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PLANT TRANSIENT ANALYSIS FOR THE DONALD C. COOK UNIT 2 REACTOR AT 3425 MWt OPERATION WITH 5% STEAM GENERATOR TUBE PLUGGING

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1.0 INTRODUCTION

Plant transient analyses are presented to support future cycle operation of the D.C. Cook Unit 2 nuclear power plant at 3425 MWt with 5% average steam generator tube plugging. The major impacts of steam generator tube plugging are a slight reduction in primary coolant flow rate and a slight degradation of primary to secondary system heat transfer. These effects motivated a re-analysis of those events previously demonstrated to be limiting with respect to thermal margin and reactor vessel pressurization. An asymmetric tube plugging distribution can adversely affect thermal margin in the locked rotor accident and is therefore assumed as the basis for analysis in the simulation of that event. Three of the steam generators were assumed 6.7% plugged, and the steam generator with the locked rotor was assumed to be unplugged. Plant transient analyses to support the 3425 MWt core power level have been reported previously(1).

Section 2.0 of this report provides a summary of the results for this analysis. Section 3.0 describes the calculational methods and input parameters employed. A more detailed description of individual event simulations is given in Section 4.0. Those events which are considered in the plant FSAR(2) but not reanalyzed here are discussed in Section 5.0.

2.0 SUMMARY

The plant transient analysis reported here has been performed to evaluate the response of the Donald C. Cook Unit 2 reactor core and reactor protection system (RPS) during anticipated operational occurrences (AOOs) and postulated accidents (PAs). The analysis supports operation of ENC reload fuel in D.C. Cook Unit 2 at a core power of 3425 MWt with an average steam generator tube plugging level of 5%. The fuel and vessel design criteria to be satisfied in the analysis are listed in Table 2.1. The key results of the analysis are given in Table 2.2. The results confirm that applicable fuel and vessel design criteria are met for the previously identified limiting FSAR transients. The least MDNBR calculated for any AOO event occurred in a slow rod withdrawal event and is well above the XNB correlation safety limit of 1.17. Peak pressure calculated for any event was 2575 psia, and occurred in the loss of load event. This is well below the criteria at maximum system pressure of 2750 psia. The MDNBRs for DNB-limiting transients reported in Table 2.2 have been calculated using ENC's automated-crossflow core thermal hydraulics simulation methodology[3]. A confirmatory analysis of the locked rotor with concurrent loss of offsite power has also been performed. Radiological release for this postulated accident is within 10 CFR 100 limit.[4]

The transient events reanalyzed and reported here comprise an adequate scope of analysis to assure the safe operation of D.C. Cook Unit 2 with 5% steam generator tube plugging. Those anticipated operational occurrences and postulated accidents whose results most closely approach specified acceptable

fuel design limits or the vessel pressurization limit have been reanalyzed and have been shown to satisfy applicable criteria. Section 5.0 demonstrates that the results of the FSAR events are bounded by the results of events which have been analyzed here or that the significant conclusions of the FSAR analyses of these events remain valid under the conditions of this analysis.

Thermal margin for the Cycle 5 (XN-2) core is significantly greater than that of the Cycle 4 core due to a reduction in the $F_{\Delta H}^{N}$ which was analyzed and to the larger number of ENC fuel assemblies present in Cycle 5. An $F_{\Delta H}^{N}$ of 1.55 was analyzed which reduces the peak assembly power by 3.1%, relative to the previous analysis(1) value of 1.60. Further, the larger number of ENC assemblies present in Cycle 5 significantly reduces the coolant mass flux penalty suffered by ENC fuel in Cycle 4. The combined effect of these factors on available thermal margin more than outweighs the effects of the calculated 1.1% RCS flow reduction due to tube plugging.

Core safety limits and the overtemperature ΔT (OT-ΔT) reactor trip setpoint reported in the previous analysis(1) are conservatively applicable to Cycle 5 and future cycles with an $F_{\Delta H}^{N}$ of 1.55. Adequate functioning of the OT-ΔT trip function has been demonstrated in the rod withdrawal analyses presented in Section 4.1.

Of the FSAR events analyzed, the slow rod withdrawal is the most limiting AOO. The calculated MDNBR for this event included allowances for uncertainties in core operating conditions as discussed in Section 3.0. The event tripped on OT-ΔT, which includes all applicable measurement and calibration uncertainties. Thus, the reported MDNBR values in Table 2.2 for the events

which tripped on OT-ΔT include a double accounting for core parameter uncertainties. Elimination of doubly accounting for these uncertainties in the MDNBR evaluation results in the MDNBR increasing to 1.44 for the slow rod withdrawal transient. The slow rod withdrawal transient remains the limiting AOO analyzed.

Table 2.1 Applicable Fuel and Vessel Design Limits

Event Class	Applicable Criteria
Anticipated Operational Occurrence	Peak System Pressure $\leq$ 2750 psia MDNBR Calculated with XNB Critical Heat Flux Correlation $\geq$ 1.17
Postulated Accident	Radiological Release below 10 CFR 100 Limits Peak System Pressure $\leq$ 2750 psia

Table 2.2 Summary of Results

Transient	Maximum Power Level (MWt)	Maximum Core Average Heat Flux (Btu/hr-ft^2)	Maximum Pressurizer Pressure (psia)	MDNBR (XNB)
Initial Conditions for Transients	3425.	197580.	2250.	1.878
Uncontrolled Rod Withdrawal @ 8.4×10^{-3} $\Delta\rho$/sec	4616.	220260.	2300.	1.625
Uncontrolled Rod Withdrawal @ 7.4×10^{-6} $\Delta\rho$/sec	4140.	234360.	2320.	1.265
Loss of Flow -				
Locked Rotor	3824.	199700.	2308.	1.276
Locked Rotor*	3996.	198690.	2380.	0.698
Loss of Load	3610.	200760.	2526.	1.878
Decreased Feedwater Heating	3743.	215910.	2252.	1.679
Decreased Feedwater Heating with Automatic Rod Control	3767.	217130.	2260.	1.627
Excessive Load Increase	4048.	232630.	2252.	1.532
Excessive Load Increase with Automatic Rod Control	4042.	229180.	2260.	1.569

* With concurrent loss of offsite power. Radiological release is within 10 CFR 100 limits.

3.0 CALCULATIONAL METHODS AND INPUT PARAMETERS

The D.C. Cook Unit 2 plant transient analysis was performed using the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR2)[5]. The PTSPWR2 code is an Exxon Nuclear digital computer program which models the behavior of pressurized water reactors under normal and abnormal operating conditions. The computer code is based on the solution of the basic transient conservation equations for the primary and secondary coolant systems. The transient conduction equation is solved for the fuel rods, and a point kinetics model is used to evaluate the core neutronics. The program calculates fluid conditions such as flow, pressure, mass inventory and steam quality, heat flux in the core, reactor power, and reactivity during the transient. Various control and safety system components are included as necessary to analyze postulated events. A hot channel model is included to trace the departure from nucleate boiling (DNB) during transients. The DNB evaluation is based on the hot rod heat flux in the high enthalpy rise subchannel and uses the XNB correlation[6] to calculate the DNB heat flux. Model features of the PTSPWR2 code are described in detail in Reference 5. Calculational methodology employed in this analysis is in accordance with ENC standard plant transient analysis methodology for PWRs[7].

A diagram of the system model used by PTSPWR2 is shown in Figure 3.1. As illustrated, the PTSPWR2 code models the reactor, two independent primary coolant loops including all major components: pressurizer, pumps, steam generators, and the steam lines, including all major valves (turbine stop valves, isolation valves, pressure relief valves, etc.). PTSPWR2 loop 2 is a lumped loop model of D.C. Cook Unit 2 primary loops 2, 3 and 4.

The present calculations were performed using the NOV76A version of ENC's PTSPWR2 code with appropriate updates. Updates are included to describe the D.C. Cook Unit 2 plant control systems.

Steady state measurement and instrumentation errors are taken into account to ensure conservatively calculated values of MDNBR. The corresponding plant initial conditions in the MDNBR calculations are as follows:

Reactor Power	=	3425 MWt + 2% (68.5 MWt) for calorometric error.
Inlet Coolant Temperature	=	542.2 + 4^{o}F for deadband and measurement error.
Primary Coolant System Pressure	=	2250 - 30 psia for steady state fluctuation and measurement errors.

Primary Coolant Flow* = 143.1 Mlbm/hr - 3.5% for measurement uncertainty.

The simultaneous application of the above parameter uncertainties minimizes the initial minimum DNB ratio in a bounding fashion. It is noted that the above steady state errors are not generally included in the plant system modeling, but rather are used to conservatively bound the calculated MDNBR. Table 3.1 shows a list of operating parameters used in the analysis.

Unless otherwise noted, the transient simulations reported herein have assumed that pressurizer spray and power-operated relief valves are fully operable in order to maintain system pressure at a minimum value. This results in the most conservative estimation of the MDNBR. These pressure control functions are assumed inoperable in those events simulated for comparison to the system pressurization criteria.

The trip setpoints incorporated into the PTSPWR2 model for D.C. Cook Unit 2 are based on the Technical Specification limits. These limiting trip setpoints are modeled in the plant transient analysis to demonstrate the adequacy of the reactor protection system for operation at a 3425 MWt rating with 5% steam generator tube plugging. Reactor trip setpoints and scram delay times associated with them are listed in Table 3.2. Adequate allowance has been made for trip instrument channel measurement uncertainties and calibration errors.

The ENC fuel design parameters for D.C. Cook Unit 2 are summarized in Table 3.3. Table 3.4 lists the neutronics parameter values which are

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*Value includes a 1.1% reduction from the current measured flow of 144.7 Mlbm/hr to account for increased loop resistance due to 5% steam generator tube plugging.

calculated to conservatively bound the D.C. Cook Unit 2 core for both the beginning and end of cycle. A design axial power profile with a peaking factor F_Z = 1.55 was used in the analysis. This profile is shown in Figure 3.2.

The scram reactivity curve used in the analysis is shown in Figure 3.3. This curve is taken from the D.C. Cook Unit 2 FSAR.(2) Scram delay times employed in the plant transient simulations are sufficiently conservative with respect to Technical Specification limits on reactor trip system performance to assure conservative simulation of reactor scram. In Figure 3.3, the scram reactivity is normalized to the total rod worth.

Table 3.1 Operating Parameters Used in PTSPWR2 Analysis of Donald C. Cook Unit 2

Core	
Total Core Heat Output, MWt	3425.
Total Core Heat Output, MBtu/hr	11,688
Heat Generated in Fuel, %	97.4
System Pressure, psia	2250
Hot Channel Factors	
Total Peaking Factor, F_Q^T	2.47
Enthalpy Rise Factor, $F_{\Delta H}^N$	1.55
Coolant Flow Rate, Mlbm/hr	138.1
Effective Core Flow Rate, Mlbm/hr	131.9
Coolant Average Temperature, ^oF	574.1
Heat Transfer	
Average Heat Flux, Btu/hr-ft^2	197,580
Steam Generators	
Total Steam Flow Mlbm/hr, per lead	3.70
Steam Temperature, ^oF	518.
Steam Pressure, psia	799.
Feedwater Temperature, ^oF	431.
Tube Plugging, %	5.0

Table 3.2 Donald C. Cook Unit 2 Trip Setpoints

	Setpoint	Used in Analysis	Delay Time
High Neutron Flux	109%	118%	0.5 sec
Low Reactor Coolant Flow	90%	87%	1.0 sec
High Pressurizer Pressure	2400 psia	2425 psia	2.0 sec
Low Pressurizer Pressure	1965 psia	1940 psia	2.0 sec
Low-Low Steam Generator Water Level	21% of span	0% of span	2.0 sec
Overtemperature ΔT*	T_{AVE_0} = 574.1^{0}F	T_{AVE_0} = 574.1^{0}F	6.0 sec
	P_0 = 2250 psia	P_0 = 2250 psia	
	K_1 = 1.267	K_1 = 1.452	
	K_3 = .000926	K_3 = .000744	

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* The overtemperature ΔT trip is a function of pressurizer pressure, coolant average temperature, and axial offset. The T_{AVE_0} and P_0 setpoints, and the setpoint bias K_1 are contained within the functional relationship. The bias constant K_1 employed in the analysis includes allowance for applicable trip channel uncertainties. Other constants in the overtemperature ΔT setpoint as it appears in the Technical Specification (gains, lead and lag constants) were incorporated without change in the analysis.

Table 3.3 Donald C. Cook Unit 2 Fuel Design Parameters - Exxon Nuclear Fuel

Fuel Radius	.1515 inches
Inner Clad Radius	.1550 inches
Outer Clad Radius	.1800 inches
Active Length	144.0 inches
Number of Fuel Rods in Core	50,952

Table 3.4 Donald C. Cook Unit 2 Kinetics Parameters Supported by the Plant Transient Analysis

Parameter	Value	
	Beginning-of-Cycle	End-of-Cycle
Moderator Coefficient ($\Delta\rho/^{o}F \times 10^4$)	+0.5	-3.9
Doppler Coefficient ($\Delta\rho/^{o}F \times 10^5$)	-1.0	-1.7
Pressure Coefficient ($\Delta\rho$/psia $\times 10^6$)	-.60	+4.3
Delayed Neutron Fraction (%)	0.61	0.510
Total Rod Worth (% $\Delta\rho$)	4.00	4.00

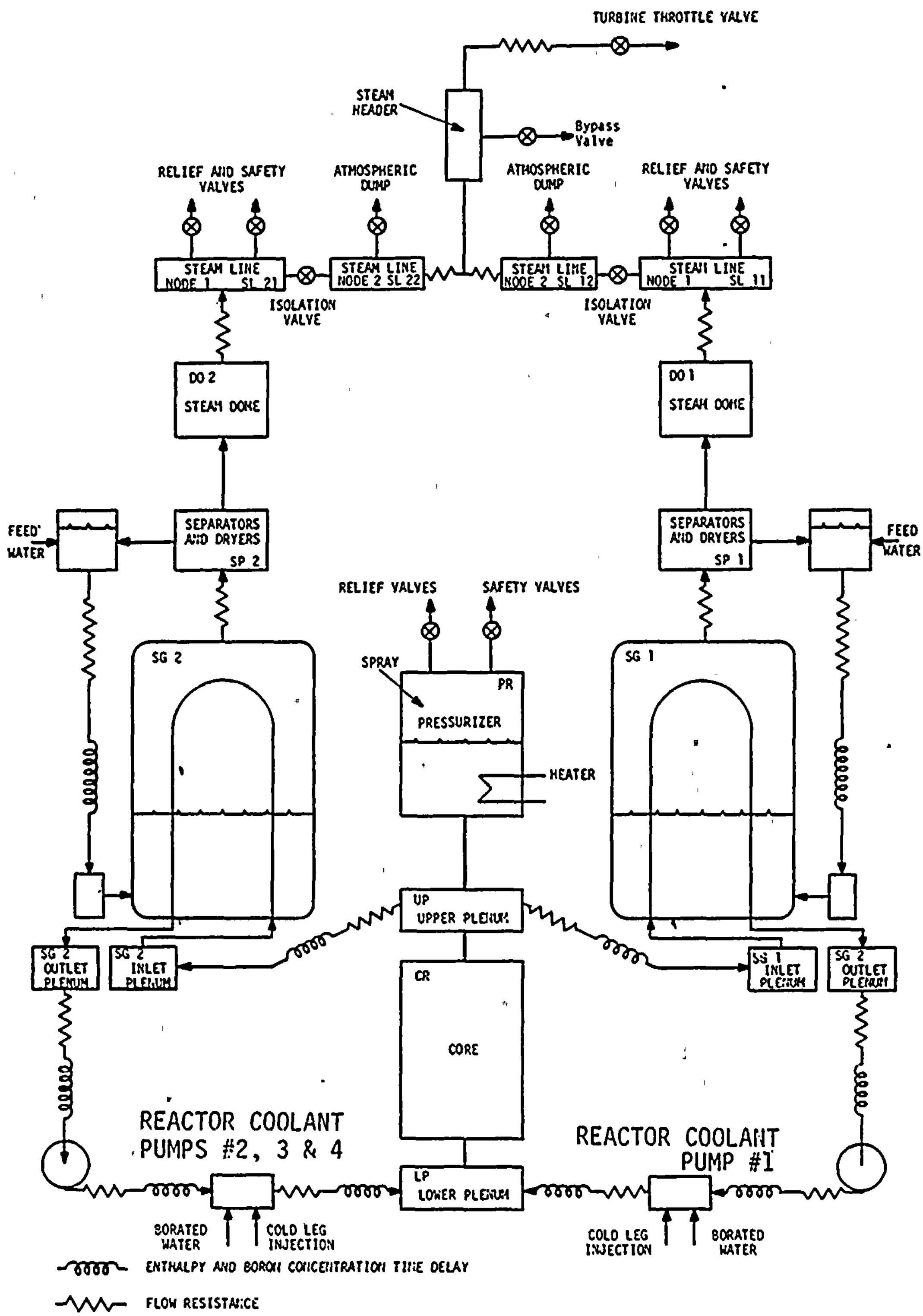

Figure 3.1 PTSPWR2 System Model

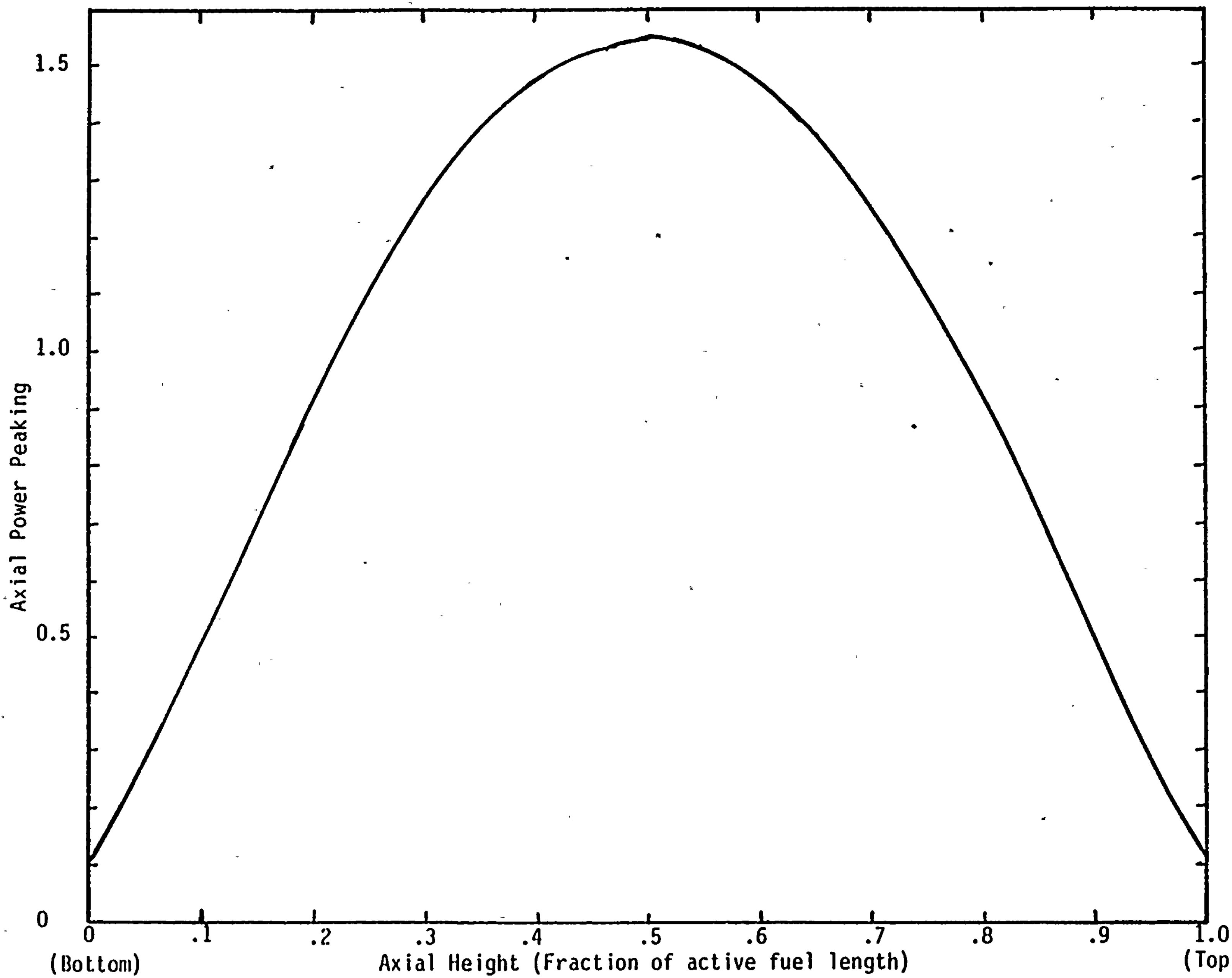

Figure 3.2 Axial power profile used in transient analysis of D. C. Cook Unit 2

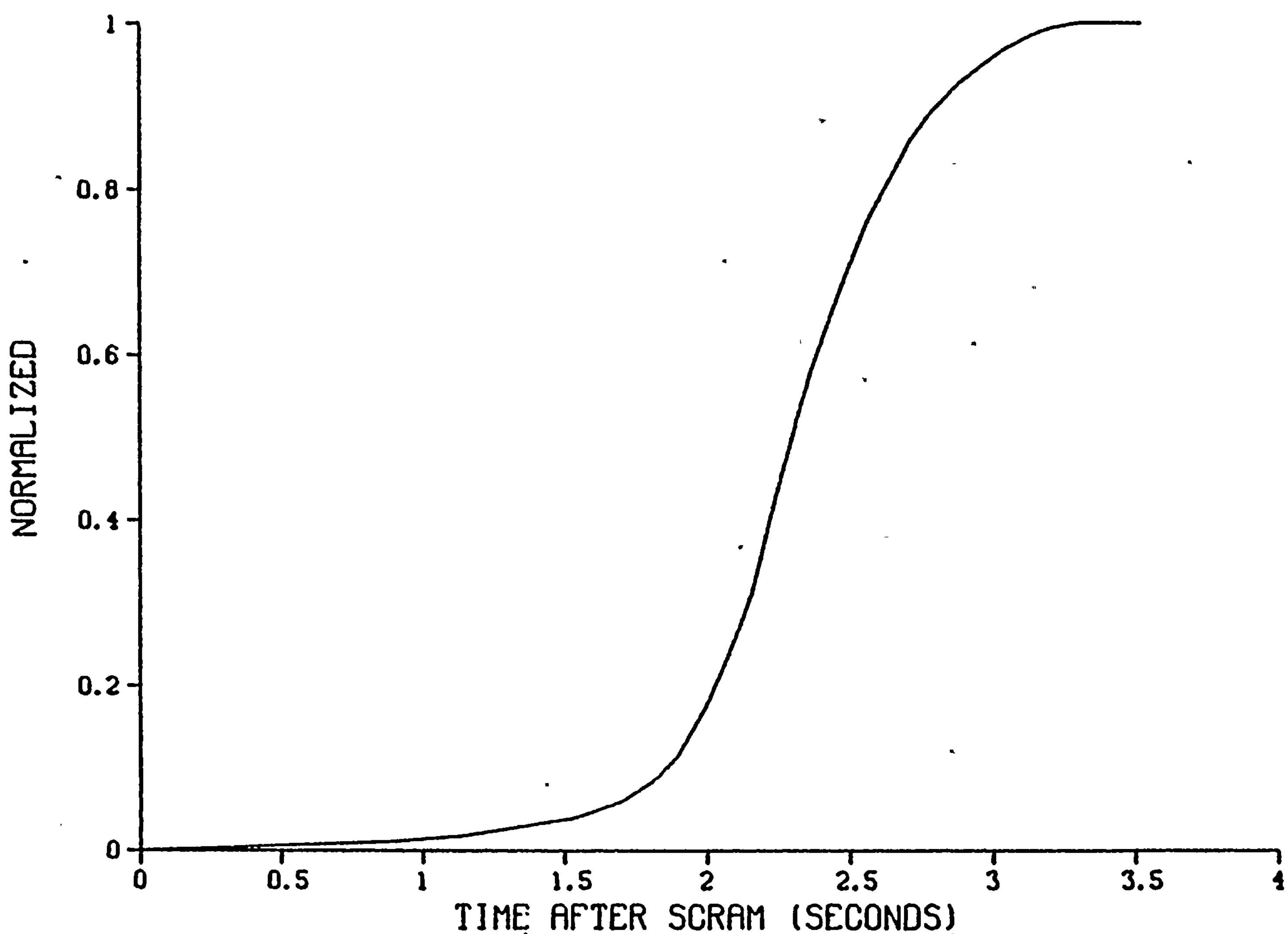

Figure 3.3 Scram Curve Used in the Transient Analysis for D.C. Cook Unit 2

4.0 TRANSIENT ANALYSIS

4.1 UNCONTROLLED ROD WITHDRAWAL

The withdrawal of control rods adds reactivity to the reactor core causing both the power level and the core heat flux to increase. Since the heat extraction from the steam generator remains relatively constant, there is an increase in primary coolant temperature. Unless terminated by manual or automatic action, this power mismatch and the resultant coolant temperature rise could eventually result in a DNB ratio of less than 1.17. While the inadvertent withdrawal of control rods is unlikely, the reactor protection system is designed to terminate such a transient while maintaining an adequate margin to DNB. Two potential causes for such an incident are: 1) operator error; and 2) a malfunction in the reactor regulating system or rod drive control system resulting in continuous withdrawal of a control rod group.

In this incident, the reactor may be tripped by an overtemperature ΔT setpoint, the high nuclear overpower setpoint, or the overpower ΔT setpoint. Additionally, the primary coolant temperature increase is limited in magnitude by the steam generator safety valve setpoint. A series of rod withdrawal simulations was performed at various reactivity insertion rates to demonstrate the adequacy of the reactor protection system for this event. Figure 4.1 summarizes the results of this study. Ample margin to DNB is demonstrated for the range of possible rod withdrawal events.

Figures 4.2 through 4.7 show plant responses for a fast rod withdrawal from full power. The reactivity insertion rate is 8.42×10^{-3} $\Delta\rho$/sec. A nuclear overpower trip (118%) occurs at 0.13 seconds. The DNB ratio

drops from an initial value of 1.878 to 1.625. Pressure increases to a maximum of 2300 psia, with core average temperature increasing 4.4°F. The event sequence for the fast rod withdrawal is given in Table 4.1.

The system responses to a slow rod withdrawal of $7.42 \times 10^{-6} \Delta\rho$/sec are depicted in Figures 4.8 through 4.14. The overtemperature ΔT trip setpoint is reached at 67.3 seconds. The minimum DNB ratio during the transient is 1.265. The event sequence for the slow rod withdrawal is given in Table 4.2.

Part power rod withdrawal analyses are discussed in Section 5.0.

4.2 LOCKED ROTOR

In the unlikely event of a seizure of a primary coolant pump, flow through the core is abruptly reduced. The reactor is tripped by the resulting low flow signal. The coolant enthalpy rises, decreasing the margin to DNB. The locked rotor transient was analyzed assuming four loop operation with instantaneous seizure of one pump from 3425 MWt. This case was shown in the reference cycle analysis to be more severe with respect to DNB penetration than a locked rotor with three loop operation. A second case was analyzed which assumed a concurrent loss of offsite power.

The transient responses for the Locked Rotor event are shown in Figures 4.15 to 4.20. The reactor is scrammed at 0.03 seconds by a low flow signal. A 1.0 second scram delay time is conservatively assumed. Core average temperature increases 11.5°F with system pressure reaching 2308 psia at 3.8 seconds. The MDNBR for the locked rotor is 1.276 at 1.9 seconds.

A second locked rotor event which additionally assumes a loss of offsite power is also simulated. The intact primary coolant pumps are assumed to initiate a coastdown concurrently with the locked rotor. Calculated MDNBR for the event is 0.698. Radiological release for this event is bounded by the LOCA accident and is within 10 CFR 100 limits.(4) This event has not been analyzed in the FSAR and is not considered to be part of the plant licensing basis. Plant response to this event with maximum pressurization assumptions is depicted in Figures 4.21 to 4.27. Pressurizer pressure control systems have been assumed inoperable in order to maximize pressure response. The MDNBR has been evaluated with minimum pressure response assumptions. The event sequences for the locked rotor cases are given in Table 4.3.

4.3 LOSS OF EXTERNAL ELECTRICAL LOAD

This simulation considers plant behavior upon a trip of the turbine-generator without a direct reactor trip. The event is simulated to assess the adequacy of the pressurizer safety valve capacity to maintain reactor coolant system pressure below the ASME code limit of 110% of design pressure (2750 psia). Transient responses are evaluated from 3425 MWt for the most severe pressurization accident: loss of load at beginning-of-cycle (BOC) with a positive moderator coefficient and no automatic reactor control.

Figures 4.28 to 4.34 depict the plant responses following a loss of load from full power. A high pressure trip occurs at 6.98 seconds, with peak pressurizer pressure reaching 2526.1 psia. The first set of steam line safety valves opens at 11.9 seconds, relieving 45% of the steam flow. The setpoint of the second set of safety valves is reached at 18 seconds. The average

primary coolant temperature increases 24.5°F above the nominal value. The MDNBR does not decrease below its initial value. The event sequence for the loss of load is given in Table 4.4.

4.4 DECREASED FEEDWATER HEATING

Failure of bleed steam to any of the six pairs of feedwater heaters could result in a 75 Btu/lb decrease in feedwater enthalpy. The event is simulated by imposing a 15 second feedwater enthalpy ramp at a rate of -5 Btu/lb-second.

Results of the decreased feedwater heating event are given in Figures 4.35 through 4.43. A new steady state is established relatively early in the transient. The MDNBR of 1.679 characterizes this steady state. The overtemperature ΔT reactor trip precludes penetration of the XNB critical heat flux correlation safety limit during transients such as the decreased feedwater heating event which are characterized by slow excursions in core power, coolant temperature, and pressure.

A second case was simulated assuming automatic rod control. A bounding EOC D-bank worth consistent with power dependent insertion limits was employed (-1.2% $\Delta\rho$). Primary side response is depicted in Figures 4.44 through 4.49. Figures 4.50 and 4.51 demonstrate RCCA control action. The MDNBR for the event is 1.627 and occurs at 258 seconds. Secondary system response is similar to that depicted in Figures 4.41 through 4.43. Event sequences for these events are given in Table 4.5.

4.5 EXCESSIVE LOAD INCREASE INCIDENT

Excessive load incidents may be initiated by sudden opening of the turbine control valves, steam dump valves, and/or the steam bypass to

condenser valve. This results in rapid increase in steam flow which causes cooldown of the primary system. Automatic Rod Control action or a large negative (EOC) moderator coefficient can result in a power increase. Protection against damage to the reactor core as a consequence of an excessive load increase is provided by the high nuclear flux, low steam generator pressure, and overtemperature ΔT setpoints.

A rapid 20% load increase is simulated at end-of-cycle conditions. To minimize the calculated MDNBR, the pressurizer heaters are assumed inoperable.

System responses to the 20% load increase are shown in Figures 4.52 through 4.60. The power increase driven by moderator cooldown continues until the high nuclear overpower trip setpoint (118% of rated power) is reached at 56.5 seconds. At the time of trip, significant primary system depressurization has further reduced available thermal margin. The decreasing primary coolant temperature mitigates the thermal margin decay, resulting in an MDNBR for the event of 1.532 shortly after reactor trip.

The case with automatic rod control is shown in Figures 4.61 through 4.66. Automatic Rod Control action is demonstrated in Figures 4.67 and 4.68. Control action resulting largely from the temperature deviation channel results in a more rapid power increase than observed in the uncontrolled case. To maximize power response, the temperature control program employed a linear gain between 0 and 120% of rated turbine demand. The reactor trips on high nuclear flux at 21.2 seconds. The MDNBR of 1.569 occurs shortly thereafter. Event sequences for the load increase transients are given in Table 4.6.

Table 4.1 Event Sequence for Fast Rod Withdrawal

Event	Time (seconds)
Uncontrolled RCCA Bank Withdrawal begins	0.0
High Neutron Flux Setpoint reached	0.13
Scram Results in Rod Motion	0.63
Minimum DNBR occurs	1.55

Table 4.2 Event Sequence for Slow Rod Withdrawal

Event	Time (seconds)
Uncontrolled RCCA Bank Withdrawal begins	0.0
Overtemperature ΔT Setpoint reached	67.3
Scram Results in Rod Motion	73.3
Minimum DNBR occurs	74.0

Table 4.3 Event Sequence for Locked Rotor

Event	Time (seconds)
CASE 1: Locked Rotor with Offsite Power Available	
Single Primary Coolant Pump seizes	0.
Loop Low Flow Trip Setpoint reached	0.03
Scram Results in Rod Motion	1.03
Minimum DNBR occurs	1.9
Peak RCS Pressure reached	3.2
CASE 2: Locked Rotor with Concurrent Loss of Offsite Power	
Single Primary Coolant Pump seizes	0.
Loop Low Flow Trip Setpoint reached	0.03
Scram Results in Rod Motion	1.03
Minimum DNBR occurs	2.4
Peak RCS Pressure reached	4.0

Table 4.4 Event Sequence for Loss of External Load

Event	Time (seconds)
Loss of Load	0.
High Pressurized Pressure Setpoint reached	6.98
Scram Results in Rod Motion	8.98
Peak Pressure reached	18.1

Table 4.5 Event Sequence for Decreased Feedwater Heating

Event	Time (seconds)
CASE 1: Uncontrolled	
Feedwater Enthalpy Begins to Decrease from the Steady State Value at 5. Btu/lb/second	0.
Feedwater Enthalpy reaches Minimum Value	15.0
Minimum DNBR reached	260.
(NOTE: Reactor scram does not occur.)	
CASE 2: Automatic Rod Control	
Feedwater Enthalpy Begins to Decrease from the Steady State Value at 5. Btu/lb/second	0.
Feedwater Enthalpy reached Minimum Value	15.0
Minimum DNBR reached	258.

Table 4.6 Event Sequence for Excessive Load Increase

Event	Time (seconds)
CASE 1: Uncontrolled	
20% Increase in Load Demand reached	10.
High Neutron Flux Trip Setpoint reached	56.0
Scram Results in Rod Motion	56.5
Minimum DNBR occurs	57.0
CASE 2: Automatic Rod Control	
20% Increase in Load Demand reached	1.0
High Neutron Flux Trip Setpoint reached	21.2
Scram Results in Rod Motion	21.7
Minimum DNBR occurs	21.5

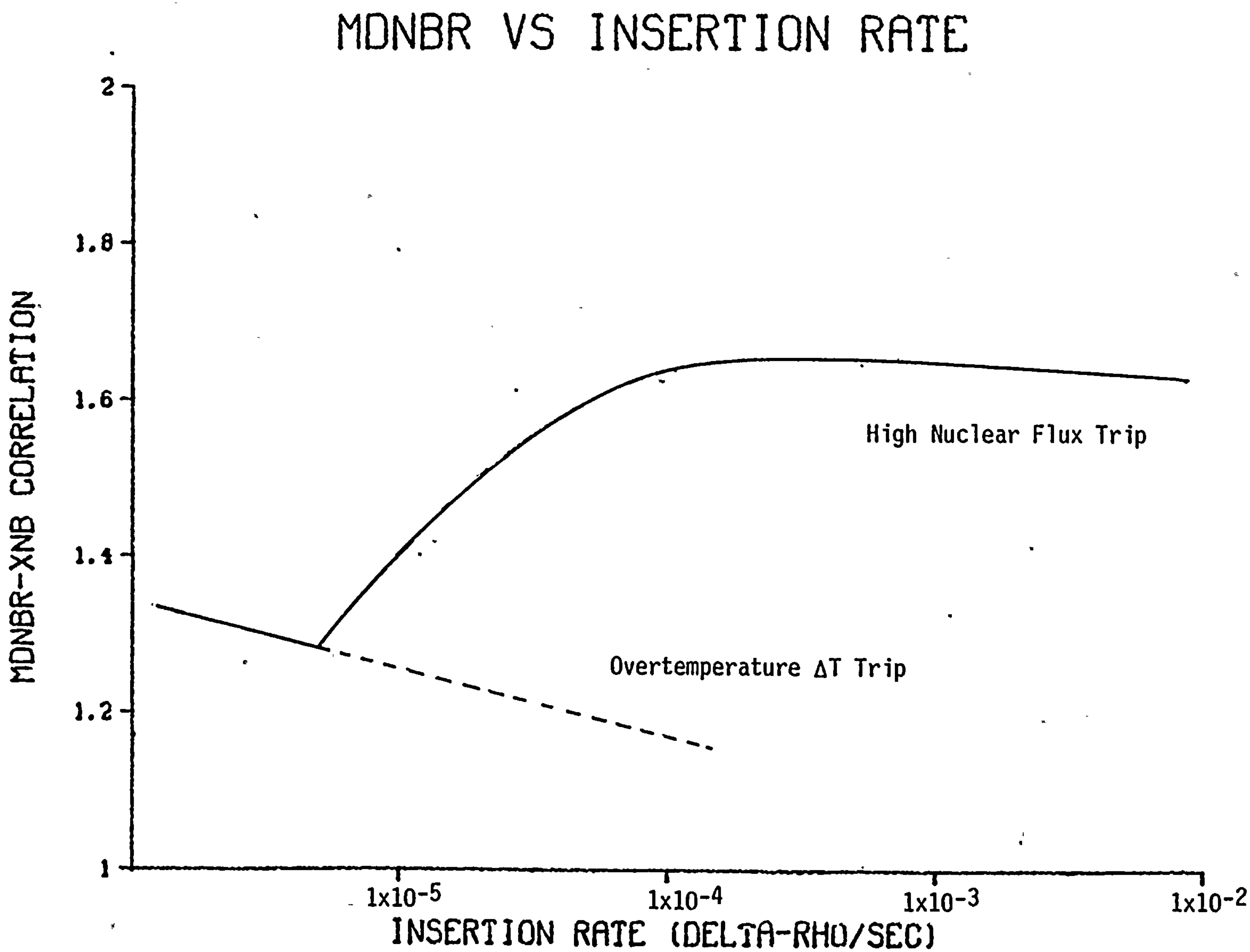

Figure 4.1 MDNBR vs. Reactivity Insertion Rate for Rod Withdrawal Transients

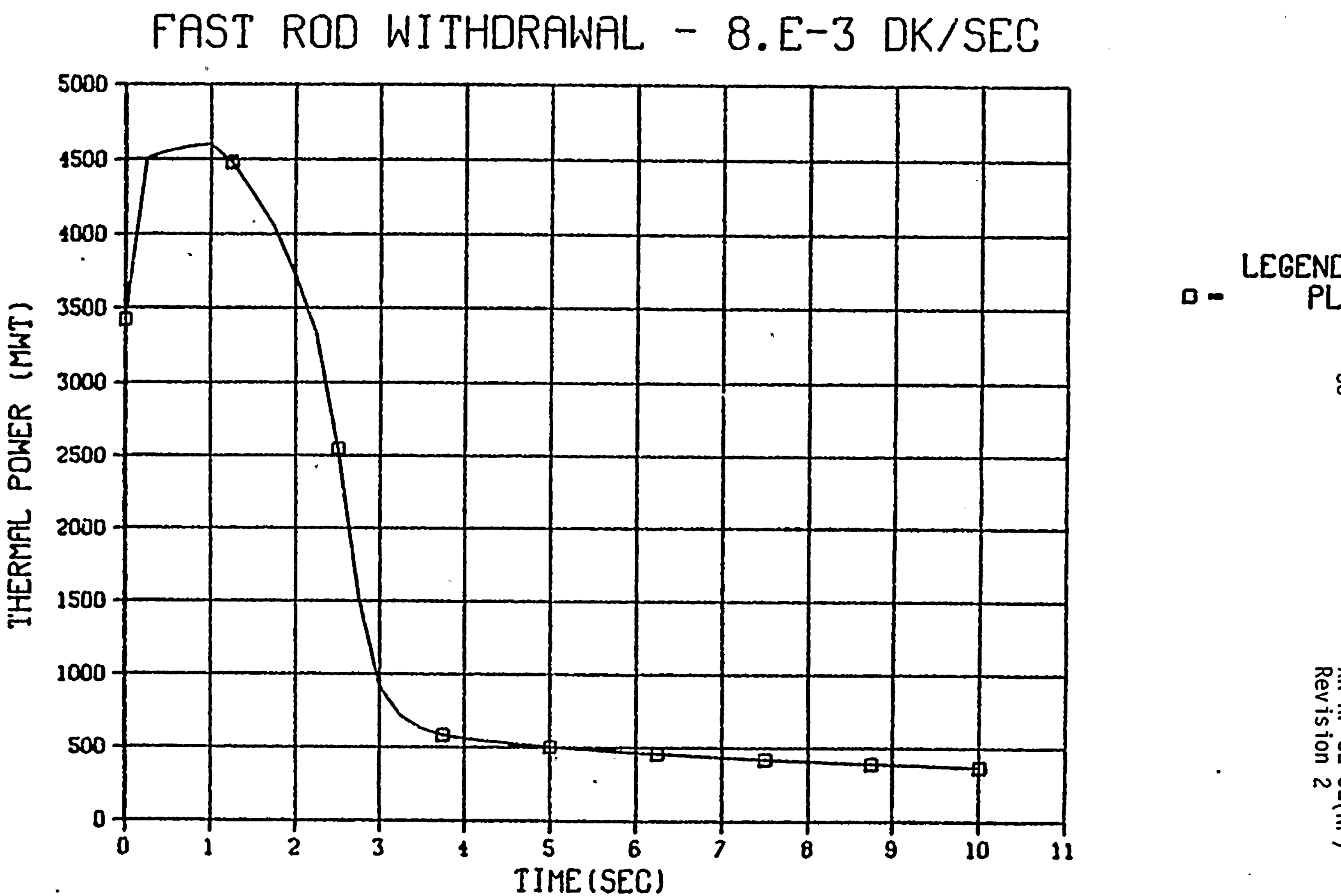

Figure 4.2 Thermal Power for Fast Rod Withdrawal

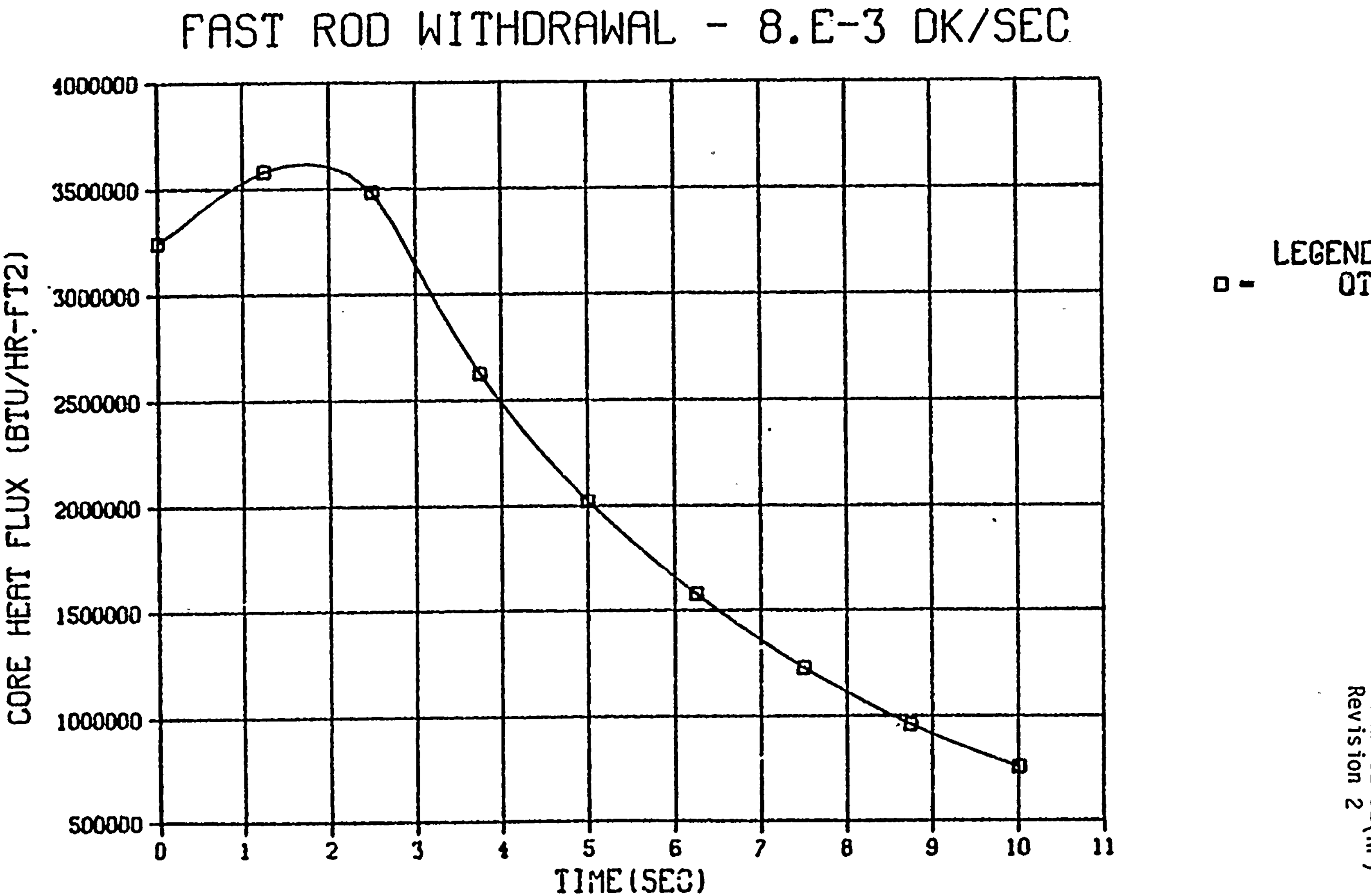

Figure 4.3 Core Heat Flux for Fast Rod Withdrawal

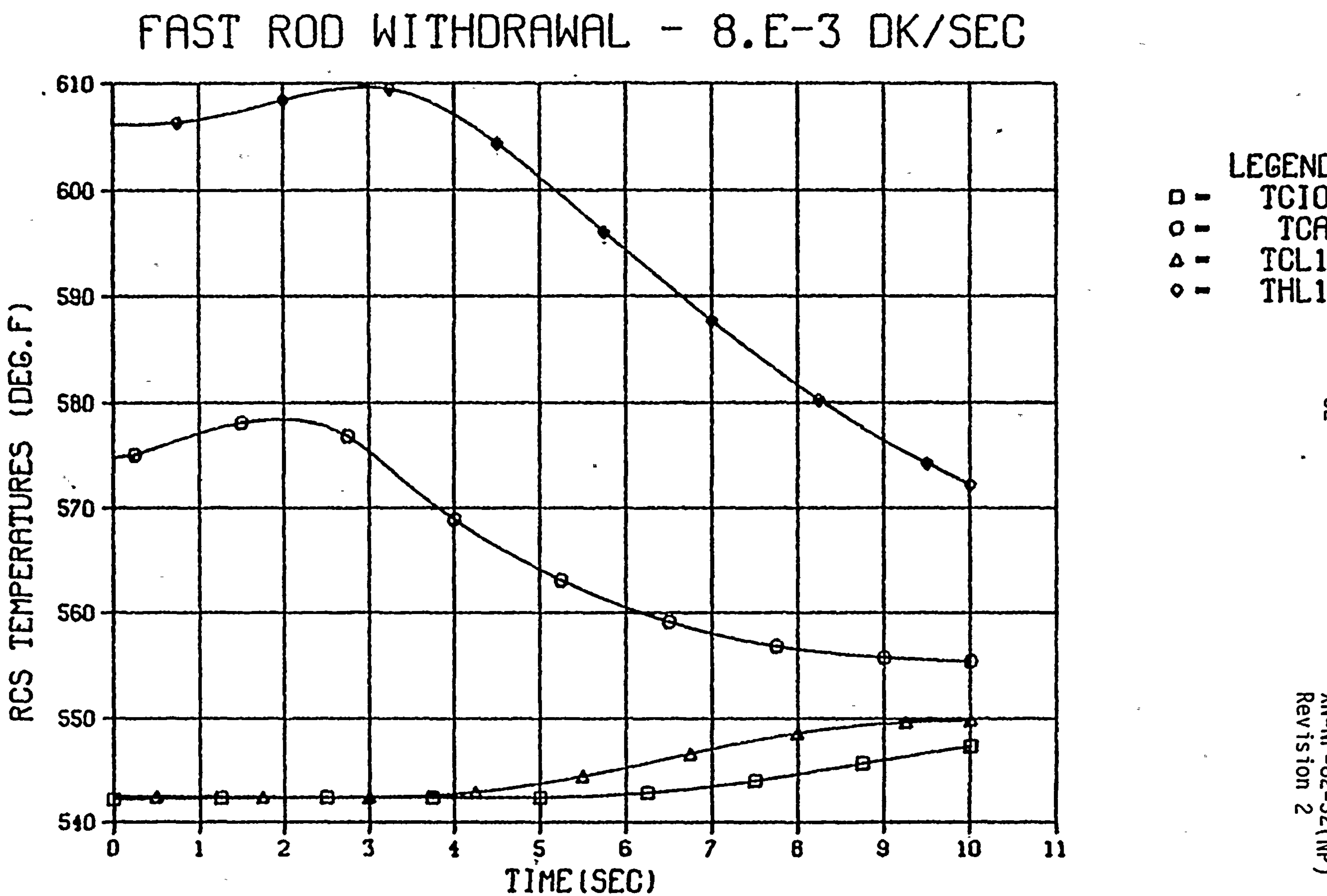

Figure 4.4 RCS Temperatures for Fast Rod Withdrawal - Hot Leg, Core Average, Cold Leg, and Core Inlet Temperature

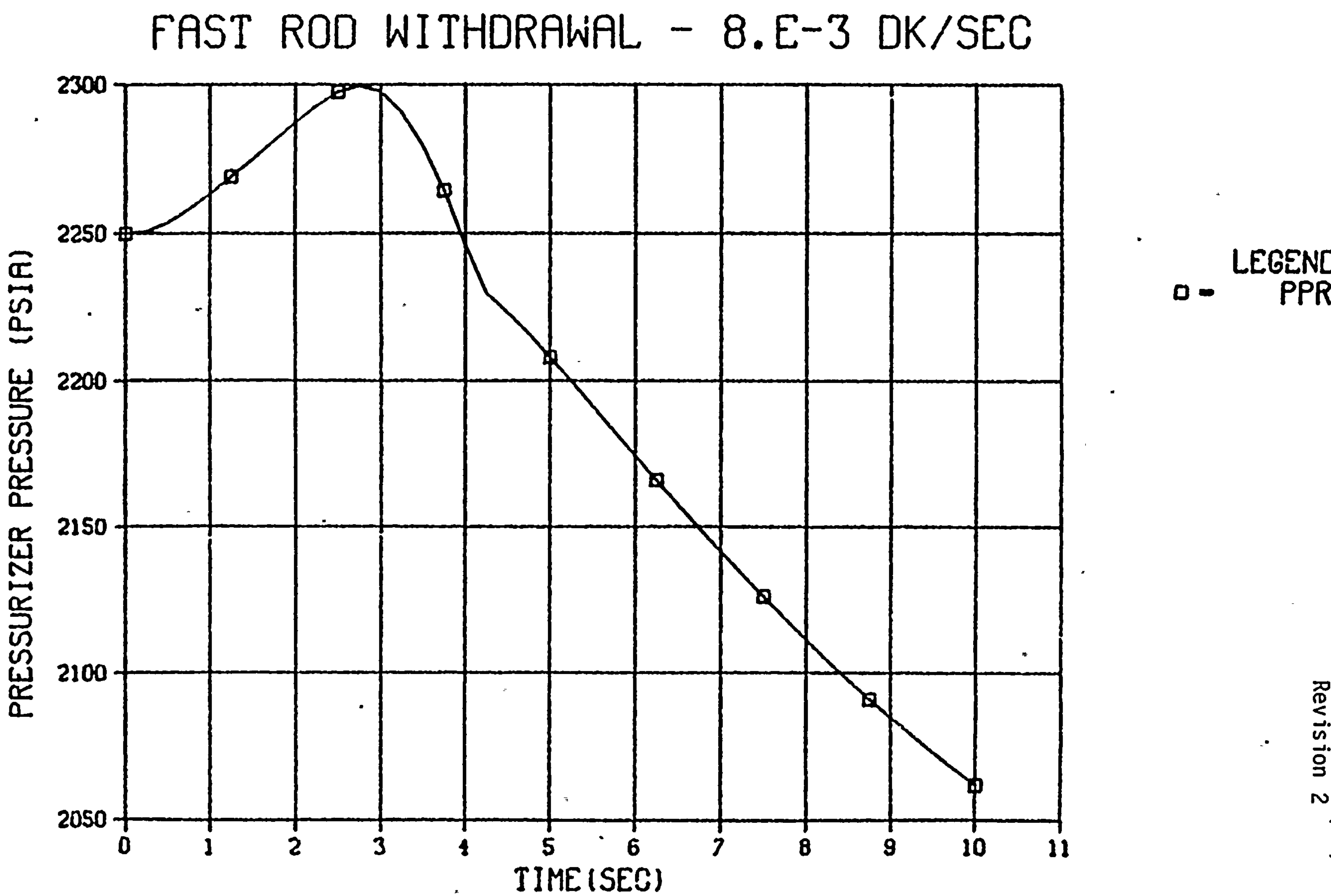

Figure 4.5 Pressurizer Pressure for Fast Rod Withdrawal

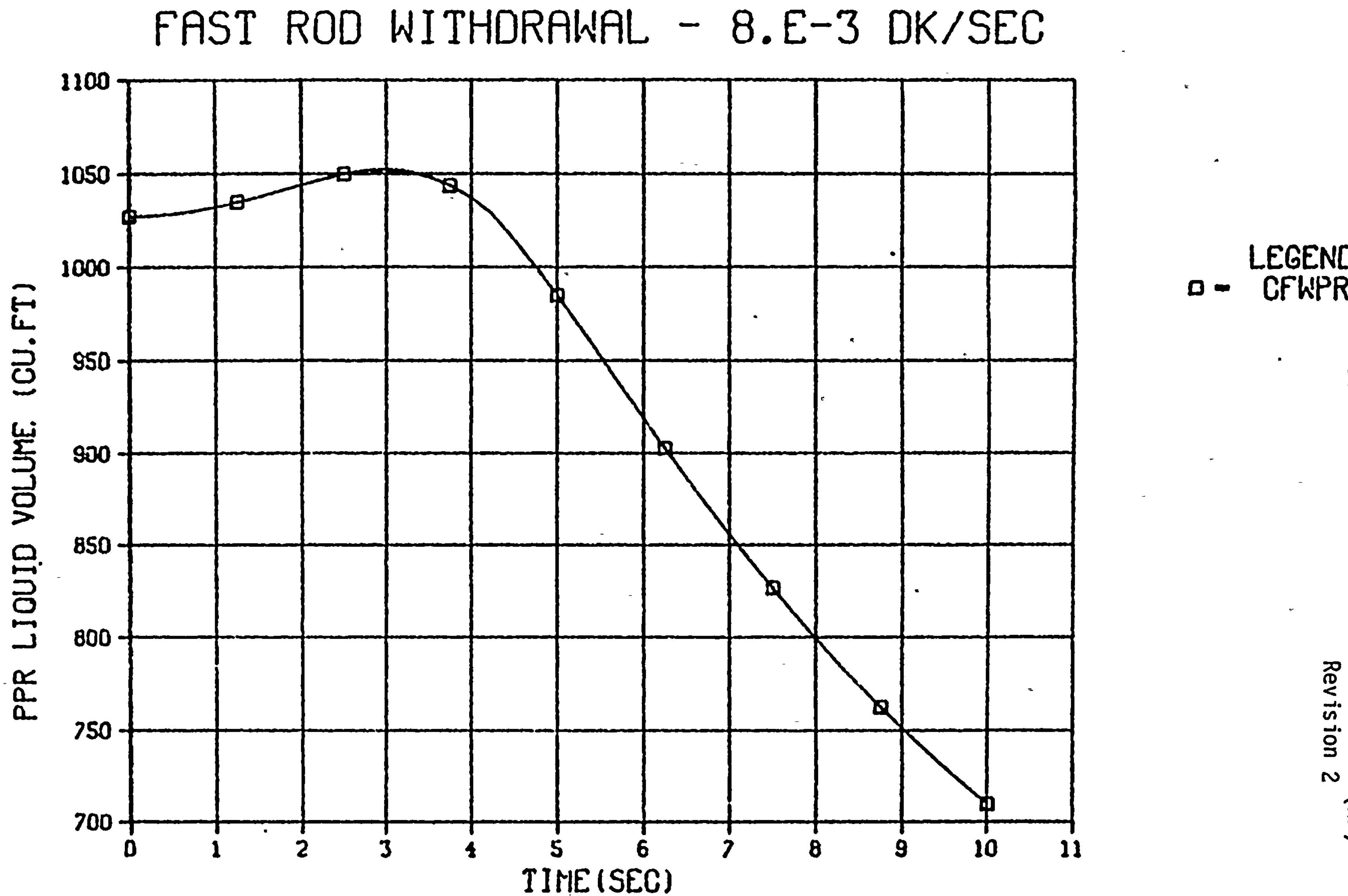

Figure 4.6 Pressurizer Liquid Volume for Fast Rod Withdrawal

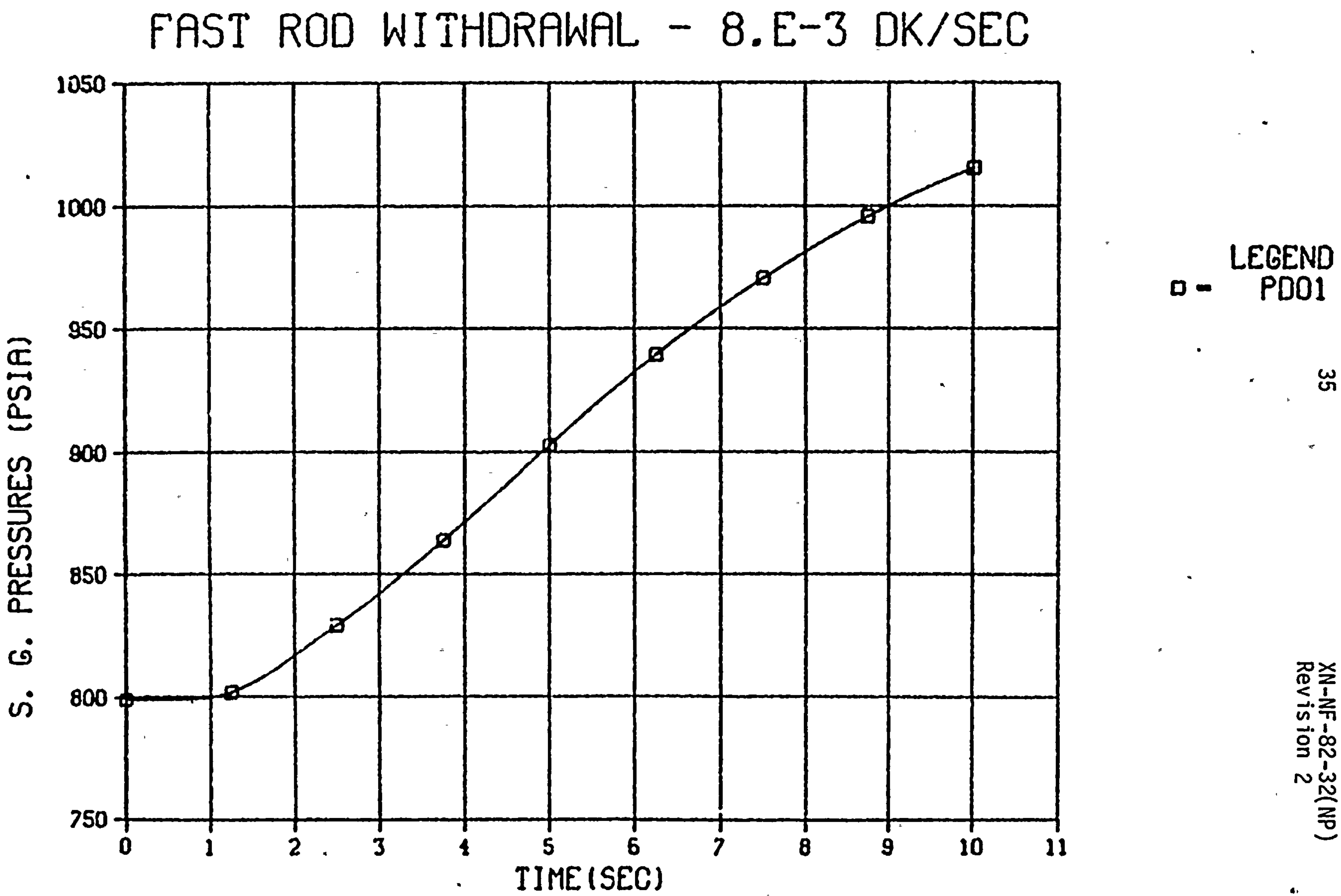

Figure 4.7 Steam Generator Pressure for Fast Rod Withdrawal

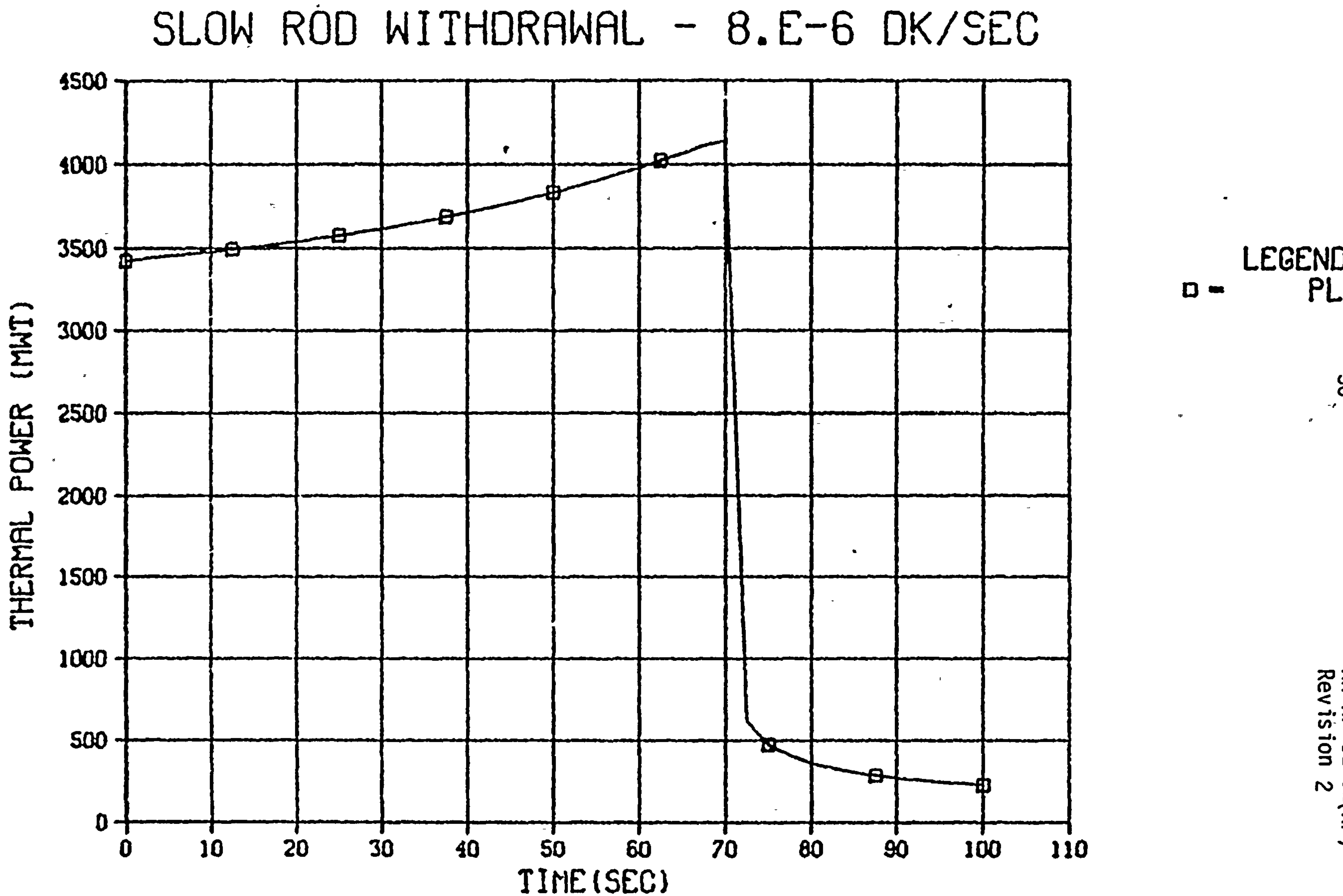

Figure 4.8 Thermal Power for Slow Rod Withdrawal

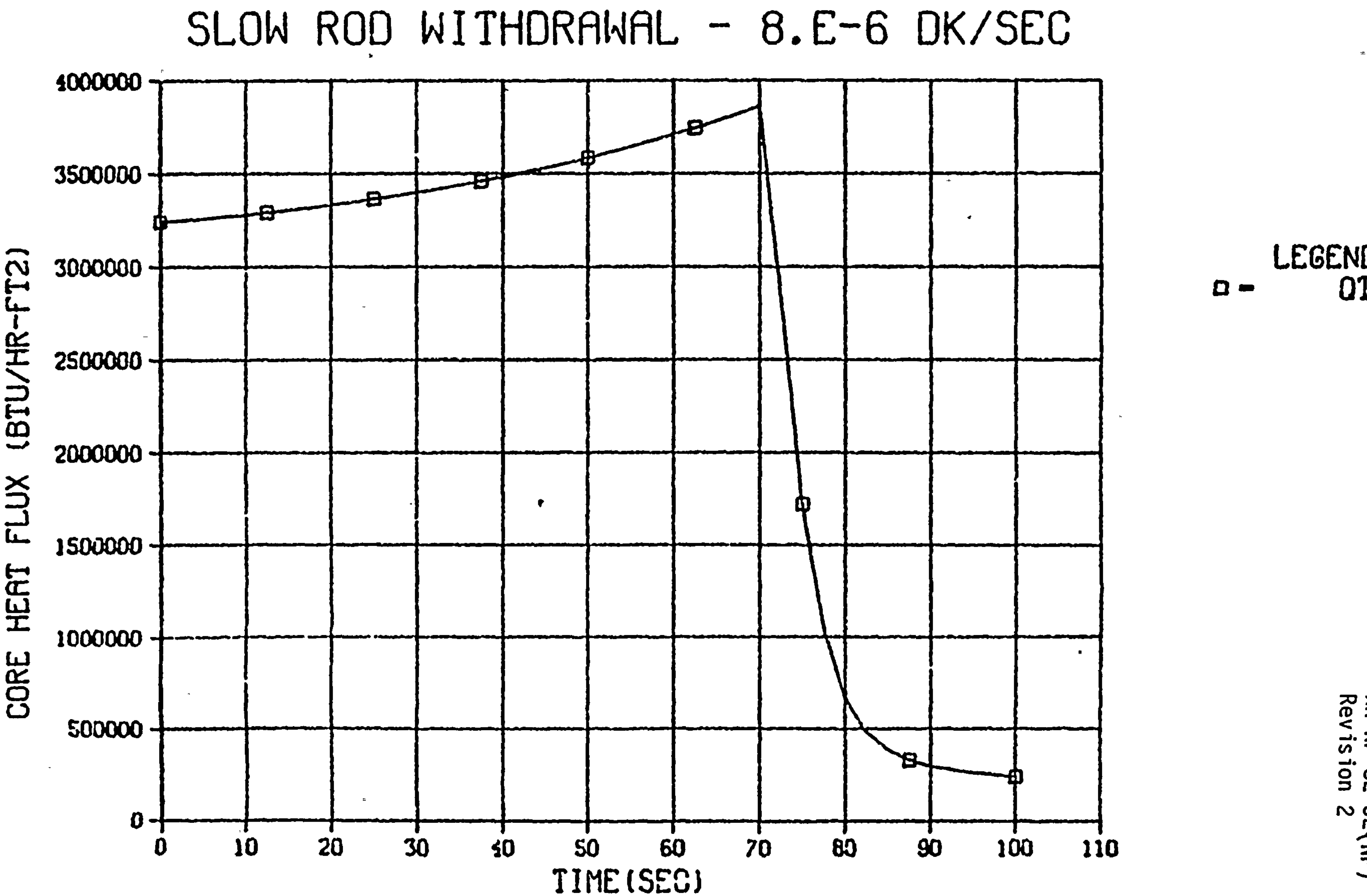

Figure 4.9 Core Heat Flux for Slow Rod Withdrawal

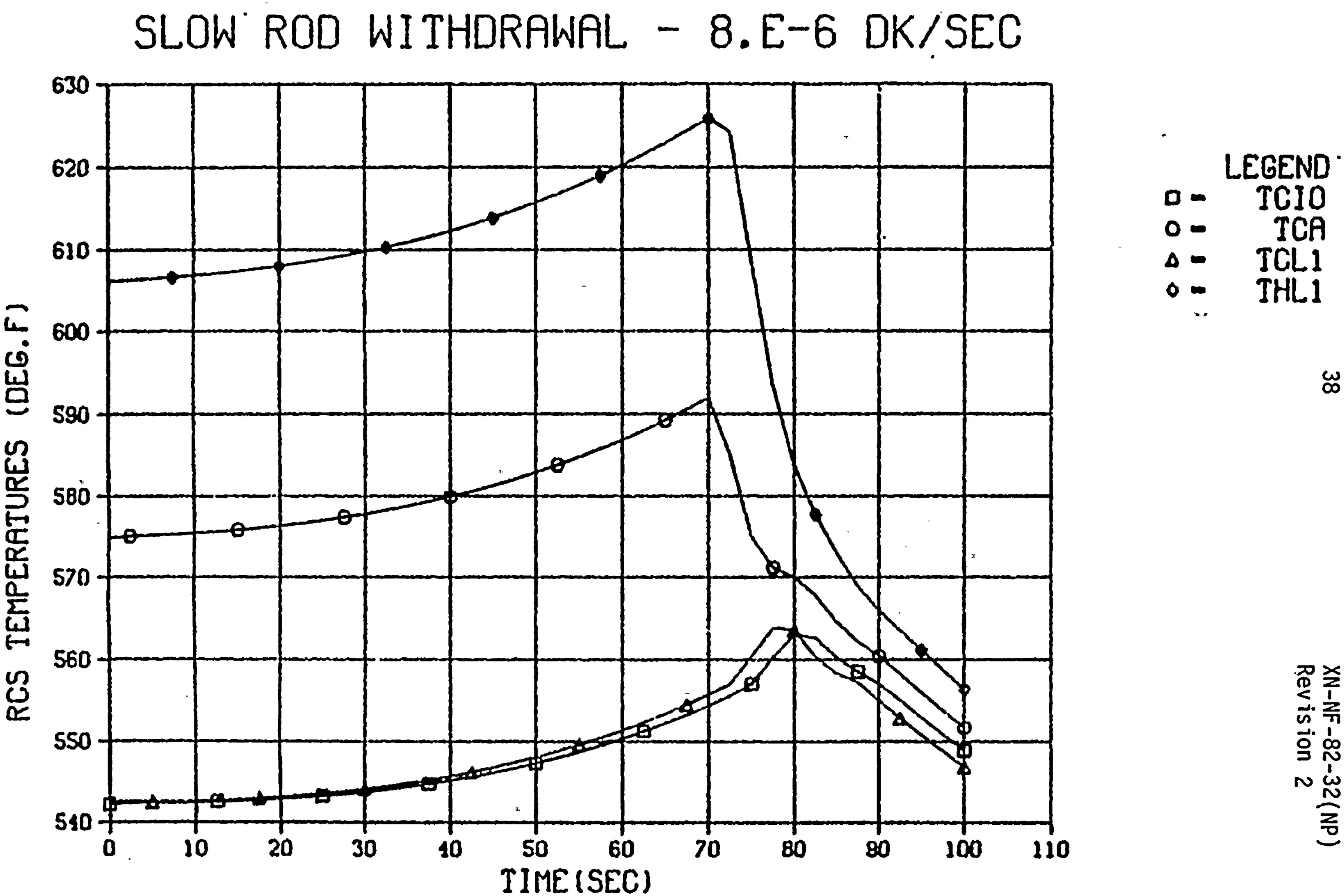

Figure 4.10 RCS Temperatures for Slow Rod Withdrawal - Hot Leg, Core Average, Cold Leg and Core Inlet Temperatures

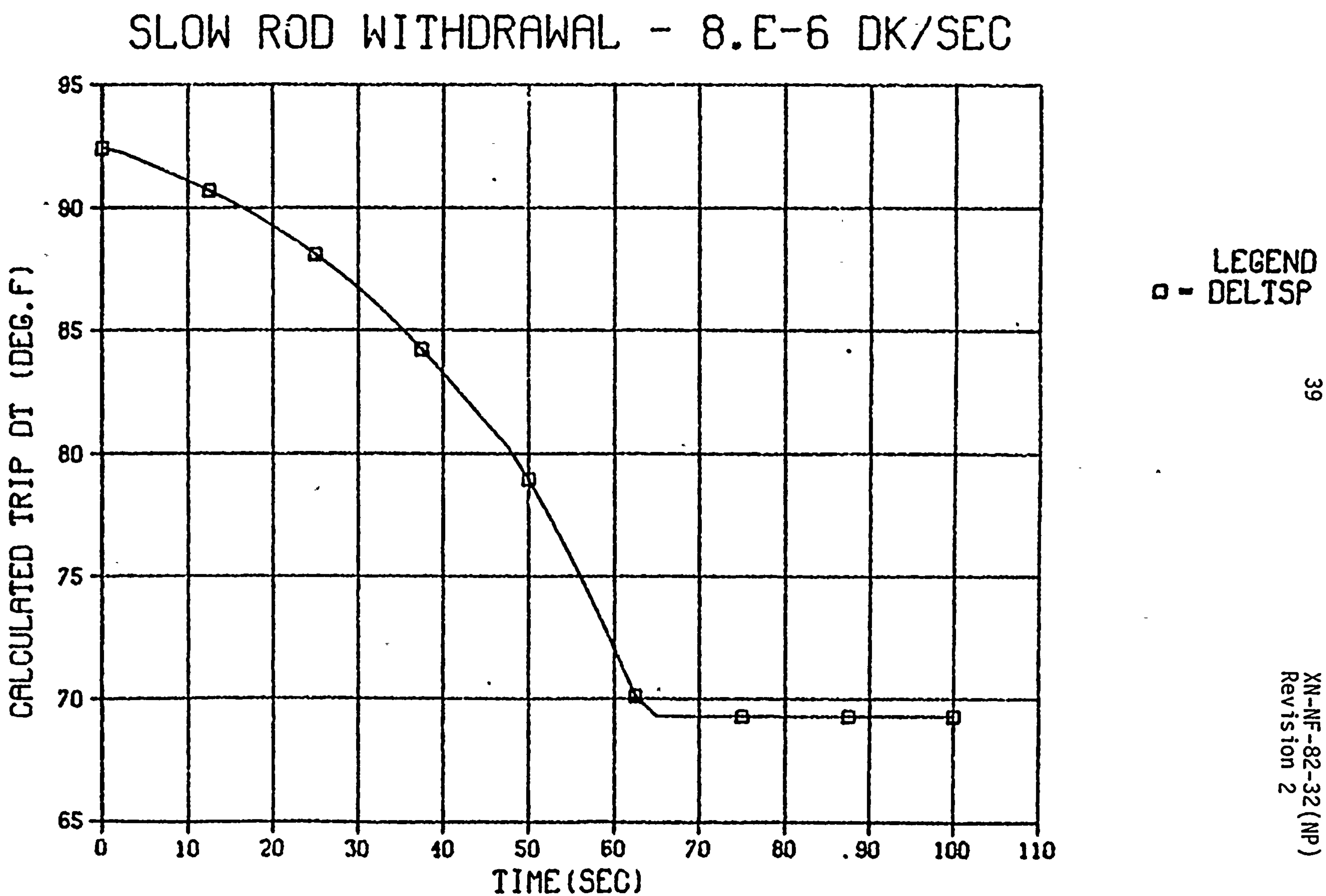

Figure 4.11 Calculated Overtemperature ΔT Setpoint for Slow Rod Withdrawal

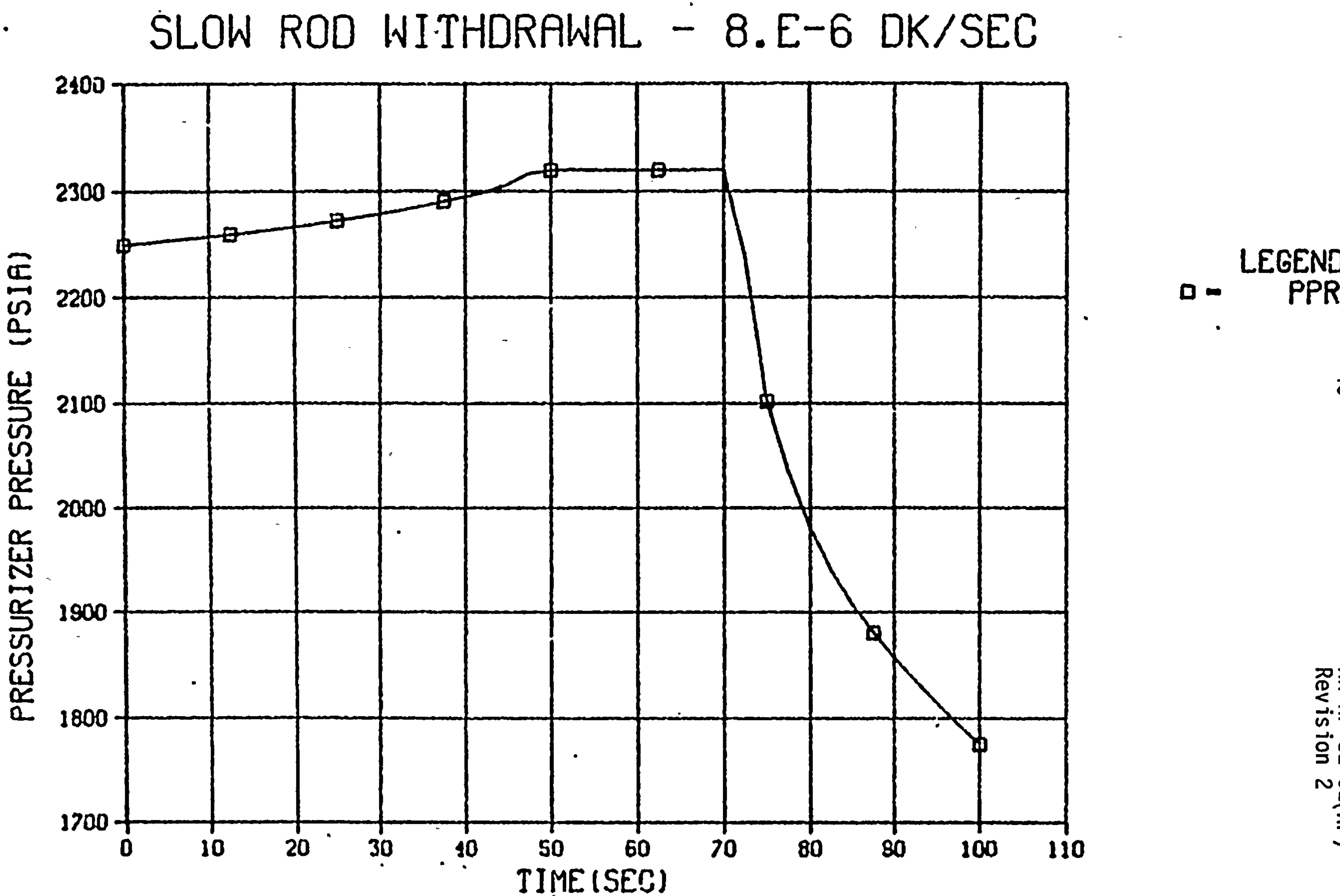

Figure 4.12 Pressurizer Pressure for Slow Rod Withdrawal

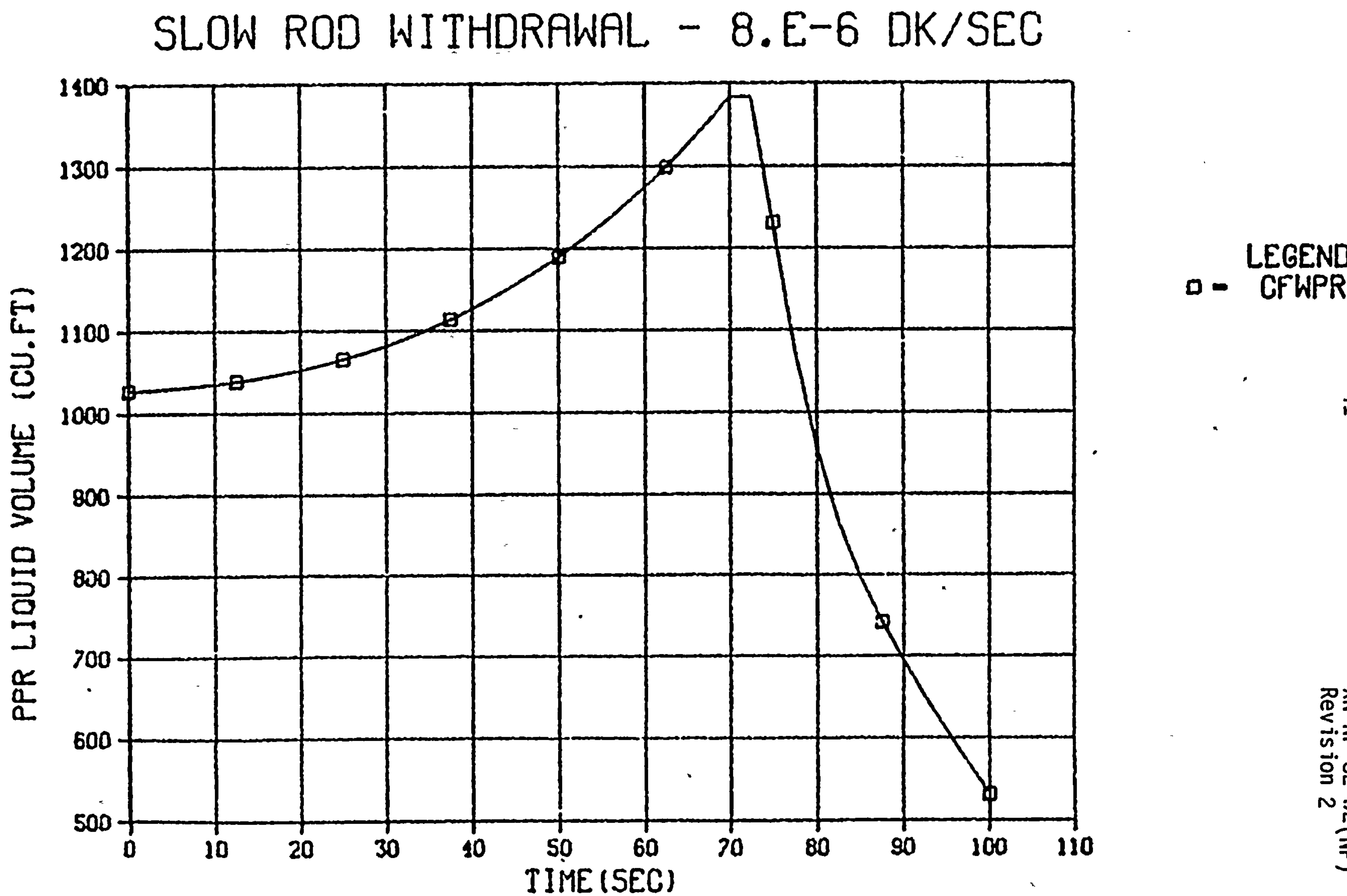

Figure 4.13 Pressurizer Liquid Volume for Slow Rod Withdrawal

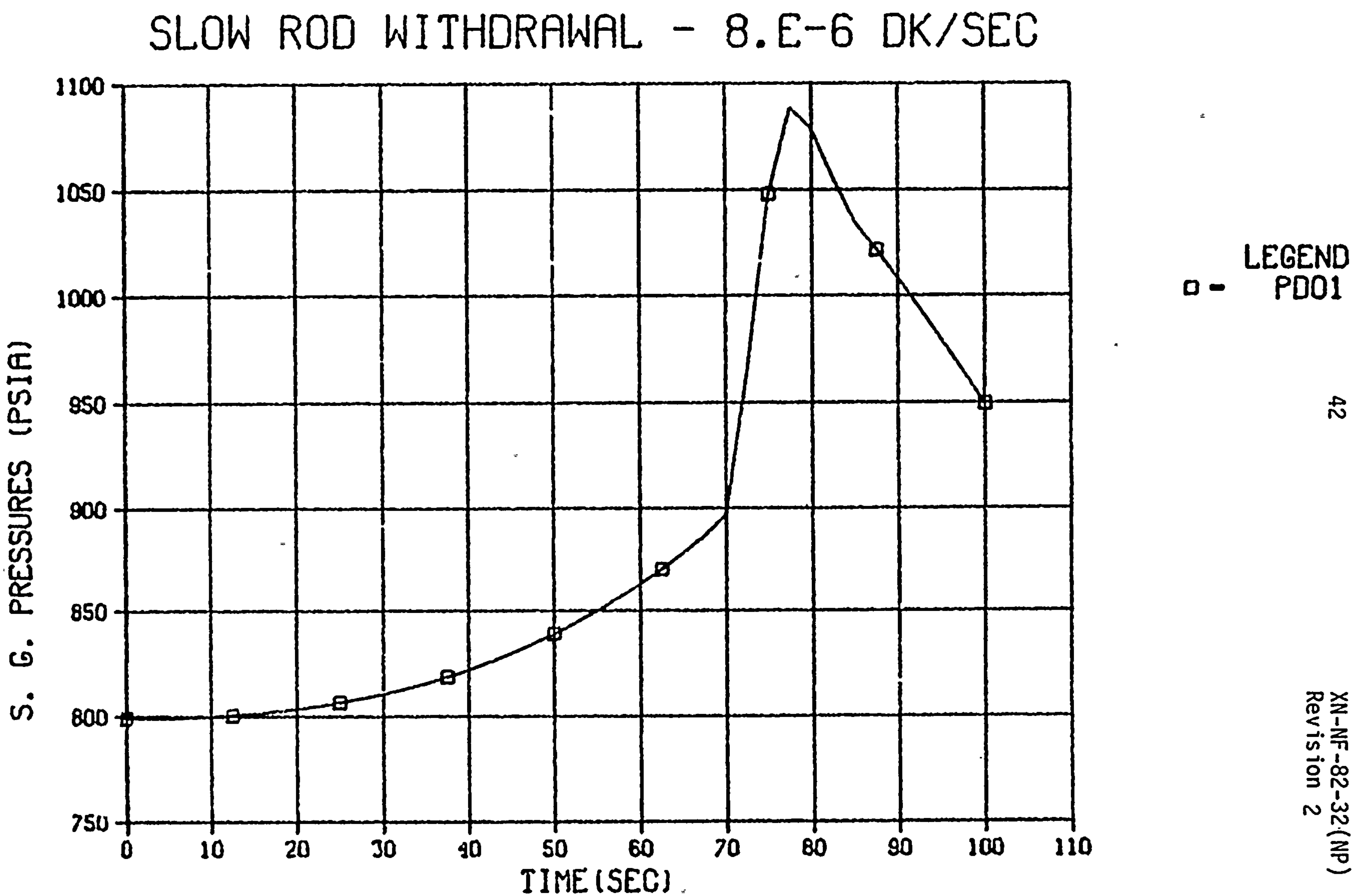

Figure 4.14 Steam Generator Pressure for Slow Rod Withdrawal

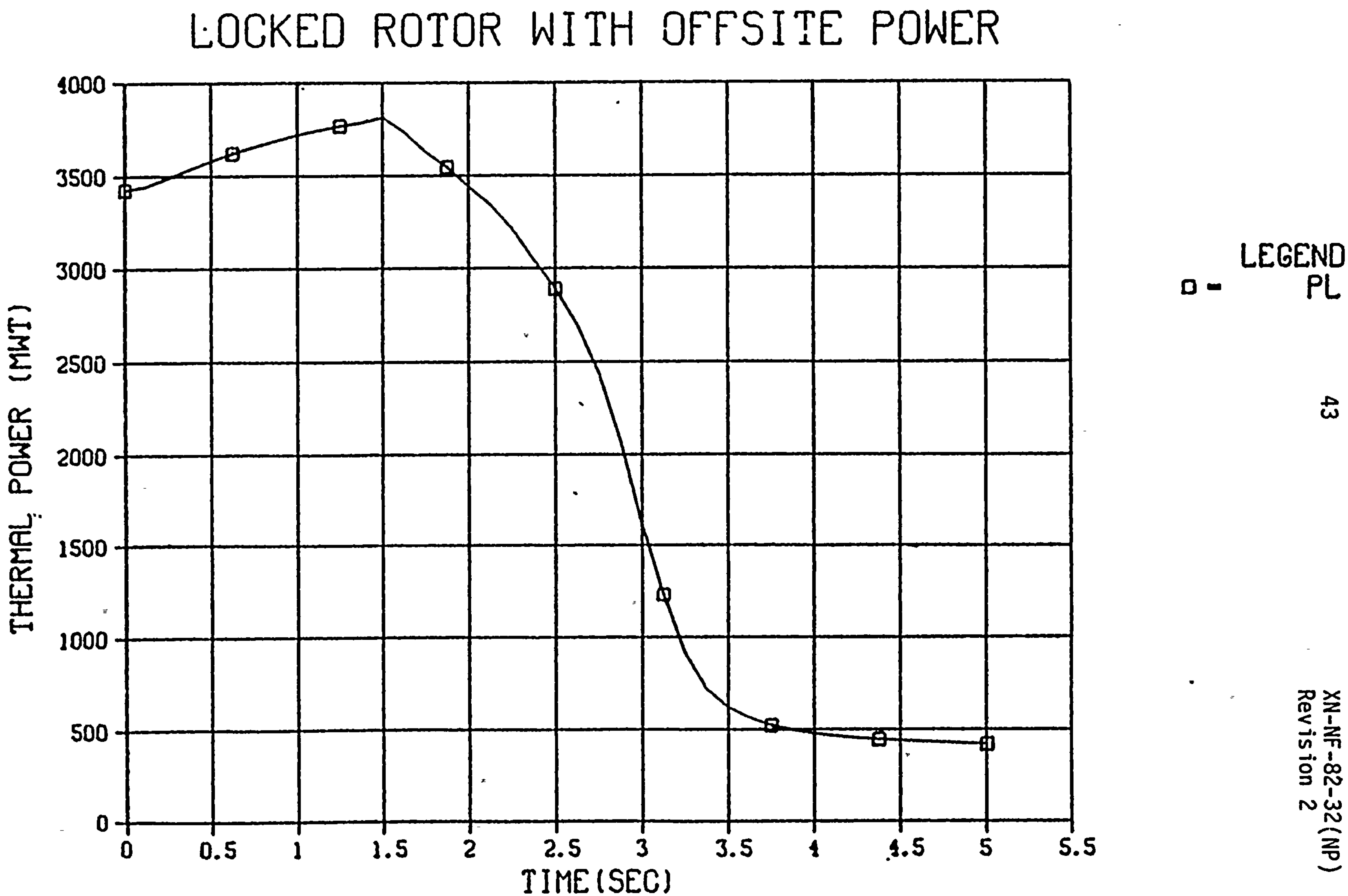

Figure 4.15 Thermal Power for Locked Rotor with Offsite Power

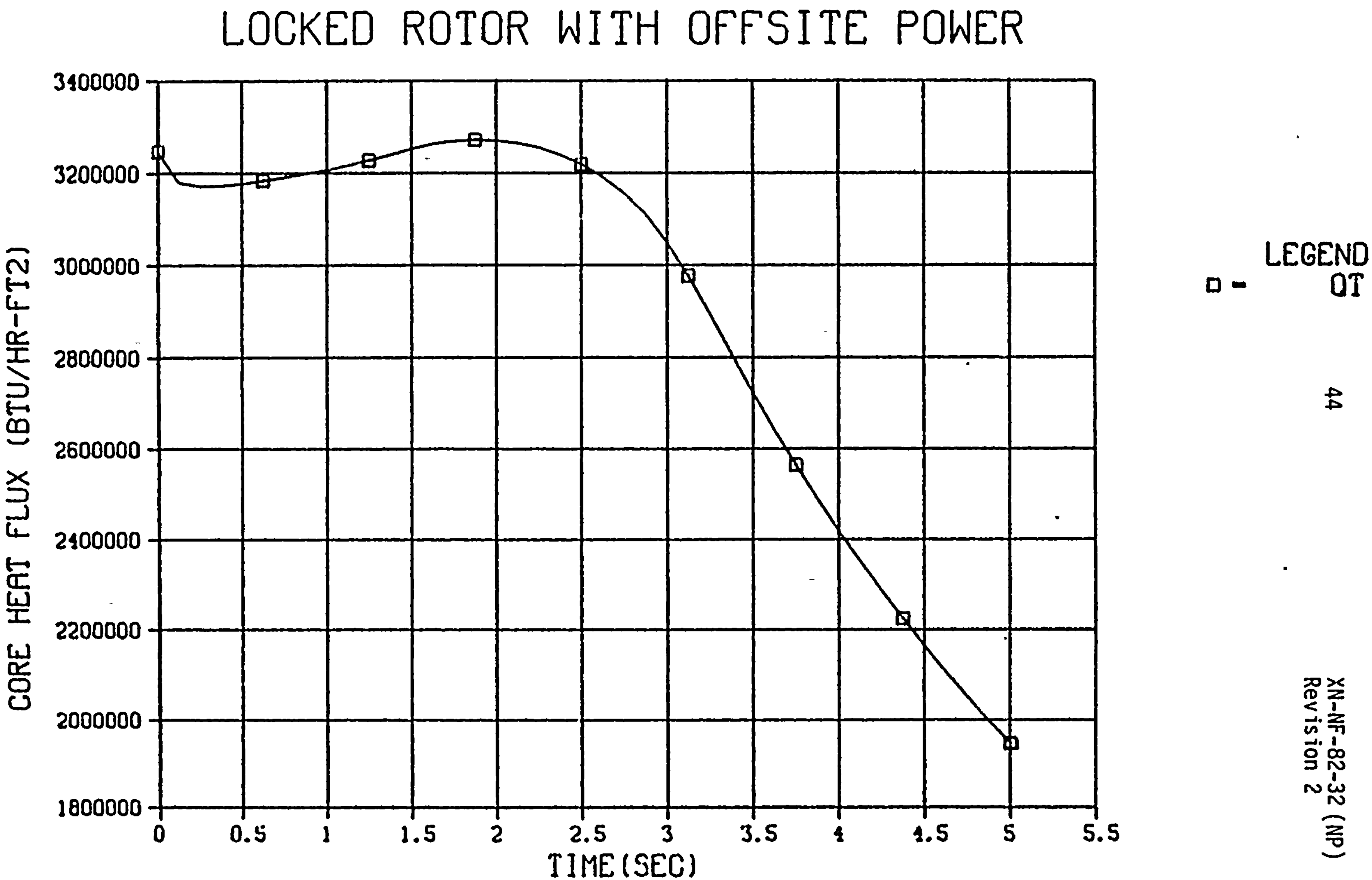

Figure 4.16 Core Heat Flux for Locked Rotor with Offsite Power

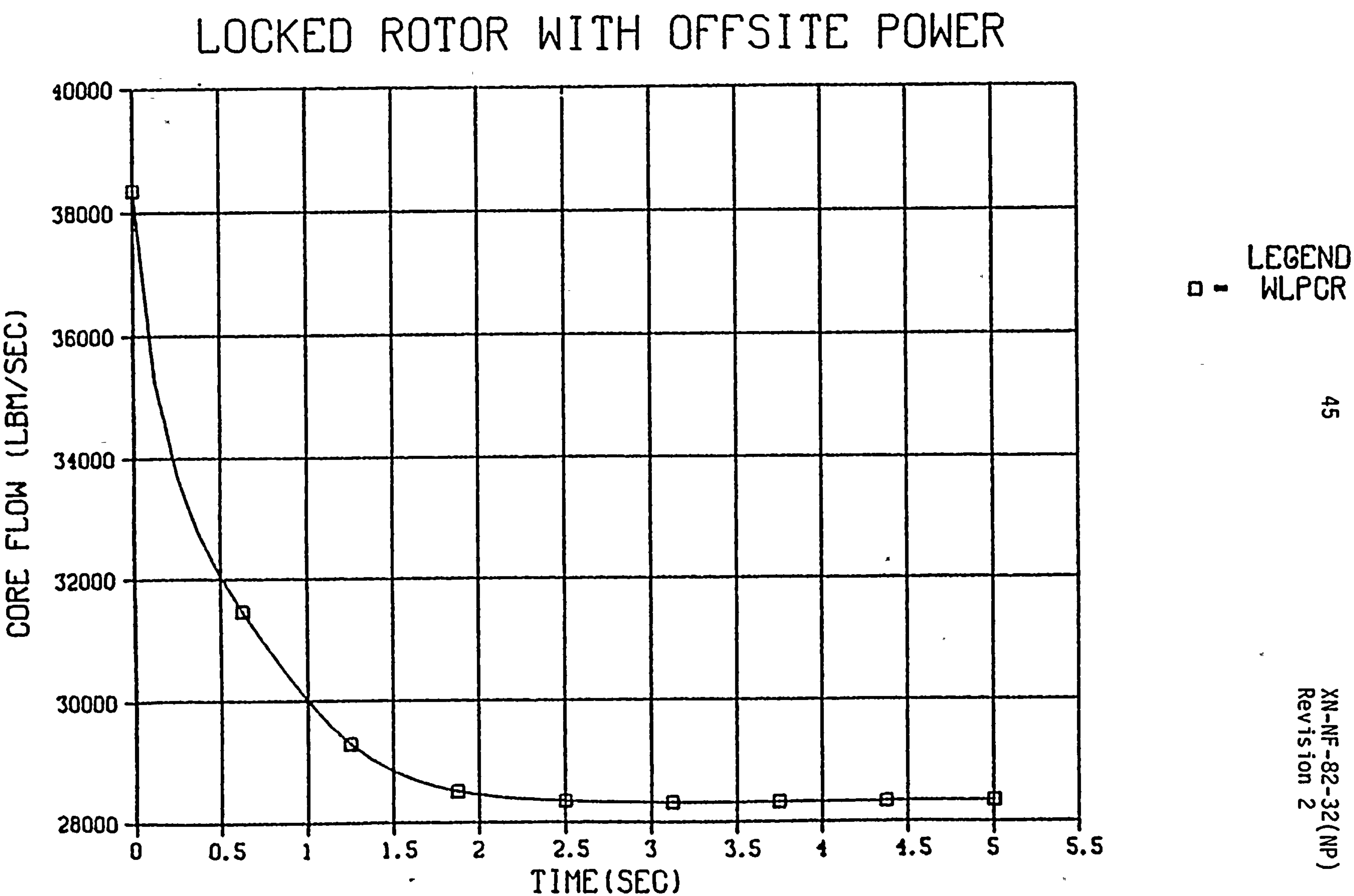

Figure 4.17 Core Flow for Locked Rotor with Offsite Power

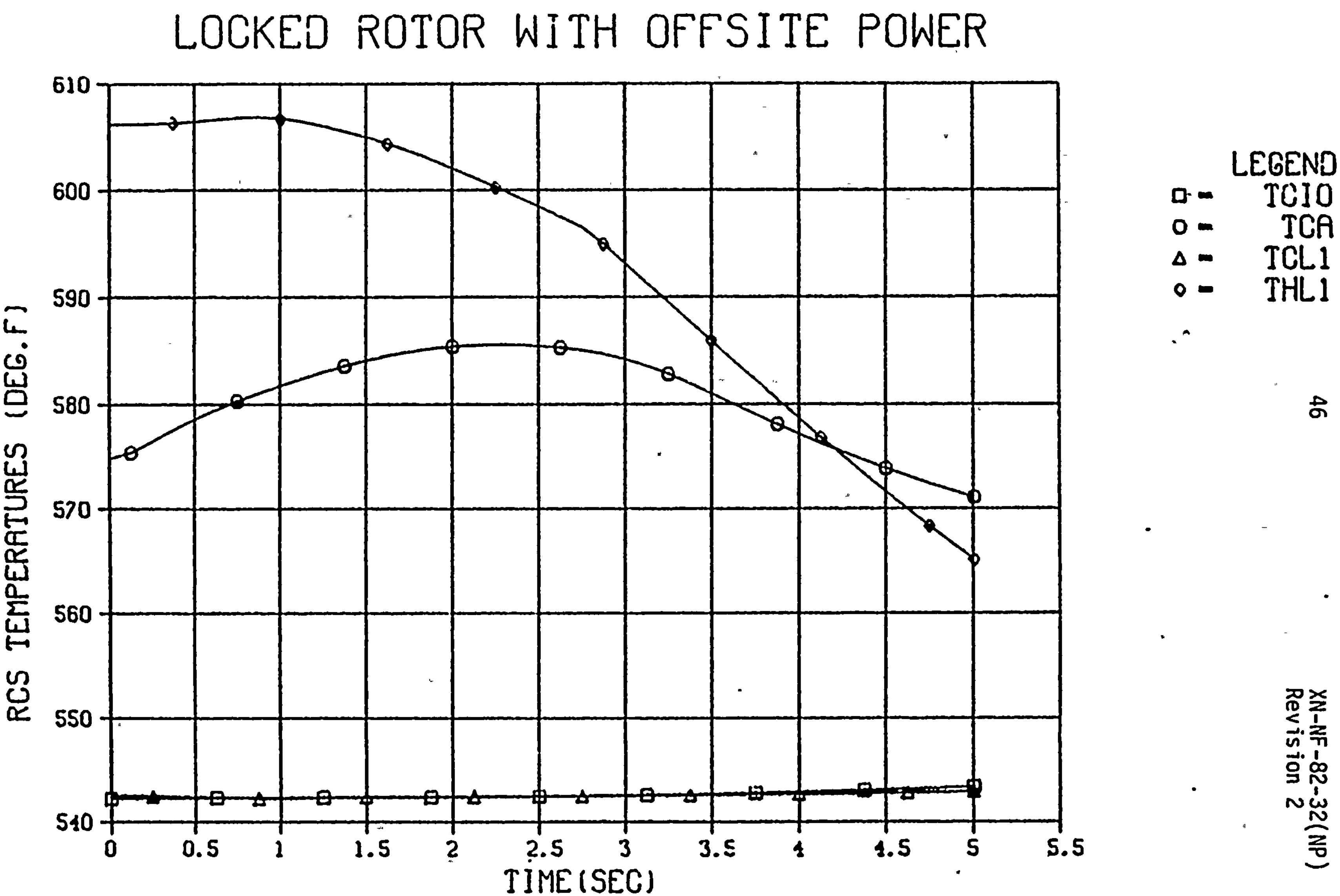

Figure 4.18 RCS Temperature for Locked Rotor with Offsite Power - Hot Leg, Core Average, Cold Leg, and Core Inlet Temperatures

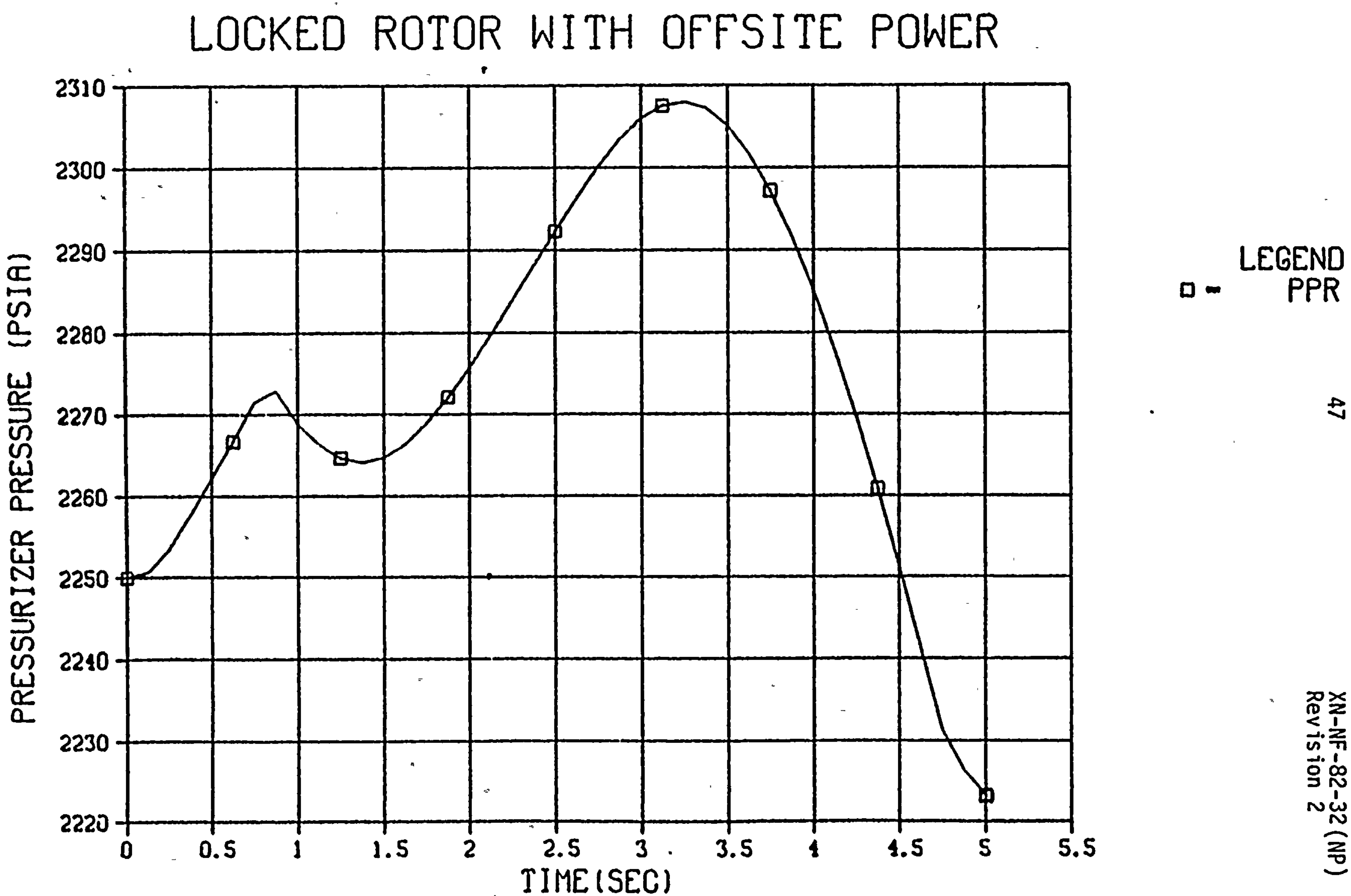

Figure 4.19 Pressurizer Pressure for Locked Rotor with Offsite Power

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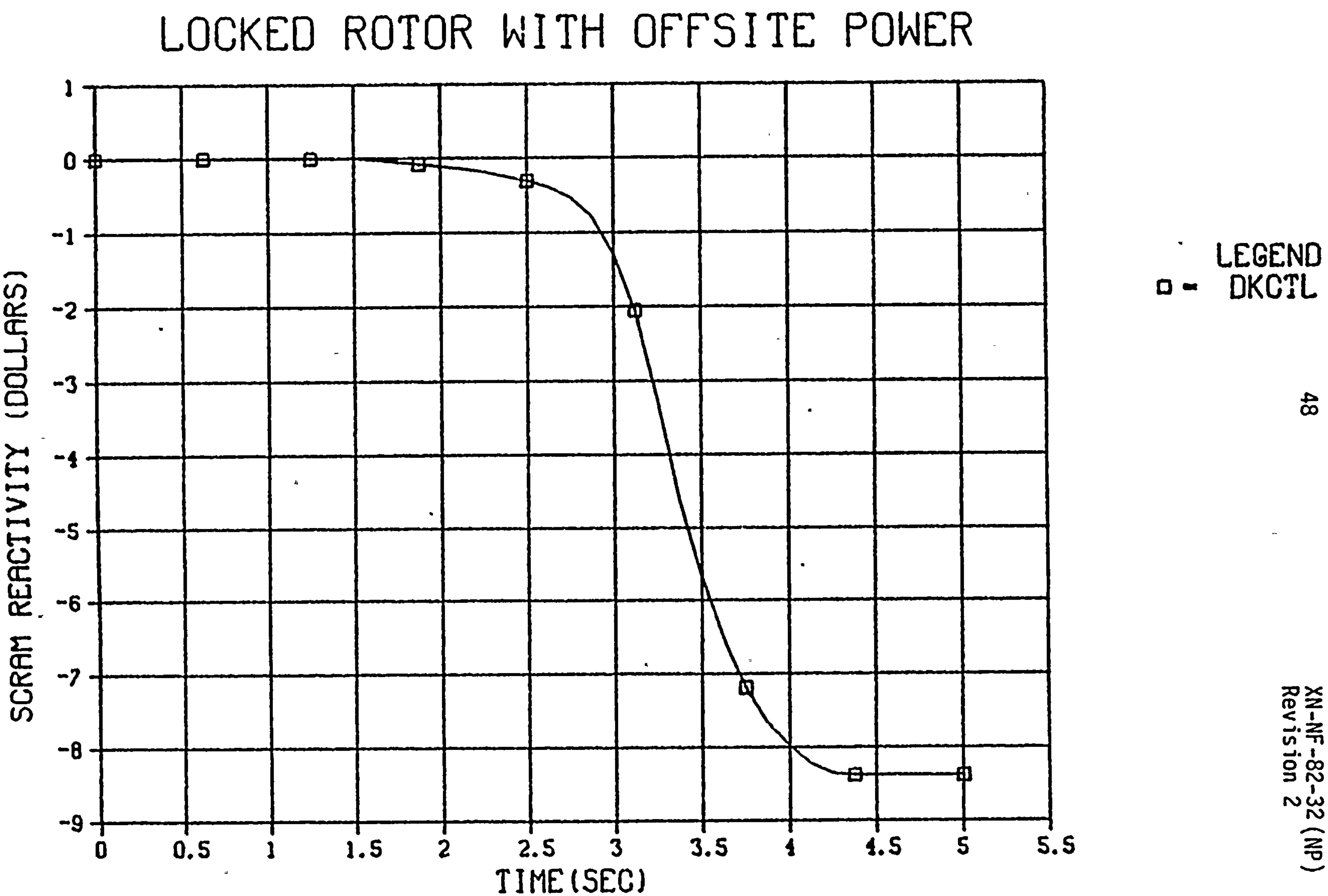

Figure 4.20 Scram Reactivity for Locked Rotor with Offsite Power

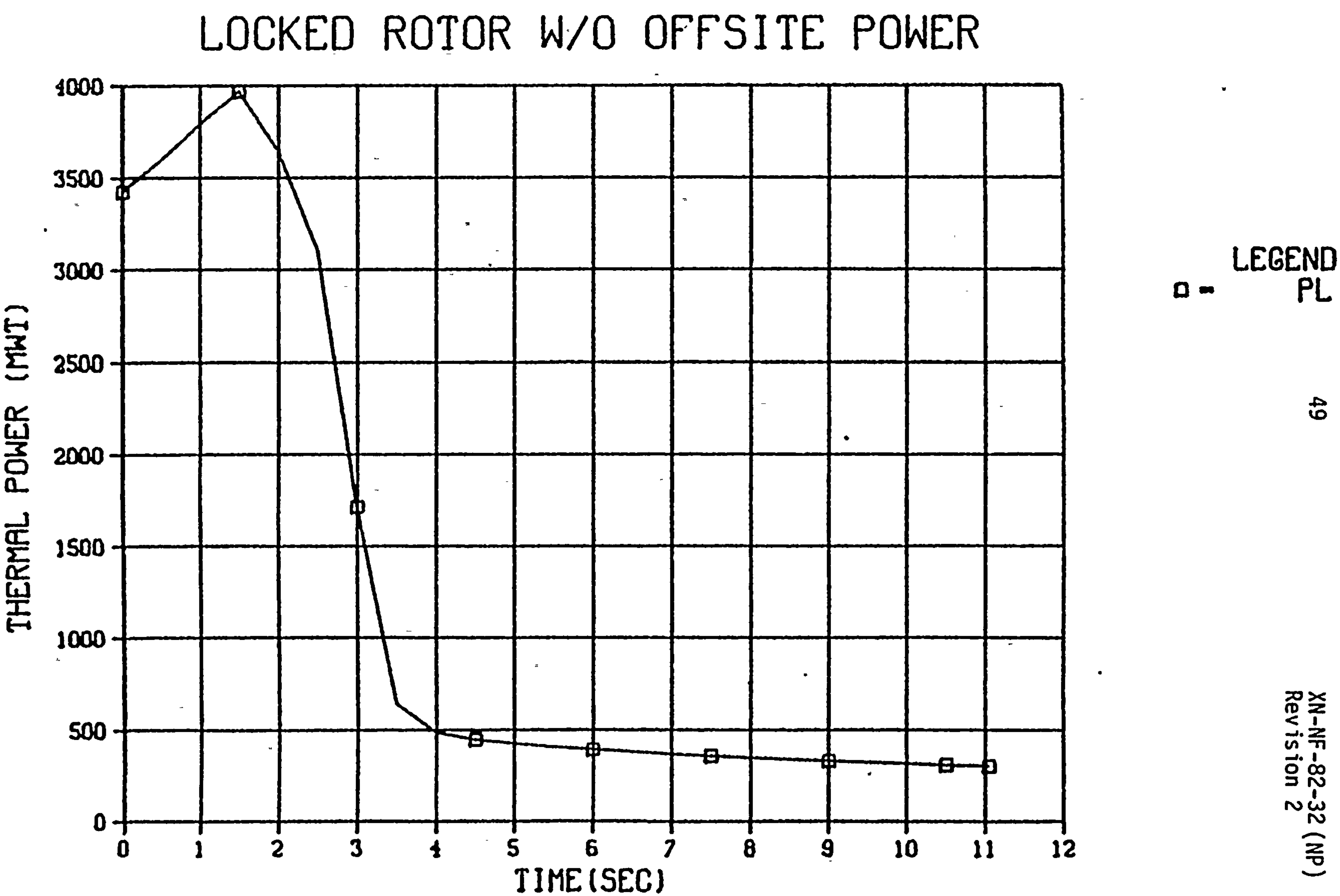

Figure 4.21 Thermal Power for Locked Rotor with Loss of Offsite Power

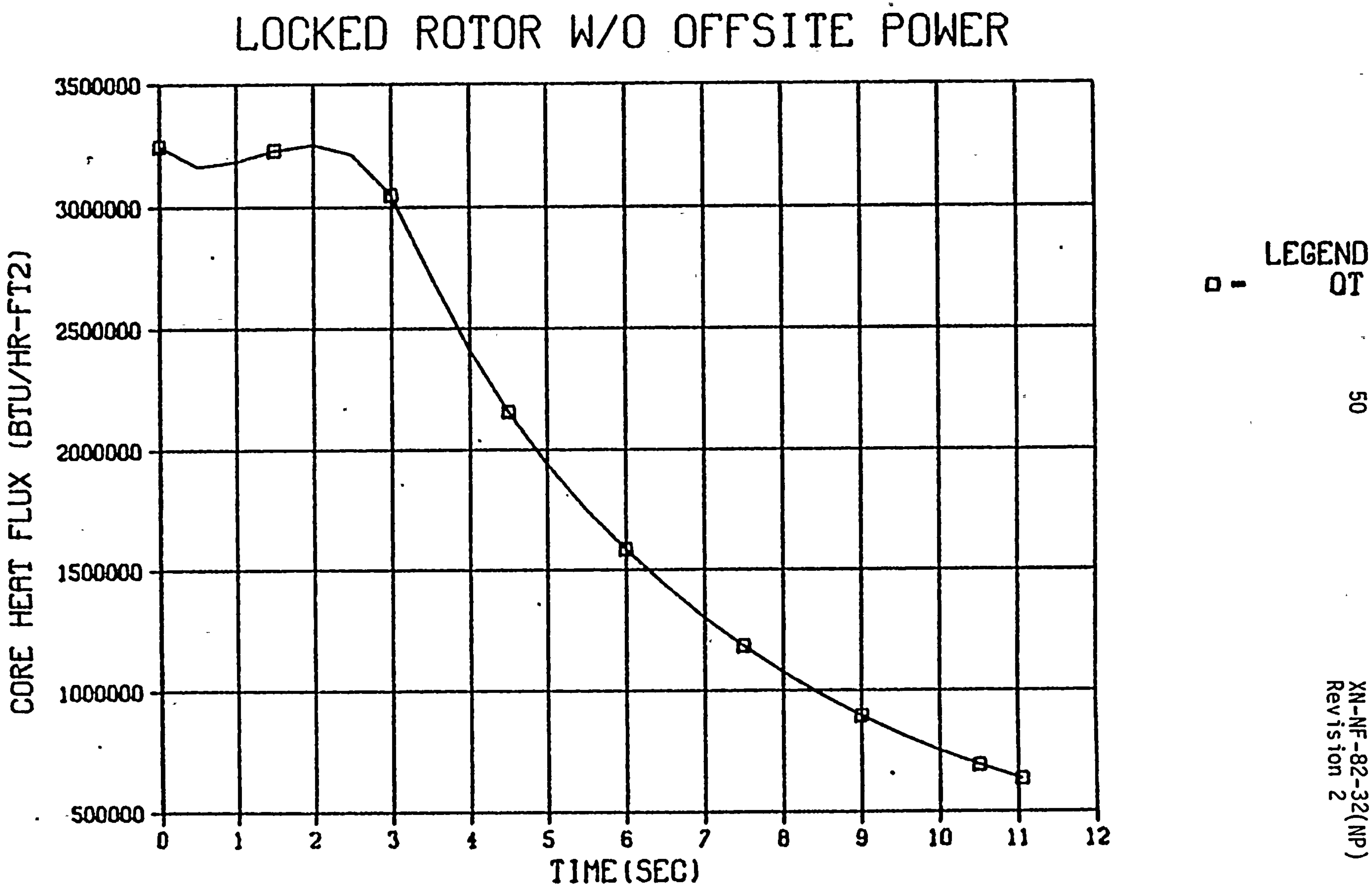

Figure 4.22 Core Heat Flux for Locked Rotor with Loss of Offsite Power

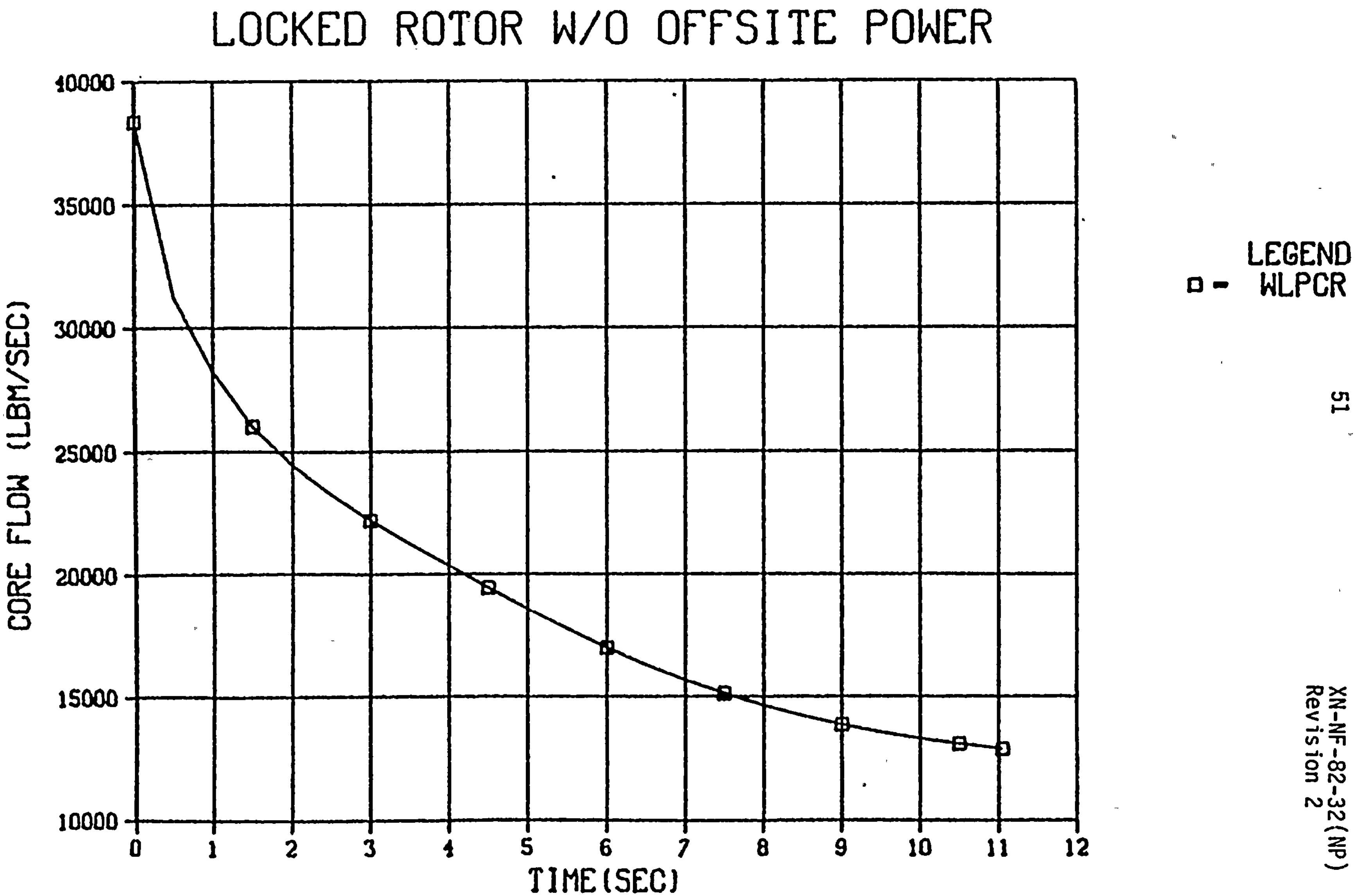

Figure 4.23 Core Flow for Locked Rotor with Loss of Offsite Power

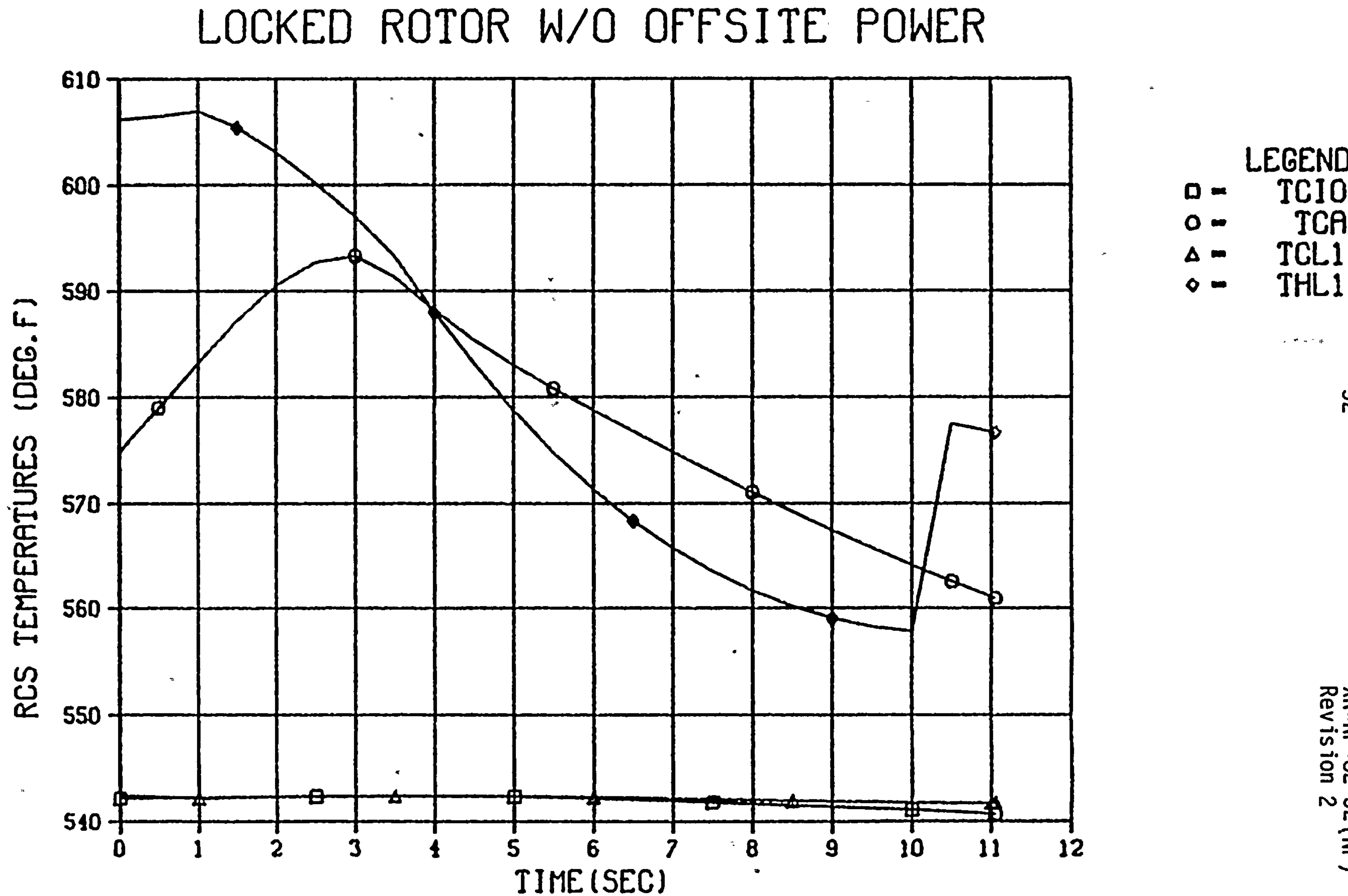

Figure 4.24 RCS Temperatures for Locked Rotor with Loss of Offsite Power - Hot Leg, Core Average, Cold Leg, and Core Inlet Temperatures

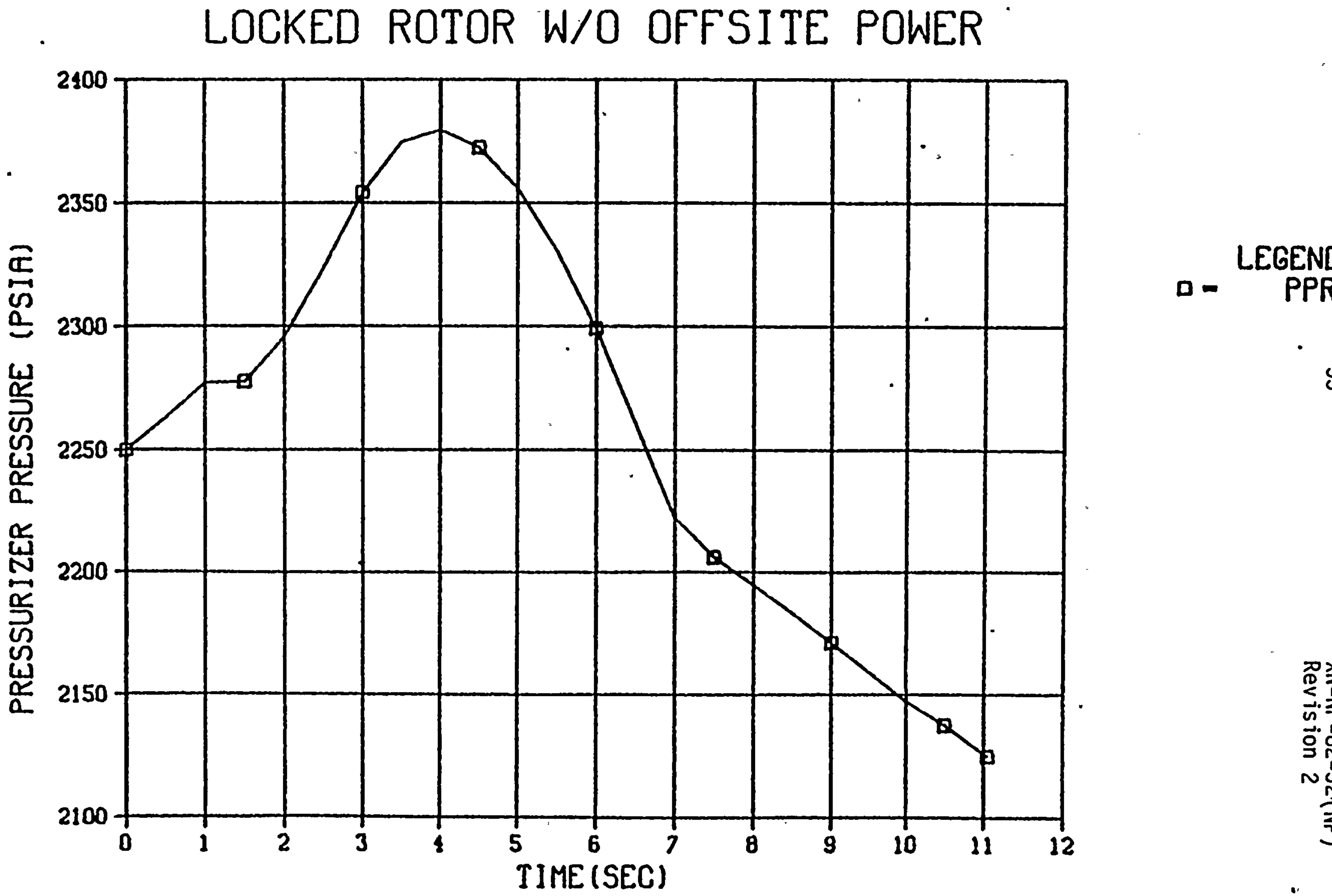

Figure 4.25 Pressurizer Pressure for Locked Rotor with Loss of Offsite Power

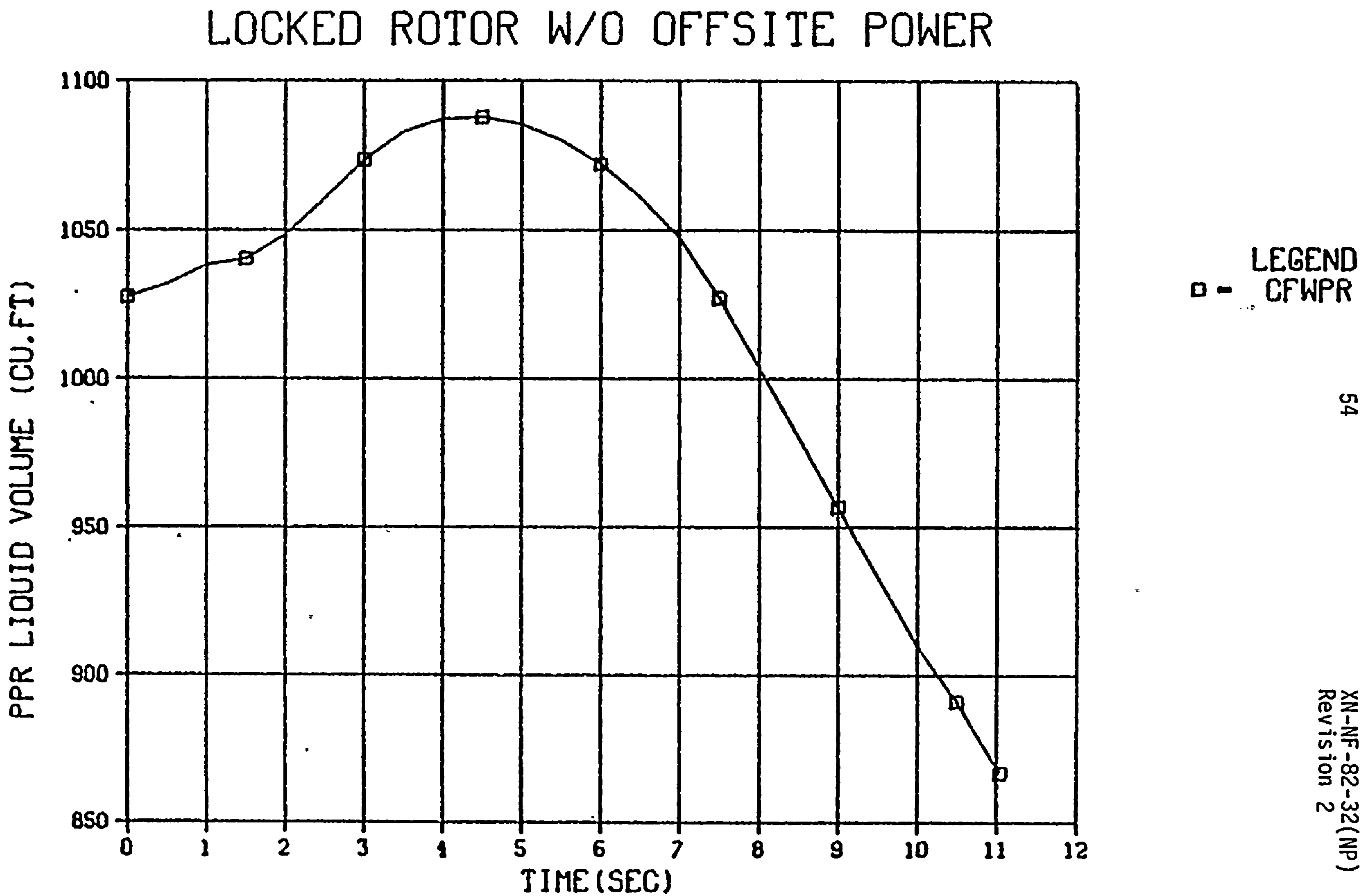

Figure 4.26 Pressurizer Liquid Volume for Locked Rotor with Loss of Offsite Power

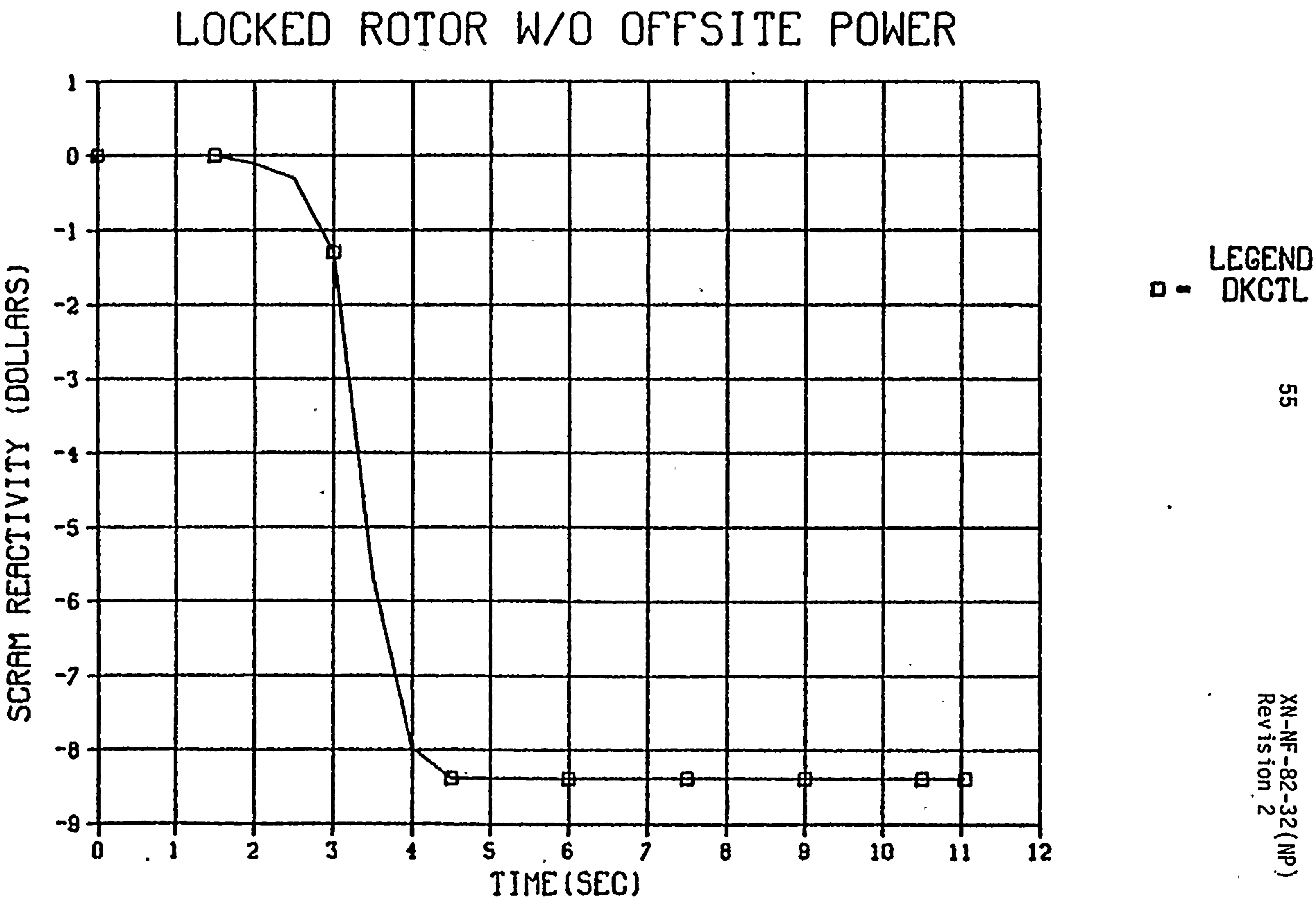

Figure 4.27 Scram Reactivity for Locked Rotor with Loss of Offsite Power

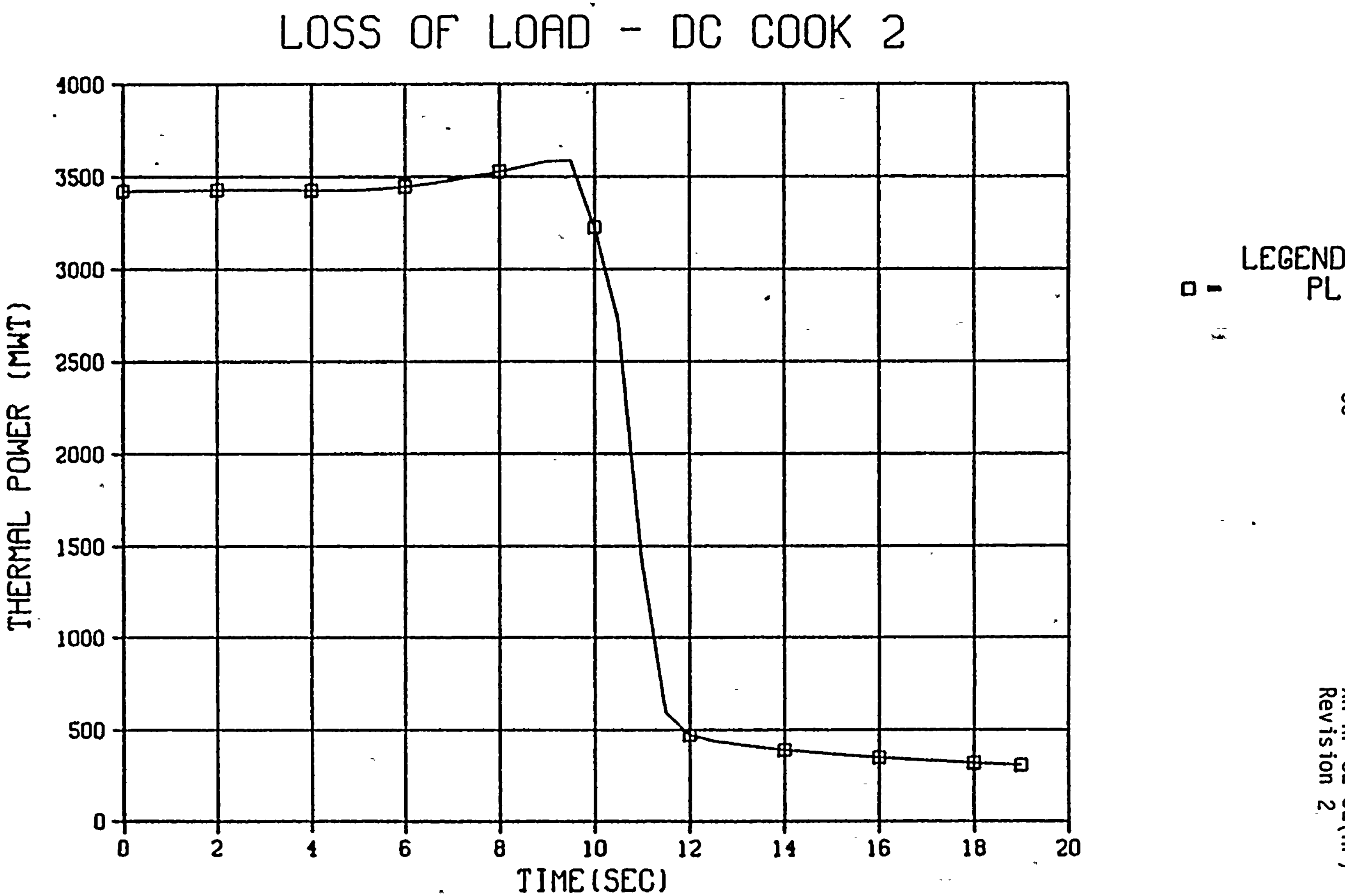

Figure 4.28 Thermal Power for Loss of Load

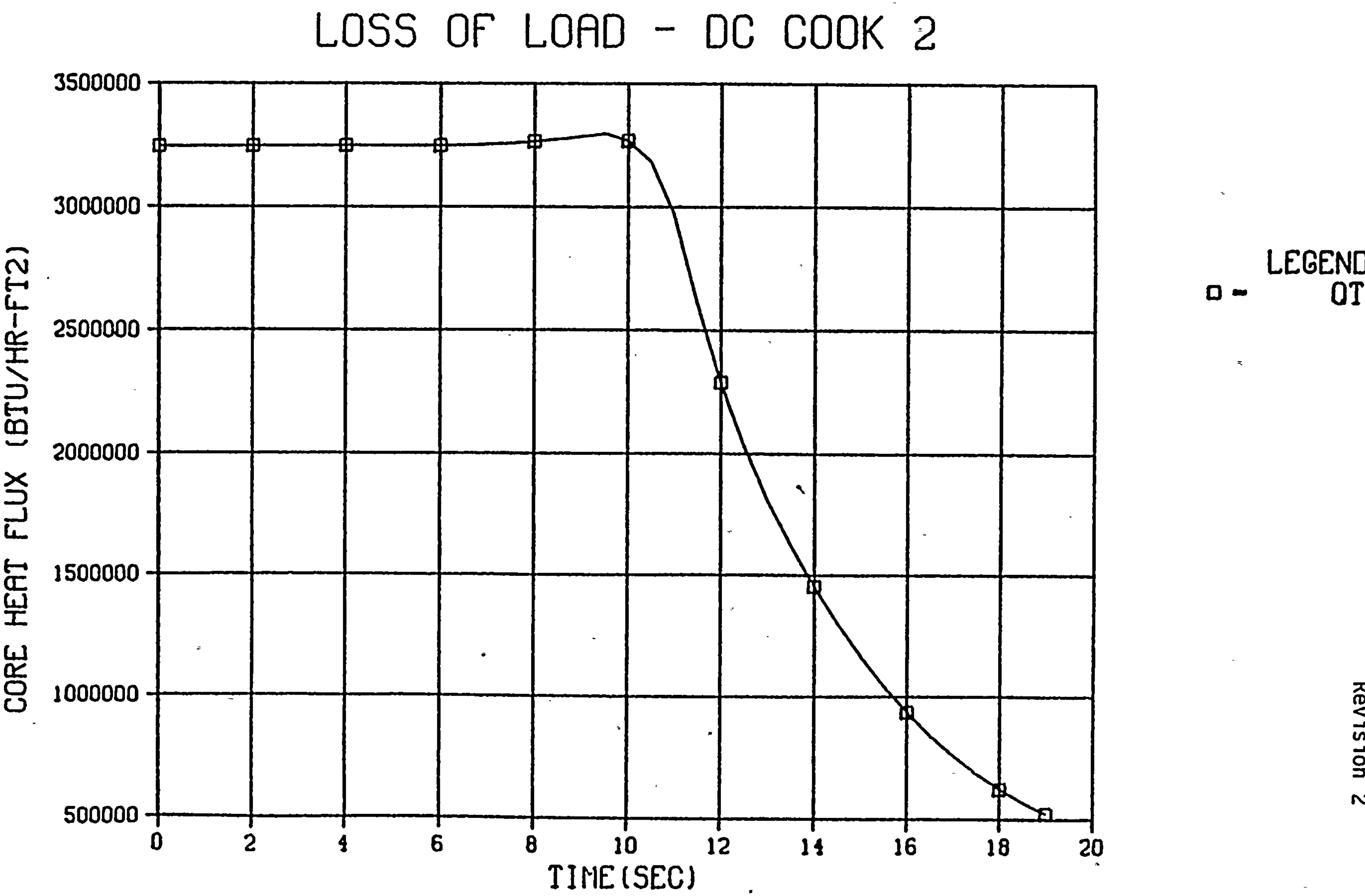

Figure 4.29 Core Heat Flux for Loss of Load

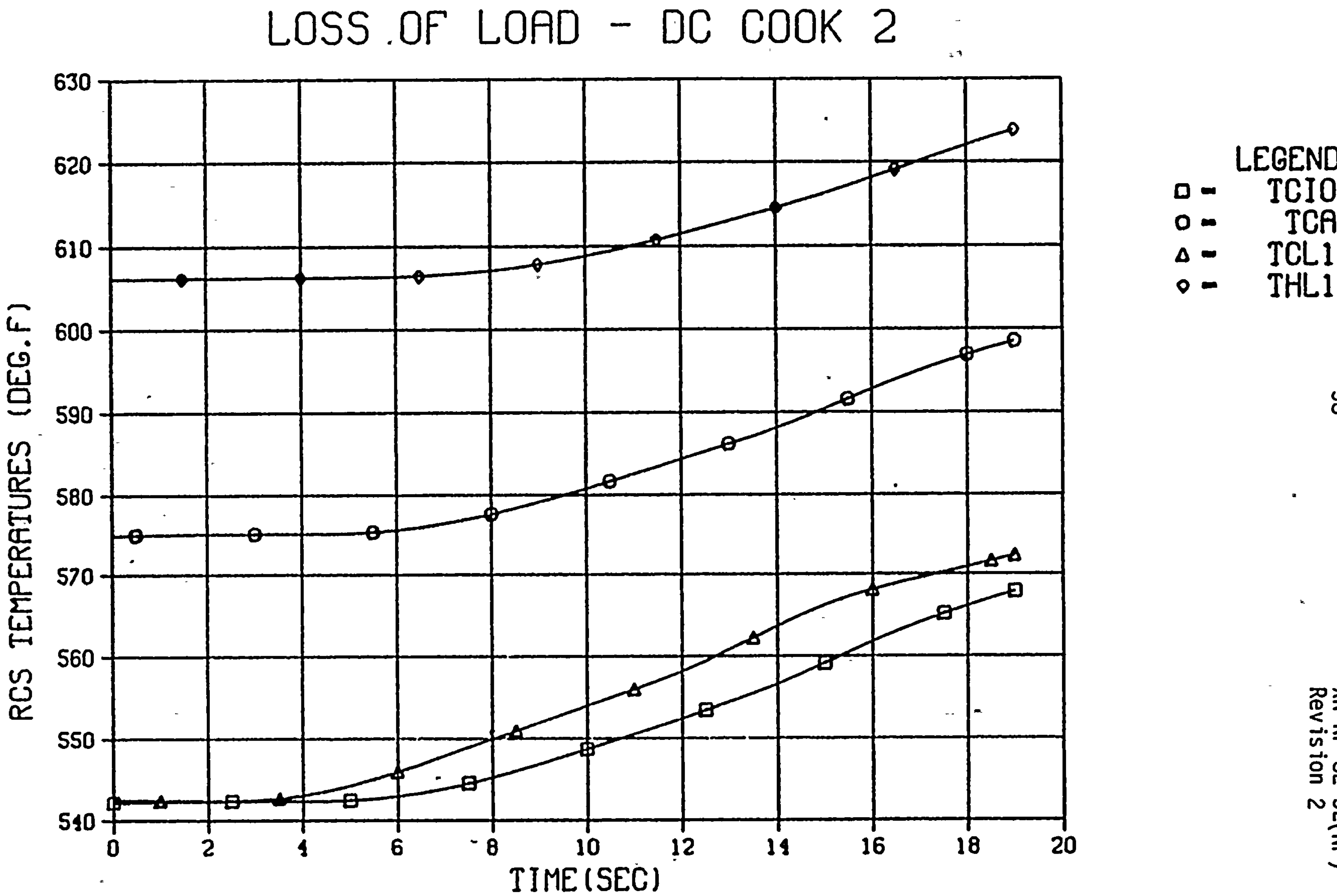

Figure 4.30 RCS Temperatures for Loss of Load - Hot Leg, Core Average, Cold Leg, and Core Inlet Temperatures

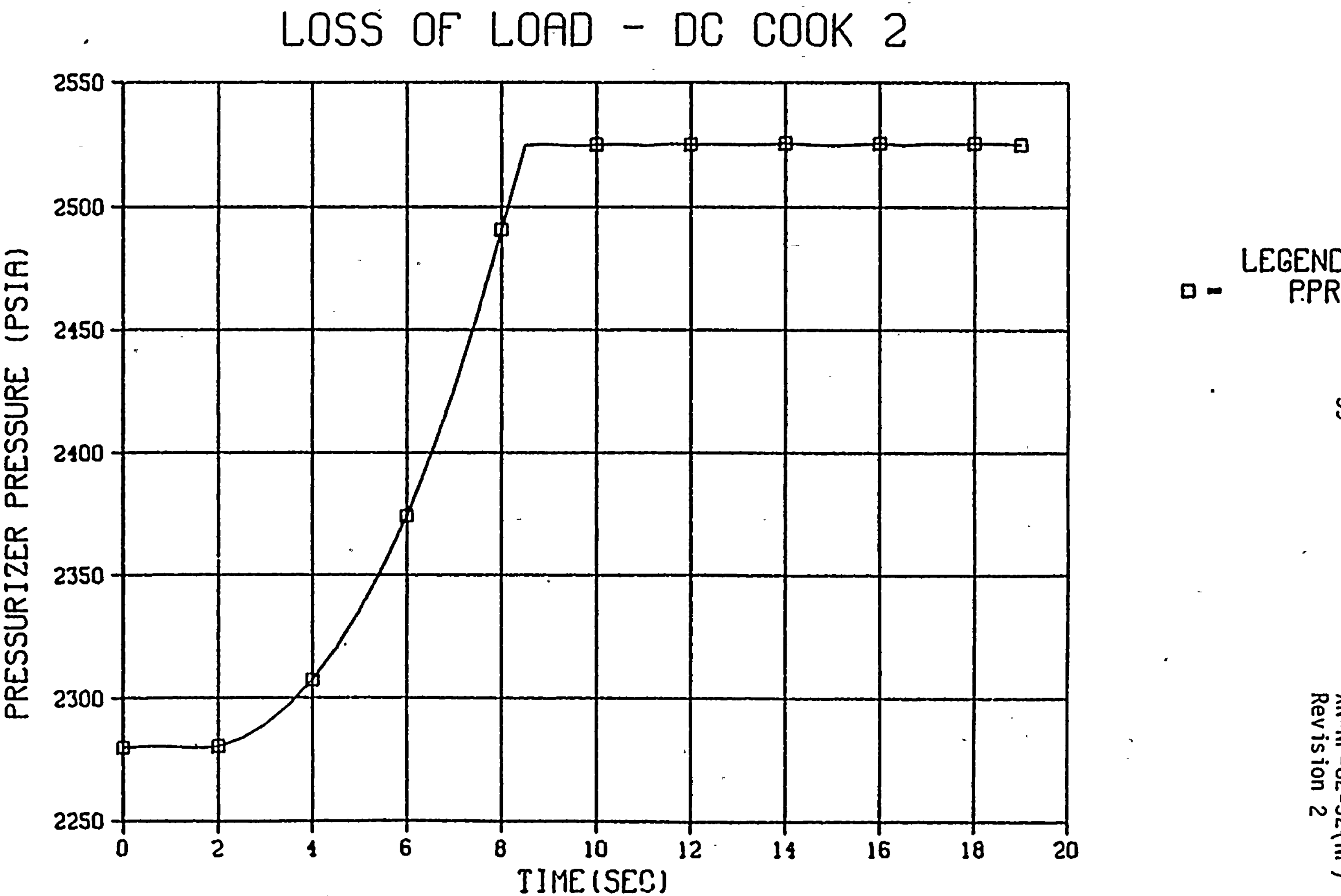

Figure 4.31 Pressurizer Pressure for Loss of Load

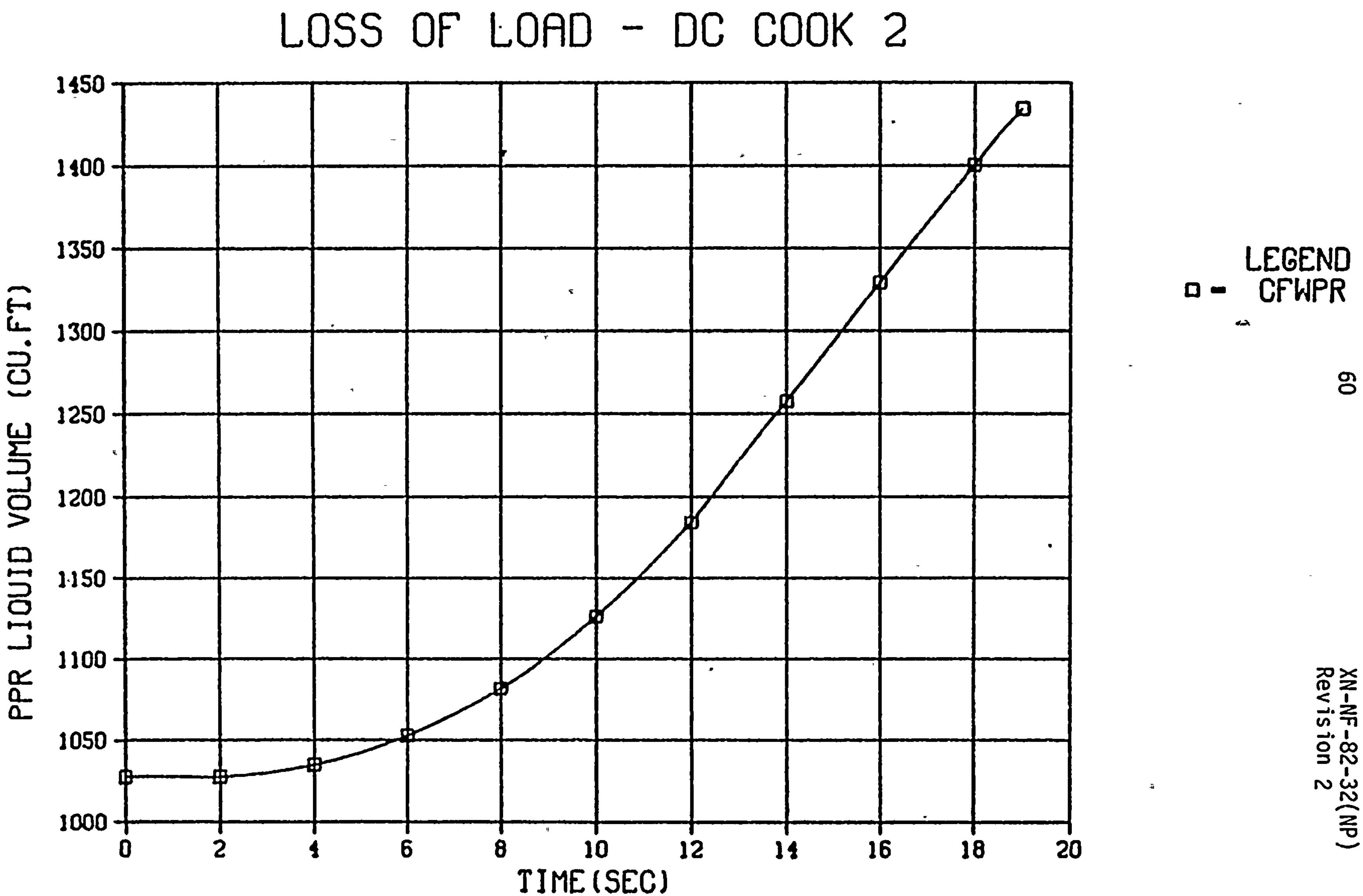

Figure 4.32 Pressurizer Liquid Volume for Loss of Load

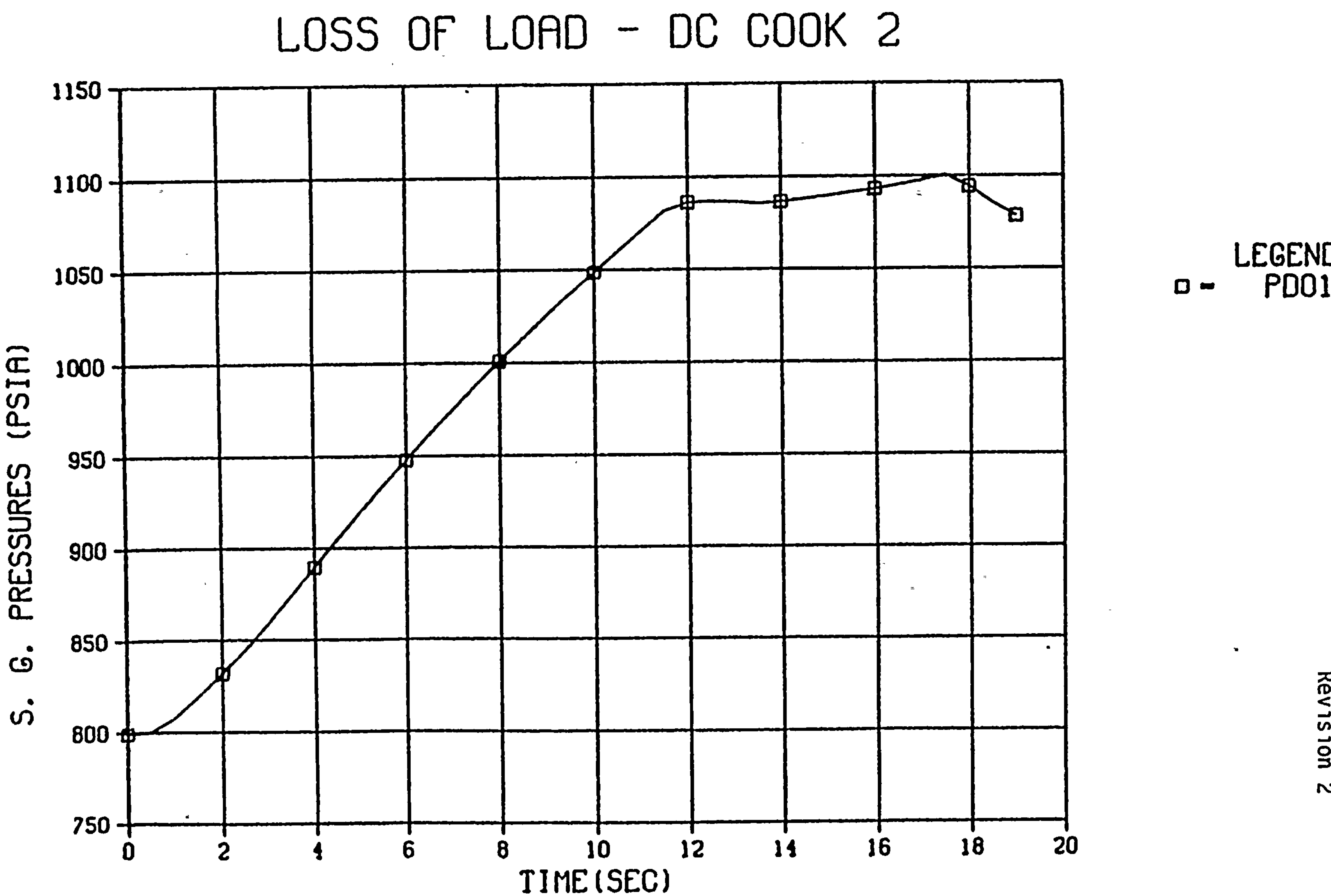

Figure 4.33 Steam Generator Pressure for Loss of Load

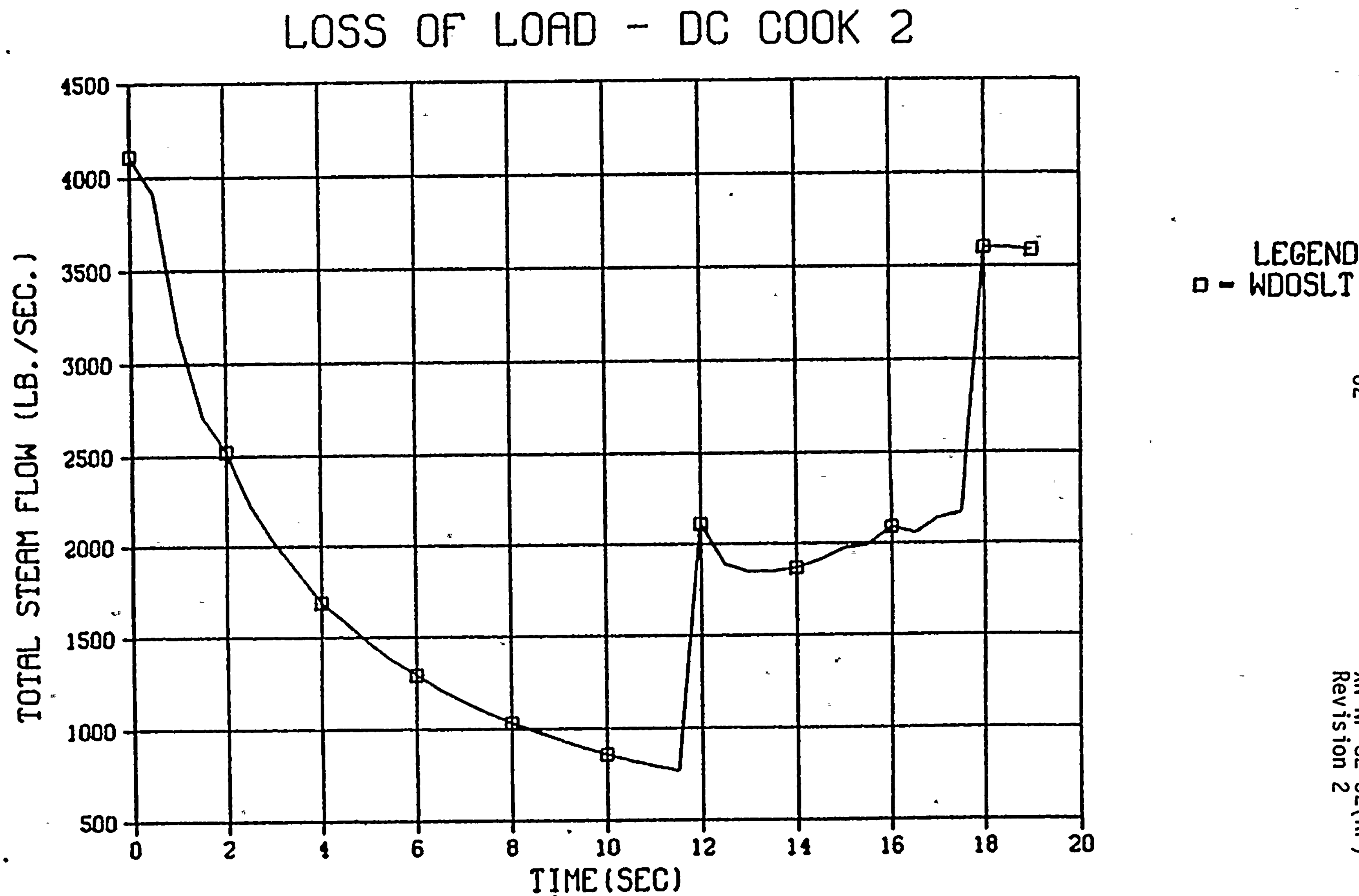

Figure 4.34 Steam Flow for Loss of Load

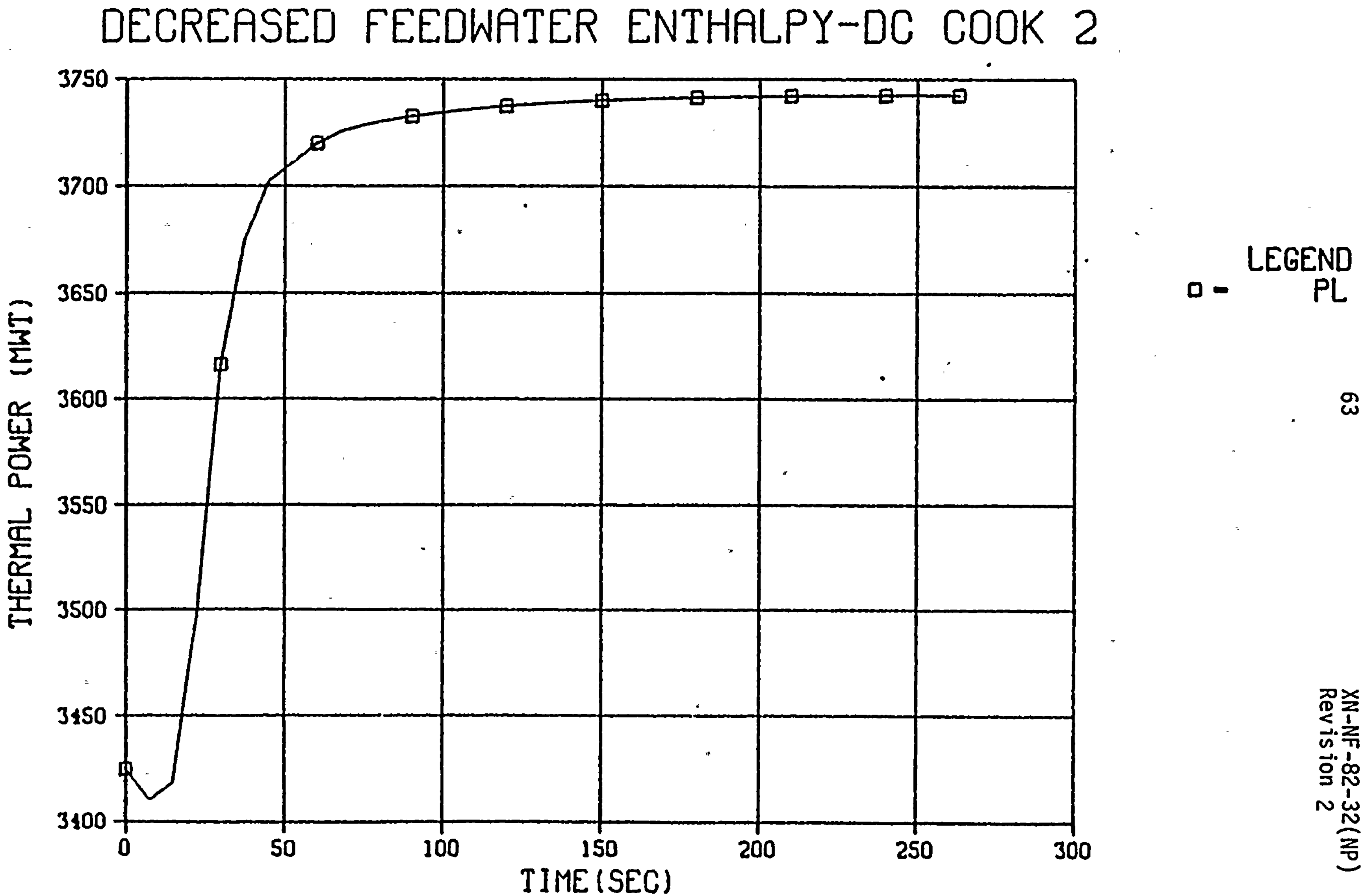

Figure 4.35 Thermal Power for Decreased Feedwater Heating

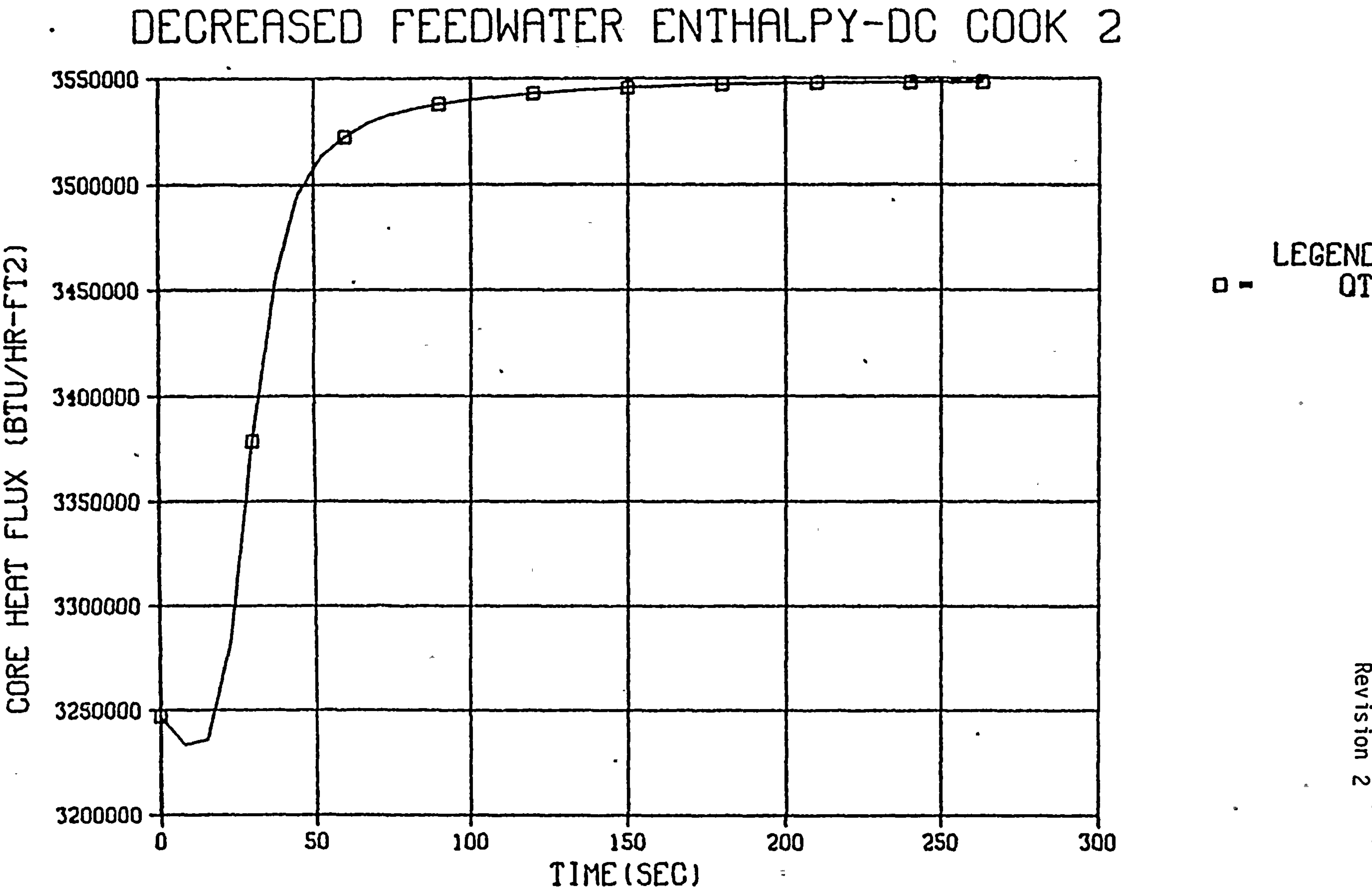

Figure 4.36 Core Heat Flux for Decreased Feedwater Heating

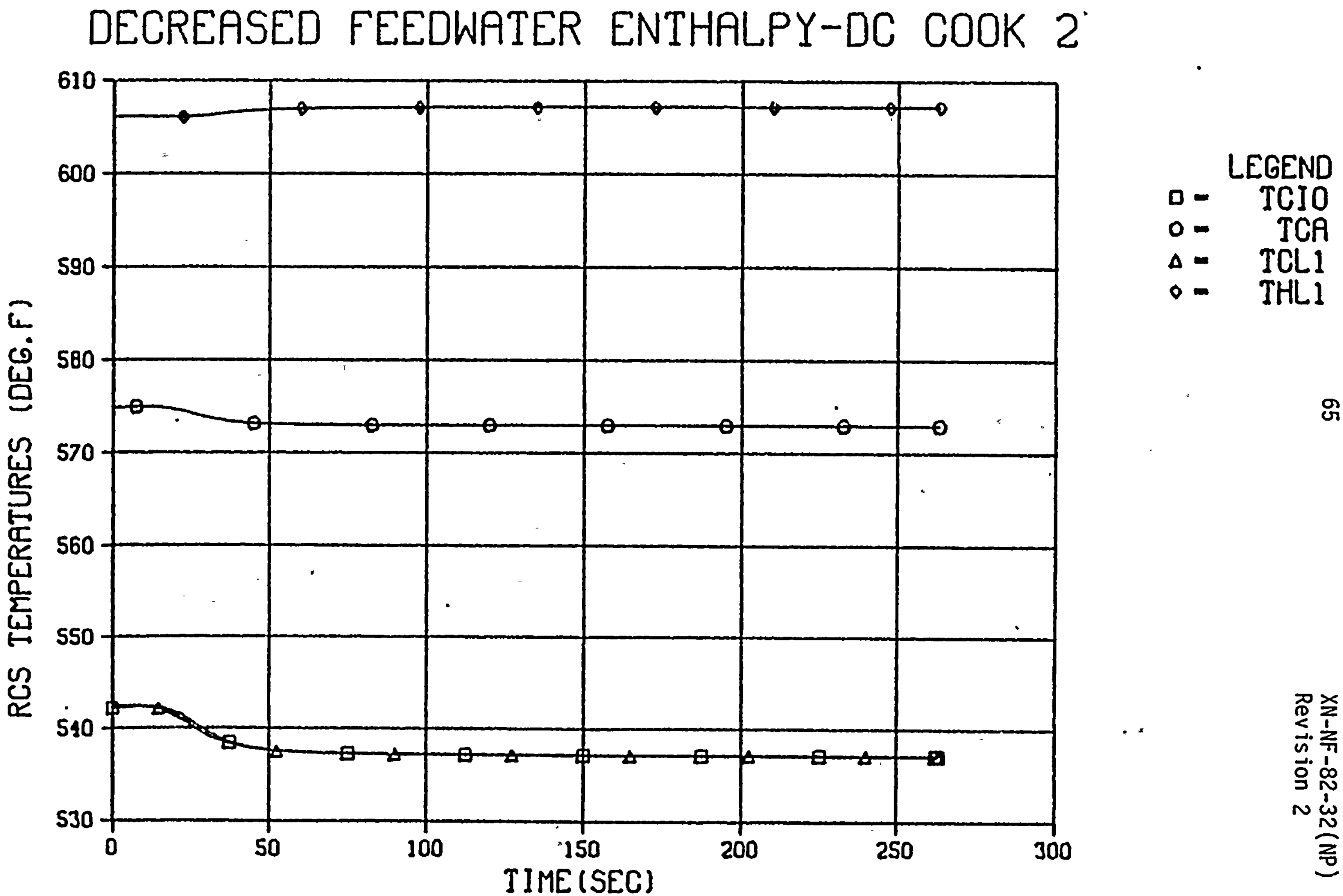

Figure 4.37 RCS Temperatures for Decreased Feedwater Heating - Hot Leg, Core Average, Cold Leg, and Core Inlet Temperatures

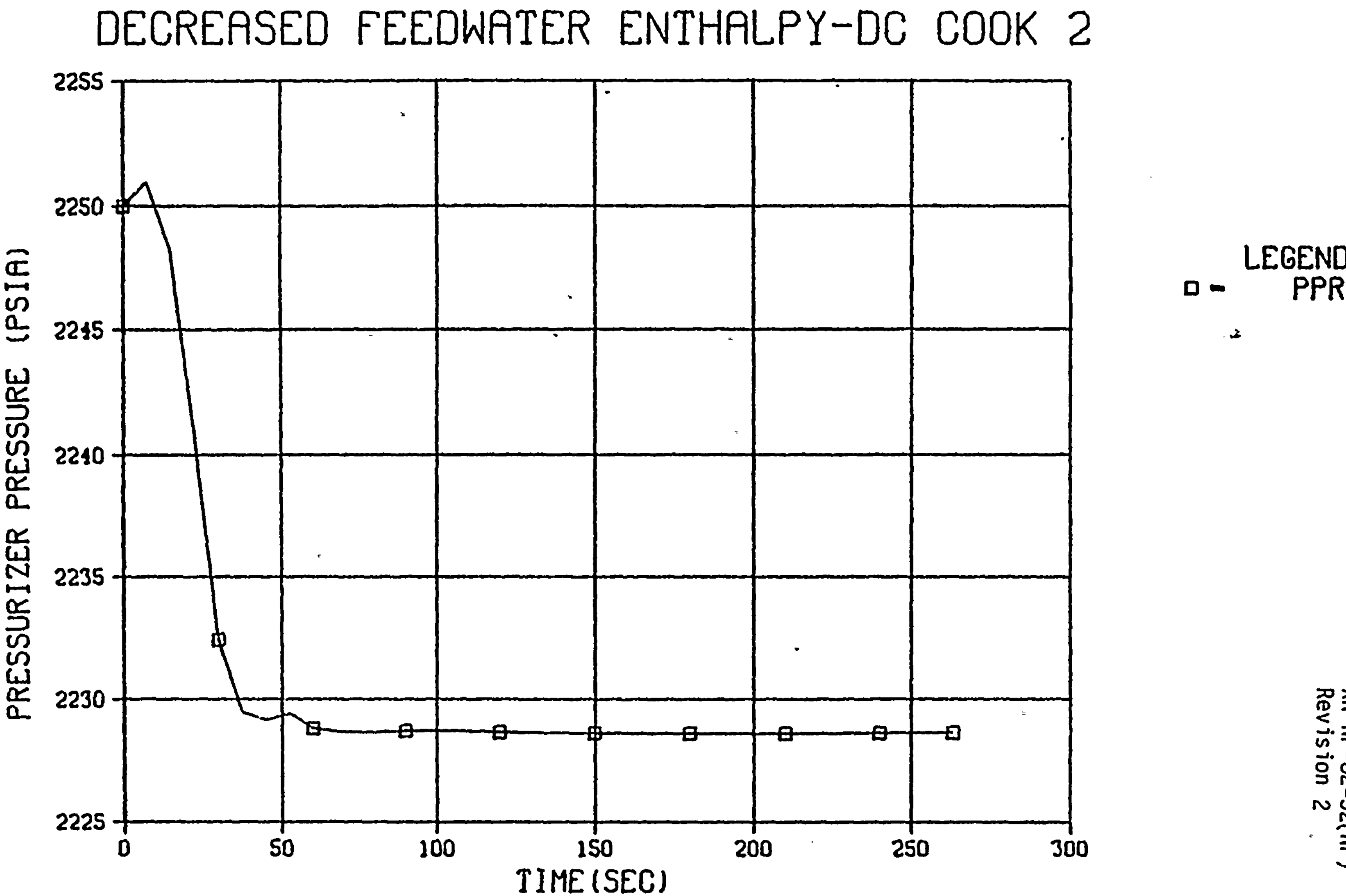

Figure 4.38 Pressurizer Pressure for Decreased Feedwater Heating

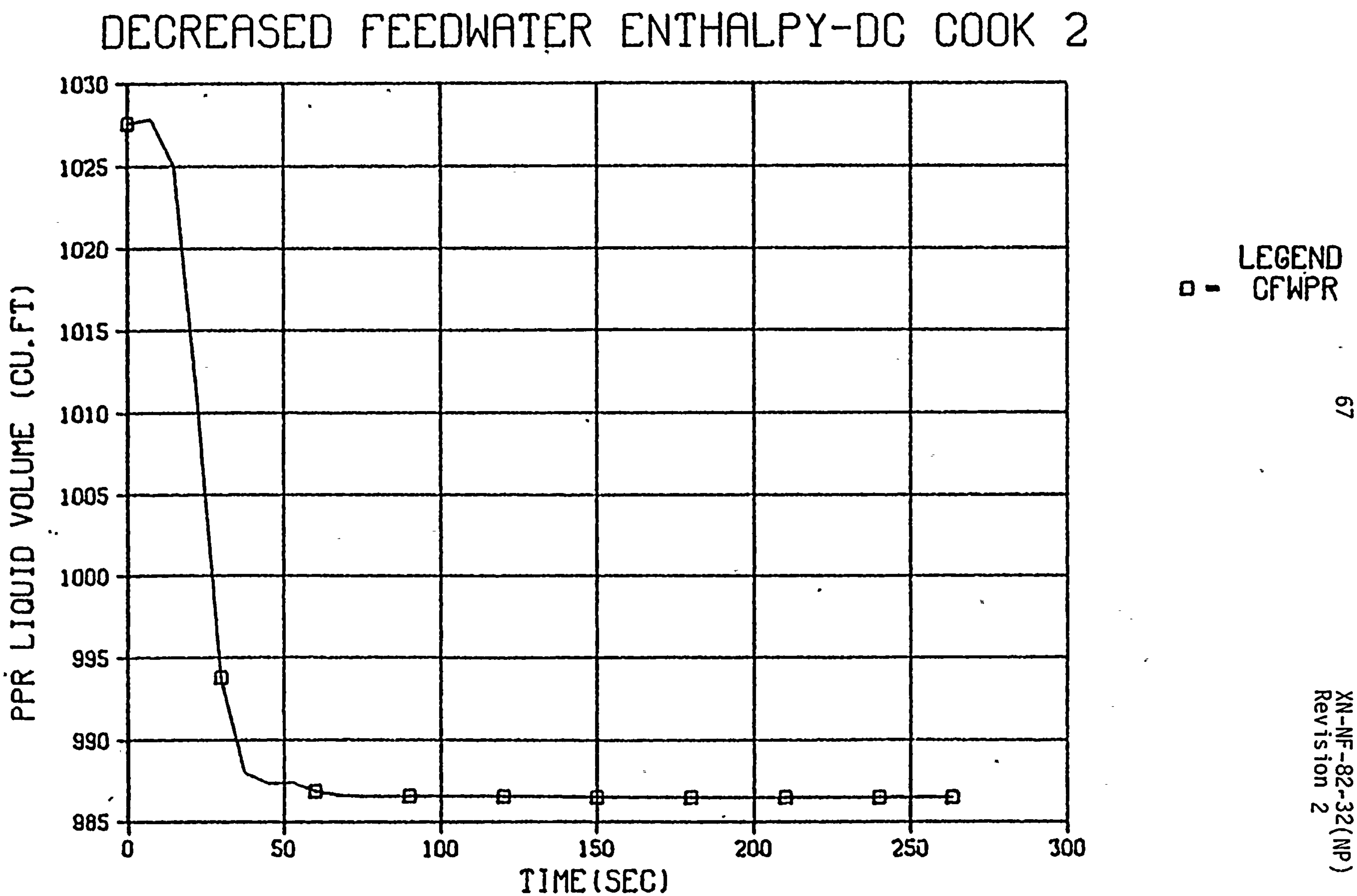

Figure 4.39 Pressurizer Liquid Volume

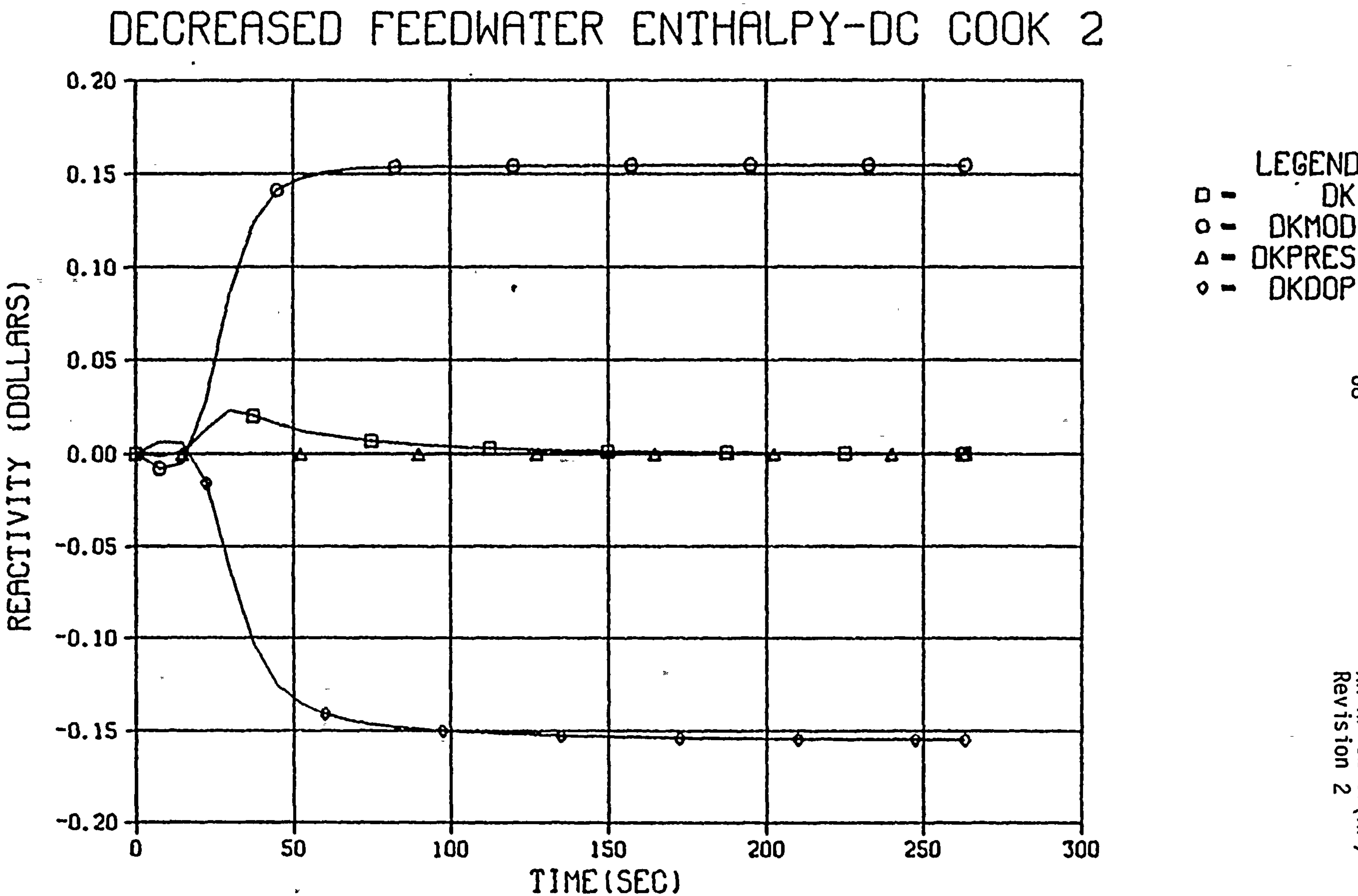

Figure 4.40 Core Reactivity for Decreased Feedwater Heating

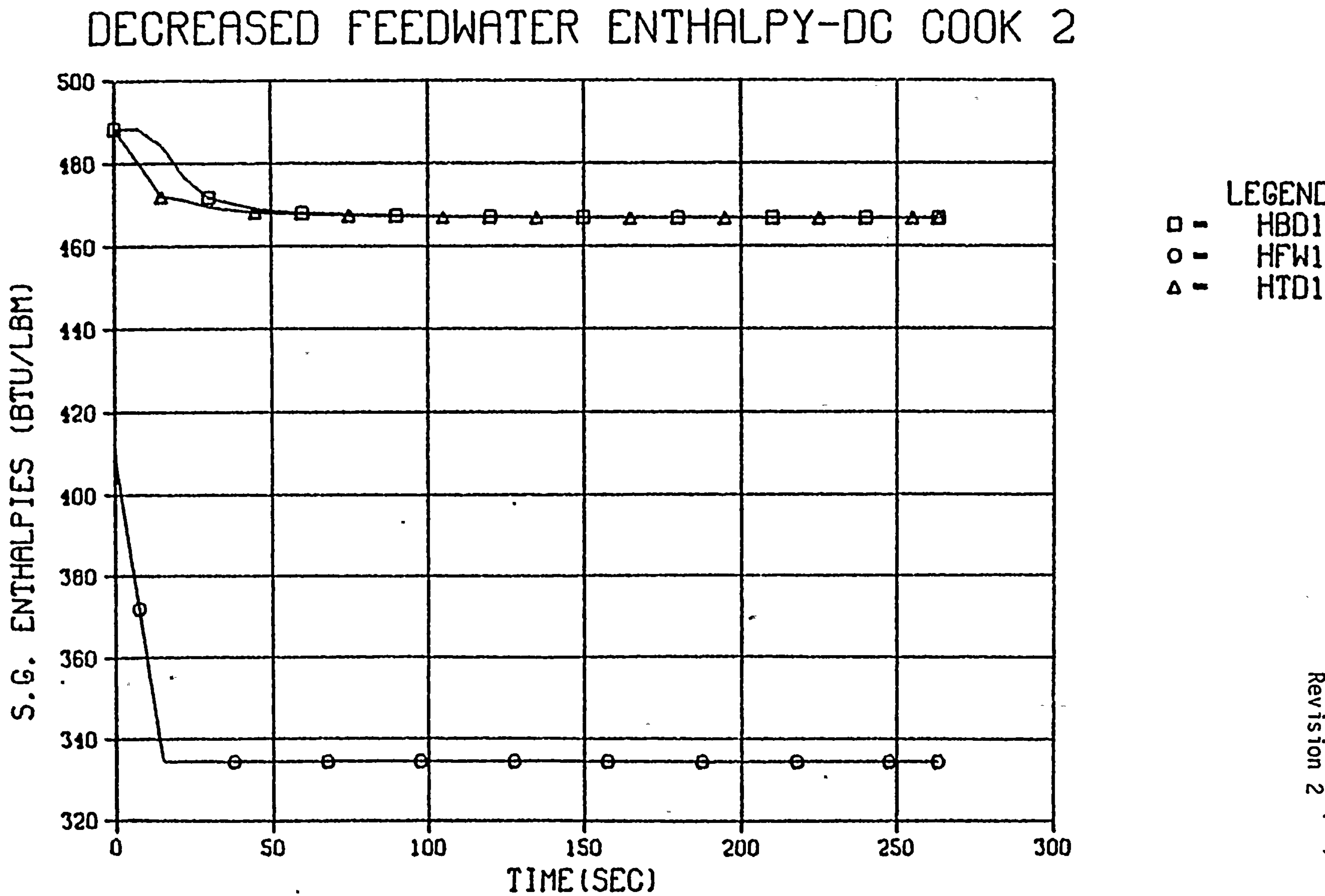

Figure 4.41 Steam Generator Enthalpies for Decreased Feedwater
Heating: Bottom of Downcomer, Top of Downcomer, and Feedwater

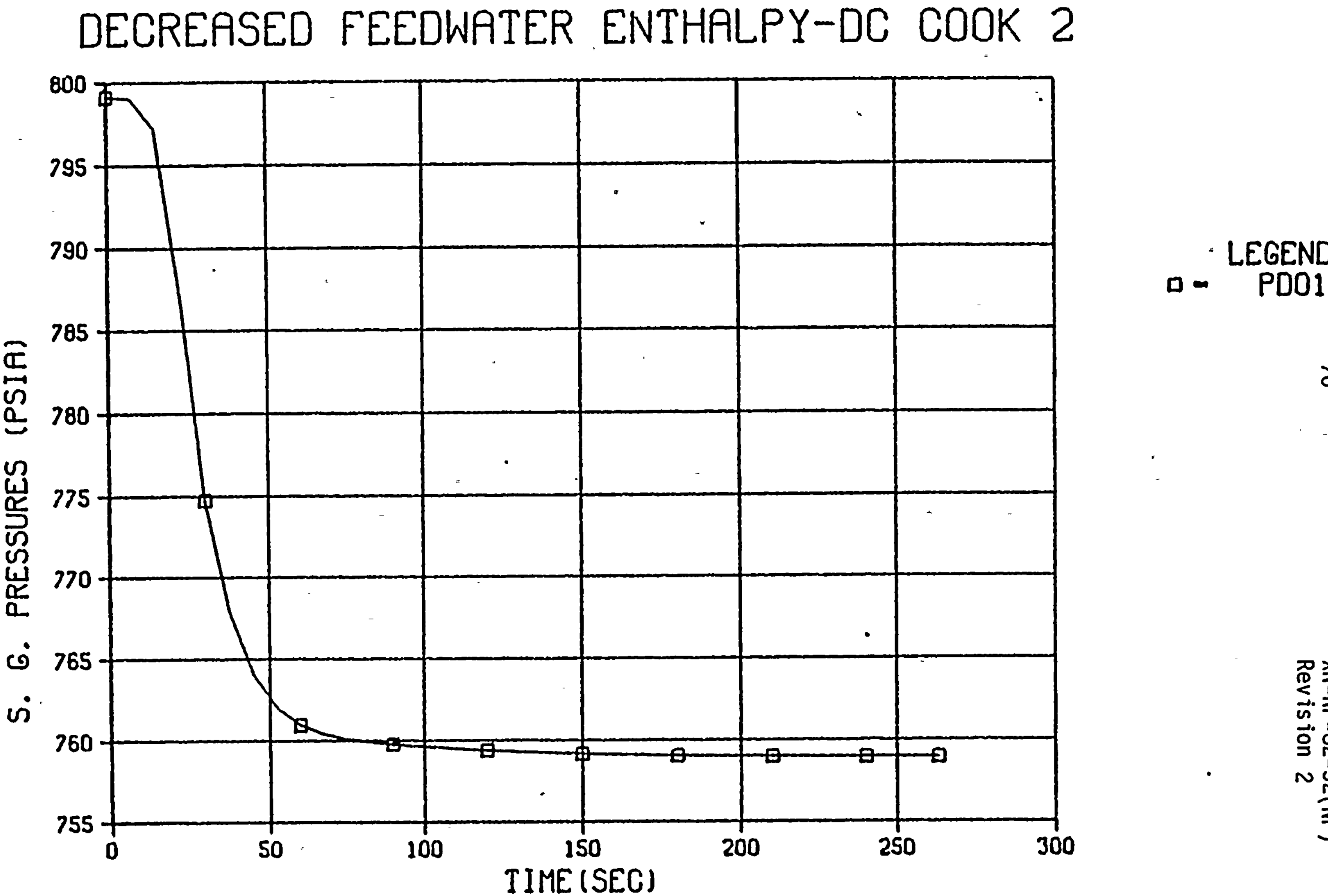

Figure 4.42 Steam Generator Pressure for Decreased Feedwater Heating

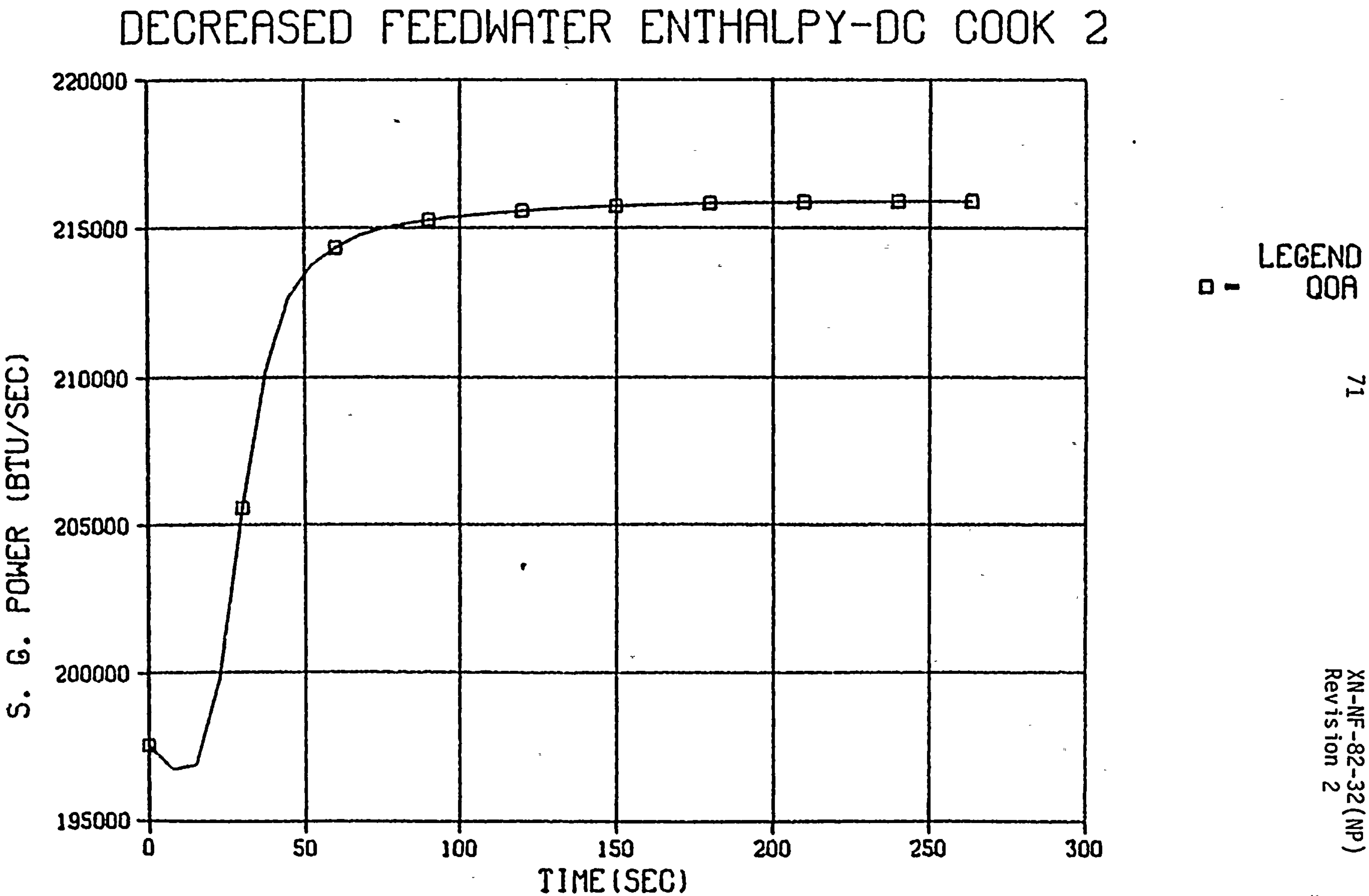

Figure 4.43 Steam Generator Power for Decreased Feedwater Heating

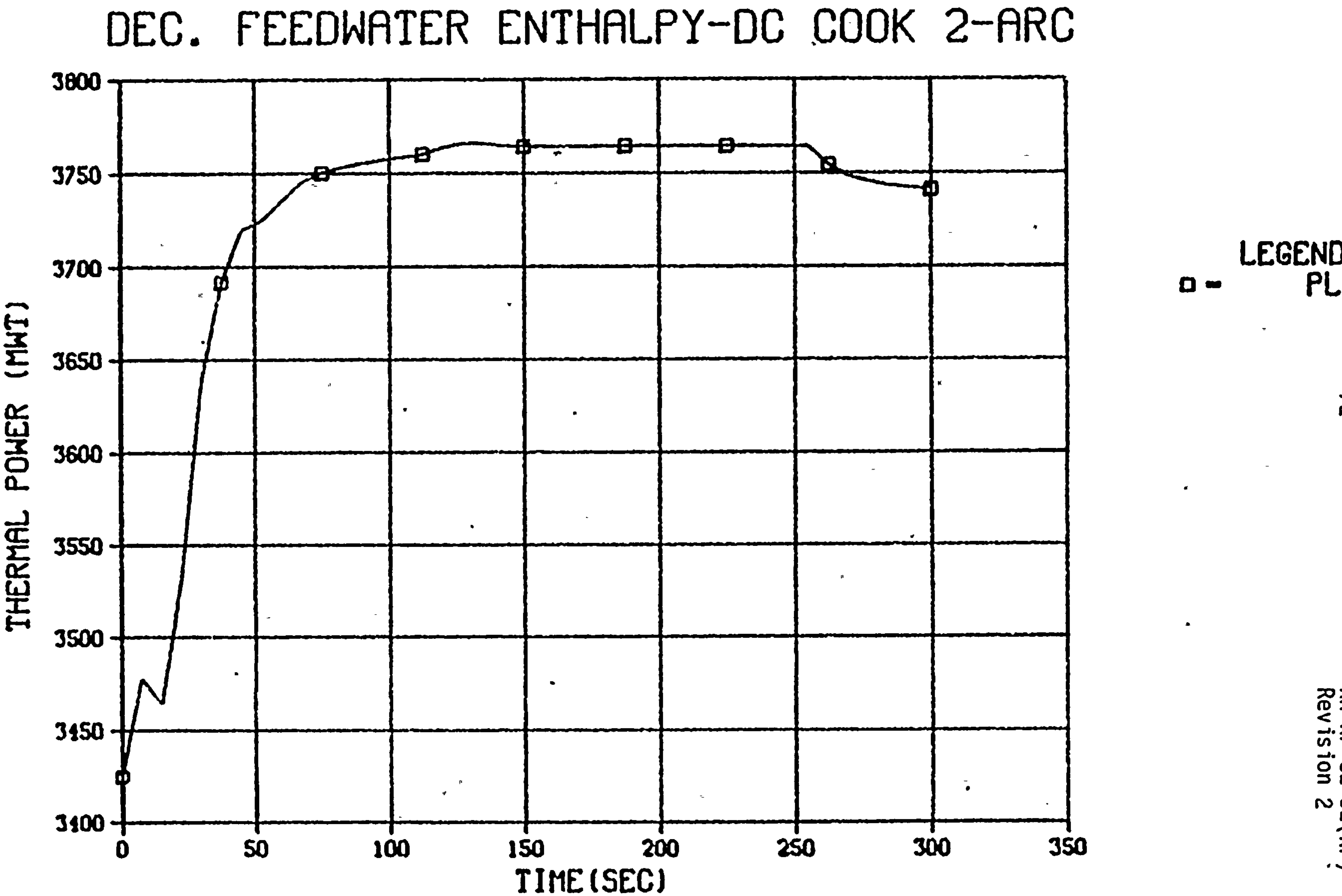

Figure 4.44 Thermal Power for Decreased Feedwater Heating with Automatic Rod Control

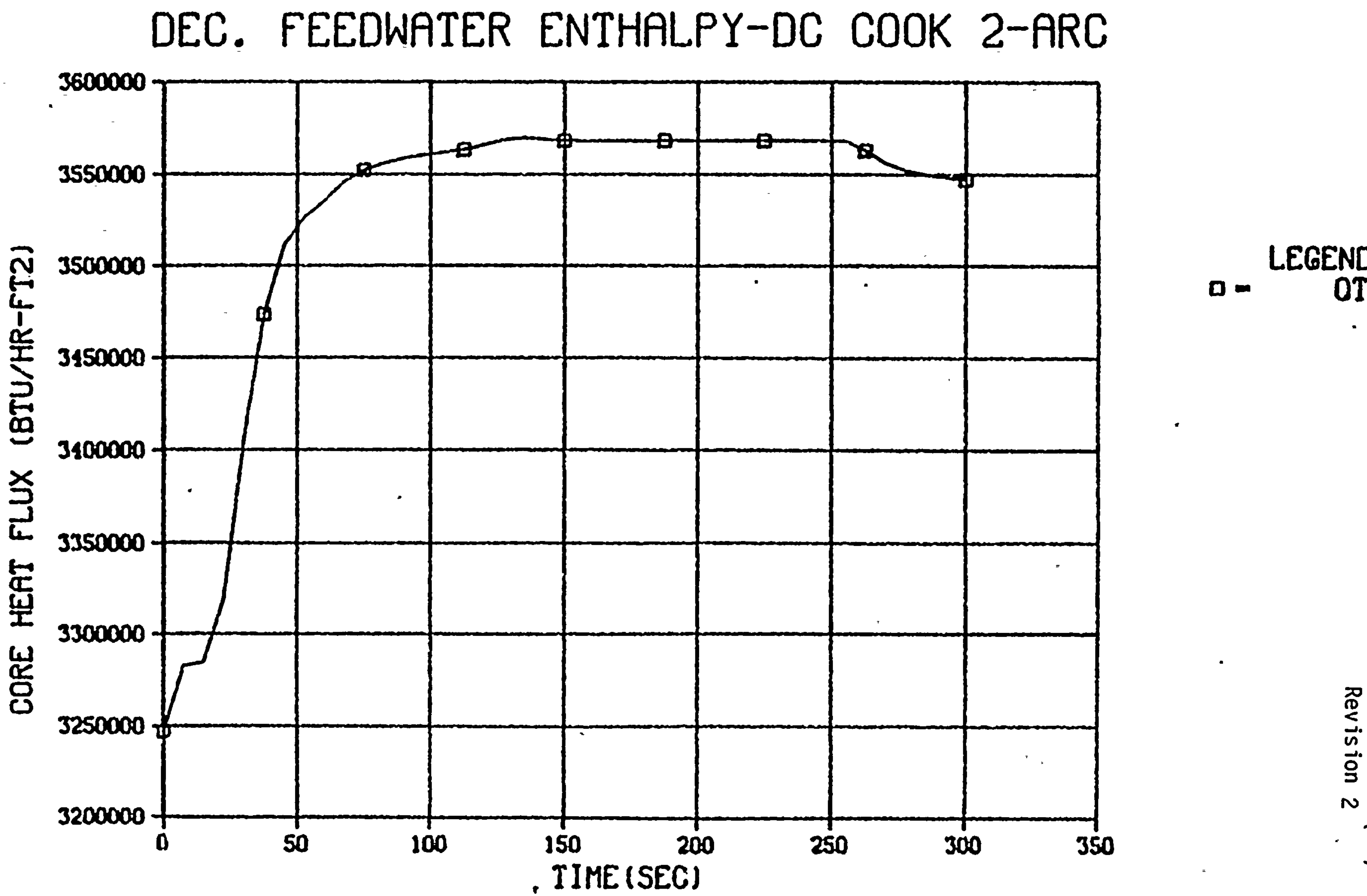

Figure 4.45 Core Heat Flux for Decreased Feedwater Heating with Automatic Rod Control

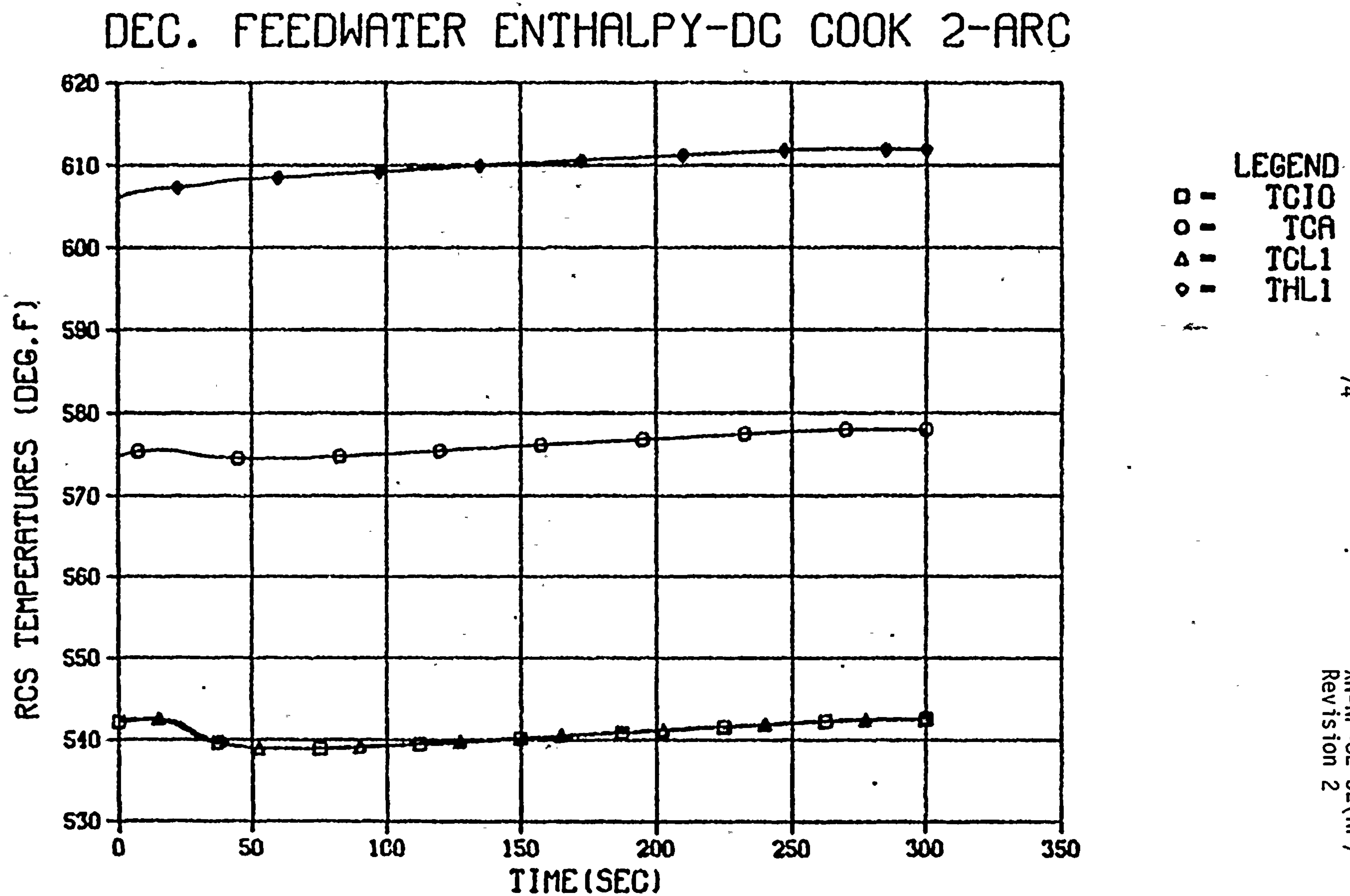

Figure 4.46 RCS Temperatures for Decreased Feedwater Heating with Automatic Rod Control - Hot Leg, Core Average, Cold Leg, and Core Inlet Temperatures

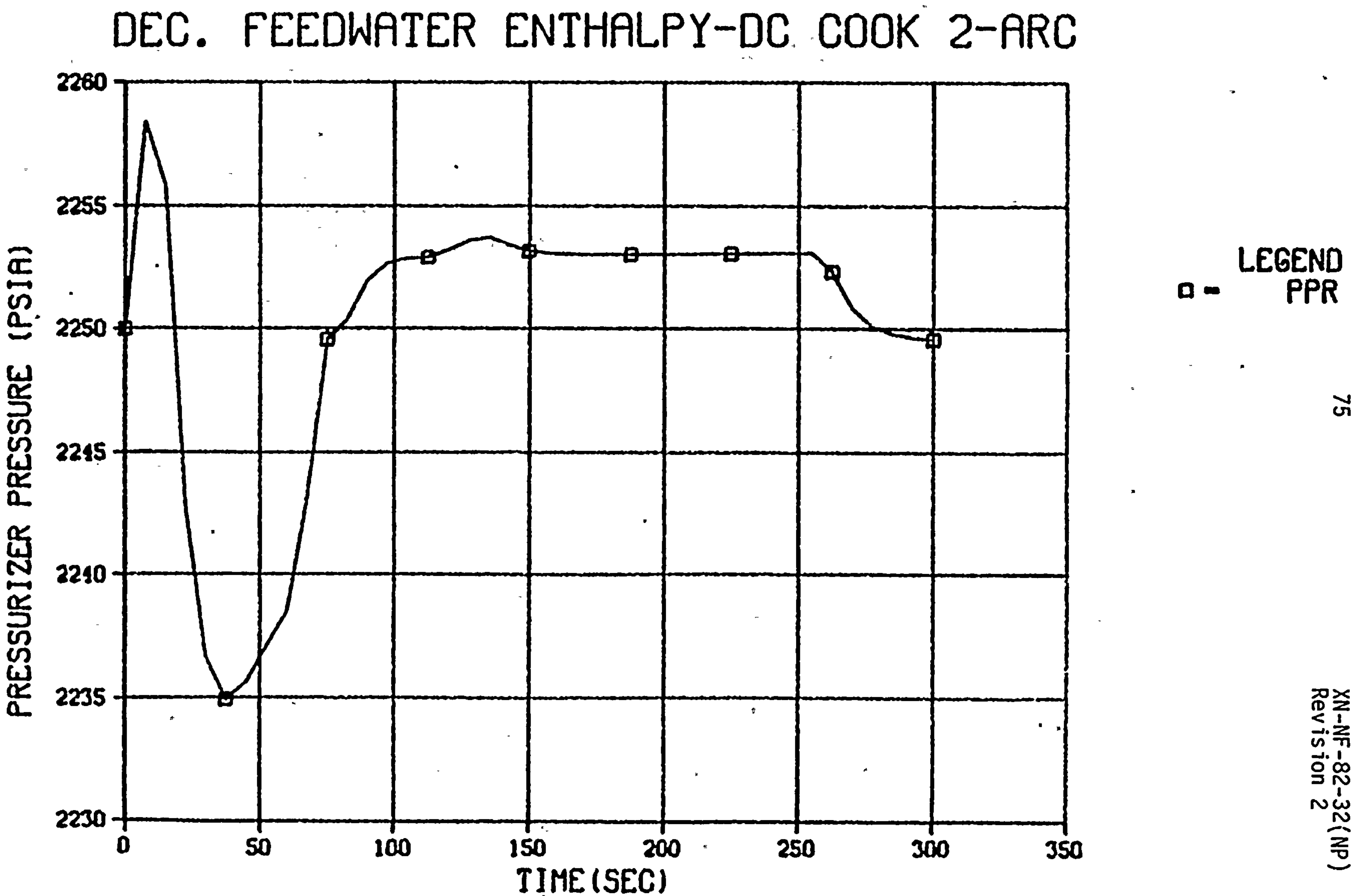

Figure 4.47 Pressurizer Pressure for Decreased Feedwater Heating with Automatic Rod Control

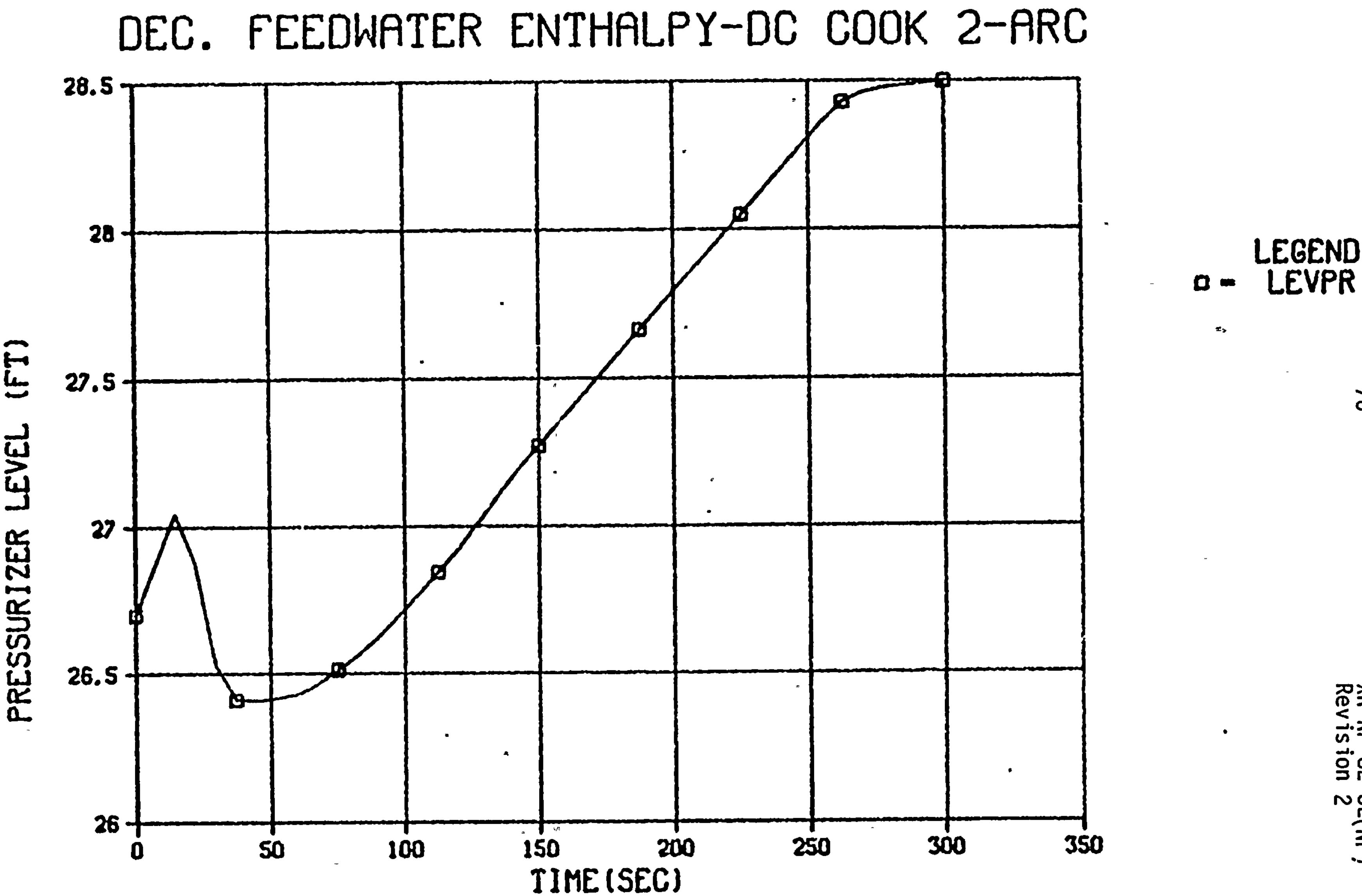

Figure 4.48 Pressurizer Liquid Level for Decreased Feedwater Heating with Automatic Rod Control

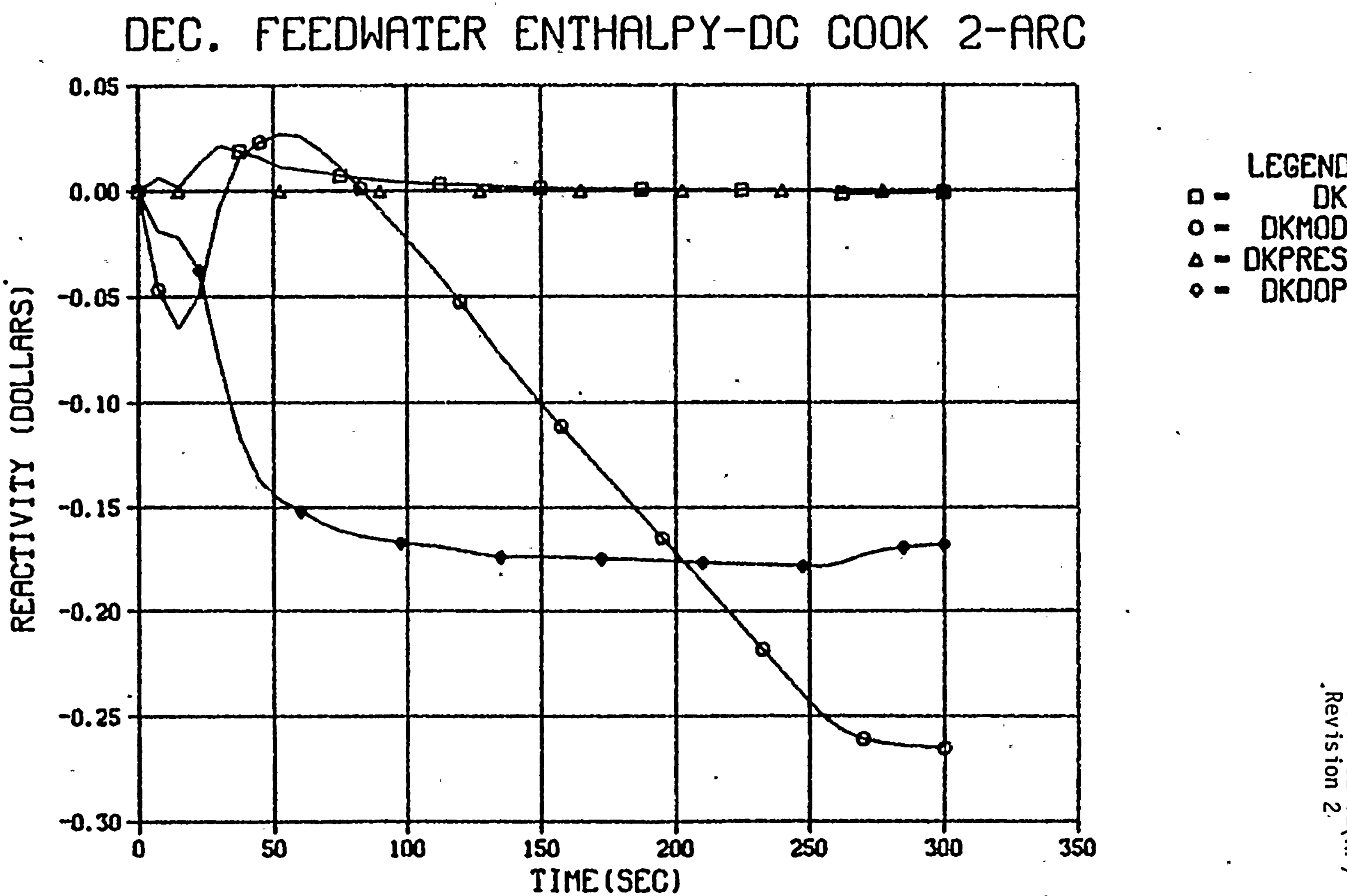

Figure 4.49 Core Reactivity for Decreased Feedwater Heating with Automatic Rod Control

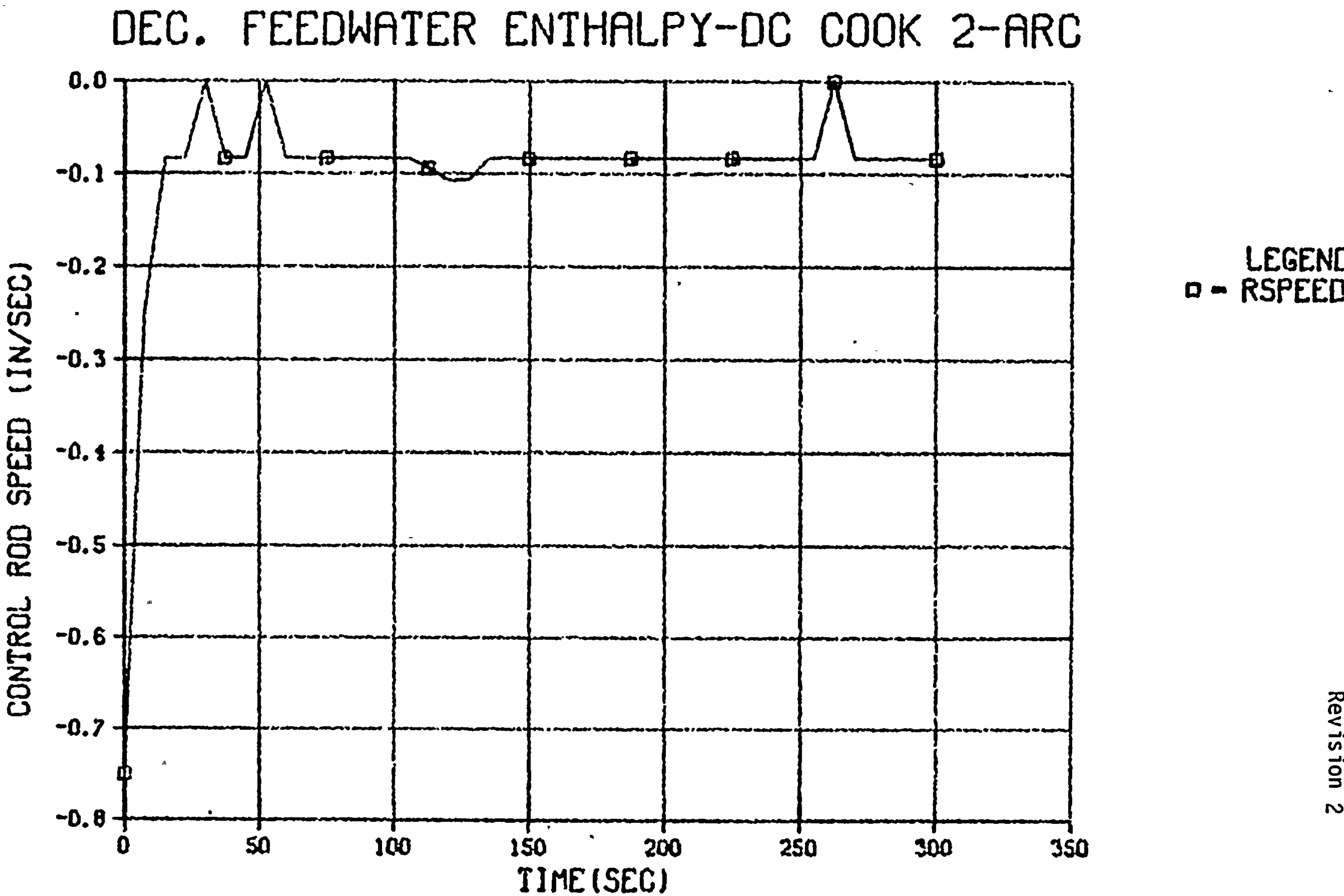

Figure 4.50 Control Rod Speed for Decreased Feedwater Heating with Automatic Rod Control

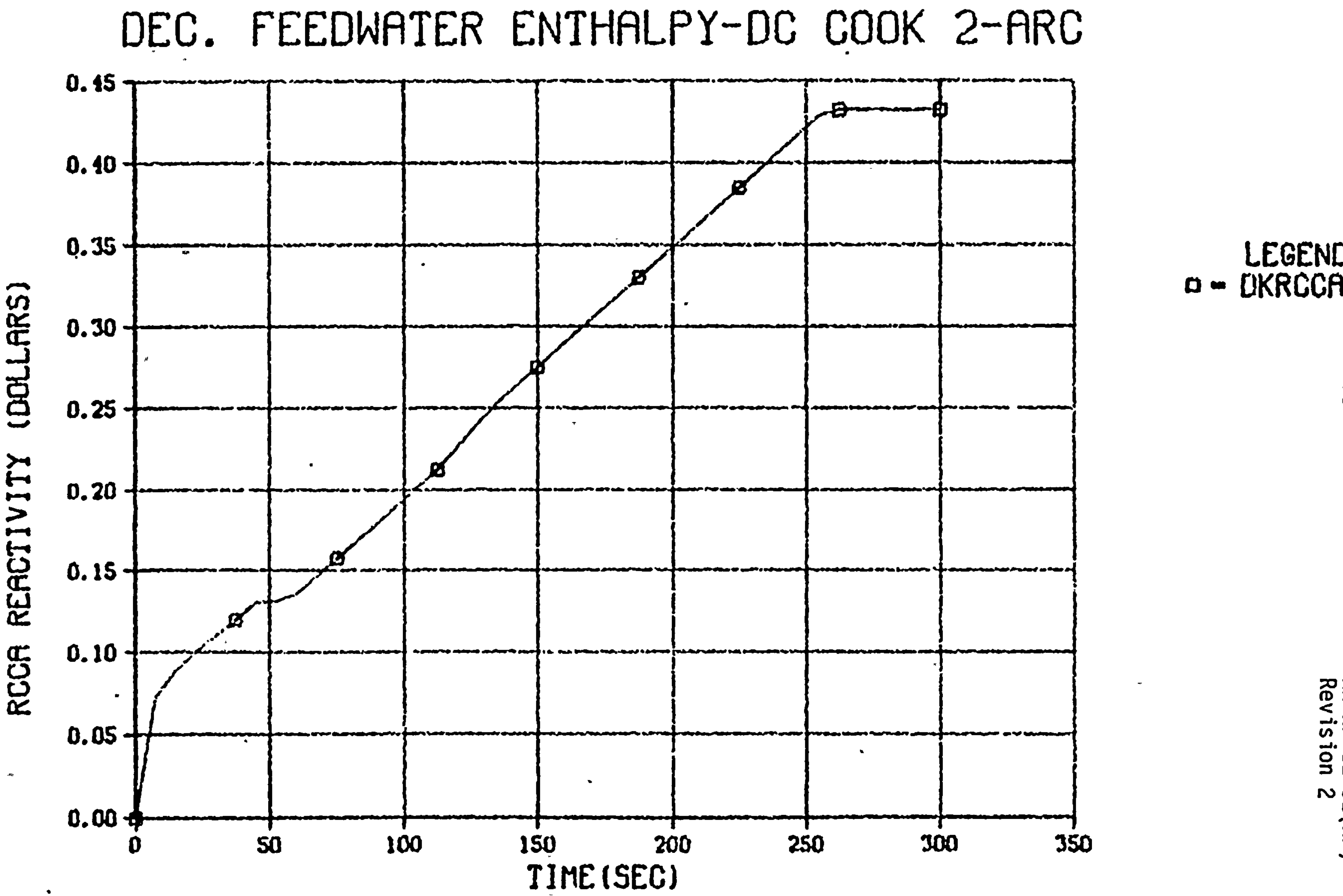

Figure 4.51 RCCA Reactivity for Decreased Feedwater Heating with Automatic Rod Control

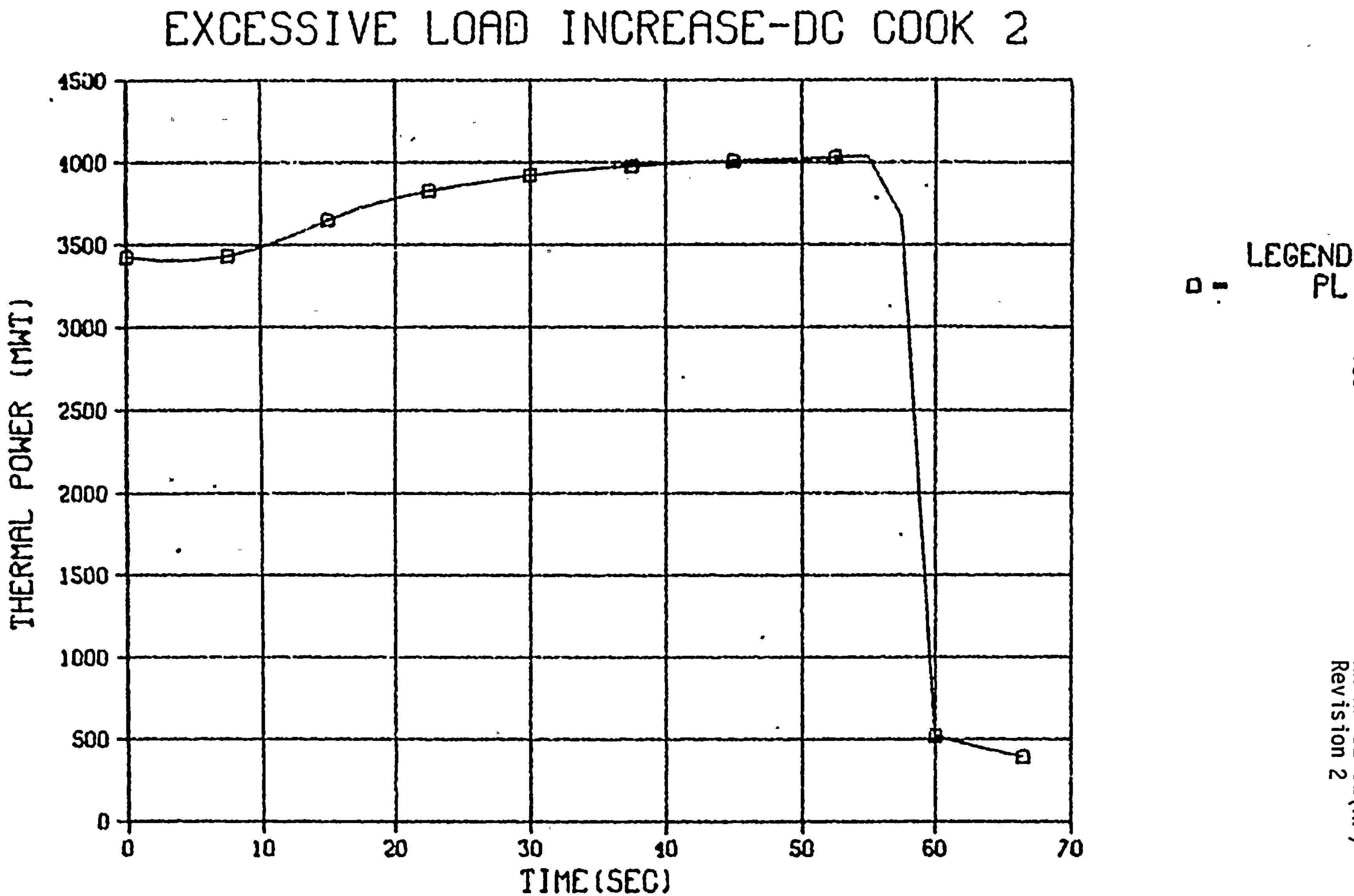

Figure 4.52 Thermal Power for Load Increase

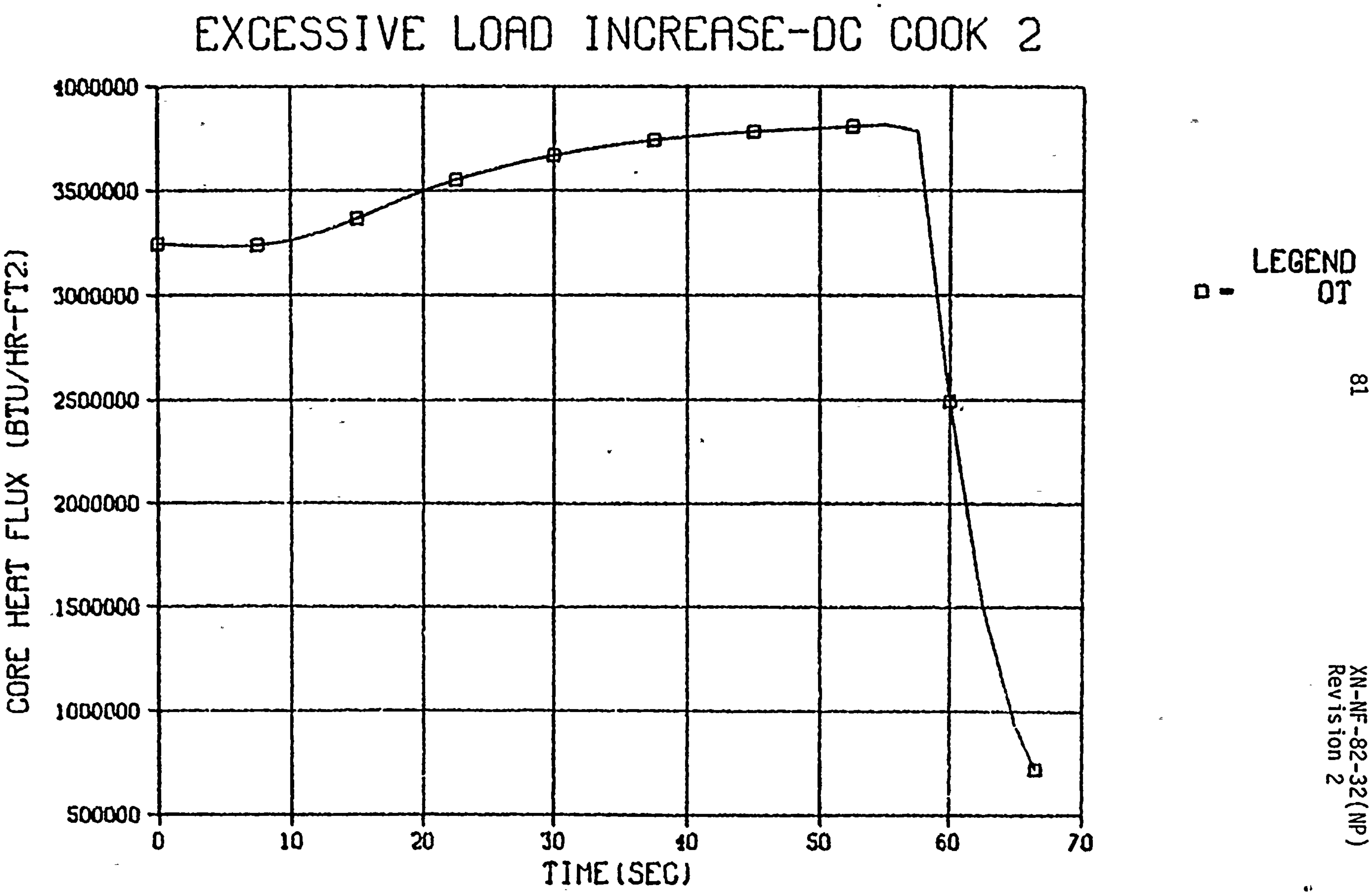

Figure 4.53 Core Heat Flux for Load Increase

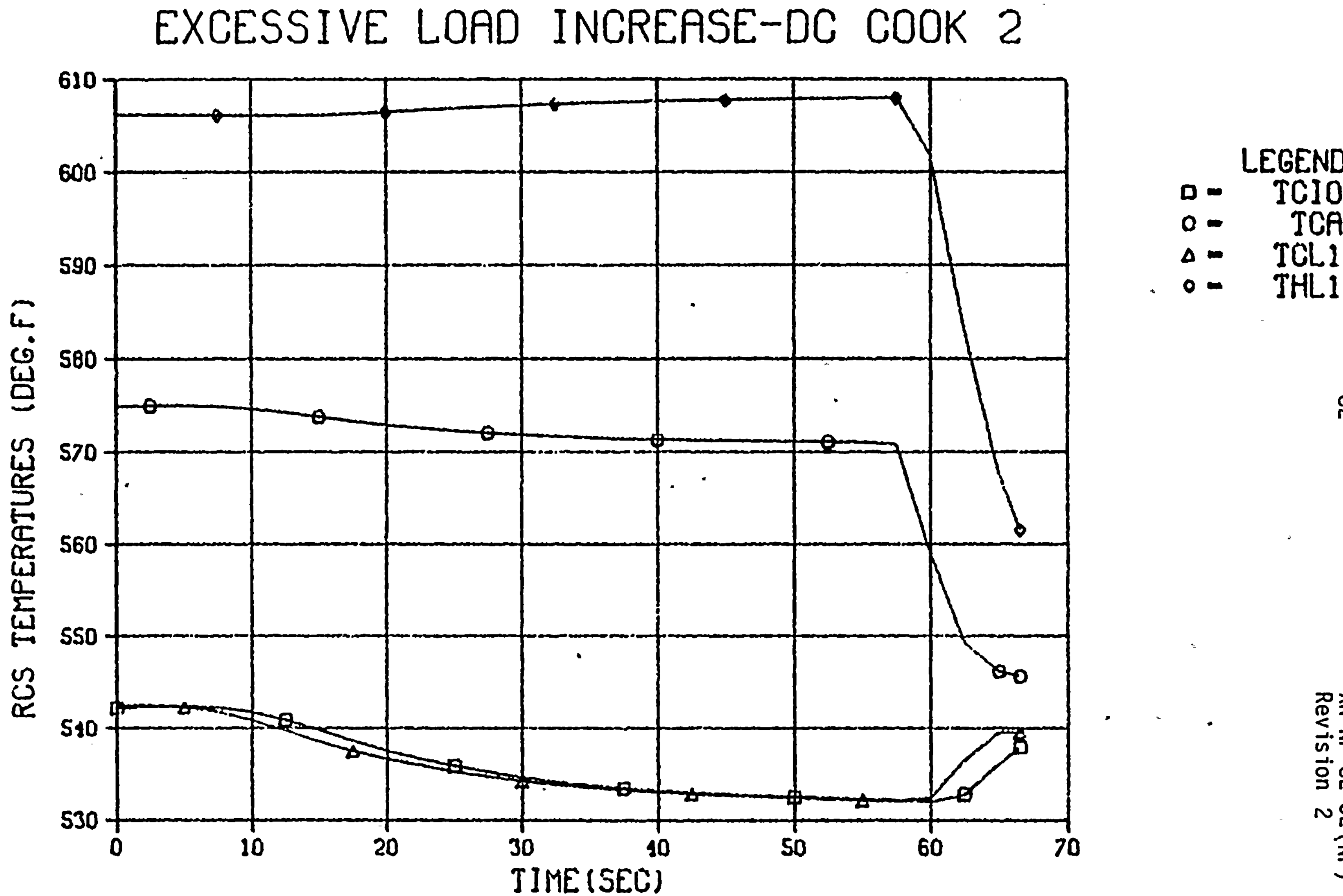

Figure 4.54 RCS Temperatures for Load Increase - Hot Leg, Core Average, Cold Leg, and Core Inlet Temperatures

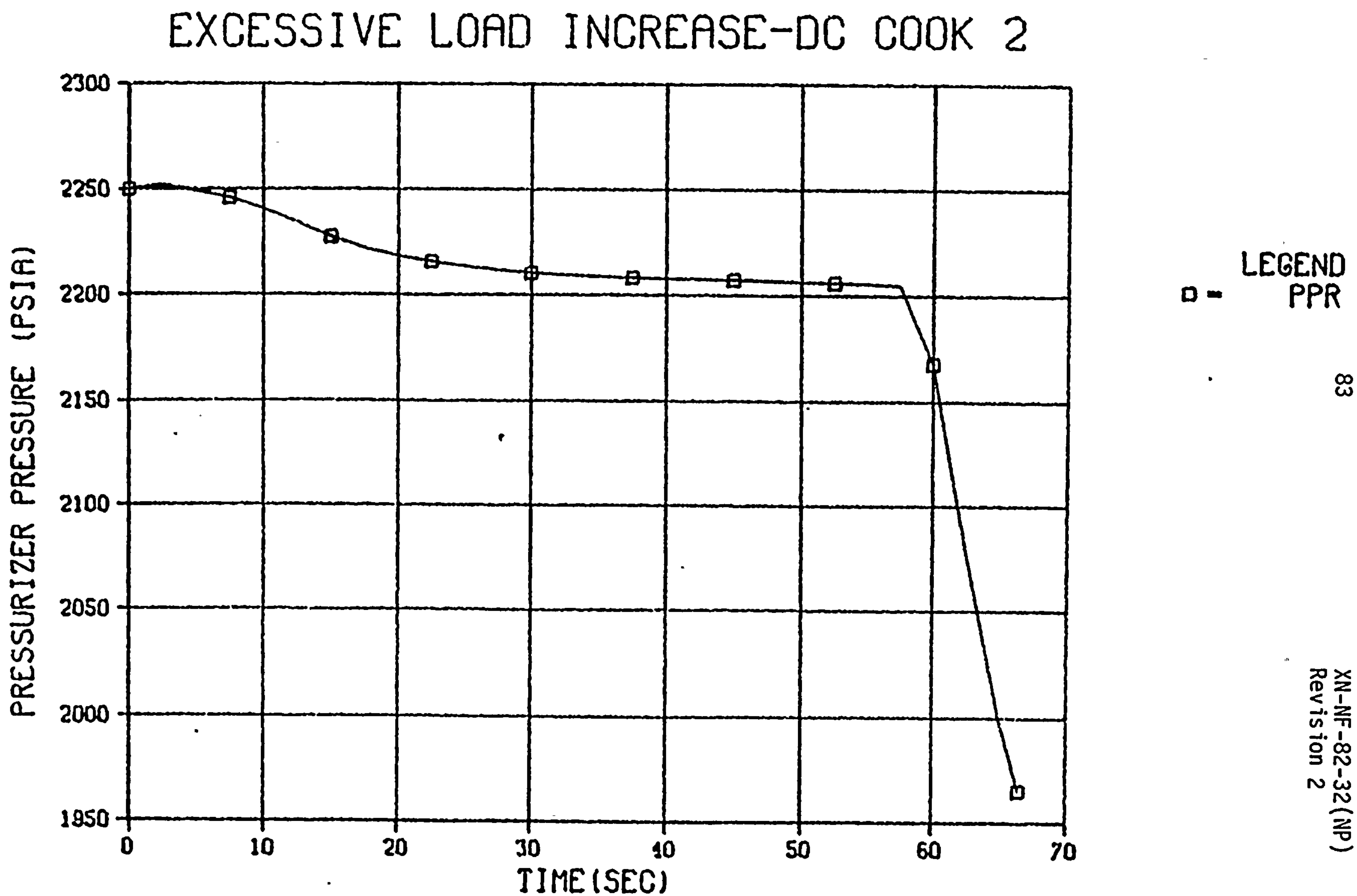

Figure 4.55 Pressurizer Pressure for Load Increase

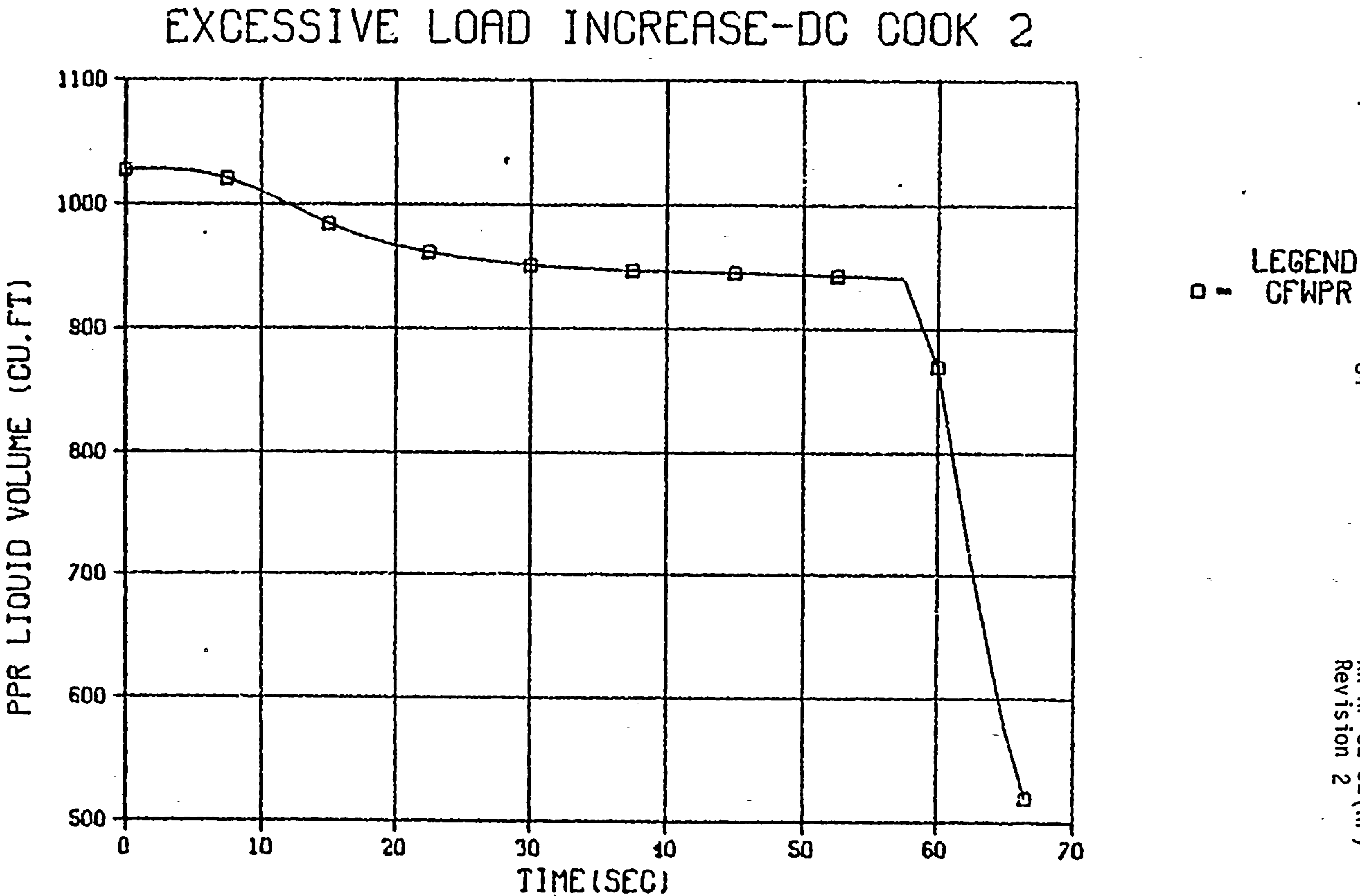

Figure 4.56 Pressurizer Liquid Volume for Load Increase

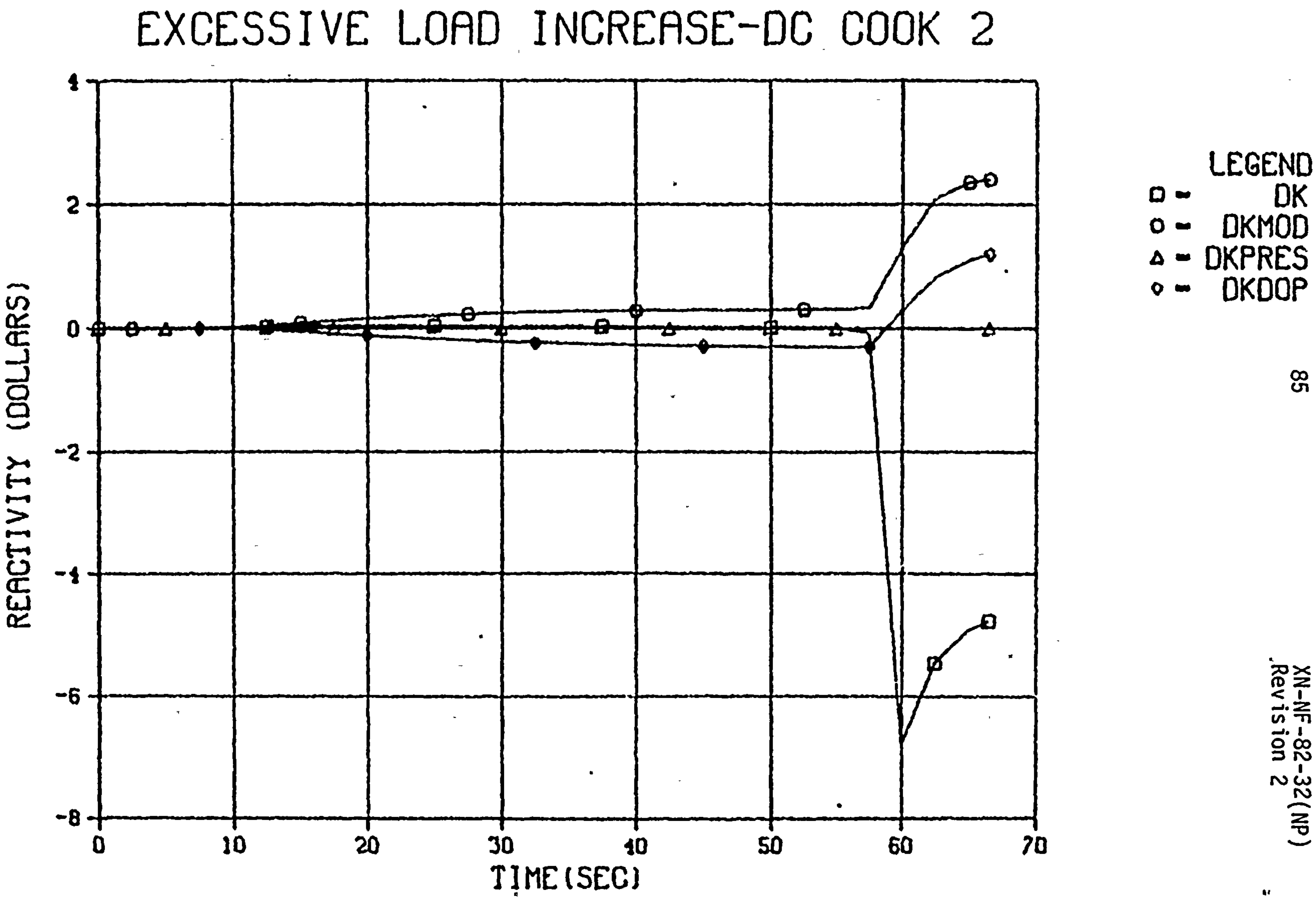

Figure 4.57 Core Reactivity for Load Increase

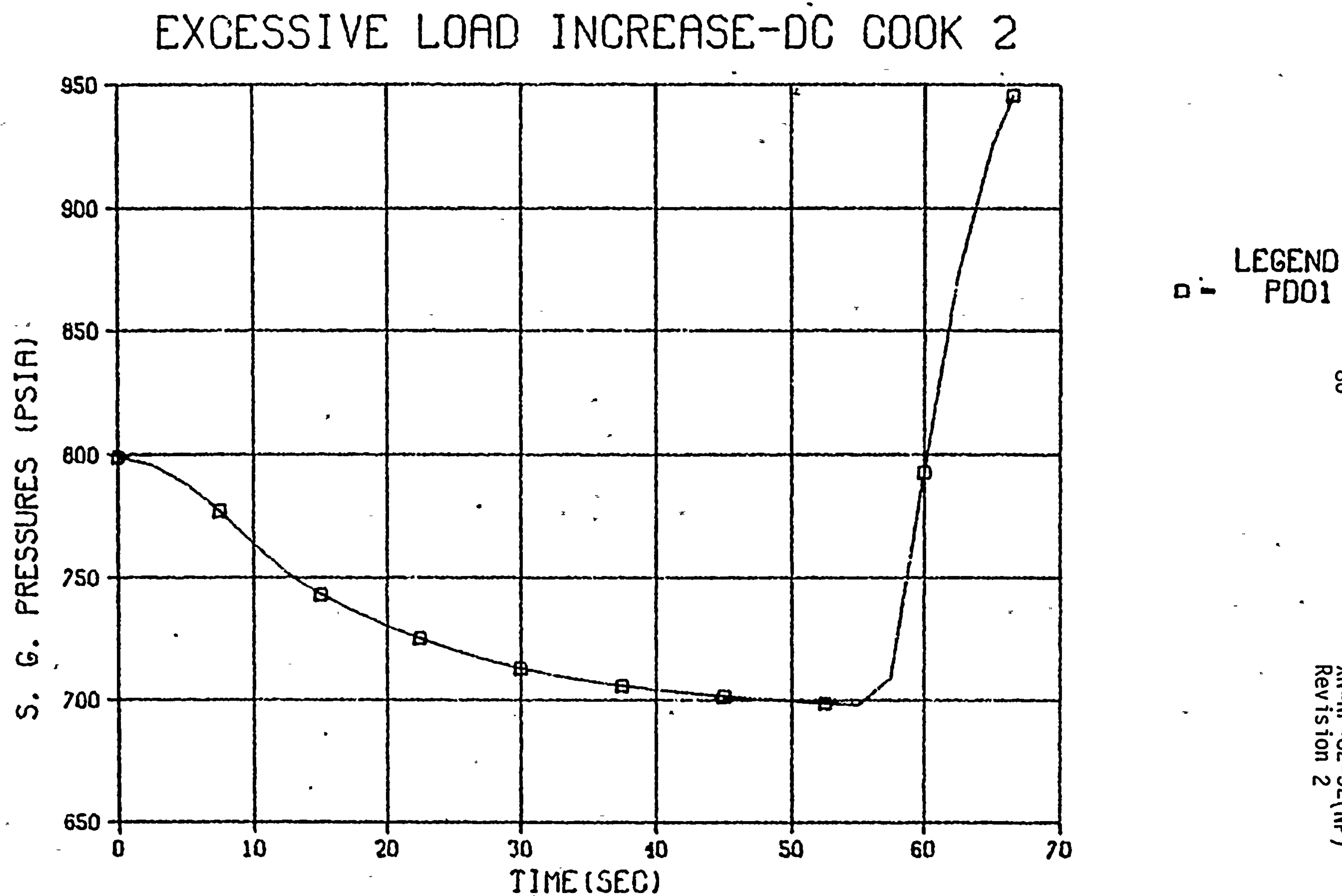

Figure 4.58 Steam Generator Pressure for Load Increase

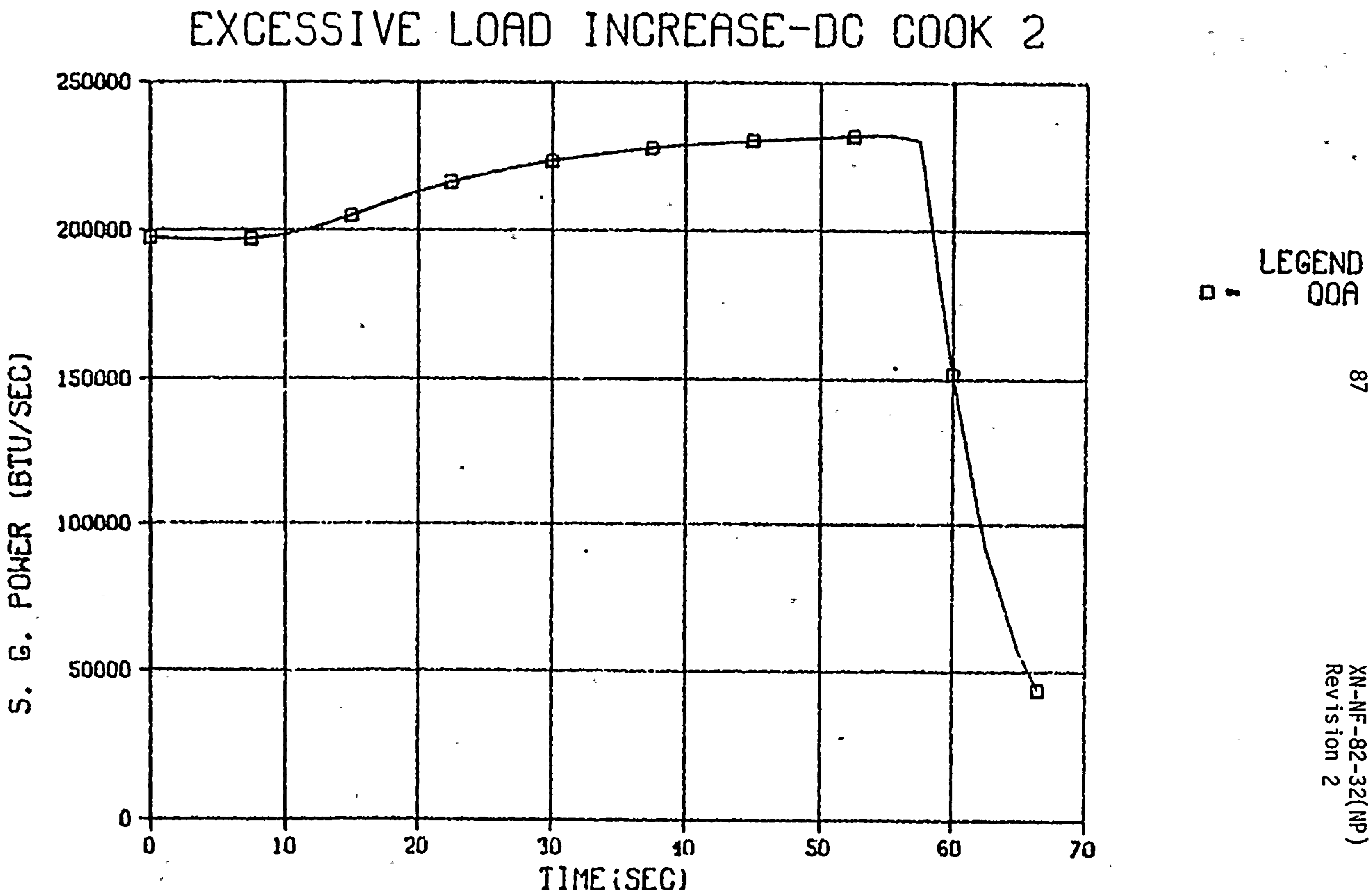

Figure 4.59 Steam Generator Power for Load Increase

EXCESSIVE LOAD INCREASE-DC COOK 2

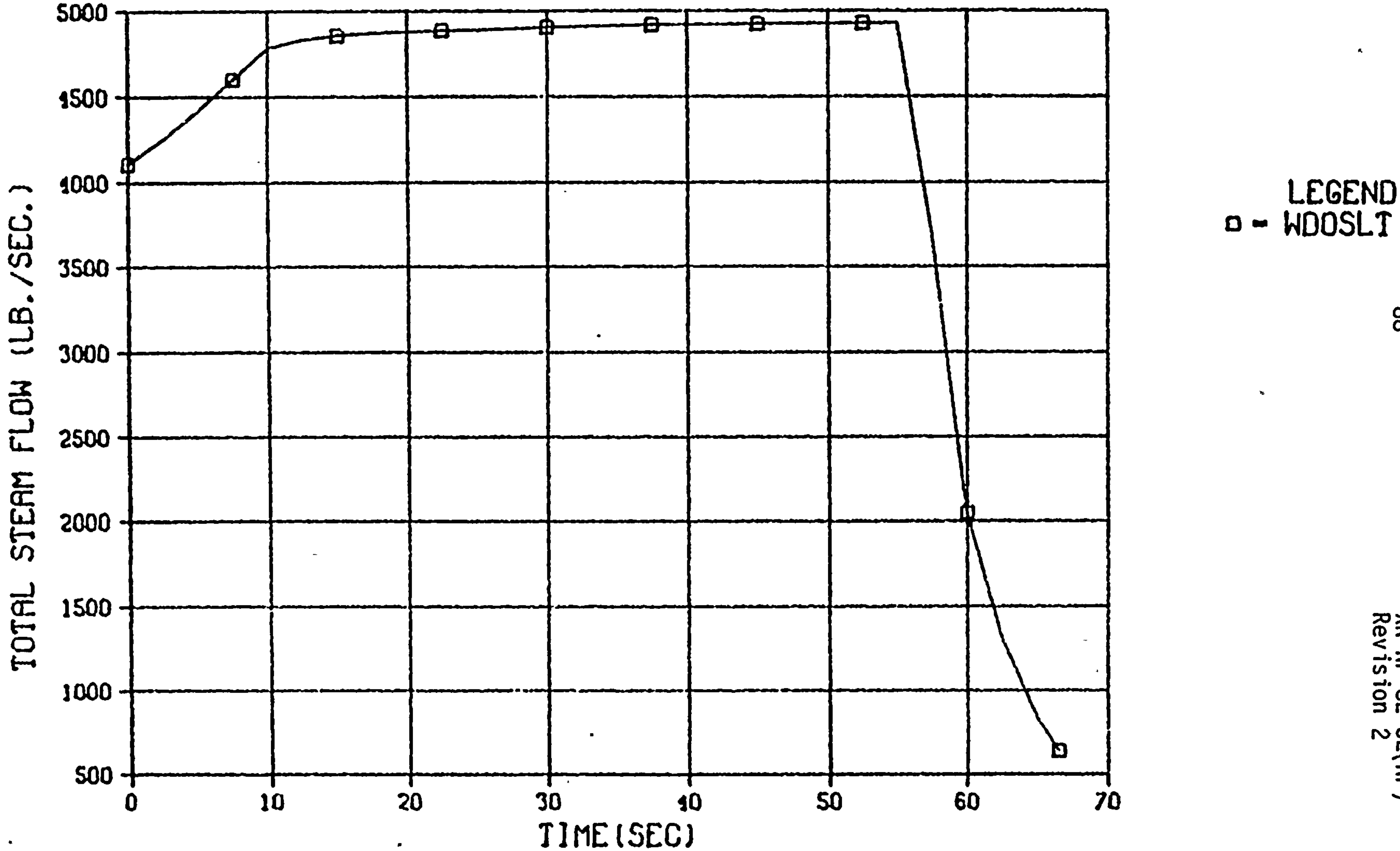

Figure 4.60 Steam Flow for Load Increase

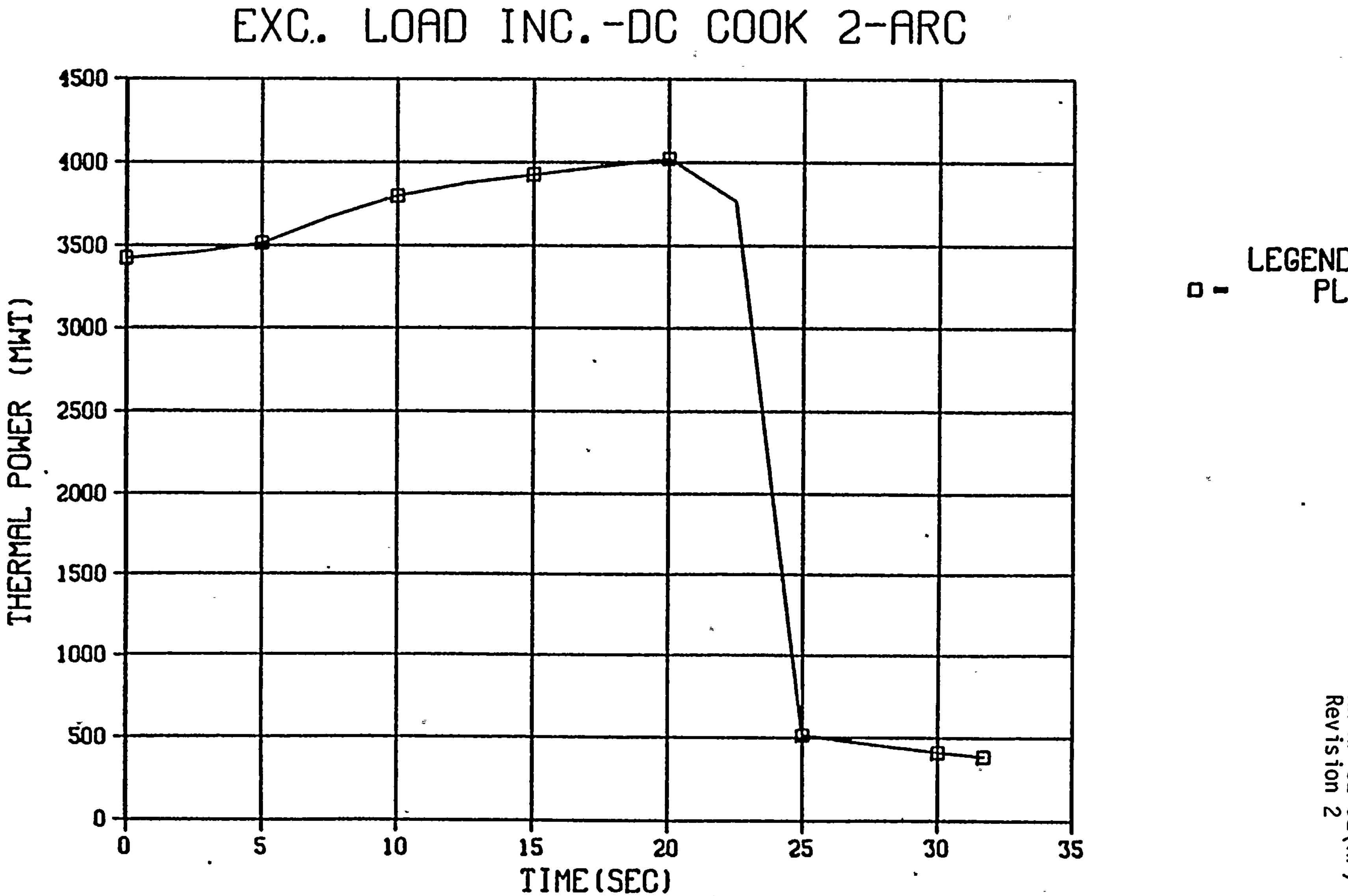

Figure 4.61 Thermal Power for Load Increase with Automatic Rod Control

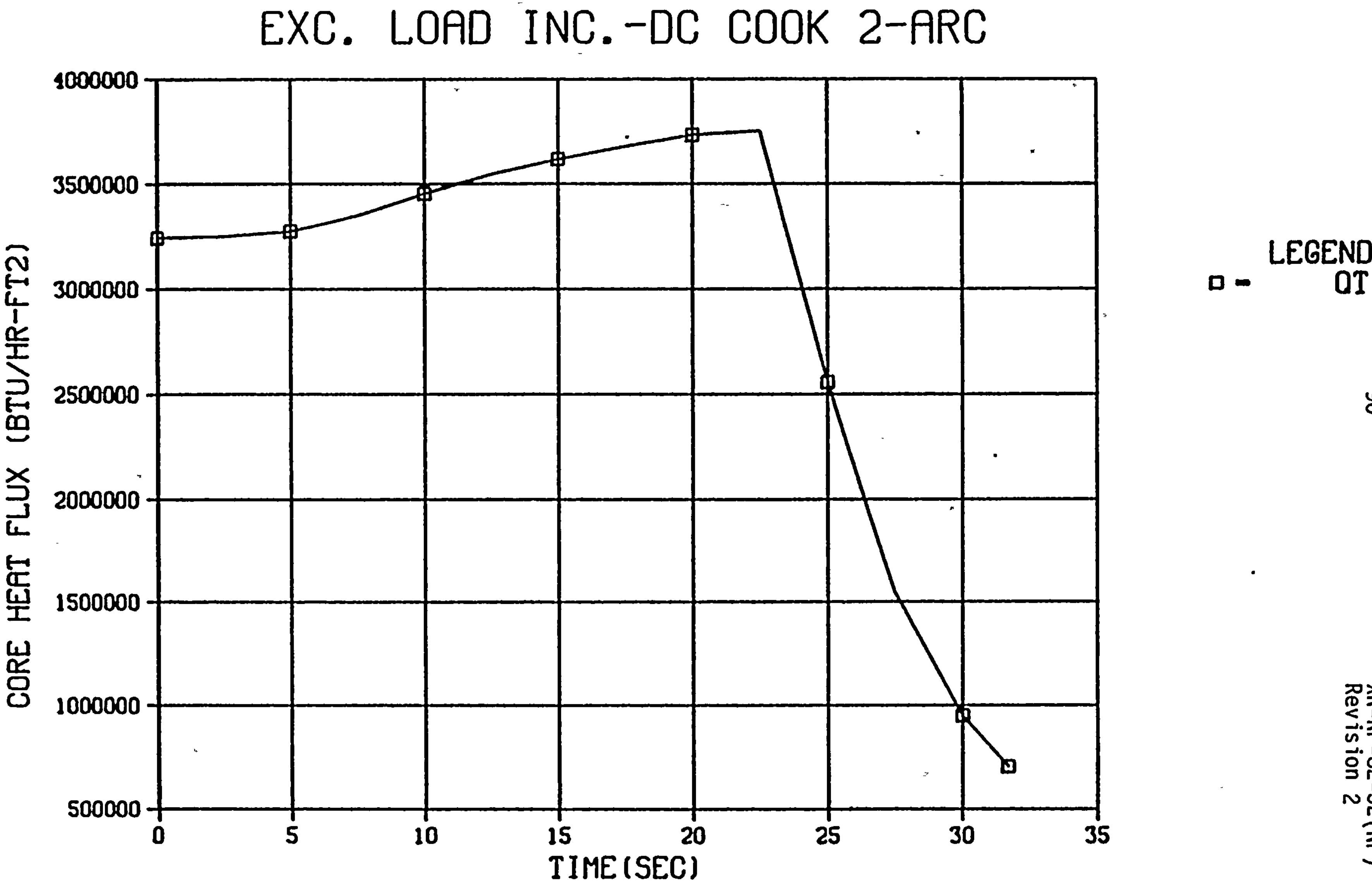

Figure 4.62 Core Heat Flux for Load Increase with Automatic Rod Control

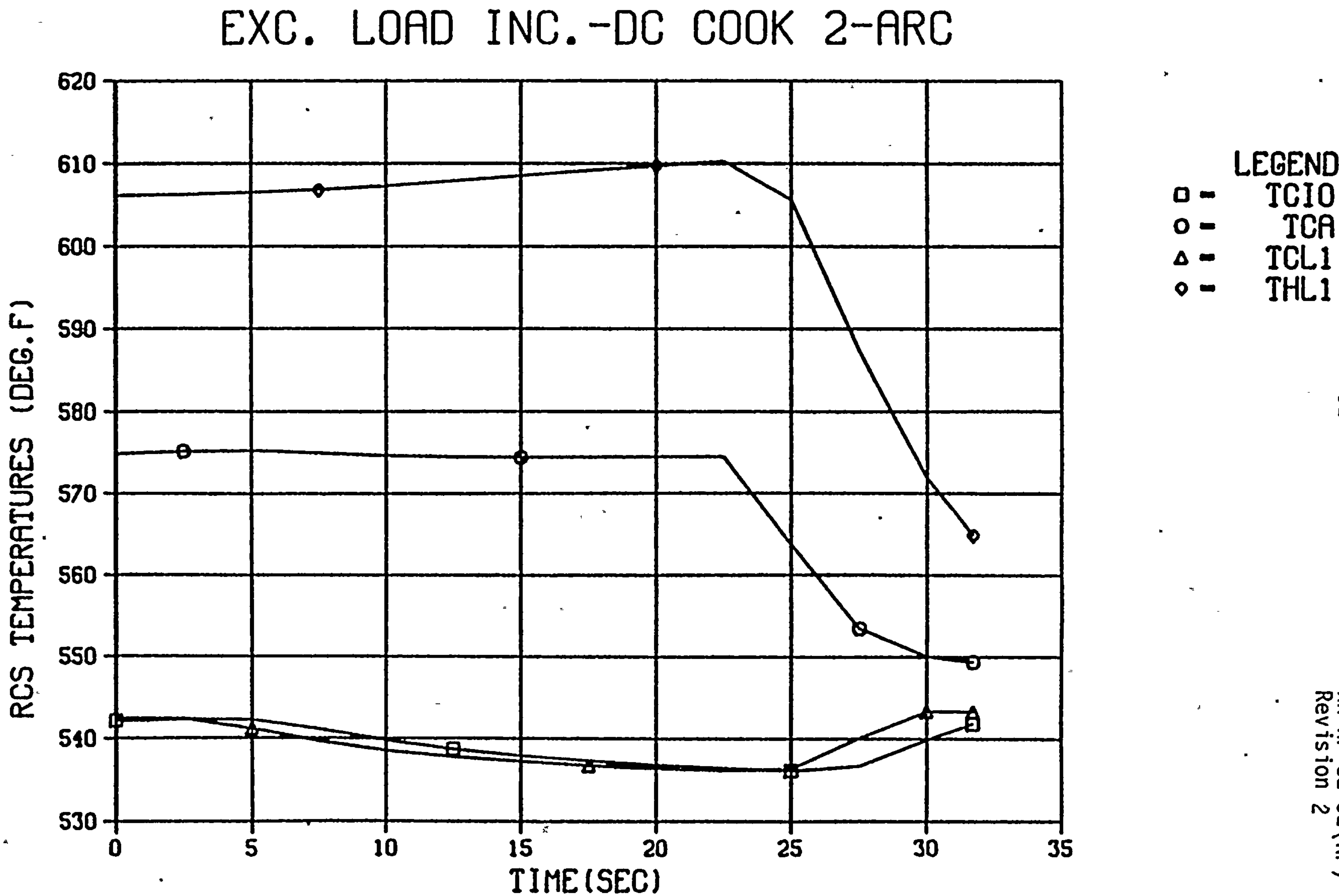

Figure 4.63 RCS Temperatures for Load Increase with Automatic Rod Control - Hot Leg, Core Average, Cold Leg, and Core

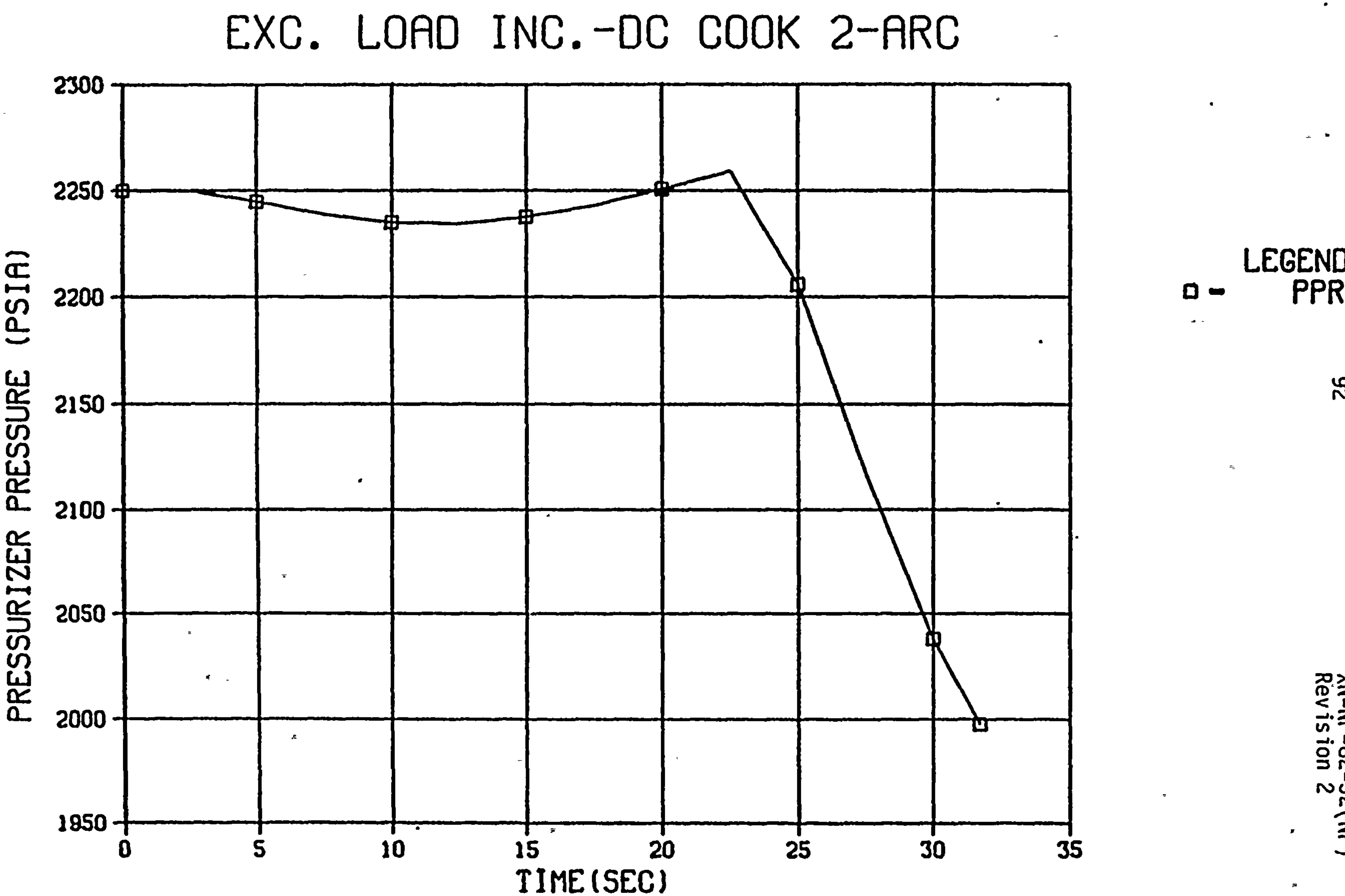

Figure 4.64 Pressurizer Pressure for Load Increase with Automatic Rod Control

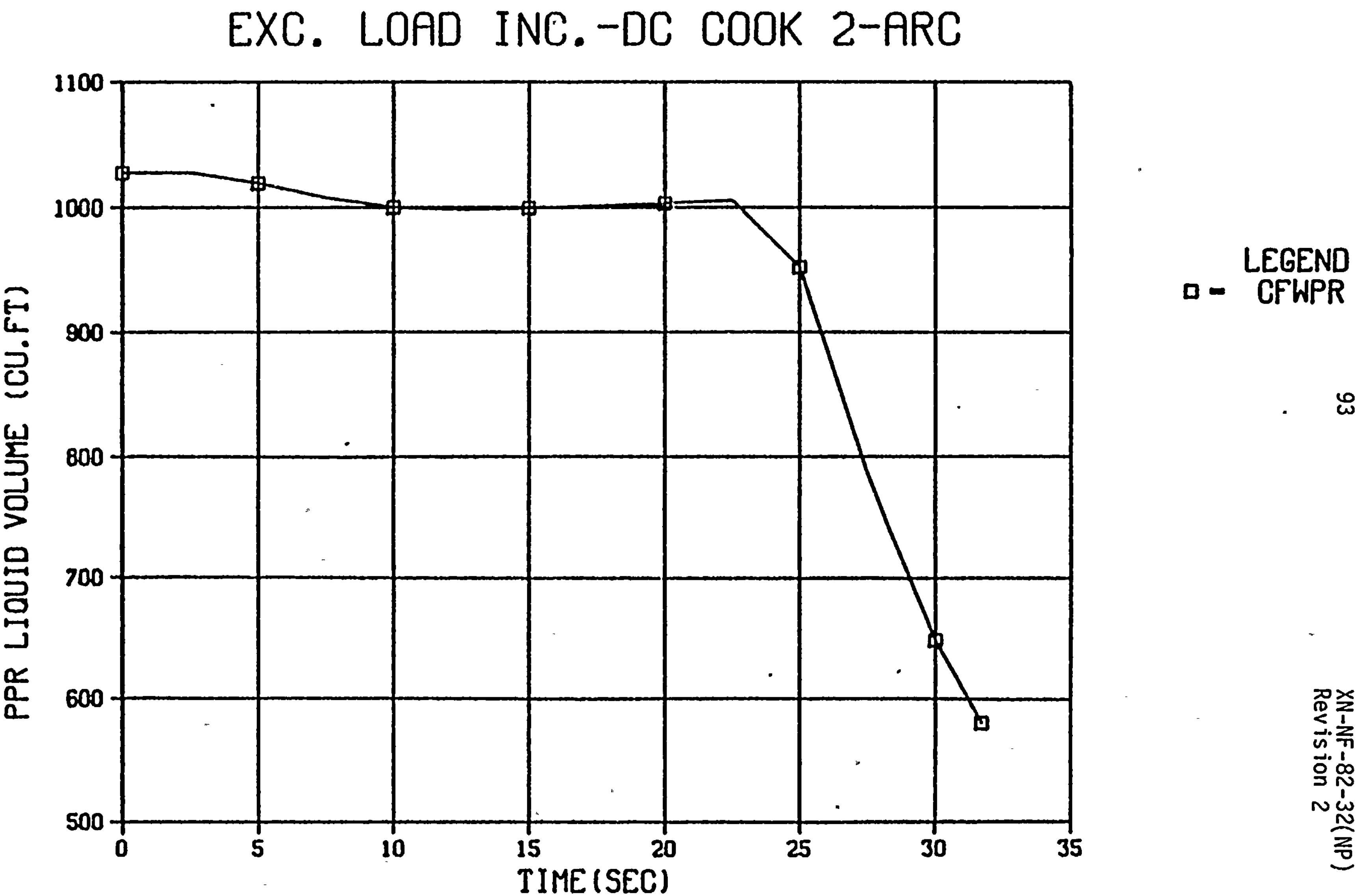

Figure 4.65 Pressurizer Liquid Volume for Load Increase with Automatic Rod Control

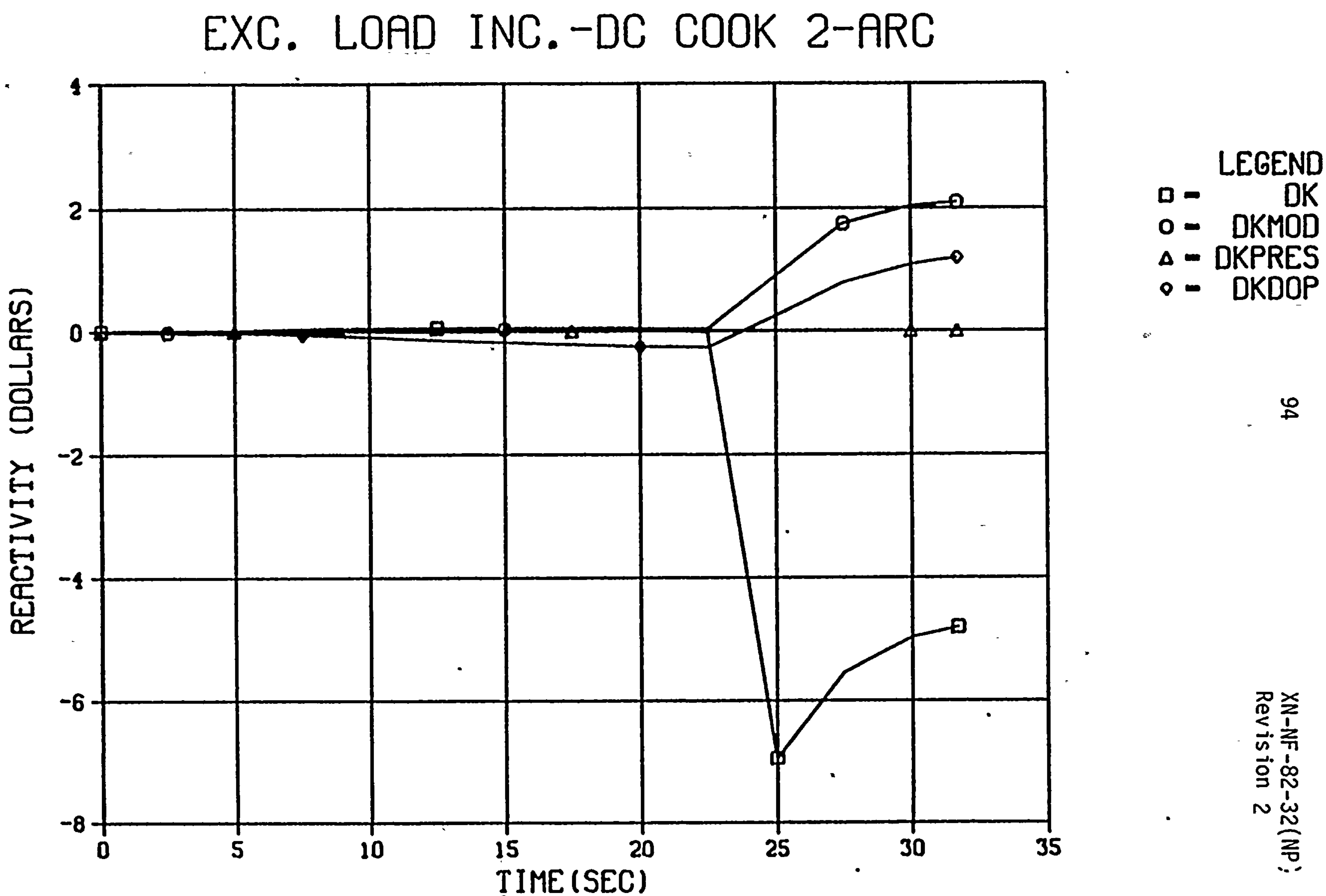

Figure 4.66 Core Reactivity for Load Increase with Automatic Rod Control

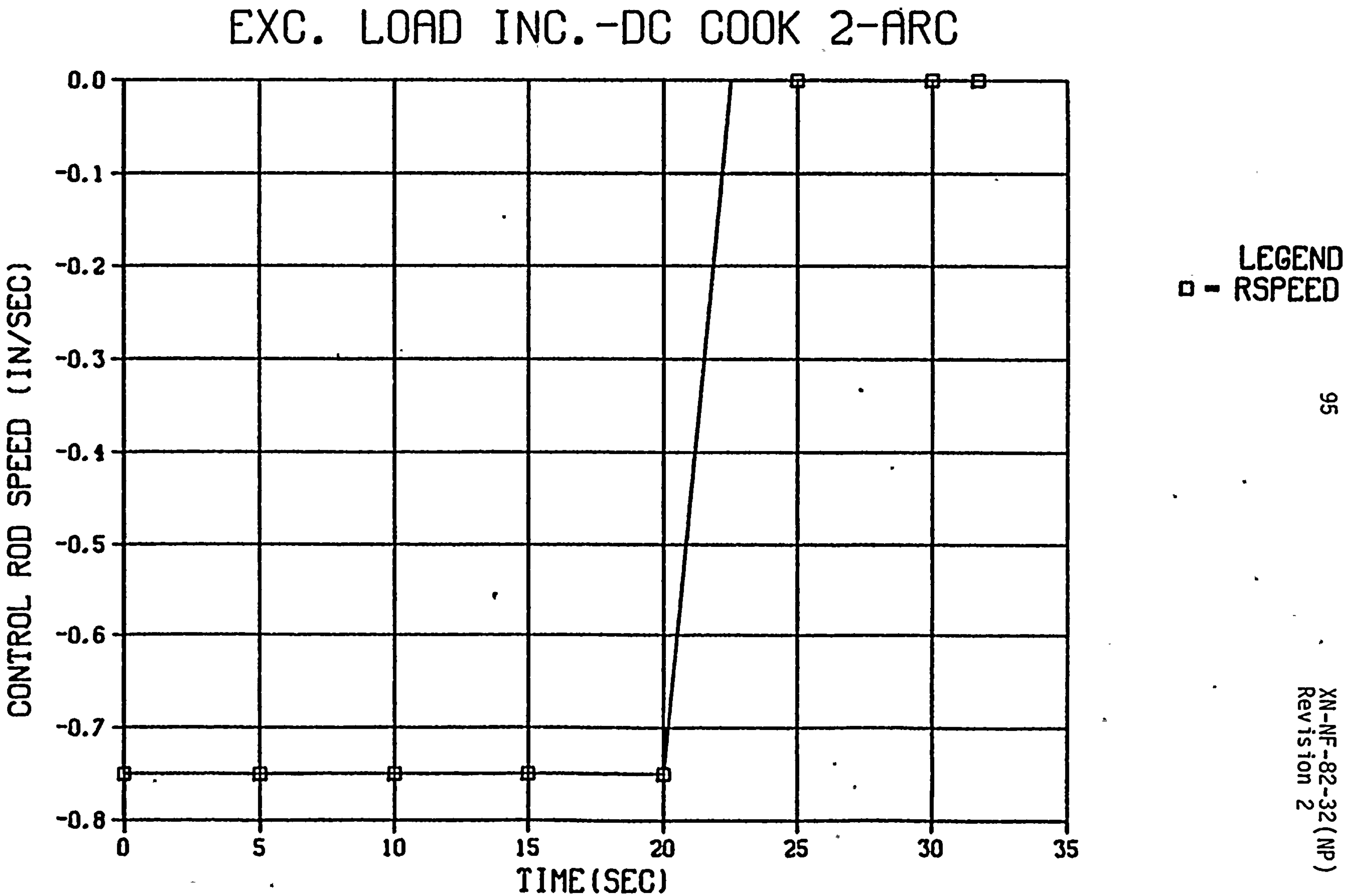

Figure 4.67 Control Rod Speed for Load Increase with Automatic Rod Control

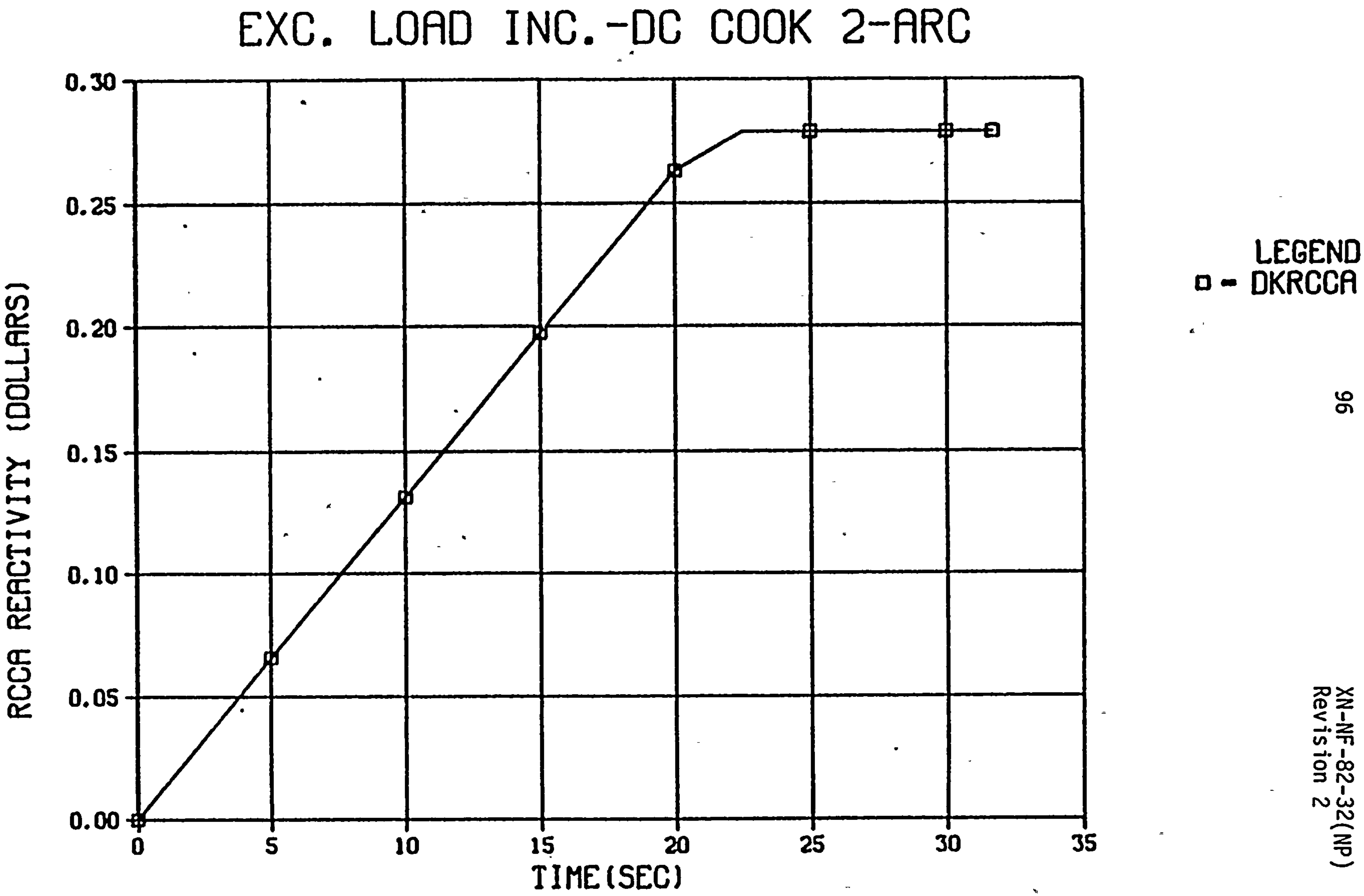

Figure 4.68 RCCA Reactivity for Load Increase with Automatic Rod Control

5.0 DISCUSSION

The transient analysis as performed by ENC for Donald C. Cook Unit 2 nuclear power plant demonstrates adequate margin to applicable fuel and vessel design limits for a mixed ENC/Westinghouse core during normal operation, anticipated operational occurrences, and postulated accidents. The following transients were analyzed using the ENC PTSPWR2 plant transient simulation model at a core power of 3425 MWt.

1) Rod Withdrawals between 8.42×10^{-3} and $7.42 \times 10^{-6}\ \Delta\rho/s$
2) Locked Primary Coolant Pump Rotor
3) Locked Primary Coolant Pump Rotor with Concurrent Loss of Offsite Power
4) Decreased Feedwater Heating
5) Excessive Load Increase
6) Loss of Load

These transients were considered because they were shown in the D.C. Cook Unit 2 FSAR(2) (reference analysis) to have the least margin to thermal margin limits. The applicable fuel and vessel design limits for the transients are a minimum DNB ratio of 1.17 calculated with the XNB critical heat flux correlation and a peak system pressure of 2750 psia. For the locked rotor accident, the fuel design criterion is that a small fraction of the core may experience boiling transition. For the locked rotor with concurrent loss of offsite power, radiological release may not exceed 10 CFR 100 limits.

Other transient events considered in the reference analysis are not reanalyzed here, either because the reference analysis results remain valid

for those events under the conditions of this analysis, or because other events which have been reanalyzed here have been shown in the reference analysis to be more limiting.

The reference analysis considered RCCA withdrawal transients initiated from a variety of core power levels equal to or less than 3391 MWt. The full power cases are shown to be the most limiting of the cases considered with respect to MDNBR. The analyzed 5% steam generator tube plugging level coupled with the loading of ENC fuel will not affect this fact, and the full power cases will continue to be the most limiting of the RCCA withdrawal events under the conditions of this analysis. Part power RCCA withdrawal cases are therefore not reanalyzed.

The results of the full power RCCA withdrawal event also bound the possible results of the Chemical and Volume Control System Malfunction transient. During this transient, reactivity is added to the core by the addition of unborated primary coolant makeup water. The system response is similar to that for the slow rod withdrawal transient analyzed in Section 4.1, with a reactivity insertion rate of about 1.0×10^{-5} Δk/sec. This insertion rate is bounded by the range of insertion rates included in this analysis.

The reference analysis of the RCCA drop transient demonstrated that, neglecting radial power distribution effects associated with the event, the MDNBR monotonically increases in time from the initial value. The MDNBR which occurs during the event may therefore be conservatively evaluated by a steady

state MDNBR calculation performed at rated initial conditions of core power, temperature, pressure, and flow, and which employs a radial peaking factor augmentation to account for the adverse core radial power distribution which characterizes the event. The radial peaking augmentation factor at 3425 MWt core power for the mixed core loading considered here is 1.2. A steady state MDNBR calculation employing this peaking augmentation factor and performed as described will result in an MDNBR well above the XNB correlation safety limit of 1.17. Since the MDNBR calculated during the transient will not exceed the steady state MDNBR thus obtained, it is concluded that the result of the RCCA drop transient at 3425 MWt meets the fuel design limit on MDNBR.

The loss of normal feedwater simulation reported in the reference analysis was performed at the Engineered Safety Features design thermal power of 105% of rated. This power level exceeds the 3425 MWt rated power assumed in this analysis. Steam generator tube plugging of 5% will result in less than 3°F higher primary coolant temperatures than shown in the reference analysis. Calculations indicate that this small increase in primary coolant temperature will not result in the expulsion of primary liquid from the pressurizer safety relief valves. Tube plugging will decrease the tendency for steam generator dryout due to reduced heat transfer effectiveness. The result of the loss of normal feedwater event presented in the reference analysis will not therefore be significantly impacted by 5% steam generator tube plugging.

The startup of an inactive loop was shown in the reference analysis to be significantly less limiting than the uncontrolled RCCA withdrawal event. Since neither the loading of ENC fuel nor the analyzed 5% steam generator tube

plugging level will alter the relative severity of these two events, the results of the uncontrolled RCCA withdrawal will continue to bound the results of the inactive loop startup. The inactive loop startup event is therefore not reanalyzed here.

The results of the loss of AC power event were shown in the reference analysis to be enveloped by the results of the four pump coastdown and loss of normal feedwater events. The flow degradation aspect of the loss of AC power event has been reanalyzed here in Subsection 4.2 as a loss of flow transient (locked rotor with loss of offsite power). Adequate long term decay heat removal is demonstrated by the loss of normal feedwater simulation reported in the reference analysis. Results of the loss of AC power event have therefore been adequately bounded by the combination of the 4 pump coastdown event reported in Subsection 4.2. The loss of normal feedwater event is discussed above..

Results of the small steam line break reported in the reference analysis are judged to remain valid for the conditions of this analysis. The event is independent of rated power, since it is initiated from hot zero power conditions. Core kinetics parameter tables employed in the reference analysis bound the core configurations considered in this analysis. The impact of steam generator tube plugging is to reduce primary to secondary heat transfer, increasing primary to secondary system temperature differences. The system temperature datum is established by the magnitude of the break flow, which is conservatively considered to be independent of tube plugging level. Primary system temperatures in this event will then be increased by tube

plugging, with consequently lesser requirements for shutdown margin. The small steam line break is therefore not reanalyzed since the reference analysis is bounding.

The main feed line break (MFLB) event was shown in the reference analysis to be independent of fuel type, since applicable fuel design limits are never approached during the event (MDNBR increases monotonically). The loss of normal feedwater (LONF) event results in greater volumetric expansion of the primary liquid than occurs during the MFLB because primary coolant expansion during the MFLB is mitigated by extraction of primary heat due to steam generator blowdown. Adequate auxiliary feedwater system capacity to prevent uncovery of the core was demonstrated in the reference analysis of the LONF event. Since the conclusion of that analysis is judged to remain valid for the 3425 MWt rating with 5% tube plugging, and bounds the primary coolant expansion and thus the potential for core uncovery in the MFLB, it is concluded that the core will remain covered throughout an MFLB initiated from the 3425 MWt level. Analysis of the MLFB event is therefore not considered.

The rod ejection transient is addressed in Reference 8.

The results of certain operational incidents are not significantly dependent on fuel type or small changes in rated power level. These include:

- RCCA Misalignment
- Turbine Generator Overspeed
- Fuel Handling Incident
- Accidental Waste Gas Release

- Radioactive Liquid Release
- Steam Generator Tube Rupture

These incidents as discussed in the reference cycle analysis were shown to be protected by administrative controls, redundancy of alarms, and/or integrity of system components. The conclusions drawn for these incidents as given in the reference analysis remain valid and these events are not reanalyzed here.

6.0 REFERENCES

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(3) XN-NF-75-21, Rev. 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," Exxon Nuclear Company, Inc., Richland, WA, September 1982.

(4) XN-NF-82-90, Supplement 1, "D.C. Cook Unit 2 Potential Radiological Consequences of Incidents Involving High Exposure Fuel," Exxon Nuclear Company, Inc., Richland, WA, February 1984.

(5) XN-74-5, Rev. 2, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," Exxon Nuclear Company, Inc., Richland, WA, May 1975.

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REACTOR AT 3425 MWt
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