VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

February 7, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 666
PSE/JOE/mjp/2000N
Docket Nos. 50-338
50-339
License Nos.: NPF-4
NPF-7

Gentlemen:

AMENDMENT TO OPERATING LICENSES NPF-4 AND NPF-7
NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGE

Pursuant to 10CFR50.90, the Virginia Electric and Power Company requests an amendment, in the form of changes to the Technical Specifications, to Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station Unit Nos. 1 and 2.

In our letter of July 17, 1984 (Serial No. 224A), we submitted the reload information description for the fifth cycle core of North Anna Unit 1. In that submittal, we indicated that the moderator temperature coefficient for the hot zero power, all-rods-out, beginning-of-life condition was calculated to be positive and therefore initial escalation to power was to be made with control rods inserted in the core in order to maintain a non-positive moderator temperature coefficient. A non-positive moderator temperature coefficient during normal operation is a basic assumption for the current UFSAR accident analyses for both the North Anna 1 and 2 cores and is therefore a current Technical Specification requirement (TS 3.1.1.4).

Since our July 17, 1984 submittal, we have reanalyzed the relevant UFSAR accidents for North Anna 1 and 2 in support of a positive moderator temperature coefficient at reduced power levels. By allowing a positive moderator temperature coefficient, the necessity of having the control rods significantly inserted in the core during initial startup and the potential for operating restrictions due to the delta flux limits associated with constant axial offset control are minimized. This would allow greater flexibility in core designs for both North Anna units in future cycles. Enclosure 1 provides the Safety Evaluation for the proposed changes. The resulting specific Technical Specification changes are given in Enclosure 2.

Rec'd w/check $150.00

A001
3/40

This request has been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Safety Evaluation and Control staff. It has been determined that this request does not involve any unreviewed safety questions as defined in 10CFR50.59 or a significant hazards consideration as defined in 10CFR50.92.

We have evaluated this request in accordance with the criteria in 10CFR170.12. A check in the amount of $150 is enclosed as an application fee.

Very truly yours,

W. L. Stewart

Enclosures:

(1) Safety Evaluation for Proposed Positive Moderator Temperature Coefficient
(2) Proposed Technical Specification Changes
(3) Voucher Check for $150

cc: Mr. James P. O'Reilly
Regional Administrator
Region II

Mr. Leon B. Engle
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North Anna Power Station

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COMMONWEALTH OF VIRGINIA)
)
CITY OF RICHMOND)

The foregoing document was acknowledged before me, in and for the City and Commonwealth aforesaid, today by W. L. Stewart who is Vice President - Nuclear Operations, of the Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 7th day of February, 19 85.

My Commission expires: 2-26, 19 85.

Ann C. McRae

Notary Public

(SEAL)

S/001

Enclosure 1

Safety Evaluation for
A Positive Moderator Temperature Coefficient

North Anna Power Station

Unit Nos. 1 and 2

SECTION I

INTRODUCTION

I. Introduction and Purpose

This safety analysis has been performed to address the safety considerations in allowing the North Anna Unit Nos. 1 and 2 to operate below 70% power with a small, positive moderator temperature coefficient (MTC). The results of this study show that power operation with a positive moderator temperature coefficient, as allowed by the attached proposed Technical Specifications changes, provides margin to UFSAR and other applicable safety limits.

The present North Anna Technical Specifications do not allow the units to be brought critical unless the moderator coefficient is negative, except during physics tests. This requirement is overly restrictive, since allowance of a small positive coefficient at reduced power levels would provide significantly increased fuel cycle flexibility, while only causing a minor effect on safety analysis results presented in the UFSAR. Nuclear design calculations for recent North Anna cycles have indicated that a positive moderator temperature coefficient may potentially be measured at beginning of cycle, hot zero power conditions with all control rods removed from the core. Control rod insertion may be used to make the coefficient negative, although plant startup is

lengthened and made more complex by restrictions on boron concentration and control rod movement. However, to facilitate future plant startups, it is highly desirable to allow a slightly positive moderator temperature coefficient at lower core power levels. 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 temperature coefficient more negative. Also, the boron concentration can be reduced as xenon builds into the core. Thus, there is less need to allow a positive coefficient as full power is approached. As fuel burnup is achieved, boron is further reduced and the moderator temperature coefficient will become negative over the entire operating power range.

The proposed Technical Specifications change, given in Enclosure 2, allows a +6 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 depicted in Figure 1. A power-dependent MTC was chosen to minimize the effect of the MTC upon accidents initiated from high power levels. Also, normal core physical phenomena described above result in MTC becoming more negative as power level increases. This Technical Specifications change is expected to provide a reasonable degree of flexibility in core design and plant operation for future cycles of North Anna Units 1 and 2. The proposed changes are similar to those which have been approved for the Trojan and Turkey

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*1 pcm = 1.0×10^{-5} dk/k

Point plants. In addition, the Surry Unit Nos. 1 and 2 Technical Specifications allow a positive moderator temperature coefficient.

SECTION II

ACCIDENT ANALYSIS

A. Introduction

The impact of a positive moderator temperature coefficient for North Anna Units 1 and 2 on the accident analyses presented in Chapter 15 of the UFSAR(1) has been assessed. Those incidents which were found to be sensitive to minimum or near-zero moderator temperature coefficients were reanalyzed. In general, these incidents are limited to transients which cause reactor coolant temperature to increase. The analyses presented herein were based on a +6 pcm/°F moderator temperature coefficient, which was assumed to remain constant for variations in temperature. The assumption of a positive moderator temperature coefficient existing at full power is conservative since the proposed Technical Specifications require that the reactor not be operated at full power if the temperature coefficient is positive.

In general, the reanalysis was based on the assumptions and methods employed in the UFSAR; exceptions are noted in the discussion of each incident. The UFSAR basis referred to in this evaluation is for plant operation at 2775 MWt core power with an RCS average temperature of 587.8 °F. This is the expected operating condition of both North Anna units upon implementation of the changes

supported by this report. Vepco computer codes were employed in the analysis of all accidents which were reanalyzed. The RETRAN code (2,3,5) was used to obtain overall RCS parameter responses to the positive MTC. Core DNB analyses were performed with the COBRA code (4). Accidents not reanalyzed included those resulting in excessive heat removal from the reactor coolant system (for which a large negative moderator temperature coefficient is conservative), and those which experience heatup following a reactor trip (which are not sensitive to the moderator temperature coefficient). Table 1 presents a list of accidents discussed in the North Anna UFSAR, and denotes those events reanalyzed for a positive coefficient.

B. Transients Not Affected By a Positive Moderator Temperature Coefficient

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

1. Rod Cluster Control Assembly Misalignment

   The peak heat flux following the drop of a control rod assembly is produced by action of the rod control system in response to the coolant average temperature decrease caused by an imbalance between core power and secondary system load. The existing analyses employed a 0.0 pcm/°F MTC, which maximizes the effect of the power overshoot for negative flux rate trip plants. Since the limiting conditions for this accident are at or near 100% power and the proposed change requires that MTC be less than 0.0 pcm/°F when above 70% power, this accident is not affected by the proposed Technical Specification and the analysis was not repeated.

2. Startup of an Inactive Reactor Coolant 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. As the most negative values of moderator reactivity coefficient produce the greatest reactivity addition, the analysis reported in the UFSAR, Section 15.2.6, represents the limiting case. Startup of an inactive loop with loop stop valves closed is effectively a boron dilution

accident, and will be discussed in Section II.C.

3. Excessive Heat Removal Due to Feedwater System Malfunctions

The addition of excessive feedwater and inadvertent opening of the feedwater bypass valve are excessive heat removal incidents, and are consequently most sensitive to negative moderator temperature coefficients. Results presented in Section 15.2.10 of the UFSAR indicate that the end of life case with a conservatively large negative moderator temperature coefficient results in the minimum margin to DNB. Therefore, this incident was not reanalyzed.

4. 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 15.2.11 of the UFSAR shows that the limiting case is with a large negative moderator temperature 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 UFSAR analysis of this case shows that the minimum DNBR is not sensitive to the moderator temperature coefficient. The UFSAR analysis cases are therefore still applicable to this incident.

5. Loss of Normal Feedwater, Loss of Offsite Power to Station Auxiliaries

The loss of normal feedwater and loss of offsite power accidents (Sections 15.2.8 and 15.2.9 of the UFSAR) are characterized by a gradual temperature rise due to decay heat production and subsequent temperature reduction to the no load average value. A positive moderator temperature coefficient will not affect these transients, since reactor trip occurs at the beginning of the transient, and the moderator reactivity coefficient will become negative following control rod insertion. Therefore, there is no reduction in shutdown margin due to the heatup of the reactor coolant system.

6. Accidental Depressurization of the Reactor Coolant System

An accidental depressurization of the reactor coolant system results from an inadvertent opening of a pressurizer safety valve (UFSAR Section 15.2.12). The most limiting case assumes the reactor is in automatic control, where the rod control system functions to keep the power and average coolant temperature essentially constant until the reactor trip. This portion of the transient is not sensitive to a positive moderator temperature coefficient. Following the reactor trip, the average coolant temperature decreases slowly. Thus, the results presented in the UFSAR represent the most limiting conditions. Therefore, this transient was not reanalyzed with a positive moderator temperature coefficient.

7. Rupture of a Main Steam Pipe/Accidental Depressurization of the Main Steam System

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 UFSAR Section 15.4.2. Similarly, the accidental depressurization of the main steam system is a temperature reduction transient. A strong negative moderator temperature coefficient results in the minimum core shutdown margin. Thus, the worst conditions for this transient are those analyzed in UFSAR Section 15.2.13.

8. Spurious Operation of Safety Injection

Analysis of a spurious operation of safety injection at power is presented in Section 15.2.14 of the UFSAR. This transient results in a decrease in average coolant temperature and is most sensitive to a negative moderator temperature coefficient. Therefore, this incident was not reanalyzed with a positive moderator temperature coefficient.

9. Rupture of a Main Feedwater Pipe

The rupture of a main feedwater pipe accident (UFSAR Section 15.4.2) is analyzed to confirm the ability of the secondary system to remove decay heat. This event is not sensitive to a

positive moderator coefficient since the reactor trip occurs early in the transient before the reactor coolant system temperature increases significantly. Therefore, this event was not reanalyzed with a positive moderator temperature coefficient.

10. Loss of Coolant Accident (LOCA)

The loss of coolant accident (UFSAR Sections 15.3.1 and 15.4.1) 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 moderator temperature coefficient be zero or negative at 70 percent power or above ensures that the previous analysis basis for this event is not affected.

C. Transients Sensitive to a Positive Moderator Coefficient

The following incidents have been identified as being potentially sensitive to a positive moderator temperature coefficient, and the consequences of these incidents were reassessed.

1. Uncontrolled Boron Dilution

For a boron dilution incident during refueling or startup, while the reactor is subcritical, Section 15.2.4 of the UFSAR shows that the operator has sufficient time to identify the problem and terminate the dilution before the reactor becomes critical. The UFSAR (Section 15.2.6) also shows that the operator has sufficient time to terminate a boron dilution during startup of an inactive loop. This incident is caused by violation of administrative procedures which require that boron concentration in the inactive loop be checked prior to opening the loop stop valve. These incidents are therefore not affected by the value of the moderator temperature coefficient. The reactivity addition due to a boron dilution at power however will cause an increase in power and reactor coolant system temperature if the reactor is in manual control. Due to the temperature increase, a positive moderator temperature coefficient would add additional reactivity, and increase the severity of the transients. However, this incident is no more severe than a rod withdrawal at power, which is analyzed in this section, and was therefore not specifically reanalyzed. Following reactor trip, the amount of time available before shutdown margin is lost is not affected by the moderator

temperature coefficient.

2. 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 15.2.1 of the UFSAR). 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. However, since the power rise is rapid and is followed by an immediate reactor trip, the moderator temperature rise is small. Therefore, the transient is only moderately sensitive to the moderator temperature coefficient.

Method of Analysis

The UFSAR states that for this transient, the highest value of peak heat flux is produced for the highest rate of reactivity insertion and lowest initial power. The analysis was reanalyzed with a +6 pcm/°F MTC and the insertion rate of 75 x 10^{-5} dk/k / sec assumed in the UFSAR. The initial power level, reactor trip instrument delays and setpoint errors used in the analysis were consistent with the UFSAR. As a result of recent Westinghouse concerns related to the number of RC pumps allowed to be operating per the Technical Specifications, this analysis was performed for the more limiting case with one RC pump

operating. In addition, a more appropriate Doppler temperature coefficient which conservatively bounds current reload cycle values was used.

Results and Conclusions

The nuclear power, coolant temperature, heat flux, fuel average temperature and clad temperature versus time for a 75×10^{-5} dk/k/sec insertion rate are shown in Figures 2 through 4. Although the nuclear power exceeds the full power nominal value for a very short period of time, the peak heat flux, peak coolant temperature and thermal power do not exceed nominal full power values. Since the heat flux does not exceed the nominal full power value and remains bounded by the UFSAR results, the conclusions presented in the UFSAR are still applicable. In addition, a detailed thermal hydraulic analysis has shown that thermal margin limits are met.

3. 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 temperature coefficient would augment the power mismatch and could reduce the margin to DNB. A discussion of this incident is presented in Section 15.2.2 of the UFSAR.

Method of Analysis

The transient was reanalyzed employing the same assumptions regarding initial conditions and instrumentation and setpoint errors used in the UFSAR. The analyses were only performed for a power level of 102% of 2775 MWt, since this is the most limiting case presented in the existing plant analyses. A constant moderator temperature coefficient of +6 pcm/°F was used in the analysis. The assumption that a positive moderator temperature coefficient exists at full power is conservative, since the moderator temperature coefficient will actually be zero or negative at full power.

Results

Figure 5 shows the minimum DNBR as a function of reactivity insertion rate for the full power cases reanalyzed. The limiting case for DNB margin is a reactivity insertion rate of

4.0 X 10^{-5} dk/k/sec. Using the conservative +6 pcm/°F moderator temperature coefficient, instead of the 0 pcm/°F limit allowed by the Technical Specification change, results in a minimum DNBR greater than the 1.30 limit value. This positive moderator temperature coefficient will therefore not lower the DNBR associated with a control rod assembly withdrawal at power below the design limit.

Conclusions

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

4. Loss of Reactor Coolant Flow

Introduction

As demonstrated in UFSAR Section 15.3.4, 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.

Method of Analysis

Analysis methods and assumptions used in the reevaluation were consistent with those employed in the UFSAR. The analysis was performed using a constant +6 pcm/°F moderator temperature coefficient coupled with the maximum Doppler temperature coefficient.

Results

For the case reanalyzed, the reactor coolant average temperature increases less than 3°F above the initial value. The impact of the positive moderator coefficient on the nuclear power transient would be limited to the initial stages of the incident during which the average reactor coolant temperature increases. This increase is terminated shortly after reactor trip. The reactor coolant system response and the transient DNBR response are similar to those for the UFSAR case which assumed a zero MTC. A lower DNBR value is expected as a result of the slight increase in the nuclear power transient. However, analysis with the positive moderator temperature coefficient confirmed that the minimum DNBR was greater than 1.30. Therefore, the effect of the positive moderator temperature coefficient on this transient is acceptable. Figures 6 through 8 show the flow coastdown, the nuclear power and heat flux transients, and the minimum DNB ratio vs. time for the +6 pcm/°F case.

Conclusions

A positive moderator temperature coefficient causes only minor changes in the results of the complete loss of flow transient, and the minimum DNBR remains above 1.30 for this incident. This case was analyzed since it is the most limiting one presented in the UFSAR. Loss of a single pump with all loops in service or with a single loop out of service and the loop stop valves open or closed were less limiting. Since this type of transient causes only a small change in core average moderator temperature, and this change does not significantly affect the nuclear power transient, the single pump loss of flow cases are not appreciably affected and therefore remain less limiting.

5. Locked Rotor

Introduction

The UFSAR (Section 15.4.4) shows that the most severe locked rotor incident is an instantaneous seizure of a reactor coolant pump rotor at 100% power with three loops operating. Following the incident, reactor coolant system temperature rises until shortly after reactor trip. A positive moderator temperature 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 analysis was performed for a +6pcm/°F moderator temperature coefficient. The initial conditions and assumptions used in this evaluation were consistent with those employed in the UFSAR. The RETRAN Hot Spot Model described in Reference 5, using the locked rotor transient assumptions, was used to evaluate the fuel rod thermal transient.

Results

Figures 9 through 12 show system response for the case reanalyzed. The nuclear power transient was most significantly affected by the positive moderator temperature coefficient. Peak nuclear power for this case was 106% of nominal. This effect resulted from the use of a conservatively large bypass flow fraction which was assumed in order to accentuate the pressure transient. This large bypass flow produces an even greater increase in core water temperature, which causes the reactivity contribution of the positive moderator temperature coefficient to be overestimated.

Table 2 provides results consistent with those presented in the UFSAR. The values of peak clad temperature and peak pressure are well below the applicable limit for this event.

Conclusions

A positive moderator temperature coefficient does not adversely affect the consequences of a locked rotor at full power with three loops operating. The integrity of the reactor coolant system is not endangered as peak pressure during the transient is 2616 psia. Since a locked rotor with three loops operating is the limiting case presented in the UFSAR, a positive moderator temperature coefficient will also not significantly affect the consequences of the two loop operation cases.

6. Loss of External Electrical Load

Introduction

Two cases, analyzed for both beginning and end of life conditions, are presented in Section 15.2.7 of the UFSAR:

1. Reactor in manual 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 pressurizer power operated relief valv

Since the moderator temperature coefficient will be negative at end of life, only the beginning of life cases were reanalyzed. 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 this reactivity addition due to a positive moderator temperature coefficient are increases in both peak nuclear power and pressurizer pressure.

Method of Analysis

A constant moderator temperature coefficient of +6 pcm/°F was assumed for the beginning of life cases reanalyzed. The method of analysis and assumptions used were otherwise in accordance with those presented in the UFSAR. These assumptions included an initial reactor power and coolant temperature which were assumed to be at their maximum values consistent with steady state full power operation, including allowances for calibration and instrument errors. The reactor coolant system pressure was assumed at its minimum value. The steam dump and direct reactor trip on turbine trip were not assumed to operate.

Results

System transient response to a total loss of load from 102% power, at beginning of life, assuming pressurizer spray and pressurizer power operated relief valves, is shown in Figures 13 and 14. The reactor trips on high pressurizer pressure, assumed to occur at 2425 psia; pressurizer pressure subsequently rises to 2520 psia. The minimum DNBR decreases from its initial value to its minimum value shortly after the reactor trip. The minimum DNBR remains greater than 1.30 during the event.

Figures 15 and 16 illustrate reactor coolant system response to a loss of load at beginning of life, assuming no credit for

pressure control. Peak pressurizer pressure reaches 2546 psia following reactor trip on high pressurizer pressure. The minimum DNBR increases from its initial value 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 1.3 limit, and the peak reactor coolant pressure is less than 110% of design. Therefore, the conclusions presented in the UFSAR are still applicable.

7. 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 reactivity addition increases nuclear power and hot spot fuel temperatures.

Method of Analysis

The method of analysis is the same as reported in Reference 5. The ejected rod worths and transient peaking factors are the same as those in the UFSAR. The acceptance criteria are the same as the Westinghouse limit criteria, which are discussed in Reference 5. Reference 6 provides the basis for these criteria. The values of significant input parameters used in the analysis are presented in Table 3. The moderator temperature coefficient used was a constant +6 pcm/°F over the range of coolant average temperature involved.

Results and Conclusions

Peak fuel and clad temperatures and nuclear power versus time for both cases are presented in Figures 17 through 20. The limiting peak hot spot clad temperature, 2493°F, was reached in the hot full power transient. Maximum fuel temperatures were also associated with the full power case. Although peak hot spot fuel centerline temperature reached 4900°F, the assumed melting point of irradiated fuel, melting was restricted to less than the innermost 10% of the pellet.

As peak fuel and clad temperatures do not exceed the fuel and clad limits presented in Section 1.3 of the Vepco rod ejection topical (5), 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 3.

SECTION III

CONCLUSIONS

To assess the effect on accident analysis of operation of North Anna Units 1 and 2 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, locked rotor, and control rod ejection. This study indicated that a small positive moderator temperature coefficient does not result in the violation of safety limits for the transients analyzed.

The analyses employed a constant moderator temperature coefficient of +6 pcm/°F, independent of power level. The results of this study are therefore conservative, since a positive moderator temperature coefficient is precluded by the proposed Technical Specifications for full power operation.

Analyses of the transients in Section 15 of the UFSAR that are affected by the change to a positive moderator temperature coefficient have been performed to demonstrate that these transients meet the appropriate transient acceptance criteria. As such, it can be concluded that the change to a positive moderator temperature coefficient will not cause safety limits to be exceeded for any incident and consequently no unreviewed safety questions as defined in 10CFR50.59 exist as a result of this proposed change.

The results of this evaluation can be stated as follows.

1. No increase in the probability of occurrence or consequences of an accident will result from this proposed change. None of the plant systems will undergo physical changes for the change to a positivie moderator temperature coefficient and therefore no change in the associated transient probabilities is expected.

2. Since the proposed change causes no other system changes (e.g., alterations in plant configuration), and given that the effects upon system accident response are fully described by the parameters evaluated, operation with this proposed change does not create the possibility of an accident of different type than any evaluated previously in the Safety Analysis Report.

3. The margin of safety as defined in the bases for the Technical Specifications is not reduced. The calculated safety parameters for the affected transients are all within the allowable limits for the respective transients.

It has been determined that the proposed change in moderator temperature coefficient does not pose a significant hazard consideration. This is based upon example vi of those types of license amendments that are considered unlikely to involve significant hazards considerations (7). Example vi partially states, " A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the systems or component specified in the Standard Review Plan." Some analysis results do show incremental increase in accident consequences. However, the analysis results clearly show that all of the acceptance criteria for these types of transients are met and the appropriate safety margins are maintained.

References

1) "Updated Final Safety Analysis Report - North Anna Power Station, Units 1 and 2," Docket Nos. 50-338, 50-339, Rev. 2, June 1984.

2) "Vepco Reactor System Transient Analysis Using the RETRAN Computer Code," VEP-FRD-41, March 1981.

3) "RETRAN-02--A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI-NP-1850-CCM, May 1981.

4) "Vepco Reactor Core Thermal-Hydraulic Analysis Using the COBRA IIIC/MIT Computer Code," VEP-FRD-33-A, October 1983.

5) "Vepco Evaluation of the Control Rod Ejection Transient," VEP-NFE-2, October 1983.

6) "An Evaluation of the Rod Ejection Accident in Westinghouse PWR's Using Spatial Kinetics Methods", WCAP-7588, Rev. 1-A, January, 1975.

7) Federal Register, Vol. 48, No. 67, April 6, 1983, p. 14864 "Standards for Determining Whether License Amendments Involve No Significant Hazards Considerations," Interim Final Rule.

TABLE 1

ACCIDENTS EVALUATED FOR
POSITIVE MODERATOR COEFFICIENT EFFECTS

FSAR	ACCIDENT	TIME IN LIFE
* 15.2.1	RCCA Withdrawal from Subcritical	BOC
* 15.2.2	RCCA Withdrawal from Power	BOC/EOC
15.2.3	RCCA Misalignment/Drop	BOC
* 15.2.4	Boron Dilution	BOC
* 15.2.5/3.4	Loss of Flow	BOC
15.2.6	Startup of an Inactive Loop	EOC
* 15.2.7	Loss of Load/Turbine Trip	BOC/EOC
15.2.8	Loss of Feedwater	-
15.2.9	Loss of Offsite Power	-
15.2.10	Feedwater Malfunction	EOC
15.2.11	Excessive Load Increase	BOC/EOC
15.2.12	Accidental Depressurization of RCS	BOC
15.2.13/4.2	Steam Line Break	EOC
15.2.14	Spurious Operation of SI	BOC
15.3.1/4.1	LOCA	BOC
15.4.2	Feed Line Break	-
* 15.4.4	Locked Rotor	BOC
* 15.4.6	RCCA Ejection	BOC/EOC

*Accidents Reanalyzed

BOC - Beginning of Cycle

EOC - End of Cycle

TABLE 2

SUMMARY OF RESULTS FOR LOCKED-ROTOR TRANSIENTS

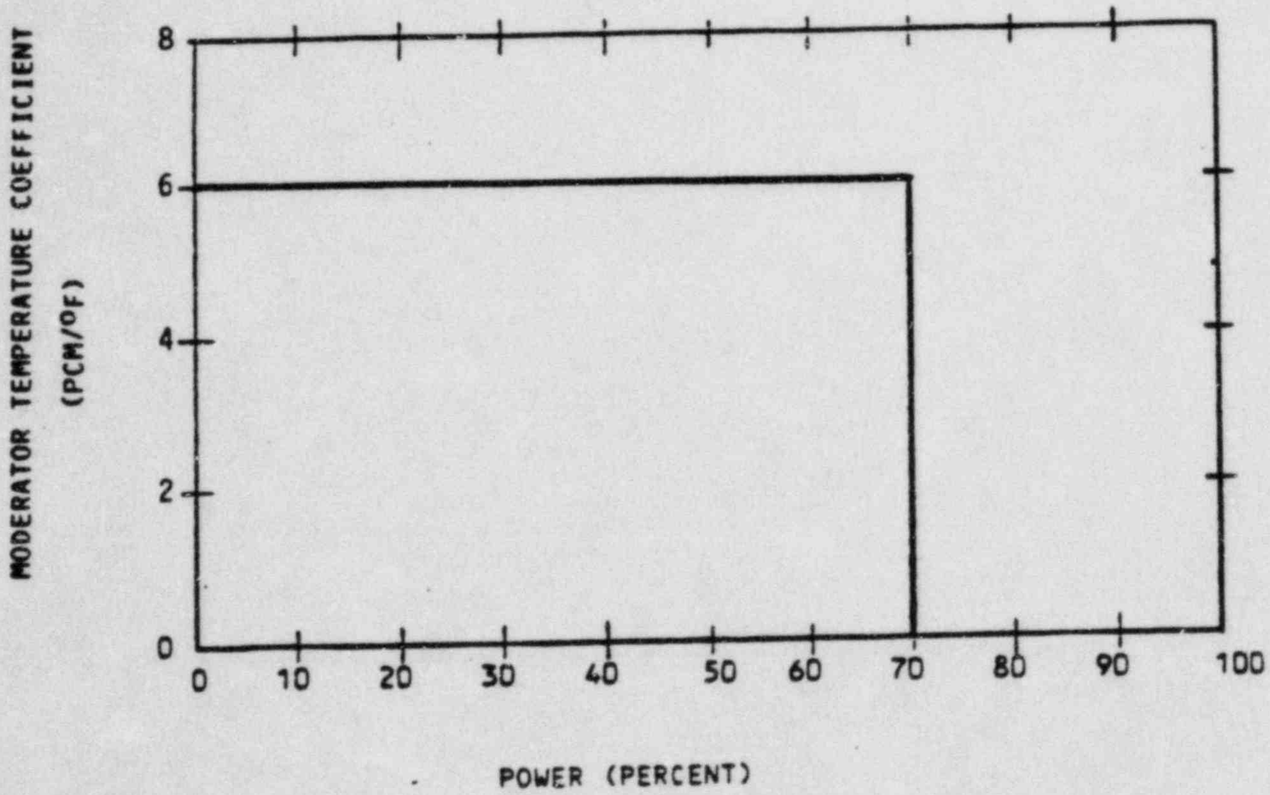
Maximum primary coolant system pressure (psia)	2616
Maximum clad temperature (^{o}F), core hot spot	2273
Amount of $Zr\text{-}H_2O$ at core hot spot (% by weight)	1.374

TABLE 3

SUMMARY OF ROD EJECTION ANALYSIS PARAMETERS AND RESULTS
BEGINNING OF CYCLE

Power Level, %	102	0
Ejected rod worth, %Δk	0.20	0.878
Delayed neutron fraction, %	.52	.52
Feedback reactivity weighting	1.68	3.23
Trip rod shutdown, %Δk	4.0	2.0
F_Q before rod ejection	2.52	--
F_Q after rod ejection	7.07	16.07
Number of operating pumps	3	2
Maximum fuel pellet average temperature, ^{o}F	4046	3502
Maximum fuel center temperature, ^{o}F	4904	4119
Maximum clad temperature, ^{o}F	2493	2486
Maximum fuel stored energy, cal/gm	188	150

FIGURE 1



MODERATOR TEMPERATURE COEFFICIENT
VS. POWER LEVEL

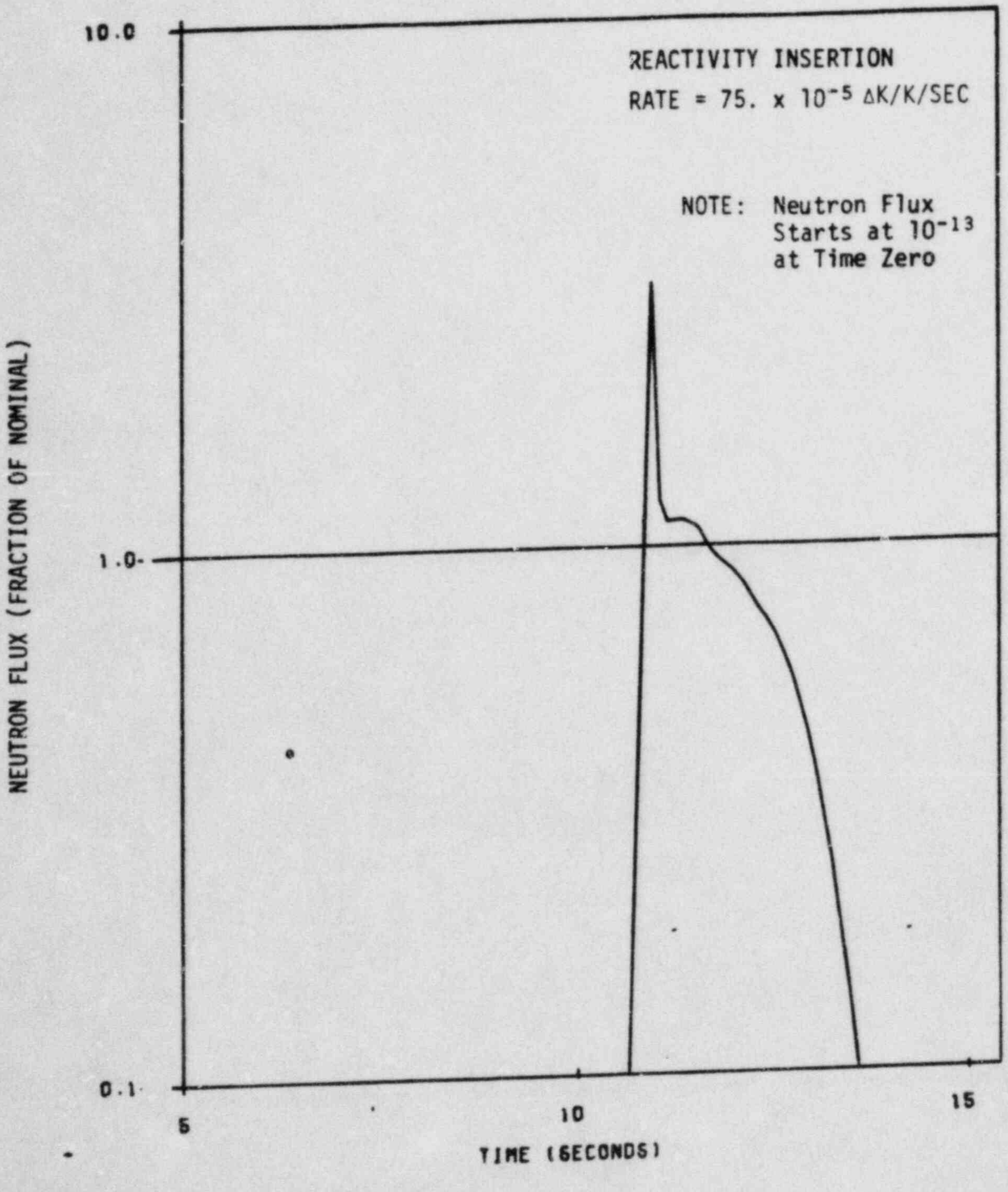


Figure 2 Uncontrolled Rod Withdrawal from a Subcritical Condition, Neutron Flux versus Time

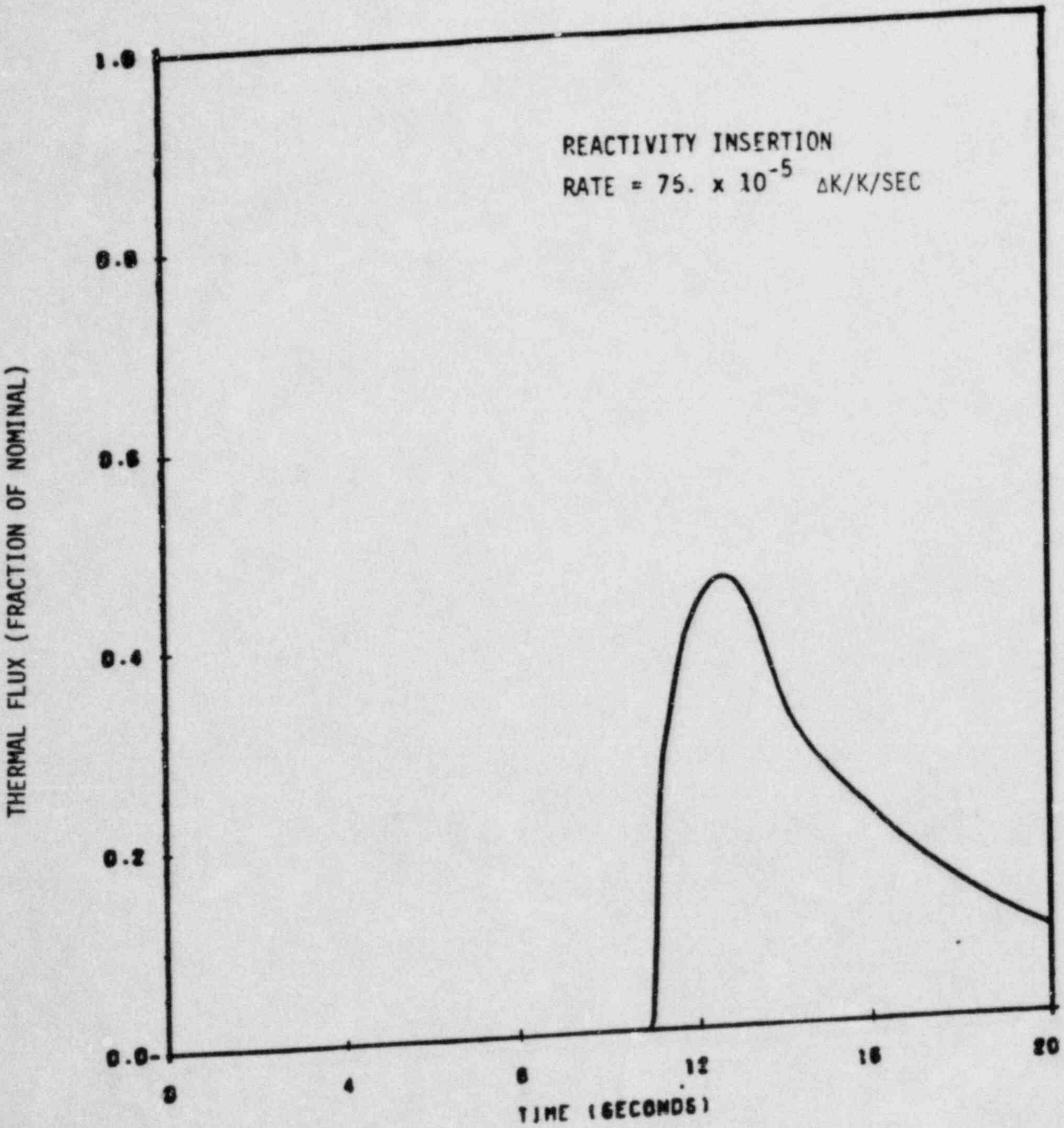


Figure 3 Uncontrolled Rod Withdrawal from a Subcritical Condition, Thermal Flux versus Time

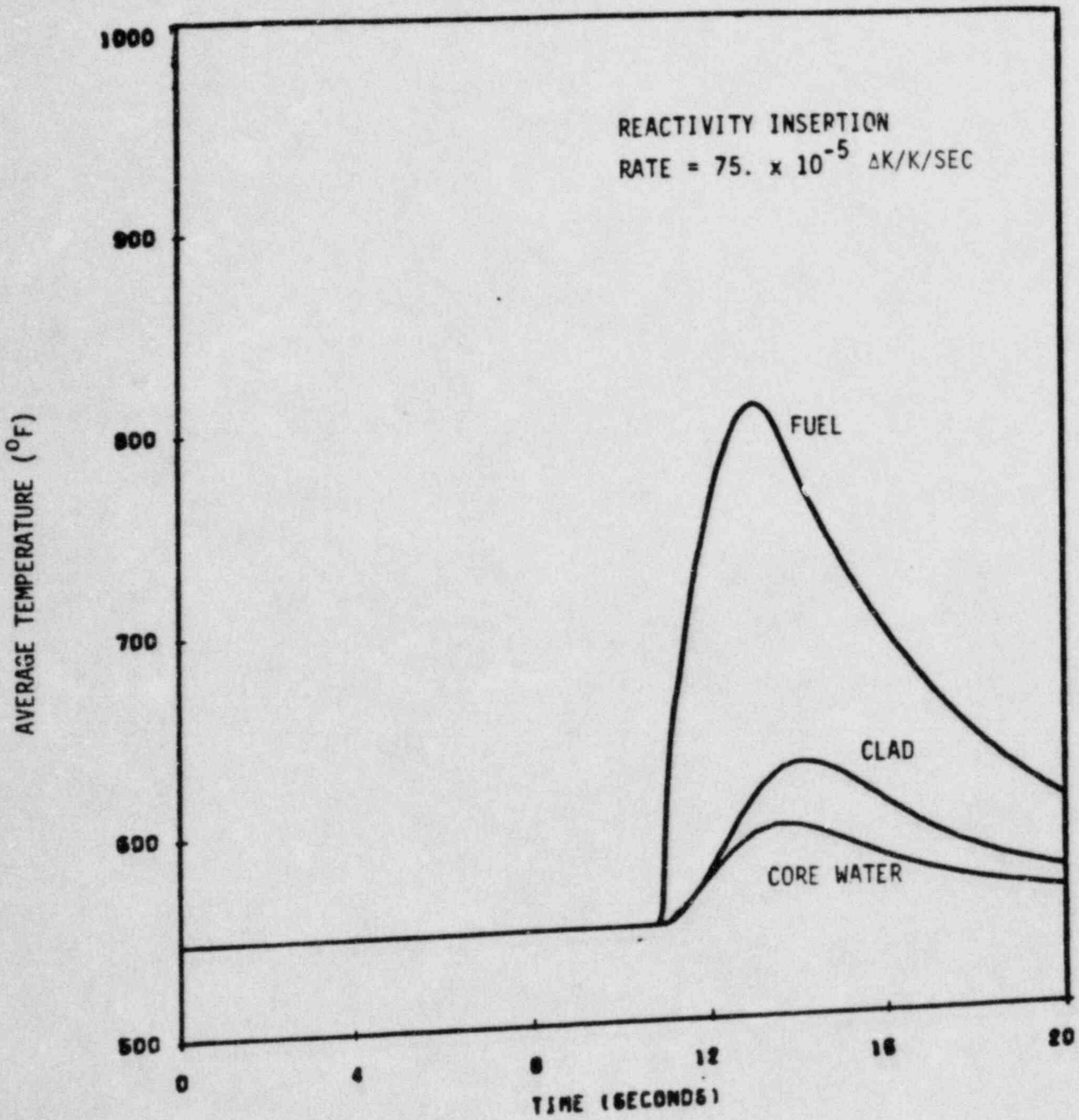


Figure 4 Uncontrolled Rod Withdrawal from a Subcritical Condtion, Temperature versus Time

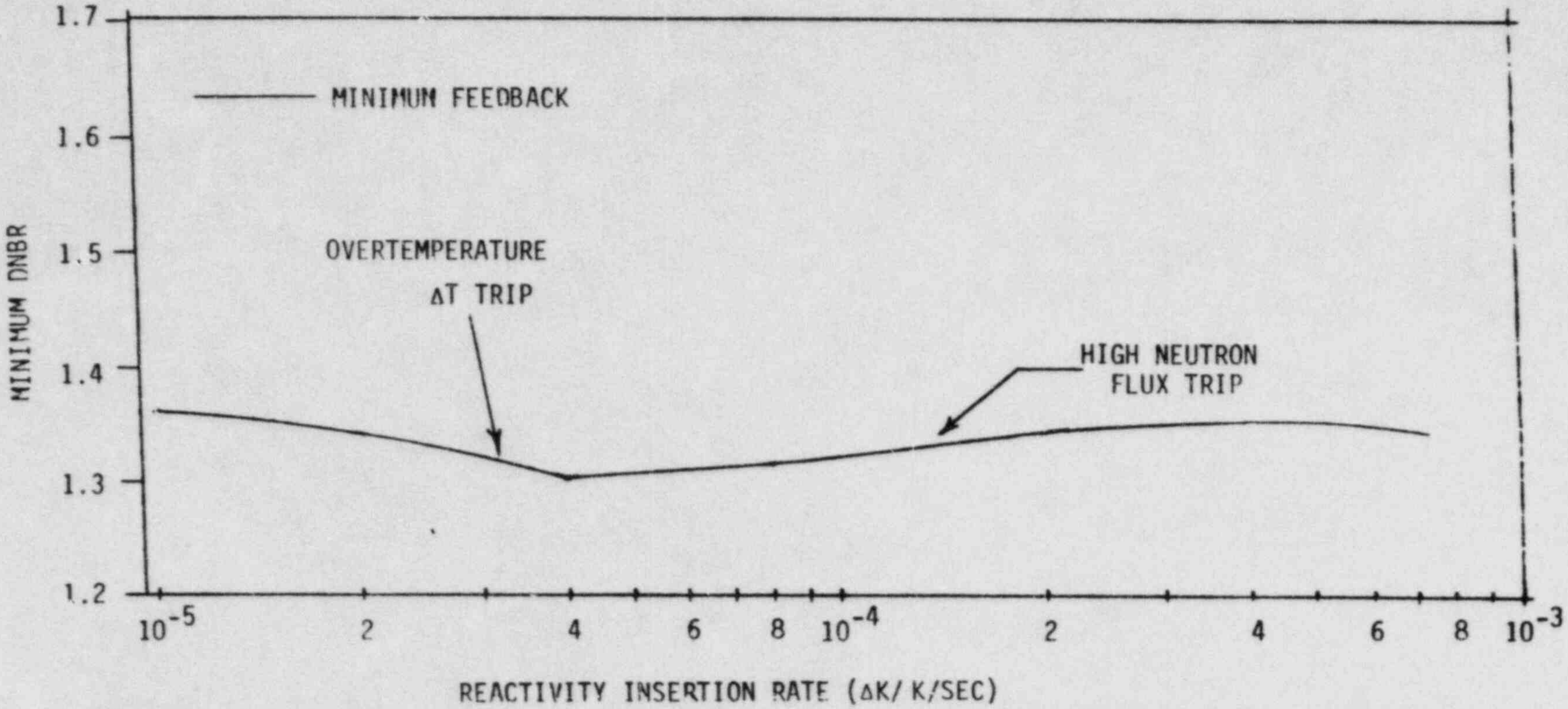


Figure 5 Effect of Reactivity Insertion Rate on Minimum DNBR for a Rod Withdrawal Accident from 100% Power

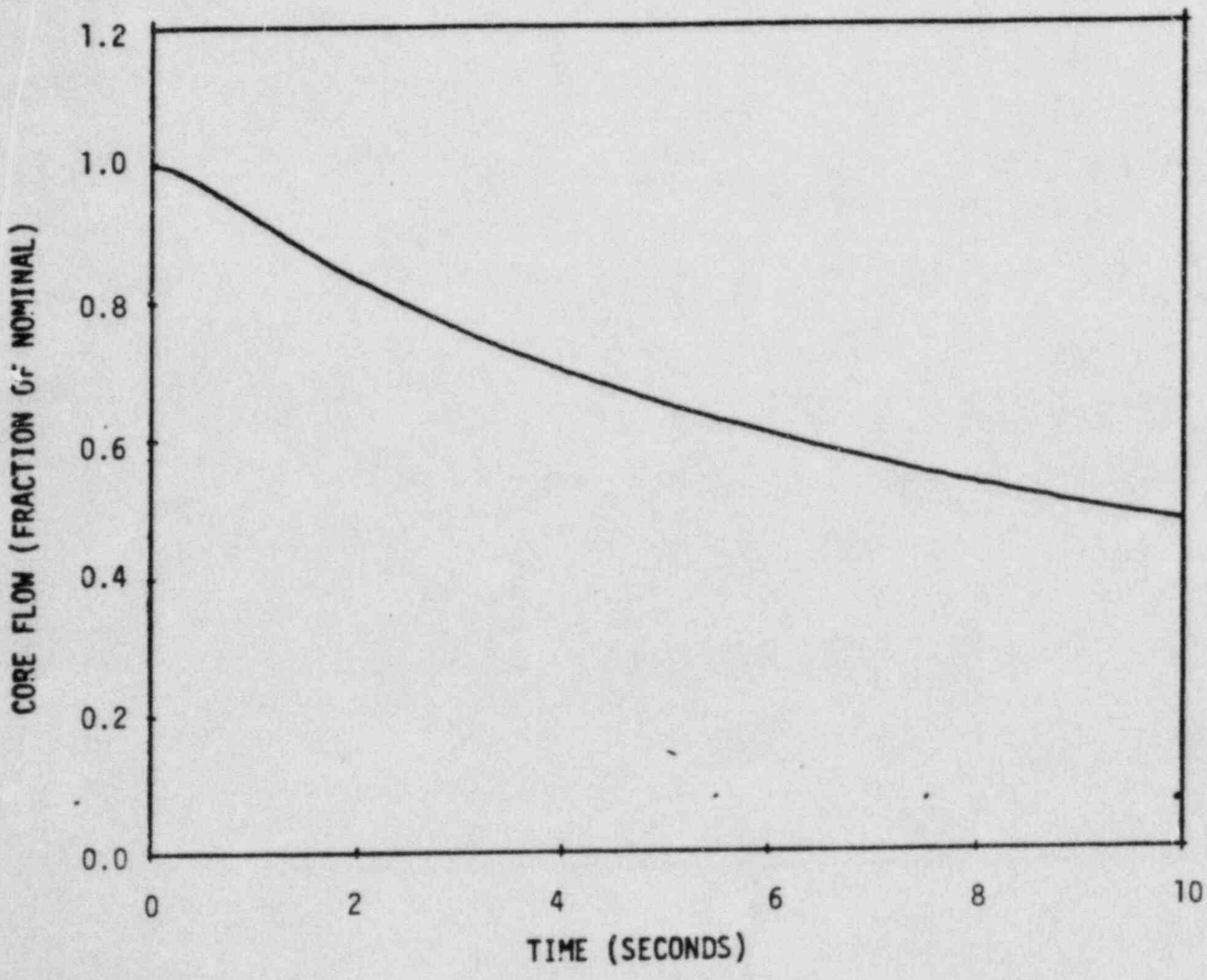


Figure 6 All Loops Operating, All Loops Coasting Down, Core Flow versus Time

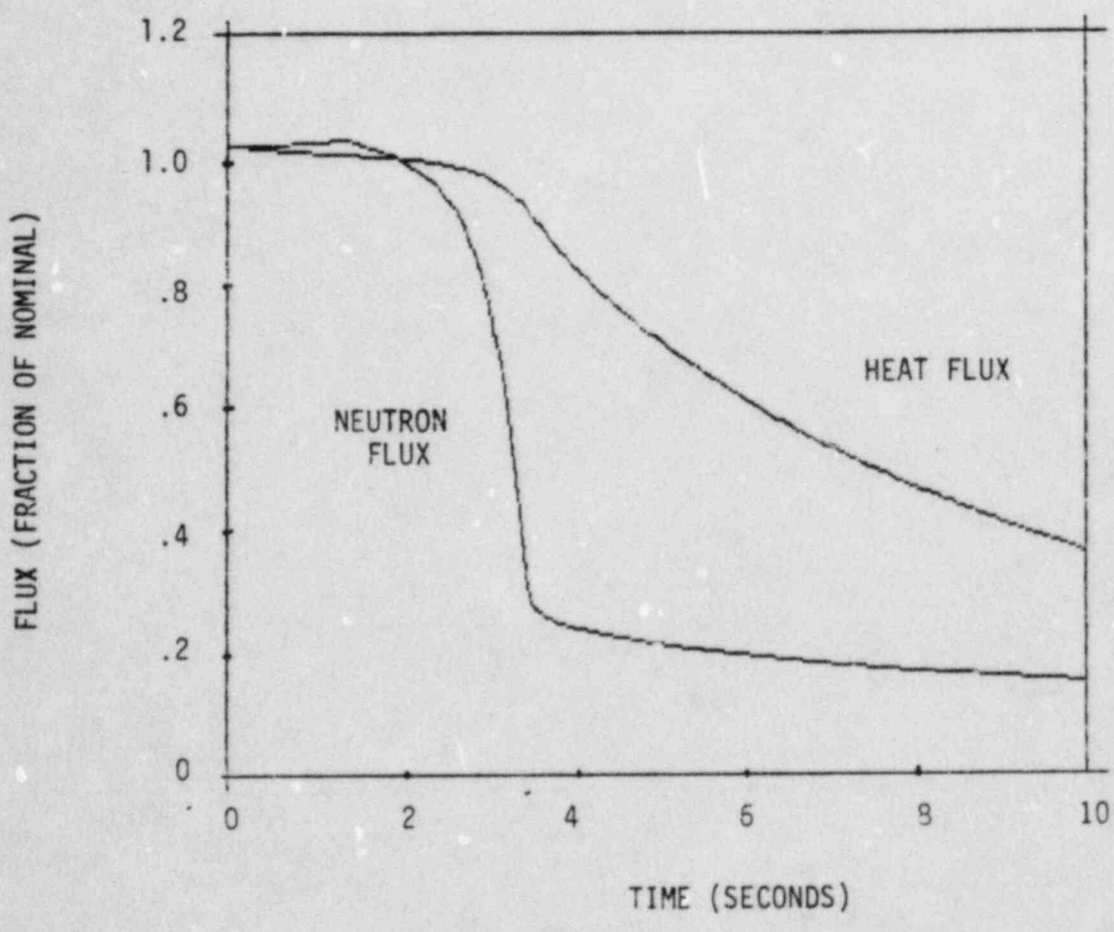


FIGURE 7 All Loops Operating, All Loops Coasting Down, Flux Transients versis Time

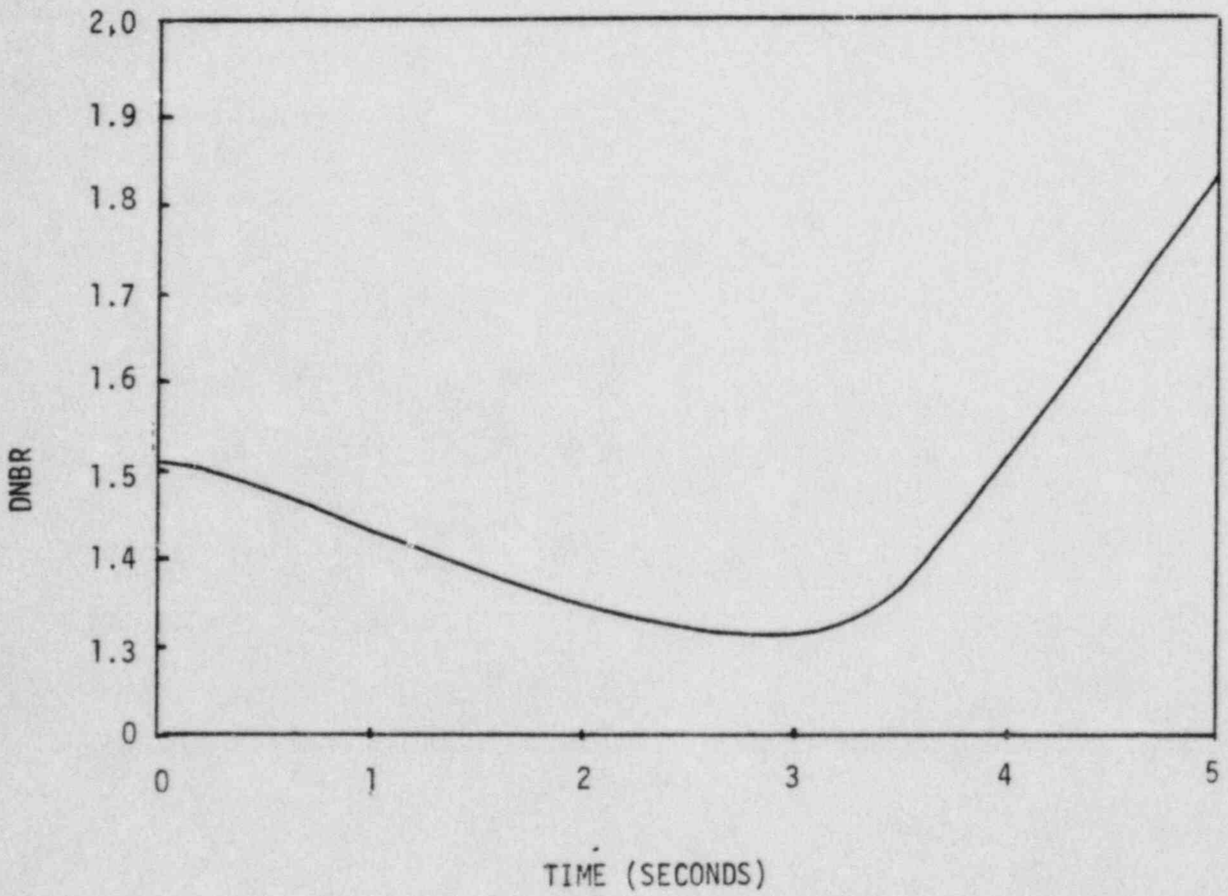


Figure 8 All Loops Operating, All Loops Coasting Down, DNBR versus Time

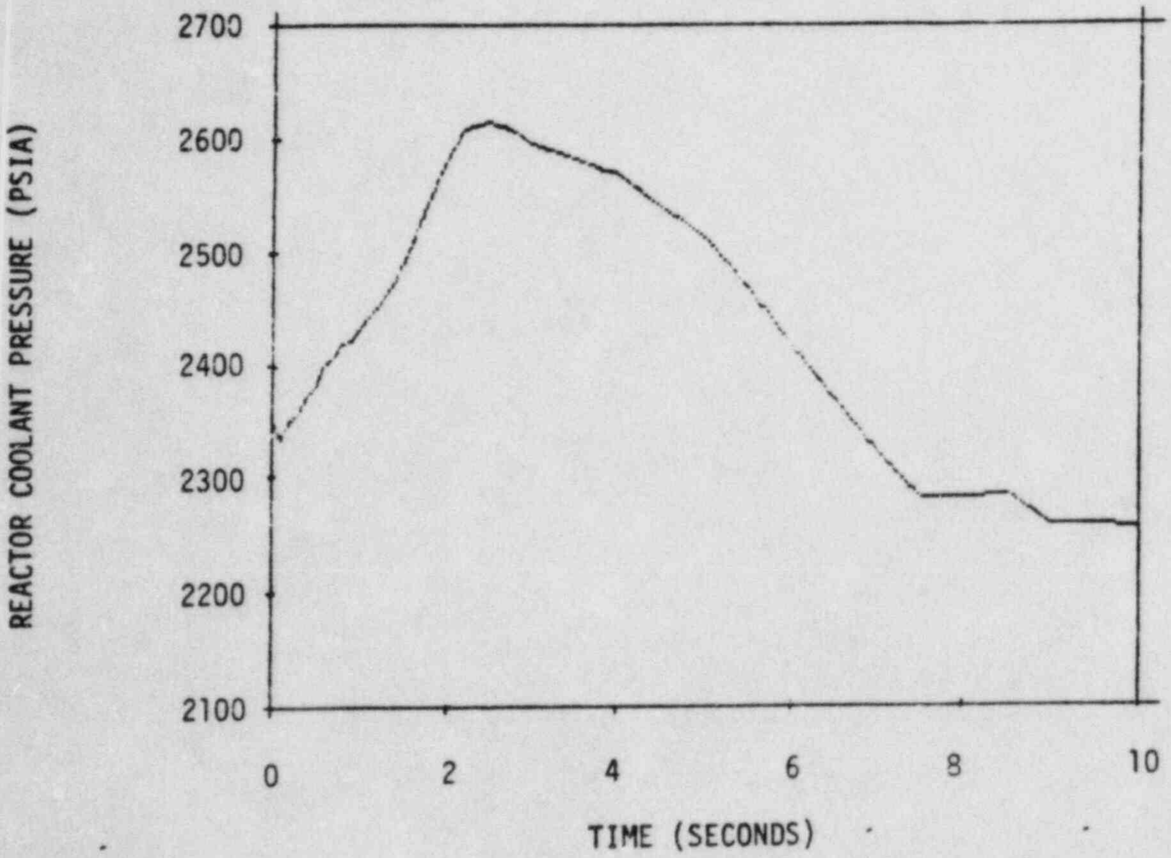


Figure 9 All Loops Operating, One Locked Rotor, Pressure versus Time

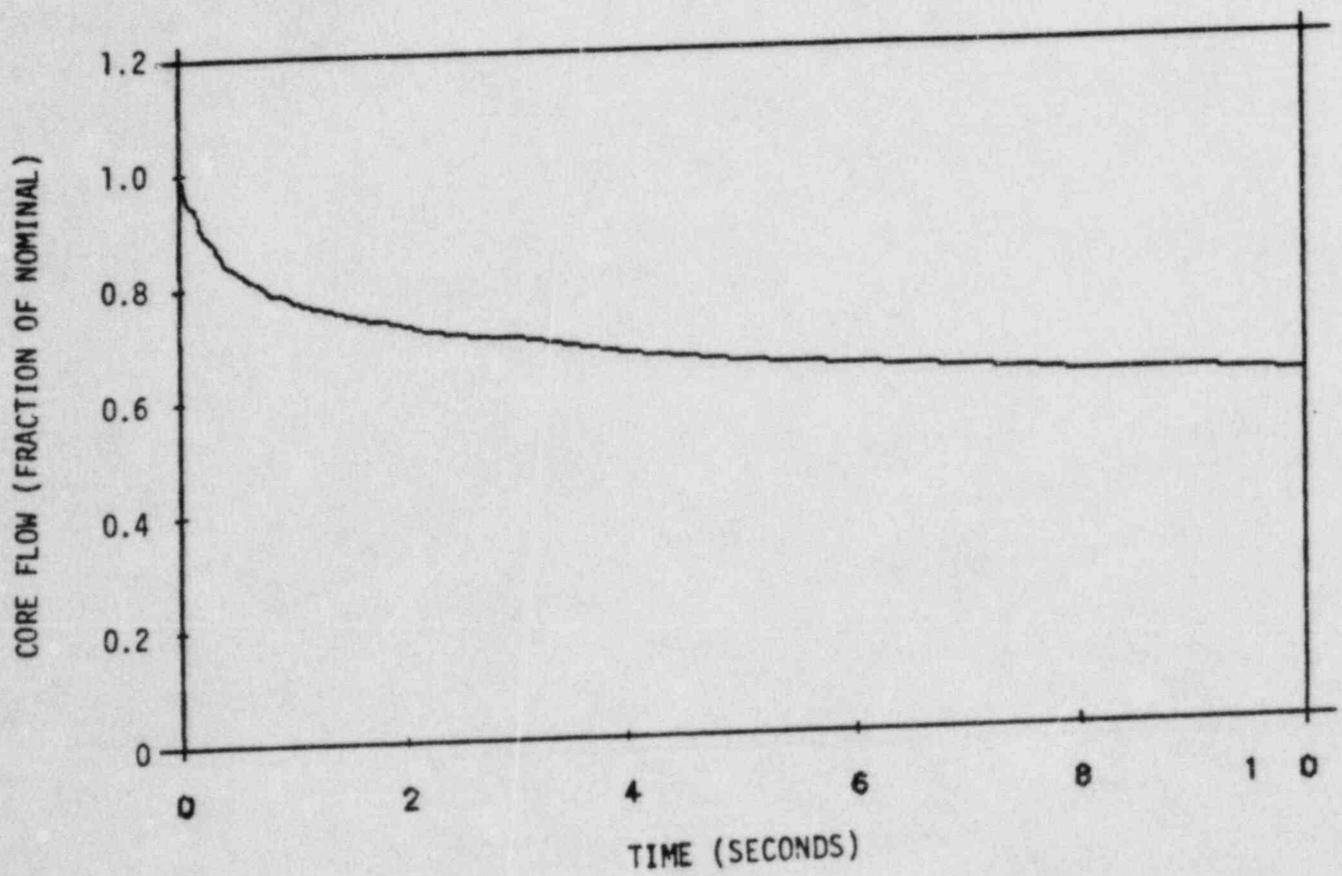


Figure 10 All Loops Operating, One Locked Rotor, Core Flow versus Time

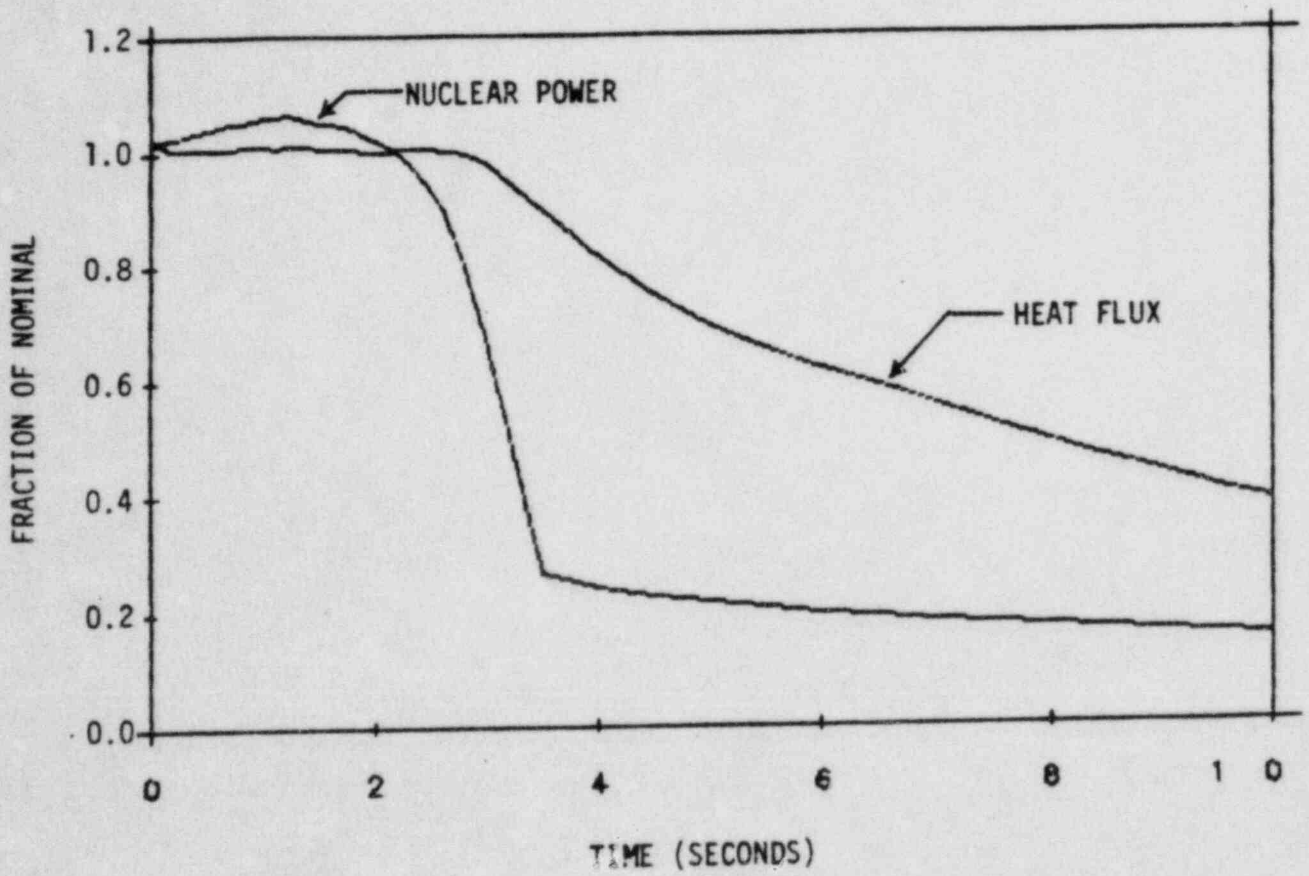


Figure 11 All Loops Operating, One Locked Rotor, Flux Transients versus Time

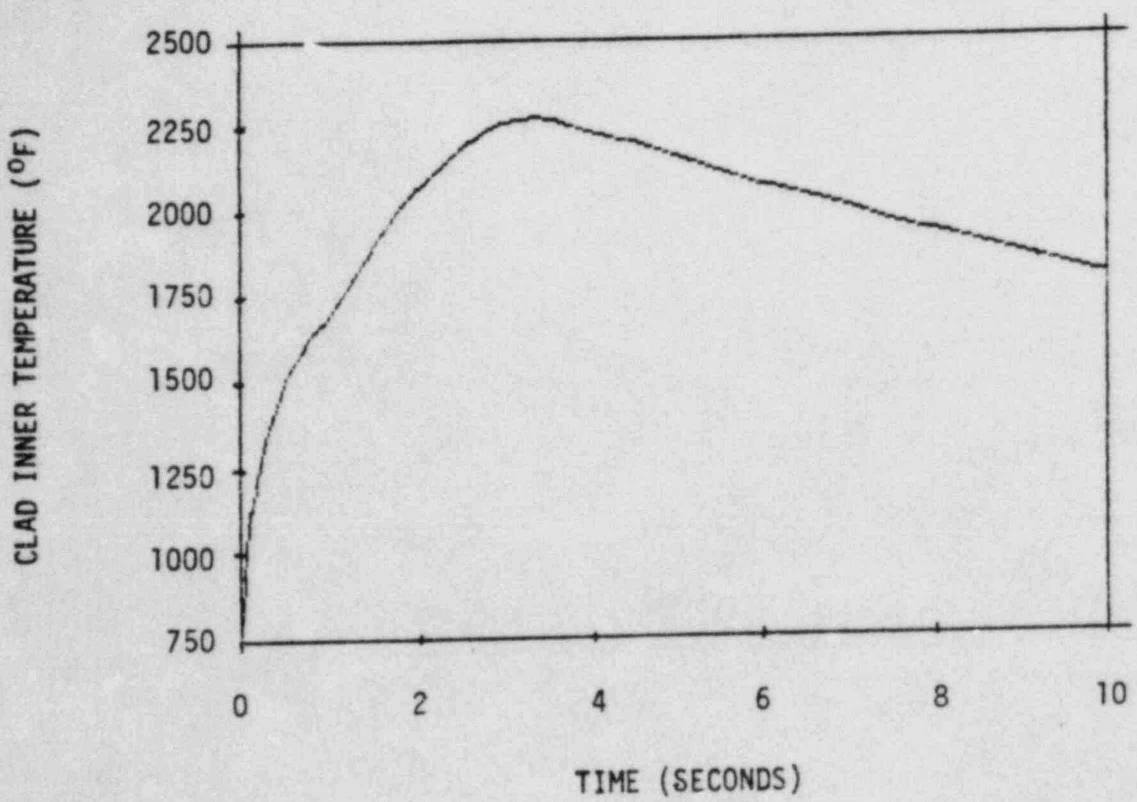


Figure 12 All Loops Operating, One Locked Rotor, Clad Temperature versus Time

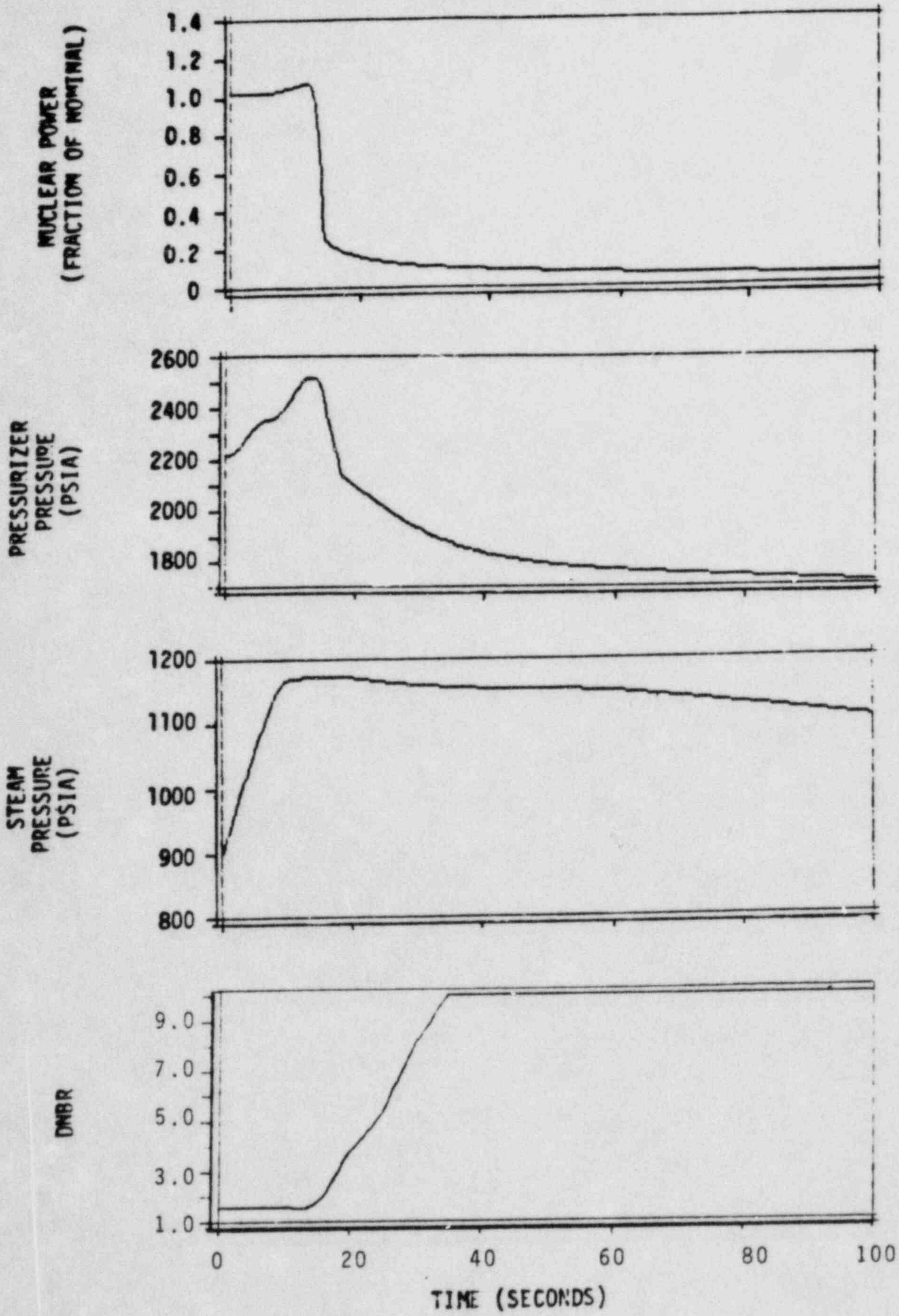


Figure 13 Loss of Load Accident, With Pressurizer Spray and Power Operated Relief Valves, Beginning of Life

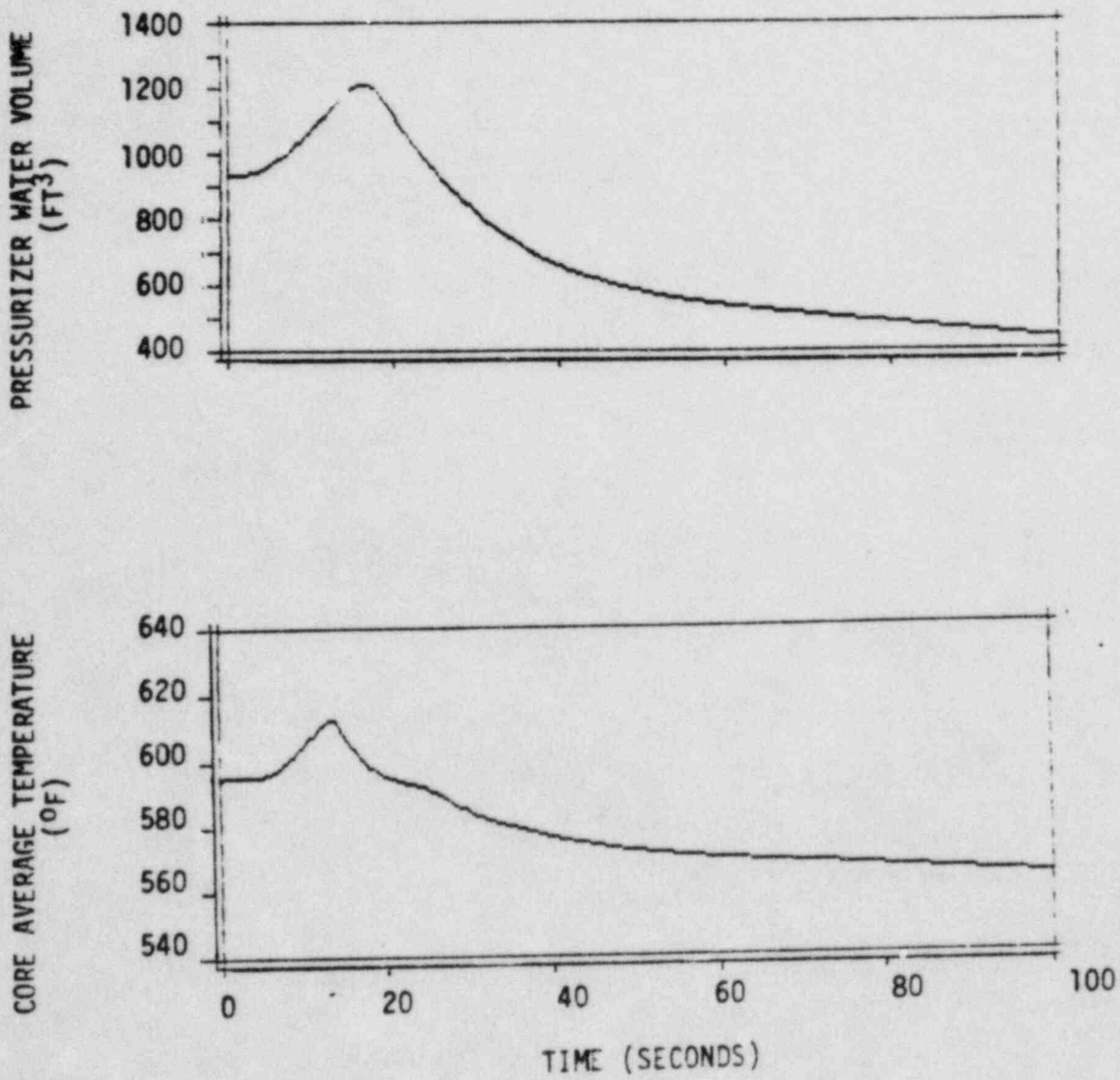


Figure 14 Loss of Load Accident, With Pressurizer Spray and Power Operated Relief Valves, Beginning of Life

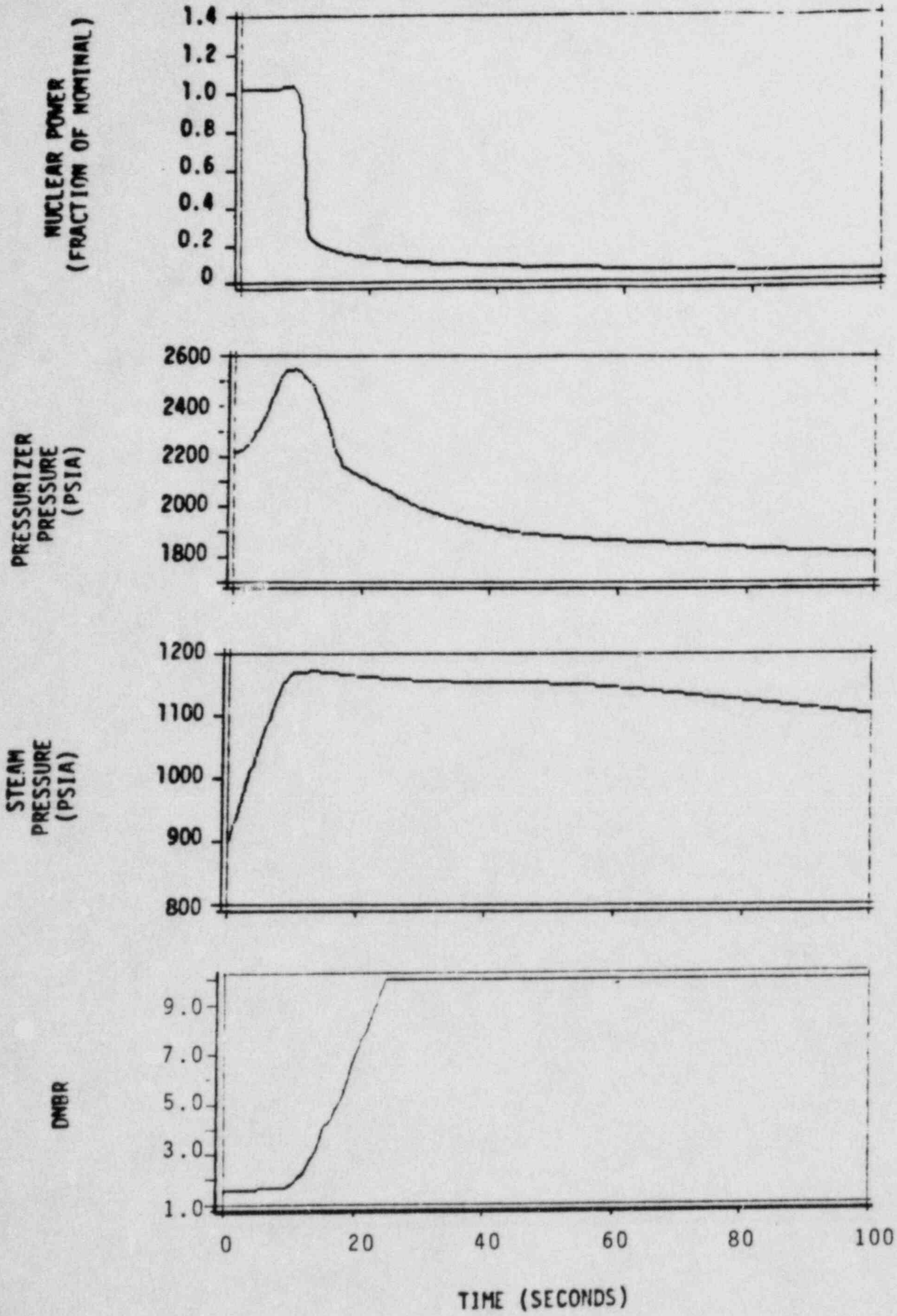


Figure 15 Loss of Load Accident, Without Pressurizer Spray and Power Relief Valves, Beginning of Life

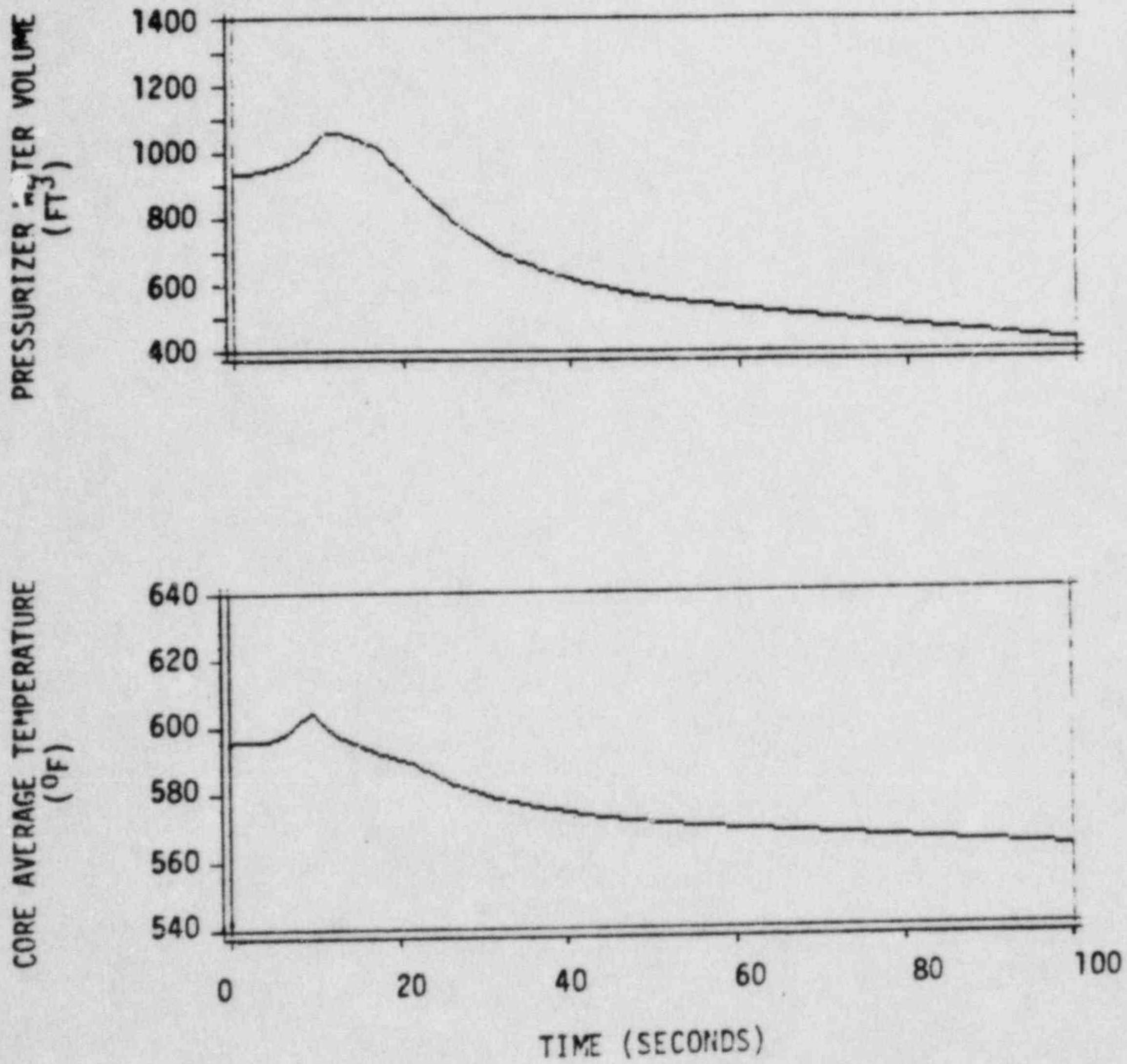


Figure 16 Loss of Load Accident, Without Pressurizer Spray and Power Operated Relief Valves, Beginning of Life

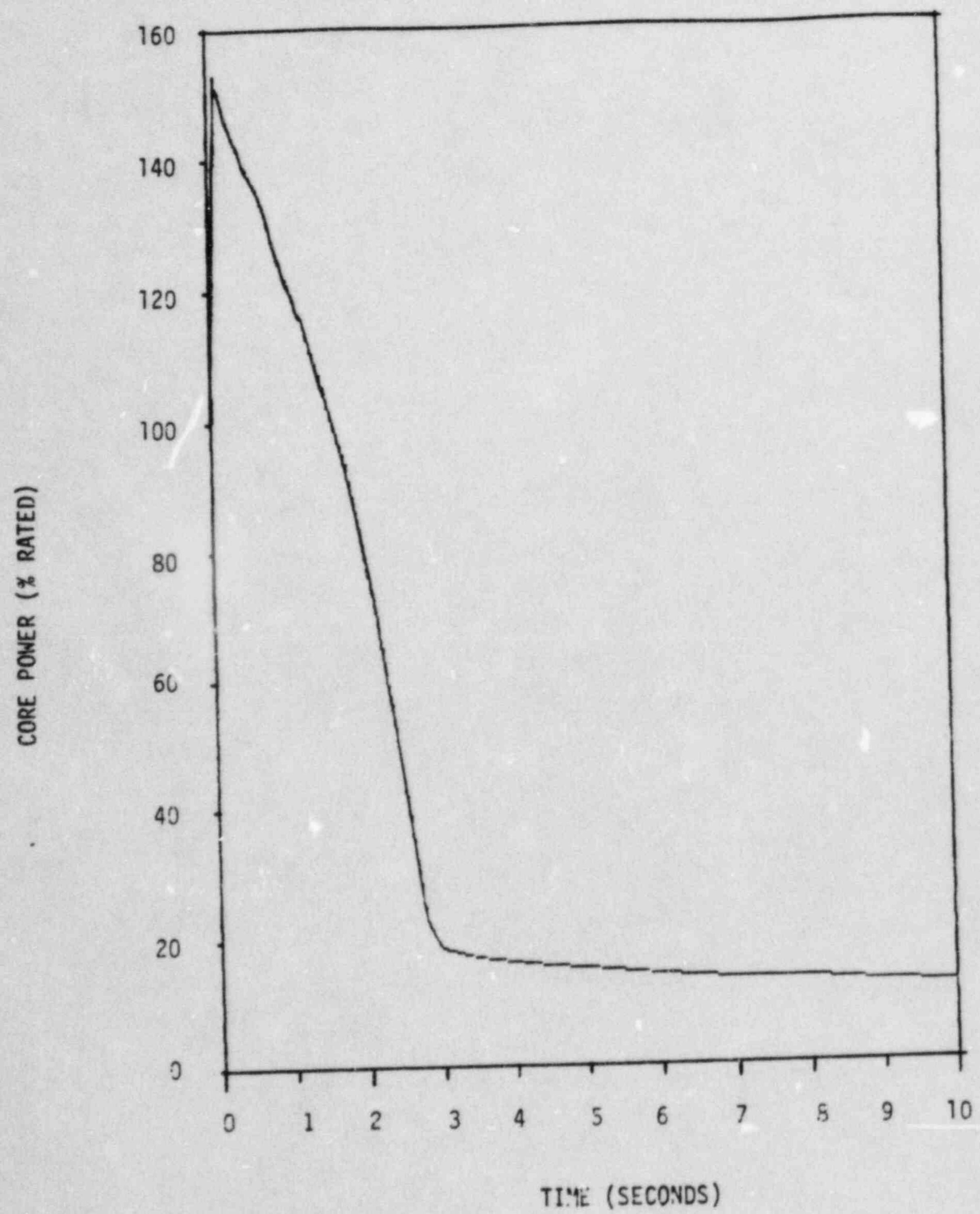


Figure 17 Nuclear Power Transient BOC HFP Rod Ejection Accident

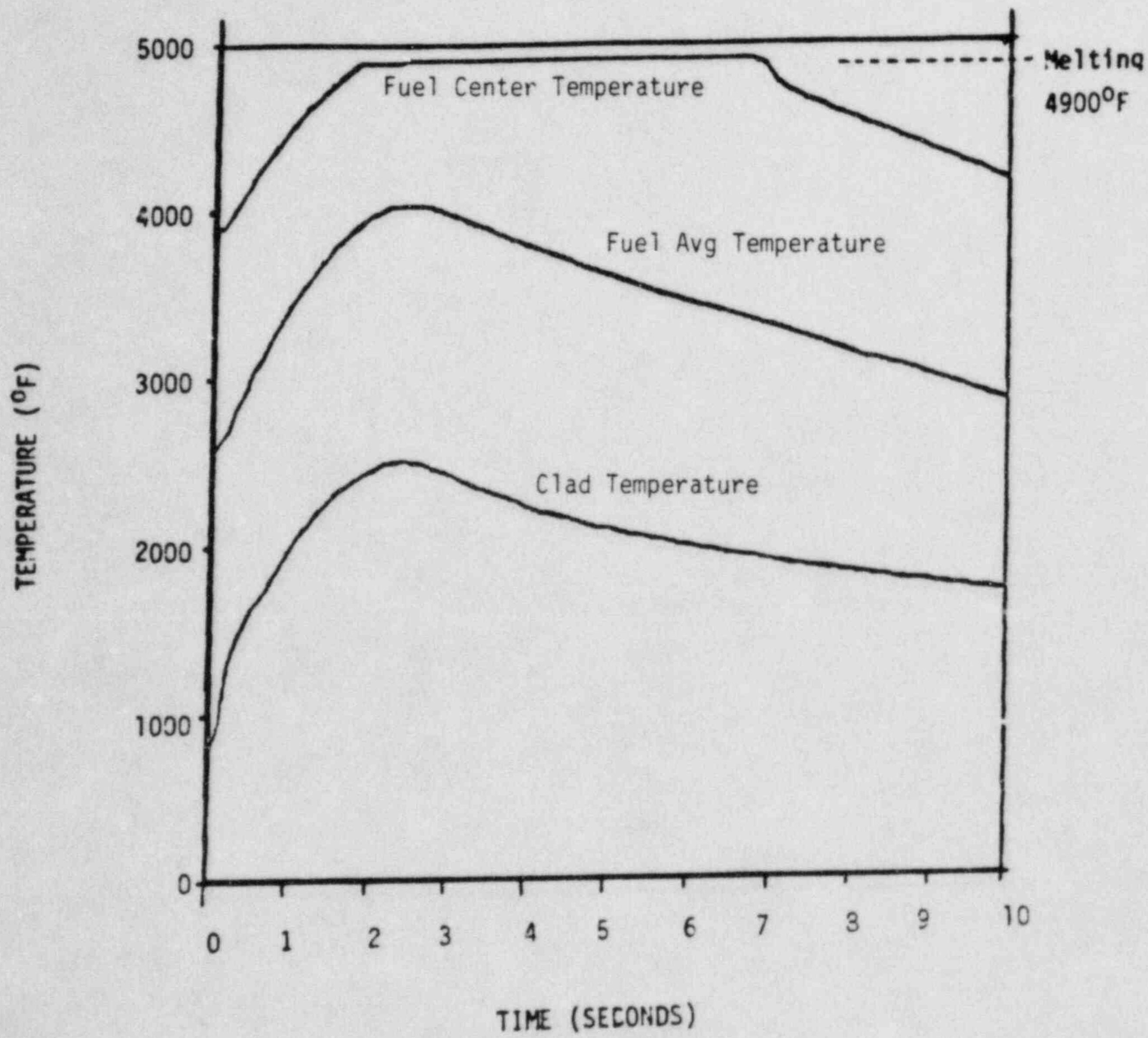


Figure 18 Hot Spot Fuel and Clad Temperatures versus Time
BOC HFP Rod Ejection Accident

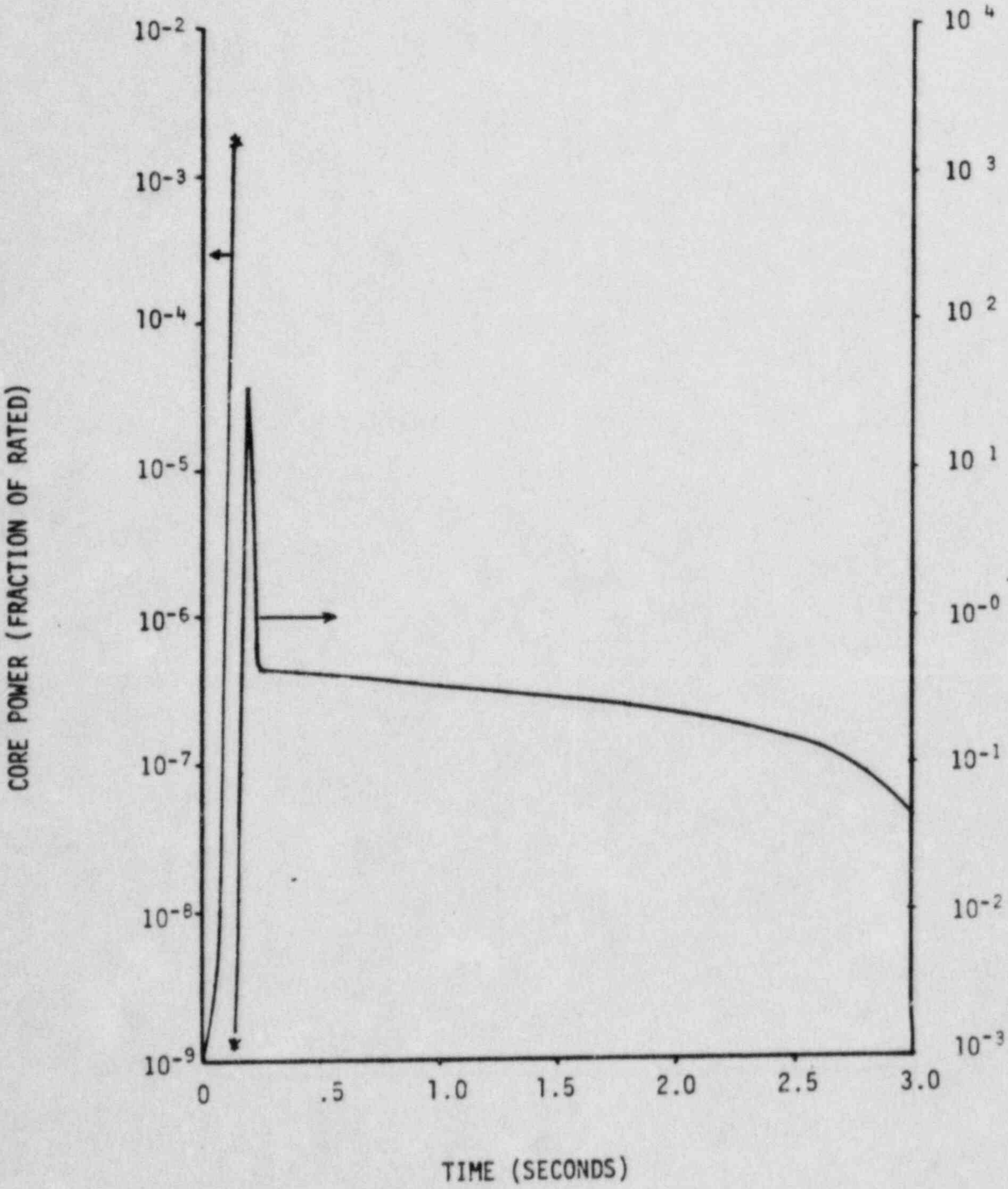


Figure 19 Nuclear Power Transient BOC HZP Rod Ejection Accident

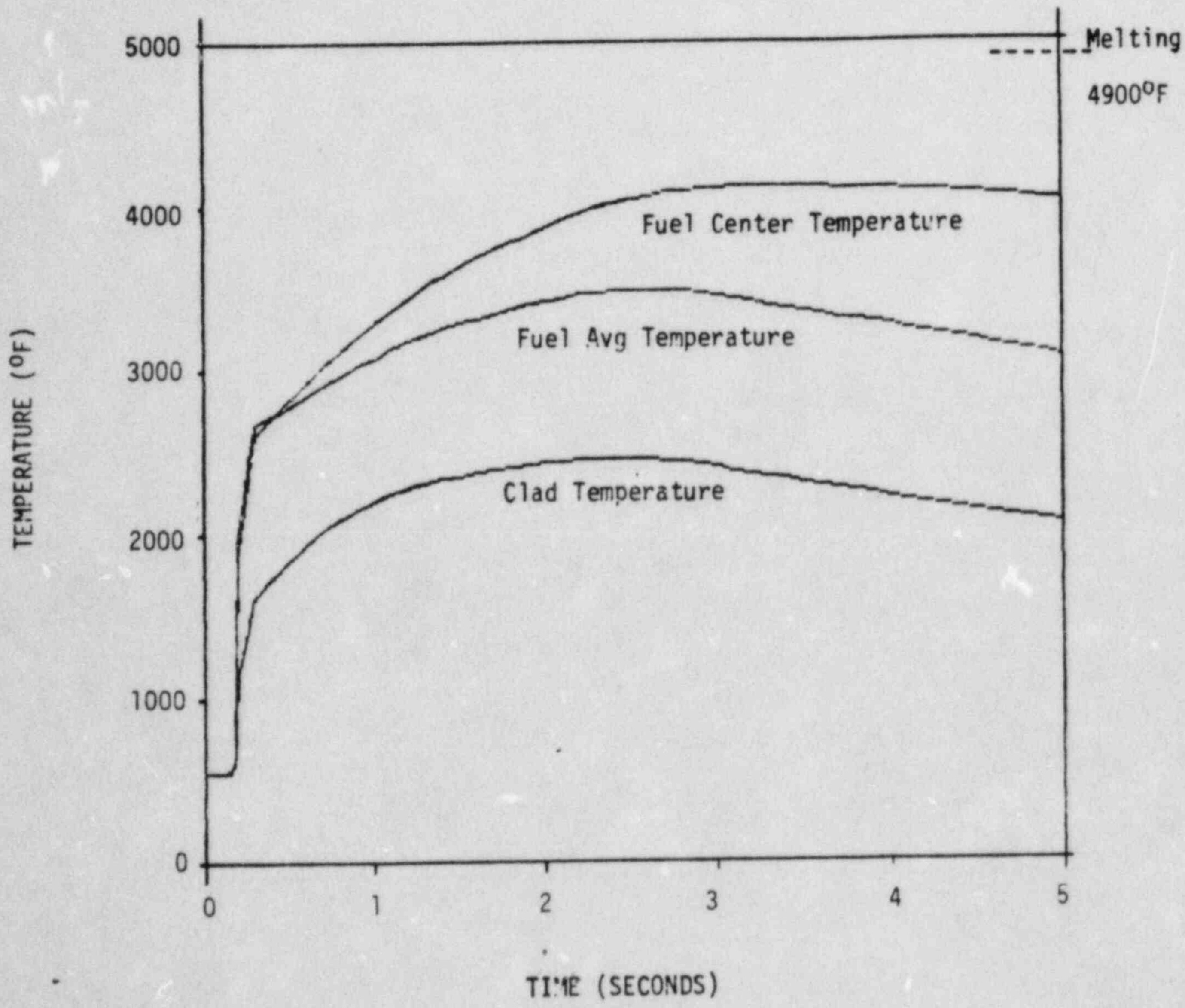


Figure 20 Hot Spot Fuel and Clad Temperature versus Time; BOC HZP Rod Ejection Accident

ENCLOSURE 2

TECHNICAL SPECIFICATIONS CHANGES FOR

A +6 PCM/^{o}F MODERATOR TEMPERATURE COEFFICIENT

NORTH ANNA POWER STATION

UNIT NOS. 1 AND 2