low power reactivity insertion increases with increasing insertion rate, the results for a single rod withdrawal are bounded by the results for a bank withdrawal (Event 15.4.1 where the insertion rate is much larger). Mode 4 operation (Hot Standby) applies when the power is between 10^{-4}% and 2% and any of the control rods are withdrawn. The peak heat flux following a rod withdrawal decreases with increasing initial power level. Since Mode 3 includes 10^{-4}% power, Mode 4 is bounded by the results of Mode 3. Finally, Mode 5 operation (Hot Shutdown) applies when the power is less than 10^{-4}% and T_{ave} is greater than 525°F. The most reactive rod worth is less than the required shutdown margin; therefore, the reactor could not become critical by the withdrawal of any single rod.

15.4.3.3 <u>Definition of Events Analyzed and Bounding Input</u>

The dropped rod and single rod misalignment events are analyzed at the rated power condition with conservative allowances applied in a direction to minimize DNBR. These biases are listed below:

Table 15.4.3-0 Conservative Assumptions for the Dropped Rod and Single Rod Misalignment Events

Power	Nominal +2%
Core Inlet Temperature	Rated +5°F
Pressure	Nominal -50 psi
Flow	Nominal -3%.

The bank misalignment event is analyzed at nominal power levels of 50% and 65% of rated using the same biases noted above.

The axial powe[illegible]ons used in these analyses correspond to those given in Section 15.0..

The single withdrawal event is analyzed at conditions that exist at the time of MDNBR as calculated by PTSPWR2 during the most limiting uncontrolled rod or bank withdrawal event. Analysis is performed for nominal power levels of 50% and 100%. The biasing used is listed in 15.4.2.3.

15.4.3.4 Analysis of Results

Minimum DNBRs obtained are given in Table 15.4.3-2 for the control rod misoperation events.

For the dropped rod and bank events, radial peaking augmentation factors are calculated at full power for BOC, MOC and EOC conditions. Table 15.4.3-3 gives the augmentation factors for single rod drop events. The maximum augmentation peaking factor of 1.15 occurs at BOC when rod P12 in bank A is dropped. The XCOBRA-IIIC MDNBR calculation further bounds the radial redistribution by using a larger augmentation factor of 1.16. The resulting MDNBR is greater than the XNB critical heat flux thermal limit of 1.17.

Table 15.4.3-4 lists the worth and augmentation factors for bank drop events. The maximum augmentation factor is 1.62 which certainly challenges the DNBR SAFDL. However, bank drops should be mitigated by the variable overpower trip. To confirm this, a transient analysis is performed that uses the minimum worth bank and EOC kinetics where the fuel and moderator temperature reactivity coefficients are most negative. These choices minimize the change in power level thereby minimizing the occurrence of the variable overpower trip.

The PTSPWR2 calculation uses bounding values of -1.76×10^{-5} $\Delta\rho/°F$ for the fuel temperature coefficient and -3.5×10^{-4} $\Delta\rho/°F$ for the moderator temperature coefficient. The minimum worth is 489 pcm which occurs when bank 4 is dropped during BOC. The resulting power level versus time is plotted in Figure 15.4.3-1.

The transient calculation does not include the variable overpower trip. However, it is implicitly included by finding the time that the overpower trip is exceeded from the power trace. The variable overpower trip function remains 10% (of rated) above the power level as it decreases, but remains constant as the power increases. Figure 15.4.3-1 shows that the power drops to 61% at 3 seconds then begins to rise. Using an uncertainty of 5.5% (Table 15.0.7-1), the variable overpower trip is activated at a power of 61% + 10% + 5.5% = 76.5% which occurs at about 6.2 seconds. Thus, the DNBR SAFDL is not challenged by the bank drop event.

For the statically misaligned rod event, augmentation factors are calculated for the full length misalignment of a rod in bank 4 (bank fully inserted, one rod positioned out of core) at full power and for BOC, MOC and EOC conditions. A maximum augmentation factor of 1.14 occurs at MOC. This event is bounded by the rod drop event, since the MDNBR calculation uses the same conditions as the rod drop event but a lower augmentation factor.

Augmentation factors are calculated for the misaligned bank event at the 50% and 65% power levels using a 16 inch offset from PDIL. The maximum augmentation factors are 1.013 for bank 4 inserted beyond PDIL at 50%, MOC and 1.018 for bank 4 inserted beyond PDIL at 65%, BOC. To further bound the power redistribution, the MDNBR calculations used an augmentation of 1.05. The MDNBR for these events is greater than the thermal limit of 1.17.

The conditions used in the evaluation of MDNBR for the single rod withdrawal event encompass reactivity insertion rates from 10^{-6} to 4.5×10^{-5} $\Delta\rho$/sec at BOC and EOC for 100% and 50% power levels. The reactivity rate of 4.5×10^{-5} $\Delta\rho$/sec conservatively represents the maximum insertion rate for this event based on the worth of a single control rod and the withdrawal speed. At 50% rated power, MDNBR occurs for single rod withdrawal with a reactivity insertion rate of 3×10^{-5} $\Delta\rho$/sec at EOC conditions. At EOC, the negative MTC tends to suppress the core power excursion for heatup events. However, when initiated from 50% power, this insertion rate does not result in a reactor trip leading to a lower MDNBR than the BOC counterpart. For the 100% power case, the limiting MDNBR for a single rod withdrawal occurs at EOC with a reactivity insertion rate of 4.5×10^{-5} $\Delta\rho$/sec. For this case, the event is terminated by a TM/LP trip after the core conditions sufficiently degrade to result in a limiting MDNBR.

The radial peaking augmentation factors used for the single control rod withdrawal event were calculated for both 100% and 50% power for BOC, MOC and EOC conditions by individually withdrawing each rod from PDILs to fully withdrawn. The worst peaking augmentation factors occurred for the 100% power case when Group 4 was at its PDIL with control rod S06 fully withdrawn; and, for the 50% power, when Groups 3 and 4 were at their respective PDILs with control rod S18 fully withdrawn. The maximum augmentation factor is 1.076 for the 100% case and 1.289 for the 50% case.

The calculated MDNBRs for these cases remain above the 1.17 thermal limit.

15.4.3.5 <u>Conclusion</u>

These moderate frequency events result in MDNBRs greater than the XNB critical heat flux correlation safety limit. Thus, the DNBR SAFDL is not penetrated. The maximum peak linear heat rate for these events is 17.4 kw/ft which is below the fuel centerline melt criterion of 21 kw/ft. Therefore, applicable acceptance criteria for these events are met.

Table 15.4.3-1 Summary of Conditions for Control Rod Misoperation Events

<u>Dropped Control Rod</u> (100% Power)	
Power, MW_t	2580.6
Core Inlet Temperature, °F	548.65
Pressurizer Pressure, psia	2010
Vessel Flow Rate, lbm/hr	120.3×10^6
Maximum Augmentation Factor (XTGPWR)	1.15
Augmentation Factor used in XCOBRA-IIIC	1.16
Hot Rod Radial Peaking Factor	2.02
<u>Statically Misaligned Control Rod</u> (100% Power)	
Power, MW_t	2580.6
Core Inlet Temperature, °F	548.65
Pressurizer Pressure, psia	2010
Vessel Flow Rate, lbm/hr	120.3×10^6
Maximum Augmentation Factor (XTGPWR)	1.14
Augmentation Factor used in XCOBRA-IIIC	1.16
Hot Rod Radial Peaking Factor	2.02
<u>Statically Misaligned Control Rod Bank (50% Power)</u>	
Power, MW_t	1315.6
Core Inlet Temperature, °F	548.65
Pressurizer Pressure, psia	2010
Vessel Flow Rate, lbm/hr	120.3×10^6
Maximum Augmentation Factor (XTGPWR)	1.018

Table 15.4.3-1 (Cont.)

Statically Misaligned Control Rod Bank(Cont.) (50% Power)	
Augmentation Factor used in XCOBRA-IIIC	1.05
Hot Rod Radial Peaking Factor	2.02
Statically Misaligned Control Rod Bank (65% Power)	
Power, MW_t	1695.1
Core Inlet Temperature, °F	548.65
Pressurizer Pressure, psia	2010
Vessel Flow Rate, lbm/hr	120.3×10^6
Maximum Augmentation Factor (XTGPWR)	1.018
Augmentation Factor used in XCOBRA-IIIC	1.05
Hot Rod Radial Peaking Factor	2.02
Single Control Rod Withdrawal (100% Power)	
Power, MW_t	2695.4
Core Inlet Temperature, °F	554.06
Pressurizer Pressure, psia	1931.2
Vessel Flow Rate, lbm/hr	116.1×10^6
Maximum Augmentation Factor (XTGPWR)	1.076
Augmentation Factor used in XCOBRA-IIIC	1.076
Hot Rod Radial Peaking Factor	1.88

Table 15.4.3-1 (Cont.)

Single Control Rod Withdrawal (50% Power)	
Power, MW_t	1427.4
Core Inlet Temperature, °F	568.45
Pressurizer Pressure, psia	1964
Vessel Flow Rate, lbm/hr	120.1×10^6
Maximum Augmentation Factor (XTGPWR)	1.289
Augmentation Factor used in XCOBRA-IIIC	1.289
Hot Rod Radial Peaking Factor	2.58

Table 15.4.3-2 Summary of MDNBRs for Control Rod Misoperation Events

Event (Power)	Operating Mode*	MDNBR
Dropped Control Rod (100%)	1	1.301
Statically Misaligned Control Rod (100%)	1	Bounded (Dropped Rod)
Statically Misaligned Bank (50%)	2	2.717
Statically Misaligned Bank (65%)	2	2.092
Rod Withdrawal (100%)	1	1.273
Rod Withdrawal (50%)	2	1.551
Rod Withdrawal (10^{-4}%)	3	Bounded (15.4.1)
Rod Withdrawal (10^{-4}%)	4	Bounded (15.4.1)
Rod Withdrawal ($\leq 10^{-4}$%)	5	Subcritical

*These modes are defined in 15.0.2.

Table 15.4.3-3 Radial Peaking Augmentation Factors for Individually Dropped Control Rods at Full Power

Control Rod		Augmentation Factor		
Group	Location	BOC	MOC	EOC
1	W15	1.13	1.12	1.12
2	P15	1.12	1.10	1.11
3	W12	1.13	1.12	1.12
3	L12	1.06	1.03	1.01
4	S18	1.08	1.12	1.13
A	W18	1.07	1.09	1.09
A	P12	1.15	1.12	1.10
B	S15	1.14	1.14	1.14

Table 15.4.3-4 Worth and Radial Peaking Augmentation Factor for Individually Dropped Control Banks at Full Power

Control	BOC		MOC		EOC	
Bank	Worth(pcm)	Augmt	Worth(pcm)	Augmt	Worth(pcm)	Augmt
1	-1145	1.62	-1193	1.46	-1222	1.38
2	- 502	1.20	- 517	1.15	- 541	1.14
3	- 740	1.15	- 704	1.13	- 706	1.14
4	- 489	1.17	- 555	1.08	- 598	1.09
A	-1534	1.54	-1567	1.36	-1586	1.23
B	-1266	1.31	-1355	1.19	-1434	1.17

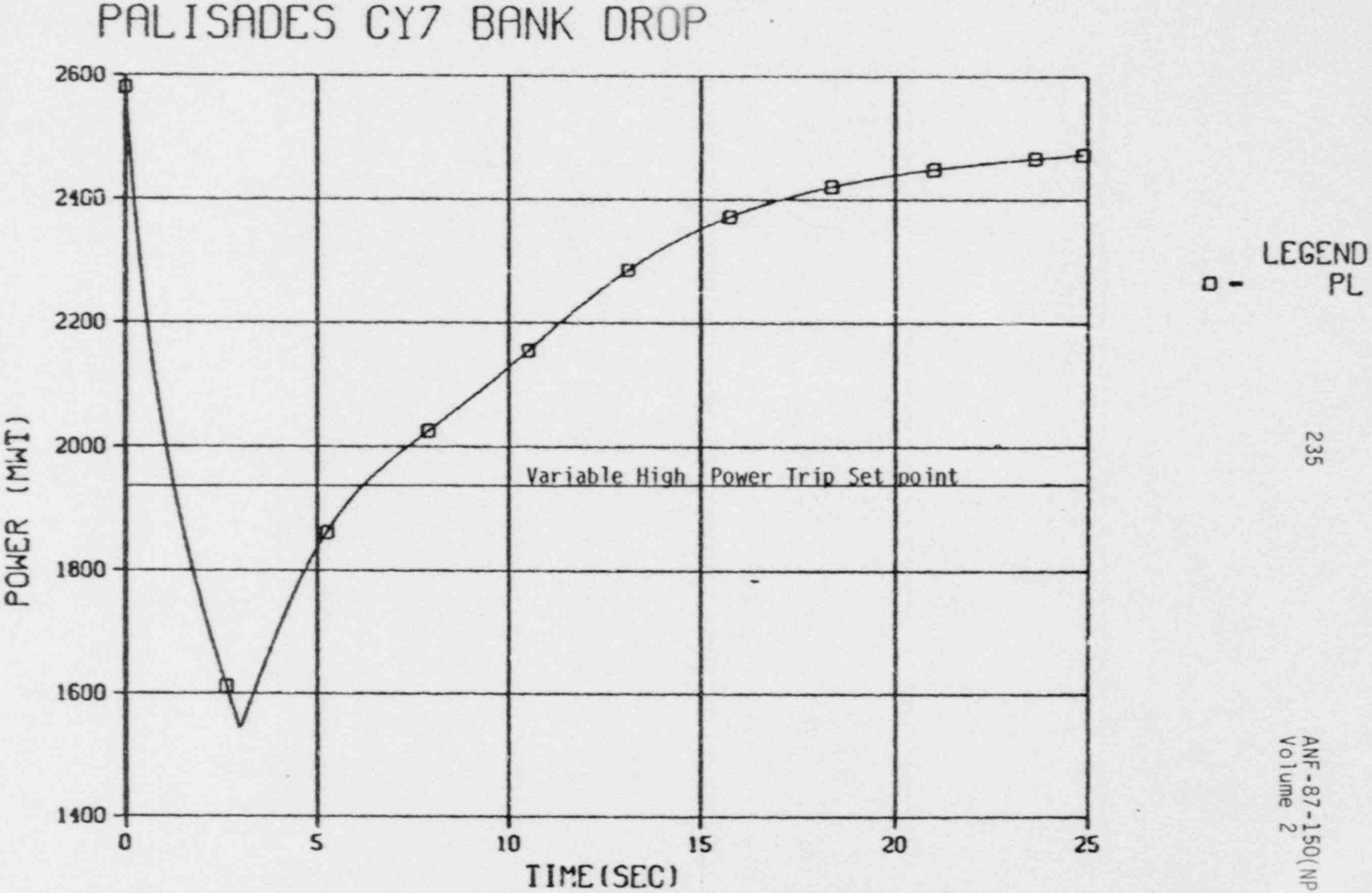


Figure 15.4.3-1 Reactor Power Level for the Minimum Worth Bank Drop Event

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT

15.4.6.1 Identification of Causes and Event Description

The Chemical and Volume Control System (CVCS) regulates both the chemistry and the quantity of coolant in the Primary Coolant System. Changes in boron concentration in the Primary Coolant System are a part of normal plant operation, compensating for long-term reactivity effects such as fuel burnup, xenon transients and plant cooldown.

Boron dilution is a manual operation, conducted under strict administrative control and in accordance with detailed operating procedures, which specify permissible limits on rate and magnitude of any increment of boron dilution. Because of the procedures involved and the numerous alarms and indications provided, the probability of a sustained erroneous dilution is very small. Administrative procedures will protect against protracted operator neglect to add boron to compensate for reactivity change included by post-shutdown cooldown or xenon decay.

The operation of the primary makeup water transfer pumps provides the normal supply of makeup water to the Primary Coolant System via charging pumps. Inadvertent dilution can be readily terminated by isolating the unborated water source or by stopping either the makeup water transfer pumps or the charging pumps.

During normal operation, concentrated boric acid solution is automatically blended with primary makeup water to the approximate concentration present in the reactor coolant and is introduced into the volume control tank discharge header automatically in response to a low-level signal from the volume control tank. A malfunction in this system (such as failure of the boric acid pumps to start or of the boric acid control valve to open) while the operator fails to observe the alarm resulting from incorrect flow, could initiate a boron dilution incident.

Boron concentration in the Primary Coolant System can be decreased by controlled feed and bleed operation or by using the deborating demineralizer. (The deborating demineralizer is used for removal of boron when the primary coolant boron concentration is below 50 ppm.)

To add primary makeup water for boron dilution, the makeup controller mode selector switch is set to DILUTE and the primary makeup water batch quantity selector is set to the desired quantity. The makeup stop valve is then opened to initiate flow. When the specified amount has been injected, the primary makeup water control valve is closed automatically. Failure of the valve to close could, on the occasion of very low pressurizer level, result in the introduction of unborated water at the maximum capacity of all three charging pumps (133 gpm), if three pumps are available.

To cover all phases of plant operation, incidents involving inadvertent boron dilution during refueling, start-up and power operation, as well as failure to add boron after shutdown, have been analyzed.

15.4.6.2 Analysis Methods

PERFECT MIXING

The dilution process due to the inadvertent injection of primary makeup water is described by the following differential equation:

$$M * \frac{dC(t)}{dt} = -W * C(t) \qquad (1)$$

so that the dilution is given by

$$T_D = \frac{M}{W} * \ln \frac{C_{initial}}{C_{critical}}$$

where: T_D = dilution time to the critical boron concentration

M = mass of water in the primary system using the applicable temperature and pressure conditions

C = boron concentration in the primary system

W = mass flow of unborated water using the applicable temperature and pressure conditions

The dilution due to the accidental transfer of the contents of the iodine removal systems to the primary coolant system is described by the following equation:

$$C_f(V_o + V_I) = C_oV_o$$

where: C_o = initial boron concentration

V_o = initial reactor coolant volume

V_I = iodine removal system volume

C_f = final boron concentration

WAVE FRONT/SLUG FLOW

If the main reactor coolant pumps are not running and one shutdown coo ing pump is running to remove decay heat, the possibility of a wave front/slug flow type dilution event exists. The perfect mixing equations described above are not applicable. In this case with symmetric charging flow the boron concentration is given by the following expression. (See Appendix A for derivation of this expression).

$$C = \left(\frac{\dot{m}_{RHR}}{\dot{m}_{RHR} + \dot{m}_c} \right)^N C_o$$

where: C_o = initial boron concentration

$\dot{m}_{RHR}$ = low Pressure Safety Injection (Residual Heat Removal Flow)

$\dot{m}_c$ = charging pump flow rate

N = the number of dilution wave fronts that have entered the core

C_N = boron concentration after N dilution wave fronts have entered the core

In the event of asymmetric charging flow, the boron concentration is given by the following expression. (See Appendix A for derivation of this expression.)

$$C_N = \left[\frac{\dot{m}_{RHR/2}}{\dot{m}_{RHR}/2 + \dot{m}_c} \right] \left[\frac{\dot{m}_{RHR}}{\dot{m}_{RHR} + \dot{m}_c} \right]^{(N-1)} C_o$$

and the terms in the expression are as previously defined.

The critical boron concentration and a conservative boron worth are determined utilizing the XTGPWR reactor simulator code. Inadvertent boron dilution for all reactor operating conditions, as well as failure to add boron after shutdown have been considered.

15.4.6.3 Definition of Events and Bounding Input

15.4.6.3.1 Dilution During Refueling

The following case has been considered: Dilution from inadvertent injection of primary water. Dilution by accidental transfer of the contents of the iodine removal systems to the primary coolant system is no longer considered since this system is double valved out during outages. As such, no single failure can result in dilution by transfer of the contents of the iodine removal systems to the primary coolant system.

For dilution to occur during refueling by primary makeup water, it is necessary to have at least one makeup water transfer pump operating, one charging pump operating, and the makeup controller set for dilution. None of these conditions are required for refueling and would be in violation of operating procedures. Nevertheless, such a dilution incident has been analyzed as follows:

1) One shutdown cooling pump is running to remove decay heat.

2) The valve in the bleed-off water header from the primary coolant pumps is closed.

3) The makeup system is set for makeup at shutdown concentration.

4) The boron concentration of the refueling water is at least 1,720 ppm (Tech Spec 3.3.1) corresponding to a shutdown of at least 5.0% with all control rods withdrawn. Periodic sampling insures that the concentration is maintained above 1,720 ppm.

5) Minimum primary coolant volume for reactor vessel head removal during refueling is considered (3300 ft^3). This is the volume necessary to fill the reactor vessel above the nozzles to insure cooling via the Shutdown Cooling System.

6) The charging dilutio flow is assumed to be 40 gpm and the wave front/slug flow approach is utilized.

The operator has adequate indication of any significant boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the main control room. The count rate is a measure of the effective multiplication factor.

In the first case, with all rods out of the core, the boron concentration must be reduced from the Safety Injection Refueling Water (SIRW) tank concentration (1720 ppm) to approximately 1365 ppm before the reactor will go critical. This would take approximately 130 minutes after arrival of the first wave front. This is ample time for the operator to recognize the audible high count rate signal and isolate the reactor makeup water source by closing valves and/or stopping the primary makeup water transfer pumps.

15.4.6.3.2 Dilution During Startup

After refueling and prior to hot standby, the primary coolant system may contain water having the boron concentration corresponding to shutdown margin of 2.55% $\Delta\rho$; namely, 1315 ppm. The maximum possible rate of introduction of unborated demineralized water is 133 gpm. The volume of reactor coolant is about 9,200 ft^3, which is the total volume of the primary coolant system excluding the pressurizer. The primary coolant pumps are assumed to be running (i.e., perfect mixing is assumed).

Under these conditions the minimum time required to reduce the reactor coolant boron concentration to 1040 ppm, when the reactor would go critical is about 1.5 hours. Boron dilution for start-up will be performed under strict procedures and administrative controls.

During dilution at hot standby or critical, the operating staff will be monitoring the nuclear instruments and the boronometer readings. Abnormal change in the reading of these instruments will inform the operator that dilution is occurring. The operator will have further indication of the process from volume control tank level and from operation of the letdown diverter valve. Further, should the makeup controller fail to close the makeup stop valve, the operator has visual indication of makeup water flow and of makeup water transfer pump operation.

In any case, should continued dilution occur, the reactivity insertion rate would be less than that considered for uncontrolled rod/rod bank withdrawals. The reactor protection provided for the rod withdrawal incident will also provide protection for the boron dilution incident.

When the primary system boron concentration is being changed, at least one shutdown cooling pump or one primary coolant pump must be functioning to provide sufficient heat removal capacity. Under the condition of one operating shutdown cooling pump, imperfect mixing is conceivable. With imperfect mixing, a shutdown cooling pump greater than or equal to 1500 gpm is required to ensure that the acceptance criteria for this event is not violated. This value was calculated by evaluating the shutdown cooling pump flow rate necessary to bring the plant to a critical state in 15 minutes(1), assuming a maximum charging flow rate of 133 gpm and a reactor coolant volume of about 9200 ft^3. The required shutdown cooling pump flow rate of 1500 gpm is half of the rated flow capacity of one shutdown cooling pump.

15.4.6.3.3 Dilution During Power Operation

Inadvertent injection of primary makeup water into the primary coolant system while the reactor is at power would result in a reactivity addition initially causing a slow rise in power, temperature and possibly pressure. Assuming that unborated water is injected at the maximum possible rate of 133 gpm and that the boron concentration in the coolant system is at a value of 1365 ppm, the rate of reactivity addition would be about 5.09×10^{-6} $\Delta\rho/s$. This is much slower than the maximum rate possible with a rod withdrawal.

The operator may also be alerted by the volume control tank level and by the indication of the letdown diverter valve. There is an alarm from the makeup controller which will alert the operator to flow deviation from the set point value. A high-level alarm from the volume control tank would also indicate that excess makeup is added. Finally, a boronometer is located in the normal purification letdown flow path. This instrument provides continuous indication of the boron concentration and a hi-low alarm.

In view of the large number of available alarms and indications, it is considered that there is ample time and information available to the operator for identification of the incident and for stopping the makeup water injection and to initiate boration.

If the operator takes no corrective action, the power, temperature and pressure would rise. However, this transient would be terminated either by the thermal margin/low pressure trip or by the overpower trip. Following a reactor trip, assuming a reactivity addition rate of 10^{-5} $\Delta\rho$/sec, which is higher than that expected for a boron dilution event, and minimum shutdown worth of -2% $\Delta\rho$, the operator would have approximately 33 minutes to terminate the dilution prior to losing shutdown margin.

15.4.6.4 Failure to Add Boron To Compensate for Reactivity Changes After Shutdown

Administrative procedures require that boron levels be set and checked by sampling before cooldown is initiated. The unlikely event of a failure to add boron before cooldown to compensate for reactivity increases due to cooldown or xenon concentration reduction, would result in a loss of shutdown margin and a return to criticality. A cooldown rate 75°F per hour was used. Assuming the end of cycle moderator temperature coefficient of reactivity at hot standby with all rods in, the maximum rate of reactivity addition during cooldown from hot standby would be 2.13×10^{-2} $\Delta\rho$/hour. The maximum rate of xenon concentration reduction occurs 10 hours after shutdown from full power operation and is approximately equivalent to the reactivity change of

0.2×10^{-2} $\Delta\rho$/hour. The reactivity addition rate due to the reduction of xenon concentration would not normally coincide with cooldown; however, with the combined effect of temperature reduction and xenon reduction at the maximum rate, it would require more than 50 minutes for the reactor to go critical, assuming a minimum 2% shutdown margin. Therefore, ample time would be available for the reactor operator to recognize the situation and initiate boration.

15.4.6.5 Results of Analysis

The results of the analysis for this event are summarized in Table 15.4.6-1. The results show that there is adequate time for the operator to manually terminate the source of dilution flow. The reactor will be in a stable condition. The operator can then initiate reboration to recover the shutdown margin. Boron dilution during power operation is bounded by the analyses presented in Sections 15.4.1 and 15.4.2. However, the results presented here demonstrate that there is adequate time for the operator to manually terminate the source of dilution flow following reactor trip.

15.4.6.6 Conclusion

Because of the administrative procedures involved and equipment safeguards provided for the boron dilution operation, the probability of an erroneous dilution is considered very small. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. In the event that the charging flow is greater than 40 gpm with the RHR system in operation and the main reactor coolant pumps are not running, then increased monitoring of the boron concentration will be taken to protect against the unlikely event of a non-uniform (i.e., wave front/slug flow) boron dilution occurrence. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the dilution and take corrective action before shutdown margin is lost or thermal margin to DNB is lost.

Table 15.4.6-1 Summary of Results for the Boron Dilution Event

Reactor Conditions	Dilution By	Time to Criticality
Refueling	Primary Water	130 minutes (Charging at 40 gpm)
Refueling and Startup with Primary Coolant System Filled	Primary Water	90 minutes (Charging at 133 gpm, main reactor coolant pumps running)
Refueling and Startup with Primary Coolant System Filled	Primary Water	>15 minutes (Charging at 133 gpm, RHR flow ≥ 1500 gpm)
Hot Standby or Critical	Primary Water	Considered in the uncontrolled rod/ rod bank withdrawal analysis
Following a trip from the Power Operation Condition		33 minutes
Failure to add boron to compensate for Reactivity changes after Shutdown		50 minutes

15.4.8 CONTROL ROD EJECTION

15.4.8.1 Identification of Causes and Event Description

The control rod ejection transient is defined as the mechanical failure of a control rod mechanical pressure housing such that the coolant system pressure ejects a control rod blade assembly and drive shaft to a fully withdrawn position. The consequences of this mechanical failure are a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The rod ejection accident is the most rapid reactivity insertion that can be reasonably postulated. The resultant core thermal power excursion is limited primarily by the Doppler reactivity effect of the increased fuel temperatures and is terminated by reactor trip of all remaining control rods, activated by neutron flux signals.

The rod ejection accident is classified as a limiting fault event (Table 15.0.11). The variable overpower trip affords protection against violation of the acceptance criteria for this event as described in Section 15.0.1.1. The criterion concerning the deposited enthalpy is addressed on a cycle specific basis in accordance with the approved ANF methodology.[26] The deposited enthalpy analysis is thus not addressed in this report. The evaluation presented herein pertains to the radiological consequence criterion, in addition to the overpressurization potential. For this analysis, the systems challenged are redundant; no single active failure in the RPS or ESF will adversely affect the consequences of the event.

15.4.8.2 Analysis Method

The analysis is performed using the PTSPWR2[10] and XCOBRA-IIIC[11] computer codes. The PTSPWR2 code models the major system components and calculates reactor power, fuel thermal response, surface heat transport, and fluid conditions, including coolant flow rate, temperature, and primary pressure. The core boundary conditions at the time of MDNBR are input into XCOBRA-IIIC to determine MDNBR.

15.4.8.3 Definition of Events Analyzed and Bounding Input

The control rod ejection accident is a reactivity insertion event which quickly inserts positive reactivity into the reactor core when the control rod mechanism fails. This reactivity insertion causes the core power level and fuel rod surface heat flux to increase, along with inducing an asymmetric radial power distribution across the core. In addition, the primary system heats up, resulting in the expansion of the primary side coolant which increases the pressurizer pressure due to the compression of the steam volume.

The event is further exacerbated by assuming the coincident loss of offsite power resulting in primary system coolant pump coastdown and a reduction of forced primary coolant. The reduction of the primary system flow correspondingly reduces the primary to secondary side heat transfer. This decrease in heat transfer capacity to the secondary side causes the primary side coolant to further heat up.

The hot full power (HFP) control rod ejection event was determined to deposit more energy into the primary system than the event initiated from hot zero power (HZP). Therefore, in terms of the event acceptance criteria, the HFP event poses a greater challenge than the HZP event. For this analysis, the event was assumed to initiate from HFP at 102% of rated full power.

To assess the acceptability of the outcome of a HFP rod ejection event, two cases were examined. The first case determines the maximum pressurization potential of the primary system during this event. The second case evaluates the radiological consequences of fuel failure due to DNB, which results from a core power excursion and a redistribution of core radial power. It was conservatively assumed that all fuel rods penetrating DNB failed. This assumption yields a conservative calculation of the number of fuel failures and offsite radiological doses.

For both the maximum pressurization and minimum DNB case, beginning-of-cycle (BOC) and end-of-cycle (EOC) kinetics were considered in order to establish the respective limiting cases. Conservative biases were applied to the reactivity coefficients and ejected rod worth. Other PTSPWR2 input was conservatively biased to bound the consequences of this event, i.e., fuel failures due to DNB and primary system pressurization.

Significant initial condition input to PTSPWR2, along with the applied bias, for the respective limiting cases is given below:

	Minimum DNB Case	Maximum Pressurization Case
Control	Manual	Manual
Core Power	Nom. + 2%	Nom. + 2%
Core Inlet Temperature	Nom. + 5°F	Nom. + 5°F
Primary Flow Rate	Nom. - 3%	Nom. - 3%
Primary Pressure	Nom. - 50 psia	Nom. + 50 psi
Kinetic Parameters		
Moderator Temperature Coefficient	EOC - 20%	BOC + 20%
Doppler Temperature Coefficient	EOC - 20%	BOC - 20%
Beta	EOC	BOC
Effective Neutron Lifetime	EOC	BOC
Ejected Rod Worth	EOC + 10%	BOC + 10%
Pellet-to-clad Heat Transfer Coeff.	Max. + 25%	Nom.
Pressurizer Heaters	Disable	Available
Pressurizer Spray	Available	Disable
Pressurizer Safety Valve Setpoint	Nom. - 3%	Max.
Secondary Relief Valve Setpoint	Nom. + 3%	Nom. + 3%

Table 15.4.8-1 provides the initial plant conditions assumed for this analysis.

15.4.8.4 Analysis of Results

The maximum pressurization case initiates with a failure of the control rod housing, causing an ejection of the affected control blade in 0.1 seconds. The ejection of a control blade results in positive reactivity being added to the core. The reactivity insertion due to the ejected control blade culminates in a maximum core neutronic power of 3727.1 MWt at 0.98 seconds. A variable overpower scram signal is generated at 0.43 seconds and the shutdown control rods begin to insert negative reactivity at 0.83 seconds.

The added reactivity from the ejected control blade, along with a positive moderator temperature coefficient at BOC, causes the core average temperature to increase to a maximum value of 583.35°F at 2.26 seconds. An increase in the coolant temperature on the primary side causes a volumetric expansion, resulting in an insurge into the pressurizer. The peak pressurizer pressure attains a value of 2452.12 psia occurring at 5.06 seconds. The pressurizer safety valves are not predicted to open since the nominal setpoint is 2500 psia. Because the pressurizer pressure does not exceed the safety valve setpoint, the primary system will not overpressurize if a control rod ejection occurs.

The event is terminated by the insertion of the remaining control blades leading to a decrease in core power, average temperature, and pressure.

The plant transient response for the maximum pressurization case is shown in Figures 15.4.8-1 through 15.4.8-3. A sequence of events pertaining to this case is given in Table 15.4.8-2.

The minimum DNB case is initiated in the same manner as the maximum pressurization case. The reactor scrams on a variable overpower signal at 0.42 seconds with the insertion of the shutdown control rods occurring at 0.82

seconds. The peak neutronic power level achieved is calculated to be 5348.1 MWt at 0.13 seconds.

The increase in core neutronic power causes a corresponding increase in the fuel rod surface heat flux. The maximum heat flux is predicted to occur at 0.90 seconds with a value of 189877 Btu/hr-ft^2.

As with the maximum pressurization case, the primary side temperatures increase due to the added reactivity of the ejected rod. The peak core average temperature is 582.76°F at 1.92 seconds. The volumetric expansion of the primary coolant causes an increase in the pressurizer pressure to a value of 2162.22 psia at 3.38 seconds. Minimum DNBR less than 1.17 is calculated to occur at 1.09 seconds.

With the core boundary conditions predicted at the time of MDNBR, along with an asymmetric core power distribution, the amount of fuel failure is calculated. It is determined that 12.2% of the fuel rods in the core will fail due to the penetration of DNB. The offsite radiological doses from this event are calculated to be 146.4 rem to the thyroid and 2.04 rem to the whole body. These doses are below the 10 CFR 100 dose limits and the whole body is less than 25% of the respective 10 CFR 100 limit. The thyroid dose calculation is based on a thyroid dose of 120 rem corresponding to the 10% fuel failure, as reported in Reference 5.

As with the maximum pressurization case, the event is terminated by the insertion of the remaining control blades leading to a decrease in core power, average temperature, and pressure.

The transient response of the plant for the DNBR case is given in Figures 15.4.8-4 through 15.4.8-6. The sequence of events for this case is given in Table 15.4.8-2.

15.4.8.5 Conclusion

The maximum pressurizer pressure does not exceed the pressurizer safety valve setpoint. Therefore, the system pressure is less than 110% of the design value of 2750 psia. The radiological doses are conservatively calculated to be less than the 10 CFR 100 dose limits. Applicable acceptance criteria are considered, therefore, to be met.

Table 15.4.8-1 Summary of Initial Operating Conditions

	Minimum DNB Case	Maximum Pressurization Case
Power (MWt)	2580.6	2580.6
Core Inlet Temperature (°F)	548.65	548.65
Pressurizer Pressure (psia)	2010	2110
RCS Flow Rate (Mlbm/hr)	116.7	116.7
Steam Dome Pressure (psia)	731.5	731.5

Table 15.4.8-2 Event Summary for a Control Rod Ejection

Maximum Pressurization Case Event Summary

Event	Time (sec)
Control Rod Ejects	0.00
Reactor Scram (rods begin insertion)	0.83
Peak Power	0.98
Peak Core Avg. Temperature	2.26
Peak Pressure	5.06
Steam line Safety Valves Open	11.65
Peak Steam Dome Pressure	11.65

Minimum DNBR Case Event Summary

Event	Time (sec)
Control Rod Ejects	0.00
Peak Power	0.13
Reactor Scram (rods begin insertion)	0.82
Peak Core Avg. Heat Flux	0.90
Minimum DNBR	1.09
Peak Core Avg. Temperature	1.92

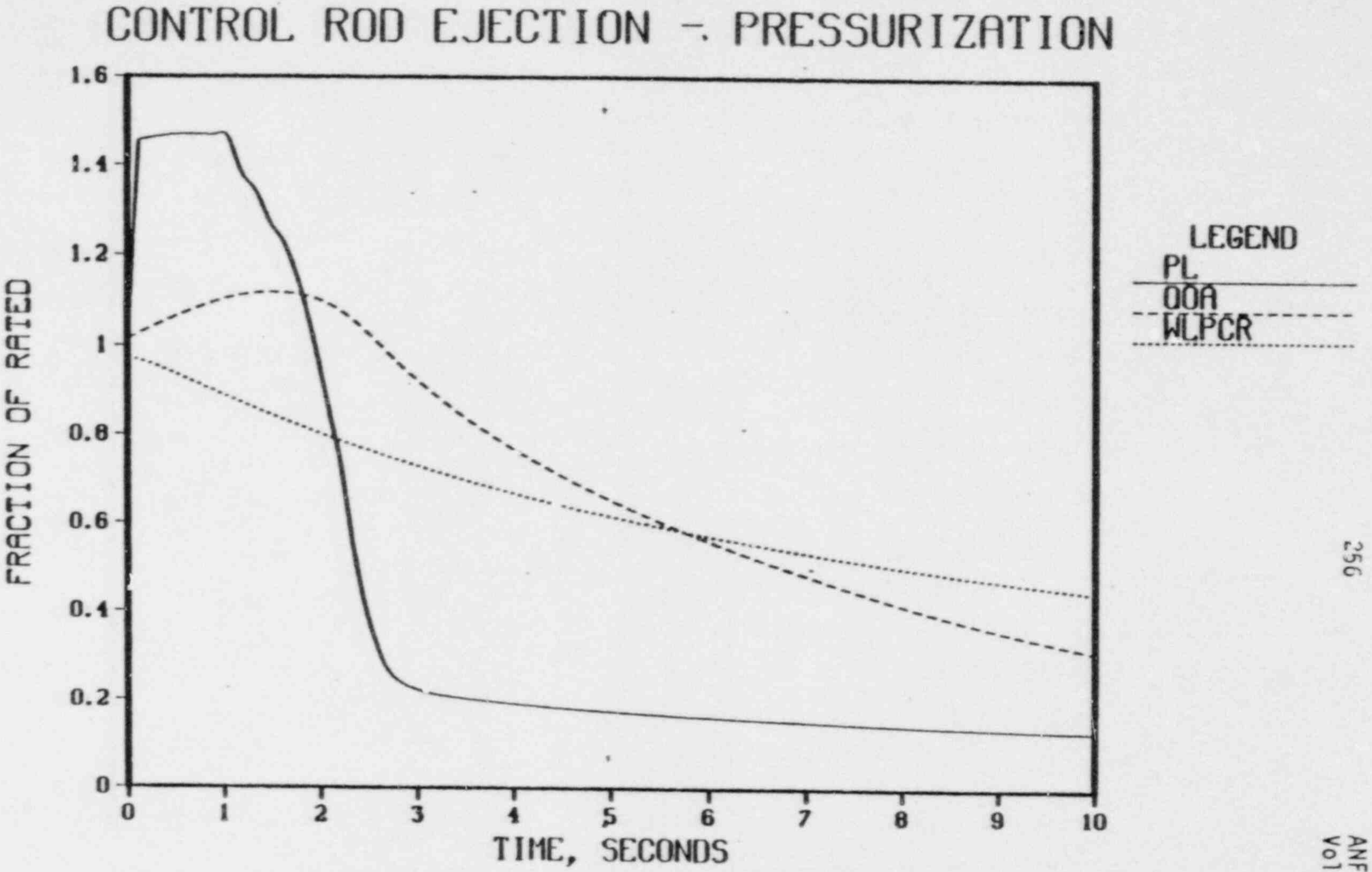


Figure 15.4.8-1 Power, Heat Flux and Flow for a Control Rod Ejection-Maximum Pressurization

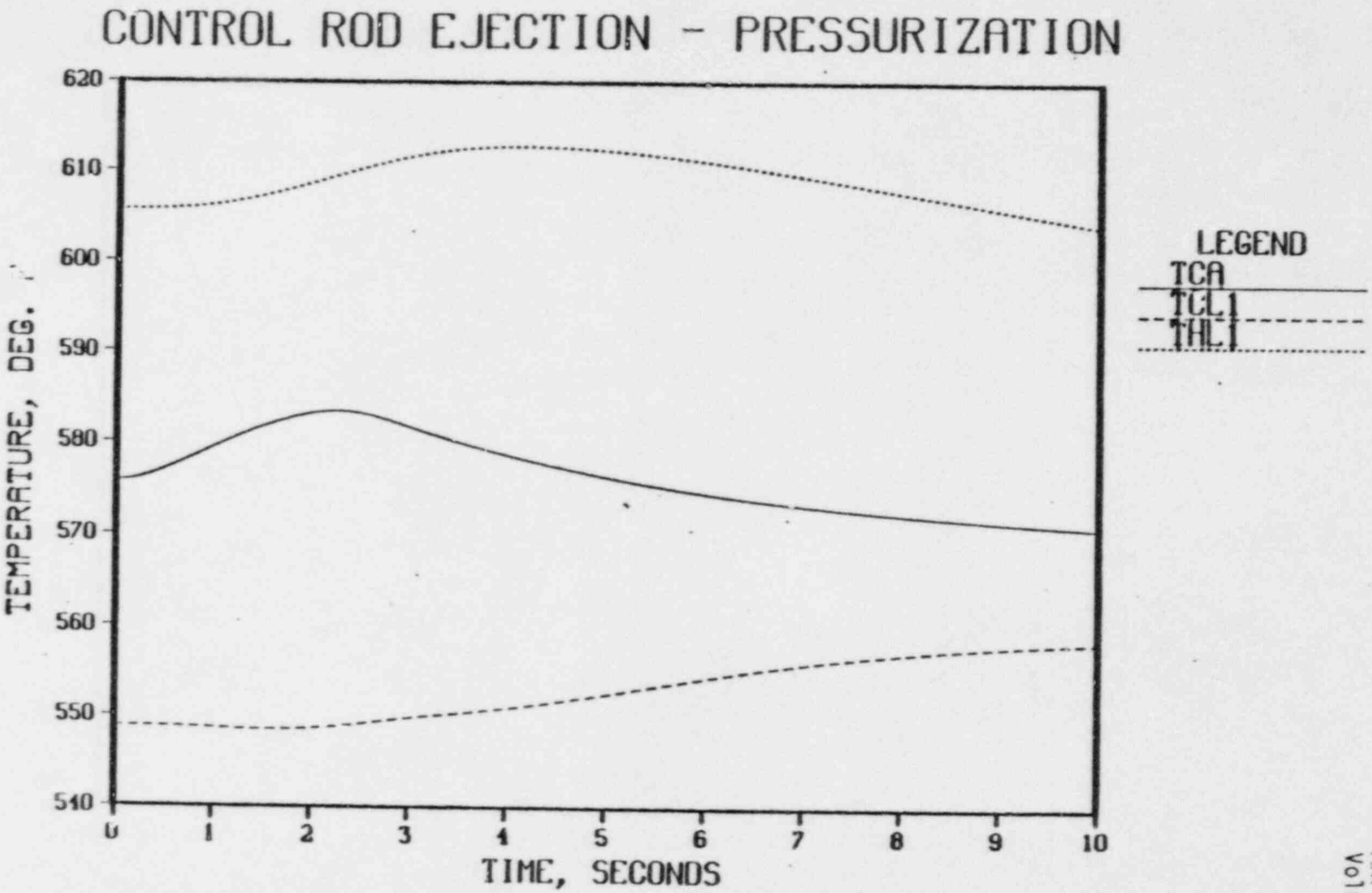


Figure 15.4.8-2 Primary Side Coolant Temperatures for a Control Rod Ejection-Maximum Pressurization

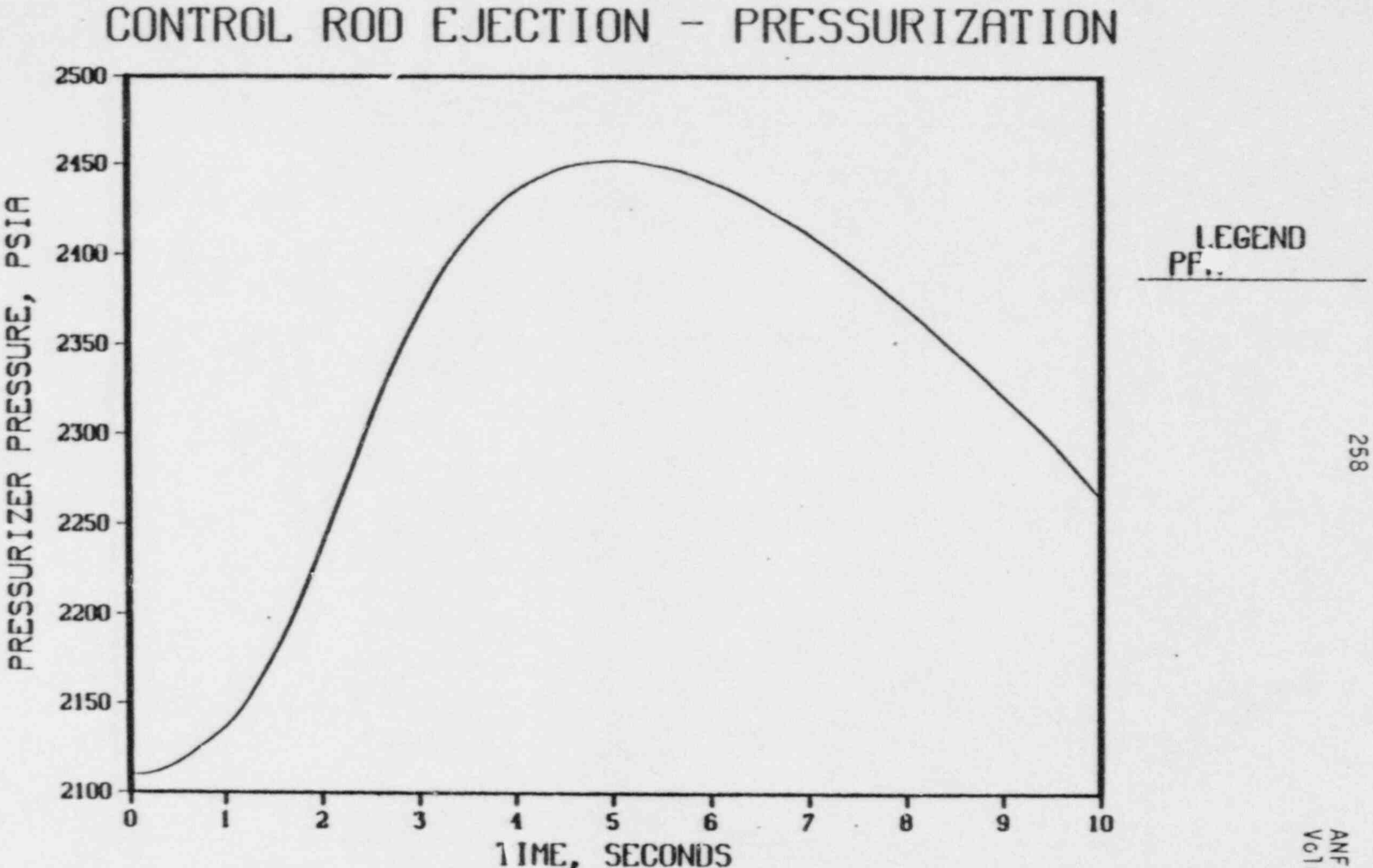


Figure 15.4.8-3 Pressurizer Pressure for a Control Rod Ejection-Maximum Pressurization

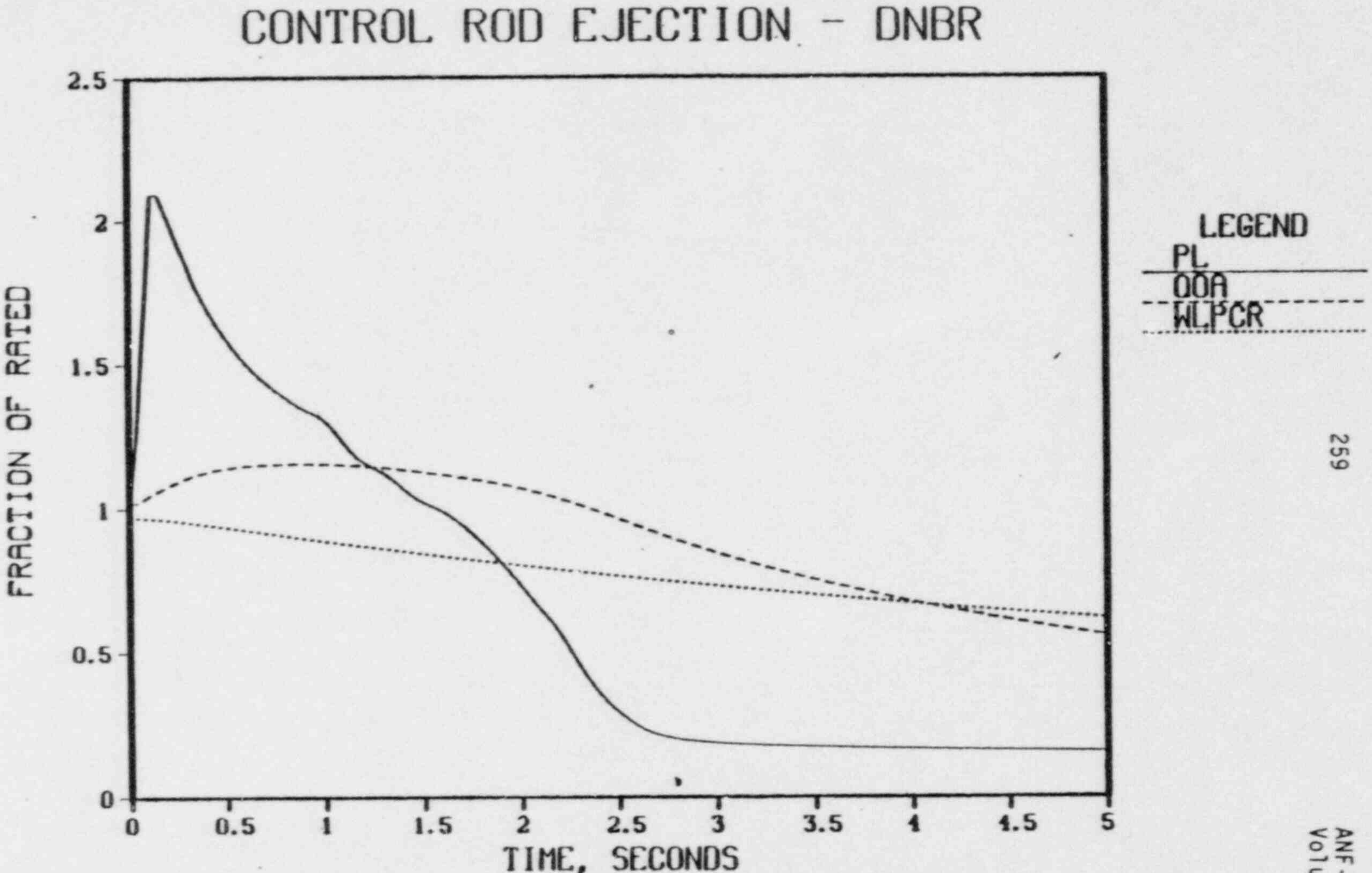


Figure 15.4.8-4 Power, Heat Flux and Flow for a Control Rod Ejection-Minimum DNBR

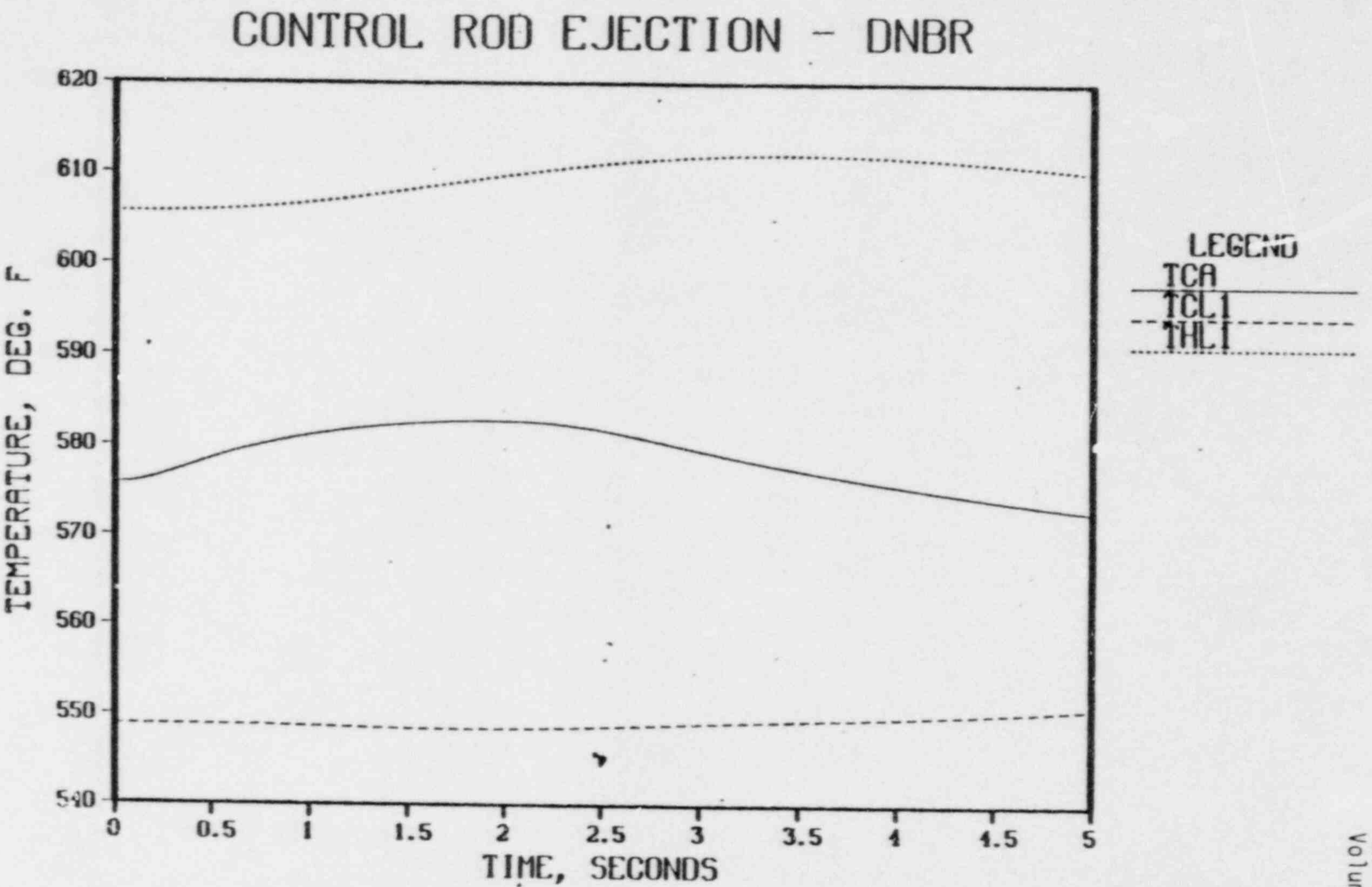


Figure 15.4.8-5 Primary Side Coolant Temperatures for a Control Rod Ejection-Minimum DNBR

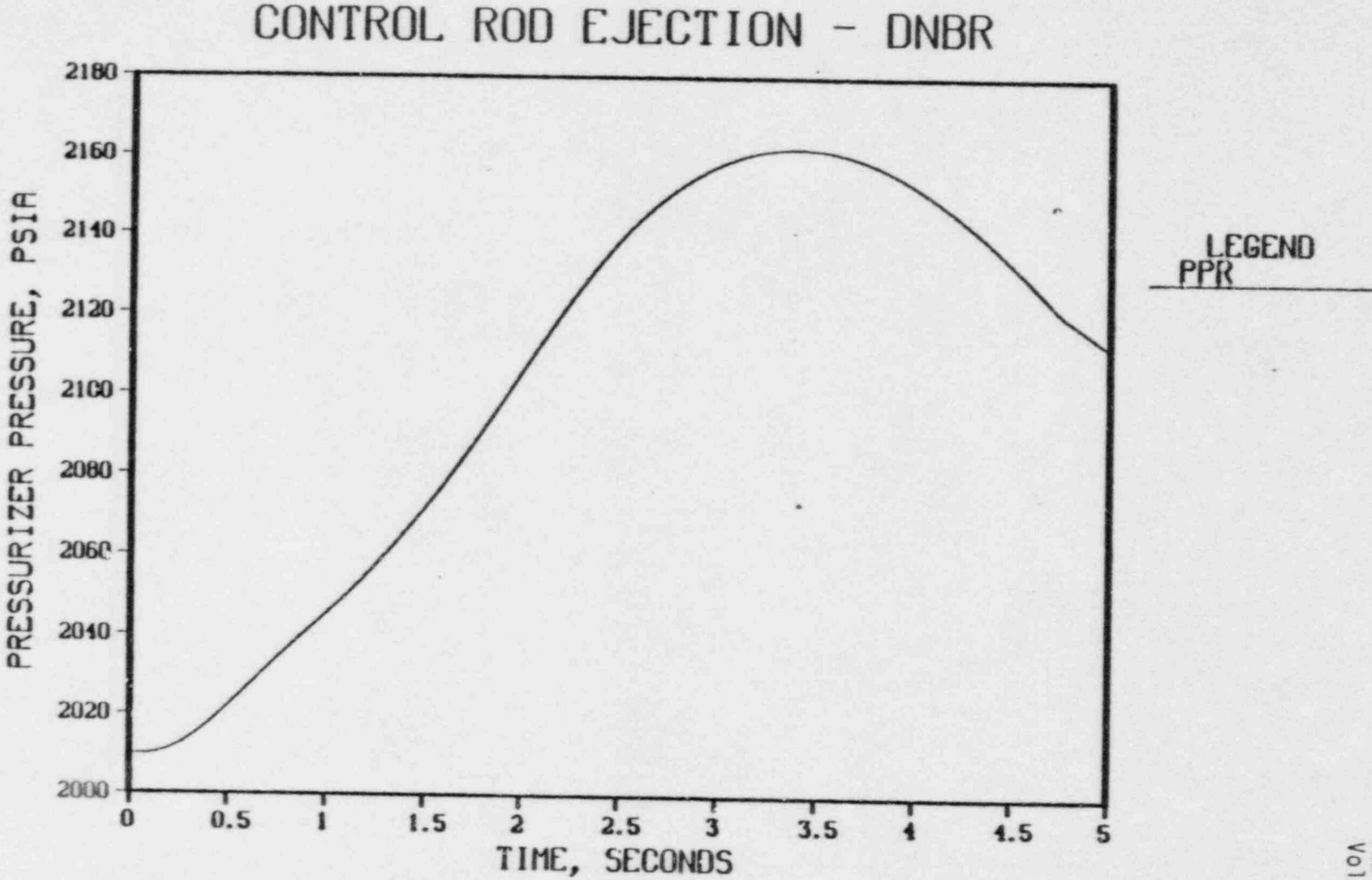


Figure 15.4.8-6 Pressurizer Pressure for a Control Rod Ejection-Minimum DNBR

4.0 REFERENCES

1. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, LWR Edition, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, July 1981.

2. "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," XN-NF-84-73(P), Rev. 1, Exxon Nuclear Company, August 1984.

3. "Palisades Modified Reactor Protection System Report - Disposition of Standard Review Plan Chapter 15 Events," ANF-87-150(P), Vol. 1, Advanced Nuclear Fuels Company, November 1987.

4. "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," XN-NF-82-21(A), Rev. 1, Exxon Nuclear Company, September 1983.

5. Palisades Final Safety Analysis Report, Updated Version (as of July 1985), Consumers Power Company.

6. "Palisades Primary Design Parameters for PTS Analysis, XN-NF-85-101(P), Exxon Nuclear Company, October 1985.

7. Palisades Plant Technical Specifications, Consumers Power Company, Appendix A to License No. DPR-20, Revision dated September 5, 1985.

8. "Exxon Nuclear DNB Correlation for PWR Fuel Design," XN-NF-621(A), Rev. 1, Exxon Nuclear Company, April 1982.

9. "Computational Procedure for Evaluating Fuel Rod Bowing," XN-NF-75-32(A), Supps. 1, 2, 3 & 4, Exxon Nuclear Company, October 1983.

10. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR), XN-NF-74-5(A), Rev. 2, Exxon Nuclear Company, October 1986, and Supplements 3-6.

11. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation", XN-NF-75-21(A), Revision 2, Exxon Nuclear Company, January 1986.

12. "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events - Sensitivity Analysis of Input Parameters," XN-NF-84-73(P), Rev. 2, Appendix A, Supp. 1, Exxon Nuclear Company, December 1984.

13. "SLOTRAX-ML: A Computer Code for Analysis of Slow Transients in PWRs," XN-NF-85-24(A), Exxon Nuclear Company, September 1986.

14. "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt", XN-NF-77-18, Exxon Nuclear Company, July 1977.

15. "Systematic Evaluation Program Design Basis Transient Reanalysis for the Palisades Reactor", XN-NF-81-25, Revision 1, Exxon Nuclear Company, May 1981.

16. "LOCA Analysis for Palisades at 2530 MWt Using the ENC WREM-II PWR ECCS Evaluation Model", XN-NF-77-24, Exxon Nuclear Company, July 1977.

17. "Analysis of Axial Power Distribution Limits for the Palisades Nuclear Reactor at 2530 MWt", XN-NF-78-16, Exxon Nuclear Company, June 1978.

18. "Analysis of Axial Power Distribution Limits for the Palisades Nuclear Reactor at 2530 MWt: Sensitivity Studies", XN-NF-78-16, Supp. 1, Exxon Nuclear Company, April 1984.

19. "ECCS and Thermal-Hydraulic Analysis for the Palisades Reload H Design", XN-NF-80-18, Exxon Nuclear Company, April 1980.

20. Letter, D.J. VandeWalle (CPCo) to Director, Nuclear Reactor Regulation, dated June 20, 1985, Docket 50-255, License DPR-20, Palisades Plant, "NUREG-0737 Item II.K.3.30 Small Break LOCA Models and Item II.K.3.31 Plant Specific Analysis".

21. Letter, John A. Zwolinski (NRC) to Mr. VandeWalle (CPCo), dated July 3, 1985, Docket 50-255, License DPR-20, Palisades Plant, "TMI Action Plan Item II.K.3.30, Small Break LOCA Analysis and II.K.3.31, Plant Specific Analysis".

22. "Determination of Palisades Thermal Margin/Low Pressure Trip Coefficients", Combustion Engineering, Inc., September 1971.

23. "CE Setpoint Methodology", CENPD-199-NP, Rev. 1-NP, Combustion Engineering, Inc., March 1982.

24. "Palisades Cycle 5 Reload Fuel Safety Analysis Report," XN-NF-81-34(P), Supp. 1, Exxon Nuclear Company, December 1981.

25. "XTG: A Two Group Three-Dimensional Reactor Simulation Utilizing Coarse Mesh Spacing (PWR Version)", XN-CC-28(A), Revision 3, Exxon Nuclear Company, January 1975.

26. "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," XN-NF-78-44, Exxon Nuclear Company, February 1978.

APPENDIX A

DERIVATION OF BORON DILUTION "WAVE FRONT"

EQUATIONS

TABLE OF CONTENTS

APPENDIX - A

A.1 Introduction

The purpose of this appendix is to present the assumptions, derivation of equations, and summary of the "wave front/slug" flow approach to the boron dilution event. This approach is supplementary to the normal uniform mixing mathematical approach.

A.2 Assumptions

Listed below are the general assumptions utilized in developing the "wave front/slug" flow approach to the boron dilution event.

1) None of the main reactor coolant pumps are in use.
2) Letdown flow is not occurring and the letdown line is not located in the same cold leg as the charging flow inlet.
3) The fluid leaving the reactor core becomes uniformly mixed in the Upper Plenum and Low Pressure Safety Injection System and Piping (Residual Heat Removal Mode) prior to re-injection into the cold legs.
4) Re-injection by the Low Pressure Safety Injection Pumps occurs in all four cold legs.
5) The volume of the Low Pressure Safety Injection System and Piping is not considered in computing the coolant loop/cycle time.
6) Both symmetric (1 cold leg - opposite loops) and asymmetric (1 cold leg only) charging flow situations are considered.
7) The symmetric dilution event considers full core dilution and the asymmetric dilution considers half core dilution with full core boron criticality requirements.
8) Low Pressure Safety Injection and Charging flow is assumed to be constant during the dilution event.
9) Slight density variations between charging flow and residual heat removal flow are neglected.
10) Previous analyses have shown that minimum RHR flow produced the most

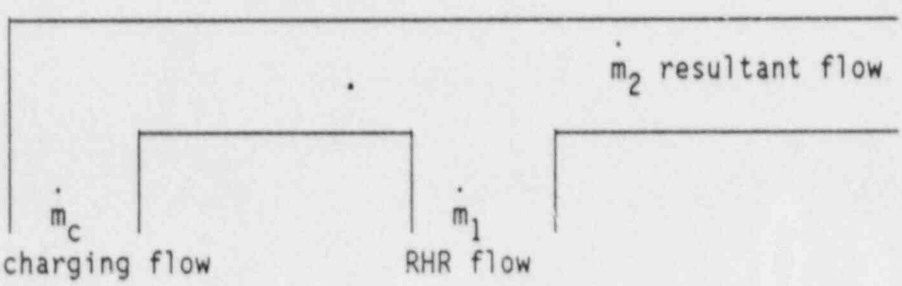
conservative results. As such, various RHR flow rates are considered.

11) Any flow in the steam generator tubes is neglected. For the cases considered herein, the steam generator U-tube are empty since the reactor coolant volume is assumed drained to the center line of the hot legs.

12) The charging flow boron concentration is assumed to be zero.

A.3 Mathematical Model - Derivation of Equations

A.3.1 Symmetric Charging Flow



Overall Mass Balance

$$\dot{m}_1 + \dot{m}_c = \dot{m}_2$$

Boron Balance

$$\dot{m}_1 C_i + \dot{m}_c C_c = \dot{m}_2 C_{(i+1)}$$

Assuming that $C_c = 0$,

$$\dot{m}_1 C_i = (\dot{m}_1 + \dot{m}_c)\, C_{(i+1)}$$

Solving for $C_{(i+1)}$ yields:

$$C_{(i+1)} = \frac{\dot{m}_1 C_i}{\dot{m}_1 + \dot{m}_c}$$

<u>1st Transit</u>

$$C_1 = \frac{\dot{m}_1 C_o}{\dot{m}_1 + \dot{m}_c}$$

<u>2nd Transit</u>

$$C_2 = \frac{\dot{m}_1 C_1}{\dot{m}_1 + \dot{m}_c}$$

$$C_2 = \frac{\dot{m}_1 \dfrac{\dot{m}_1 C_o}{\dot{m}_1 + \dot{m}_c}}{\dot{m}_1 + \dot{m}_c}$$

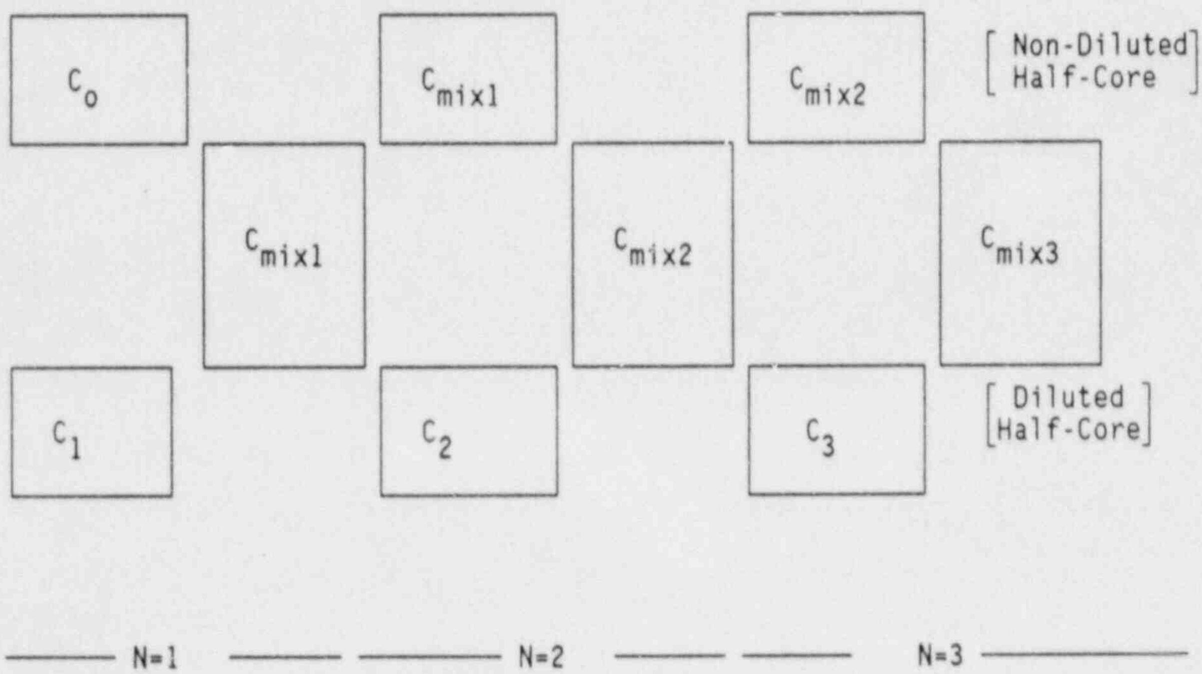
Rearranging,

$$C_2 = \frac{\dot{m}_1 \dot{m}_1}{(\dot{m}_1 + \dot{m}_c)(\dot{m}_1 + \dot{m}_c)} C_o$$

By induction,

$$C_N = \left[\frac{\dot{m}_1}{\dot{m}_1 + \dot{m}_c} \right]^N C_o \qquad N = 0, 1, 2 \ldots m \quad \text{Loop Transient}$$

A.3.2 Asymmetric Charging Flow



Schematic of calculation process to achieve boron concentration in diluted half core sector as a function of N; where N is the number of boron dilution wave fronts that enter core.

N = 1

dilute

$$C_1 = \frac{\dot{m}_1}{\dot{m}_1 + \dot{m}_c} C_o$$

$$C_1 = \frac{.5 * \dot{m}_{RHR}}{.5 * \dot{m}_{RHR} + \dot{m}_c} C_o$$ where $\dot{m}_1 = .5 * \dot{m}_{RHR}$ for diluted half core asymmetric calculation

Define Mixing factor #1 = XM1 = $\dfrac{.5 * \dot{m}_{RHR}}{.5 * \dot{m}_{RHR} + \dot{m}_c}$

$$\boxed{C_1 = XM1\ C_o}$$

Assume Mixing in the Upper Plenum and RHR System Piping
(Equivalent to symmetric calculation for N=1)

$$C_{mix1} = \left[\frac{\dot{m}_1}{\dot{m}_1 + \dot{m}_c}\right] C_o$$

$$C_{mix1} = \left[\frac{\dot{m}_{RHR}}{\dot{m}_{RHR} + \dot{m}_c}\right] C_o$$

Define Mixing Factor #2 = XM2 = $\dfrac{\dot{m}_{RHR}}{\dot{m}_{RHR} + \dot{m}_c}$

$$C_{mix1} = XM2\ C_o$$

N = 2

dilute

$$C_2 = XM1\ C_{mix1} \qquad \text{with } C_o = C_{mix1}$$

$$\boxed{C_2 = XM1\ XM2\ C_o}$$

Assume mixing in the Upper Plenum and RHR System Piping
(Equivalent to symmetric calculation for N = 2)

$$C_{mix2} = \left(\frac{\dot{m}_1}{\dot{m}_1 + \dot{m}_c}\right)^2 C_o$$

$$C_{mix2} = \left(\frac{\dot{m}_{RHR}}{\dot{m}_{RHR} + \dot{m}_c}\right)^2 C_o$$

$$C_{mix2} = XM2^2\ C_o$$

N = 3

dilute

$$C_3 = XM1\ C_{mix2} \qquad \text{with } C_o = C_{mix1}$$

$$\boxed{C_3 = XM1\ XM2^2\ C_o}$$

N = N In general C_N can be expressed, by induction, as

$$\boxed{C_N = XM1\ XM2^{(N-1)} C_o}$$

Concentration of dilute half core sector in terms of N the number of dilution waves entering core.

A.4 Summary

In general, the wave front/slug flow approach to the boron dilution event is only applicable if the main coolant pumps are not in operation. Uniform mixing is more probable than wave front/slug flow if the main coolant pumps are in operation. Application of the wave front/slug flow dilution equations indicates that increased operator response time is available if: (1) Low Pressure Safety Injection (Residual Heat Removal-- RHR) flow is maintained near the nominal value of 3000 gpm; and, (2) the possibility of charging flow being more than 40 gpm is eliminated or minimized by administrative procedures.

ANF-87-150(NP)
Volume ?

Issue Date: 6/13/88

PALISADES MODIFIED REACTOR PROTECTION SYSTEM REPORT:

ANALYSIS OF CHAPTER 15 EVENTS

Distribution

RA Copeland

RC Gottula

JS Holm

JW Hulsman

JD Kahn

TR Lindquist

LA Nielsen

LD O'Dell

FB Skogen

GN Ward

HE Williamson

CPCo/HG Shaw (20)

Document Control (5)