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ROD WITHDRAWAL TRANSIENT REANALYSIS FOR THE PALISADES REACTOR

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ROD WITHDRAWAL TRANSIENT

RE-ANALYSIS FOR THE

PALISADES REACTOR

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1.0 INTRODUCTION

This report details the results of a reanalysis of rod withdrawal transients for the Palisades reactor operating at 2530 MWT. The reanalysis was undertaken to account for the time response of the resistance temperature detectors (RTD) in the hot and cold legs. The previous transient analyses performed in 1977[1] had not explicitly accounted for the RTD response time. The RTD response time was not an issue until after the Three Mile Island incident in 1979. The hot and cold leg temperature RTD measurements are used in the thermal margin/low pressure (TM/LP) trip function for termination of rod withdrawal transients. The results of this analysis, including a description of the models used in the previous and in the current analysis, are summarized below and presented in greater detail in subsequent sections of this report. The analysis shows margin exists to DNBR limits for the transients analyzed when RTD time response is considered.

In the 1977 Exxon Nuclear Company (ENC) performed a plant transient analysis of the Palisades reactor operation at 2530 MWT. The analysis was performed with the PTSPAL version of the PTSPWR2 transient code and with the W-3 DNB correlation. The analysis did not explicitly include the RTD response time. For rod withdrawal transients part power peaking values were reduced as reactor power increased. Calculations for the most limiting transient, the rod withdrawal transient initiated from 52% of rated power, resulted in an MDNBR of 1.32, compared to the W-3 limit of 1.30.

The current rod withdrawal analysis is performed in a more conservative manner than the previous analysis in 1977. The current analysis accounts for the effect of the RTD response time on the TM/LP trip. To further add conservatism, radial peaking values are held constant during rod withdrawal transients and not allowed to decrease as reactor power increases. The analysis was performed at the current Palisades Technical Specification[2] part power peaking limits.

The current analysis included the use of the ENC XNB correlation to calculate DNB, and an update to the algorithm for calculating reactor coolant flow. In addition to the above changes, the UFEB82 version of the PTSPWR2 code(3) which has been used in licensing analyses of other CE reactors was adopted. The new version of the code includes a pressurizer model which more accurately calculates the pressure increase in the pressurizer during rod withdrawal transients.

The use of an improved basis for DNBR analysis of the rod withdrawals transient does not compromise the previous 1977 analysis of the other limiting transients which is based upon models and a DNB correlation containing excess conservatisms.

2.0 SUMMARY

In order to assess the adequacy of the reactor protection system with RTD time delay, the following incidents were analyzed:

1) Control rod withdrawal transients initiated from 102% of rated power at reactivity addition rates bounding the possible range.

(2) Control rod withdrawal transients initiated from 52% of rated power at reactivity addition rates bounding the possible range.

Control rod withdrawal transients from other power levels were not evaluated since a previous analysis concluded these were bounding.(4) Beginning of cycle (BOC) and end of cycle (EOC) kinetics parameters were used in the analysis and represent bounding values for minimum and maximum negative feed back, respectively. An additional set of kinetics representative of mid-cycle (MC) conditions was also evaluated.

The PTSPWR2 DNBR results for the 52% of rated initial power level are presented in Figure 2.1. The corresponding results for the 102% of rated power cases are presented in Figure 2.2. A more detailed calculation made with XCOBRA-IIIC based on the worst DNBR producing conditions calculated with PTSPWR2 resulted in the lowest MDNBR value calculated in this analysis of 1.40. XCOBRA-IIIC is ENC's program for calculating detailed thermal hydraulic conditions within subchannels, fuel assemblies and entire cores.

The most limiting transients were initiated from 52% power with constant reactivity addition rates of less than 3.0×10^{-5}/sec and the mid-cycle kinetics parameters. Total rod bank worth was limited to 1.0%$\Delta\rho$ for the mid-cycle cases. This corresponds to a conservative sum of the calculated rod bank worths of banks 3 and 4 inserted to their power dependent insertion limits (PDILS). Reactivity addition rates less than 3.0 x 10^{-5}/sec result in an automatic TM/LP reactor trip. The 1.40 value is to be compared to the XNB DNBR limit of 1.17. The calculation shows adequate margin to DNB limits.

For transients initiated from 52% of rated power, the MC kinetics cases at low reactivity rates result in lower DNBR values than the EOC cases due to the following. For maximum negative feedback conditions (EOC) and low reactivity addition rates, pressurizer spray capacity is adequate to stabilize primary system pressure. The power and average fuel and moderator temperatures in the primary system continue to rise until the rod banks are fully withdrawn. After this point in time, power and system temperature will stabilize provided a TM/LP trip set point is not reached. No reactor trip set point is reached for EOC kinetics and low reactivity addition rates.

For the MC kinetics cases and low reactivity addition rates, higher equilibrium power and temperature (fuel and moderator) conditions would be obtained for the same amount of total rod bank worth withdrawn compared to the EOC cases provided a trip set point is not reached. The MC kinetics parameters produce less negative feedback than the EOC parameters and therefore higher power levels are achieved at equilibrium for the MC kinetics cases. System pressure would be similar for both cases since it would be within the range of the pressurizer spray controller. For the MC cases, a total rod bank worth of 1%$\Delta\rho$ is withdrawn and the reactor system approaches equilibrium; however, a slight reduction in primary system pressure occurrs as the reactor system enters the stabilization phase resulting in a TM/LP trip. Since the MC cases were at a higher resultant power and primary system temperature than the EOC cases for about the same pressure, lower DNBR values were calculated for the MC case. If the TM/LP trip had not occurred, the calculated DNBR value in this MC case would have been similar to the tripped MC case.

The analysis performed supports Cycle 6 operation considering the RDT time delay and the current Technical Specification power peaking limits. It also supports future cycles whose operating parameters do not exceed the bounds upon which this analysis is based.

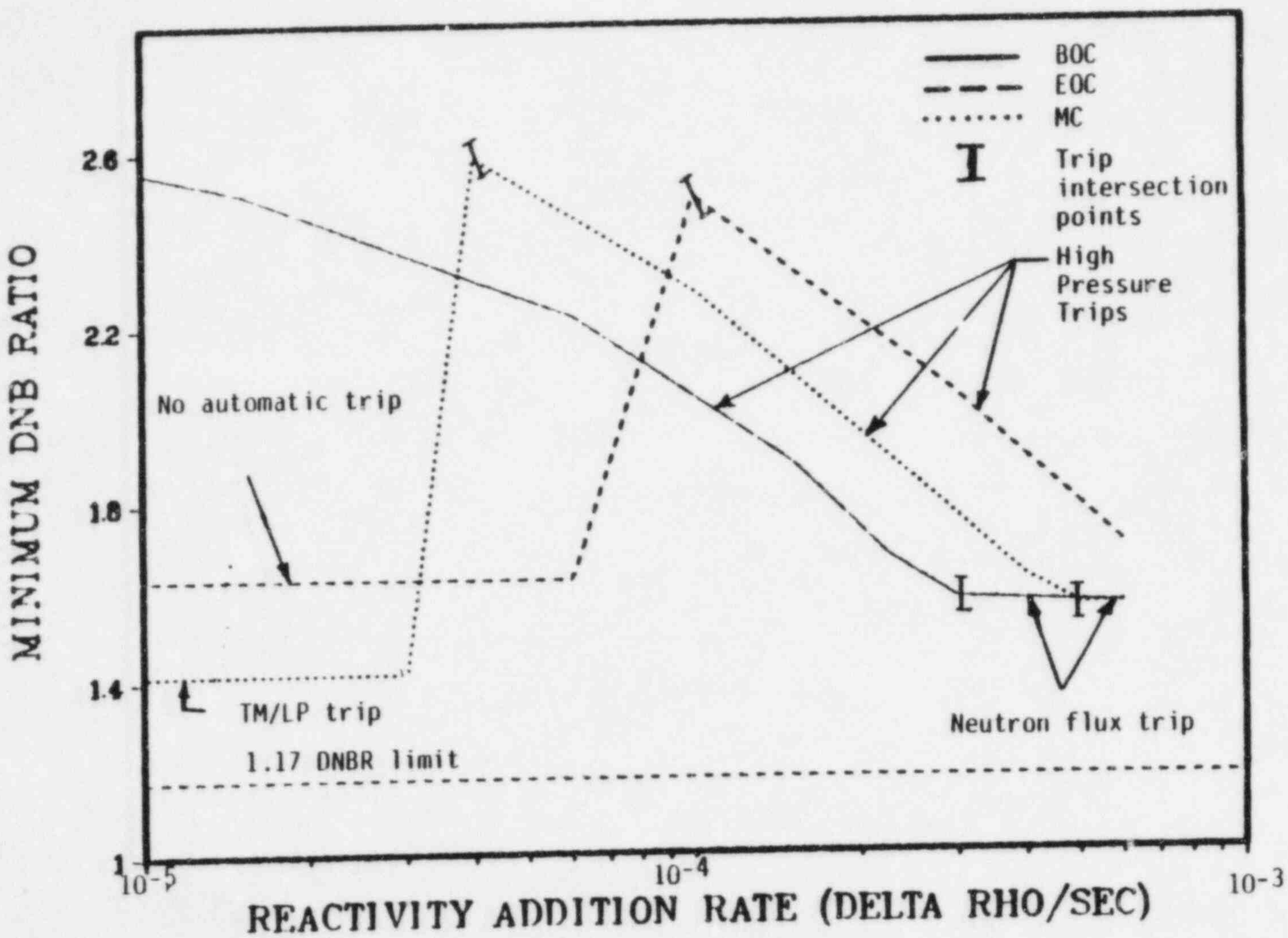


Figure 2.1 MDNBR Values for Rod Withdrawal Transients Initiated from 52% of Rated Power

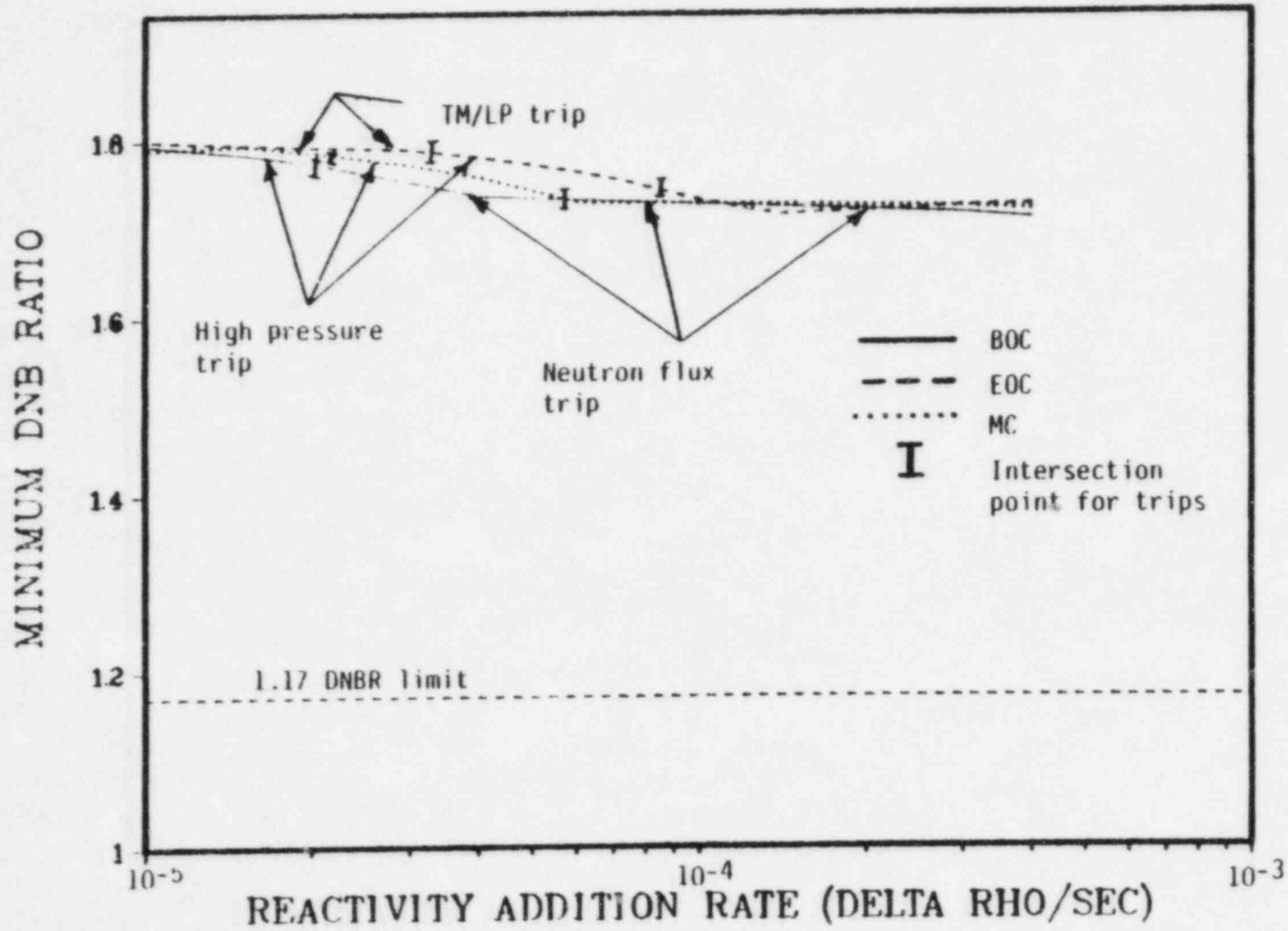


Figure 2.2 MDNBR Values for Rod Withdrawal Transients Initiated from 102% of Rated Power

3.0 CALCULATIONAL METHODS AND INPUT

3.1 PREVIOUS CODES

The 1977 analysis was conducted with the PTSPAL APR77A version of the PTSPWR2 computer program and the JULY75B version of the XCOBRA-IIIC computer program. PTSPWR2 is ENC's program for simulating pressurized water reactor transients. The XCOBRA-IIIC is ENC's program for calculating detailed thermal hydraulic conditions within subchannels, fuel assemblies, and entire cores. These versions were not found to be suitable for the current analysis. The UFEB82 version of PTSPWR2 and the UMAR82 version of XCOBRA-IIIC were used in this analysis for the below discussed reasons.

3.2 XCOBRA-IIIC VERSION - UMAR82

MDNBR values were calculated in this analysis with the XNB DNB correlation at the request of CPC. The XNB DNB correlation was developed using the UMAR82 version of XCOBRA-IIIC. The XNB correlation and data base is discussed in detail in a separate report.(5) The application of XNB specifically to Palisades is presented in an additional separate report.(6)

3.3 PTSPWR2 VERSION - UFEB82

Initial calculations in this analysis were made with the PTSPAL APR77A version of the PTSPWR2. Review of these calculations showed that the primary system pressure increased much slower than expected. It was determined that a more realistic pressurizer model was needed, such as the model currently available in the UFEB82 version.(7) The Palisades input was therefore integrated into the UFEB82 version of the code.

The new pressurizer model treats the compression of steam within the pressurizer, which is not condensed by the sprays as an adiabatic compression. Use of the new pressurizer model results in higher primary system pressure which under some rod withdrawal transients results in a high pressure trip. A high pressure trip results in improved DNB margin.

3.3.1 RTD Model

The temperatures measured by the resistance temperature detectors in the primary system hot and cold legs are used as input into the thermal margin low pressure trip. The trip function used in previous 1977 analysis and also this analysis is:

P1 = 61.66*THLRTD - 37.16*TCLRTD - 14725 - 165 + 25

P2 = 23.13*THLRTD - 11885 - 165 + 25

165 = the measurement uncertainty

25 = the additional bias required to meet DNB criteria in 1977 analysis

THLRTD = Hot leg RTD measurement, ^{o}F

TCLRTD = Cold leg RTD measurement, ^{o}F

P1 = Pressure which the primary system must be above or reactor trip will occur, psia

P2 = Pressure which the primary system must be above or reactor trip will occur, psia

The plant trip system logic auctioneers the two pressures (P1 and P2) and trips the reactor if pressure falls below the higher value.

The model for the hot and cold leg RTD temperatures in terms of the hot and cold leg coolant temperatures is a built-in first order function in the PTSPWR2 computer code. The basic finite difference form of the model is:

$$TRTD(N+1) = \frac{\Delta t}{\tau} * (TF(N+1) - TRTD(N)) + TRTD(N)$$

TRTD(N+1) = Temperature of RTD at time (N+1)

TRTD(N) = Temperature of RTD at time (N)

TF(N+1) = Temperature of primary coolant at time (N+1)

Δt = Differential time step

τ = Time constant for the RTD

3.3.2 Secondary Safety Valve Model Update

During the control rod withdrawal transient, power increases but turbine flow is relatively constant. As a result the primary and secondary systems will increase in temperature and pressure until power reaches a maximum and/or equilibrium or until the secondary safety valves open. Upon opening of the secondary safety valves, pressure and temperature on the secondary side will stabilize. The reactor system response therefore is impacted by the operation of these valves during a rod withdrawal transient.

At Palisades there are three sets of safety valves in each of two steam lines.(4) Each set contains 4 valves. Each set begins to open at a particular set point and the set points are nominally 985, 1005, and 1025 PSIG for the three sets respectively. The bank with the highest pressure set point will be fully open at a nominal pressure of 1055 PSIG.

The 1977 analysis modeled these valves as follows. The valves were assumed to begin opening at 1030 PSIA and be fully open .1 second thereafter. The valves were assumed to begin closing at 1000 PSIA and be completely closed within .1 sec. For certain cases in the current analysis involving relatively moderate pressure/time derivatives in the secondary system, this model resulted in repeated, and rather rapid, opening and closing of the valves and attendent numerical stability problems. This evidenced itself as a pressure spike in the primary system.

A more realistic proportional model for the valves was utilized which assumes the valves open in proportion to the ratio of the pressure difference between steam line pressure and the lowest pressure set point and the difference between the full open pressure and the lowest pressure setpoint. This model was added to the PTSPWR2 version utilized in this analysis by multiplying the previously calculated valve flow by the following proportionality factor to obtain the updated flow:

$$\text{Flow factor} = \frac{\text{PSL} - \text{PSP}}{\text{PFO} - \text{PSP}}$$

PSL = Absolute pressure in the steam line, PSIA

PSP = Nominal absolute pressure set point

985. + 14.7 = 999.7 PSIA

PFO= Nominal full open pressure for the last set of valves to open, 1025 + 30 + 14.7 = 1069.7 PSIA

3.4 INPUTS

Only inputs that have some difference compared to that used in the 1977 analysis for XCOBRA-IIIC or PTSPWR2 are discussed here. For the 1977 analysis inputs, the reader should refer to Reference (1).

3.4.1 Kinetics Parameters

The kinetics parameters utilized in this analysis consisted of three sets:

(1) Beginning of cycle (BOC) or minimum feedback

(2) End of cycle (EOC) or maximum feedback, and

(3) Mid-cycle (MC).

The basis for the parameters used in the analysis are listed in Table 3.1. Also listed are calculated parameters for the next cycle, Cycle 6, which is representative of expected values for future cycles. The BOC, EOC and MC values used are listed in Table 3.2. The BOC and EOC values are identical to those used in the 1977 analysis with exception of the EOC doppler coefficient. A review of the kinetics parameters at the EOC suggest that a doppler coefficient of $-2.11 \times 10^{-5}\ \Delta\rho/^{o}F$ be utilized rather than the previous value of $-1.66 \times 10^{-5}\ \Delta\rho/^{o}F$. The calculated value at hot full power ($-1.51 \times 10^{-5}\ \Delta\rho/^{o}F$) was more positive than the value used in the 1977 analysis, but calculated values at hot zero power were slightly more negative ($-1.76 \times 10^{-5}\ \Delta\rho/^{o}F$). Therefore, in order to insure bounding of feedback effects, the hot zero power value was made more negative by 20% and that value is the basis for this analysis.

During the execution of this analysis CPC requested some additional parallel analysis for the current cycle, Cycle 5, subject to different assumptions and conditions. The kinetics utilized in that analysis were approximately representative of values mid-way between maximum and minimum feedback values. Those values were also utilized in this analysis and are termed mid-cycle (MC) in this report.

3.4.2 Rod Bank Worth

For rod withdrawal transients from the 52% initial power level, power will increase as long as the rods continue to be withdrawn with the exception that negative feedback effects can act in a delayed fashion to reduce power overshoots. However, as the rods continue to be withdrawn power will once again increase until the rods are fully withdrawn at which time feedback effects will bring the power to some equilibrium value assuming a reactor trip set point has not been attained. The PDILs for rod banks 3 and 4 are 20% and 80% respectively at 50% power based on Figure 3.1.[3] In the 1977 analysis a conservative value of combined rod bank (3 and 4) worth of 1.5% was assumed and was based on the rods being fully inserted. This assumption was maintained for this analysis for BOC and EOC kinetics. However, for MC kinetics for transients initiated from 52% power, DNB criteria could not be met. Therefore, rod bank worths for banks 3 and 4 at their PDILs were reviewed and found to be less than 1% $\Delta\rho$. Therefore a value of 1% $\Delta\rho$ was used in the analysis for MC kinetics and the 52% power level, and found to result in DNB values above the limit.

3.4.3 Power Distributions and Radial Peaking Factors

The axial power distributions used in this analysis are shown in Figures 3.2 and 3.3 for transients initiated from 102% and 52% power respectively. These axial distributions were held constant during the transient. The 102% power case axial distribution is the same as used in recent analyses for Palisades reloads and is the nominal distribution utilized in the sensitivity study of axial power distributions in 1978.[7] The axial distribution for the 52% power case is the same as that used for the 1977 analysis.

The radial peaking factors used in this analysis are in compliance with the Technical Specification allowables for full and part power.[3] The local radial power distribution is taken from neutronics analysis of a B_4C shimmed, 208 rod, "I" reload fuel assembly. The calculation was made for an infinite array of similar fuel assemblies. Previous similar DNB analysis has shown that fuel assemblies with 208 rods are the most limiting and that infinite array power distribution maps provide the worst case local power distribution. The fuel rod and channel numbering system utilized in this analysis is illustrated in Figure 3.4. The local power distribution map is shown in Figure 3.5. Previous analysis has shown that interior rods are the most DNB limiting. Rod number 11 adjacent to channel 17 is the most limiting for this selected assembly as has been shown to be the case for previous analyses. Following precedence established by previous analysis, a 3% increase in power above the Technical Specification local allowable has been added to the worst rod, i.e. rod 11.

3.4.4 RTD Time Constant

CPC requested that the analysis be conducted with an RTD time constant of 7±2 seconds. Based on the TM/LP trip functions, the hot leg RTD time constant was taken to be 9 seconds and the cold leg time constant was taken to be 5 seconds in order to be conservative during the rod withdrawal transient.

3.4.5 Secondary Valve Flow Rate Capacity

The 24 secondary safety valves each have a flow rate capacity of 135 lbm/sec[4]. The 1977 analysis assumed the maximum flow capacity for the total of 12 valves in each steam line to be 540lbm/sec. This appears to have been an oversight since this value corresponds to the total flow of a single group of four valves. The analysis presented in this report assumed a maximum total flow capacity of 1622 lbm/sec for each set of 12 valves in each steam line.

3.4.6 Application of Coolant Flow Update

The primary system flow rate decreases as power increases due to temperature effects on the hydraulics in the primary loop. The flow rate at a particular power level is calculated on the basis of a maximum required flow at hot zero power of 126.9×10^6 lbm/hr corrected to a primary coolant temperature of 532°F.[3] The calculation method used in the 1977 analysis has been updated. The updated method utilized in this analysis, and the previous method are described in detail in Appendix A.

The updated coolant flow method was applied as follows. First primary system flow rates were calculated at the 52% and 102% power levels using the updated methodology. Average assembly mass velocities were next calculated for each power level. Correction factors were then applied to obtain the hot assembly mass velocities at each power level as follows:

GHC = GAVE * FCBC * FMU * FMD

GHC = Hot assembly inlet mass velocity

GAVE = Average assembly inlet mass velocity without corrections

FCBP = Core bypass factor, .97

FMU = Primary system measurement uncertainty, .97

FMD = Flow maldistribution factor between assemblies, .97

The correction factors were established on the basis of previous analyses. These factors continue to provide conservatism to the flow values utilized in the analysis.

The hot assembly mass velocity was input to XCOBRA-IIIC to calculate the hot channel DNB ratio and other parameters utilized to calibrate the hot channel model in PTSPWR2. The transient was next run on PTSPWR2 and the minimum DNB point established. Conditions at the minimum DNB point were then input to XCOBRA-IIIC to calculate more detailed local fluid conditions and a more accurate DNB value. The higher flow calculated at the beginning of the transient results in higher flows at the minimum DNB condition, and thereby improves the DNB value at that point.

Table 3.1 Kinetics Parameter Comparison

	1977 Analysis Kinetics (1)		Kinetics Utilized in Cycle 5 Reanalysis		Kinetics Calculated for Cycle 6 Safety Analysis	
	BOC	EOC	Nominal Values at 10,000 MWD/MT	Bounding* @10,000 (Min. Feedback)	BOC	EOC
Doppler Coefficient $(\Delta\rho/^{o}F) \times 10^{-5}$	-0.87 (least negative)	-1.66 (most negative)	-1.46(HFP)	-1.17	-1.30 -1.59	-1.51 HFP -1.76 HZP
Moderator Temperature Coefficient $(\Delta\rho/^{o}F) \times 10^{-4}$	+0.5 (least negative)	-3.5 (most negative)	-2.54	-2.03	-0.57 +0.28	-2.76 HFP ----- HZP
Moderator Pressure Coefficient $(\Delta\rho/PSIA) \times 10^{-6}$	-1.00 (most negative)	+7.00 (most positive)	+5.08	+4.06	+1.14 -0.56	+5.52 HFP ----- HZP
Delayed Neutron Fraction	0.0075 (highest)	0.0045 (lowest)	0.0053	0.0064	0.0061	0.0053
Prompt Neutron Lifetime (sec)	41.89	19.93	24.4	41.89	21.7	24.6
Net Rod Worth %	2.90	-2.90	-5.31	-2.90	-4.35	-5.07

* Nominal values plus biasing 20% in the BOC direction.

Table 3.2 Kinetics Parameters Utilized in This Analysis

	BOC	EOC	MC
Doppler Coefficient $(\Delta\rho/^{o}F) \times 10^{-5}$	-0.87 (least negative)	-2.11 (most negative)	-1.17
Moderator Temperature Coefficient $(\Delta\rho/^{o}F) \times 10^{-4}$	+0.5 (least negative)	-3.5 (most negative)	-2.03
Moderator Pressure Coefficient $(\Delta\rho/PSIA) \times 10^{-6}$	-1.00 (most negative)	+7.00 (most negative)	+4.06
Delayed Neutron Fraction	0.0075 (highest)	0.0045 (lowest)	0.0064
Prompt Neutron Lifetime (sec)	41.89	19.93	41.89
Net Rod Worth % $\Delta\rho$	-2.90	-2.90	-2.90

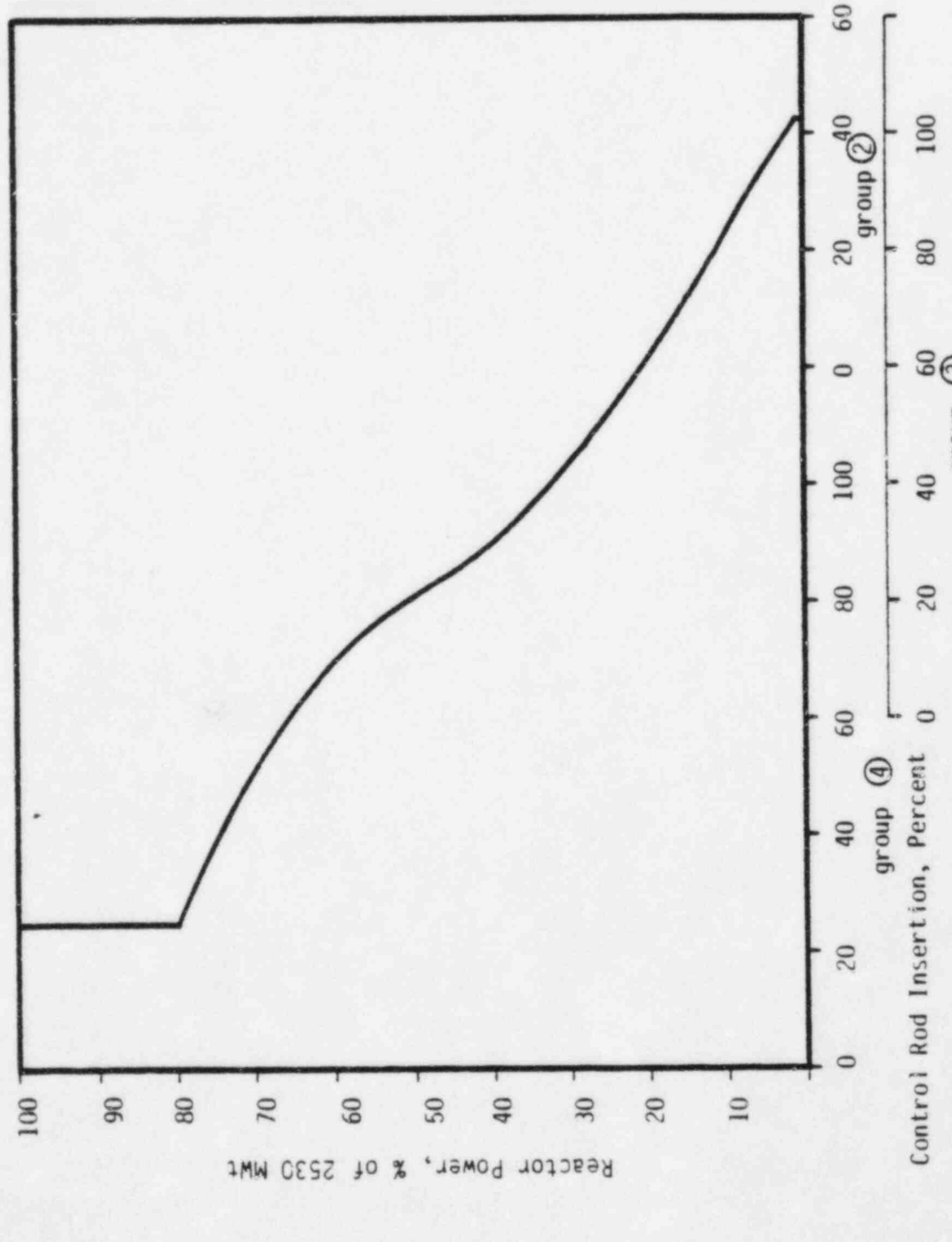


Figure 3.1 Power Dependent Insertion Limits (PDILs)(3)

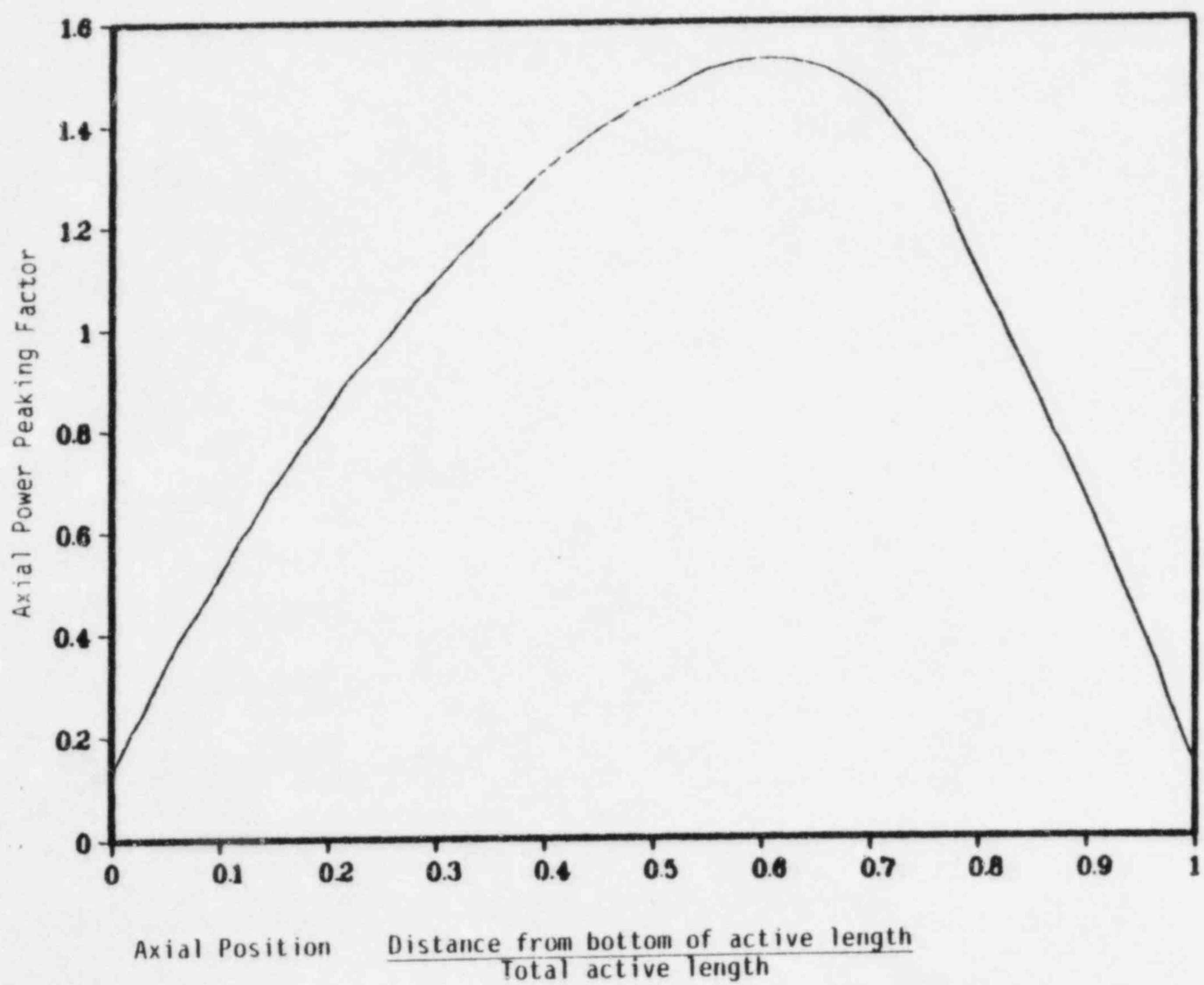


Figure 3.2 Axial Power Profile for Transients Initiated from 102% of Rated Power

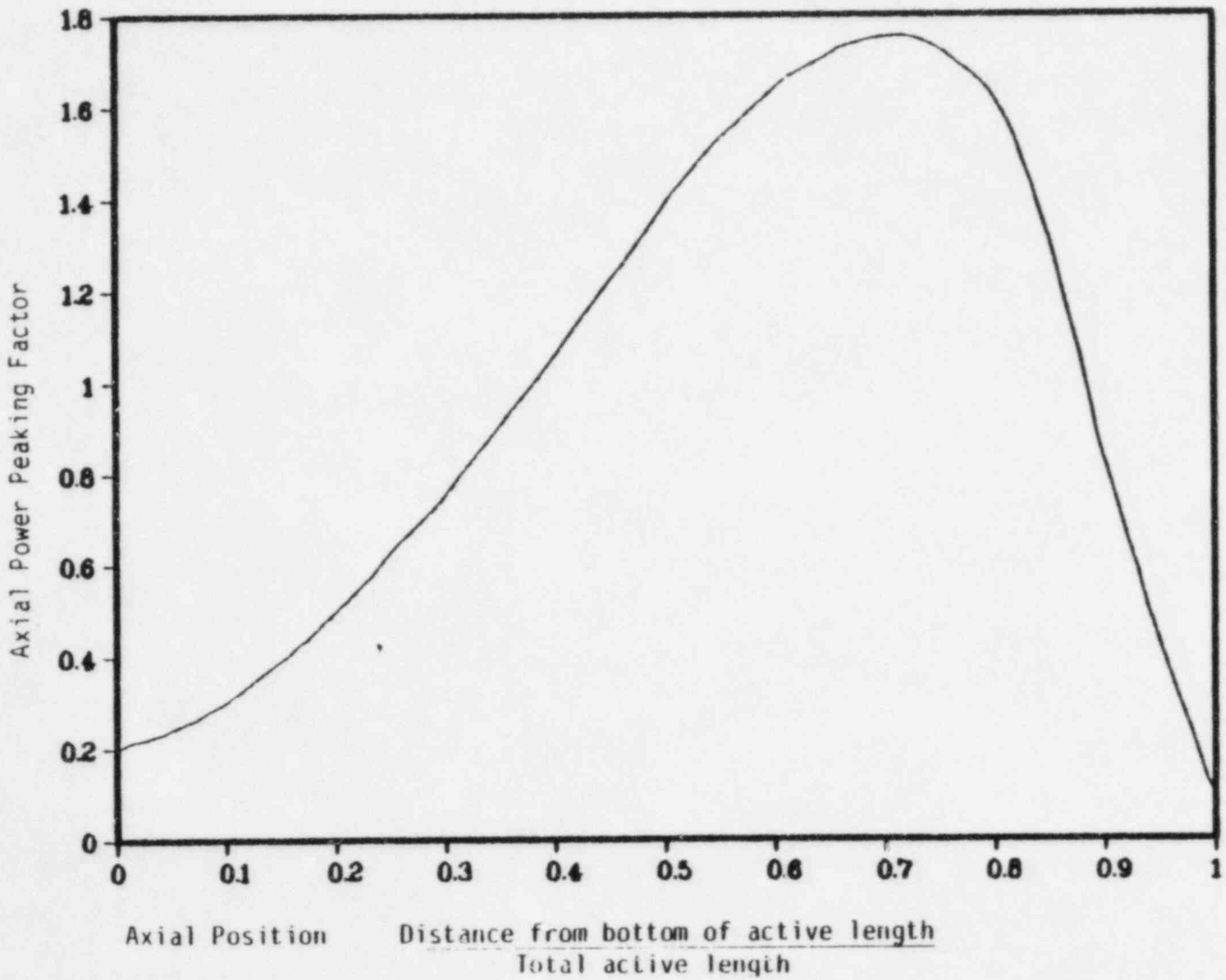


Figure 3.3 Axial Power Profile for Transients Initiated from 52% of Rated Power

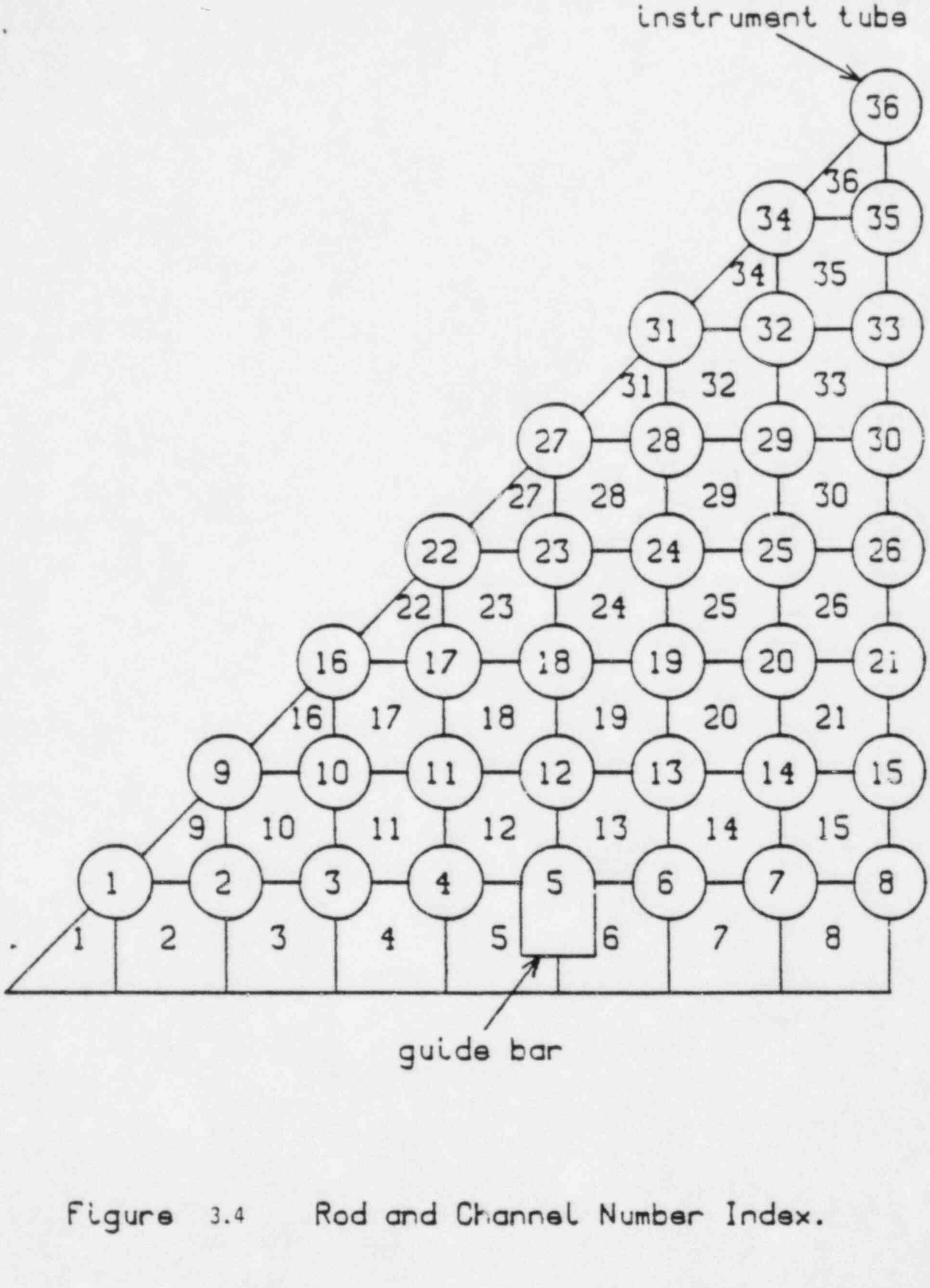
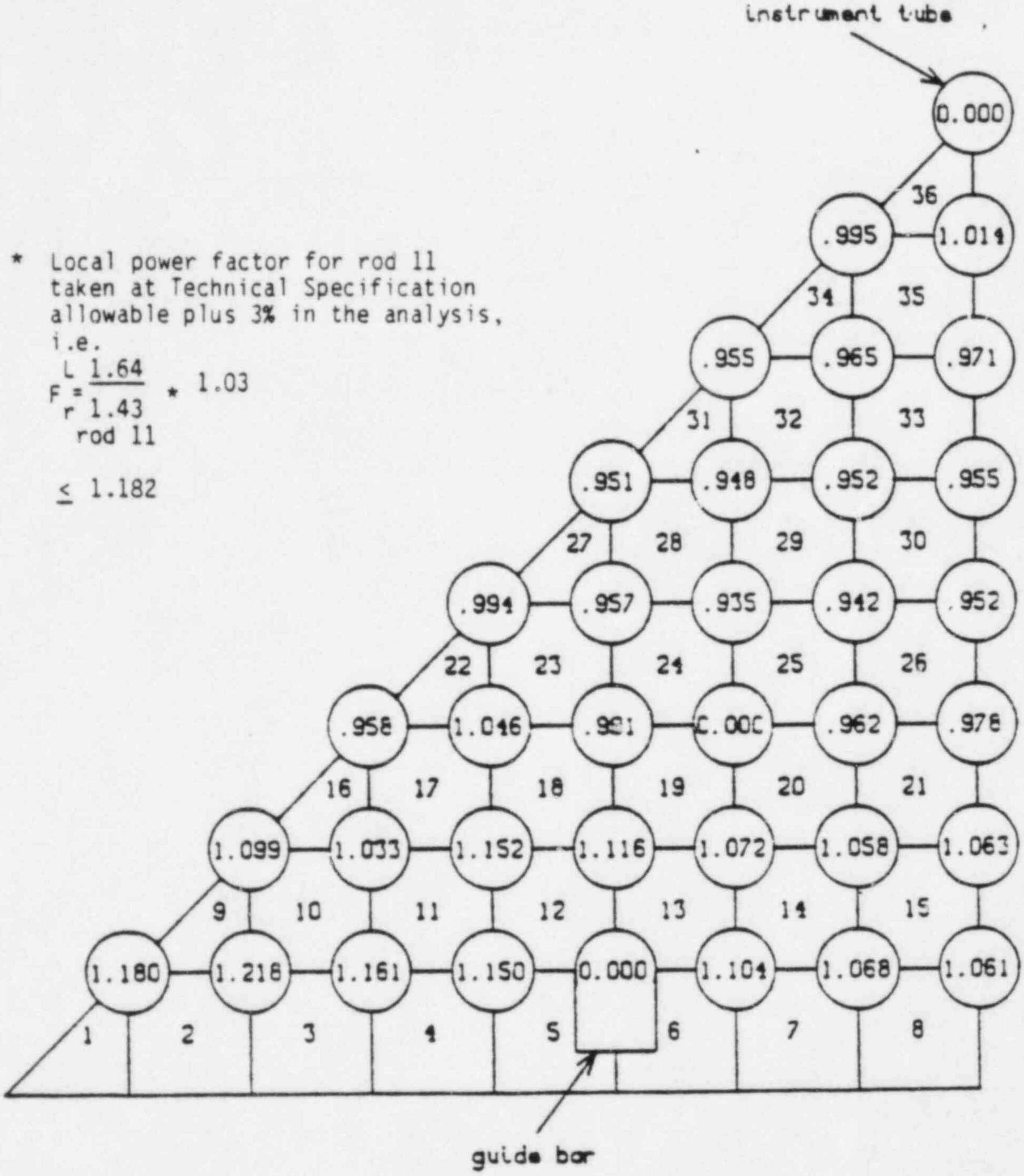


Figure 3.4 Rod and Channel Number Index.



* Local power factor for rod 11 taken at Technical Specification allowable plus 3% in the analysis, i.e.

$$F^{L}_{r\ \text{rod 11}} = \frac{1.64}{1.43} * 1.03$$

$\leq$ 1.182

Figure 3.5 Local Rod Power Distribution.

4.0 TRANSIENT ANALYSIS RESULTS

The transient analysis was conducted at two initial power levels, 52% and 102% of rated. Control rod withdrawal transients initiated from other power levels were not analyzed since previous analysis found that the above bounded the problem.[4]

4.1 CONTROL ROD WITHDRAWALS FROM 52% OF RATED POWER

Uncontrolled rod bank withdrawals initiated from 52% of rated power were conducted over the same rate of reactivity insertion rates utilized in the 1977 analysis (i.e. 1.0×10^{-5} thru $6.0 \times 10^{-4}\ \Delta\rho$/sec). The analysis was conducted for the BOC, EOC, and MC sets of kinetics previously discussed. Total rod banks worth was limited to 1.5% for the BOC and EOC kinetics cases. For the MC case DNB criteria could not be met with this total worth value. A review of the total worth of the sum of rod banks 3 and 4 inserted to their Technical Specification allowed power dependent insertion limits (PDILS)[2] indicated that the total worth based on the PDIL restriction would be less than 1.0% $\Delta\rho$. This value was utilized for the MC cases.

The axial power distribution utilized in the 1977 analysis for part power conditions was used for the 52% power case and held constant during the transient. This distribution is illustrated in Figure 3.2. The assembly radial peaking factor utilized for the 52% power case was based on the Technical Specification[2] allowable for power levels above 50% of rated:

$$F_r^A\ (P) = F_r^A\ (100\%) \times (1. + .5\ (1. - P))$$

$$F_r^A\ (50\%) = F_r^A\ (100\%) \times (1. + .5\ (1. - P))$$

F_r^A (50%) = 1.7875

F_r^A (P) = Assembly radial peaking factor at P > 50% of rated

P = Power level in fraction of rated. (A value of 50% used here, since the 52% value includes 2% for uncertainty)

F_r^A (100%) = Technical Specification allowed value for 208 rod assembly at full power 1.43

This F_r^A value, 1.7875, was held constant during the transient to insure conservatism.

The results for the 52% power case are illustrated in Figure 2.1 which plots minimum calculated DNB ratio versus reactivity addition rate. The minimum MDNBR condition for all transients considered occurred for the MC kinetics case for reactivity addition rates less than $3 \times 10^{-5} \Delta\rho$/sec. The MDNBR values graphed are from PTSPWR2 hot channel calculations. A more detailed analysis of local fluid conditions using XCOBRA-IIIC and the assembly boundary conditions established with PTSPWR2 at the most limiting point in time of the transient result in a more accurate MDNBR value of 1.40. Plant response for the MC kinetics cases which resulted in minimum DNBR values for this analysis are typified in Figures 4.1 thru 4.5 for a reactivity addition rate of $2.5 \times 10^{-5} \Delta\rho$/sec.

Sample plant system response during a rod withdrawal transient from 52% power are illustrated in Figures 4.6 thru 4.10 for a reactivity addition rate of $6.0 \times 10^{-5} \Delta\rho$ /sec and EOC kinetics.

All BOC, EOC, and MC kinetics cases initiated from 52% of rated power with high to intermediate reactivity addition rates terminate on the over power neutron flux trip or the high pressure trip. For very high reactivity addition rates, power increases rapidly and the temperature and pressure increase of the primary system lag in time such that the overpower trip is reached before the TM/LP or high pressure trip.

For the BOC kinetics and low reactivity addition rates, transients also trip on the high pressure trip. The high pressure trip results under the following circumstances. During reactivity additions via control rod withdrawal, power increases, and primary system temperature increases. Water in the primary system expands and flows into the pressurizer, reducing the volume available for the steam thereby acting to compress the steam and increase the pressure. As the pressure reaches certain set points, the primary pressurizer sprays are actuated. Water from the primary system is sprayed into the steam volume condensing part of the steam thereby acting to stabilize the pressure. If the rate of power increase, primary system temperature increase, and resultant flow of water into the pressurizer continues at a high enough rate for a sustained time period, the capacity of the sprays to stabilize pressure will be exceeded. Pressure will increase rather than stabilize and a high pressurizer trip will result provided the over power trip is not reached first.

For transients initiated from 52% of rated power, the MC kinetics cases at low reactivity rates result in lower DNBR values than the EOC cases due to the following. For maximum negative feedback conditions (EOC) and low reactivity addition rates, pressurizer spray capacity is adequate to stabilize primary system pressure. However, power and temperatures in the primary system continue to rise until the rod banks are fully withdrawn. After this point power and system temperature, etc. will stabilize provided a TM/LP trip set point is not reached. No reactor trip set point is reached for EOC kinetics and low reactivity addition rates.

For the MC kinetics cases and low rectivity addition rates, higher equilibrium power and temperature conditions would be obtained for the same amount of total rod bank worth withdrawn compared to the EOC cases provided a trip set point is not reached. The MC kinetics parameters produce less negative feedback than the EOC parameters and therefore higher power levels are achieved at equilibrium for the MC kinetics cases. System pressure would be similar for both cases since it would be within the range of the pressurizer spray controller. For the MC cases, a total rod bank worth of 1% is withdrawn prior to reaching the TM/LP trip, however, a slight reduction in primary system pressure occurred as the reactor system entered the stabilization phase and resulted in a TM/LP trip. Since the MC cases were at a higher resultant power and primary system temperature than the EOC cases for about the same pressure, lower DNBR values were calculated for the MC case. If the TM/LP trip had not occurred, the calculated DNBR value in this MC case would have been similar to the tripped MC case.

4.2 CONTROL ROD WITHDRAWALS FROM 102% OF RATED POWER

The uncontrolled rod bank withdrawal initiated from 102% of rated power was analyzed for BOC, EOC and MC kinetics. No limits were placed on rod bank worths for the 102% power cases. The range of reactivity addition rates utilized was the same as the 1977 analysis (i.e. 1.0×10^{-5} to 3×10^{-4} $\Delta\rho$/sec). Pressurizer sprays were included in the analysis to minimize pressure increases in the primary system and thereby minimize MDNBR values. The results of the analysis are presented graphically in Figure 2.2. Plotted is the minimum DNB ratio versus reactivity addition rates for the three different sets of kinetics. In all cases the reactor trip system functioned to terminate the transient as noted in the figure. Sample reactor system transient response characteristics for a reactivity insertion rate of 3.0 x 10^{-5} $\Delta\rho$/sec with EOC kinetics are illustrated in Figures 4.11 through 4.15.

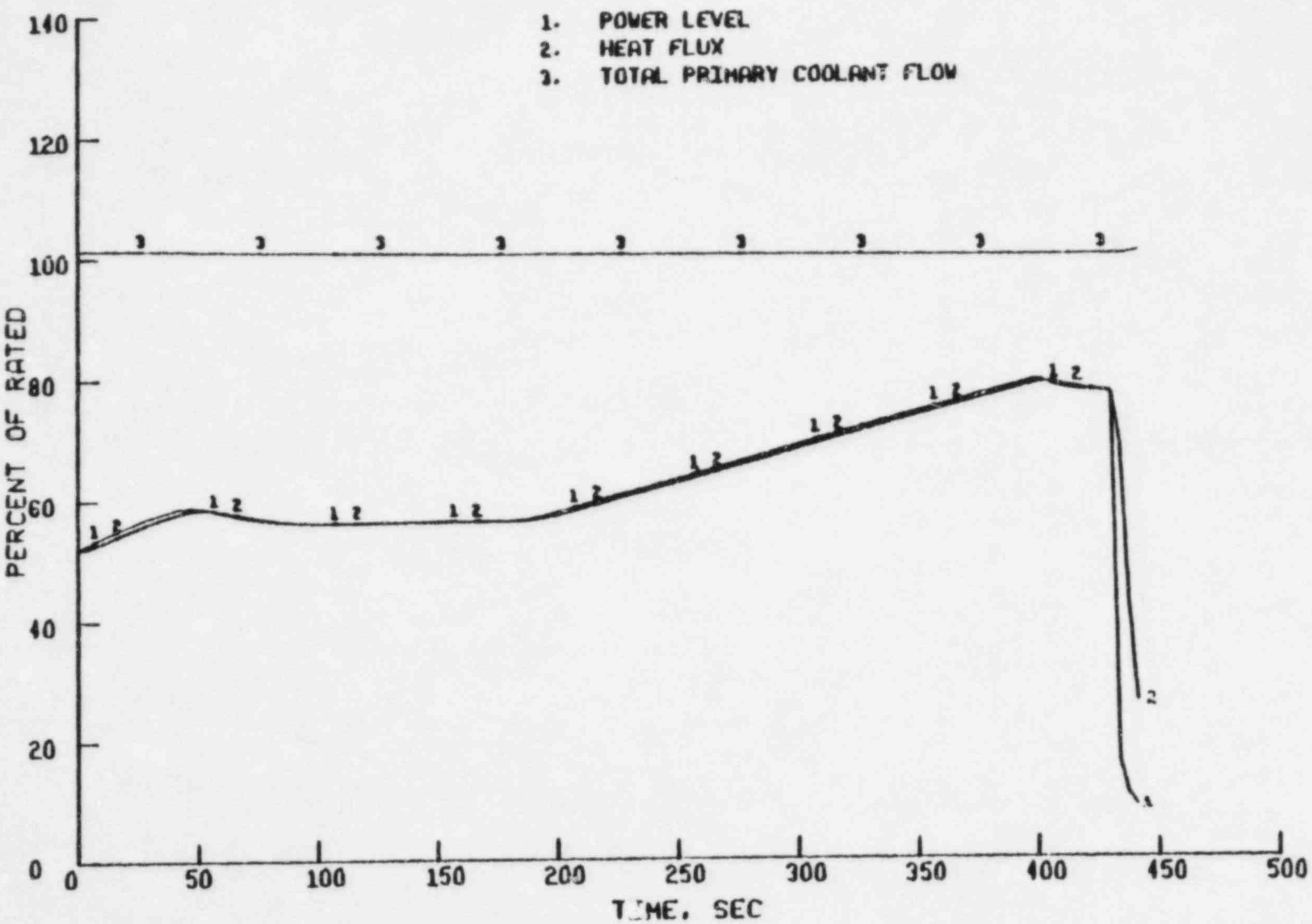


Figure 4.1 Power, Heat Flux, and Primary Flow - 52% Power, MC Kinetics, $2.5*10^{-5}$ $\Delta\rho$/sec

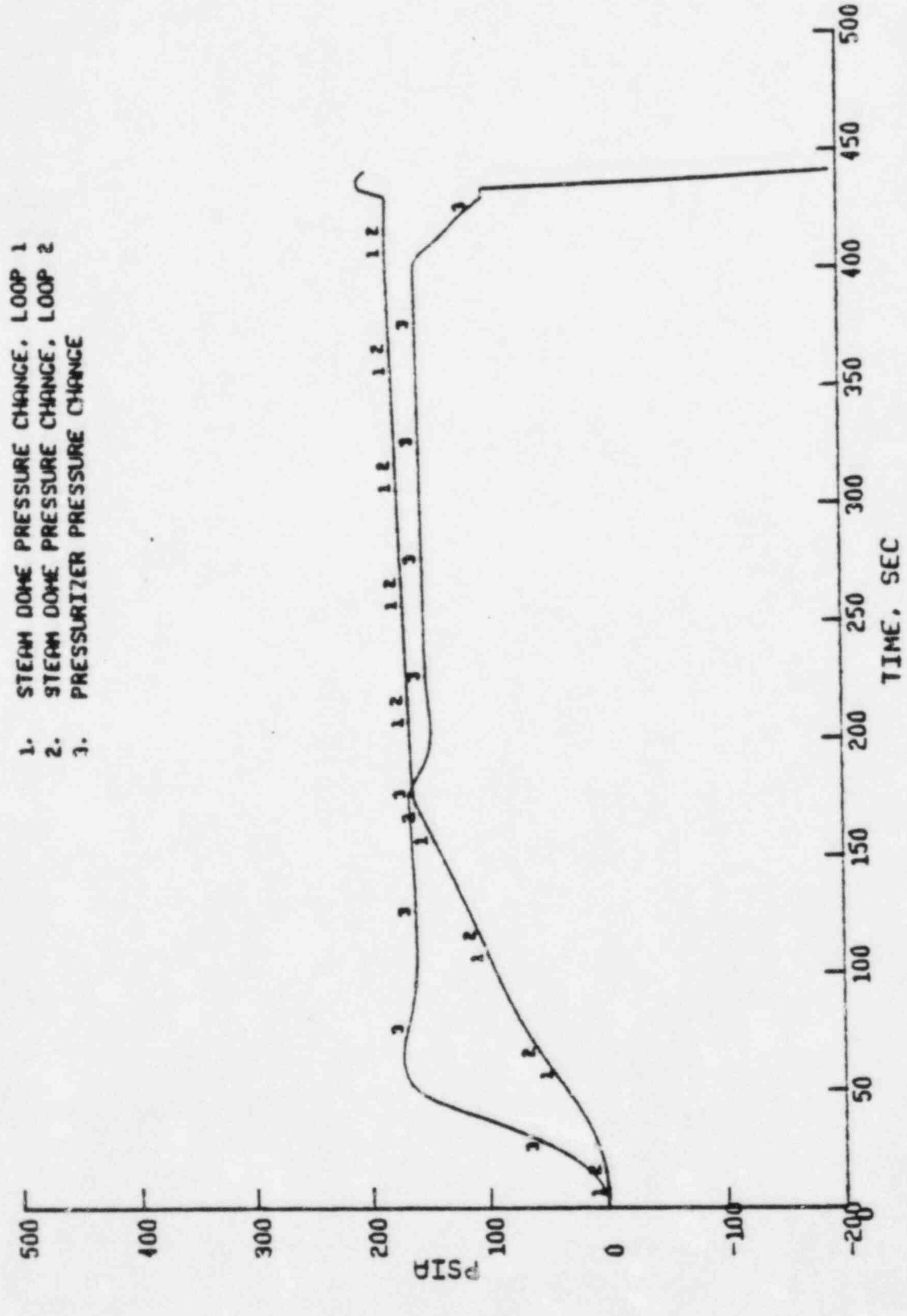


Figure 4.2 Steam Dome and Pressurizer Pressure Change - 52% Power, MC Kinetics, $2.5*10^{-5}$ $\Delta\rho$/sec

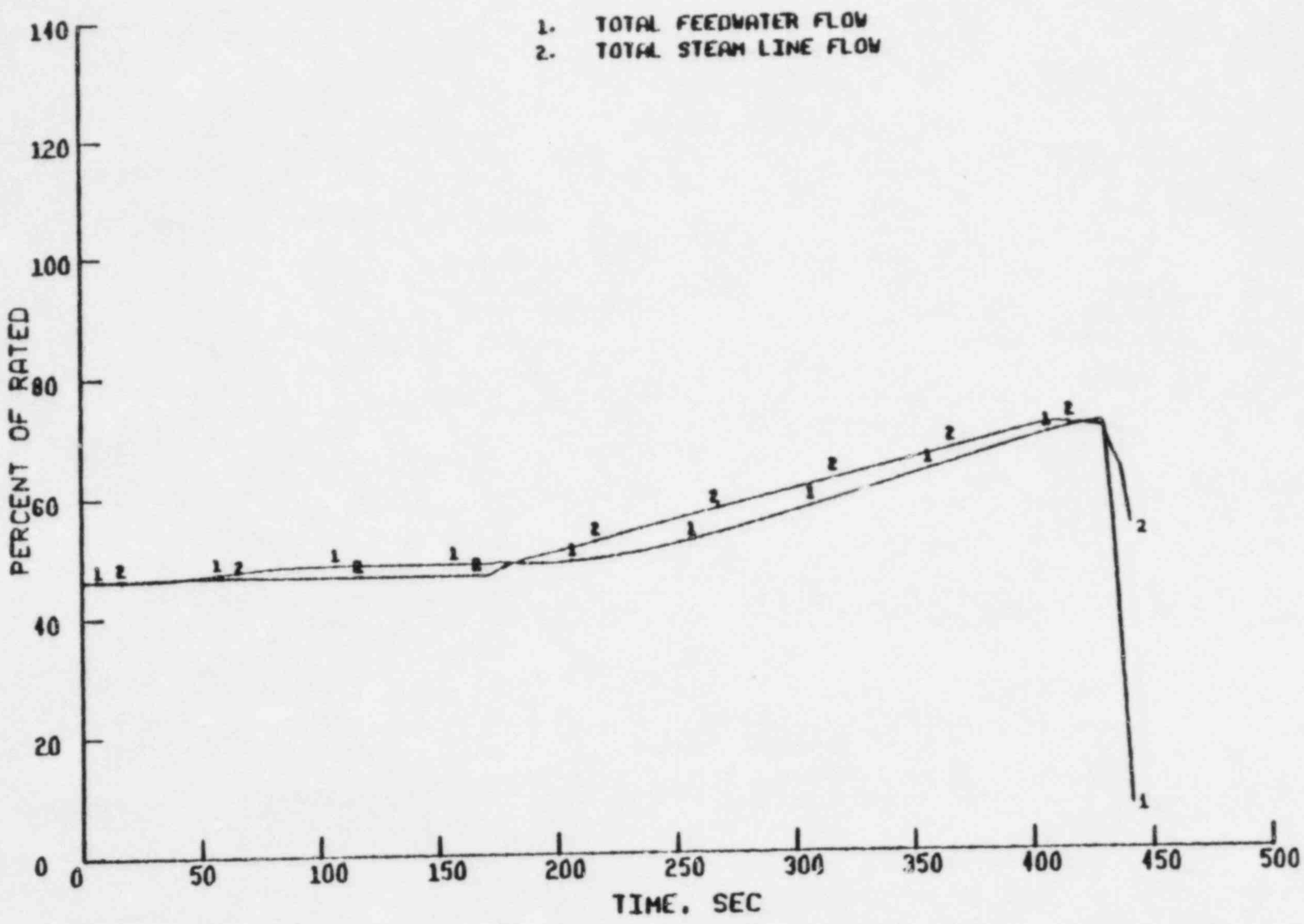


Figure 4.3 Feedwater and Steamline Flow - 52% Power, MC Kinetics, $2.5*10^{-5}$ $\Delta\rho$/sec

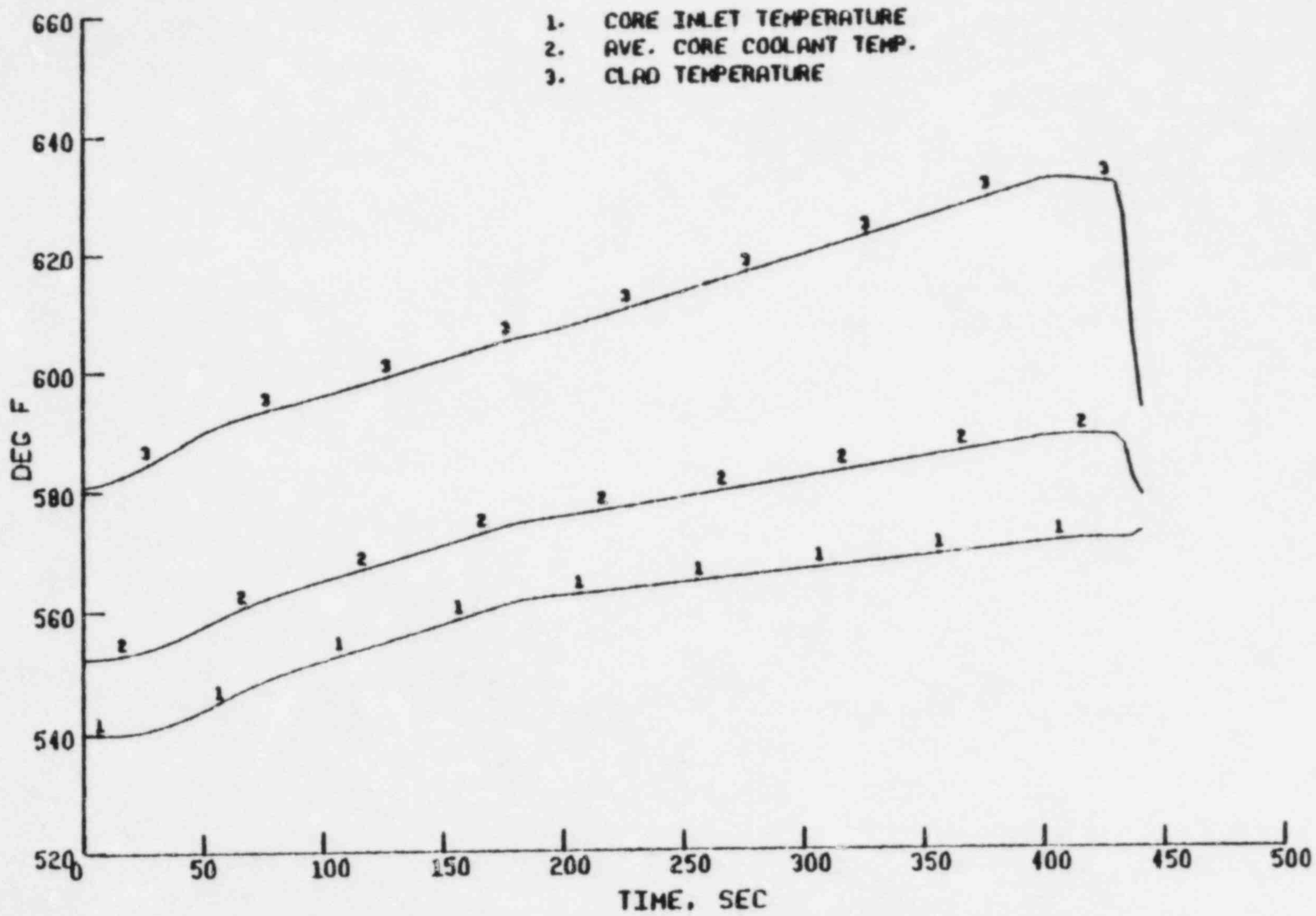


Figure 4.4 Core Inlet, Core Coolant, and Clad Temperature
52% Power, MC Kinetics, $2.5*10^{-5}$ $\Delta\rho$/sec

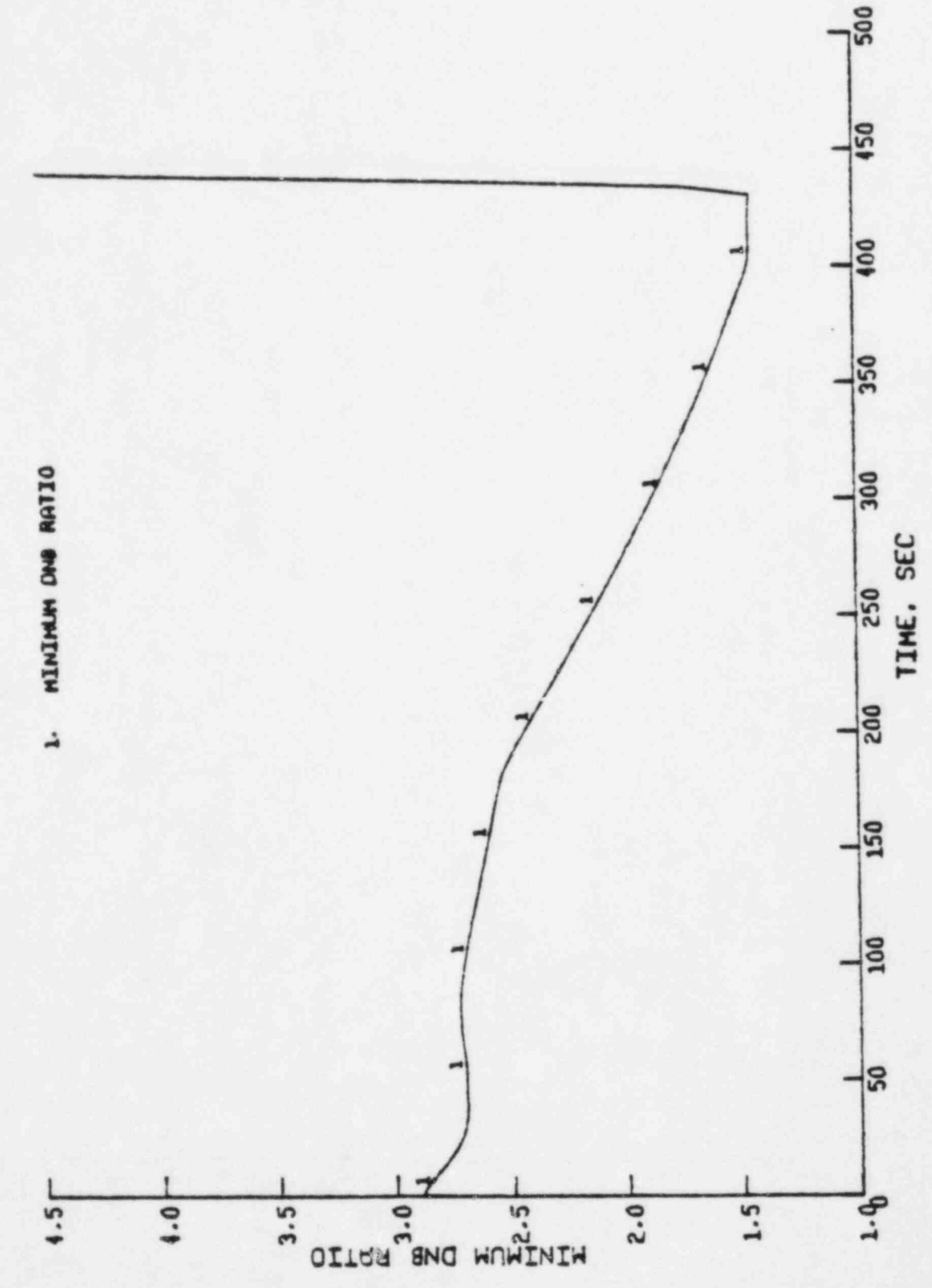


Figure 4.5 MDNBR - 52% Power, MC Kinetics, $2.5*10^{-5}$ $\Delta\rho$/sec

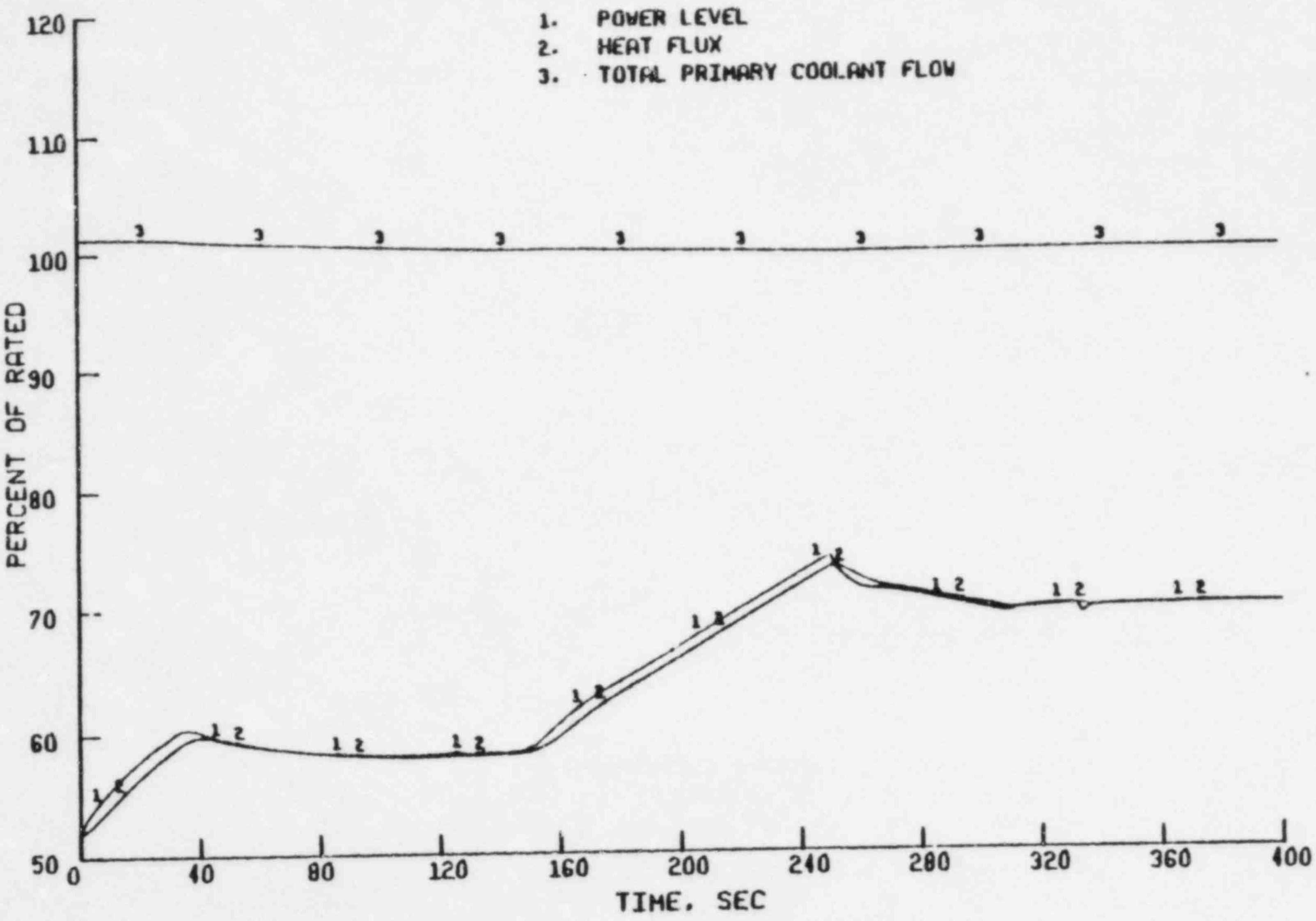


Figure 4.6 Power, Heat Flux, and Primary Flow - 52% Power, EOC Kinetics, $6 \times 10^{-5}\ \Delta\rho$/sec

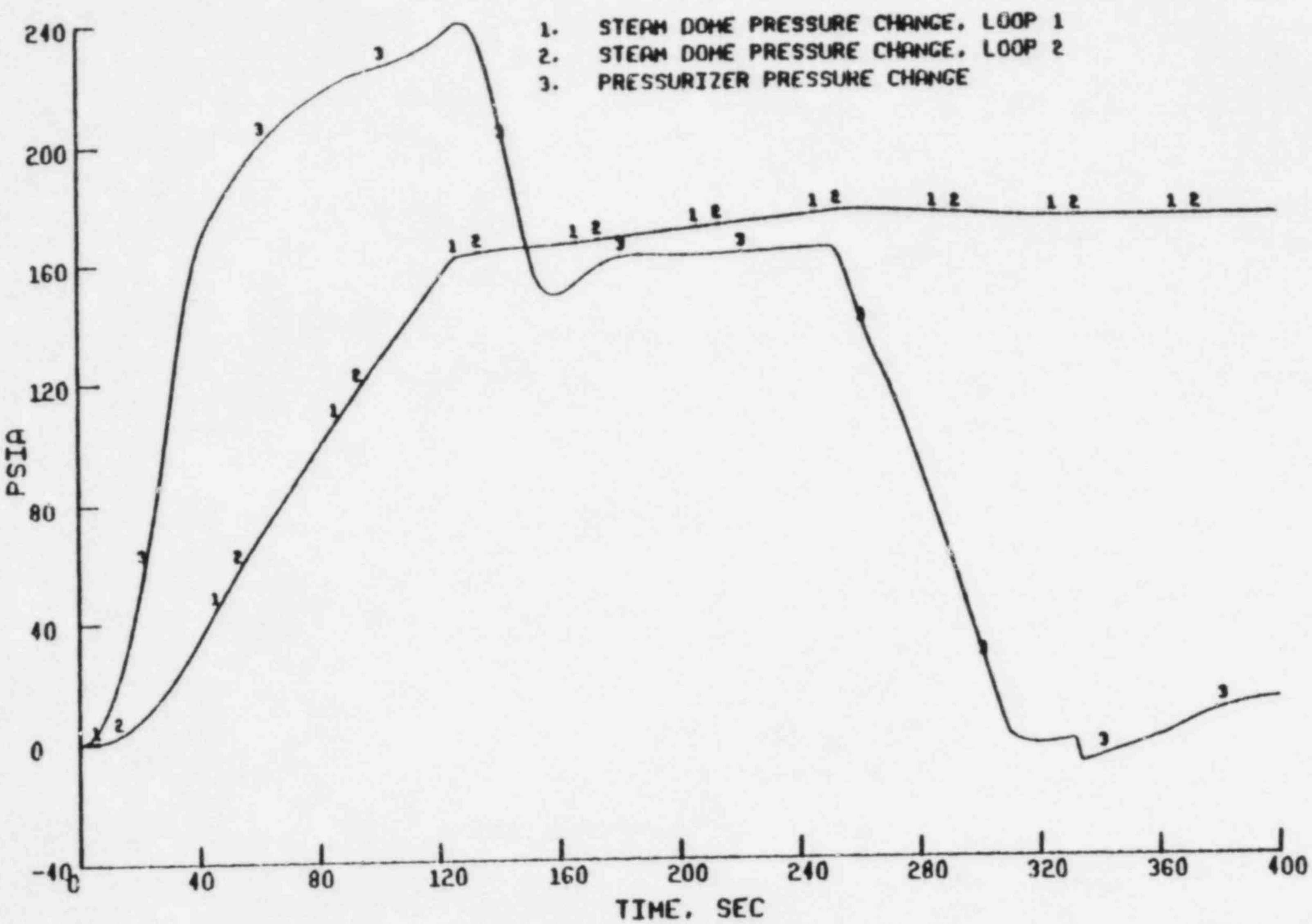


Figure 4.7 Steam Dome and Pressurizer Pressure Change - 52% Power, EOC Kinetics, 6 x 10^{-5} $\Delta\rho$/sec

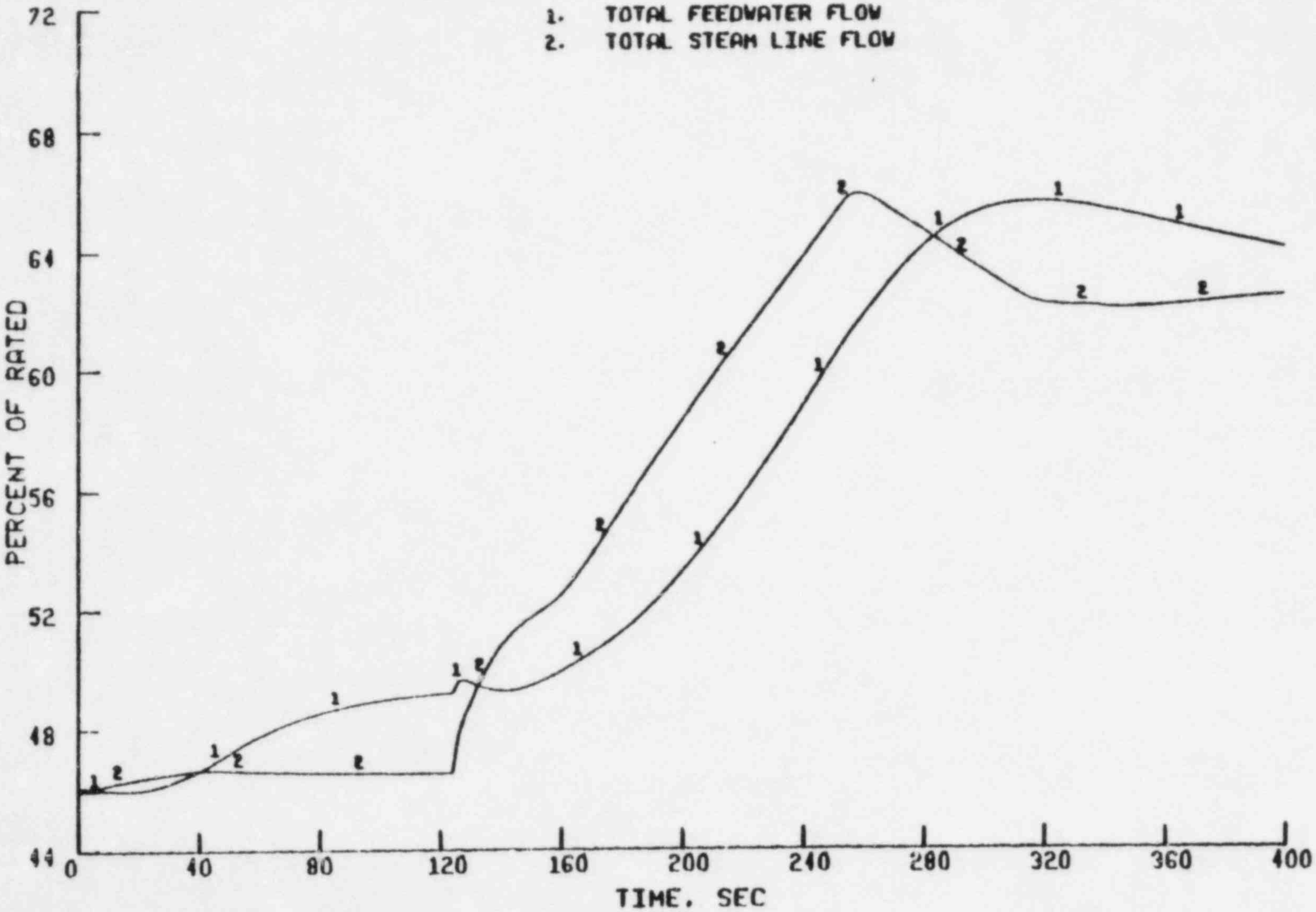


Figure 4.8 Feedwater and Steamline Flow -
52% Power, EOC Kinetics, $6 \times 10^{-5}\,\Delta\rho$/sec

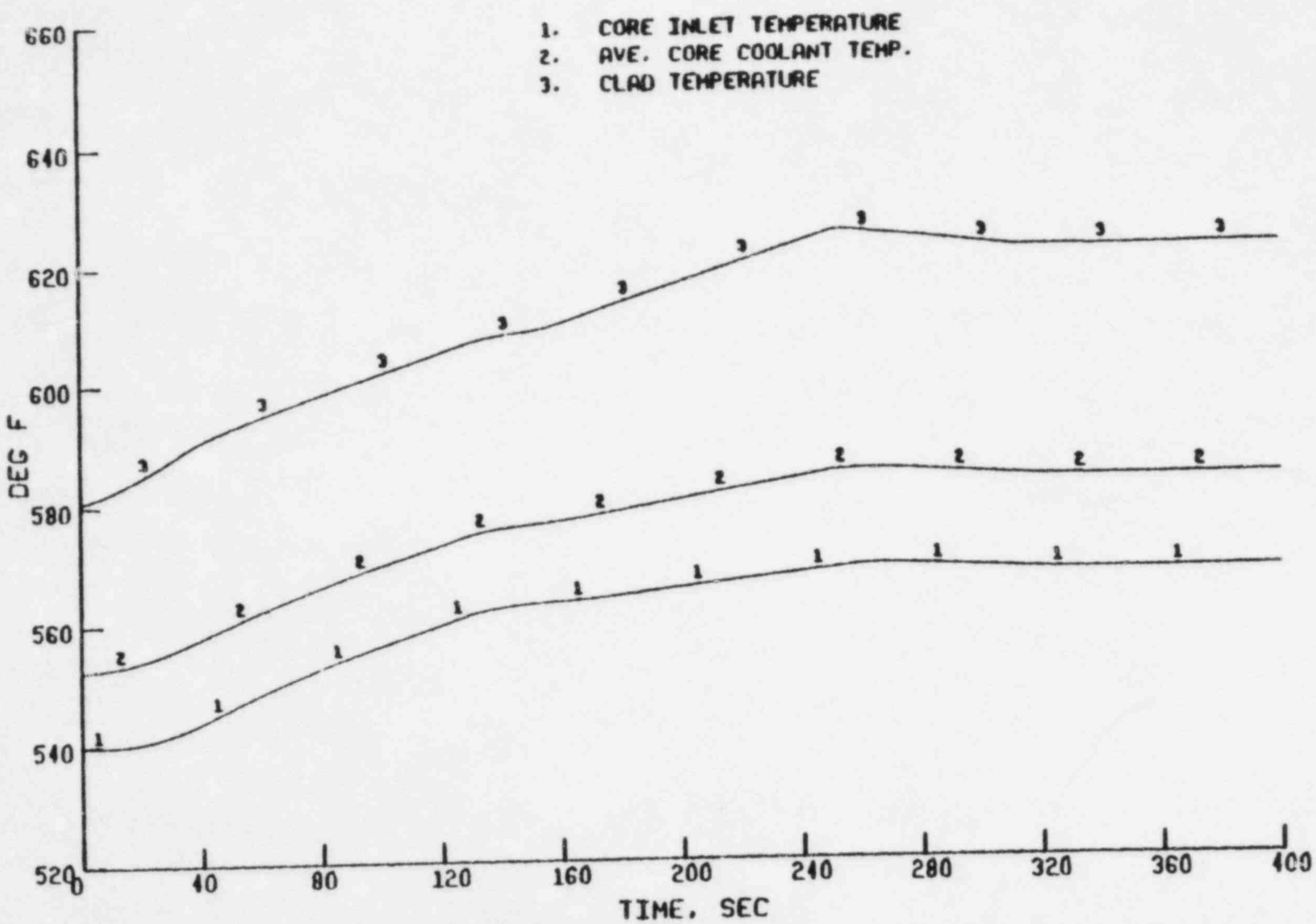


Figure 4.9 Core Inlet, Core Coolant, and Clad Temperature - 52% Power, EOC Kinetics, $6 \times 10^{-5} \Delta\rho$ /sec

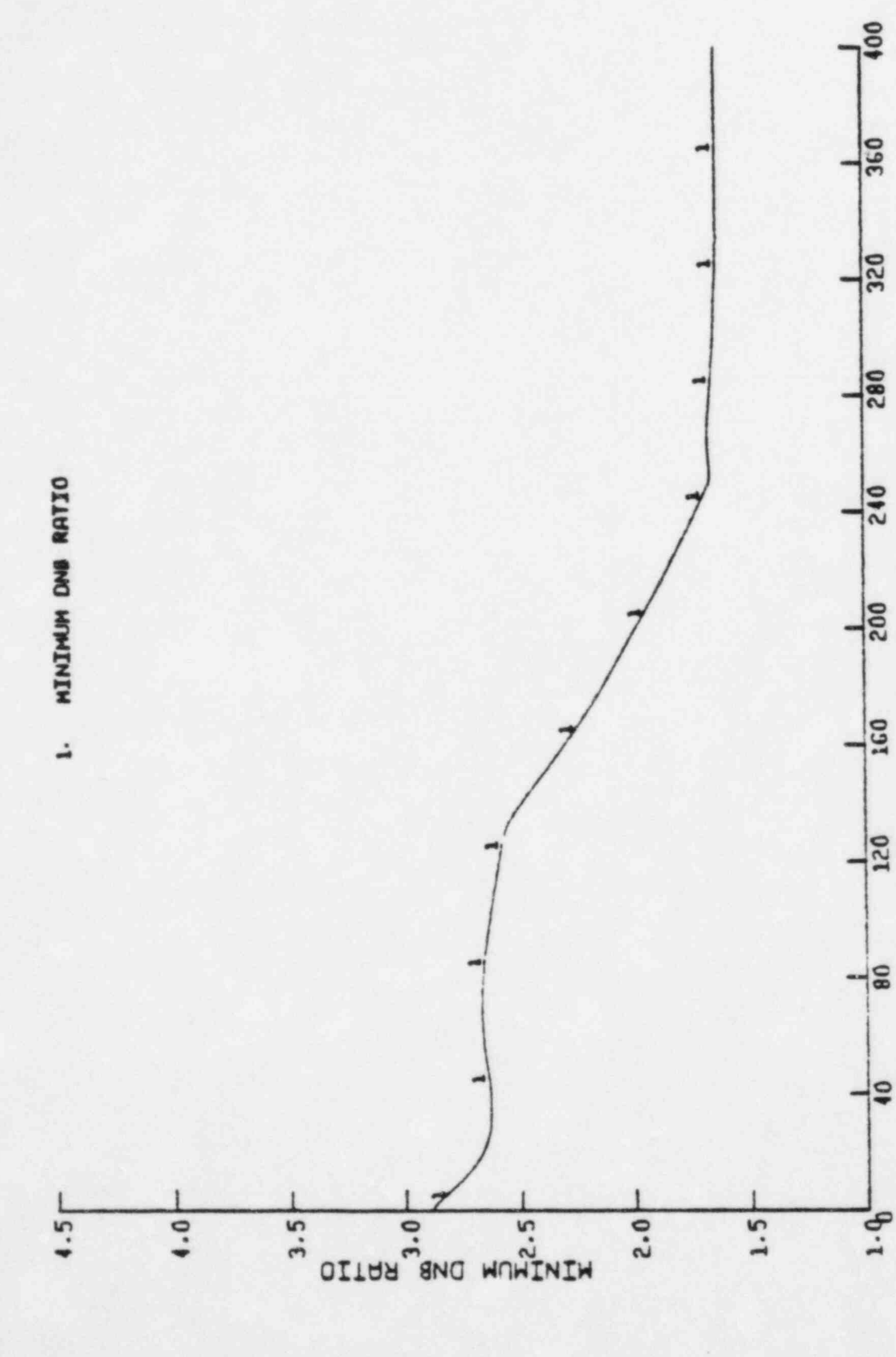


Figure 4.10 MDNBR - 52% Power, EOC Kinetics, 6×10^{-5} $\Delta\rho$/sec

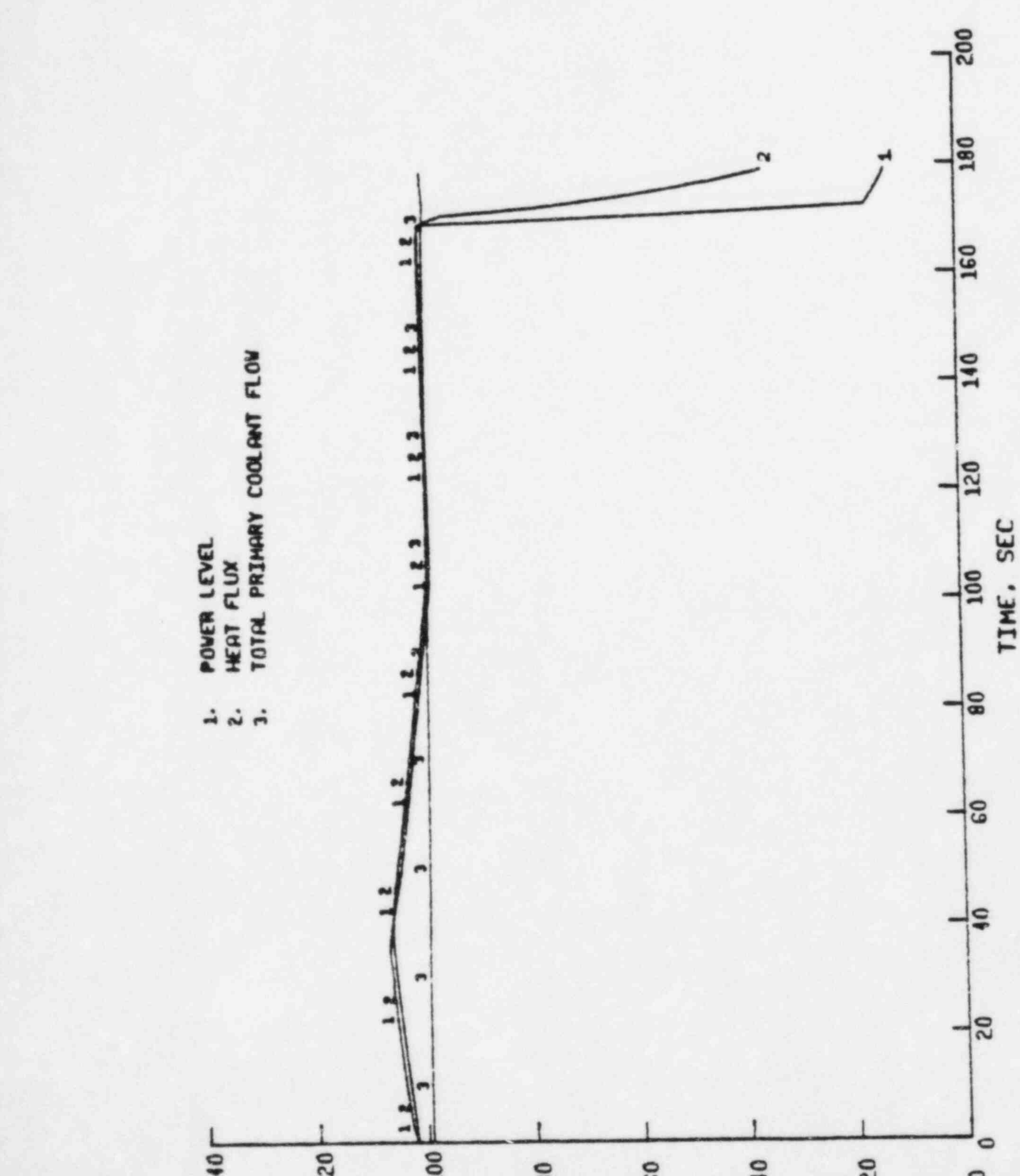


Figure 4.11 Power, Heat Flux, and Primary Flow - 102% Power, EOC Kinetics, 3×10^{-5} $\Delta\rho$/sec

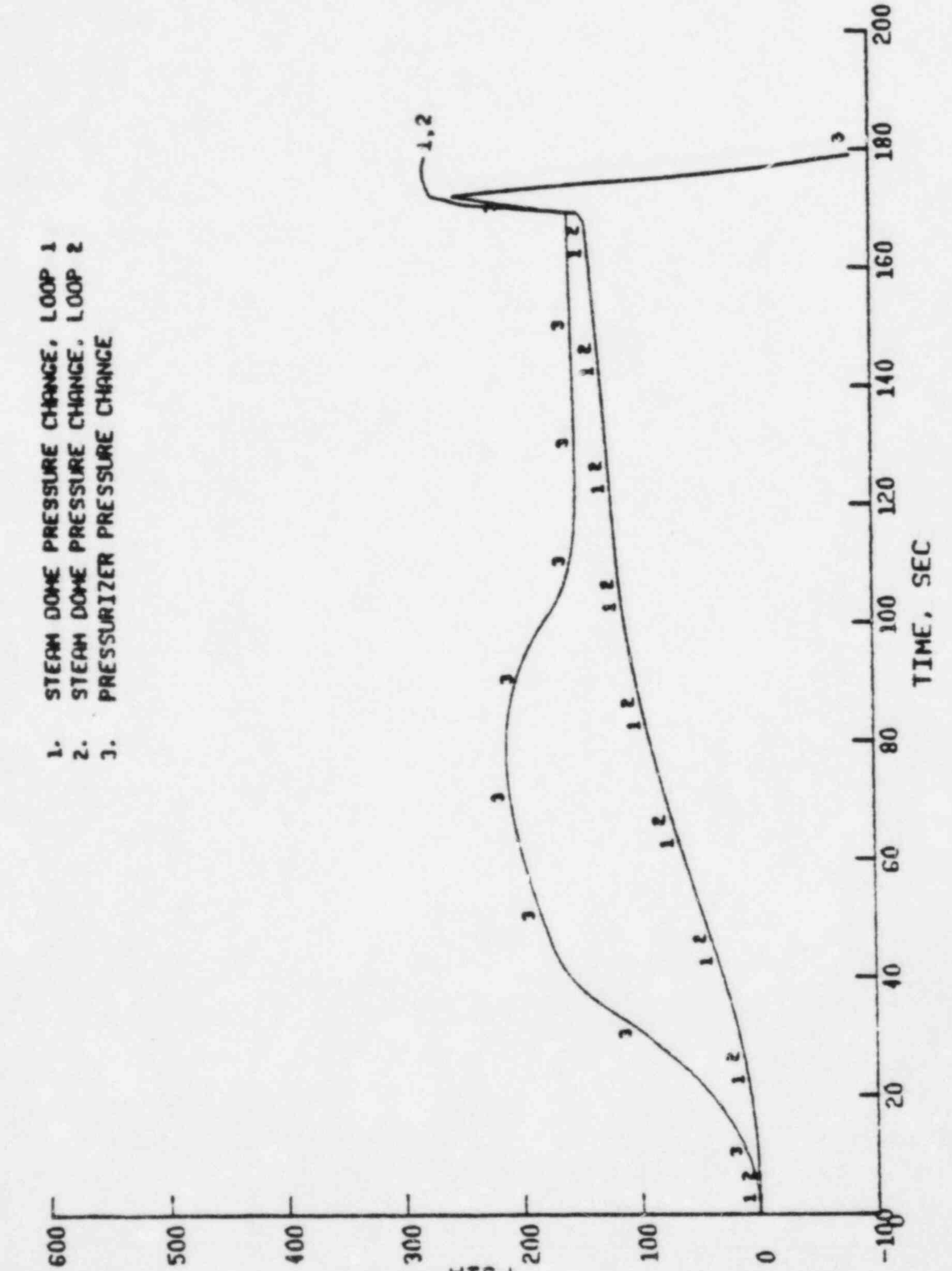


Figure 4.12 Steam Dome and Pressurizer Pressure Change - 102% Power, EOC Kinetics, 3×10^{-5} $\Delta\rho$/sec

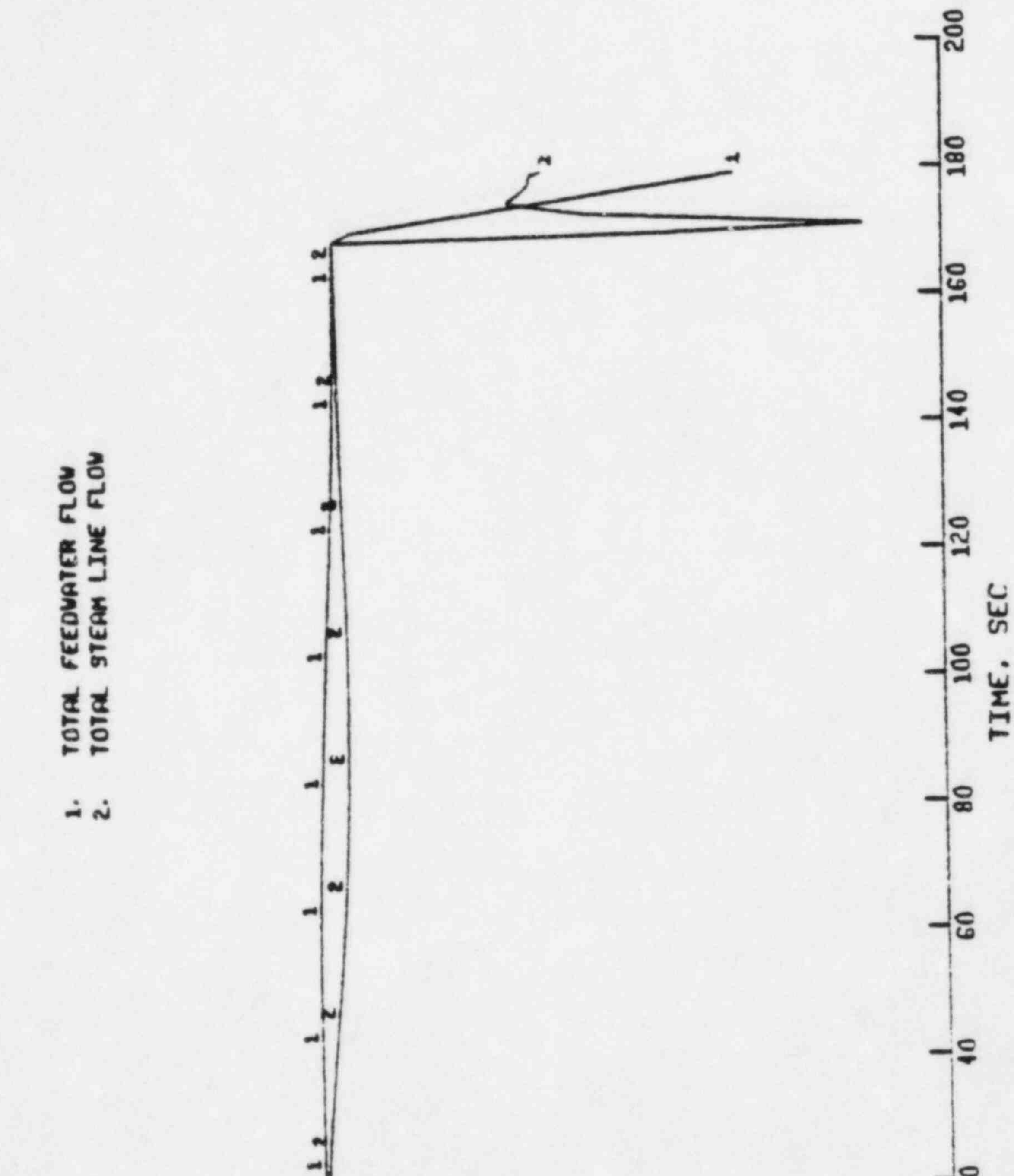


Figure 4.13 Feedwater and Steamline Flow - 102% Power, EOC Kinetics, 3×10^{-5} $\Delta\rho$/sec

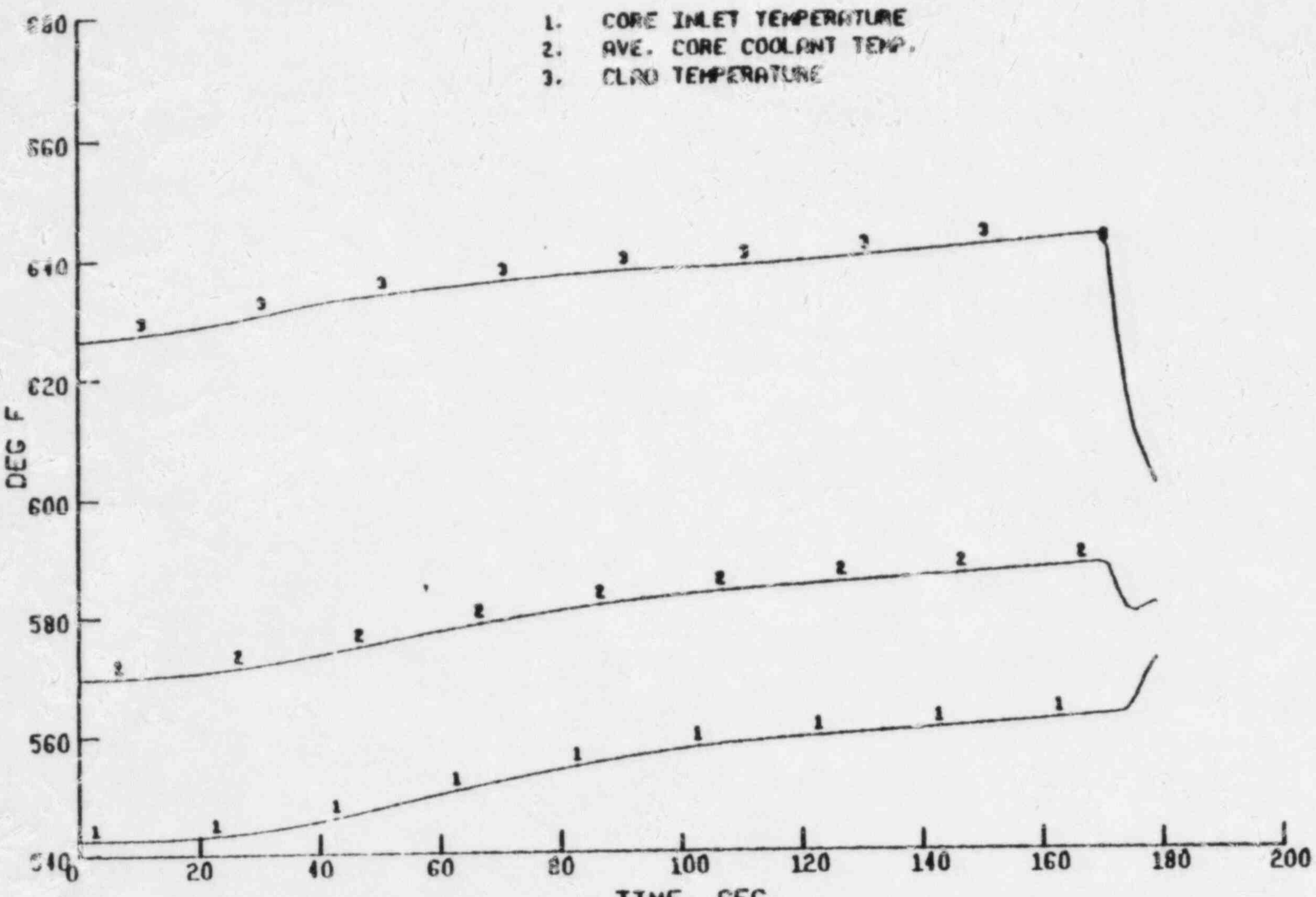


Figure 4.14 Core Inlet, Core Coolant, and Clad Temperature - 102% Power, EOC Kinetics, 3 x 10^{-5} $\Delta\rho$/sec

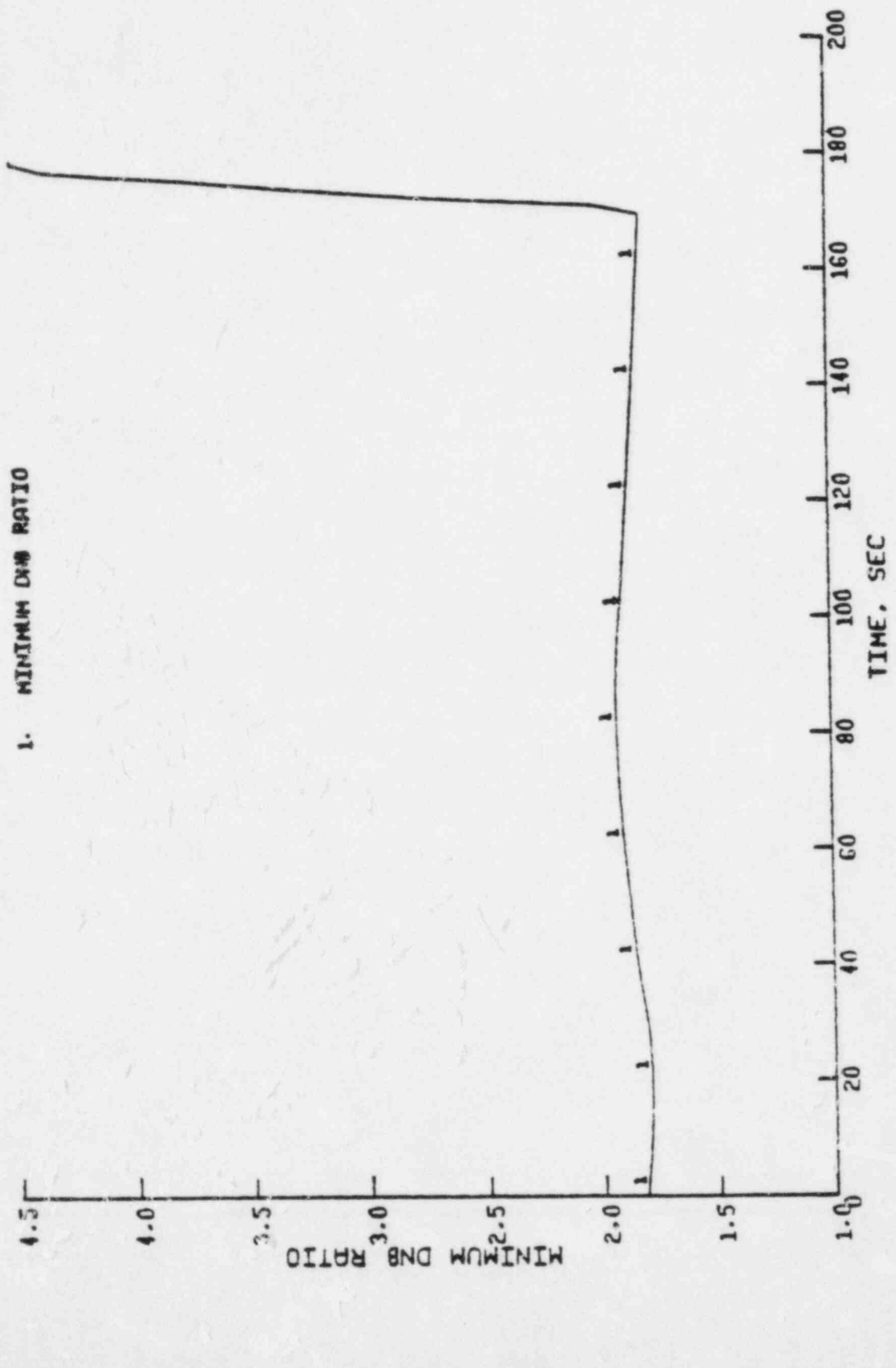


Figure 4.15 MDNBR - 102% Power, EOC Kinetics, 3×10^{-5} $\Delta\rho$/sec

5.0 CONCLUSIONS

It is concluded from this analysis that there is at least a 95% probability with a 95% confidence level that no fuel rod in the Palisades core will experience DNB during an uncontrolled rod bank withdrawal transient. The analysis also indicates that the new PTSPWR2 pressurizer model, the XNB correlation and coolant flow update more than compensate for the DNBR reducing effects of the RTD time response, and the more conservative assumptions regarding part power radial peaking factors used in this analysis compared to the 1977 analysis.

6.0 REFERENCES

1. Koester, G.E., et al., "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 Mwt", XN-NF-77-18, July, 1977.

2. Palisades Plant Technical Specification; Appendix A to Provisional Operating License DPR-20 for the Palisades Plant Consumers Power Company, Docket No. 50-255 - License DPR-20, Including Revisions and Ammendments thru June 13, 1983.

3. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors", (PTS-PWR), XN-74-5, Revision 2, August, 1983.

4. Palisades Plant Final Safety Analysis Report, Consumers Power Company.

5. Macduff, R. B., "Exxon Nuclear DNB Correlation for PWR Fuel Designs", XN-NF-621(P), Rev. 1, April, 1982.

6. Macduff, R.B., "Justification of XNB Correlation for Palisades", XN-NF-709, Rev. 0, 5/12/83.

7. Kelley, R.H., "Analysis of Axial Power Distribution limits for the Palisades Nuclear Reactor at 2530 Mwt", XN-NF-78-16 June, 1978.

8. Porten, D.R. et. al, "Palisades Cycle 6 Safety Analysis Report", XN-NF-83-0-3, January 24, 1983.

Appendix A

A.0 COOLANT FLOW UPDATE

The method used to calculate primary coolant system (PCS) flow at power is described and compared to the previous method used in the 1977 analysis in this appendix. Primary system flow at power is a basic input parameter to DNBR calculations. An increase in mass flow rate will result in an increase in MDNBR. The new method results in an ~2% increase in primary system flow at rated power compared to the previous method.

Plant Technical Specifications(2) currently require that the primary system pumps circulate at least 126.9 x 10^6 lbm/hr of water (when corrected to 532°F). Compliance measurements are taken at hot zero power (near 532°F) and corrected to a uniform primary system water temperature of 532°F. A comparison is then made to the Technical Specification value before bringing the reactor to power.

As the reactor is brought to power, temperature gradients are established in the primary loop water, and the temperature level at the pump may rise. The result will be a reduction in mass flow rate due both to decreases in water density at any given pump volumetric flow rate and increases in loop pressure drop, which must be in equilibrium with the pressure/drop flow characteristics of the primary system pumps. In addition, the density differences in vertical portions of loop components give rise to gravitational head differences which act to aid the pump. The resultant flow at power is calculated on the basis of the flow at hot zero power and corrections for the above effects. This flow becomes the basis for DNBR calculations.

The previous flow calculation method used in the 1977 analysis and a more accurate updated method used in this analyses are presented in the following.

A.1 PREVIOUS METHOD

The previous algorythm for calculating corrected primary coolant system mass flow rate was:

$$W = A - .185\ (T-532) - .00164\ P$$

where:

W = PCS mass flow rate, 10^6 lb/hr

A = measured nominal PCS mass flow rate at zero power and 532°F
= 126.9 used for this study, 10^6 lb/hr. Also used in previous analysis since it is minimum Tech Spec value allowed.

T = core coolant inlet temperature, °F

P = reactor power, MWt

The previous method assumed a constant head pump, excess conservatism in the impact of density changes on head loss, and no credit for gravitational head differences.

A.2 NEW METHOD

The approach used in this analysis will be described with the aid of Figure A.1. At hot zero power (HZP), the intersection of the pump curve and hydraulic loss curve establish operating point "A". For a fixed cold leg temperature, as the reactor is brought to full power, reductions in density in the steam generator, core and hot leg occur, resulting in a new hydraulic loss curve and a new operating point "B".

Increases in cold leg temperature at any fixed power level result in similar temperature increases throughout the primary system. No volumetric flow rate changes will occur. The mass flow rate will change in proportion to the density change at the cold leg (or pump).

Density differences over corresponding vertical components at full power act to assist the pump. When this additional driving force is added, operating point "C" is established. The approach taken in this analysis was to establish these operating points and thereby the flow at full power.

The calculation proceeded on a normalized basis as that approach eliminated the calculation of intermediate quantitative values. Pump volumetric flow and head at HZP were used as normalizing parameters. The Technical Specification minimum allowed flow at HZP was used for conservatism, together with the corresponding head taken from the plant pump curve. The following order was used.

1. Linearized and normalized pump curves were developed.
2. Normalized hydraulic head loss curves or functions were developed.
3. Operating point "B" was determined.
4. Normalized hydrostatic or gravity head functions were developed.
5. Operating point "C" was determined.

The following summarized relations are required to compute the mass flow rate at full power conditions based on conditions at hot zero power.

Solution for Point "B"

$$M_B = \left(\frac{\rho_{CL}}{\rho_{CL_{HZP}}}\right)\left(\frac{Q_B}{Q_{HZP}}\right) M_{HZP}$$

$$Q_B = \left[\frac{+ C + \sqrt{C^2 - 4 \frac{\rho_{CL}}{\rho_{AV}} (\overline{C} - 1)}}{2 \left(\frac{\rho_{CL}}{\rho_{AV}}\right)}\right]$$

$\overline{C}$ = Slope of normalized pump curve, normalized to HZP flow conditions, -.93882

ρ_{CL} = Cold leg water density at power

$\rho_{CL_{HZP}}$ = Cold leg water density at HZP

ρ_{HL} = Hot leg water density at power (taken at top of core average for conservatism)

ρ_{AV} = $\dfrac{\rho_{HL} + \rho_{CL}}{2}$

HZP = Hot zero power conditions corrected to 532° with pressure assumed at 2100 psia.

Q_{HZP} = Volumetric flow rate at HZP at the cold leg or pump

Q_B = Volumetric flow rate at full power, Point B, at the cold leg or pump

M_{HZP} = Mass flow rate at HZP = 126.9×10^6 lbm/hr.

M_B = Mass flow rate at B, lbm/hr.

Solution for Point "C"

$$M_C = \left(\frac{\rho_{CL}}{\rho_{CL_{HZP}}}\right) \left(\frac{Q_C}{Q_{HZP}}\right) M_{HZP}$$

$$\frac{Q_B}{Q_{HZP}} = \frac{Q_B}{Q_{HZP}} + d$$

$$d = \underline{b} \cos k$$

$$\underline{b} = \underline{c} \frac{\sin \beta}{\sin \gamma}$$

$$\underline{c} = \frac{(1 - \frac{\rho_{HL}}{\rho_{CL}})}{(\frac{\Delta P_{TOT}}{\rho_{CL}})_{HZP}} \left[h_{CL} + 1/2\ (h_{sg} + h_{core}) \right]$$

$$\gamma = 180 - \alpha - \beta$$

$$\beta = 90 - \eta$$

$$\eta = -TAN^{-1} (\overline{C})$$

$$\alpha = 90 - \kappa$$

$$\kappa = TAN^{-1} \left[2\left(\frac{\rho_{CL}}{\rho_{AV}}\right)\left(\frac{Q_B}{Q_{HZP}}\right) \right]$$

$\left(\frac{\Delta P_{TOT}}{CL}\right)_{HZP}$ = Pump head at HZP = 249.7 ft.

h_{CL} = Cold leg gravity head. Vertical distance from top of active core to plenum side of steam generator tube sheet - 10.725 ft.

h_{sg} = Steam generator gravity head. Vertical distance from steam generator tube sheet to top of tubes - 27.763 ft.

h_{core} = Core gravity head. Vertical distance across active core - 10.983 ft.

M_C = Mass flow rate at Point C, lbm/hr.

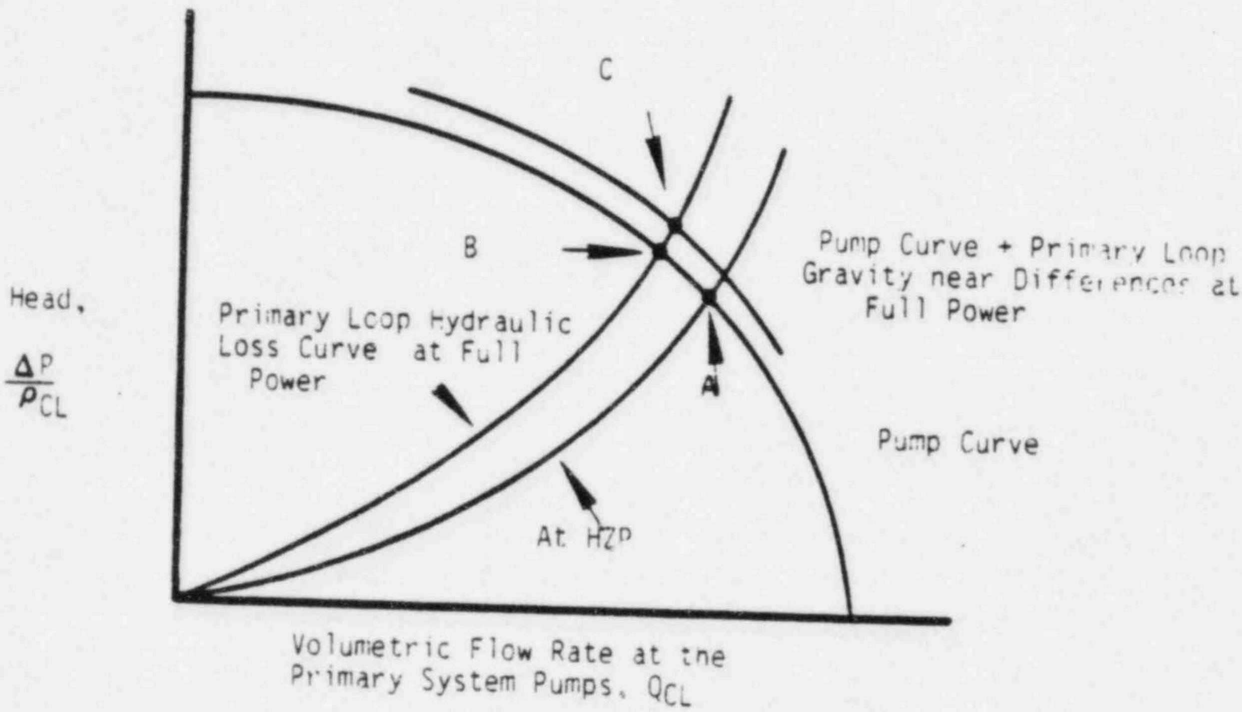


Figure A.1 Pump Curve, Gravity Head, and Hydraulic Loss Curve Schematic Utilized in Development of Coolant Flow Upgrade

XN-NF-83-57

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ROD WITHDRAWAL TRANSIENT

RE-ANALYSIS FOR THE

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