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ADVANCED NUCLEAR FUELS CORPORATION

H.B. ROBINSON UNIT 2
CHAPTER 15
OVERTEMPERATURE ΔT TRIP EVENT ANALYSIS FOR ELIMINATION OF RTD BYPASS PIPING

JULY 1988

ADVANCED NUCLEAR FUELS CORPORATION

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OVERTEMPERATURE ΔT TRIP EVENT ANALYSIS
FOR ELIMINATION OF RTD BYPASS PIPING

By

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June 1988

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1.0 INTRODUCTION

Presented in this report are the results of Standard Review Plan (SRP)[1] Chapter 15 event analyses performed for H.B. Robinson Unit 2 to support a revised RTD installation design which eliminates the bypass piping. The analyses include a simulation of the processing of temperature signals input to the Overtemperature ΔT trip. The associated lag constant and delay times are shown in Table 2.1. The analyses are structured to support a Technical Specification $F_{\Delta H}$ limit of 1.65.

Basic assumptions used in the analyses include (a) precluding the withdrawal function of automatic rod control, (b) a non-positive moderator temperature coefficient above 50% power for the rod drop transient, and (c) a K1 value in the OTΔT trip function of 1.24 including uncertainties.

The results of the analyses are summarized in Section 2.0 of this report. The detailed event case descriptions and results are provided in Section 3.0. References for this report are listed in Section 4.0.

2.0 SUMMARY

The DNBR limiting cases of the loss of external load, uncontrolled control rod withdrawal, and control rod drop events were simulated to determine the validity of the Overtemperature ΔT reactor trip function for the new RTD installation which eliminates the bypass piping. These events were identified in the analysis of record[2] as the most limiting DNBR events which also trip on the Overtemp ture ΔT reactor protection failure. The calculations employed stan d ANF thermal hydraulic and Chapter 15 event calculation methodology[3 MDNBRs were calculated with the XNB critical heat flux correlation[5], which is applicable to ANF fuel in the H.B. Robinson Unit 2 reactor[6]. Operating conditions and event acceptance criteria are as given in Reference 2 unless otherwise noted.

Event results are summarized in Table 2.2. Calculated MDNBRs for the events analyzed are above the XNB critical heat flux correlation safety limit of 1.17. Based on these results, it is concluded that applicable acceptance criteria are met with the current Technical Specification limit on $F_{\Delta H}$ of 1.65 and a Technical Specification Overtemperature ΔT trip function with a K1 value of 1.24 including uncertainties.

TABLE 2.1 OVERTEMPERATURE ΔT REACTOR TRIP
DELAY TIMES AND LAG CONSTANTS

Component	
Thermal lag representing both thermal transport through the thermowell and the RTD response time	4.0 sec
Electronics delay representing electronic signal processing, trip breaker operation, and control rod drive shaft gripper release	0.75 sec

TABLE 2.2 SUMMARY OF EVENT ANALYSIS RESULTS

Transient Event	MDNBR (XNB)
15.2.1 Loss of External Load	1.19
15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power	1.19
15.4.3 Full Length RCCA Drop (Manual)	1.23

3.0 ANALYSIS OF PLANT TRANSIENTS

The FSAR Chapter 15 events were reviewed to determine the limiting DNBR events which also trip on the Overtemperature ΔT reactor protection feature. The events identified were:

(1) Loss of Load - MDNBR Case;
(2) Uncontrolled Control Rod Bank Withdrawal From Full Power; and
(3) Dropped Full Length RCCA.

These events were analyzed using the PTSPWR2 plant transient simulation code[7]. The version of PTSPWR2 used incorporates a model which simulates the time lags and delays resulting from the RTD installation in a thermowell as well as electronic signal processing, trip breaker operation, and control rod drive shaft gripper release. This section provides the results of the analysis of the three limiting events. Subsections in the report are enumerated in accordance with the SRP in order to facilitate review. Single failures were considered for each of the three limiting events.

4.0 REFERENCES

(1) "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

(2) XN-NF-84-74, Rev. 1, "Plant Transient Analysis for H.B. Robinson Unit 2 at 2300 MWt With Increased $F^N_{\Delta H}$," Exxon Nuclear Company, Richland, WA, April 1986.

(3) XN-NF-82-21(A), "Application Of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Richland, WA, September 1983.

(4) XN-NF-84-73(P), "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Exxon Nuclear Company, Richland, WA, December 1984.

(5) XN-NF-621(A), Rev. 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, Richland, WA, September 1983.

(6) XN-NF-711(P), "Extension of XNB Correlation to PWR Fuel Assembly Designs With Spacer Pitch Greater Than 22 Inches," Exxon Nuclear Company, Richland, WA, May 1983.

(7) XN-NF-74-5(A) & Supps. 1-6, Rev. 2, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)," Exxon Nuclear Company, Richland, WA, October 1986.

(8) File Number NF-2484/NF-1077, Rev. 3, "Single Failure Analysis, H.B. Robinson Unit 2," Carolina Power and Light Company, April 1985.

(9) XN-CC-28, Rev. 5, "XTG: A Two-Group Three-Dimensional Reactor Simulator Utilizing Coarse-Mesh Spacing (PWR Version)," Exxon Nuclear Company, Richland, WA, July 1979.

15.2.1 LOSS OF EXTERNAL LOAD

The analysis addresses the DNBR branch of the Loss of External Load event. That event was shown to result in an Overtemperature ΔT reactor trip in the analysis of record.(2) The event is analyzed to verify the OTΔT trip function with the new RTD installation which eliminates the bypass piping.

15.2.1.1 Identification of Causes and Event Description

A loss of external load can result from loss of the generator 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 valves close, 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.

Two event sequences may be postulated, one leading to a challenge of the vessel pressurization criterion, the second leading to a challenge to the DNBR limit. Only the second event sequence resulted in an Overtemperature ΔT reactor trip in the analysis of record.(2) Only the event challenging the DNBR limit is analyzed here.

If the reactor is not tripped when the turbine is tripped, the primary system temperature continues to rise. The primary liquid will expand and the pressurizer steam space is compressed, causing the pressurizer pressure to rise. In the event sequence considered, the mitigative features of the pressurizer spray and pressurizer relief valves are assumed to function. This minimizes the pressurization of the primary system, resulting in a conservative evaluation of the MDNBR for this event. Energy is removed during the early phase of the transient through the steam generator safety valves when the steam generator pressure exceeds the safety valve opening setpoint.

The challenge to the specified acceptable fuel design limit (SAFDL) on DNBR is evaluated because of the increasing core inlet temperature and the potential for the reactor core power to increase prior to reactor trip. Reactor control is assumed to be in the manual mode so the reactor power will not be reduced when the primary system average temperature begins to increase.

This event is a moderate frequency (Condition II) event. The acceptance criterion for this event sequence is that the MDNBR during the transient must be above 1.17. The cited[8] single failure for this event does not affect the results of the analysis of the event since the Engineered Safety Features (ESF) are not challenged within the time period of interest.

15.2.1.2 Analysis Method

This event is analyzed with the PTSPWR2 computer program.[7] The core thermal-hydraulic boundary conditions from the PTSPWR2 calculation are used as input to the XCOBRA-IIIC methodology[3] to predict the minimum DNBR for the event.

15.2.1.3 Definition of Events Analyzed and Bounding Input

This event is analyzed to ensure that reactor protection systems are properly set to prevent penetration of the SAFDLs. The analysis takes no credit for the turbine bypass system or for direct reactor trip on turbine trip.

The input parameters are biased to maximize the increase in reactor power during the transient consistent with minimizing event MDNBR. Also, the parameters and the equipment operational states are selected to reduce the primary system pressurization to provide a conservative estimation of the minimum DNBR during the transient.

The bounding operating mode for this event is full power initial conditions with the reactor control system in the manual mode.

15.2.1.4 <u>Analysis of Results</u>

The event initiates with closure of the turbine control valves. Steam line pressure increases until the secondary side safety valves open at 6.1 sec. The maximum pressure in the steam dome of the steam generators is not achieved until 19.9 sec. The pressurization of the secondary side results in decreased primary to secondary heat transfer and a substantial rise in cold leg temperature.

The rapid increase in primary side temperatures result in a large insurge into the pressurizer, compressing the steam space and pressurizing the primary system. The pressurizer PORVs open at 4.7 sec. and the pressurizer safety valves are opened at 8.8 sec. The valves have enough capacity to mitigate the pressure transient and limit the pressure to the safety valve setpoint value.

Pressurizer PORVs and spray limit the pressure rise, preventing a reactor scram on high pressure. Reactor scram occurs on Overtemperature ΔT, with rod insertion commencing at 15.5 sec. The reactor power reaches about 117.5%. The steam generator safety valve flow limits the core average temperature rise to 24.0°F at 16.7 sec. The DNBR challenge results from the core power and primary coolant temperature increase. The challenge is exacerbated by the action of the pressure control systems.

Plant initial operating conditions assumed in the analyses are summarized in Table 15.2.1-1.

The transient response to this event is shown in Figures 15.2.1-1 to 15.2.1-5. An event summary is shown in Table 15.2.1-2. The minimum DNBR was computed to be 1.19. This is above the DNBR limit of 1.17 for the XNB correlation.

15.2.1.5 Conclusion

The minimum DNBR is greater than the XNB DNB correlation safety limit. Therefore, the DNBR acceptance criterion is met.

TABLE 15.2.1-1 LOSS OF EXTERNAL LOAD - SUMMARY OF INITIAL OPERATING CONDITIONS

Parameter	Value
Power (MWt)	2346
Core Inlet Temperature (°F)	550.2
Pressurizer Pressure (psia)	2220
Reactor Coolant System Flow Rate (lbm/hr)	97.29×10^6
Steam Dome Pressure (psia)	828.3

TABLE 15.2.1-2 LOSS OF EXTERNAL LOAD EVENT SEQUENCE

Event	Time (sec)
Turbine Trip	0.0
Pressurizer PORVs Open	4.7
Steam Line Safety Valves Open	6.1
Pressurizer Safety Valves Open	8.8
Peak Pressure	15.0
Reactor Scram (Begin Rod Insertion)	15.5
Peak Power	15.5
Minimum DNBR	16.3
Peak Core Average Temperature	16.7
Peak Steam Dome Pressure	19.9

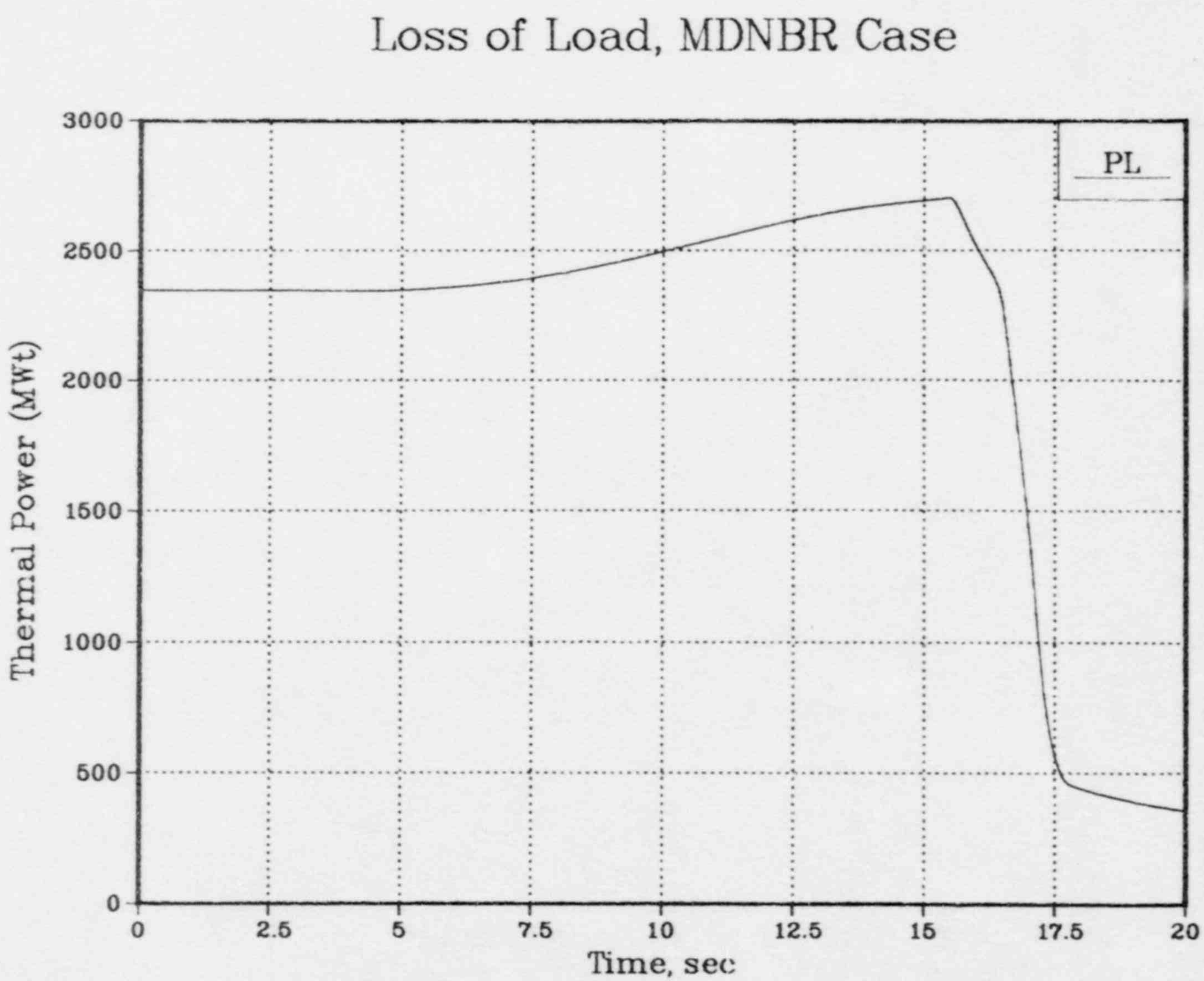

FIGURE 15.2.1-1 LOSS OF LOAD, REACTOR POWER LEVEL

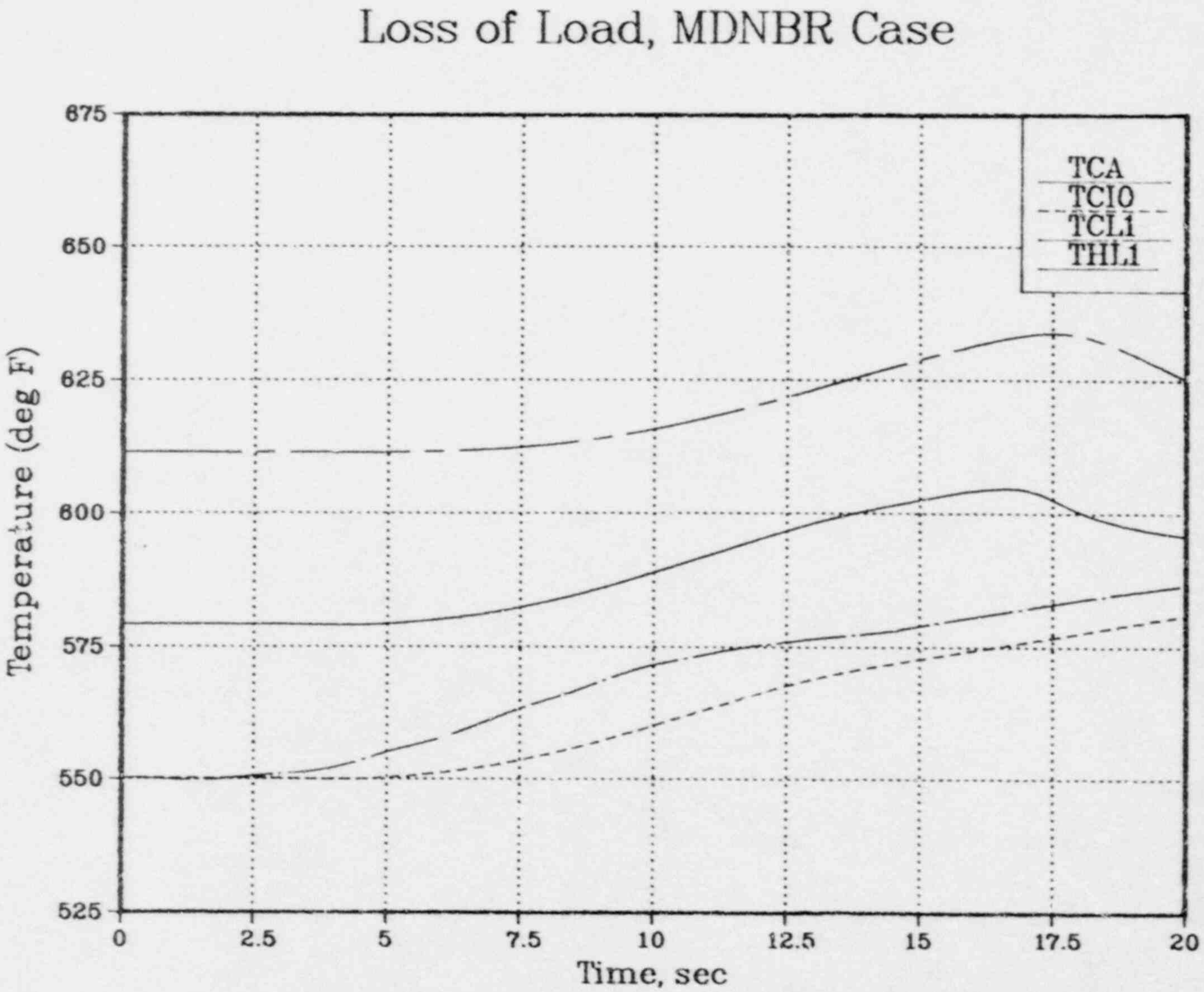

FIGURE 15.2.1-2 LOSS OF LOAD, CORE INLET, AVERAGE, COLD AND HOT LEG TEMPERATURES

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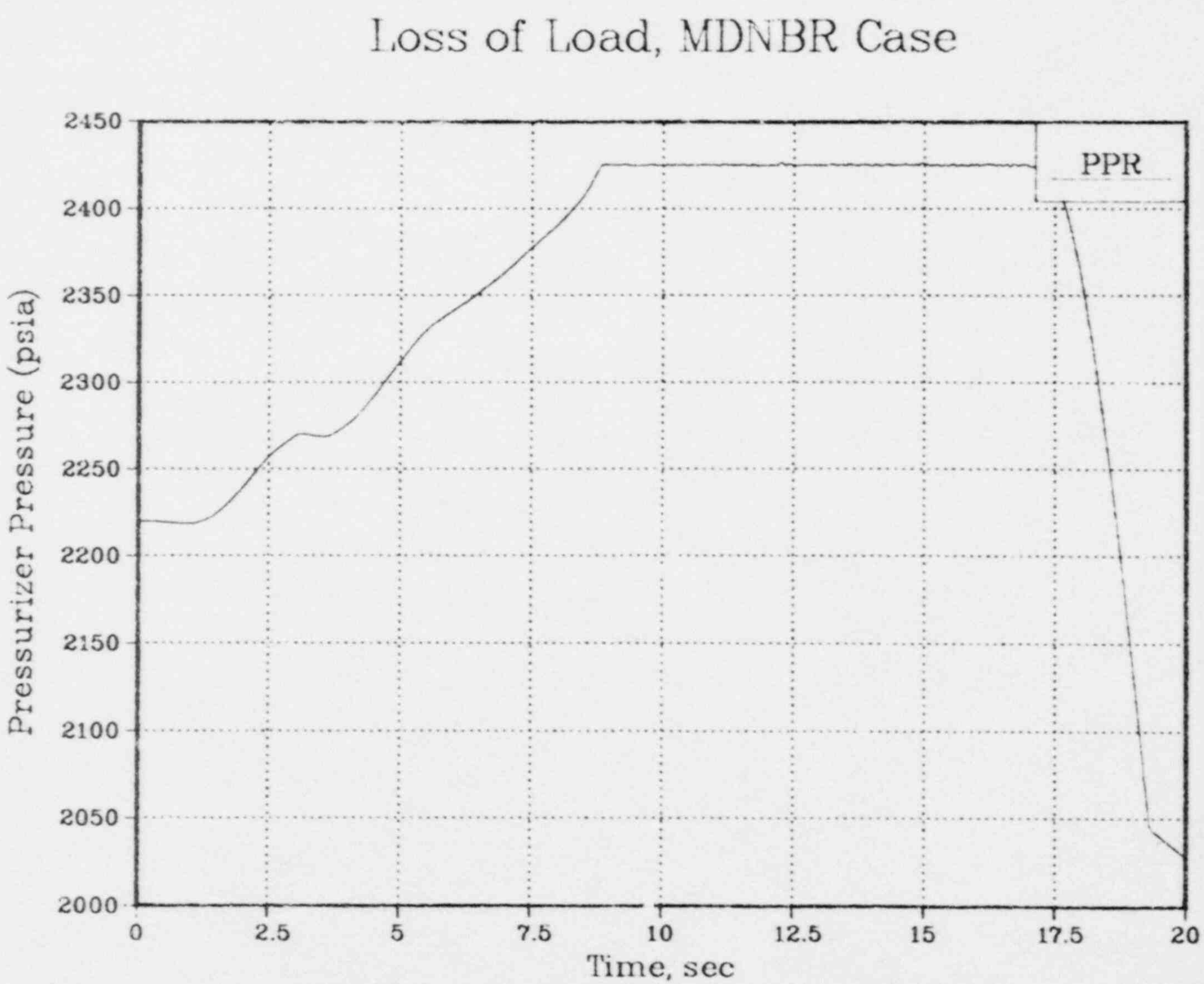

FIGURE 15.2.1-3 LOSS OF LOAD, PRESSURIZER PRESSURE

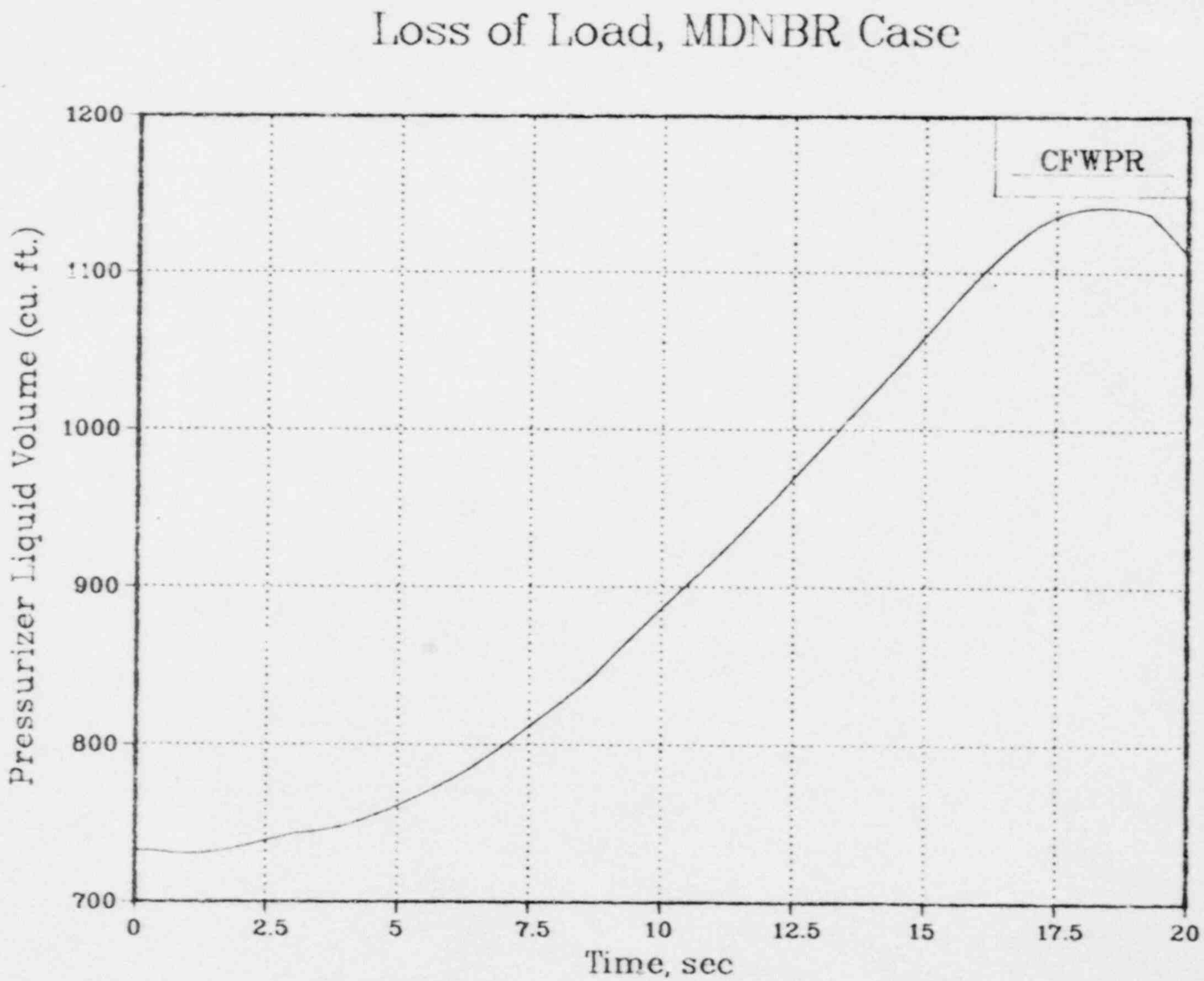

FIGURE 15.2.1-4 LOSS OF LOAD, PRESSURIZER LIQUID VOLUME

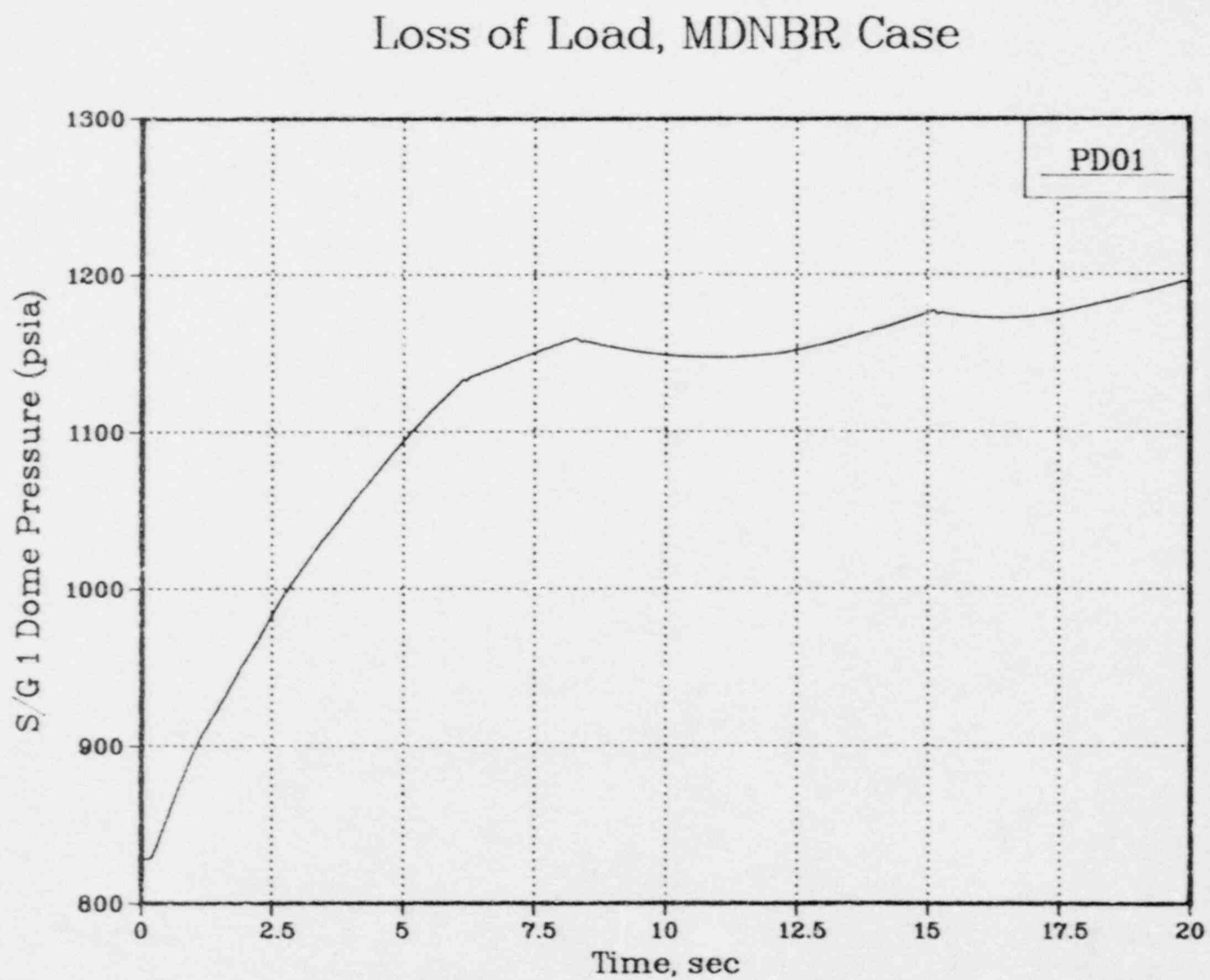

FIGURE 15.2.1-5 LOSS OF LOAD, STEAM GENERATOR DOME PRESSURE

15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER

The analysis addresses the limiting uncontrolled rod withdrawal transient resulting in reactor trip on the Overtemperature ΔT reactor trip. That case was determined by review of the reference analysis[2] to be a rated power case at BOC conditions. The reactivity insertion rate is that resulting in minimum DNBR, and results in simultaneous Overtemperature ΔT and power range high flux trips. The event is discussed below.

15.4.2.1 Identification of Causes and Event Description

This event is defined to result from an uncontrolled control bank withdrawal at full power. The event could be caused by misoperation of the most reactive control rod banks wired in common withdrawing at up to the maximum rate.

The reactor protection system is designed and set to preclude penetration of the SAFDLs. The Overtemperature ΔT and power range (high setting) high flux trips are principally challenged. Both trip setpoints include allowance for process variable measurement, processing channel drift, and operating variances from that indicated.

The Overtemperature ΔT function is designed and set to protect against DNB. Principal DNB parameters such as power (measured as core coolant temperature rise), core coolant temperature, primary pressure and core power distribution are measured, and the function decreases margin to trip setpoint when process variables indicate a decrease in operating margin. This function is established based on the core protection boundaries, operation within which assures protection of the SAFDLs.

Reactivity insertion rates are large enough at less than the maximum rate that core temperature rise lags behind nuclear power. The power range reactor trip protects the system from these events.

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 limiting rod bank withdrawal transient was analyzed to assess the impact of the new RTD installation which eliminates the bypass piping.

This event is classified as a Condition II event. The acceptance criterion is that the SAFDLs must not be penetrated. This will be assured if the minimum DNBR is above 1.17. The systems challenged in this event are redundant and no single active failure will adversely affect the consequences of the event.[8]

15.4.2.2 Analysis Method

The analysis is performed using the PTSPWR2[7] code and the XCOBRA-IIIC methodology.[3] 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 transposed to the XCOBRA-IIIC methodology[3] for calculation of the MDNBR.

15.4.2.3 Definition of Events Analyzed and Bounding Input

The limiting rod bank withdrawal event is from full power initial conditions with an insertion ramp of 2 pcm/sec and positive reactivity feedback.

Additional cases were analyzed to verify the trends of the reference analysis[2] plot of MDNBR versus reactivity insertion rate. This ensured that the limiting case was selected for analysis.

15.4.2.4 <u>Analysis of Results</u>

The limiting rod bank withdrawal transient was analyzed using a modified version of PTSPWR2 which simulates the new RTD installation. The limiting event was a reactivity insertion ramp of 2 pcm/sec from full power initial conditions with positive reactivity feedback. Initial conditions for the event are summarized in Table 15.4.2-1.

Thermal power increases steadily throughout the transient in response to the reactivity insertion until the occurrence of reactor scram. Coolant temperatures also increase steadily due to the primary-to-secondary system power mismatch. The pressure increase due to coolant expansion and insurge flow to the pressurizer is limited to a maximum of 2274 psia by the primary PORVs. Increasing core power and temperature result in a reactor trip on the Overtemperature ΔT reactor trip at 27.2 sec. The MDNBR occurs shortly after the beginning of scram.

Figures 15.4.2-1 through 15.4.2-6 show the plant responses for the limiting rod bank withdrawal transient. Table 15.4.2-2 presents the sequence of events for this event. The calculated MDNBR is 1.19.

Calculations were performed for various reactivity insertion rates to verify that the limiting reactivity insertion rate, which results in a trip on the Overtemperature ΔT trip function, determined in the reference analysis[2] (2 pcm/sec) is still the limiting reactivity insertion rate.

15.4.2.5 <u>Conclusion</u>

The MDNBR is greater than the XNB correlation safety limit. Therefore, the DNBR acceptance criterion is met.

Table 15.4.2-1 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL - SUMMARY OF INITIAL OPERATING CONDITIONS

Parameter	Value
Power (MWt)	2346
Core Inlet Temperature (°F)	550.2
Pressurizer Pressure (psia)	2220
Reactor Coolant System Flow Rate (lbm/hr)	97.29 × [illegible]

Table 15.4.2-2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL EVENT SEQUENCE

Event	Time (sec)
Uncontrolled RCCA Bank Withdrawal begins	0.0
Overtemperature ΔT Setpoint reached	26.5
Scram Results in Rod Motion	27.2
Minimum DNBR occurs	27.4

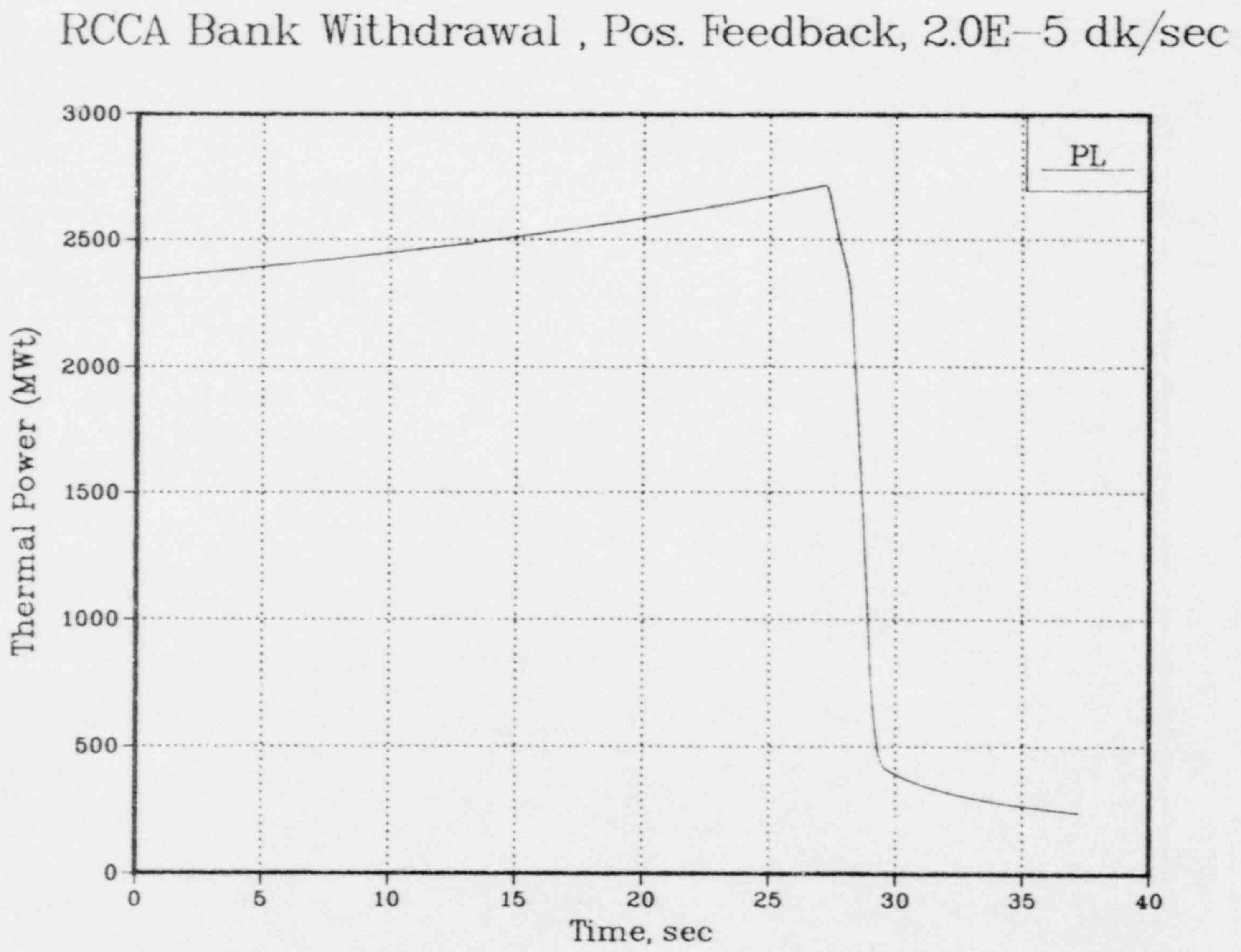

FIGURE 15.4.2-1 SLOW ROD WITHDRAWAL, 100% RATED POWER, REACTOR POWER

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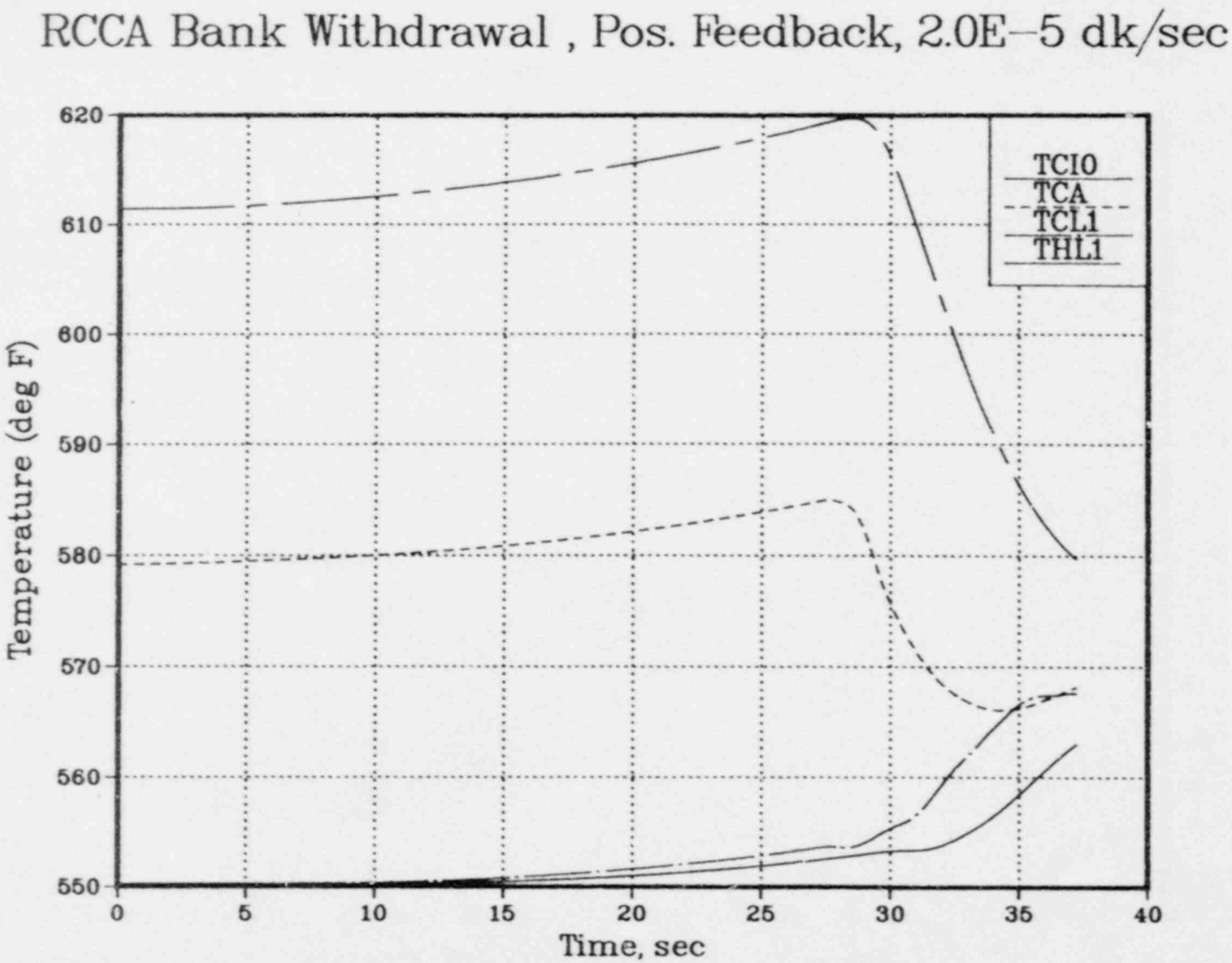

FIGURE 15.4.2-2 SLOW ROD WITHDRAWAL, 100% RATED POWER, PRIMARY COOLANT CORE INLET, AVERAGE, COLD AND HOT LEG TEMPERATURES

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RCCA Bank Withdrawal , Pos. Feedback, 2.0E-5 dk/sec

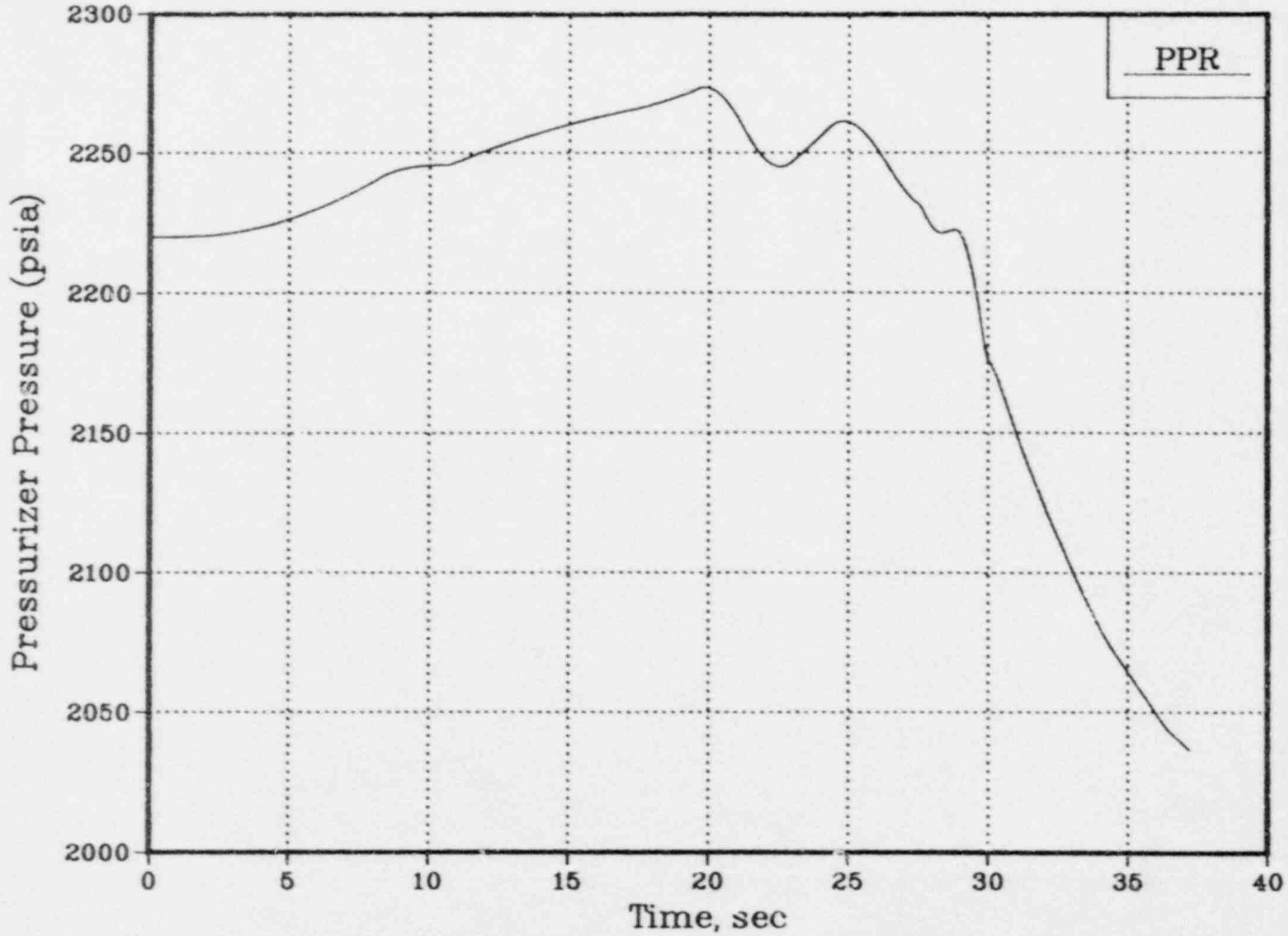

FIGURE 15.4.2-3 SLOW ROD WITHDRAWAL, 100% RATED POWER, PRESSURIZER PRESSURE

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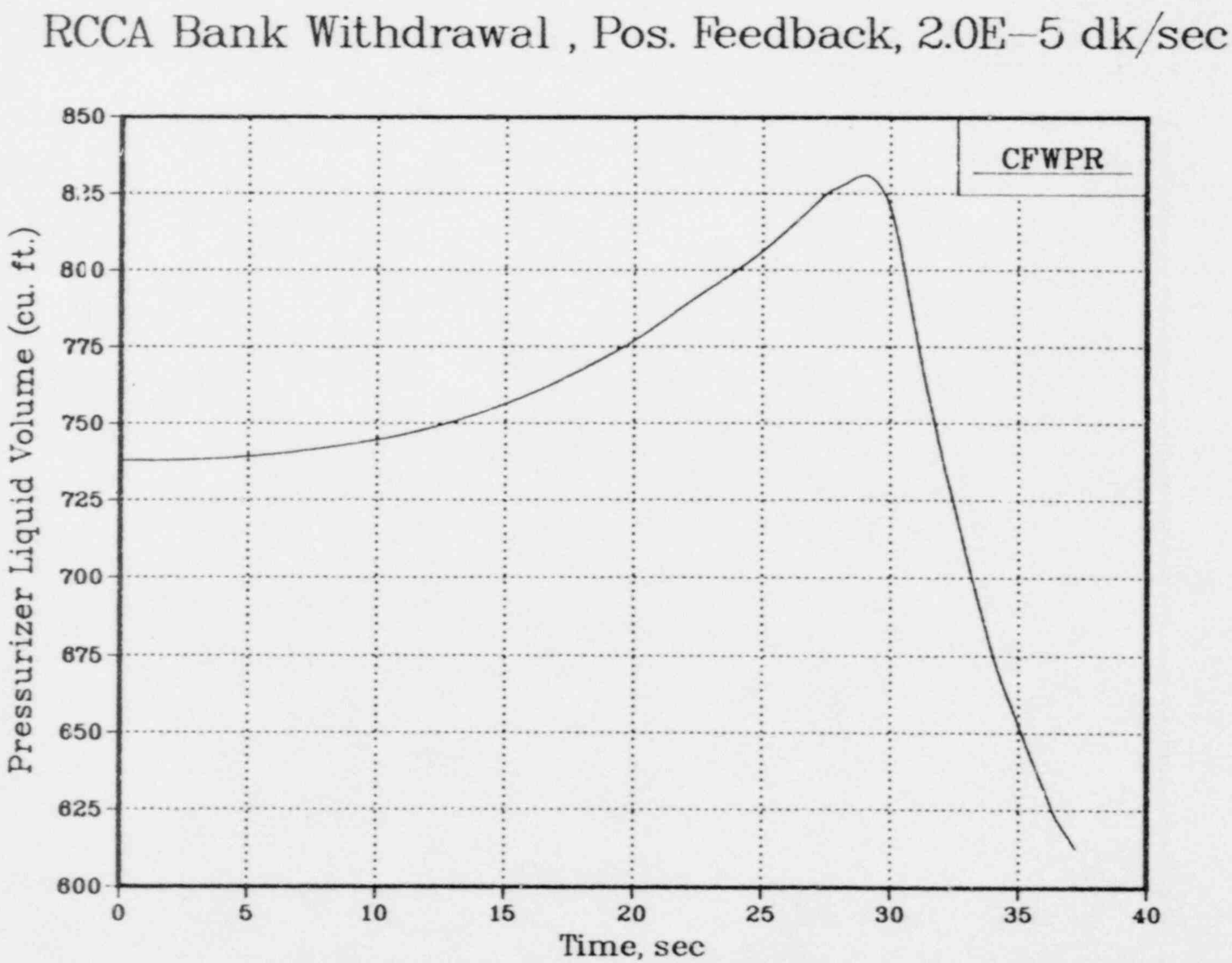

FIGURE 15.4.2-4 SLOW ROD WITHDRAWAL, 100% RATED POWER, PRESSURIZER LIQUID VOLUME

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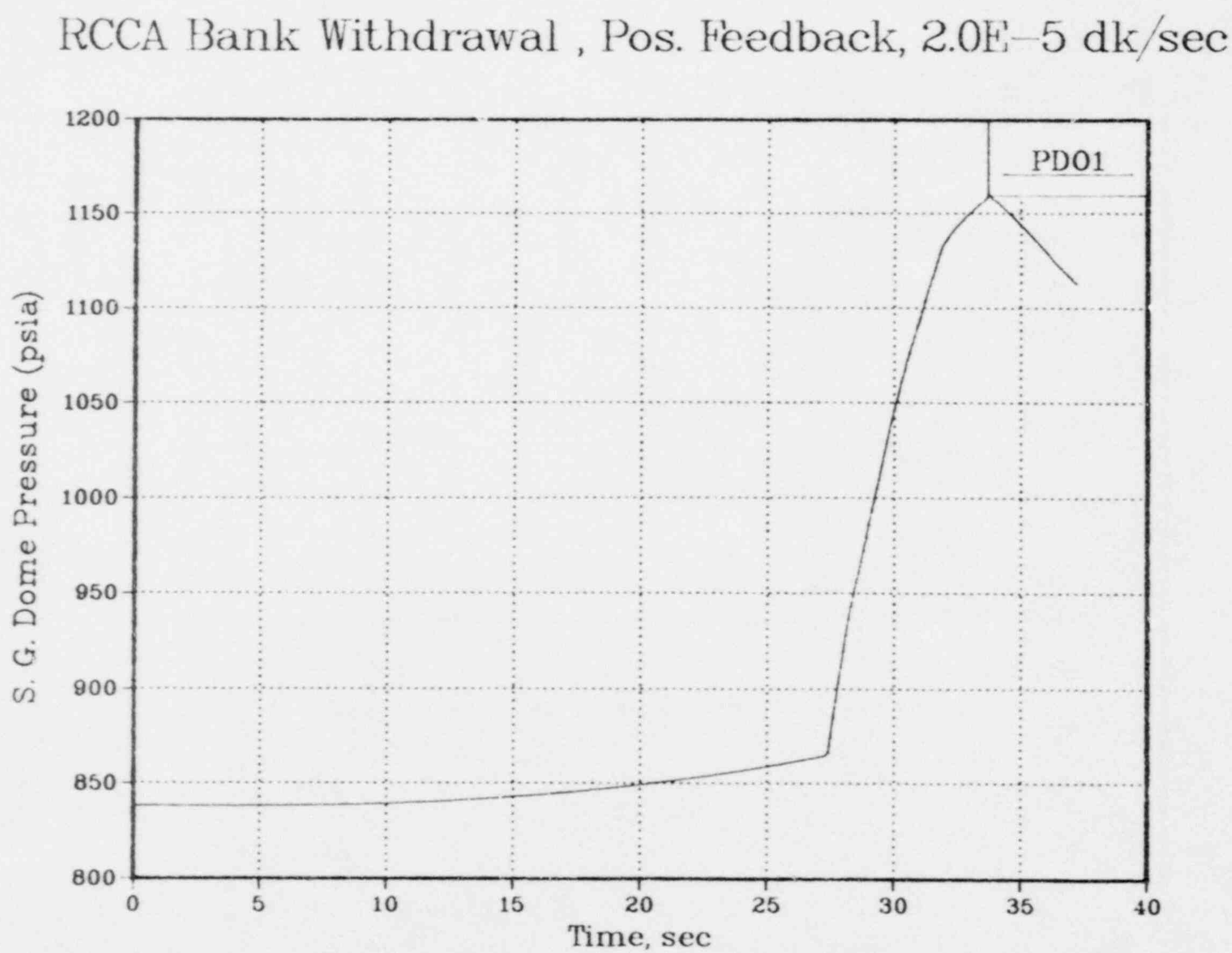

FIGURE 15.4.2-5 SLOW ROD WITHDRAWAL, 100% RATED POWER, STEAM GENERATOR DOME PRESSURE

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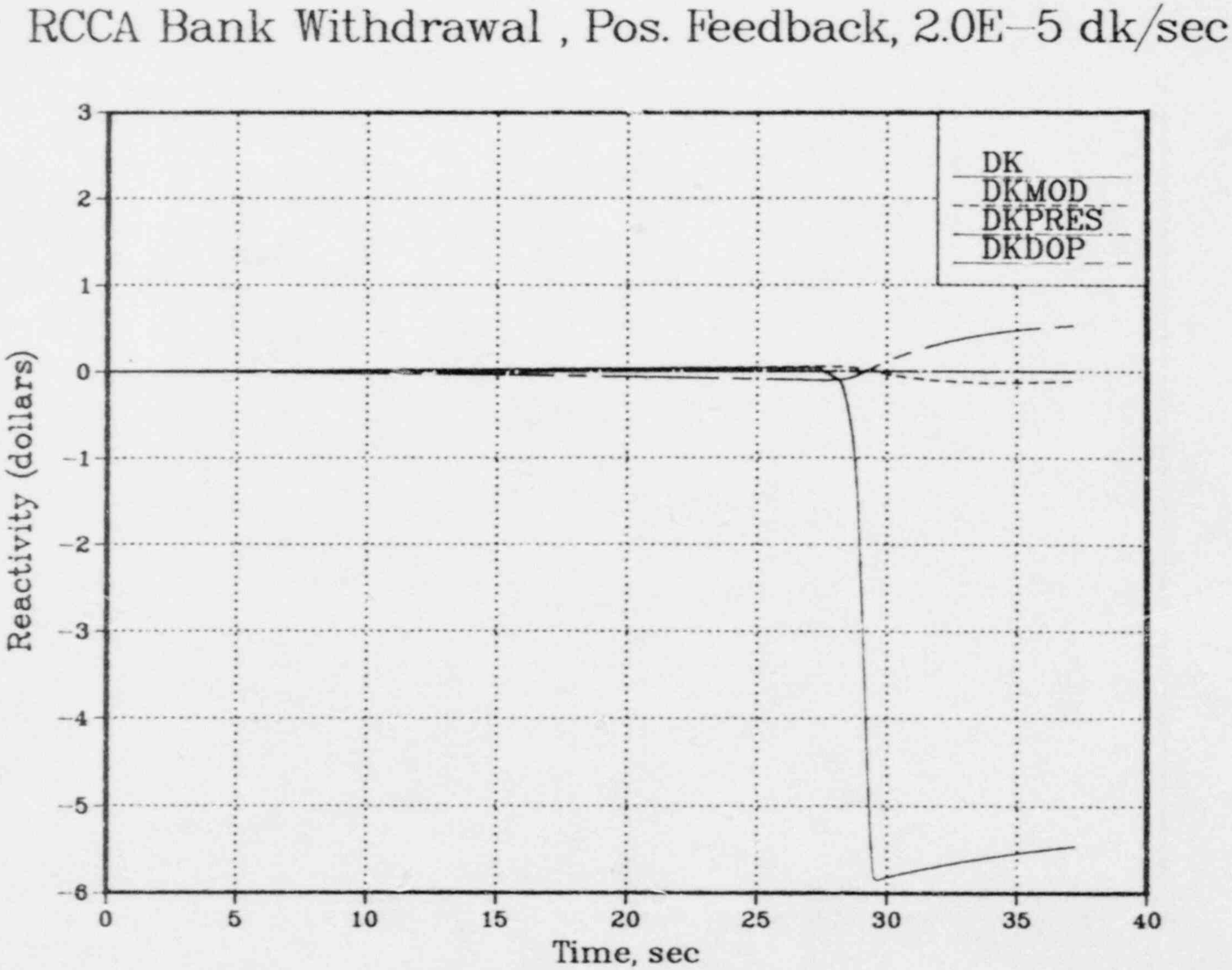

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FIGURE 15.4.2-6 SLOW ROD WITHDRAWAL, 100% RATED POWER, REACTIVITY ADDITIONS

15.4.3 CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

The analysis addresses the limiting rod drop transient resulting in reactor trip on the Overtemperature ΔT reactor trip. The limiting case assumes manual rod control, a non-positive moderator temperature coefficient above 50% power, and active turbine runback. The event is analyzed to verify the OTΔT trip function with the new RTD installation which eliminates the bypass piping.

15.4.3.1 Identification of Causes and Event Description

The event is defined to be initiated by a dropped RCCA. The dropped RCCA promptly inserts negative reactivity which reduces reactor power and disturbs the power distribution, resulting in increased local power peaking. The rod bottom and negative flux rate signals can independently initiate turbine runback to 70% of full power. With turbine runback, the reduction in load initially results in a load mismatch if the dropped rod reactivity does not match that required for the runback power level. If reactivity insertion is greater than that required to match the runback power level, T_{avg} initially decreases. If reactivity is less, T_{avg} initially increases. The reactor protection system will limit consequences should conditions approach setpoint values.

This event is classified as a Condition II event. The acceptance criteria for this event is that the MDNBR is greater than 1.17.

15.4.3.2 Analysis Method

The analyses are performed by coupling a conservative power peak to transient response and DNBR 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 reactor simulator.[9] Standard neutronic methodology is used to calculate neutronics parameters such as control rod worth and power peaking.

The system response to a single dropped RCCA is analyzed with the PTSPWR2[7] code. The DNB analysis is performed using the XCOBRA methodology[3], using the operating conditions from the PTS calculation. Local power redistribution effects due to the dropped rod are input to the XCOBRA methodology by a local power augmentation factor. The Technical Specification value of the allowed $F_{\Delta H}$ is multiplied by this augmentation factor.

15.4.3.3 Definition of Events Analyzed and Bounding Input

For control rod misoperation events, the maximization of power peaking results in a reduction in the DNBR. To assure that bounding values are determined for the radial power peaking, the following approach is used. The increase in power peaking above that associated with equilibrium steady state conditions is determined for a spectrum of cycle exposures and applicable control rod configurations. Based on these results, a conservative augmentation factor is derived.

The limiting Condition II event analyzed is a dropped full length RCCA of low worth with turbine runback. No single failure assumption is required since manual rod control is assumed.

15.4.3.4 Analysis of Results

The event initiates with a step negative reactivity insertion representing a single minimum worth dropped rod. At event initiation, a turbine runback signal is assumed. Turbine load demand reaches its programmed value at 9 sec. Average coolant temperature at first decreases in response to the power reduction caused by the dropped rod, but later increases due to the reduced secondary load demand. The temperature increase causes insurge to the pressurizer, resulting in a pressure increase sufficient to open the PORVs at

16.1 sec. Reactor scram on Overtemperature ΔT occurs at 61.1 seconds. The reactor power continues to decrease after the rod is dropped.

The MDNBR as calculated by the XCOBRA methodology is 1.23, greater than the 1.17 XNB DNBR limit. The minimum DNBR occurs at 61.2 seconds.

A summary of initial conditions is presented in Table 15.4.3-1. The sequence of events for the limiting case is presented in Table 15.4.3-2.

15.4.3.5 Conclusion

The minimum DNBR is greater than the XNB correlation safety limit of 1.17. Therefore, the event acceptance criterion on DNBR is met.

Table 15.4.3-1 DROPPED FULL LENGTH RCCA (MANUAL) - SUMMARY CF INITIAL CONDITIONS

Condition	Value
Power, MWt	2346
Core Inlet Temperature, °F	550.2
Pressurizer Pressure, psia	2220
Reactor Coolant System Flow Rate, lb/hr	97.29×10^6

Table 15.4.3-2 DROPPED FULL LENGTH RCCA (MANUAL) EVENT SEQUENCE

Event	With Turbine Runback Time, sec
Dropped RCCA Fully in	0.0
Turbine Runback Begins	0.0
Turbine Runback Reaches Low Load Limit	9.0
Pressurizer PORVs Open	16.1
Peak Pressurizer Pressure	17.0
Reactor Scram and Rods begin to Fall (Overtemperature ΔT)	61.1
Peak Core Power Level	0.0
Steam Generator Safety Valves Open	57.5
Minimum DNBR Occurs	61.2

RCCA Drop, 0.0 MTC, no ARC, w/ Turbine Runback

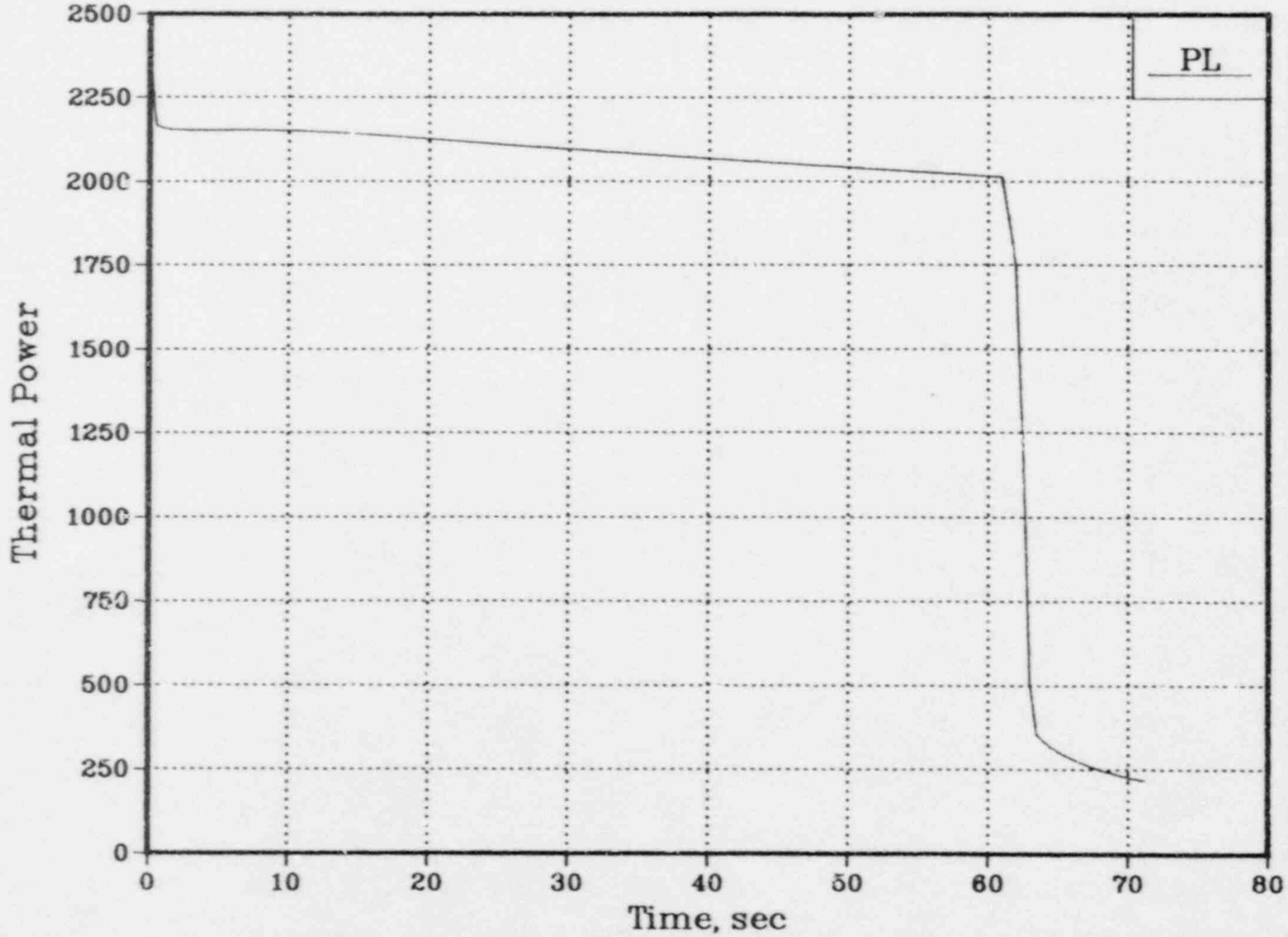

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FIGURE 15.4.3-1 RCCA DROP, REACTOR POWER

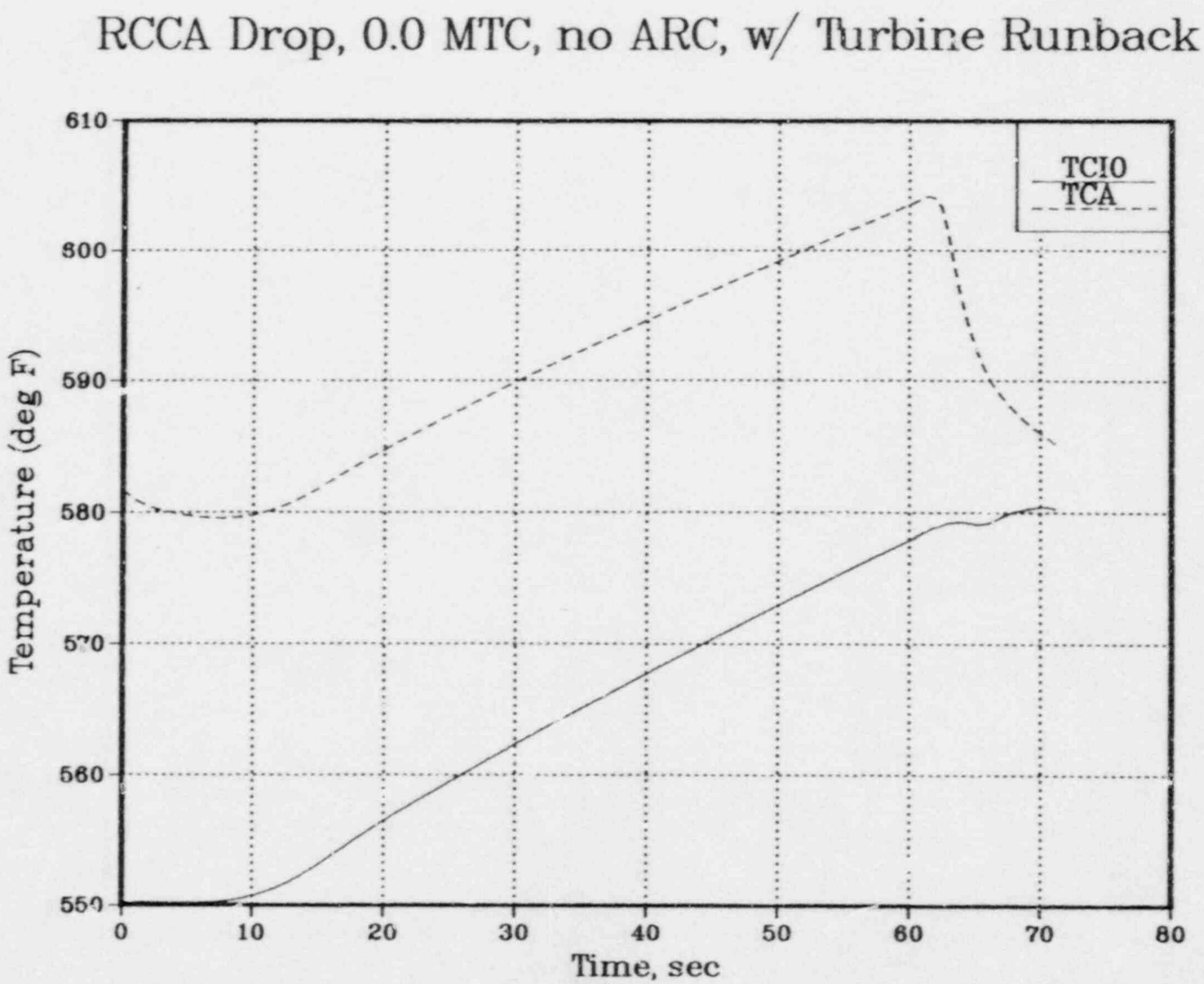

FIGURE 15.4.3-2 RCCA DROP, CORE AVERAGE AND INLET COOLANT TEMPERATURES

RCCA Drop, 0.0 MTC, no ARC, w/ Turbine Runback

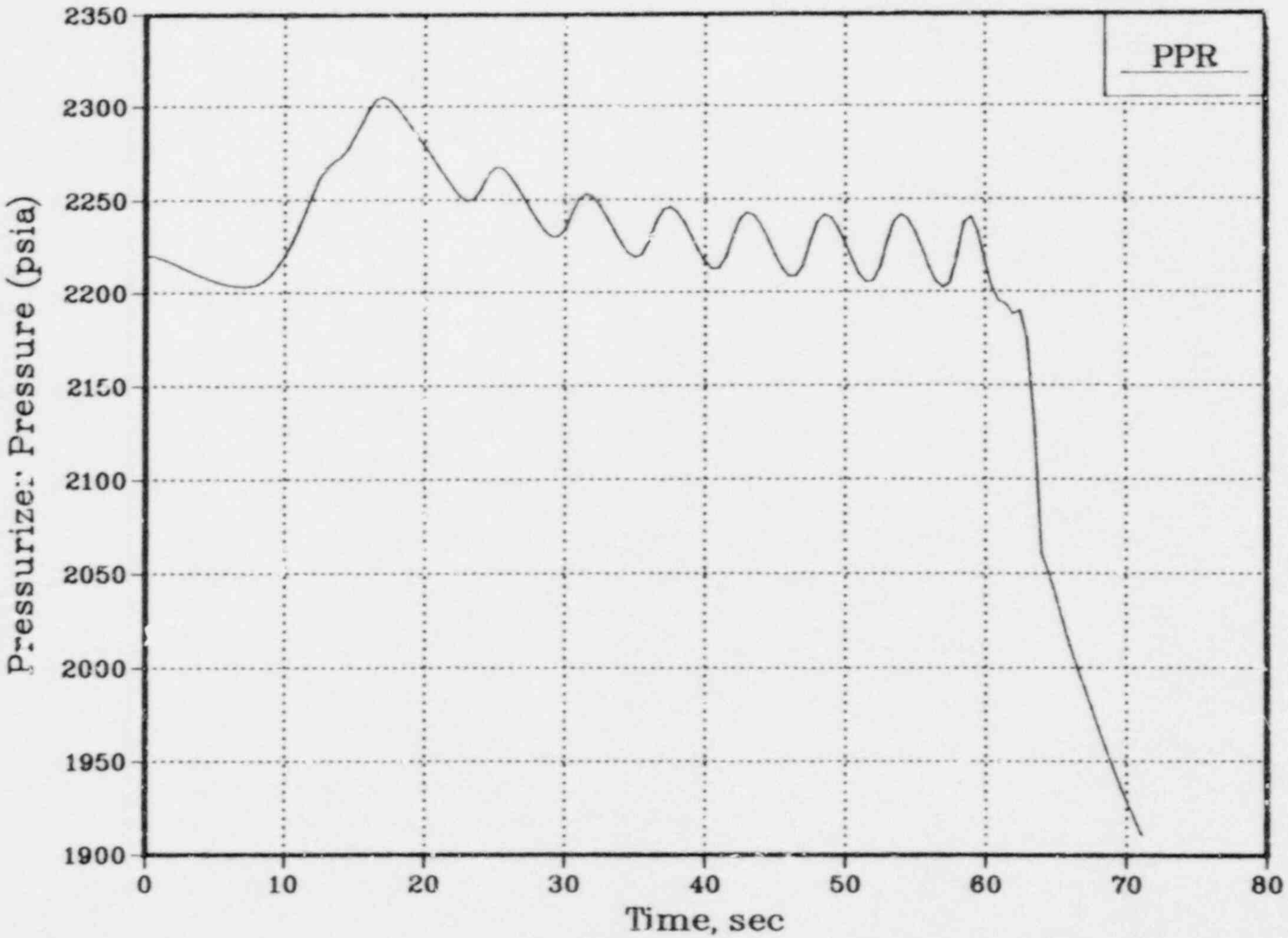

FIGURE 15.1.3-3 RCCA DROP, PRESSURIZER PRESSURE

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