XN-NF-82-45

REVISION 1

# PLANT TRANSIENT ANALYSIS FOR OPERATION OF THE R.E. GINNA UNIT 1 NUCLEAR POWER PLANT AT REDUCED PRESSURE AND TEMPERATURE

JULY 1982

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

XN-NF-82-45

Revision 1

ISSUE DATE:
07/22/82

# PLANT TRANSIENT ANALYSIS FOR OPERATION OF THE R.E. GINNA UNIT 1 NUCLEAR POWER PLANT AT REDUCED PRESSURE AND TEMPERATURE


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## 1.0 INTRODUCTION AND SUMMARY

This document presents plant transient analysis to support operation of the R.E. Ginna Unit 1 nuclear power plant with reduced primary coolant temperature and pressure. Specifically this document supports a full load (1520 MWt) $T_{AVE}$ reduction from 573.5°F to 558.5°F and a primary system pressure reduction from 2250 psia to 2000 psia. The results of the analysis show generally greater thermal margins during limiting transient events for the new conditions than for normal temperature and pressure operation. This improvement in margin is because the 15°F reduced temperature condition outweighs the slight adverse effect on DNBR of reduced primary system pressure. In performing the analysis reductions in primary temperature of 15°F to 50°F were considered. Thermal margin results for 15°F reduced temperature were found to bound the results for larger reductions in primary coolant temperature.

The reference $T_{AVE}$ and pressure inputs into the calculated overtemperature $\Delta T$ and overpower $\Delta T$ trip functions were changed to reflect reduced temperature and pressure operation in order to maintain the steady state margin to trip and so that improvements in the initial DNBR also applied during transients that were protected by these two trip functions.

Plant transient analyses for ENC reload fuel at R.E. Ginna Unit 1 are documented in References 1 and 2. These analyses covered normal temperature and pressure operation. Substantial margin improvements for 50°F reduced temperature and 2000 psia pressure operation were shown in Reference 3. The present analysis supports operation with 15°F to 50°F reduced temperature operation at 2000 psia and hence bounds the analysis of Reference 3.

As in the case of prior analyses, the present analysis was performed using the Exxon Nuclear plant transient simulation code, PTSPWR2[4] with supporting subchannel analysis using the standard ENC methodology[5]. In order to support operation over a range of $T_{AVE}$ schedules, two bounding schedules representing the highest and lowest schedules of interest were considered. The higher schedule was found to be more limiting with respect to thermal margin because it has a much less favorable margin at power, and because moderator feedback is much stronger at higher temperatures. Consequently the higher schedule results in lower calculated MDNBRs for all events initiated from full power and, during the steam line break transients, the core has a greater tendency to return to power than any lower temperature schedule. Thus, this analysis supports operation with $T_{AVE}$ reduced from 15°F to 50°F below the current $T_{AVE}$ schedule.

The design basis events, listed below, as well as the input parameters used to simulate the reactor system, are reported herein:

| Event | Incident Class* |
|---|---|
| 1. Uncontrolled Control Rod Withdrawal | |
| • Fast Rod Withdrawal | II |
| • Slow Rod Withdrawal | II |
| 2. Loss of Coolant Flow | III |
| 3. Locked Rotor | IV |
| 4. Loss of External Electric Load | II |
| 5. Large Steam Line Break | IV |
| 6. Small Steam Line Break | IV |

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* Consistent with current FSAR incident classification for PWRs.

Events 1, 2, 3 and 4 were initiated from full power, while events 5 and 6 were initiated from hot zero power (HZP). The criteria for Class II and III events are:

(1) Peak System pressure should not exceed 2750 psia (= 110% of Design); and

(2) the minimum departure from nucleate boiling ratio (MDNBR) should be greater than the 95/95 value of 1.3 for the W-3 correlation[5].

In the case of Class IV accidents, some fuel damage is acceptable provided it is confined to a limited number of fuel rods in the core.

The criterion for steam line breaks is that shutdown margin must be sufficient to limit the occurrence of boiling transition in the core so that the extent of potential core damage is small for a large steam line break and that the core does not go critical following a small steam line break.

The analyses are based on an equilibrium ENC fueled core using conservative neutronic parameters calculated for ENC fuel. The results of the calculations are summarized in Table 1.1. The lowest MDNBR for Class II and III events initiated at 1520 MWt was 1.70 for the slow rod withdrawal transient. Evaluation of the bounding pressure transient, loss of electric load, indicates that peak primary system pressures will not exceed the 2750 psia vessel integrity limit. The locked rotor accident, a Class IV event, was analyzed and the MDNBR was found to be 1.26. While this value did not meet the 1.3 criterion, based on the W-3 correlation, the result is acceptable in light of the low probability of the event and the extremely short time the DNBR was below 1.3. The large steam line break resulted in a minimum critical heat flux ratio of 1.10 based on the Modified MacBeth Correlation[6,7]. Less than 1% of the fuel rods undergo boiling transition at this DNBR. The small steam line break did not result in the reactor going critical. The results of the analysis show that the transients initiated from full power at reduced $T_{AVE}$ have increased thermal margins relative to prior analysis for normal temperature and pressure operation.

Finally it is noted that both the present and prior analyses have significant conservatism relative to thermal margins that might be expected if one of the postulated transients actually occurred. This conservatism is built into the analysis by stacking plant operating condition uncertainties to minimize MDNBR, by using conservative reactor trip setpoints and trip delays, and by using bounding reactor kinetics parameters. In addition, the transients generally represent worst case scenarios (e.g. stuck control rod for steam line breaks, neglect of a direct trip for pump coastdown, etc.)

Table 1.1 Summary of Results for $T_{AVE}$ Reduction of 15°F

| Transient Class | Maximum Power Level (MWt) | Core Average Heat Flux (Btu/hr-ft$^2$) | Maximum Pressurizer Pressure (psia) | MDNBR (W-3) |
|---|---|---|---|---|
| Initial Conditions for Transients | 1520.0 | 177,608 | 2000 | 2.07 |
| Uncontrolled Rod Withdrawal (II) @ $6.0 \times 10^{-4}$ $\Delta\rho$/sec | 1861.5 | 191,492 | 2360 | 1.84 |
| Uncontrolled Rod Withdrawal (II) @ $5 \times 10^{-5}$ $\Delta\rho$/sec | 1747.3 | 198,802 | 2366 | 1.70 |
| Loss of Flow (III) 2-Pump Coastdown | 1520.0 | 177,608 | 2355 | 1.74 |
| Loss of Flow - (IV) Locked Rotor | 1520.0 | 177,608 | 2384 | 1.26 |
| Loss of Load* (II) | 1550.7 | 181,160 | 2528 | 2.07 |
| Large Steam Line Break (IV) | 685.2 | 74,165 | 2000 | 1.10** |
| Small Steam Line Break (IV) | -- | -- | 2000 | --*** |

* Initiated from 102% power

** Calculated with the Modified MacBeth Critical Heat Flux Correlation

*** Does not go critical

## 2.0 CALCULATIONAL METHODS AND INPUT PARAMETERS

The analysis of R.E. Ginna transient performance was performed using the Exxon Nuclear Company plant transient simulation model for pressurized water reactors, PTSPWR2, a digital computer program developed to model the behavior of pressurized water reactors under normal and abnormal operating conditions. The model is based on the solution of the basic transient conservation equations for the primary and secondary coolant systems. The transient conduction equation is solved for the fuel rods, and a point kinetics model is used to calculate the core neutronics. The program calculates fluid conditions such as flow, pressure, mass inventory and steam quality, heat flux in the core, reactor power, and reactivity during the transient. Various control and safety system components are included as necessary to analyze postulated events. A hot channel model is included to trace the departure from nucleate boiling (DNB) during transients. The DNB evaluation is based on the hot rod heat flux in the high enthalpy rise subchannel and uses the W-3 correlation to calculate the DNB heat flux for pressures greater than 1450 psia and, as an option, a modified MacBeth correlation for pressures less than 1450 psia.

The PTSPWR2 code models the reactor, two independent primary coolant loops (including all major components such as the pressurizer, both pumps, and the piping), two steam generators, and their steam lines (including all major valves such as turbine stop valves, isolation valves, and pressure relief valves). Figure 2.1 is a system schematic representing the model elements in PTSPWR2 and their interaction. For a more thorough discussion of the model details of the PTSPWR2 code, see Reference 4.

Several updates were included in the present analysis to (1) improve the initial steady state plant balance; (2) correct pressurizer surge flow calculations; (3) to incorporate the effects of pressurizer heaters on pressure control; and (4) to terminate steam generator heat losses after the steam inventory was exhausted. The pressurizer surge flow calculations are conservative for the loss-of-electric-load event, where the quantity of interest is the maximum primary system pressure. The maximum pressure attained is limited by surge flow or by safety relief valve capacity versus plant heatup rate.

For steam line breaks, the pressurizer control tries to maintain the pressure by turning the backup pressurizer heaters full on. The resulting expansion of the water delays safety injection, which occurs during the rapid pressure drop accompanying the emptying of the pressurizer of water. The pressurizer control has a slightly conservative effect on the transients from power, except for the loss-of-electric-load event, because pressurizer spray tends to hold down the pressure at the time of MDNBR, thereby lowering the MDNBR. The correction of heat loss to the steam generator has no impact, since it occurs after the boron from the safety injection has reached the reactor.

Conservative approximations are applied for predicting those system responses which contribute to minimum values of the DNB ratio. These approximations are categorized as either: (1) generic approximations applied to the steady state DNBR to account for plant instrumentation errors; (2) approximations which conservatively bound R.E. Ginna Unit 1 neutronics parameters; or (3) conservative operation of plant control or safety systems, in a transient-specific fashion.

The generic approximations (Category 1) are applied to all full power transients to account for steady state and instrumentation errors. The initial DNBR conditions are obtained by adding the maximum steady state errors to rated values as follows:

| | | |
|---|---|---|
| Reactor Power | = | 1520 MWt + 2% (30.4 MWt) for calorimetric error. |
| Average Coolant Temperature | = | 558.5°F - Schedule A<br>523.5°F - Schedule B<br>+ 4°F for deadband and measurement error |
| Primary Coolant System Pressure | = | 2000 - 30 psia for steady state fluctuation and measurement errors, |

where Schedule A and Schedule B refer to the $T_{AVE}$ schedules defined in Table 2.1.

The combination of the above parameters acts to minimize the initial minimum DNB ratio. It should be noted that none of the above steady state errors are explicitly included in the plant transient modeling, except for the loss-of-electric load event, but they are used to conservatively bound the initial MDNBR. Table 2.2 shows a list of operating parameters used in the analysis.

The trip setpoints incorporated into the PTSPWR2 model for R.E. Ginna Unit 1 are based on the Technical Specification limits and have been revised for the changed system conditions. These limiting trip setpoints with their associated time delays for each trip function are listed in Table 2.3.

The overtemperature and overpower $\Delta T$ trips are calculated from the parameters in Table 2.3 as

$$\Delta T_0 \left[K_1 - \frac{1+25S}{1+5S} K_3 (T_{AVE} - T_{SETPOINT}) + K_2 (P_{pr} - P_{SETPOINT}) - f(\Delta I)\right]$$

and

$$\Delta T_0 \left[K_4 - K_6 \frac{10S}{1+10S} T_{AVE} - K_5 (T_{AVE} - T_{SETPOINT}) - f(\Delta I)\right],$$

respectively. The function, $f(\Delta I)$, depends on the integrated top to bottom power skew, $\Delta I$, and exacts a 2% penalty for each 1% that $\Delta I$ falls outside the range -18% to 8%.

In the present analysis, the values of the constants $K_1$ to $K_6$ include an allowance for a 4% error in the trip signal calculation. This error allowance was not included in the prior analysis. The effect of this 4% allowance is to cause the slow rod withdrawal transient to have a calculated MDNBR that is 2.0% lower than the two-pump coastdown transient.

The reference setpoint values for $T_{AVE}$ and $P_{PR}$ were set to the new reduced temperature and pressure values in order that the margin to trip would not be increased for the reduced temperature and pressure conditions.

The ENC fuel design parameters for R.E. Ginna Unit 1 are summarized in Table 2.4. Table 2.5 lists the neutronics parameter values which conservatively bound the R.E. Ginna Unit 1 core for both the beginning and end of cycle, and those used for transient analysis. A design axial power profile with a peaking factor $F_Z$ = 1.64 at X/L = 0.6 was used in the analysis.

The approximations in Category 2 refer to the reactivity feedback effects from moderator temperature changes and Doppler broadening of the absorption resonances. For full power transients, the moderator temperature and pressure coefficients were set to zero. This provides a conservative estimate of the moderator density feedback for all transients from full power. The conservative choice for the Doppler feedback coefficient depends on the transient being analyzed. Table 2.6 summarizes the moderator and Doppler feedback coefficients applicable for full power transients. Note that feedback for steam line breaks is treated differently.

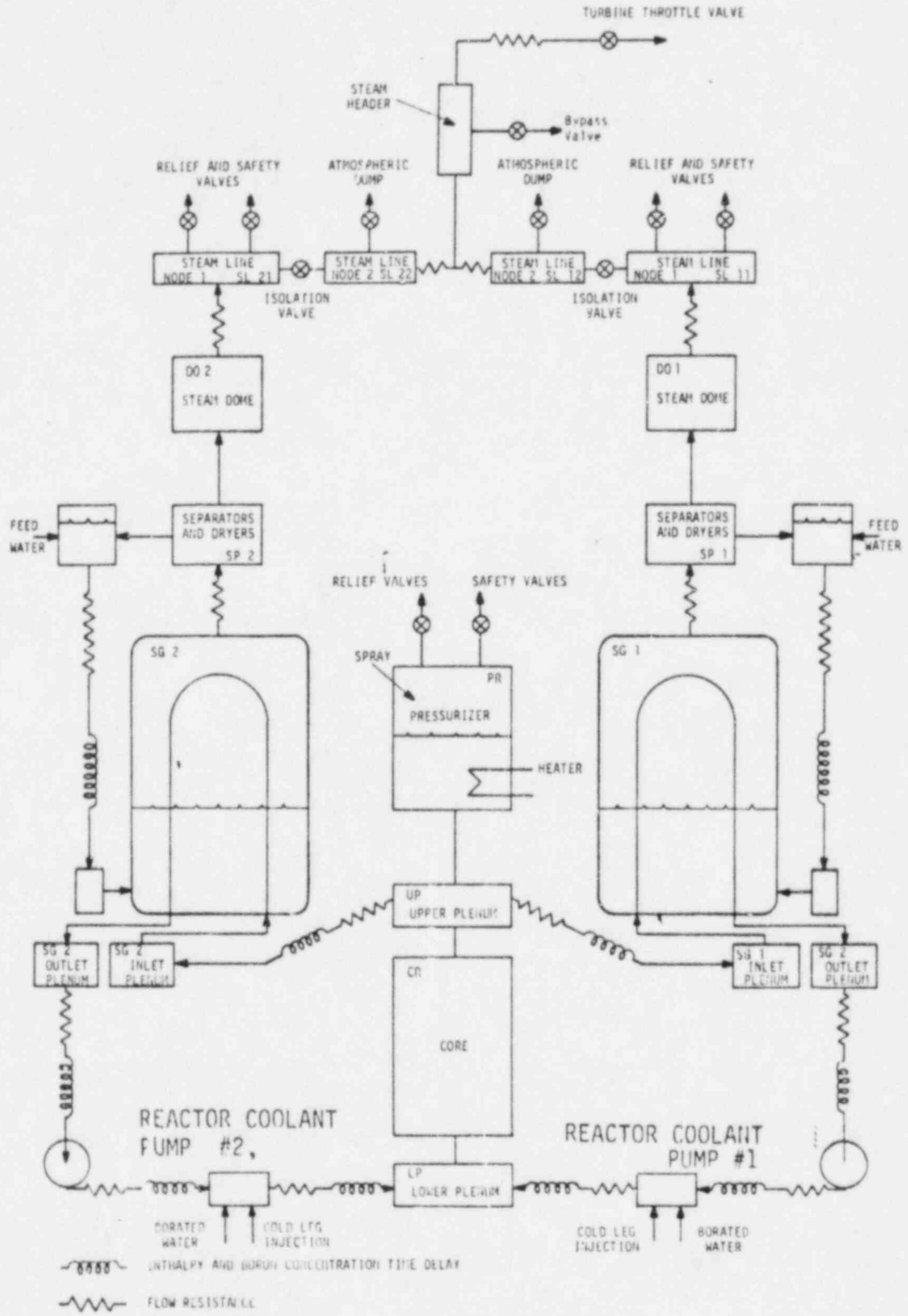


Figure 2.1 PTSPWR2 System Model

Table 2.1 $T_{AVE}$ Schedules for Operation at 2000 psia

| Schedule* | $T_{AVE}$ @ HZP | $T_{AVE}$ @ HFP |
|---|---|---|
| A | 547 | 558.5 |
| B | 514** | 523.5 |

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* $T_{AVE}$ varies linearly with power

** This value is conservative with respect to prior estimates of approximately 500°F(3) and represents the same decrease in primary cold leg temperature as in Schedule A.

Table 2.2 Thermal Parameters for Operation of R.E. Ginna Unit 1 at Reduced Average Temperature and Primary System Pressure

| | | |
|---|---|---|
| **CORE** | | |
| Total Core Heat Output (MW) | 1520 | |
| Heat Generated in Fuel (%) | 97.4 | |
| Pressurizer Pressure (psia) | 2000 | |
| **HOT CHANNEL FACTORS** | | |
| Total Peaking Factor, $F_Q^T$ | 2.80 | |
| Enthalpy Rise Factor, $F_{\Delta H}^N$ | 1.66 | |
| **SYSTEM PARAMETERS FOR $T_{AVE}$ SCHEDULE** | A | B |
| $T_{AVE}$ at full power ($^oF$) | 558.5 | 523.5 |
| at zero power | 547 | 514 |
| Total Primary Flow at full power (mlb/hr) | 69.2 | 71.8 |
| Active Core Flow at full power (mlb/hr) | 66.0 | 68.5 |
| Pressurizer level at full power (% of span) | 49 | 49 |
| at zero power | 33 | 36 |
| **STEAM GENERATORS** | | |
| Total steam flow (mlb/hr) | 6.60 | |
| Steam temperature ($^oF$) | 503.8 | |
| Feedwater Temperature ($^oF$) | 432.3 | |
| Steam Dome Pressure at full power (psia) | 700 | |
| Tube Plugging (%) | 10 | |

Table 2.3 Reactor Trip Setpoints

| Function | Delay Time (sec) | Tech. Spec. value | Schedule A | Schedule B |
|---|---|---|---|---|
| • High neutron flux (%) | 0.5 | 109 | 116 | 116 |
| • Low coolant flow (%) | 0.6 | 90 | 87 | 87 |
| • High pressure (psia) | 1.0 | 2400 | 2400 | 2400 |
| • Low pressure (psia) | 1.0 | 1880 | 1700 | 1700 |
| • High Water Level (% of span) | 1.0 | 88 | 100 | 100 |
| • Low-low S.G. water level (% of N.R. span) | 1.0 | 16 | 0 | 0 |
| • Reactor $\Delta T$ Trips | | | | |
| $\Delta T_0$ ($^oF$) | | Full Power Value | 57.4 | 58.4 |
| $T_{SETPOINT}$ ($^oF$) | | 573.5 | 558.5 | 523.5 |
| $P_{SETPOINT}$ (psia) | | 2250 | 2000 | 2000 |
| Overtemperature $K_1$ | | 1.12 | 1.165 | 1.165 |
| Overtemperature $K_2$ | | $7.356\times10^{-4}$ | $6.82 \times 10^{-4}$ | $6.82 \times 10^{-4}$ |
| Overtemperature $K_3$ | | .01577 | 0.0141 | 0.0141 |
| Overpower $K_4$ | | 1.083 | 1.134 | 1.134 |
| Overpower $K_5$ | | .001 | .001 | .001 |
| Overpower $K_6$ | | .0262 | .0273 | .0273 |
| • HPSI-Actuation (psia) | | 1738 | 1730 | 1730 |

Table 2.4 Exxon Nuclear Reload for R.E. Ginna Unit 1
Fuel Design Parameters

| | | |
|---|---|---|
| Fuel Radius | 0.1782 | inch |
| Inner Clad Radius | 0.1820 | inch |
| Outer Clad Radius | 0.2120 | inch |
| Active Length | 142.0 | inch |
| Number of Fuel Rods in Core | 21,659 | |

Table 2.5 R.E. Ginna Unit 1 Kinetics Parameters

| Parameters | Cycle 12 LTP Nominal Value BOC | Cycle 12 LTP Nominal Value EOC | Values Used In Transient Analyses: Rod Withdrawal and Loss of Load Transients, BOC | Values Used In Transient Analyses: Loss of Primary Flow Transients, BOC | Values Used In Transient Analyses: Steam Line Break Transients from HZP, EOC |
|---|---|---|---|---|---|
| Moderator Temperature Coefficient (pcm/$^{o}$F) | -5.7 | -24.8 | 0 | 0 | ** |
| Moderator Pressure Coefficient (pcm/psia) | +.09 | +.35 | 0 | 0 | * |
| Doppler Coefficient (pcm/$^{o}$F) | -1.4 | -1.6 | -1.0 | -1.5 | *** |
| Boron Worth Coefficient (pcm/ppm) | -7.9 | -8.4 | * | * | -7 |
| Scram Worth (pcm) | -3467 | -4827 | -1600 | -1600 | * |
| Shutdown Margin (pcm) | 1000 | 1900 | * | * | 1600 |
| Delayed Neutron Fraction | .0059 | .0052 | .0059 | .0059 | .0049 |

* Not Applicable to this transient.
** See Figure 3.32.
*** See Figure 3.33.

Table 2.6 Moderator and Doppler Coefficients

| Transient | Desired Moderator Feedback Effect | Resulting* Coefficient $\Delta\rho/^{o}F \times 10^{-5}$ | Desired Doppler Feedback Effect | Resulting Coefficient $\Delta\rho/^{o}F \times 10^{-5}$ |
|---|---|---|---|---|
| Fast Rod Withdrawal | Minimum | 0.0 | Minimum | -1.0 |
| Slow Rod Withdrawal | Minimum | 0.0 | Minimum | -1.0 |
| Loss of Coolant Flow | Minimum | 0.0 | Maximum | -1.5 |
| Locked Rotor | Minimum | 0.0 | Maximum | -1.5 |
| Loss of Load | Minimum | 0.0 | Minimum | -1.0 |
| Steam Line Breaks | Maximum | *** | Minimum | ** |

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* For minimal effect no moderator feedback is allowed

** See Figure 3.32

*** See Figure 3.33

# 3.0 TRANSIENT ANALYSIS

## 3.1 INITIAL CONDITIONS

Evaluation of the effects of the proposed reduced temperature and pressure operation on steady state thermal margin was performed in accordance with standard ENC thermal hydraulic calculational methodology described in Reference 5. All events were initiated with a pressure of 2000 psia compared to the 2250 psia used in References 1 and 2. A range of average temperatures was considered from Schedule A to Schedule B of Table 2.1. References 1 and 2 used $T_{AVE}$ = 573.5$^{o}$F. Since, in all cases, Schedule A was more limiting, due to either initial DNBR or nonlinearity of moderator feedback, Schedule A was analyzed for all transients.

With Schedule A, the initial MDNBR at full power operation, 1520 MWt, was 2.07 compared to 2.0[1,2] while Schedule B provided an initial MDNBR of 2.63. The improvements in MDNBR are due to the decrease in $T_{AVE}$ which more than compensates for the 250 psi decrease in pressure. The improvement in initial DNBR is reflected directly in the MDNBR for those events (1,2,3 and 4) initiated from full power. The overpower and overtemperature $\Delta T$ trip functions were adjusted for the reduced temperature and pressure conditions in order to maintain the margin to trip and thus transients terminated by the $\Delta T$ trip tend to preserve the improvements in margin asociated with the improved initial DNBR. The analysis results presented in the following subsections include the most limiting rod withdrawals at full power conditions, the most limiting loss of flow accidents, which have previously been shown as the worst accidents relative to thermal margin, the loss of load transient, and the steam line break transients, which demonstrate the characteristics of reactor coolant cooldown incidents.

The transient events discussed in this Section are summarized in Tables 3.1 to 3.8.

### 3.2 UNCONTROLLED ROD WITHDRAWAL

The withdrawal of a control rod bank adds reactivity to the reactor core, causing both the power level and the core heat flux to increase. Since the heat extraction from the steam generator remains relatively constant, there is an increase in primary coolant temperature. Unless terminated by manual or automatic action, this power mismatch and the resultant coolant temperature rise could eventually result in an unacceptable loss of the thermal margin. While the inadvertent withdrawal of a control rod bank is unlikely, the reactor protection system is designed to terminate such a transient while maintaining an adequate margin to DNB.

In the rod withdrawal incident the reactor may be tripped by the overtemperature $\Delta T$ function, by the nuclear overpower function, or by another reactor protective safety system setpoint. Both a fast rod withdrawal and a slow rod withdrawal were analyzed from an initial power level of 1520 MWt. Beginning-of-cycle kinetics coefficients were used with a minimum value for Doppler feedback.

Figures 3.1 to 3.6 show plant responses for a fast rod withdrawal ($6 \times 10^{-4}\ \Delta\rho$/sec) from 1520 MWt. A nuclear overpower trip (116% setpoint) occurred at 1.99 seconds. The DNB ratio dropped from an initial value of 2.07 to 1.84. Pressure increased to a maximum of 2360 psia with core average temperature increasing by less than 2.5$^{o}$F. Pressurizer control was active during the transient simulation, resulting in lower system pressures and lower calculated MDNBR. The parameters for the two linear control functions are listed in Table 3.9, along with those of the pressure relief valves. Following the reactor trip, reactivity was inserted via a programmed curve, depicted in Figure 3.17.

The system responses to a slow rod withdrawal of $5 \times 10^{-5}$ $\Delta\rho$/sec are depicted in Figures 3.8 to 3.13. The overtemperature $\Delta T$ function initiated the reactor trip at 24.27 seconds, and the minimum DNB ratio during the transient was 1.7. Sizing of the parameters in the overtemperature $\Delta T$ trip function is such that for a reactivity insertion of $5 \times 10^{-5}$ $\Delta\rho$/sec it is nearly coincident with the nuclear overpower trip (116%). For the rod withdrawal accidents at reduced coolant temperature condition, power peaking increases about the withdrawn rod are not expected to be more than a few percent, and these are more than offset by the significantly improved DNBR resulting from lower $T_{AVE}$.

### 3.3 LOSS OF COOLANT FLOW

The loss of coolant flow transient is postulated to occur as a result of a loss of electric power to the primary coolant pumps. The transient results in an increase in coolant temperature which, in combination with the decrease in flow, reduces the margin to DNB. Only the limiting case has been analyzed. This case is the loss of power to both pumps when the reactor system is operating at 1520 MWt. Beginning-of-cycle values of kinetics coefficients are assumed with a conservative choice for the Doppler coefficient. The loss of power to all pumps would ordinarily result in a reactor trip due to either under-voltage or under-frequency at the bus. No credit was taken for these protective functions and the trip allowed to occur on a low flow signal. This delay resulted in a further flow reduction at full power, and a more conservative calculation of margin to DNB. The pressurizer pressure control was retained to provide a more conservative MDNBR.

Figures 3.14 to 3.19 depict plant responses after the loss of all pump power. A reactor trip occurred at 3.57 seconds. A minimum DNB ratio of 1.74 was reached 4.45 seconds after the beginning of coastdown. System pressure peaked at 2355 psia.

### 3.4 LOCKED ROTOR

In the unlikely event of a seizure of a primary coolant pump, flow through the core would be drastically reduced, resulting in a reactor trip on a low flow signal. The coolant enthalpy would rise, thus decreasing the margin to DNB. The locked rotor transient was analyzed assuming two loop operation with instantaneous seizure of one pump from 1520 MWt. Beginning-of-cycle kinetics coefficients were used as the BOC moderator coefficient is the most adverse. A conservative value for Doppler feedback was used and pressurizer pressure control was retained.

The responses for the locked rotor transient are shown in Figures 3.20 to 3.25. The reactor trip occurred at 0.64 seconds on the low flow function. Core average temperature increased by 18.7°F with system pressure reaching 2384 psia. The DNB ratio in the analysis reached a minimum of 1.26 at 1.85 seconds, and recovered the initial DNBR of 2.07 by 3.75 seconds. The total exposure of the core to a DNBR of 1.3 or less was approximately 1 second.

### 3.5 LOSS OF ELECTRIC LOAD

Loss of electric load involves plant behavior following a trip of the turbine-generator without a direct reactor trip. The major consequence of the loss of heat sink is a rapid increase in $T_{AVE}$ and an associated rise in pressurizer level and pressure due to expansion of the primary coolant. Conceivably DNBR could be a problem, since rising temperatures adversely affect thermal margins. However, for R.E. Ginna Unit 1, the pressurization transient is sufficiently strong even with pressurizer pressure control functioning to cause an increase in DNBR with increasing $T_{AVE}$. Thus the purpose for evaluating this transient is to assess peak pressure versus the vessel integrity limit of 2750 psia. In calculating the peak pressurizer pressure, spray control was turned off, and the power operated relief valves (PORVs) disabled.

The transient was initiated from 1550.4 MWt (102% power) and 2030 psia, with a conservative Doppler feedback and no moderator feedback. The reactor tripped on a high pressurizer pressure signal at 4.81 seconds and peak pressure was 2520 psia at 6.89 seconds. DNBR never dropped below the initial value. Figures 3.26 to 3.31 show the system responses to a loss of electric load. Control of maximum pressure was exercised by the safety relief valves whose capacity far exceeds the rate at which the coolant can expand. After 10 seconds the pressurizer was only 68% full (up from 49% operating level) and there was no danger of "packing" the pressurizer.

### 3.6 STEAM LINE BREAKS

A break of a steam pipe (or safety valve failure) would result in a sharp reduction in steam inventory in a steam generator. This pressure decrease, which accompanies the loss of heat via ejected steam, would cause a heat loss from the primary coolant, reducing primary coolant temperature and pressure. With a negative moderator temperature coefficient, the reduced temperature would lead to a reactivity insertion into the core which could lead to criticality and core damage if unchecked.

Steam line break transients are simulated with the PTSPWR2 plant transient simulation code. As a worst case, the steam line break was assumed to occur at hot zero power conditions corresponding to a core average temperature of 547°F when the steam generator secondary side water inventory was at a maximum, thus prolonging the duration and increasing the magnitude of the primary loop cooldown. For conservatism, the most reactive control rod was assumed to be stuck out of the core when evaluating the shutdown margin of the control rods.

The reactivity as a function of core average temperature and the variation of reactivity as a function of core power used in this analysis are shown in Figures 3.32 and 3.33. A shutdown margin of 1.6% was used for conservatism. Table 2.5 summarizes the kinetics parameters used in steam line break analysis. Minimum capability of the boron injection system, which is based on two of three high pressure safety injection (HPSI) pumps being available, was assumed. A low pressurizer pressure signal initiated HPSI. The entry of borated water at 20,000 ppm into the primary loop cold legs was delayed by the necessity of first sweeping the injection lines of low concentration borated water. The delay time is dependent on the difference between the pressure and the pump shut-off head, 1400 psia. Initial system pressure is not an important factor in this analysis, since depressurization of the primary loop to the low pressure trip setpoint occurs rapidly once the pressurizer empties and the time is essentially independent of initial system pressure. The effects of initial temperature are discussed for the large steam line break.

Flow from steam line breaks was calculated based on a fixed break area and the Moody curve(8) for choke flow. The break area corresponded to a double-ended guillotine rupture of the steam line at the exit of the steam generator in the large break analysis. The small break analysis represented a failed safety relief valve.

### 3.6.1 Large Steam Line Break

For the large break the steam flow was calculated from the Moody curve for critical flow of saturated steam, based on the flow area for a break inside of containment. Initially the intact steam generator also blew down until the main steam isolation valve closed. This case, which retained pump power, was shown to give the greatest return to power[9].

Figures 3.34 to 3.39 show the transient response for a large steam line break initiated from 547°F (Schedule A). A conservative choice of shutdown margin (1.6%) was used although the Technical Specification limit is 1.9%. The initial steam flow (8615 lb/second) induced a rapid cooldown to ~400°F. Accompanying this cooldown was a rapid rise in moderator reactivity such that the reactor went critical at 16 seconds and reached a peak power of 685 MWt, or 45.1% of 1520 MWt, before the borated water shut the reactor down. The HPSI signal was received at 6.6 seconds on the emptying of the pressurizer. By 16.6 seconds, HPSI had occurred. The core parameters at the time of peak power were outside the range of validity for the W-3 correlation and a Modified MacBeth Correlation with a conservative local hot rod peaking factor, $F_Q^T$, of 14 was used. The minimum critical heat flux (CHF) ratio of 1.1 occurred at 43.8 seconds. Less than 1% of the fuel rods underwent boiling transition.

Figures 3.40-3.45 show the transient response for a large steam line break initiated from 514°F (Schedule B). The steam flow for this case was based on the Moody curve as with Schedule A. Because of reduced steam pressure at saturation, the initial flow is significantly less (5848 lbs/sec). A rapid cooldown to ~400°F occurred following the break. Accompanying the cooldown was an increase in moderator reactivity which was less than the Schedule A insertion after a large break. The moderator reactivity as a function of temperature is shown in Figure 3.33. Because of the curvature, a change of 50°F from a $T_{AVE}$ of approximately 550°F produced about a 30% larger change in reactivity than a change of 50°F from 500°F. In fact, it requires about a 70°F change from 500°F to produce the same change in reactivity that 50°F produces from 550°F. The reactor went critical at approximately 24 seconds and reached a peak power of 271.9 MWt, or 17.9% of 1520 MWt. HPSI occurred at 18.4 seconds with the borated water shutting the reactor down at 41.7 seconds. The MCHFR was 3.21 for this transient.

The transient return to power from HZP on Schedule B was less than an Schedule A. Similarly, because of lower steam pressures and the shape of the moderator reactivity curve, the tendency to return to power on Schedule B will be less for all steam line breaks.

3.6.2 Small Steam Line Break

The small steam line break summarized in Figures 3.46 to 3.51 corresponds to a failed safety relief valve with a capacity of 228.5 lbs/sec at 1100 psia, during single loop operation on Schedule A. The shutdown margin of 1.6%, compared to the end-of-cycle Technical Specification limit of 2.4% for one loop operation, prevented the reactor from returning to power. No HPSI signal was generated because the pressurizer never emptied. As with the large break, Schedule B results, not shown, displayed even less of a tendency to return to power.

Table 3.1 Uncontrolled Rod Withdrawal (Fast)
Event Table

| Time (sec.) | Event | Value |
|---|---|---|
| 0 | Initiate Reactivity Insertion | $6 \times 10^{-4}$ $\Delta\rho$/sec. |
| 1.99 | Nuclear Overpower Trip | 116% |
| 2.11 | Peak Power Level | 1861.5 MWt |
| 2.95 | Minimum DNB Ratio | 1.84 |
| 3.11 | Peak Heat Flux (average) | 191,492 Btu/hr-$ft^2$ |
| 6.13 | PORV Opened | 2350 psia |
| 6.46 | Peak Pressurizer Pressure | 2360 psia |

Table 3.2 Uncontrolled Rod Withdrawal (Slow)
Event Table

| Time (sec.) | Event | Value |
|---|---|---|
| 0 | Initiated Reactivity Insertion | $5 \times 10^{-5}$ /sec. |
| 24.27 | Overtemperature $\Delta T$ trip | 115% |
| 24.28 | Peak Power Level | 1747.3 MWt |
| 24.40 | Minimum DNB Ratio | 1.7 |
| 24.46 | Peak Heat Flux (average) | 198,802 Btu/HR-ft$^2$ |
| 28.53 | PORV Opened | 2350 psia |
| 37.13 | Peak Pressurizer Pressure | 2366 psia |

Table 3.3 Loss of Coolant Flow
Event Table

| Time (sec.) | Event | Value |
|---|---|---|
| 0 | Loss of Pumping Power | - |
| 3.57 | Low Flow Trip | 87% |
| 4.45 | Minimum DNB Ratio | 1.74 |
| 4.96 | Peak Core Temperature (average) | 563°F |
| 7.26 | PORV Opened | 2350 psia |
| 7.45 | Peak Pressurizer Pressure | 2355 psia |

Table 3.4 Locked Rotor - Event Table

| Time (sec.) | Event | Value |
|---|---|---|
| 0 | Pump Seizure in Loop 1 | - |
| 0.64 | Low Flow Trip | 82% |
| 1.85 | Minimum DNB Ratio | 1.26 |
| 3.08 | PORV Opened | 2350 psia |
| 3.60 | Peak Pressurizer Pressure | 2384 psia |

Table 3.5 Loss of Electric Load
Event Table

| Time (sec.) | Event | Value |
|---|---|---|
| 0 | Loss of all Electric Load | - |
| 3.81 | High Pressure Trip Signal Generated | 2400 psia |
| 4.39 | Pressurizer Safety Valve Opened | 2500 psia |
| 4.81 | High Pressure Trip | - |
| 6.89 | Peak Pressurizer Pressure | 2526 psia |

Table 3.6 Large Steam Line Break (547°F)
Event Table

| Time (sec.) | Event | Value |
|---|---|---|
| 0 | Double-Ended Break in Loop 1 | - |
| 5.00 | MSIV Closed on Loop 1 | - |
| 6.61 | Low Pressurizer Pressure HPSI signal generated | 1698 psia |
| 15.77 | Reactor went Critical | - |
| 16.60 | HPSI Pump reached Operating Head | 1400 psia |
| 42.70 | Boron Entered the Loop | - |
| 42.72 | Peak Power Level | 685.2 MWt |
| 43.75 | Minimum DNB Ratio* | 1.10 |
| 43.83 | Peak Heat Flux (average) | 74,166 Btu/hr-ft$^2$ |

* Calculated with modified MacBeth correlation

Table 3.7 Large Steam Line Break (514°F)
Event Table

| Time (sec.) | Event | Value |
|---|---|---|
| 0 | Double-Ended Break in Loop 1 | - |
| 5.00 | MSIV Closed on Loop 1 | - |
| 8.42 | Low Pressurizer Pressure HPSI signal generated | 1697 psia |
| 18.41 | HPSI Pump reached Operating Head | 1400 psia |
| 24.30 | Reactor went Critical | - |
| 41.71 | Boron Entered the Loop | - |
| 41.82 | Peak Power Level | 271.9 MWt |
| 45.00 | Minimum DNB Ratio* | 3.21 |
| 45.06 | Peak Heat Flux (average) | 27,937 Btu/hr-ft$^2$ |

* Calculated with modified MacBeth correlation

Table 3.8 Small Steam Line Break
Event Table

| Time (sec.) | Event | Value |
|---|---|---|
| 0 | Steam Line Safety Valve Opened | - |
| 168.75 | Maximum Reactivity | -$0.84 |

Table 3.9 Pressure Control and Safety Parameters

| | Value Used in PTSPWR2 |
|---|---|
| PRESSURIZER | |
| SPRAY | |
| On, psia | 2048 |
| Off, psia | 1998 |
| Maximum, lbs/sec | 44.98 |
| HEATER | |
| Pressure full on, psia | 1981 |
| Pressure full off, psia | 2011 |
| Maximum, kw | 800 |
| PORVs | |
| Setpoint, psia | 2350 |
| Total Capacity, lbs/sec | 99.44 |
| SAFETY RELIEF VALVES | |
| Setpoint, psia | 2500 |
| Total Capacity, lbs/sec | 160 |
| STEAM GENERATOR | |
| SAFETY RELIEF VALVES | |
| Setpoint, psia | 1100 A<br>1118 B<br>1136 C<br>1154 D |
| Valve Capacity, lbs/sec | 228.45 |
| RELIEF VALVE | |
| Setpoint, psia | 1065 (open)<br>777 (close) |
| Total Capacity, lbs/sec | 228.45 |

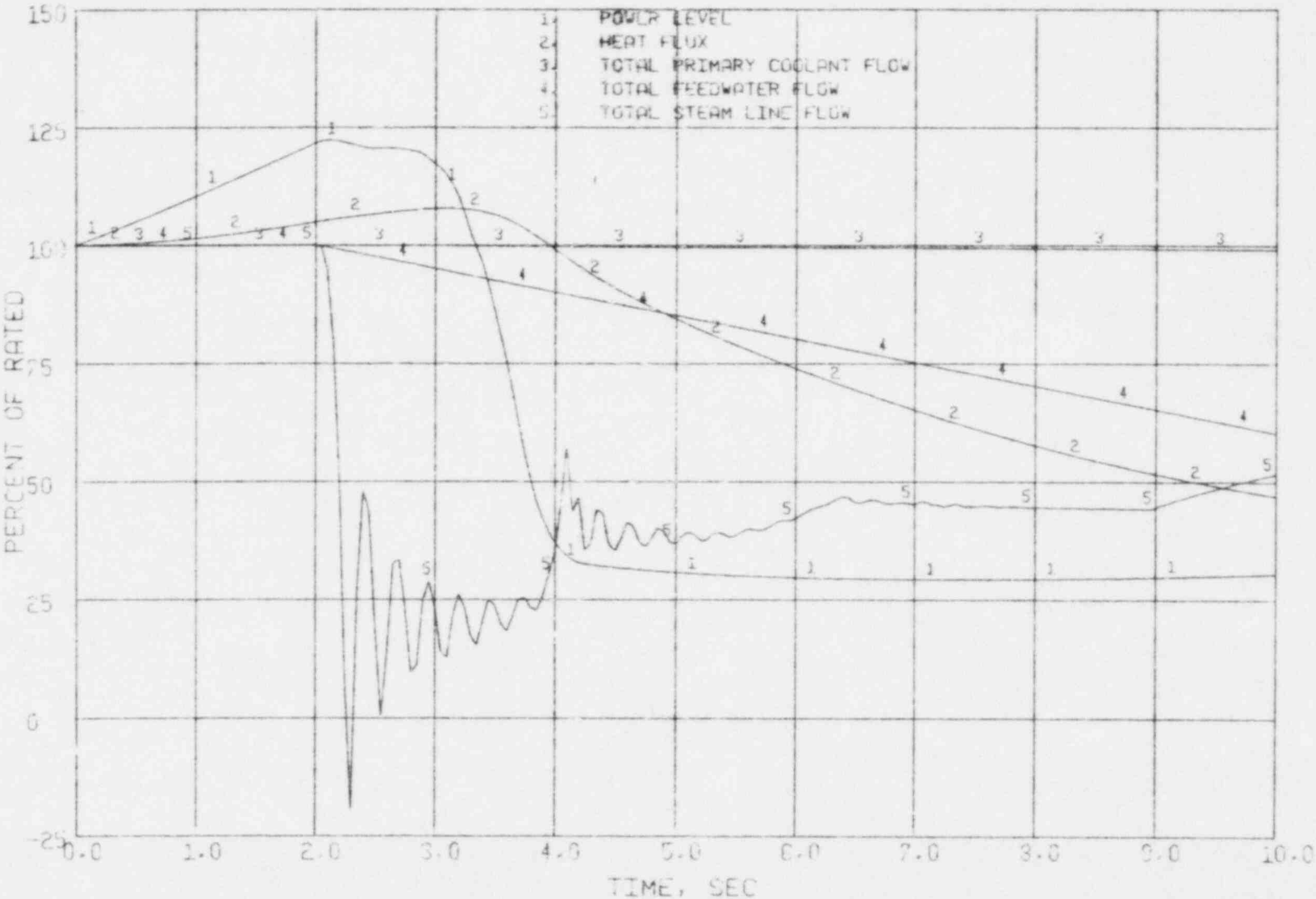


Figure 3.1 Uncontrolled Rod Withdrawal (Fast)
Power, Heat Flux and System Flows

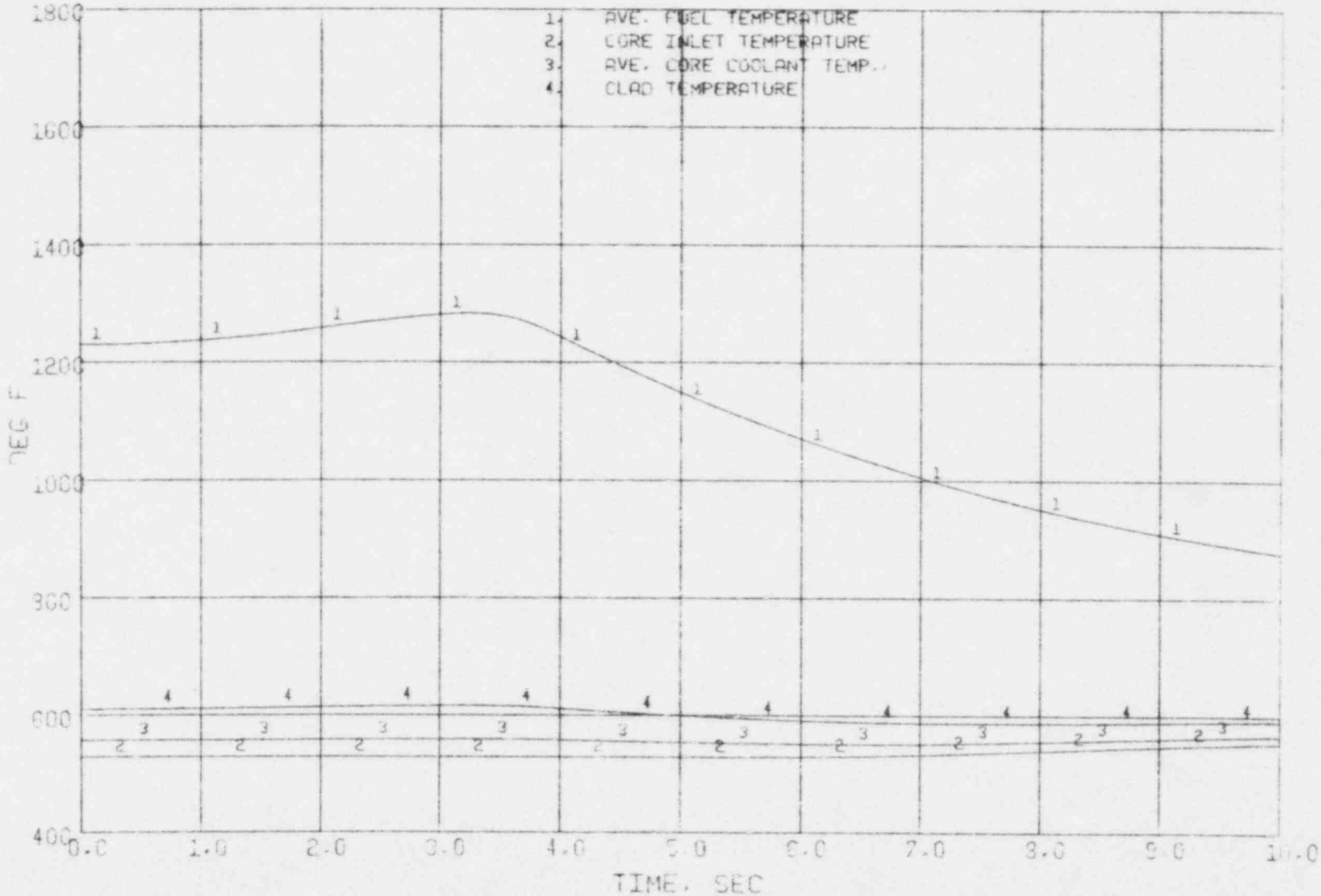


Figure 3.2 Uncontrolled Rod Withdrawal (Fast)
Core Temperature Response

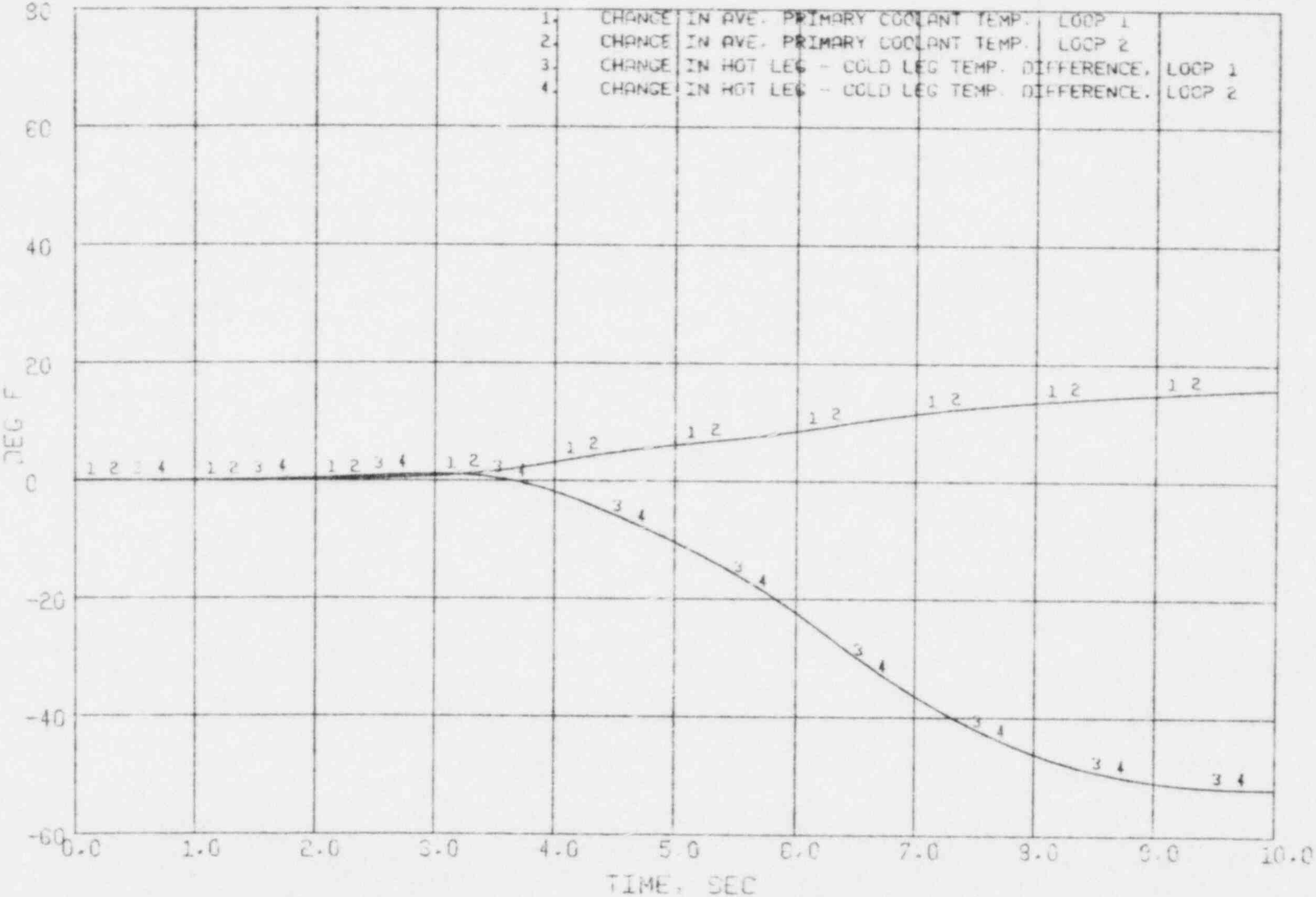


Figure 3.3 Uncontrolled Rod Withdrawal (Fast)
Primary Loop Temperature Changes

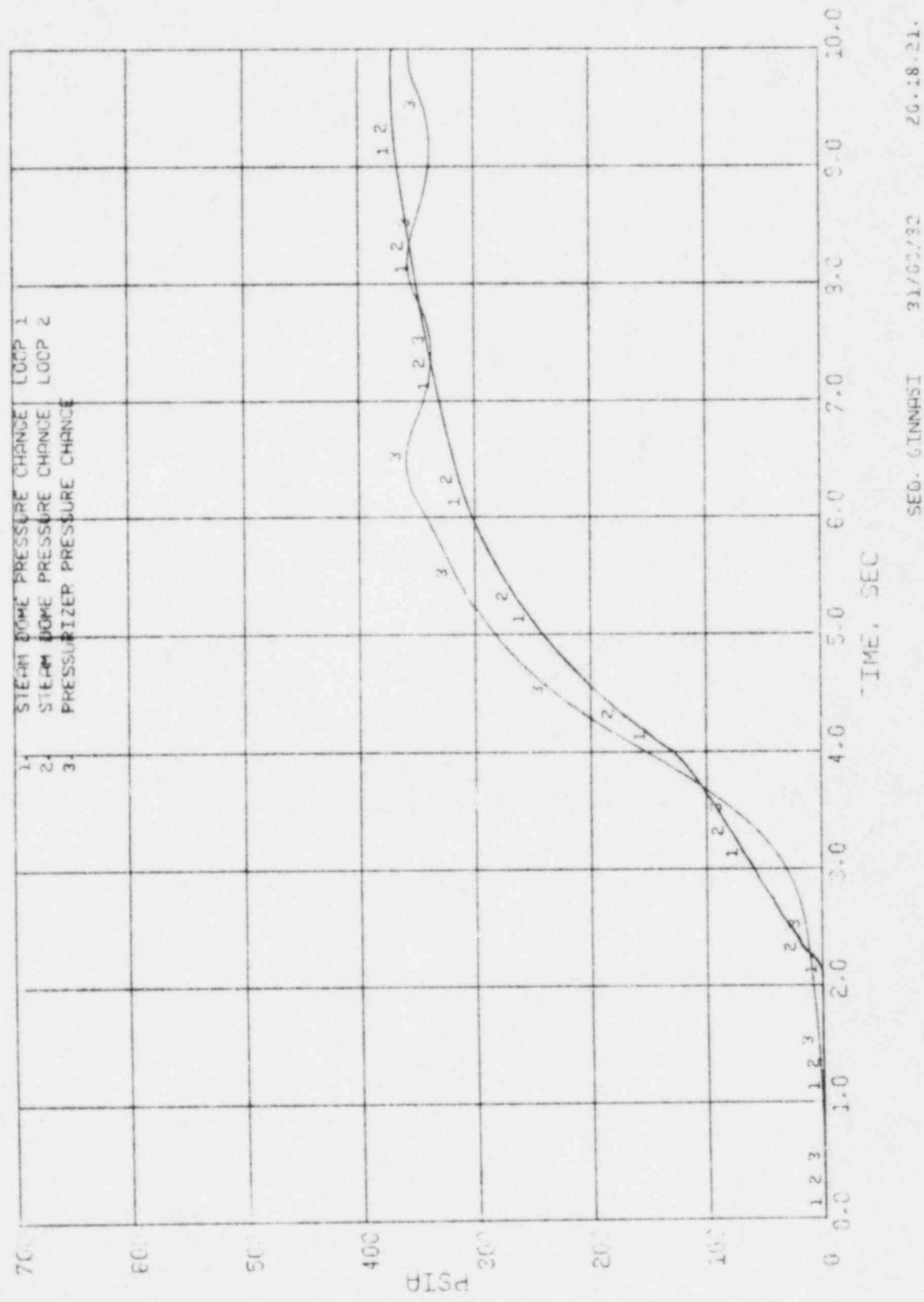


Figure 3.4 Uncontrolled Rod Withdrawal (Fast) Pressure Changes

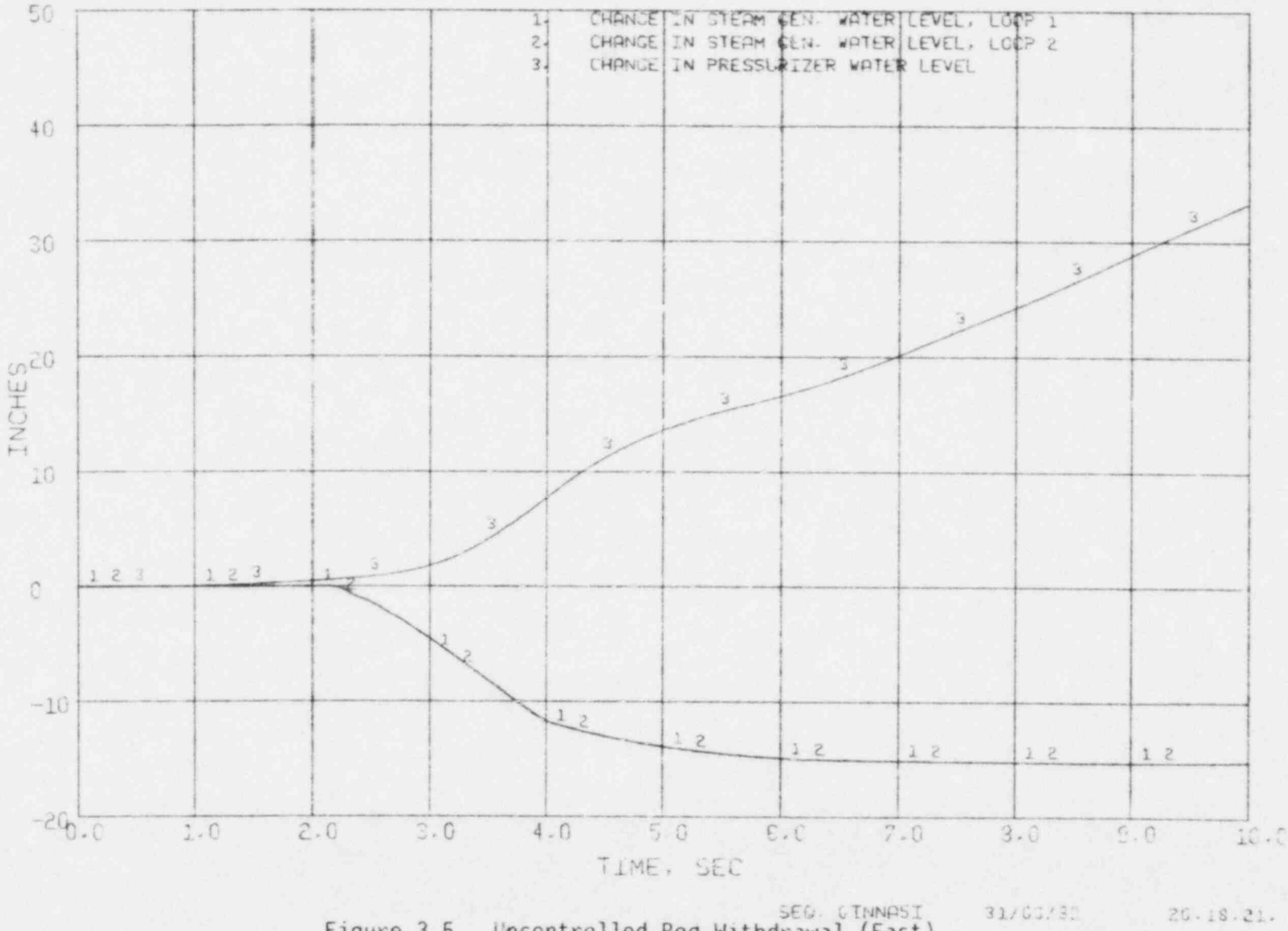


Figure 3.5 Uncontrolled Rod Withdrawal (Fast) Level Changes

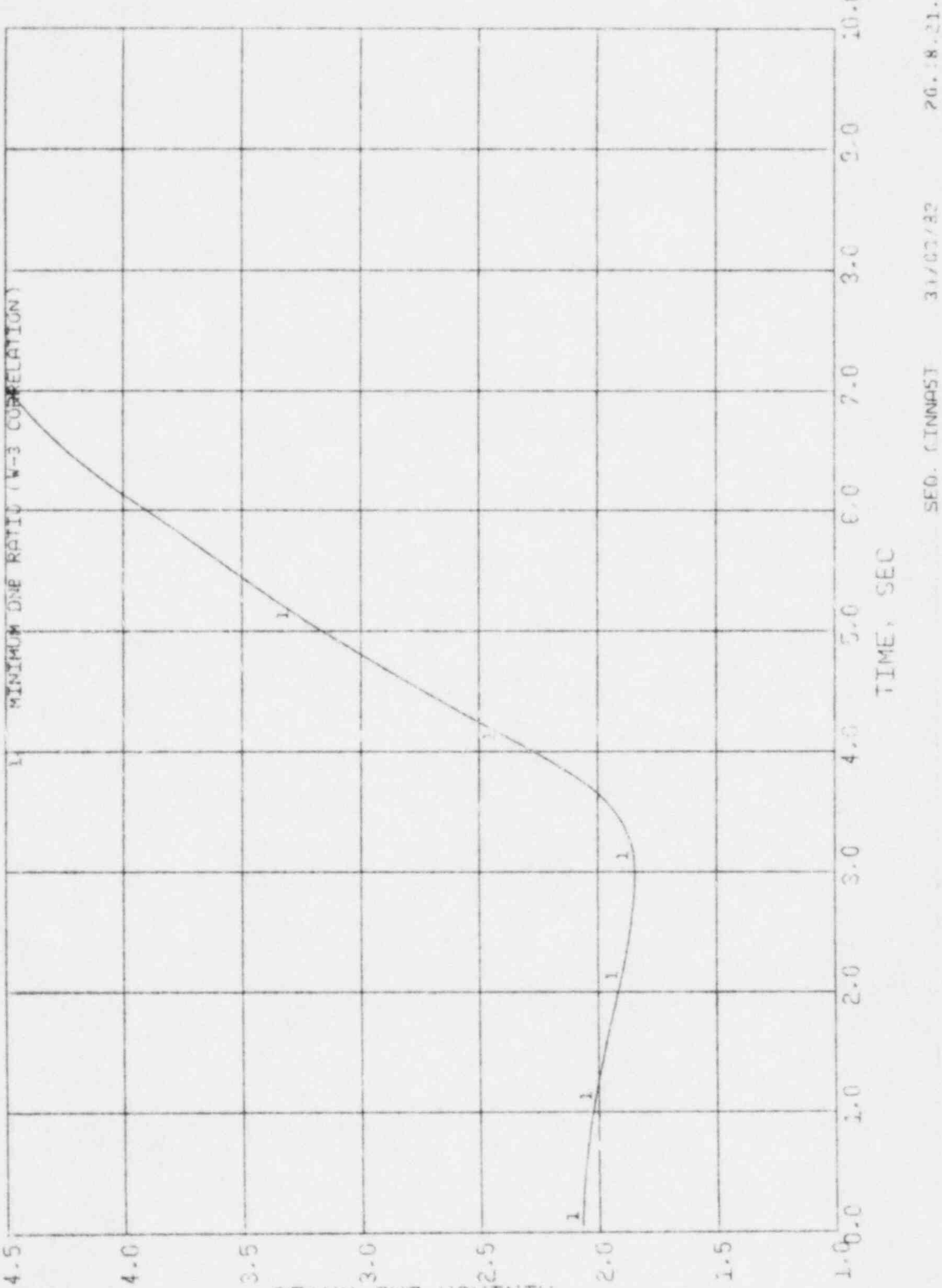


Figure 3.6 Uncontrolled Rod Withdrawal (Fast)
Minimum DNB Ratio

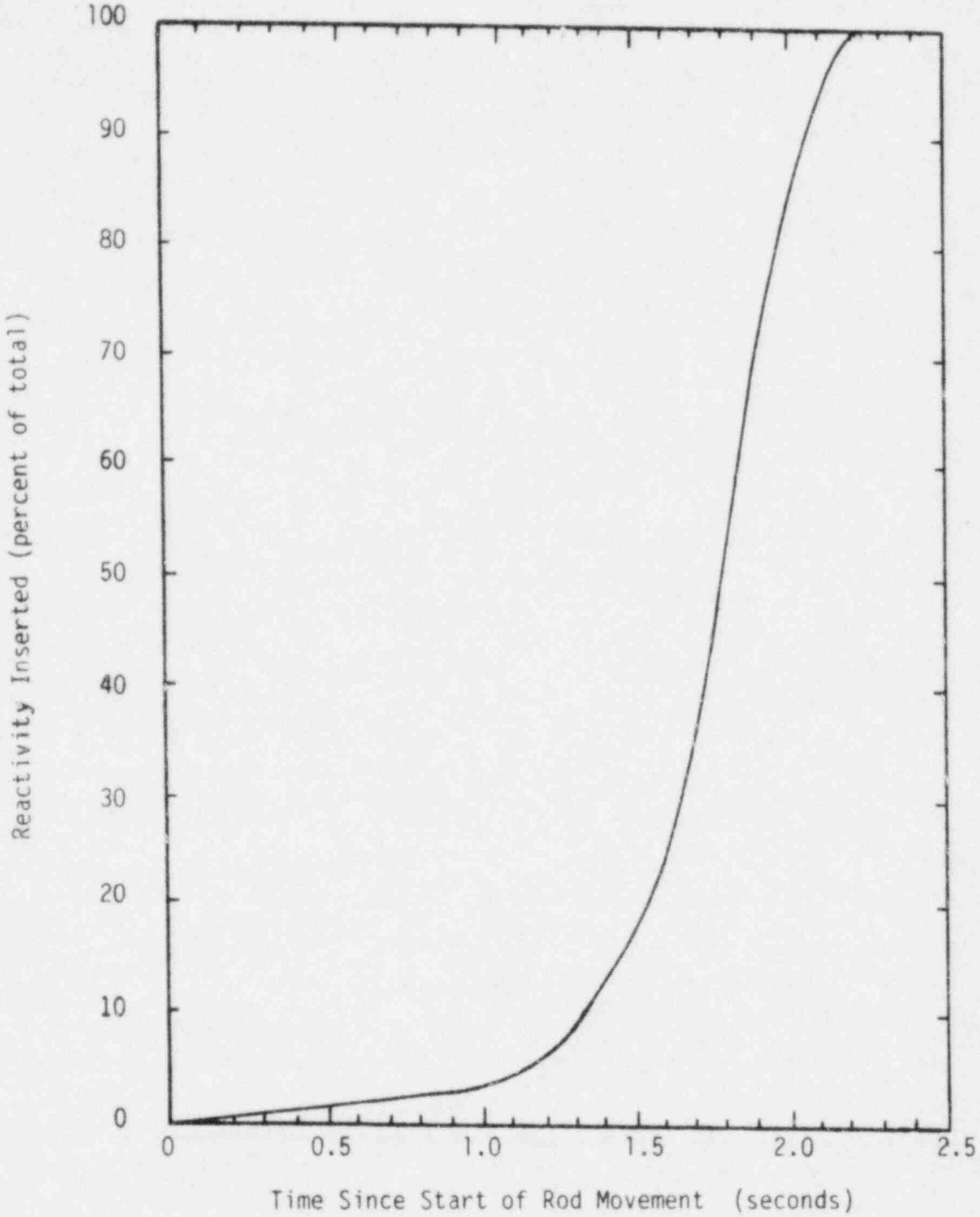


Figure 3.7 Scram Curve used in R.E. Ginna Unit 1 Transient Analysis

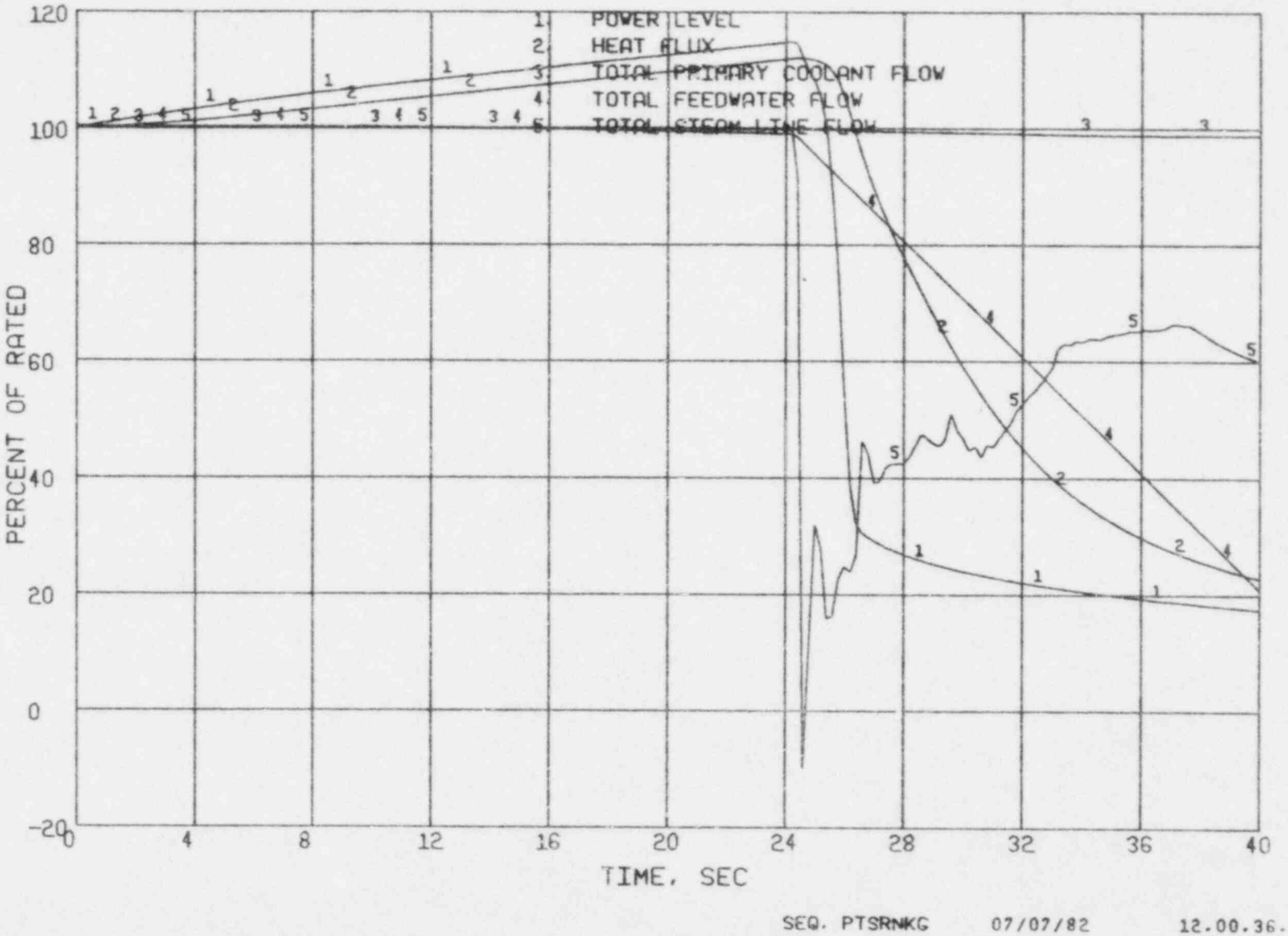


Figure 3.8 Uncontrolled Rod Withdrawal (Slow) - Power, Heat Flux and System Flows

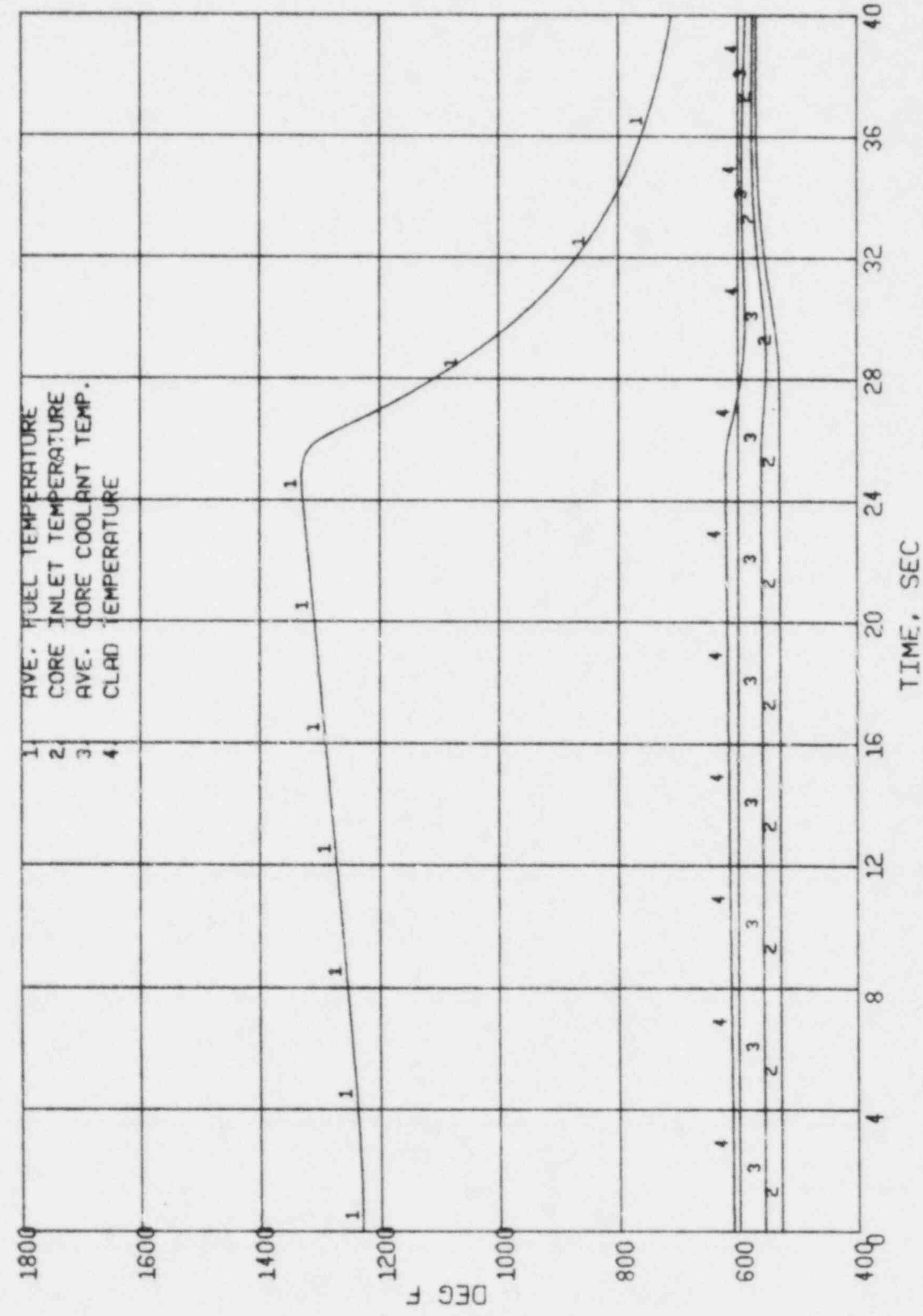


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Figure 3.9 Uncontrolled Rod Withdrawal (Slow) - Core Temperature Response

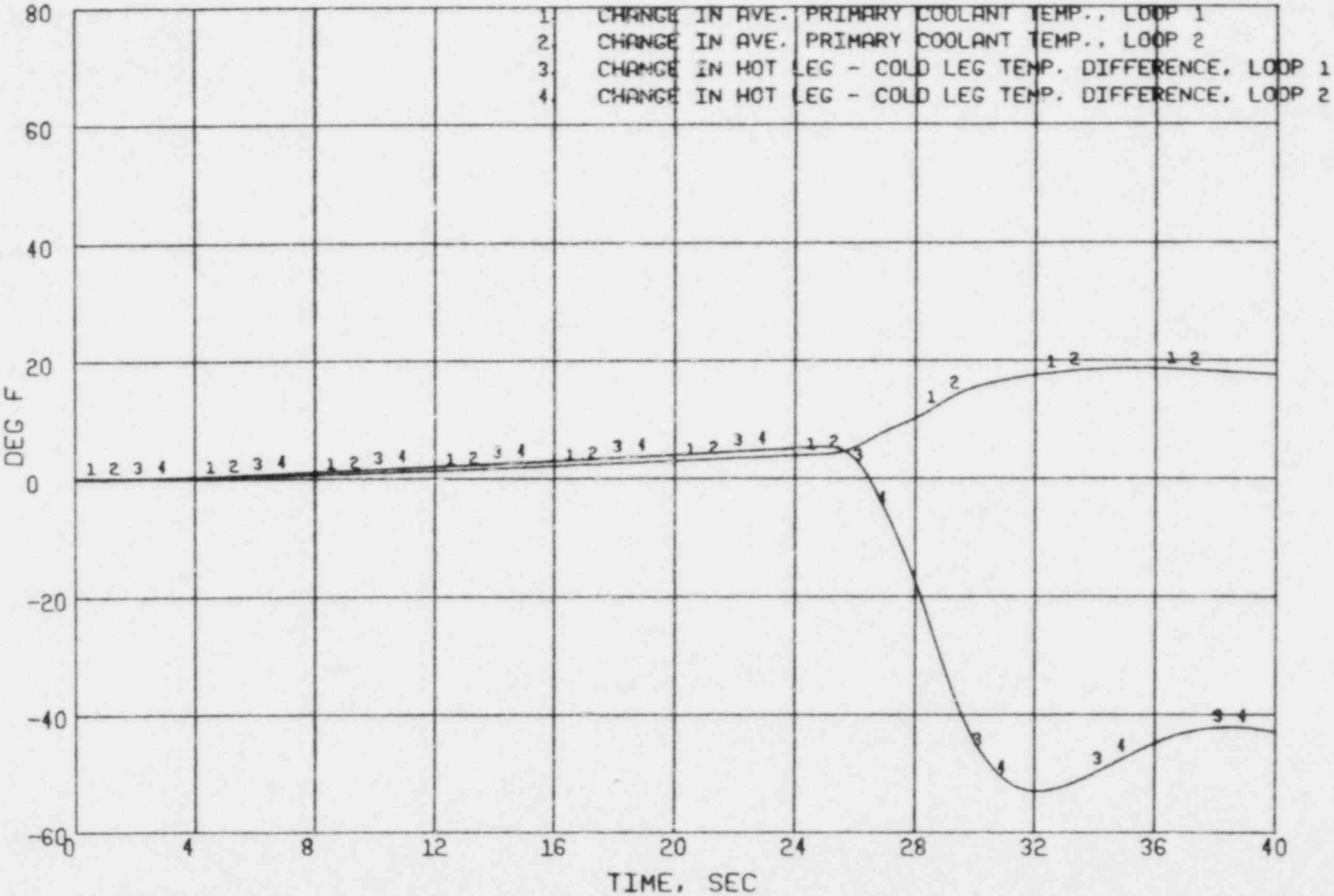


Figure 3.10 Uncontrolled Rod Withdrawal (Slow) - Primary Loop Temperature Changes

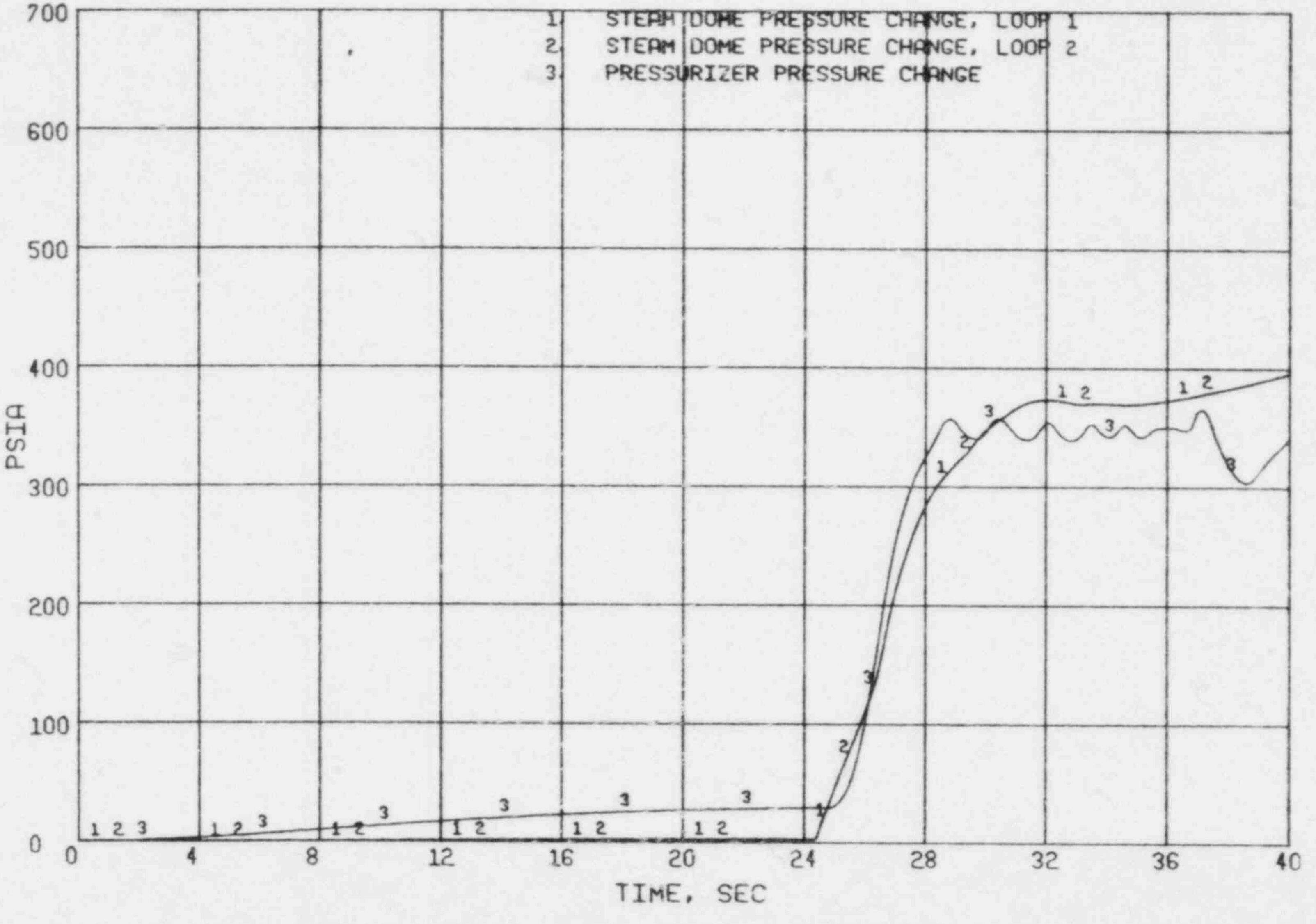


Figure 3.11 Uncontrolled Rod Withdrawal (Slow) -
Pressure Changes

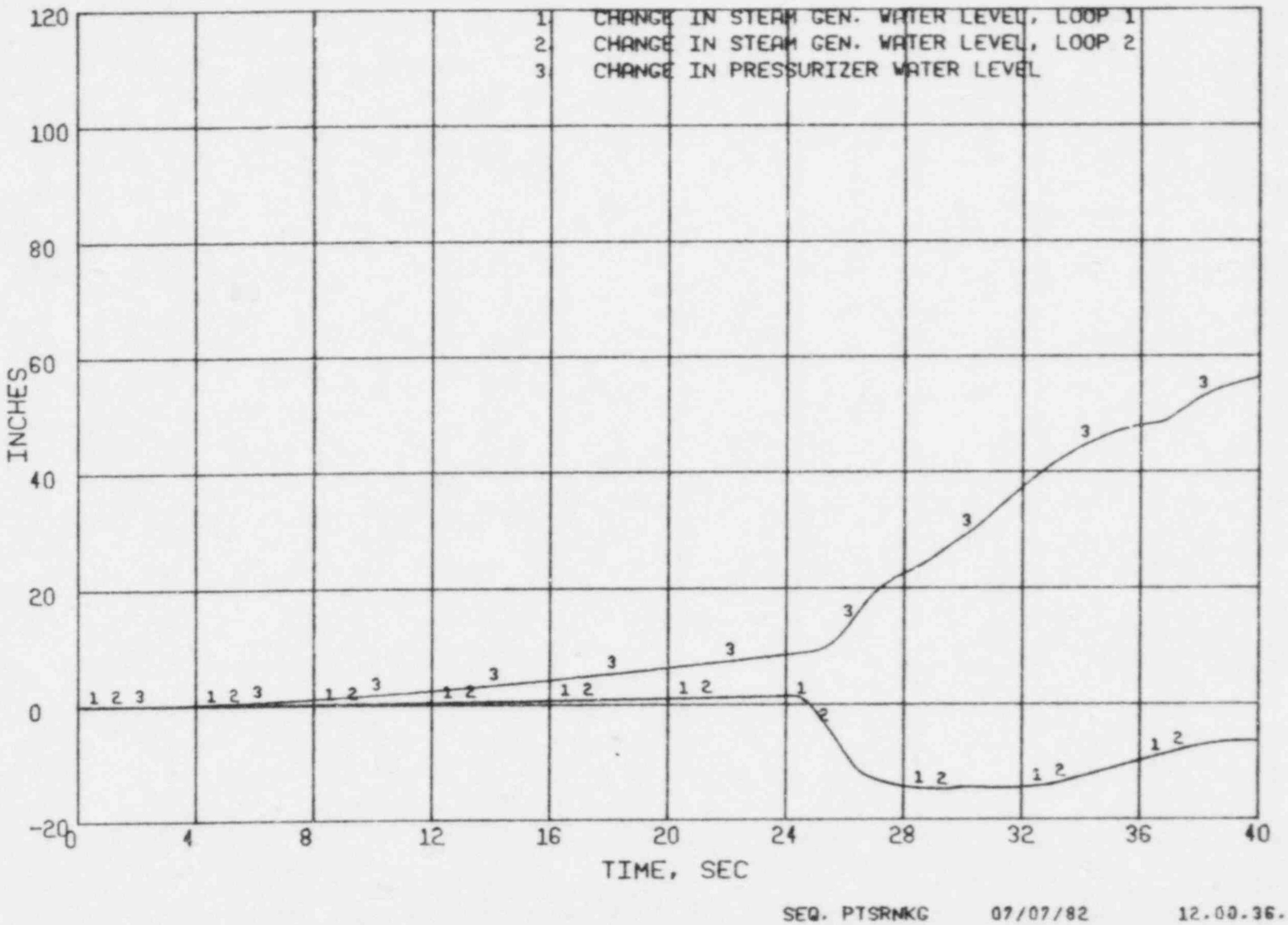


Figure 3.12 Uncontrolled Rod Withdrawal (Slow) - Level Changes

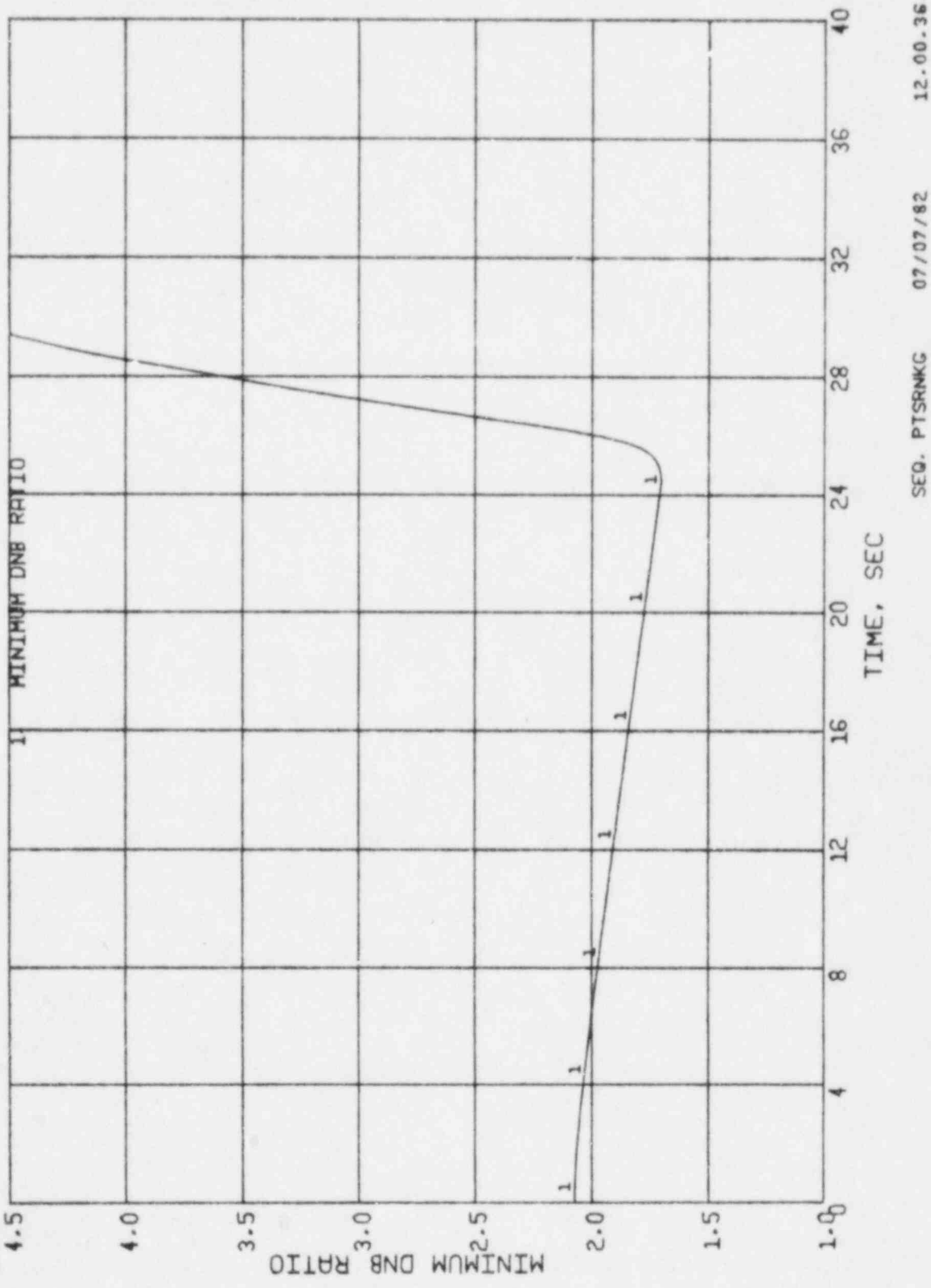


Figure 3.13 Uncontrolled Rod Withdrawal (Slow) - Minimum DNB Ratio

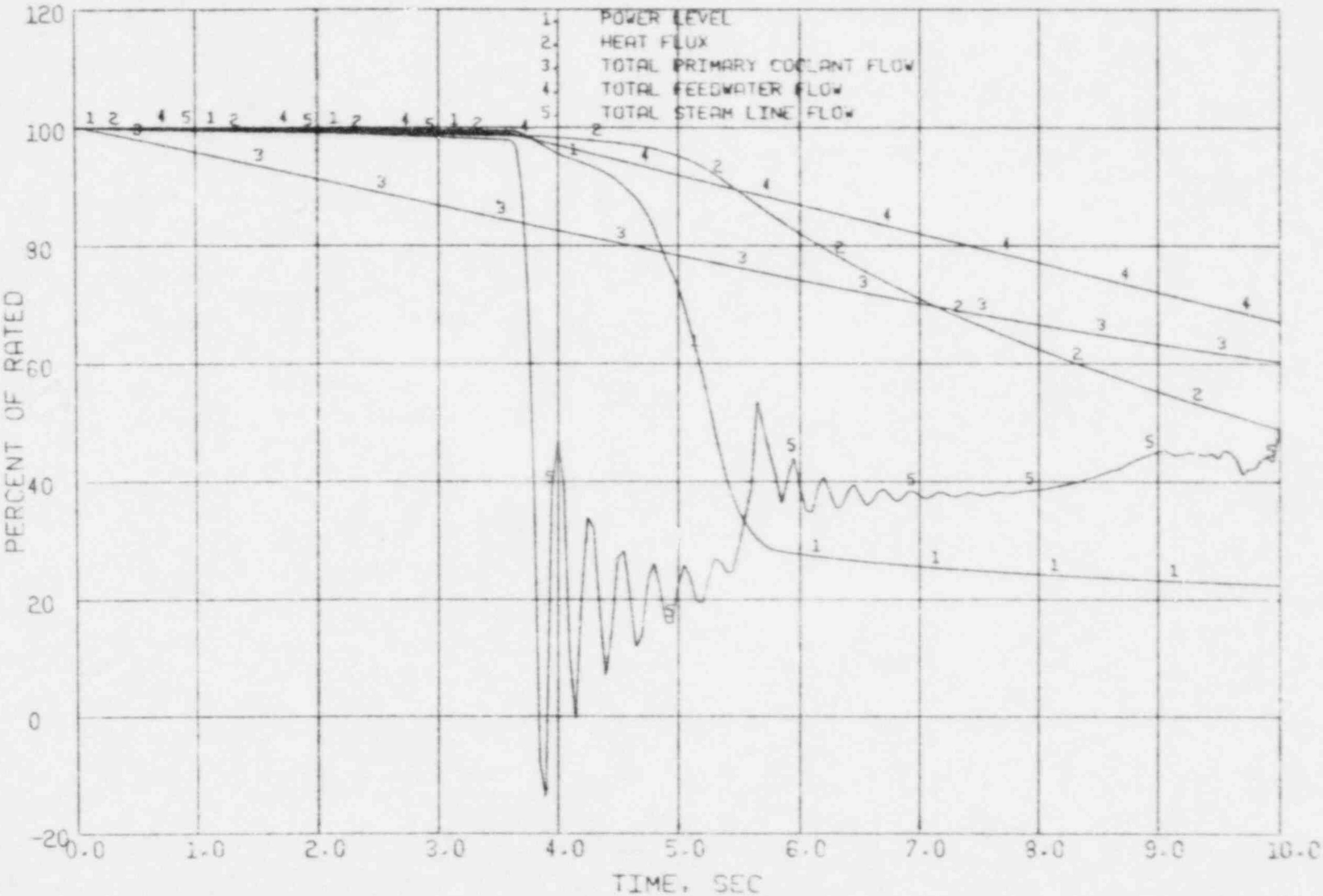


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Figure 3.14 Loss of Coolant Flow - Power, Heat Flux and System Flows

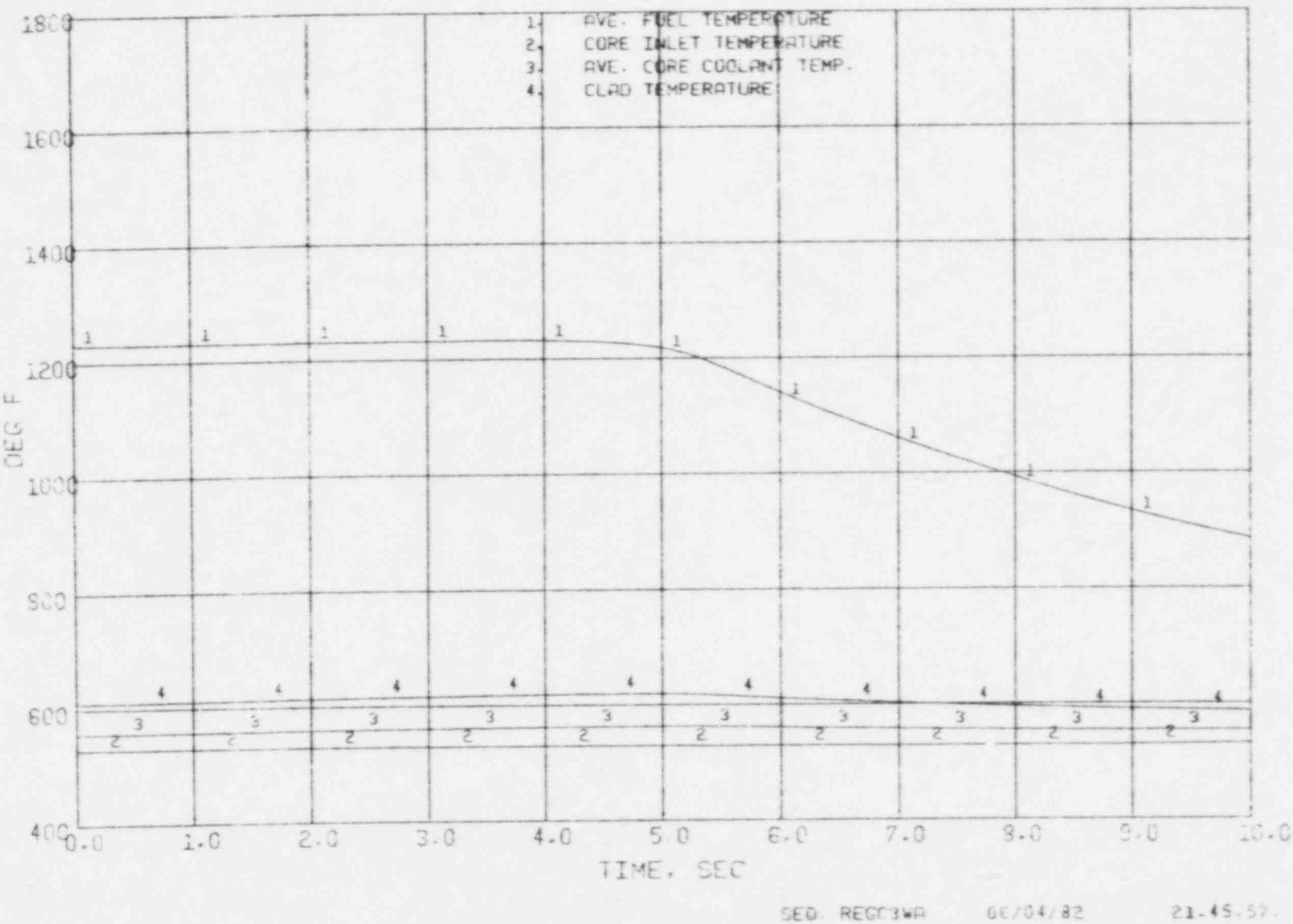


Figure 3.15 Loss of Coolant Flow - Core Temperature Response

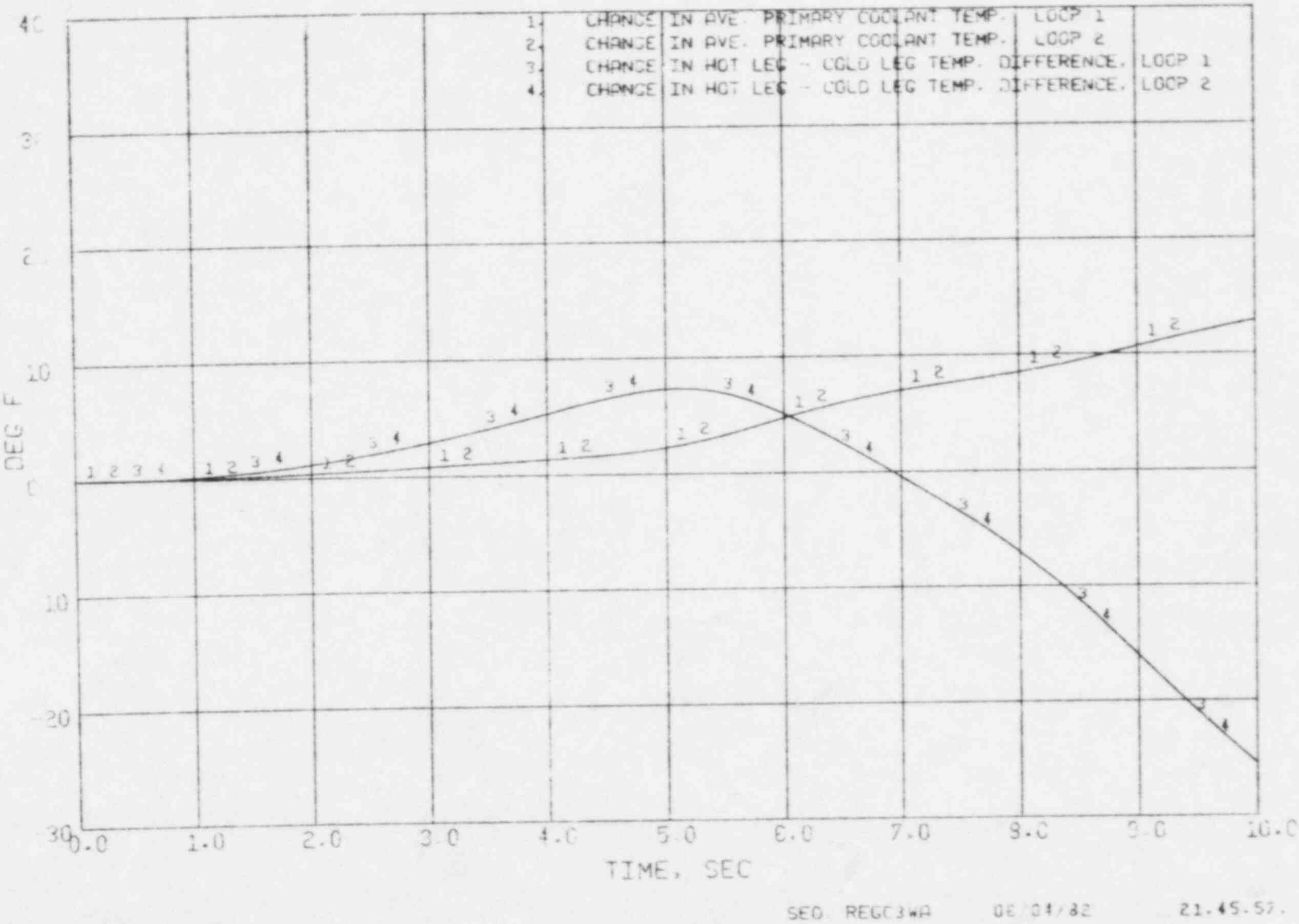


Figure 3.16 Loss of Coolant Flow - Primary Loop Temperature Changes

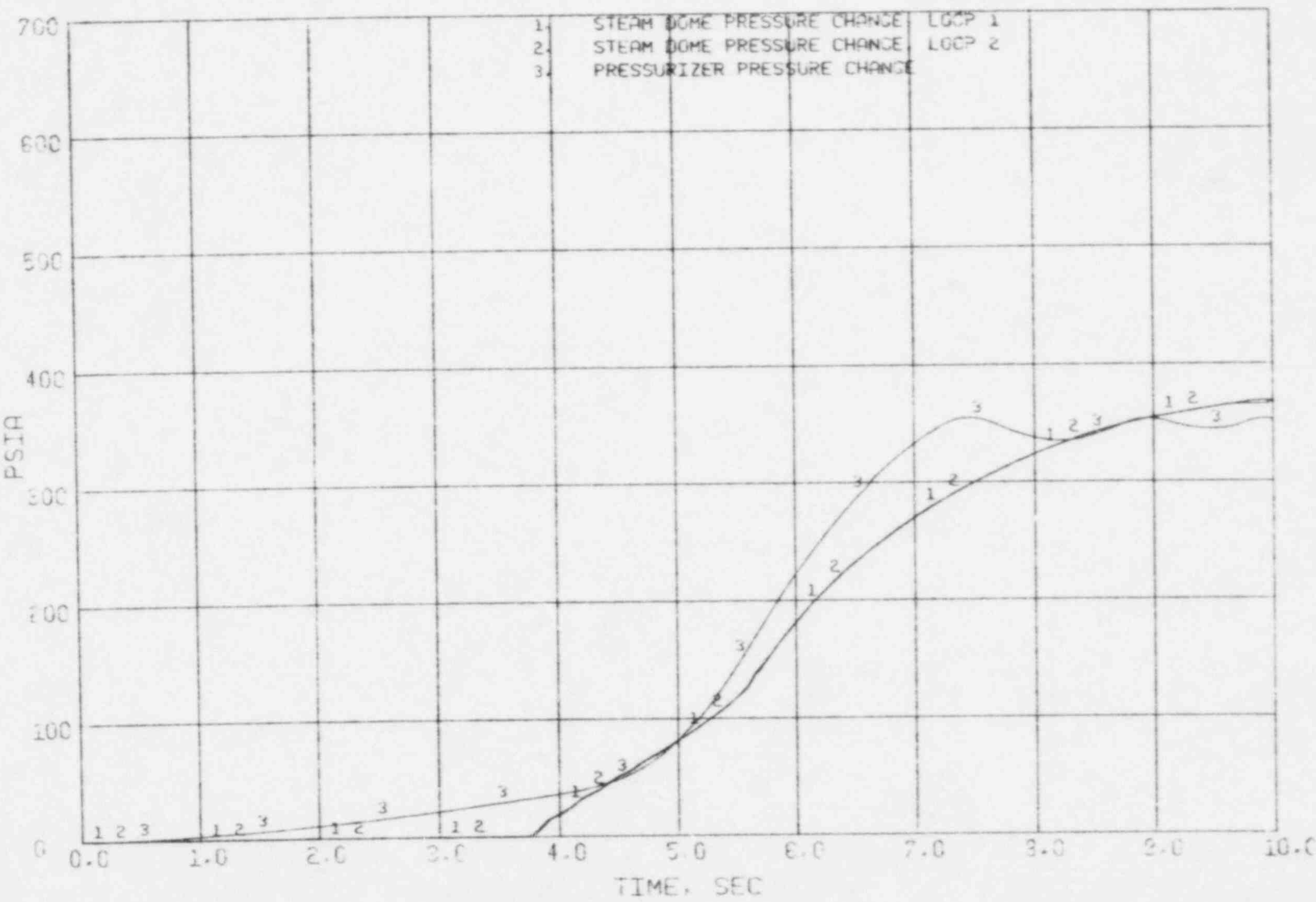


Figure 3.17 Loss of Coolant Flow - Pressure Changes

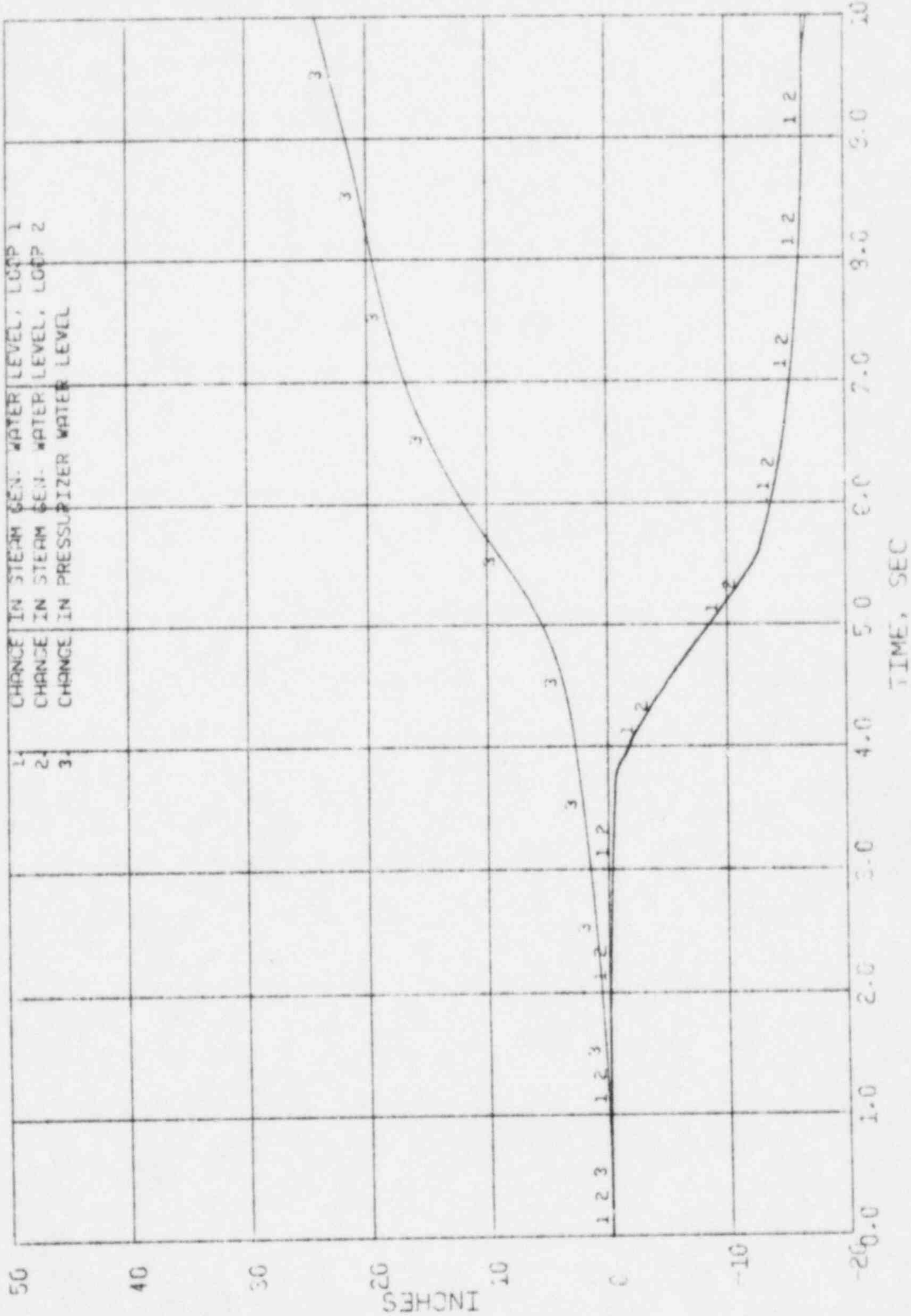


Figure 3.18 Loss of Coolant Flow - Level Changes

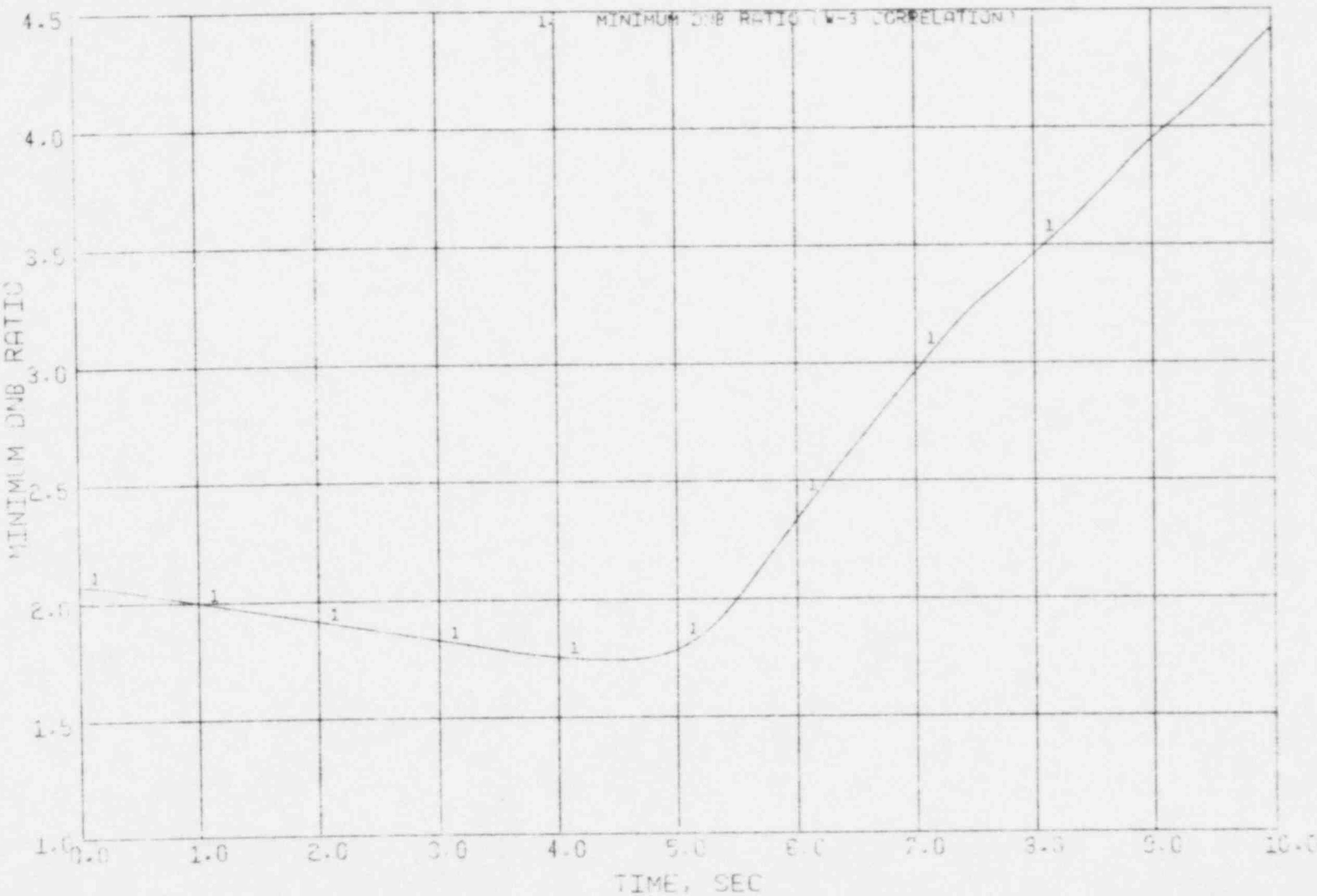


Figure 3.19 Loss of Coolant Flow - Minimum DNB Ratio

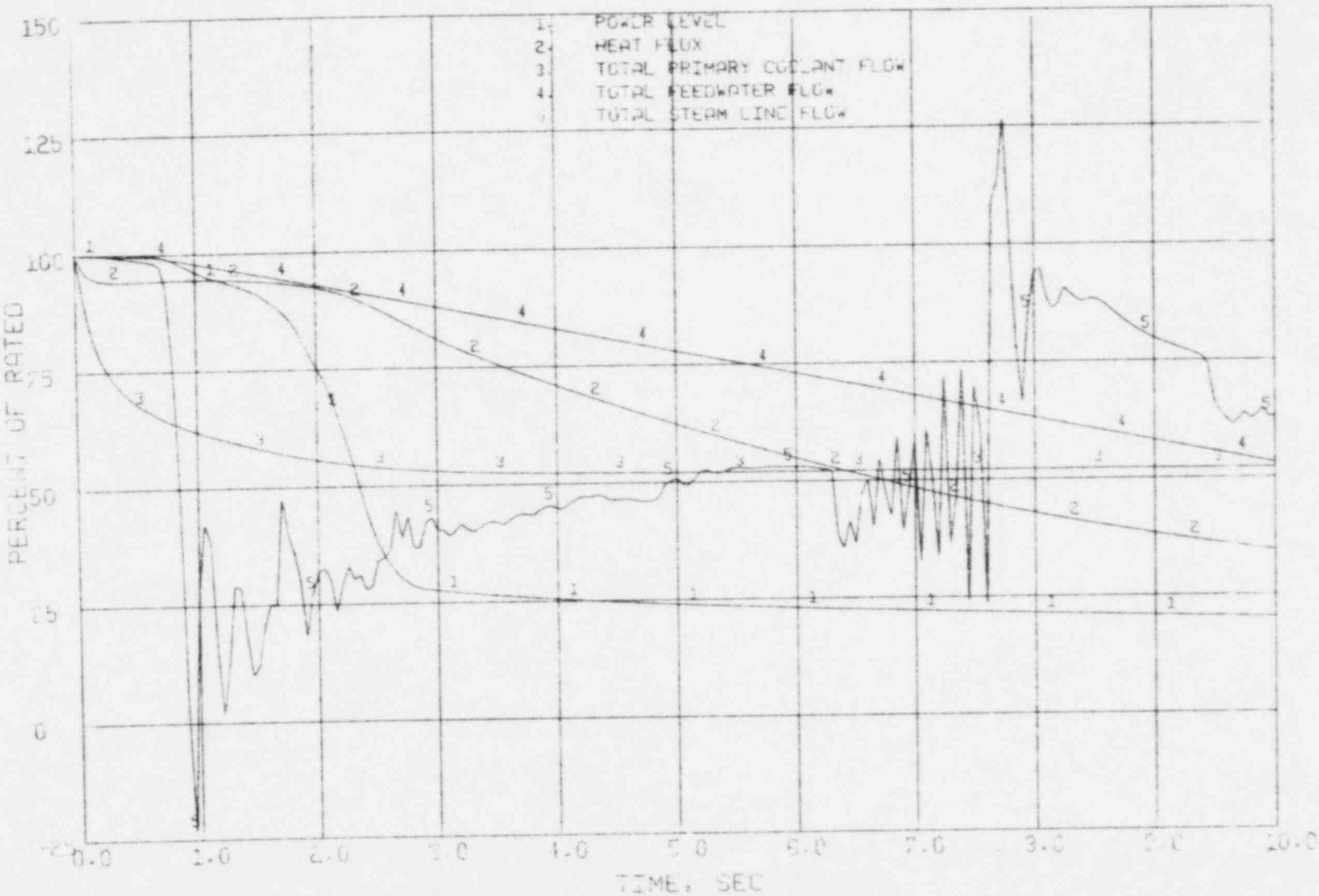


Figure 3.20 Locked Rotor - Power, Heat Flux and System Flows

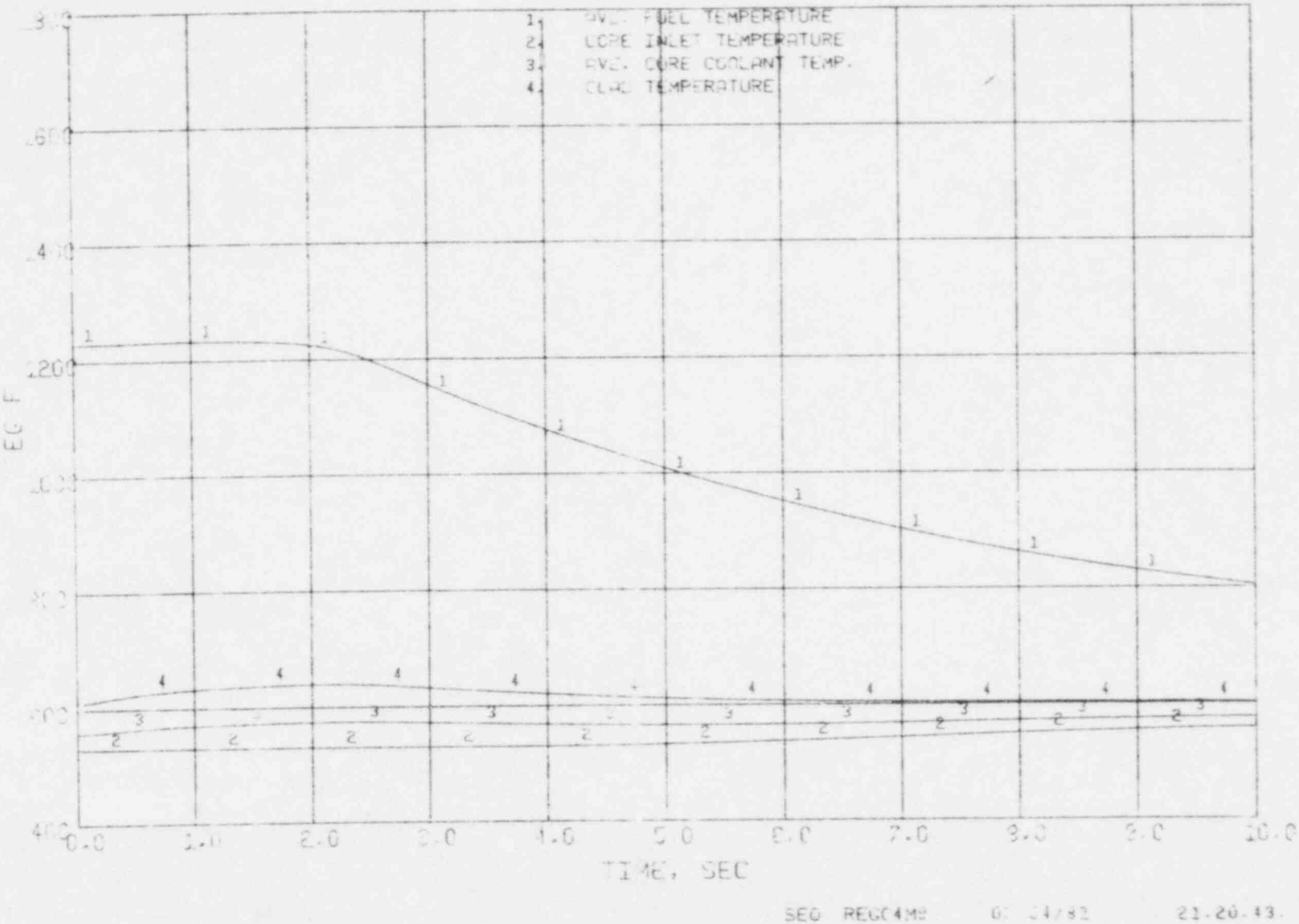


Figure 3.21 Locked Rotor - Core Temperature Response

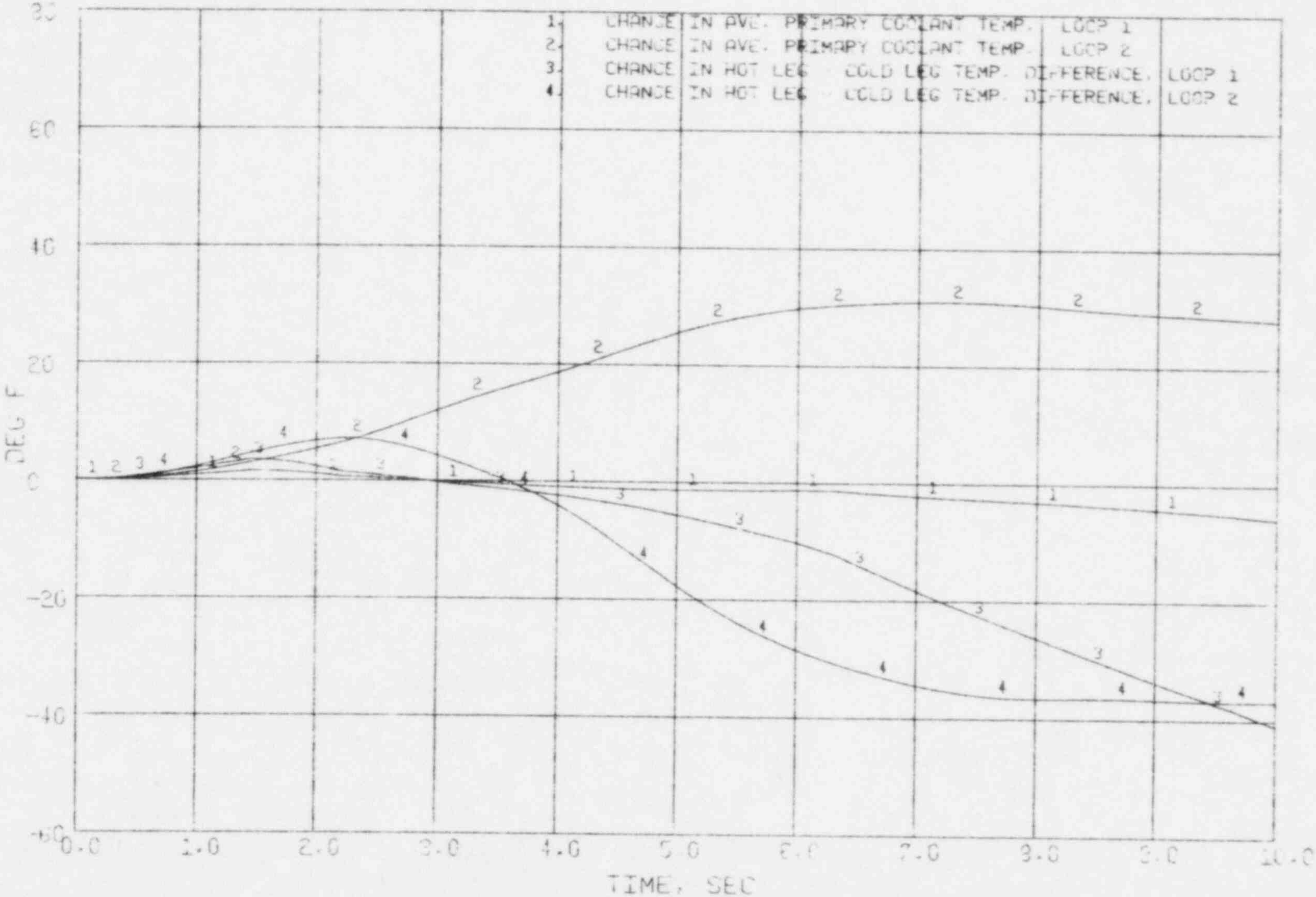


Figure 3.22 Locked Rotor - Primary Loop Temperature Changes

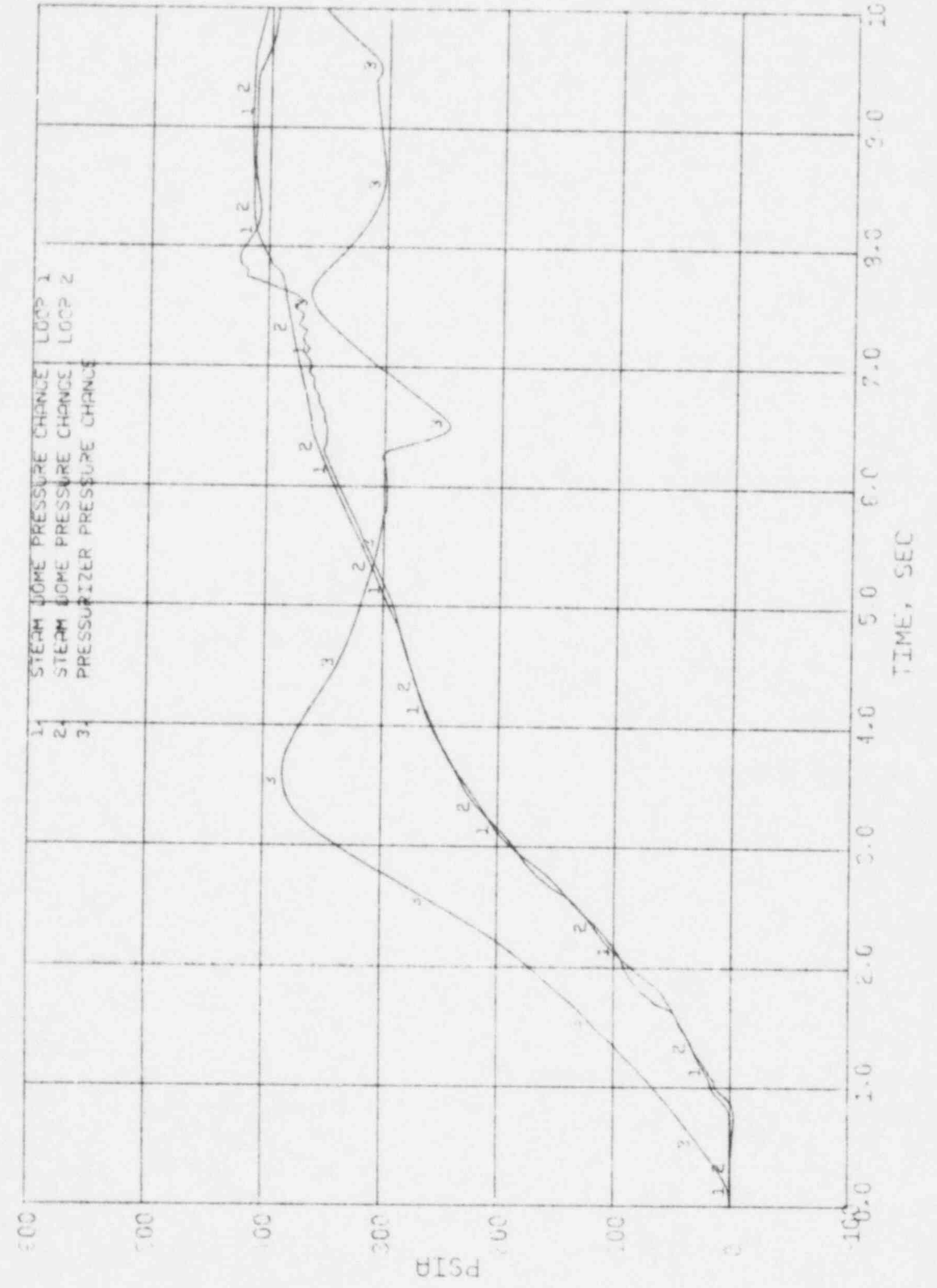


Figure 3.23 Locked Rotor - Pressure Changes

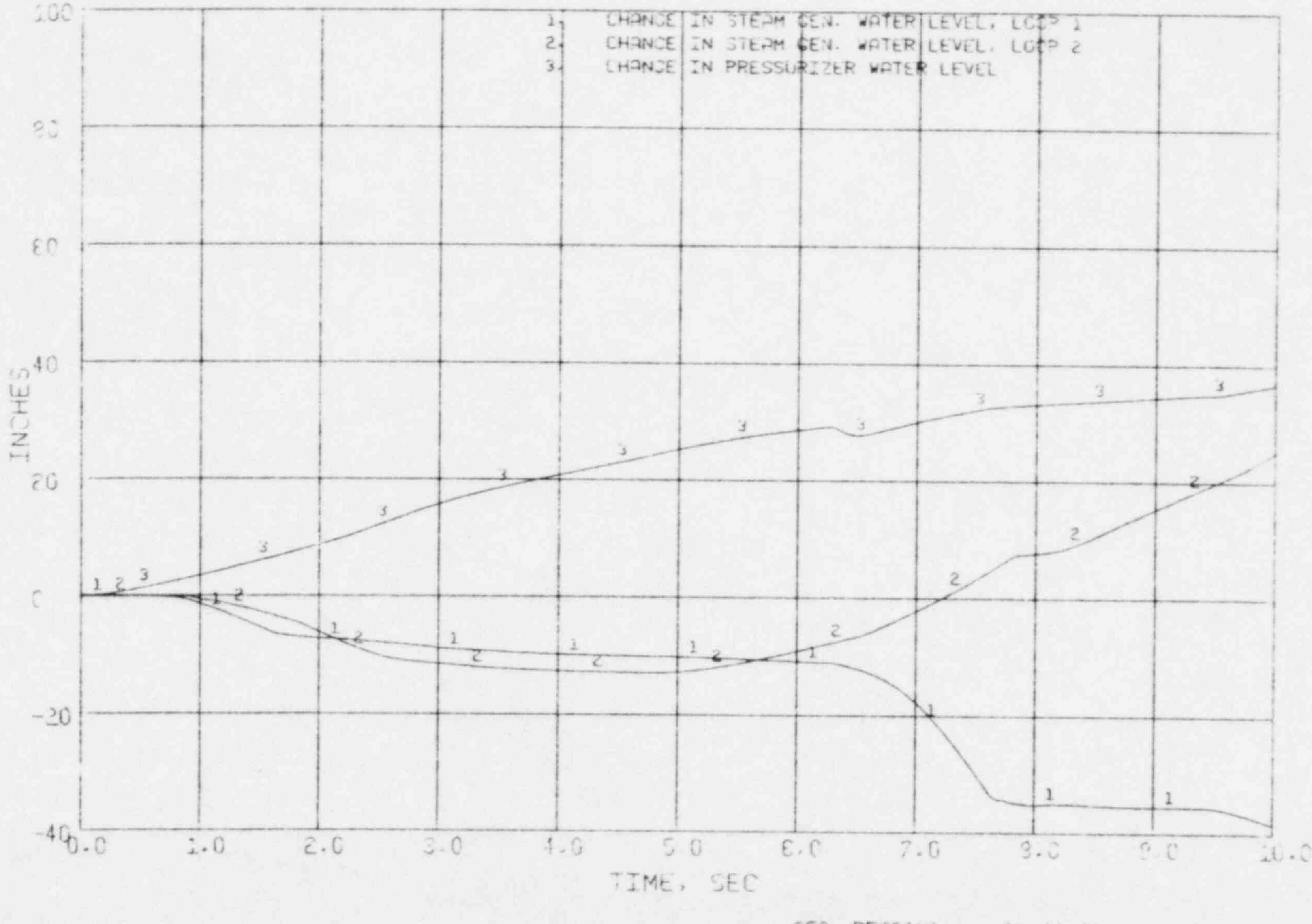


Figure 3.24 Locked Rotor - Level Changes

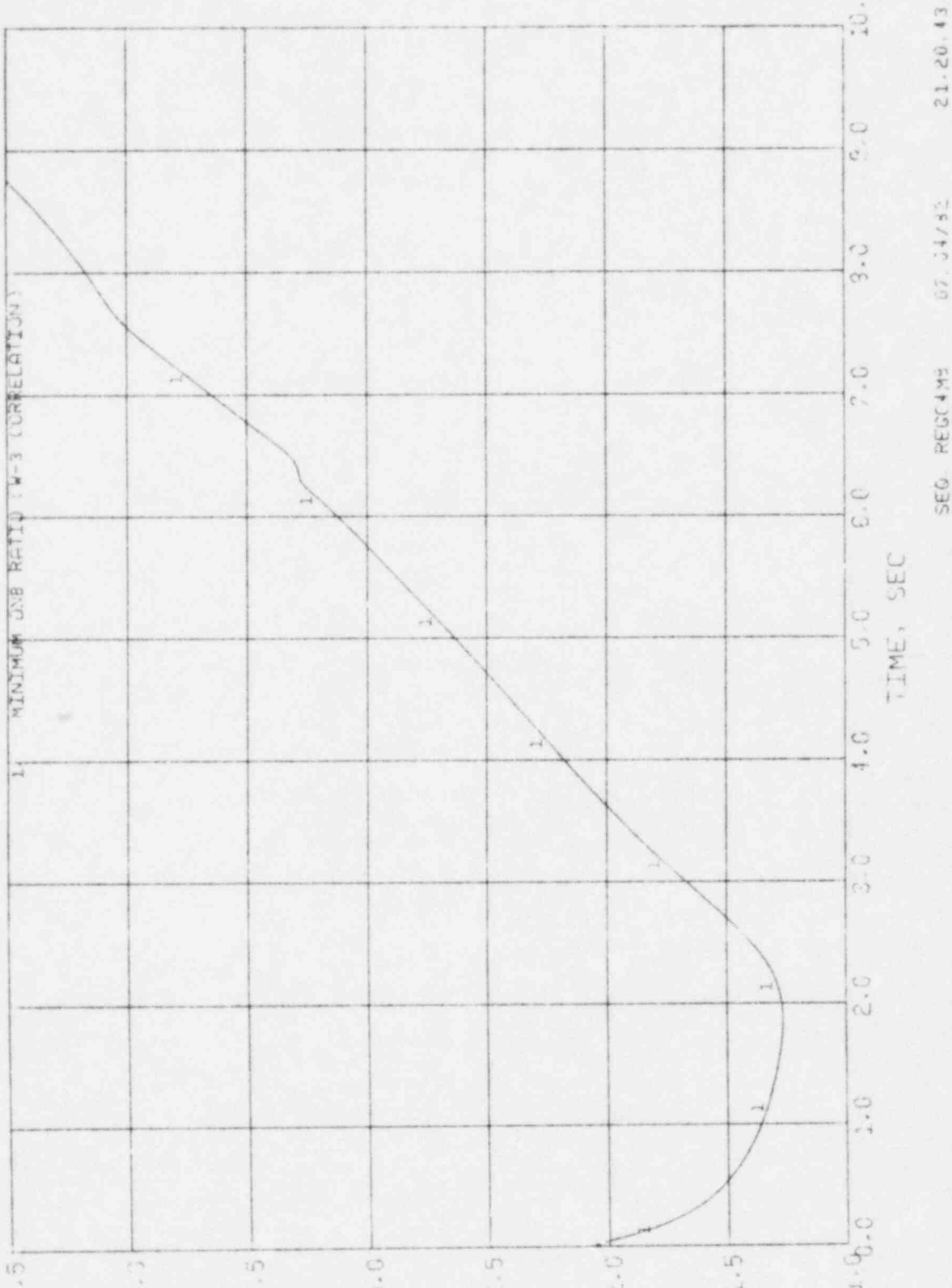


Figure 3.25 Locked Rotor - Minimum DNB Ratio

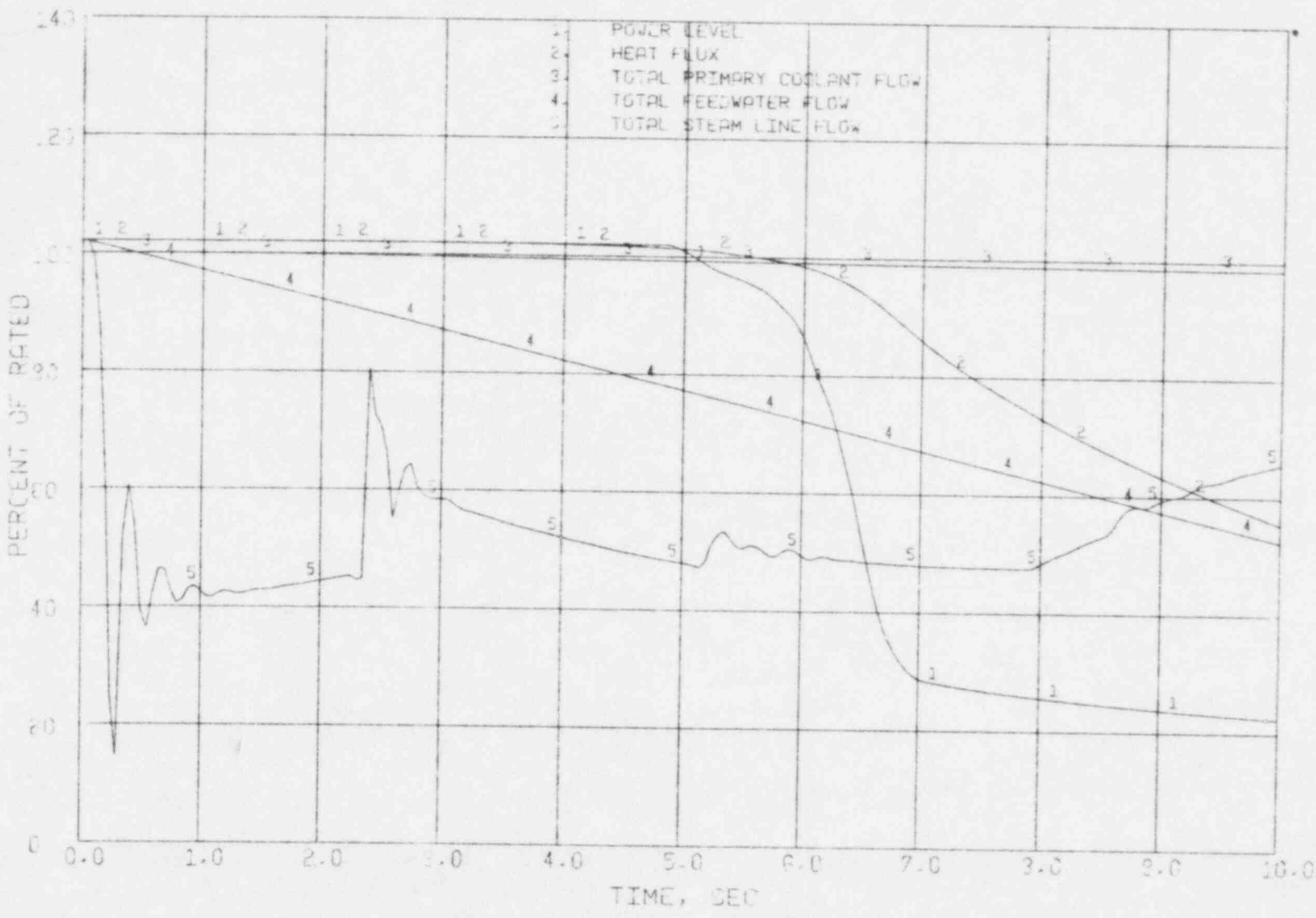


Figure 3.26 Loss of Electric Load
Power, Heat Flux and System Flows

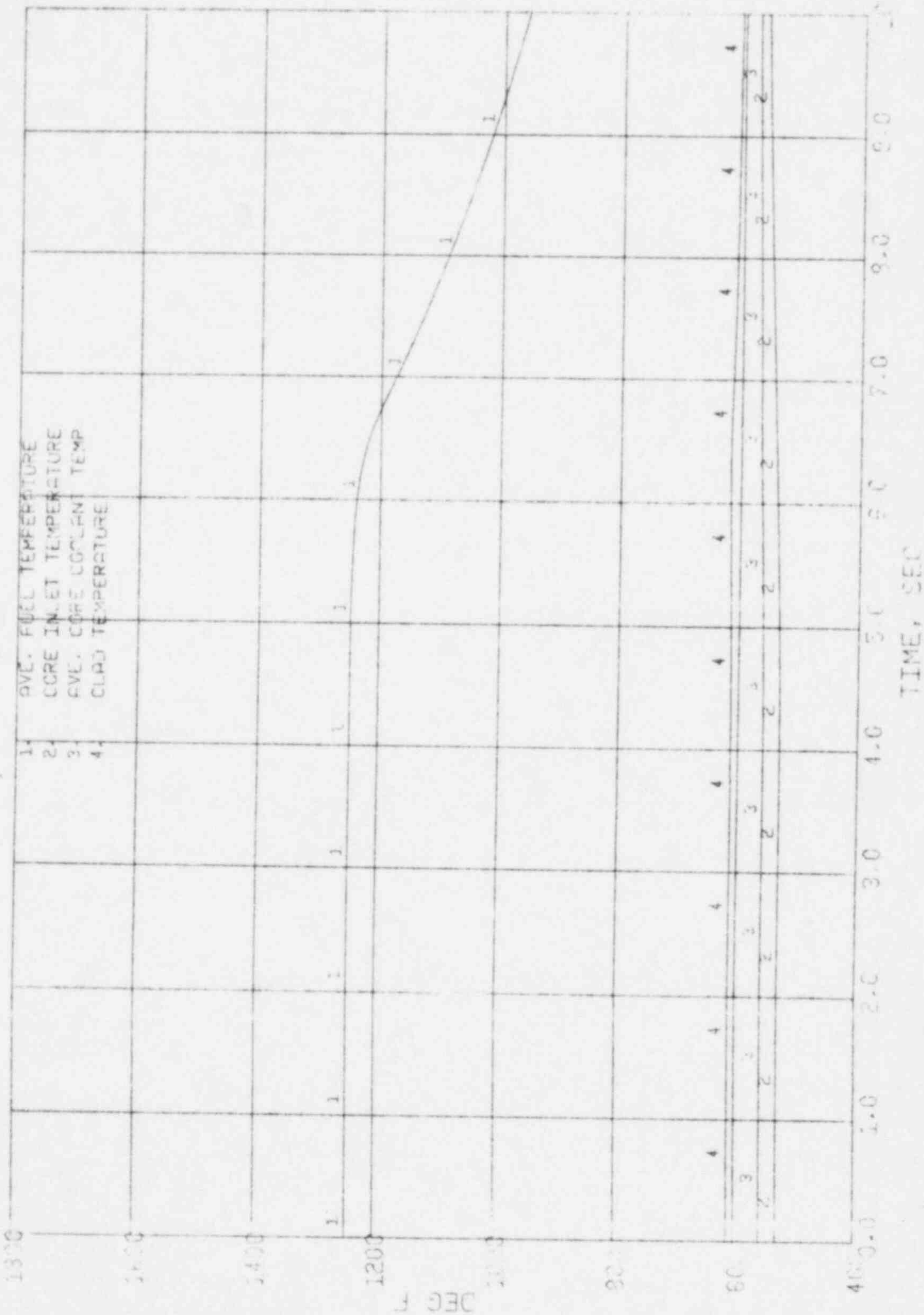


Figure 3.27 Loss of Electric Load - Core Temperature Response

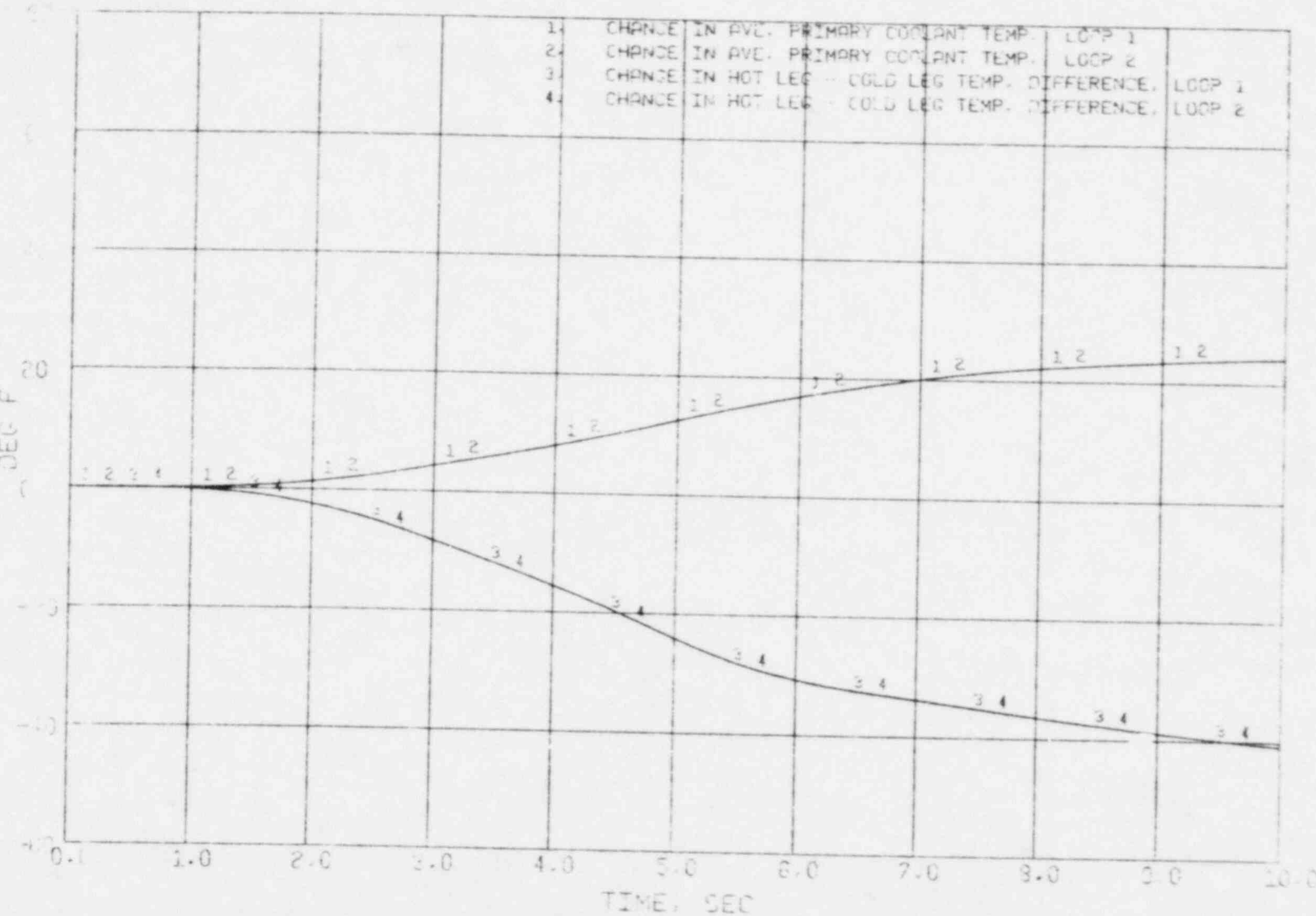


Figure 3.28 Loss of Electric Load - Primary Loop Temperature Changes

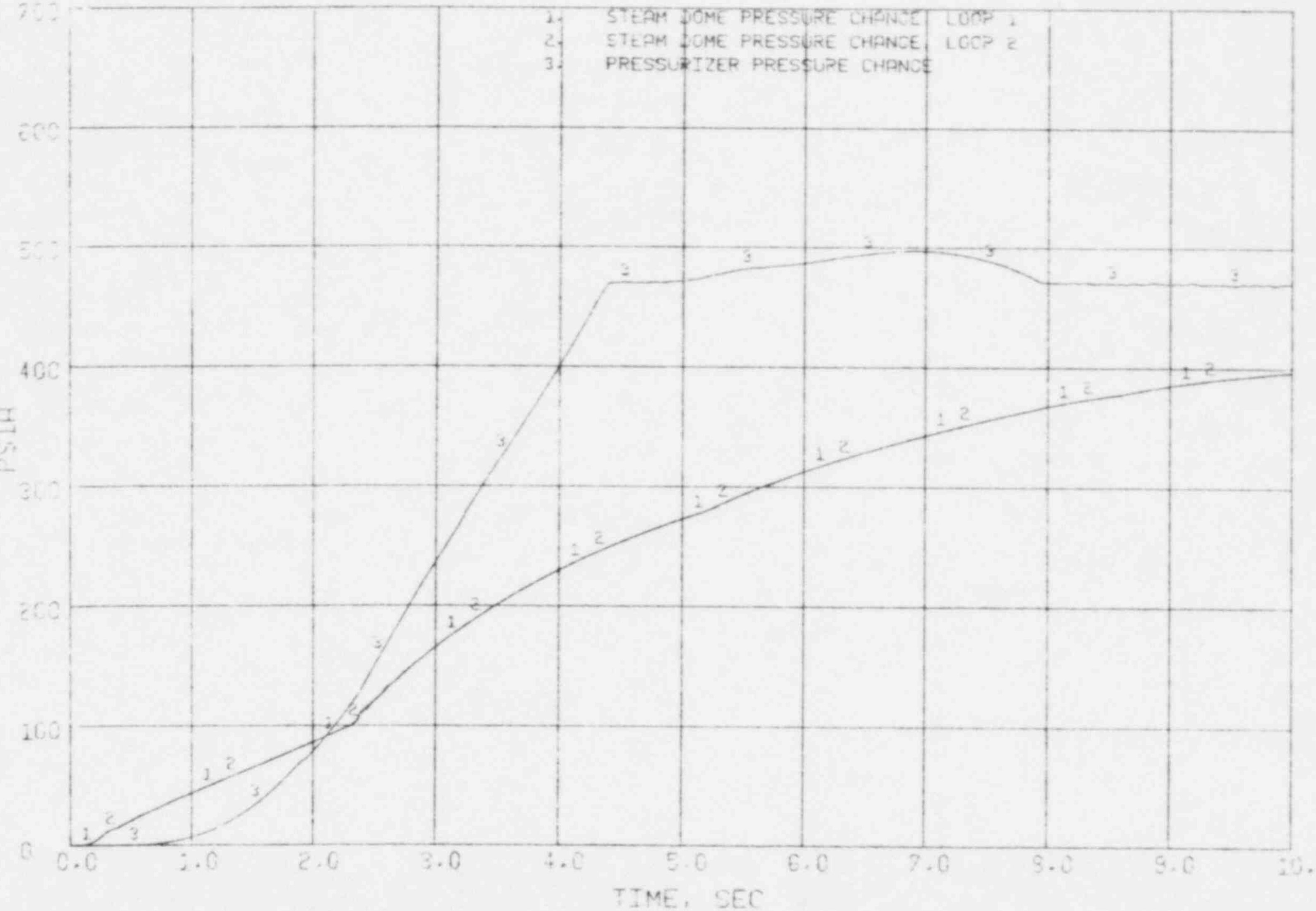


Figure 3.29 Loss of Electric Load - Pressure Changes

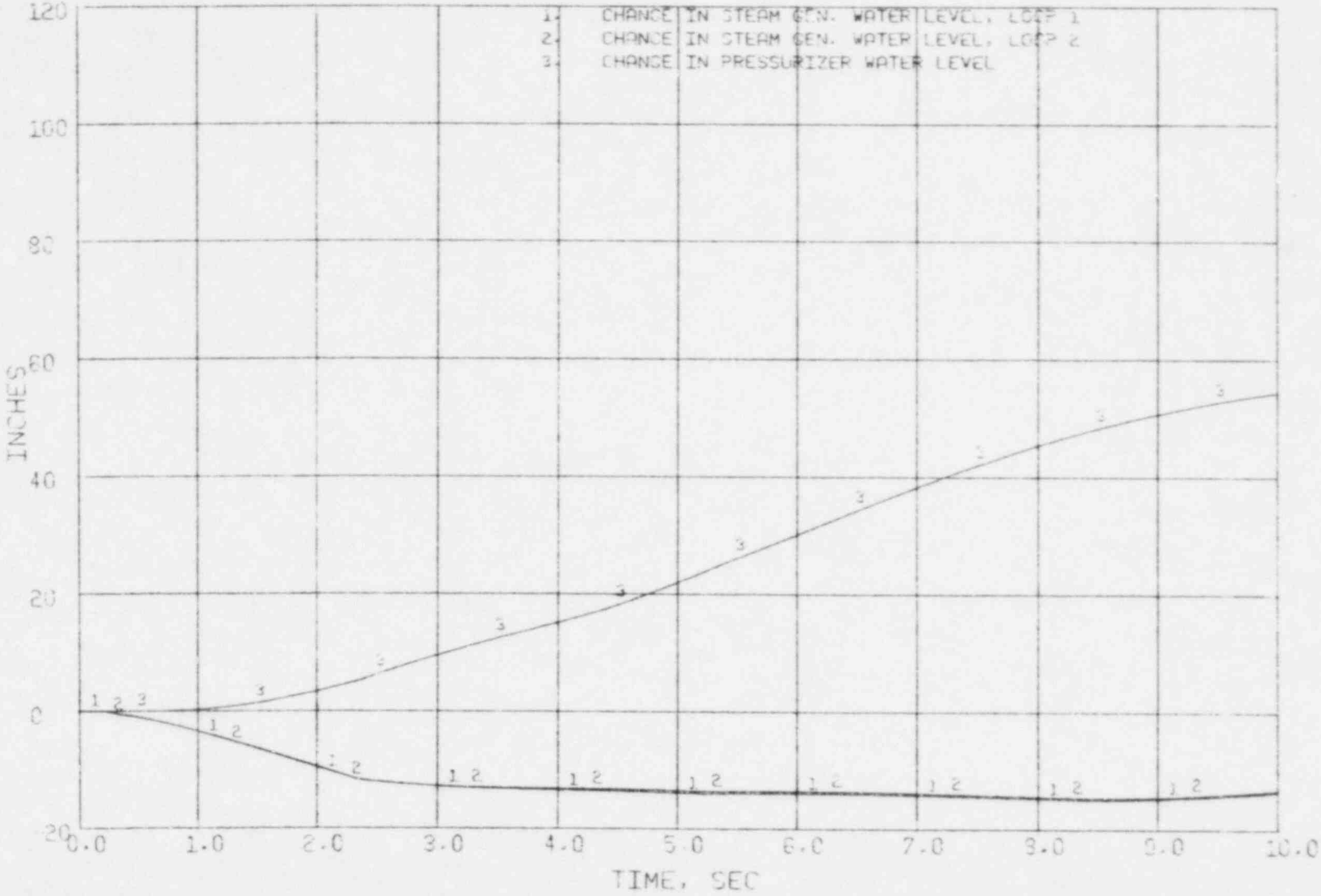


Figure 3.30 Loss of Electric Load - Level Changes

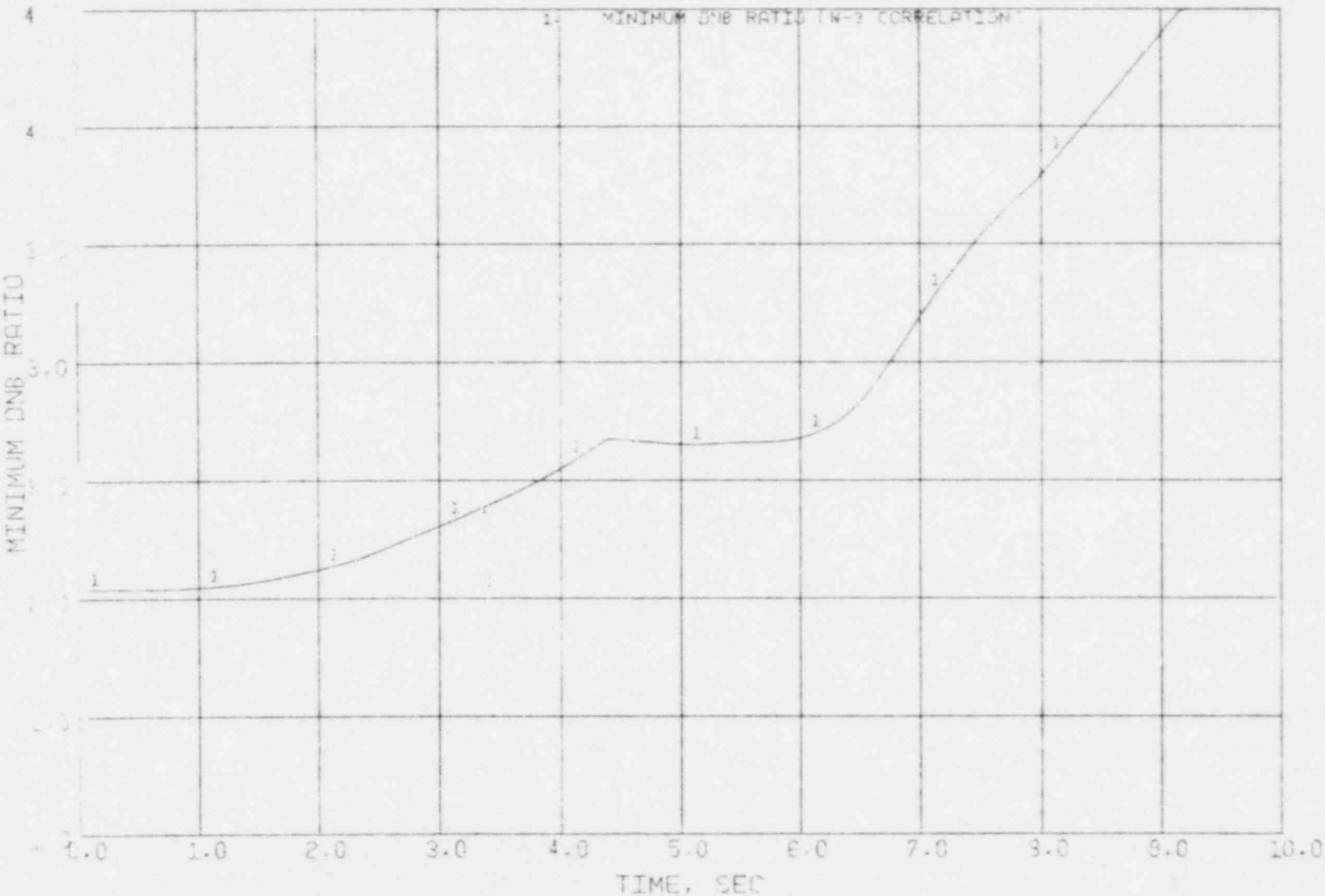


Figure 3.31 Loss of Electric Load - Minimum DNB Ratio

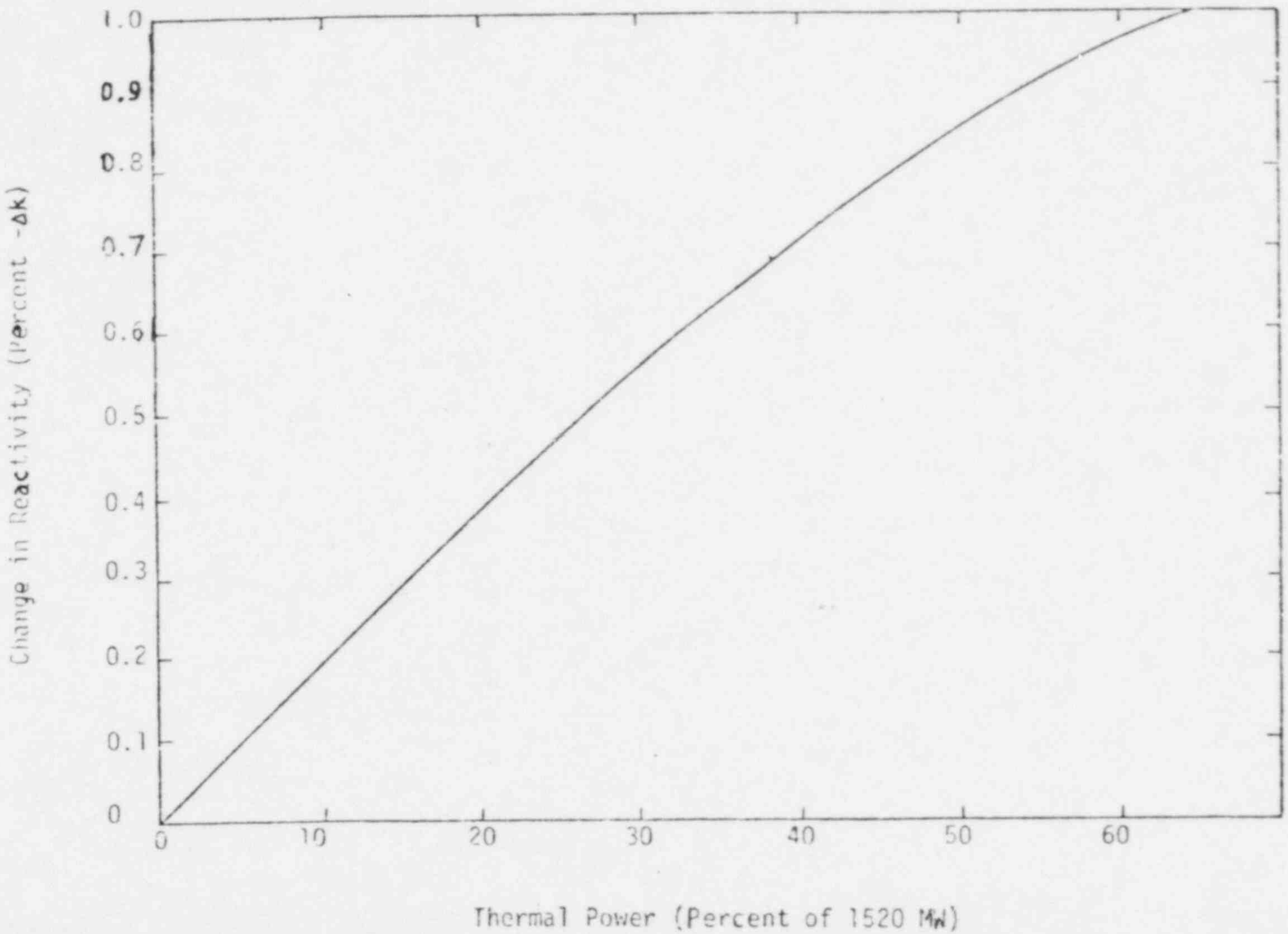


Figure 3.32 Variation of Reactivity with Power
At Constant Core Average Temperature

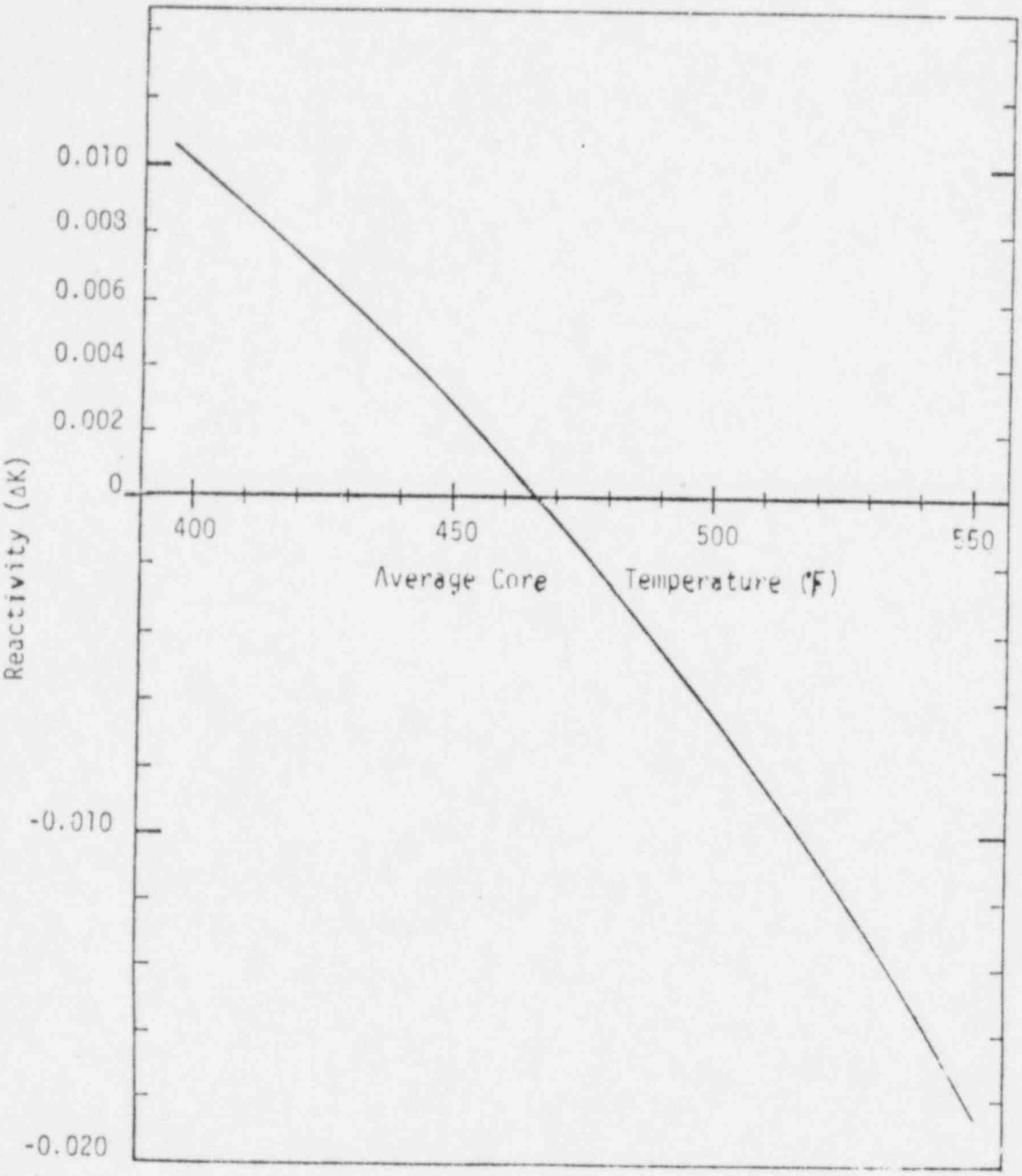


Figure 3.33 Variation of Reactivity with Core Average Temperature at the End of the Cycle

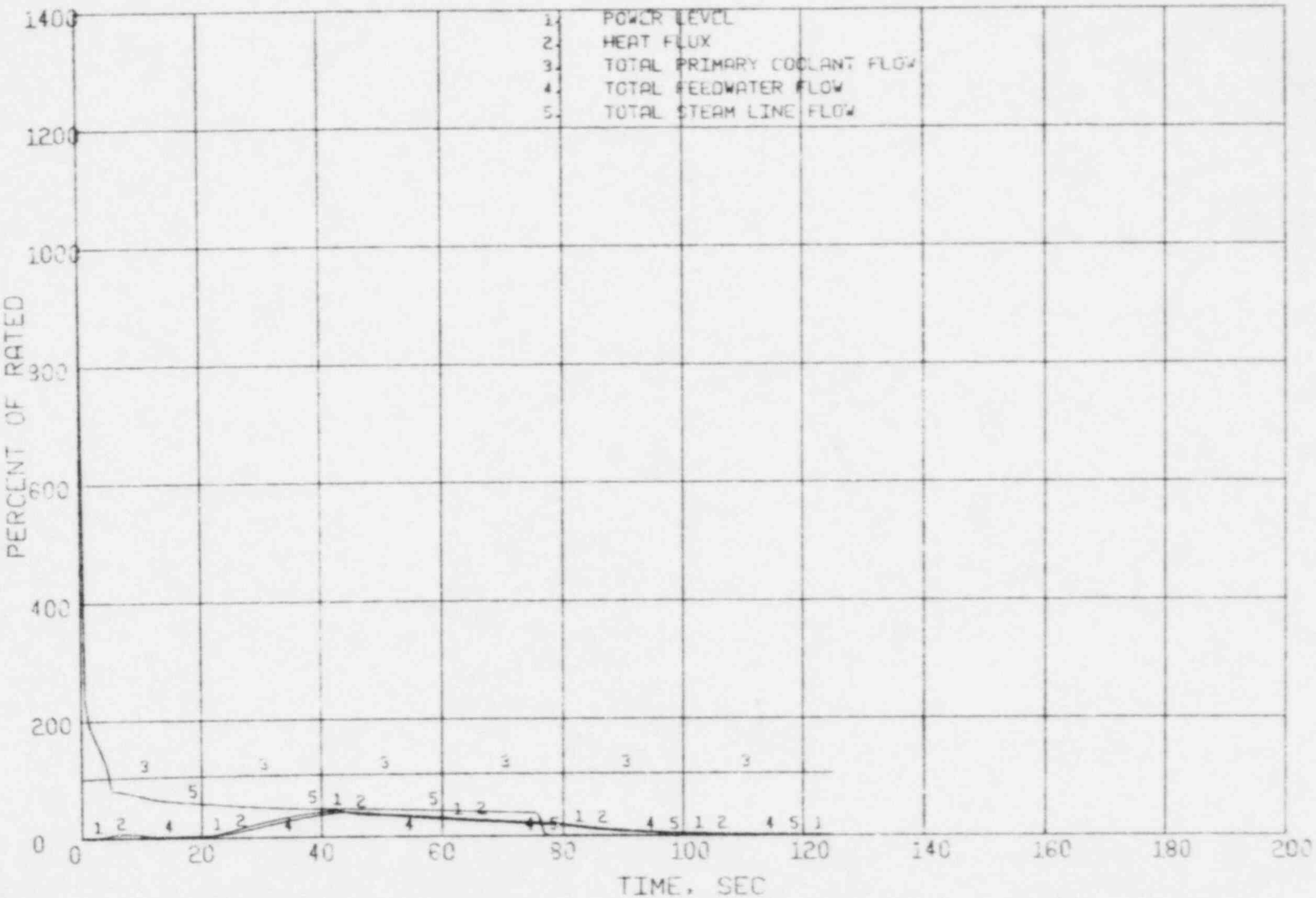


Figure 3.34 Large Steam Line Break (547°) -
Power, Heat Flux and System Flows

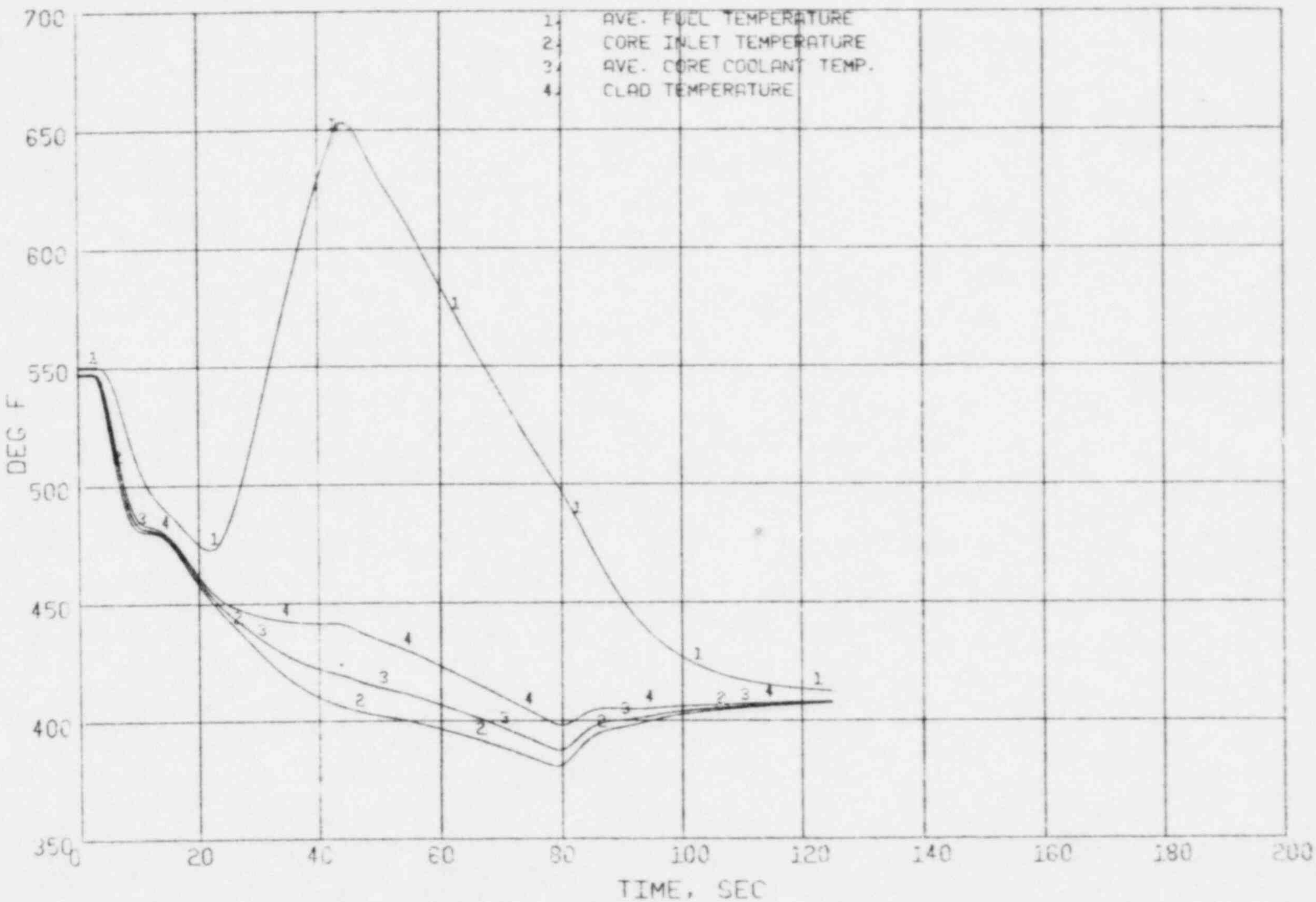


Figure 3.35 Large Steam Line Break ($547^0$) - Core Temperature Response

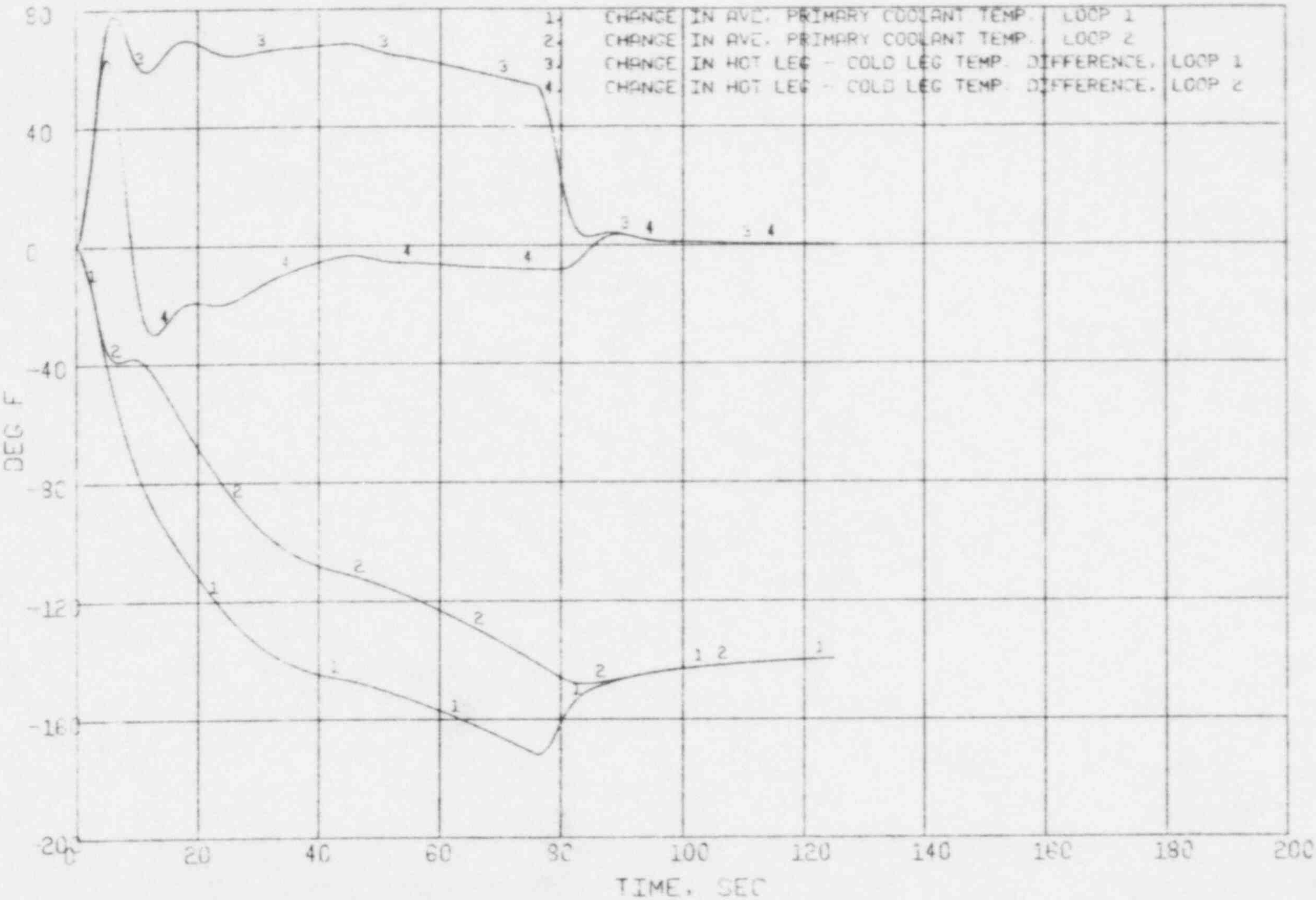


Figure 3.36 Large Steam Line Break (547$^{b}$) - Primary Loop Temperature Changes

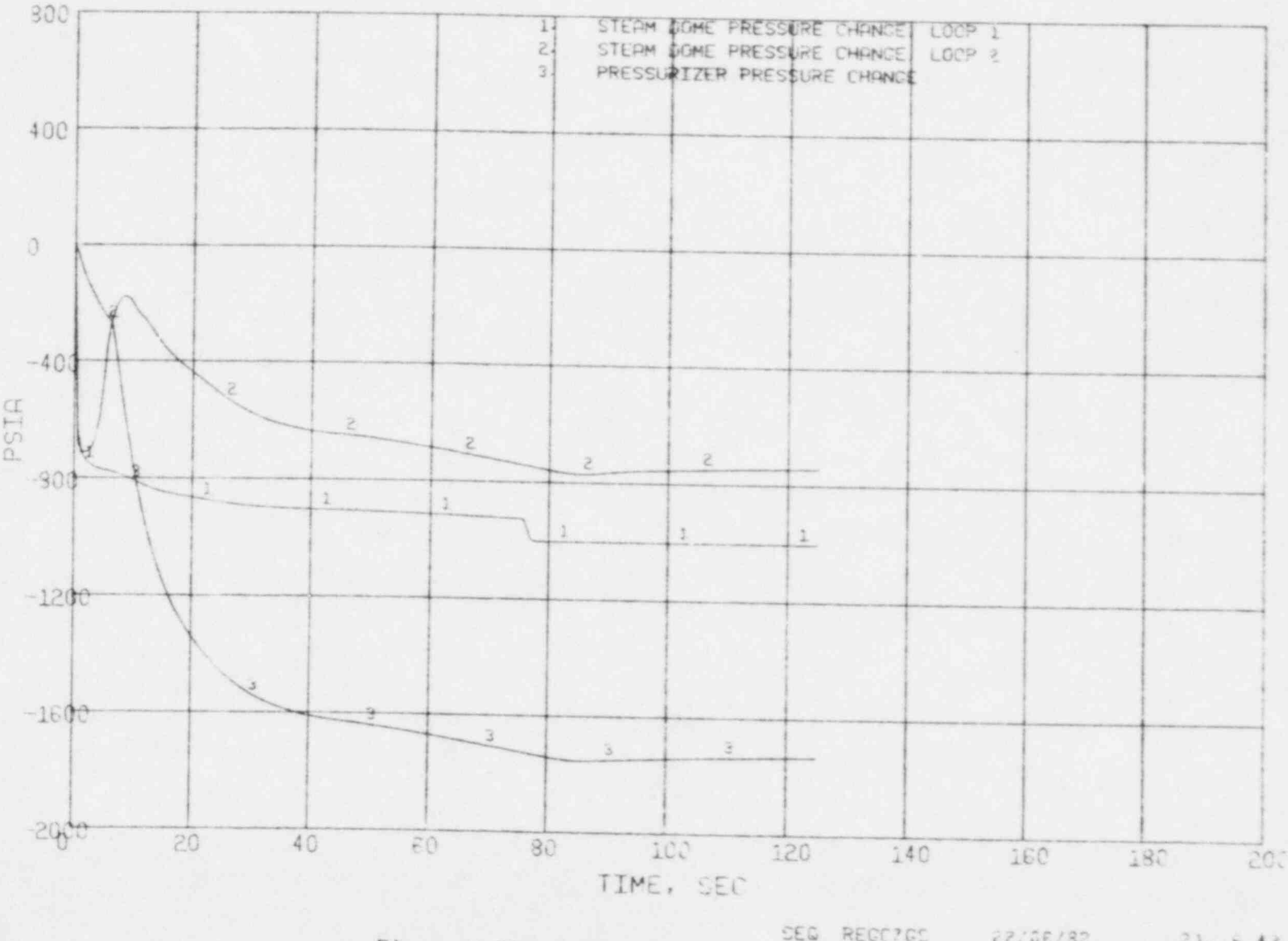


Figure 3.37 Large Steam Line Break (547$^{0}$) - Pressure Changes

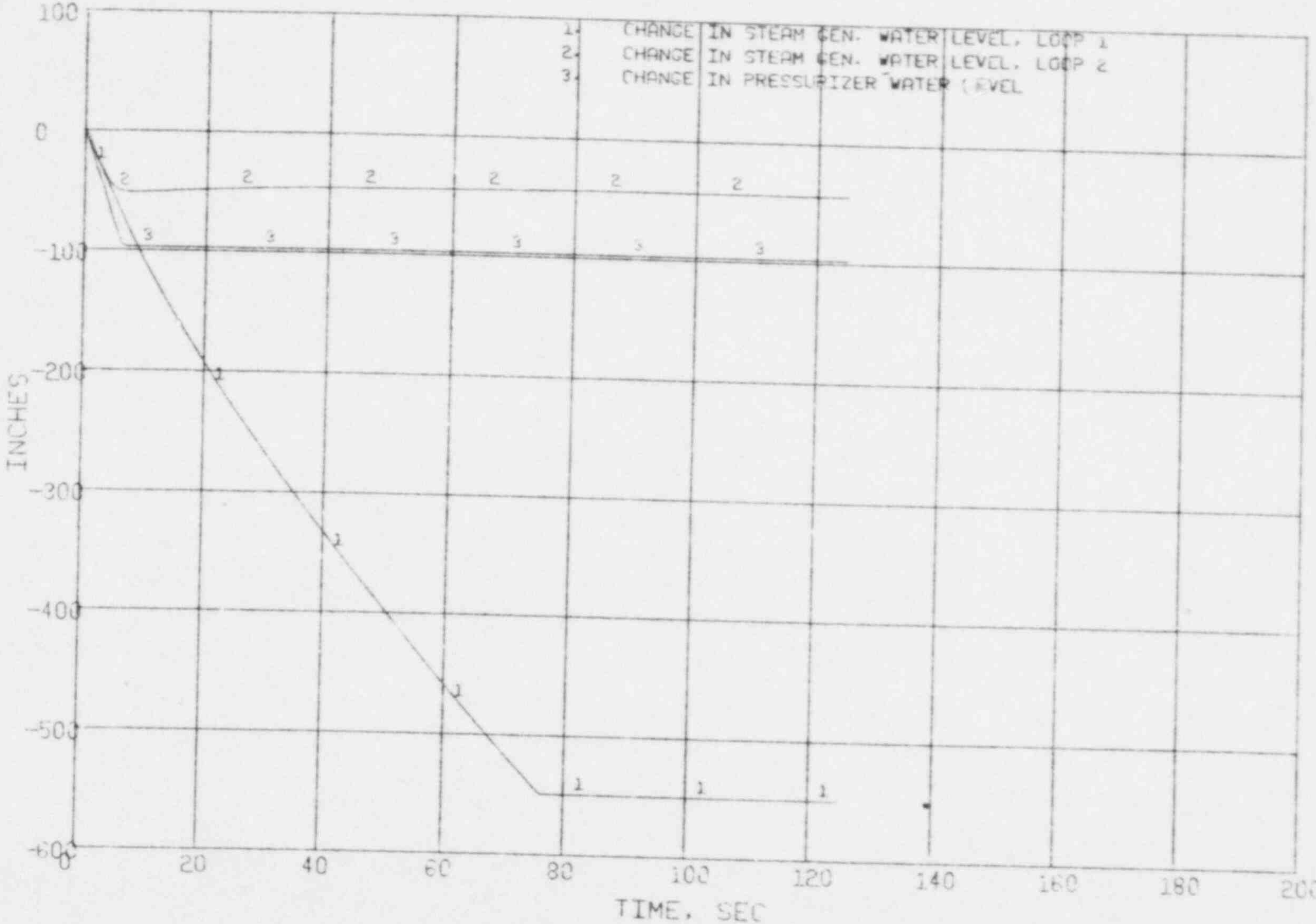


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Figure 3.38 Large Steam Line Break ($547^{o}$) - Level Changes

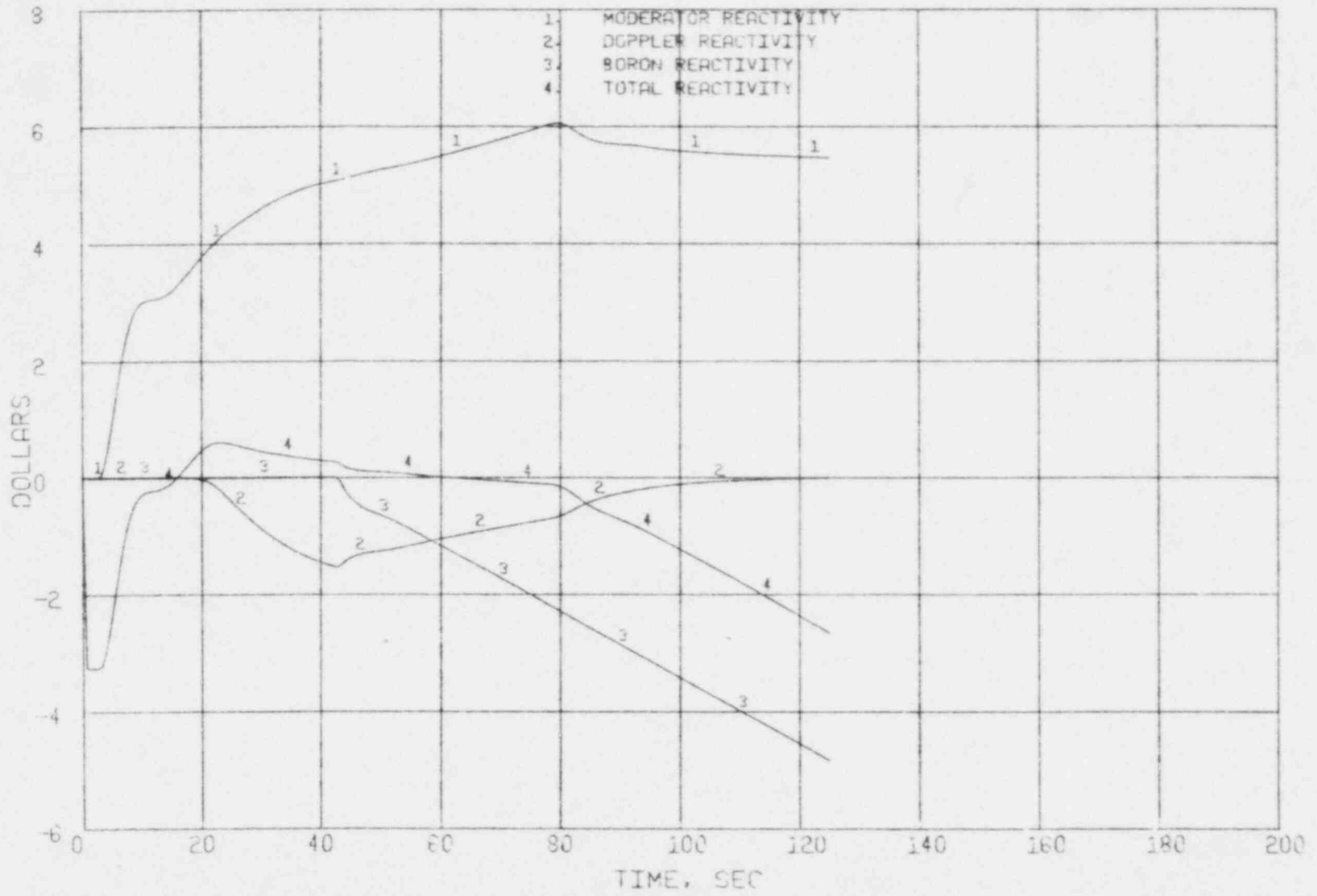



Figure 3.39 Large Steam Line Break (547°) - Reactivity

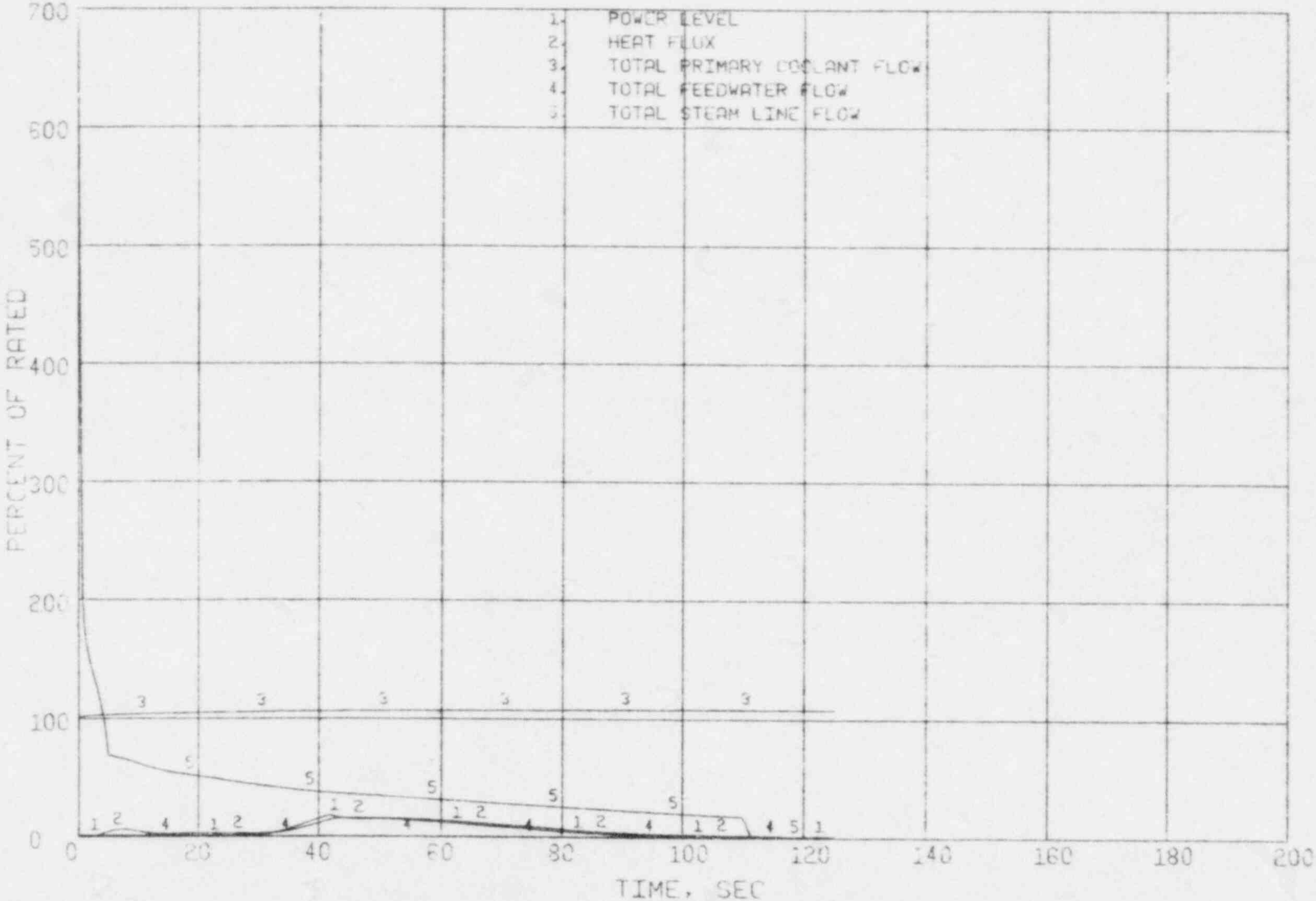
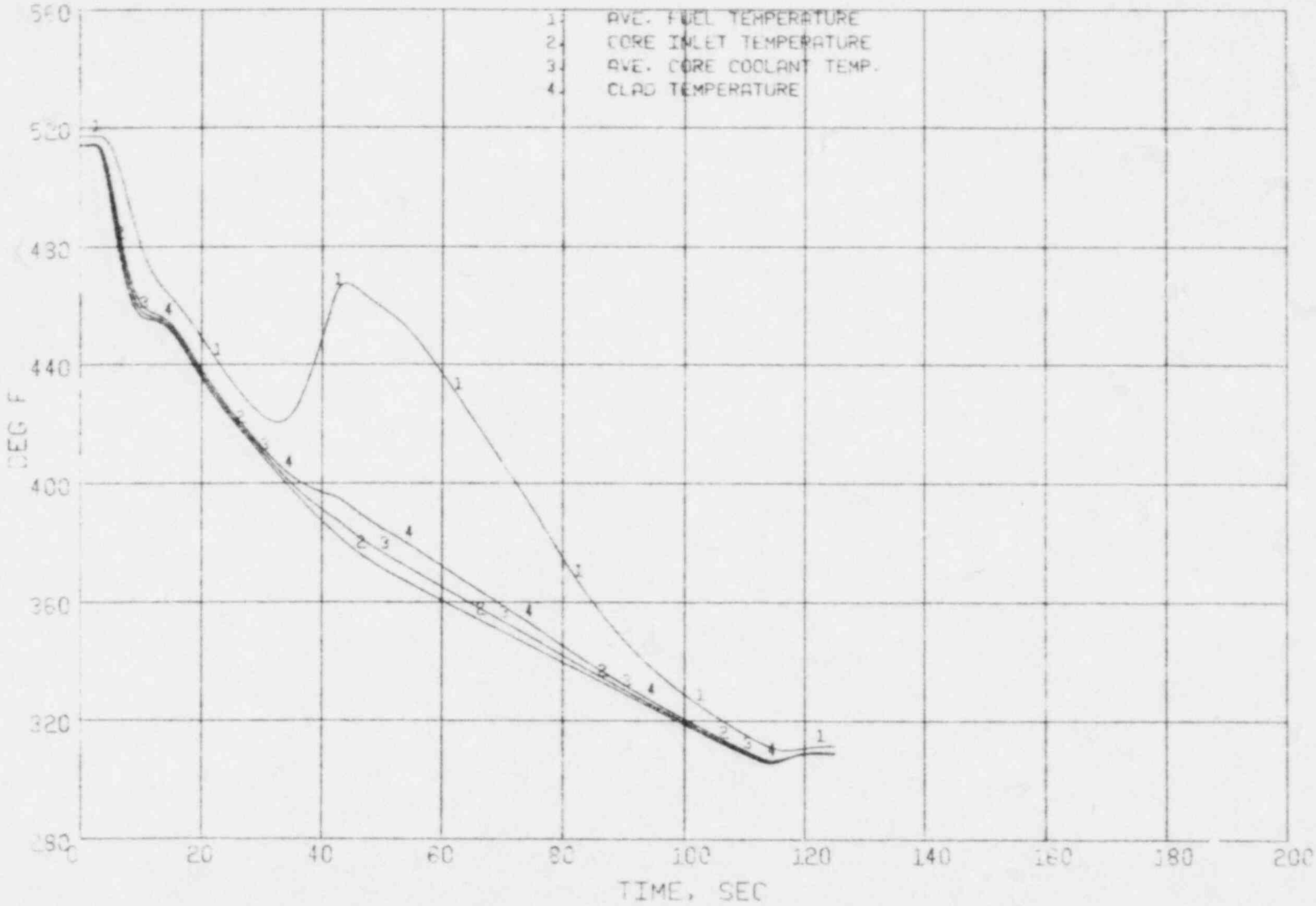
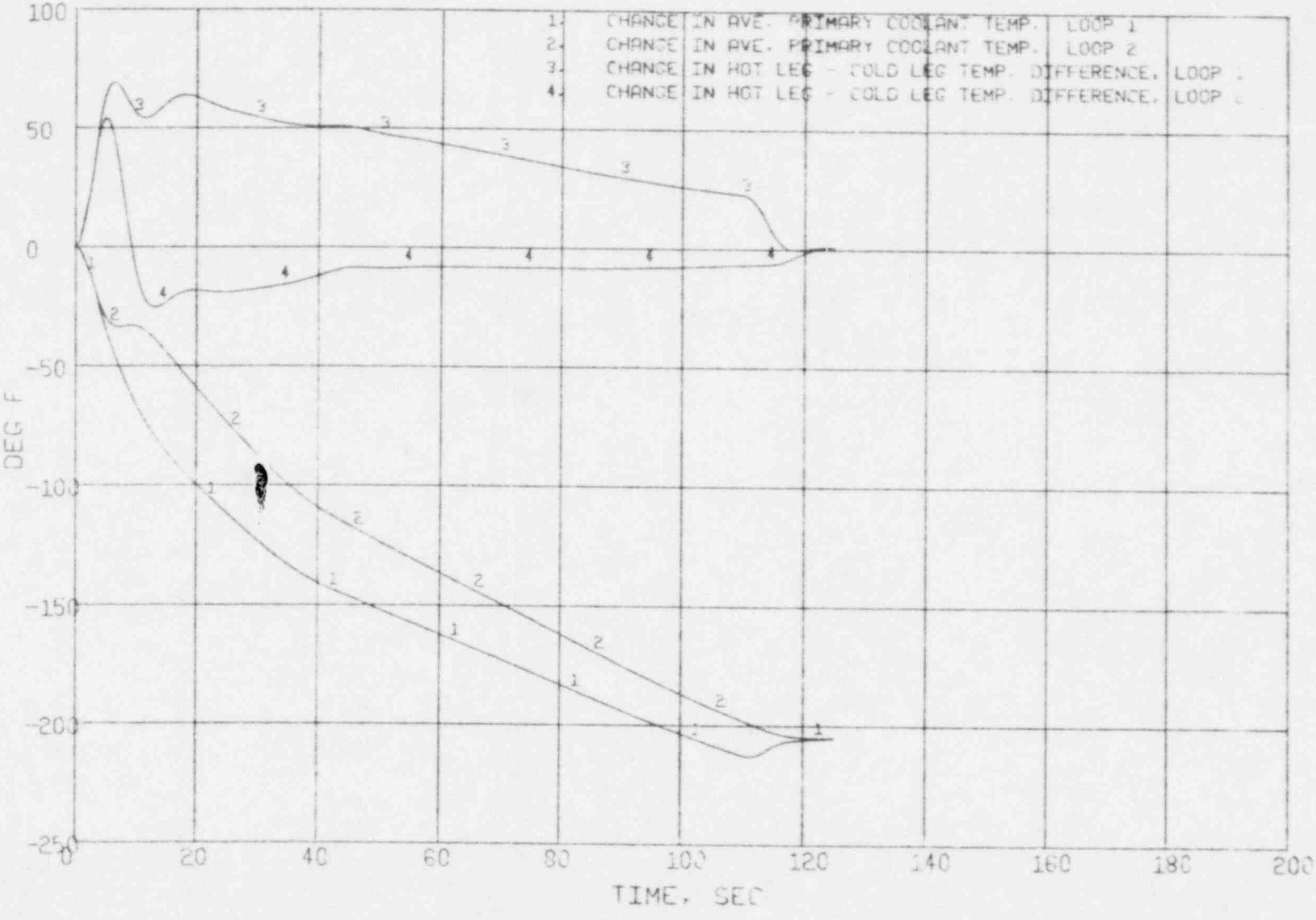



Figure 3.40 Large Steam Line Break ($514^{o}$) - Power, Heat Flux and System Flows




SEQ. REGC7KR 22/06/82 21.00.58.

Figure 3.41 Large Steam Line Break ($514^{o}$) - Core Temperature Response




SEQ REGC7KR 22/06/82 21.30.58.

Figure 3.42 Large Steam Line Break ($514^{0}$) - Primary Loop Temperature Changes

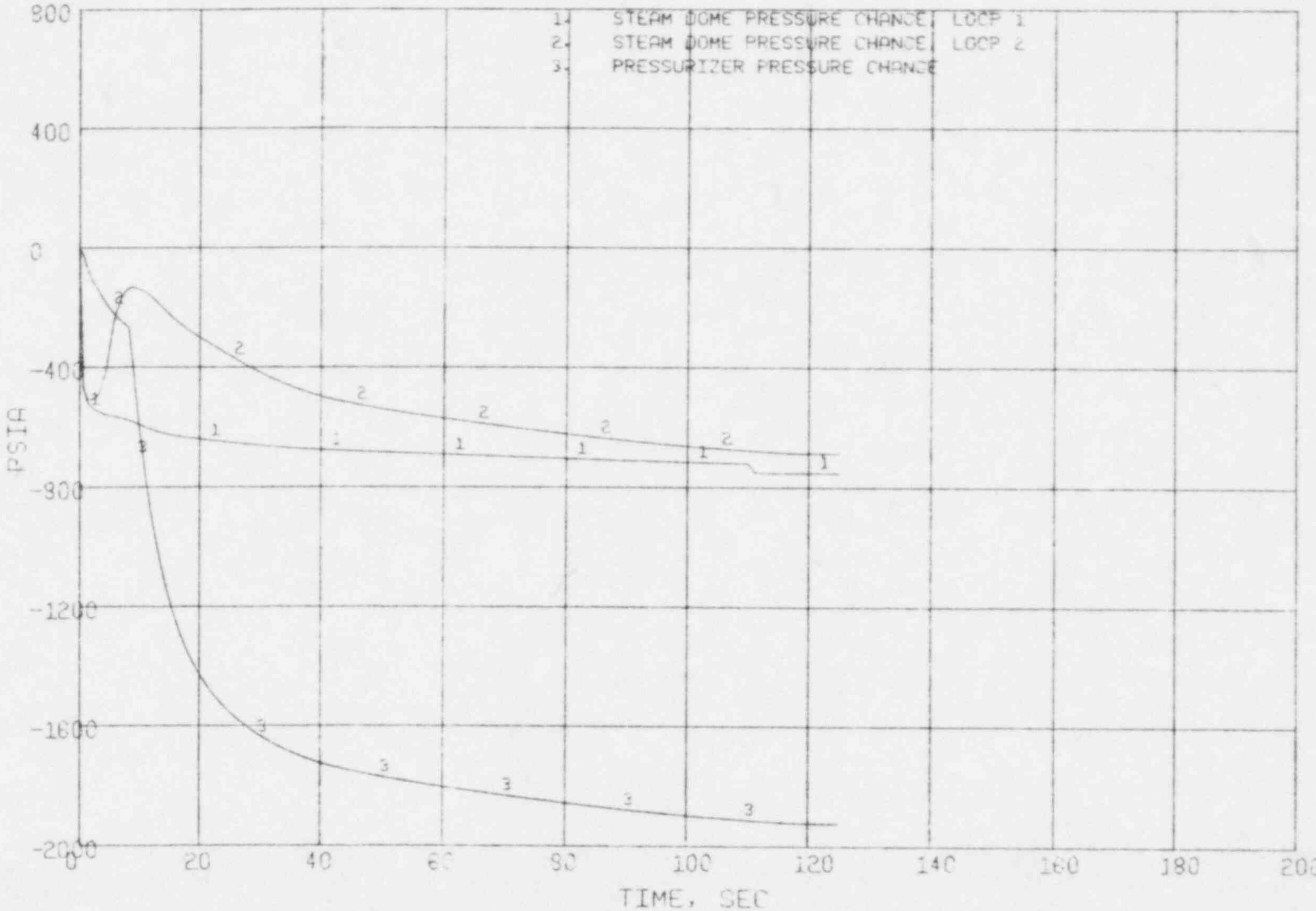



Figure 3.43 Large Steam Line Break (514$^{0}$) - Pressure Changes

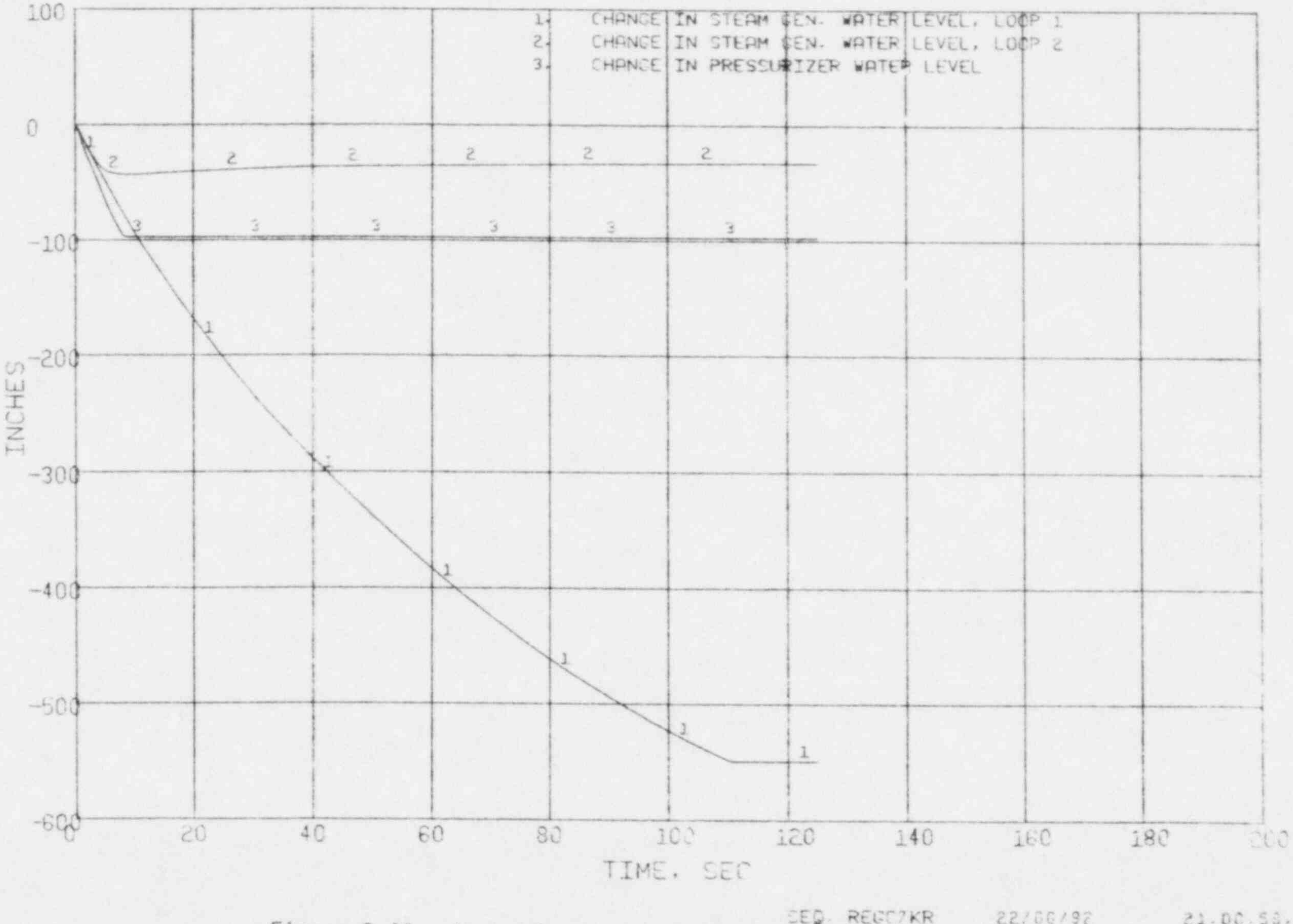



Figure 3.44 Large Steam Line Break ($514^{0}$) - Level Changes

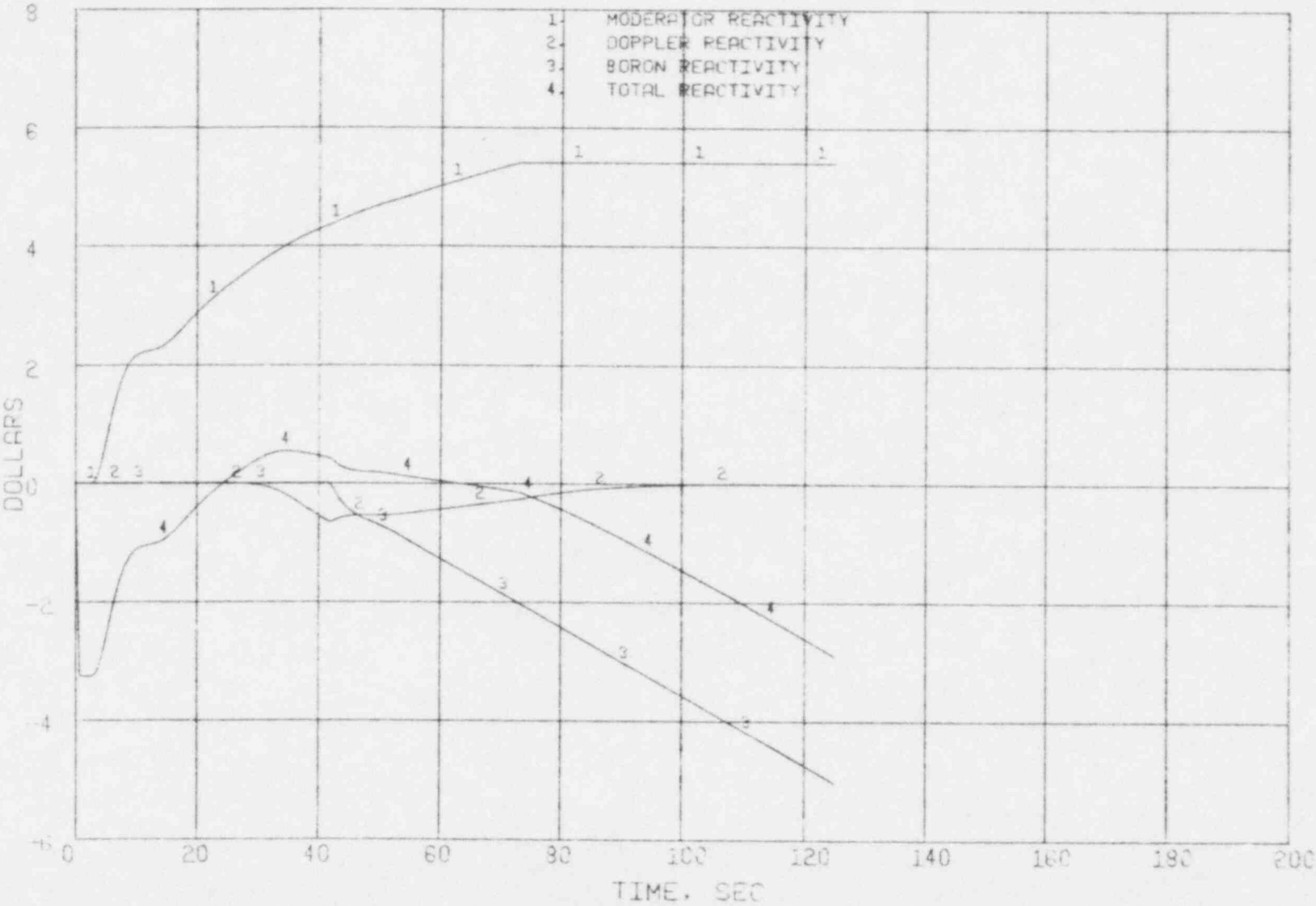



Figure 3.45 Large Steam Line Break (514$^{o}$) - Reactivity

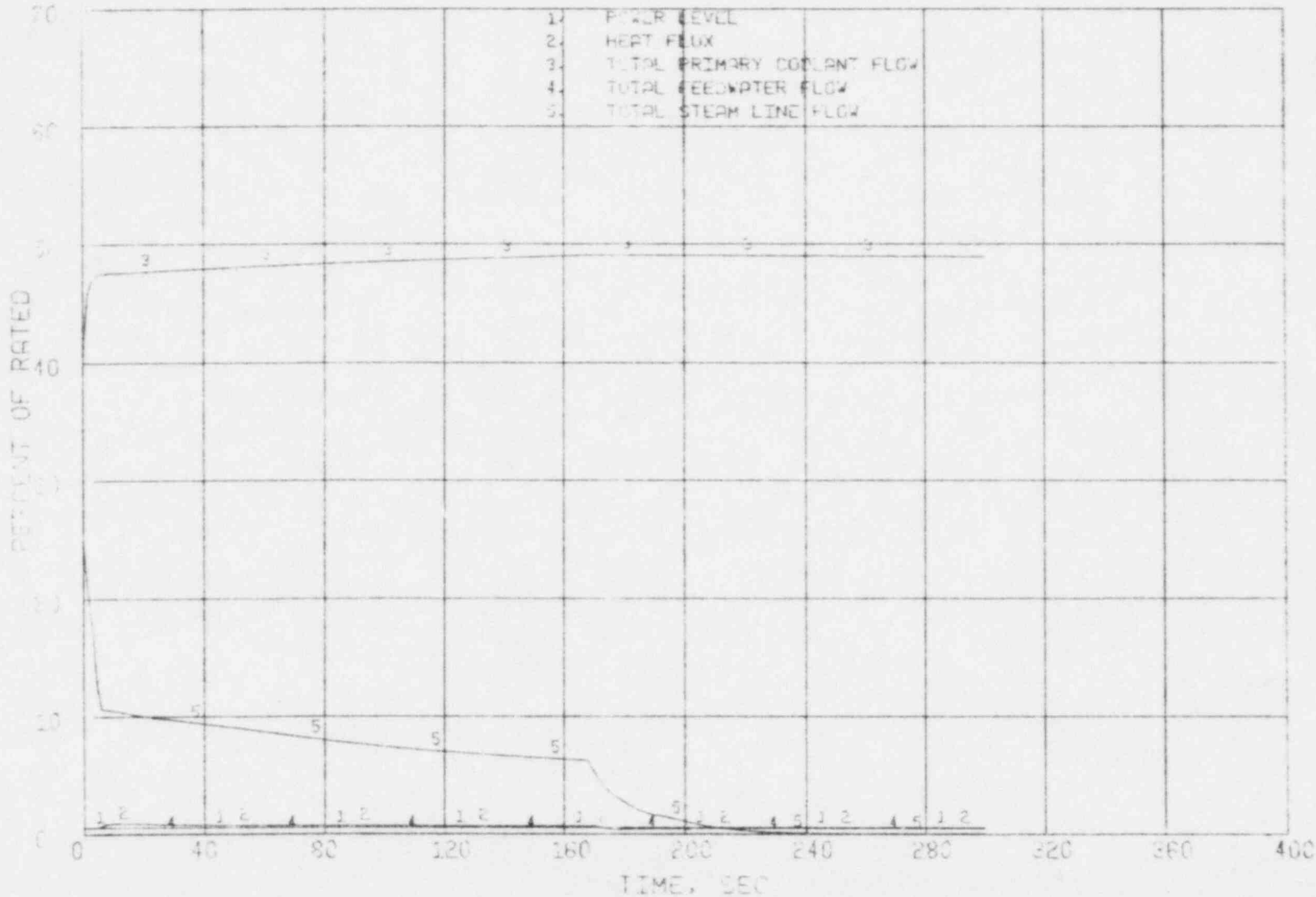



Figure 3.46 Small Steam Line Break (547$^{o}$) - Power, Heat Flux and System Flows

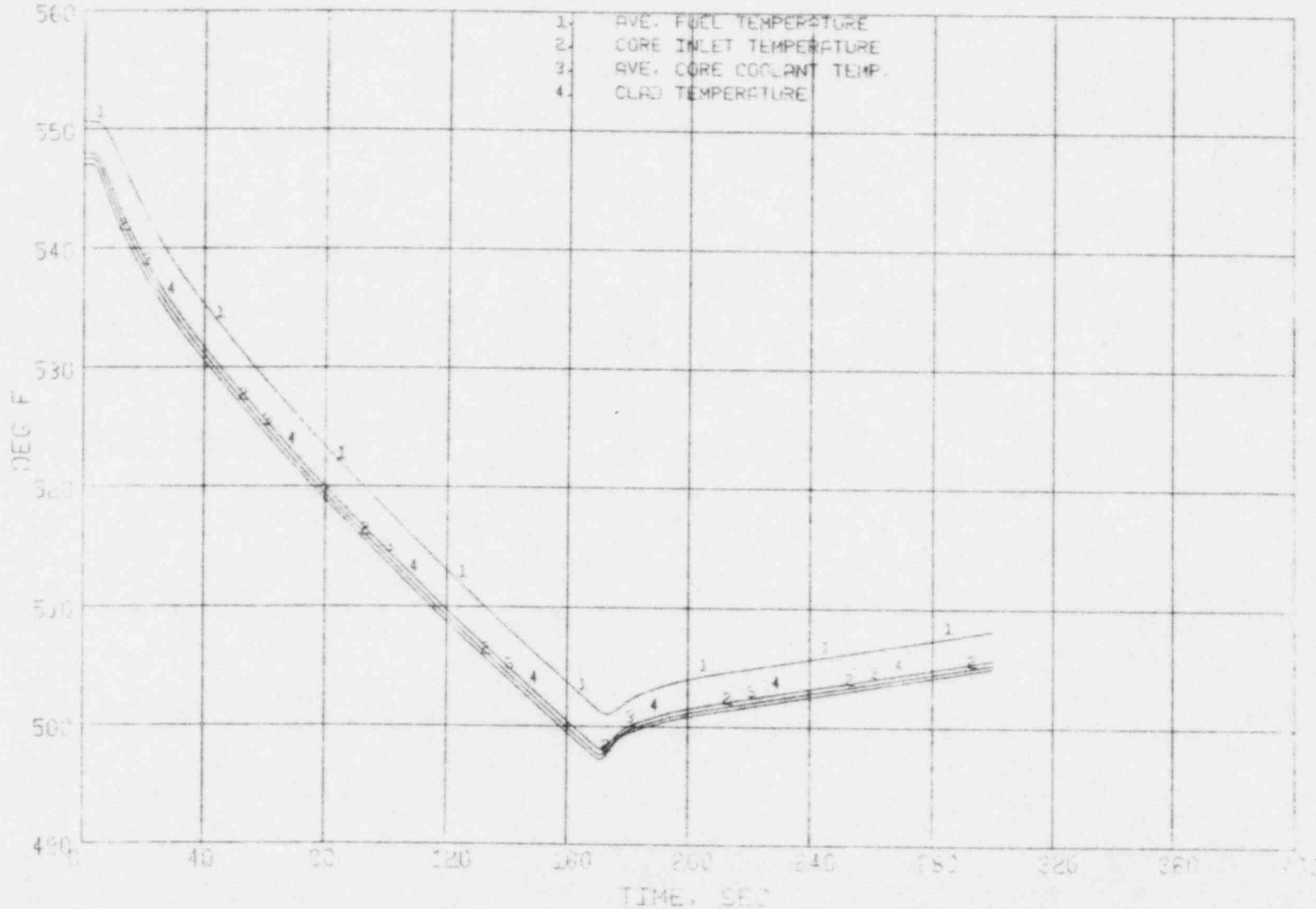



Figure 3.47 Small Steam Line Break (547°) - Core Temperature Response

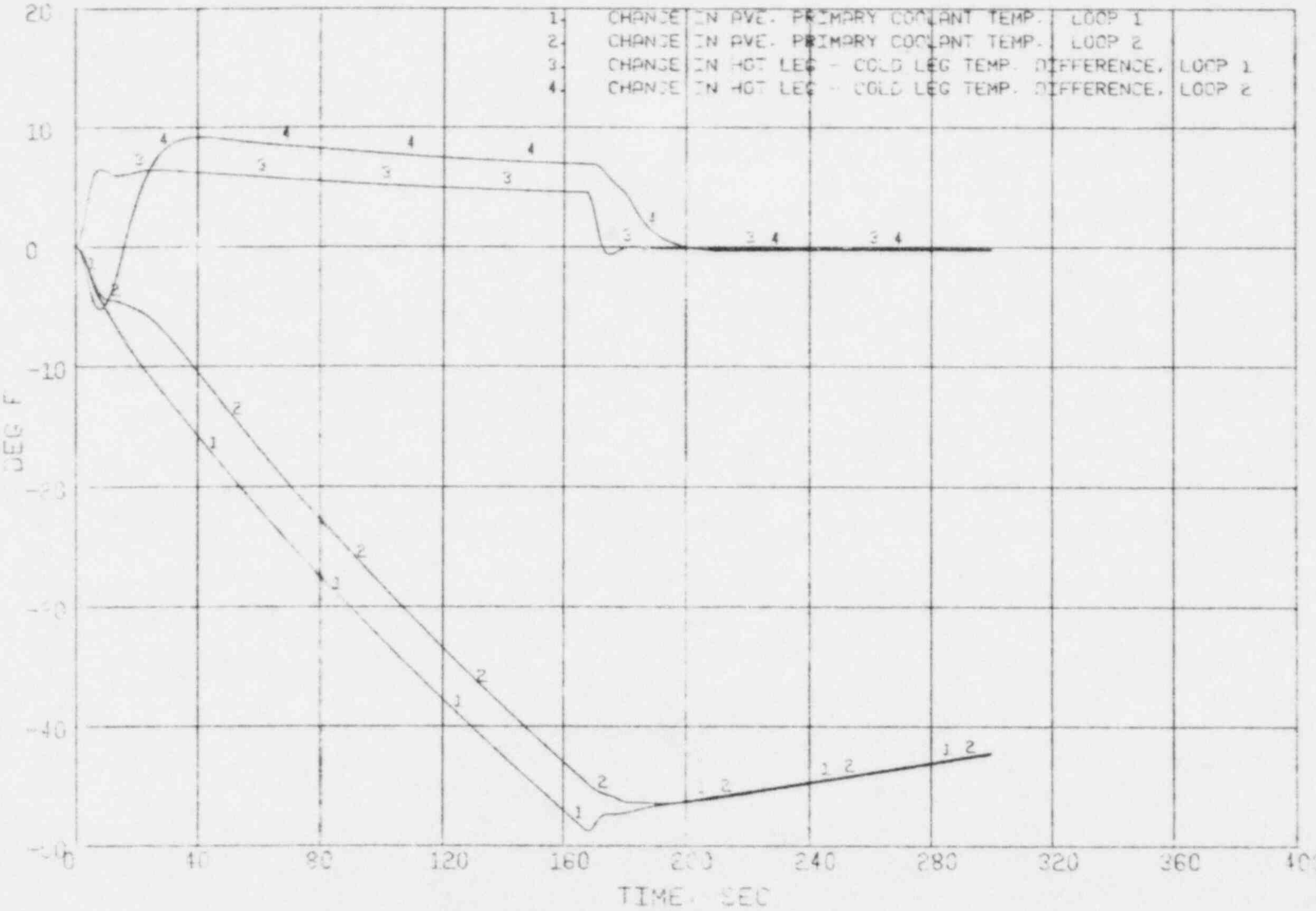



Figure 3.48 Small Steam Line Break (547$^{o}$) - Primary Loop Temperature Changes

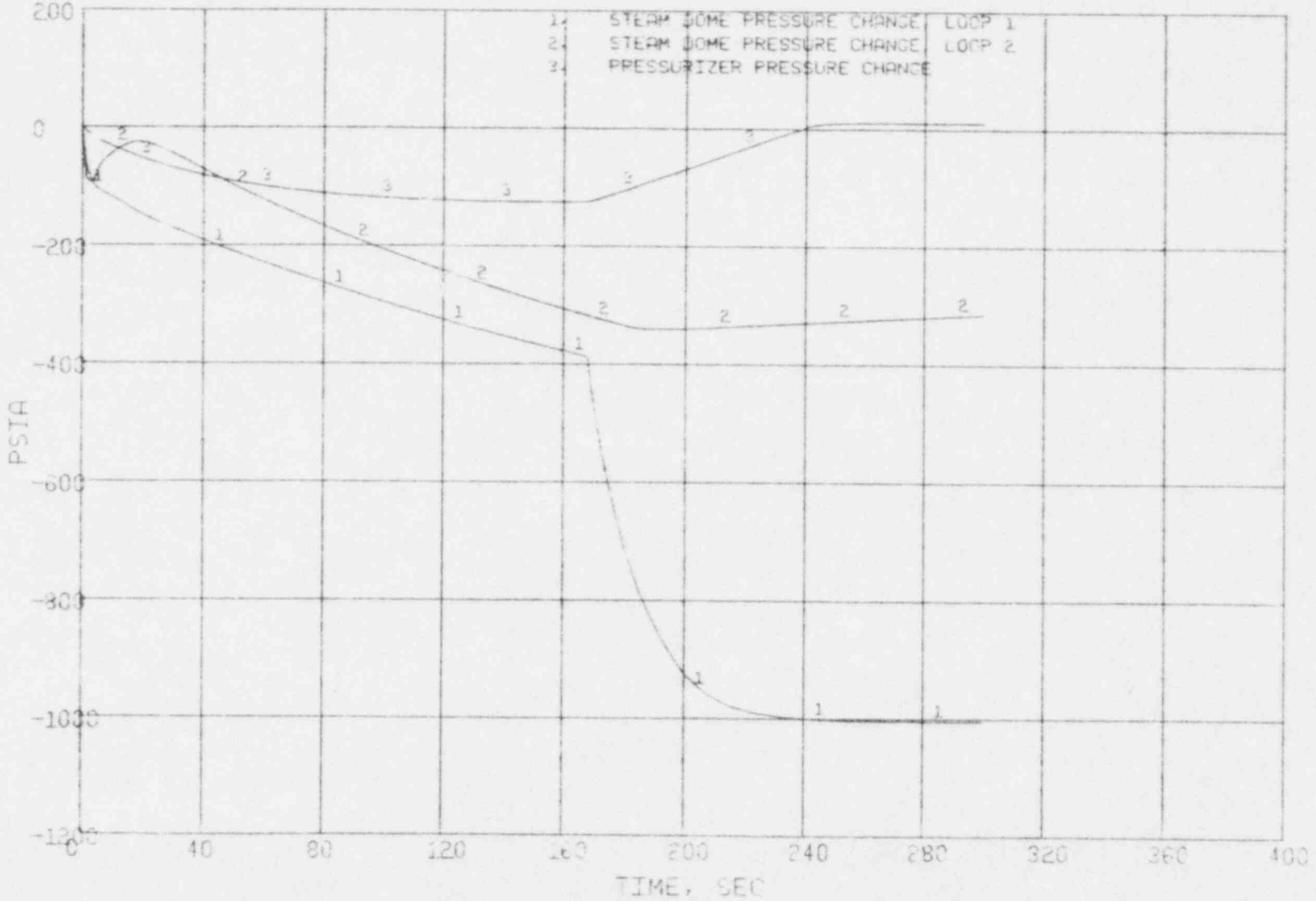



Figure 3.49 Small Steam Line Break (547$^{0}$) - Pressure Changes

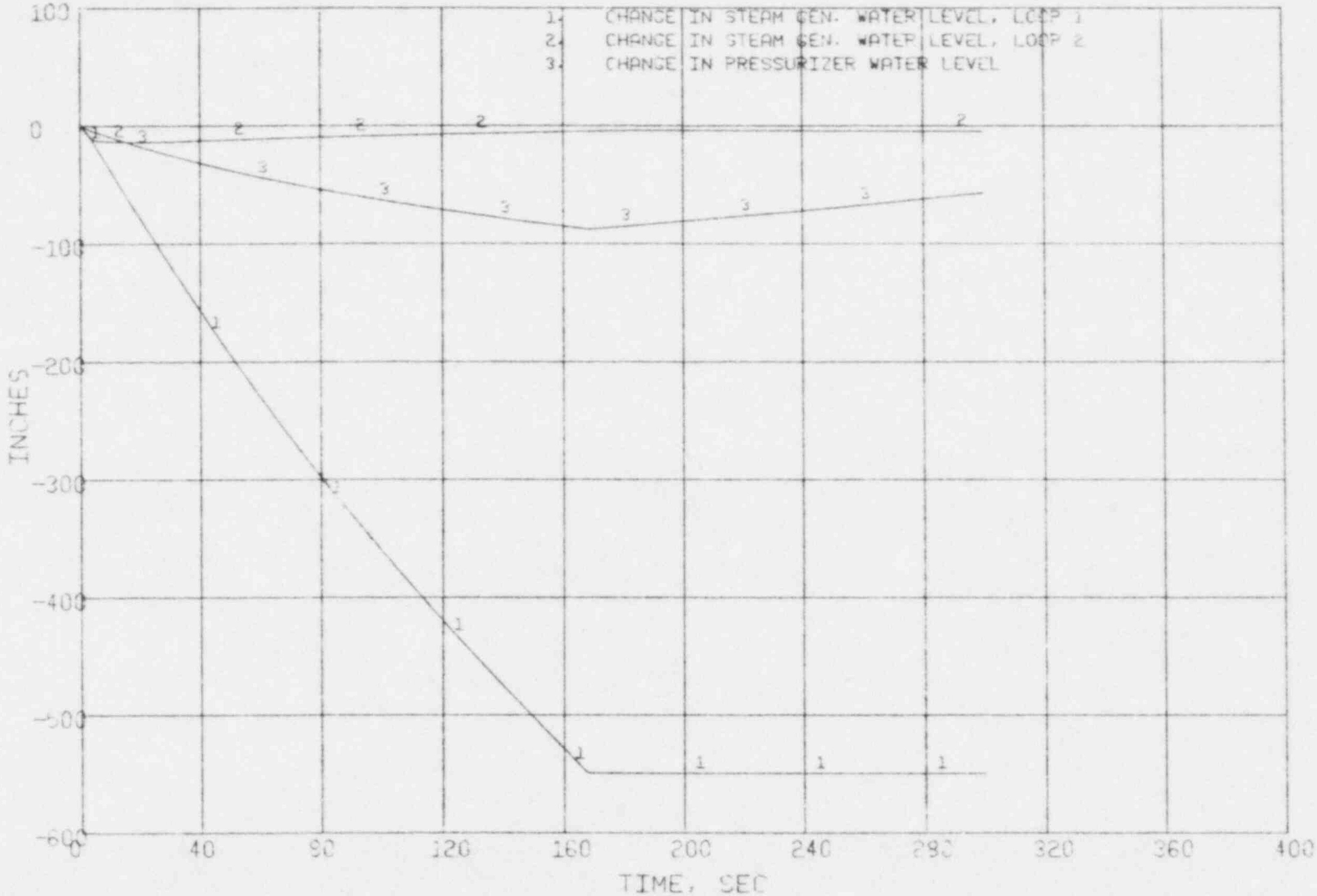



Figure 3.50 Small Steam Line Break (547°) - Level Changes

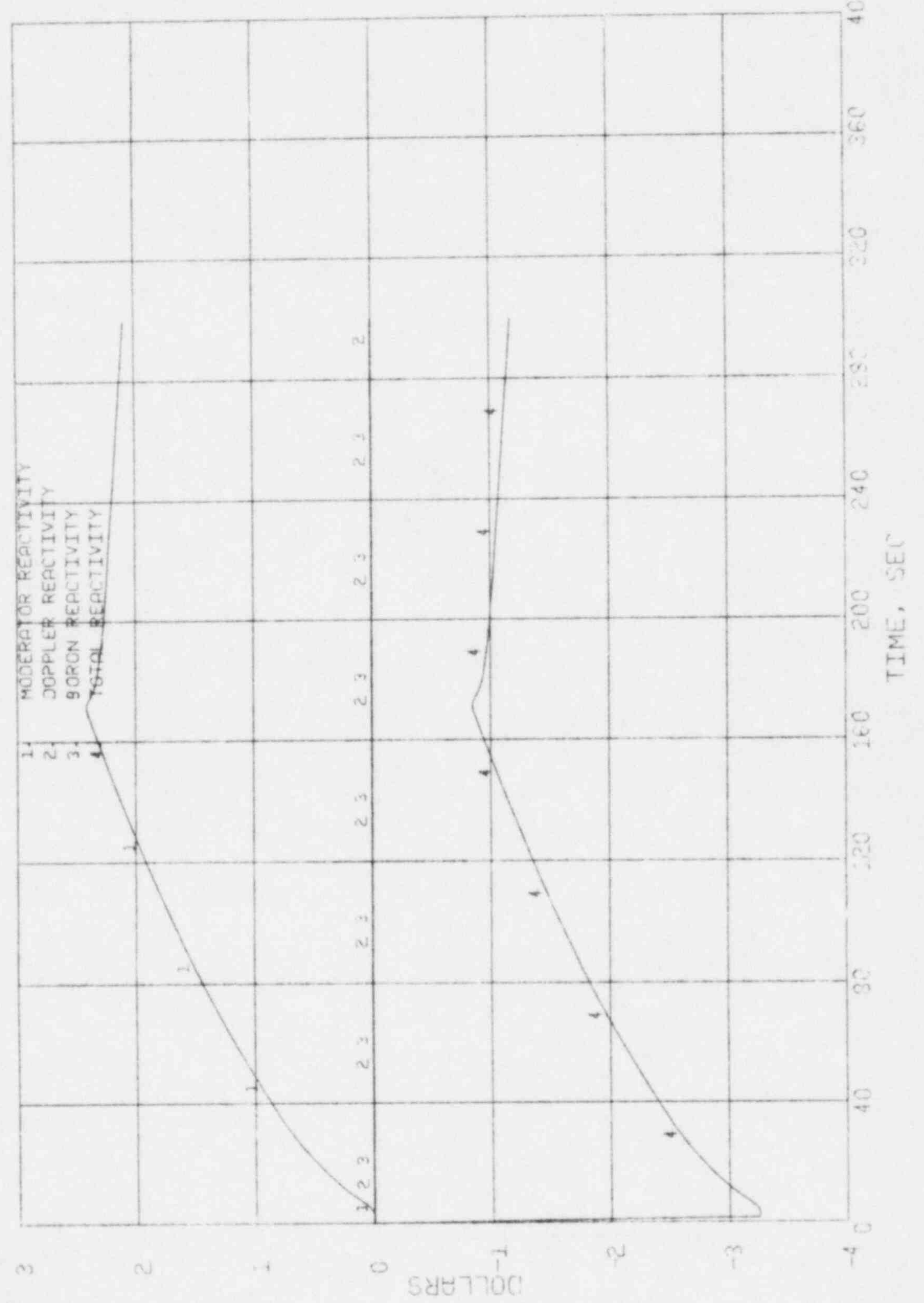



Figure 3.51 Small Steam Line Break (547°) - Reactivity

## 4.0 CONCLUSIONS

The transient analyses for the R.E. Ginna Unit 1 Nuclear Power Plant for conditions of reduced primary coolant temperature, Schedule A, and pressure show adequate margin to safety limits. The neutronics data used in these analyses are consistent with, or conservative with respect to the previous analysis[3]. For reduced primary coolant temperature and pressure the limiting transient analyses reported in Section 3 showed generally increased margins when compared to the previous analyses[1,2].

Several additional transients commonly analyzed were not treated in Section 3. These included:

- startup of an inactive loop
- loss of feedwater
- RCCA drop
- loss of A.C. power
- chemical and volume control system malfunction
- reduction in feedwater enthalpy accident.

They were not limiting transients in prior analyses (see References 1 and 2) and should remain non-limiting for reduced temperature and pressure since their rate of reactivity insertion is enveloped by the limiting transients discussed in Section 3. Further, the steady state MDNBR increased with reduced temperature and pressure, contributing additional margin.

Two average temperature schedules, which bound the proposed operating range for R.E. Ginna, were treated in the analysis. The lower schedule, B, was shown to be bounded by the higher schedule, A, because of improved initial DNBR for transients from power and because of the shape of the moderator reactivity curve for steam line breaks.

The transient analysis supports operation of the R.E. Ginna Unit 1 nuclear power plant at reduced temperature and pressure using the trip setpoint parameters listed in Table 4.1.

Operation of R.E. Ginna at reduced temperature and pressure is more favorable from a plant transient point of view. A reduction of 250 psi in pressure is more than compensated for in the DNBR by a decrease in the average temperature of 15°F. Further reductions in average temperature will improve thermal margins in all transients from power and decrease the reactivity insertion in steam line breaks.

Table 4.1 PPS Trip Settings for Reduced T, P

| Function | Value Supported by Analysis |
| --- | --- |
| • Low Pressure Trip (psia) | 1730 |
| • Reactor $\Delta T$ Trips | |
| $\Delta T_0$ ($^{o}F$) | Full power value |
| $T^{*}_{setpoint}$ ($^{o}F$) | 523.5-558.5 |
| $P_{setpoint}$ (psia) | 2000 |
| $K_1$ | 1.12 |
| $K_2$ | $6.56 \times 10^{-4}$ |
| $K_3$ | 0.0136 |
| $K_4$ | 1.09 |
| $K_5$ | 0.001 |
| $K_6$ | 0.0262 |

------------------

* equal to plant average temperature

## 5.0 REFERENCES

1) "Plant Transient Analysis for the R.E. Ginna Unit 1 Nuclear Power Plant", XN-NF-77-40, Rev. 0, Exxon Nuclear Company, Inc., Richland, Washington 99352, November 1979.

2) "Plant Transient Analysis for the R.E. Ginna Unit 1 Nuclear Power Plant", XN-NF-77-40, Supplement 1, Exxon Nuclear Company, Inc., Richland, Washington 99352, March 1980.

3) "Reduced Primary System Temperature and Pressure Analyses for R.E. Ginna Nuclear Power Unit", XN-NF-80-44, Exxon Nuclear Company, Inc., Richland, Washington 99352, September 1980.

4) "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)", XN-74-5, Revision 1, Exxon Nuclear Company, Inc., Richland, Washington 99352

5) "Definition and Justification of Exxon Nuclear Company DNB Correlation for Pressurized Water Reactors", XN-75-48, Exxon Nuclear Company, Inc., Richland, Washington 99352, October 1975.

6) MacBeth, R.V., "Burnout Analysis Part 5. Examination of Published World Data for Rod Bundles", AEEW-R-358, June 1964.

7) MacBeth, R.V., "An Appraisal of Forced Convection Burn-out Data", Proc. Inst. Mech. Engs., Vol. 18, Part 3E, 1965-1966.

8) Moody, F.J., ASME Transactions, p. 134, February 1965.

9) "Rochester Gas and Electric Corporation Proposed Change to Shutdown Margin Requirements", Ginna Nuclear Plant Technical Specification, Amendment Application dated September 22, 1975.

PLANT TRANSIENT ANALYSIS FOR OPERATION OF THE RE GINNA UNIT 1 NUCLEAR POWER PLANT AT REDUCED PRESSURE AND TEMPERATURE

XN-NF-82-45
REVISION 1
ISSUE DATE: 07/22/82

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