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Safety Analysis for Operation of Turkey Point Units 3 and 4 With a Positive Moderator Coefficient

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SAFETY ANALYSIS FOR OPERATION OF TURKEY POINT UNITS 3 AND 4 WITH A POSITIVE MODERATOR COEFFICIENT

SECTION I
INTRODUCTION

I. Introduction and Purpose

This safety analysis has been performed to support the proposed Technical Specification change for Turkey Point Units 3 and 4 which would allow a small, positive moderator temperature coefficient to exist at power levels below 70 percent power. The results of the analysis, which are presented in the following section, show that the proposed change can be accommodated with margin to applicable FSAR safety limits.

The present Turkey Point Units 3 and 4 Technical Specifications require the moderator temperature coefficient (MTC) to be zero or negative at all times while the reactor is critical. This requirement is overly restrictive, since a small positive coefficient at reduced power levels could result in a significant increase in fuel cycle flexibilty, but would have only a minor effect on the safety analysis of the accident events presented in the FSAR.

The proposed Technical Specifications change allows a +5 pcm/°F* MTC below 70 percent of rated power, changing to a 0 pcm/°F MTC at 70 percent power and above. This MTC is diagrammed in Figure 1. A power-level dependent MTC was chosen to minimize the effect of the specification on postulated accidents at high power levels. Moreover, as the power level is raised, the average core water temperature becomes higher as allowed by the programmed average temperature for the plant, tending to bring the moderator coefficient more negative. Also, the boron concentration can be reduced as xenon builds into the core. Thus, there is less

* 1 pcm = 10^{-5} $\Delta k/k$

need to allow a positive coefficient as full power is approached. As fuel burnup is achieved, boron is further reduced and the moderator coefficient will become negative over the entire operating power range.

SECTION II

ACCIDENT ANALYSIS

I. Introduction

The impact of a positive moderator temperature coefficient on the accident analyses presented in Chapter 14 of the Turkey Point Units 3 and 4 FSAR has been assessed. Those incidents which were found to be sensitive to minimum positive or near-zero moderator coefficients were reanalyzed. In general, these incidents are limited to transients which cause reactor coolant temperature to increase. With one exception, the analyses presented herein were based on a +5 pcm/°F* moderator temperature coefficient, which was assumed to remain constant for variations in temperature.

The control rod ejection analysis was based on a coefficient which was +5 pcm/°F at zero power nominal average temperature, and which became less positive for higher temperatures. This was necessary since the TWINKLE computer code, on which the analysis is based, is a diffusion-theory code rather than a point-kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature. For all accidents which were reanalyzed, the assumption of a positive moderator temperature coefficient existing at full power is conservative since as shown in Appendix A, the proposed Technical Specification requires that the coefficient be zero or negative at or above 70 percent power.

In general, reanalysis was based on the analysis methods, computer codes, and assumptions employed in the FSAR and subsequent safety analyses; any exceptions are noted in the discussion of each incident. Accidents not reanalyzed included those resulting in excessive heat removal from the reactor coolant system for which a large negative moderator coefficient is conservative, and those for which heatup

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* 1 pcm = 1.0×10^{-5} $\Delta k/k$.

effects following reactor trip have been investigated, and found to be not sensitive to the moderator coefficient. Table I gives a list of accidents presented in the Turkey Point Units 3 and 4 FSARs, and denotes those events reanalyzed for a positive coefficient.

II. Transients Not Affected By a Positive Moderator Coefficient

The following transients were not reanalyzed since they either result in a reduction in reactor coolant system temperature, and are therefore sensitive to a negative moderator temperature coefficient, or are otherwise not affected by a positive moderator temperature coefficient.

A. RCCA Misalignment/Drop

Only the RCCA drop case presented in Section 14.1.3 of the FSAR is potentially affected by a positive moderator temperature coefficient. Use of a positive coefficient in the analysis would result in a larger reduction in core power level following the RCCA drop, thereby increasing the probability of a reactor trip. For the return to power automatic rod control case, a positive coefficient (which would only exist below 70 percent power) would result in a small increase in the power overshoot. Since the limiting conditions for this accident are at or near 100 percent power where the moderator temperature coefficient must be zero or negative, this accident is unaffected by the proposed Technical Specification and thus the analysis was not repeated.

B. Startup of an Inactive Reactor Coolant Loop

An inadvertent startup of an idle reactor coolant pump results in a decrease in core average temperature. As the most negative values of moderator reactivity coefficient produce the greatest reactivity addition, the analysis reported in the FSAR, Section 14.1.6, represents the limiting case.

C. Excessive Heat Removal Due to Feedwater System Malfunctions

The addition of excessive feedwater or the reduction of feedwater temperature are excessive heat removal incidents, and are

consequently most sensitive to a negative moderator temperature coefficient. Results presented in Section 14.1.7 of the FSAR, based on a negative coefficient, represent the limiting case. Therefore, this incident was not reanalyzed.

D. Excessive Load Increase

An excessive load increase event, in which the steam load exceeds the core power, results in a decrease in reactor coolant system temperature. With the reactor in manual control, the analysis presented in Section 14.1.8 of the FSAR shows that the limiting case is with a large negative moderator coefficient. If the reactor is in automatic control, the control rods are withdrawn to increase power and restore the average temperature to the programmed value. The analysis of this case in the FSAR show that the minimum DNBR is not sensitive to moderator temperature coefficient. Therefore, the results presented in the FSAR are still applicable to this incident.

E. Loss of Normal Feedwater, Loss of Offsite Power

The loss of normal feedwater and loss of offsite power accidents (Sections 14.1.11 and 14.1.12 of the FSAR) are analyzed to determine the ability of the secondary system to remove decay heat. These events are not sensitive to a positive moderator coefficient since the reactor trip occurs at the beginning of the transient before the reactor coolant system temperature increases significantly. Therefore, these events were not reanalyzed.

F. Rupture of a Main Steam Pipe

Since the rupture of a main steam pipe is a temperature reduction transient, minimum core shutdown margin is associated with a strong negative moderator temperature coefficient. The worst conditions for a steamline break are therefore those analyzed in the FSAR (Section 14.2.5).

G. Loss of Coolant Accident (LOCA)

The loss of coolant accident (Section 14.3 of the FSAR) is analyzed to determine the core heatup consequences caused by a rupture of the reactor coolant system boundary. The event results in a depressurization of the RCS and a reactor shutdown at the beginning of the transient. This accident was not reanalyzed since the Technical Specification requirement that the temperature coefficient be zero or negative at 70 percent power or above ensures that the previous analysis basis for this event is not affected.

III. Transients Sensitive to a Positive Moderator Coefficient

A. Boron Dilution

As stated in Section 14.1.5 of the FSAR, an uncontrolled boron dilution incident cannot occur during refueling due to administrative controls which isolate the reactor coolant system from the potential source of unborated water. If a boron dilution incident occurs during startup, the FSAR shows that the operator has sufficient time to identify the problem and terminate the dilution before the reactor returns critical. Therefore, the value of the moderator coefficient has no effect on a boron dilution incident during startup. The reactivity addition due to a boron dilution at power will cause an increase in power and reactor coolant system temperature. Due to the temperature increase, a positive moderator coefficient would add additional reactivity and increase the severity of the transient. With the reactor in automatic control, however, the rod insertion alarms provide the operator with adequate time to terminate the dilution before shutdown margin is lost. A boron dilution incident with the reactor in manual control is no more severe than a rod withdrawal at power, which is analyzed in Section III.C, and therefore this case was not specifically analyzed. Following reactor trip, the amount of time available

before shutdown margin is lost is not affected by the moderator coefficient.

B. Control Rod Withdrawal From a Subcritical Condition

Introduction

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 14.1.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient. The power excursion causes a heatup of the moderator and fuel. The reactivity addition due to a positive moderator coefficient could result in increases in peak heat flux, peak fuel, and clad temperatures. The time the core is critical before a reactor trip is very short so that the coolant temperature does not increase significantly. Hence, the effect of a positive moderator coefficient is small.

Method of Analysis

The analysis was performed in the FSAR for a reactivity insertion rate of 60×10^{-5} Δk/sec. This reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). A constant moderator temperature coefficient of +5 pcm/°F was used in the analysis. The digital computer codes, initial power level, and reactor trip instrument delays and setpoint errors used in the analysis were the same as used in WCAP-8284 Rev. 2, "Florida Power and Light - Turkey Point Units 3 and 4 - Precautions, Limitations, and Set Points".

Results and Conclusions

The nuclear power, coolant temperature, heat flux, fuel average temperature, and clad temperature versus time for a 60 x 10-5 Δk/sec insertion rate are shown in Figures 2 through 4. This insertion rate, coupled with a positive moderator temperature coefficient of +5 pcm/°F, yields a peak heat flux which does not exceed the nominal value. Therefore the conclusions presented in the FSAR are still applicable.

C. Uncontrolled Control Rod Assembly Withdrawal at Power

Introduction

An uncontrolled control rod assembly withdrawal at power produces a mismatch in steam flow and core power, resulting in an increase in reactor coolant temperature. A positive moderator coefficient would augment the power mismatch and could reduce the margin to DNB. A discussion of this incident is presented in Section 14.1.2 of the FSAR.

Method of Analysis

The transient was reanalyzed employing the same digital computer code and assumptions regarding instrumentation and setpoint errors used for the FSAR and subsequent safety analyses. This transient was analyzed at 80 percent power with a positive moderator coefficient since this case is the most limiting of those presented in the FSAR. A constant moderator coefficient of +5 pcm/°F was used in the analysis. The assumption that a positive moderator coefficient exists at full power is conservative since at full power, the moderator coefficient will actually be zero or negative.

Results

Figure 5 shows the minimum DNBR as a function of reactivity insertion rate. The limiting case for DNB margin is a reactivity insertion rate of 7.2×10^{-5} Δk/sec from full power initial conditions which results in a minimum DNBR of 1.44. These results provide larger margin to DNB than the FSAR results primarily due to a lower nuclear peaking factor (1.55) than the FSAR value (1.75). The peaking factor was reevaluated in WCAP-8074 to account for fuel pellet axial densification. A positive moderator coefficient therefore does not lower the DNBR associated with a control rod assembly withdrawal at power below the limit value of 1.30.

Conclusions

These results demonstrate that the conclusions presented in the FSAR are still valid. That is, the core and reactor coolant system are not adversely affected since nuclear flux and overtemperature ΔT trips prevent the core minimum DNB ratio from falling below 1.30 for this incident.

D. Loss of Reactor Coolant Flow

Introduction

As demonstrated in the FSAR, Section 14.1.9, the most severe loss of flow transient is caused by the simultaneous loss of electrical power to all three reactor coolant pumps. This transient was reanalyzed to determine the effect of a positive moderator temperature coefficient on the nuclear power transient and the resultant effect on the minimum DNBR reached during the incident. The effect on the nuclear power transient would be limited to the initial stages of the incident during which reactor coolant temperature increases; this increase is terminated shortly after reactor trip.

Method of Analysis

Analysis methods and assumptions used in the reevaluation were consistent with those employed in the FSAR and subsequent safety analyses.

The current digital computer codes and assumptions used to calculate the flow coastdown and resulting system transient were in accordance with those used to perform the FSAR and subsequent safety analyses. The analysis was done with a constant moderator coefficient of +5 pcm/°F.

Results

For the analysis performed with a +5 pcm/°F moderator coefficient, the reactor coolant average temperature increases less than 6°F above the initial value. Therefore, a positive moderator coefficient does not appreciably affect the reactor coolant system response or the minimum DNBR reached during the transient. For this case, a minimum DNBR of 1.64 was obtained. Figure 6 through 8 show the flow coastdown, the nuclear power and heat flux transients, and the minimum DNBR versus time.

Conclusions

A positive moderator temperature coefficient does not appreciably affect the result of the complete loss of flow transient, and the minimum DNBR remains above the limit value of 1.30 for this incident. This case was analyzed since it is the most limiting one presented in the FSAR. Since the transient causes only a small change in core average moderator temperature, and the positive moderator coefficient does not appreciably affect the nuclear power transient, the partial loss of flow cases will also not be appreciably affected.

E. Locked Rotor

Introduction

The FSAR (Section 14.1.9) shows that the most severe locked rotor incident is an instantaneous seizure of a reactor coolant pump rotor at 100 percent power with three loops operating. Following the incident, reactor coolant system temperature rises until shortly after reactor trip. A positive moderator coefficient will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the incident. The transient was reanalyzed, however; due to the potential effect on the nuclear power transient and thus on the peak reactor coolant system pressure and fuel temperatures.

Method of Analysis

The digital computer codes and assumptions used in the reanalysis to evaluate the pressure transient and thermal transient were in accordance with those used in the FSAR and subsequent safety analyses. An analysis was done at 70 percent power with a moderator/coefficient of +5 pcm/°F to show that this case is not more limiting than the 100 percent power, 0 pcm/°F case presented in the FSAR. This case is sufficient to illustrate the impact on the transient by a positive moderator coefficient, since the moderator coefficient will actually be zero or negative at full power.

Results and Conclusions

Table II compares results obtained for this case with those presented in the FSAR. As shown in the table, the FSAR analysis at full power with a zero moderator coefficient results in higher reactor coolant pressure than the 70 percent power case with a positive moderator coefficient with the peak clad temperature well below the 2700°F limit for +5 pcm/°F moderator coefficient study. Therefore, the conclusions presented in the FSAR are still applicable.

F. Loss of External Electrical Load

Introduction

Two cases, analyzed for both beginning and end of life conditions, are presented in Section 14.1.10 of the FSAR:

1. Reactor in automatic rod control with operation of the pressurizer spray and the pressurizer power operated relief valves; and

2. Reactor in manual rod control with no credit for pressurizer spray or power operated relief valves.

As the moderator temperature coefficient will be negative at end of life, only beginning of life cases were repeated. The result of a loss of load is a core power level which momentarily exceeds the secondary system power extraction causing an increase in core water temperature. The consequences of the reactivity addition due to a positive moderator coefficient are increases in both peak nuclear power and pressurizer pressure.

Method of Analysis

A constant moderator temperature coefficient of +5 pcm/°F was assumed. The method of analysis and assumptions used were otherwise in accordance with those presented in the FSAR and subsequent safety analyses.

Results

The system transient response to a total loss of load from 102 percent power, with control rods in automatic control, assuming pressurizer relief and spray valves, is shown in Figures 9 and 10. Peak pressurizer pressure reaches 2443 psia following a reactor trip on overtemperature ΔT. This compares to a value of 2395 psia presented in the FSAR. A minimum DNBR of 1.74 is reached shortly after reactor trip.

Figures 11 and 12 illustrate reactor coolant system response to a loss of load with rods in manual control, assuming no credit for pressure control. Peak pressurizer pressure reaches 2534 psia following reactor trip on high pressurizer pressure. The peak pressure reached in the FSAR analysis for this case was 2517 psia. The minimum DNBR is initially 1.86 and increases throughout the transient.

Conclusions

The analysis demonstrates that the integrity of the core and the reactor coolant system pressure boundary during a loss of load transient will not be affected by a positive moderator reactivity coefficient since the minimum DNB ratio remains well above the 1.30 limit, and the peak reactor coolant pressure is less than 110 percent of design. Therefore, the conclusions presented in the FSAR are still applicable.

G. Rupture of a Control Rod Drive Mechanism Housing Control Rod Ejection

Introduction

The rod ejection transient is analyzed at full power and hot standby for both beginning and end of life conditions. Since the moderator temperature coefficient is negative at end of life, only the beginning of life cases were reanalyzed. The high nuclear power levels and hot spot fuel temperatures resulting from a rod ejection are increased by a positive moderator coefficient. A discussion of this transient is presented in Section 14.2.6 of the FSAR.

Method of Analysis

The digital computer codes, ejected rod worths, and transient peaking factors for analyses of the nuclear power transient and

hot spot heat transfer are the same as those used in Turkey Point Unit 4 Cycle 4 RSE. The moderator coefficient used for this transient was +5 pcm/°F at zero power nominal average temperature, decreasing to approximately +4 pcm/°F at full power T-average. This is still a conservative assumption since the moderator coefficient actually is zero or negative above 70 percent power.

Results and Conclusions

Peak fuel and clad temperatures and nuclear power versus time for both full power and hot standby are presented in Figures 13 through 16 and are consistent with those in Unit 4 Cycle 4 RSE. The limiting peak hot spot clad temperature, 2210°F, was reached in the hot full power case. Maximum fuel temperatures were also associated with the full power case. Although the peak hot spot fuel centerline temperature for this transient exceeded the melting point, melting was restricted to less than the innermost 10 percent of the pellet.

As fuel and clad temperature do not exceed the fuel and clad limits specified in the FSAR, there is no danger of sudden fuel dispersal into the coolant, or consequential damage to the primary coolant loop. The results are summarized in Table III.

SECTION III
CONCLUSIONS

To assess the effect on accident analysis of operation of Turkey Point Units 3 and 4 with a slightly positive moderator temperature coefficient, a safety analysis of transients sensitive to a zero or positive moderator coefficient was performed. These transients included control rod assembly withdrawal from subcritical, control rod assembly withdrawal at power, loss of reactor coolant flow, loss of external load, and control rod ejection. This study indicated that a small positive moderator coefficient does not result in the violation of safety limits for the transients analyzed.

Except as noted, the analyses employed a constant moderator coefficient of +5 pcm/°F, independent of power level. The results of this study are conservative for the accidents investigated at full power, since the proposed Technical Specification shown in Appendix A required that the coefficient be zero or negative at or above 70 percent power.

TABLE I

ACCIDENTS EVALUATED FOR POSITIVE MODERATOR COEFFICIENT EFFECTS

	FSAR	Accident	Time in Life
*	14.1.1	RCCA Withdrawal from Subcritical	BOC
*	14.1.2	RCCA Withdrawal from Power	BOC/EOC
	14.1.3/4	RCCA Misalignment/Drop	BOC
*	14.1.5	Boron Dilution	BOC
	14.1.6	Startup of an Inactive Loop	EOC
	14.1.7	Reduction in Feedwater Enthalpy	EOC
	14.1.8	Excessive Load Increase	BOC/EOC
*	14.1.9	Loss of Flow/Locked Rotor	BOC
*	14.1.10	Loss of Load/Turbine Trip	BOC/EOC
	14.1.11	Loss of Feedwater	-
	14.1.12	Station Blackout	-
	14.2.5	Steam Line Break	EOC
*	14.2.6	RCCA Ejection	BOC/EOC
	14.3.2	LOCA	BOC

* Accidents Evaluated

BOC - Beginning of Cycle
EOC - End of Cycle

TABLE II

COMPARISON OF RESULTS FOR LOCKED ROTOR ANALYSES

	This Study	FSAR
Moderator temperature coefficient, $\Delta k/°F$	5×10^{-5}	0
Initial power level, percent of nominal	70	100
Peak clad temperature during transient, °F	1587	1810
Peak reactor coolant system pressure, psia	2430	2440

TABLE III

SUMMARY OF ROD EJECTION RESULTS BEGINNING OF CYCLE

	Hot Zero Power	Hot Full Power
Maximum fuel pellet average temperature, °F	2169	4091
Maximum fuel center temperature, °F	2565	5185
Maximum clad average temperature, °F	1624	2367
Maximum fuel enthalpy, cal/gm	84	177
Fuel pellet melting, percent	0	< 10

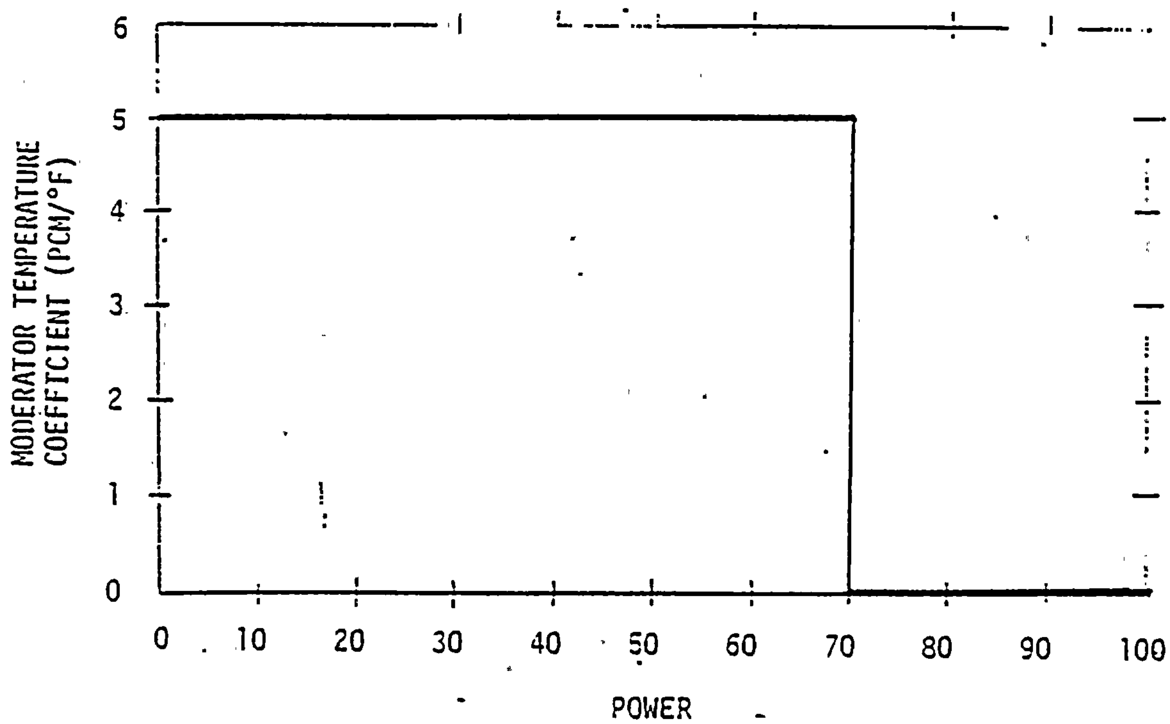

FIGURE 1 MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

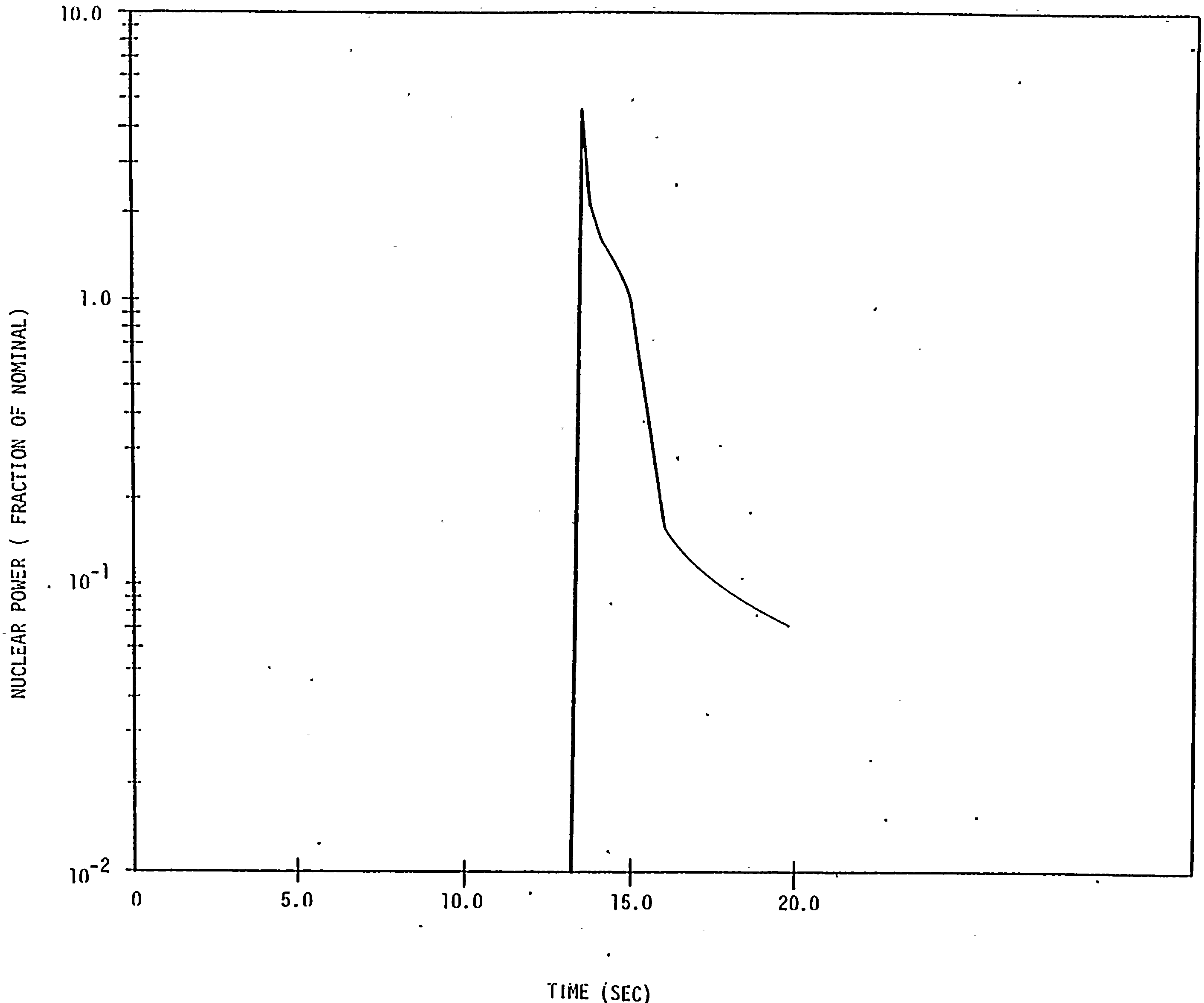

FIGURE 2 ROD WITHDRAWAL FROM SUBCRITICAL NUCLEAR POWER VS TIME

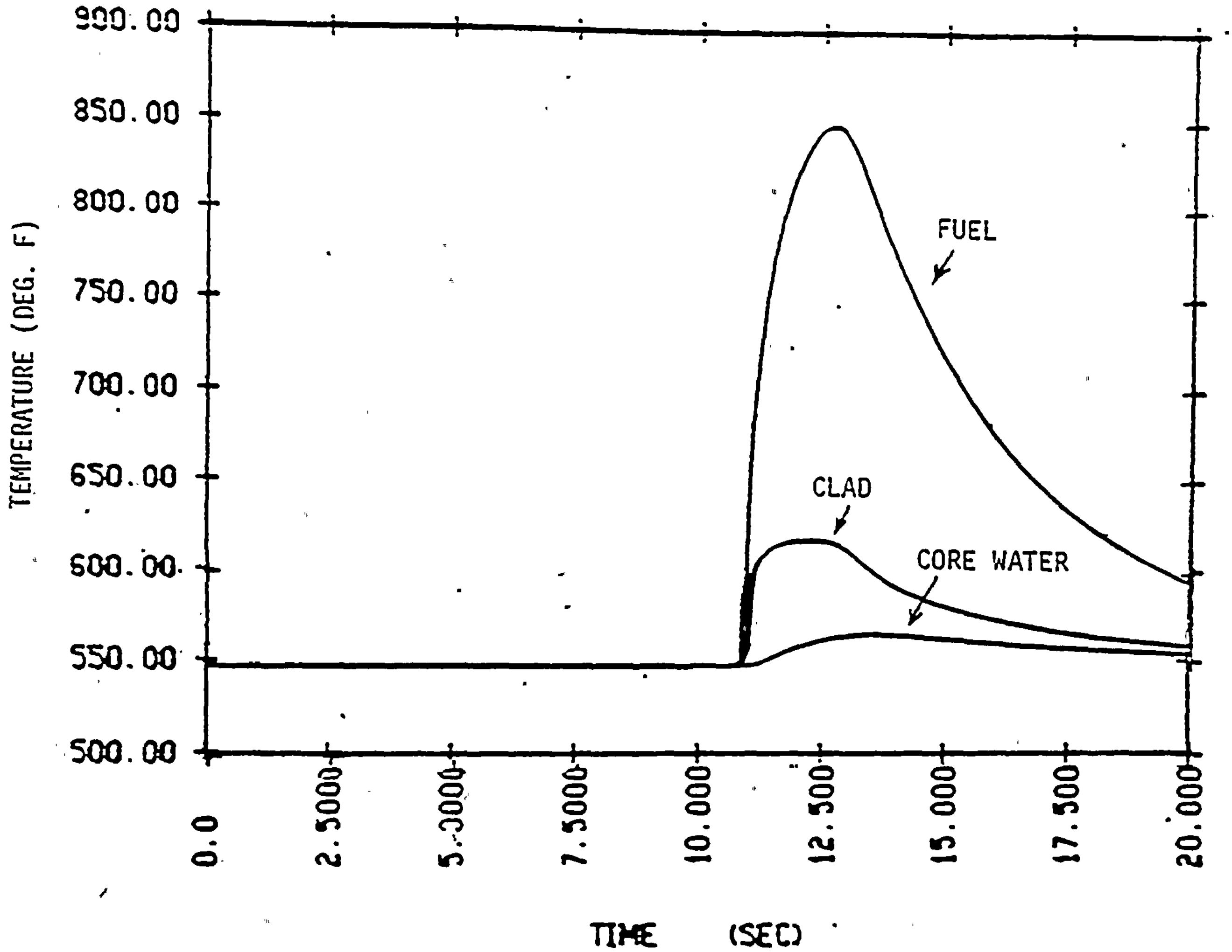

FIGURE 3 ROD WITHDRAWAL FROM SUBCRITICAL
TEMPERATURE VS TIME

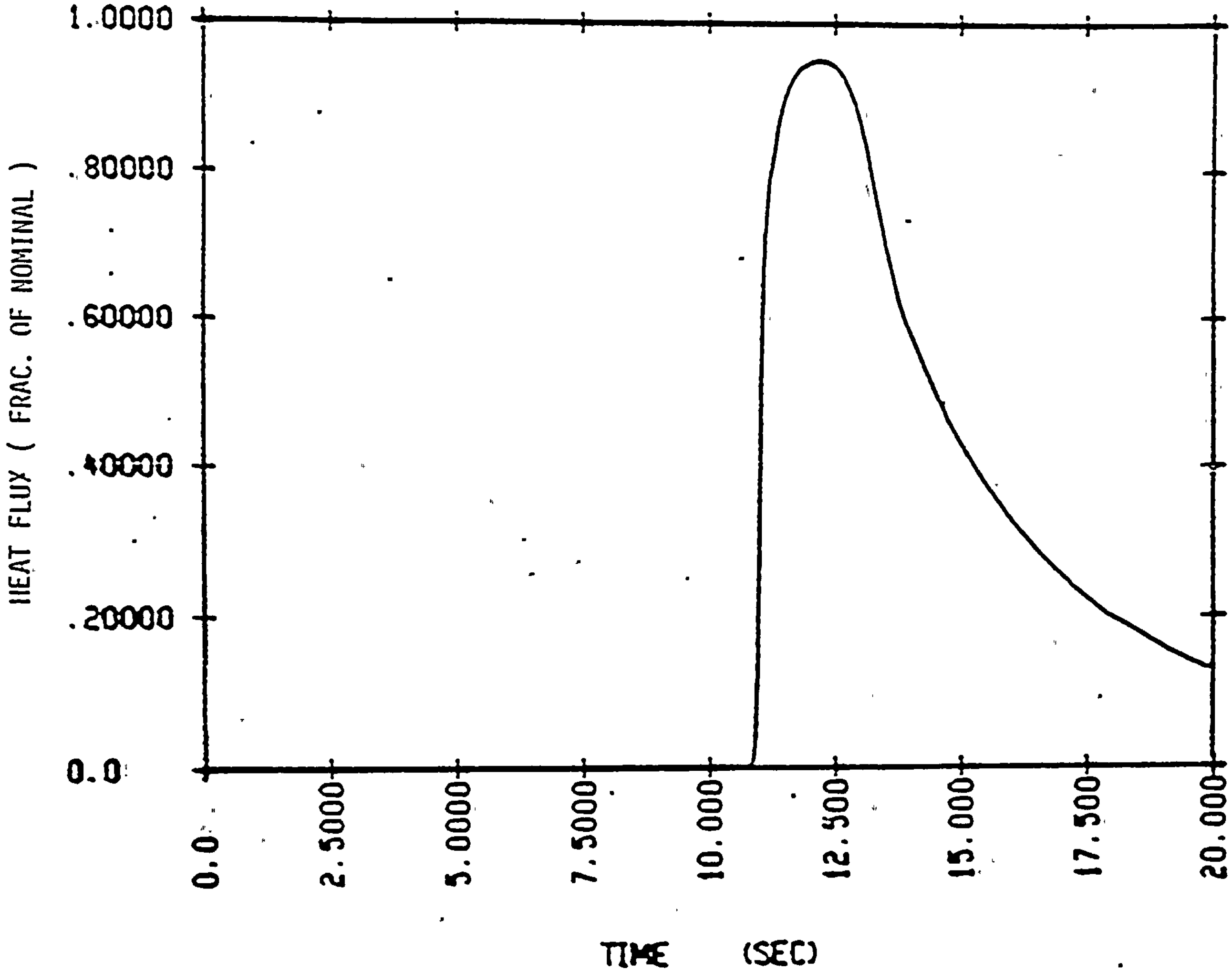

FIGURE 4 ROD WITHDRAWAL FROM SUBCRITICAL
HEAT FLUX VS TIME

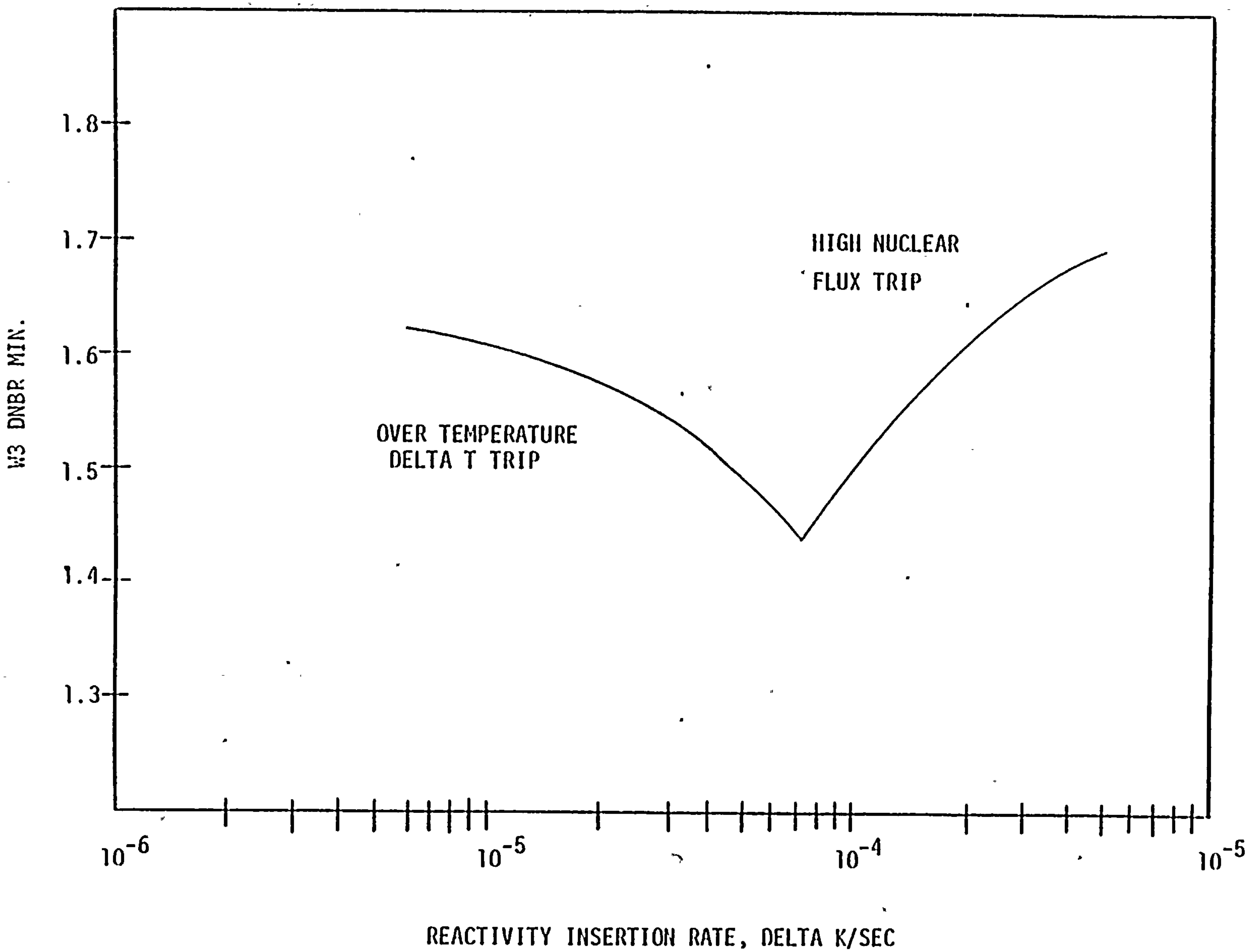
W3 DNBR MIN.
1.8
1.7
1.6
1.5
1.4
1.3
HIGH NUCLEAR
FLUX TRIP
OVER TEMPERATURE
DELTA T TRIP
10^{-6}
10^{-5}
10^{-4}
10^{-5}
REACTIVITY INSERTION RATE, DELTA K/SEC

FIGURE 5

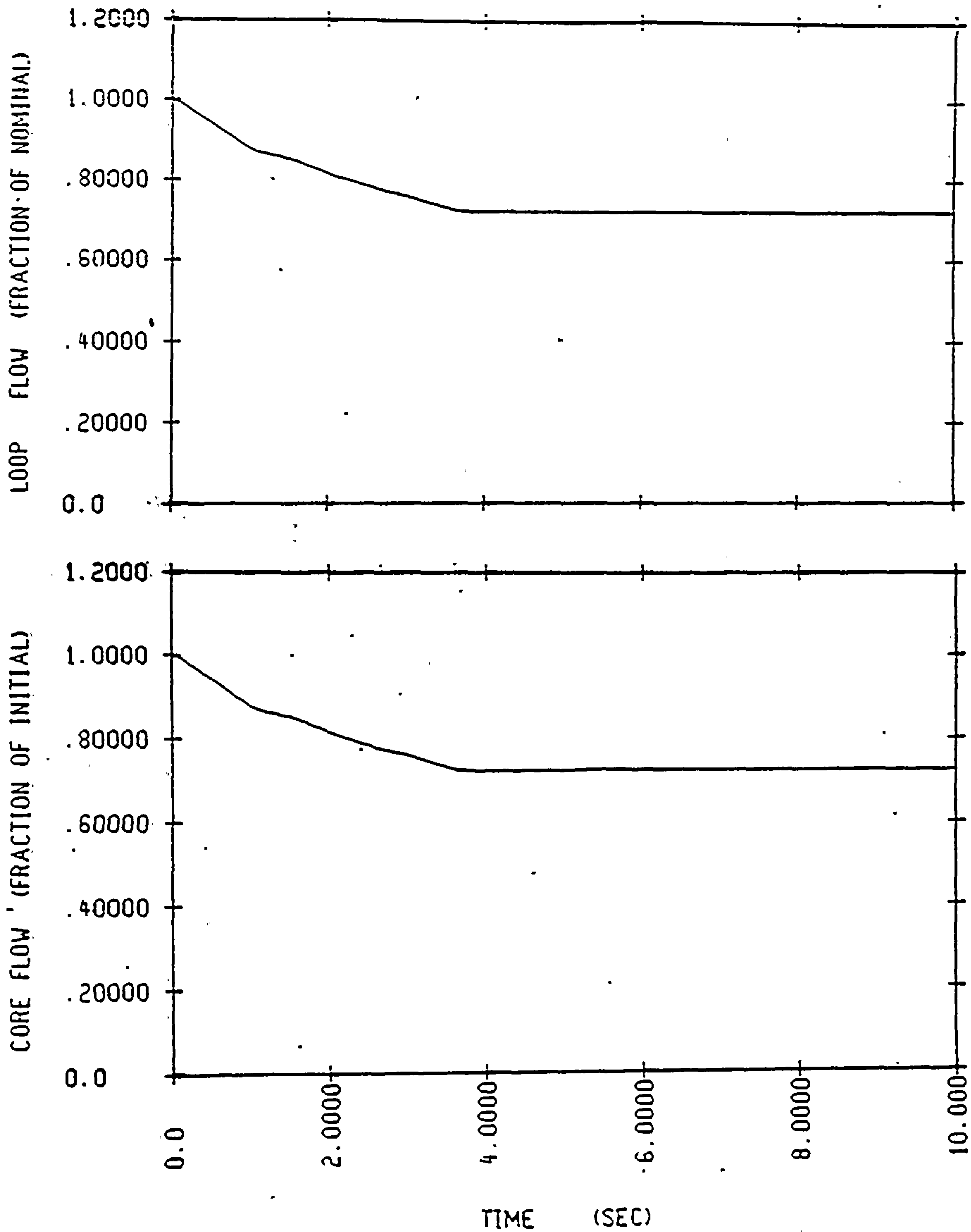

FIGURE 6 LOSS OF FLOW
FLOW VS TIME

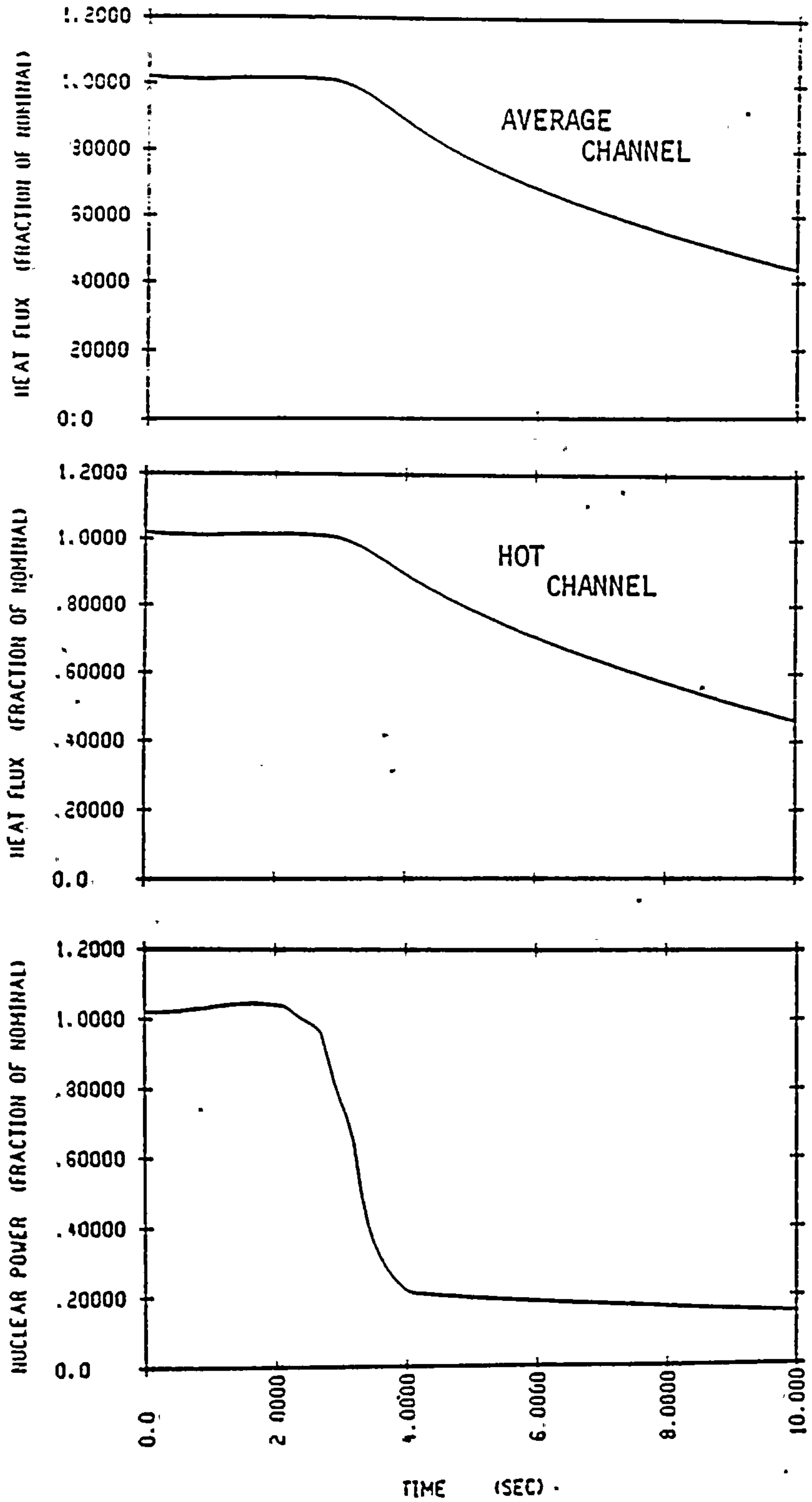

FIGURE 7 LOSS OF FLOW
HEAT FLUX VS TIME
NUCLEAR POWER VS TIME

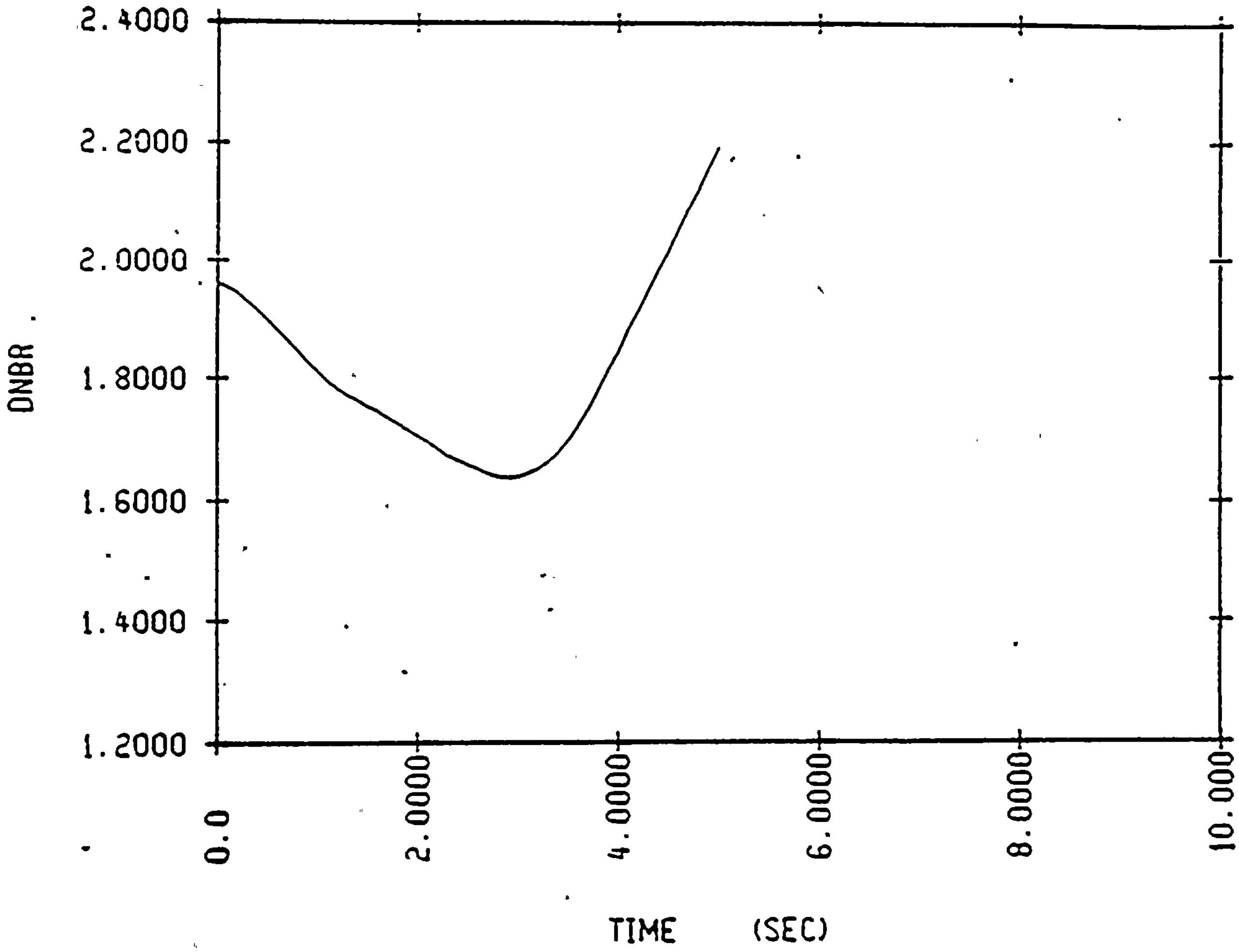

FIGURE 8 LOSS OF FLOW
DNBR VS TIME

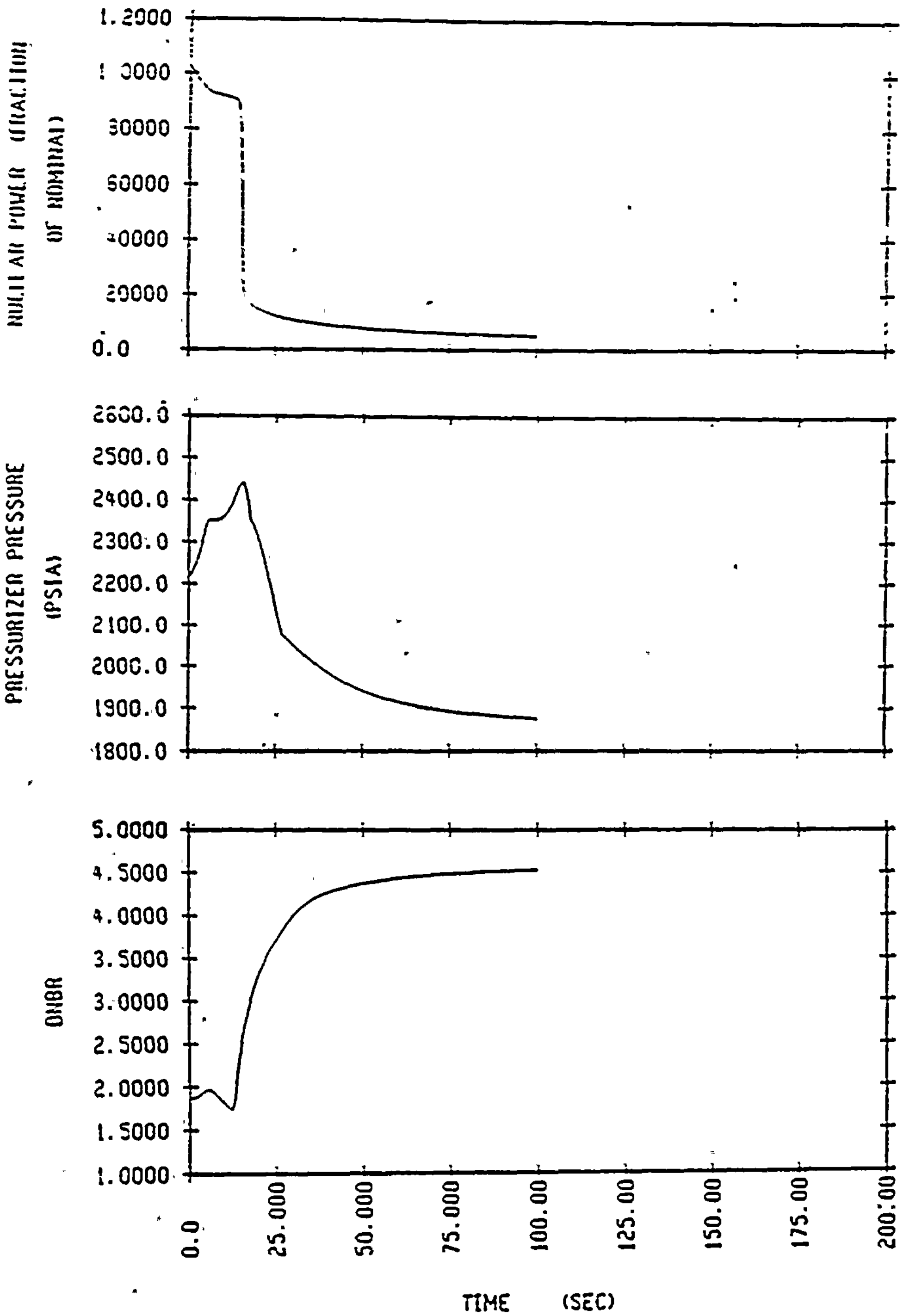

FIGURE 9 LOSS OF LOAD
WITH PRESSURE CONTROL

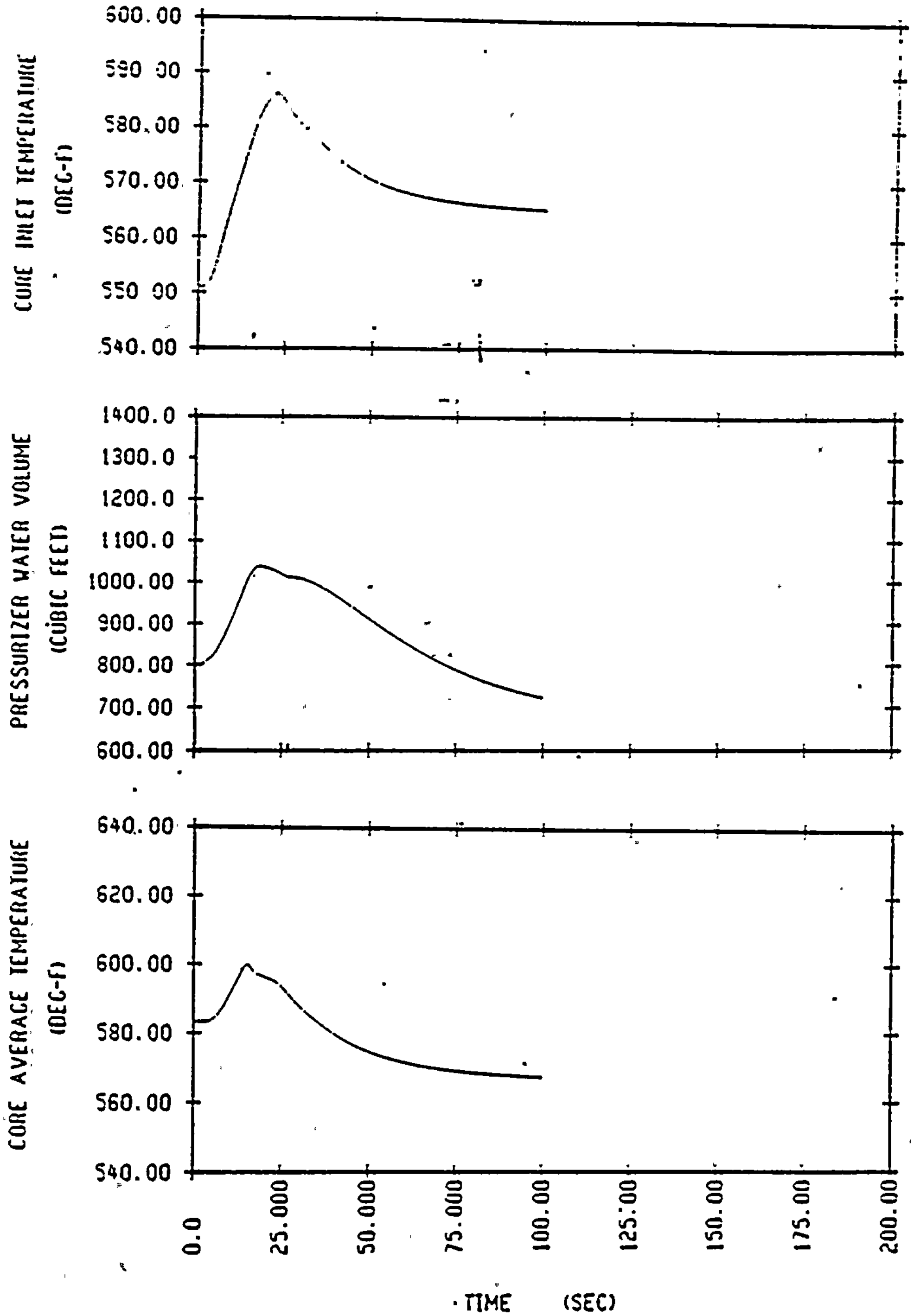

FIGURE 10 LOSS OF LOAD
WITH PRESSURE CONTROL

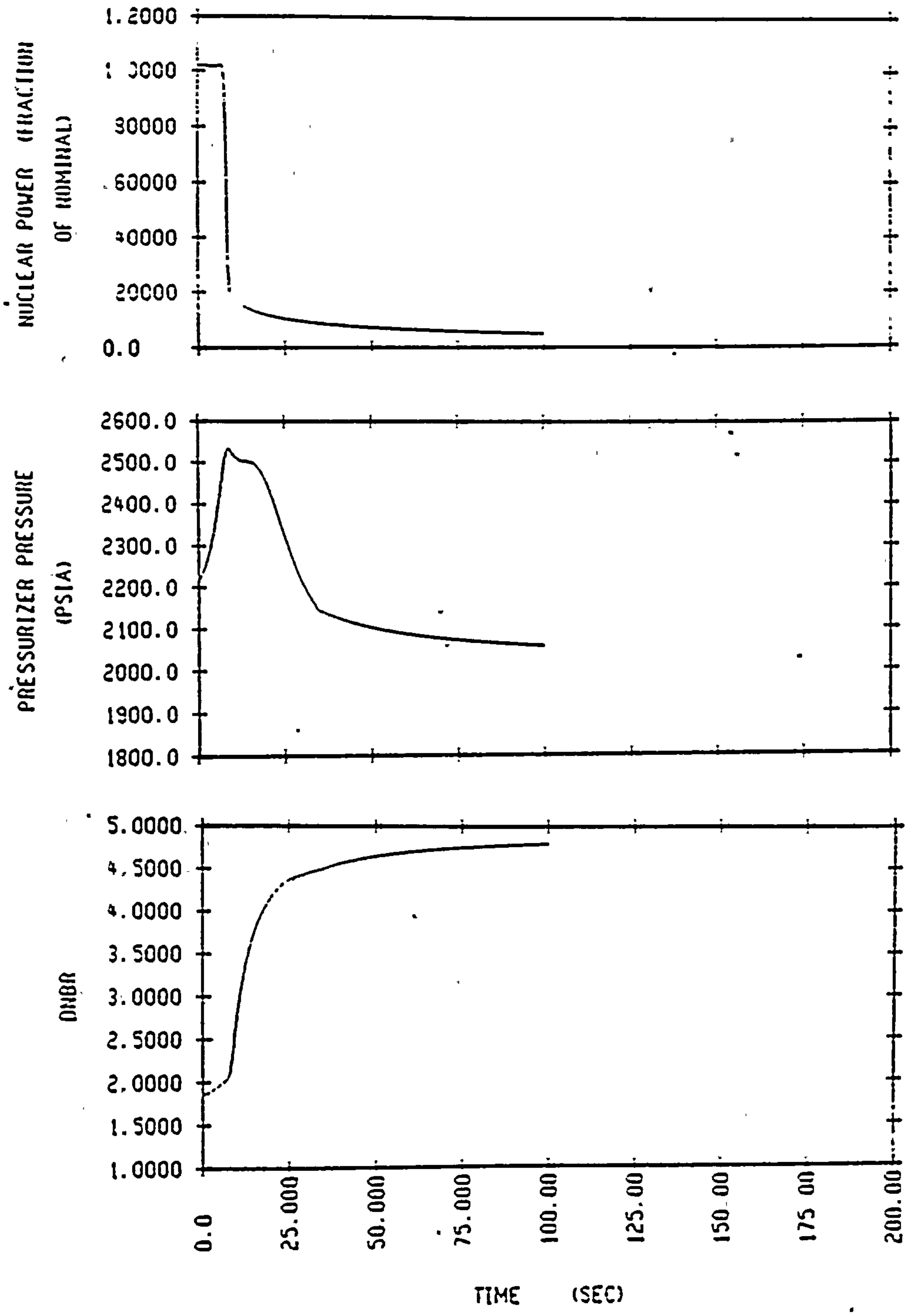

FIGURE 11 LOSS OF LOAD
MANUAL ROD CONTROL
NO PRESSURIZER RELIEF OR SPRAY

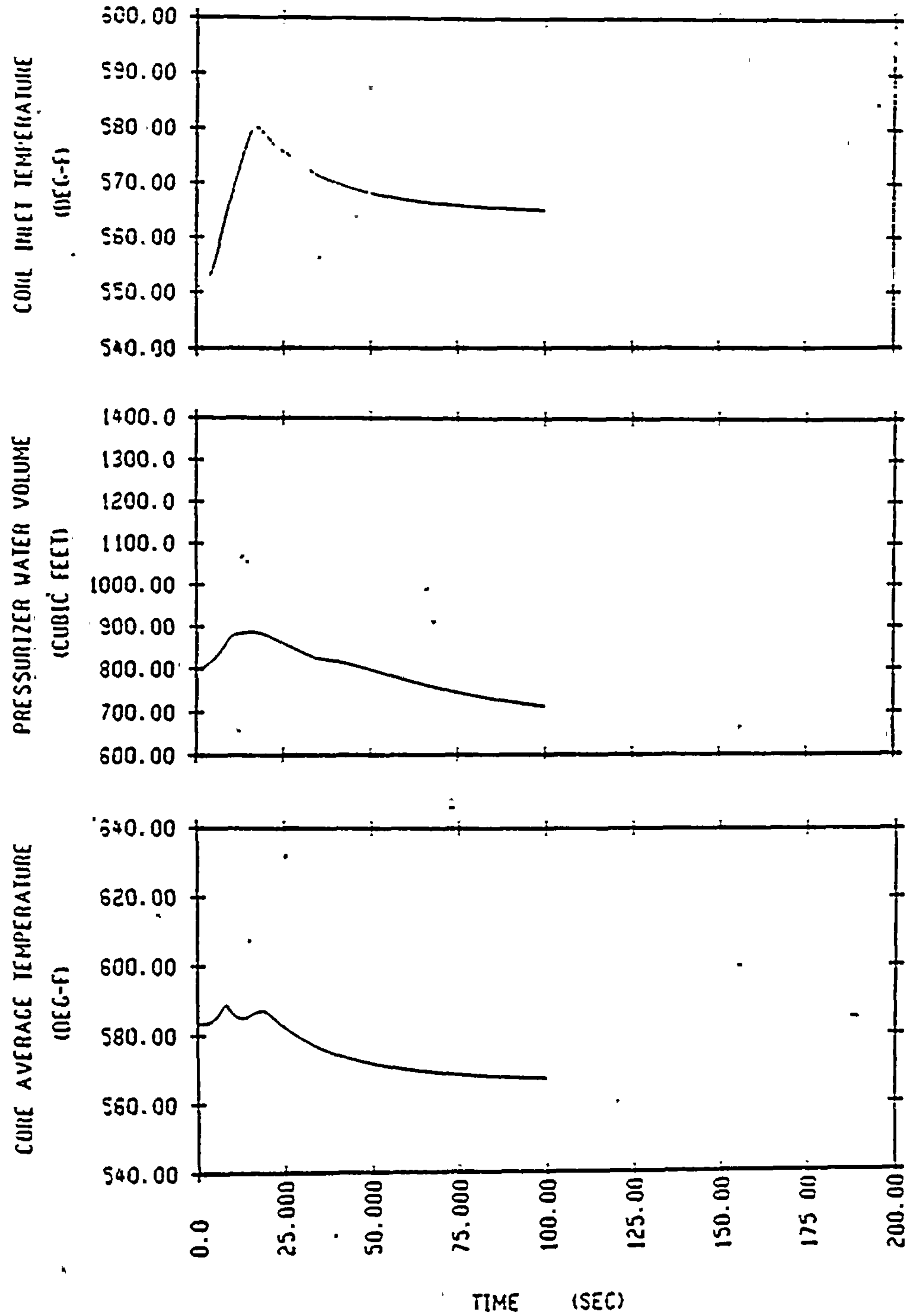

FIGURE 12 LOSS OF LOAD
MANUAL ROD CONTROL
NO PRESSURIZER RELIEF OR SPRAY

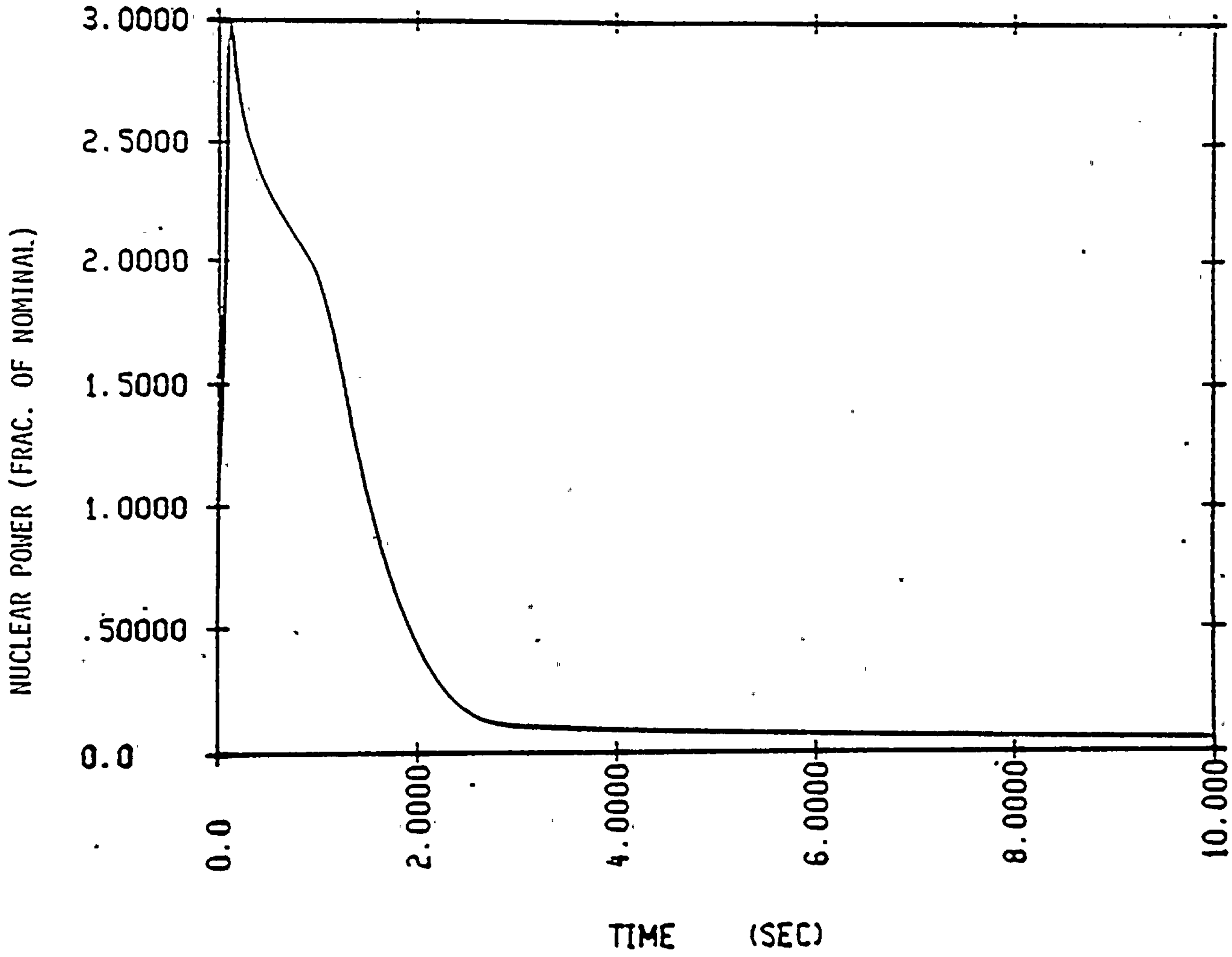

FIGURE 13 ROD EJECTION BOL HFP
NUCLEAR POWER VS TIME

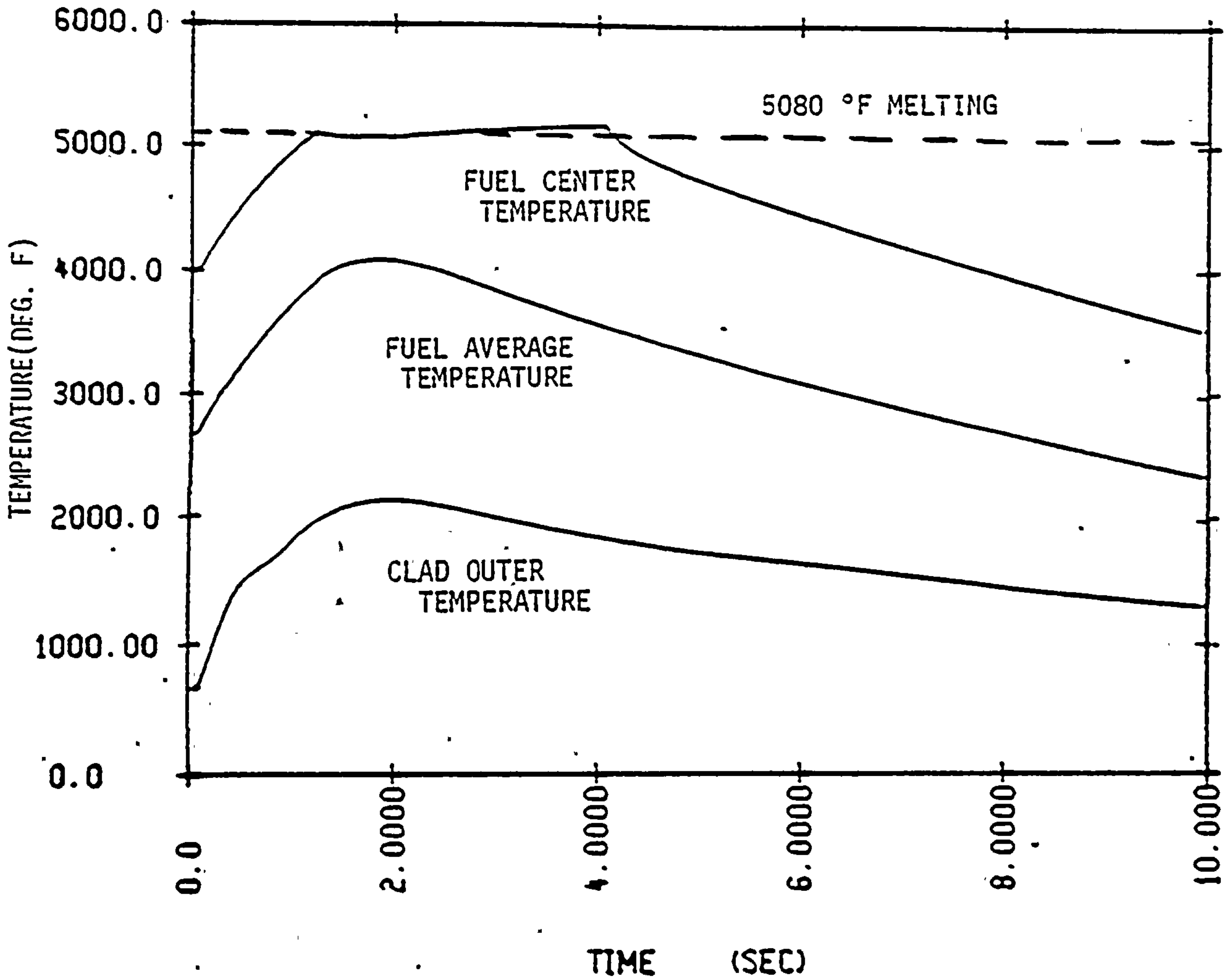

FIGURE 14 ROD EJECTION BOL HFP
TEMPERATURE VS TIME

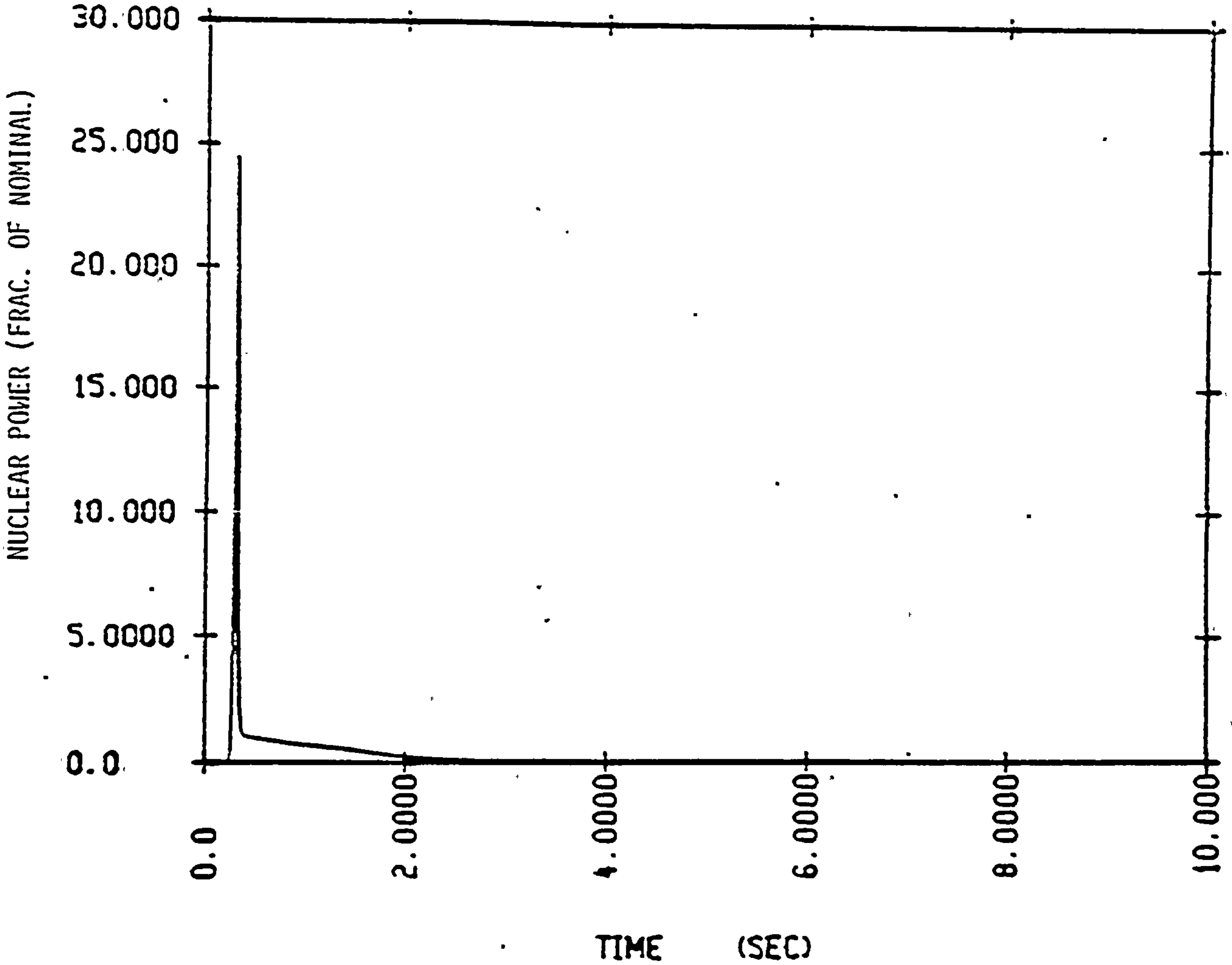

FIGURE 15 ROD EJECTION BOL HZP
NUCLEAR POWER VS TIME

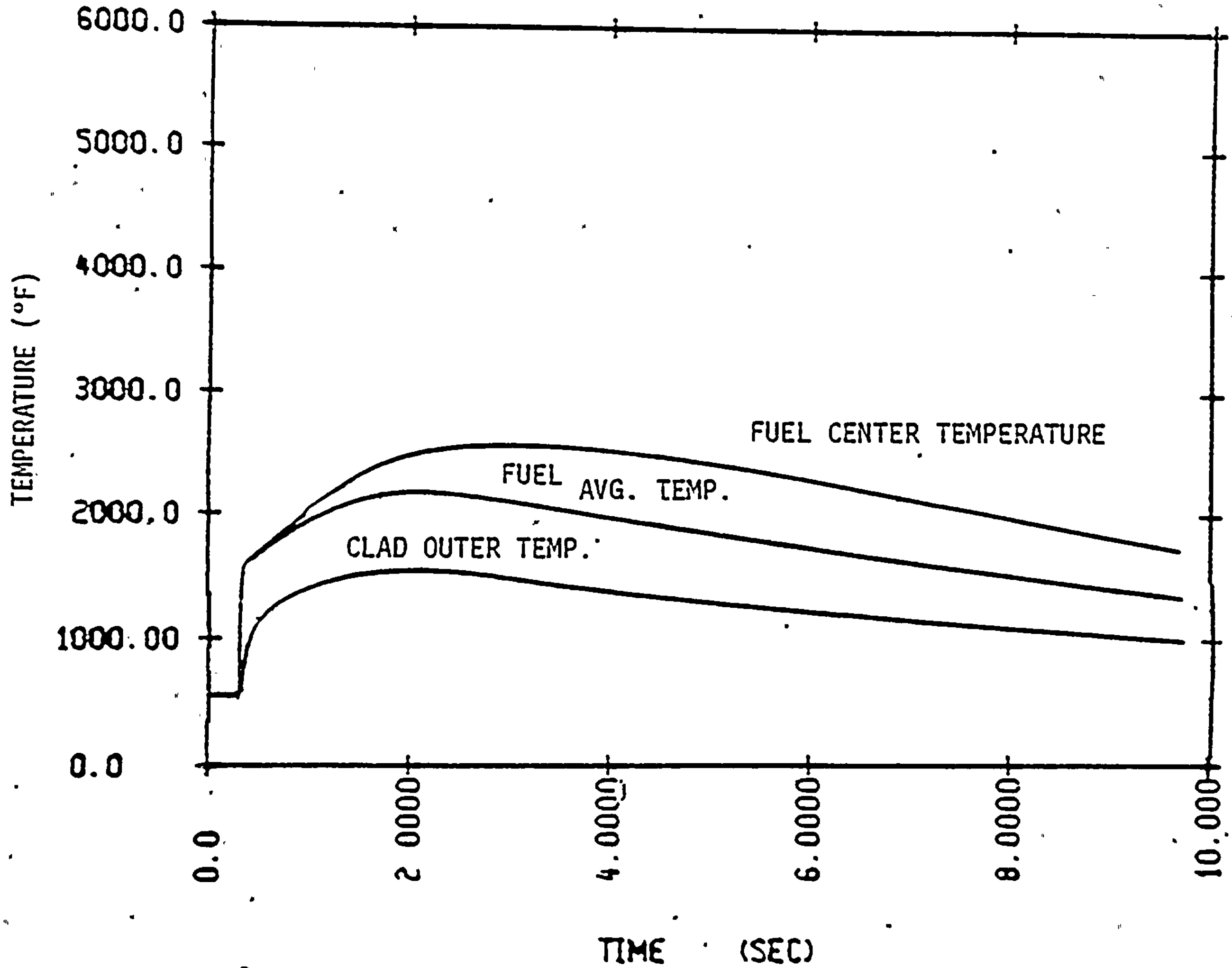

FIGURE 16 ROD EJECTION BOL HZP
TEMPERATURE VS TIME