ENCLOSURE 1

NORTH ANNA UNIT 1 AND 2
587.8°F REACTOR COOLANT SYSTEM
NSSS SAFETY EVALUATION SUMMARY

WESTINGHOUSE ELECTRIC CORPORATION

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A. OBJECTIVE

The objective of this report is threefold as described below:

1) To provide an update of the NSSS documentation and analyses to reflect a net reactor coolant pump (RCP) heat input of 12 MWt. Based on analysis performed subsequent to the original license, the net RCP heat input to the NSSS is 12 MWt. The license presently reflects 10 MWt.

2) To provide a description of the proposed change in the North Anna Unit 1 and 2 reactor coolant system average temperature (i.e., 580.3°F → 587.8°F) and to assess the associated impact on the NSSS. This information is to be utilized by the Virginia Electric and Power Company in their request for amendment of the plant operating licenses in accordance with the requirement of 10CFR50.90.

3) To provide a technical basis for the determination that the proposed change in reactor coolant system average temperature and updating of the net reactor coolant pump heat input does not involve an unreviewed safety question in accordance with the requirement of 10CFR50.59.

B. CONCLUSION

The proposed change in reactor coolant system average temperature and updating of net reactor coolant pump heat input have been reviewed and evaluated in detail with respect to the following:

1) NSSS Accident Analyses
2) NSSS System Adequacy
3) NSSS Component Integrity
4) NSSS/Balance of Plant Interfaces
5) Technical Specification Impact

Based on the fact that the proposed change does not result in violation of any NSSS system or equipment design criteria, and that it is not necessary to revise any of the plant operating procedures it was concluded that:

1) The probability of accidents or malfunctions of equipment previously evaluated in the FSAR is not increased.

2) The possibility of an accident or equipment malfunction of a different type than any previously evaluated in the FSAR is not created.

3) The consequences of accidents or malfunctions of equipment evaluated in the FSAR are not increased.

4) The margins of safety as defined in the bases to the plant Technical Specifications are not reduced.

Therefore this change does not reduce the plant safety margins and involves no unreviewed safety question as defined by 10CFR50.59.

1 INTRODUCTION

The Virginia Electric and Power Company (VEPCO) is pursuing a performance optimization program to maximize the electrical output of North Anna Units 1 and 2. One of the areas being investigated is the impact of increased Nuclear Steam Supply System (NSSS) steam generator outlet steam pressure on the electrical generation of the units. Increasing the NSSS steam generator outlet steam pressure at the full thermal load of 2785 MWt is accomplished by increasing the NSSS reactor coolant system average temperature.

A test was conducted at North Anna Unit 2 on October 29, 1980, to determine the effect of varying steam generator outlet steam pressure on gross electrical generation. The test revealed that at 2785 MWt a 7°F variation in NSSS reactor coolant system average temperature resulted in a 50 psi change in steam generator outlet pressure and a 5 MVA variation in gross electrical generation. This increase in gross electrical generation was solely attributable to the additional energy contained in the higher pressure steam. There was no increase in the thermal output of the NSSS.

In addition to increasing the RCS average temperature by 7.5°F to 587.8°F verification and/or confirmatory analyses were performed to demonstrate that a net reactor coolant pump heat input of 12 MWt to the NSSS does not result in any safety concerns. By incorporating the additional 2 MWt of net RCP heat input the NSSS rating is increased to 2787 MWt (currently 2785 MWt) with reactor power remaining unchanged (2775 MWt). This results in an additional 0.6 MVA of gross electrical generation.

Changing the North Anna 1 and 2 NSSS documentation to reflect a 587.8°F reactor coolant system average temperature and updating the net reactor coolant pump heat input would result in approximately 5.6 MVA increased electrical generation from each unit with no hardware modifications to the NSSS. The effort required to implement the change is to perform the

accident analyses and verify the adequacy of the NSSS systems and components relative to the regulatory codes, standards and design criteria in effect at the time of the original license.

The Westinghouse scope of effort in this report is as follows:

1) Verify margins for NSSS accident analyses.

2) Confirm NSSS systems adequacy.

3) Confirm NSSS component integrity.

4) Identify any Balance of Plant (BOP) interface revisions.

5) Cite recommended revisions to the North Anna 1 and 2 Technical Specifications.

2 COMPARISON OF PARAMETERS

The NSSS steam generator outlet steam pressure is determined by the temperatures and temperature differentials between the primary and secondary side of the steam generator tubes. The control systems of the plant are based on producing essentially "dry" steam for input to the turbine, therefore, the steam pressure for practical purposes is fixed by the temperatures on the primary side of the steam generator tubes. The steam generator steam flow is primarily dependent on the NSSS thermal power. The reactor thermal rating will remain at its currently licensed value of 2775 MWt. However, the NSSS rating would be increased from 2785 MWt to 2787 MWt to reflect the actual net reactor coolant pump heat input. At rated thermal load, increasing the average temperature on the primary side of the steam generator tubes by 7.5°F will increase the temperature of the steam on the secondary side by approximately 6.8°F, which corresponds to a 50 psi increase in steam pressure.

Table 2.1 contains a comparison of the current and proposed Reactor Coolant System (RCS) temperatures and flow rates at rated thermal power. From the table it can be seen that the RCS thermal rating, pressure and "no load" temperature remain at the current values. The core inlet volumetric flow rate has been increased to reflect the actual performance of the reactor coolant pumps. The total core inlet thermal flow rate is a conservatively low design flow utilized for thermal and hydraulic analyses (e.g., DNB evaluations). Based on North Anna Unit 1 and 2 calorimetric data the measured core inlet volumetric flow rate is 302,100 gpm with 2.8 percent of the steam generator tubes plugged. If the steam generator tube plugging level was increased to 5 percent, the measured flow would decrease by less than 1 percent. The North Anna Units employ a calorimetric - ΔT method to determine the core inlet flow rate. For this flow measurement technique the maximum uncertainty in the total flow measurement is $\pm$ 1.75 percent. Accounting for a 5 percent steam generator tube plugging level and the maximum flow measurement error of 1.75 percent, a total core inlet thermal flow rate of 285,000 gpm is conservatively low. Therefore, a flow of 285,000 gpm can be utilized as a design thermal flow rate for the proposed RCS average

temperature increase. The RCS average temperature has been increased from 580.3°F to 587.8°F. The variations in inlet temperature and temperature rises are attributable to the thermodynamic properties of compressed liquid water and the increased core inlet volumetric flow rate. Figure 2-1 graphically depicts the reactor vessel cold leg, average and hot leg fluid temperatures as a function of power level for both the current and proposed operating conditions. The variations in temperature between operation at the proposed and current parameters decreases to zero as power is reduced from rated thermal load to no load conditions.

The information presented in Table 2.1 and Figure 2-1 represent only the first level of impact on the plant parameters and operating characteristics. The remainder of this report reviews the impact of the 7.5°F increase in RCS average temperature on NSSS accident analyses, NSSS systems, NSSS components, NSSS/BOP interfaces and Technical Specifications.

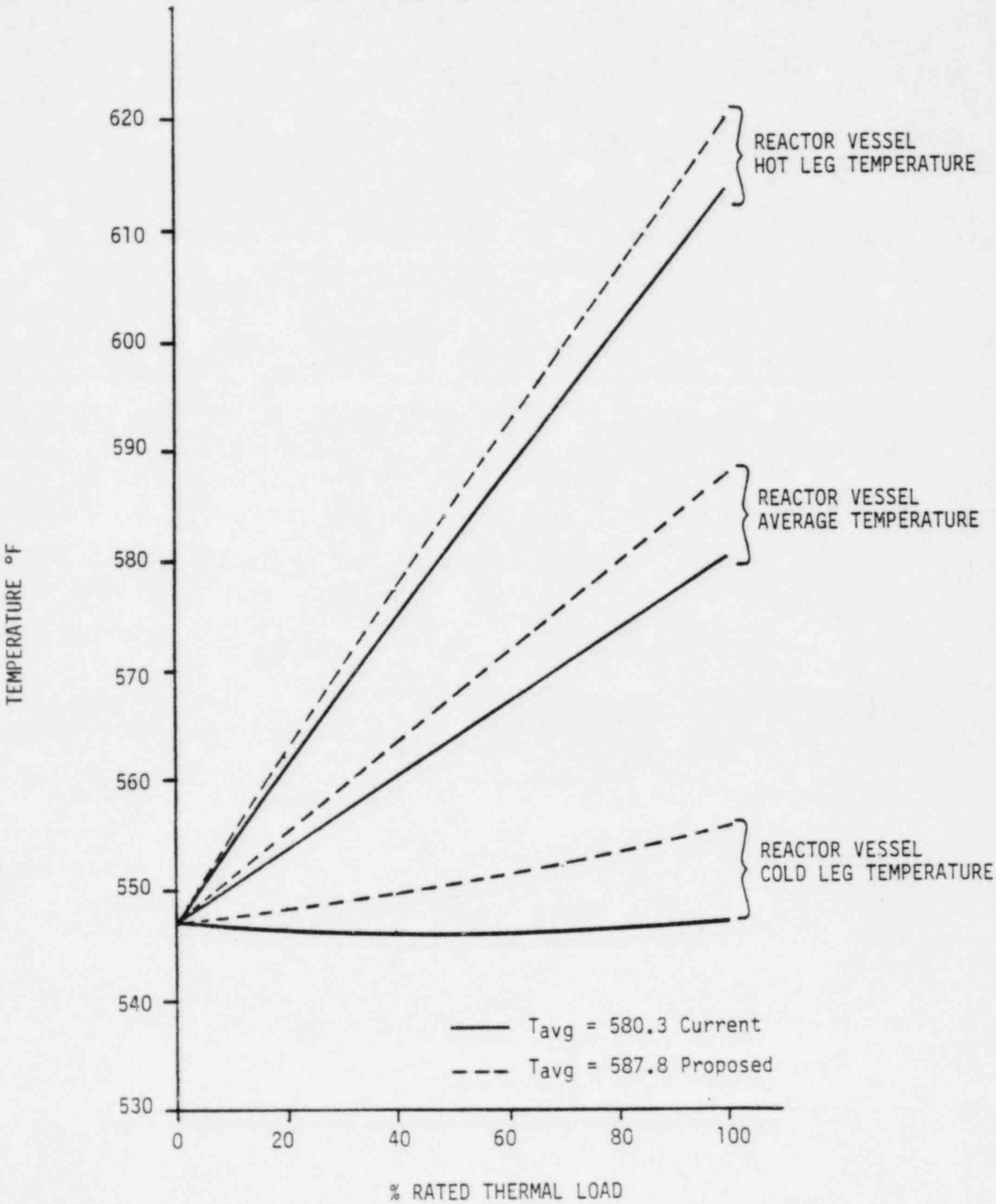
The impact of the additional 2 MWt of net reactor coolant pumps heat input was reviewed in conjunction with the 7.5°F temperature increase. In all instances, it was determined that the additional 2 MWt of pump heat had an insignificant effect and did not result in any reduction in safety margin. The impact of the increased Reactor Coolant System average temperature constitutes essentially the entire change. Therefore, all subsequent discussions in this report will only refer to the average temperature increase. However, all evaluations performed included both the increased net pump heat and the 7.5°F increase in RCS temperature.

TABLE 2.1

COMPARISON OF REACTOR COOLANT SYSTEM PARAMETERS

Thermal and Hydraulic Design Parameters	Design Conditions	
	Current	Proposed
NSSS Power, MWt	2785	2787
Net Reactor Coolant Pump Heat Input, MWt	10	12
Reactor Core Heat Output, MWt	2775	2775
System Pressure, Nominal psia	2250	2250
System Pressure, Min., Steady State, psia	2220	2220
Total Core Inlet Thermal Flow Rate, gpm	278,400	285,000
Total Core Inlet Thermal Flow Rate, lbm/hr	105.1×10^6	106.3×10^6
Core Effective Flow Rate for Heat Transfer, lbm/hr	100.4×10^6	101.5×10^6
Reactor Coolant System Temperatures, °F		
Nominal Reactor Vessel/Core Inlet	546.9	555.5
Average Rise in Vessel	66.9	64.5
Average Rise in Core	69.7	67.2
Average in Core	583.6	591.1
Average in Vessel	580.3	587.8
No Load	547.0	547.0

FIGURE 2-1
NORTH ANNA I AND II
REACTOR COOLANT TEMPERATURES
- VS -
% OF RATED THERMAL LOAD



3 NSSS ACCIDENT ANALYSES

Evaluations and analyses were performed to assess the impact of a 7.5°F increase in RCS average temperature (i.e., 580.3°F to 587.8°F) on the docketed North Anna Unit 1 and 2 postulated accident analyses. In conjunction with the RCS average temperature increase the impact of a net reactor coolant pump heat input of 12 MW_t (currently 10 MW_t reflected in the FSAR) was assessed. A safety evaluation has been performed to address the safety considerations in allowing North Anna Units 1 and 2 to operate at an RCS average temperature of 587.8°F with an NSSS power of 2787 MW_t and a corresponding reactor power of 2775 MW_t. In this study, several transients which were determined to be sensitive to an increase in Tavg were reanalyzed assuming a 900 psia secondary steam pressure, 12 MW_t net reactor pump heat, a thermal design flow of 285,000 gpm. (plant total) and a 7.5°F increase in Tavg. The reanalysis and evaluation techniques used are consistent with the methods detailed in the FSAR and subsequent safety analysis. The results of this study demonstrate that the proposed Tavg increase and revised reactor coolant pump heat input can be accommodated with margin to applicable FSAR safety limits. An assessment of the impact of these changes on the Chapter 15 FSAR transients is presented in the following discussions.

3.1 Transients Not Reanalyzed

For a number of the Chapter 15 postulated transients it was not deemed appropriate to perform a reanalysis. Each of these transients is discussed and the justification for not reanalyzing for the 7.5°F RCS average temperature increase are provided below:

A. Spurious SIS Operation at Power

The spurious operation of the safety injection system at power could be caused by operator error or a false electrical actuating signal. Following the actuation signal, the high-head safety injection pumps force highly concentrated boric acid solution in the cold legs of each loop.

Previous analytical results have shown that, because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. Thus, this accident is not a limiting event.

Although reactivity feedback parameters affect the time at which reactor trip occurs, the increased steam pressure has not affected these parameters. The results of this transient show that the DNBR continues to increase from its initial value. Thus, the conclusions presented in the North Anna FSAR remain valid.

B. Startup of an Inactive Loop

An inadvertent startup of an idle reactor coolant pump with loop stop valves open results in the injection of cold water into the core. This accident need not be addressed due to Technical Specification restrictions which prohibit power operation with a loop out of service. However, startup of an inactive loop is not a limiting transient.

C. Uncontrolled Boron Dilution at Power

The Boron Dilution transient is analyzed to ensure that adequate time is available for the operator to terminate the inadvertent addition of unborated makeup water to the RCS before losing all shutdown margin. If a boron dilution event should occur while the reactor is at power in manual control, an overpower transient may result. However, the consequences of the overpower portion of the event are bounded by those of the RCCS Bank Withdrawal at Power transient.

D. Partial Loss of Flow

A partial loss of flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump or pumps supplied by a single reactor coolant pump bus.

The results of this analyses are less limiting and therefore are bounded by the analysis of the Complete Loss of Flow transient which results from the simultaneous loss of electrical supplies to all reactor coolant pumps.

E. Single RCCA Withdrawal at Power

The single rod withdrawal accident can occur due to multiple failures of the control equipment. The result is both a positive reactivity insertion tending to increase core power and coolant temperature, and an increase in the local power density in the core region covered by the RCCA. A trip will ultimately ensue on overtemperature ΔT, however the power distribution may be sufficiently worse than the design value such that DNB will occur.

Evaluation of the transient at the power and coolant conditions at which the overtemperature ΔT trip would be expected to trip the plant shows that the upper limit for the number of rods with a DNBR less than 1.30 remains below the 5 percent assumed for the docketed analysis.

F. Small LOCA

A loss of coolant accident is the result of a rupture of the Reactor Coolant System (RCS) piping or of any line connected to the system. Ruptures of small cross section (less than 3/8 inch diameter) will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps. These pumps maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown.

The small break LOCA need not be reanalyzed since it is not the limiting accident. Also, rapid pressurization of the steam generator secondary side up to the safety valve setpoint occurs in small break LOCA's, well before core uncovery. Therefore, the effect of a 50 psi increase in the initial pressure would be negligible.

G. Steam Generator Tube Rupture

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with one percent of the fuel rods defective. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the Reactor Coolant System. In the event of a coincident loss of off-site power, or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

Steam generator tube rupture need not be reanalyzed. Studies performed and reported in engineering calculations indicate a slight benefit for an increase in initial steam pressure for steam generator tube rupture LOCA's.

Zero Power Transients

The following transients were not analyzed for the proposed 7.5°F Tavg increase because they are initiated at zero power conditions (which have not changed):

- Uncontrolled RCCA Bank Withdrawal from Subcritical
- Steamline Rupture
- Rod Ejection (HZP cases)
- Feedwater System Malfunctions (zero power case)
- Uncontrolled Boron Dilution during refueling, hot and cold shutdown

3.2 Non-LOCA Accident Reanalyses

As mentioned previously several non-LOCA transients which were determined to be sensitive to the proposed 7.5°F increase in RCS average temperature were reanalyzed. The following is a summary of those transients.

A. Reactivity and power distribution anomalies

   1. Rod Withdrawal at Power
   2. Dropped Rod(s)
   3. Rod Ejection

B. Decrease in heat removal by the secondary (heat-up transients)

   1. Loss of Load
   2. Loss of Normal Feedwater/Station Blackout
   3. Feedline rupture

C. Increase in heat removal by the secondary (cooldown transients)

   1. Feedwater system malfunctions
   2. Excessive load increase

D. Decrease in RCS flowrate

   1. Complete Loss of flow
   2. Locked Rotor

E. Decrease in Reactor Coolant Inventory

   1. Accidental Depressurization of the RCS

These transients were analyzed to determine their sensitivity to the changes in NSSS conditions associated with the 7.5°F RCS average temperature increase.

In the appendix to this enclosure, details of the analyses are presented via proposed page changes to the FSAR. In all cases, the conclusions presented in the FSAR remain valid. Note that the proposed changes do not reflect the recent FSAR update. Page revisions reflecting changes from these reanalyses will be formally incorporated in the FSAR during the next update.

In addition, the following pages of the North Anna FSAR Chapter 15 will be deleted due to the reanalysis performed for the 50 psi steam pressure increase:

Figure 15.1-8	Nuclear Power Following Trip for Input to the BLKOUT Code
Table 15.2-3 (page 15.2-92)	Natural Circulation Flow
Figure 15.2-11	

3.3 Large Break LOCA Accident Reanalysis

3.3.1 INTRODUCTION

A reanalysis of the ECCS performance for the postulated large break Loss of Coolant Accident (LOCA)[1] has been performed which is in compliance with Appendix K to 10 CFR 50. The results of this reanalysis are presented herein and are in compliance with 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors. This analysis was performed with the NRC approved (Ref. 2 , 11) 1981 version of the Wesinghouse LOCA- ECCS evaluation model. The analytical techniques used are in full compliance with 10 CFR 50, Appendix K.

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur and include such items as the core peaking factors, the containment pressure, and the performance of the emergency core cooling system (ECCS). All assumptions and initial operating conditions used in this reanalysis were the same as those used in the previously docketed LOCA-ECCS analysis (Ref. 3) with the following exceptions: 1) reactor coolant system conditions corresponding to a Tavg of 587.8°F; 2) a Thermal Design Flow of 95,000 gpm per loop was used; and 3) 5 percent steam generator tube plugging was assumed.

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[1]The reanalysis of the small break LOCA is not necessary and therefore the analysis of this accident submitted by Reference 1 remains applicable.

3.3.2 DESCRIPTION OF POSTULATED MAJOR REACTOR COOLANT PIPE RUPTURE (LOSS OF COOLANT ACCIDENT - LOCA)

A LOCA is the result of a rupture of the reactor coolant system (RCS) piping or of any line connected to the system. The system boundaries considered in the LOCA analysis are defined in the FSAR. Sensitivity studies (Reference 4) have indicated that a double-end cold leg guillotine (DECLG) pipe break is limiting. In the unlikely event of a DECLG break, a rapid depressurization of the RCS will result. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection system (SIS) signal is actuated when the appropriate setpoint is reached and the high head safety injection pumps are activated. The actuation and subsequent activation of the ECCS, which occurs with the SIS signal, assumes the most limiting single failure event. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. (It should be noted, however, that no credit is taken in the analysis for the insertion of control rods to shut down the reactor).

2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the reactor coolant system. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by

forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calcuated, consistent with Appendix K of 10 CFR 50. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms. During the refill period, it is assumed that rod-to-rod radiation is the only core heat transfer mechanism. The heat transfer between the reactor coolant system and the secondary side may be in either direction depending on the relative temperatures. For the case of continued heat addition to the secondary side, secondary side pressure increases and the main safety valves may actuate to reduce the pressure. Makeup to the secondary side is automatically provided by the auxiliary feedwater system. Coincident with the safety injection signal, normal feedwater flow is stopped by closing the main feedwater control valves and tripping the main feedwater pumps. Emergency feedwater flow is initiated by starting the auxiliary feedwater pumps. The secondary side flow aids in the reduction of reactor coolant system pressure. When the reactor coolant system depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. A conservative assumption is then made that the injected accumulator water bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50. In addition, the reactor coolant pumps are assumed to be tripped at the initiation of the

accident and effects of pump coastdown are included in the blowdown analysis.

The water injected by the accumulators cools the core and subsequent operation of the low head safety injection pumps supplies water for long term cooling. When the RWST is nearly empty, long term cooling of the core is accomplished by switching to the recirculation mode of core cooling, in which the spilled borated water is drawn from the containment sump by the low head safety injection pumps and returned to the reactor vessel.

The containment spray system and the recirculation spray system operates to return the containment environment to a subatmospheric pressure.

The large break LOCA transient is divided, for analytical purposes, into three phases: blowdown, refill, and reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of inter-related computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339(Ref. 5). This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with

10 CFR 50, Appendix K. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in WCAP-8306 (Ref, 6), WCAP-8326(Ref. 7), WCAP-8171(Ref. 8), and WCAP-8305(Ref. 9), respectively. These codes are able to assess whether sufficient heat transfer geometry and core amenability to cooling are preserved during the time spans applicable to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.

SATAN-VI is used to determine the RCS pressure, enthalpy, and density, as well as the mass and energy flow rates in the RCS and steam generator secondary, as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator mass and pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end of blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow rates that are assumed to be vented to the containment. Since the mass flow rates to the containment depends upon the core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. WREFLOOD is also linked to the LOCTA-IV code in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature.

LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel and clad temperature of the hottest rod in the core. The input to LOCTA-IV consists of appropriate thermal-hydraulic output from SATAN-VI and WREFLOOD and conservatively selected initial RCS operating conditions. These initial conditions are summarized in Table 1 and Figure 1. (The axial power shape of Figure 1 assumed for LOCTA-IV is a cosine curve which has been previously verified(Ref. 10) to be the shape that produces the maximum peak clad temperature).

The COCO code, which is also used in the LOCA analysis, calculates the containment pressure. Input to COCO is obtained from the mass and energy flow rates assumed to be vented to the containment as calculated by the SATAN-VI and WREFLOOD codes. In addition,

conservatively chosen initial containment conditions and an assumed mode of operation for the containment cooling system are input to COCO. These initial containment conditions and assumed modes of operation are provided in Table 2.

3.3.3 DISCUSSION OF SIGNIFICANT INPUT

Significant differences in input between this analysis and the previously docketed analysis are delineated in Section 1.0 and discussed in more detail below. The changes made in the analysis are consistent with those made in performing the Reference 3 analysis with the 1981 model. Additional changes were made to reflect system conditions associated with an increase in plant Tavg. The steam generator tube plugging level was changed to 5 percent in order to be consistent with the assumptions for the non-LOCA analyses and to reduce the impact on total peaking factor, FQ.

The notable change for this analysis is the use of the plant conditions corresponding to operation with a Tavg of 587.8°F. The appropriate changes in initial plant fluid system conditions input were incorporated in the analysis. This includes an increase in assumed Thermal Design Flow to 95,000 gpm per loop, and an increase in steam generator secondary side steam pressure resulting from the increased Tavg.

When the above changes were incorporated into the analysis, it was found that the assumed heat flux hot channel factor could be kept at the 2.20 value used in the Reference 3 analysis and still ensure compliance with the 10 CFR 50.46 acceptance criteria.

3.3.4 RESULTS

Tables 1 and 2 and Figure 1 present the initial conditions and modes of operation that were assumed in the analysis. Table 3 presents the time sequence of events and Table 4 presents the results for the double-ended cold leg guillotine break (DECLG) for the CD=0.4 discharge coefficient. All previous LOCA-ECCS submittals for the North Anna units have resulted in the CD=0.4 discharge coefficient being the limiting break size. The applicability of this conclusion (i.e. CD=0.4 is the limiting break size) for this analysis was explicitly verified. Consequently, only the results of the most limiting break size are presented in the figures and remaining tables in this submittal. The current analysis resulted in a limiting peak clad temperature of 2174.8°F, a maximum local cladding oxidation level of 7.60%, and a total core metal-water reaction of less than 0.3%. The detailed results of the LOCA reanalysis are provided in Tables 3 through 6 and Figures 2 through 18.

3.3.5 CONCLUSIONS

For breaks up to and including the double-ended severance of a reactor coolant pipe and for the operating conditions specified in Table 1 and 2, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel rod clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and the long-term decay heat is removed for an extended period of time.

3.3.6 REFERENCES

1. Final Safety Analysis Report, North Anna Power Station, Units 1 and 2, Virginia Electric and Power Company.

2. Letter from J. R. Miller(NRC) to E. P. Rhae(Westinghouse), dated December 1, 1981.

3. Letter from R. H. Leasburg(Vepco) to H. R. Denton(NRC), Serial No. 490, August 16, 1982.

4. Buterbaugh, T. L., Johnson, W. J., and Kopelic, S. D., "Westinghouse ECCS Plant Sensitivity Studies," WCAP-8356, July 1974.

5. Bordelon, F. M., et. al., "Westinghouse ECCS Evaluation Model-Summary," WCAP-8339, July, 1974.

6. Bordelon, F. M., et.al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8306, June 1974.

7. Bordelon, F. M., and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8326, June 1974.

8. Kelly, R. D., et. al., "Calculation Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8171, June 1974.

9. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974.

10. Letter from C. M. Stallings(Vepco) to E. G. Case (NRC), Serial No. 092, February 17, 1978.

11. Eicheldinger C., "Westinghouse ECCS Evaluation Model - 1981 Version", WCAP-9220-P-A Revision I (Proprietary Version), WCAP-9221-A Revision I (Non-Proprietary Version), February, 1982.

TABLE 1

INITIAL CORE CONDITIONS ASSUMED FOR THE
DOUBLE-ENDED COLD LEG GUILLOTINE BREAK (DECLG)

CALCULATIONAL INPUT

Core Power (MWt, 102% of)	2775
Peak Linear Power (kw/ft, 102% of)	11.98
Heat Flux Hot Channel Factor (F_Q)	2.20
Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)	1.55
Accumulator Water Volume (ft^3, each)	1025
Reactor Vessel Upper Head Temperature Equal to Thot	

LIMITING FUEL REGION AND CYCLE	CYCLE	REGION
Unit 1	ALL	ALL Regions
Unit 2	ALL	ALL Regions

TABLE 2

CONTAINMENT DATA

NET FREE VOLUME	1.916×10^6 ft^3
INITIAL CONDITIONS[1]	
Pressure	9.5 psia
Temperature	90°F
RWST Temperature	35°F
Outside Temperature	-10°F
SPRAY SYSTEM[1]	
Number of Pumps Operating	2
Runout Flow Rate (per pump)	2000 gpm
Time in which spray is effective	59 secs
STRUCTURAL HEAT SINKS[1]	
Thickness (In)	Area (Ft^2), w/uncertainty
6 Concrete	8,393
12 Concrete	62,271
18 Concrete	55,365
24 Concrete	11,591
27 Concrete	9,404
36 Concrete	3,636
.375 Steel, 54 Concrete	22,039
.375 Steel, 54 Concrete	28,933
.500 Steel, 30 Concrete	25,673
26.4 Concrete, .25 Steel, 120 Concrete	12,110
.407 Stainless Steel	10,527
.371 Steel	160,328
.882 Steel	9,894
.059 Steel	60,875

[1]See the response to Comment S6.106 of the FSAR for a detailed breakdown of the containment heat sinks and for justification of the other input parameters used to calculate containment pressure.

TABLE 3

TIME SEQUENCE OF EVENTS

	DECLG CD=0.4 (Sec)
Start	0.0
Reactor Trip	0.75
S. I. Signal	2.15
Acc. Injection	16.20
Pump Injection	27.15
End of Bypass	31.00
End of Blowdown	31.00
Bottom of Core Recovery	44.55
Acc. Empty	55.05

TABLE 4

RESULTS FOR DECLG

	CD=0.4
Peak Clad Temp, °F	2174.8
Peak Clad Location, Ft.	7.25
Local Zr/H20 RXN (max), %	7.60
Local Zr/H20 Location, Ft.	7.5
Total Zr/H20 RXN, %	<0.3
Hot Rod Burst Time, sec.	39.20
Hot Rod Burst Location, Ft.	6.0

TABLE 5

REFLOOD MASS AND ENERGY RELEASES

DECLG (CD= 0.4)

TIME(SEC)	TOTAL MASS FLOWRATE (LB/SEC)	TOTAL ENERGY FLOWRATE (10 BTU/SEC)
44.552	0.0	0.0
45.677	0.675	0.0088
51.353	35.37	0.4603
61.396	75.51	0.9428
77.596	162.23	1.174
97.296	263.19	1.384
119.696	276.33	1.351
144.496	283.20	1.301
202.496	296.28	1.196
280.496	315.28	1.089
477.646	350.53	0.8803

TABLE 6

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT

DECLG, CD=0.4

TIME(SEC)	MASS FLOWRATE* (LBM/SEC)
0.00	4010.1
1.01	3622.4
3.01	3105.5
5.01	2762.4
7.01	2510.0
10.01	2225.6
15.01	1897.5
20.01	1673.8
25.01	1518.8
30.01	1559.4 **

*For energy flowrate multiply mass flowrate by a constant of 59.60 BTU/lbm.

**For energy flowrate at this time multiply mass flowrate by 54.00 BTU/lbm.

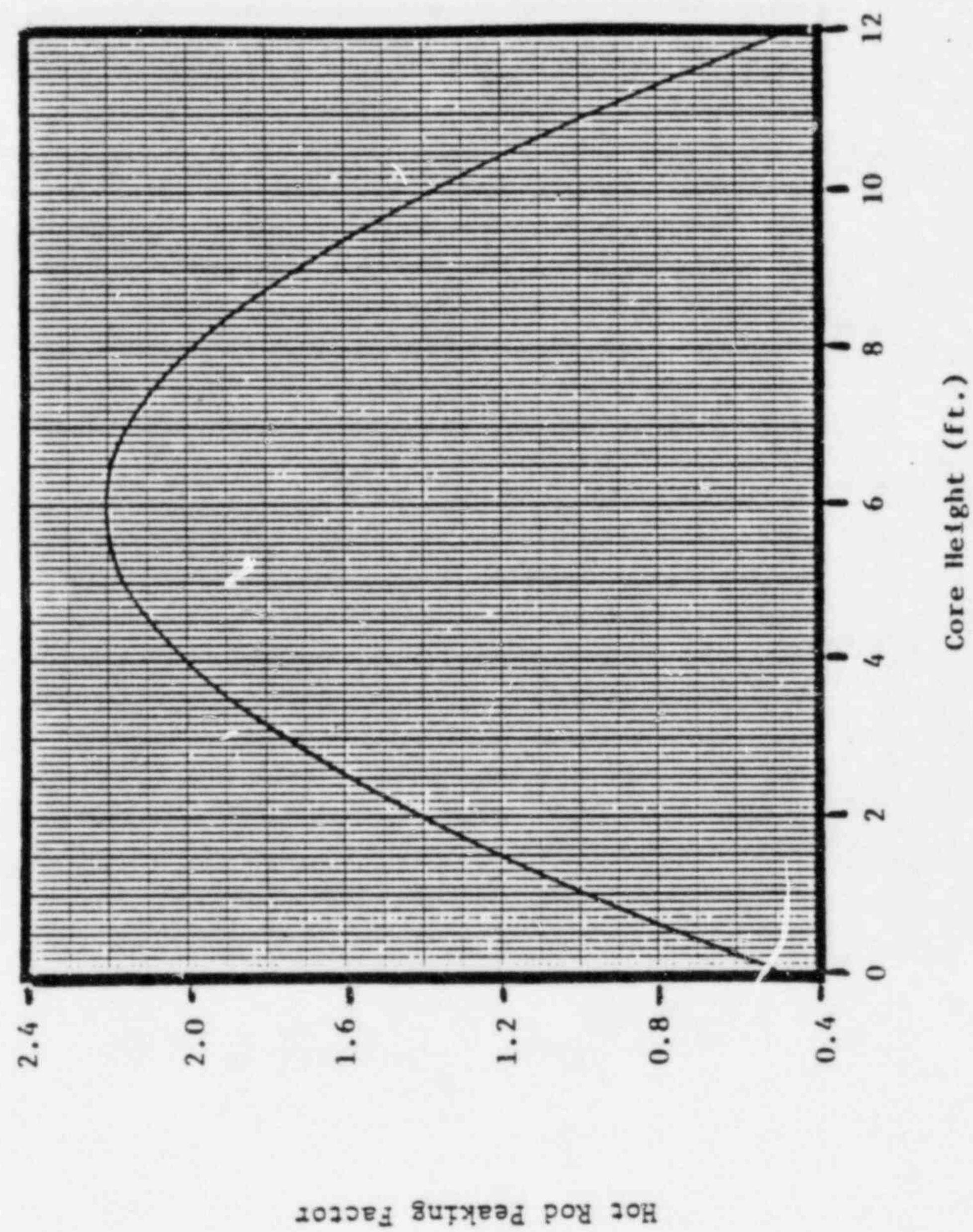


Figure 1: Peaking Factor versus Core Height - FQ=2.20

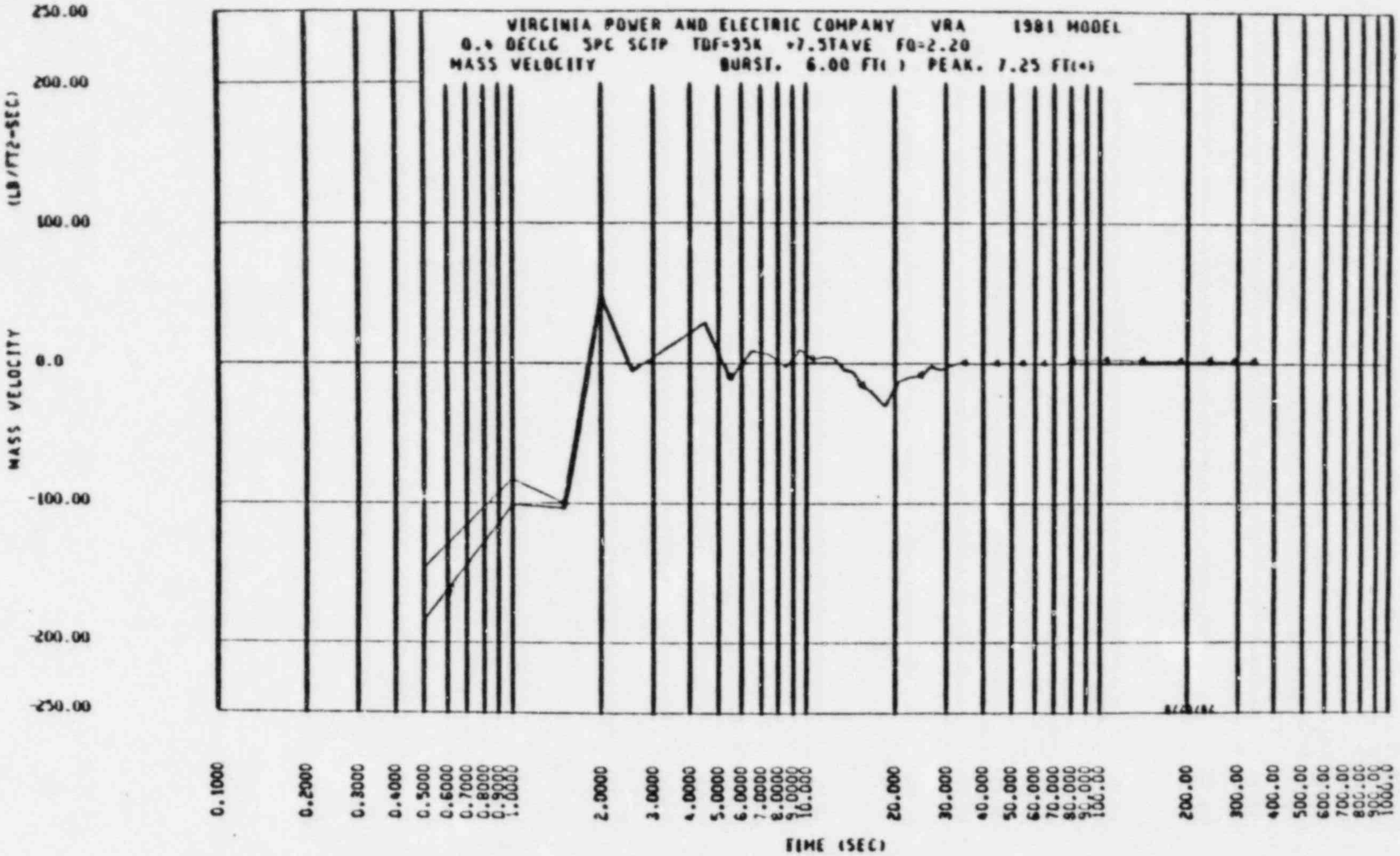


FIGURE 2 : MASS VELOCITY
DECLG (CD = 0.4)

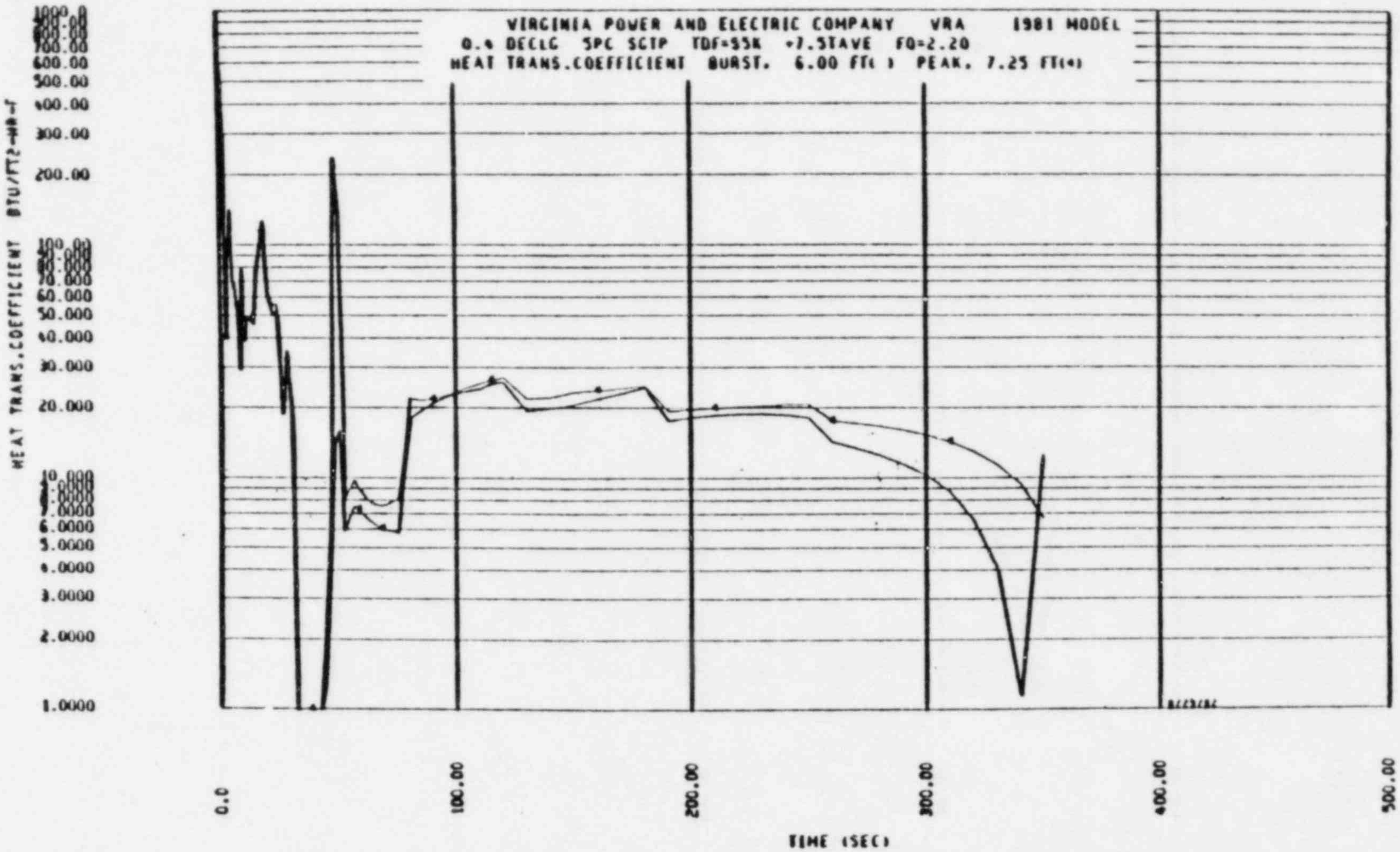


FIGURE 3 : HEAT TRANSFER COEFFICIENT
DECLG (CD = 0.4)

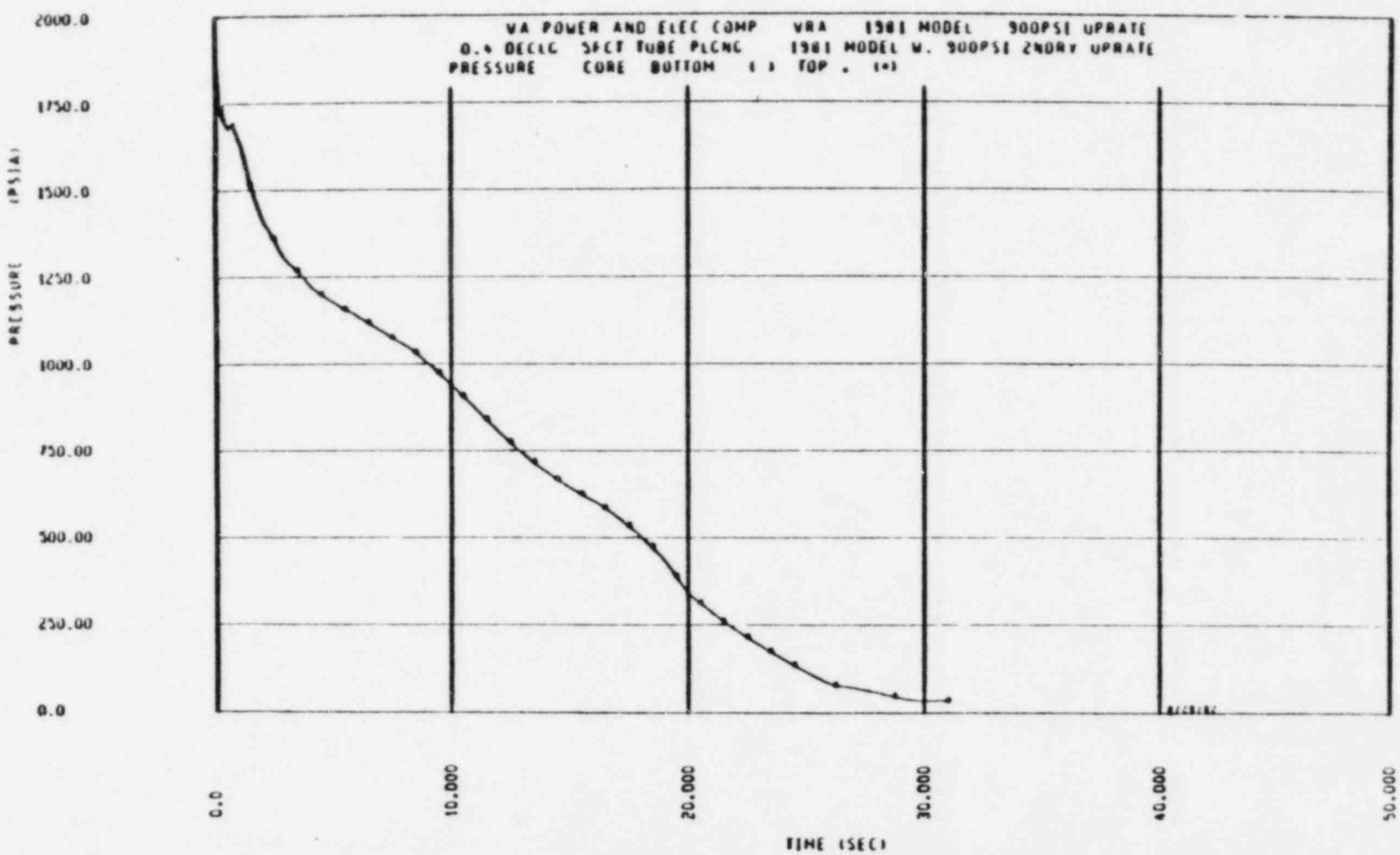


FIGURE 4 : CORE PRESSURE
DECLG (CD = 0.4)

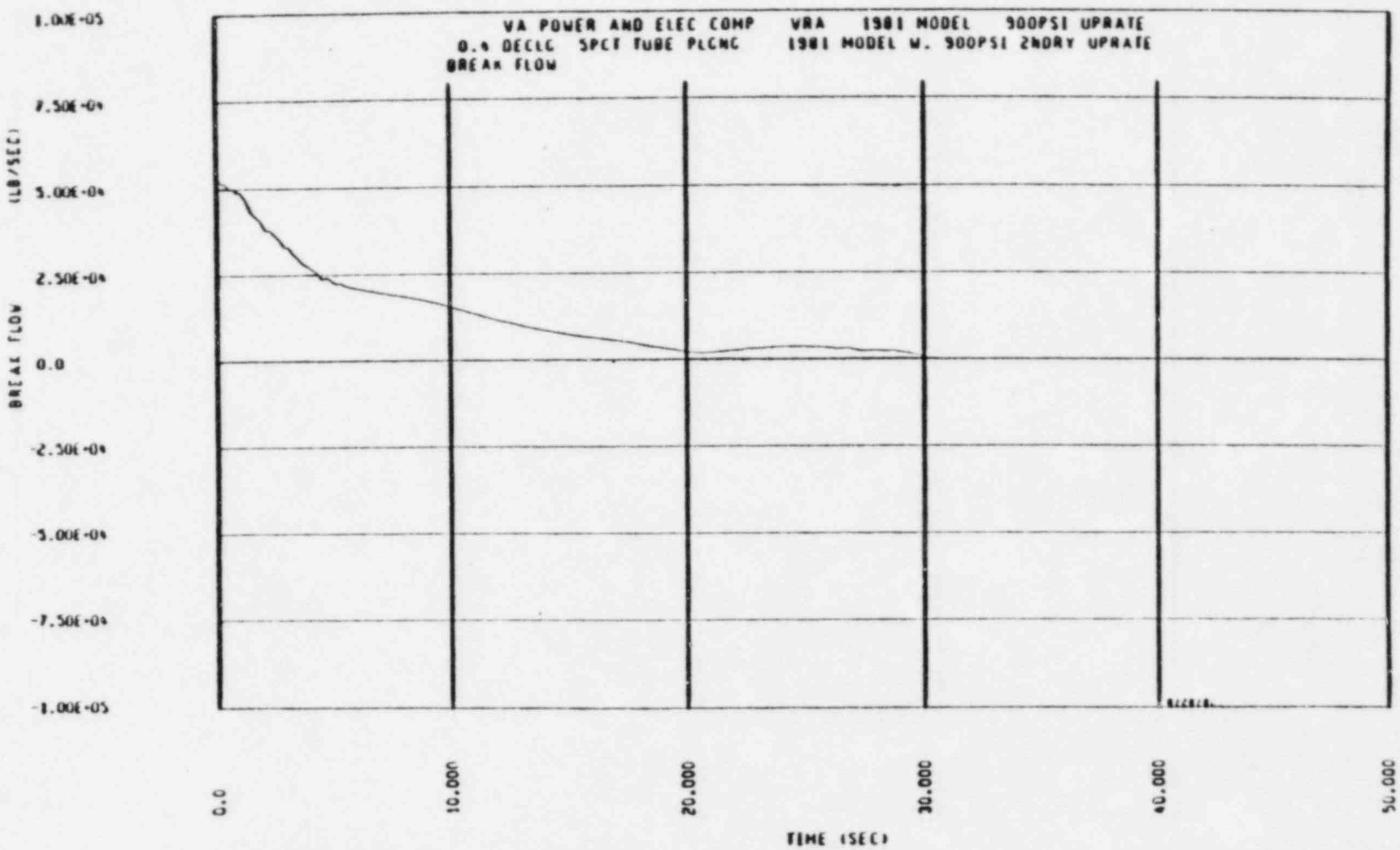


FIGURE 5 : BREAK FLOW RATE
DECLG (CD = 0.4)

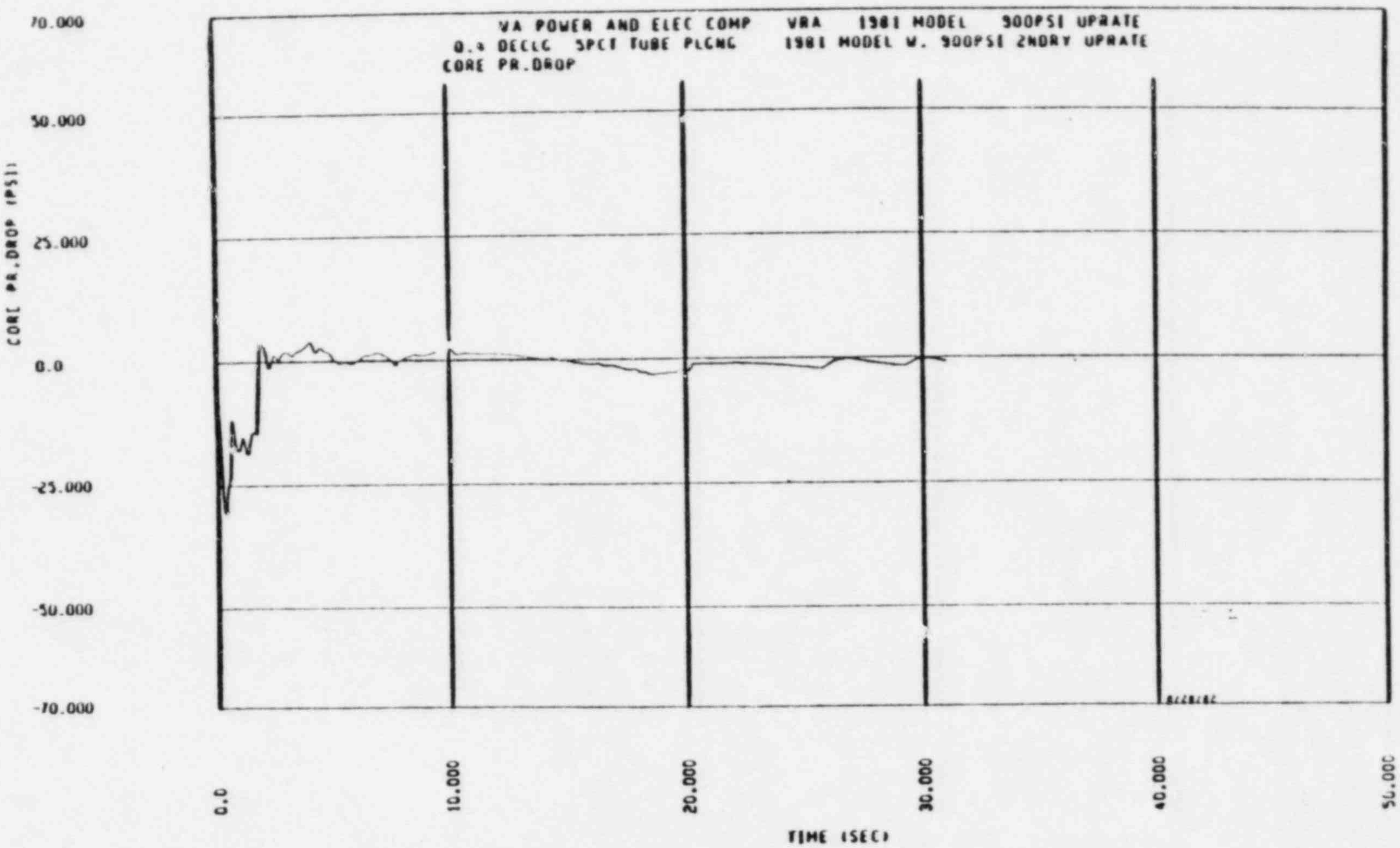


FIGURE 6 : CORE PRESSURE DROP
DECLG (CD = 0.4)

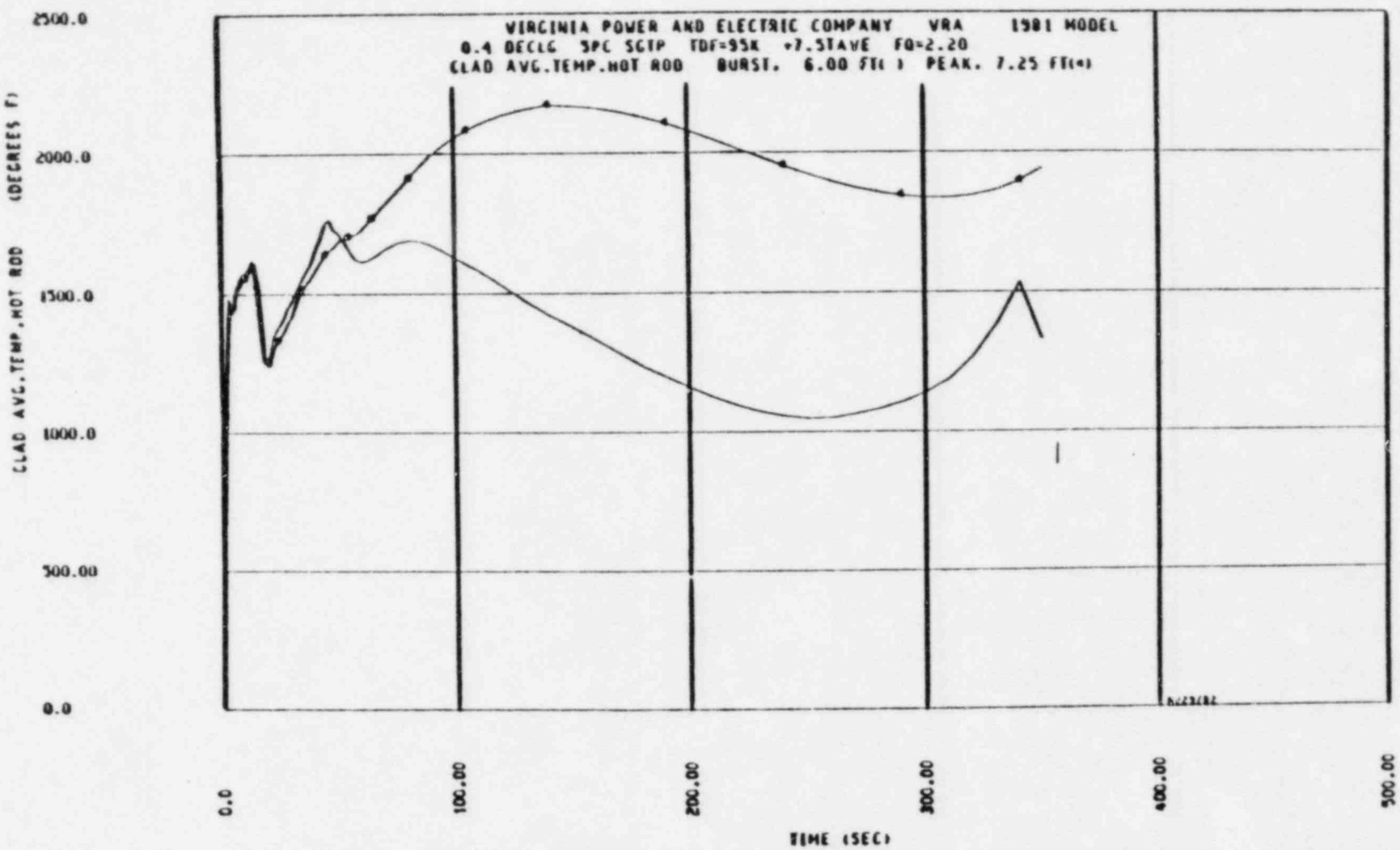


FIGURE 7 : PEAK CLAD TEMPERATURE
DECLG (CD = 0.4)

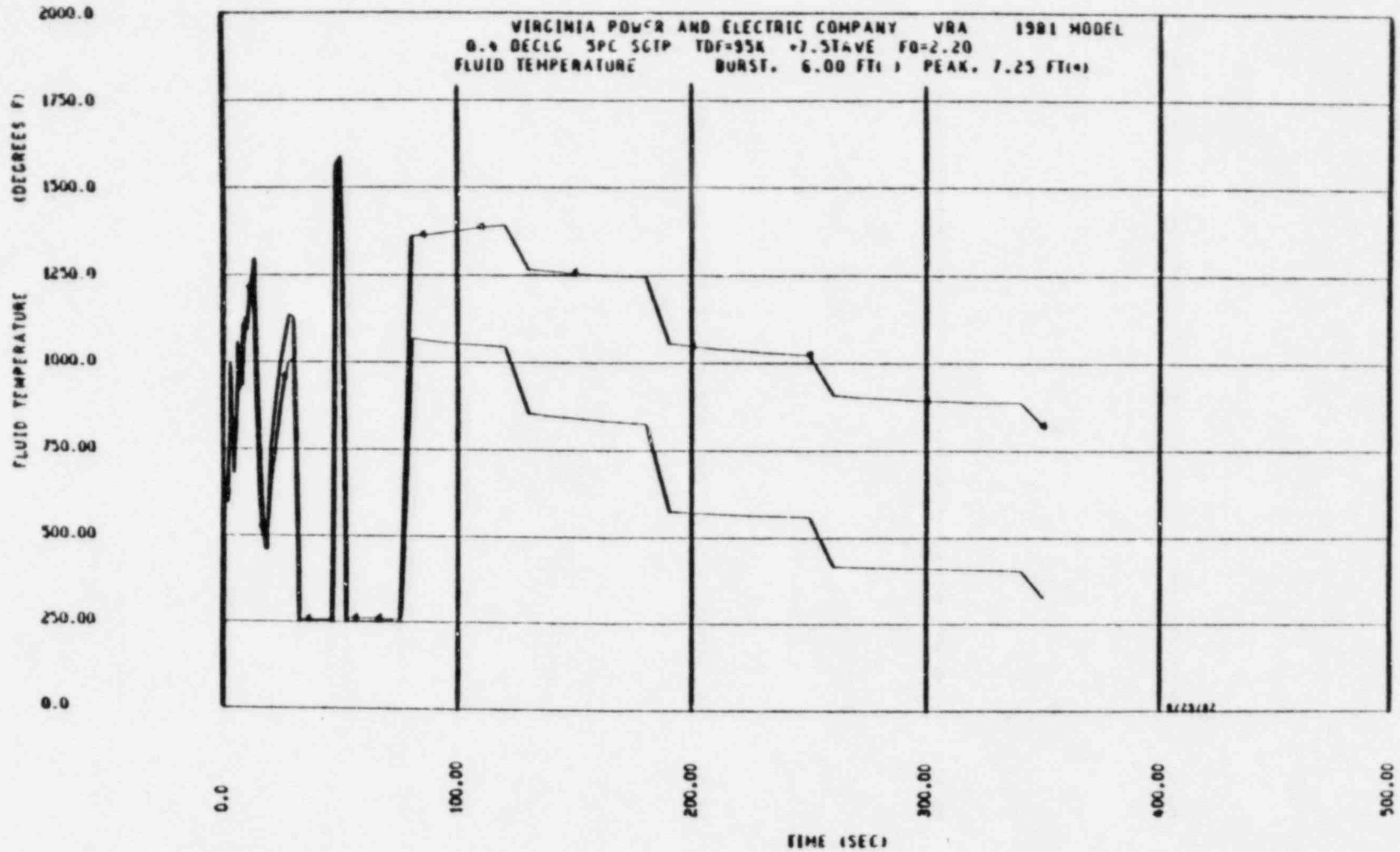


FIGURE 8 : FLUID TEMPERATURE
DECLG (CD = 0.4)

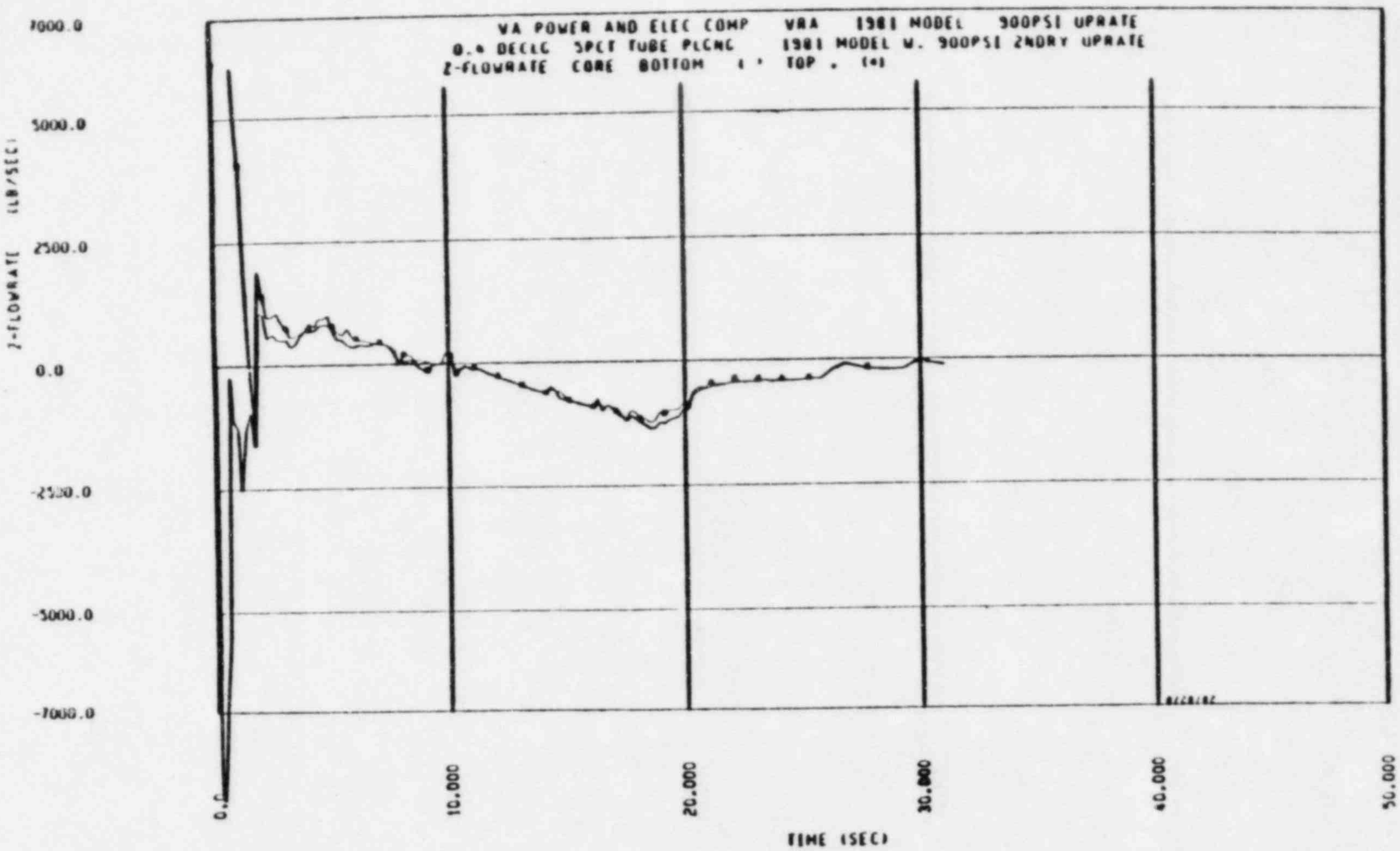


FIGURE 9 : CORE FLOW (TOP AND BOTTOM)
DECLG (CD = 0.4)

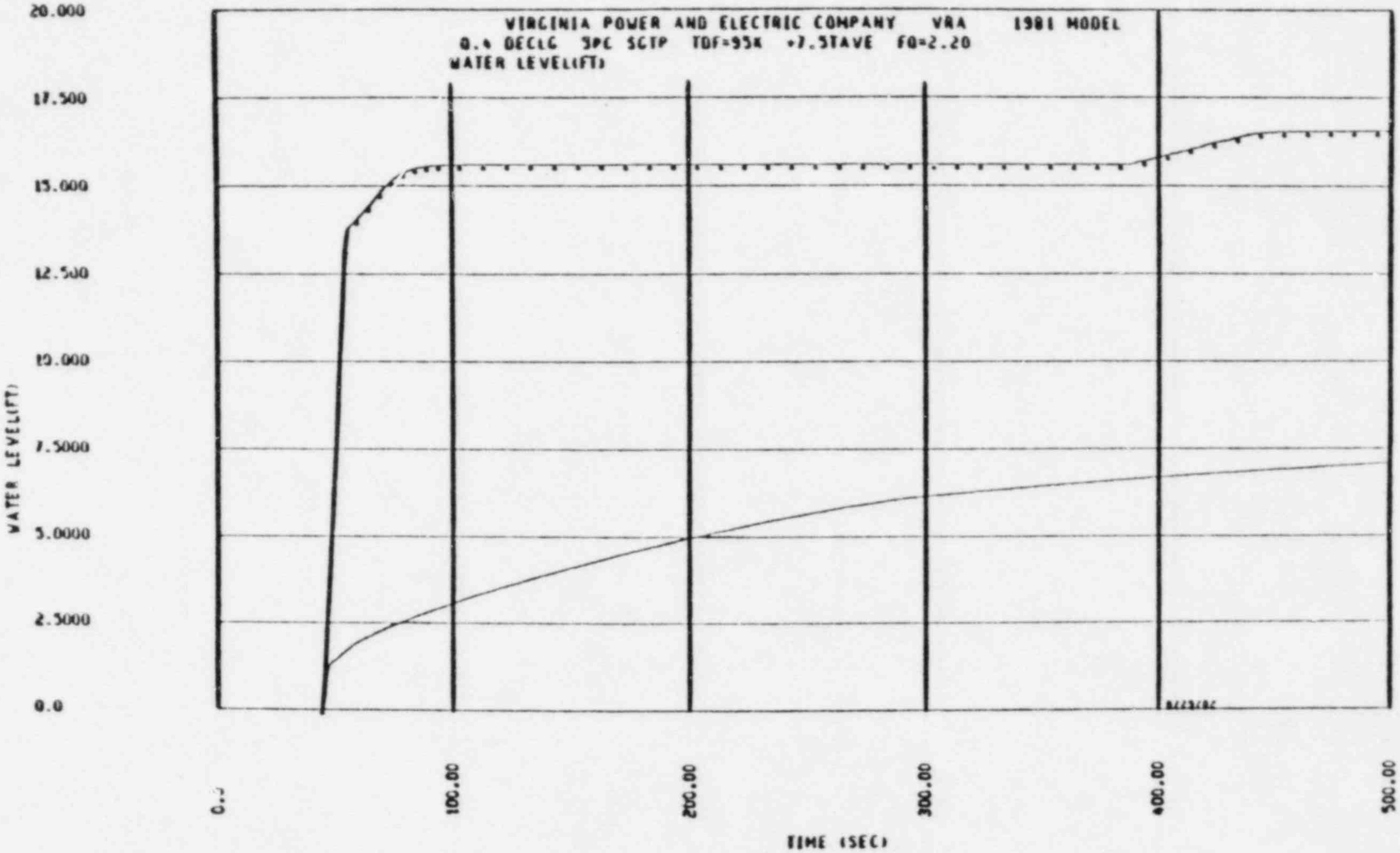


FIGURE 10 : REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG (CD = 0.4)

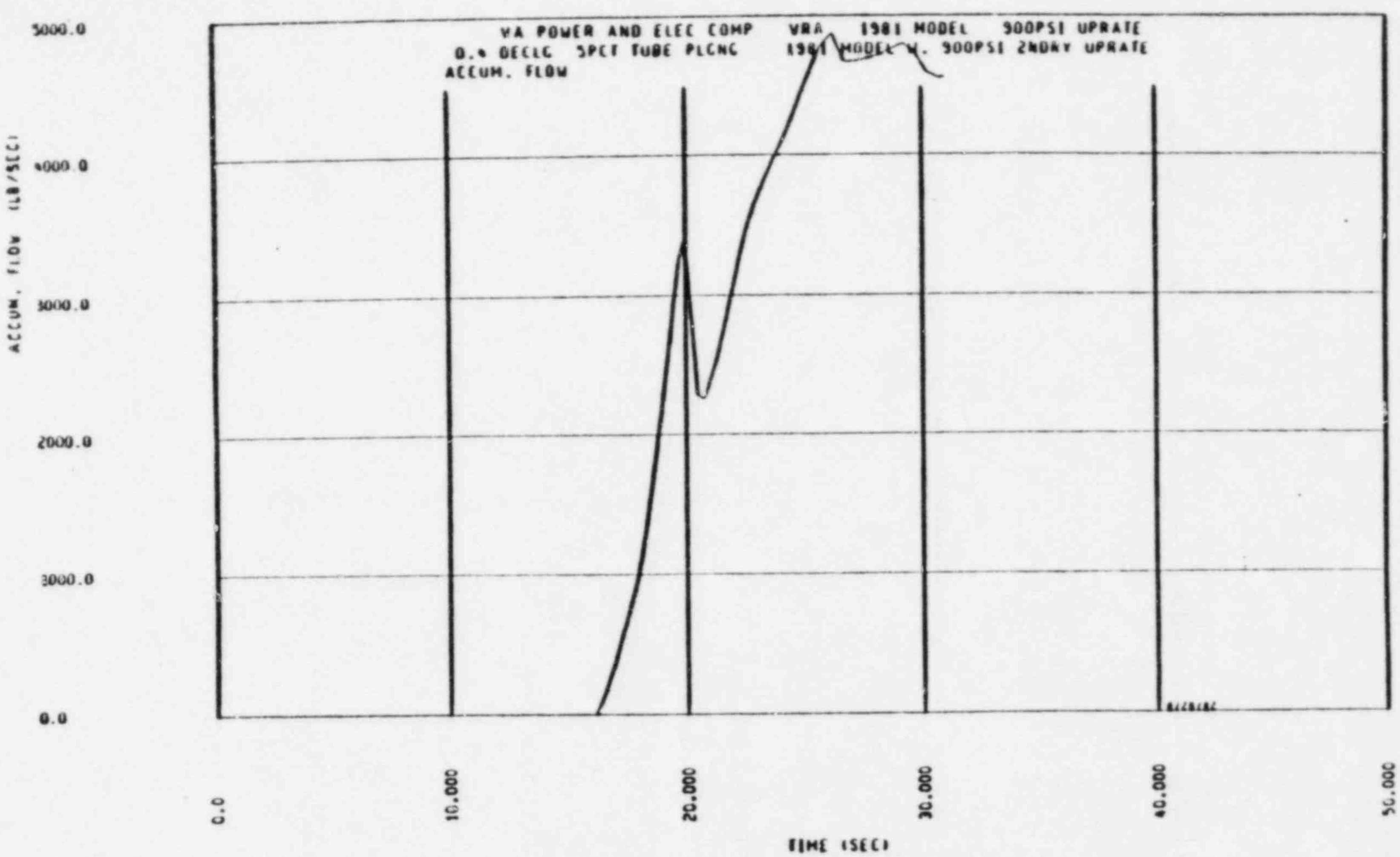


FIGURE 11 : ACCUMULATOR FLOW (BLOWDOWN)
DECLG (CD = 0.4)

VRA, 900 PSI SECONDARY PRESSURE

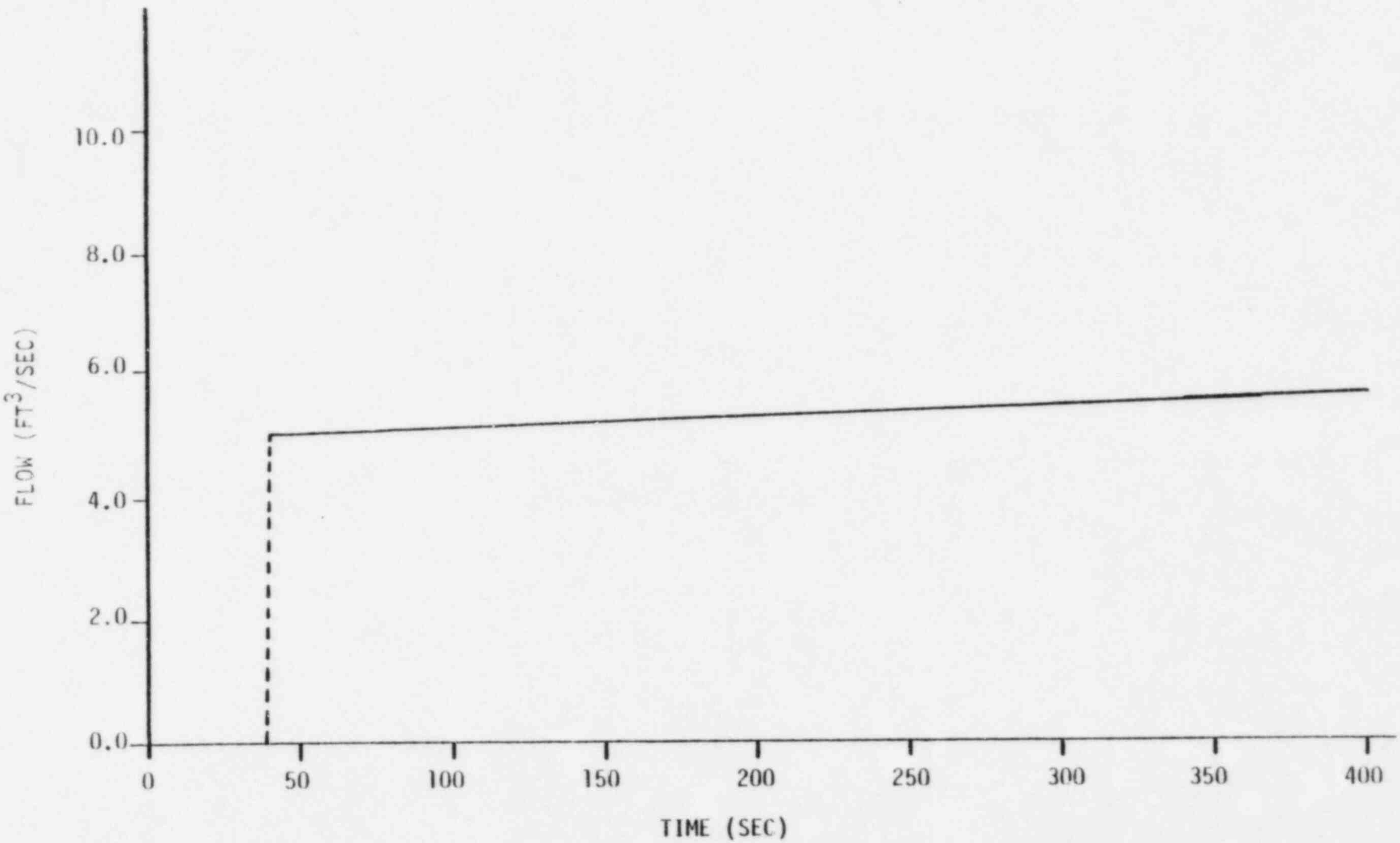


FIGURE 12: PUMPED ECCS FLOW (REFLOOD)
DECLG(CD = 0.4)

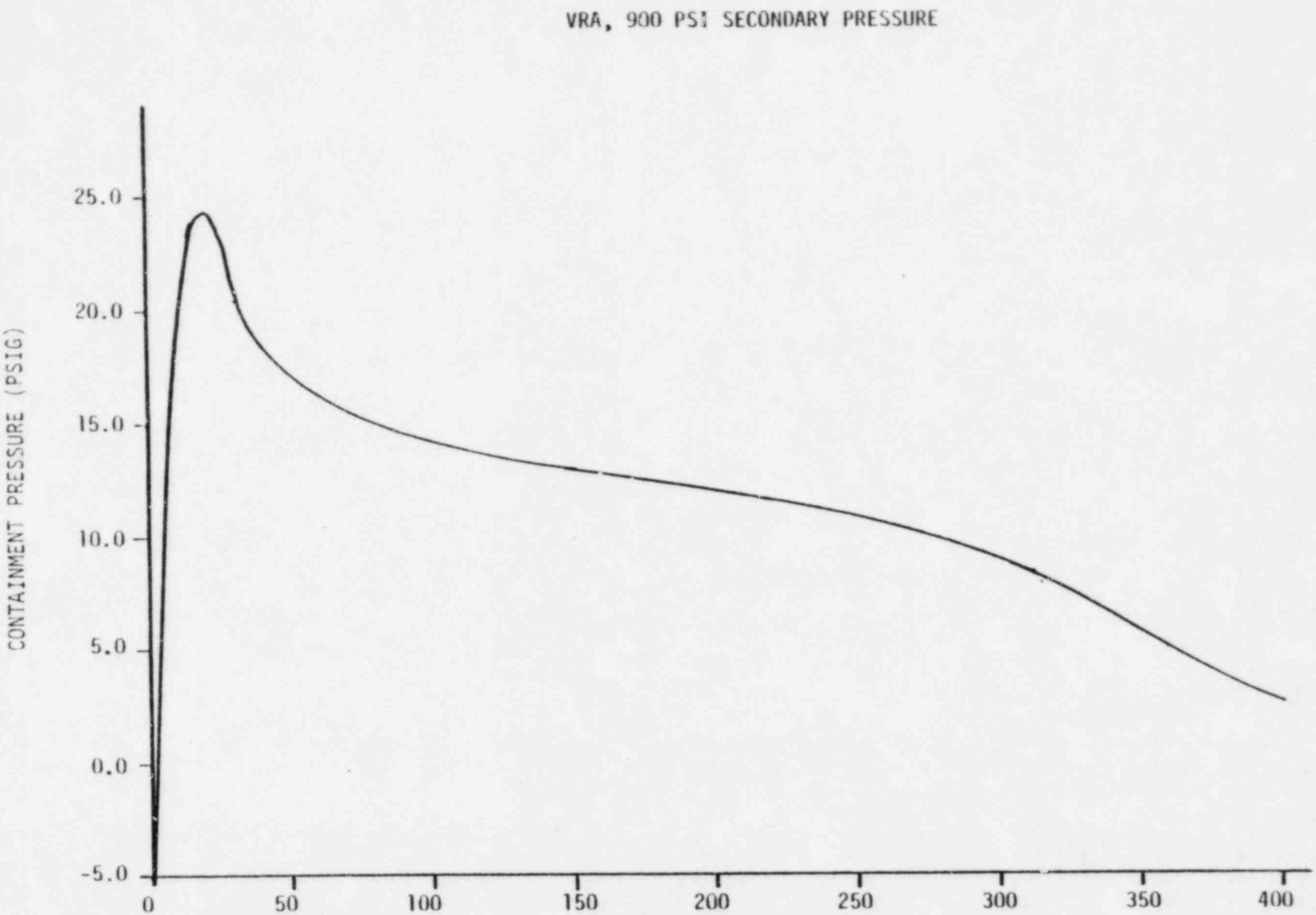


FIGURE 13: CONTAINMENT PRESSURE
DECLG(CD = 0.4)

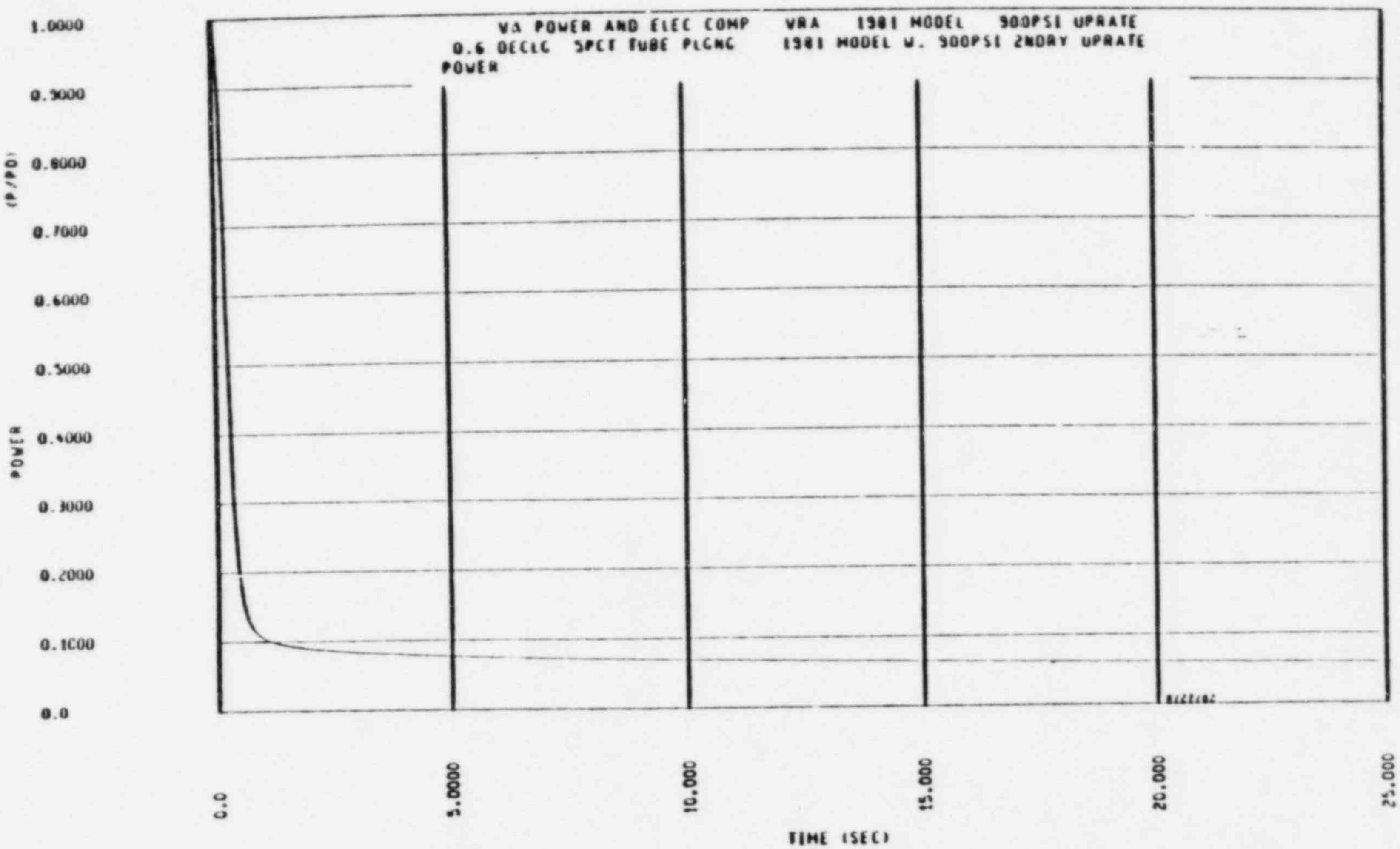


FIGURE 14 : CORE POWER TRANSIENT
DECLG (CD = 0.4)

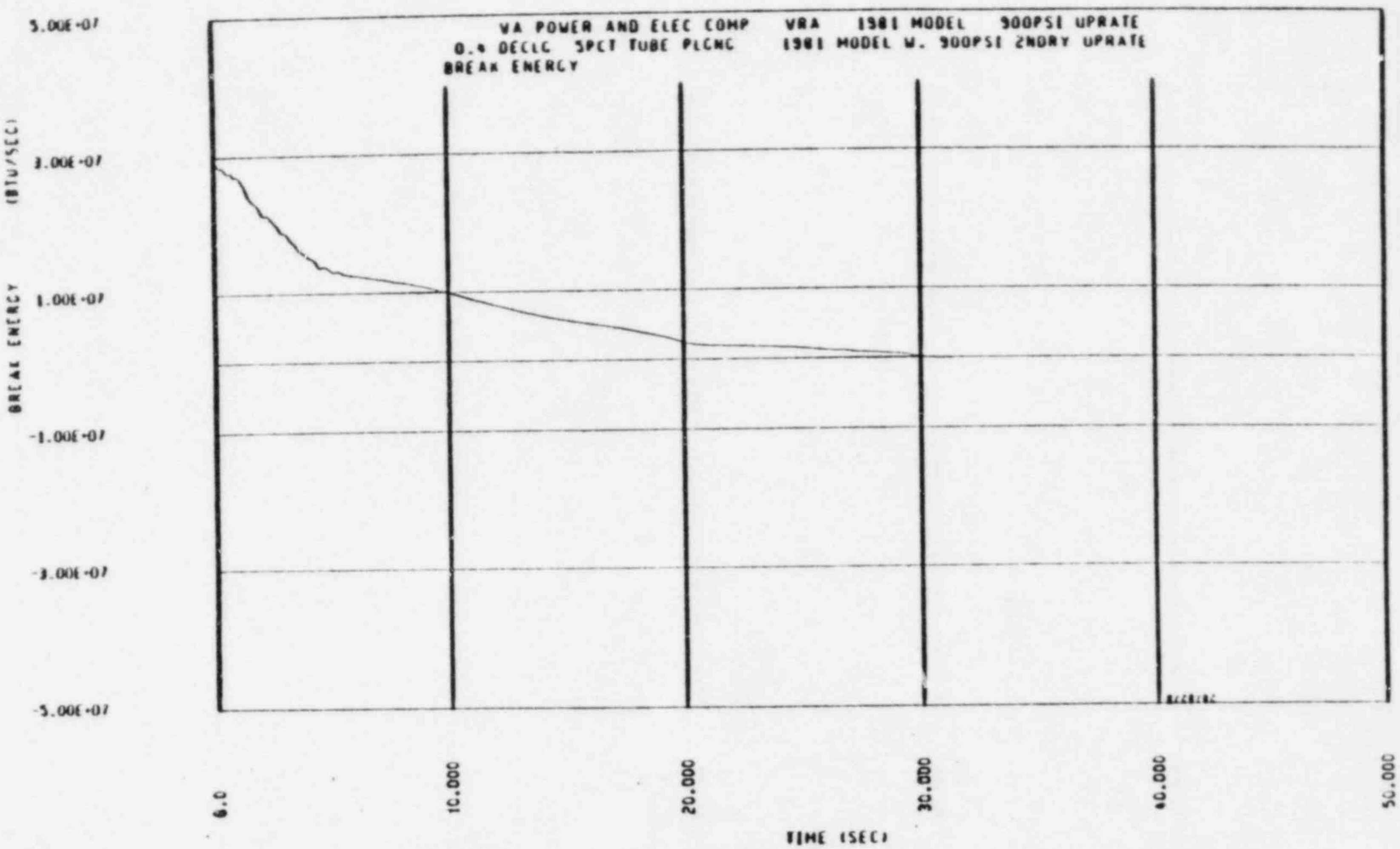


FIGURE 15 : BREAK ENERGY RELEASED TO CONTAINMENT

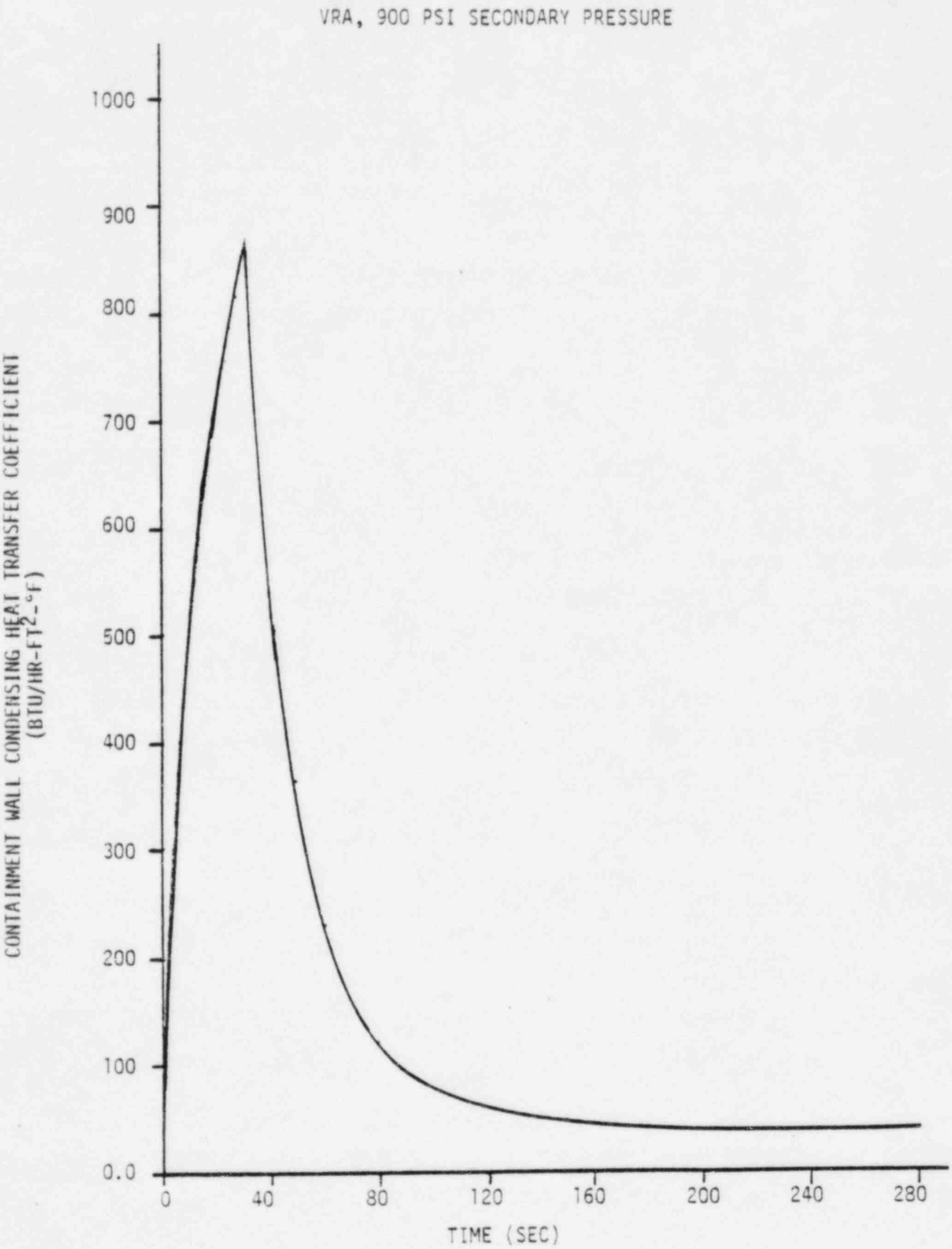


FIGURE 16 : CONTAINMENT WALL
HEAT TRANSFER COEFFICIENT

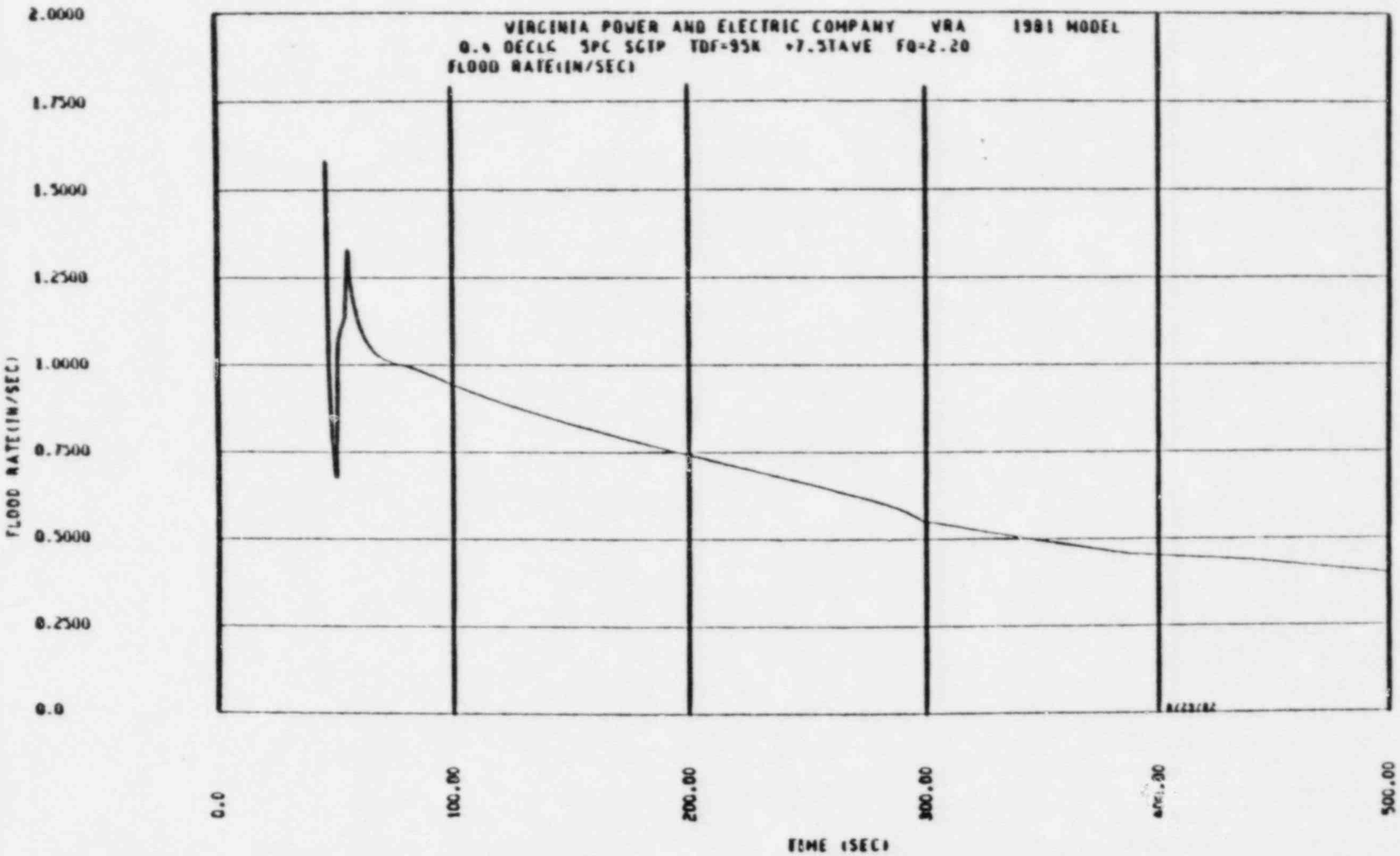


FIGURE 17: REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG (CD = 0.4)

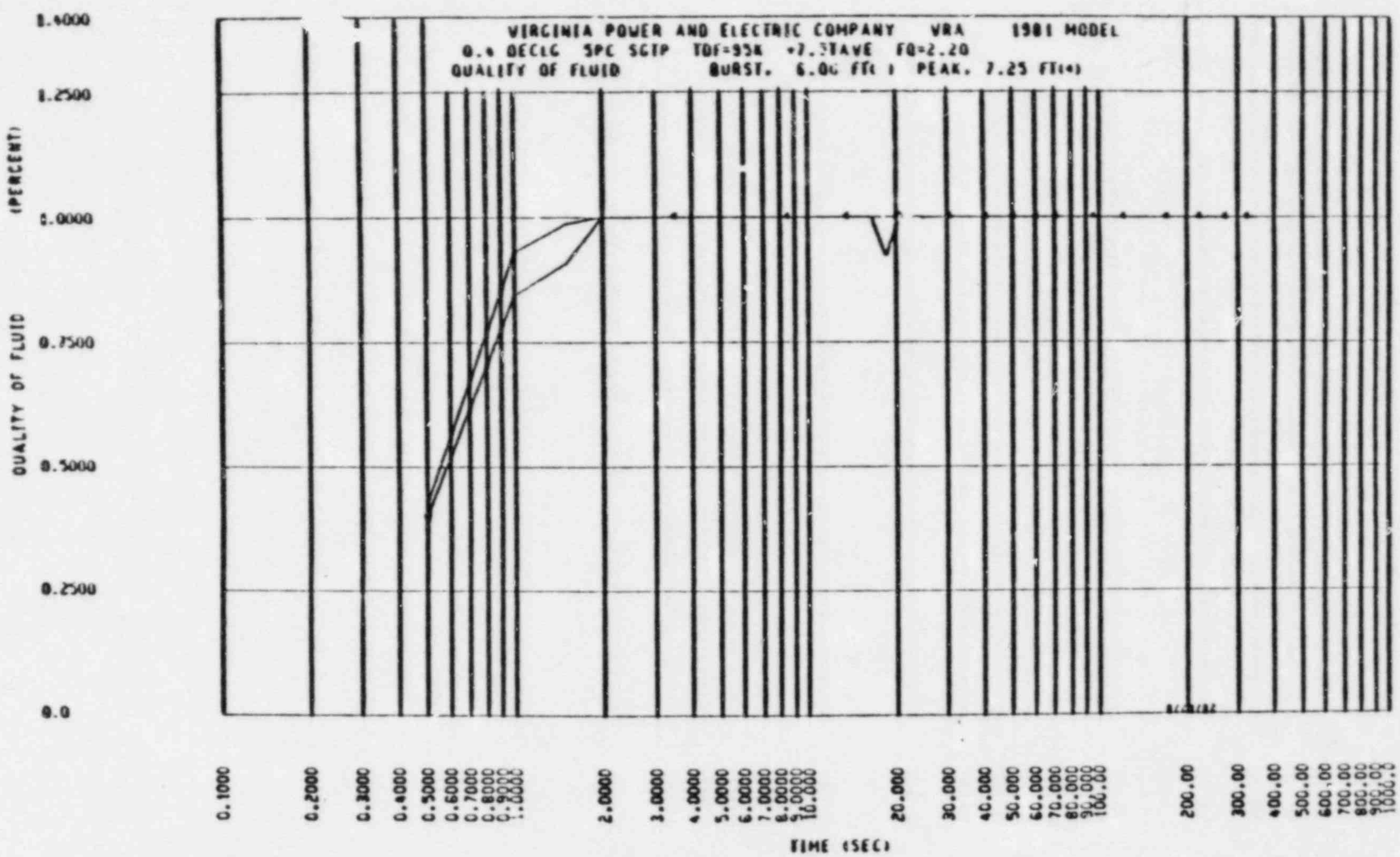


FIGURE 18: FLUID QUALITY
DECLG (CD = 0.4)

3.4 Accident Analyses - Conclusions

To assess the impact on the accident analysis of operation at North Anna with an increased RCS average temperature and 12 MWt net reactor coolant pump heat input, safety analyses of transients sensitive to the increase were performed. These transients included rod withdrawal at power, dropped rod, rod ejection, loss of load, loss of normal feedwater/station blackout, feedline rupture, feedwater system malfunctions, excessive load increase, accidental depressurization of the RCS, complete loss of flow, locked rotor, and large LOCA. This study has indicated that the 7.5°F increase in Tavg increase (and the resulting 50 psi steam pressure increase does not result in any violations of safety limits for the transients analyzed.

4 NSSS SYSTEMS IMPACT

An evaluation was performed to determine the influence of increasing the Reactor Coolant System average temperature by 7.5°F to 587.8°F on the NSSS systems. The systems reviewed were:

Reactor Coolant System
Chemical and Volume Control System
Residual Heat Removal System
Safety Injection System
Reactor Control and Protection System

Each system evaluation verified the adequacy of the system design during normal and postulated transient operating conditions. In addition any revisions to system functional requirements and/or design criteria associated with the increased temperature were identified and reviewed by the cognizant component design function.

The 7.5°F temperature increase has no impact on either the Residual Heat Removal or Safety Injection Systems. These systems are not functional during normal operation. The function of the RHR system is to cool the NSSS from approximately 350°F to a cold shutdown condition. The prime function of the SI system is to provide cooling flow to the core in the event of severe transients. The cooldown and ECCS functional requirements and design criteria are not impacted by this normal operating temperature change.

With respect to the Chemical and Volume Control System (CVCS) the 7.5°F increased temperature has a minor impact on the operating temperatures of the CVCS heat exchangers. The need for revised functional requirements was evaluated relative to the existing heat exchangers (i.e., regenerative, non-regenerative, excess letdown and seal water). It was determined that the installed North Anna CVCS heat exchangers are in compliance with the original system design bases when operated at the elevated temperatures. Operation of the CVCS heat exchangers with an RCS average temperature of 587.8°F will result in minor increases in heat load to the Component Cooling Water System (CCWS) relative to

current operation at 580.3°F. The increased CCWS heat loads are discussed in Section 6. The components downstream of the CVCS heat exchangers (e.g., piping, tanks, filters, pumps) are unaffected by the increased temperatures.

Evaluation of the Reactor Coolant System at the increased temperature revealed that no significant revisions in functional requirements and design bases are necessary. For example, with respect to postulated Reactor Coolant System transients for systems and equipment design (e.g., heat-up, cooldown, reactor trip, etc.) the original design bases reflected temperatures and core thermal power in excess of those associated with the proposed change. The revised nominal operating temperatures were forwarded to the cognizant RCS systems and component design functions for review. The result of the equipment evaluation is presented in Section 5. Based on this evaluation the increased RCS average temperature was found to be within the original design envelope of the Reactor Coolant System.

In addition to the evaluation of the increased temperatures on the Reactor Protection System setpoints which is discussed in Section 7, the Reactor Control System was reviewed to provide a summary of recommended revisions to control functions. An example of this type of review is the development of the RCS average temperature as a function of thermal power based on the increased full thermal load temperature with the no-load temperature remaining at the current value of 547°F. The majority of the control system information is contained in the North Anna Precautions, Limitations and Setpoints (PLS) document. Recommended modifications to the Reactor Control System software will be accomplished via revisions to the North Anna PLS. The North Anna Reference Operating Instructions were reviewed and it was determined that no revisions were required for an RCS average temperature of 587.8°F.

In summary, review of the NSSS systems indicates that the proposed Reactor Coolant System average temperature of 587.8°F is enveloped by the original design bases, criteria and functional requirements. Where appropriate, revisions to the North Anna Precautions Limitations and Setpoints document and Balance of Plant interfaces will be prepared prior to implementation of the 587.8°F RCS average temperature.

5 NSSS COMPONENT IMPACT

The cognizant NSSS component design functions determined the impact of a 587.8°F RCS average temperature on the NSSS equipment. The evaluation assumed that the standards and design criteria applied to the original license were applicable for the proposed change. The system information of prime importance to the equipment designer is the design conditions (temperature, pressure, etc.), seismic response, LOCA loadings, transient events and nominal temperatures. The proposed increase in RCS average temperature has no impact on the design conditions and the seismic response. An increase in fluid temperature results in a decrease in LOCA loadings, and therefore the original LOCA loads are conservative for the proposed 587.8°F RCS average temperature. This is because the decompression wave which hydraulically loads the vessel internals, loop and component supports is proportional to the difference between normal operating pressure and saturation pressure. Operation at higher temperatures causes a reduction in this differential pressure and thus reduced loads.

During the original design an evaluation was performed to determine the type and number of occurences of plant operational transients which constitute the bases for analyzing and evaluating the cyclic behavior of the NSSS components. A description of these transient events is provided in Section 5.2 of the FSAR. The transient evaluation was performed based on temperatures, pressures and power levels in excess of the actual plant rating. The transient analyses provide each component design group with information regarding variations in temperature, power, pressure and flow rate of the unit for use in structural integrity evaluations of the NSSS components. As discussed in Section 4, review of the transients applicable to North Anna revealed that the original design basis transients remain conservative for an RCS average temperature of 587.8°F.

The nominal operating cold leg, hot leg and average temperature of the RCS have increased. As a result it was necessary to verify that each installed component and the associated analyses are in compliance with

the design codes, standards and criteria in effect at the time of the original license for the revised nominal operating conditions. In a few instances it was necessary to revise the documented analyses to account for the increased RCS temperatures. Three levels of effort were utilized for this review. Each of the three levels and the components in each level are discussed below.

A. The first level of effort was to identify for which NSSS systems and associated components no change in the original design bases and functional requirements was required. For these components and/or systems, which are listed below, no additional effort was required with respect to the RCS average temperature increase.

    1) Residual Heat Removal System
    2) Safety Injection System
    3) Pressurizer Spray, Power Operated Relief and Safety Valves
    4) Chemical and Volume Control System except for Heat Exchangers

B. The second level of effort was to identify for which NSSS components the increased RCS temperatures were bounded by analyses performed for a generic design or where a unit with the identical component was evaluated at duty ratings equal to or greater than those associated with the proposed change. The components in this category are:

    1) Reactor Coolant Pumps
    2) Control Rod Drive Mechanisms
    3) Reactor Coolant Loop Piping
    4) Reactor Coolant Loop Isolation Valves
    5) CVCS Heat Exchangers
    6) Pressurizer
    7) Upper Reactor Internals Assembly

C. The third level of effort was to confirm analytically compliance with the applicable design codes, standards and criteria for specific instances where the increased RCS temperatures were not bounded by analyses performed for a generic design or for a unit with the identical components at duty ratings equal to or greater than

those associated with the proposed change. The three components in this category and a description of the associated effort is discussed below:

1) Reactor Vessel

Review of the reactor vessel documentation revealed that the existing analyses bound the proposed change with the exception of the unit loading transient. For this particular transient the proposed temperature change (T_{HOT} minus $T_{NO\ LOAD}$) from no load to full load conditions is approximately 6°F greater than originally evaluated. Analyses were performed for an even greater temperature change than proposed. These analyses indicate that the change has an insignificant impact on the stress range and fatigue life of the vessel, and therefore is acceptable.

2) Lower Reactor Internals Assembly

The North Anna 1 and 2 lower reactor internals assembly were reviewed relative to other 3 loop internals at duty ratings in excess of those associated with the subject temperature increase. With the exception of the item discussed below, the North Anna Unit 1 and 2 lower reactor internals are geometrically identical to higher duty rated structures. Therefore, with the exception of the item in the following discussion, the analyses performed for higher duty 3 loop plants bound the proposed North Anna 1 and 2 increased temperatures. With respect to the one geometrical variation, the following discussion demonstrates that the increased temperatures have no significant impact on either the existing component or the associated analyses.

<u>Thermal Shield</u> - The higher duty rated structures which were reviewed all utilize neutron panels in lieu of the thermal shield employed on the North Anna design. The prime concerns relative to the thermal shield are related to flow loadings and

the potential for flow induced vibrations. An 8.6°F increase in temperature has an insignificant impact on the flow rate, and therefore creates no hardware concern. In addition, an 8.6°F variation in temperature is within the accuracy of the existing thermal analyses. Therefore, the installed North Anna 1 and 2 lower reactor internals assemblies are in compliance with the design codes, standards and criteria applied at the time of the original license.

3) Steam Generator

Review of the steam generator documentation revealed that the current documentation bounds the proposed pressure and temperature changes with the exception of specific evaluations for the steam generator shell in the vicinity of the upper lateral supports and the feedwater nozzle. For these exceptions it was necessary to perform detailed analyses to verify compliance with design criteria for the increased pressure loading at NSSS full thermal power. These analyses consider loads imparted simultaneously to the components due to supports, weight, temperature, flow, pressure and seismic. The analyses performed assumed the original design basis loadings on the component remain unchanged, with the exception of the pressure loadings. A pressure value of 1020 PSIA was assumed with full thermal load steam flow in the generator. This pressure corresponds to the no load value and represents an upper bound value for the evaluation. The results of the analyses demonstrate that the steam generators remain in compliance with the applicable design criteria.

In summary, review of the majority of the NSSS components indicates that the proposed Reactor Coolant System average temperature of 587.8°F is enveloped by either the original North Anna analyses or analyses for other 3 loop plants with identical structures at a higher duty rating.

For specific components where additional analyses were necessary, it was determined that the structures remain in compliance with the design codes, standards and criteria applied at the time of the original license.

5 NSSS/BOP INTERFACE

In an effort to coordinate the NSSS review with the related review of the Balance of Plant (BOP), a program was established to identify areas in which the 7.5°F increase in Reactor Coolant System (RCS) average temperature could have an impact on the BOP design. This temperature increase corresponds approximately to a 50 psi increase in steam generator outlet steam pressure at the full thermal loading of 2787 MWt.

During the course of the evaluation it was determined that a number of NSSS/BOP interfaces were not impacted by the 7.5°F increase in RCS average temperature. The interfaces in this category and a brief explanation of why the existing interface data is not impacted are discussed below:

1) Mass and Energy Release Data - The original Loss of Coolant Accident data for containment integrity evaluations was based on an NSSS rating of 2910 MWt/850 psi steam pressure. This data bounds the proposed temperature increase on the RCS at 2787 MWt. The Main Steamline Break data was based on the event occurring at the no load condition which is unchanged by the increased nominal operating temperatures.

2) Auxiliary Feedwater System - The original auxiliary feedwater system requirements were based on an NSSS rating of 2910 MWt/850 psi steam pressure. The 7.5°F RCS average temperature increase at 2787 MWt is bounded by the original analyses. Therefore, the auxiliary feedwater system requirements are unchanged.

3) Source Terms for Offsite Dose Evaluations - The data currently in the FSAR are based on a core power of 2900 MWt. These source terms are essentially a function of core power and burnup only. The impact of the 7.5°F temperature increase on burnup is negligible. Therefore, the current FSAR data remain unchanged.

4) Spent Fuel Pit Decay Heat Loads - The decay heat of the fuel in the spent fuel pit is a function of core power and burnup and is independent of the 7.5°F average temperature increase. The data employed in the original design evaluation remain unchanged.

5) Steam System Design Transient - The steam system transients provided in the Westinghouse Steam Systems Design Manual are unchanged.

6) RCS Loop Pipe Loads, Thermal Displacements and Desig, Data - Based on a detailed review it was determined than any changes in loadings and piping/support thermal displacements and other design data are within the bounds of the original evaluation.

7) Flux Mapping System - It was determined that the effect of the increased temperature is within the accuracy of the original design data.

A number of areas have been identified for which it is recommended that the Balance of Plant systems be reviewed to evaluate the effects of the proposed change. These areas are:

1) Condensate and Feedwater Systems - Review the system performance and component adequacy at the increased pressure to ensure satisfactory operation.

2) Main Steam System - Perform the same review as outlined for the condensate and feedwater system.

3) Component Cooling Water System - The increased RCS inlet temperature does increase the heat loadings on the Component Cooling Water System (CCWS) due to the Chemical and Volume Control System heat exchangers. Table 6.1 provides a summary of the heat loadings on the CCWS for the current RCS cold leg

temperature of 546.8°F and a value of 555.5°F. The heat loads are based on a maximum CCWS inlet temperature of 105°F to each heat exchanger. The impact of the increased CVCS heat exchanger heat loads on the CCWS should be reviewed to assure that the maximum inlet temperature of 105°F remains unchanged.

An additional area of review was the Westinghouse supplied turbine generators. Based on a detailed review, it was determined that the assumptions, analyses and evaluations (e.g., turbine missiles) performed to verify the operating characteristics and structural integrity of the turbine generator at the current operating parameters bound the conditions resulting from the 7.5°F RCS average temperature increase to 587.8°F.

TABLE 6.1

INCREASED CCWS HEAT LOADS RESULTING FROM CVCS OPERATION
UNDER INCREASED RCS TEMPERATURES

Heat Exchanger/Condition	Heat Transfer Rate (BTU/HR x 106)	
	RCS Cold Leg Temp. 546.8°F	RCS Cold Leg Temp. 555.5°F
Non-Regenerative:		
Normal Operation	5.10	5.28
Maximum Purification	11.1	11.1
Heat Up (Design)	16.1	16.1
Excess Letdown:		
Design	3.08	3.26
Seal Water Return:		
Normal Operation	0.92	1.02
Design	1.25	1.37

7 TECHNICAL SPECIFICATION REVISIONS

A review of the North Anna Unit 1 and 2 Technical Specifications was performed to establish revisions necessary to reflect the 7.5°F increase in Reactor Coolant System average temperature to 587.8°F at full thermal load of 2787 MWt. The required revisions are identified in Enclosure 3.

Appendix

to

Enclosure 1

North Anna FSAR Changes

for

Tavg of 587.8°F

are concerned, the "Engineered Safety Features design rating" plus allowance for error is assumed. The thermal power values used for each transient analyzed are given in Table 15.1-16.

15.1.2.2 Initial Conditions

For accident evaluation, the initial conditions are obtained by adding maximum steady state errors to rated values. The following steady state errors are considered:

a) Core power	± 2 percent allowance for calorimetric error
b) Average Reactor Coolant System temperature	± 4°F allowance for deadband and measurement error
c) Pressurizer pressure	± 30 psi allowance for steady state fluctuations and measurement error

TABLE 15.1-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Guaranteed Nuclear Steam Supply System thermal power output	2787 MWt
The Engineered Safety Features design rating (maximum calculated turbine rating)	2910 MWt
Thermal power generated by the reactor coolant pumps	12 MWt
Guaranteed Core Thermal Power	2775 MWt

TABLE 15.1-2

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay (Seconds)
Power Range High Neutron Flux, High Setting	118%	0.5
Power Range High Neutron Flux, Low Setting	35%	0.5
Overtemperature ΔT	Variable see Figure 15.1-1	6.0*
Overpower ΔT	Variable see Figure 15.1-1	6.0*
High pressurizer pressure	2410 psig	2.0
Low pressurizer pressure	1845 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Undervoltage Trip	68% nominal	1.2

*Total time delay (including RTD bypass loop fluid transport delay, effect bypass loop piping thermal capacity, RTD time response, and trip circuit channel electronics delay) from the time the temperature difference in the coolant loops exceed the trip setpoint until the rods are free to fall.

TABLE 15.1-6

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR(1) TEMPERATURE ($\Delta k/^{\circ}F$)	MODERATOR(1) DENSITY ($\Delta K/gm/cc$)	DOPPLER(2)	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWT)
CONDITION II					
Uncontrolled RCC Assembly Bank Withdrawal from a Subcritical Condition	WIT-6, FACTRAN	$+1 \times 10^{-5}$	---	lower	0
Uncontrolled RCC Assembly Bank Withdrawal at Power	LOFTRAN	---	0 and 0.43	lower and upper	2787
RCC Assembly Misalignment	THINC, TURTLE, LOFTRAN	---	0	upper	2785
Uncontrolled Boron Dilution	NA	NA	NA	NA	0 and 2785
Partial Loss of Forced Reactor Coolant Flow	PHOENIX, LOFTRAN	---	0	upper	1810 and 2785
Start-up of an Inactive Reactor Coolant Loop	MARVEL, THINC	---	0.43	lower	1810
Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	---	0 and 0.43	lower and upper	2787
Loss of Normal Feedwater	LOFTRAN	---	0	upper	2910
Loss of Off-Site Power to the Plant Auxiliaries (Plant Blackout)	LOFTRAN	---	0	upper	2910

TABLE 15.1-6 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR[1] TEMPERATURE ($\Delta k/^\circ F$)	MODERATOR[1] DENSITY (ΔK/gm/cc)	DOPPLER[2]	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWT)
CONDITION II (Cont'd.)					
Excessive Heat Removal Due to Feedwater System Malfunctions	LOFTRAN	---	0.43	lower	0 and 2787
Excessive Load Increase	LOFTRAN	---	0 and 0.43	lower and upper	2787
Accidental Depressurization of the Reactor Coolant System	LOFTRAN	---	0	upper	2787
Accidental Depressurization of the Main Steam System	MARVEL	---	Function of Moderator Density. See Sec. 15.2.13 (Fig. 15.2-41)	-2.2 pcm/°F	0 (Subcritical)
Inadvertent Operation of ECCS During Power Operation	LOFTRAN	---	0	lower	
CONDITION III					
Loss of Reactor Coolant From Small Ruptured Pipes or from Cracks in Large Pipe which Actuate Emergency Core Cooling	WFLASH, LOCTA-IV				2785

TABLE 15.1-6 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR(1) TEMPERATURE (Δk/°F)	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR(1) DENSITY (ΔK/gm/cc)	DOPPLER(2)	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWT)
CONDITION III (Cont'd.)					
Inadvertent Loading of a Fuel Assembly Into an Improper Position	LEOPARD, TURTLE	---	NA	NA	2785
Complete Loss of Forced Reactor Coolant Flow	LOFTRAN, THINC, FACTRAN	---	0	upper	1810 and 2787
Waste Gas Decay Tank Rupture	NA	---	NA	NA	2910
Single RCC Assembly Withdrawal at Full Power	TURTLE, THINC, LEOPARD	---	NA	NA	2785
CONDITION IV					
Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss of Coolant Accident)	SATAN COCO REFLOOD LOCTA	Function of Moderator Density See Section 15.4.1		Function of Fuel Temp. See Section 15.4.1	2787
Major Secondary System Pipe Rupture up to and Including Double-Ended Rupture (Rupture of a Steam Pipe)	MARVEL, THINC	Function of Moderator Density See Section 15.2.13 (Fig. 15.2-41)		-2.2 pcm/°F	0 (Subcritical)
(Major Rupture of a Feedwater Pipe)	LOFTRAN		.43	upper	2910

TABLE 15.1-6 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED MODERATOR(1) TEMPERATURE ($\Delta k/^\circ F$)	MODERATOR(1) DENSITY ($\Delta K/gm/cc$)	DOPPLER(2)	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWT)
CONDITION IV (Cont'd.)					
Steam Generator Tube Rupture	NA	NA	NA	NA	2910
Single Reactor Coolant Pump Locked Rotor	LOFTRAN, THINC, FACTRAN	---	0	upper	1810 and 2787
Fuel Handling Accident	NA	NA	NA		2910
Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	TWINKLE, FACTRAN	-1 pcm/°F BOL -26 pcm/°F EOL	---	Consistent with lower limit shown Fig. 15.1-5	0 and 2787

NOTES:

(1) Only one is used in an analysis i.e., either moderator temperature or moderator density coefficient.

(2) Reference Figure 15.1-5

a) Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure, resulting in the minimum initial margin to DNB.

b) Reactivity coefficients - Two cases are analyzed:

1. Minimum reactivity feedback. A zero moderator coefficient of reactivity is assumed corresponding to the beginning of core life. A least negative Doppler power coefficient is used in the analysis.

2. Maximum reactivity feedback. A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

c) The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.

d) The rod cluster control assembly trip insertion characteristics is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

e) The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at maximum speed.

The effects of rod cluster control assembly movement on the axial core power distribution is accounted for by causing a decrease in overtemperature and overpower ΔT trip setpoints proportional to a decrease in margin to DNB.

Results

Figures 15.2-4 and 15.2-5 show the response of neutron flux, pressure, average coolant temperature, and DNB ratio to a rapid rod cluster control assembly withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and a large margin to DNB is maintained.

The response of neutron flux, average coolant temperature, pressure and DNB ratio for a slow rod assembly withdrawal from full power is shown in Figures 15.2-6 and 15.2-7. Reactor trip on overtemperature ΔT occurs after a longer period and rise in temperature and pressure is consequently larger than for rapid rod cluster control assembly withdrawal.

Figure 15.2-8 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for the minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT trip channels. The minimum DNBR is never less than 1.30.

Figures 15.2-9 and 15.2-10 show the minimum DNBR as a function of reactivity insertion rate for rod cluster control assembly withdrawal incidents starting at 60 and 10 percent power respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below 1.30.

15.2.2.3 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than 1.30.

15.2.3 <u>Rod Cluster Control Assembly Misalignment</u>

15.2.3.1 Identification of Causes and Accident Description

Rod cluster control assembly misalignment accidents include:

a) Dropped full-length assemblies;

b) A dropped full-length assembly bank;

c) Statically misaligned full length assembly (See Table 15.2-2).

Method of Analysis

Steady state power distributions are analyzed for this event using the TURTLE[5] code. The peaking factors calculated by TURTLE are then used by the THINC code to calculate the DNBR. For the transient response to a dropped RCCA or RCCA bank the LOFTRAN[4] code is used. The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valve, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Results

A dropped rod cluster control assembly typically results in a reactivity insertion of -0.15 percent $\Delta k/k$. Extensive analyses were performed to show that in manual control the minimum DNBR occurs near the end of the transient when the system has essentially returned to a new steady state equilibrium condition. Without automatic rod control, the system will return to a new equilibrium condition at a reduced primary temperature as a result of the moderator reactivity feedback. As typical of a PWR uncontrolled response, the return to power is monotonic and therefore power overshoot is not a condition for this case.

A power overshoot after a dropped rod cluster assembly incident can only result from the action of the automatic rod controller. For a given PWR system, the power overshoot is essentially a function of the rod controller characteristics. As a result of the unreviewed safety issue which identified the potential for dropped rod events leading to power overshoots, resulting in calculated DNB ratios lower than those reported in the current licensing documents, the following interim procedural solution has been instituted:

a) In manual reactor control, there is no change from current procedures.
b) In automatic control above 90% reactor power, Bank D must be withdrawn to greater than or equal to 215 steps.

By using this restricted operating strategy, the potential for large overshoots is minimized.

Sensitivity studies have confirmed that the maximum power overshoot occurs for the following conditions:

a) Minimum moderator reactivity feedback corresponding to beginning of core life conditions.
b) Least negative doppler only power coefficient.

Figure 15.2-11 illustrates the transient for the following limiting conditions:

a) Initial Power: 92 percent of rated power.
b) Zero Moderator Reactivity Coefficient.
c) Control Bank Reactivity Worth: 6 pcm/step.

The DNBR was computed along the transient assuming constant hot channel factors to illustrate that limiting DNB conditions occur at the point of maximum return to power. Therefore, only the peak power conditions need to be analyzed with the penalty associated for a dropped rod cluster control assembly condition.

A dropped rod cluster control assembly group typically results in a reactivity insertion of -1.2 percent $\Delta k/k$ which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.5 seconds following the drop of a rod cluster assembly group. The core is not adversely affected during this period, since power is decreasing rapidly.

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which group D is inserted to its insertion limit with one assembly fully withdrawn. The insertion limits in the Technical Specifications are chosen based on a number of limiting criteria. One of these criteria is that control bank D may be inserted to its low-low insertion limit at full power with any one assembly fully withdrawn without the DNBR following below 1.30. This is demonstrated in Table 15.2-2. Multiple independent alarms, including a bank insertion limit alarm, alert the operator before the postulated conditions are approached.

DNB calculations have not been performed specifically for assemblies missing from other banks, however, power shape calculations have been done as required for the rod cluster control assembly ejection analysis. Inspection of the power shapes shows that the DNB and peak kw/ft situation is less severe than the group D case discussed above assuming insertion limits on the other groups.

15.2.3.3 Conclusions

It is shown that in all cases of dropped assemblies, the DNBR remains greater than 1.30 and, consequently, dropped assemblies do not cause core damage.

For all cases of dropped groups, the reactor is tripped by the power range negative neutron flux rate trip and consequently dropped banks do not cause core damage.

For all cases of any group inserted to its rod insertion limit with any single rod cluster control assembly in that group fully withdrawn, the DNBR remains greater than 1.30. Thus, rod misalignments do not result in core damage.

15.2.4 Uncontrolled Boron Dilution

pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 102 percent of full power without a direct reactor trip primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins.

Assumptions are:

a) Initial Operating Conditions - the initial reactor power and Reactor Coolant System temperatures are assumed to their maximum values consistent with the steady state full power operation including allowances for calibration and instrument errors (See Section 15.1.2). The initial Reactor Coolant System pressure is assumed at a minimum value consistent with the steady state full power operation including allowances for calibration and instrument errors. These assumptions result in the maximum power

difference for the load loss, and the minimum margin to core protection limits at the initiation of the accident.

b) Moderator and Doppler Coefficients of Reactivity - the total loss of load is analyzed for both beginning of life and end of life conditions. Moderator temperature coefficients of zero at beginning of life and a large (absolute value) negative value at end of life are used. A conservatively large (absolute value) Doppler power coefficient is used for beginning of life and a least negative value is used for end of life.

c) Reactor Control - from the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control.

d) Steam Release - no credit is taken for the operation of the steam dump system or steam generator power operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value (~1100 psia).

e) Pressurizer Spray and Power Operated Relief Valves - two cases for both the beginning and end of life are analyzed:

1. Full credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure.

2. No credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure. Pressurizer heater operation is assumed since heater operation on high pressurizer water level will tend to increase the maximum surge rate through the pressurizer safety valves.

f) Feedwater Flow - main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur. The auxiliary feedwater flow would remove core decay heat following plant stablilization.

Reactor trip is actuated by the first Reactor Protection System trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip.

Results

The transient responses for a total loss of load from 102 percent full power operation are shown for four cases; two cases for the beginning of core life and two cases for the end of core life, in Figures 15.2-23 through 15.2-30.

Figures 15.2-23 and 15.2-24 show the transient response for the total loss of load at beginning of life with zero moderator temperature coefficient assuming full credit for the pressurizer spray and pressurizer power operated relief valves. No credit is taken for steam dump. The reactor is tripped by the overtemperature ΔT trip channel. The minimum DNBR is well above the 1.30 design value.

Figures 15.2-25 and 15.2-26 show the response for the total loss of load at end of life assuming a large (absolute value) negative moderator temperature coefficient and a least negative doppler only power coefficient. All other plant parameters are the same as the case above. The reactor is tripped by the overtemperature ΔT trip channel. The DNBR increases throughout the transient and never drops below its initial value.

The pressurizer safety valves are not actuated in the transients shown in Figures 15.2-23 through 15.2-26.

The total loss of load accident was also studied assuming the plant to be initially operating at 102% of full power with no credit taken for the pressurizer spray, pressurizer power operated relief valves, or steam dump. The reactor is tripped on the high pressurizer signal. Figures 15.2-27 and 15.2-28 show the beginning of life transient with zero moderator coefficient. The neutron flux remains constant at 102% of full power until the reactor is tripped. The DNBR increases throughout the transient. In this case the pressurizer safety valves are actuated.

Figures 15.2-29 and 15.2-30 show the transient at the end of life with the other assumptions being the same as in Figures 15.2-27 and 15.2-28. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated.

d) One turbine driven auxiliary feedwater pump (capable of delivering at least 700 gpm) which is started on the same signals as the motor driven pumps.

The motor driven auxiliary feedwater pumps are supplied by the diesels if a loss of offsite power occurs and the turbine-driven pump utilizes steam from the secondary system. Both types of pumps are designed to start within one minute even if a loss of all AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the secondary steam to the atmosphere. The auxiliary pumps take suction from a condensate water storage tank for delivery to the steam generators.

The analysis shows that following a loss of normal feedwater, the auxiliary feedwater system is capable of removing the stored and residual heat thus preventing either overpressurization of the Reactor Coolant System or loss of water from the reactor core.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN[4] code is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation

describes the plant thermal kinetics, Reactor Coolant System including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Assumptions:

a) The initial steam generator water level (in all steam generators) at the time of reactor trip is at the narrow range low level tap, i.e., a conservatively low level.

b) The plant is initially operating at 102 percent of 2910 MWt the engineered safeguards design rating.

c) A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

d) Only two motor driven auxiliary feedwater pumps are available one minute after the accident.

e) Auxiliary feedwater is delivered to two steam generators.

f) Secondary system steam relief is achieved through the self-actuated safety valves. Note that steam relief will, in fact, be through the

power operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, for the sake of analysis these have been assumed unavailable.

g) The initial reactor coolant average temperature is 4°F higher than the nominal value since this results in a greater expansion of Reactor Coolant System water during the transient and, thus, in a higher water level in the pressurizer.

Results

Figure 15.2-31a shows plant parameters following a loss of normal feedwater with offsite power available.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, the auxiliary feedwater pumps are automatically started, reducing the rate of water level decrease.

The capacity of the auxiliary feedwater pumps are such that the water level in the feed steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the Reactor Coolant System relief or safety valves.

The auxiliary feedwater system is started automatically as discussed in the loss of normal feedwater analysis. The steam driven auxiliary feedwater pump has a capacity of 700 gpm and utilizes steam from the secondary system, exhausting directly to the atmosphere. The two motor driven auxiliary feedwater pumps have a capacity of 340 gpm each and are supplied by power from the diesel generators. The pumps take suction directly from a condensate storage tank for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

15.2.9.2. Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the LOFTRAN[4] code is done to obtain the plant transient following a station blackout. The simulation describes the plant thermal kinetics, Reactor Coolant System including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The first few seconds of the transient will closely resemble a simulation of the complete loss of flow incident (See Section 15.3.4), i.e., core

damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual heat must be removed to prevent damage to either the Reactor Coolant System or the core. The assumptions used in the analyses are similar to the loss of normal feedwater flow incident, except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip.

Results

Figure 15.2-31 shows plant parameters following a station blackout.

The LOFTRAN results show that natural circulation flow available is sufficient to provide adequate core decay heat removal following a reactor trip and RCP coastdown.

15.2.9.3 Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage. The DNBR is maintained above 1.30. The Reactor Coolant System is not overpressurized and no water relief will occur through the pressurizer relief or safety valves. Thus there will be no cladding damage and no release of fission products to the Reactor Coolant System.

15.2.10 Excessive Heat Removal Due to Feedwater System Malfunctions

2.10.1 Identification of Causes and Accident Description

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the Reactor Coolant System. The overpower - overtemperature protection (neutron overpower, overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than 1.30.

15.2.10.2 Analysis of Effects and Consequences

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer code LOFTRAN[4]. This code simulates a multi-loop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to evaluate plant behavior in the event of feedwater system malfunction. Feedwater temperature reduction due to feedwater heater bypass valve actuation in conjunction with an inadvertent trip of the heater drain pump is considered.

Excessive feedwater addition due to a control system malfunction or operator error which allows a feedwater control valve to open fully is considered. Two cases are analyzed as follows:

a) Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large negative moderator temperature coefficient.

b) Accidental opening of one feedwater control valve with the reactor in automatic control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

with the power unbalance during the transient. The time at which reactor trip occurs is of no concern for this accident. At lighter loads coolant contraction will be slower resulting in a longer time to trip.

15.2.14.2 Analysis of Effects of Consequences

Method of Analysis

The spurious operation of the SIS system is analyzed by employing the detailed digital computer program LOFTRAN[4]. The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the safety injection system. The program computes pertinent plan variables including temperatures, pressures, and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. Analysis of several cases shows the results are relatively independent of time to trip.

A typical transient is presented representing conditions at beginning of core life. Results at end of life are similar except that moderator feedback effects result in a slower transient.

The assumptions are:

15.2.15 References

1. W. C. Gangloff, "An Evaluation of Anticipate Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486, May 1971.

2. D. B. Fairbrother, H. C. Hargrove, "WIT-6 Reactor Transient Analysis Computer Program Description," WCAP-7980, November 1972.

3. C. Hunin, "FACTRAN", "A Fortran N Code for Thermal Transients in UO_2 Fuel Rod," WCAP-7908, June 1972.

4. S. T. Maher, "LOFTRAN Code Description," WCAP-7878, Rev. 3, July 1981.

5. S. Altomare, R. F. Barry, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7758, September 1971.

6. F. M. Bordelon, "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP-7969, September 1972.

7. M. A. Mangan, "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October 1971.

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (sec.)
Uncontrolled RCCA Bank Withdrawal at Power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (70 pcm/sec)	0
	Power range high neutron flux high trip point reached	1.7
	Rods begin to fall into core	2.2
	Minimum DNBR occurs	3.0
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3.0 pcm/sec)	0
	Overtemperature ΔT reactor trip signal initiated	24.7
	Rods begin to drop into core	26.7
	Minimum DNBR occurs	26.8

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (sec.)
Loss of External Electrical Load		
1. With pressurizer control (BOL)	Loss of electrical load	0
	Initiation of steam release from steam generator safety valves	6.5
	Overtemperature ΔT Reactor Trip Point Reached	6.9
	Rods begin to drop	8.9
	Minimum DNBR occurs	10.0
	Peak pressurizer pressure occurs	9.5
2. With pressurizer control (EOL)	Loss of electrical load	0
	Initiation of steam release from steam generator safety valves	6.5
	Overtemperature ΔT Reactor Trip Point Reached	6.9
	Rods begin to drop	8.9

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (sec.)
	Minimum DNBR occurs	(1)
	Peak pressurizer pressure occurs	7.5
3. Without pressurizer control (BOL)	Loss of electrical load	0
	Initiation of steam release from steam generator safety valves	6.5
	High pressurizer pressure reactor trip point reached	5.1
	Rods begin to drop	7.1
	Minimum DNBR occurs	(1)
	Peak pressurizer pressure occurs	8.0

(1) DNBR does not decrease below its initial value.

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (sec.)
4. Without pressurizer control (EOL)	Loss of electrical load	0
	Initiation of steam release from steam generator valves	6.5
	High pressurizer pressure reactor trip point reached	5.1
	Rods begin to drop	7.1
	Minimum DNBR occurs	(1)
	Peak pressurizer pressure occurs	7.5

(1) DNBR does not decrease below its initial value.

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (sec.)
Loss of Normal Feedwater (With offsite power available)	Main feedwater flow stops	10.0
	Low steam generator water level trips	48.0
	Rods begin to drop	50.0
	Peak water level in pressurizer occurs	52.0
	Two steam generators begin to receive auxiliary feed from two motor driven auxiliary feedwater pumps	108.0
	Core decay heat plus reactor coolant pump heat decreases to auxiliary feedwater heat removal capacity	~400.0

TABLE 15.2-1 (Coninued)

TIME SEQUENCE OF EVENTS FOR

CONDITION II EVENTS

Accident	Event	Time (sec.)
Loss of offsite power	Main feedwater stops	10.0
	Low steam generator water level trip	48.0
	Rods begin to drop	50.0
	Reactor coolant pumps begin to coast down	50.0
	Peak water level in pressurization occurs	52.0
	Two steam generators begin to receive auxiliary feed from two motor driven auxiliary feedwater pumps	108.0
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~256.0

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (sec.)
Excessive feedwater at full load	One main feedwater control valve fails fully open	0
	Minimum DNBR occurs	21.0
	Feedwater flow isolated due to high-high steam generator level	23.0

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (sec.)
Excessive Load Increase		
1. Manual Reactor Control (BOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	200
2. Manual Reactor Control (EOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	100
3. Automatic Reactor Control (BOL)	10% step load increase	0
	Equilibrium conditions reached	200
4. Automatic Reactor Control (EOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	200

TABLE 15.2-1 (Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Events	Time (sec.)
Accidental depressurization of the Reactor Coolant System	Inadvertent Opening of one RCS Safety Valve	0
	Reactor Trip	22.2*
	Minimum DNBR occurs	22.8*
Accidental depressurization of the Main Steam System	Inadvertent Opening of one main steam safety or relief valve	0
	Pressurizer Empties	168
	20,000 ppm boron reaches RCS loops	237

* Includes two second delay from reaching trip setpoint to beginning of rod motion.

Permissive 8 a breaker open signal from any two pumps will actuate a reactor trip.

Normal power for the reactor coolant pumps is supplied through busses from a transformer connected to the generator. Each pump is on a separate bus. When generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core.

Following any turbine trip, where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

15.3.4.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by three digital-computer codes. First the LOFTRAN(8) code is used to calculate the loop and core flow during the transient, the time of the reactor trip, and the nuclear power transient following reactor trip. The FACTRAN(9) code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The R grid spacer factor is applied to the W-3 correlations. The transients presented represent the minimum of the typical or thimble cell.

The following loss of flow cases are analyzed:

a) Loss of three pumps from a nominal Reactor Coolant System heat output of 100% (2787 MWt) with three loops operating.

b) Loss of two pumps from a nominal Reactor Coolant System heat output of 60% (1671 MWt) with two loops operating; no loops stop valves closed.

c) Loss of two pumps from a nominal Reactor Coolant System heat output of 65% (1810 MWt) with two loops operating; loop stop valves closed in one loop.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.2.5, except that, following the loss of supply to all pumps at power, a reactor trip is actuated by either bus undervoltage or bus under-frequency.

Results

The calculated sequence of events is shown in Table 15.3-2 for the three cases analyzed. Figures 15.3-23 through 15.3-31 show the loop coastdowns, the reactor vessel coastdowns, the nuclear power coastdowns and the average and hot channel heat flux coastdowns for each of the three cases. The reactor is assumed to trip on the undervoltage signal. The DNBR curve for each of the cases is not less than 1.30.

15.3.4.3 Conclusions

The analysis performed has demonstrated that for the complete loss of

TABLE 15.3-2

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Accident	Event	Time (sec.)
1. Complete loss of forced reactor coolant flow all 3 loops operating, all 3 pumps coasting down		
	Coastdown begins	0
	Rod Motion begins	1.2
	Minimum DNBR occurs	3.1
2. Two out of three pumps operating, two pumps coasting down, loop stop valves open		
	Coastdown begins	0
	Rod Motion begins	1.2
	Minimum DNBR occurs	2.3
3. Two out of three pumps operating, two pumps coasting down, loop stop valves closed		
	Coastdown begins	0
	Rod Motion begins	1.2
	Minimum DNBR occurs	2.3

(Refer to Chapter 7 for a description of the actuation system.)

b) An auxiliary feedwater system to provide an assured source of feedwater to the steam generators for decay heat removal.
(Refer to chapter 10 for description of the auxiliary feedwater system.)

15.4.2.2.2 Analysis of Effects and Consequences

Method of Analyis

A detailed analysis using the LOFTRAN(4) code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics, reactor coolant system including natural circulation, pressurizer, steam generators and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions are:

a) The plant is initially operating at 102 percent of the engineered safeguards design rating.

b) Initial reactor coolant average temperature is 4°F above the nominal value, and the initial pressurizer pressure is 30 psi above its nominal value.

c) No credit is taken for the pressurizer power-operated relief valves or pressurizer spray.

d) Initial pressurizer level is at the nominal programmed value +2 percent (error); initial steam generator water level is at the nominal value +5 percent in the faulted steam generator and at the nominal -5 percent in the intact steam generators.

e) No credit is taken for the high pressurizer pressure reactor trip. Note: This assumption is made for calculational convenience. Pressurizer power-operated relief valves and spray could act to delay the high pressure trip. Assumptions 3 and 5 permit evaluation of one hypothetical, limiting case rather than two possible cases: one with a high pressure trip and no pressure control; and one with pressure control but no high pressure trip.

f) Main feed to all steam generators is assumed to stop at the time the break occures (all main feedwater spills out through the break).

g) A conservative feedwater line break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time the trip setpoint is reached. After the trip occurs, a saturated liquid discharge is assumed until all the water inventory is discharged from the affected steam generator. This minimizes the heat removal capability of the affected steam generator.

h) No credit is taken for the low-low water level trip on the affected steam generator until essentially all liquid is discharged from the steam generator shell.

i) The worst possible break area is assumed; i.e., one that empties the affected steam generator (and causes a reactor trip on low-low steam generator water level as assumed above) at the same time as the fluid inventory in the unaffected steam generators drops to the trip point

for low steam generator level. This assumption minimizes the steam generator fluid inventory at the time of trip, and thereby maximizes the resultant heatup of the reactor coolant. The reactor trip was assumed to be actuated when the water level in the unaffected steam generator decreases to the low-low level trip setpoint minus 10% of narrow range span.

j) No credit is taken for heat energy deposited in reactor coolant system metal during the reactor coolant system heatup.

k) No credit is taken for charging, letdown, or safety injection.

l) Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases.

m) Conservative core residual heat generation is assumed based upon long term operation at the initial power level preceding the trip.

n) The auxiliary feedwater is assumed to be actuated by the low-low steam generator water level signals with the feed rate of 340 gpm, the capacity of one motor driven auxiliary feed pump. A 60 second delay was assumed following the low-low level signal to allow time for startup of stand by diesel generators and auxiliary feedwater pumps.

Results

Figure 15.4.2-7a and b show the calculated plant parameters following a feedline rupture. Results for the case with offsite power available are presented on Figure 15.4.2-7a. Results for the case where offsite power is lost are presented on Figure 15.4.2-7b. The reactor core remain covered with water throughout the transient, as water relief due to thermal expansion is limited by the heat removal capability of the AFW.

The time sequence of events for both cases is shown on Table 15.4.2-3

15.4.2.2.3 Conclusion

Results of the analysis show that for the postulated feedline rupture, the assumed auxiliary feedwater system capacity is adequate to remove decay heat, to prevent overpressurizing the reactor coolant system, and to prevent uncovering the reactor core.

TABLE 15.4.2-3

TIME SEQUENCE OF EVENTS FOR POSTULATED FEEDLINE RUPTURE

(Sheet 1 of 2)

Accident	Event	Time (sec.)
Feedwater System Pipe Break		
1. With offsite power available	Main feedline rupture occurs	10
	Low-low steam generator level trip setpoint reached in ruptured steam generator	14.5
	Rods begin to drop	16.5
	Low steamline pressure setpoint reached in ruptured steam generator	73
	Auxiliary feedwater is delivered to intact steam generators	74.5
	All main steamline isolation valves close	80
	Steam generator safety valve setpoint reached in intact steam generators	226
	Pressurizer safety valve setpoint reached	367
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	>4500

TABLE 15.4.2-3

TIME SEQUENCE OF EVENTS FOR POSTULATED FEEDLINE RUPTURE

(Sheet 2 of 2)

Accident	Event	Time (sec.)
Feedwater System Pipe Break		
2. Without offsite power available	Main feedline rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	14.5
	Rods begin to drop, power lost to the reactor coolant pumps	16.6
	Low steamline pressure setpoint reached in ruptured steam generator	42
	All main steamline isolation valves close	49
	Auxiliary feedwater is delivered to intact steam generators	74.5
	Steam generator safety valve setpoint reached in intact steam generators	82
	Pressurizer safety valve setpoint reached	225
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~2900

15.4.4.2 Analysis of Effects and Consequences

Method of Analysis

Two digital-computer codes are used to analyze this transient. The LOFTRAN[35] code is used to calculate the resulting loop and core coolant flow following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN[23] code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

The following cases are analyzed:

a) Locked Rotor Accident from a nominal Reactor Coolant System heat output of 100% (2787 MWt) with three loops operating.

b) Locked Rotor Accident from a nominal Reactor Coolant System heat output of 60% (1671 MWt) with two loops operating, no loop stop valves closed.

c) Locked Rotor Accident from a nominal Reactor Coolant System heat output of 65% (1810 MWt), loop stop valves in one loop closed.

TABLE 15.4.4-1

SUMMARY OF RESULTS FOR
LOCKED ROTOR TRANSIENTS

	3 Loops Operating Initially	2 Loops Operating Initially, No Isolated Loops	2 Loops Operating Initially, One Isolated Loop
Maximum Primary Coolant System Pressure (psia)	2642	2690	2734
Maximum Clad Temperature (^oF) Core Hot Spot	2165	*	*
Amount of $Zr-H_2O$ at Core Hot Spot (% by weight)	0.6%	*	*

*Substantially lower than for all loops operating

15.4.6.2.3 Results

The values of parameters use in the analysis, as well as the results of the analysis, are presented in Table 15.4.6-1 and discussed below.

Beginning of Cycle Full Power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were 0.20%ΔK and 7.07 respectively. The peak hot spot clad average temperature was 2562°F. The peak hot spot fuel center temperature exceeded the beginning of life melt temperature of 4900°F. However, melting was restricted to less than 10% of the pellet.

Beginning of Cycle, Zero Power

For this condition, control bank D was assumed to be fully inserted and C was at its insertion limit. The worst ejected rod is located in control bank D and has a worth of 0.785%ΔK and a hot channel factor of 13.0. The peak hot spot clad temperature reached only 2270°F.

End of Cycle, Full Power

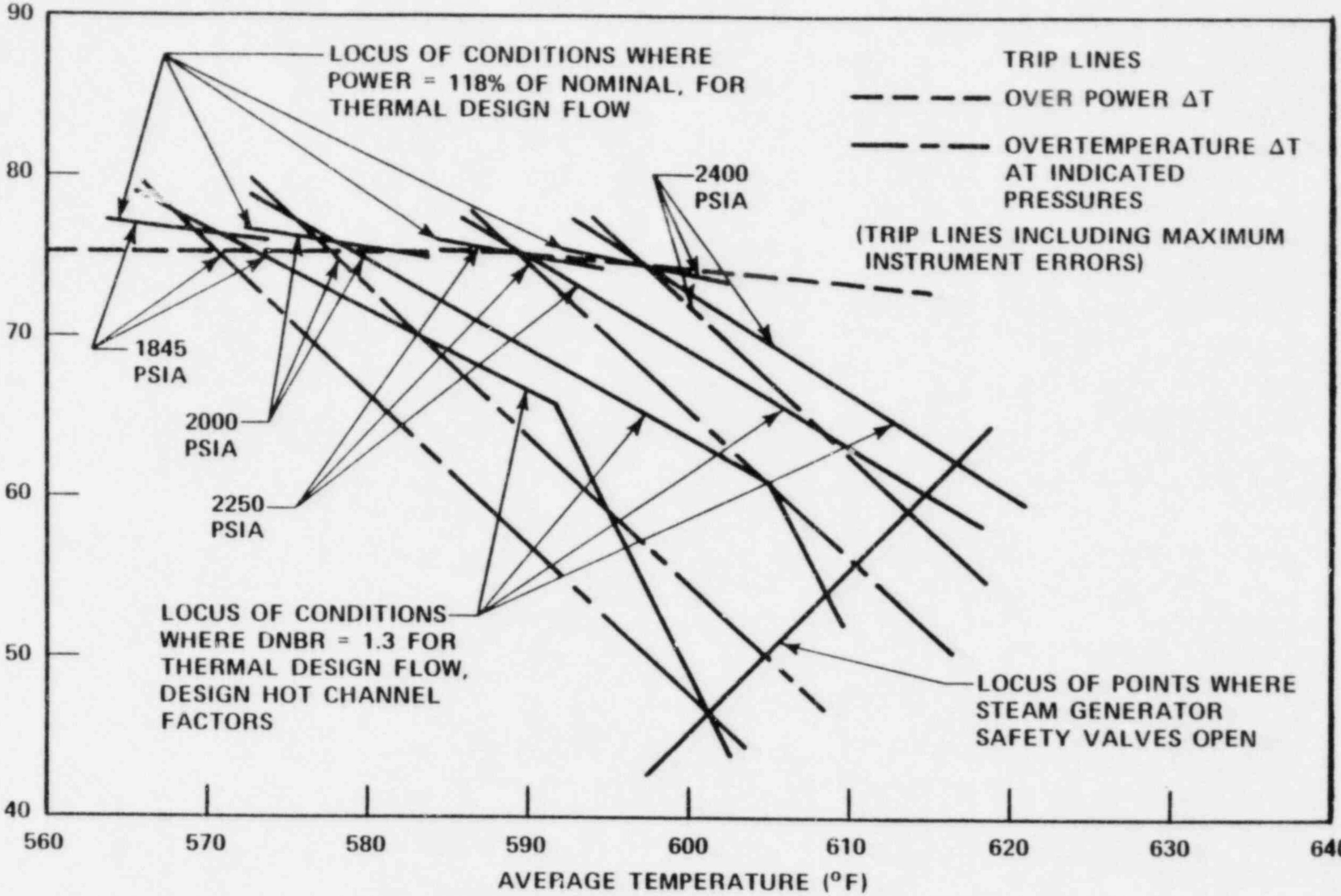
Control bank D was assumed to be inserted to its insertion limit. Conservative values of ejected rod worth and hot channel factor, 0.21%ΔK and 7.88 respectively, were used. This resulted in a peak clad temperature of 2562°F.

TABLE 15.4.6-1

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY

EJECTION ACCIDENT

Time in Life	Beginning	Beginning	End	End
Power Level	102%	0%	102%	0%
Ejected Rod Worth %Δk	0.20	0.785	0.21	0.98
Delayed neutron fraction %	0.52	0.52	0.44	0.44
Feedback reactivity weighting	1.30	2.40	1.60	3.55
Trip Reactivity % Δk	4.0	2.0	4.0	2.0
F_q before rod ejection	2.55	--	2.55	--
F_q after rod ejection	7.07	13.0	7.88	18.7
Number of operational pumps	3	2	3	2
Max. fuel pellet average temperature, ^oF	4053	3055	4067	3495
Max. fuel center temperature, ^oF	4900	3560	4800	3985
Max. clad average temperature, ^oF	2562	2270	2562	2670
Max. fuel stored energy, cal/gm	177.1	127	178	148



17,6C4-1

Figure 15.1-1. Illustration of Overtemperature and Overpower ΔT Protection

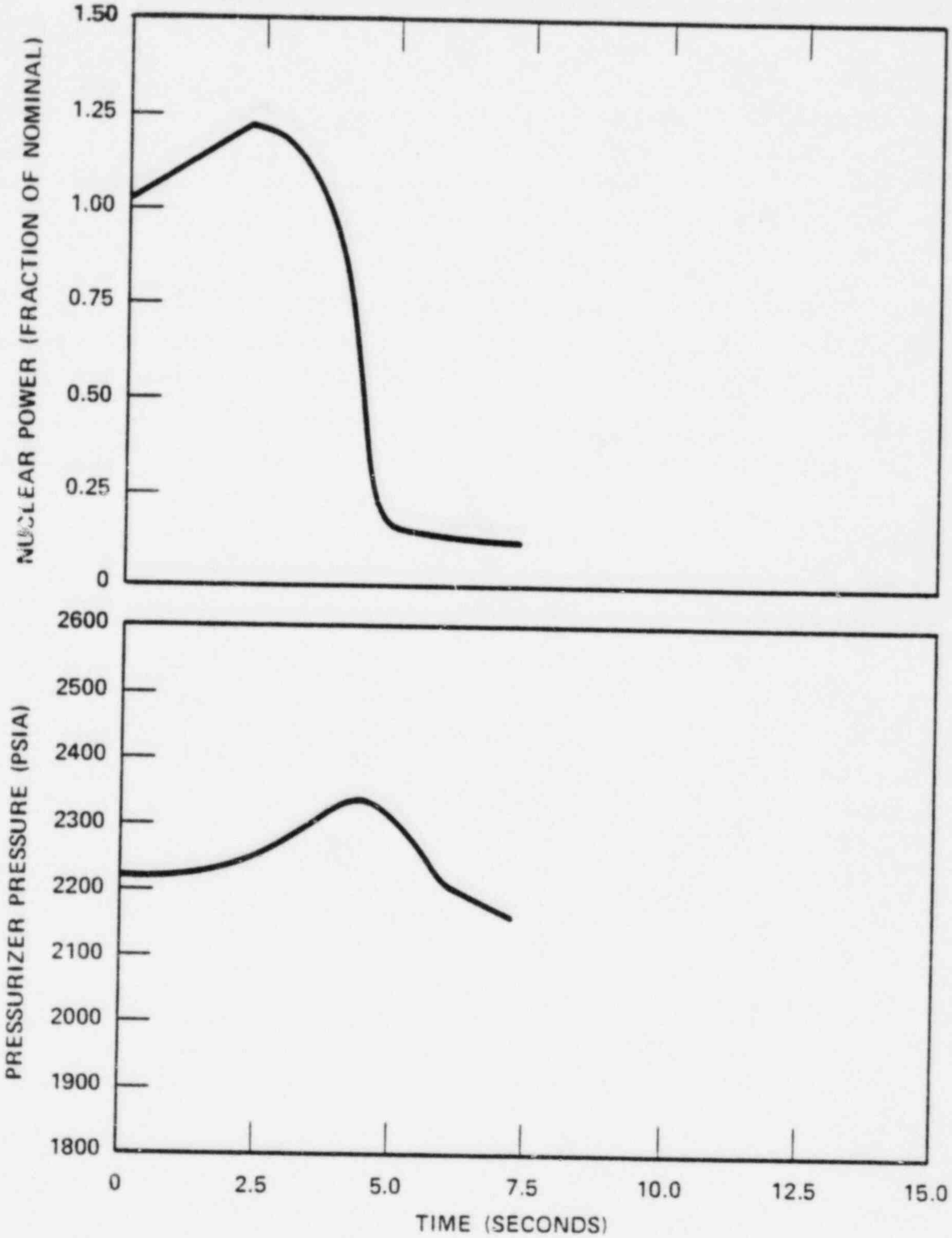


Figure 15.2-4 Transient Response for Uncontrolled Rod Withdrawal from Full Power With Minimum Reactivity Feedback and 70 pcm/sec Withdrawal Rate

17,537-2

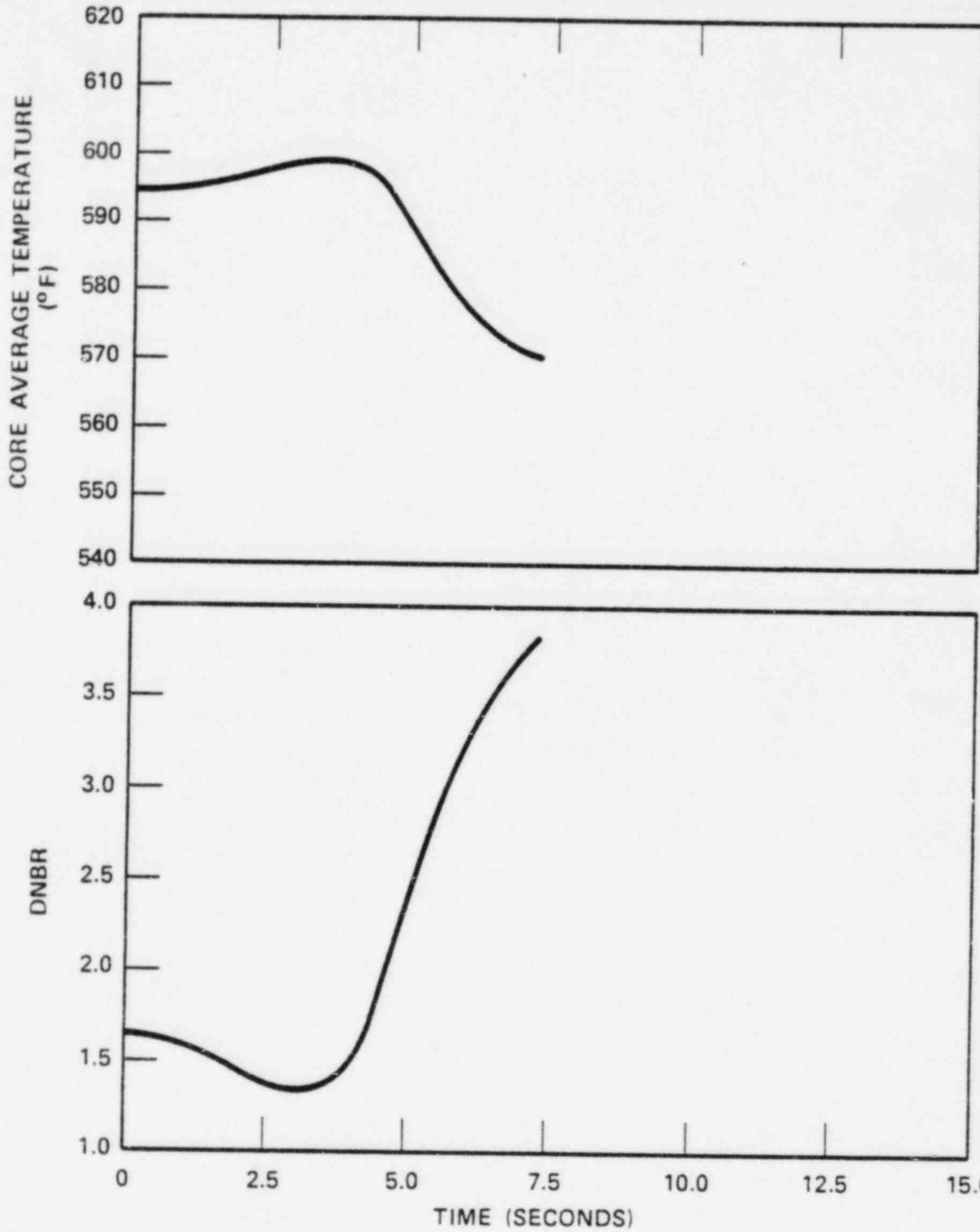


Figure 15.2-5 Transient Response for Uncontrolled Rod Withdrawal from Full Power With Minimum Reactivity Feedback and 70 pcm/sec Withdrawal Rate

17,537-3

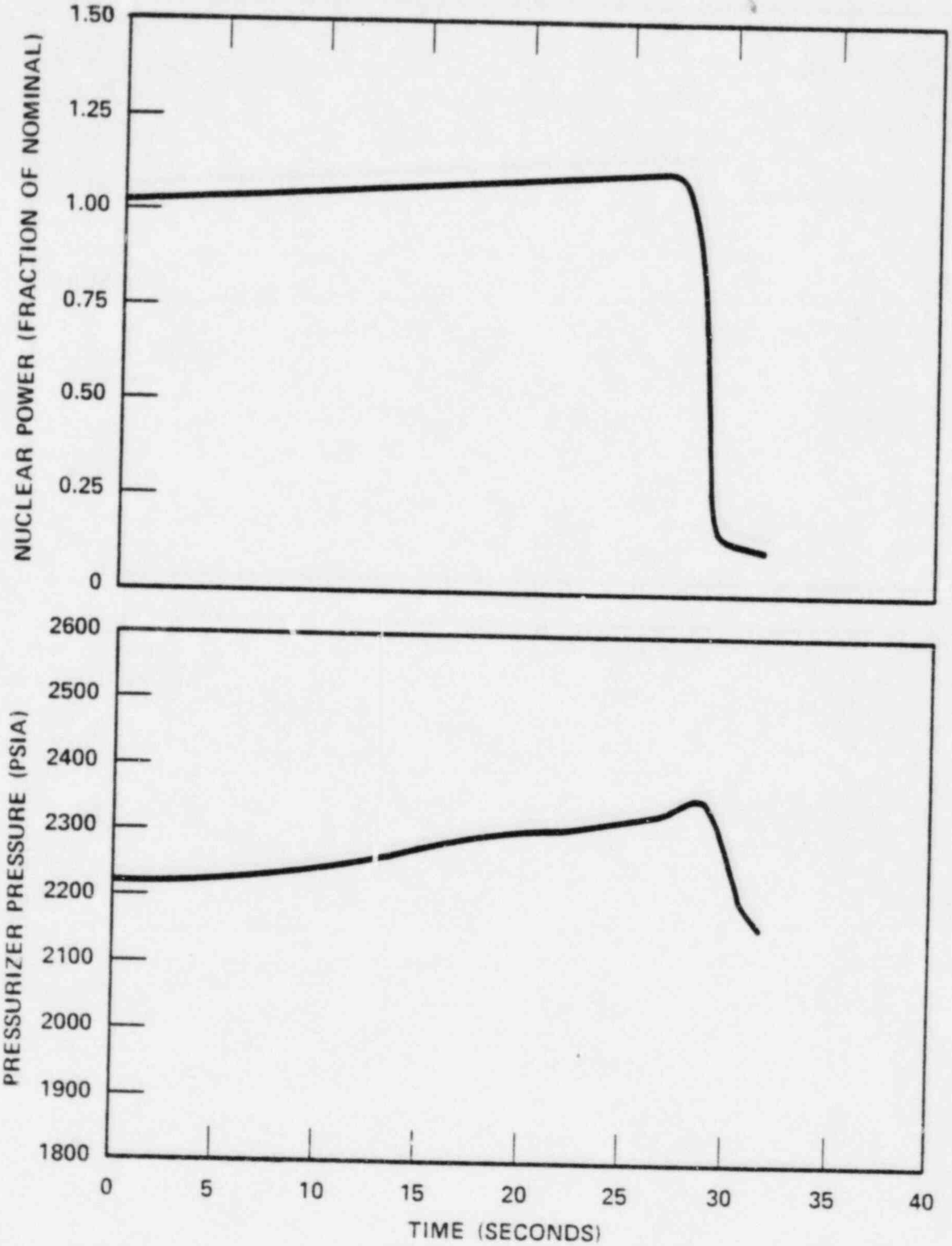


Figure 15.2-6 Transient Response for Uncontrolled Rod Withdrawal from Full Power With Minimum Feedback and 3 pcm/sec Withdrawal Rate

17,537-4

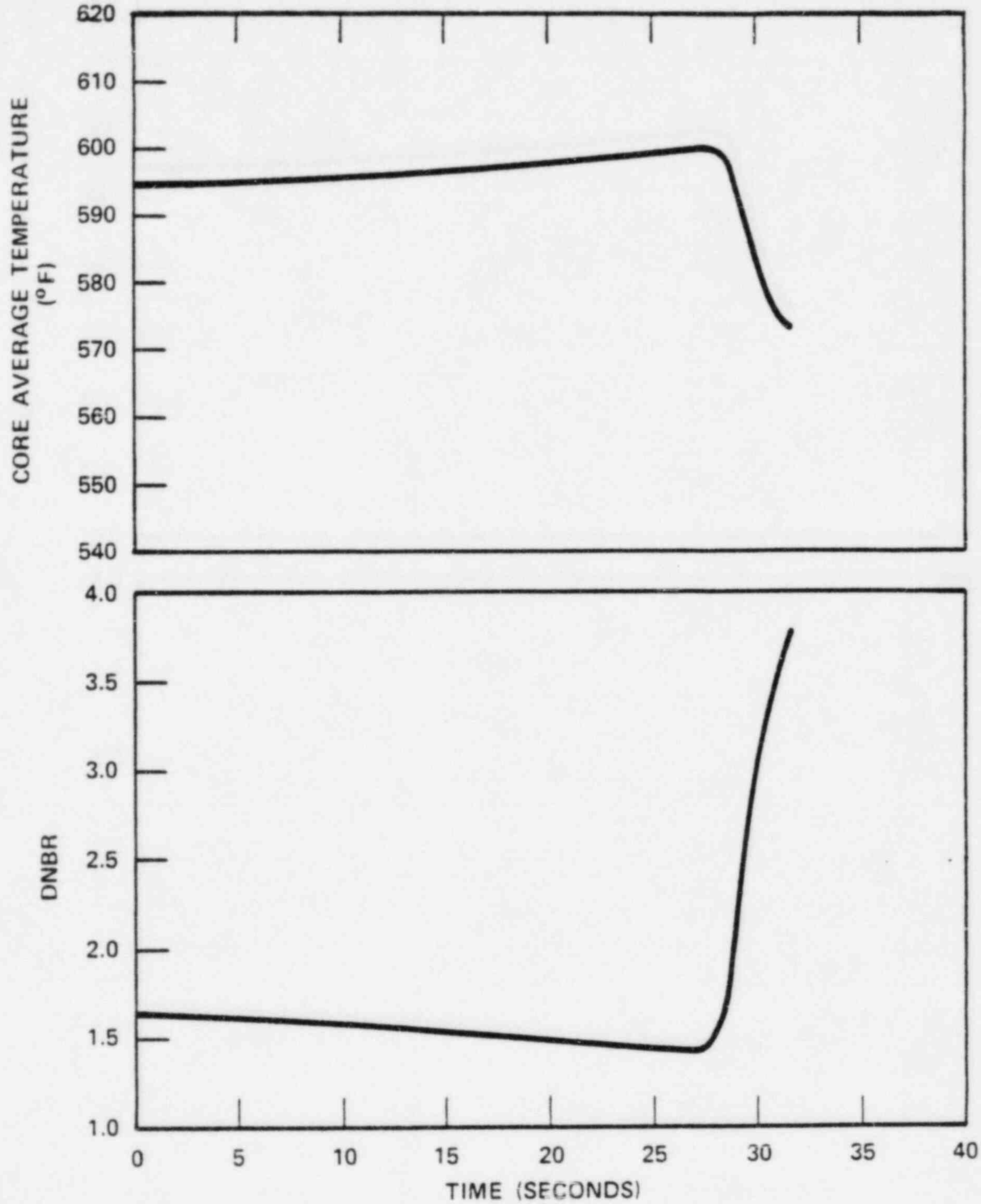


Figure 15.2-7 Transient Response for Uncontrolled Rod Withdrawal from Full Power With Minimum Reactivity Feedback and 3 pcm/sec Withdrawal Rate

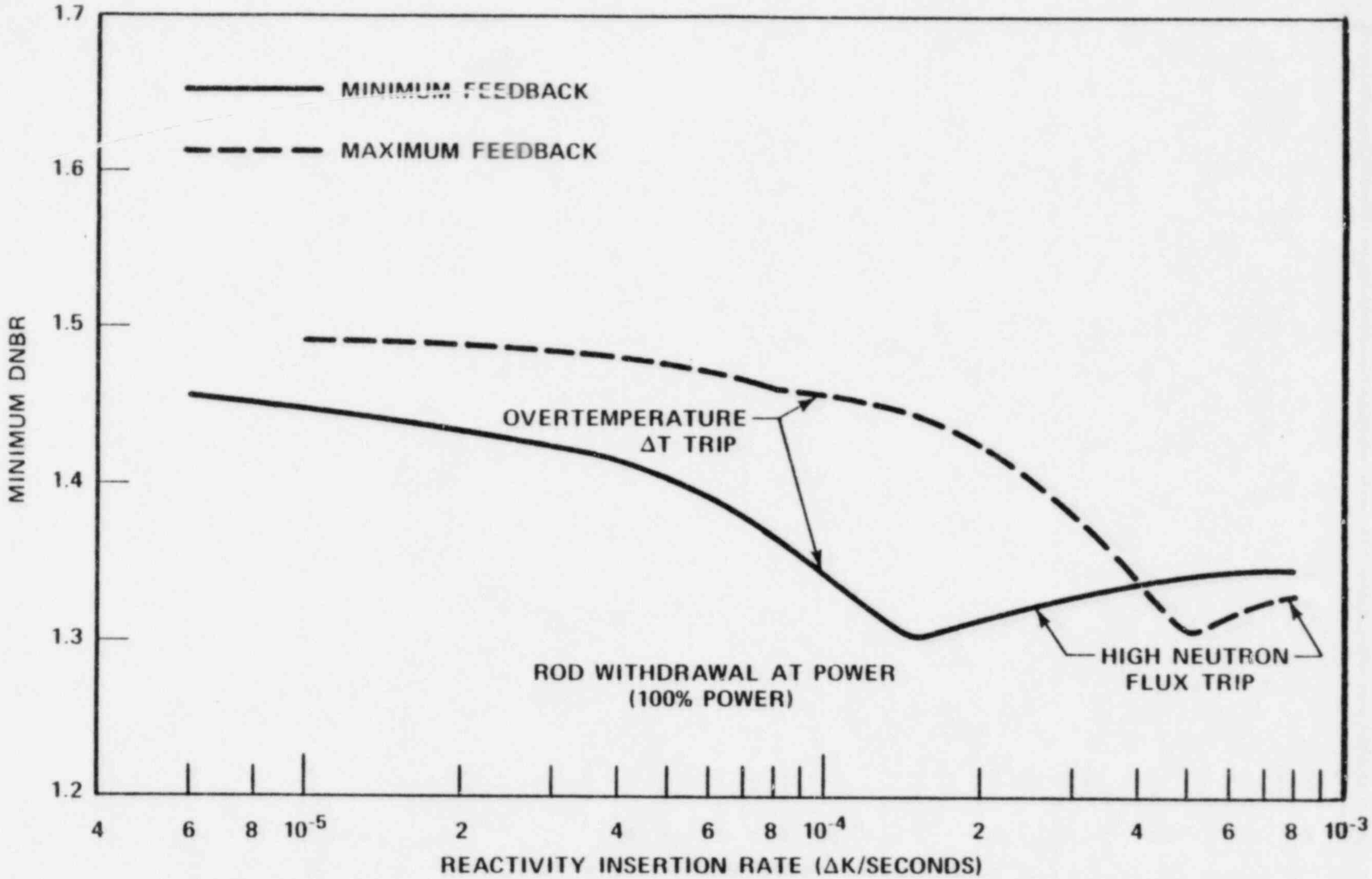


Figure 15.2-8. Effect of Reactivity Insertion Rate on Minimum DNBR for a Rod Withdrawal Accident from 100% Power

17,664-3

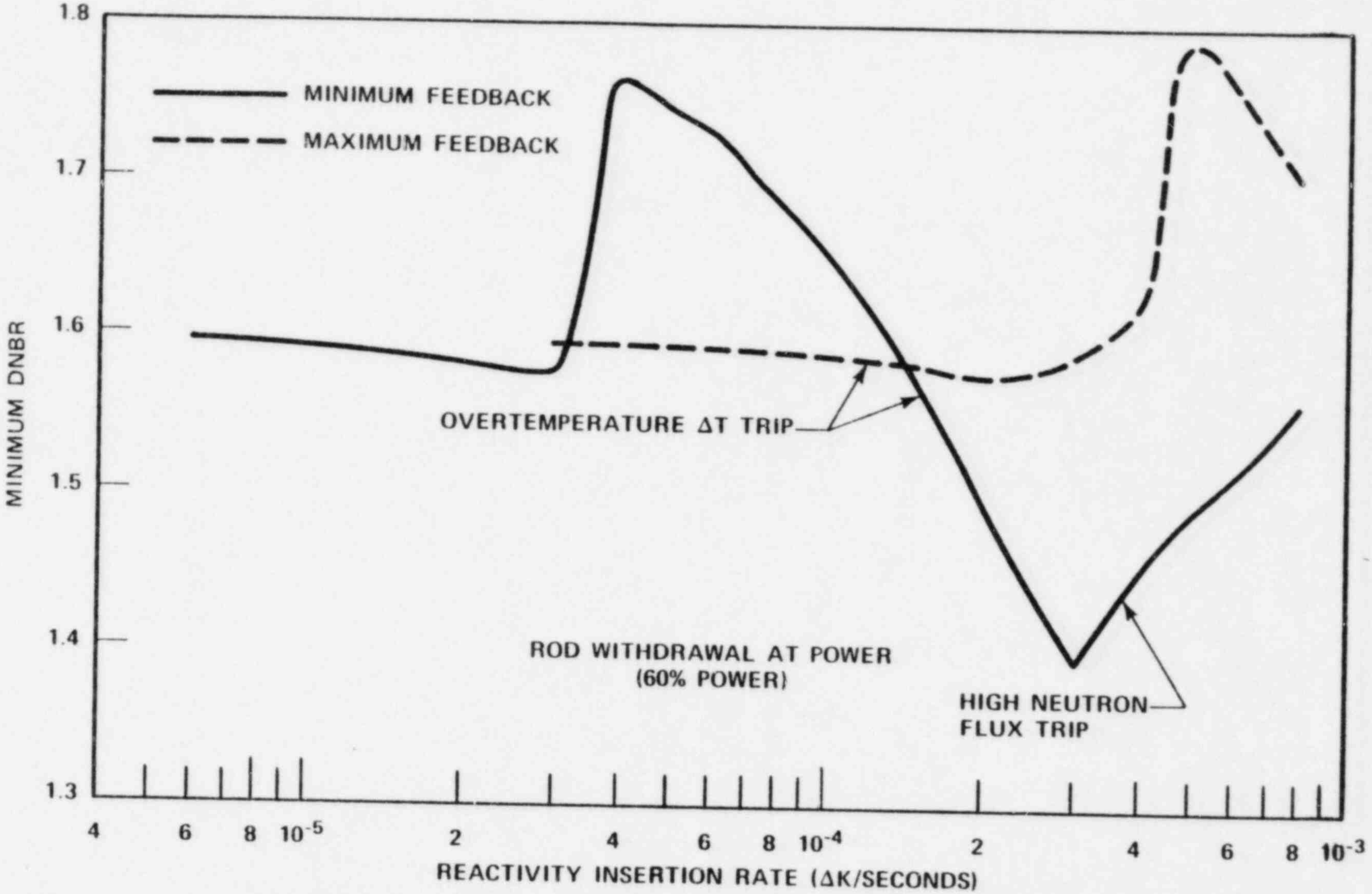


Figure 15.2-9. Effect of Reactivity Insertion Rate on Minimum DNBR for a Rod Withdrawal Accident from 60% Power

17,664-4

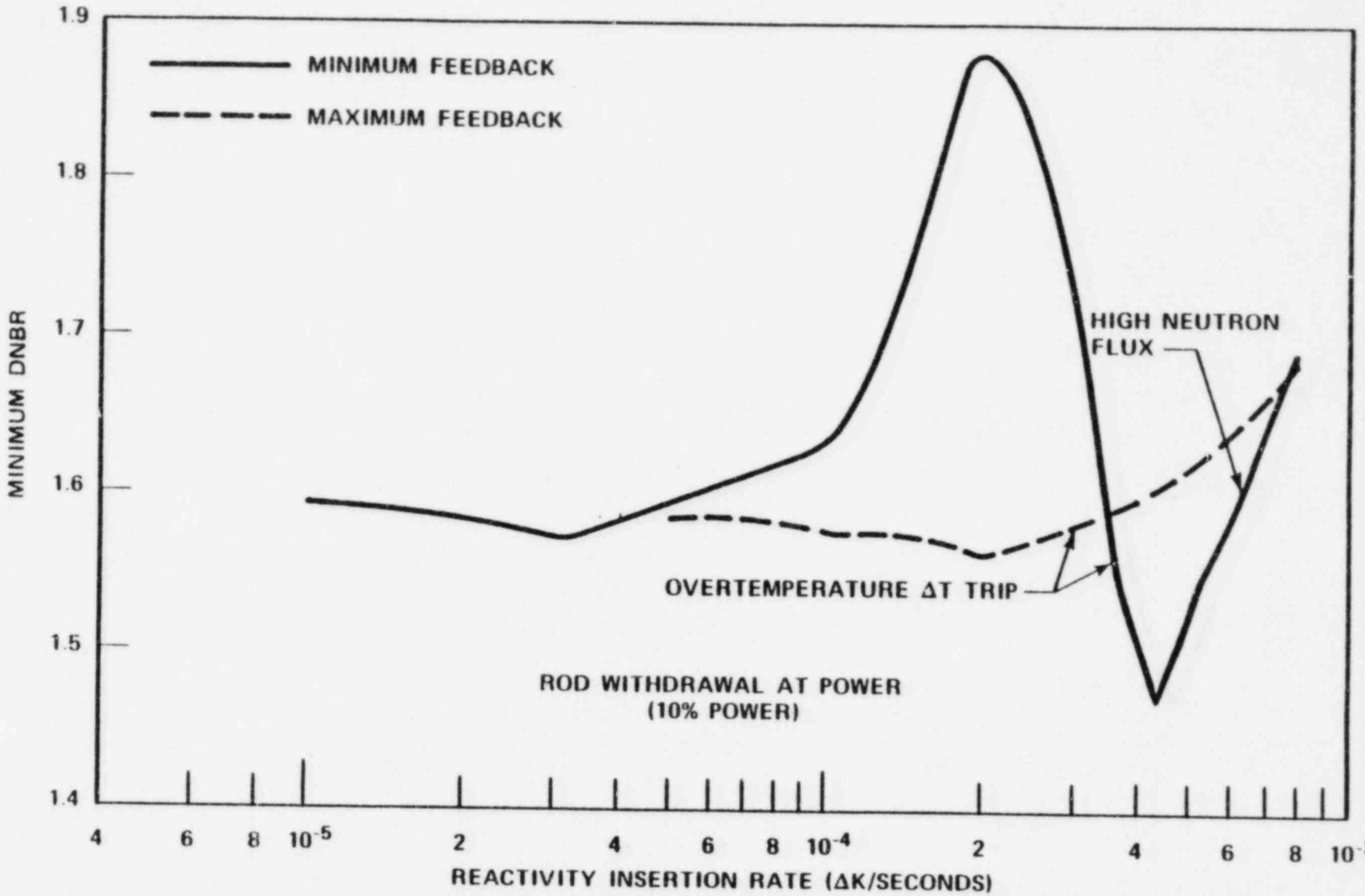


Figure 15.2-10. Effect of Reactivity Insertion Rate on Minimum DNBR for Rod Withdrawal Accident from 10% Power

17,664-2

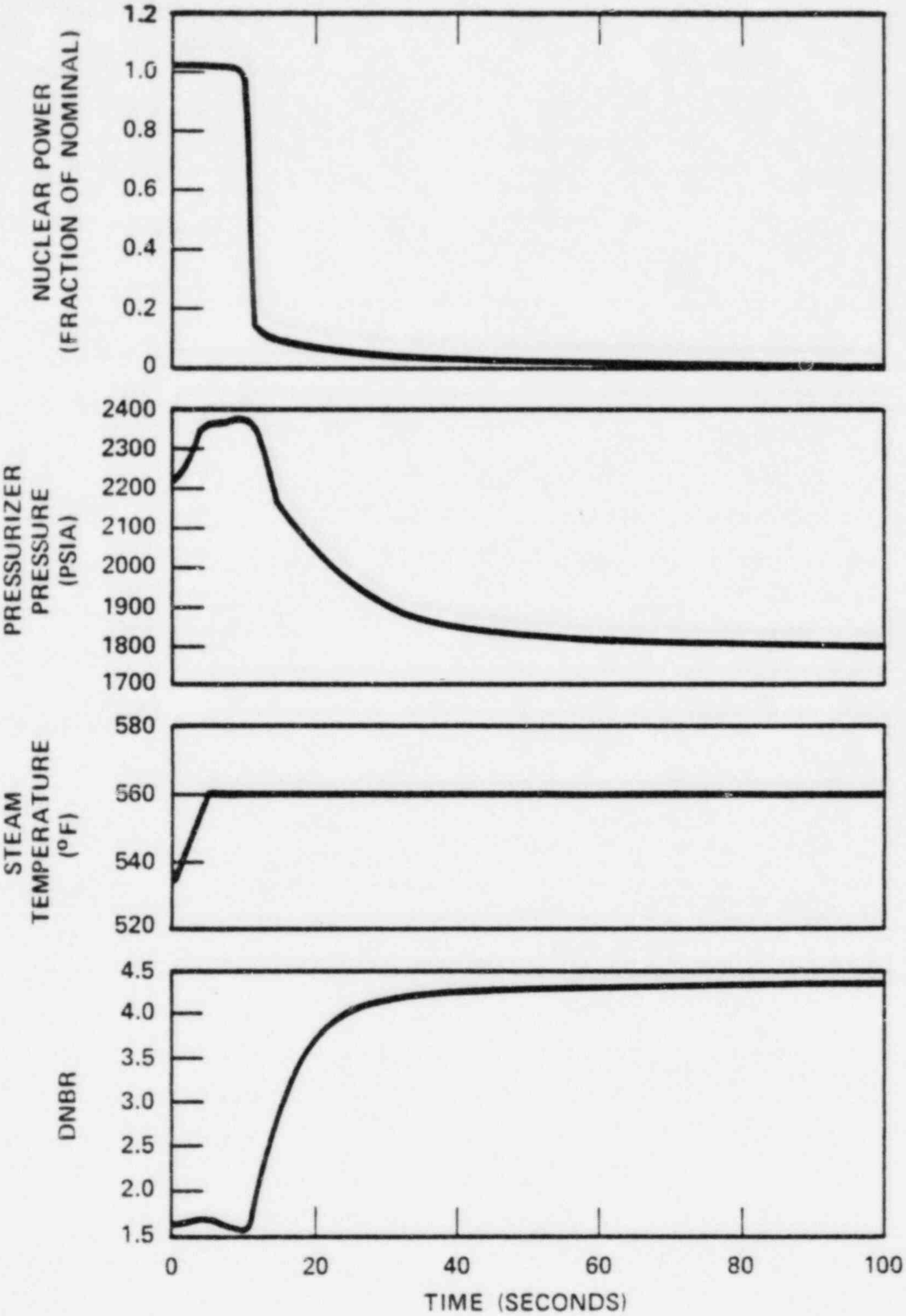


Figure 15.2-23 Loss of Load Accident, With Pressurizer Spray and Power Operated Relief Valves, Beginning of Life

17,537-6

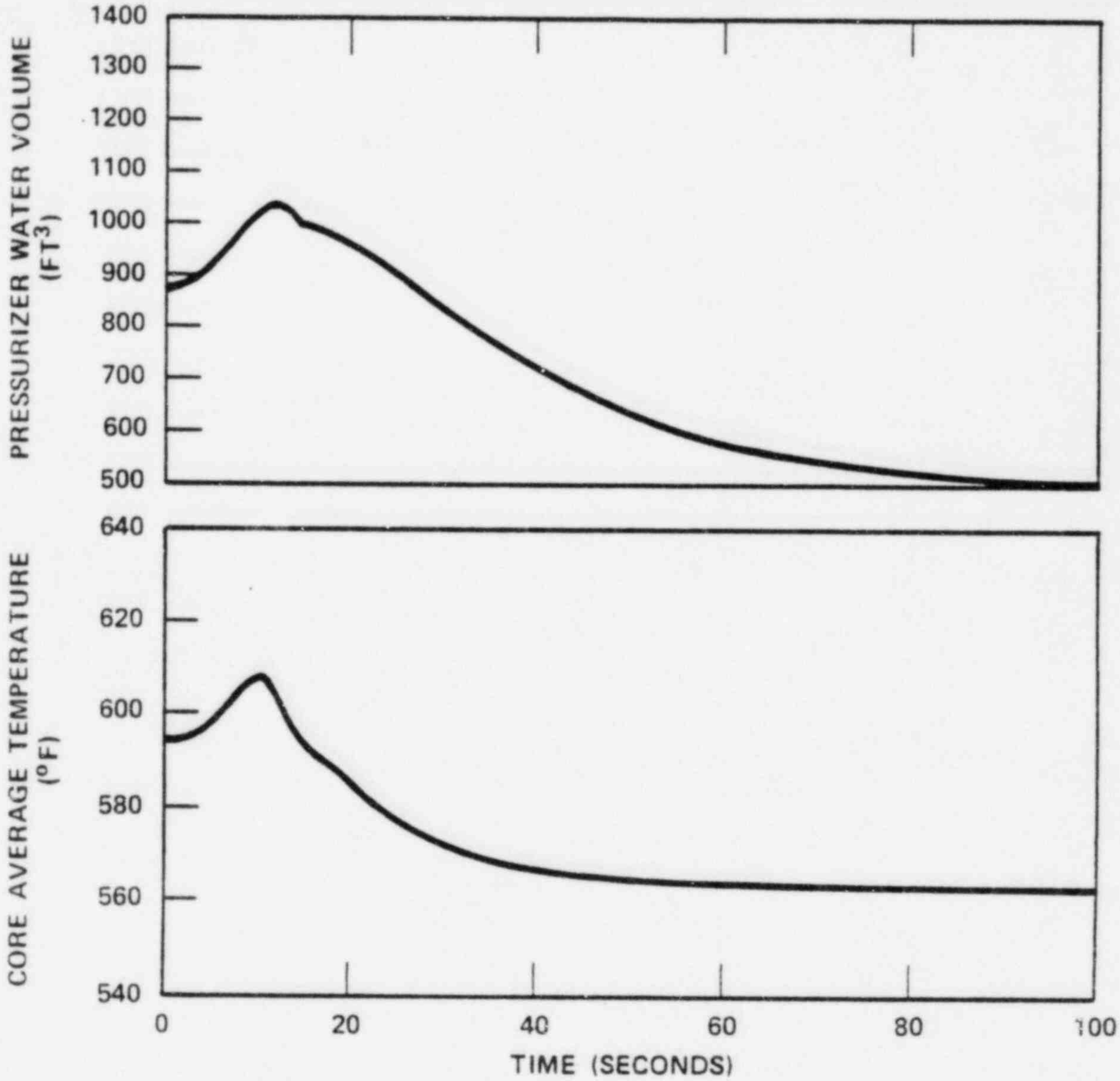


Figure 15.2-24 Loss of Load Accident, With Pressurizer Spray and Power Operated Relief Valves, Beginning of Life

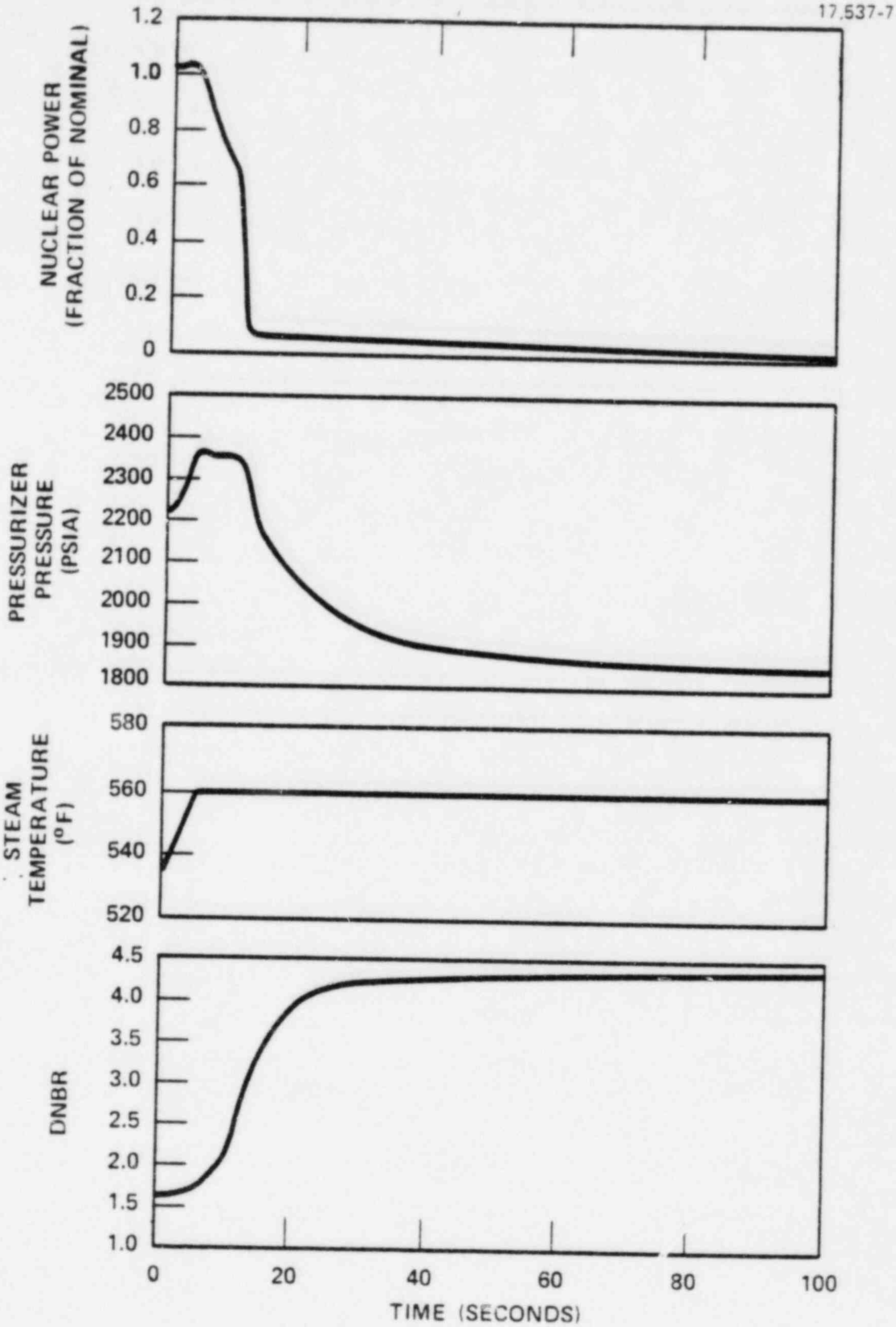


Figure 15.2-25 Loss of Load Accident, With Pressurizer Spray and Power Operated Relief Valves, End of Life

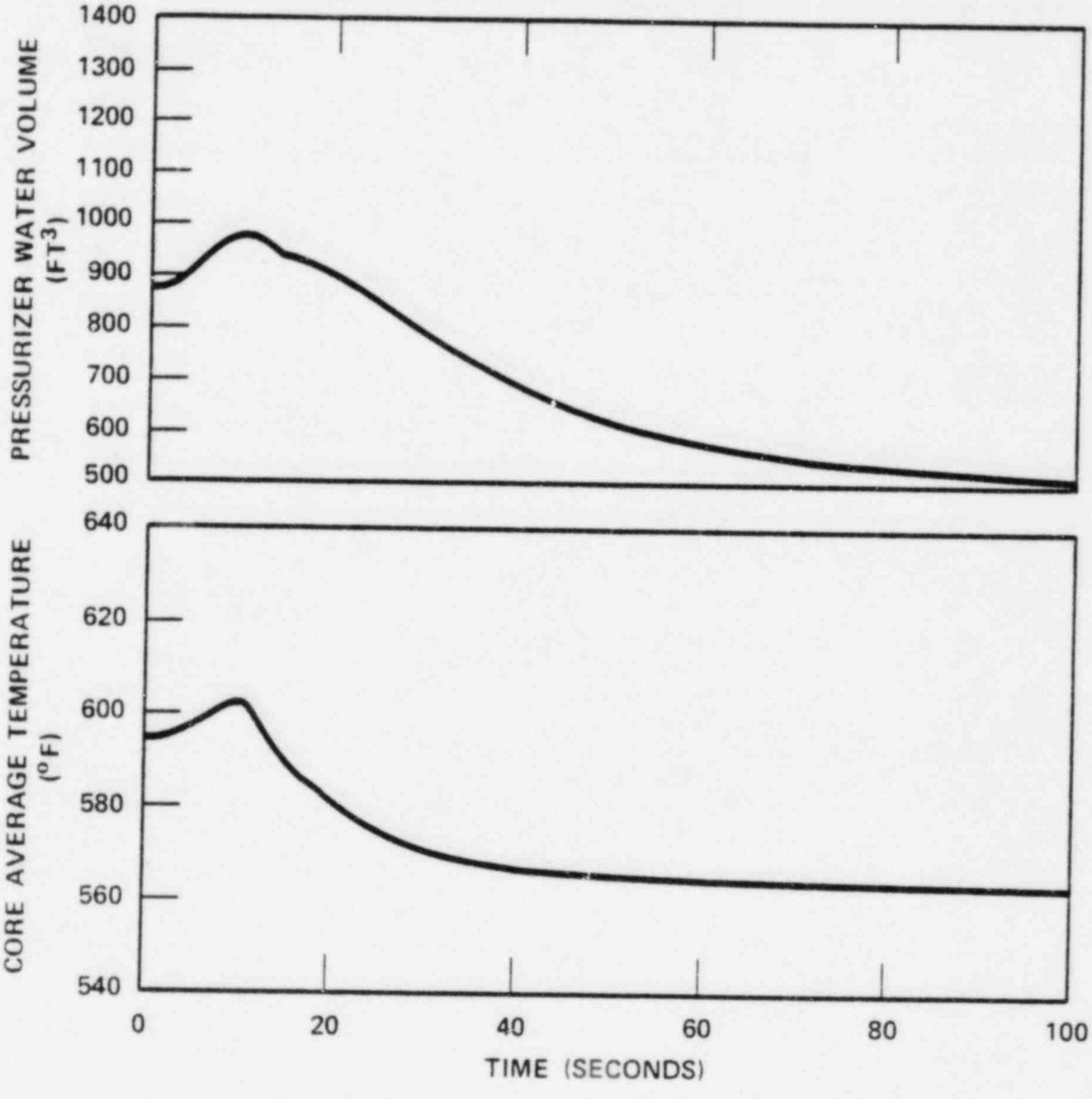


Figure 15.2-26 Loss of Load Accident, With Pressurizer Spray and Power Operated Relief Valves, End of Life

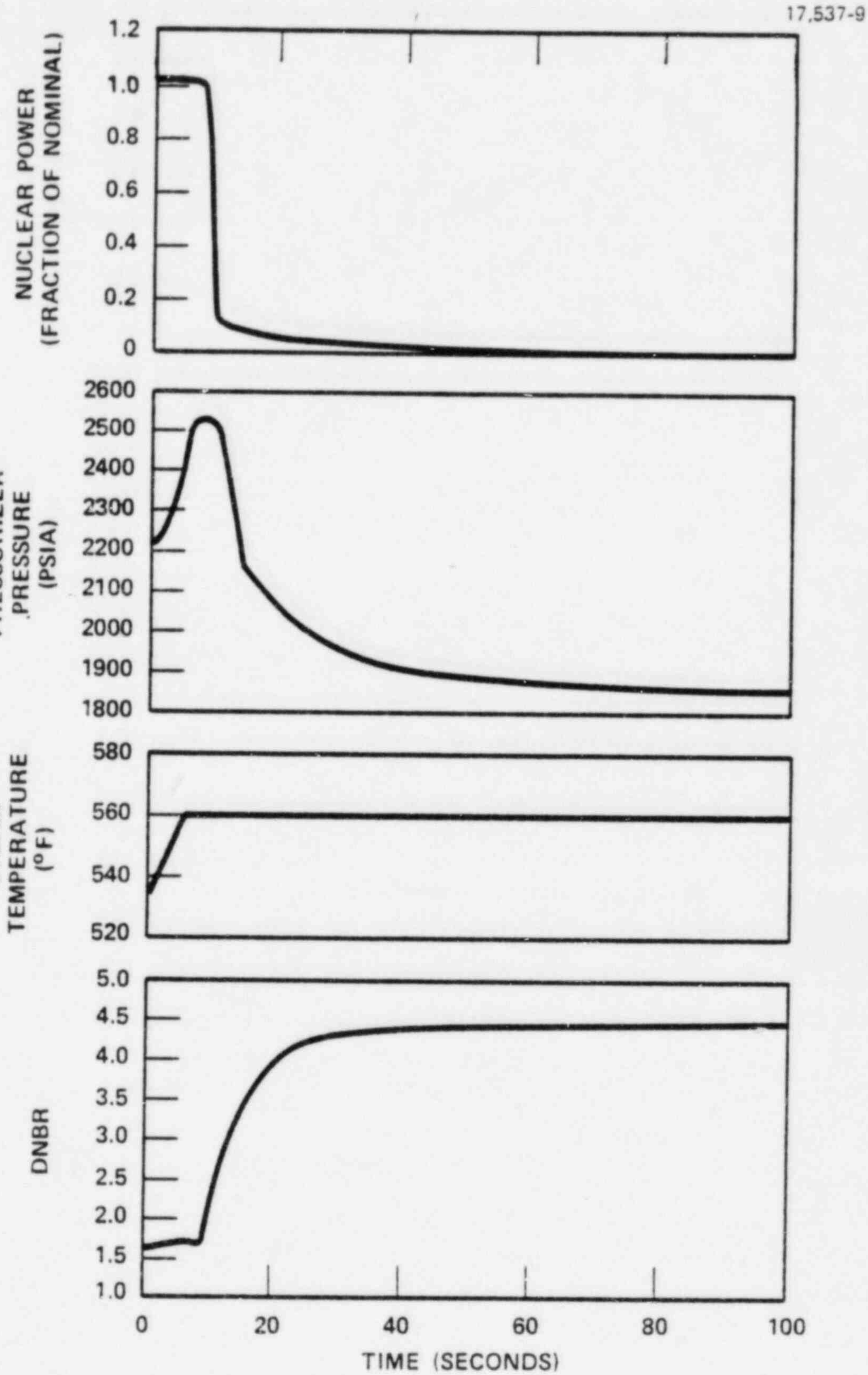


Figure 15.2-27 Loss of Load Accident, Without Pressurizer Spray and Power Relief Valves, Beginning of Life

17,537-10

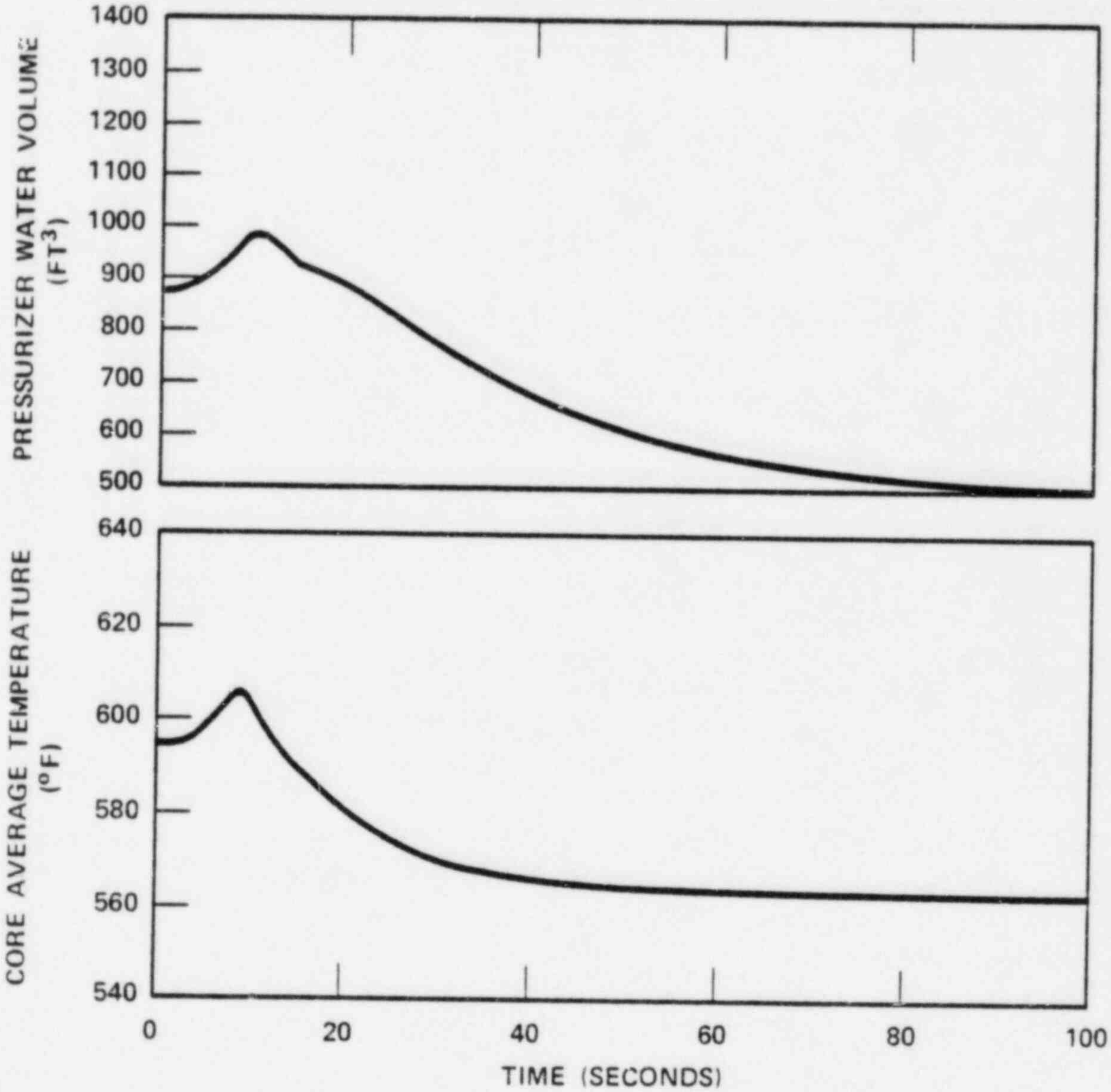


Figure 15.2-28 Loss of Load Accident, Without Pressurizer Spray and Power Operated Relief Valves, Beginning of Life

17,537-11

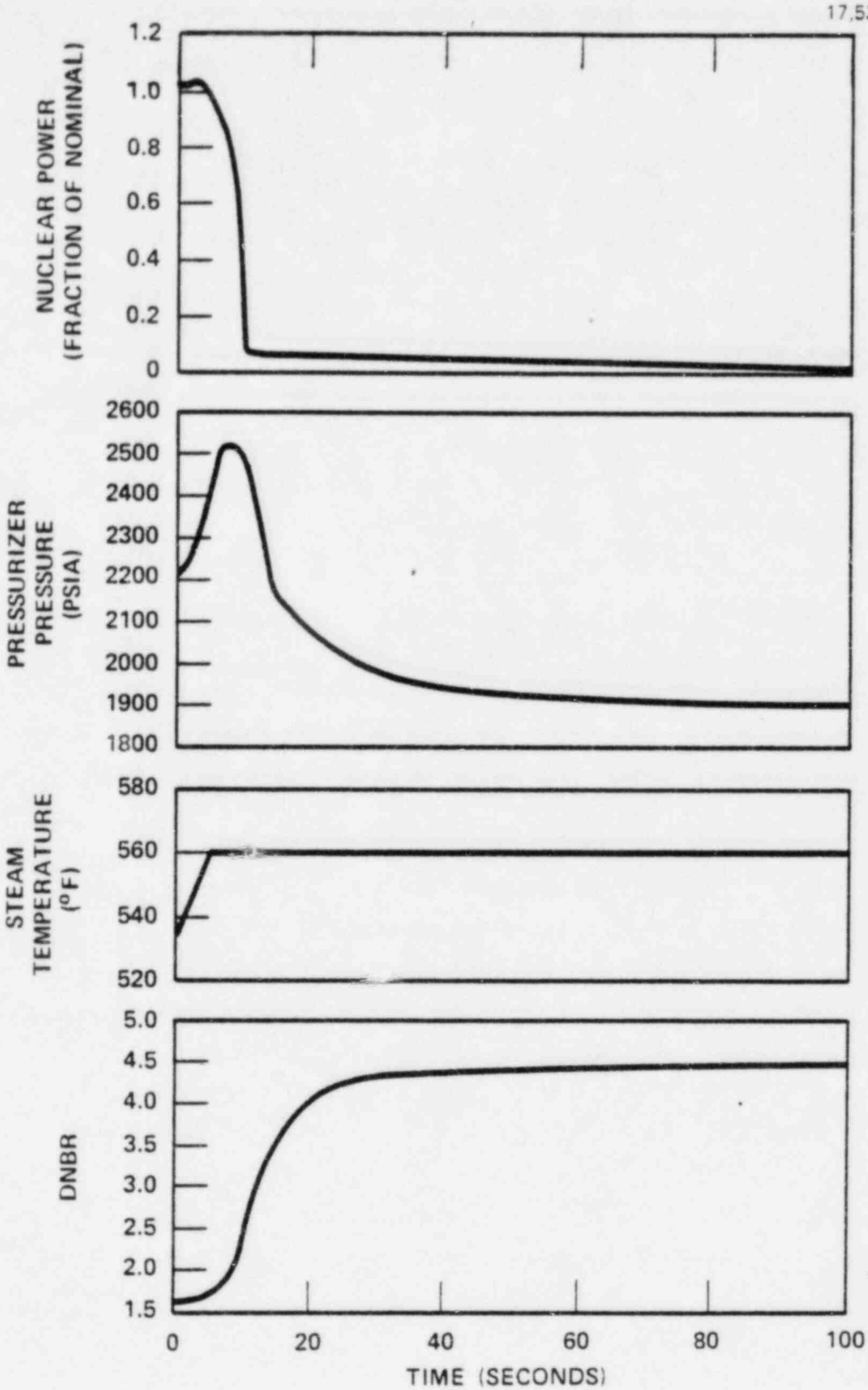


Figure 15.2-29 Loss of Load Accident, Without Pressurizer Spray and Power Operated Relief Valves, End of Life

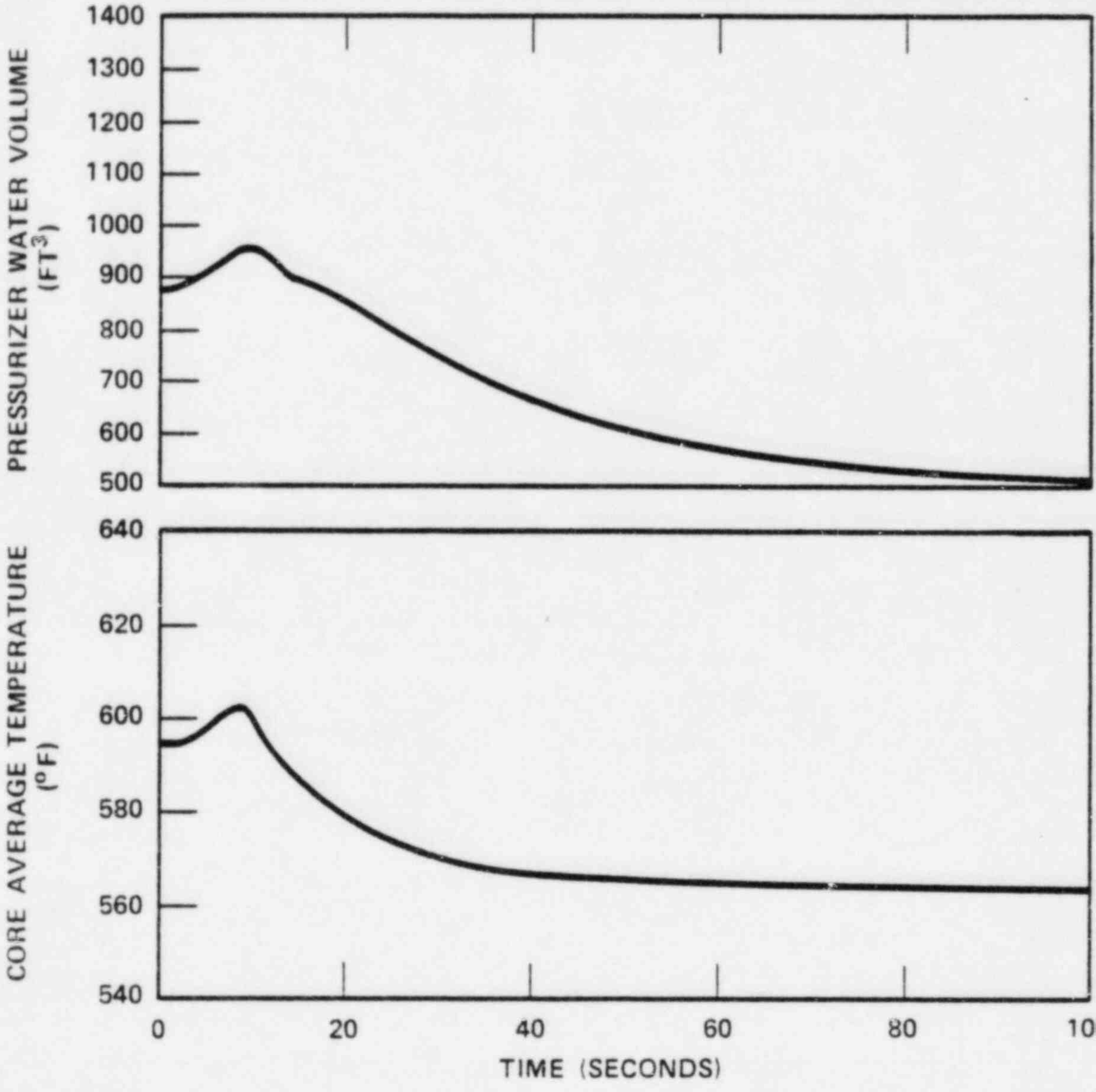


Figure 15.2-30 Loss of Load Accident, Without Pressurizer Spray and Power Operated Relief Valve, End of Life

21323-1

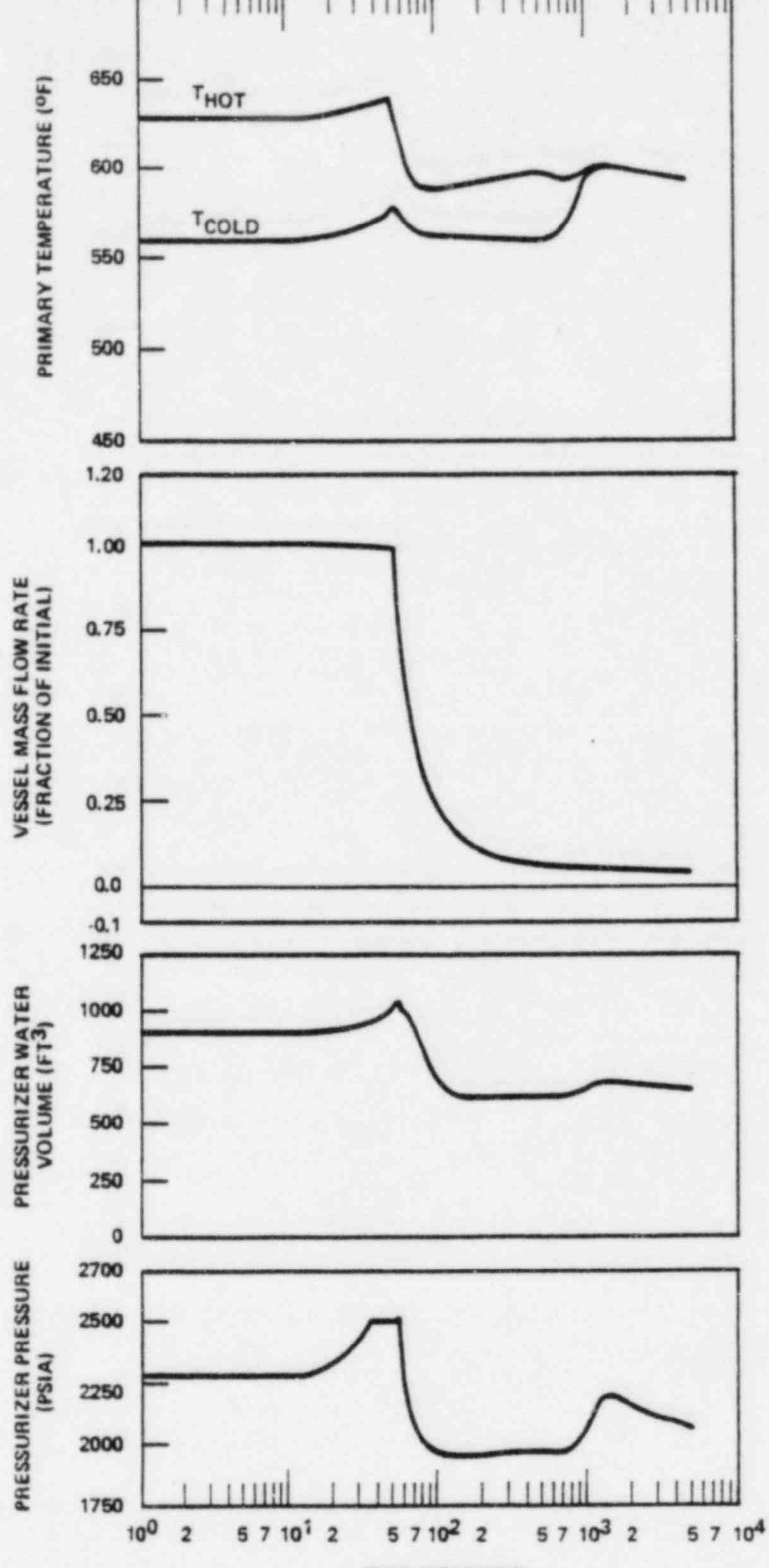


Figure 15.2-31. Primary Temperature, of Loops not Receiving Auxiliary Feedwater, Vessel Mass Flow Rate, Pressurizer Pressure and Water Volume as a Function of Time, Loss of Normal Feed Accident (Without Offsite Power Available)

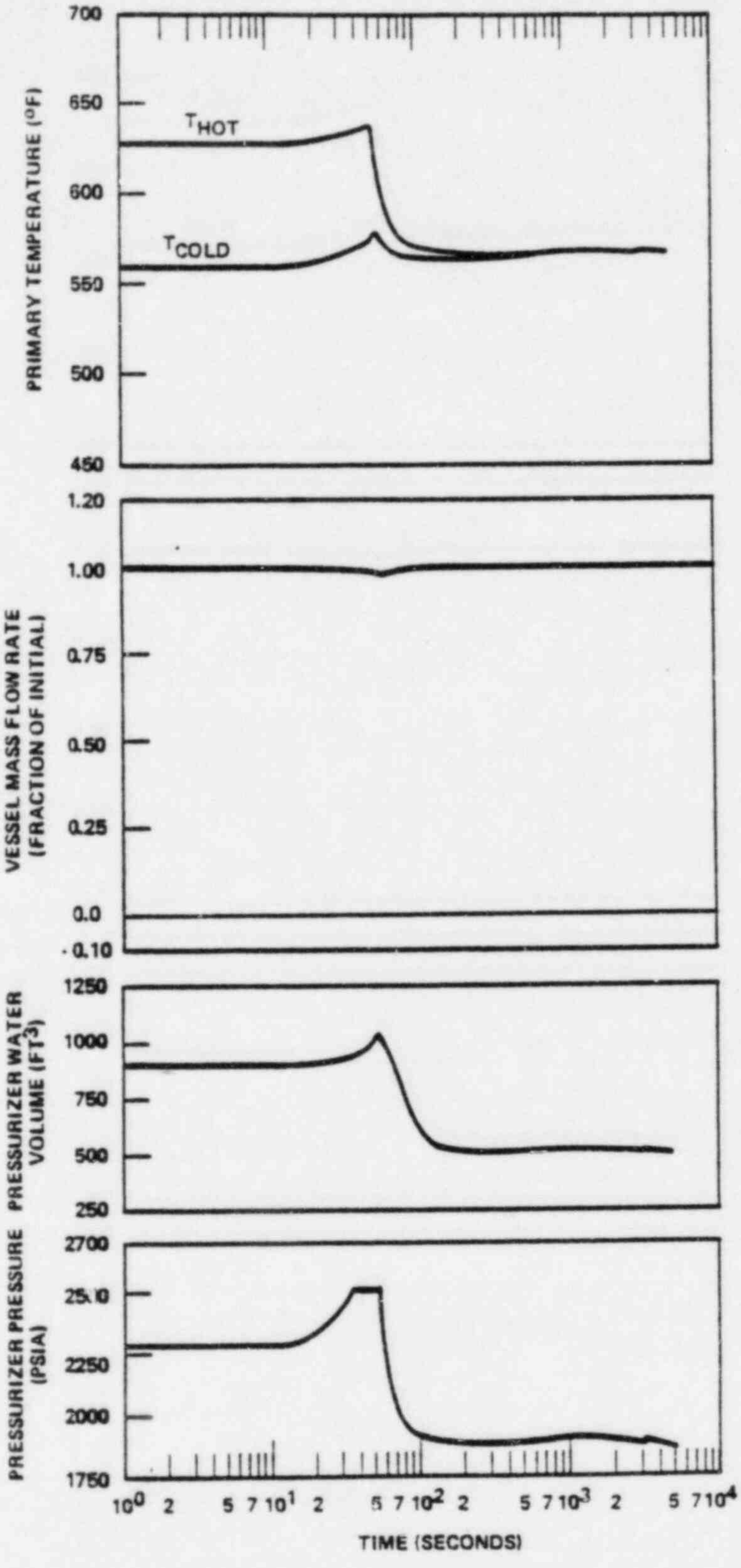


Figure 15.2-31a. Primary Temperature, of Loops not Receiving Auxiliary Feedwater, Vessel Mass Flow Rate, Pressurizer Pressure and Water Volume as a Function of Time, Loss of Normal Feed Accident (With Offsite Power Available)

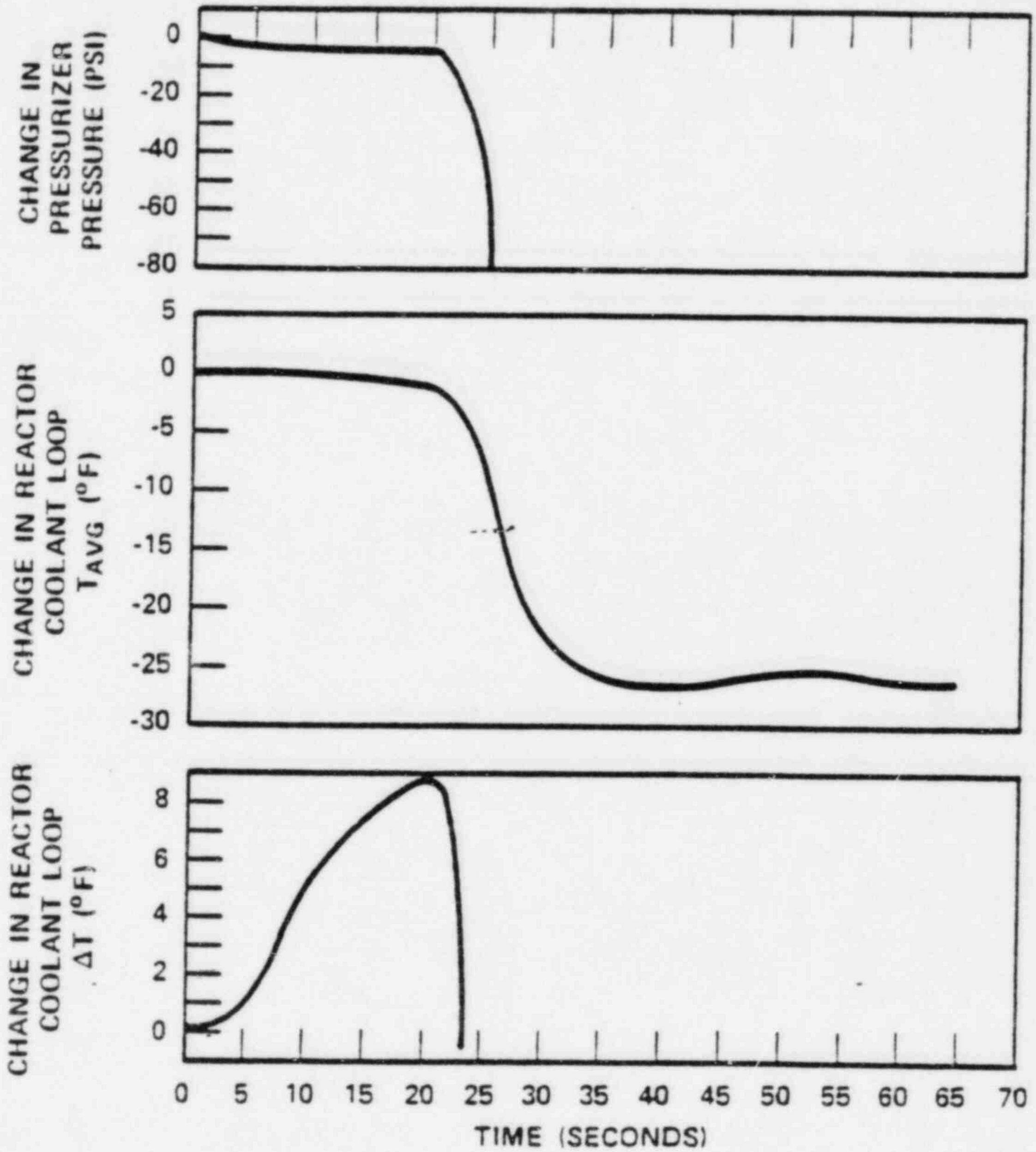


Figure 15.2-32 Feedwater Control Valve Malfunctions 17 X 17 Core

17,664-14

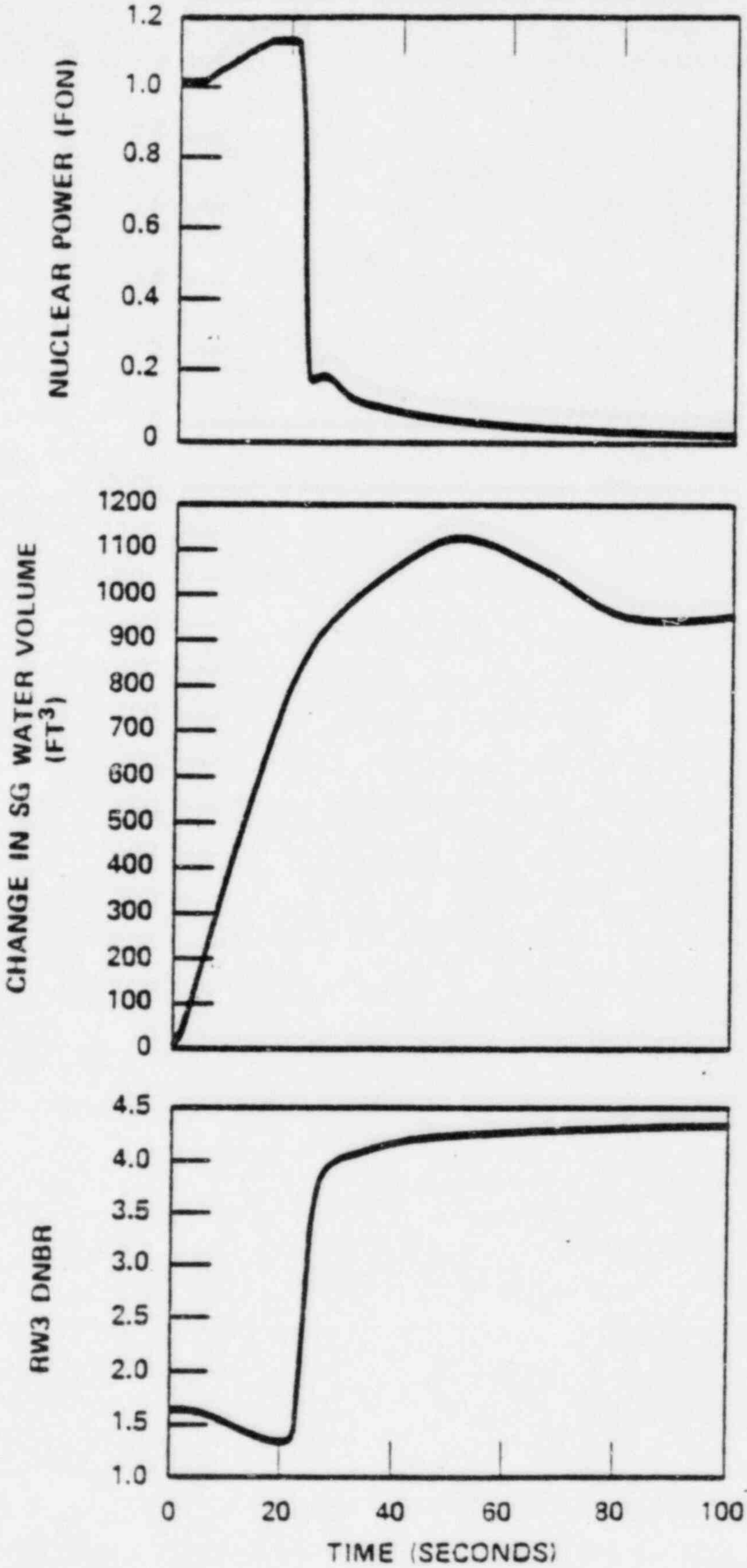


Figure 15.2-32. Feedwater Control Valve Malfunctions 17 X 17 Core (Continued)

17,537-13

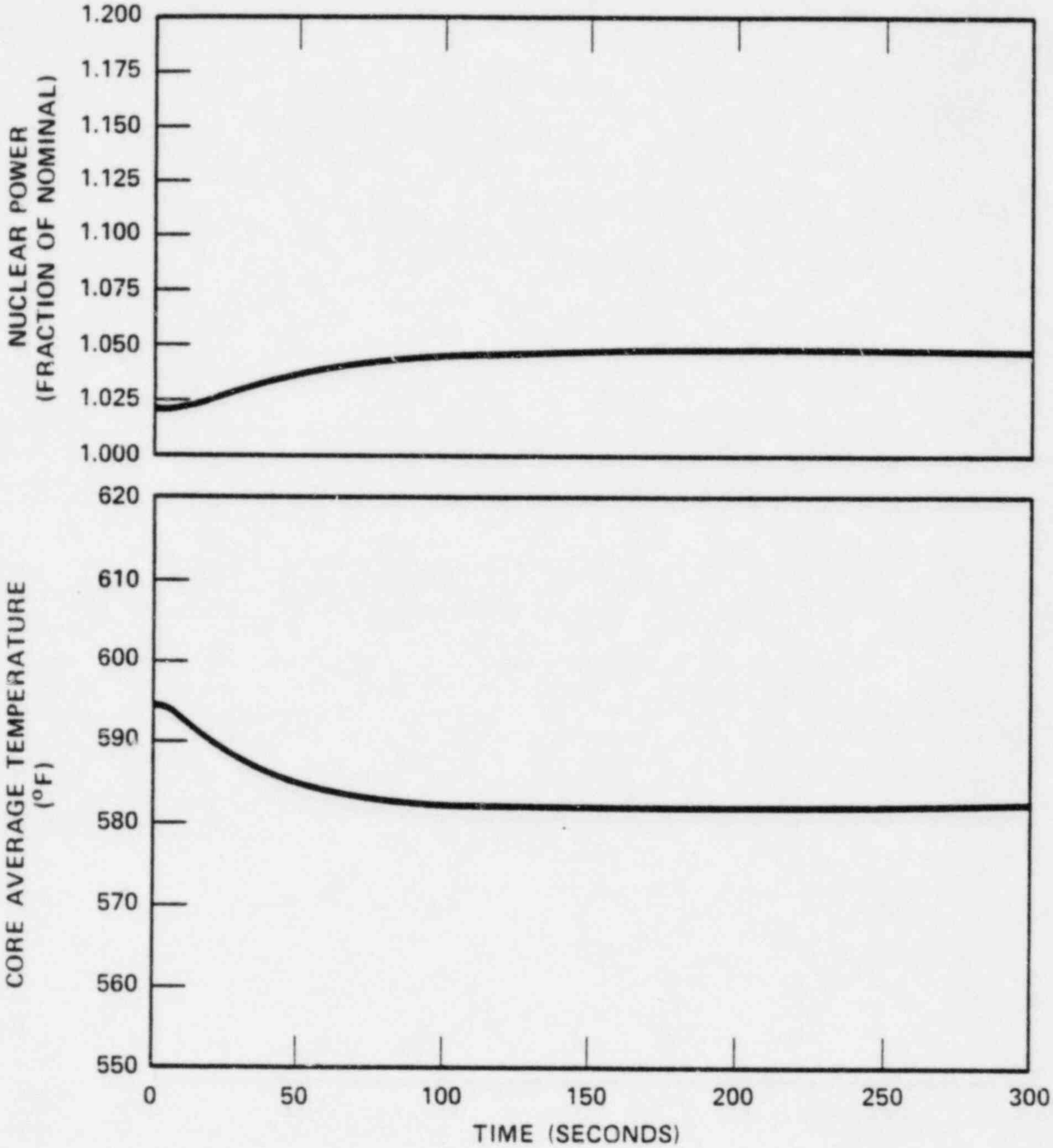


Figure 15.2-33 Excessive Load Increase Without Rod Control, Beginning of Life

17,537-14

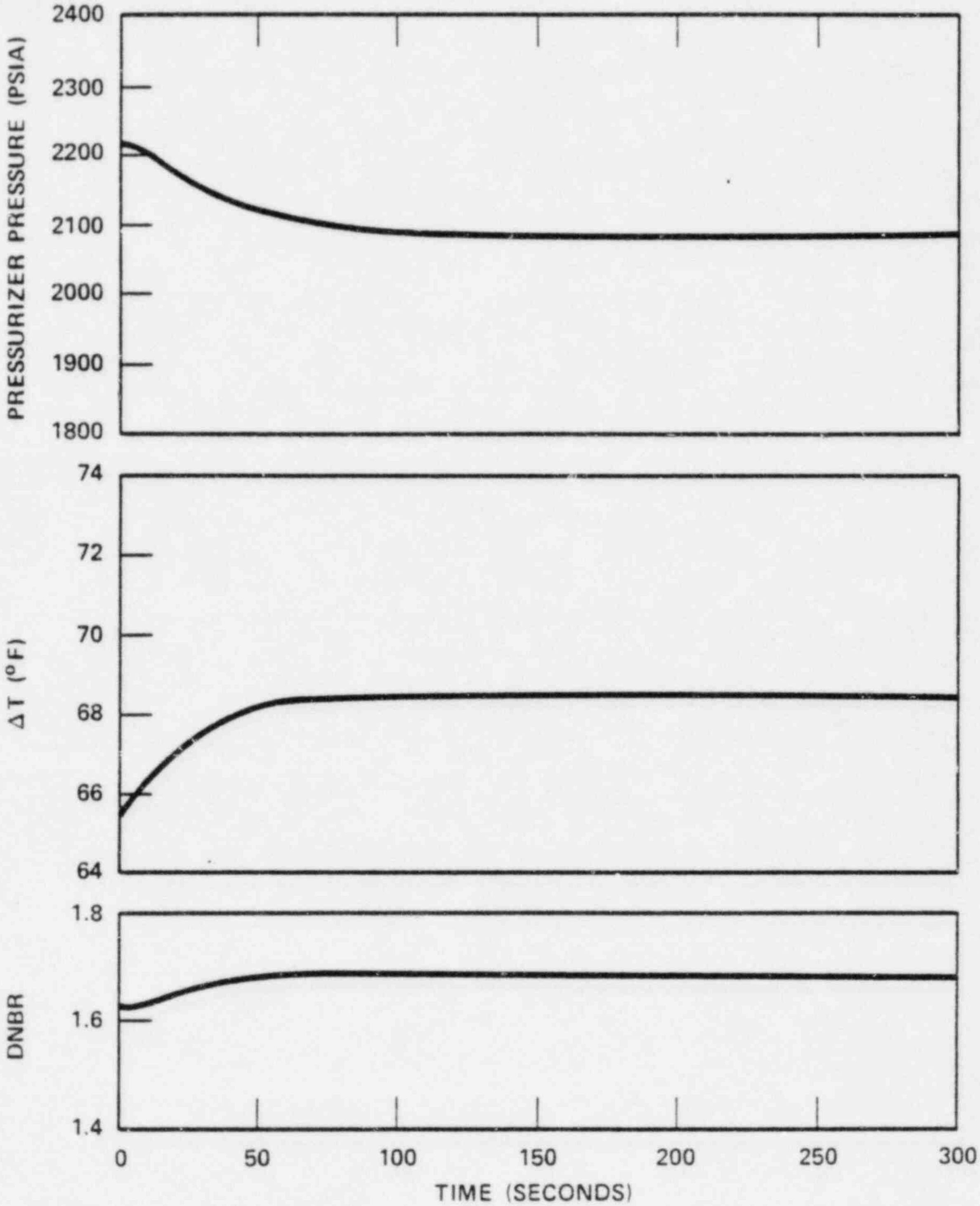


Figure 15.2-34 Excessive Load Increase Without Rod Control, Beginning of Life

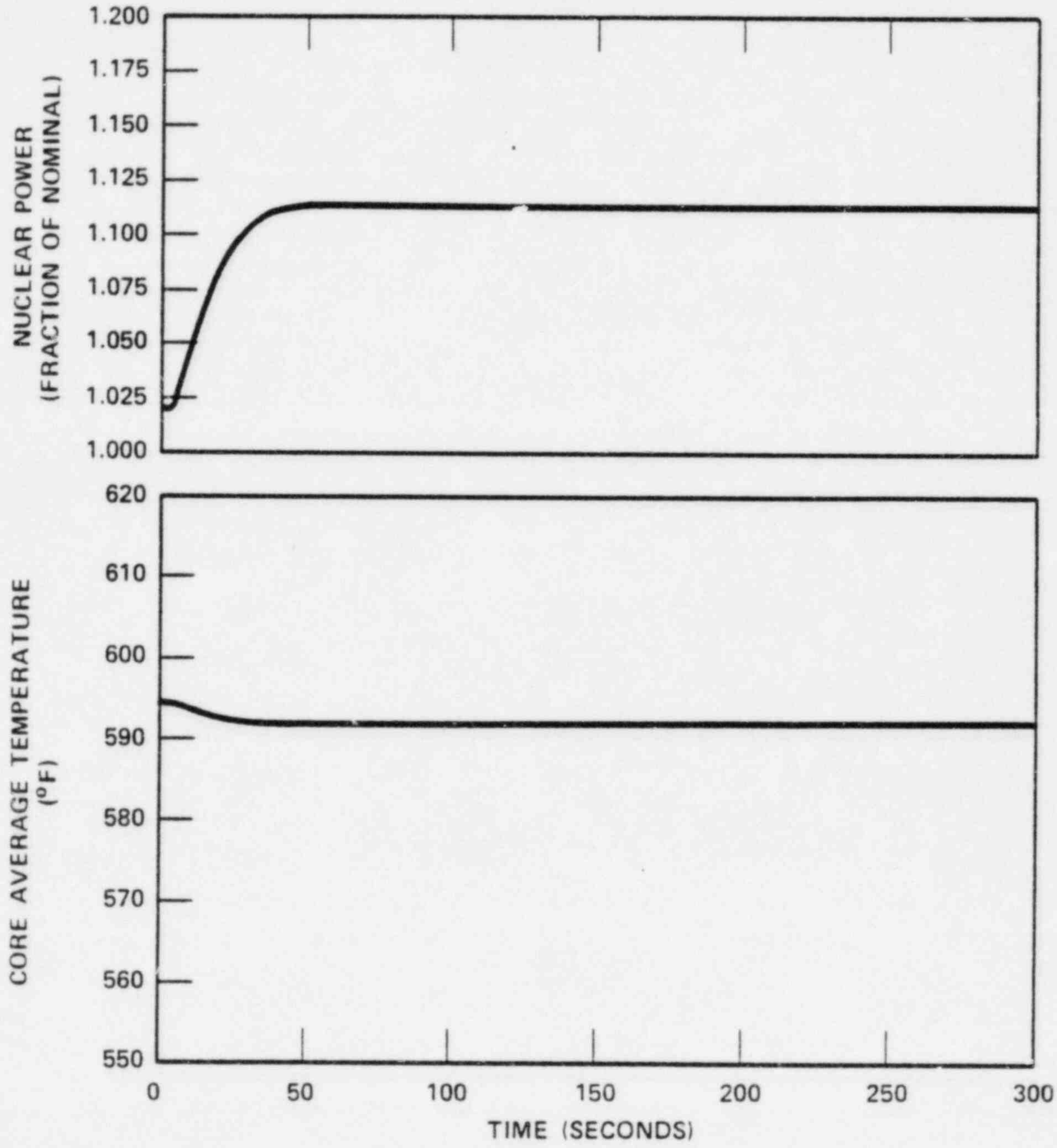


Figure 15.2-35 Excessive Load Increase Without Rod Contro', End of Life

17,537-16

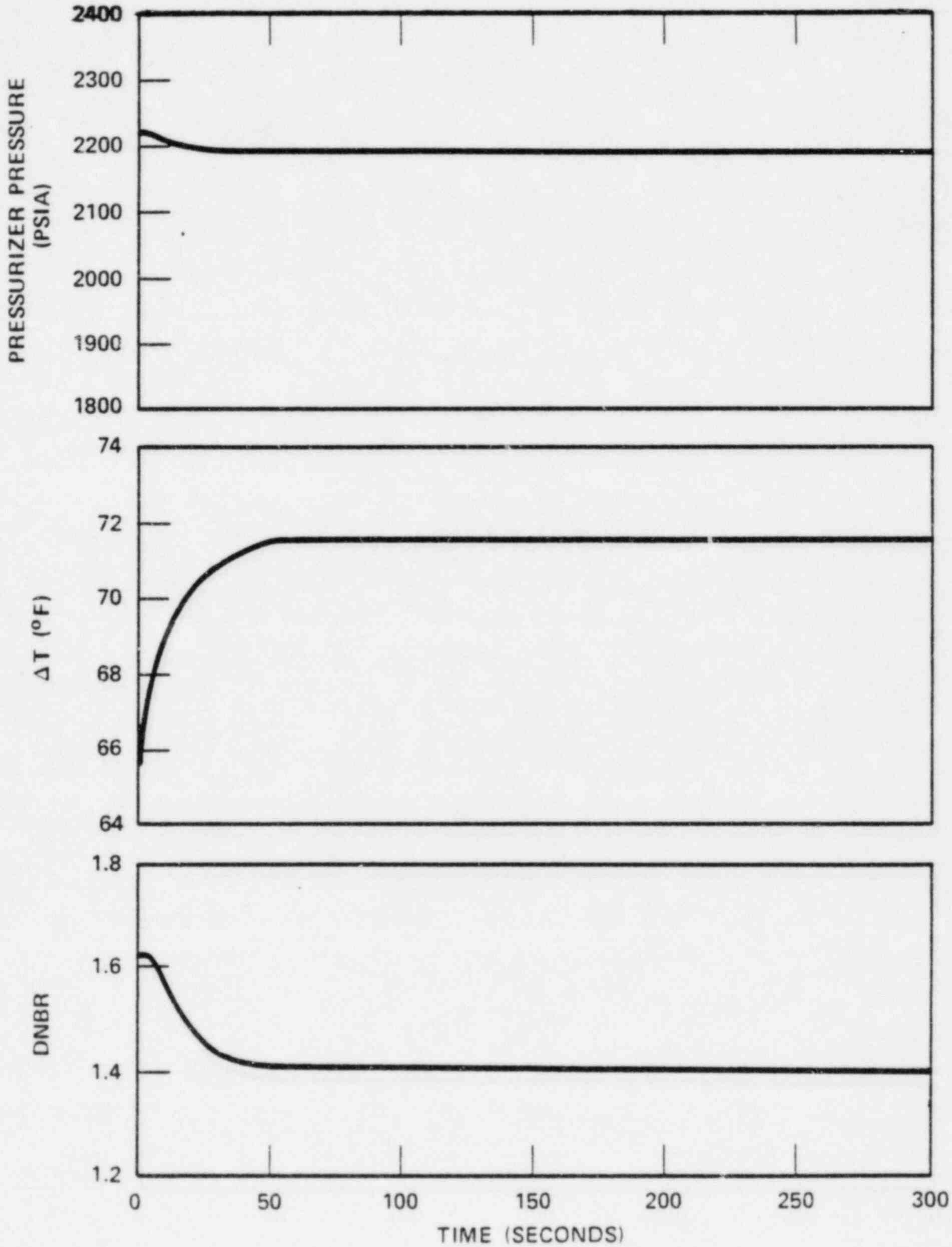


Figure 15.2-36 Excessive Load Increase Without Rod Control, End of Life

17,537-17

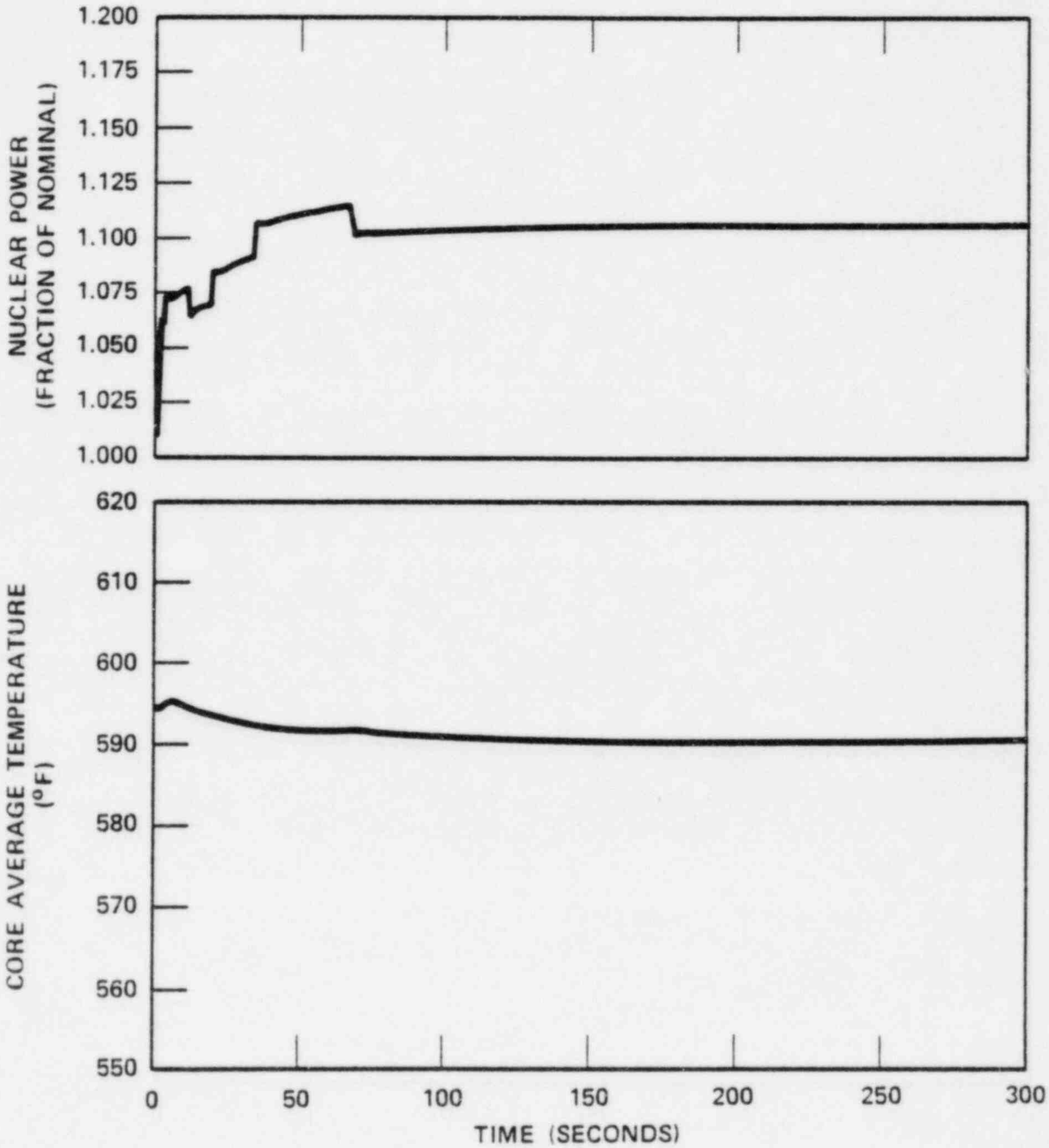


Figure 15.2-37 Excessive Load Increase With Rod Control, Beginning of Life

17,537-18

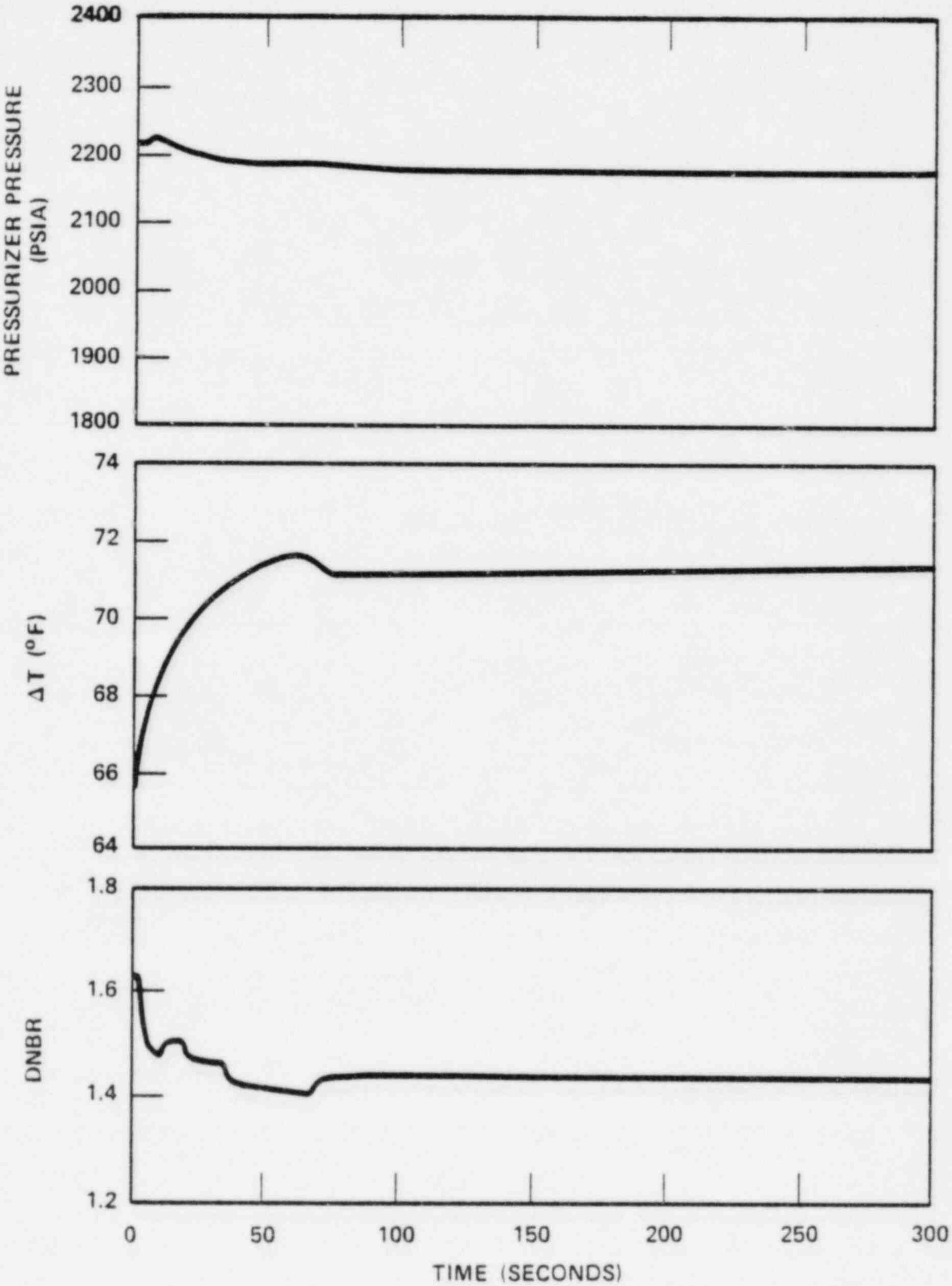


Figure 15.2-38 Excessive Load Increase With Rod Control, Beginning of Life

17,537-19

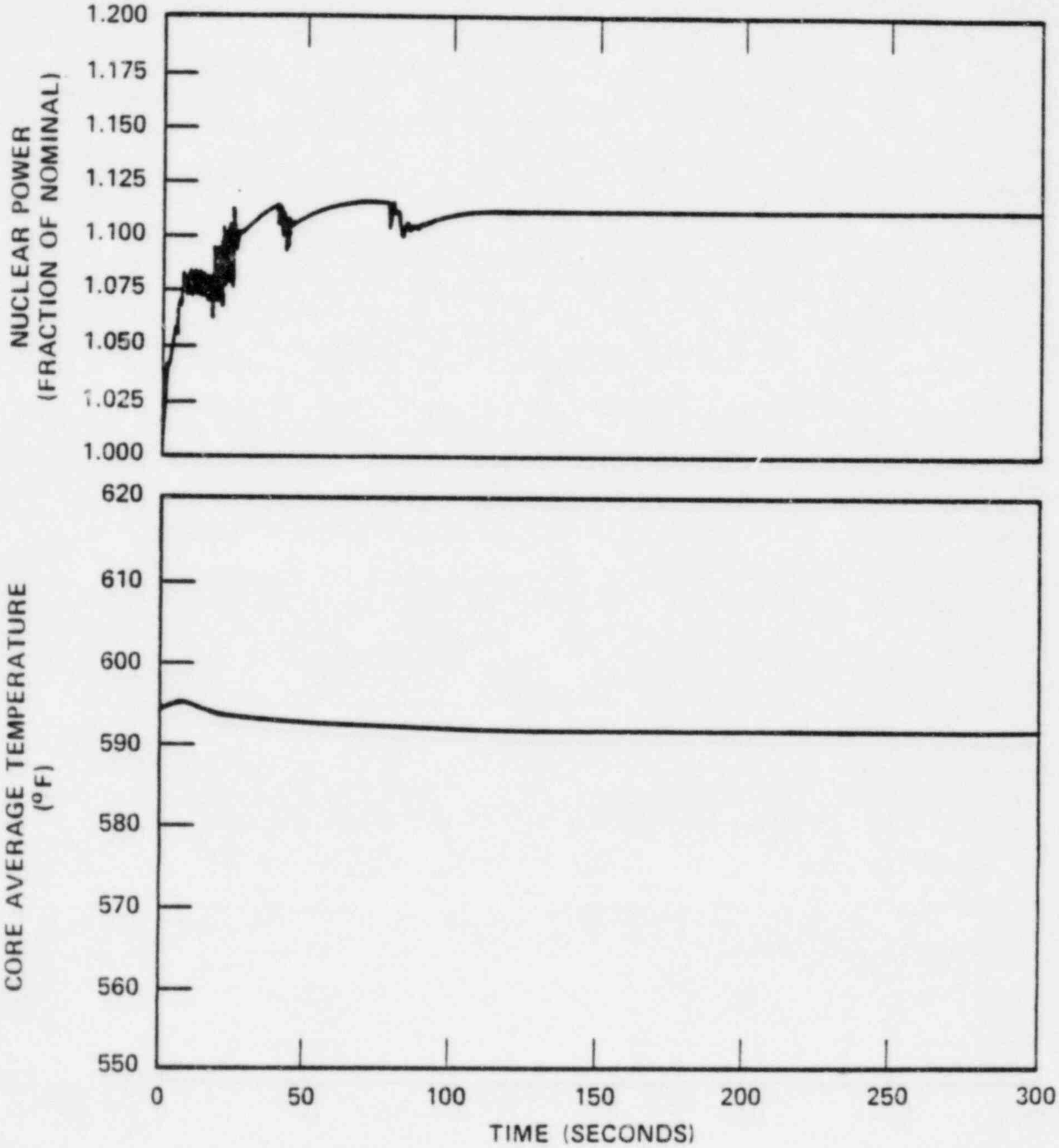


Figure 15.2-39 Excessive Load Increase With Rod Control, End of Life

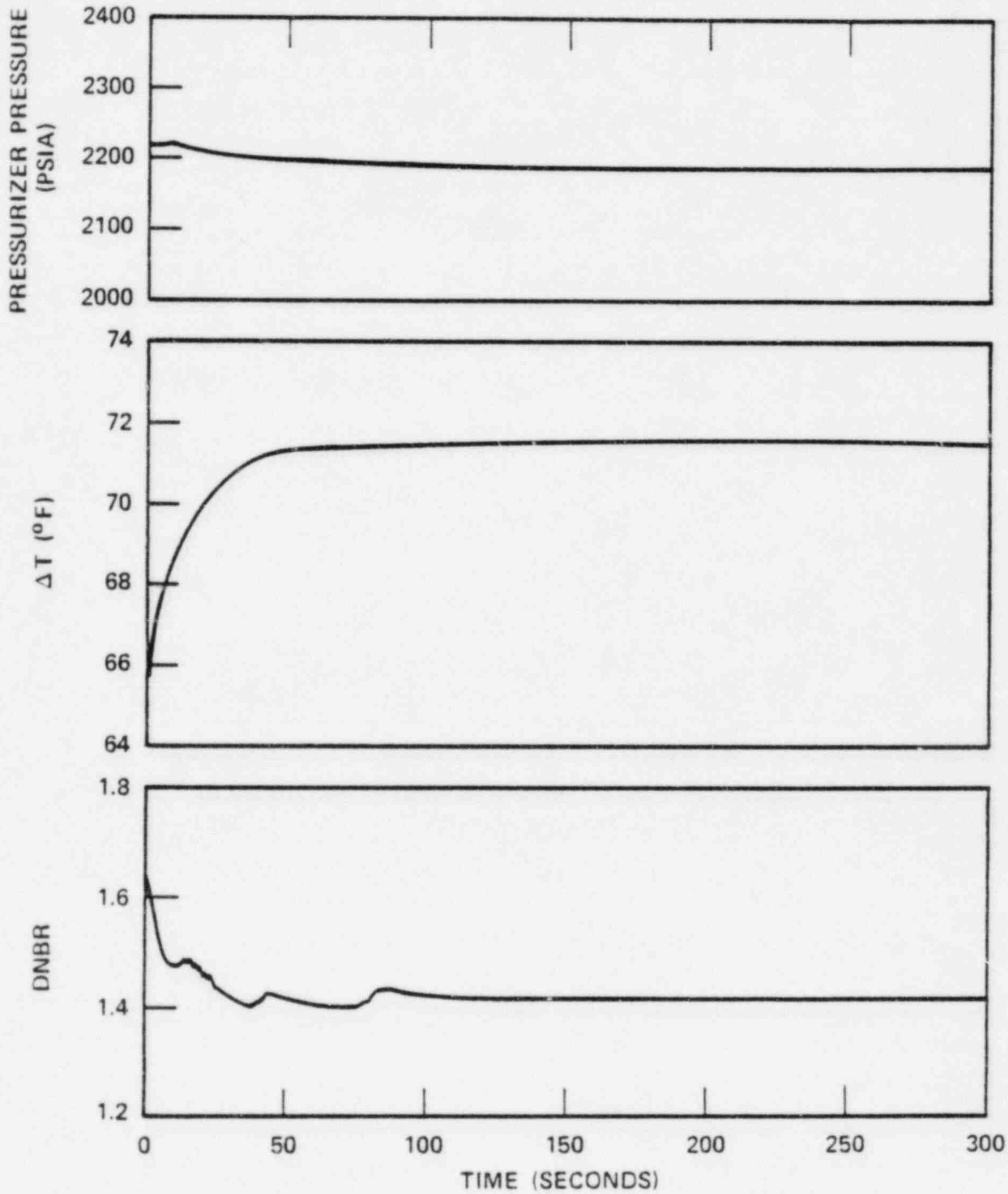


Figure 15.2-40 Excessive Load Increase With Rod Control, End of Life

21323-3

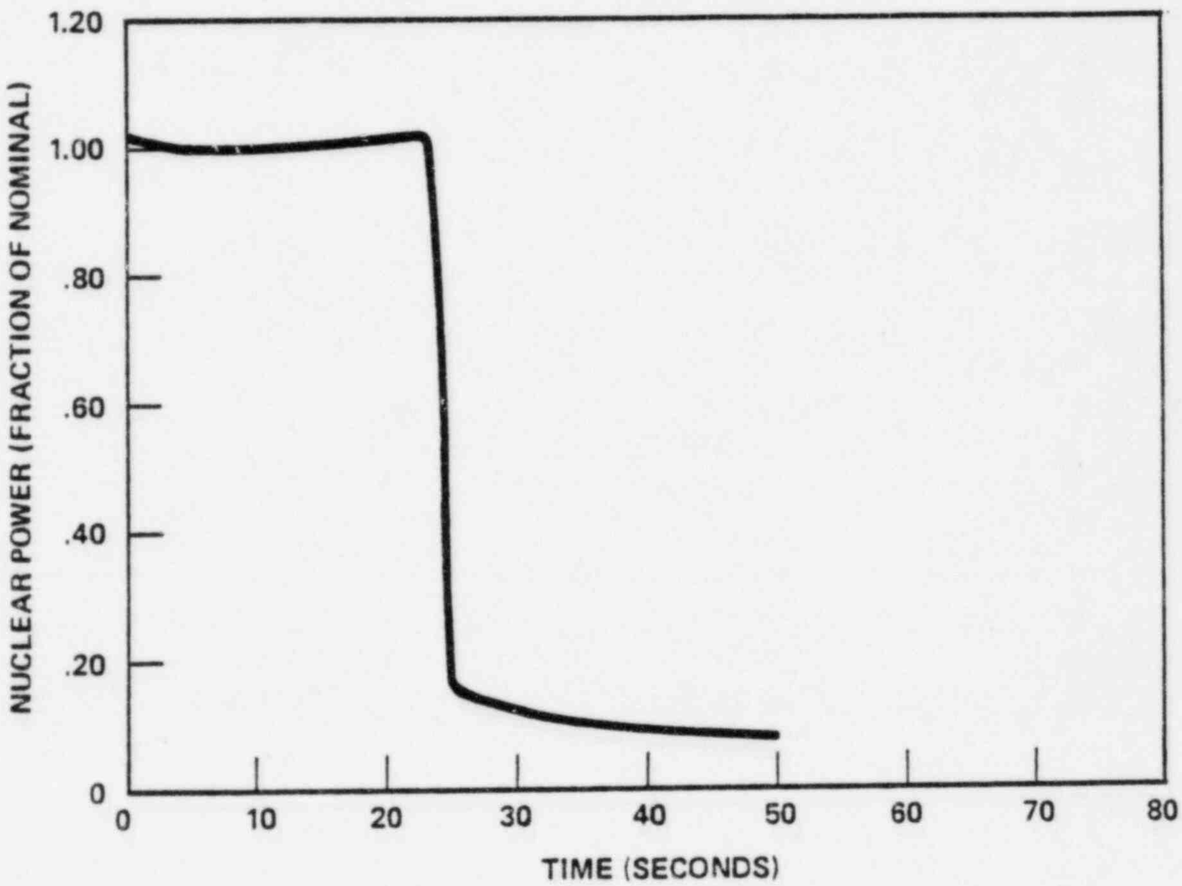


Figure 15.2-41. Flux Transient for Accidental Depressurization

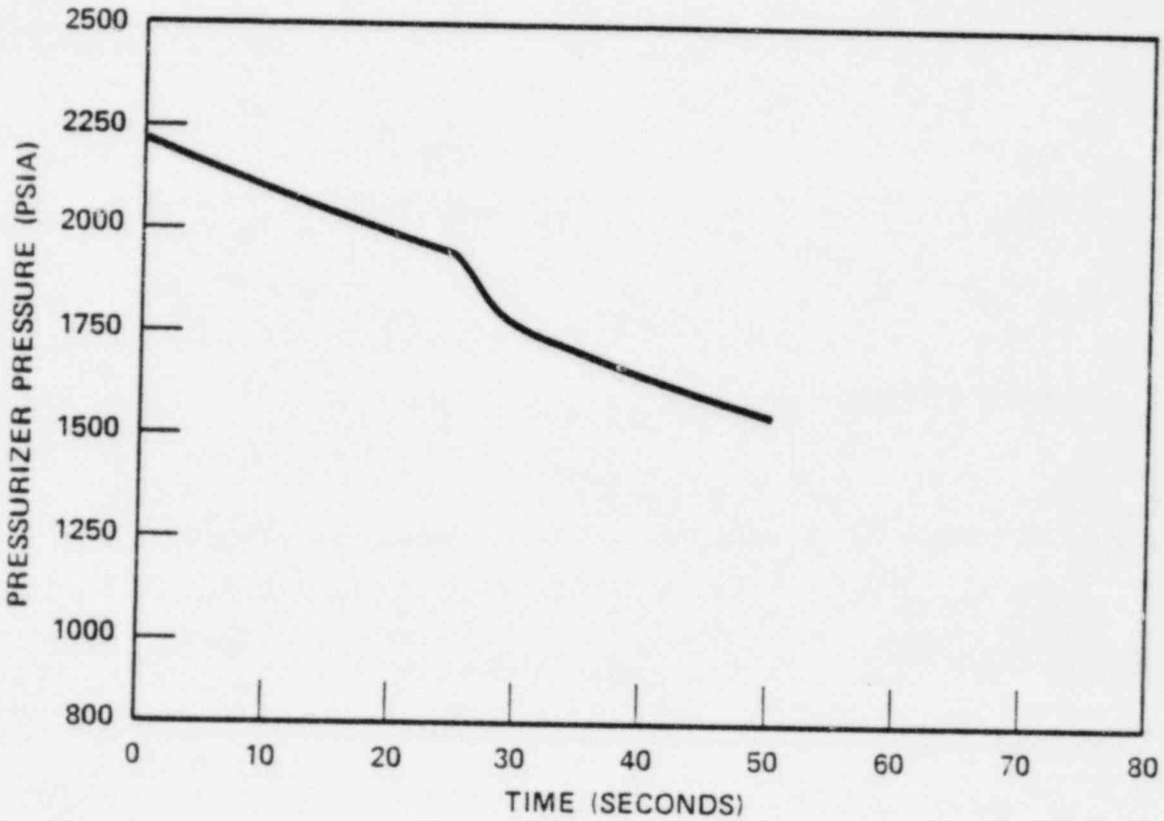


Figure 15.2-42 Pressurizer Pressure Transient for Accidental Depressurization

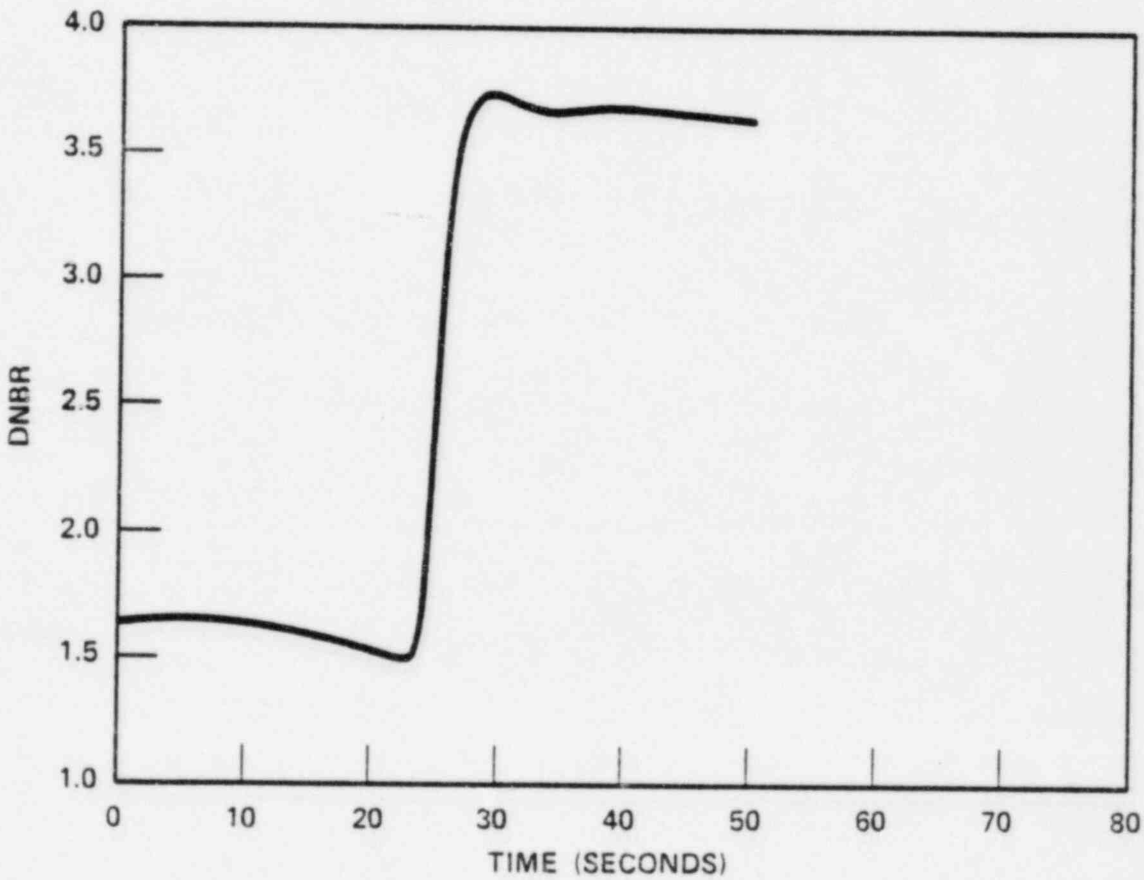


Figure 15.2-43 DNBR Transient for Accidental Depressurization

21323-6

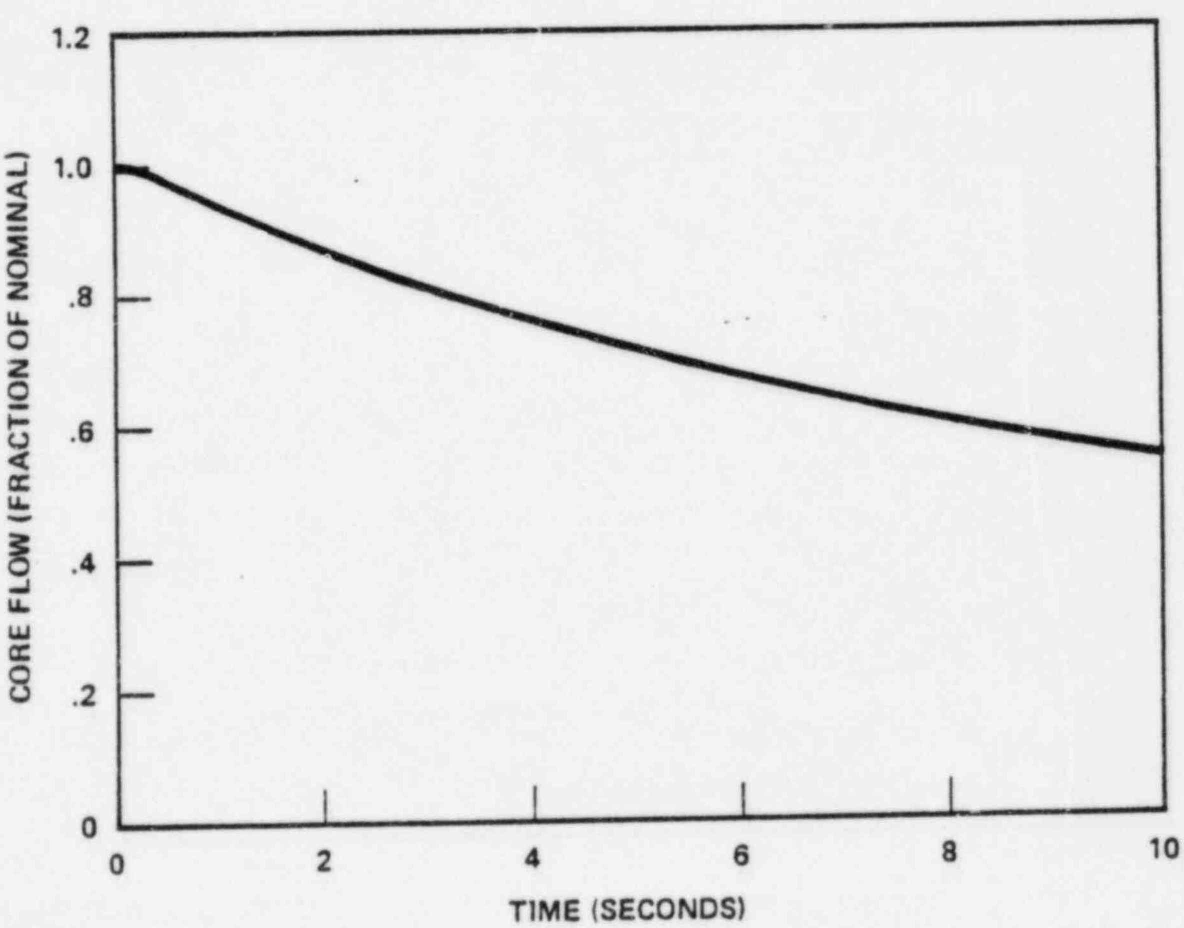


Figure 15.3-23. All Loops Operating, All Loops Coasting Down, Core Flow versus Time

21323-4

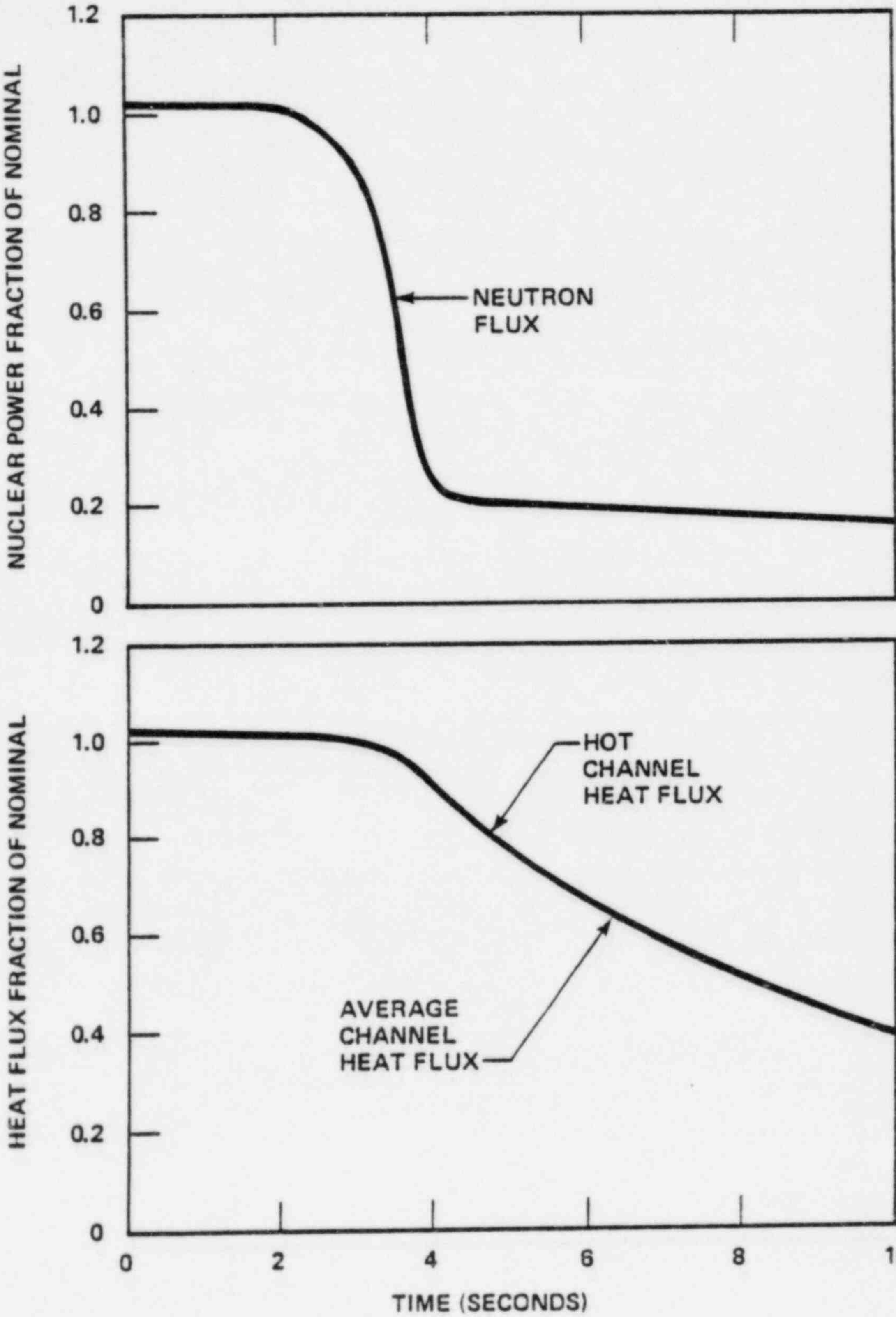


Figure 15.3-24. All Loops Operating, All Loops Coasting Down, Flux Transients versus Time

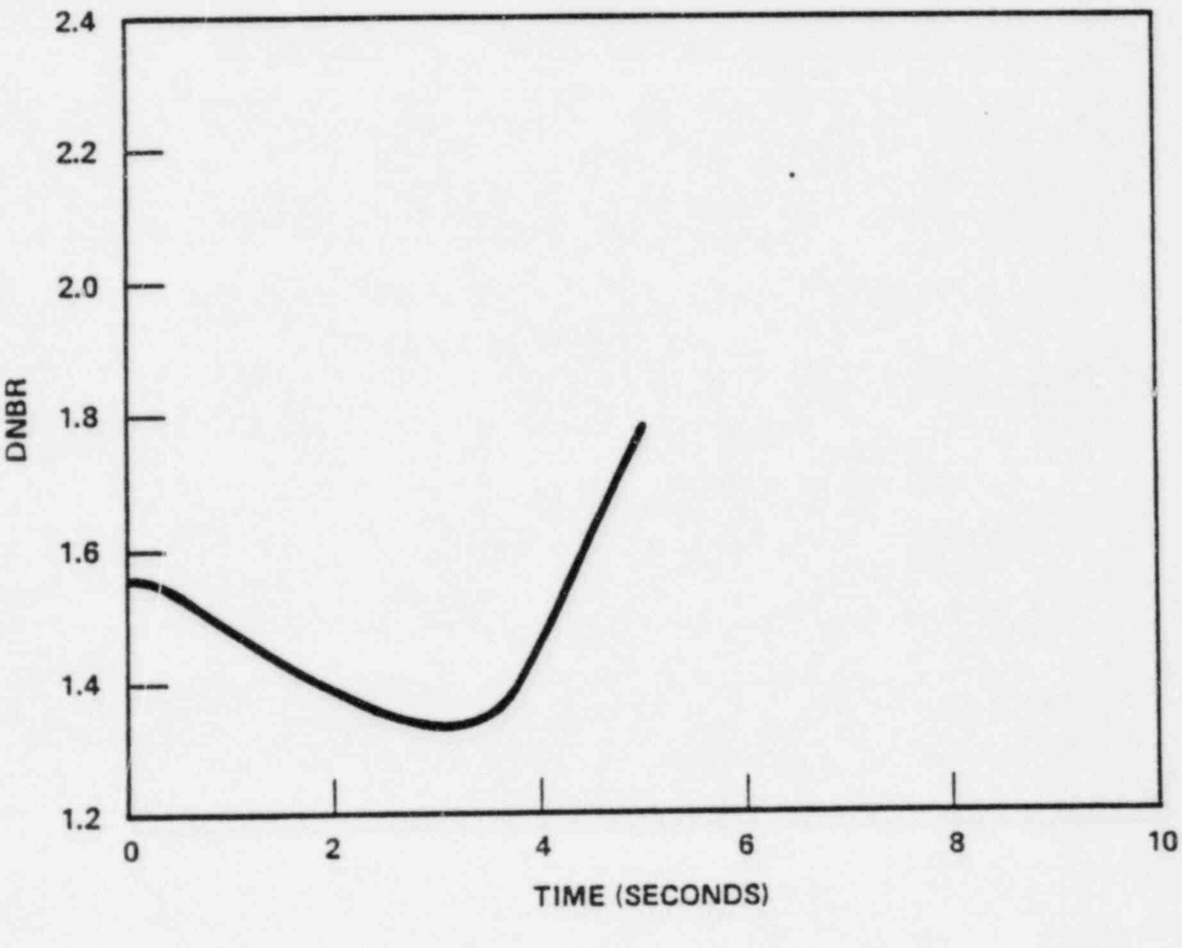


Figure 15.3-25. All Loops Operating, All Loops Coasting Down, DNBR versus Time

17,664-9

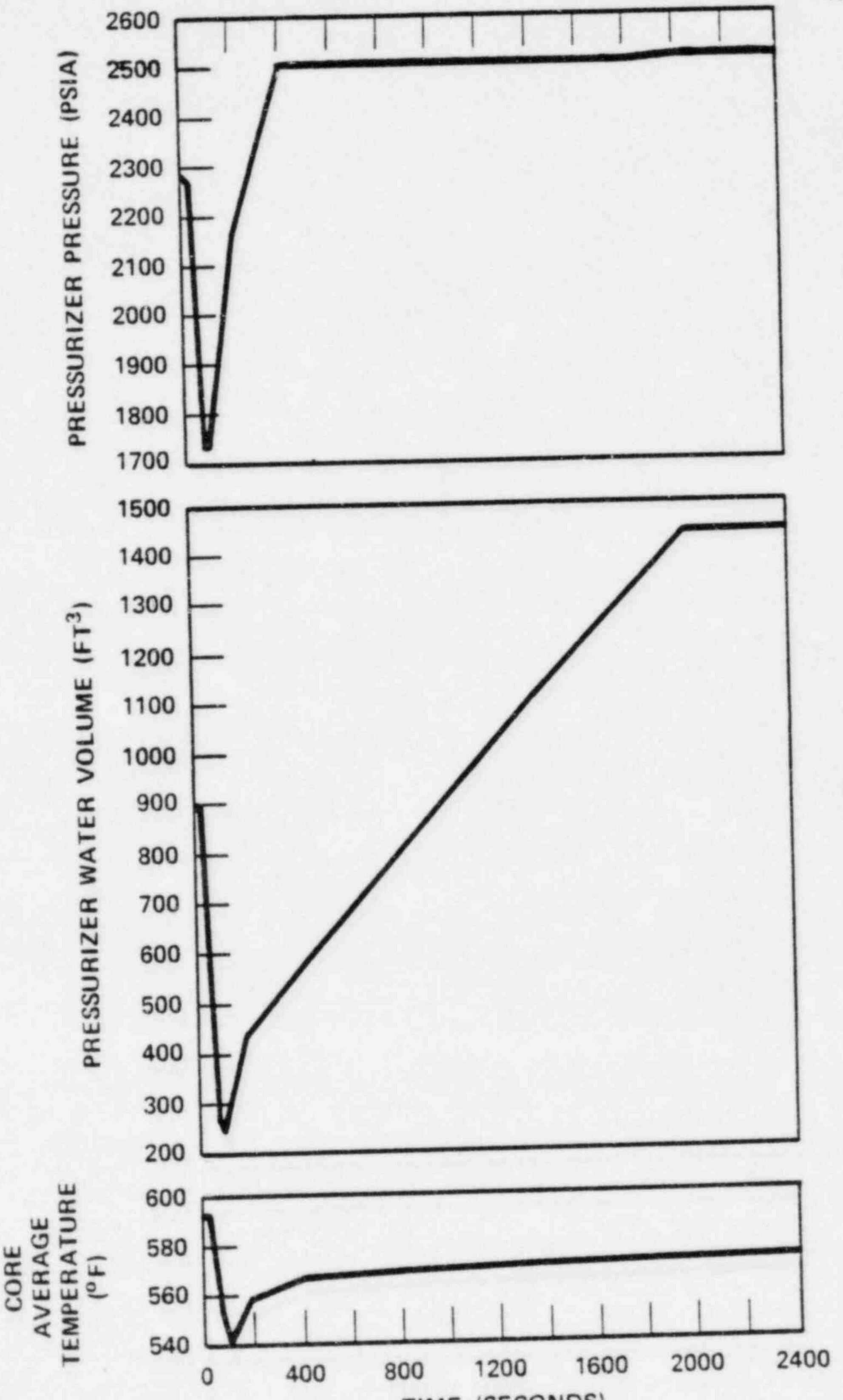


Figure 15.4.2-7a. Main Feedline Rupture Accident – Core Average Temperature Pressurizer Pressure and Pressurizer Water as a Function of Time – With Offsite Power

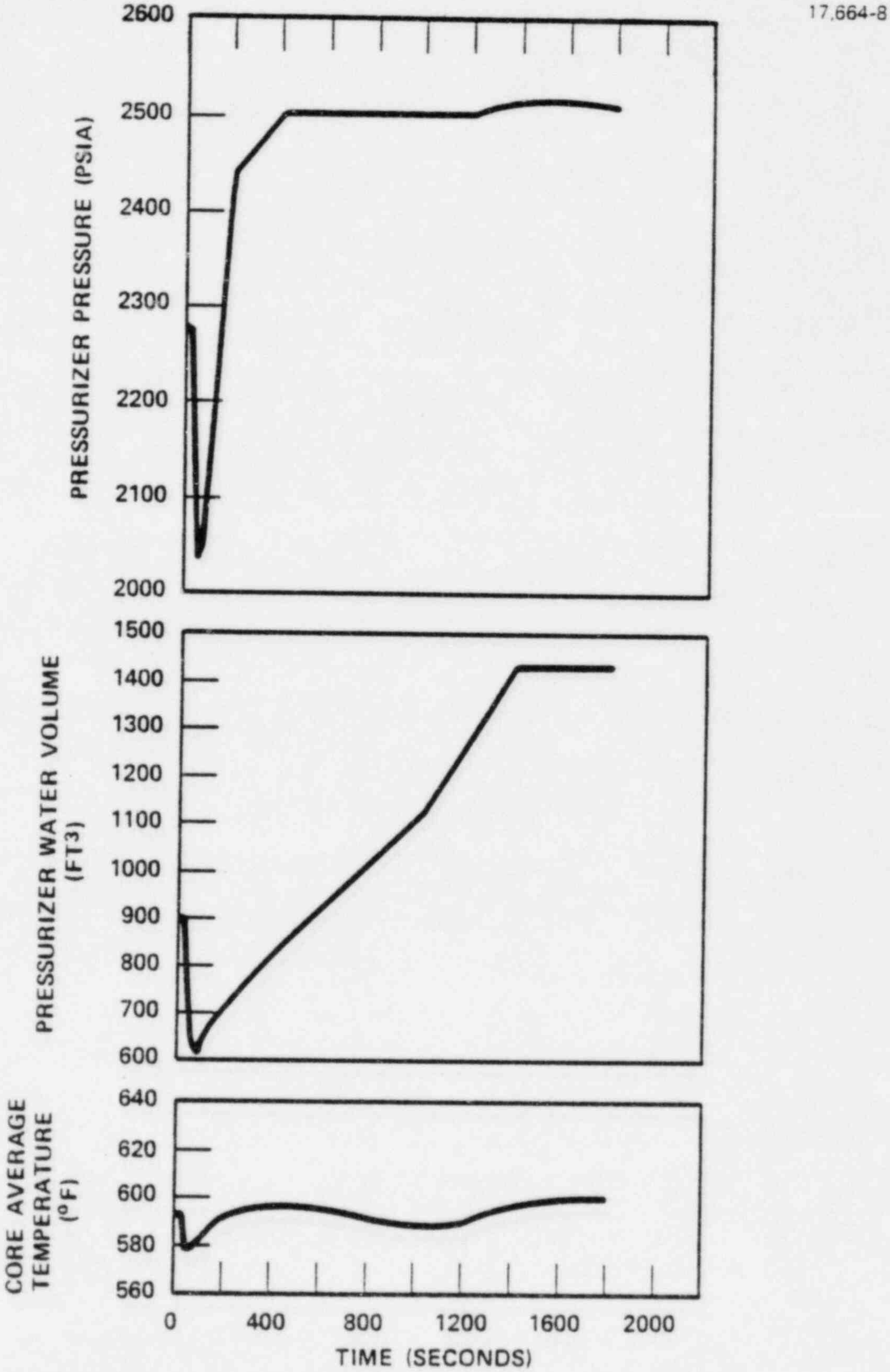


Figure 15.4.2-7b. Main Feedline Rupture Accident – Core Average Temperature, Pressurizer Pressure and Pressurizer Water as a Function of Time – Without Offsite Power

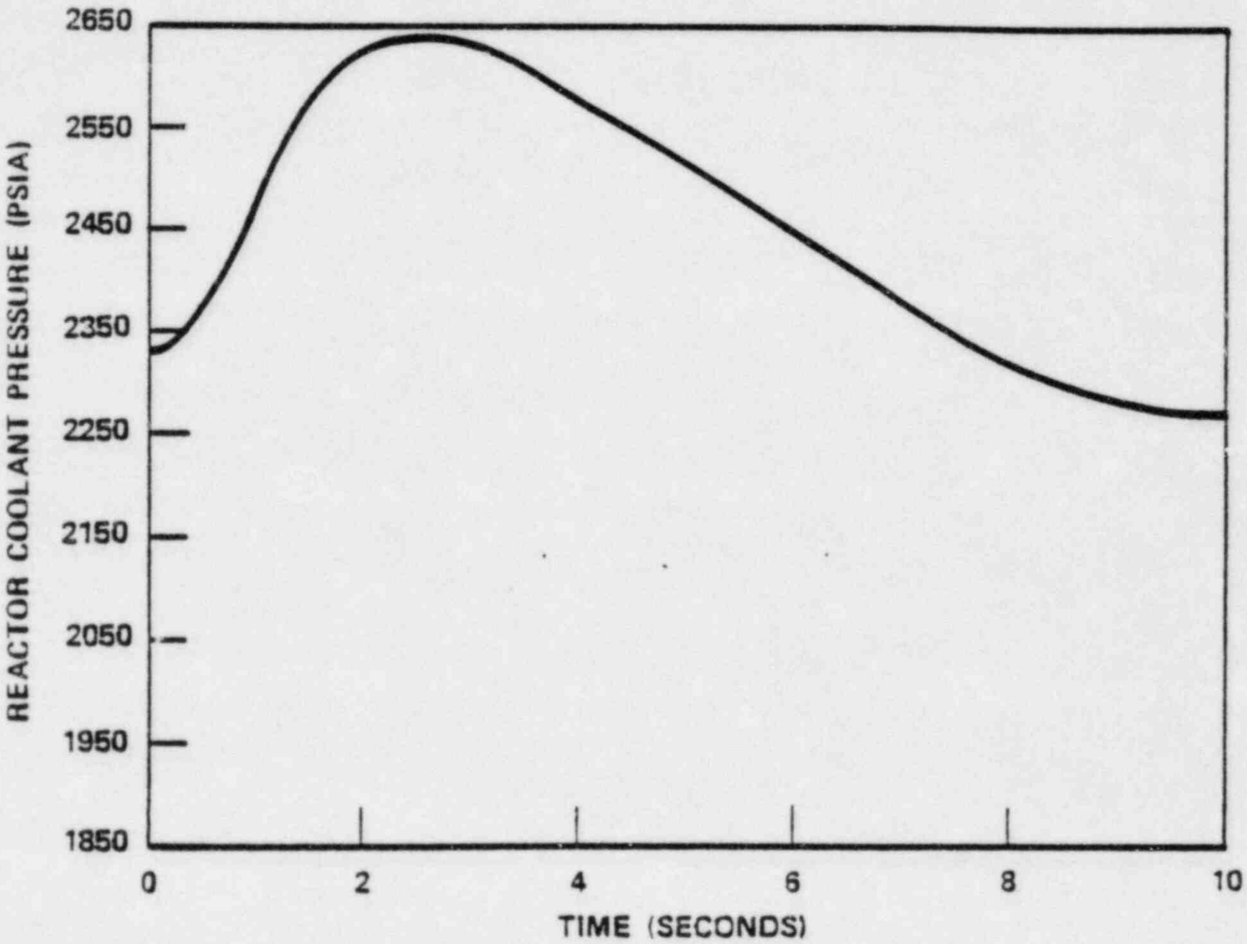


Figure 15.4.4-1. All Loops Operating, One Locked Rotor, Pressure versus Time

17,664-13

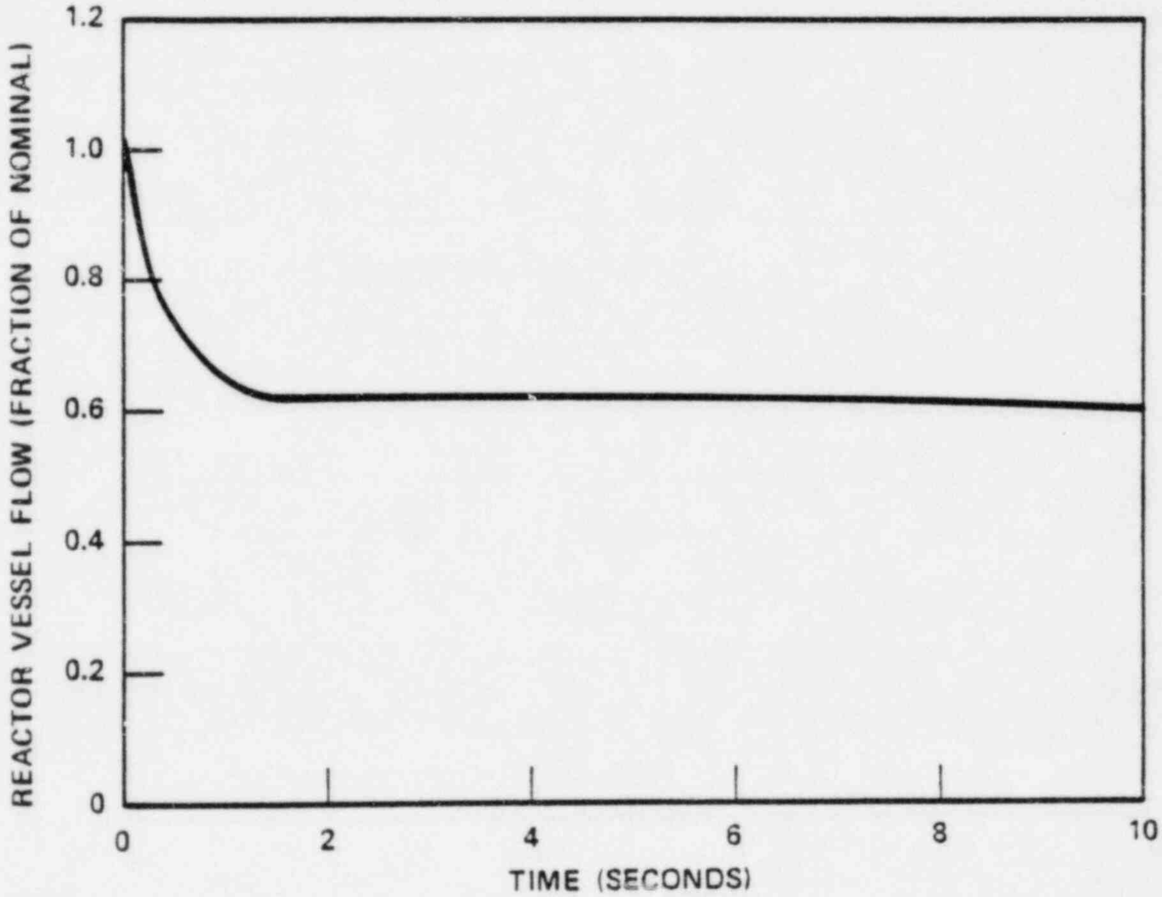


Figure 15.4.4-4. All Loops Operating, One Locked Rotor, Core Flow versus Time

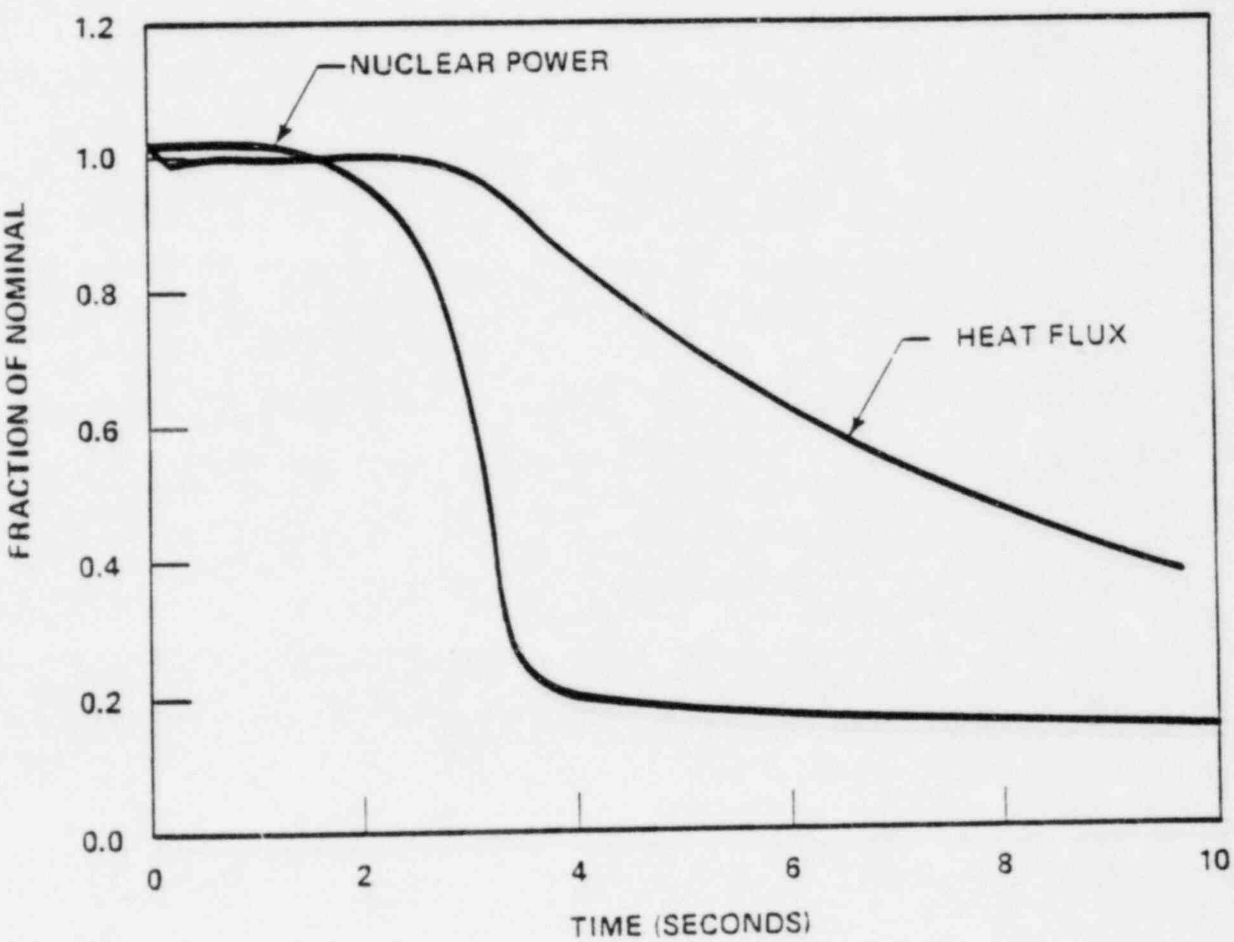


Figure 15.4.4-5 All Loops Operating, One Locked Rotor, Flux Transients Versus Time

17 664-11

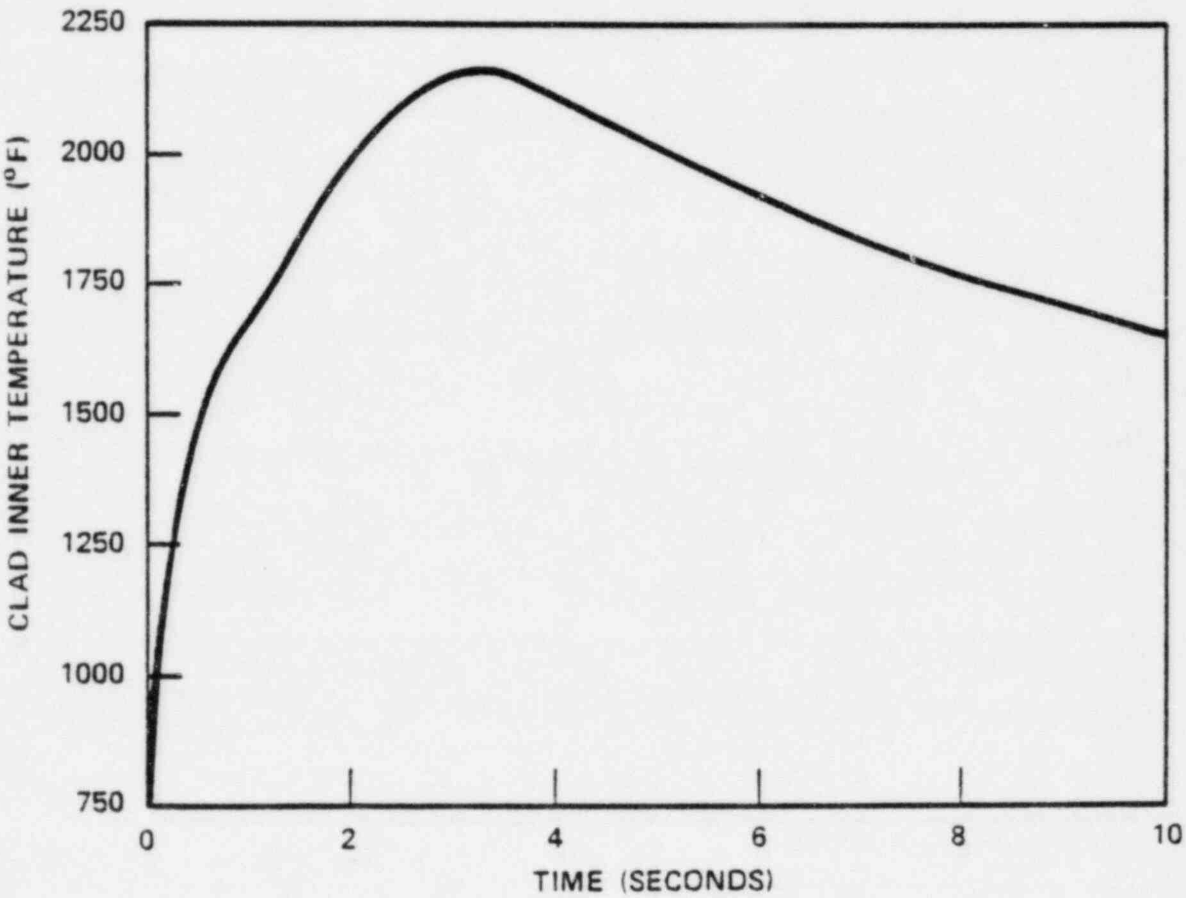


Figure 15.4.4-10. All Loops Operating, One Locked Rotor, Clad Temperature versus Time

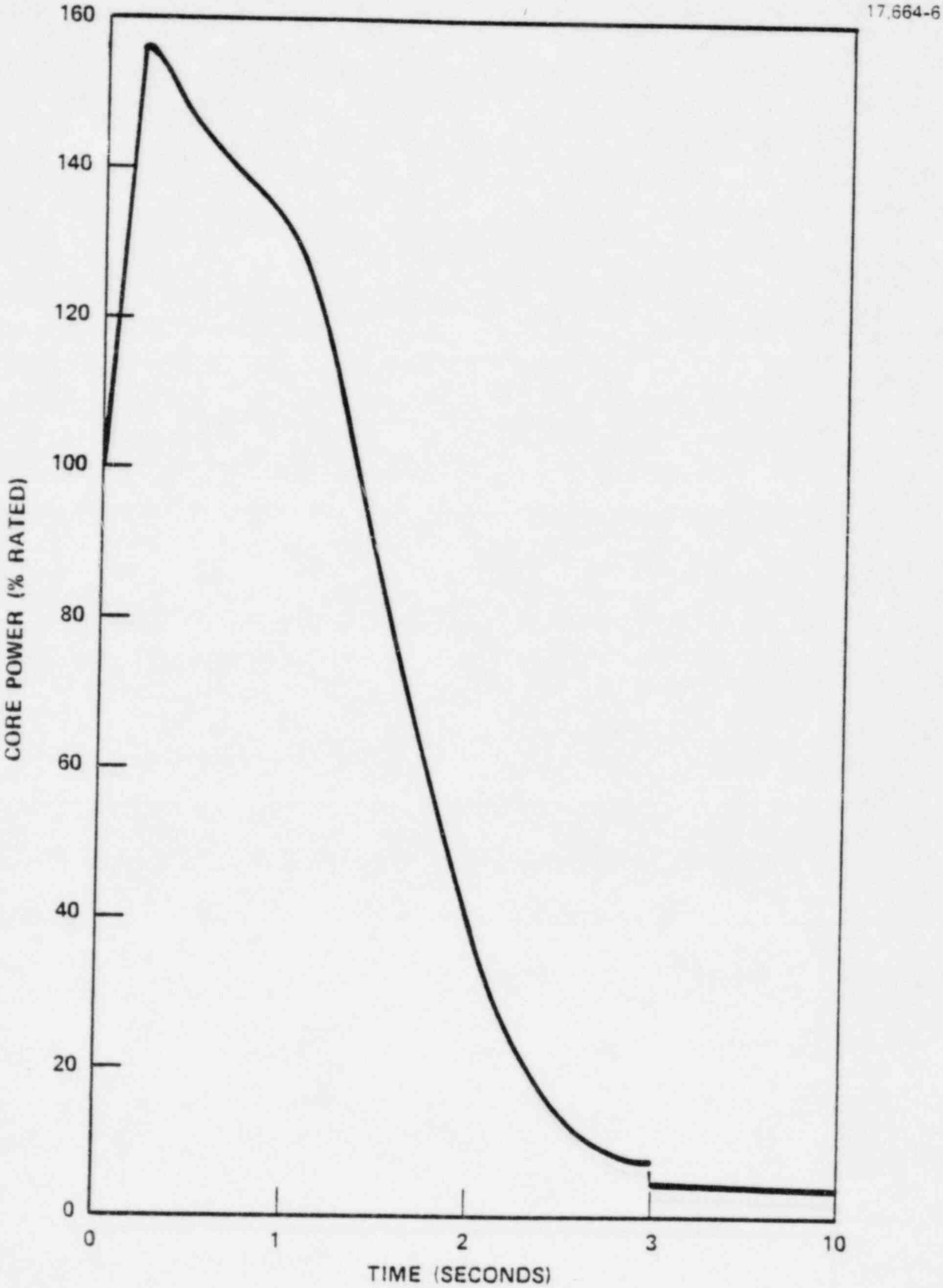


Figure 15.4.6-1. Nuclear Power Transient BOL HFP Rod Ejection Accident

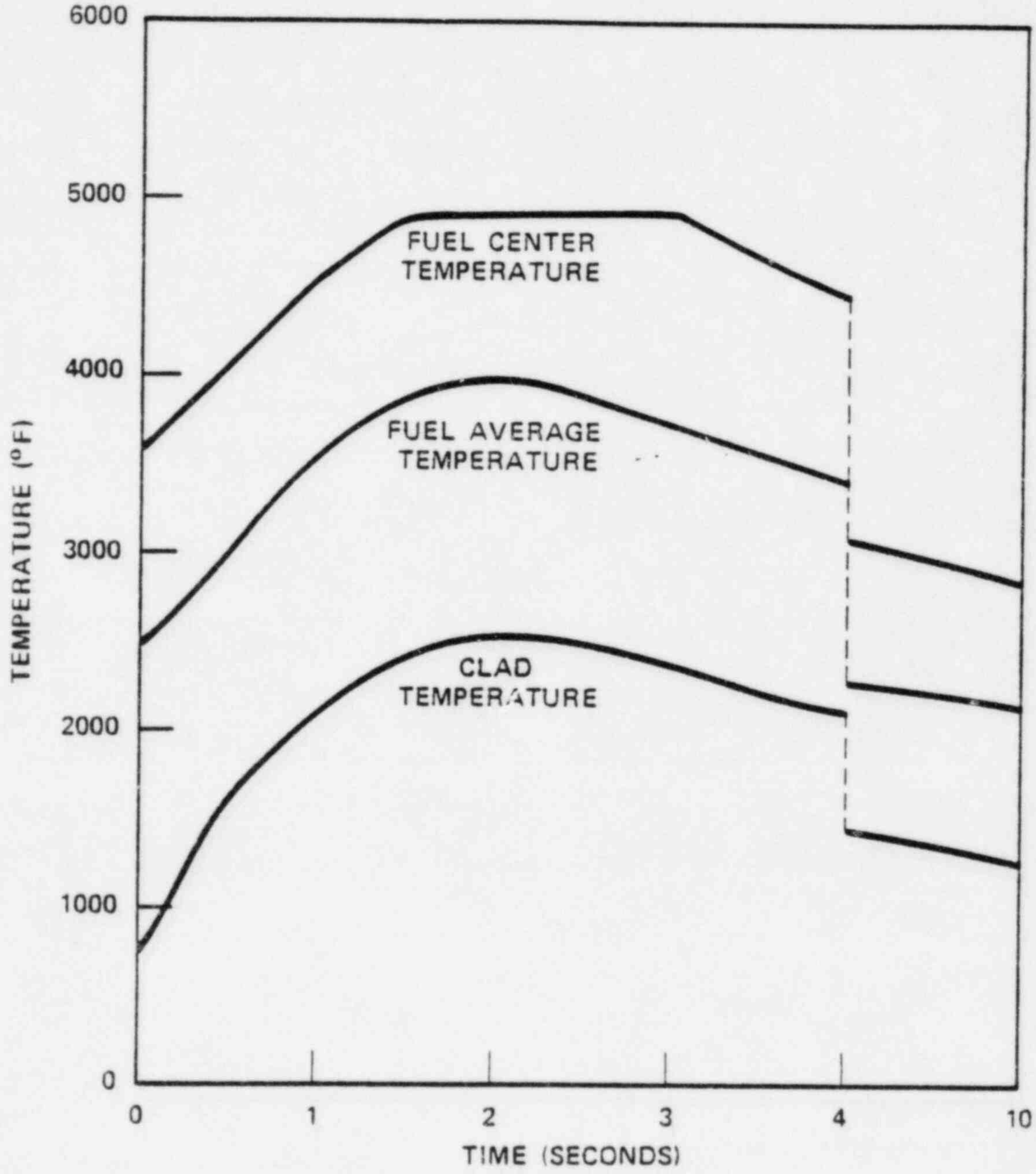


Figure 15.4.6-2. Hot Spot Fuel and Clad Temperature versus Time, BOL HFP Rod Ejection Accident