EVALUATION OF TRANSIENT NUCLEAR INSTRUMENTATION POWER RANGE FLUX ERROR

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1.0 INTRODUCTION

This evaluation addresses the concern that the error assumed in the FSAR transient and accident analyses for the nuclear instrumentation (NI) power range flux signal may be nonconservative for certain transient scenarios. Current analyses assume the worst case reactor power level to be 112% when the high flux trip setpoint is reached. The 112% value is based on the following components:

105.5%	- Tech Spec high flux trip setpoint
0.5%	- Setpoint error
2.0%	- Heat balance error
2.0%	- Steady state NI flux error
2.0%	- Transient NI flux error
112.0%	

The 2.0% error contribution due to transient NI flux error effects may not conservatively bound the magnitude of the effects for certain transients. If the transient NI flux error exceeds 2.0%, then the worst case reactor power would exceed 112%. For those transients causing this effect that rely on the high flux trip for mitigation, the potential for delaying or missing a high flux trip and resulting in operation at power levels above 112% raises a safety concern. Adequate safety margins to center fuel melt and DNBR limts have not been demonstrated above 112% power.

The Reactor Protective System (RPS) monitors core power utilizing nuclear instrumentation power range flux channels to measure neutron leakage flux in four locations around the core. Transient NI flux errors can be induced by two mechanisms. The first type involves those transients which result in a reduction in the coolant temperature entering the reactor vessel. When the reactor vessel downcomer temperature decreases there is an increased attenuation of the neutron leakage flux. As a result the NI channels will not accurately indicate the true core power level. The second type of transient induces a flux error by perturbing the radial power distribution in the core and thereby affecting the leakage flux. The only transient which can induce a significant flux error in this manner is the rod ejection transient.

A program was undertaken to evaluate this potential safety concern. The first task was to quantify the magnitude of the NI flux error for the two types of initiating mechanisms. The second task was to review all plant transients to determine those that would be affected by an NI flux error effect and rely on the high flux trip. The third task was to perform plant specific analyses for each affected transient to determine the impact of the flux error. The fourth task was to determine the margins to CFM and DNBR for the worst case transients.

2.0 EVALUATION OF NI FLUX ERROR FOR OVERCOOLING TRANSIENTS

2.1 Quantification of NI Flux Error From Plant Data

In order to quantify the magnitude of the flux error, an effort was made to compile any data involving a temperature change which occurred at power from

which the NI flux error could be determined. Three types of test programs were identified. The first type was the temperature coefficient of reactivity at power test, for which data are available for different units and cycles. The second test program was the Oconee 1 ΔTc tests which investigated a mismatch condition between steam generators. The third type was the thermal mixing test program performed at Oconee 1. These data are presented in Table 2.1-1. The change in the indicated NI power obtained from these data were corrected for the change in the core thermal power and the change in incore flux tilt (as measured by the incore detector system) occurring during the temperature transient and the resulting NI flux error were normalized to full power conditions. Figure 2.1-1 is the normalized NI flux error of $\Delta NI(\%FP) = 0.54 \times \Delta T(°F) - 0.3$. The dashed lines represent the 90% confidence band to the fit, characterizing a maximum temperature induced NI flux error at 95% confidence level of $\Delta NI(\%FP) = 0.54 \times \Delta T(°F) + 3.3$.

2.2 Review of Affected Transients

Transients which cause a reduction in coolant temperature and may rely on the high flux trip can be separated into three general categories. The first category involves all failures which result in a loss of secondary pressure control. These include a steam pressure regulator failure, a failure in the turbine bypass system, a failed main steam safety valve, and steam line break transients. For these transients, the result is a decrease in the coolant temperature exiting the steam generator. The rate of decrease in pressure determines the rate the temperature decreases, which in turn causes an increase in the NI flux error. Therefore, the NI flux error increases with increasing steam line break size. Also, the assumed moderator temperature coefficient is important in that it determines the rate and magnitude of any core power increase. The combinations of steam break size and moderator temperature coefficient determine whether or not the high flux trip is required, or if the transient will be terminated by the low pressure trip or the variable pressure-temperature trip. The second category is the loss of feedwater heater transient which causes a reduction in feedwater temperature. The third category is the feedwater controller failure which is an uncontrolled increase in feedwater flow. The loss of feedwater heater and feedwater controller failure transients are not severe transients but must be analyzed since the potential for developing an NI flux error exists.

2.3 Review of System Response and Performance

In order to evaluate scenarios which predict a potential for operating at power levels above 100% core power, it is necessary to consistently evaluate the performance of the other systems, in particular the secondary side of the plant, at these higher power levels. If, for instance, the main feedwater system cannot deliver the necessary flowrate to maintain operation at a higher power level, then it is not realistic to consider transients which stabilize and maintain a higher core power level. Reactor trips other than high power, control system responses, alarms, and limits which are involved in the transients must also be evaluated, including consideration of off-normal responses which adversely affect the transient. Some of the interactions which have a potential for influencing the transient response include:

A. Reactor Trips

1. High NI flux with temperature compensation - $110\% + f(\Delta T) \times \Delta T$

2. Anticipatory reactor trips on turbine and main feedwater pump trips

   a. Low main feedwater pump discharge pressure - 700 psig
   b. High steam generator level - 90%

3. Low RCS Pressure - 1806 psig

4. P-T limit - $P \leq 11.14\ T_H - 4706$

B. Alarms

1. High NI power (set by operators) - ~102%

2. Low T_{ave} - 574°F

3. Low pressure - 2055 psig

4. Low pressurizer level - 200 inches

5. High temperature 608°F

6. Low turbine header pressure - 835 psig

7. Low main feedwater pump discharge pressure - 1000 psig

8. High SG level - 82%

9. Megawatt error - -10% FP

10. ICS in track

11. Turbine in manual

12. Neutron error cross limit

13. Feedwater error cross limit

14. BTU limit

C. Limits

1. Rod withdrawal - 103% NI power

2. Turbine in auto - ΔP swing < 75 psi

3. SG level - ~85-90% proportional

4. Feedwater cross limit

5. Feedwater BTU limit

D. ICS Performance

1. Pressure Control - Normal pressure control is accomplished by the turbine control valves to maintain turbine header pressure. If a depressurization exceeds the response time of the turbine control valves and turbine header pressure decreases by 75 psi, turbine control transfers from auto to manual and the ICS goes into track. Pressure control is then only accomplished by the operator. If no operator action occurs, the turbine will follow the steam available and generated megawatts will decrease.

2. Feedwater control - With the ICS in auto, feedwater demand is controlled by megawatt demand and turbine header pressure. If the ICS goes into track, feedwater demand increases in the case of decreasing turbine header pressure, until the feedwater error cross limit is reached.

3. Reactor control - With the ICS in auto, reactor demand is controlled by the megawatt demand and T-ave. With the ICS in track, rods are withdrawn in the case of decreasing T-ave until T-ave is within the control band or NI power reaches 103%, the rod withdrawal limit.

2.4 Plant Specific Analysis

The system response for each transient scenario was simulated using the RETRAN-Ø1-MODØØ3 code developed by EPRI. The model for Oconee is illustrated in Figure 2.4-1. Steady state initial conditions are described by the following data:

Initial power level	-	2568 MWt (100% FP)
RCS pressure	-	2155 psig
Pzr level	-	220 inches
T-ave	-	579°F
RCS flow	-	106.5% of design
Moderator coefficient	-	-3.0×10^{-4} Δk/k/°F (unless specified)
Doppler coefficient	-	-1.2×10^{-5} Δk/k/°F
SG pressure	-	910 psig
SG level	-	60% operating range
Feedwater flow	-	11.0×10^{6} lb/hr

Since the transients of interest are primarily initiated and controlled by steam generator conditions, the modeling of the ICS - - pressure regulation, turbine control, feedwater control, and reactor control are included. For each transient, the expected response of the ICS and the response considering different failure assumptions are detailed. The assumptions used in the analyses were selected to enhance the NI flux error effect. An assumed bounding value of 1% FP/°F for the temperature induced NI flux error was initially utilized to identify the maximum power level reached without a reactor trip on high flux during the transient.

2.4.1 Failure of a Main Steam Safety Valve

The transient is initiated by the failure of a main steam safety valve which constitutes a steam leak of approximately 7% main steam flow. This causes a drop in steam generator pressure which initiates turbine control valve closure to maintain turbine header pressure at the setpoint. The depressurization is not large enough to cause the turbine to transfer from auto to manual. Megawatts generated drops by 7% which initiates a demand for increased feedwater flow and reactor power. Since there is a demand for increased reactor power, rods are withdrawn until inhibited by the 103% limit. Feedwater flow increases until it is restricted by the feedwater cross limit to 5% greater than reactor power. Since reactor power is limited to 103%, feedwater being delivered at a rate greater than 103% causes an excess heat sink, and T-cold begins to decrease to a maximum ΔTc of -2.5°F. In response to the temperature reduction, reactor power increases due to the negative moderator coefficient, and reaches a maximum thermal power of 107.4%. At the 107.4% power level, the megawatt error signal no longer exists since the turbine has recovered the lost megawatts, and the system stabilizes at the new conditions. The alarms which would be actuated during this transient to alert the operators would be: 1) high NI flux 2) neutron error cross limit 3) feedwater error cross limit. A neutron flux error of 2.5% occurs, but since reactor power does not exceed 112%, there is no unanalyzed potential for fuel damage.

The transient response can be affected by considering a number of failures within the ICS. If these failures are assumed, the response to the failure of a main steam safety valve would be bounded by the response to a larger steam leak. For this reason, discussion of failure modes and the resulting transient response are given in the analysis of the turbine bypass failure.

2.4.2 Failure of the Turbine Bypass System

The transient is initiated by all turbine bypass valves failing open, resulting in a steam leak of 23.4% main steam flow. This causes a drop in steam generator pressure and the ICS responds by throttling turbine control valves to maintain turbine header pressure. Megawatts generated decreases by 23.4%. In response to the loss of megawatts, the ICS increases feedwater flow until restricted by the cross limit, and pulls rods until inhibited by the rod withdrawal limit. The system stabilizes at a new steady state. Megawatts generated recovers to 86% of initial capacity. Due to the increase in feedwater flow, T-cold decreases by -4.5% °F, resulting in an NI flux error of 4.5%. Reactor power increases to a maximum of 110.3%. The alarms which would be actuated during the transient to alert the operators would be: 1) high NI flux 2) megawatt error 3) neutron error cross limit 4) feedwater error cross limit. This scenario represents the expected response of the system to the failure of the turbine bypass system, assuming a 1% FP/°F flux error. The transient has also been analyzed considering failures within the ICS which have an adverse impact on the magnitude of the flux error and the maximum power level.

The transient has been reanalyzed assuming the failure of the ICS to throttle turbine control valves. This could result from turbine control transferring from auto to manual with the ICS in track. Steam generator pressure decreases,

and megawatts generated drops to 88% of initial capacity. Feedwater flow increases in response to low steam generator pressure and turbine header pressure error, and reactor power increases due to the negative moderator coefficient and the decrease in T-cold. The feedwater error cross limit restricts feedwater flow to 112.4%. Low steam generator pressure and increased feedwater flow reduce T-cold by -8.7°F. Reactor power is limited to 116.0% power. The alarms which would be actuated during this transient to alert the operators would be: 1) high NI flux 2) low turbine header pressure 3) ICS in track 4) turbine in manual 5) megawatt error 6) low main feedwater pump discharge pressure 7) neutron error cross limit 8) feedwater error cross limit.

In addition to the assumption of the failure of the ICS to maintain steam generator pressure, the failure of the feedwater cross limit is also assumed. This constitutes the worst case response to the failure of the turbine bypass system. Feedwater flow increases in response to low steam generator pressure and turbine header pressure error. The ICS is assumed not to limit feedwater by the cross limit so that feedwater increases to its maximum capacity (120%) until throttled back to avoid tripping the turbine on high steam generator level. Megawatts generated decreases by 11% and then recovers to 97% of initial capacity. Low steam generator pressure and increased feedwater flow decreases T-cold by -10.9°F, and reactor power increases due to the negative moderator coefficient to a maximum of 120.0%, Figure 2.4.2-1. The alarms which would be actuated during this transient to alert the operators would be: 1) high NI flux 2) low turbine header pressure 3) low main feedwater pump discharge pressure 4) high SG level 5) megawatt error 6) ICS in track 7) turbine in manual 8) low T-ave 9) neutron error cross limit 10) low pressurizer level.

The transient causes an NI flux error of 10.9% to develop which results in the power level exceeding 112%. Since the thermal/hydraulic analysis is based on a maximum of 112%, it is necessary to show adequate margin to DNBR and CFM limits. In order to calculate the minimum DNBR for the transient, a thermal/hydraulic model for the hot channel was developed using the COBRA IIIC/MIT code. The model is based on the following parameters:

- 1.714 design radial pin peak
- 1.5 cosine axial shape
- 140.3 inch densified fuel stack length
- 2.7% direct moderator heating
- 1.011 average pin power hot channel factor
- 1.014 local heat flux hot channel factor
- +1°F inlet temperature error
- -45 psi pressure error
- 106.5% RCS flow
- 5% flow maldistribution penalty to the hot assembly
- 2% flow area reduction factor to the hot channel

The model calculates a conservative value for the minimum DNBR, shown in Figure 2.4.2-2. The figure shows that a sufficient margin to the DNBR limit of 1.43 (design limit of 1.3 plus 10.2% margin to account for fuel rod bowing effects) is maintained during the transient.

The current CFM limit is 20.15 kw/ft, corresponding to a low burnup condition near the beginning of the fuel cycle. For this transient, the maximum linear heat rate, including the necessary conservatisms, can be calculated as follows:

$$MLHR = \overline{q'} \times P \times F_Q \times F_D \times F_q$$

where: $\overline{q'}$ = average kw/ft at 100% power

P = fractional power level

F_Q = design total peaking factor

F_D = densification power spike factor

F_q = hot channel power factor

$$MLHR = 19.05 \frac{kw}{ft} = 5.80 \times 1.20 \times 2.571 \times 1.05 \times 1.014$$

The worst case maximum linear heat rate is less than the 20.15 kw/ft limit. Since the transient is analyzed using an end of cycle core condition when the linear heat rate limit is greater than 20.15 kw/ft, there is additional margin to be credited if required for this case.

This analysis demonstrates that a steam leak equivalent to the complete failure of the turbine bypass system, including adverse failures within the ICS, and including a conservative NI flux error effect, does not result in exceeding the fuel failure criteria.

The transient response of this event based on the temperature induced flux error of $\Delta NI(\%FP) = 0.54 \times \Delta T(°F) + 3.3$ results in a reactor trip at 130 seconds when the reactor power reaches 118.7% FP with a downcomer temperature reduction of -10.1 °F and an NI flux error of 8.8% FP, Figure 2.4.2-3.

2.4.3 Steam Line Break

The steam line break transient includes the complete spectrum of break sizes with the response of the system strongly influenced by the break size and the moderator coefficient. For the purpose of evaluating the NI flux error effect, small breaks up to approximately 23.4% main steam flow are covered by the preceeding analysis of the turbine bypass system failure. Larger breaks are characterized initially by two dominating effects. Initiation of steam generator blowdown causes a rapid increase in the heat removal rate such that RCS pressure and cold leg temperature decrease. The rate of decrease is a function of break size. When the cooler water enters the core, reactor power increases due to the negative moderator coefficient. The course the transient takes is, therefore, determined initially by these two effects. The decrease in reactor vessel downcomer temperature can cause an NI flux error. The reactor trips which are required for termination of steam line breaks are high flux, pressure-temperature, and low pressure.

Very large breaks result in a rapid depressurization of the RCS and reactor trip occurs on low pressure or pressure-temperature before the cooler water enters the reactor vessel. As the break size decreases, the rate of depres-

surization decreases and there is enough time for the cooler water to reach the core and initiate an increase in core power. Depending on the break size and the moderator coefficient, the power increase may be fast enough to limit the RCS depressurization and prevent a reactor trip on pressure-temperature. For these combinations of break size and moderator coefficient, the only fast reactor trip available for terminating the transient is the high NI flux trip. Since the cooler water entering the downcomer will cause the indicated NI flux signal to decrease, the indicated NI power can include a significant error in relation to real core power level. The relationship of importance is the rate and magnitude of core power increase determined by the moderator coefficient, compared to the magnitude of the reduction in indicated NI power as a result of the NI flux error. As the moderator coefficient becomes less negative, the power increase is slower and the flux error will prevent a high flux trip. However, the reduced rate of power increase will not turn the RCS depressurization around as rapidly, so that the transient is mitigated by the pressure-temperature trip. A spectrum of break sizes and moderator coefficients were analyzed to determine the case resulting in the highest power level without tripping the reactor on either pressure-temperature or high flux including the conservative assumption of a 1.0%FP/°F NI flux error. A summary of the cases analyzed is presented in Table 2.4.3-1. The transient responses for two scenarios are presented. The first case bounds those steam line breaks which result in the highest power level without tripping the reactor on high flux. The second case bounds those breaks that result in the highest power level without tripping the reactor on pressure-temperature. As discussed in the analysis of the turbine bypass system failure, the assumptions regarding control system response and boundary conditions on the secondary system are important, in particular for those steam line breaks which do not result in a fast reactor trip. Feedwater control is assumed to respond in the manner most aggravating to the transient. Turbine control and reactor control are similarly assumed to respond adversely.

2.4.3.1 1.0 ft^2 Steam Line Break with -1.5×10^{-4} $\Delta k/k/°F$ Moderator Temperature Coefficient

This case represents a 57% main steam flow leak with a moderator temperature coefficient corresponding to middle of cycle conditions. The power transient shown in Figure 2.4.3.1-1 shows the NI indicated power to increase to 109.6% at 10.4 seconds, narrowly missing the NI flux error compensated high power trip at 110%. A larger break size with the assumed temperature coefficient would result in a rapid trip, similarly the same break size with a more negative moderator coefficient. A smaller break size with a more negative moderator coefficient would result in a smaller flux error and a lower peak power level. The same break size with a less negative moderator coefficient is analyzed in Section 2.4.3.2.

The decrease in steam generator pressure and increased feedwater flow results in a T-cold reduction of -22.4°F, a maximum NI flux error of 22.4%, a peak power of 127.5%, and a sustained power level of 125%. The alarms which would be actuated during this transient to alert the operators would be: 1) high NI flux; 2) low turbine header pressure; 3) low main feedwater pump discharge pressure; 4) megawatt error; 5) ICS in track; 6) turbine in manual; 7) low T-ave; 8) neutron error cross limit; 9) low pressurizer level.

Since the power level exceeds 112%, a thermal/hydraulic analysis was performed to demonstrate adequate margin to CFM and DNBR limits. The transient DNBR response is shown in Figure 2.4.3.1-2. The minimum DNBR does not exceed the 1.43 limit. Using the same method as Section 2.4.2, the worst case maximum linear heat rate is 20.24 kw/ft. This value appears to exceed the 20.15 kw/ft limit; however, the 20.15 limit is based on beginning of cycle conditions. The CFM limit characteristically increases with burnup as shown in Figure 2.4.3.1-3, such that the limit consistent with the assumed middle of cycle moderator coefficient is substantially greater than the predicted 20.24 kw/ft MLHR. This transient does not exceed the CFM limit.

The transient response of the event based on the temperature induced flux error of $\Delta NI(\%FP) = 0.54 \times \Delta T(°F) + 3.3$ results in a reactor trip at 9.2 seconds when the reactor power reaches 120.0% FP with a downcomer temperature reduction of -11.4°F and an NI flux error of 9.4% FP, Figure 2.4.3.1-4.

2.4.3.2 1.0 ft^2 Steam Line Break with -1.0×10^{-4} $\Delta k/k/°F$ Moderator Temperature Coefficient

This case is similar to 2.4.3.1 with the exception that the moderator temperature coefficient is assumed to be -1.0×10^{-4} $\Delta k/k/°F$ corresponding to an earlier time in the cycle. The power transient shown in Figure 2.4.3.2-1 shows that the core power does not increase as rapidly with the less negative moderator coefficient, and as a result, the NI flux error prevents approaching the high power trip setpoint. The slower increase in power allows the RCS pressure to decrease such that a trip on pressure-temperature is narrowly missed. A larger break size or a less negative moderator coefficient would result in a pressure-temperature trip.

The decrease in steam generator pressure and increased feedwater flow results in a T-cold reduction of -24.2°F, a maximum NI flux error of -24.2%, a peak power of 127.2%, and a sustained power level of 125%. The alarms which would be actuated during this transient are the same as 2.4.3.1 with the addition of the low RCS pressure alarm. The minimum DNBR does not exceed the 1.43 limit as shown in Figure 2.4.3.2-2. The worst case MLHR is 20.19 kw/ft. Similar to the argument presented previously, the 20.15 limit increases with fuel burnup such that when the time in cycle corresponding to the -1.0×10^{-4} $\Delta k/k/°F$ moderator coefficient occurs, the transient will not exceed the CFM limit.

The transient response of this event based on the temperature induced flux error of $\Delta NI(\%FP) = 0.54 \times \Delta T(°F) + 3.3$ results in a reactor trip at 27.6 seconds when the reactor power reaches 125.9% FP with a downcomer temperature reduction of -22.7°F and an NI flux error of 15.5% FP, Figure 2.4.3.2-3.

2.4.4 Loss of Feedwater Heaters

The loss of feedwater heaters transient results in an increase in the heat removal capability of the steam generators and causes T-cold to decrease. This condition creates a potential for the NI flux error effect to develop. The scenario assumes a feedwater temperature reduction of 85°F as a result of bypassing feedwater heaters A and B. This transient results in an NI flux error of 6.6% and a peak power of 112%, and, therefore, does not represent a safety concern.

2.4.5 Feedwater Controller Failure

The feedwater controller failure transient assumes a ramp increase in feedwater flow to 110% flow. The ICS limits the flowrate to avoid tripping the turbine on high steam generator level. The increase in feedwater increases the heat sink and causes T-cold to decrease by 3.5°F. Due to the negative moderator coefficient, reactor power increases to 106.5% power. The transient is then limited by high steam generator level. Since the power level does not exceed 112%, there is no safety concern.

2.5 Summary and Conclusions for Cooldown Treatments

The analyses presented have simulated the system response for all transients capable of inducing the NI flux error effect due to a reduction in the reactor vessel downcomer temperature. The most severe transient responses occurred for intermediate size steam line breaks. Comprehensive sensitivity studies were undertaken to characterize the effects of interest. The assumptions utilized in the simulations were selected to enhance the NI flux error and maximize the core power response. The modeling of the ICS included the most adverse turbine control, feedwater control, and reactor control. No credit was taken for operator action. The evaluations of center fuel melt and DNBR limits included appropriate conservatism and resulted in the fuel failure criteria not being exceeded.

It must be recognized that the analyses presented herein are worst case, as the result of the combination of many conservative assumptions. The simultaneous occurrence of all the assumptions is a very low probability event. The impact of some of the individual assumptions are as follows:

1) 1%/°F NI flux error - The plant data given in Figure 2.2-1, illustrates the conservatism of this assumption. The upper dashed line statistically predicts the value of the flux error to be less than that given by the relationship, $\Delta NI(\%) = 0.54\ \Delta T + 3.3$ at a 95% confidence level. With this value the reduction in NI indicated power is much less, and all limiting transients which did not previously trip on high flux would result in a trip.

2) 110% + ΔNI high power trip - All components of the high core power trip error are at maximum value. This maximizes the mismatch between indicated NI power and real core power.

3) Failure of the ICS feedwater cross limit - The feedwater cross limit normally limits the feedwater to within ±5% of NI power. If functional, feedwater would be throttled and the transient response would be less severe.

4) Failure of turbine header pressure control - Turbine control valves respond to a decrease in steam generator pressure by throttling down on steam flow to the turbine. If functional, the depressurization would be reduced and the transient response would be less severe.

5) No operator action - The operators would be alerted to the occurrence of the transient by several alarms which clearly identify the nature of the transient. Station emergency procedures (EP/O/A/1800/08) specify the operator actions which would be promptly undertaken. The required actions are manual reactor trip if not automatic, and isolation of feedwater to the affected steam generator. These actions would mitigate any contribution of an NI flux error to the transient response.

Based on the analyses presented and the considerations outlined above, it can be concluded that the occurrence of an NI flux error during overcooling transients does not result in an increase in fuel failures for any of the affected transients.

3.0 EVALUATION OF NI FLUX ERROR FOR ROD EJECTION TRANSIENT

3.1 Review of Rod Ejection Transient

The initiating event in a rod ejection transient is the rupture of the RCS pressure boundary in the vicinity of a control rod drive mechanism and the resultant ejection of a control rod. This causes an increase in core power due to the positive reactivity addition, and an abnormal power distribution because of the loss of control rod pattern symmetry. The alarms which would be actuated include the high NI flux alarm and the asymmetric rod position alarm. Power peaking increases near the ejected rod location so that the relative power away from the ejected rod location decreases. Since the NI indicated power is actually a measurement of the leakage flux at the core perimeter, a rod ejection on one side of the core can occur and cause a core power increase without the NI detectors on the other side of the core indicating the real magnitude of the power increase. In this scenario, the high flux trip can be delayed or may not occur at all. The core power response is determined by the worth of the rod, the Doppler feedback which limits the excursion, and the location of the ejected rod. Reactor trip will occur when the two lowest NI flux signals exceed the worst case trip setpoint of 110%.

The rod ejection transient is mainly limited by the criterion that the peak fuel pellet enthalpy does not exceed 280 cal/gm. The peak pellet enthalpy increases with the local peaking factor which increases with the ejected rod worth. The peak power and rate of the core power excusions also increase with increasing rod worth. It is also apparent that the magnitude of the NI flux error increases with increasing rod worth in a given core location. Each NI channel will indicate a flux which is the core average flux multiplied by the average peaking factor on the perimeter of the core in the vicinity of the NI detector.

Large ejected rod worths cause a very rapid power excursion. Core power increases so rapidly that even considering an NI flux error of a large magnitude, the delay in the reactor trip will only increase by a few hundredths of a second. Figure 3.1-1 (FSAR Figure 14-29) shows that the sensitivity of a small increase in the trip delay is minimal. As the ejected rod worth decreases, it is less obvious what the potential trip delay is. An analysis has been performed to quantify the increase in the time to trip on NI high flux.

3.2 Quantification of NI Flux Error

Unlike the cooldown transients, there exists no plant data for quantifying the NI flux error which can occur as a result of a rod ejection transient, so that an analytical estimate of the effect was undertaken. To determine the magnitude of the NI flux error resulting from a rod ejection, twenty-two three dimensional EPRI-NODE rod ejection simulations from 01C6, 02C5, and 03C6 were analyzed. These results are given in Table 3.2-1 and in an example power distribution in Figure 3.2-1. These simulations provide an estimate of the worst case NI flux error as a function of ejected rod worth.

3.3 Rod Ejection Analysis

A series of rod ejection simulations have been performed using the RETRAN code to obtain the core average power response according to the point kinetics model as a function of ejected rod worth. These results are shown in Figure 3.3-1. Combining the RETRAN and EPRI-NODE results, we can determine the additional trip delay for the cases analyzed, Table 3.3-1. The ejected rod is assumed to be fully ejected in 0.15 seconds. The high flux trip occurs when the two lowest NI channels reach 110% including the flux error, in comparison with the FSAR assumption of 112%. The kinetic response indicates that the power excursion has essentially peaked when the rod is fully ejected. The Doppler feedback limits the excursion at that time. Therefore, the additional trip delay does not increase the magnitude of the excursion. The delay only increases the fuel enthalpy by the energy generated at the already established power level. Simulations for very low worth ejected rods which do not result in a reactor trip, either with or without the NI flux error, do not present a safety concern since the power level and peaking factor do not result in a core condition where fuel design limits could be exceeded.

3.4 Conclusions for Rod Ejection Transients

The existing rod ejection analysis assumes a high flux trip delay of 0.3 seconds. The analyses presented show an incremental trip delay due to the NI flux error of a few hundredths of a second for small ejected rod worths. For higher worth rods, the incremental delay will be comparable. This analysis demonstrates that the impact of the NI flux error on the rod ejection transient is minimal and well within the sensitivity studies included with the FSAR analysis.

4.0 SUMMARY AND CONCLUSIONS

This document provides a comprehensive evaluation of the continued safe operation of the Oconee Nuclear Station considering the potential for an NI flux error to develop and impact on the transient analyses documented in the FSAR. The affected transients have been identified and reanalyzed using a very conservative approach to conservatively bound the worst case transient response. It has been demonstrated that the center fuel melt and DNBR fuel performance limits have not been exceeded. Therefore, the operating limits presently specified in the Technical Specifications, as derived from the FSAR safety analysis, remain valid.

Table 2.1-1.

Plant NI Flux Error Data

Moderator Temperature Coefficient Test Data	Heat Balance Power (%FP)	NI Power (%FP)	T-ave (^{o}F)
Oconee 1 Cycle 2	98.63	98.68	574.92
	98.23	96.60	572.73
Oconee 1 Cycle 3	99.28	99.82	579.09
	98.71	96.29	575.30
Oconee 1 Cycle 6	97.31	99.85	582.13
	97.22	96.20	577.27
Oconee 2 Cycle 1	75.15	75.60	579.3
	74.33	78.23	584.6
Oconee 2 Cycle 2	99.23	99.1	581.30
	99.70	95.6	575.76
Oconee 2 Cycle 5	97.40	99.74	581.62
	97.41	95.50	576.40
Oconee 3 Cycle 1	97.58	95.79	578.54
	98.15	92.93	574.20
Oconee 3 Cycle 5	97.10	98.86	581.28
	97.16	96.26	576.53

Oconee 1 ΔT_c Test Program Data

Heat Balance	NI Power (%FP)				Cold Leg Temperatures (^{o}F)				Incore Tilt (%)			
Power (%FP)	NI-5	NI-6	NI-7	NI-8	T_cA1	T_cA2	T_cB1	T_cB2	WX	XY	YZ	ZW
73.82	75.66	75.53	75.16	75.25	565.0	565.6	560.1	560.4	-0.14	-1.20	1.27	0.09
74.36	76.06	75.81	75.34	75.56	568.4	568.5	558.7	558.7	-0.70	-2.20	2.11	0.80
74.97	74.19	74.19	74.87	74.78	561.0	561.9	561.8	561.1	0.06	-0.82	0.88	-0.12
73.77	72.94	73.12	73.97	73.75	557.4	558.0	563.4	563.4	0.69	0.20	-0.06	-0.82
75.98	72.53	72.81	73.62	73.41	553.5	554.7	564.0	564.0	1.20	0.99	-0.80	-1.39
76.24	74.87	74.97	74.44	74.50	560.8	561.5	561.5	561.5	0.69	0.09	0.09	-0.86

(continued)

Table 2.1-1.

Plant NI Flux Error Data

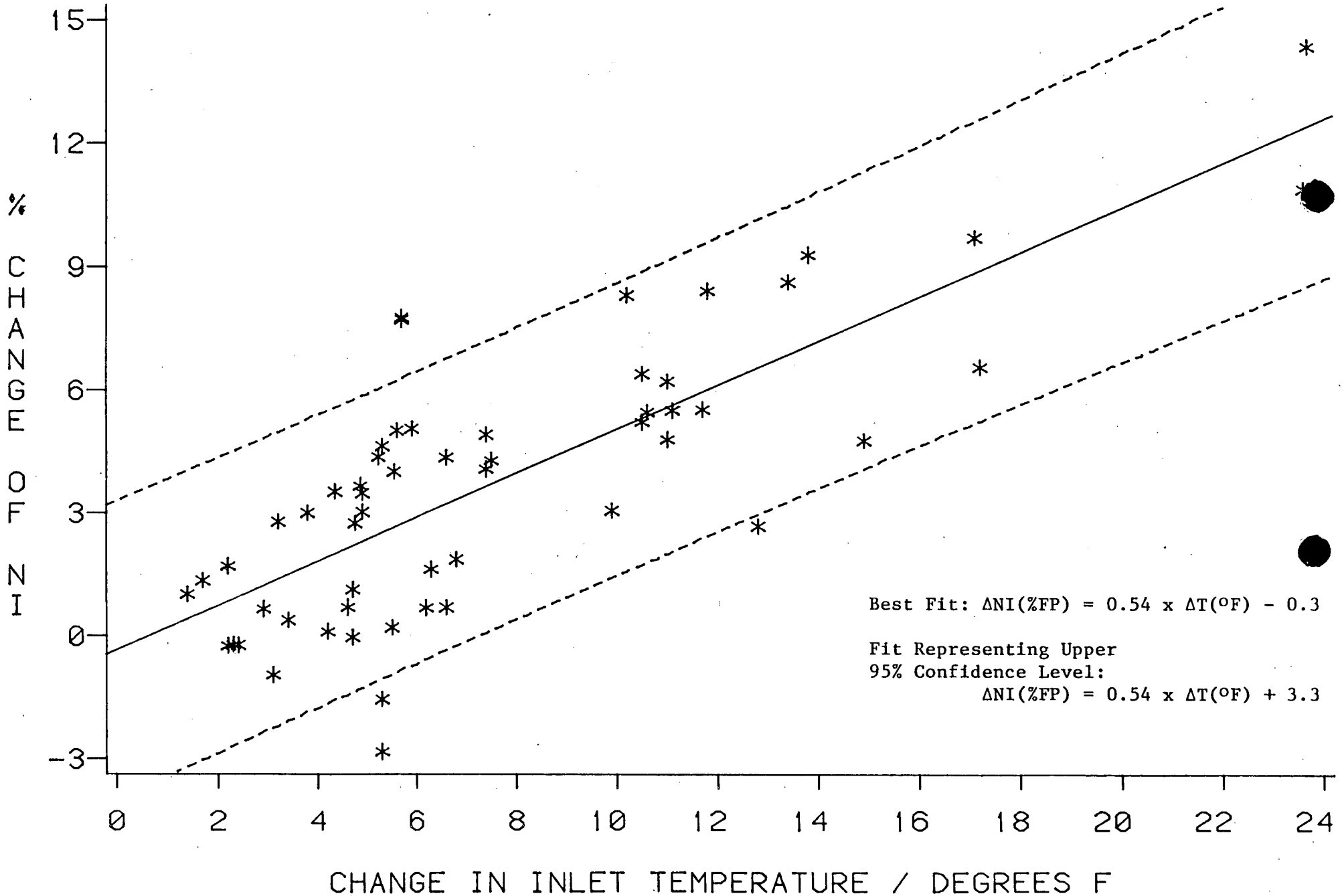
(continued)

Oconee 1 Thermal Mixing Test Program Data

Heat Balance Power (%FP)	NI Power (%FP) NI-5	NI-6	NI-7	NI-8	Cold Leg Temperatures (°F) T_cA1	T_cA2	T_cB1	T_cB2	Incore Tilt (%) WX	XY	YZ	ZW
38.99	40.3	39.1	39.1	38.5	570.9	571.0	571.0	571.0	1.35	0.32	-1.22	-0.43
38.96	40.3	39.0	38.9	38.3	573.1	573.4	567.9	567.9	0.27	-0.43	-0.47	0.63
38.95	40.2	38.8	38.7	38.2	575.4	575.4	565.4	565.3	-0.88	-1.27	0.32	1.84
38.90	39.5	38.2	37.9	37.4	577.7	577.6	560.5	560.0	-2.34	-2.45	1.41	3.39
38.86	40.3	38.8	39.0	38.4	571.4	571.4	571.5	571.5	1.46	-0.11	-1.18	-0.16
38.21	37.8	37.7	37.8	37.8	572.0	574.4	566.2	566.3	2.82	0.80	-2.12	-1.50
39.17	39.1	38.9	39.0	39.0	567.5	567.7	577.6	577.6	4.26	1.71	-3.05	-2.92
39.18	39.9	39.7	40.1	39.9	564.3	564.8	584.2	584.0	6.59	3.35	-4.57	-5.36
38.75	39.3	39.2	38.6	38.7	572.1	572.1	572.3	572.1	2.54	0.71	-1.94	-1.31

Figure 2.1-1.

OCONEE NI FLUX ERROR DATA



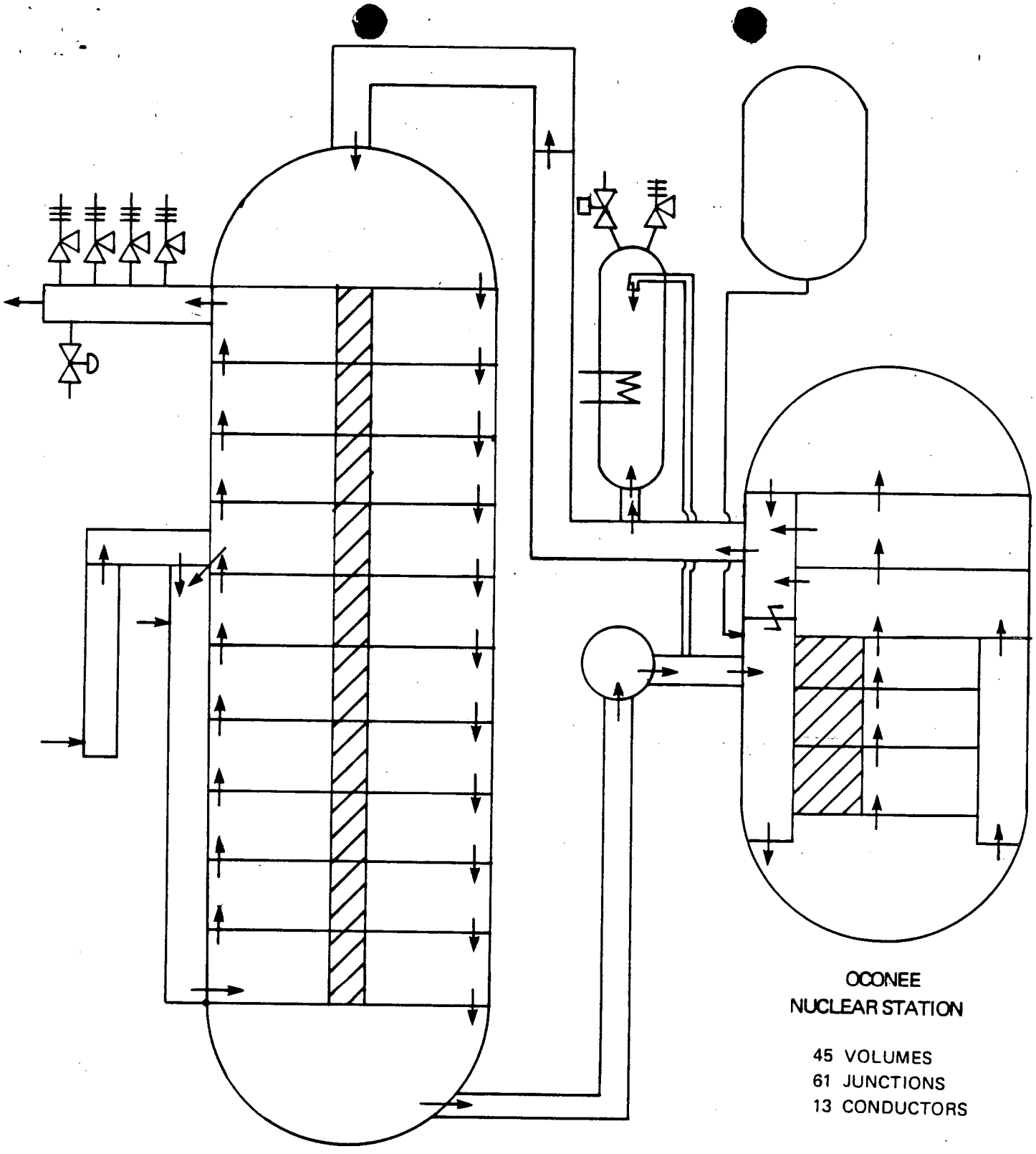


Figure 2.4-1. Oconee RETRAN Model

Figure 2.4.2-1.

Turbine Bypass Failure

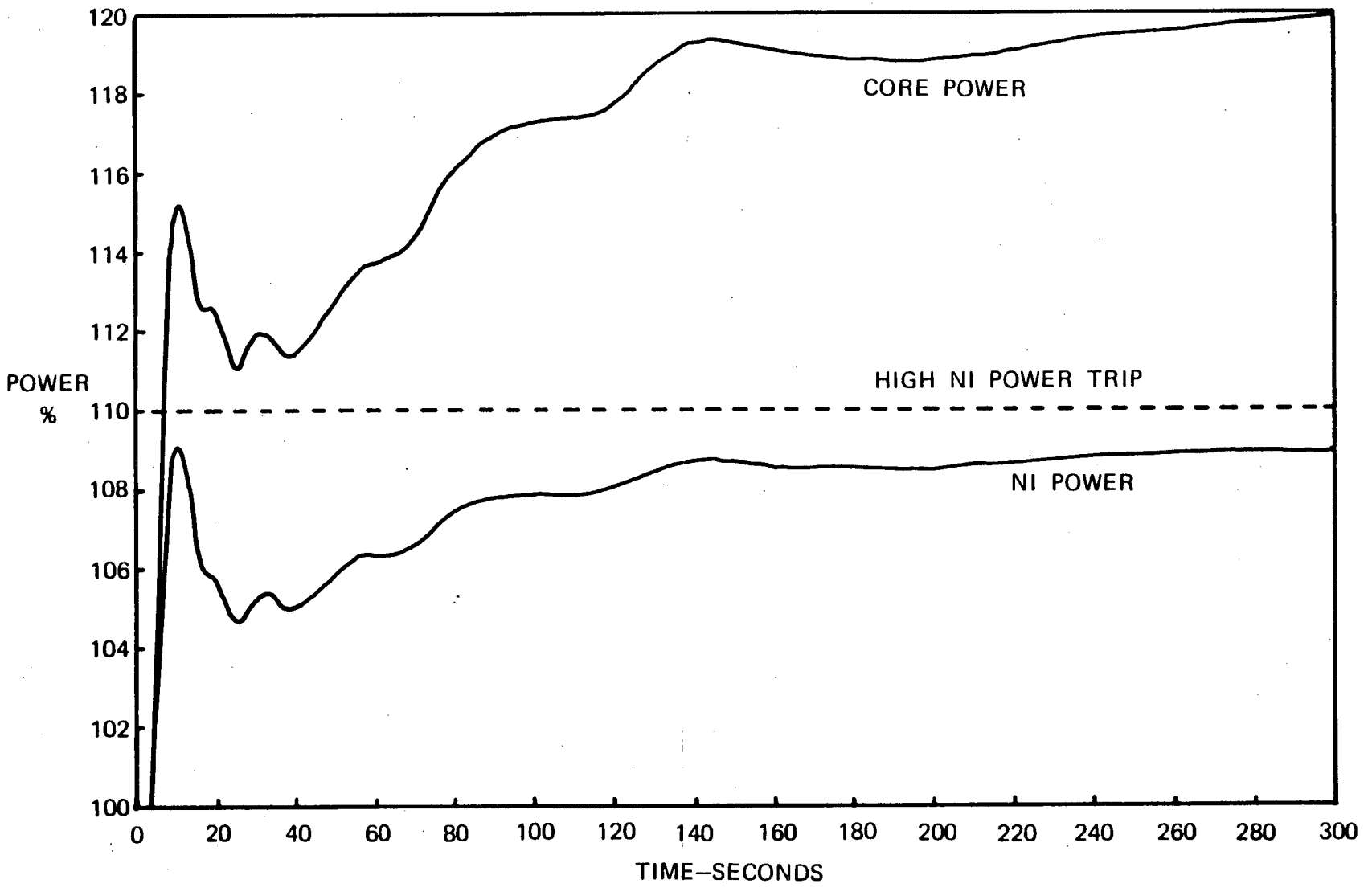


Figure 2.4.2-2.

Turbine Bypass Failure

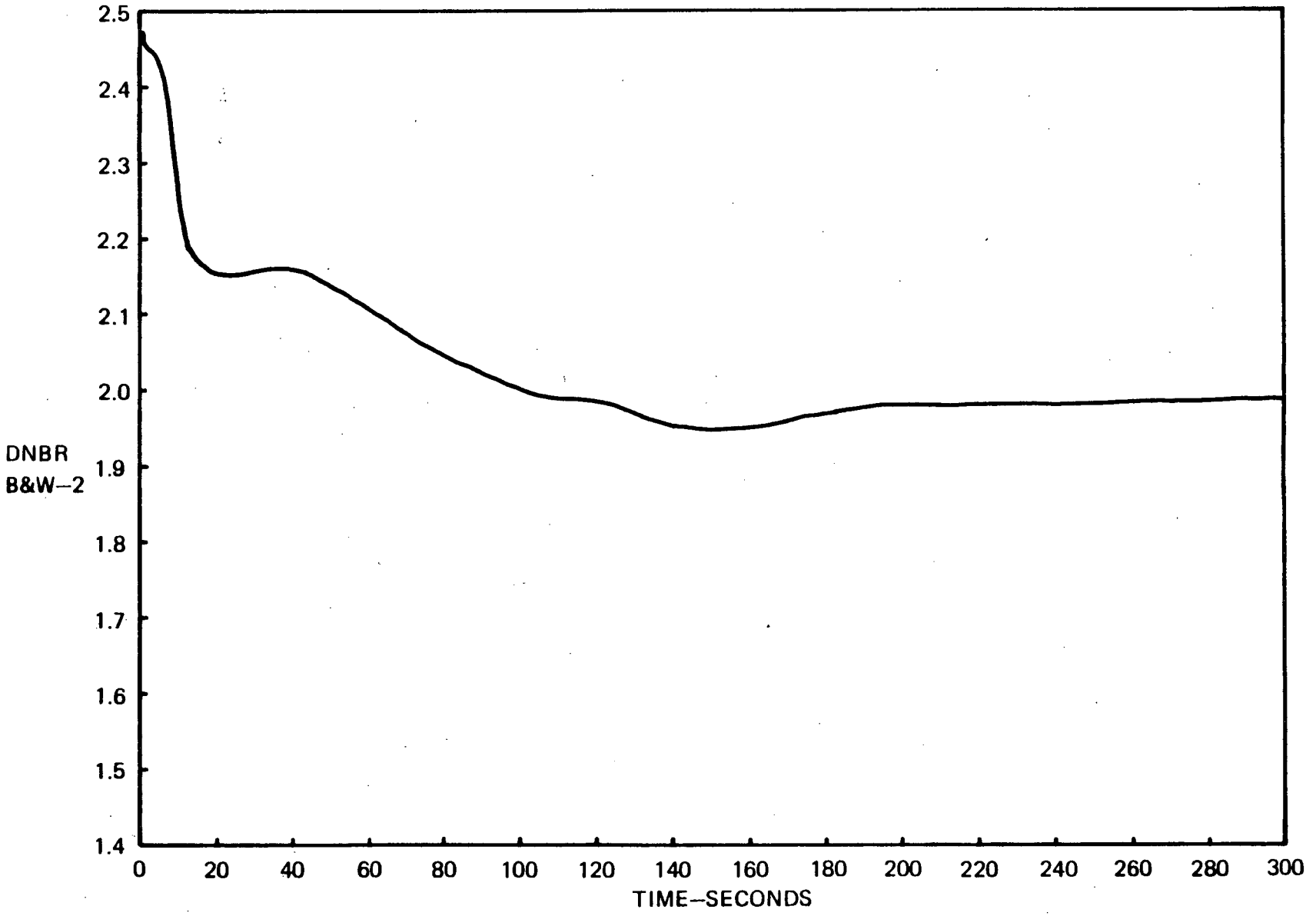


Figure 2.4.2-3.

Turbine Bypass Failure

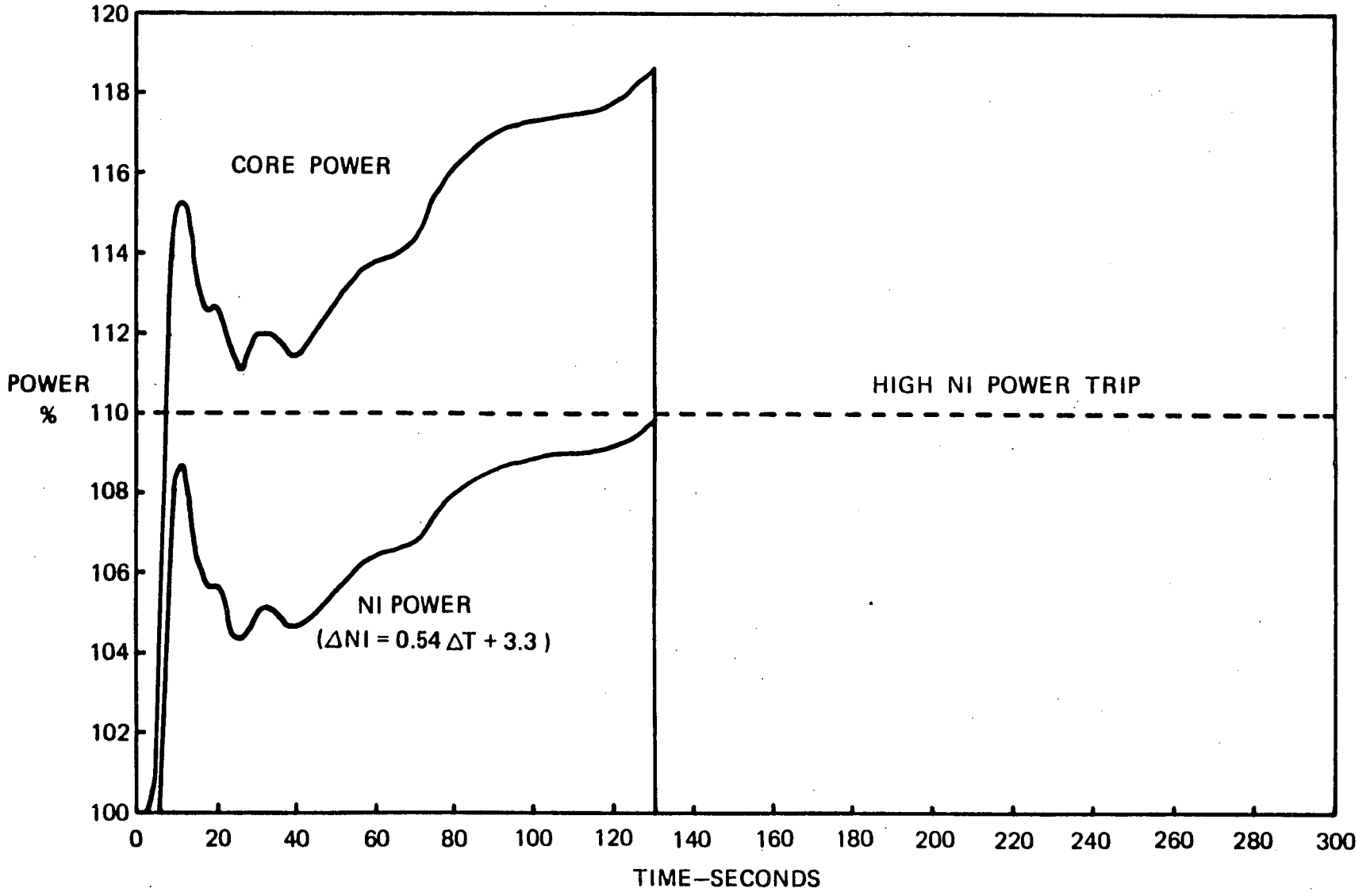


Table 2.4.3-1.

Steam Line Break Case Analyses

Case	Break Size ft^2	Moderator Coefficient 10E-4 Δk/k	Reactor Control	Trip Type Time NI Delay	Duration P > 112% (sec)	Rx Power Peak/ Sustained	Comments
1	1.0	-3.0	Yes*	High Flux 7.3 sec 0.4 sec Delay	1.0	119.8% / N/A	Mitigated by Fast Trip
2	0.75	-3.0	Yes*	High Flux 7.7 sec 0.7 sec Delay	1.0	117.0% / N/A	Mitigated by Fast Trip
3	0.60	-3.0	Yes*	High Flux 8.2 sec 1.5 sec Delay	1.5	116.9% / N/A	Mitigated by Fast Trip
4	0.50	-3.0	Yes*	High Flux 9.3 sec 2.2 sec Delay	2.0	116.5% / N/A	Mitigated by Fast Trip
5	0.45	-3.0	No	None	Extended	120.3% / 120.0%	NI Peak Flux 109.6% (trip at 110%)
6	1.0	-1.5	No	None	Extended	127.5% / 125.0%	NI Peak Flux 109.6% (trip at 110%)
7	1.0	-1.0	Yes	None	Extended	127.2% / 125.0%	Very Near P-T Trip at 10.0 sec
8	1.0	-0.5	Yes	P-T Trip 10.5 sec	1.5	114.6% / N/A	Mitigated by Fast Trip

Note: Yes* = reactor control insignificant

Figure 2.4.3.1-1.

1.0 ft^2 Steam Line Break

$\alpha_T = -1.5 \times 10^{-4}\ \Delta k/k/^oF$

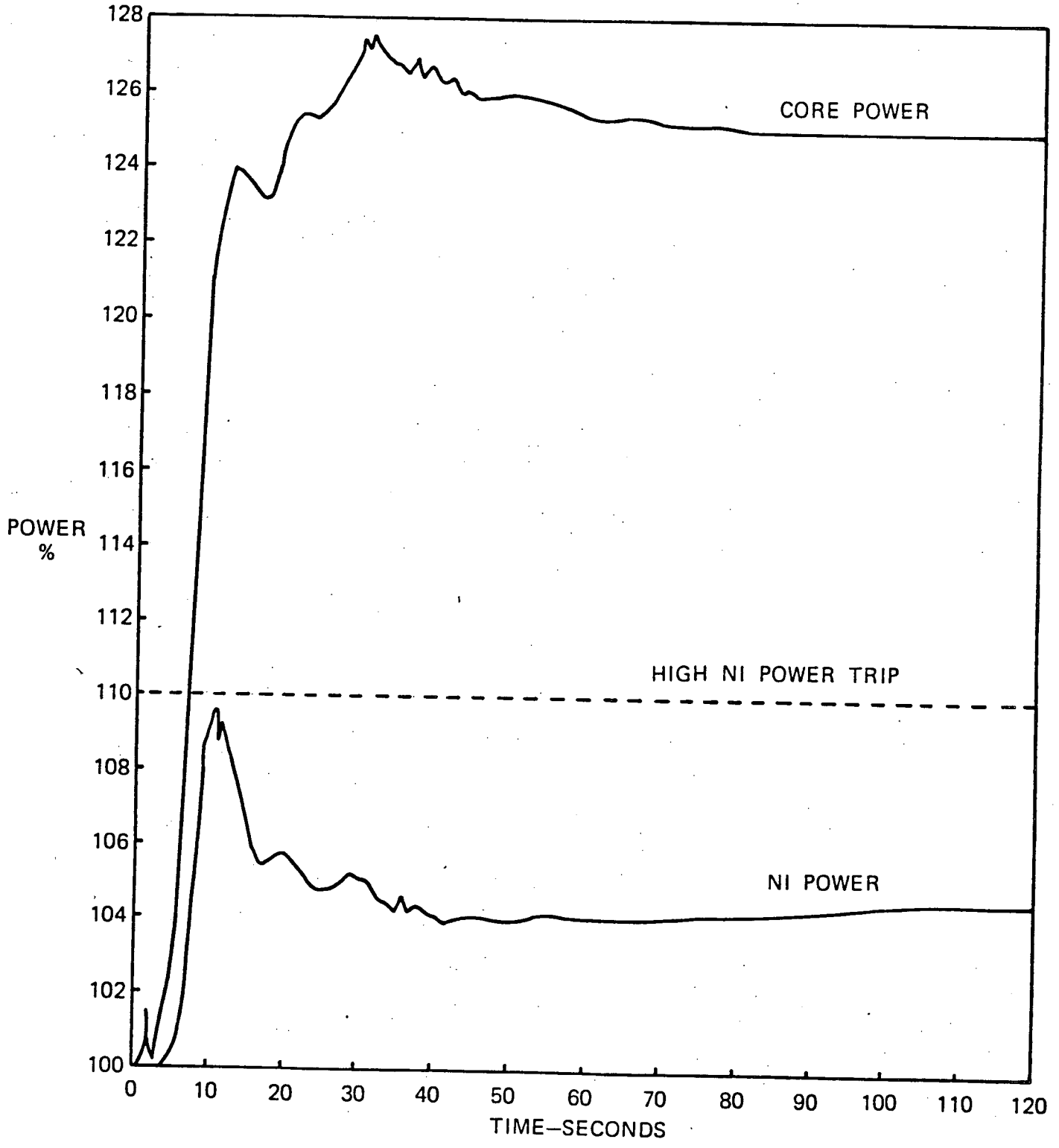


Figure 2.4.3.1-2.

1.0 ft^2 Steam Line Break

$\alpha_m = -1.5 \times 10^{-4}\ \Delta k/k/^oF$

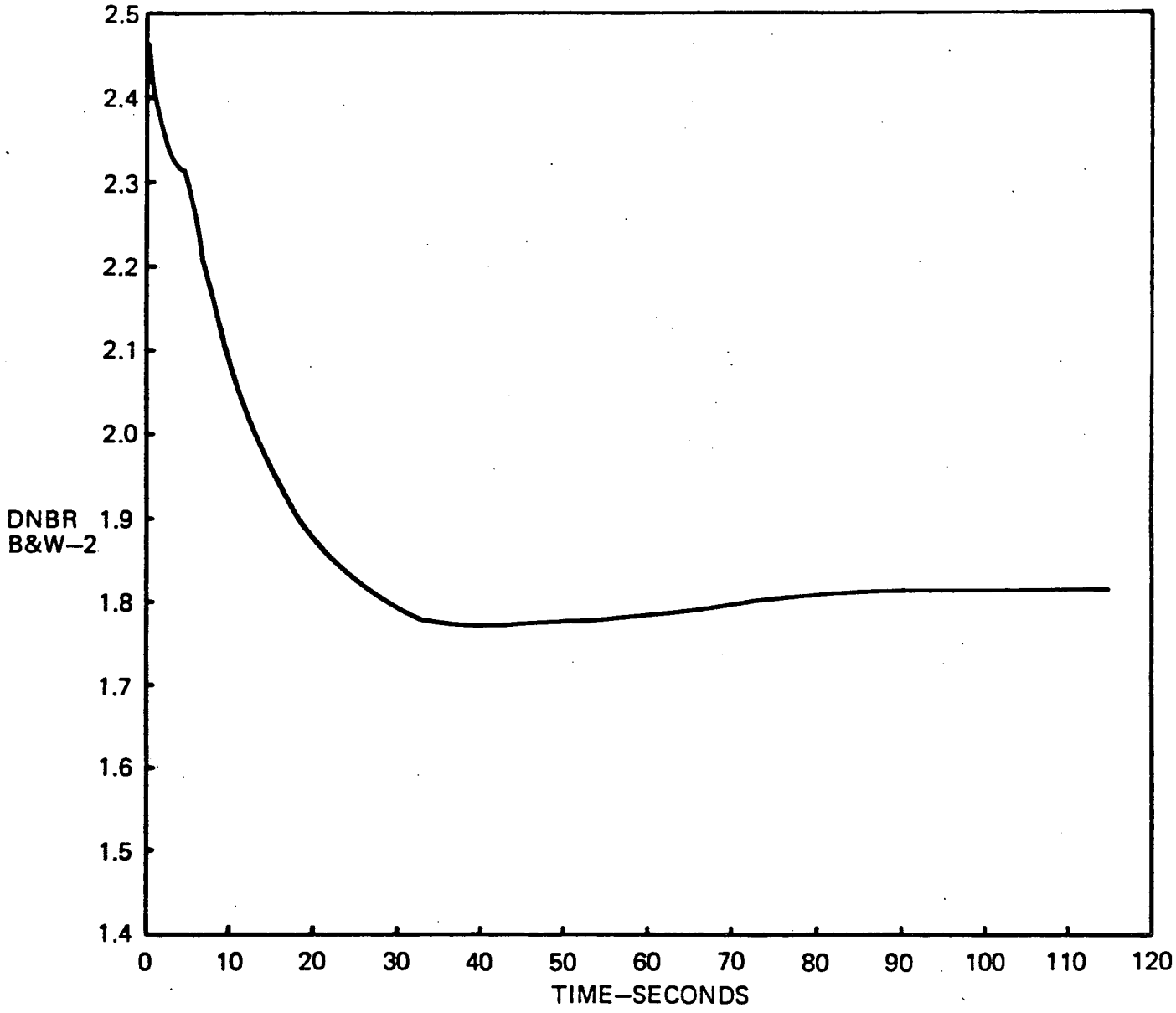


Figure 2.4.3.1-3.

Typical Center Fuel Melt Limit Vs. Burnup

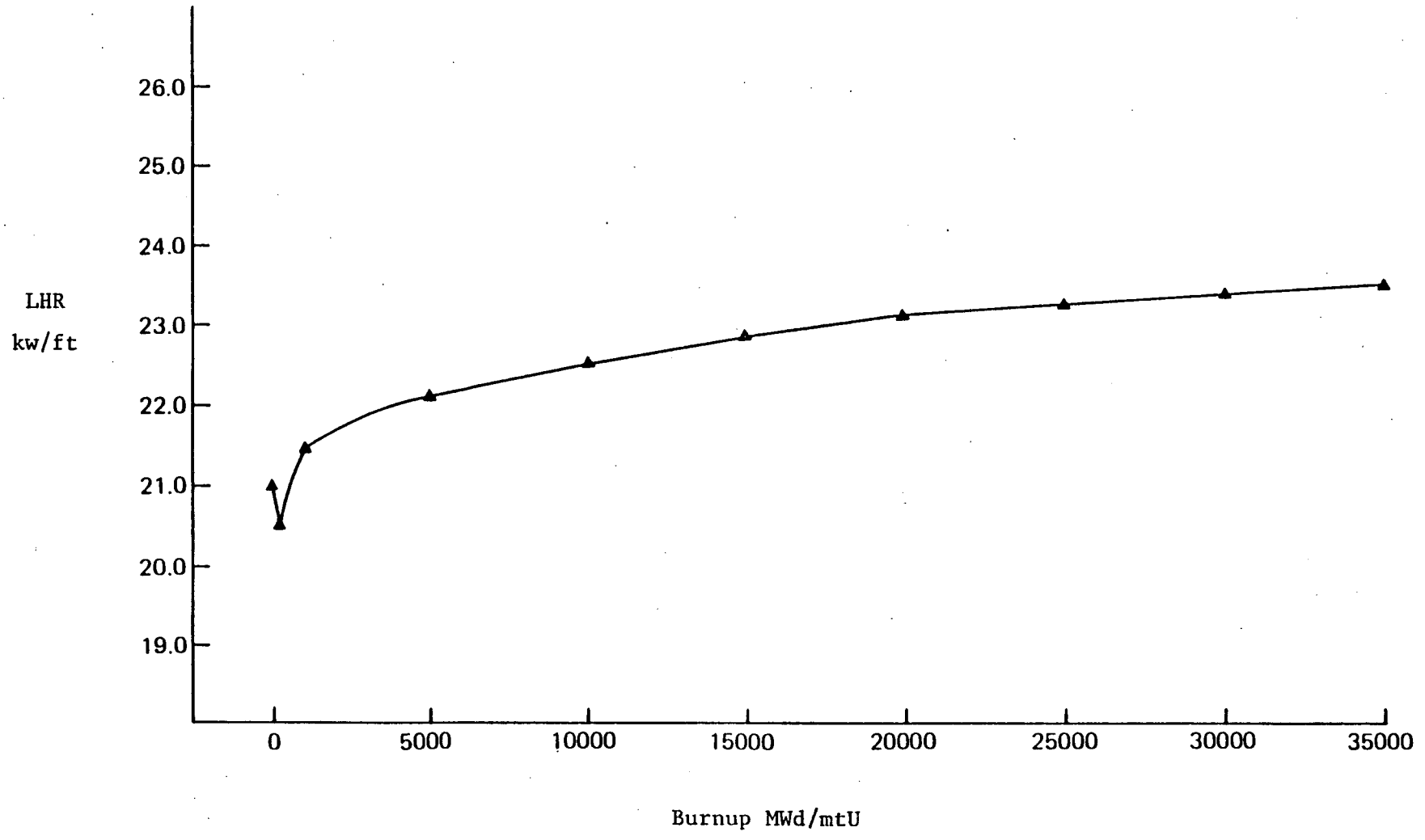


Figure 2.4.3.1-4.

1.0 ft^2 Steam Line Break

$\alpha_T = -1.5 \times 10^{-4}\ \Delta k/k/^oF$

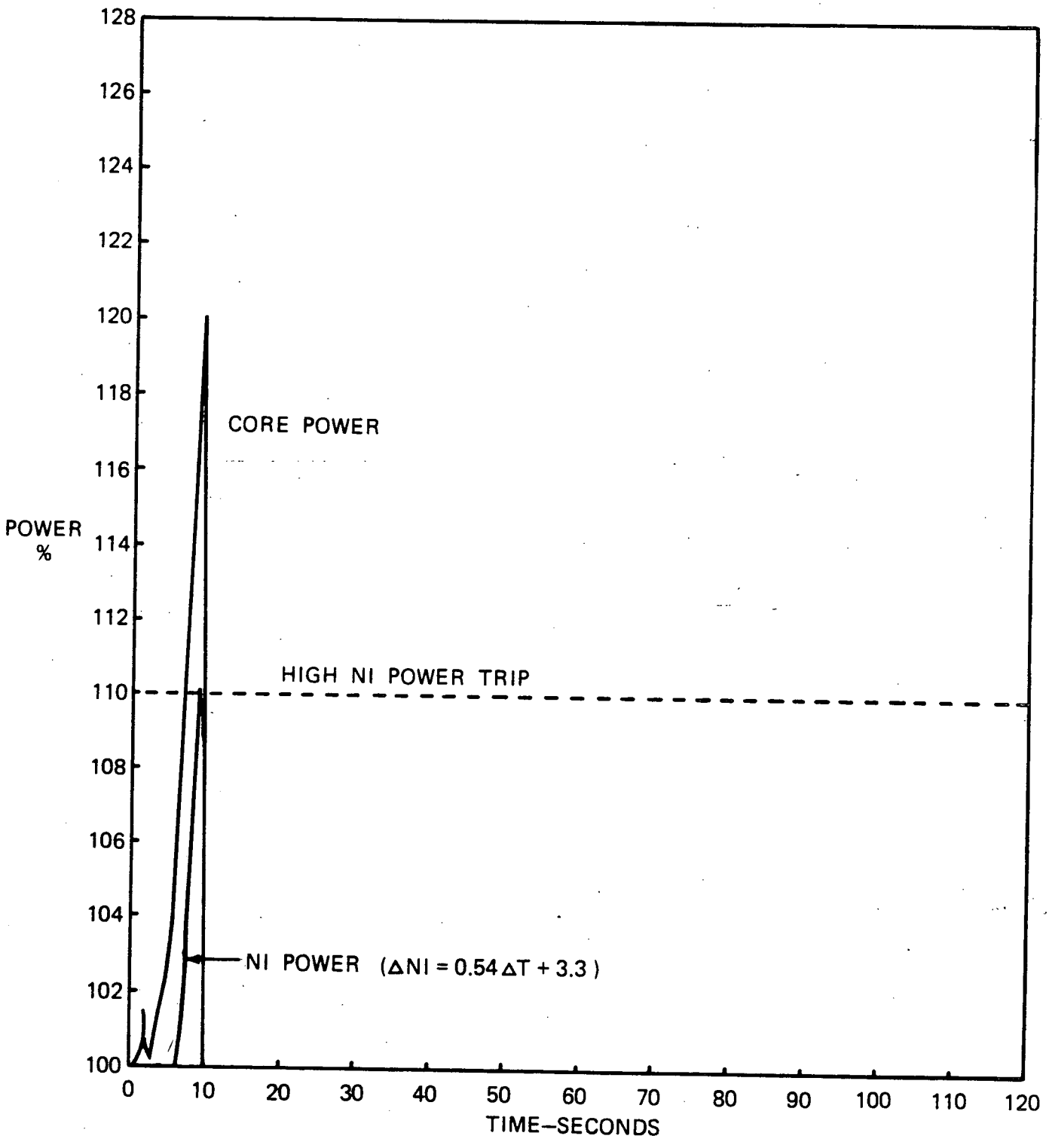


Figure 2.4.3.2-1.

1.0 ft^2 Steam Line Break

$\alpha_T = -1.0 \times 10^{-4}$ $\Delta k/k/^oF$

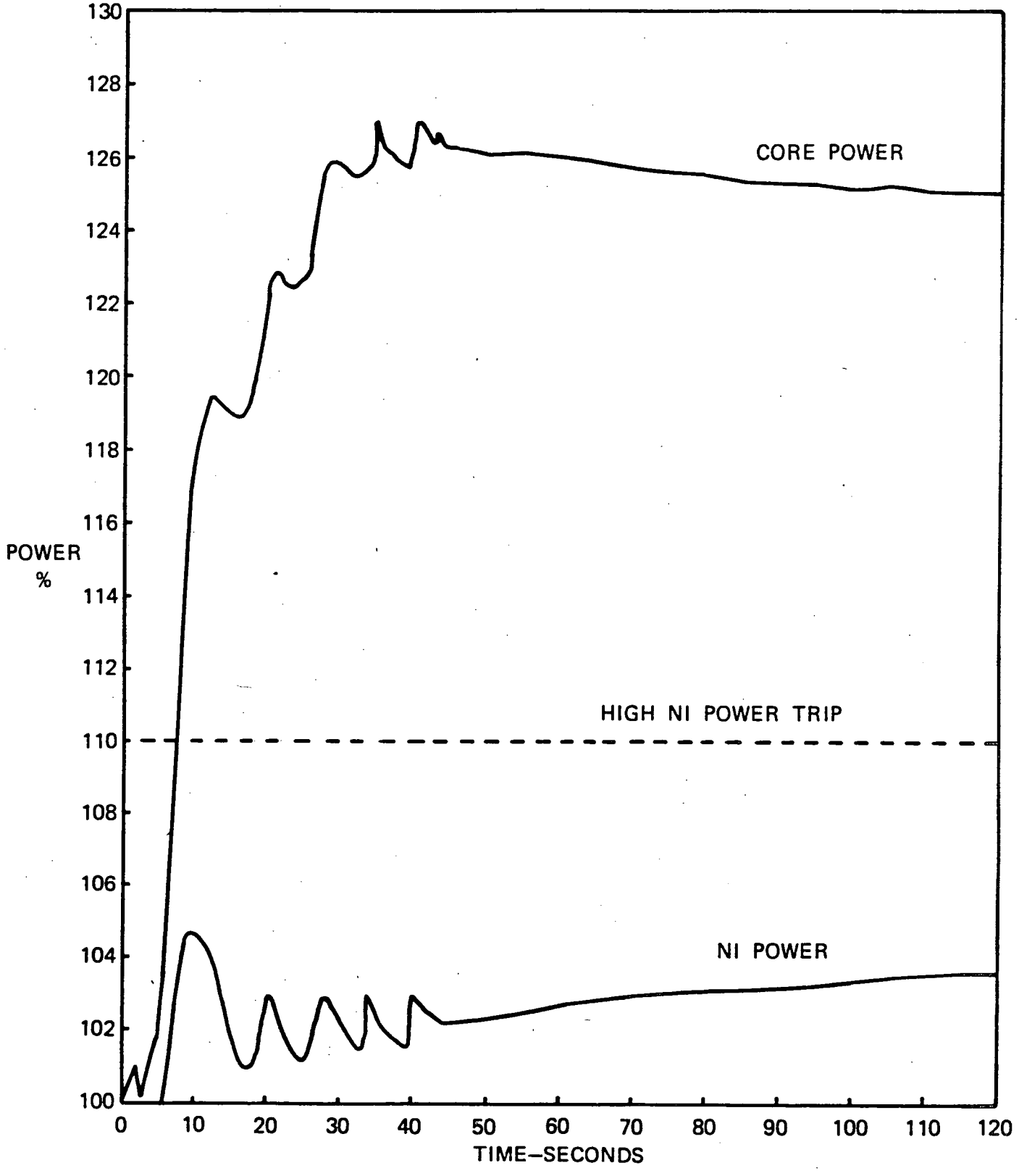


Figure 2.4.3.2-2.

1.0 ft^2 Steam Line Break

$\alpha_m = -1.0 \times 10^{-4}\ \Delta k/k/^oF$

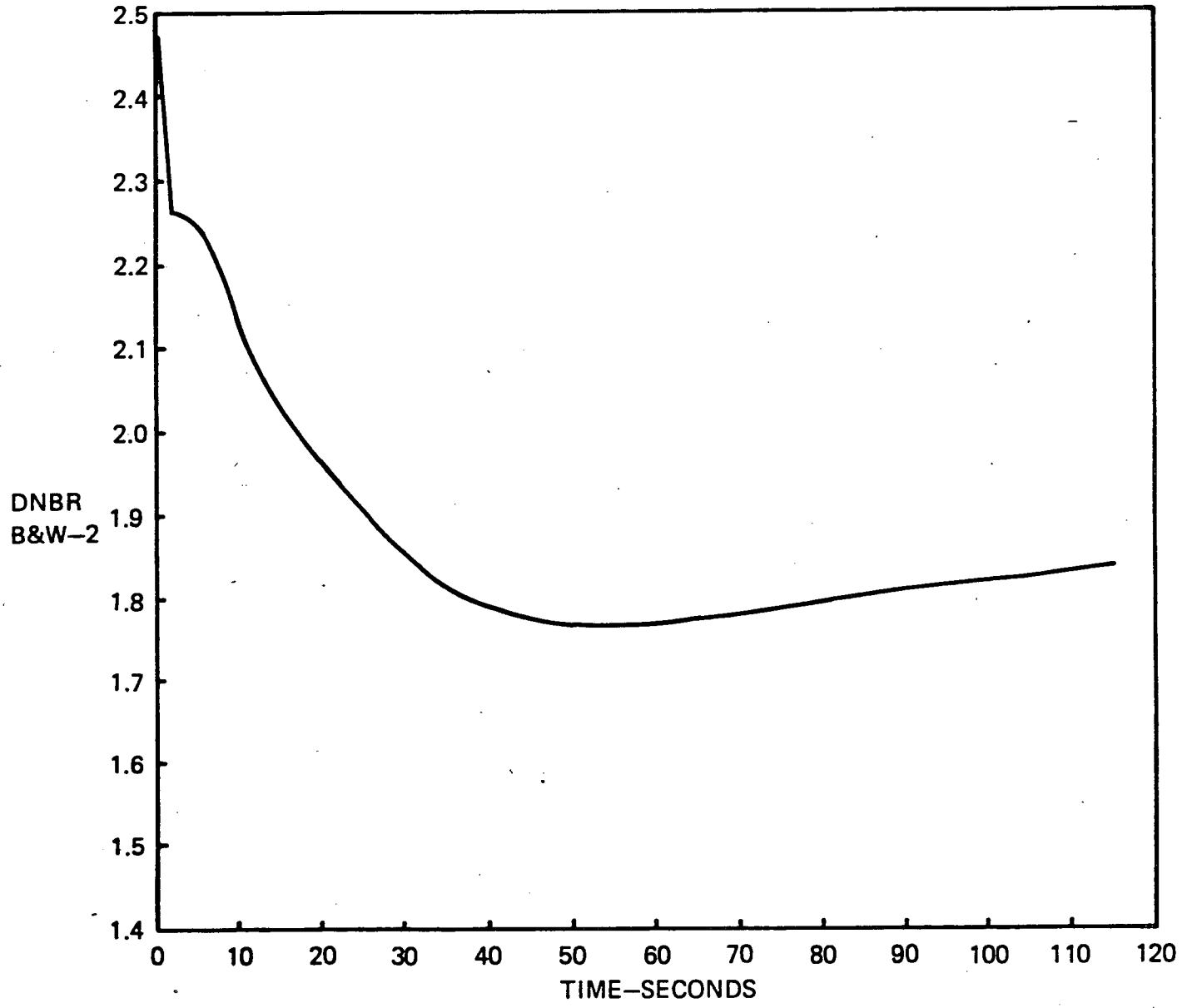


Figure 2.4.3.2-3.

1.0 ft^2 Steam Line Break

$\alpha_T = -1.0 \times 10^{-4}\ \Delta k/k/^oF$

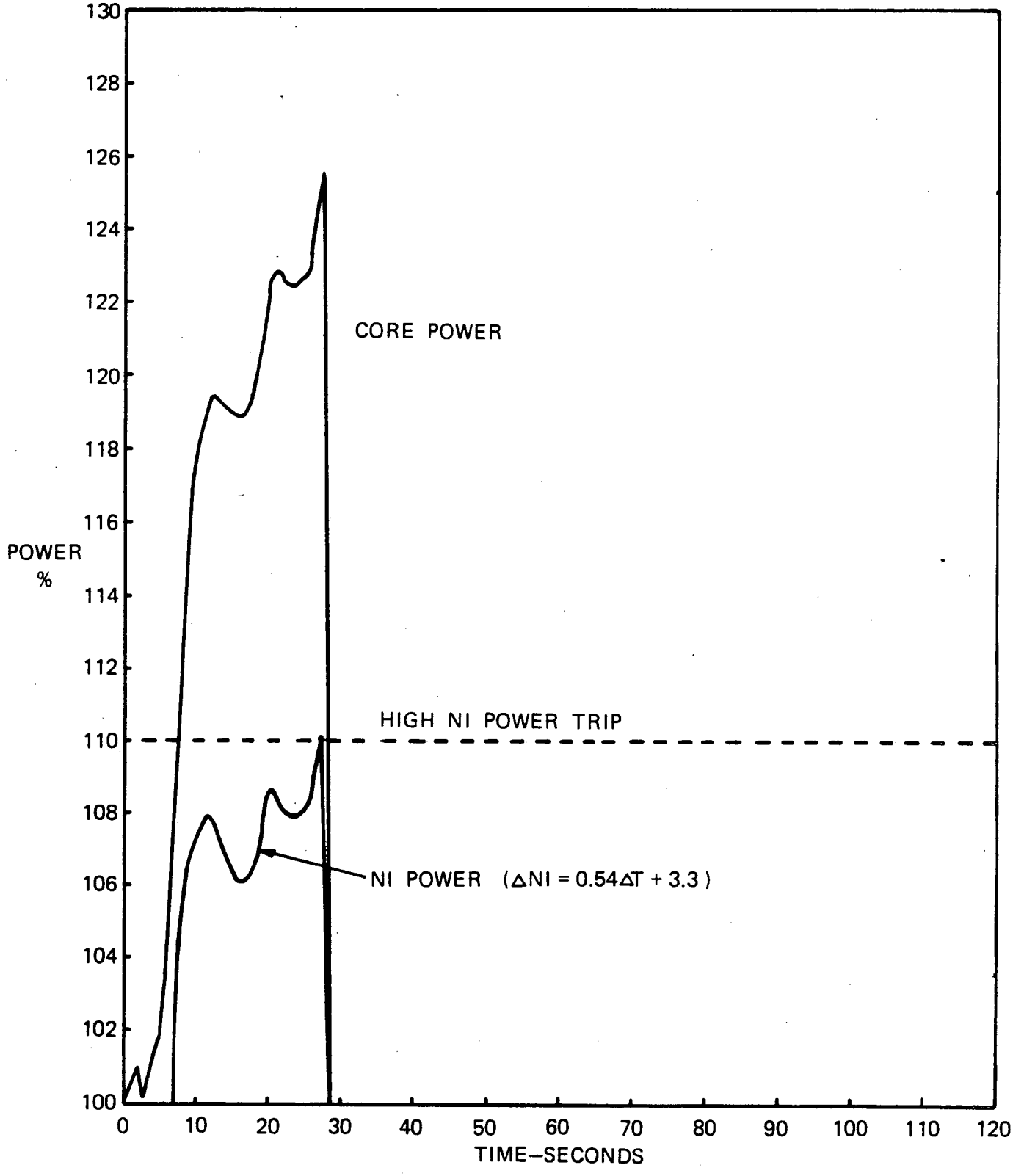
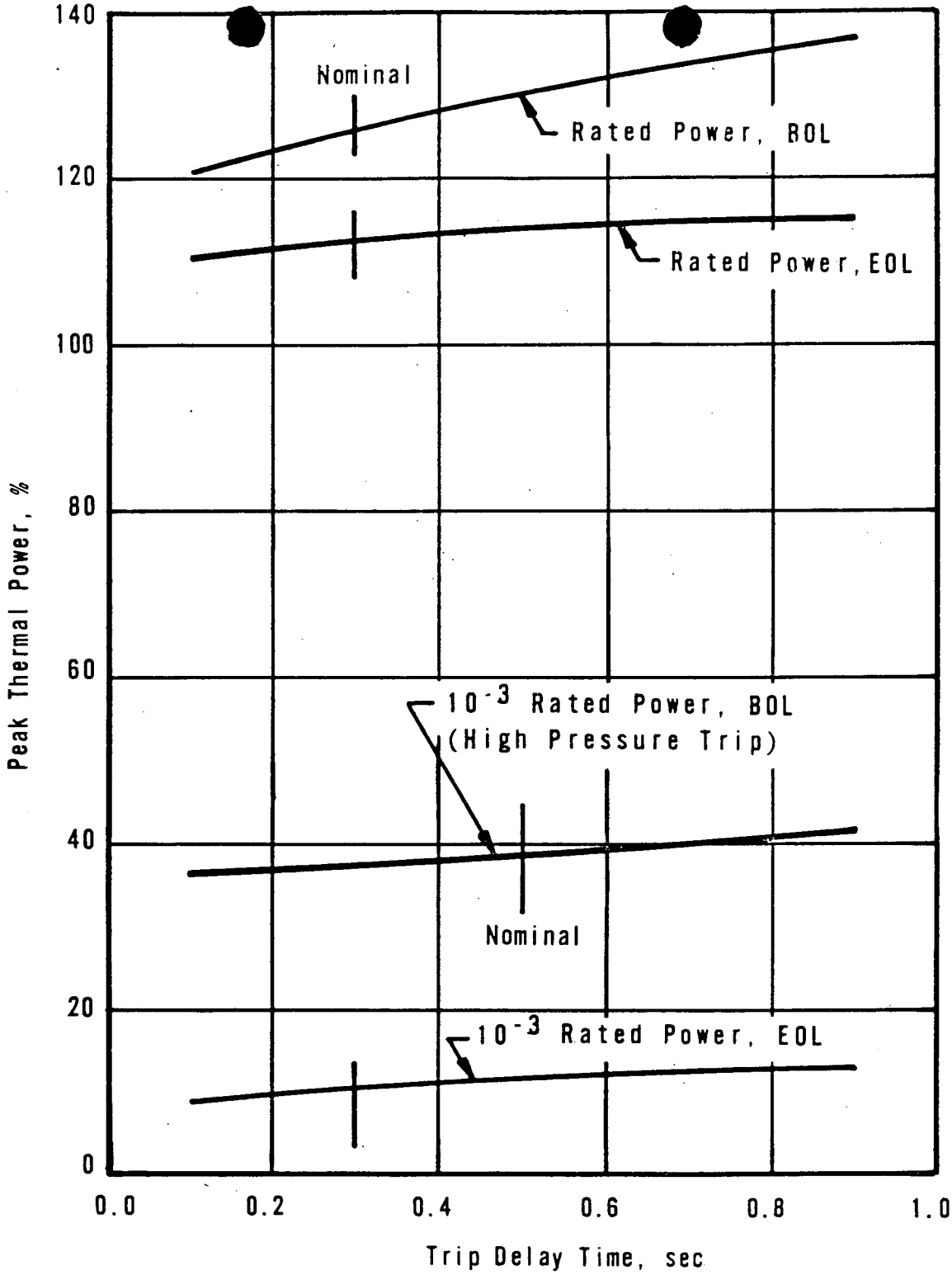


Figure 3.1-1.



EFFECT ON PEAK THERMAL POWER OF VARYING THE TRIP DELAY TIME FOR AN EJECTED ROD WORTH OF 0.56% $\Delta k/k$ AT 10^{-3} RATED POWER AND 0.46% $\Delta k/k$ AT RATED POWER

(Reference Supplement 9 Revisions for Oconee 3)

Table 3.2-1.

Rod Ejection NI Flux Error

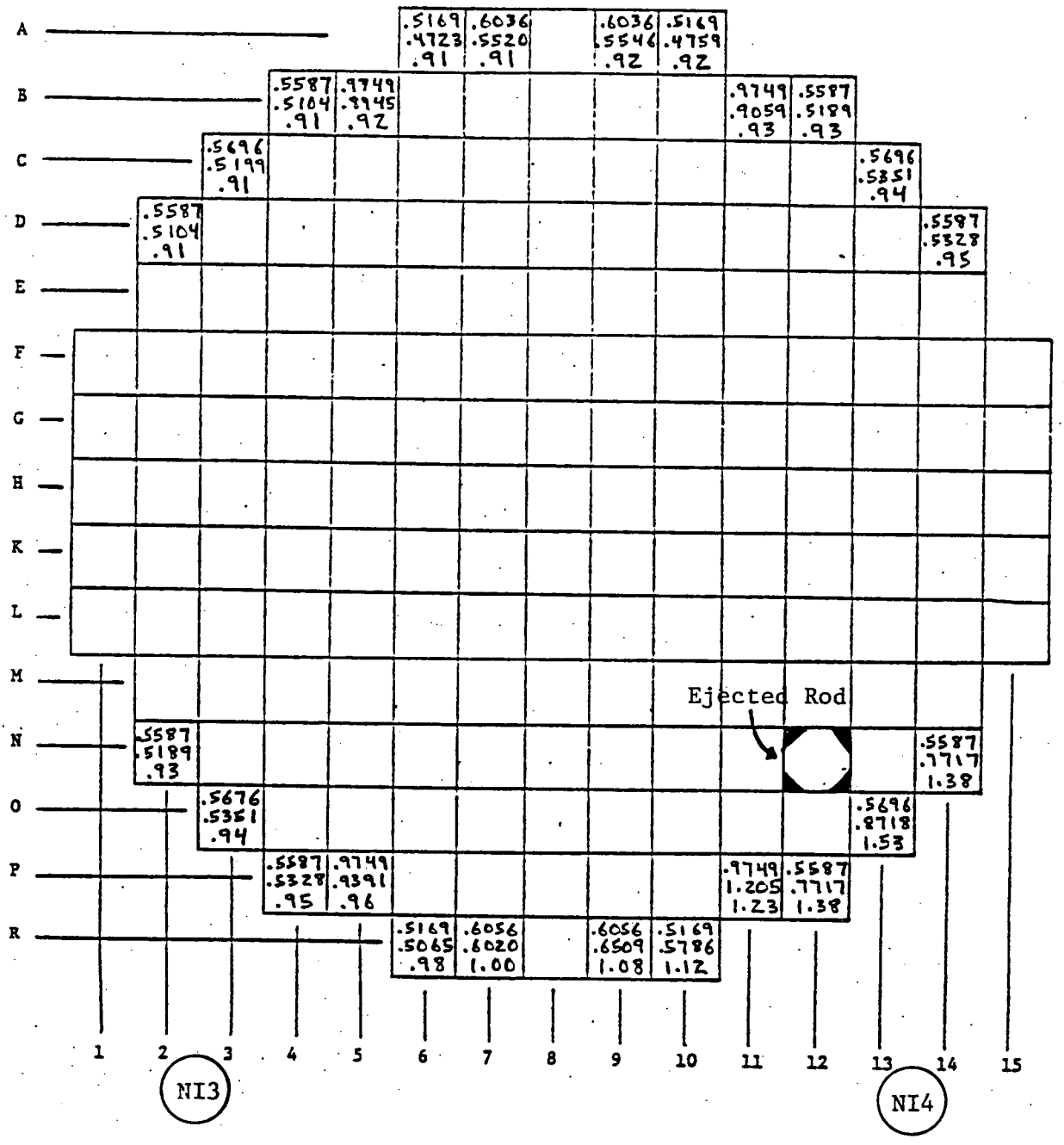
Unit Cycle	BOC EOC	Worth % Δk/k	Percent Withdrawn	Location	NI1 Δϕ	NI2 Δϕ	NI3 Δϕ	NI4 Δϕ	NI Flux Error
03C6	BOC	0.128%	0%	H-10	.98	.94	symmetric		.94
		0.051%	50%	H-10	.99	.98	symmetric		.98
		0.021%	75%	H-10	1.00	.99	symmetric		.99
		0.156%	0%	H-14	1.00	.91	symmetric		.91
		0.062%	50%	H-14	1.00	.96	symmetric		.96
		0.024%	75%	H-14	1.00	.99	symmetric		.99
		0.138%	0%	N-12	.93	.91	.96	1.28	.92
		0.055%	50%	N-12	.97	.96	.98	1.11	.96
		0.022%	75%	N-12	.99	.99	.99	1.04	.99
	EOC	0.136%	0%	H-10	.98	.95	symmetric		.95
		0.162%	0%	H-14	.99	.93	symmetric		.93
		0.154%	0%	N-12	.94	.93	.96	1.26	.93
01C6	BOC	0.156%	0%	K-13	.97	.94	.94	.97	.94
		0.130%	0%	L-10	.96	.95	.97	1.03	.95
	EOC	0.163%	0%	K-13	.97	.94	.95	1.02	.94
		0.145%	0%	L-10	.96	.95	.97	1.02	.95
02C5	BOC	0.194%	0%	L-10	.94	.92	.96	1.04	.93
		0.135%	0%	H-14	.99	.94	symmetric		.94
		0.164%	0%	N-12	.94	.93	.96	1.29	.93
	EOC	0.174%	0%	L-10	.95	.94	.96	1.02	.94
		0.153%	0%	H-14	.99	.95	symmetric		.95
		0.167%	0%	N-12	.94	.94	.96	1.25	.94

Note: NI#Δϕ = ratio of the average power of the six peripheral fuel assemblies nearest the NI detector location after/before the rod is ejected.

NI Flux Error = the average of the two lowest NI#Δϕ values.

Figure 3.2-1.

Oconee 3 Cycle 6
HFP Group 7 In
N-12 = 0.138% Δk/k
100% Power



.XXXX
.YYYY
.ZZ

Initial Radial Peaking Factor
Final Radial Peaking Factor
Assembly Flux Ratio

NI1 Flux Ratio = .93
NI2 Flux Ratio = .91
NI3 Flux Ratio = .96
NI4 Flux Ratio = 1.28

Table 3.3-1.

Rod Ejection NI Flux Error Trip Delay

Unit Cycle	BOC EOC	Worth %Δk/k	Percent Withdrawn	Location	NI Flux Error	112% Power Trip Time	NI Flux Error Trip Setpoint	NI Flux Error Trip Time	Trip Delay Sec
03C6	BOC	0.128%	0%	H-10	.94	0.402	117.0	0.418	0.016
		0.051%	50%	H-10	.98	N/A	112.2	N/A	N/A
		0.021%	75%	H-10	.99	N/A	111.1	N/A	N/A
		0.156%	0%	H-14	.91	0.394	120.9	0.417	0.023
		0.062%	50%	H-14	.96	N/A	114.6	N/A	N/A
		0.024%	75%	H-14	.99	N/A	111.1	N/A	N/A
		0.138%	0%	N-12	.92	0.399	119.6	0.427	0.028
		0.055%	50%	N-12	.96	N/A	114.6	N/A	N/A
		0.022%	75%	N-12	.99	N/A	111.1	N/A	N/A
	EOC	0.136%	0%	H-10	.95	0.401	115.8	0.420	0.019
		0.162%	0%	H-14	.93	0.393	118.3	0.409	0.016
		0.154%	0%	N-12	.93	0.394	118.3	0.410	0.016
01C6	BOC	0.156%	0%	K-13	.94	0.394	117.0	0.407	0.013
		0.130%	0%	L-10	.95	0.403	115.8	0.422	0.019
	EOC	0.163%	0%	K-13	.94	0.393	117.0	0.406	0.013
		0.145%	0%	L-10	.95	0.398	115.8	0.411	0.013
02C5	BOC	0.194%	0%	L-10	.93	0.388	118.3	0.403	0.015
		0.135%	0%	H-14	.94	0.400	117.0	0.415	0.015
		0.164%	0%	N-12	.93	0.393	118.3	0.409	0.016
	EOC	0.174%	0%	L-10	.94	0.391	117.0	0.404	0.013
		0.153%	0%	H-14	.95	0.395	115.8	0.405	0.010
		0.167%	0%	N-12	.94	0.392	117.0	0.405	0.013

Figure 3.3-1.

Rod Ejection Transient

Neutron Power vs. Time

High Flux Trip 120%