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PLANT TRANSIENT ANALYSIS FOR ST. LUCIE UNIT 1

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PLANT TRANSIENT ANALYSIS FOR ST. LUCIE UNIT 1

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1.0 INTRODUCTION AND SUMMARY

The plant transient analysis reported here was performed to support operation of the St. Lucie Unit 1 nuclear power plant with a mixed core of 84 Exxon Nuclear Company (ENC) assemblies (XN-1) and 133 Combustion Engineering (CE) assemblies.

Because of the higher grid spacer loss associated with the ENC fuel compared to the resident CE fuel, ENC suffers a flow diversion penalty. Since Cycle 6 is the first cycle in which ENC fuel will be loaded, Cycle 6 will have a lower ratio of ENC to CE fuel assemblies than future cycles, and therefore a larger penalty. Succeeding cycles will experience decreasing penalties until all the fuel in the core is ENC fuel, at which point there will be no flow penalty and the thermal margin for ENC fuel will be significantly improved. For this reason, it is anticipated that the Cycle 6 core will be more thermal-hydraulically limiting than future cycles, and thus the present analysis will be bounding of for Cycle 7 and subsequent cycles at St. Lucie Unit 1. Cycle-specific safety analyses will be performed to confirm this expectation. The criteria used in the analysis are based on protecting the specified acceptable fuel design limits (SAFDLs), listed in Table 1.1, for anticipated operational occurrences (AOOs); and on demonstrating an acceptably low level of fuel damage in the event of a postulated accident (PA).

The purpose of this analysis is to examine core thermal margins for Cycle 6 in St. Lucie Unit 1. Based on thermal-hydraulic considerations alone, ENC fuel assemblies in Cycle 6 will be somewhat more thermal-margin limiting than either co-resident CE fuel in Cycle 6 or CE fuel in the all-CE core of Cycle 5. Thus verification of adequate thermal margin in Cycle 6 has been a necessity. Because of the importance of the pressure criterion, peak pressurization transients are addressed, although ENC reload fuel is not expected to significantly impact system pressure response.

The key results of the analysis are summarized in Table 1.2 and confirm that the criteria are met. The lowest value of MDNBR for any AOO is 1.326 for the loss-of-flow event. This value is well above the 95/95 limit for ENC's XNB critical heat flux correlation[1].

Thermal margin results for the locked rotor accident are such that less than 1.0% of the fuel in the core would be expected to experience DNB. A significant return to power did not occur in the analysis for the large steam line break accident using a 3.6% $\Delta\rho$ shutdown margin.

The transient events reanalyzed for this cycle are the most limiting events and comprise an adequate set of simulations to assure safe operation of St. Lucie Unit 1. The events considered for the operation of St. Lucie Unit 1 at 2700 MWt are discussed in Section 4.0 and the events analyzed in this report are shown to bound the set discussed in the stretch power submittal[2]. The reactor protection system (RPS) setpoints for Cycle 6 were not changed so the setpoints considered by ENC are the same setpoints taken into account by CE. To ensure conservatism, MDNBRs were calculated using the most limiting axial shapes from the set of 1374 different axial shapes analyzed by ENC.

The analysis of the limiting transients is described in Section 3. The present simulation shows substantially the same plant response to the transients with major differences related to the limiting assembly thermal-hydraulic performance. A description of the transient calculational methods and the input parameters is provided in Section 2.

Analysis of the limiting transients has shown that there exists a safe margin to the SAFDLs during AOOs and that fuel damage is less than 1% for the PAs. The thermal margin for the Cycle 6 reload is sufficient.

Table 1.1 Fuel and Vessel Design Limits

Event Class	Criteria
Anticipated Operational Occurrences (AOOs)	• Specified acceptable fuel design limits (SAFDLs) • MDNBR, based on XNB, >1.17 • Local power density 21 kW/ft • Pressure $<$ 2750 psia
Postulated Accident (PA)	• Fuel damage is limited to a small fraction of the fuel in the core • Pressure $<$ 2750 psia

Table 1.2 Summary of Results

Transient	Maximum Power Level (MWt)	Maximum Core Average Heat Flux (BTU/hr-ft^2)	Maximum System Pressure (psia)	MDNBR (XNB)
Loss of Load	3446.5	210544	2657	2.035*
Excess Load	3482.7	200512	2250	1.385
RCS Depressurization	40.5	5829	2250	1.389
Loss of Coolant Flow	2735	189766	2401	1.326
CEA Withdrawal	3131.8	199908	2363	1.590
CEA drop	2708.5	190301	2250	1.465
Seized Rotor (DNB) †	2757.3	189766	2338	1.189
(Pressure)	2822.2	199753	2397	1.450
Steam Line Rupture †	37.2	9682	2250	>4.5
Steady State Operation	2700	189759	2250	1.72

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* Transient conditions which produce the highest average RCS temperatures for loss-of-load events were used.

† Postulated Accidents

2.0 CALCULATIONAL METHODS AND INPUT PARAMETERS

2.1 CODE DESCRIPTION

The transient analysis for St. Lucie Unit 1 was performed using PTSPWR2[3] the Exxon Nuclear Company plant transient simulation model for pressurized water reactors. The simulation code models the behavior of pressurized water reactors under both normal and abnormal conditions by solving the transient conservation equations for the primary and secondary systems numerically. Core neutronics behavior is modeled using point kinetics, and the transient conduction equation is solved for fuel temperatures and heat fluxes. State variables such as flow, pressure, temperature, mass inventory, steam quality, heat flux, reactor power and reactivity are calculated during the transient. Where appropriate the reactor protection system (RPS) and control system are modeled to describe the transients. The departure from nucleate boiling ratio (DNBR) is calculated for the hot channel during the transients using a hot channel model and the XNB[1] correlation.

The system model used by PTSPWR2, shown in Figure 2.1, models the reactor, both primary coolant loops, both steam generators and both steam lines. All major components (pressurizer, coolant pumps, and all major valves) are also modeled.

The present calculations were performed using the NOV76A version of the PTSPWR2 code, along with appropriate updates. These updates include:

(1) An improved pressurizer model, described in Reference 4.

(2) A correction to the mass balance on the secondary side of the steam generator.

(3) A modified set of trip functions to describe a Combustion Engineering plant.

(4) Axial shape-dependent scram curves.

(5) A dynamic flow coastdown model; and

(6) Appropriate changes to the primary loop, hydraulic behavior to describe the 2 hot leg - 4 cold leg configuration of St. Lucie Unit 1.

Updates 1 and 2 have been included in recent ENC transient analyses. Updates 3-6 were prepared specifically for this analysis.

The trip functions used in this analysis consist of calculated trips set in conjunction with the limiting conditions of operation (LCOs) which protect the specified acceptable fuel design limits (SAFDLs) based on local power density (LPD) and departure from nucleate boiling (DNB) and trips based on single state variables. These latter trip setpoints are listed in Table 2.1 aong with the trip time delay appropriate for all of the RPS trips. The reactor trip setpoints for Cycle 6 at St. Lucie Unit 1 are unchanged from Cycle 5.

The calculated trips for St. Lucie Unit 1 consist of an LPD trip and a Thermal Margin/Low Pressure (TM/LP) trip. The LPD trip protects against a power excursion exceeding the local power density limit of 21 kW/ft. The trip trip is based on core power, Q, defined as the larger of the neutron flux power and the thermal power and on the peripheral axial shape index (ASI); which is defined as,

$$ASI = \frac{P_{LOW} - P_{UP}}{P_{LOW} + P_{UP}} , \qquad (2.1)$$

where P_{LOW} and P_{UP} are the output from the bottom and top ex-core flux sensors, respectively. Figure 2.2 shows the trip function.

The TM/LP trip is based on the same auctioneered core power as the LPD trip. In addition, it also depends upon the ASI, and the inlet temperature

T_{IN}. The form of the trip function is,

$$P_{VAR} = 2061\ A1\ (ASI)\ QR1\ (Q) + 15.85\ T_{IN} - 8950, \qquad (2.2)$$

where A1 and QR1 are shown in Figures 2.3 and 2.4, respectively. Pressurizer pressure is the system variable which is compared to the trip setpoint, P_{VAR}. The TM/LP trip protects the core from the onset of DNB with at least a 95% probability as long as the plant is operated within the appropriate limiting conditions of operation (LCO) shown in Figure 2.5.

A set of ASI-dependent scram curves, shown in Figure 2.6, provides a conservative scram curve for each ASI.

Two basic kinds of axial power distributions were considered in the analysis. For transients and accidents where thermal margin (DNB) is the limiting factor, top peaked axial power distributions were limiting. For peak pressurization transients, bottom peaked power distributions with delayed scram were limiting. For each of these cases, the scram curve was interpolated from the curves given in Figure 2.6.

The pump response to a loss of power was modeled by setting the shaft rotation speed derivative equal to the pumping torque, divided by the effective inertia. The flow in each of the four cold legs was calculated based on the pump head and the required pressure drop. The effective inertia was then adjusted to provide a good fit to plant data(4).

2.2 MODELING UNCERTAINTIES

The present plant transient analysis is basically a deterministic analysis. Thus, steady state measurement and instrumentation errors were taken into account in an additive fashion to ensure conservative calculations of MDNBR. The plant uncertainties related to initial conditions in the MDNBR calculations are:

Power	+ 2% for calorimetric error
Inlet coolant temperature	+ $2^{o}F$ for deadband and measurement error
RCS pressure	- 22 psi for steady-state measurement errors.

Combined with design flow, these parameter uncertainties minimize the initial minimum DNBR. These uncertainties are not included in the plant system modeling explicitly, rather they are used to establish a conservative bound to the initial minimum DNBR. Table 2.2 is a list of operating parameters used in this analysis.

The trip setpoints are based on Technical Specification Limits(5) and are unchanged from Cycle 5. Statistical verification of the calculated trips (LPD and TM/LP) is presented in the Safety Analysis Report (SAR)(8). These trip setpoints are modeled conservatively in the transient analysis to provide bounding simulations of the plant response.

The pressurizer control system was modeled in such a fashion that it could not ameliorate the effects of transients. The spray system was operable during DNBR transients while the heaters were off, thus tending to minimize DNBR. For pressurization transients, e.g. loss-of-electric load, the spray system and pressurizer relief valves were removed from the simulation.

Additional conservatisms in the pressurization transient include the conservative modeling of the high pressure trip (2422 psia), higher initial power (102%), a conservative choice of kinetics parameters, and a bottom-peaked core to delay termination of the transient as long as possible.

2.3 DESIGN PARAMETERS

The ENC fuel design parameters for St. Lucie Unit 1 are summarized in Table 2.3. Table 2.4 lists the neutronics parameters, both nominal and bounding values for beginning of cycle (BOC) and end of cycle (EOC) conditions. The values used in the analysis for the moderator temperature coefficient and shutdown margin are consistent with the new Cycle 6 Technical Specification limits for these parameters. Three axial power distributions, which were found to give minimum steady-state DNBRs, were used along with the radial peaking factor appropriate for each. The radial peaking factors used correspond in each case to the Technical Specification Limit of 1.7 allowing for a 7% uncertainty. The three DNB limiting axial profiles are shown in Figures 2.7 through 2.9. The axial profiles shown in Figures 2.7 through 2.9 are specifically for the hot rod. The quoted ASI is the peripheral ASI and contains the effects of rod shadowing and shape annealing. The ASI one would calculate based on the axial shape would, therefore, not agree with the quoted ASI. However, the local hot rod power is preserved over the entire length of the core.

Table 2.1 St. Lucie Unit 1 Trip Functions

Function	Allowable Values		Values Used in the Analysis	
	Setpoint	Delay* (seconds)	Setpoint	Delay* (seconds)
Variable High Power Trip (% of rated)	107	0.9	112	0.9
Low Flow Trip (% of design)	95	1.15	93	1.15
High Pressurizer Pressure (psia)	2400	1.4	2422	1.4
Low Steam Generator Pressure (psia)	600	1.4	578	1.4
Low Steam Generator Water Level (% of span)	37	1.4	31.5	1.4
LPD (described in text)	-	0.9	-	0.9
TM/LP (described in text)	-	1.4	-	1.4
Steam Generator Pressure Difference (psi)	135	1.4	185	1.4

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* includes a 0.5 second allowance for the holding coils to release

Table 2.2 St. Lucie Unit 1 Operating Parameters used in PTSPWR2 Analysis

CORE	
Total Heat Output (MWt)	2700
Heat generated in fuel (%)	97.5
Coolant Flow Rate (Mlb/hr)	134.8
Unrodded Pin Radial Peaking Factor	1.70
Average Heat Flux (BTU/hr-ft^2)	189759
REACTOR COOLANT SYSTEM	
Coolant Flow Rate (Mlb/hr)	139.4
Pressure (psia)	2250
Average Temperature (^oF)	572
STEAM GENERATORS	
Feedwater Enthalpy (Btu/lb)	410.4
Pressure (psia)	880.1
Steam Flow (Mlb/hr) @ 2700 MWt	11.72

Table 2.3 ENC Fuel Design Parameters for St. Lucie Unit 1, Cycle 6

Fuel Pellet Diameter (in)	0.370
Outer Clad Diameter (in)	0.440
Inner Clad Diameter (in)	0.378
Active Fuel Length (in)	136.7
Number of Fuel Rods in the Core	37,008

Table 2.4 Neutronics Parameters for St. Lucie Unit 1, Cycle 6

Parameter		Bounding Values BOC	Bounding Values EOC	Cycle 6 Nominal Value BOC	Cycle 6 Nominal Value EOC
Moderator Temperature Coefficient					
($\Delta\rho/^{o}F \times 10^4$)	HZP	0.7	-2.8	0.48	-1.44
	HFP	0.2	-2.8	-0.13	-2.07
Doppler Temperature Coefficient					
($\Delta\rho/^{o}F \times 10^5$)	HFP	-1.0	-2.0	-1.2	-2.07
	HZP	--	--	-1.6	-1.9
Pressure Coefficient					
($\Delta\rho$/psi $\times 10^6$)		-1.4	5.0	-1.0	4.3
Boron Worth Coefficient					
($\Delta\rho$/ppm $\times 10^4$)		-0.8	-0.9	-0.88	-1.07
Delayed Neutron Fraction (Beta)		0.0071	0.0045	0.0058	0.0050
Total Rod Worth		-0.05078	-0.05671	-0.0649	-0.0712
Shutdown Margin		--	0.0355	--	0.036

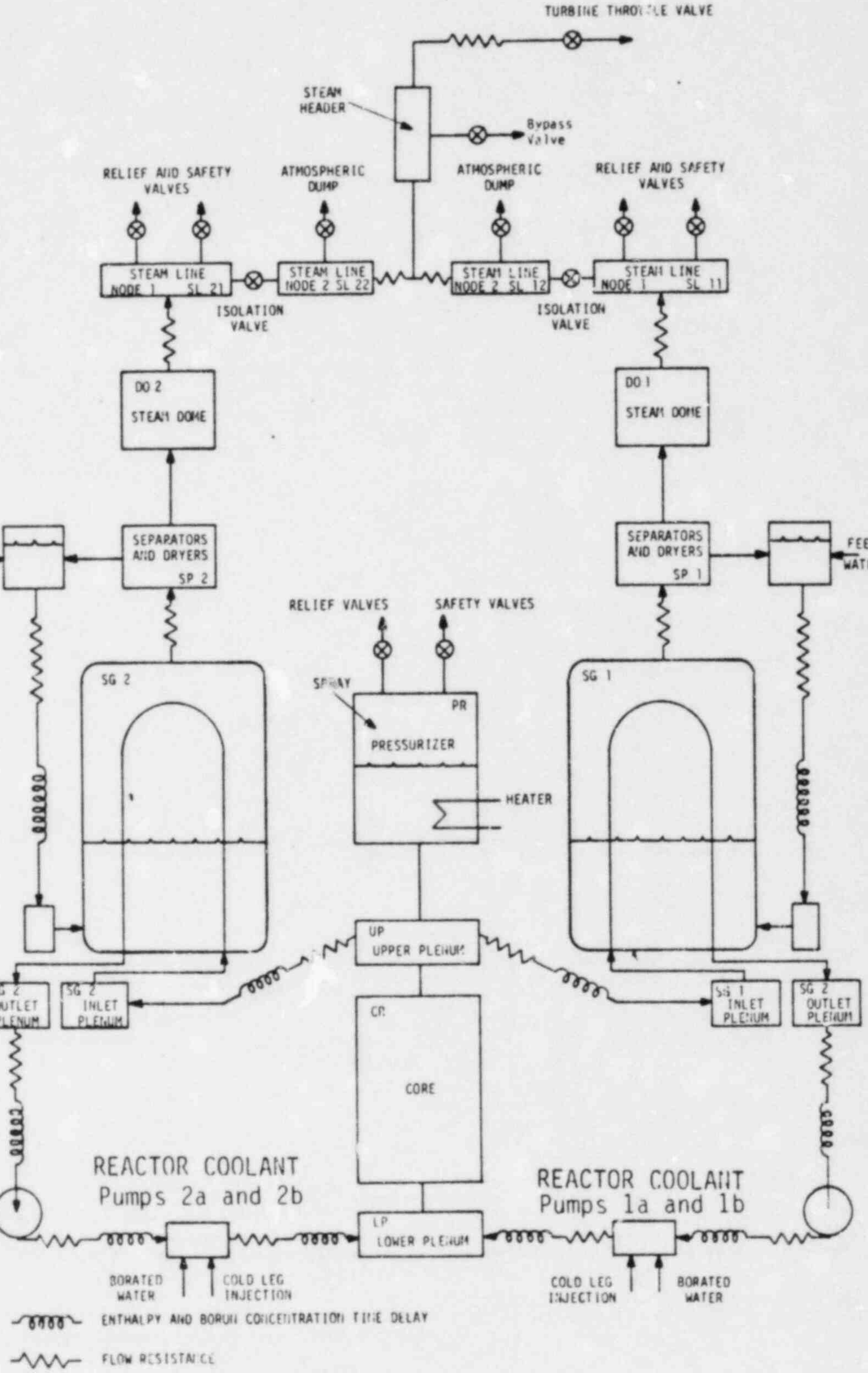

Figure 2.1 PTSPWR2 System Model

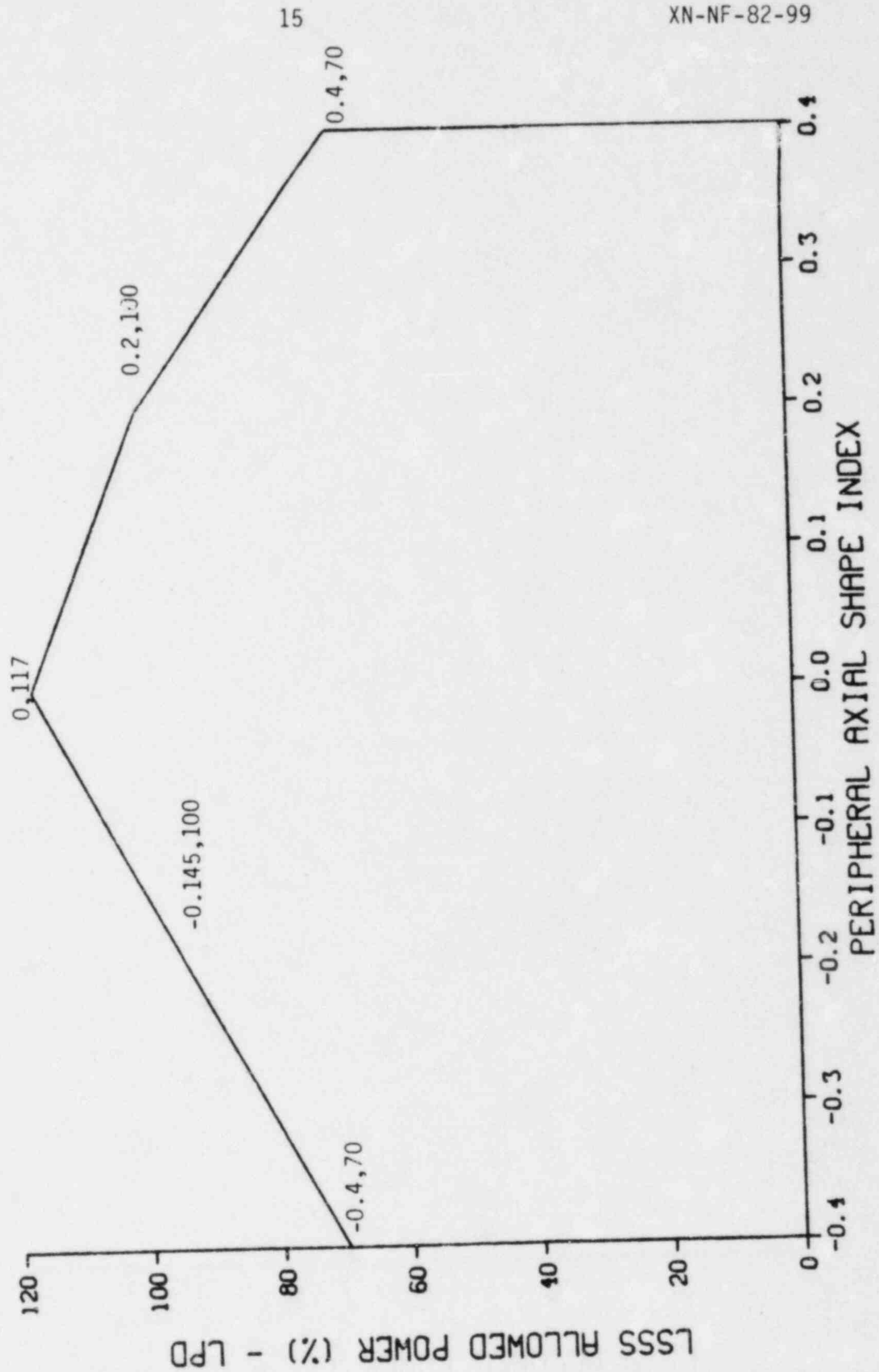

Figure 2.2 St. Lucie Unit 1 - LSSS Based On LPN

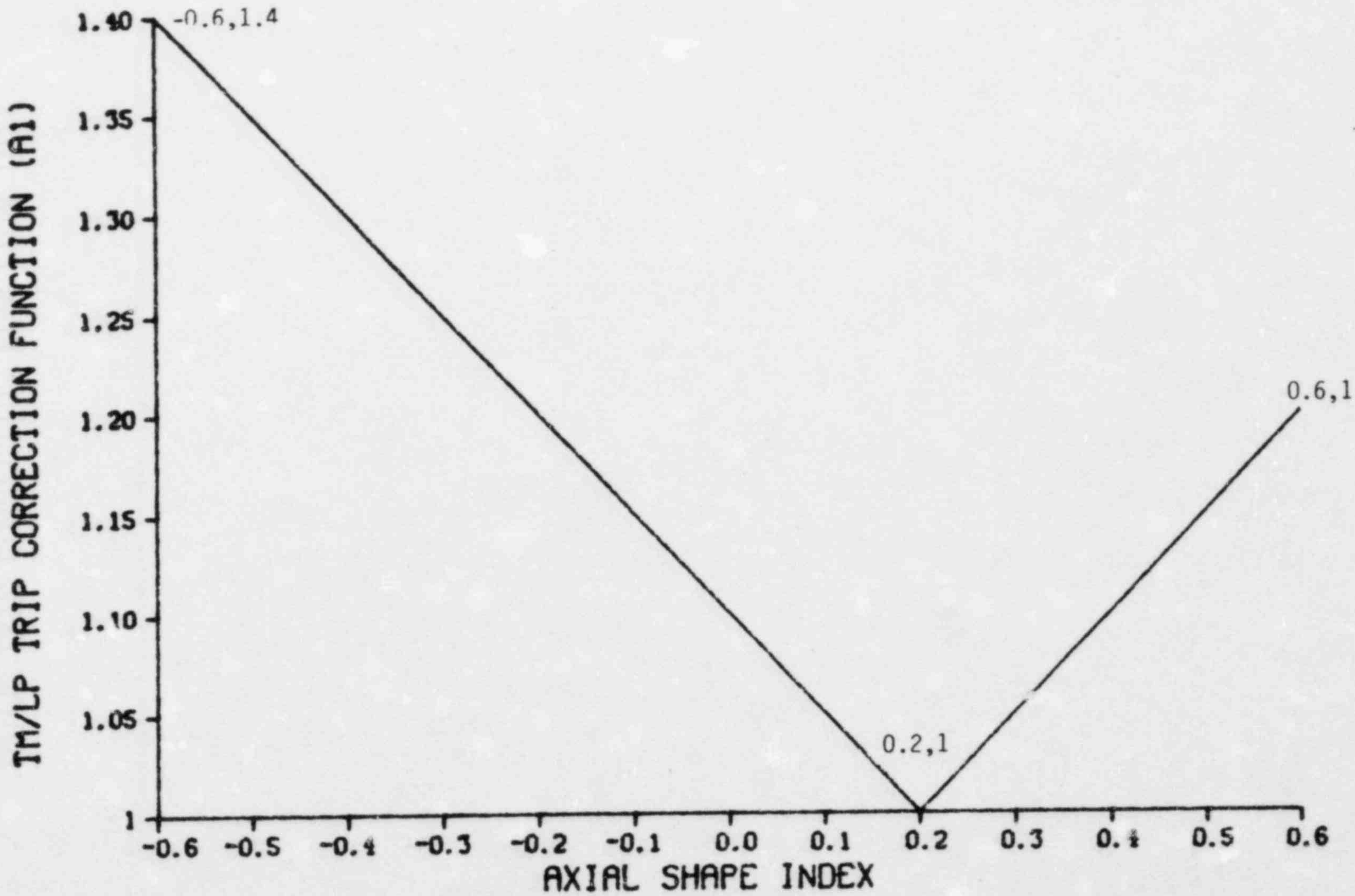

Figure 2.3 St. Lucie Unit 1 - TM/LP Correction Function A1

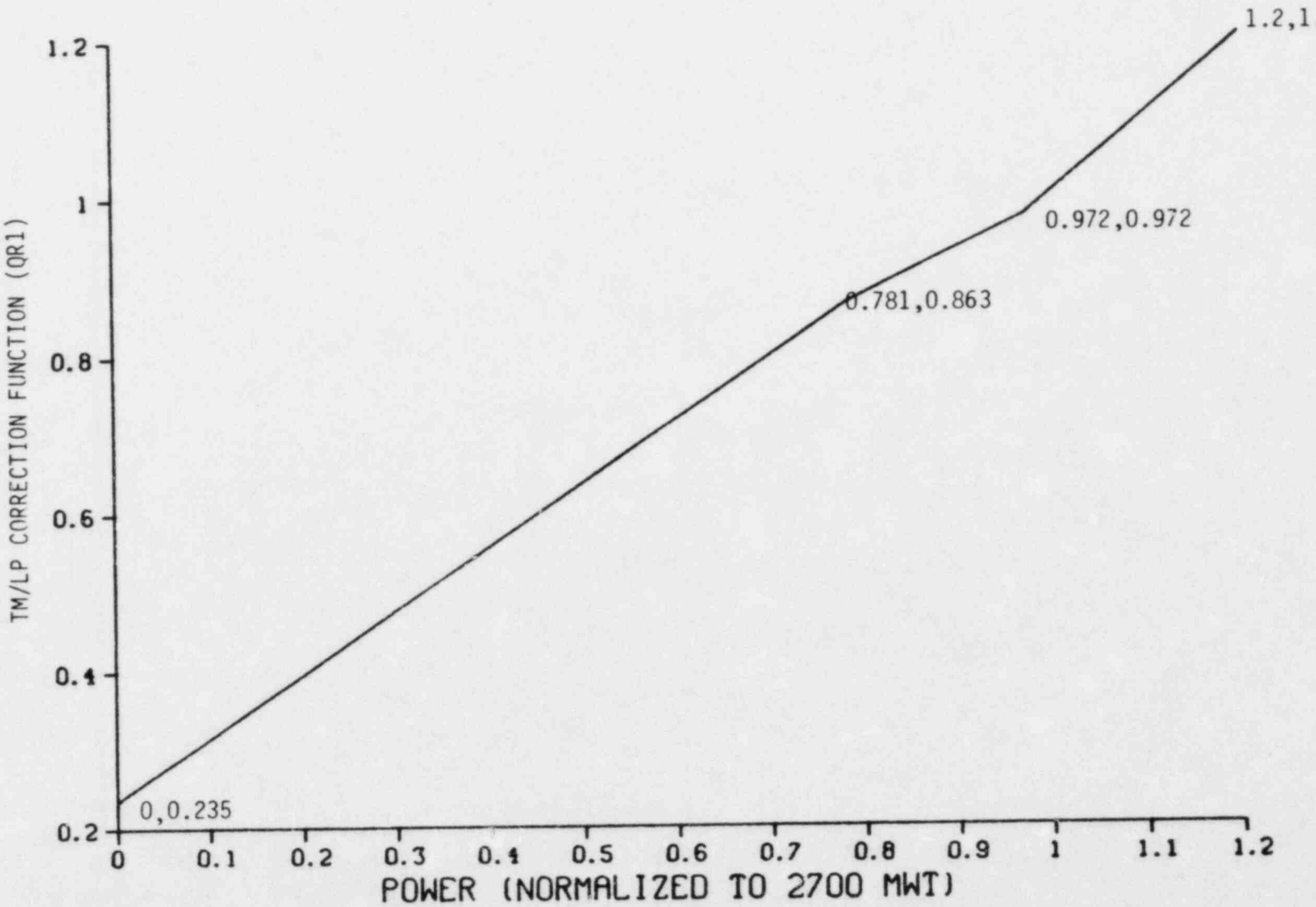

Figure 2.4 St. Lucie Unit 1 - TM/LP Correction Function QR1

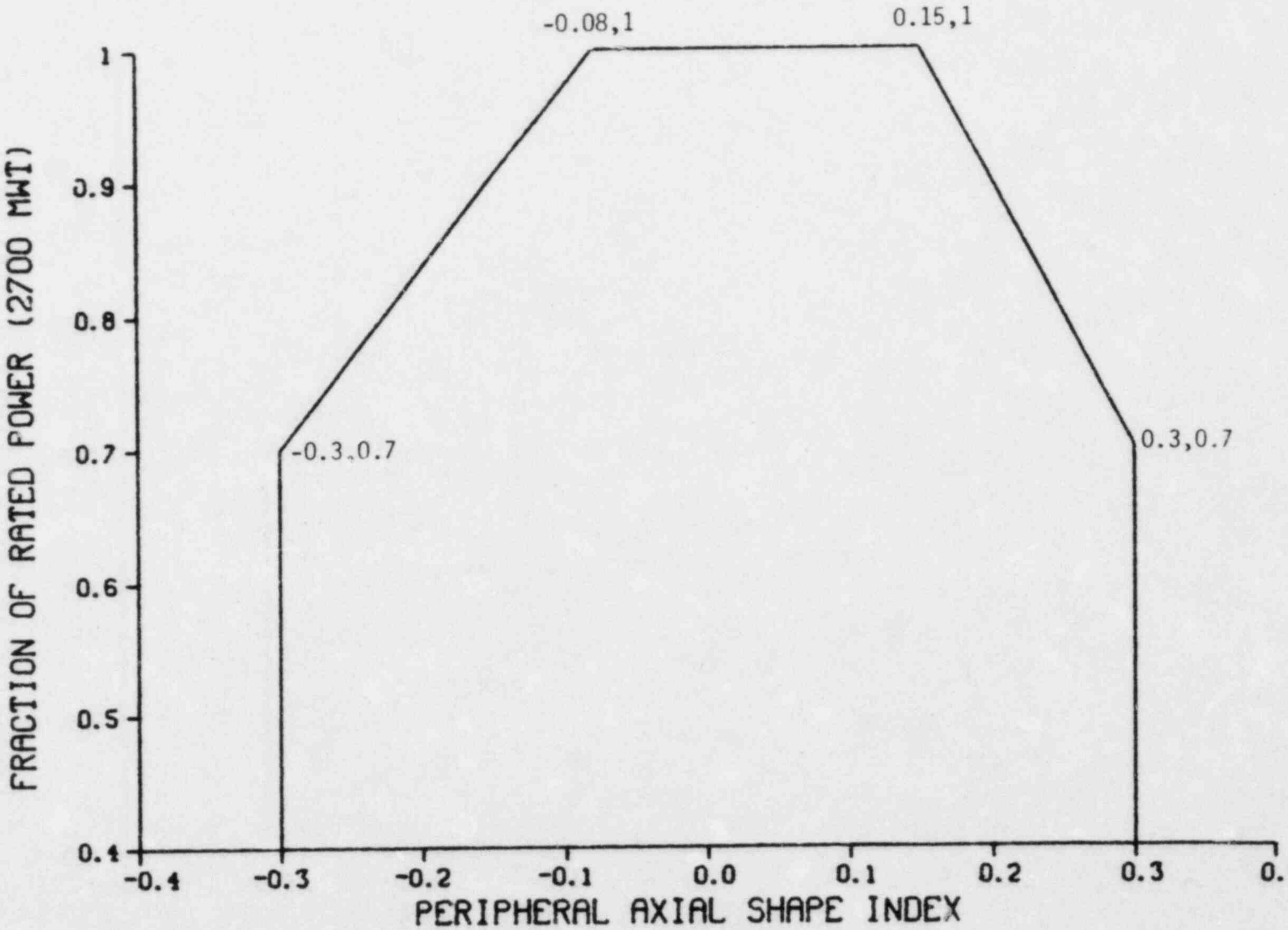

Figure 2.5 DNB Limiting Condition Of Operation For St. Lucie Unit 1

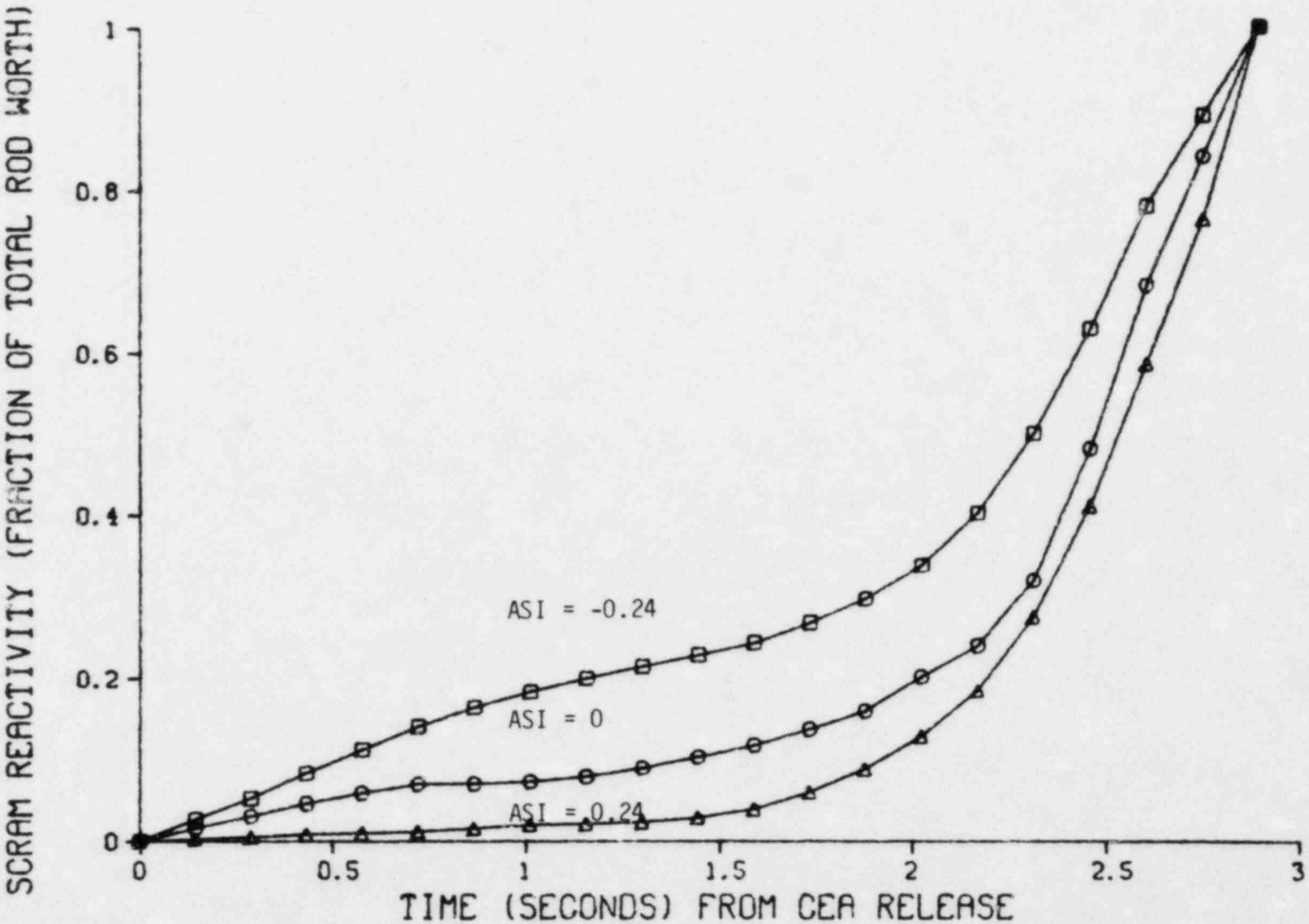

Figure 2.6 Scram Curves For St. Lucie Unit 1 - Cycle 6

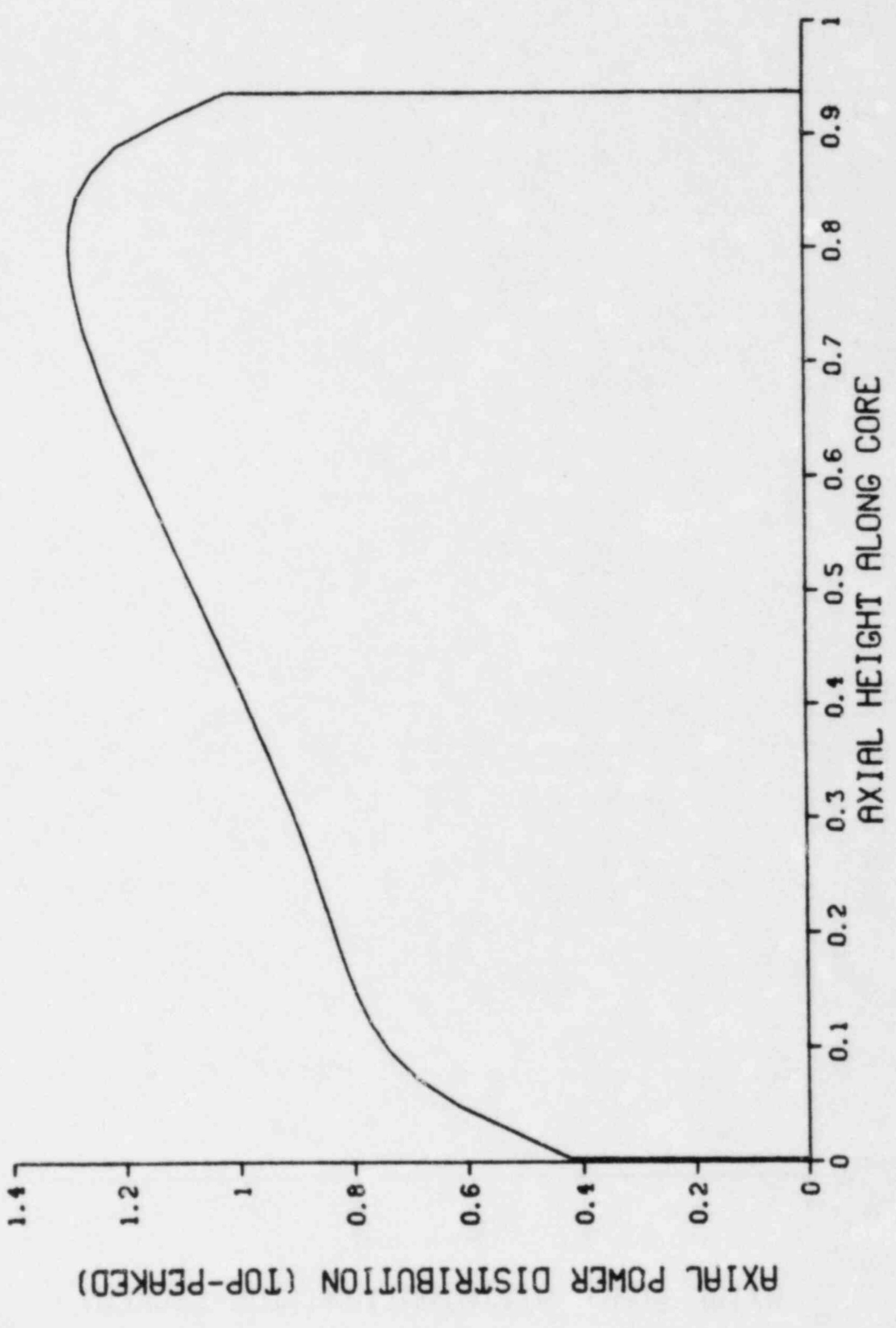

Figure 2.7 St. Lucie Unit 1 - Cycle 6 - DNB Limiting Top Peaked Core (ASI = -0.18)

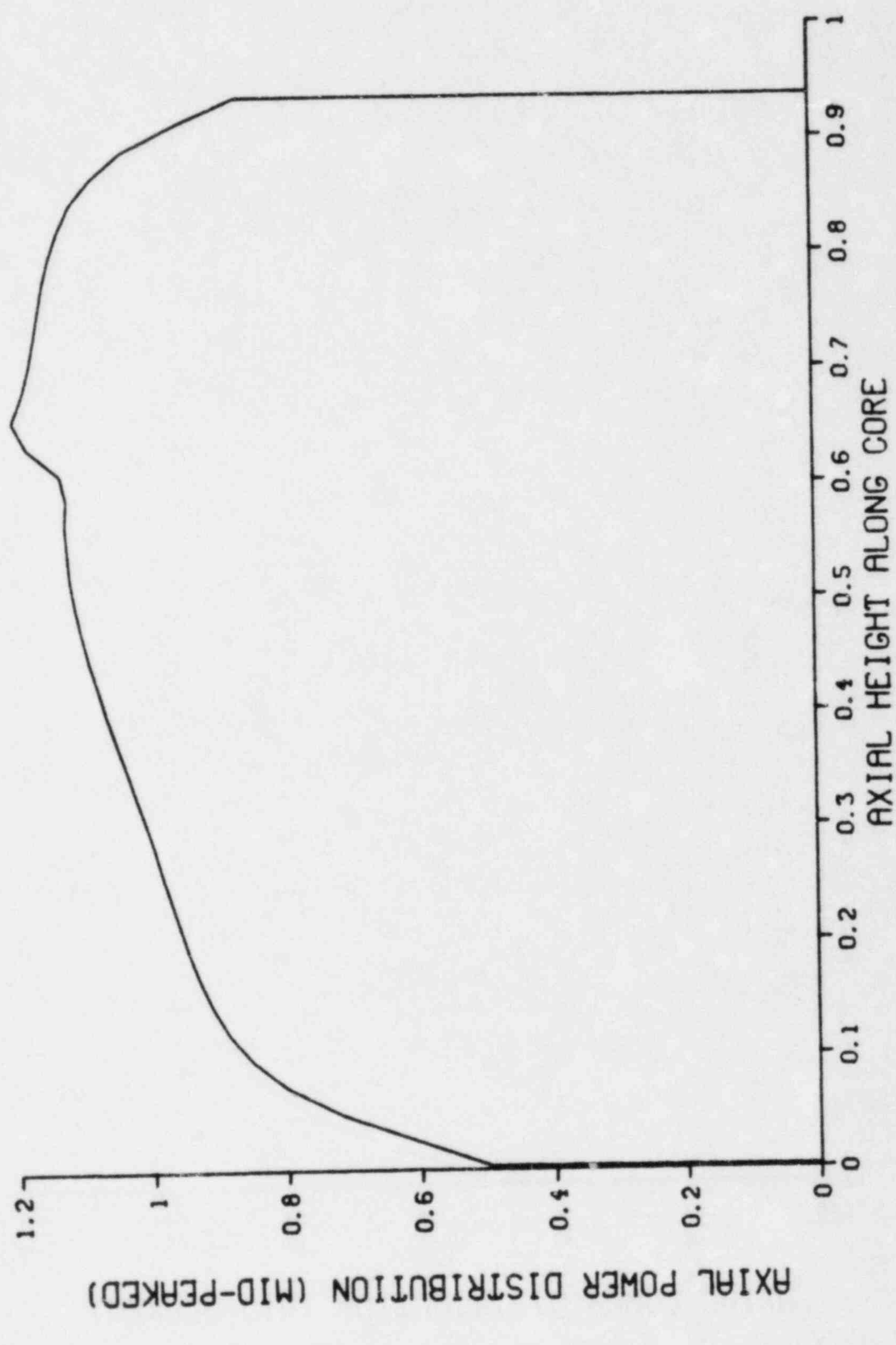

Figure 2.8 St. Lucie Unit 1 - Cycle 6 - DNB Limiting Mid-Peaked Core (ASI = 0)

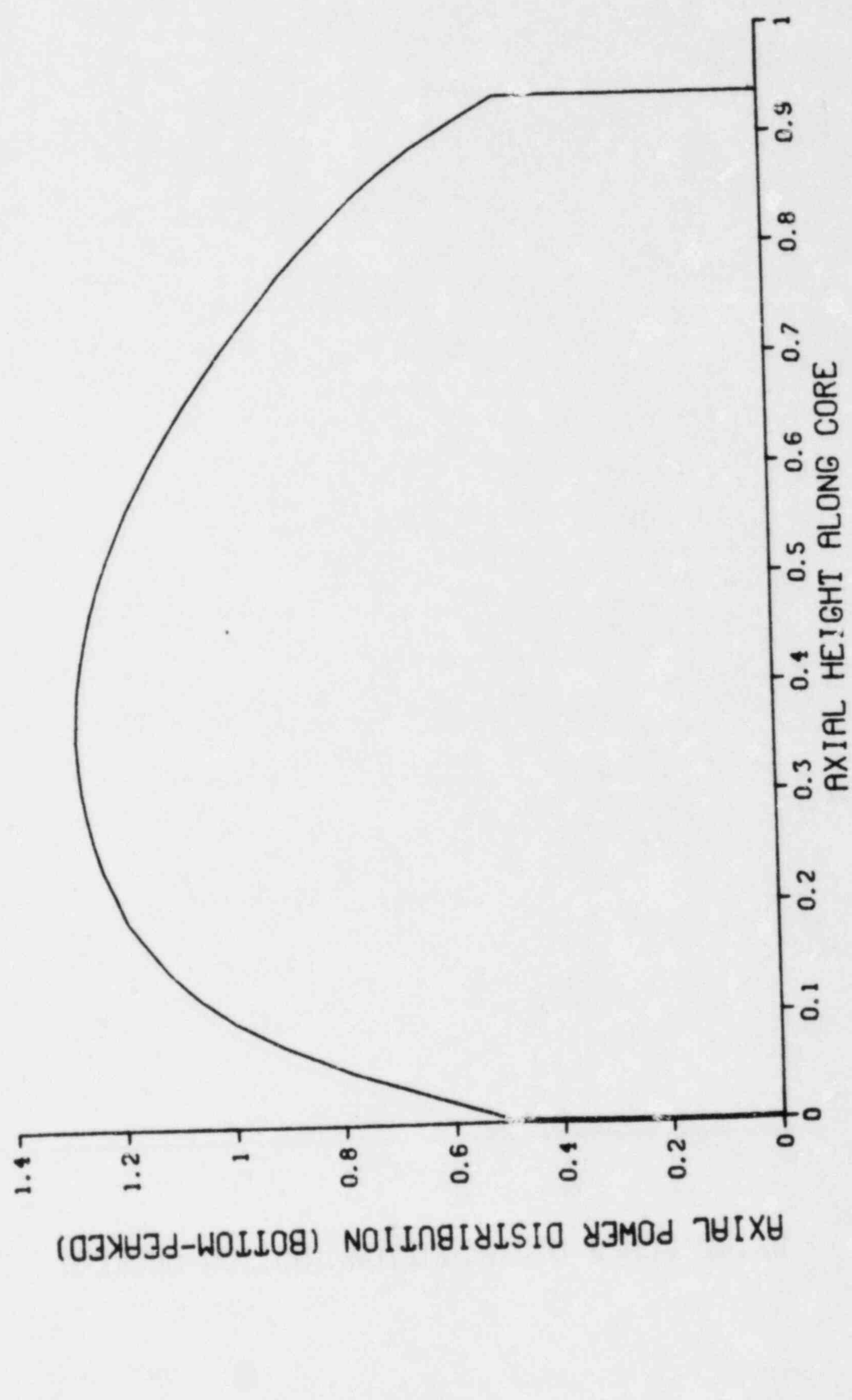

Figure 2.9 St. Lucie Unit 1 - Cycle 6 - DNB Limiting Bottom-Peaked Core (ASI = 0.14

3.0 TRANSIENT ANALYSIS

The transients analyzed for St. Lucie Unit 1 are categorized as either Anticipated Operation Occurrences (AOOs) or Postulated Accidents (PAs). The AOOs are further categorized as either requiring only the action of the reactor protection system (RPS) to meet the Specified Acceptable Fuel Design Limits (SAFDLs) or those requiring RPS action and/or observance of the Limiting Conditions of Operation (LCO).

Table 3.1 lists the transient events considered and summarizes the disposition of each transient. The Boron dilution event is not analyzed since as a reactivity insertion transient at power, it is bounded by the CEA withdrawal transient. For shutdown modes, there is sufficient shutdown margin to meet applicable operator action time criteria for the boron dilution event. Other transients not re-analyzed include the loss-of-feedwater and excess feedwater flow transients producing heatup and cooldown rates which are less severe than those produced by the loss-of-load and the excess load transients, respectively. Further, the loss-of-A.C.-power event was not considered since it is bounded by the loss-of coolant-flow transient with respect to thermal margin. Asymmetric steam generator events test the effectiveness of the asymmetric steam generator protective trip (ASGPT) in providing a scram signal sufficiently in advance of the time that mismatched cold leg temperatures would reach the core thereby assuring that highly tilted core power distributions do not occur. Ample margin was demonstrated for prior cycles(2) and ENC fuel will not affect the operation of the ASGPT. Hence asymmetric steam generator transients, other than the steam line rupture, have not been re-analyzed. The steam tube rupture is unchanged since the

TM/LP trip still protects against fuel damage[8], thus removing any fuel dependence in this transient.

3.1 ANTICIPATED OPERATIONAL OCCURRENCES REQUIRING ONLY RPS ACTION

The transients analyzed which fall into this category are: the loss-of-load transient, the excess load transient and the RCS-depressurization transient.

3.1.1 LOSS OF LOAD EVENT

This event was analyzed to simulate plant performance upon a turbine trip without a direct reactor trip. The abrupt loss-of-heat sink results in a rapid rise in the reactor coolant system (RCS) temperature and an expansion of the coolant which produces an insurge of water into the pressurizer and, ultimately, an increase in pressurizer pressure. The criterion employed is that the peak transient pressure must not exceed the ASME code limit of 110% of design pressure (i.e. 2750 psia). The SAFDLs were not approached in this transient since power was appreciably less than that required to reach 21 kW/ft and MDNBR occurred at steady state operation at the start of the event. The transient was initiated from 102% power with bounding EOC conditions and the bottom-peaked core shown in Figure 2.9. The pressurizer spray was turned off and the effects of the relief valves were also ignored in order to produce as high a pressure as possible during the simulated transient. The steam dump and bypass were also removed from the model for the same reason. The kinetics parameters used in this analysis are listed in Table 3.2.

Figures 3.1 to 3.11 show the simulated plant response for this event. A high pressure trip occurred at 4.45 seconds and the peak pressure reached was 2657 psia. The pressure was limited by the operation of the safety valve (See Figure 3.8). The MDNBR, Figure 3.10, did not fall below the initial value and the primary temperature increased by 15.5°F. Table 3.3 summarizes the events during the transient.

3.1.2 Excess Load Event

Inadvertent opening of the turbine control valve, steam dump valves and/or the steam bypass valve would result in increased steam flow and increased heat extraction. The resultant cooldown of the RCS would produce a positive reactivity insertion at EOC conditions when a large, negative moderator feedback coefficient exists. Protection against core damage is provided by the variable high power trip (VHPT), the low steam generator pressure trip, and the TM/LP trip.

The mid-plane-peaked axial power shape used in this analysis is shown in Figure 2.8. This particular shape has an ASI of zero and prevents the local power density (LPD) trip from occurring during the event. A top-peaked axial power shape, such as shown in Figure 2.7, would have provided a lower initial MDNBR, but would have led to a rapid LPD trip, mitigating the transient effects. The pressurizer heaters are assumed to be inoperable to provide a conservative MDNBR calculation. The kinetics parameters used in the simulation are listed in Table 3.4.

The limiting excess-load transient is the simultaneous opening of steam dump and bypass valves. The plant response to this event was simulated by rapidly ramping steam flow to 143.4% of rated flow. Figures 3.12 to 3.22 show the simulated plant response. The reactor tripped on the VHPT at 7.42 seconds and the peak power, Figure 3.12, reached 129%. The MDNBR, Figure 3.21, fell to 1.385 at 7.5 seconds. The reactor scram would have occurred much sooner were it not for the 12 second time delay associated with the cold leg RTDs. The TM/LP trip is also nearly simultaneous with the VHP trip. The event sequence is given in Table 3.5.

3.1.3 RCS Depressurization Event

The RCS depressurization event was used in assessing the bias term in the TM/LP trip(2). Trip processing delays and measurement uncertainties were used to establish the value of that bias.

The event simulated was a failure of both pressurizer relief valves fully open. The kinetics parameters listed in Table 3.6 are bounding BOC values and were used in the simulation. The pressurizer heater capacity was set to zero to allow a more rapid depressurization, and a top-peaked core power distribution, as shown in Figure 2.7, was used to minimize initial MDNBR.

Upon the failure of the relief valve, the RCS pressure fell rapidly, as shown in Figure 3.30, and a reactor trip on the LPD function occurred at 10.9 seconds. The MDNBR was 1.389 at 10.9 seconds. Figures 3.23 to 3.33 show the simulated plant response for this event. A summary of the transient events sequence is given in Table 3.7.

The trip uncertainties were treated in a deterministic fashion for this case. Had the TM/LP, the operation of which was verified statistically(8), been treated statistically, the trip would have occurred on the TM/LP rather than the LPD trip. Hence, even without the LPD trip, the TM/LP trip would still protect the DNBR limit for the XNB critical heat flux correlation. It is thus concluded that the bias in the TM/LP is sufficient to protect the core during this event.

3.2 ANTICIPATED OPERATIONAL OCCURRENCES REQUIRING RPS ACTION AND/OR OBSERVANCE OF THE LCOs

The transients discussed in this subsection require the observance of the LCOs for DNB and for linear heat rates in order to protect the SAFDLs, and consist of: the loss-of-coolant flow event, the CEA withdrawal event, and the CEA drop event.

3.2.1 Loss-of-Coolant-Flow Event

Flow reductions result in an increase in enthalpy rise across the core and a subsequent increase in coolant temperature in the hot leg of the RCS. The increased local enthalpy and decreased flow result in a reduction of margin to DNB in the core. The most severe transient, a loss of power to all four RCS pumps simultaneously, was evaluated by simulating a coastdown of all four RCS pumps in the PTSPWR2 model and observing the MDNBR for the transient.

Bounding BOC kinetics, listed in Table 3.8, were used along with the top-peaked axial power distribution shown in Figure 2.7.

The event sequence for the transient is summarized in Table 3.9. Figures 3.34 to 3.44 show the simulated plant responses to the four-pump coastdown. The reactor tripped in 2.1 seconds with the minimum DNBR, Figure 3.43 reaching 1.26 in 2.25 seconds. The pressurizer pressure increases to 2401 psia at 5.71 seconds.

3.2.2 CEA-Withdrawal Event

An inadvertent withdrawal of a bank of CEAs introduces positive reactivity which increases both core power and heat flux. Two potential initiators of this event are: (1) operator error; and (2) a malfunction of either the CEA drive mechanism or of the drive control system which results in an uncontrolled, continuous withdrawal of a CEA bank. Heat extraction through the steam generator remains constant and the increased power is converted to heat in the RCS. Protection against violation of either of the SAFDLs is provided by the variable high power trip (VHPT), the TM/LP trip, or the LPD trip.

An uncontrolled rod withdrawal was simulated with PTSPWR2 by increasing the reactivity linearly at a rate which conservatively bounds that which can be achieved in the reactor. Bounding BOC kinetics, Table 3.10, were used in conjunction with a mid-peak axial shape, Figure 2.8, to provide a conservative estimate of the reactor performance. This choice of shape prevents an almost instantaneous LPD trip from occurring. Further, since slow withdrawals are protected by the TM/LP trip, a fast withdrawal rate was simulated to provide the greatest power overshoot associated with the scram delay.

The simulated plant response for a reactivity insertion, $1.63 \times 10^{-4}\ \Delta\rho$/sec, from full power is displayed in Figures 3.45 to 3.55. The overpower transient is terminated by the VHPT at 116% power in 3.86 seconds. The MDNBR falls to 1.59 at 3.6 seconds (see Figure 3.54) and the pressure rises to 2363 psi at 7.3 seconds. The sequence of events for this simulation is summarized in Table 3.11.

3.2.3 CEA Drop Event

A failure in the CEA drive mechanism can result in an inadvertent full-length insertion of a CEA during power operation. Fixed demand from the turbine would cause a cool-off transient in the RCS and, for negative moderator feedback, a return to the original power with a significantly greater radial peaking on the core. Since the power initially decreases following the dropping of the CEA, no reactor trip occurs and protection of the SAFDLs is provided solely by the LCOs.

This event was simulated by introducing a step decrease in total reactivity at a steady-state, full power. Bounding EOC kinetics parameters, Table 3.12, were used and the reactivity insertion was selected to conservatively bound that due to the most reactive CEA being inserted. A radial peaking factor of 110% was included during the return to power. A top-peaked axial distribution, Figure 2.7, was used to model the hot channel in order to provide a conservative DNBR trace. During the cooldown transient, inlet temperature fell, mass flow rose and pressure, which was not controlled in this transient, fell. The increased radial peaking and reduced pressure tended to decrease the DNBR while the decreased inlet temperature and increased flow tended to increase the DNBR.

Table 3.13 summarizes the event sequence of the transient. Flow, Figure 3.62, increased due to the cool-off, Figure 3.58, and pressure, Figure 3.63 fell to a minimum of 2215 psia at 20 seconds, and recovers to 2242 psia by 200 seconds. The DNBR fell to 1.485 at 115 seconds and recovered to 1.49 by 200 seconds. Figures 3.56 to 3.66 depict the plant response for the transient.

3.3 POSTULATED ACCIDENTS

The events discussed in this subsection are assumed to occur infrequently and are not required to meet the SAFDLs. The ultimate criterion applied to these transients is a radiation exposure limit. In assessing the safety of the XN-1 reload fuel, a comparison of expected pin failure with prior cycles is used to judge the acceptability of the fuel performance. Fuel failure is conservatively assumed coincident with the occurrence of DNB. Hence, for the two accidents analyzed in this subsection, the expected number of fuel pins undergoing DNB was used as the evaluation criterion.

3.3.1 Primary-Pump-Seizure Event

The instantaneous loss of pumping power caused by disintegration of the pump impeller or a complete seizure of the pump shaft would result in a rapid flow decrease through the affected cold leg, and would cause a reactor trip due to low flow in that loop. The flow reduction rate would be more drastic than in a total loss of pumping power and would create a more rapid approach to DNB. The increase in enthalpy of the RCS coolant would also result in a pressure increase. The transient was therefore analyzed once to minimize DNBR and once to maximize RCS pressure. For both analyses, bounding BOC kinetics parameters, Table 3.14, were used to maximize the power excursion and delay the shutdown of the power following the trip.

The transient was first simulated by stopping one of the four pumps at full-power operation, and by using the axial power distribution shown in Figure 2.7. This shape provides the most conservative MDNBR. Pressurizer pressure control was retained so that the spray would decrease the pressure transient. The results of the simulation are shown in Figures 3.67 to 3.77. The event sequence is summarized in Table 3.15. A minimum DNBR of 1.189 at 1.2 seconds was obtained using the XNB correlation. Fuel damage due to DNB would have been significantly less than 1% for this DNBR. Core average temperature increased $6^{o}F$ and peak pressurizer pressure reached 2338 psia at 3.92 seconds.

The transient was also simulated by stopping one of the four pumps at full power operation, and by using the axial power distribution shown in Figure 2.9. This shape delays the reactor shutdown since the core power is peaked much lower. The pressurizer spray and relief valves were removed from the model to increase the pressure transient. The results of the simulation are shown in Figures 3.78 to 3.88. The event sequence is summarized in Table 3.16. A minimum DNBR of 1.45 occurred at 1.2 seconds and the core average temperature rose $10.7^{o}F$ resulting in a peak pressure of 2397 psia at 4.6 seconds.

3.3.2 Steam-Line-Break Event

A break of a steam line pipe would result in an increase in heat removal which would reduce the RCS coolant temperature as it withdrew more heat than was being produced by the reactor. For a negative moderator temperature coefficient, this cooldown would result in positive reactivity insertion and could lead to a return to criticality following the reactor trip and could result in core damage caused by DNB occurring as a result of the loss of pressure control. A large double-ended guillotine break of the large steam pipe at the steam generator exit at hot zero power (HZP) and EOC conditions has been shown to be the most limiting accident for a return to power(2). The secondary side of the steam generator is at its highest pressure (902 psia), i.e. has the greatest inventory of cold water and the moderator temperature coefficient is most negative.

For increased conservatism the most reactive CEA was assumed to be stuck fully withdrawn and a conservative radial peaking of 17 was applied to the hot channel for which the top-peaked axial shape shown in Figure 2.9 was used. Boron injection was modeled with one of the 3 injection pumps operating to introduce borated water (1720 ppm) via a safety injection line (volume=12.0 cu.ft) which was assumed to be boron-free initially. High Pressure Safety Injection (HPSI) was initiated by a low pressurizer pressure signal and the pump performance curve in Figure 3.89 was used to calculate the injection rate. The initial pressurizer level was set to the HZP level of 33% of span. The boron makeup tanks were not included in the model.

Passive injection of highly borated water (approximately 15,000 ppm) from the safety injection tanks was also modeled. This protective feature did not enter since assumed stratification in the reactor head at natural circulation caused the system pressure to remain above the 215 psia maintained in the safety injection tanks.

The transient was initiated from HZP (T_{AVE}=532°F) by introducing a break (6.35 ft^2 for S.G. #1 and 2.35 ft^2 for S.G. #2) in the steam line. The moderator temperature feedback is given by Figure 3.90 and the Doppler feedback by Figure 3.91. Since Doppler feedback for zero power cannot be calculated from Figure 3.91, because the reactor was subcritical, Doppler feedback was calculated as, $-2.3 \times 10^{-3}\ \Delta T\ \%\ \Delta\rho$. The break flows were calculated using the Moody curve for critical flow of saturated steam(6). The carry-over fraction at the break was conservatively assumed to be zero to increase the total cooldown. The steam flow from the intact generator, which had to pass through the flow restrictors in both steam lines, was terminated after 7 seconds by the action of the main Steam Line Isolation Valves (MSIVs).

The system responses are shown in Figures 3.92 to 3.101 and the transient event sequence is summarized in Table 3.17. Initial steam line flow was 467% of rated. The steam generator with the broken line emptied in 176 seconds. After the pressurizer emptied, the RCS pumps were tripped and, after a 3 second delay, the HPSI was initiated. Following pressurizer emptying, pressure control was lost and system pressure fell to hot leg saturation. At 60 seconds flow reversed in the intact loop and the core inlet temperature approached the cold leg temperature of the broken loop. At 188 seconds auxiliary feedwater flow was initiated and a second cooldown transient ensued. The maximum return to power, 37.2 MWt, occurred at 221 seconds. This power level was not sufficient to cause significant fuel or coolant heating.

3.4 BOUNDING MODERATOR TEMPERATURE ANALYSIS

The bounding moderator temperature coefficient (MTC) for BOC at hot zero power (HZP) described in Table 2.4 was verified by simulating several full-power transients which are sensitive to positive moderator temperature coefficient using 7×10^{-5} $\Delta\rho$ for the MTC. The transients are: the depressurization event, the loss-of-coolant-flow event, and the seized-rotor event. Since the MTC decreases from HZP to HFP, the results bound the possible consequences of these events.

For the depressurization transient, peak power was 2843.2 MWt and the MDNBR was 1.377. For the loss-of-coolant-flow transient, the peak power was 2859.5 MWt, and the MDNBR dropped from 1.326 to 1.266. For the seized-rotor transient, the peak power was 1955.3 MWt, and the MDNBR dropped from 1.1.89 to 1.127.

Of these three transients, only the seized-rotor transient produces a measureable probability of DNB occurring in the hot channel. Using conservative statistics for the XNB correlation, less than 1% of the pins would be expected to undergo DNB.

Analysis of transients at HFP indicates that the bounding MTC for HZP is acceptable.

Table 3.1 Transient Events

Transient	Disposition
AOOs Requiring Only RPS Action	
Boron Dilution	Not Analyzed
Loss of Load	Analyzed
Loss of Feedwater	Not Analyzed
Excess Load	Analyzed
Excess Feedwater	Not Analyzed
RCS Depressurization	Analyzed
AOOs Requiring RPS Action and/or LCO	
Loss of Coolant Flow	Analyzed
Loss of A.C. Power	Not Analyzed
CEA Withdrawal	Analyzed
CEA Drop	Analyzed
Asymmetric S.G. Transients	Not Analyzed
PAs	
Seized Rotor	Analyzed
Steam Line Rupture	Analyzed
S.G. Tube Rupture	Not Analyzed

Table 3.2 Kinetics Parameters for the Loss-of-Load Event

Parameter	Value
Moderator Temperature Coefficient	$1.6 \times 10^{-4}\ \Delta\rho/^{o}F$
Doppler Coefficient	$-8 \times 10^{-6}\ \Delta\rho/^{o}F$
Moderator Pressure Coefficient	$5 \times 10^{-6}\ \Delta\rho/^{o}F$
Beta (effective)	4.5×10^{-3}

Table 3.3 Event Table for a Loss-of-Load

Time (seconds)	Event	Value
2.38	Steam Line Safety Valves Opened	---
3.05	High pressure signal generated	2422 psia
3.86	Pressurizer safety valve opened	2500 psia
3.87	Peak power level occurred	3446.5 MWt
4.45	Reactor trip occurred (high pressure)	---
	Peak heat flux occurred	210544 Btu/hr-ft^2
7.43	Peak pressurizer pressure occurred	2657 psia
7.50	Peak average temperature occurred	590.4^{o}F
8.62	Peak core average temperature occurred	586.1^{o}F
29.52	Peak steam dome pressure occurred	1304 psia

Table 3.4 Kinetics Parameters for the Excess Load Event

Parameter	Value
Moderator Temperature Coefficient	-2.8×10^{-4} $\Delta\rho/^{\circ}F$
Doppler Coefficient	-8×10^{-6} $\Delta\rho/^{\circ}F$
Moderator Pressure Coefficient	3.6×10^{-6} $\Delta\rho$/psi
Beta (effective)	4.5×10^{-3}

Table 3.5 Event Table for an Excess Load

Time (seconds)	Event	Value
0	Open steam pump and bypass	43.4% of rated flow
7.42	Reactor Trip occurred (VHPT) Peak Power Level occurred	3482.7 MWt
7.50	MDNBR occurred	1.385
7.51	Peak Average Core Flux occurred	200512 BTU/hr-ft^2
12.69	Steam line safety valves opened	---
13.00	Peak steam dome pressure occurred	1004 psia

Table 3.6 Kinetics Parameters for the RCS Depressurization Event

Parameter	Value
Moderator Temperature Coefficient	$2 \times 10^{-5}\ \Delta\rho/^{o}F$
Doppler Coefficient	$8 \times 10^{-6}\ \Delta\rho/^{o}F$
Moderator Pressure Coefficient	$-1.4 \times 10^{-6}\ \Delta\rho/psi$
Beta (effective)	4.5×10^{-3}

Table 3.7 Event Table for RCS Depressurization

Time (seconds)	Event	Value
0	Failure of pressurizer relief valve	--
10.90	MDNBR occurred	1.389
10.91	Peak power level occurred	2841 MWt
	Peak core heat flux occurred	195829 Btu/hr-ft^2
	Reactor trip occurred (LPD)	--
10.93	Peak core average temperature occurred	572.8^{o}F
11.25	Peak average temperature occurred	574.7^{o}F
13.84	Steam line safety valves opened	--

Table 3.8 Kinetics Parameters for the Loss-of-Coolant Flow Event

Parameter	Value
Moderator Temperature Coefficient	$+2 \times 10^{-5}\ \Delta\rho/^{o}F$
Doppler Coefficient	$-8 \times 10^{-6}\ \Delta\rho/^{o}F$
Moderator Pressure Coefficient	$-1.4 \times 10^{-6}\ \Delta\rho/psi$
Beta (effective)	4.5×10^{-3}

Table 3.9 Event Table for a Loss-of-Coolant Flow

Time (seconds)	Event	Value
0	Loss of pumping power to all four pumps	---
2.10	Reactor trip occurred (low flow)	---
	Peak power occurred	2735 MWt
2.25	Minimum DNBR occurred	1.326
2.56	Peak core average temperature occurred	576.5°F
5.75	Peak average temperature occurred	577.4°F
5.71	Peak pressurizer pressure occurred	2401 psia
6.76	Steam line safety valves opened	---

Table 3.10 Kinetics Parameters for the CEA Withdrawal Event

Parameter	Value
Moderator Temperature Coefficient	$+2 \times 10^{-5}$ $\Delta\rho/^{o}F$
Doppler Coefficient	-6×10^{-6} $\Delta\rho/^{o}F$
Moderator Pressure Coefficient	-6×10^{-7} $\Delta\rho$/psi
Beta (effective)	4.5×10^{-3}

Table 3.11 Event Table for CEA Withdrawal

Time (seconds)	Event	Value
0	CEA withdrawal initiated	1.63×10^{-4} /second
3.85	Minimum DNBR occurred	1.590
3.86	Power level occurred	3131.8 MWt
	Peak core heat flux occurred	199908 Btu/hr-ft^2
	Reactor trip occurred (VHPT)	---
3.98	Peak core average temperature occurred	574.6^{o}F
7.20	Steam line safety valves opened	---
7.25	Peak average temperature occurred	575.6^{o}F
7.29	Peak RCS pressure occurred	2363 psia
12.37	Peak steam dome pressure occurred	1016.6 psia

Table 3.12 Kinetics Parameters for the CEA-Drop Event

Parameter	Value
Moderator Temperature Coefficient	$-2.8 \times 10^{-5} \Delta\rho/^{o}F$
Doppler Coefficient	$-2 \times 10^{-5} \Delta\rho/^{o}F$
Moderator Pressure Coefficient	$3.6 \times 10^{-6} \Delta\rho/psi$
Beta (effective)	4.5×10^{-3}

Table 3.13 Event Table for CEA Drop

Time (seconds)	Event	Value
0	CEA dropped	-0.00105
115	MDNBR occurred	1.485

Table 3.14 Kinetics Parameters for the Seized-Rotor Event

Parameter	Value
Moderator Temperature Coefficient	$+2 \times 10^{-5}$
Doppler Coefficient	$-8 \times 10^{-6}\ \Delta\rho/^{o}F$
Moderator Pressure Coefficient	$-1.4 \times 10^{-6}\ \Delta\rho/psi$
Beta (effective)	4.5×10^{-3}

Table 3.15 Event Table for Seized Rotor (DNB)

Time (seconds)	Event	Value
0	Seizure of Pump 1a	--
1.20	Minimum DNBR	1.189
1.23	Reactor trip (low flow) occurred	--
1.39	Peak core average temperature occurred	578.2°F
3.92	Peak pressure occurred	2338 psia
5.95	Steam line safety valve (loop 2) opened	--
5.99	Steam line safety valve (loop 1) opened	--

Table 3.16 Event Table for Seized Rotor (Pressure)

Time (seconds)	Event	Value
0	Seizure of Pump 1A	--
1.20	Minimum DNBR occurred	1.450
1.23	Reactor trip (low flow) occurred	--
1.49	Peak core average temperature occurred	584.1°F
4.60	Peak pressurizer pressure occurred	2397 psia
6.39	Steam line safety valve (loop 2) opened	--
6.45	Steam line safety valve (loop 1) opened	--

Table 3.17 Event Table for Steam Line Break

Time (seconds)	Event	Value
0	Large steam line break occurred	6.35 ft^2 SG # 1
		2.35 ft_2 SG # 2
7.00	Main steam isolation valves closed	-----
7.79	Peak core heat flux occurred	9682 Btu/hr-ft^2
8.00	Pressurizer emptied	-----
8.30	HPSI signal generated	1576 psia
18.30	Operator tripped RCS pumps	-----
44.54	Boron entered the loop	1720 ppm
60.0	Flow reversed in intact loop	-----
112.0	Peak reactivity reached	0.774 % $\Delta\rho$
124.0	Peak power reached	15.1 MWt
172.0	Flow recovered in intact loop	-----
176.0	Second peak reactivity reached	0.0817 % $\Delta\rho$
188.0	Auxiliary feedwater started	253.6 lbs/sec
216.0	Flow reversed in intact loop	-----
220.0	Second peak power reached	37.2 MWt

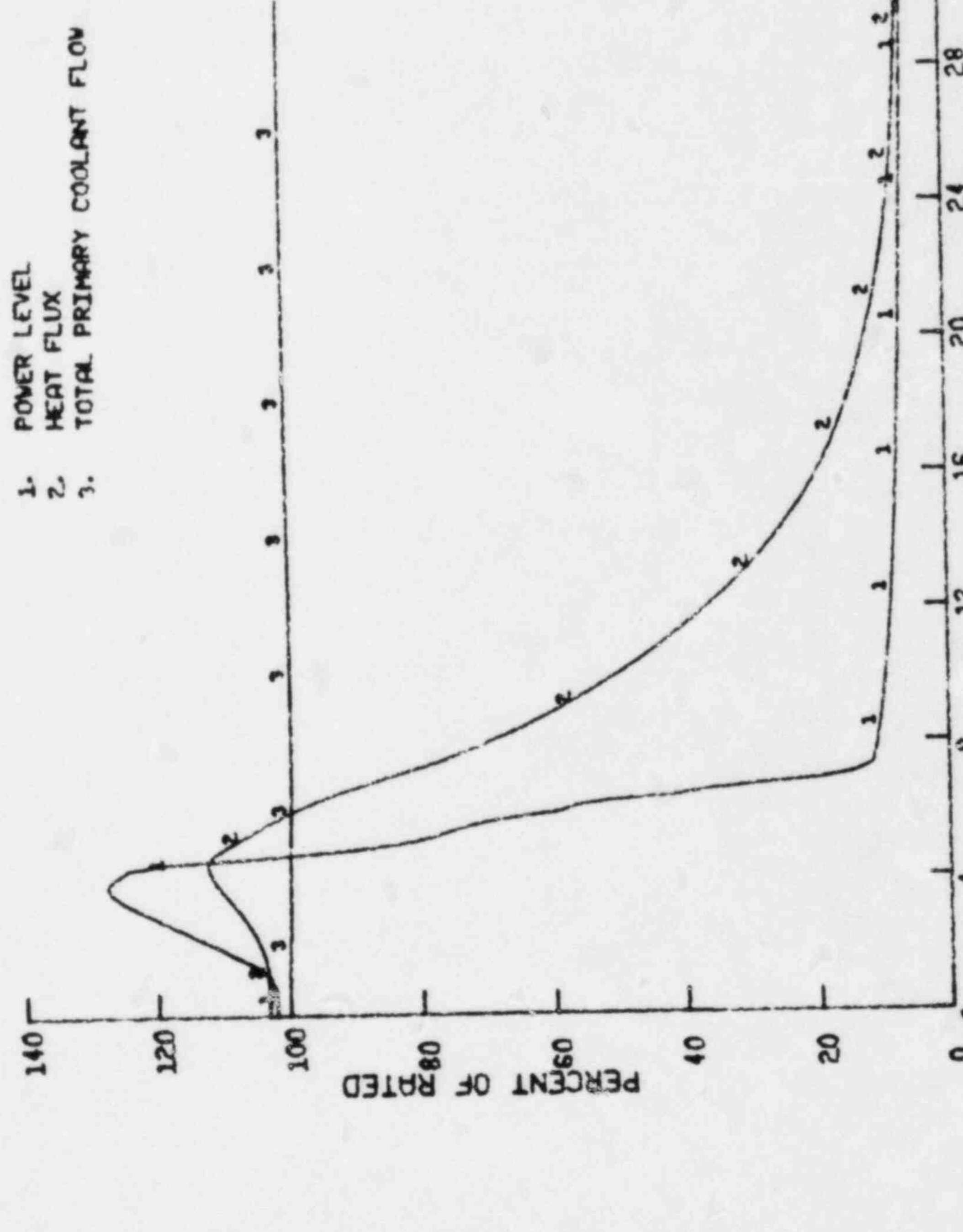

Figure 3.1 St. Lucie Unit 1 - Power, Heat Flux And Flow

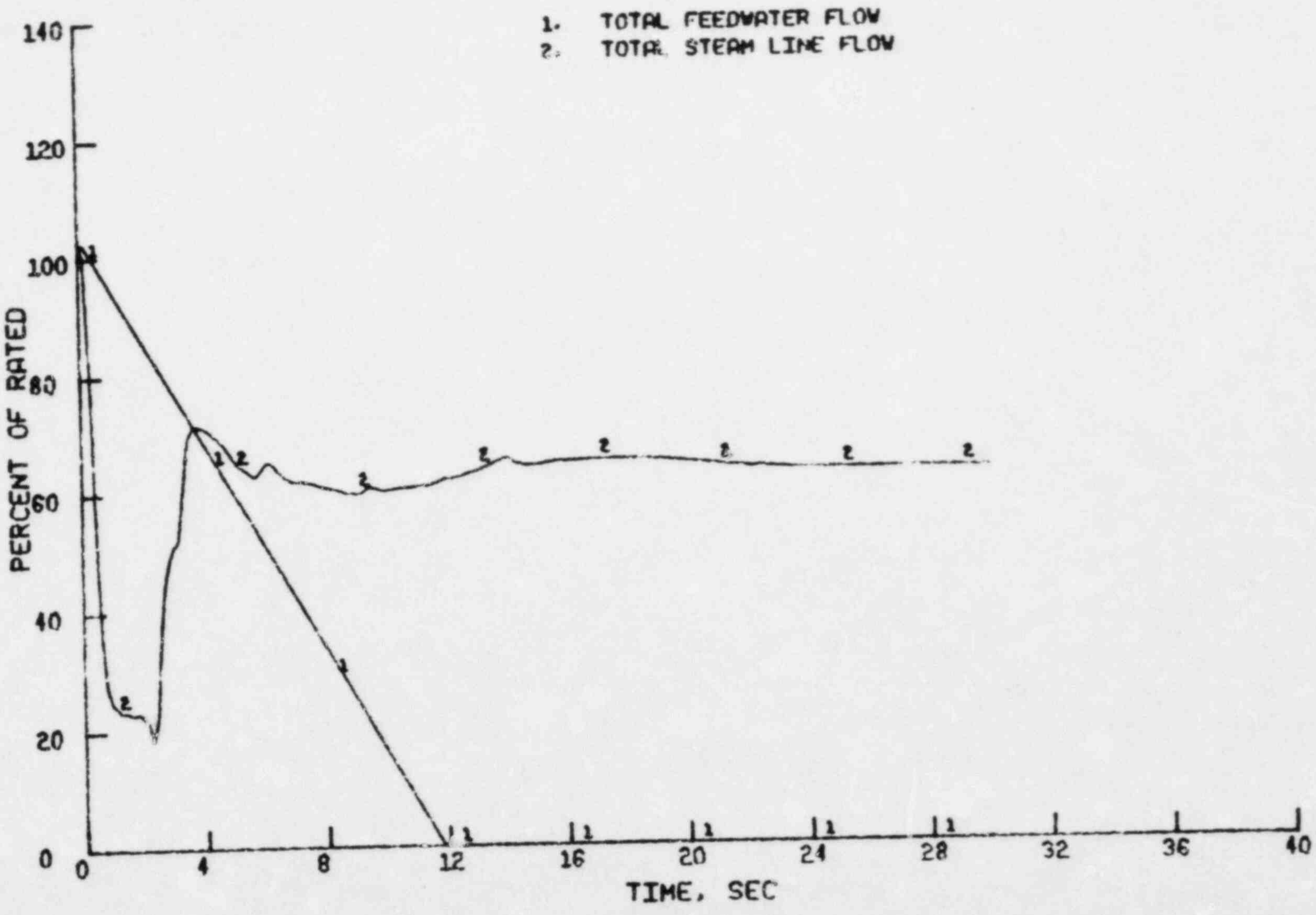

Figure 3.2 St. Lucie Unit 1 - S. G. Flow - Loss Of Electric Load

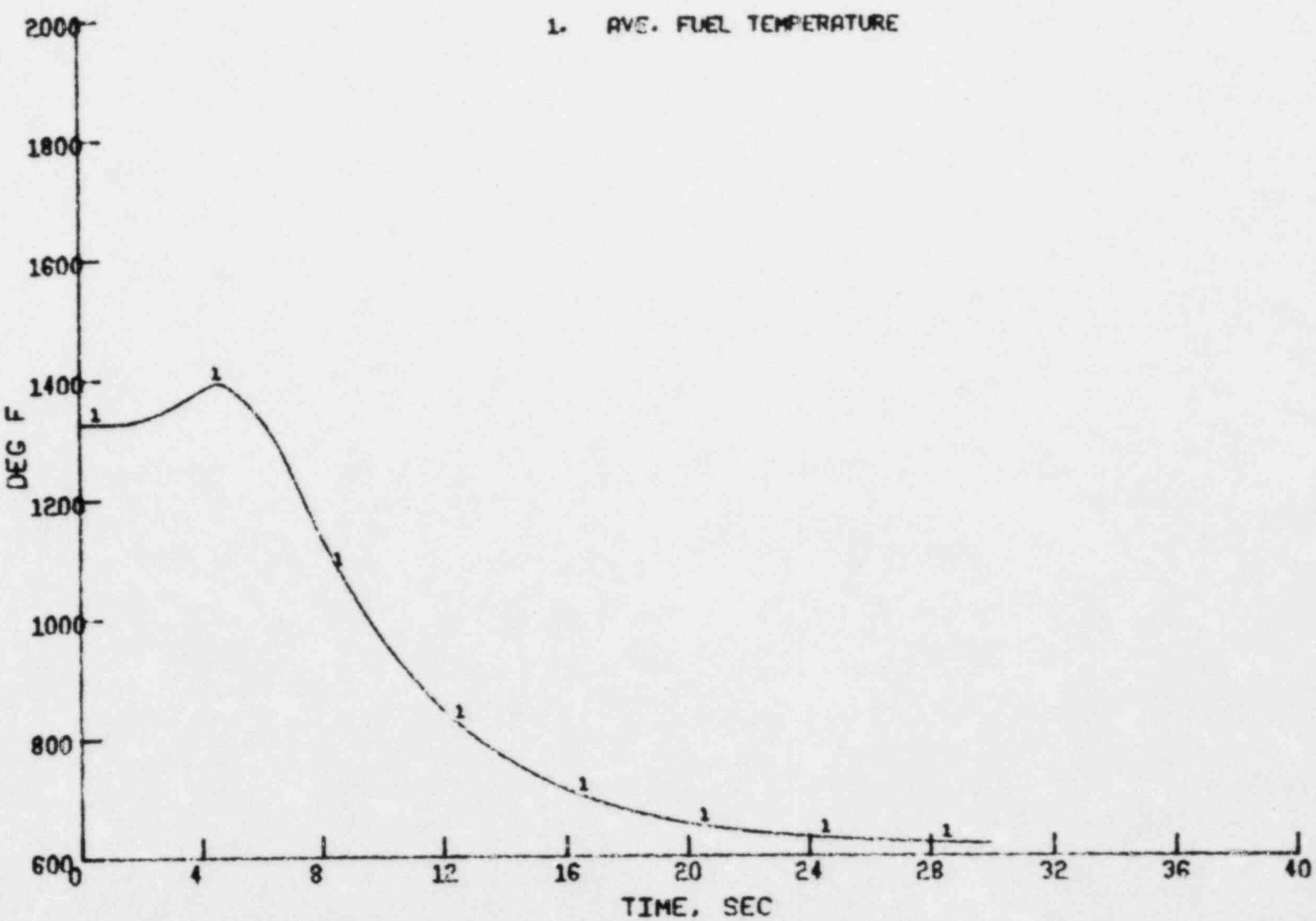

Figure 3.3 St. Lucie Unit 1 - Fuel Temperature - Loss Of Electric Load

16.54.44.

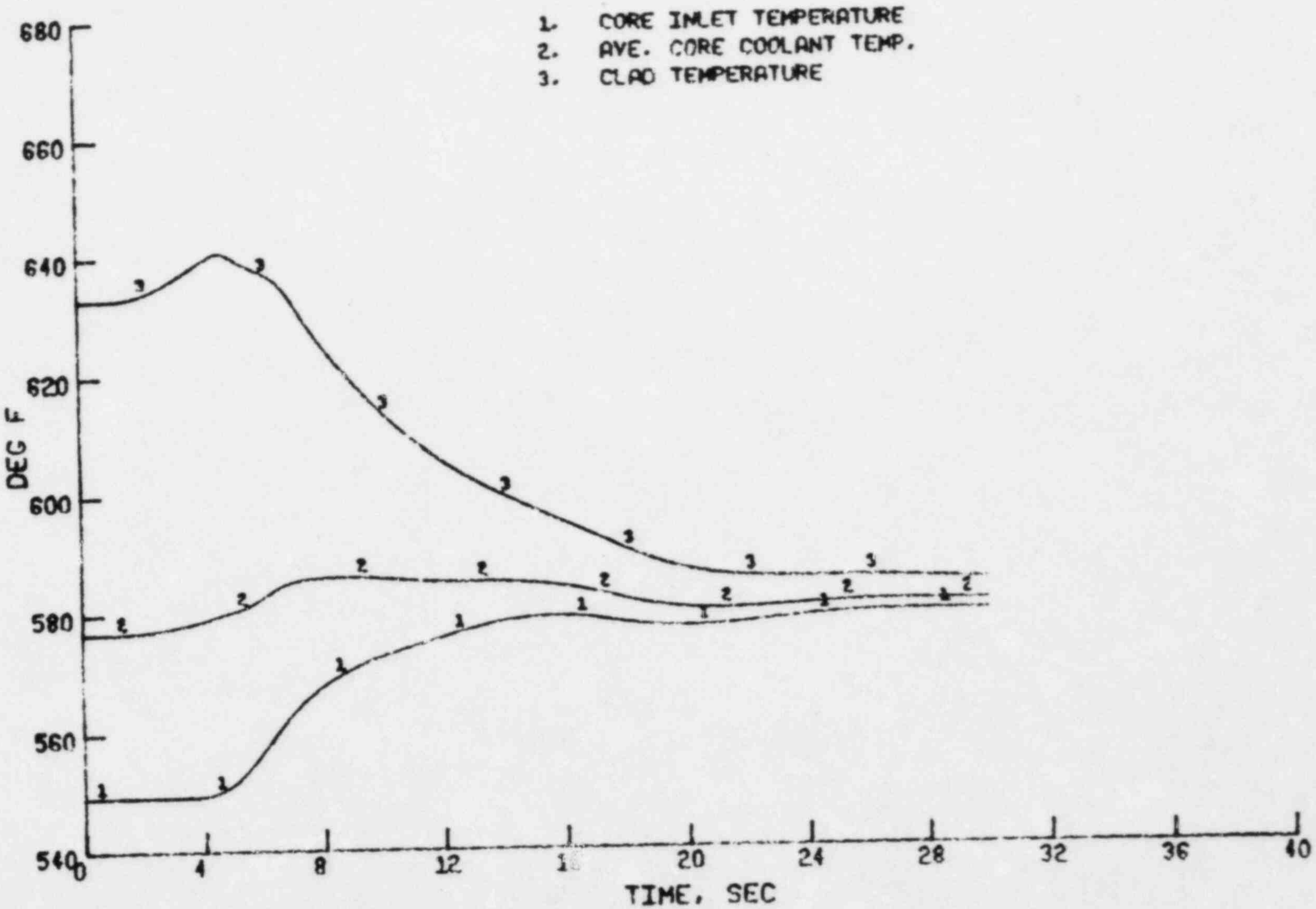

Figure 3.4 St. Lucie Unit 1 - Core Temperature - Loss Of Electric Load

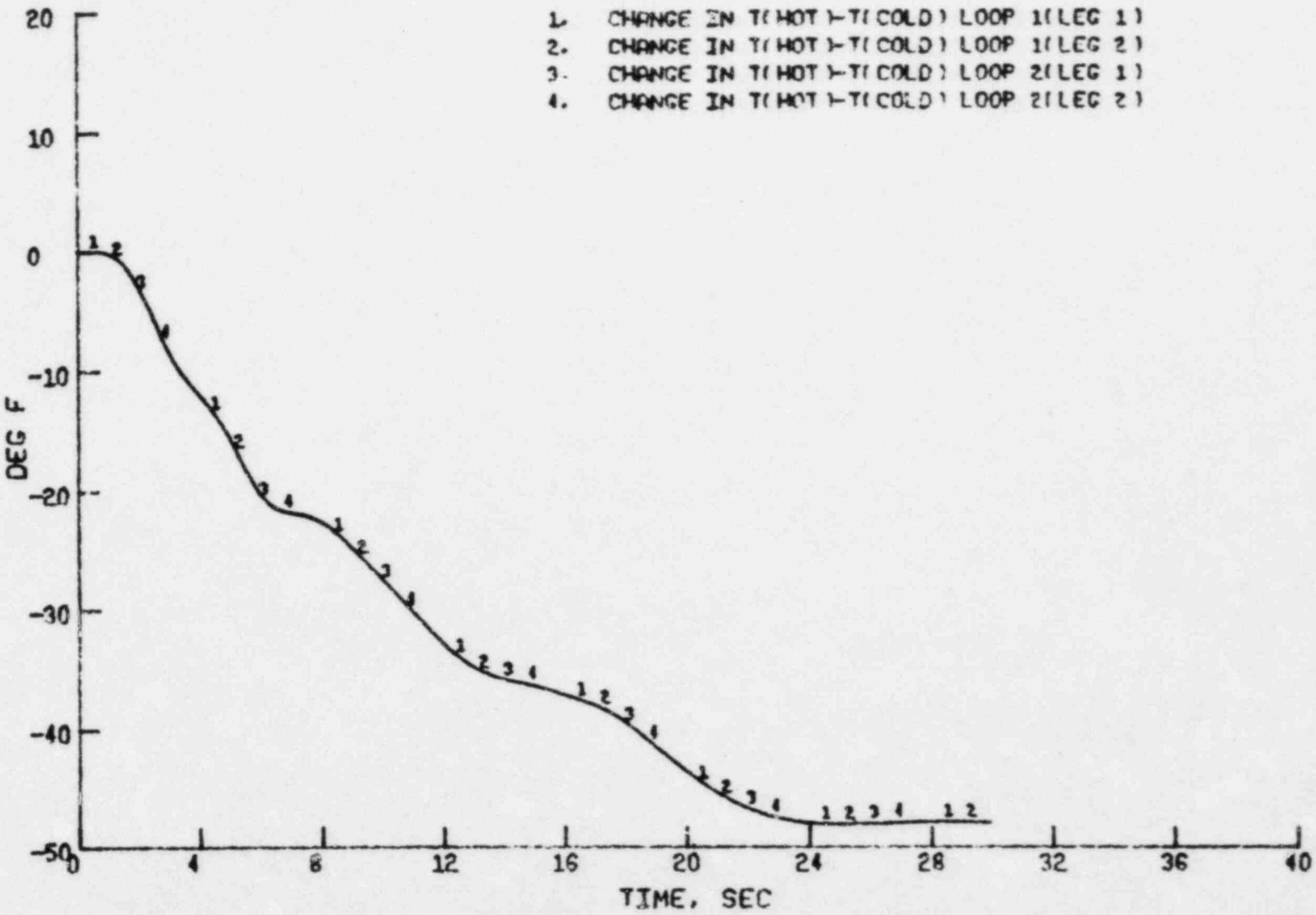

Figure 3.5 St. Lucie Unit 1 - Loop Temperature Differences - Loss Of Electric Load

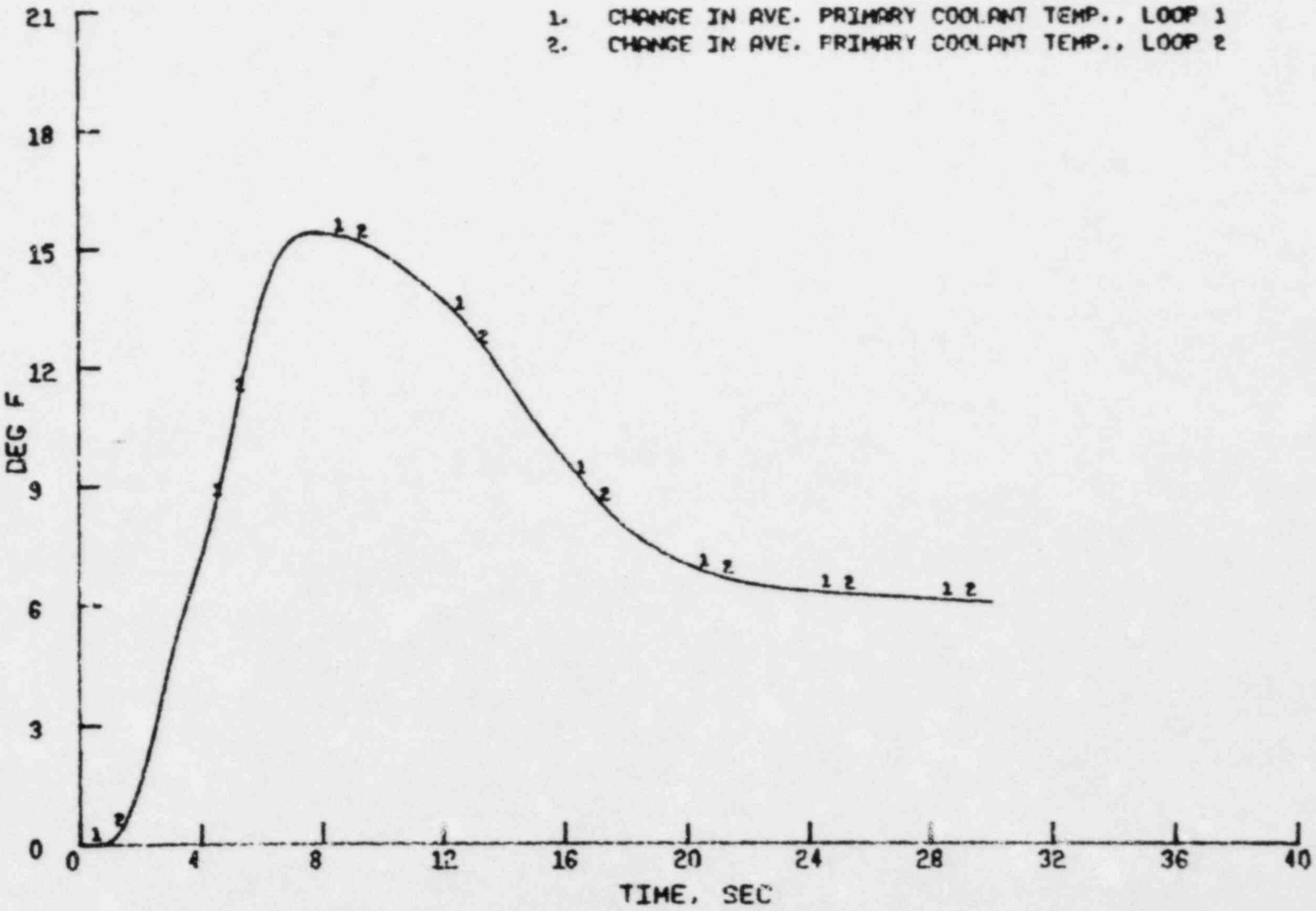

Figure 3.6 St. Lucie Unit 1 - Average Temperatures - Loss Of Electric Load

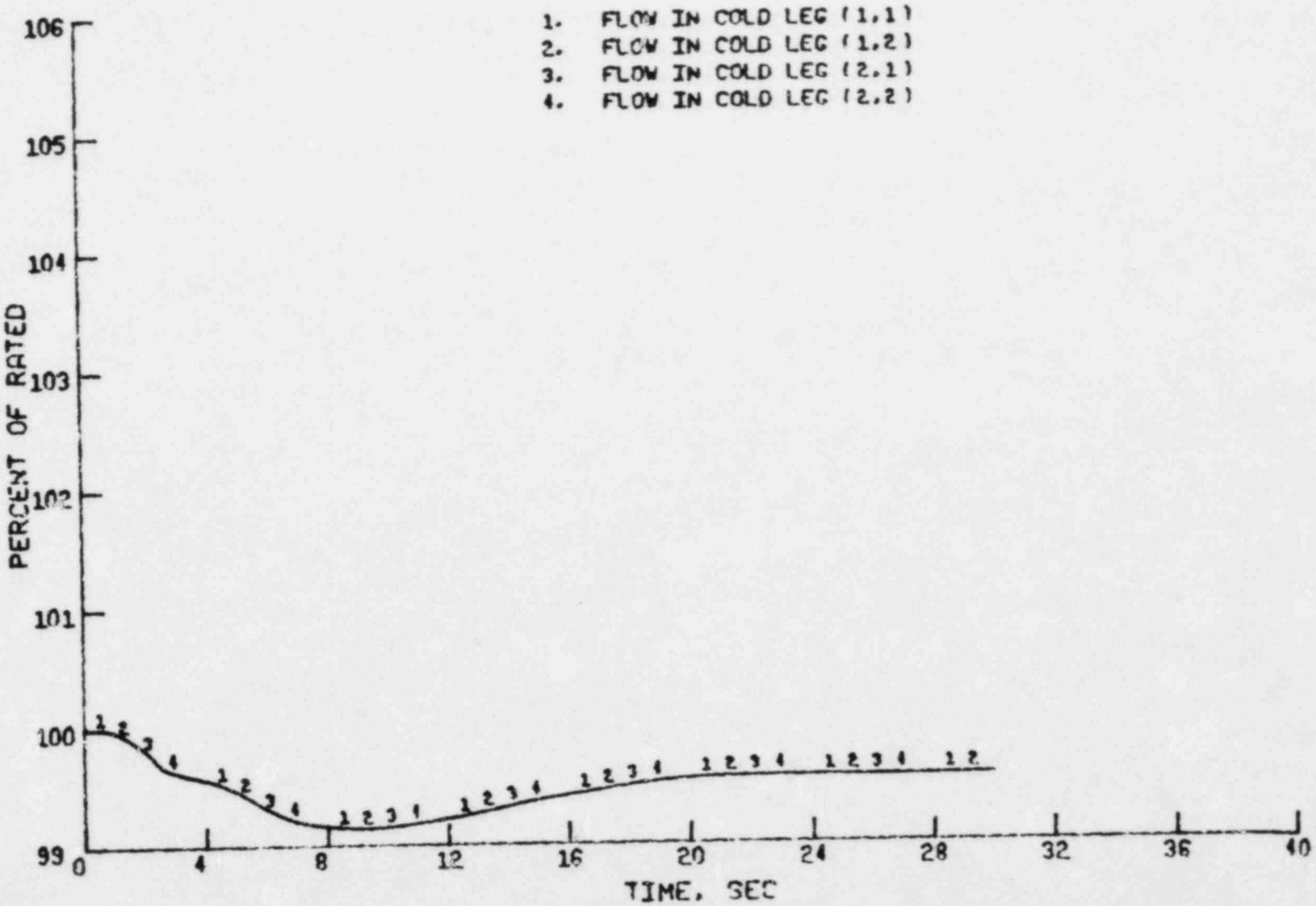

Figure 3.7 St. Lucie Unit 1 - Cold Leg Flows - Loss Of Electric Load

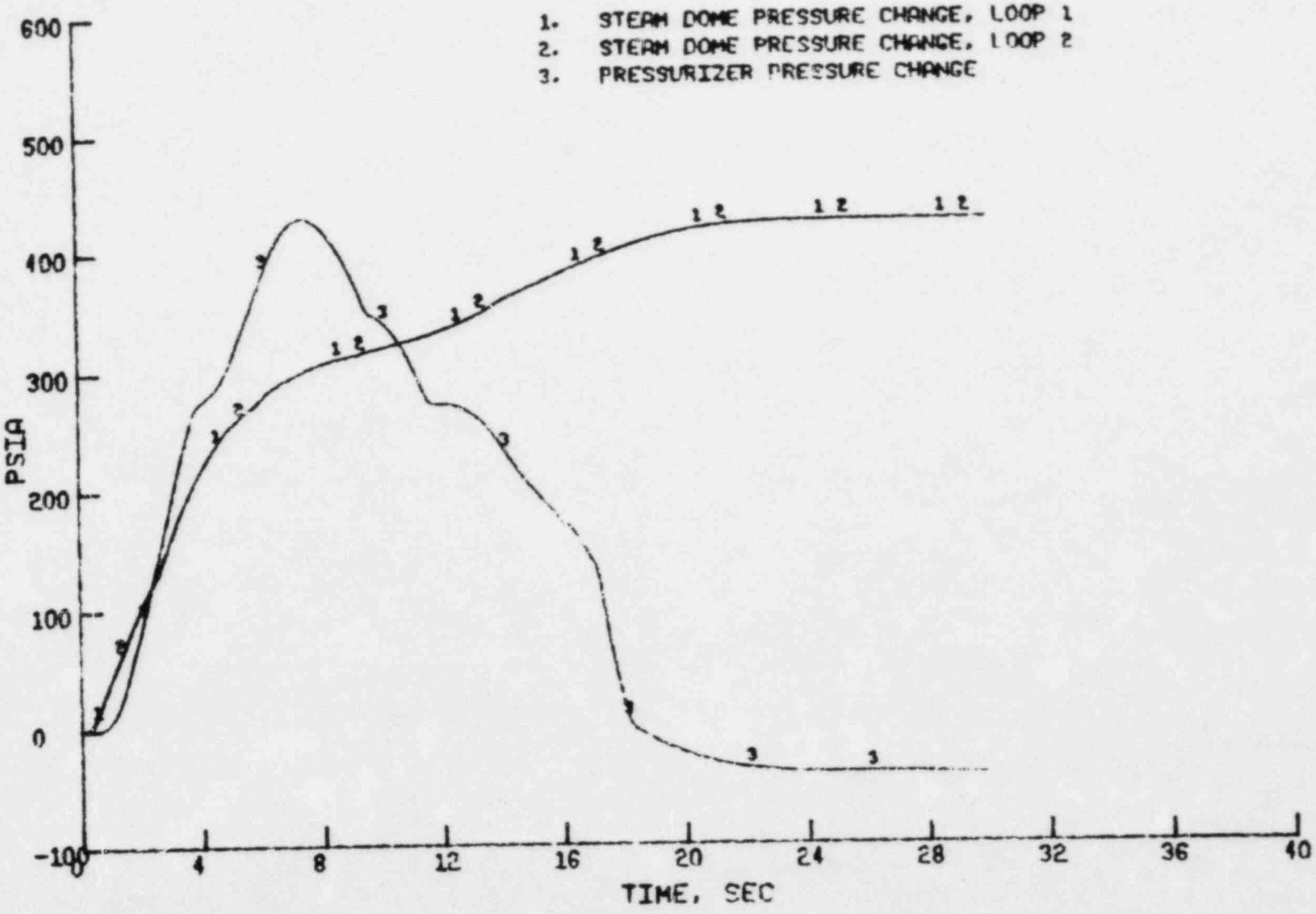

Figure 3.8 St. Lucie Unit 1 - Pressures - Loss Of Electric Load

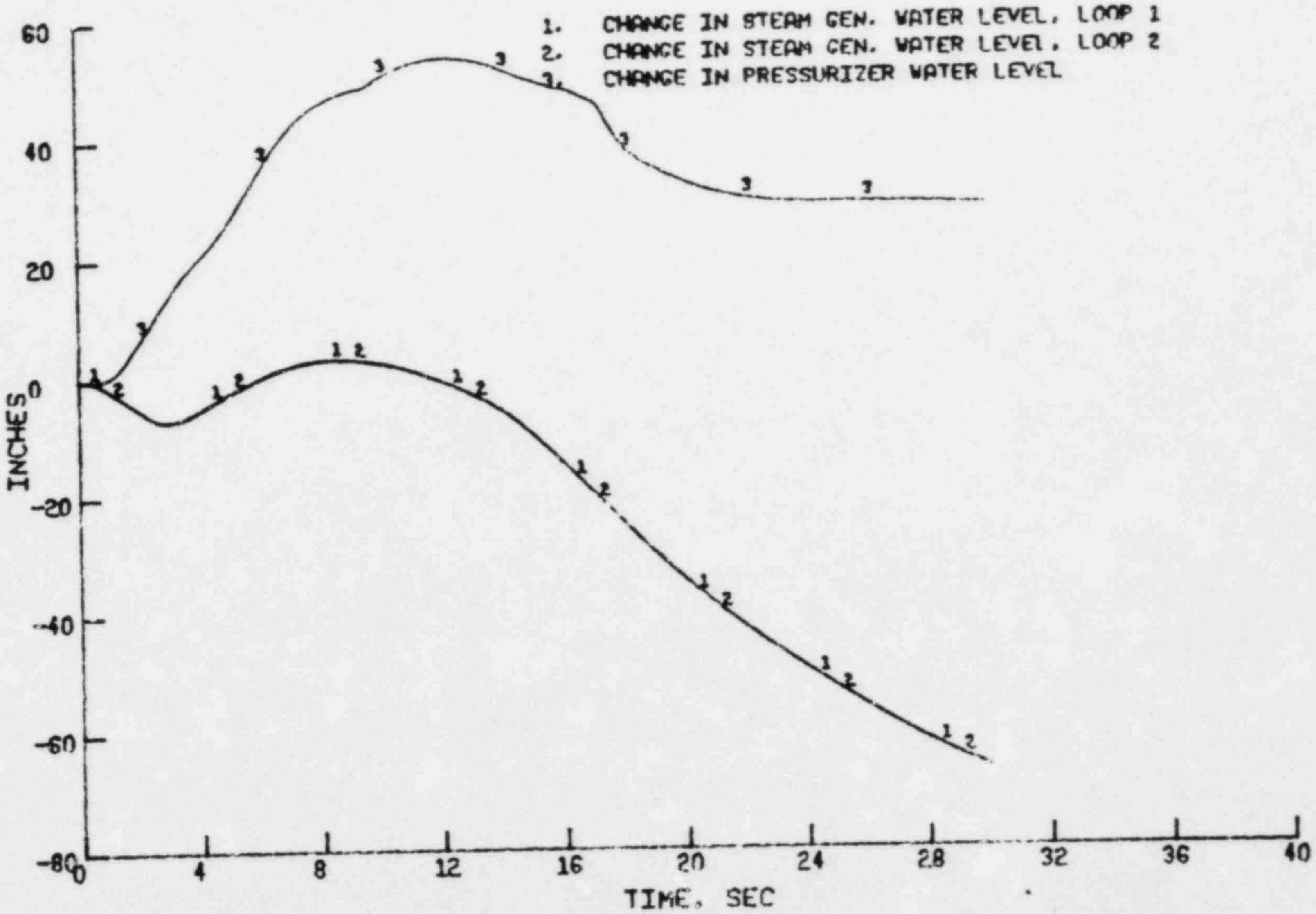

Figure 3.9 St. Lucie Unit 1 - Water Levels - Loss Of Electric Load

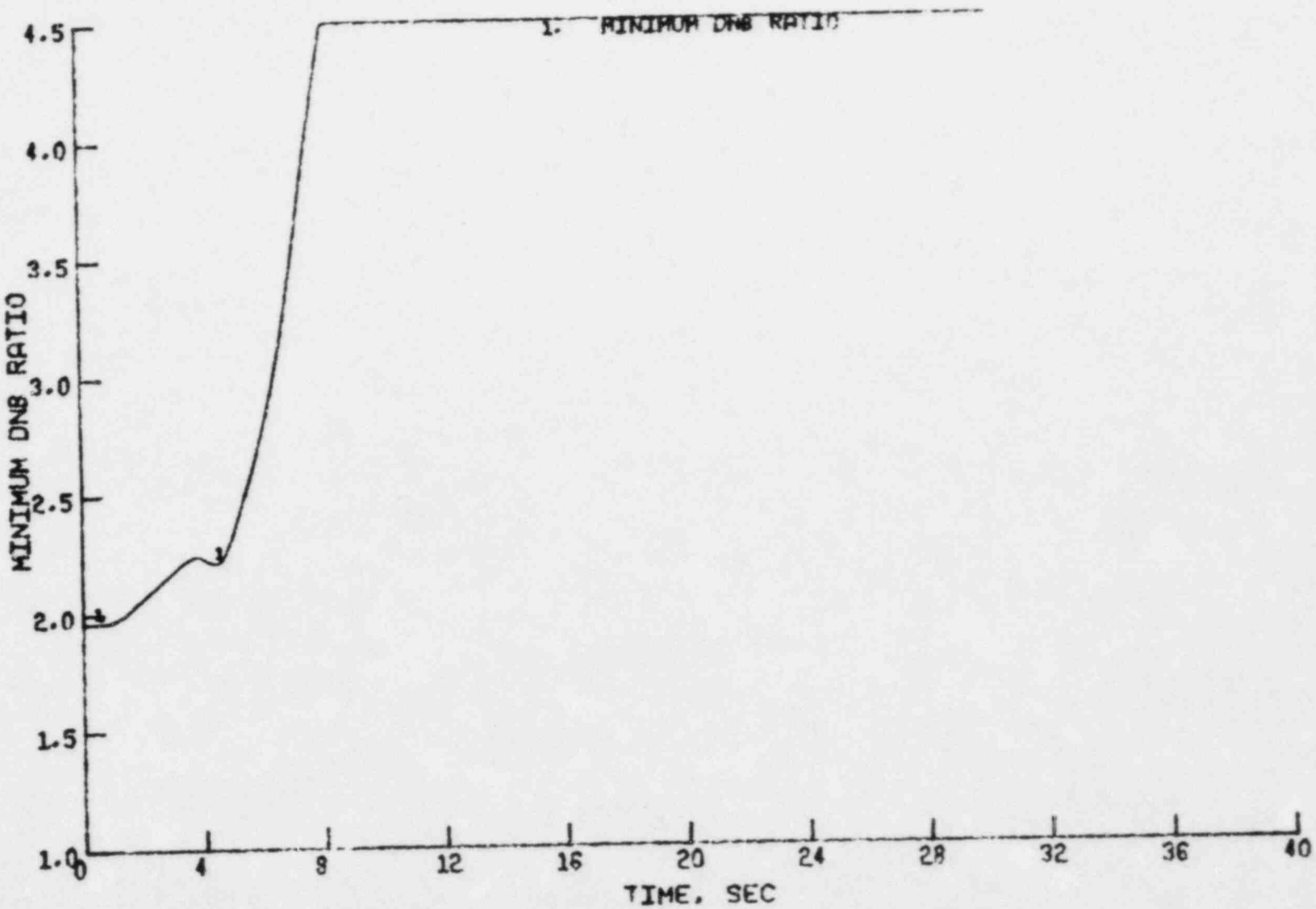

Figure 3.10 St. Lucie Unit 1 - DNBR - Loss Of Electric Load

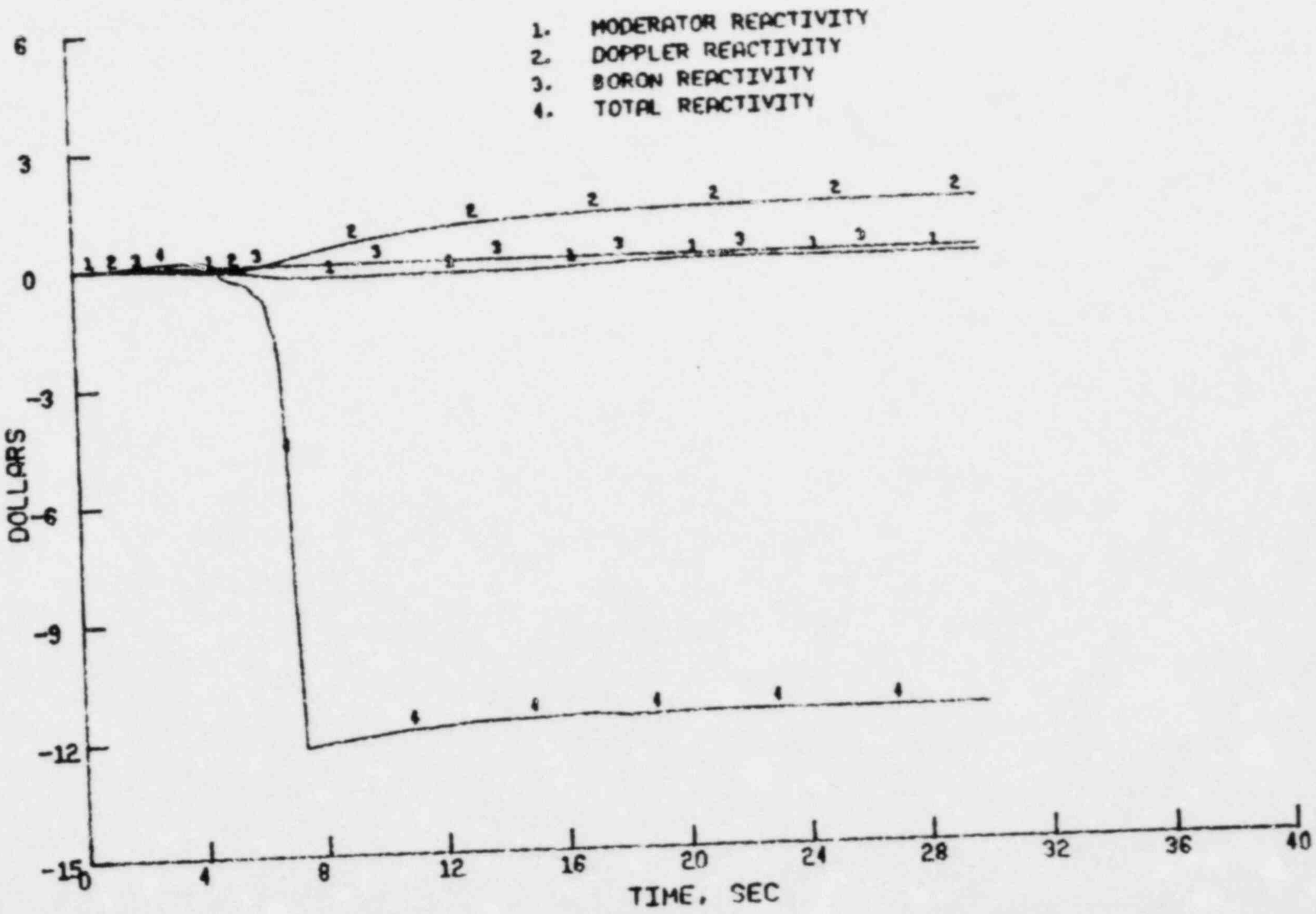

Figure 3.11 St. Lucie Unit 1 - Reactivity - Loss Of Electric Load

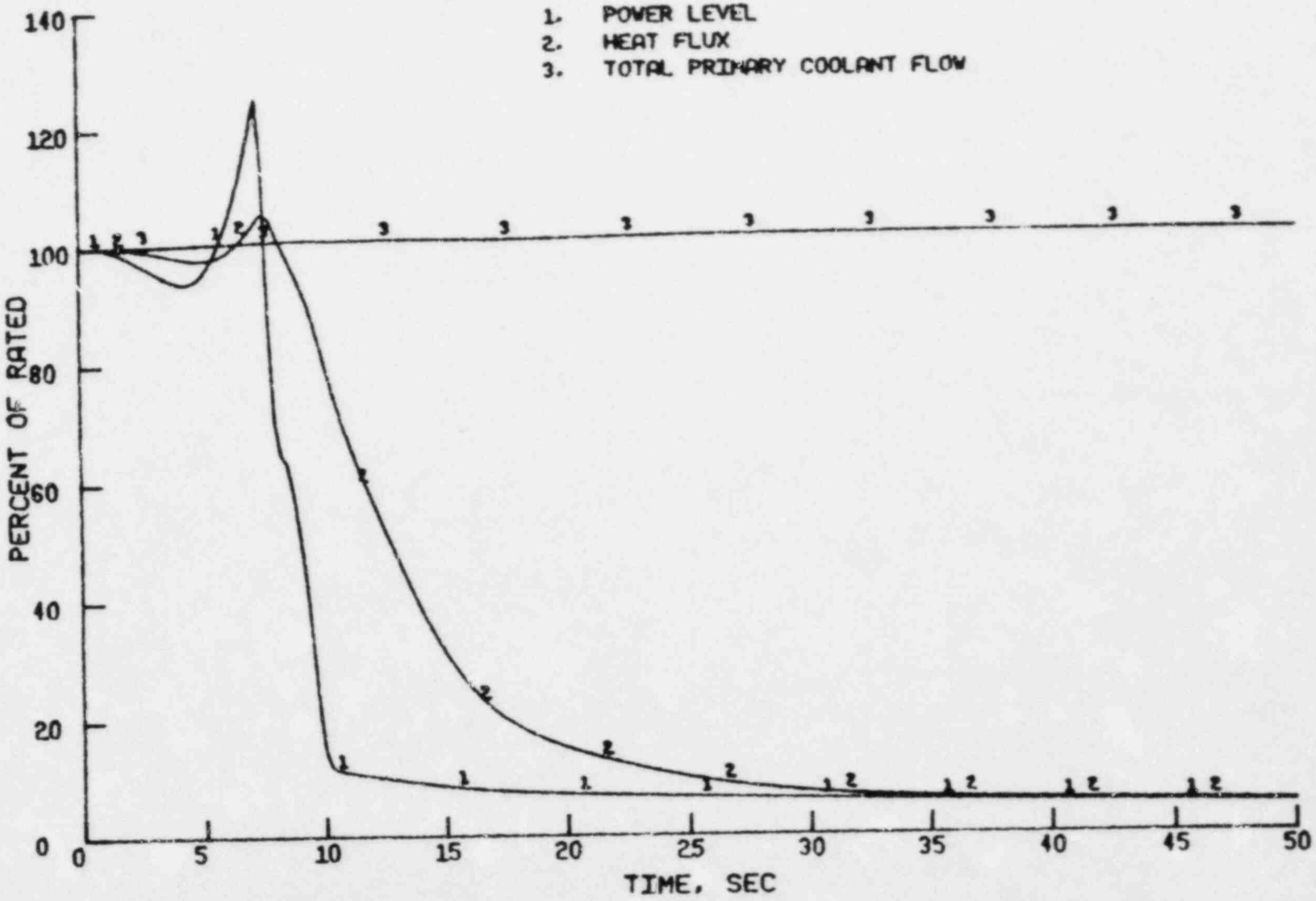

Figure 3.12 St. Lucie 1 - Power, Heat Flux and Flow - Excess Load

STLUCJ7 14/12/82 20.35.50.

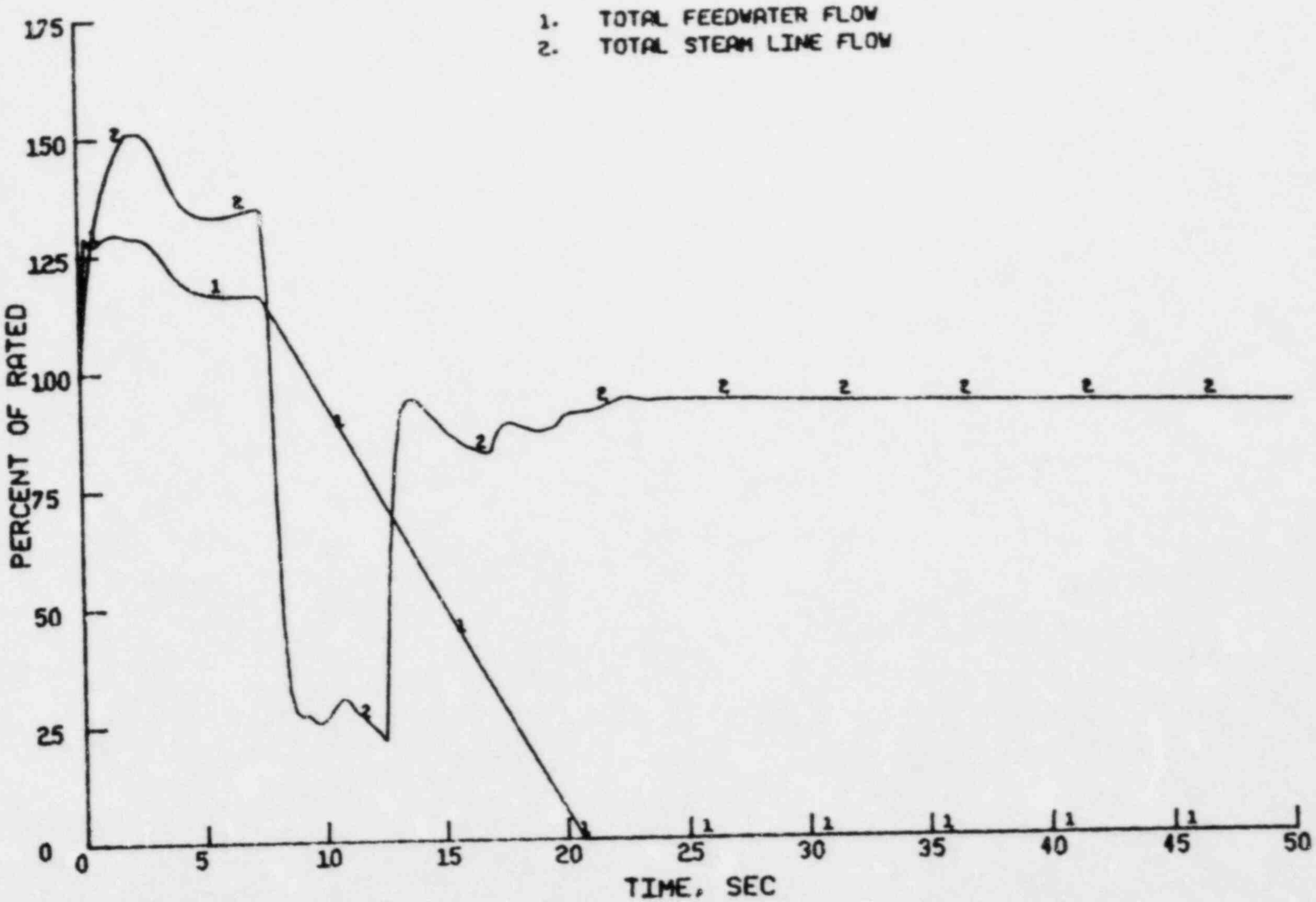

Figure 3.13 St. Lucie 1 - S. G. Flows - Excess Load

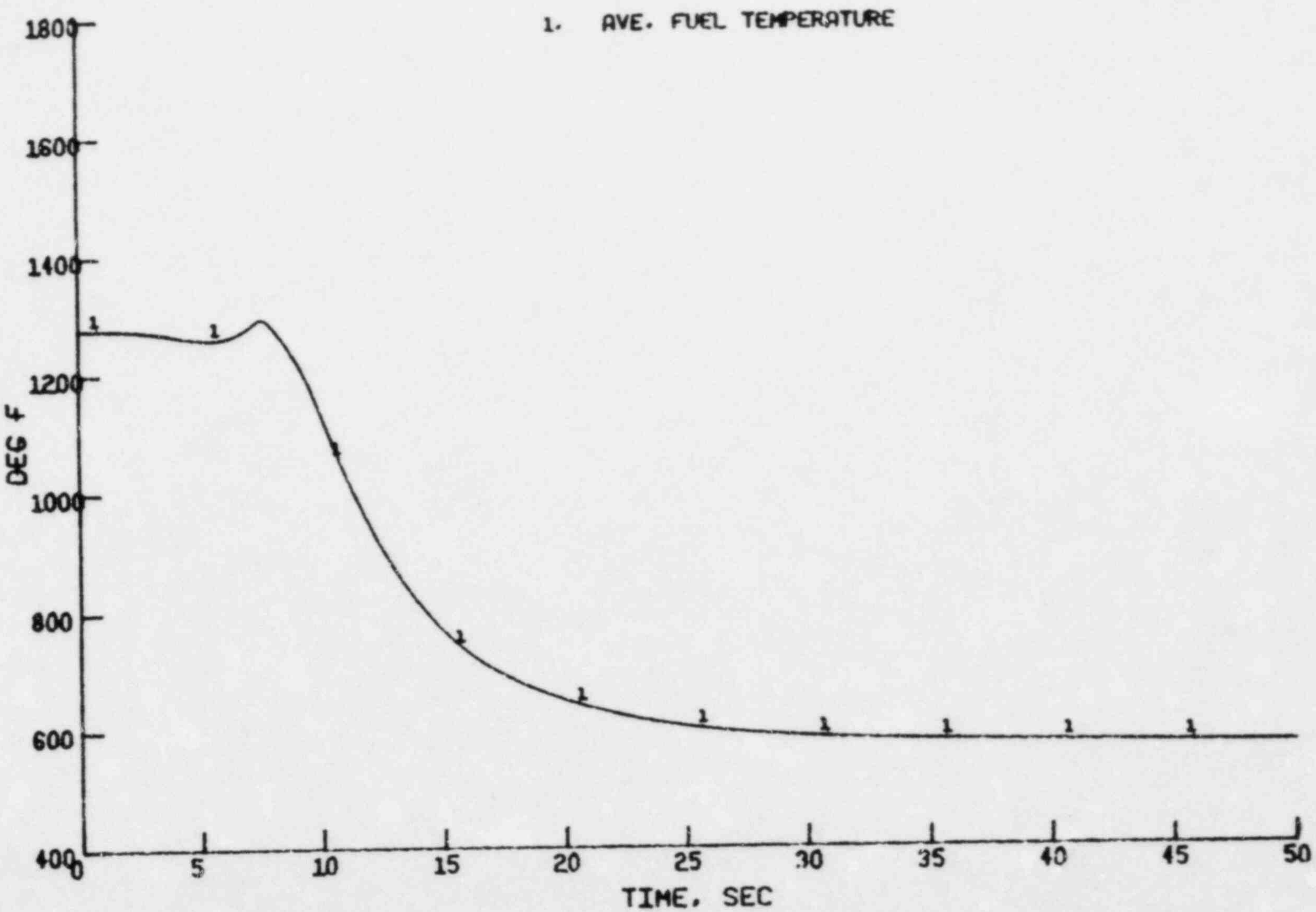

Figure 3.14 St. Lucie 1 - Fuel Temperature - Excess Load

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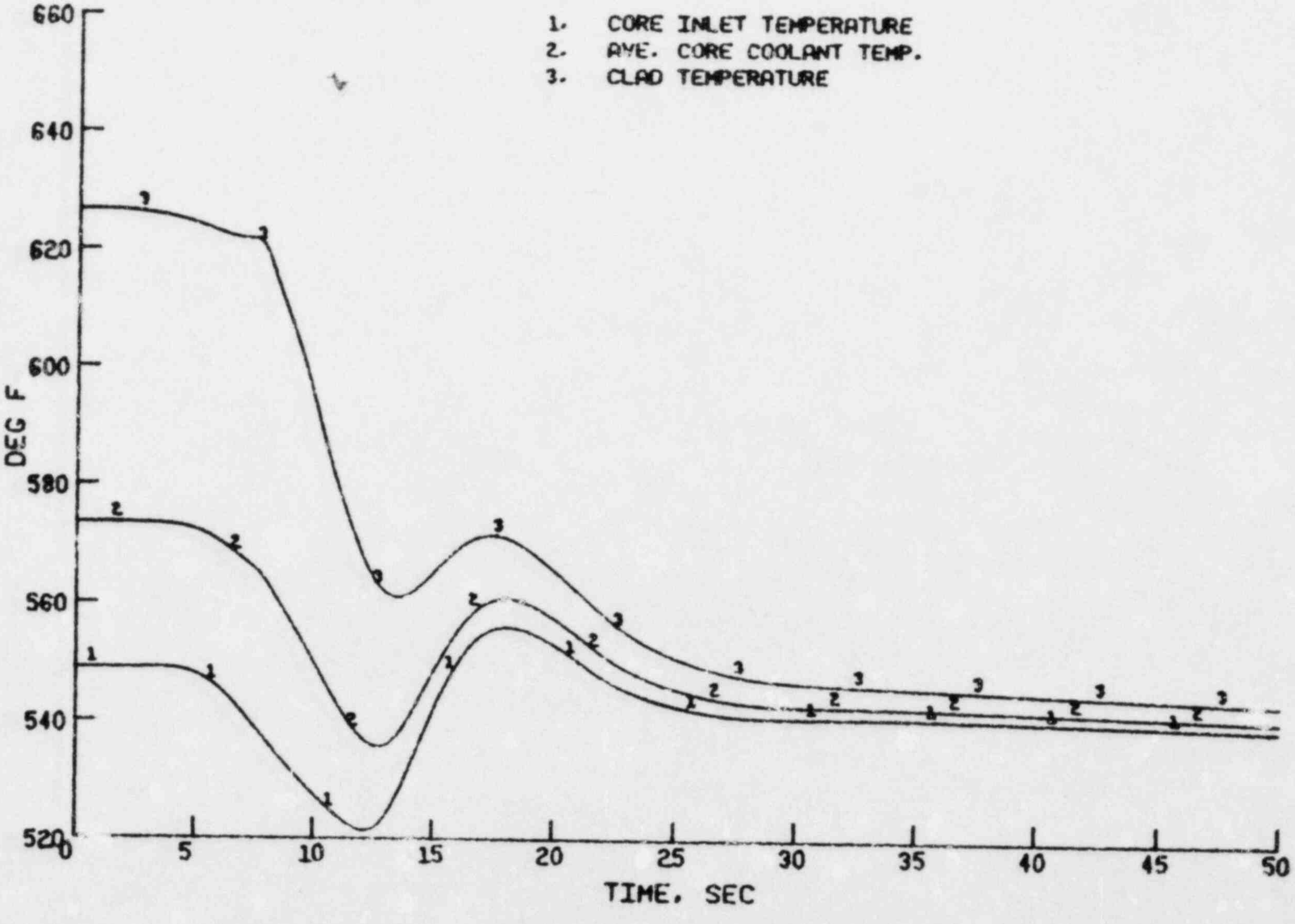

Figure 3.15 St. Lucie 1 - Loss Of Coolant Flow - Excess Load

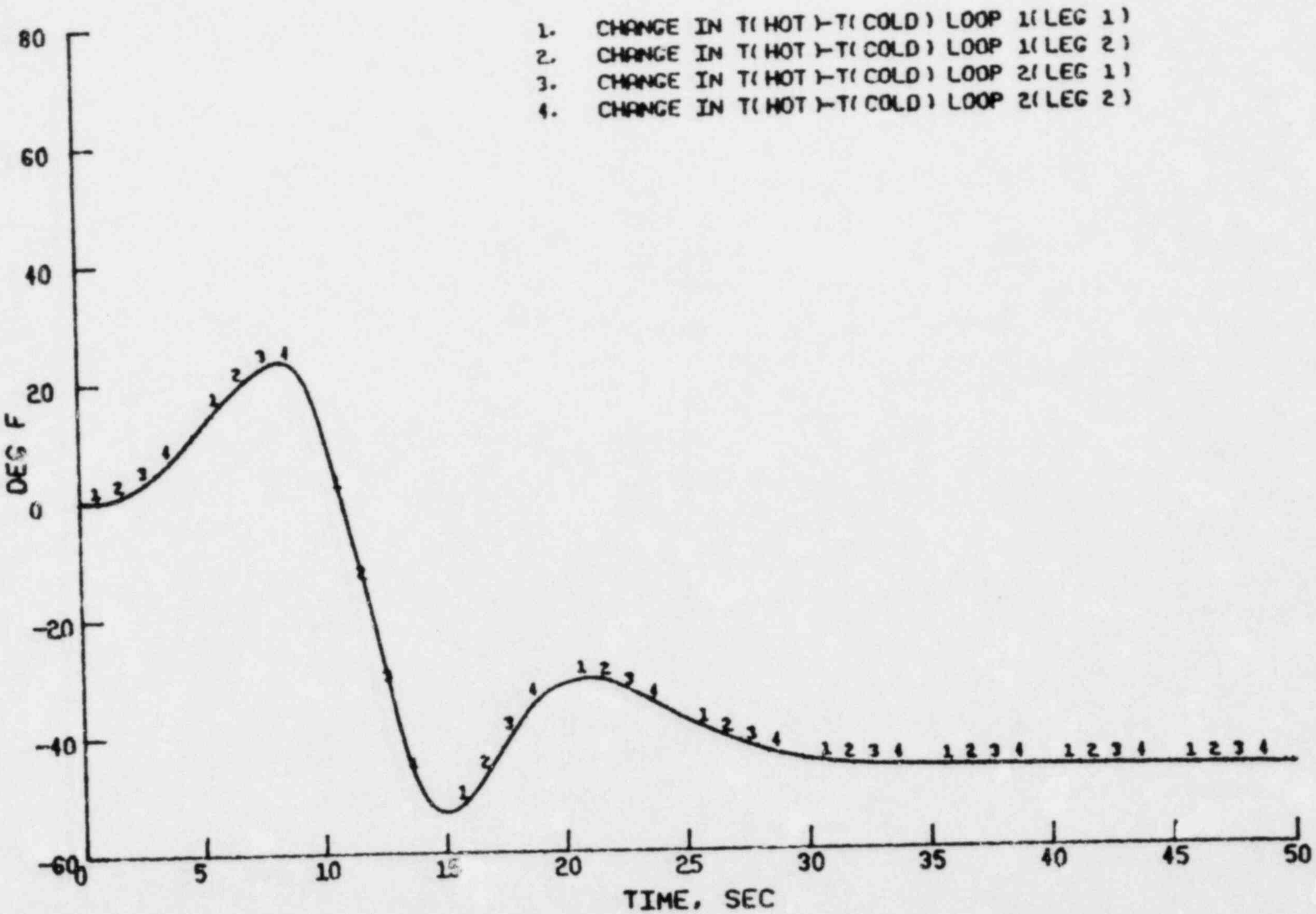

Figure 3.16 St. Lucie 1 - Loop Temperature Differences - Excess Load

STLUCJ2 14/12/82 20.35.50.

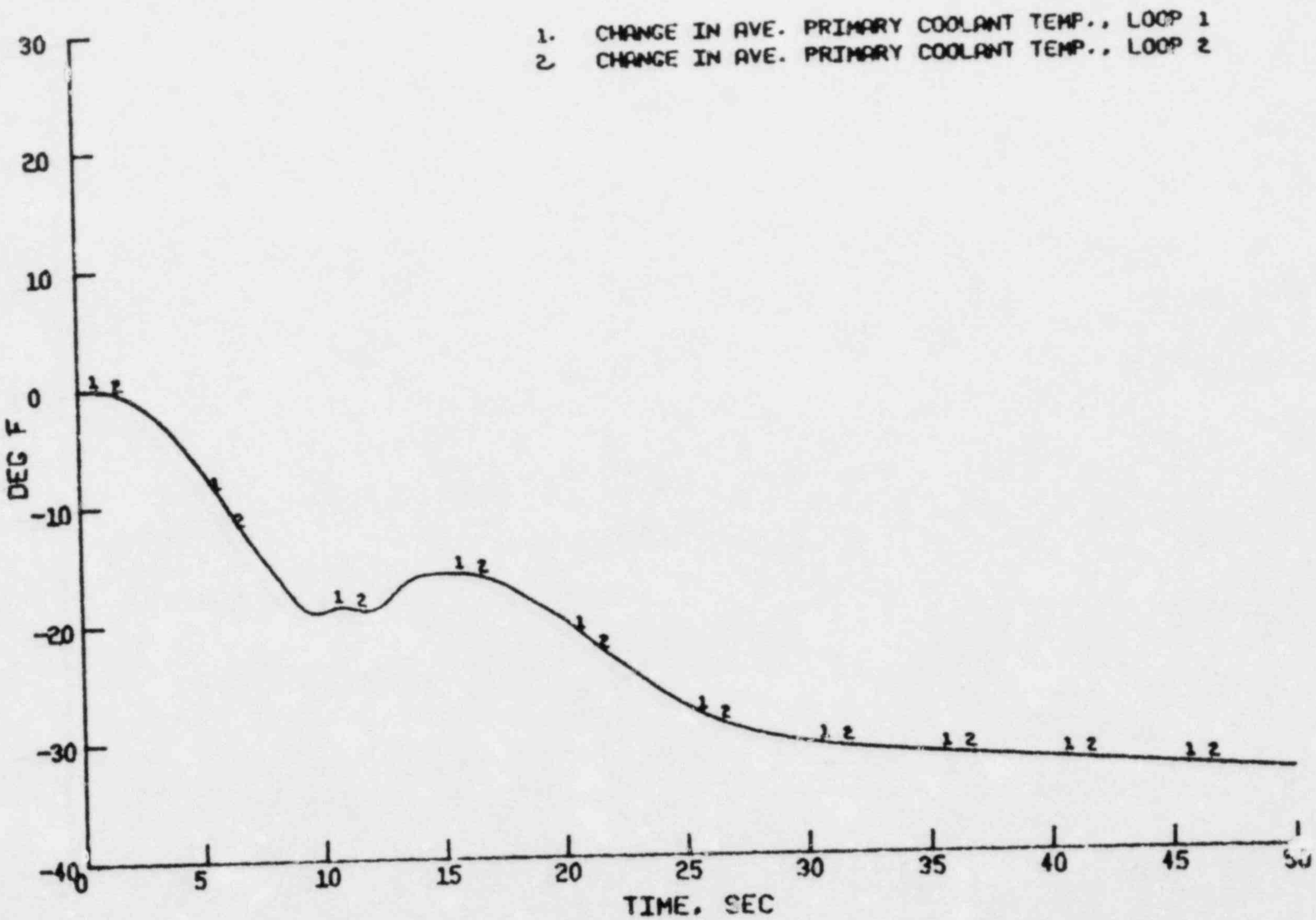

Figure 3.17 St. Lucie 1 - Average Temperatures - Excess Load

STLUCJZ 14/12/82 20.35.50.

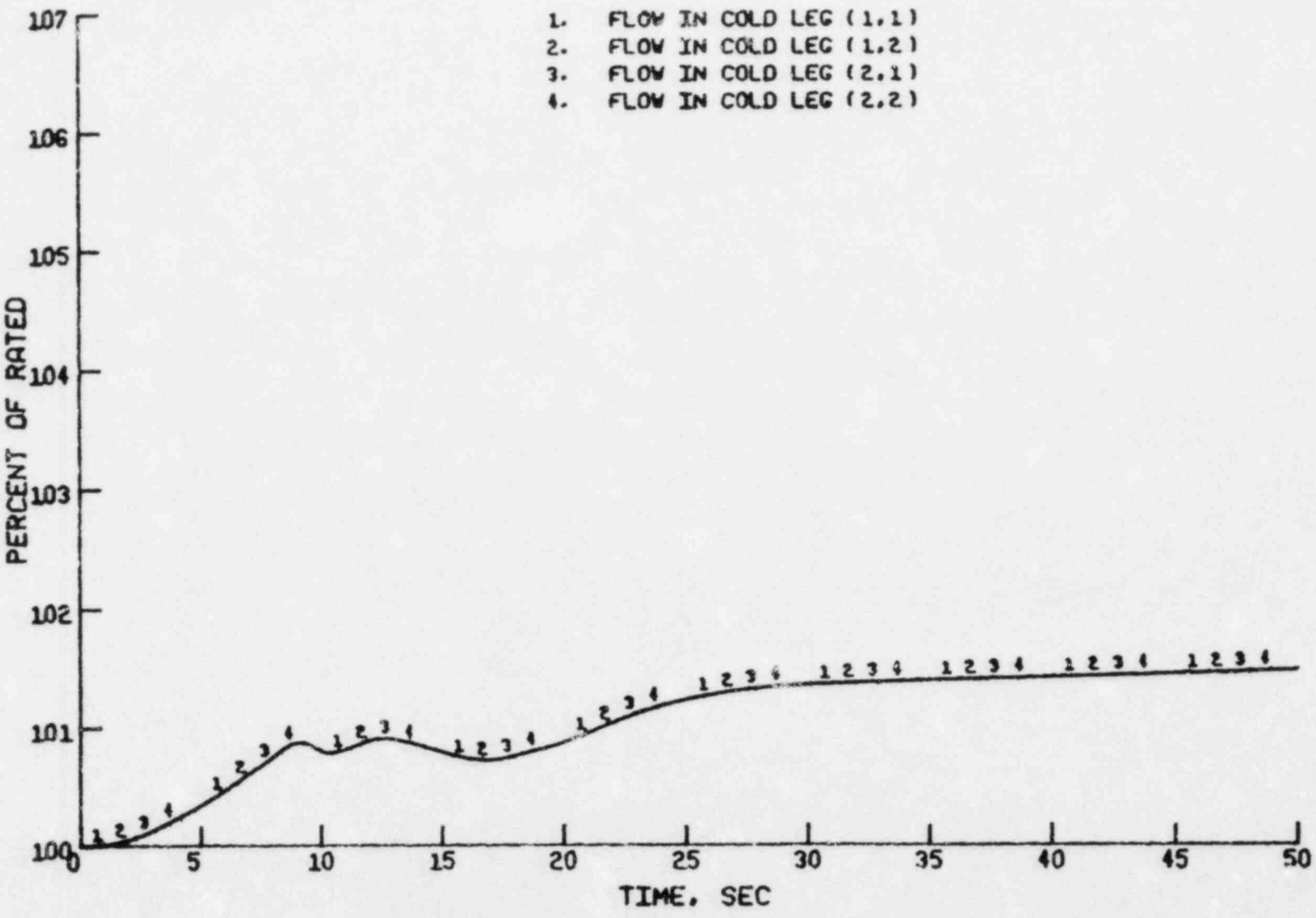

Figure 3.18 St. Lucie Unit 1 - Cold Leg Flows - Excess Load

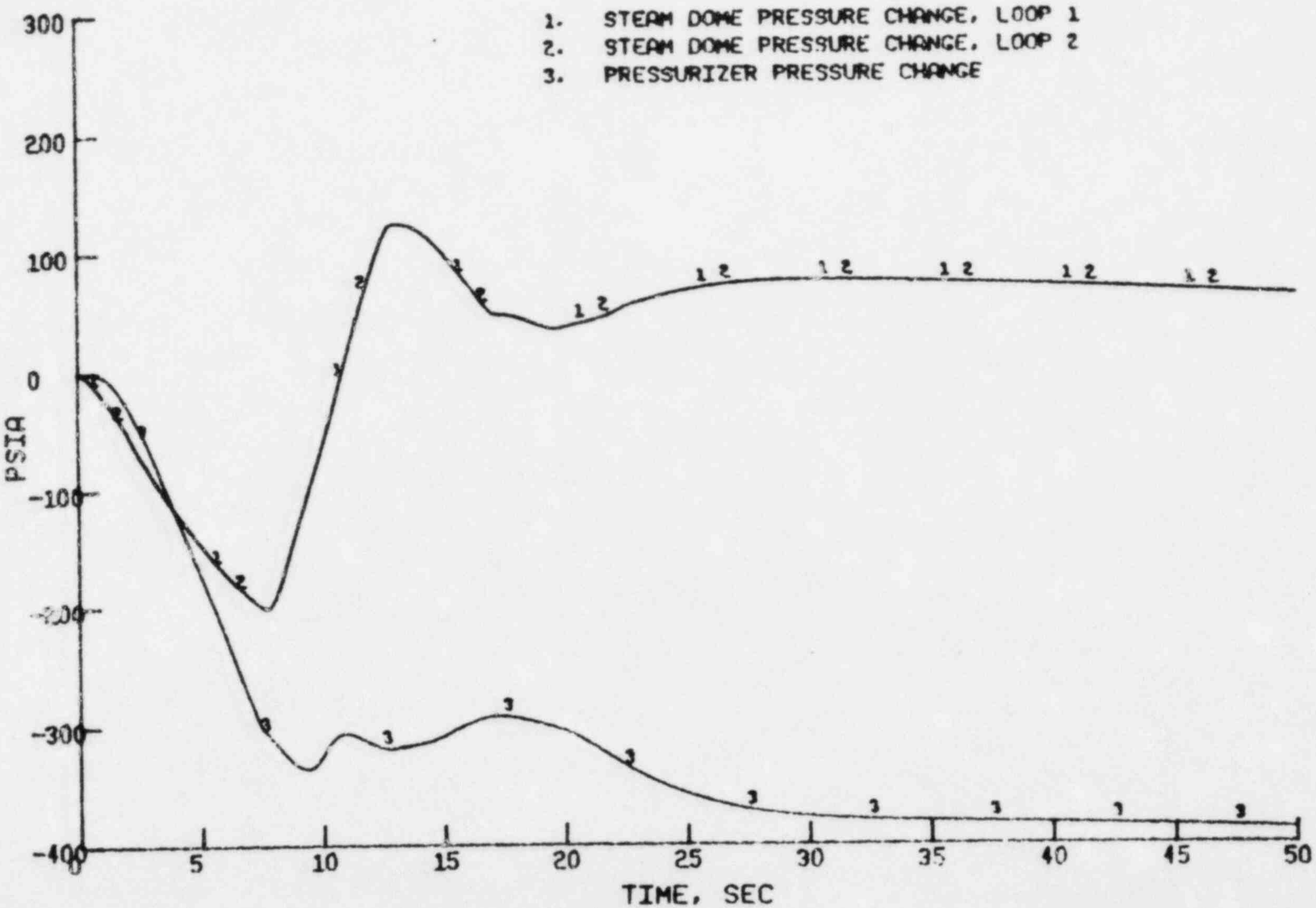

Figure 3.19 St. Lucie Unit 1 - Pressures - Excess Load

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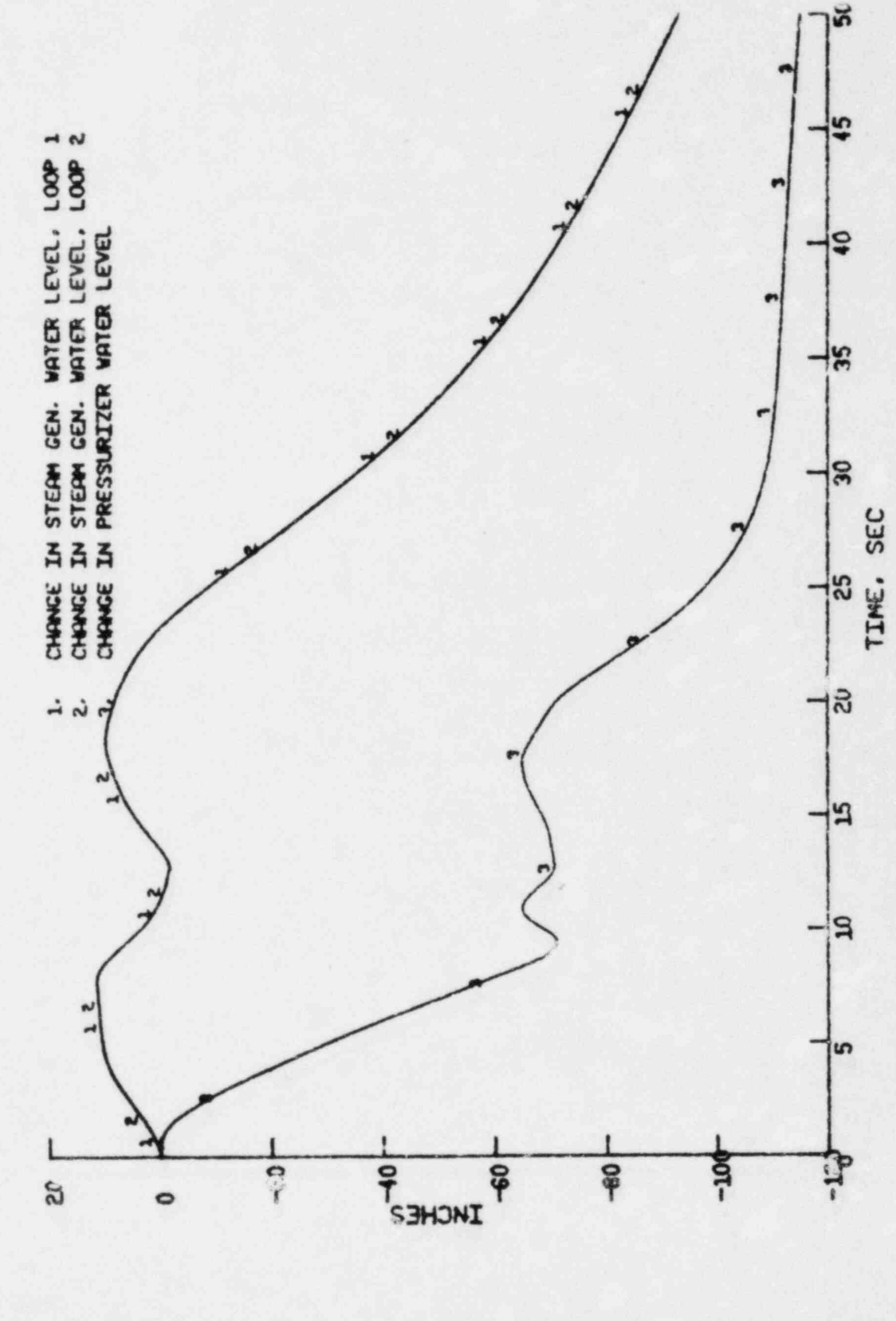

Figure 3.20 St. Lucie Unit 1 - Water Levels - Excess Load

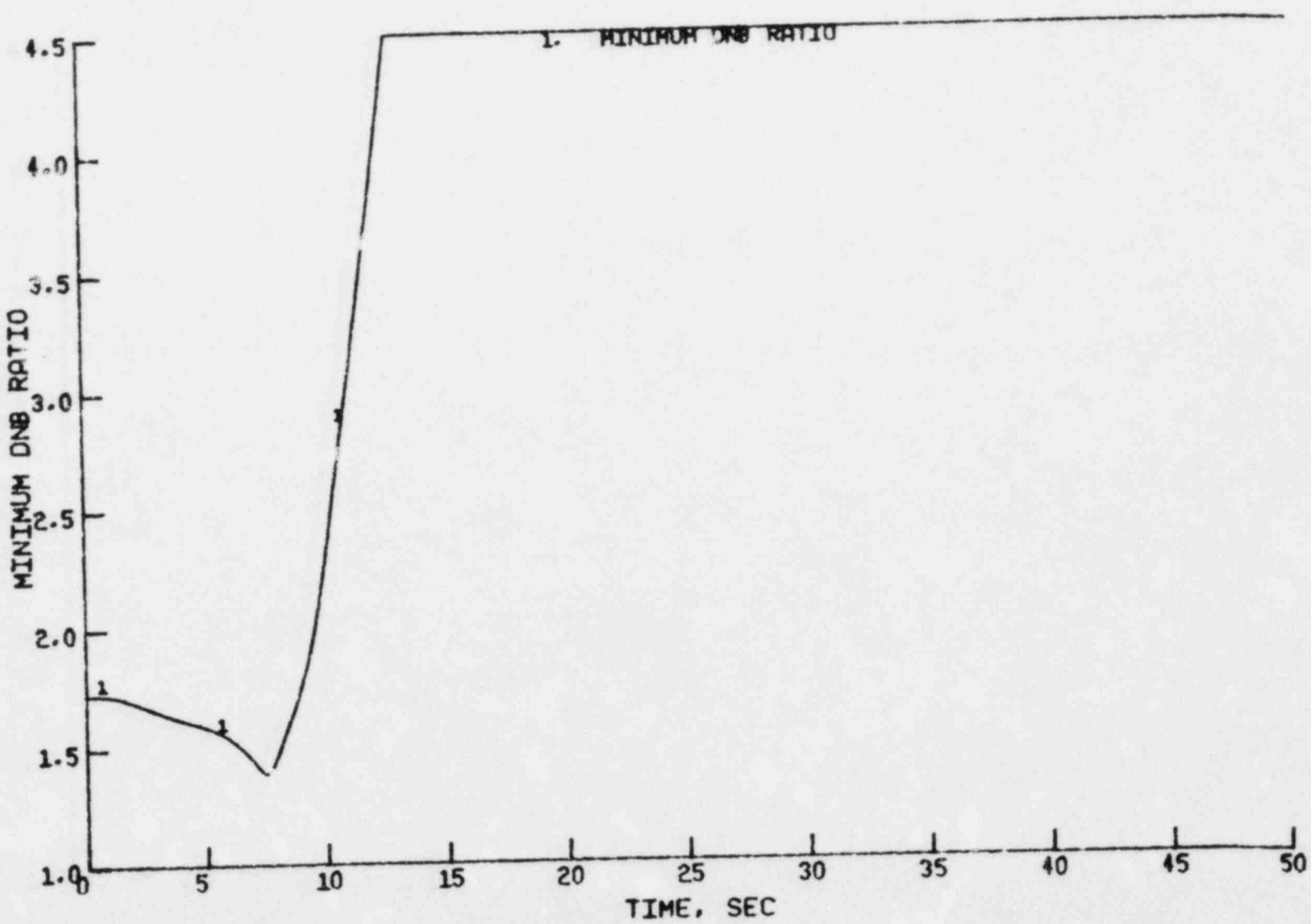

Figure 3.21 St. Lucie Unit 1 - DNBR - Excess Load

STLUCJ2 14/12/82 20.35.51

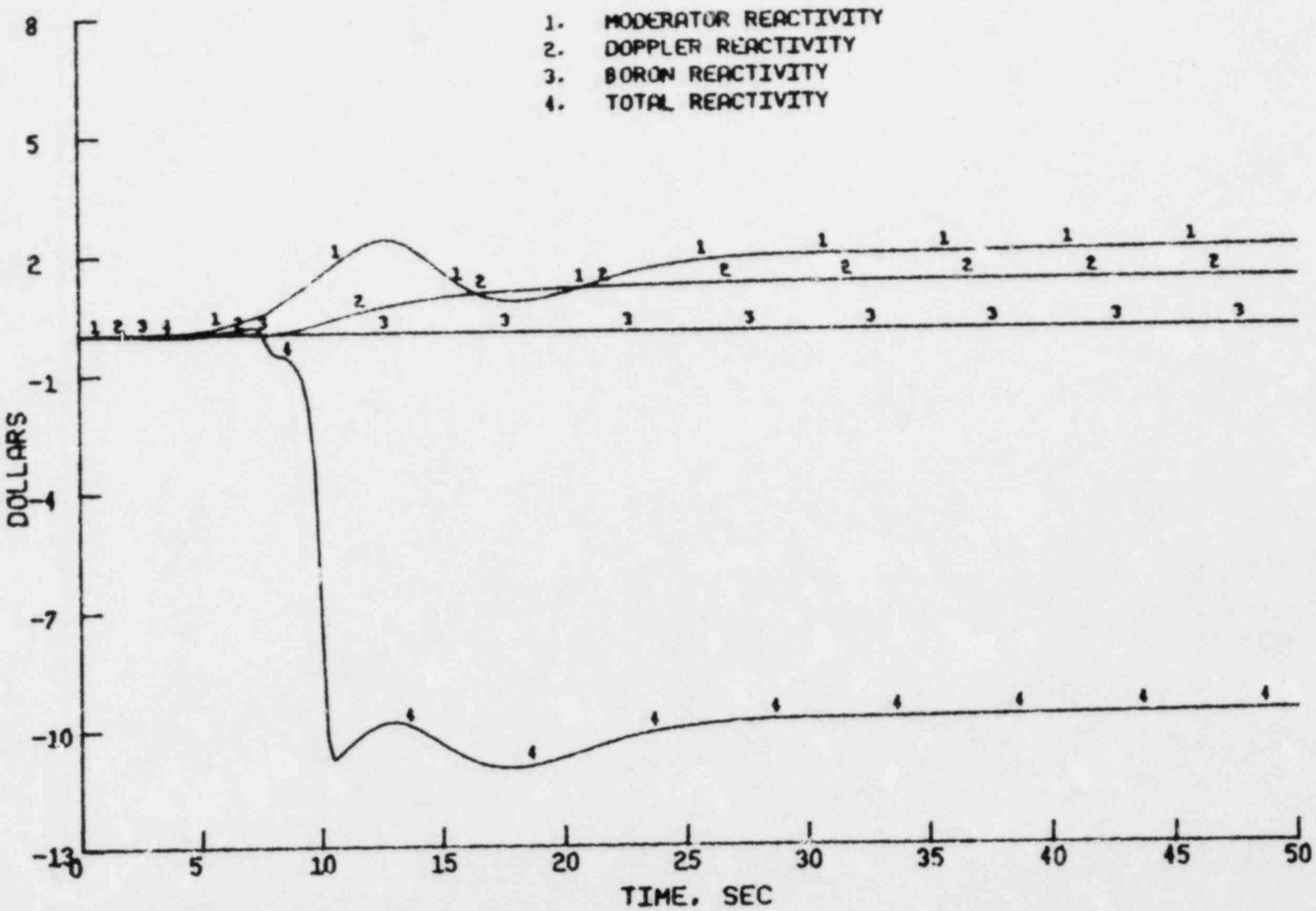

Figure 3.22 St. Lucie Unit 1 - Reactivity - Excess Load

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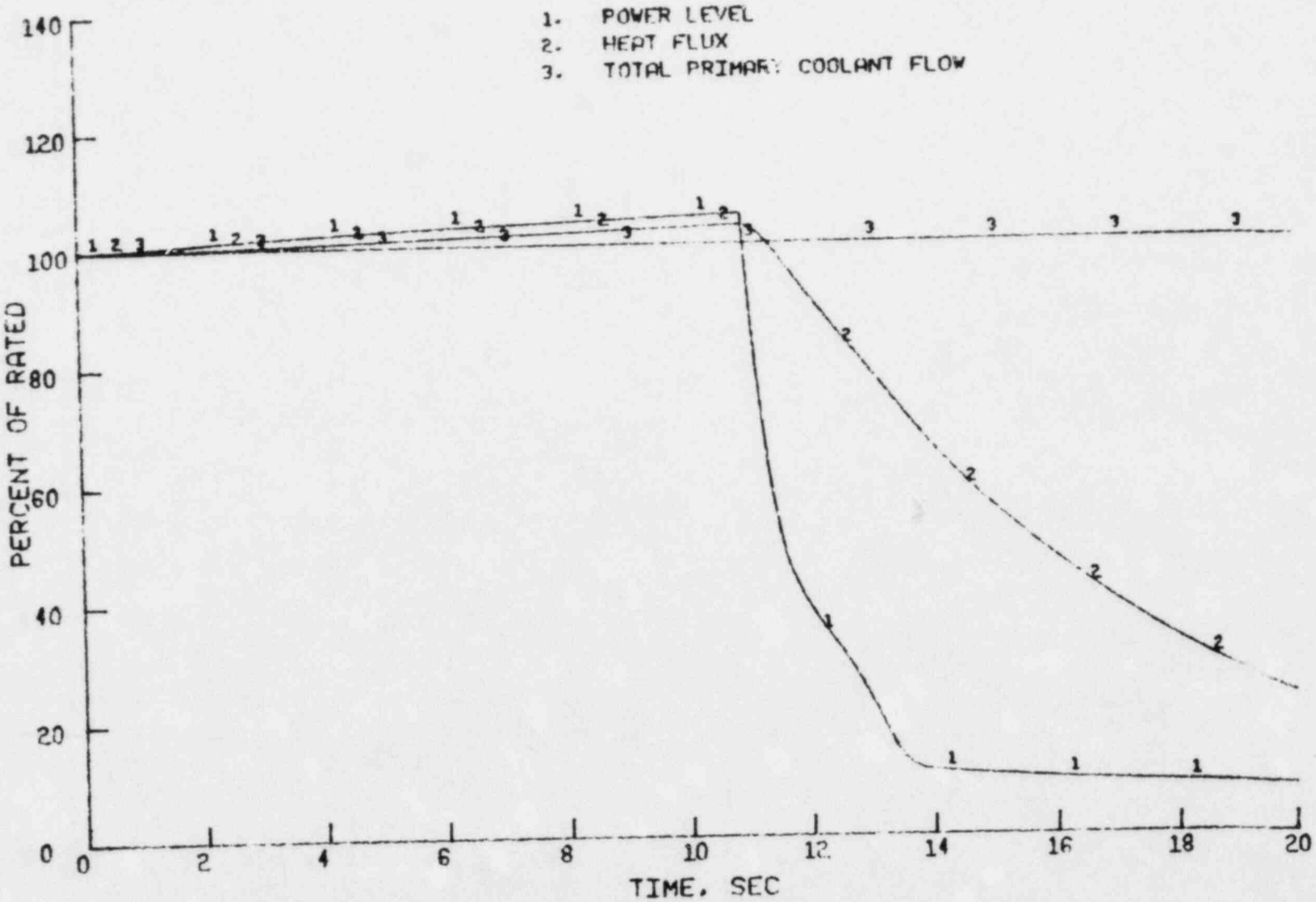

Figure 3.23 St. Lucie Unit 1 - Power, Heat Flux and Flow - RCS Depressurization

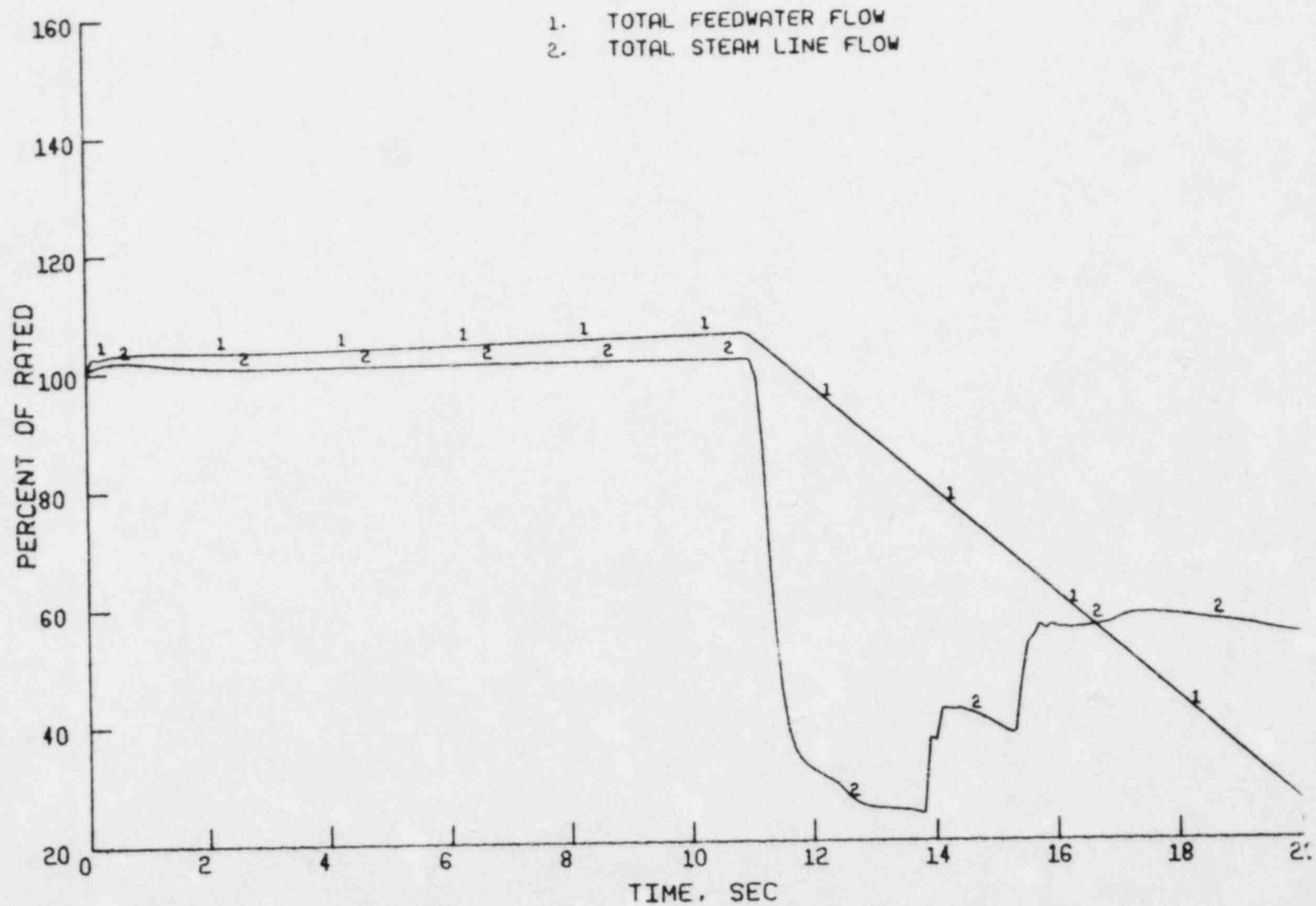

Figure 3.24 St. Lucie Unit 1 - S. G. Flows - RCS Depressurization

STLUCO8 20/12/82 20.53.30

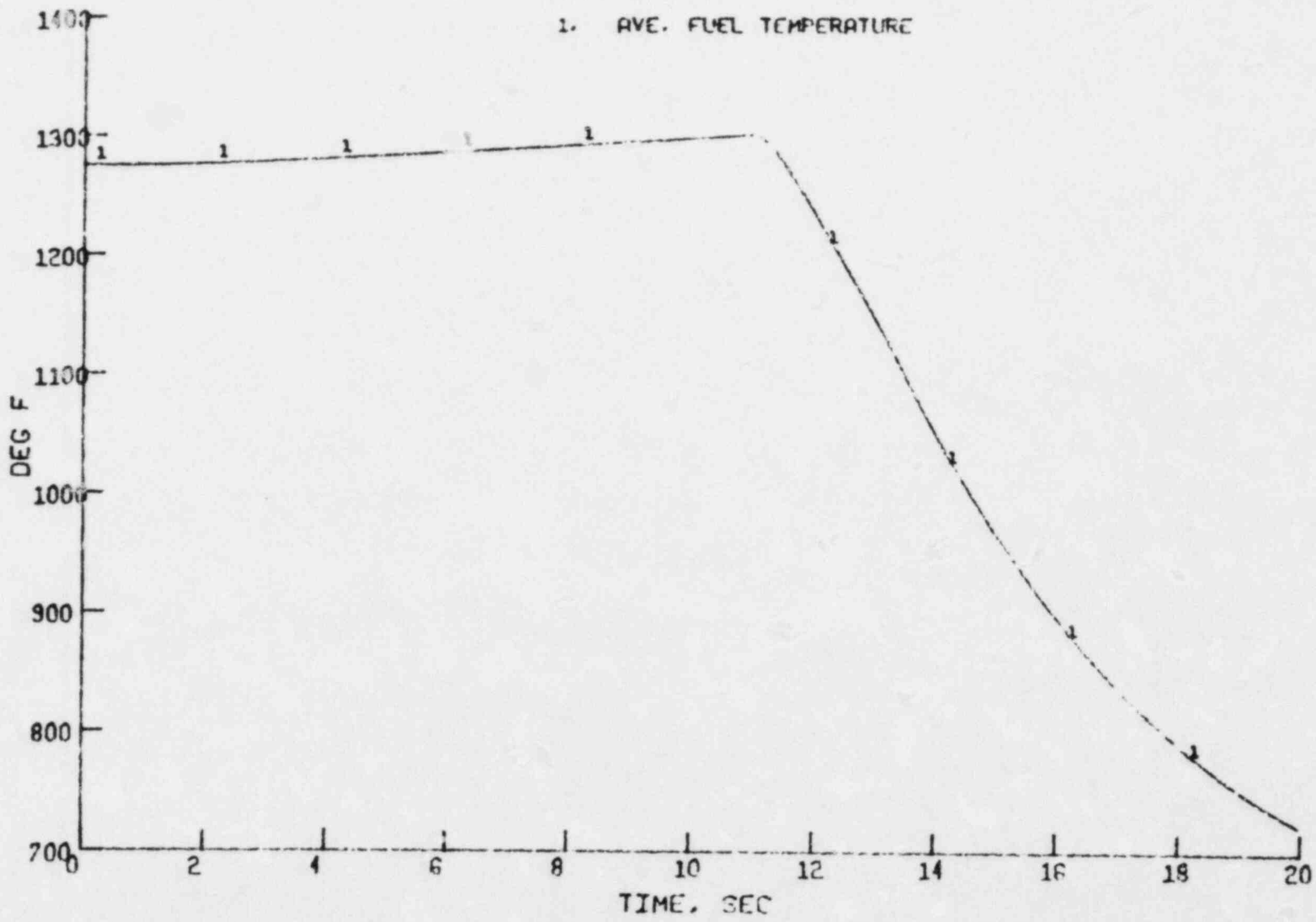

Figure 3.25 St. Lucie Unit 1 - Fuel Temperature - RCS Depressurization

STLUCO8 20/12/82 20.53.30.

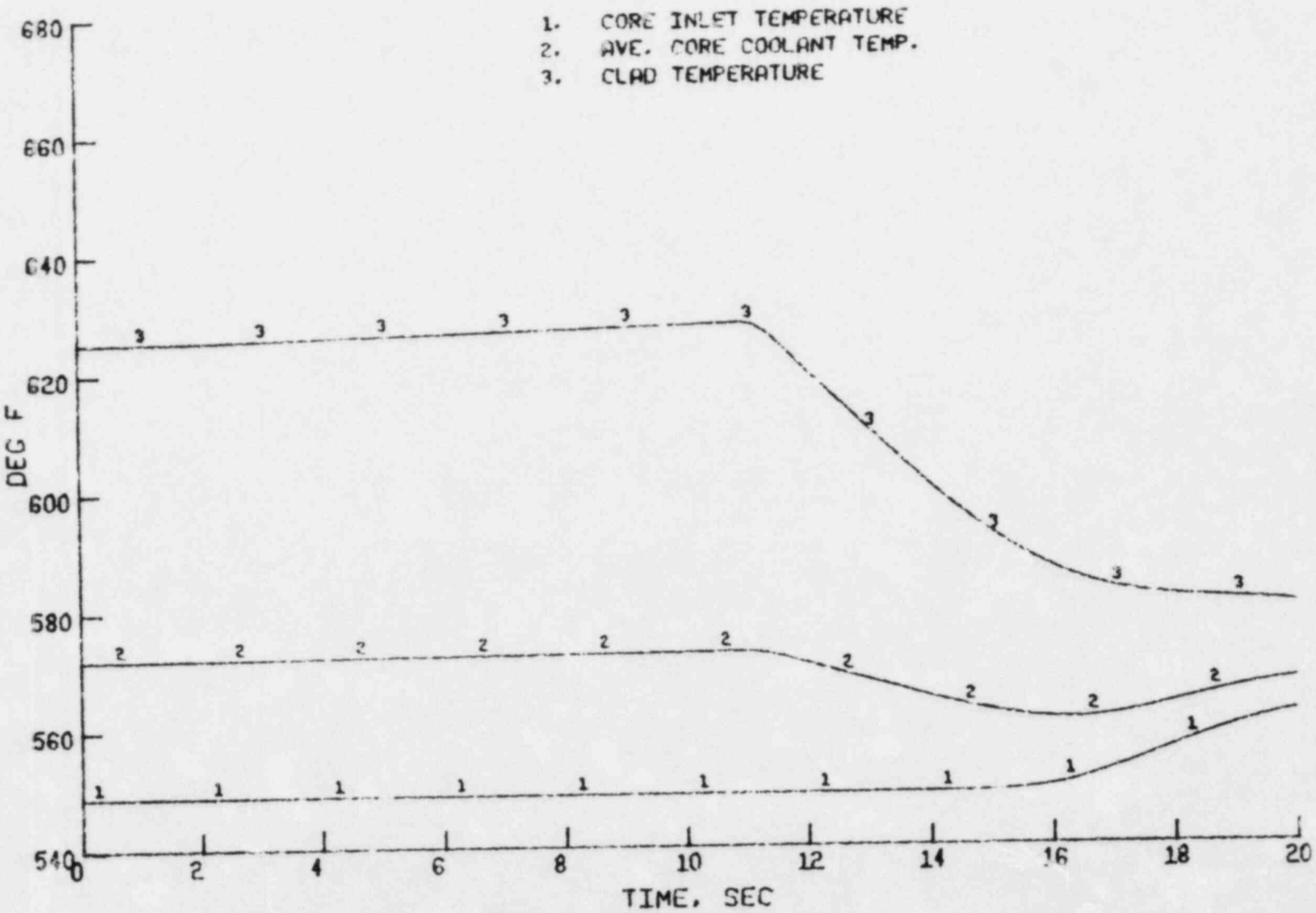

Figure 3.26 St. Lucie Unit 1 - Core Temperatures - RCS Depressurization

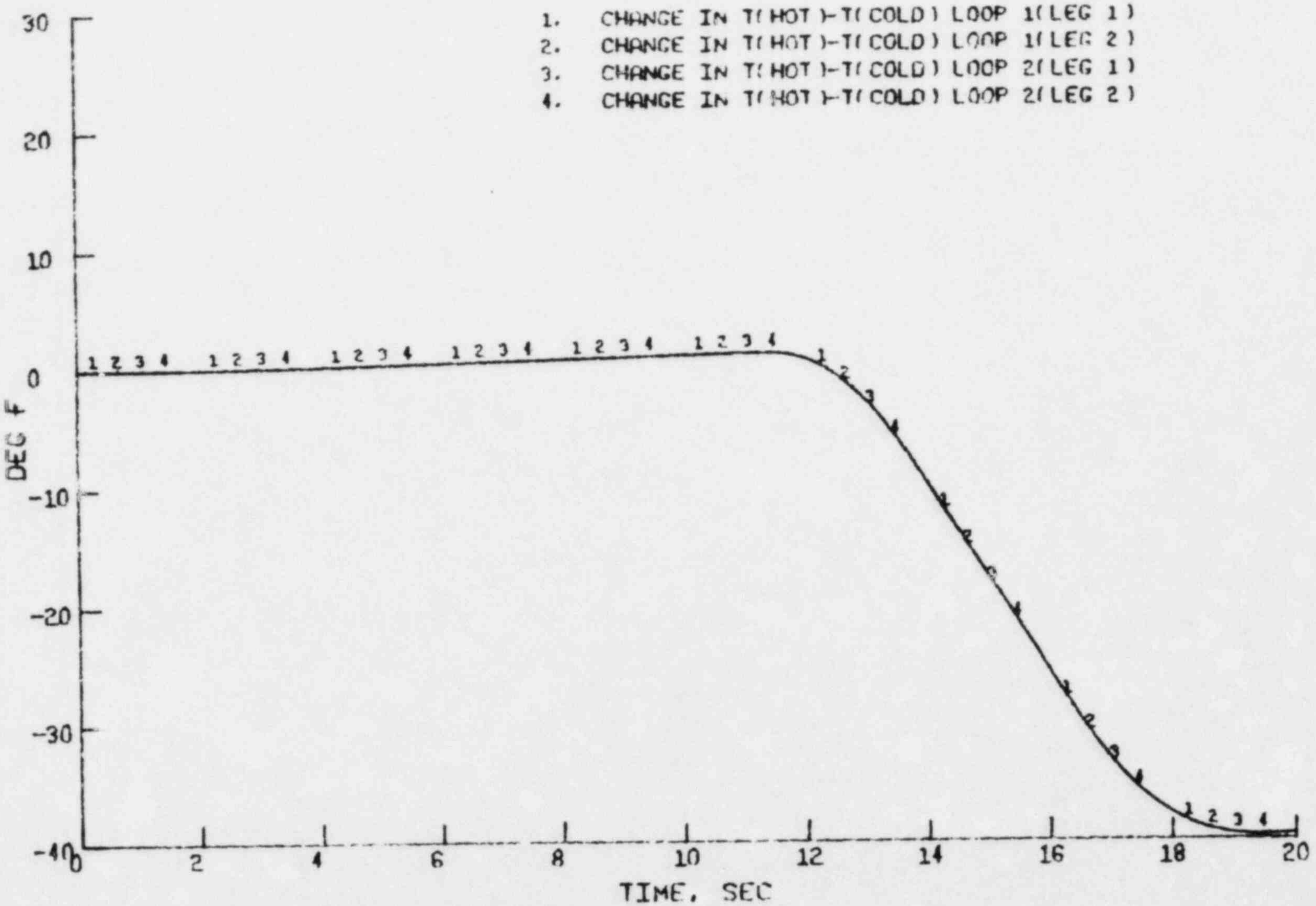

Figure 3.27 St. Lucie Unit 1 - Loop Temperature Differences - RCS Depressurization

STLUC08 20/12/82 20.53.39

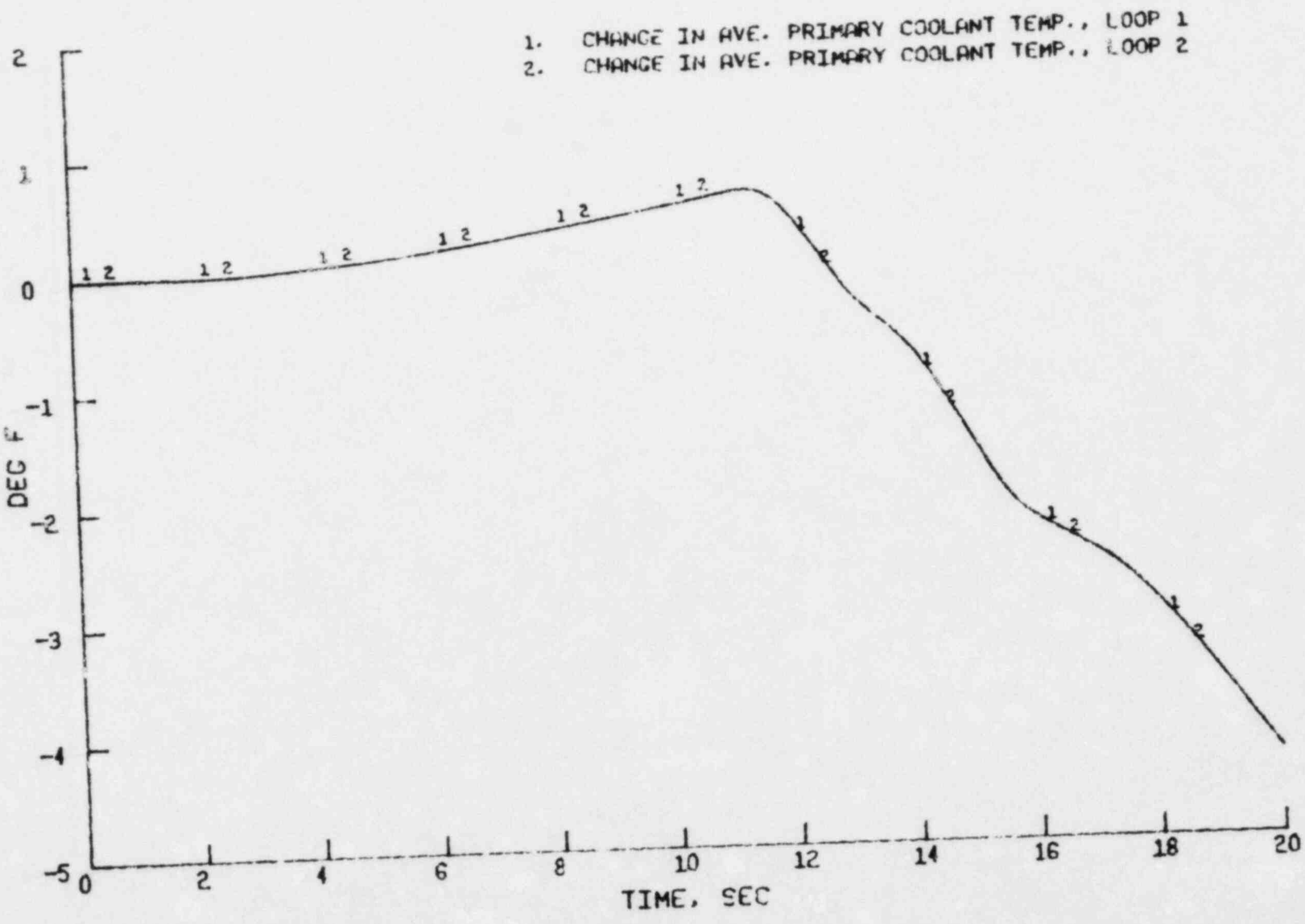

Figure 3.28 St. Lucie Unit 1 - Average Temperatures - RCS Depressurization

STLUCO8 20/12/82 20.53.30.

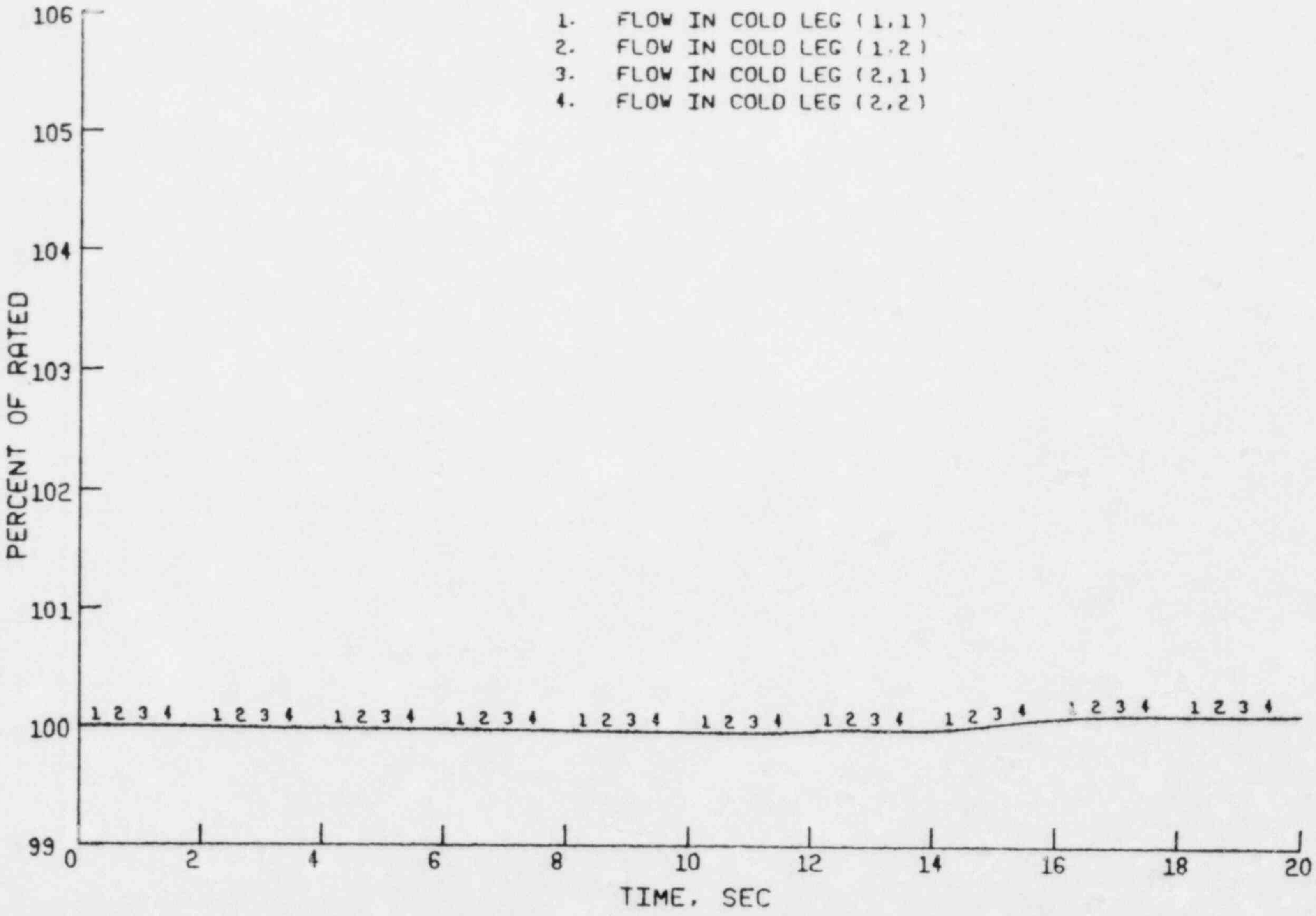

Figure 3.29 St. Lucie Unit 1 - Cold Leg Flows - RCS Depressurization

STLUCO8 20/12/82 20.53.30.

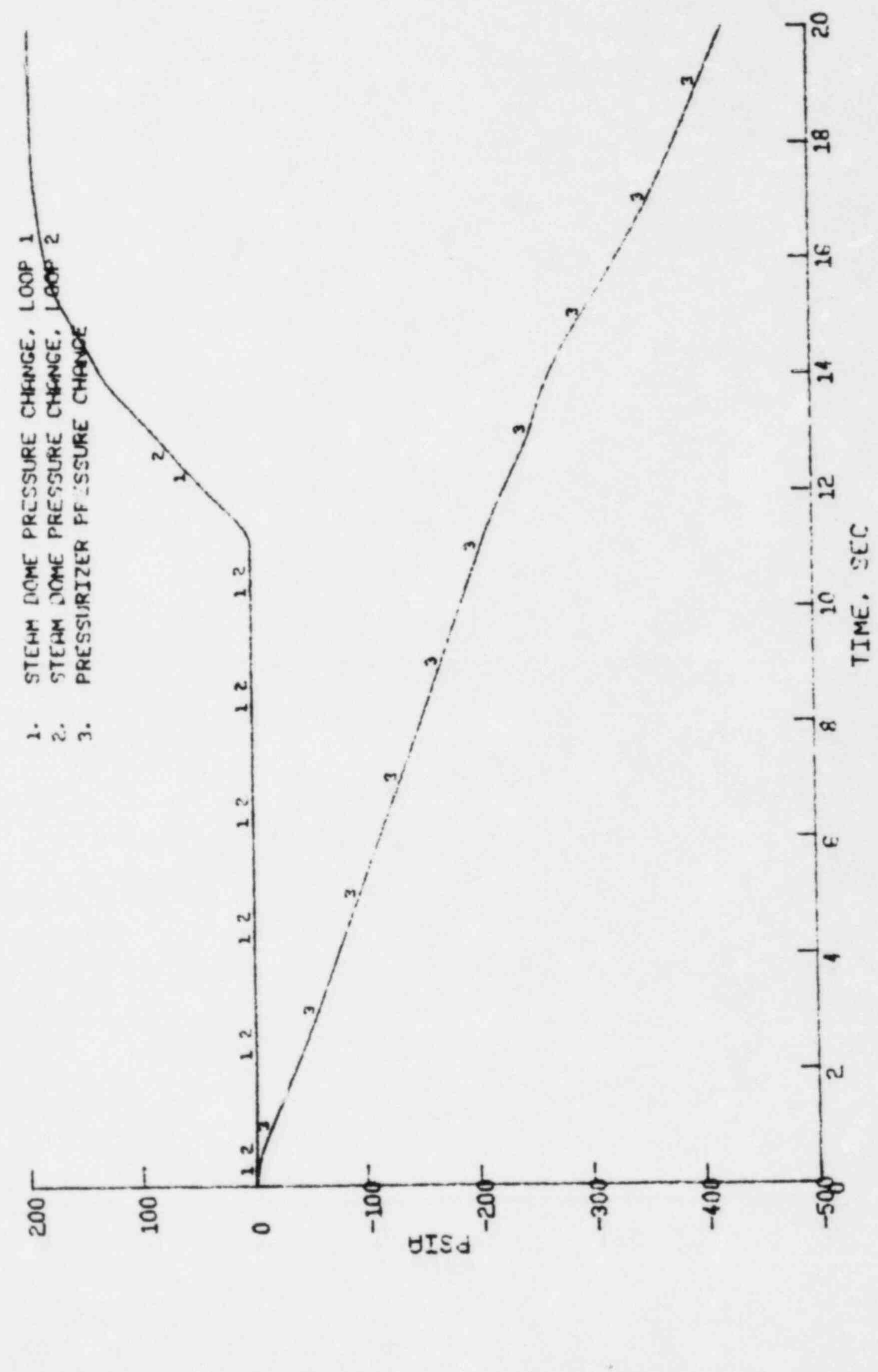

Figure 3.30 St. Lucie Unit 1 - Pressures - RCS Depressurization

STLUC08 20/12/82 20.53.30.

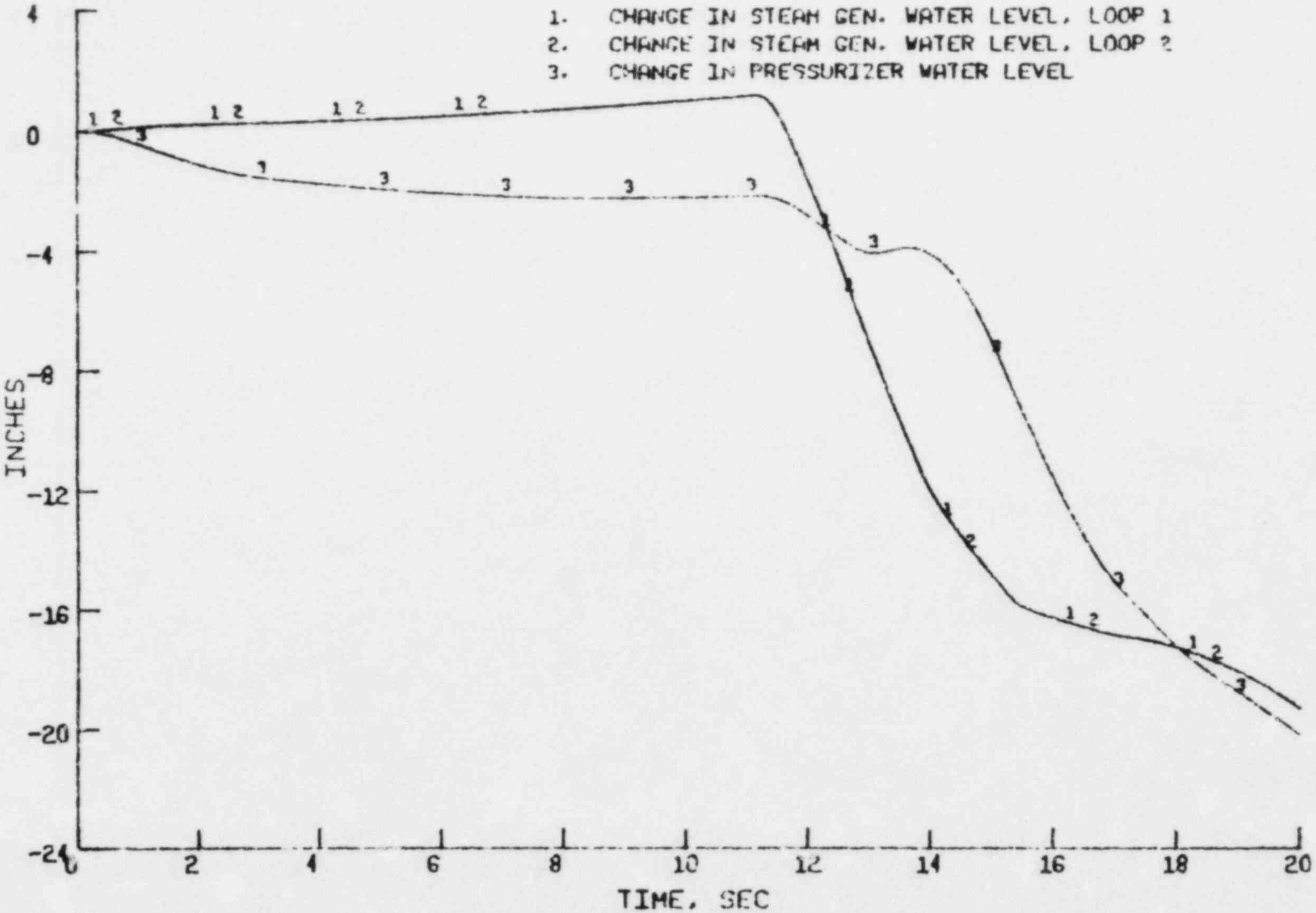

Figure 3.31 St. Lucie Unit 1 - Water Levels - RCS Depressurization

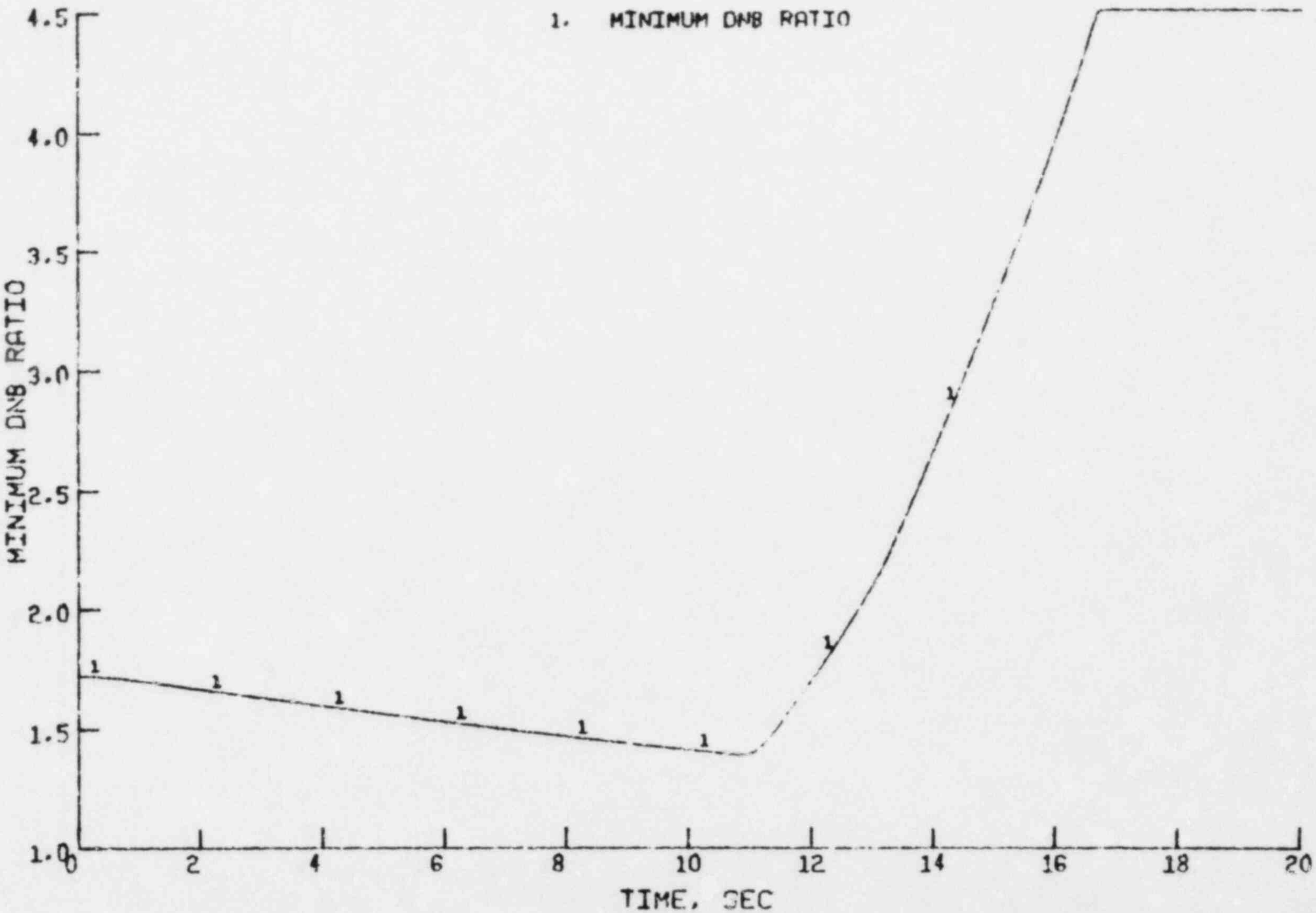

Figure 3.32 St. Lucie Unit 1 - DNBR - RCS Depressurization

STLUCO8 20/12/82 20.53.30.

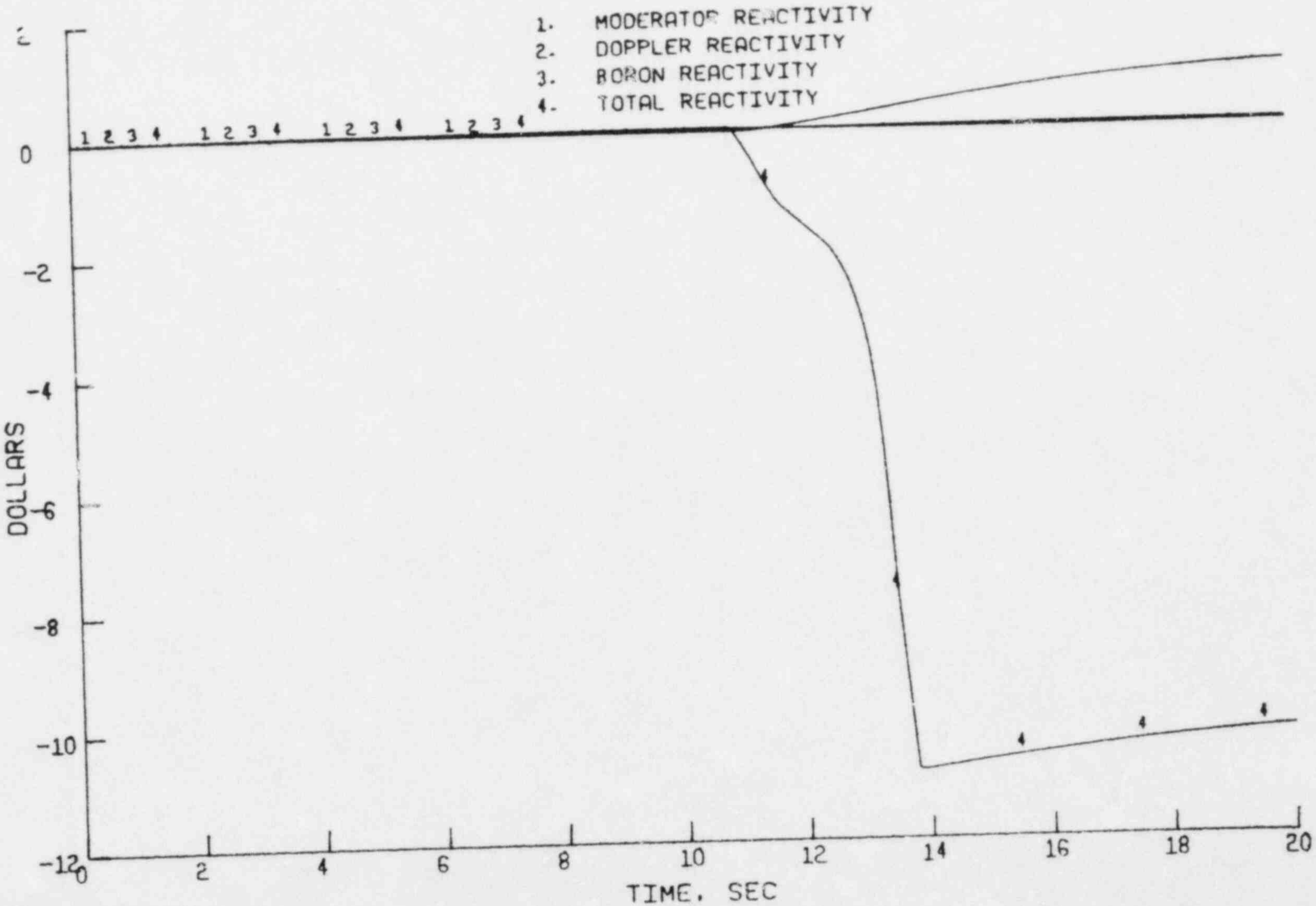

Figure 3.33 St. Lucie Unit 1 - Reactivity - RCS Depressurization

STLUCOR 20/12/82 20.53.30.

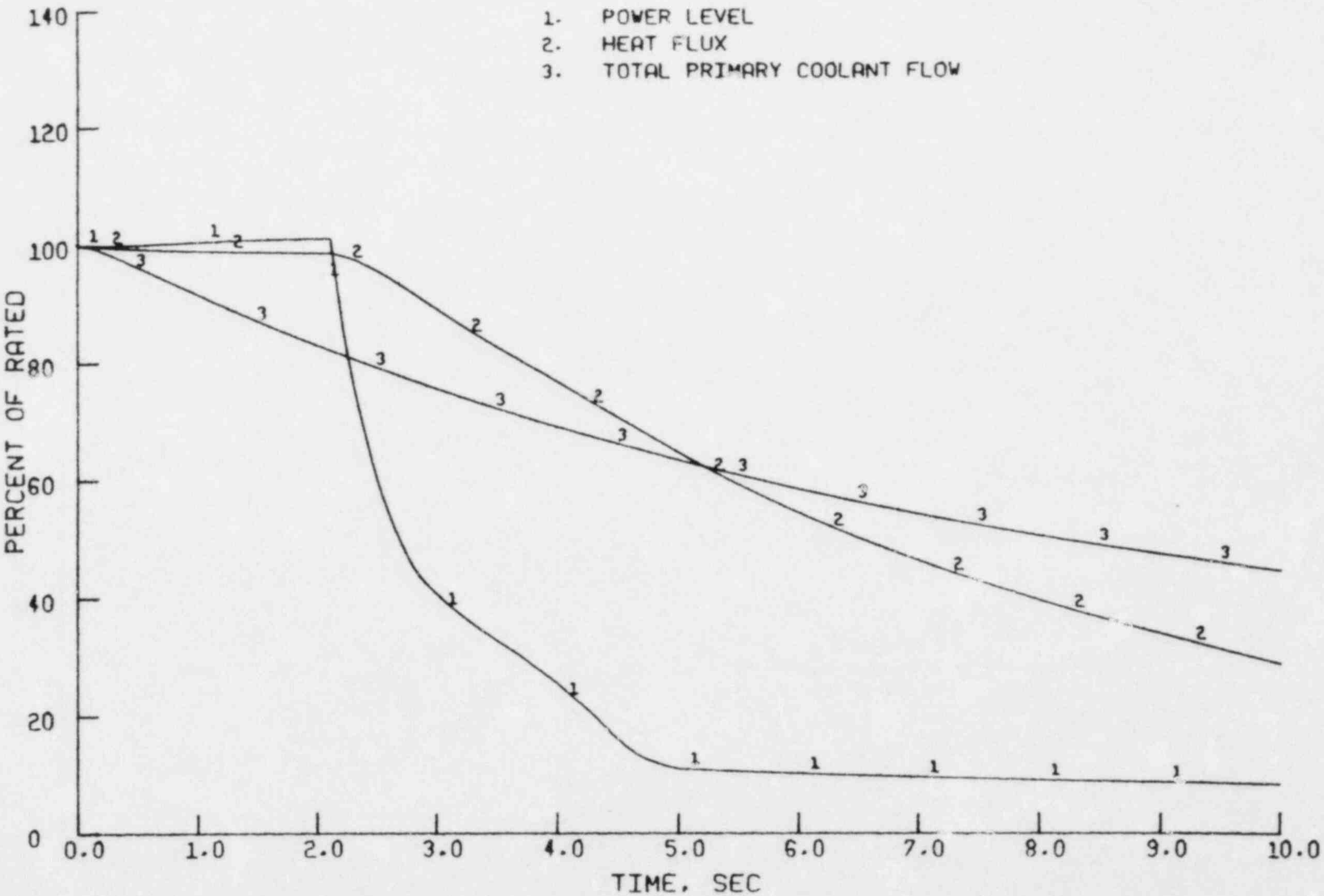

Figure 3.34 St. Lucie Unit 1 - Power, Heat Flux and Flow - Loss Of Coolant Flow

STLUCPJ 20/12/82 20.31.15.

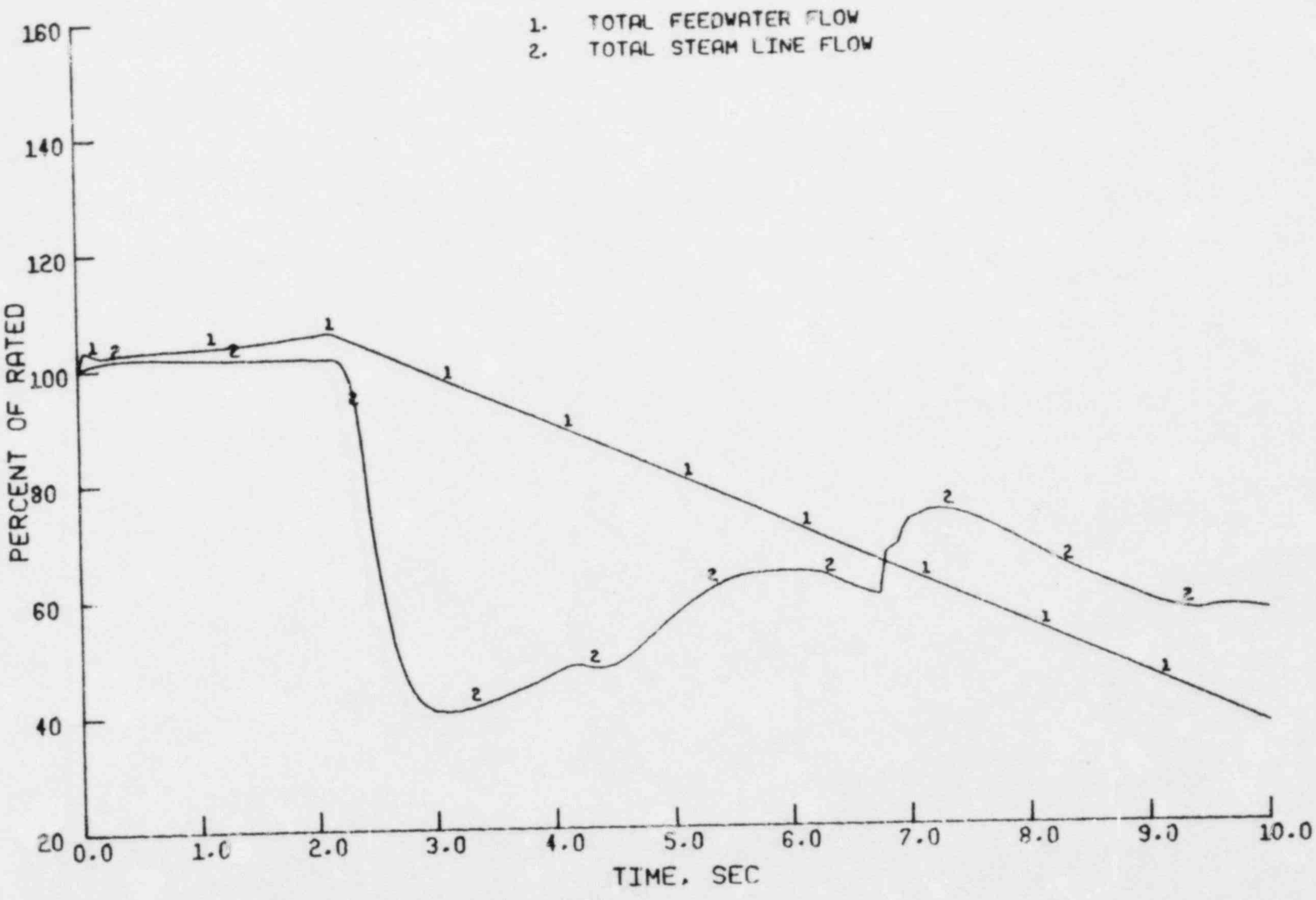

Figure 3.35 St. Lucie Unit 1 - S. G. Flows - Loss Of Coolant Flow

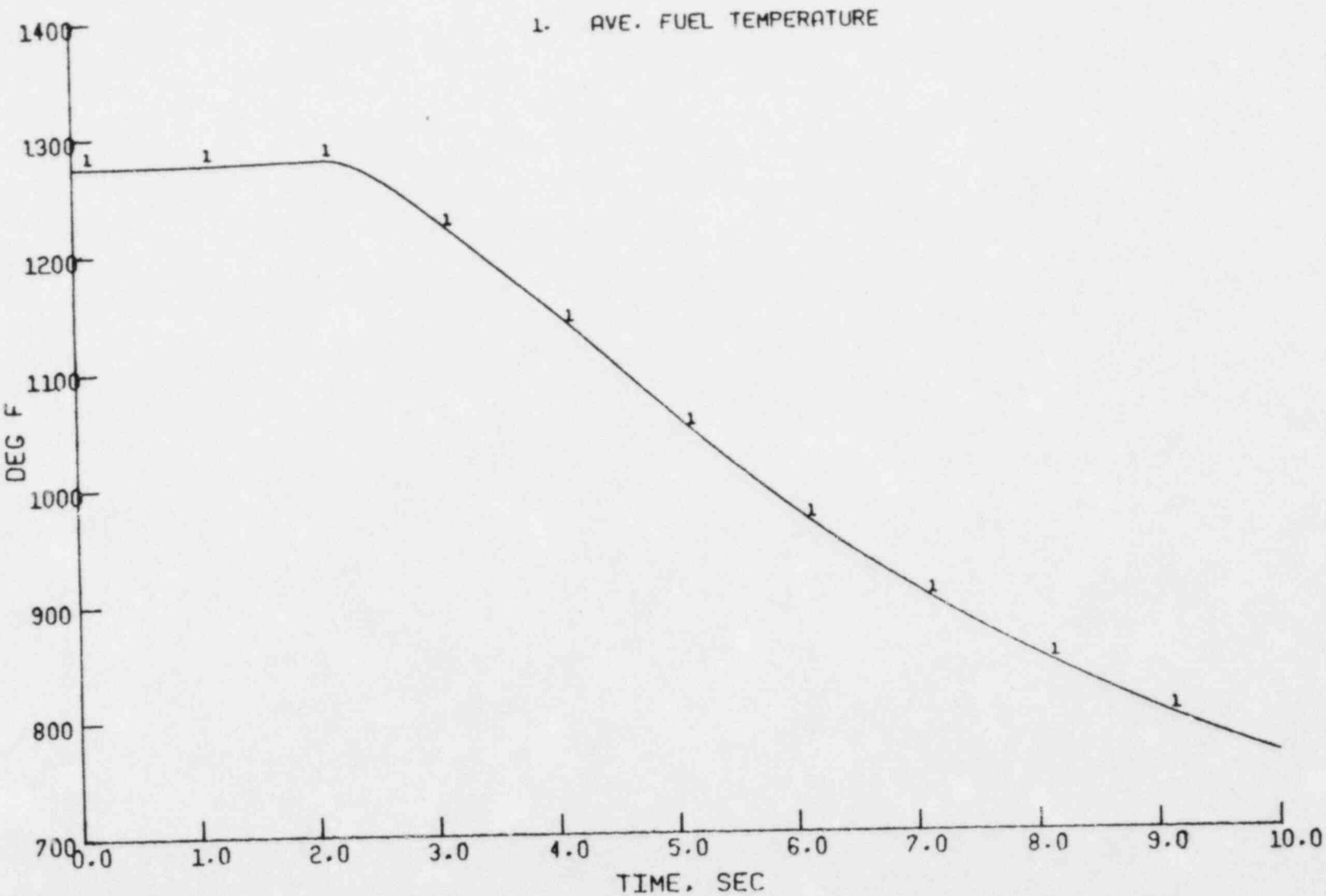

Figure 3.36 St. Lucie Unit 1 - Fuel Temperature - Loss Of Coolant Flow

20.31.15.

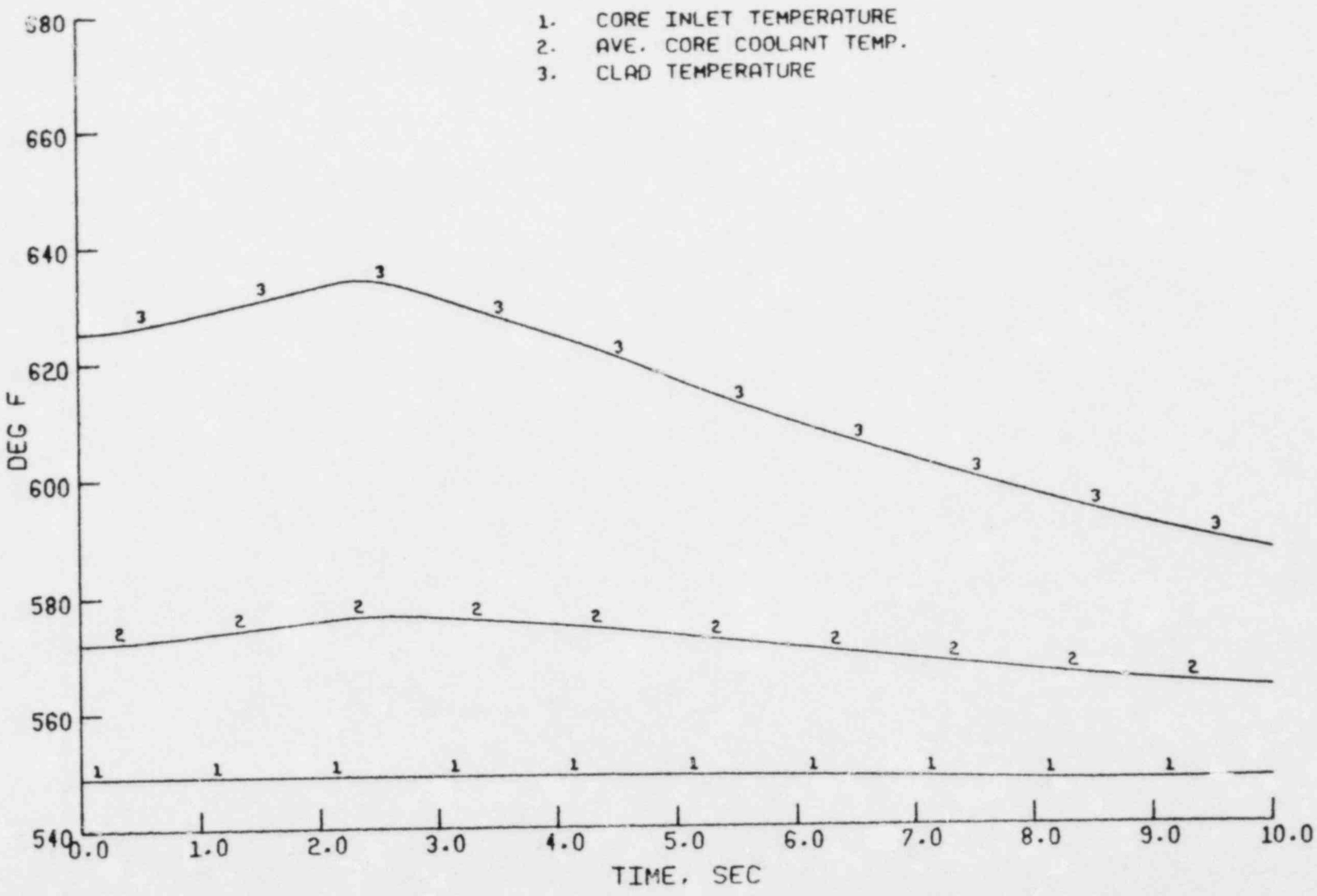

Figure 3.37 St. Lucie Unit 1 - Core Temperatures - Loss Of Coolant Flow

STLUCPJ 20/12/82 20.31.15.

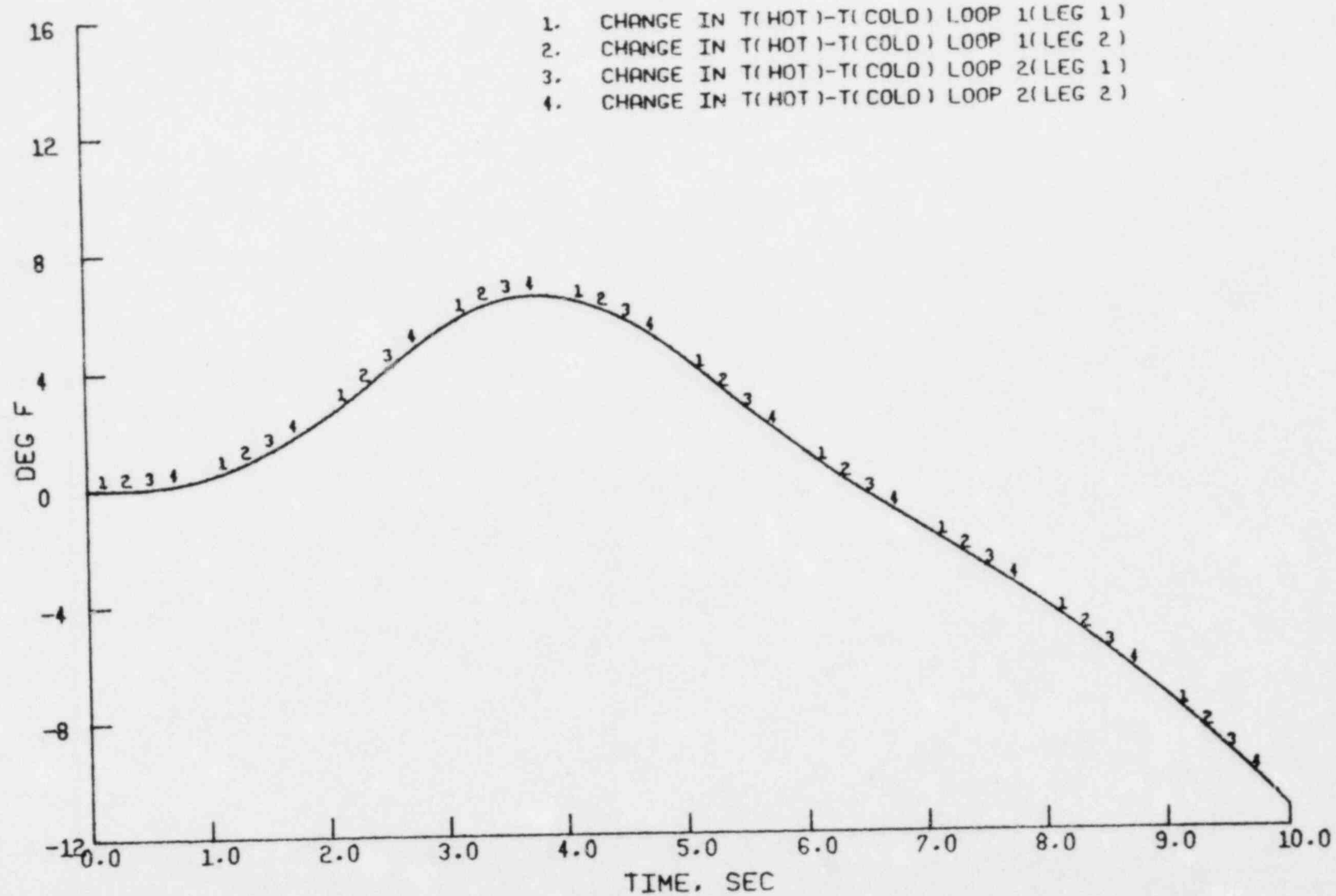

Figure 3.38 St. Lucie Unit 1 - Loop Temperature Differences - Loss Of Coolant Flow

STLUCPJ 20/12/82 20.31.15.

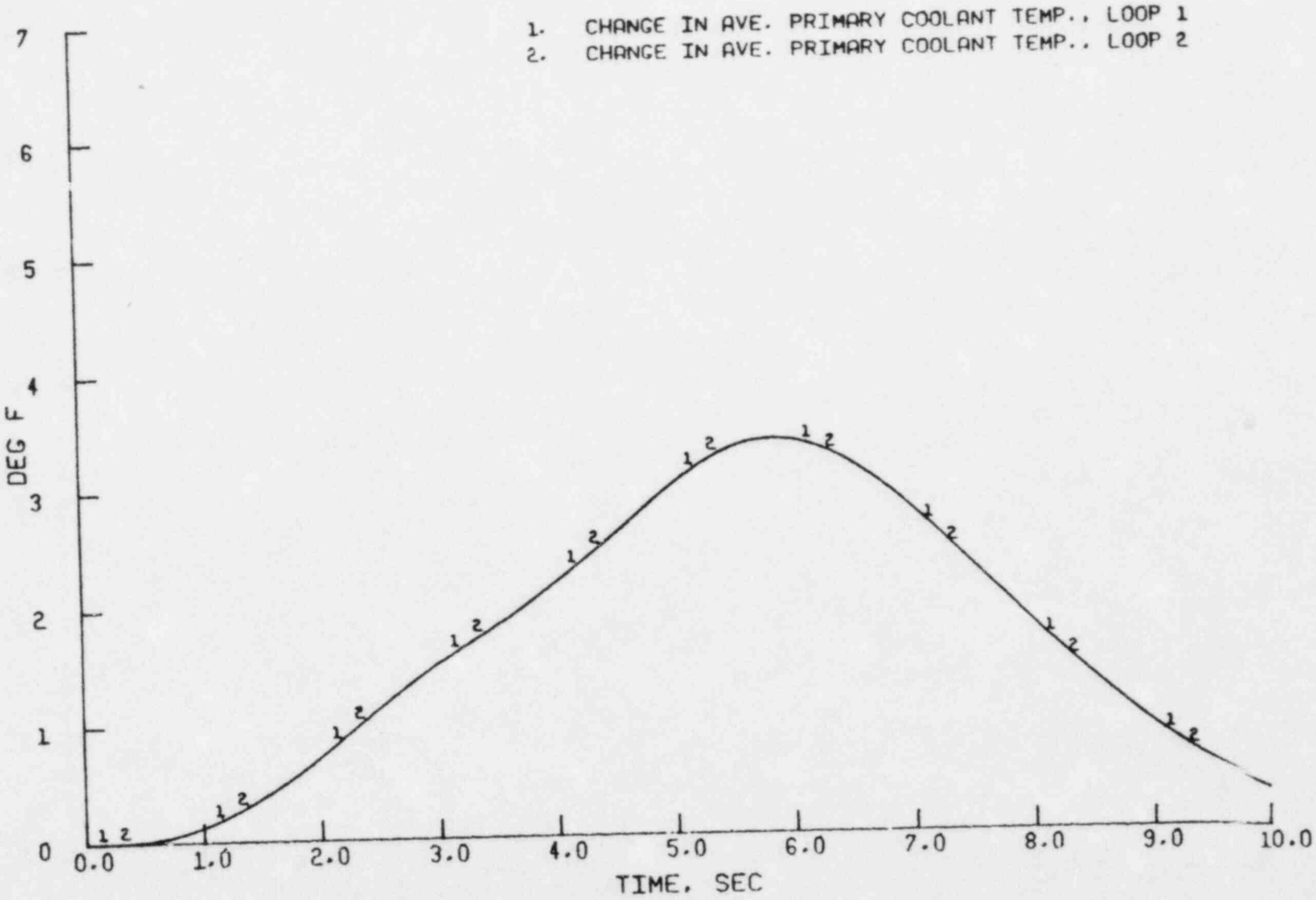

Figure 3.39 St. Lucie Unit 1 - Average Temperatures - Loss Of Coolant Flow

STLUCPJ 20/12/82 20.31.

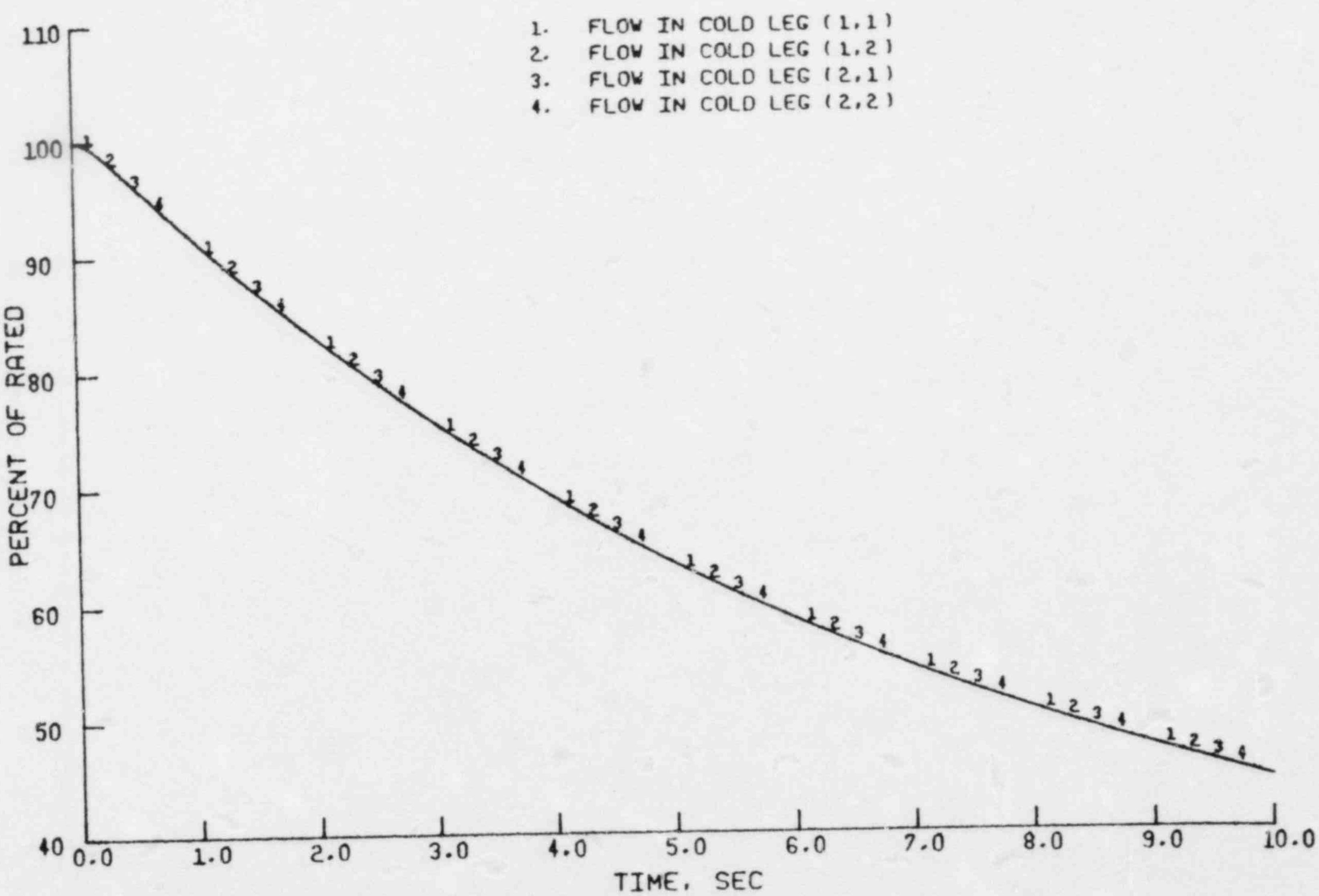

Figure 3.40 St. Lucie Unit 1 - Cold Leg Flows - Loss Of Coolant Flow

STLUCPJ 20/12/82 20.31.15.

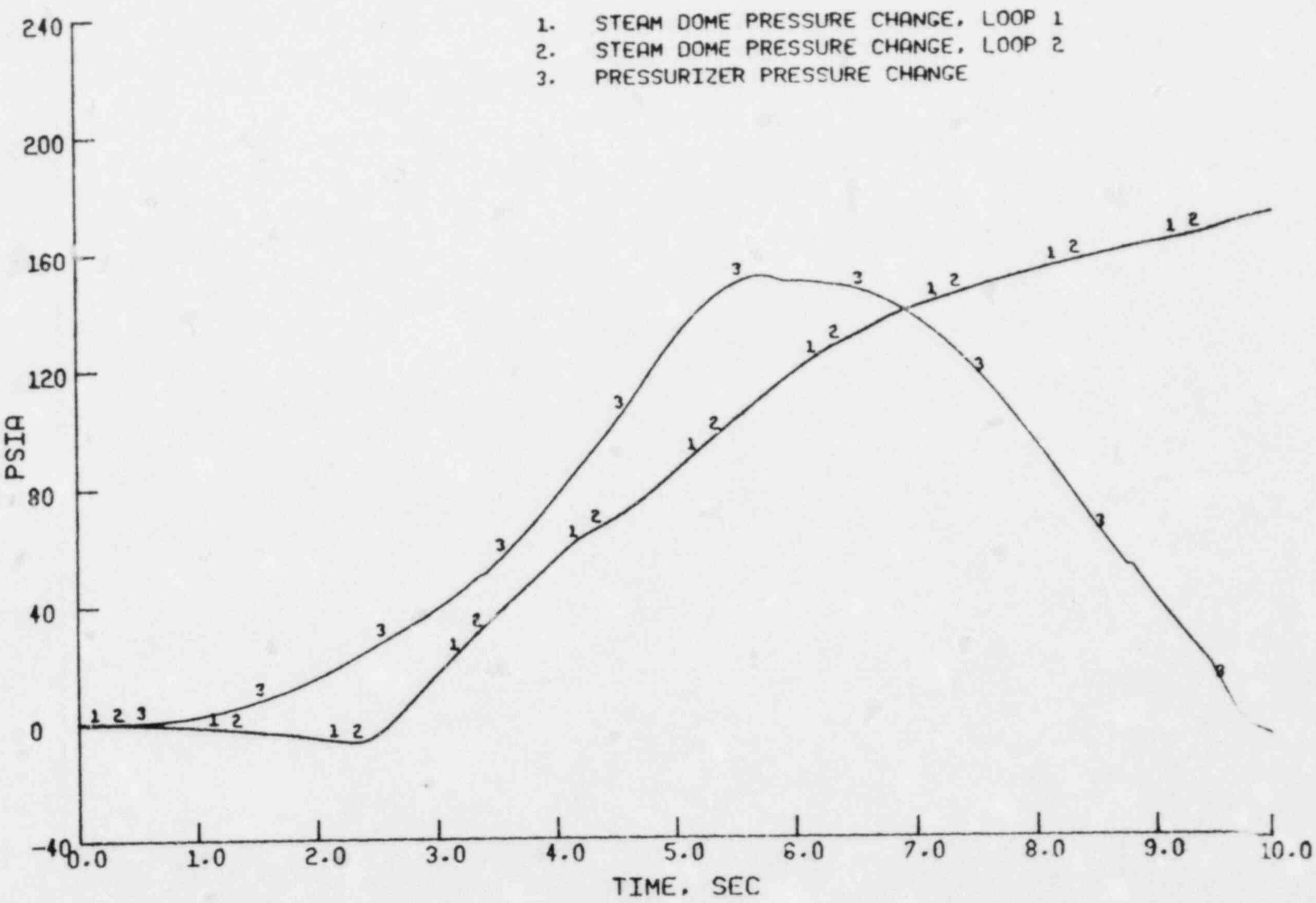

Figure 3.41 St. Lucie Unit 1 - Pressures - Loss Of Coolant Flow

STLUCPJ 20/12/82 20.31.15

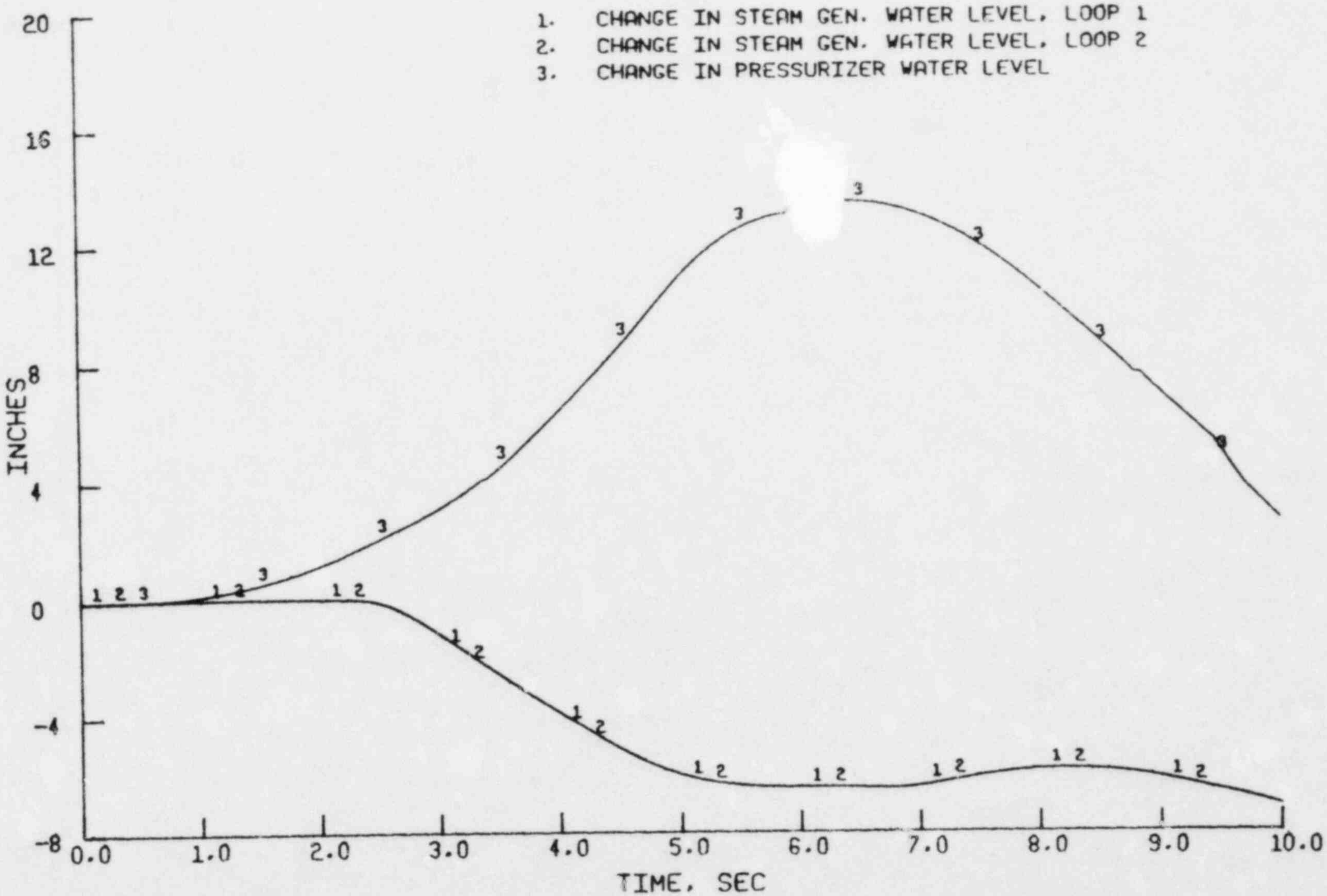

Figure 3.42 St. Lucie Unit 1 - Water Levels - Loss Of Coolant Flow

STLUCPJ 20/12/82 20.31.15.

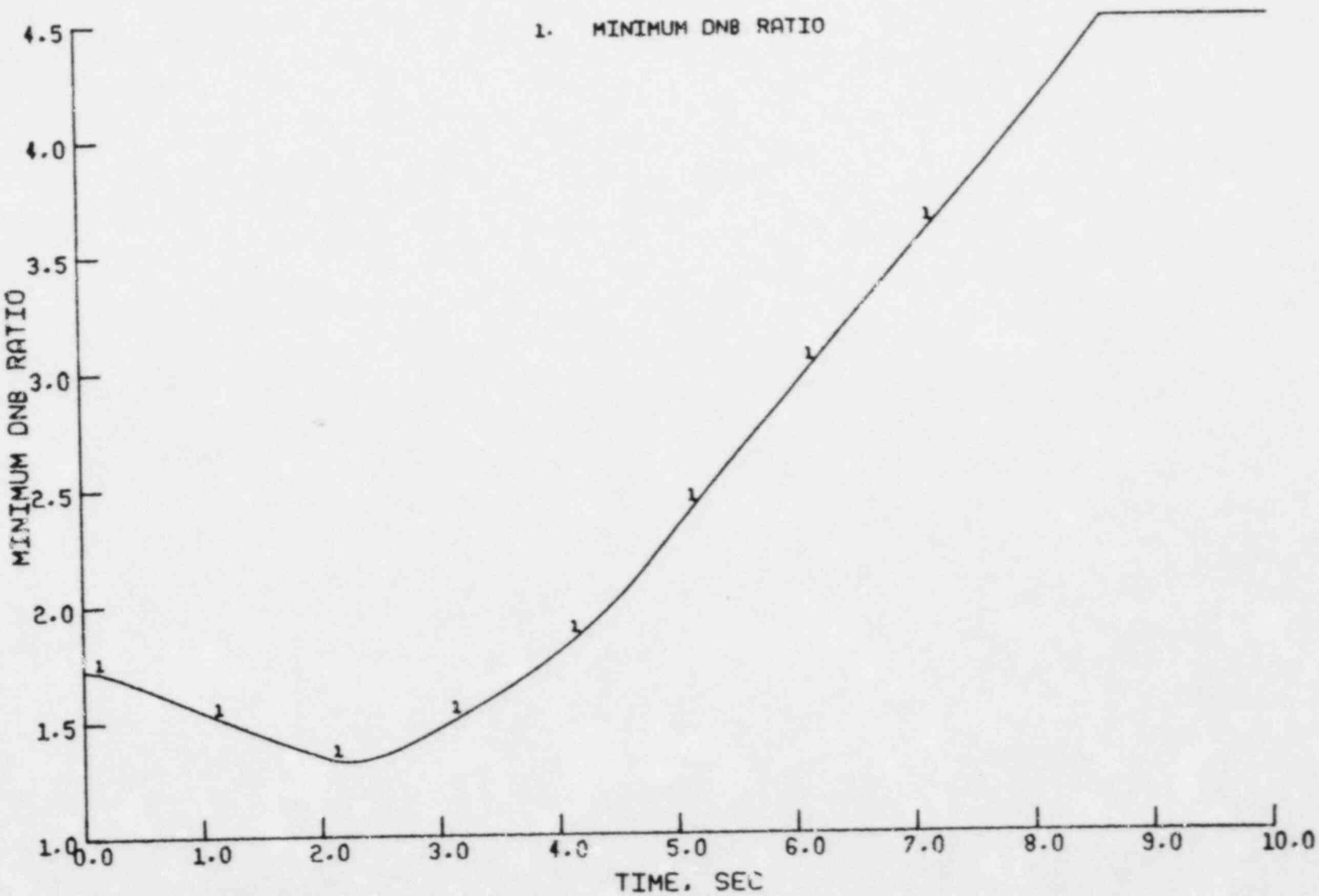

Figure 3.43 St. Lucie Unit 1 - DNBR - Loss Of Coolant Flow

S?LUCPJ 20/12/82 20.31.15.

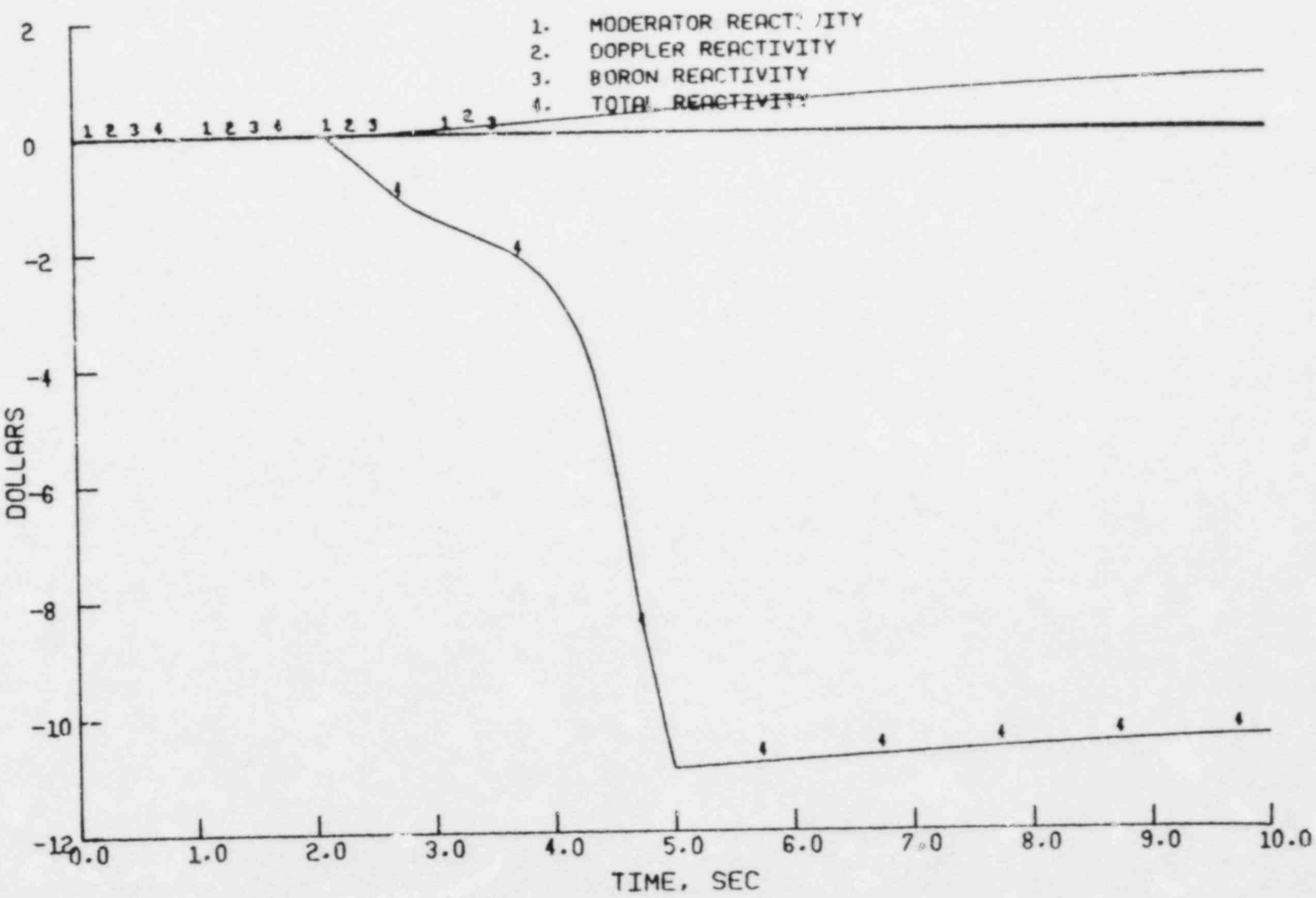

Figure 3.44 St. Lucie Unit 1 - Reactivity - Loss Of Coolant Flow

STLUCPJ 20/12/82 20.31.15.

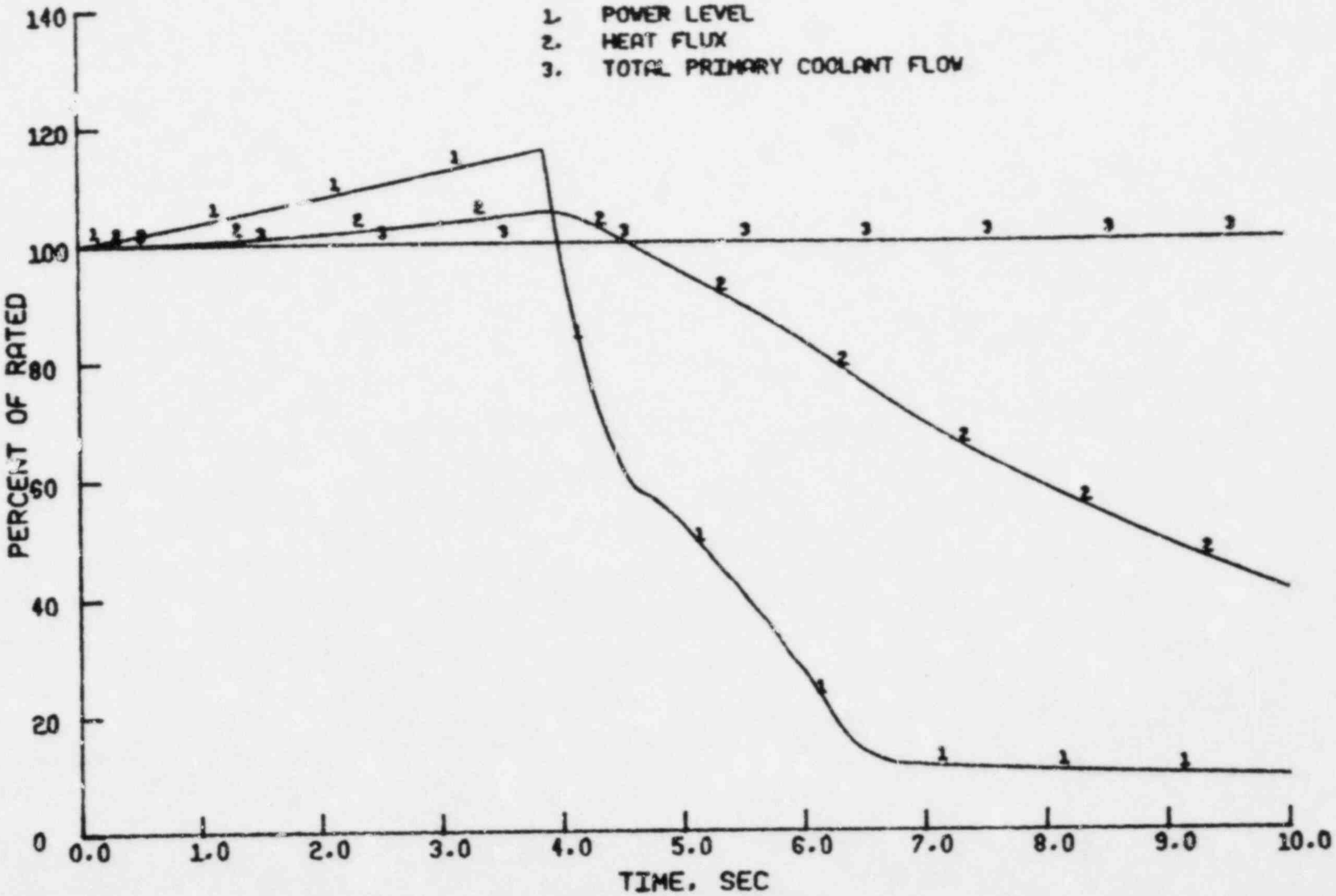

Figure 3.45 St. Lucie Unit 1 - Power, Heat Flux and Flow - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

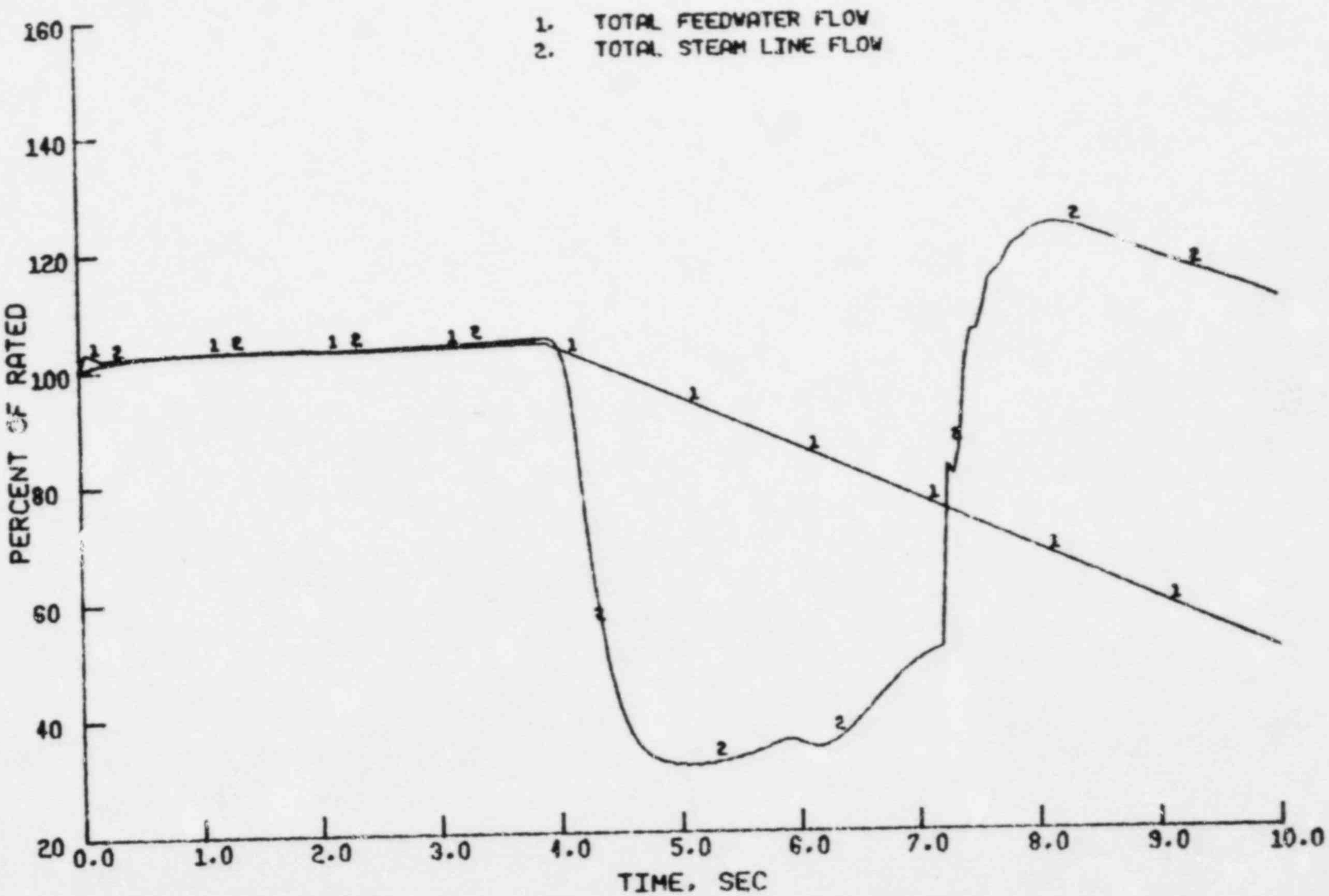

Figure 3.46 St. Lucie Unit 1 - S. G. Flows - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

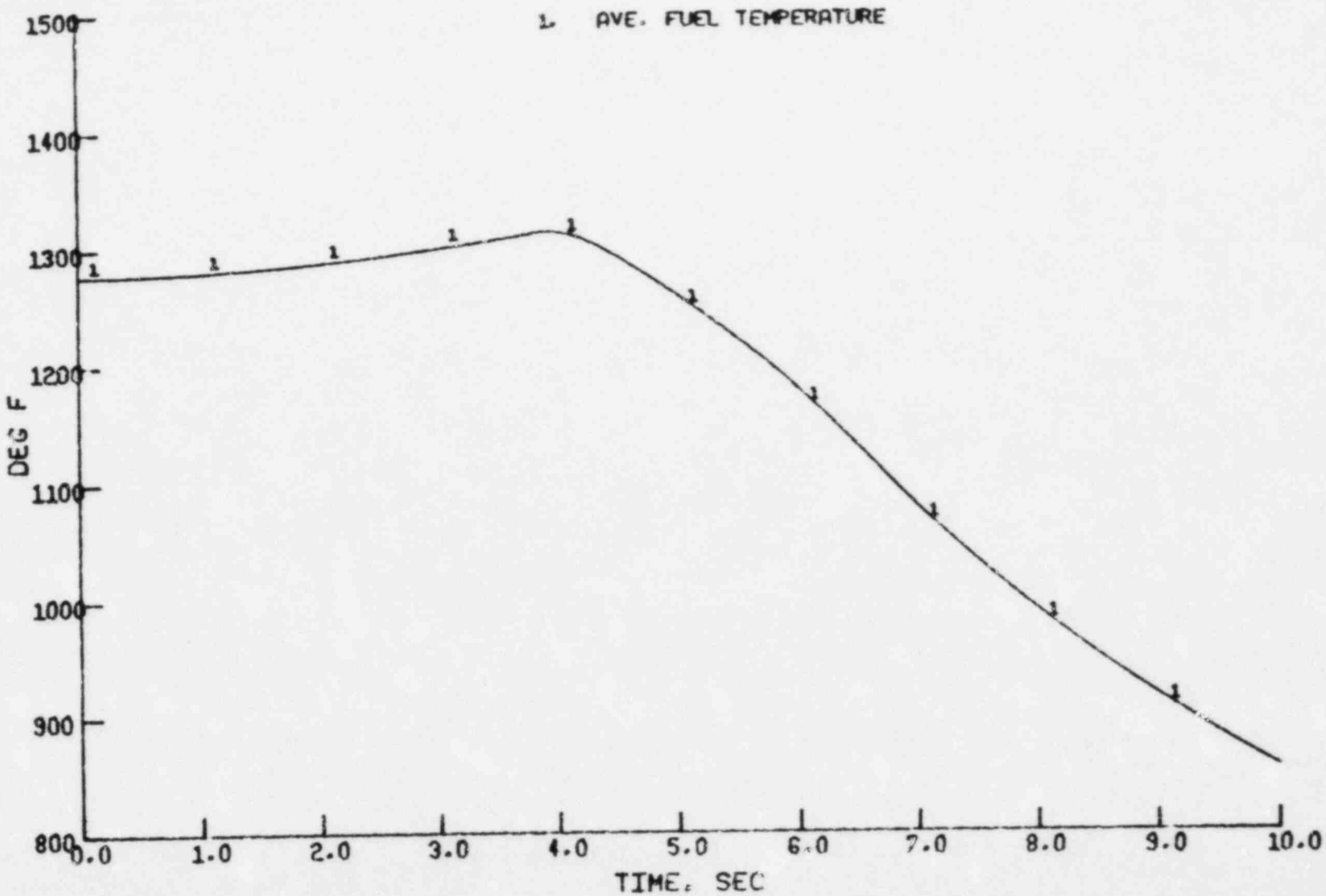

Figure 3.47 St. Lucie Unit 1 - Fuel Temperature - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

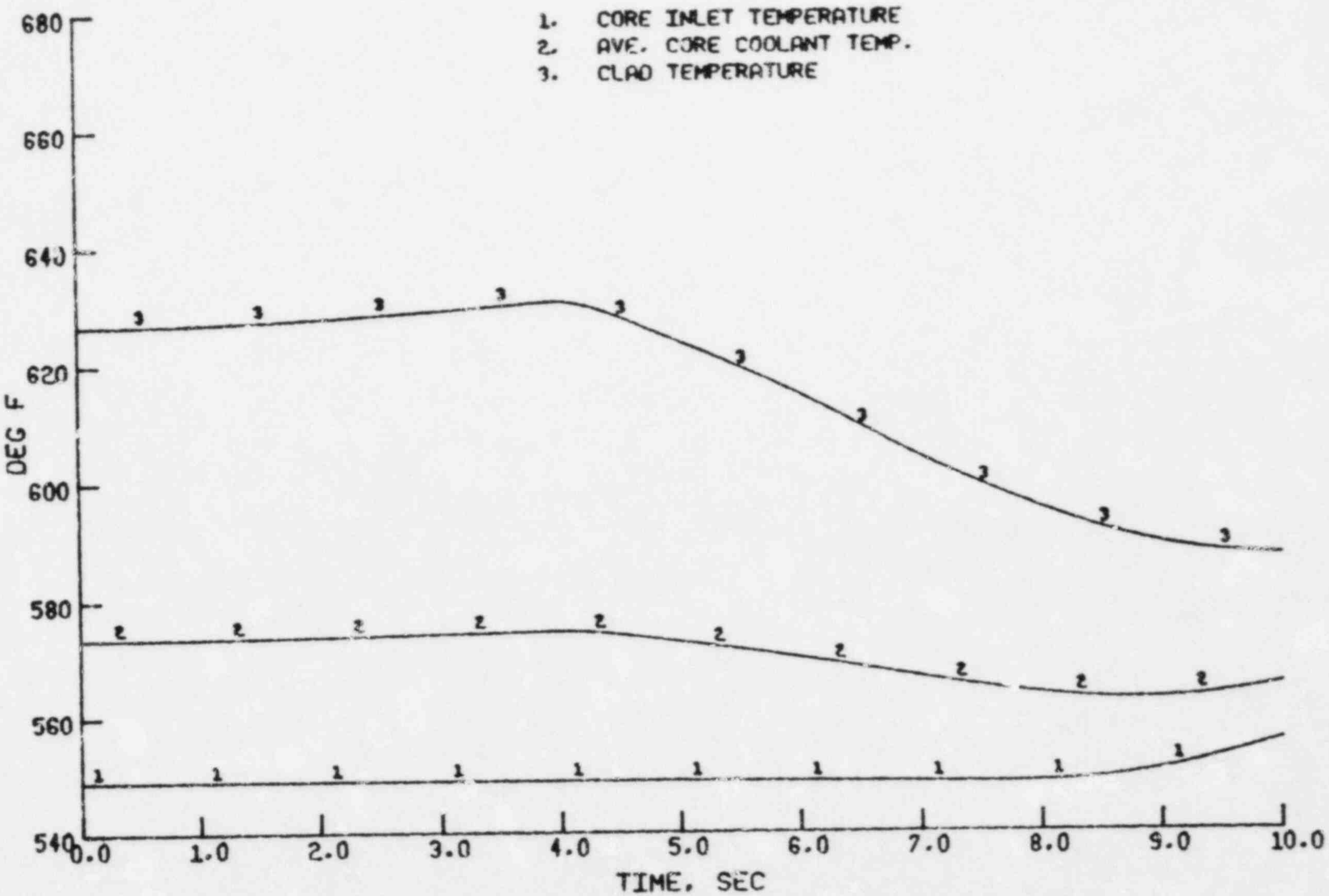

Figure 3.48 St. Lucie Unit 1 - Core Temperatures - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

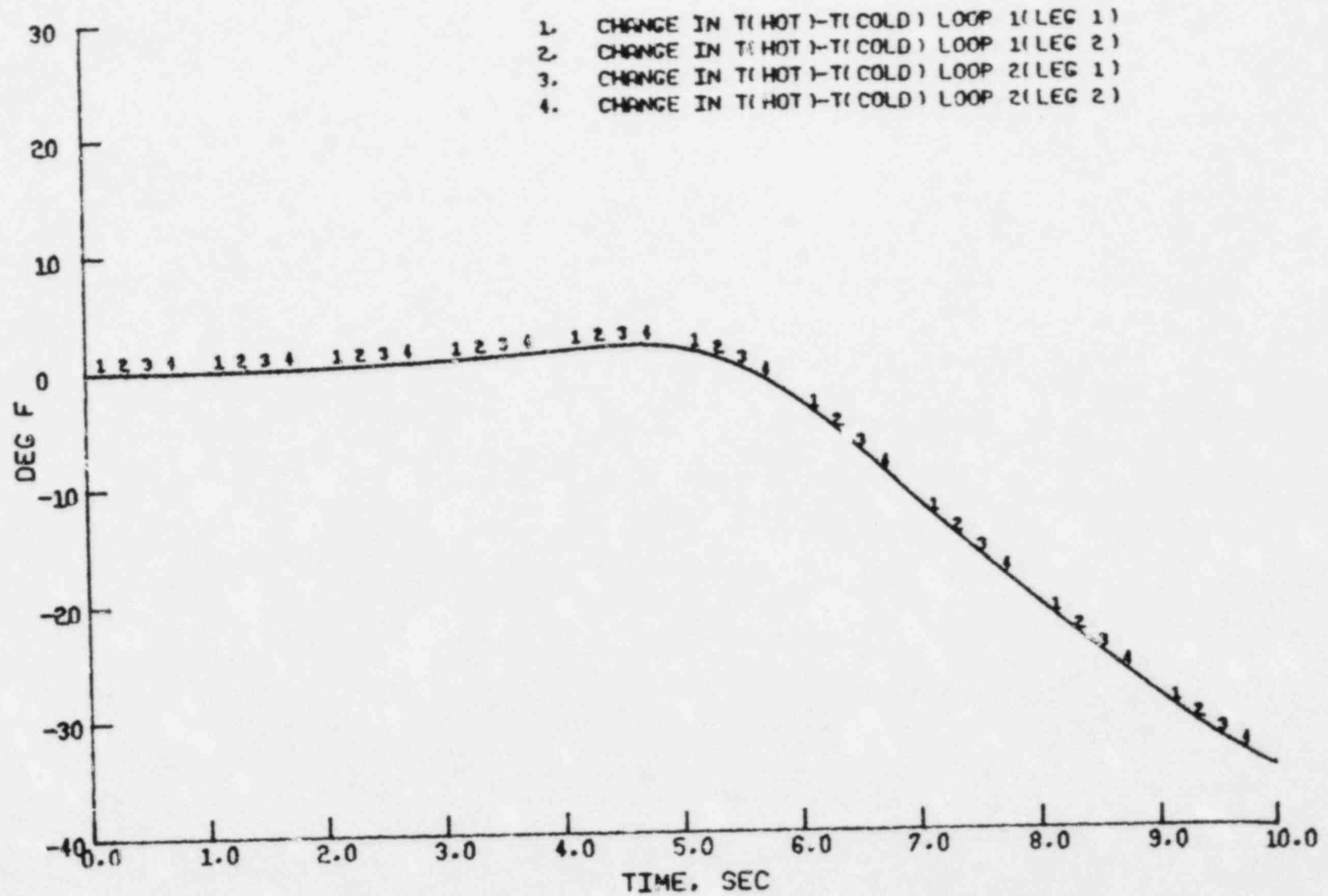

Figure 3.49 St. Lucie Unit 1 - Loop Temperature Differences - CEA Withdrawal

STLUCIU 15/12/82 20.22.54

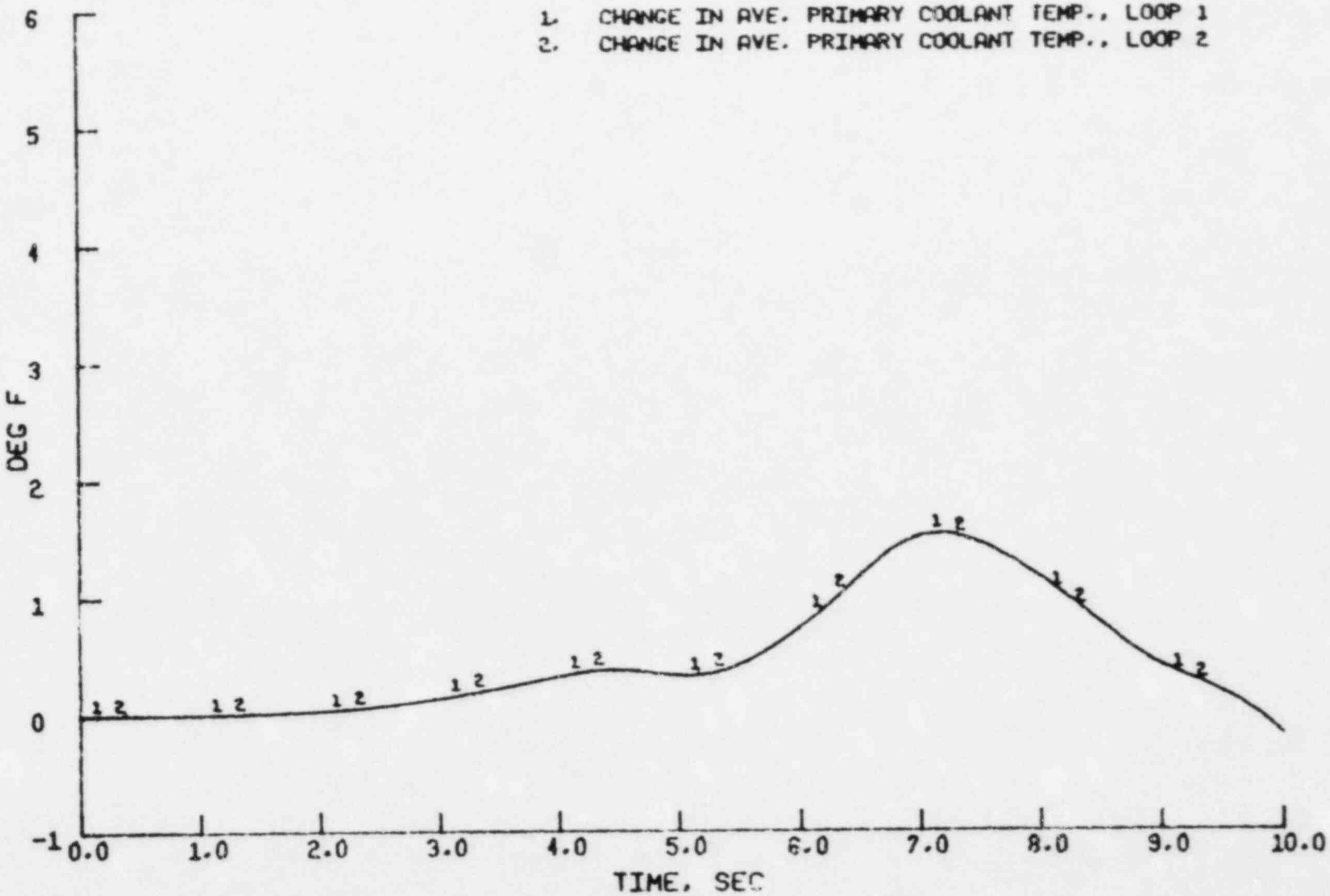

Figure 3.50 St. Lucie Unit 1 - Average Temperatures - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

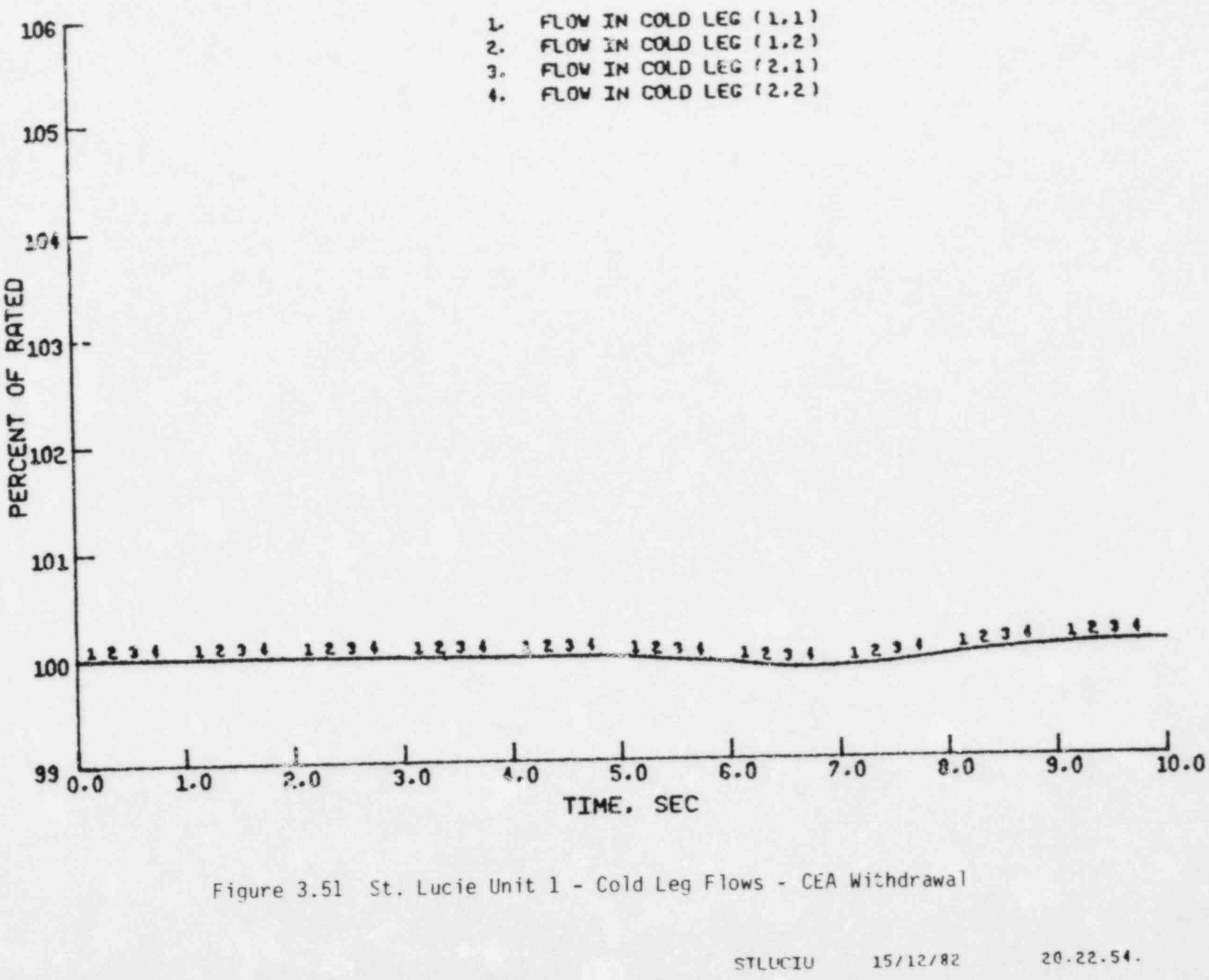

Figure 3.51 St. Lucie Unit 1 - Cold Leg Flows - CEA Withdrawal

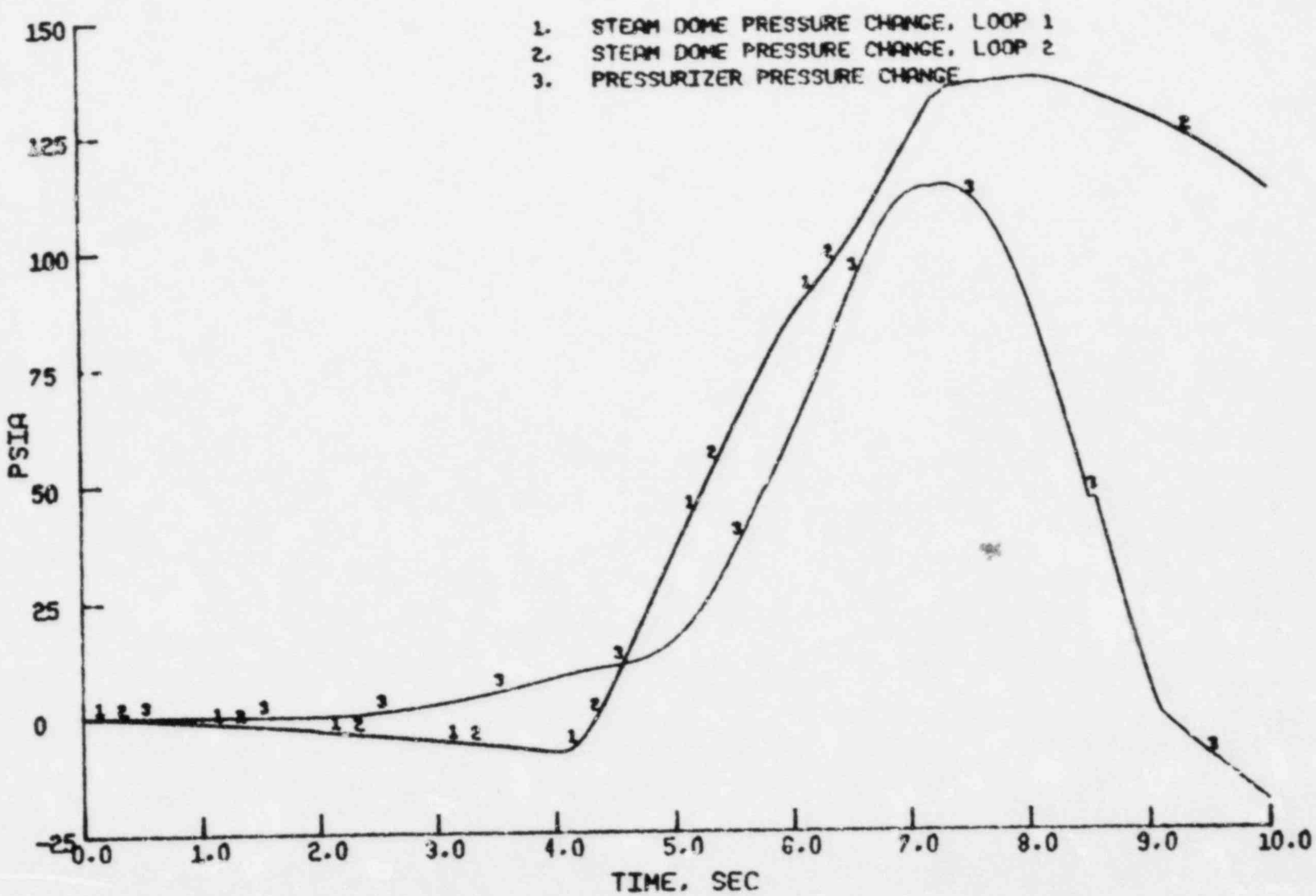

Figure 3.52 St. Lucie Unit 1 - Pressures - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

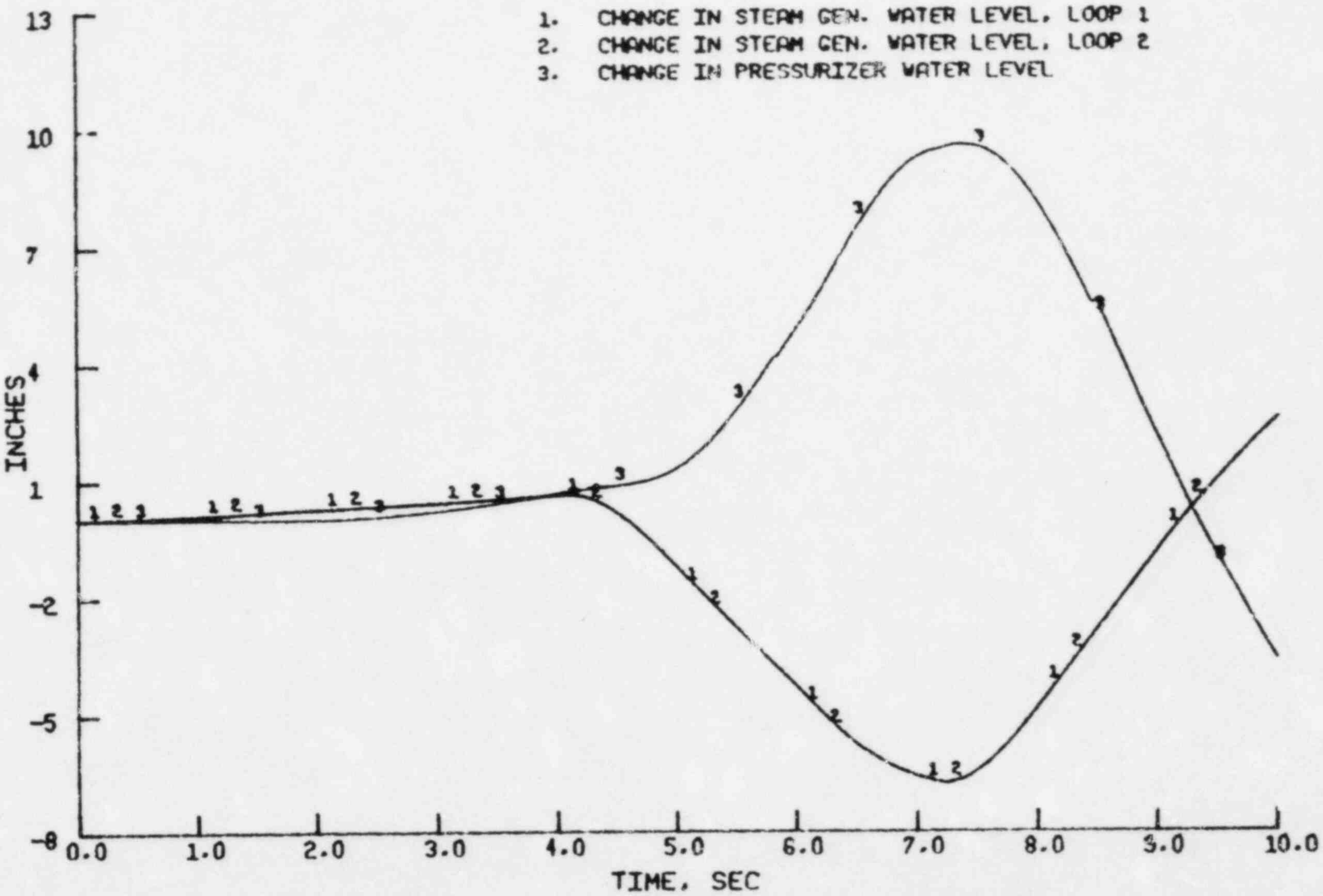

Figure 3.53 St. Lucie Unit 1 - Water Levels - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

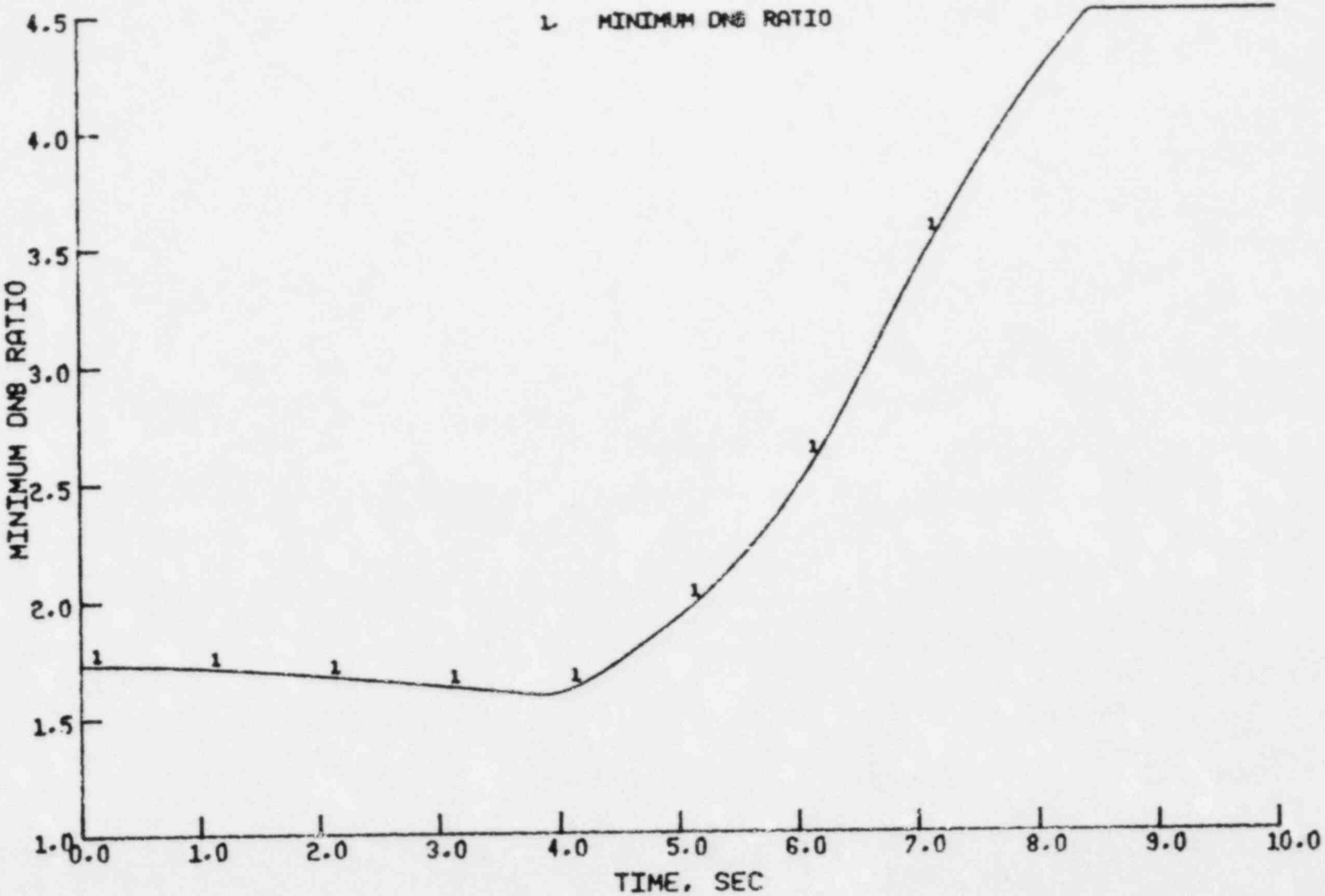

Figure 3.54 St. Lucie Unit 1 - DNBR - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

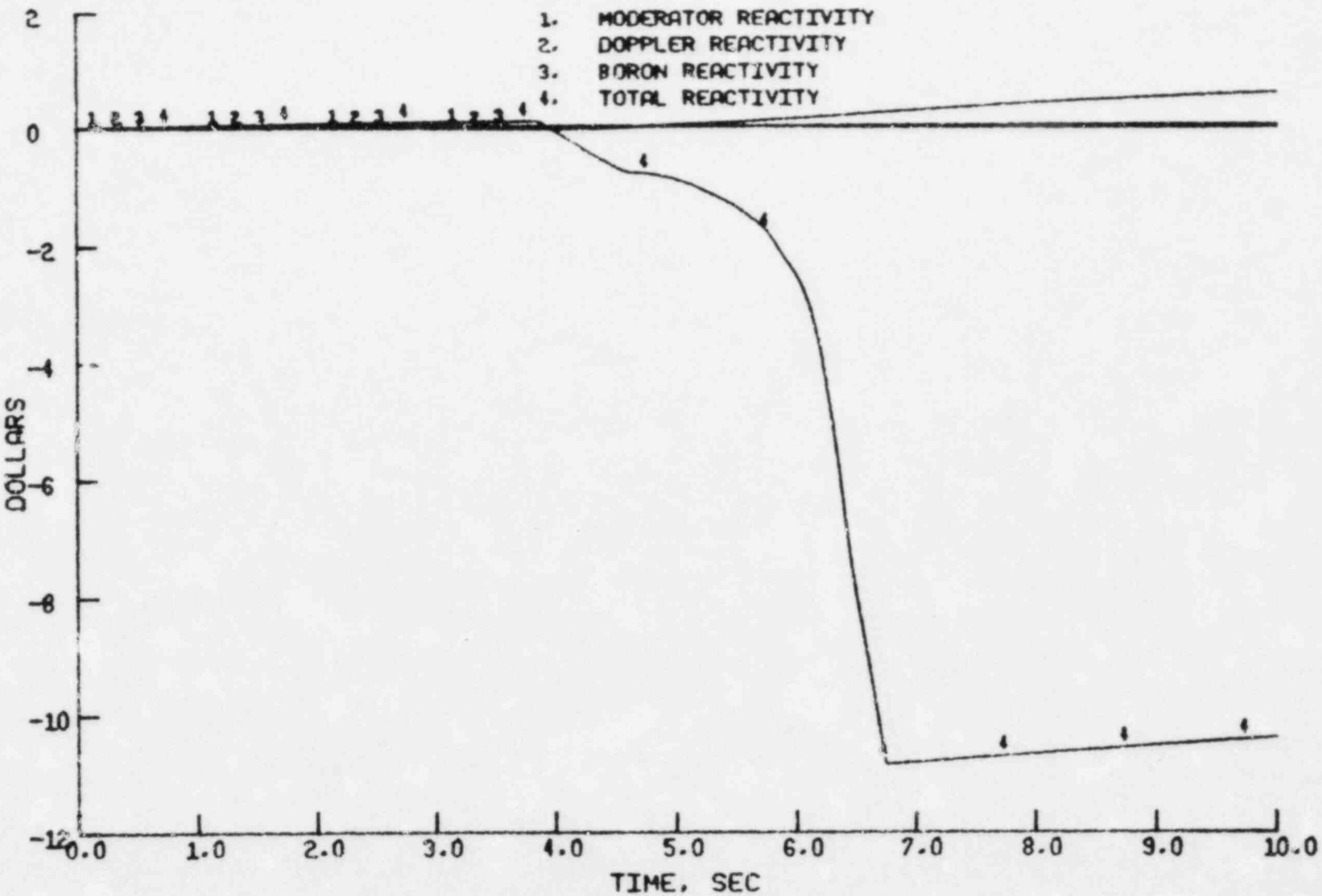

Figure 3.55 St. Lucie Unit 1 - Reactivity - CEA Withdrawal

STLUCIU 15/12/82 20.22.54.

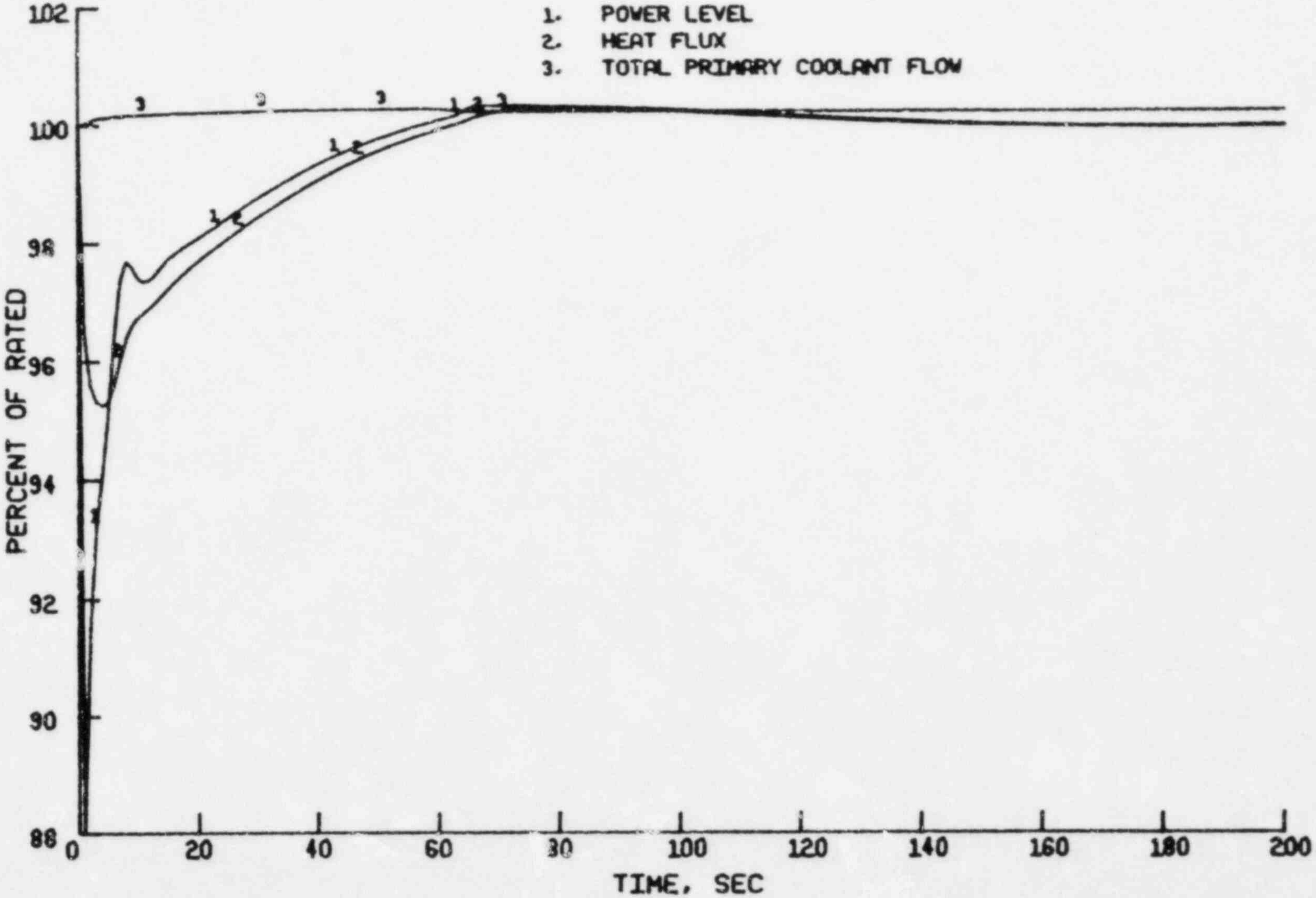

Figure 3.56 St. Lucie Unit 1 - Power, Heat Flux and Flow - CEA Drop

STLUCLK 02/12/82 20.57.52

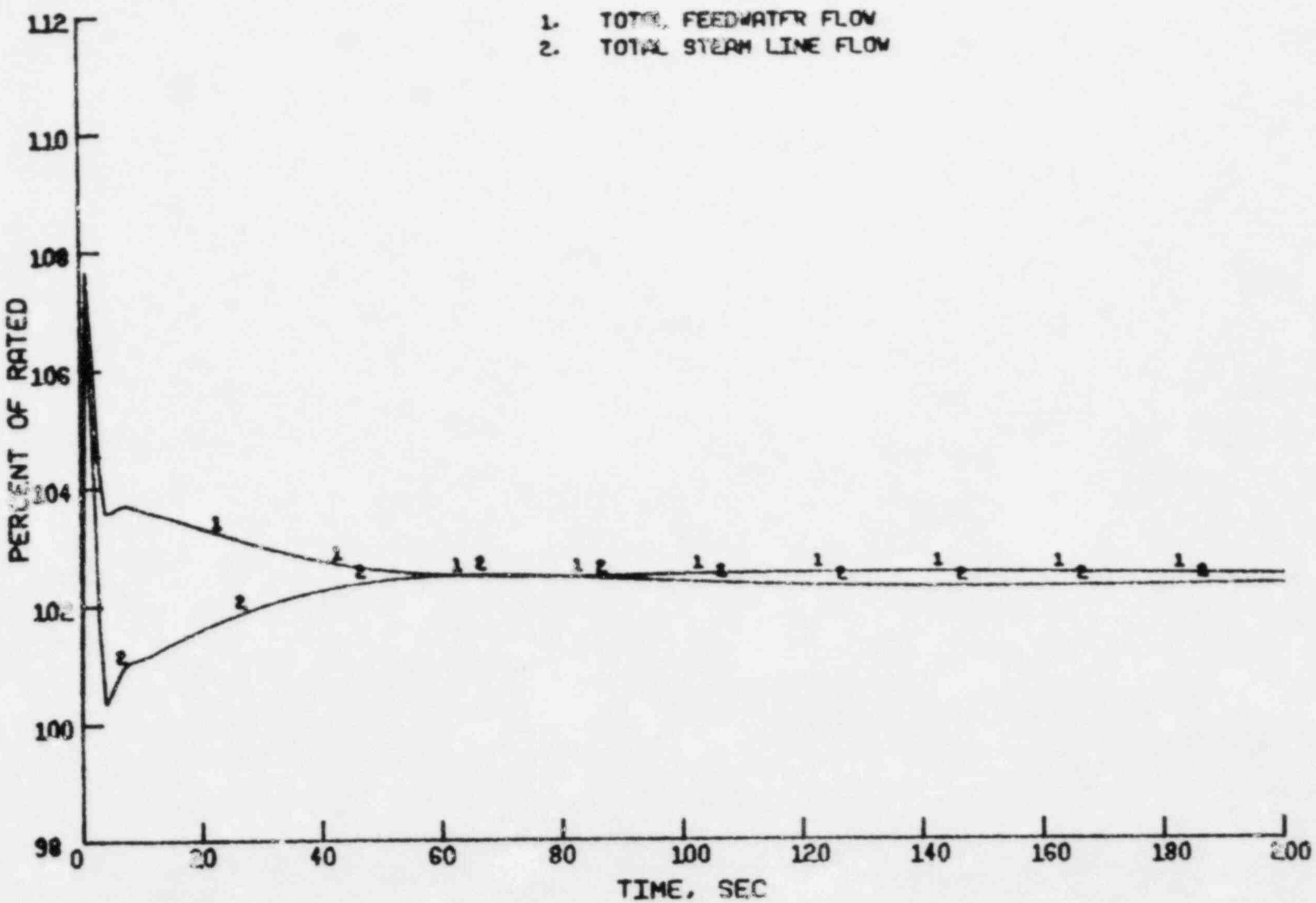

Figure 3.57 St. Lucie Unit 1 - S. G. Flows - CEA Drop

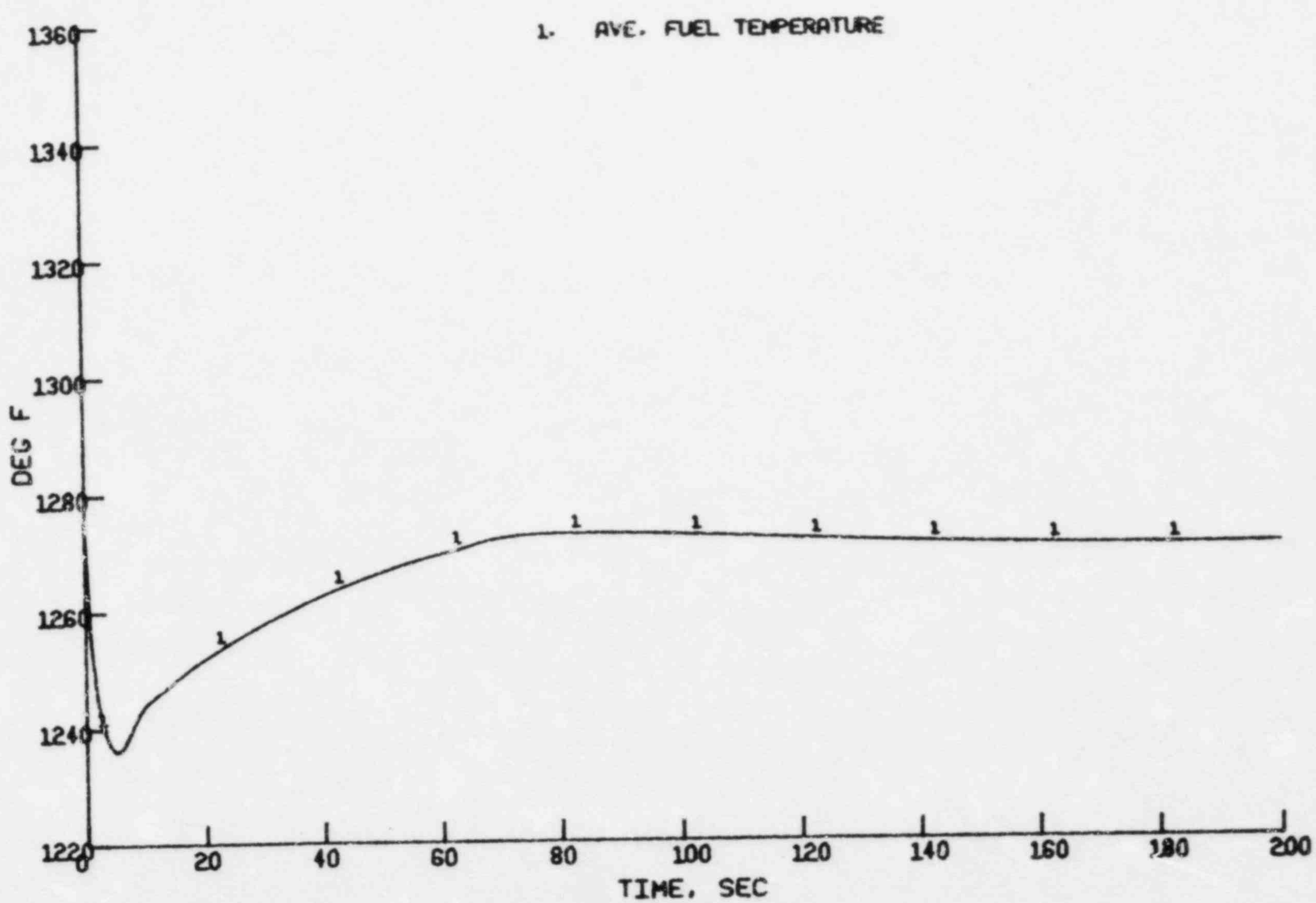

Figure 3.58 St. Lucie Unit 1 - Fuel Temperature - CEA Drop

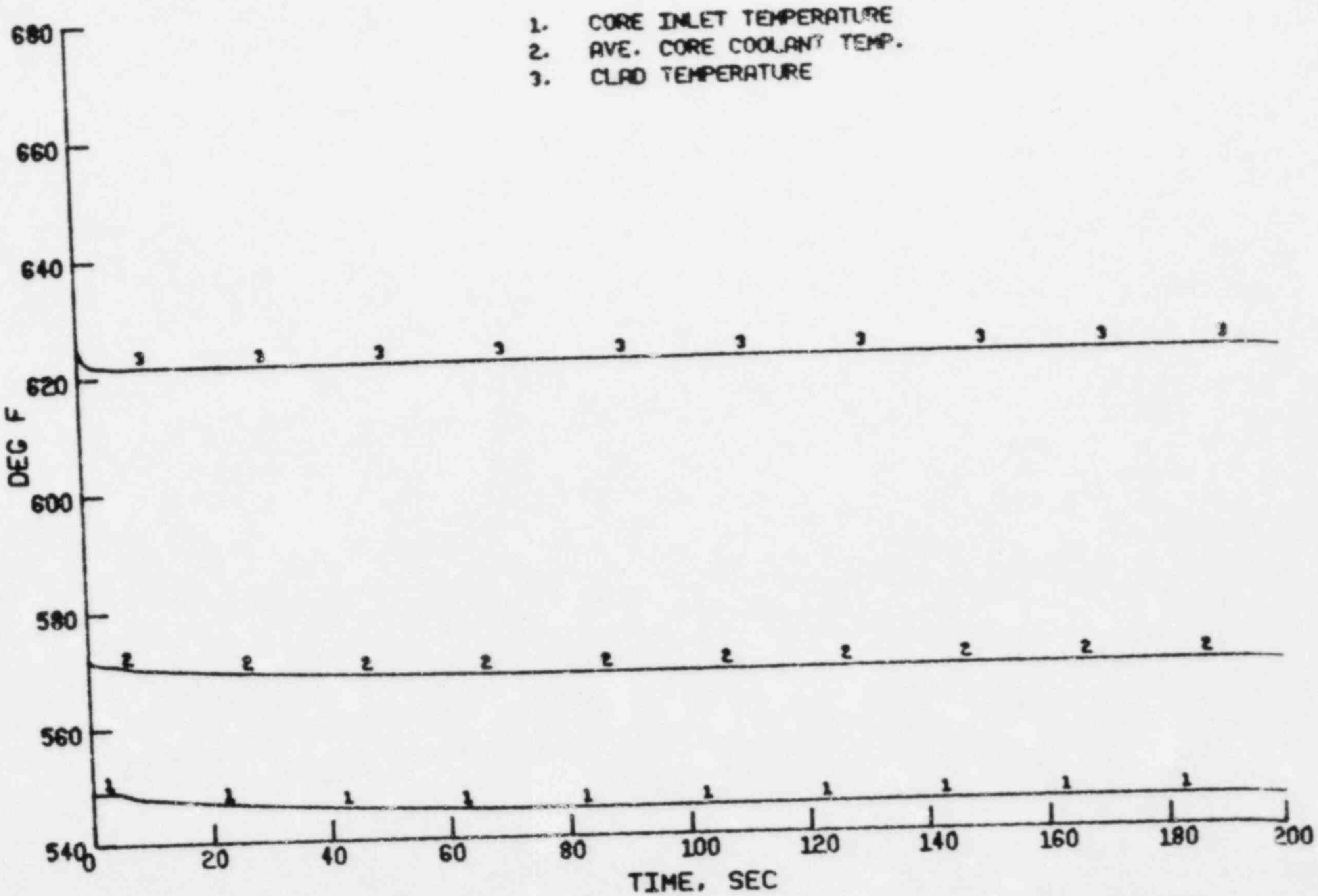

Figure 3.59 St. Lucie Unit 1 - Core Temperatures - CEA Drop

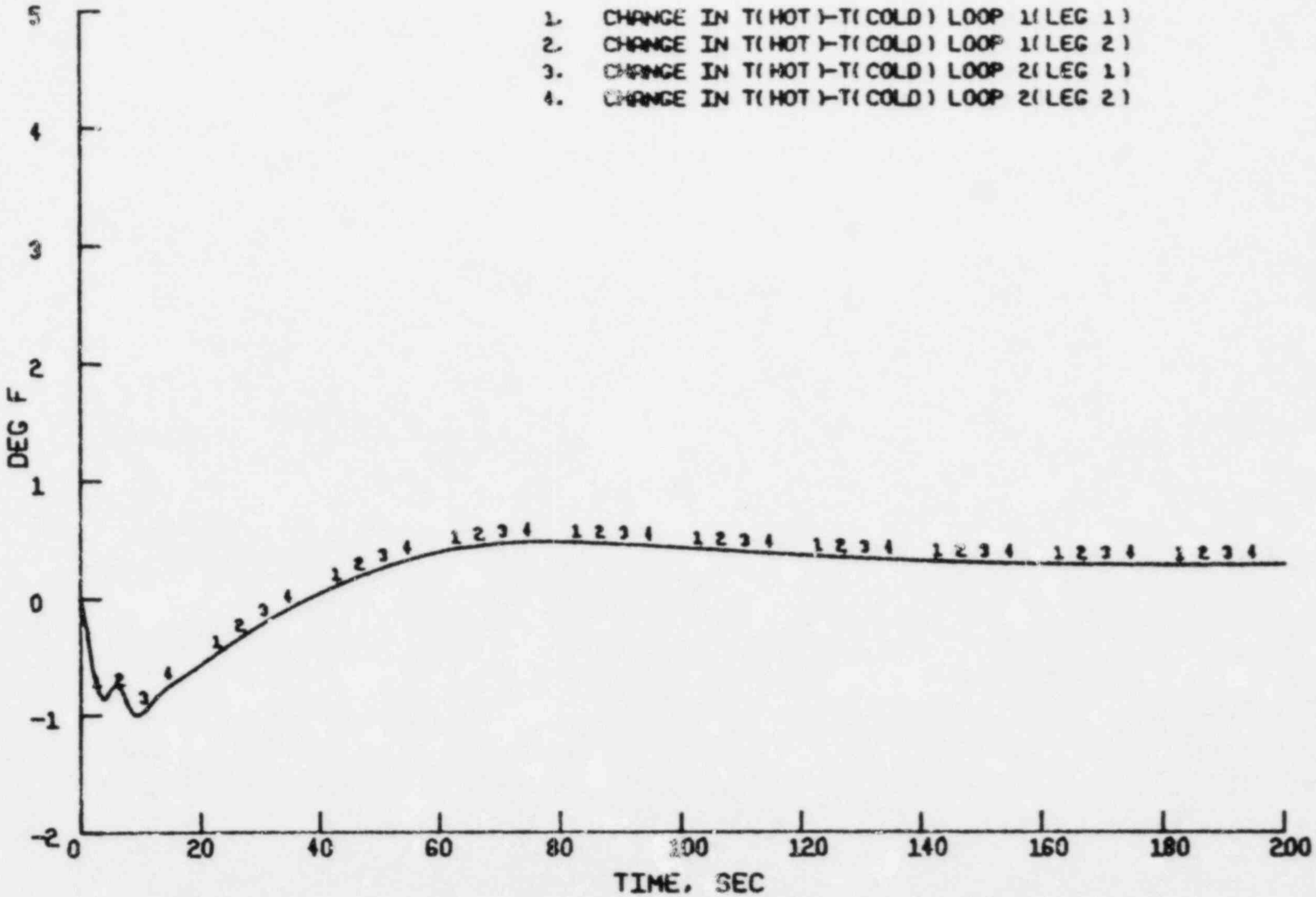

Figure 3.60 St. Lucie Unit 1 - Loop Temperature Differences - CEA Drop

STLUCLK [illegible] 20.57.52.

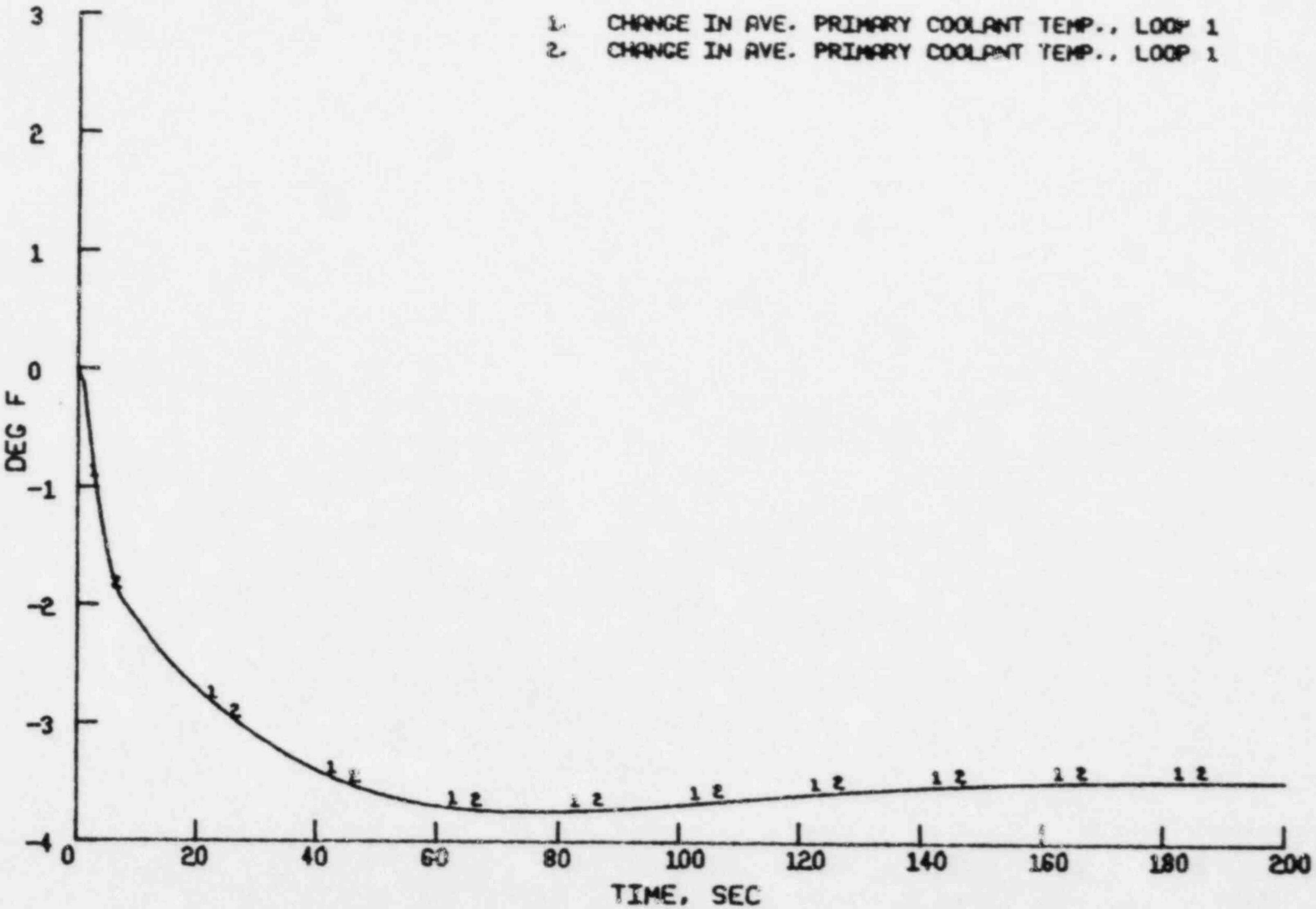

Figure 3.61 St. Lucie Unit 1 - Average Temperatures - CEA Drop

STLUCLK 02/12/82 20.57.52.

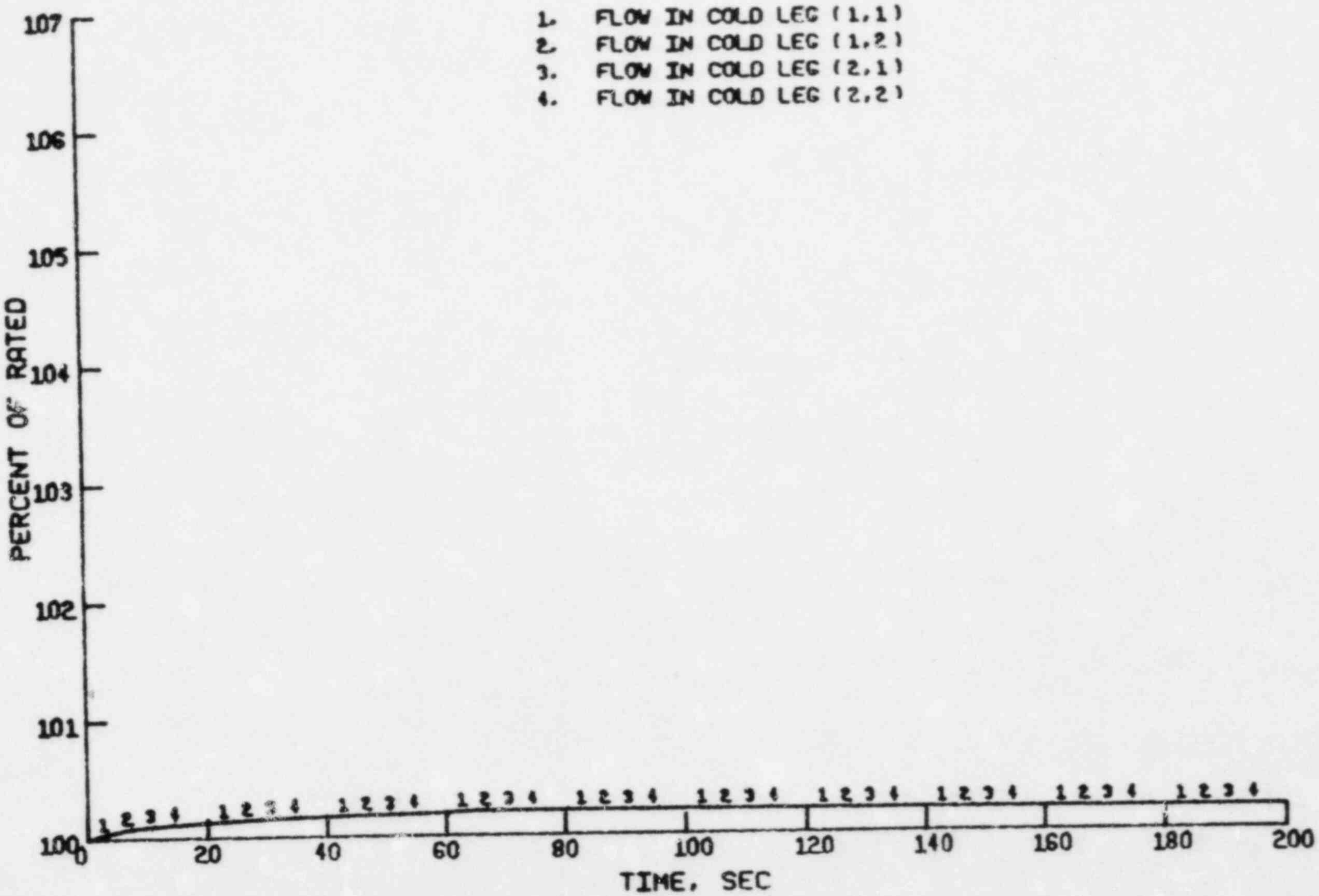

Figure 3.62 St. Lucie Unit 1 - Cold Leg Flows - CEA Drop

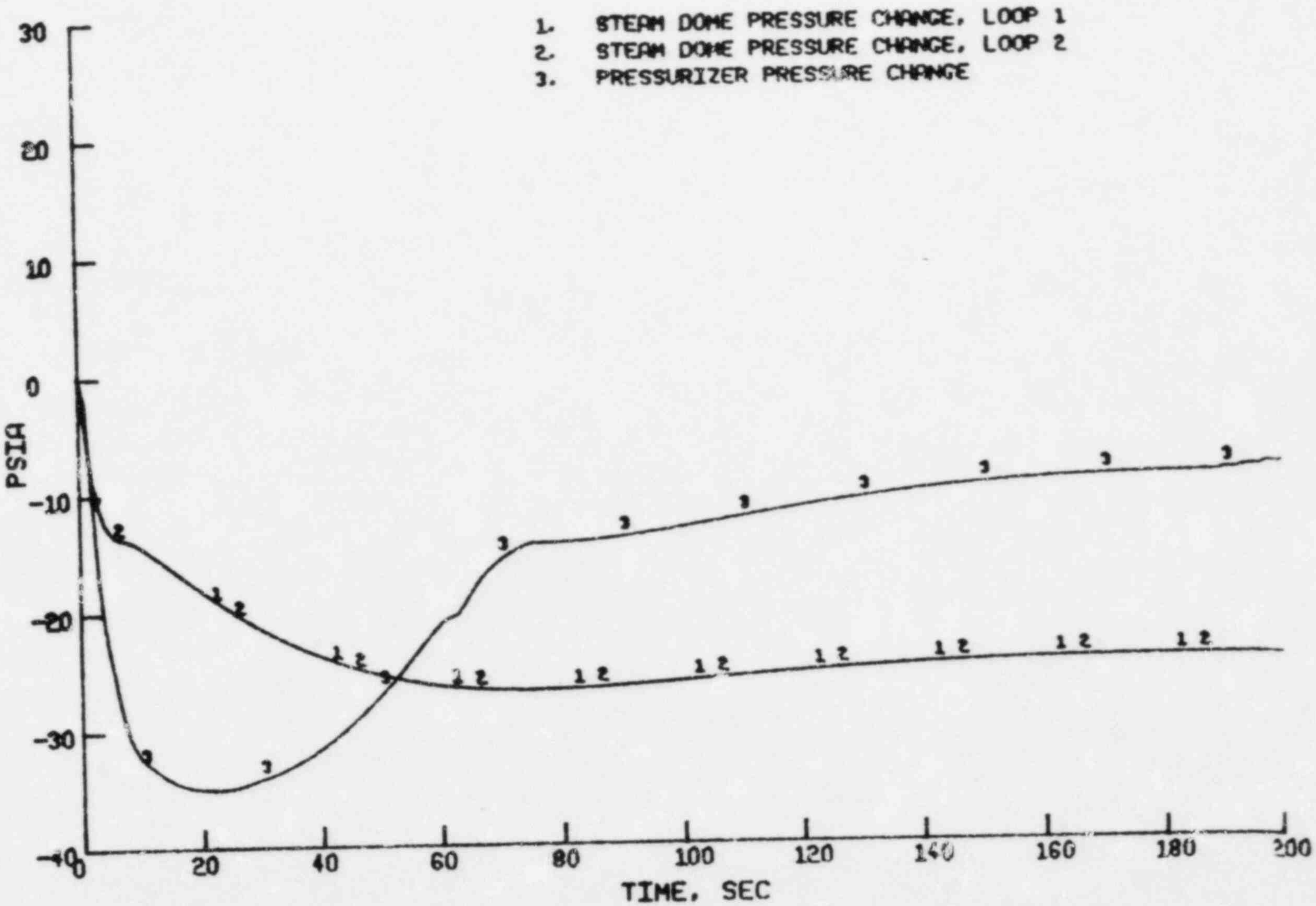

Figure 3.63 St. Lucie Unit 1 - Pressures - CEA Drop

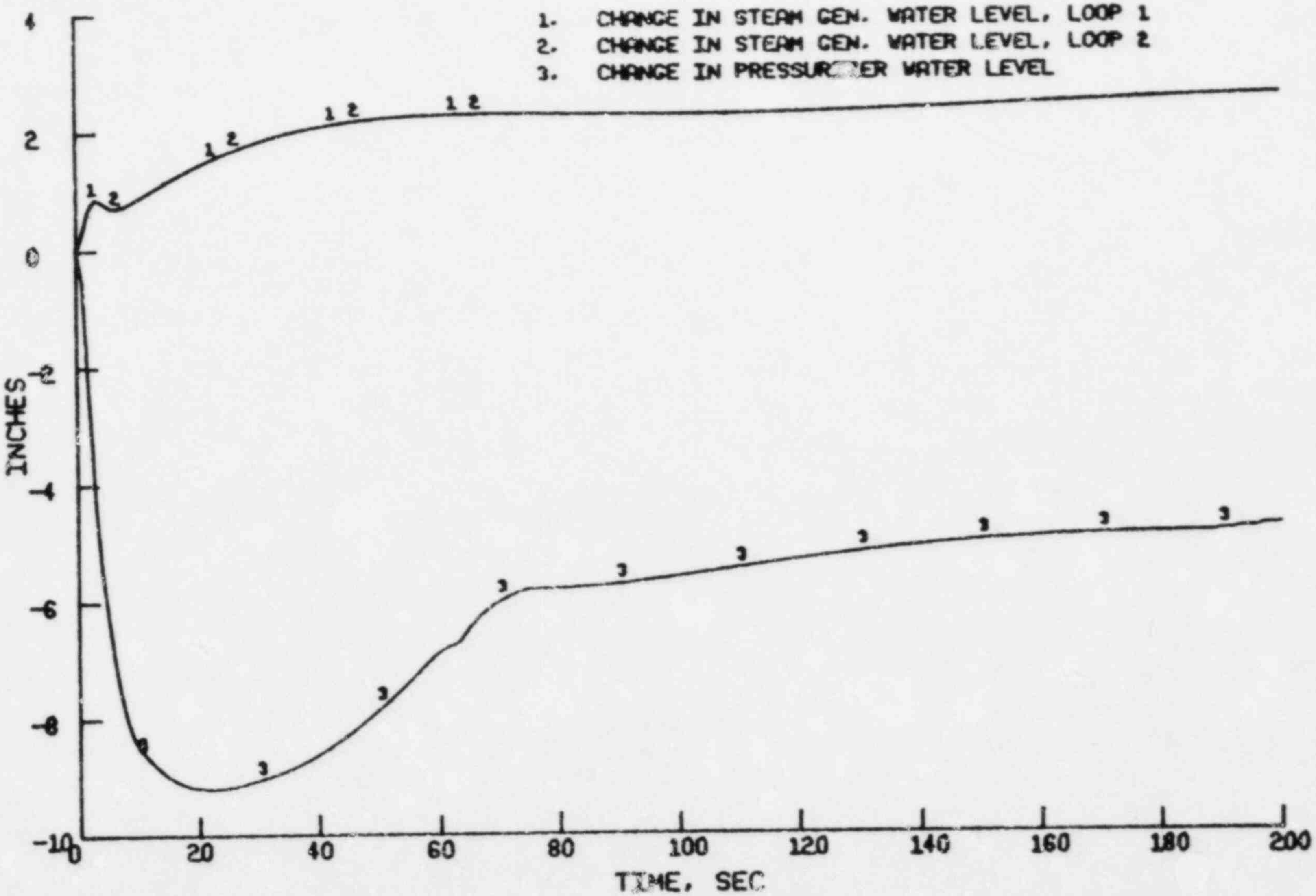

Figure 3.64 St. Lucie Unit 1 - Water Levels - CEA Drop

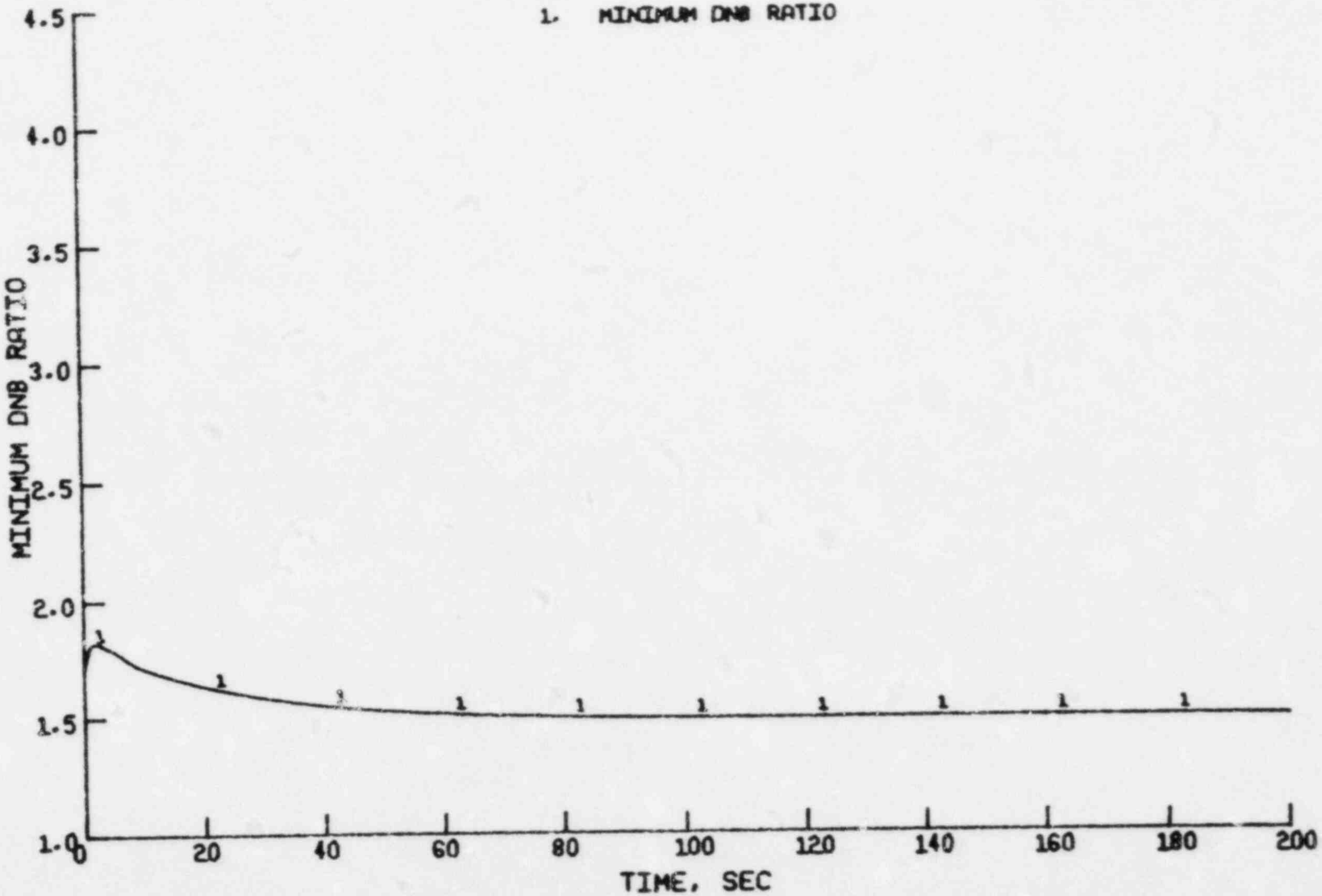

Figure 3.65 St. Lucie Unit 1 - DNBR - CEA Drop

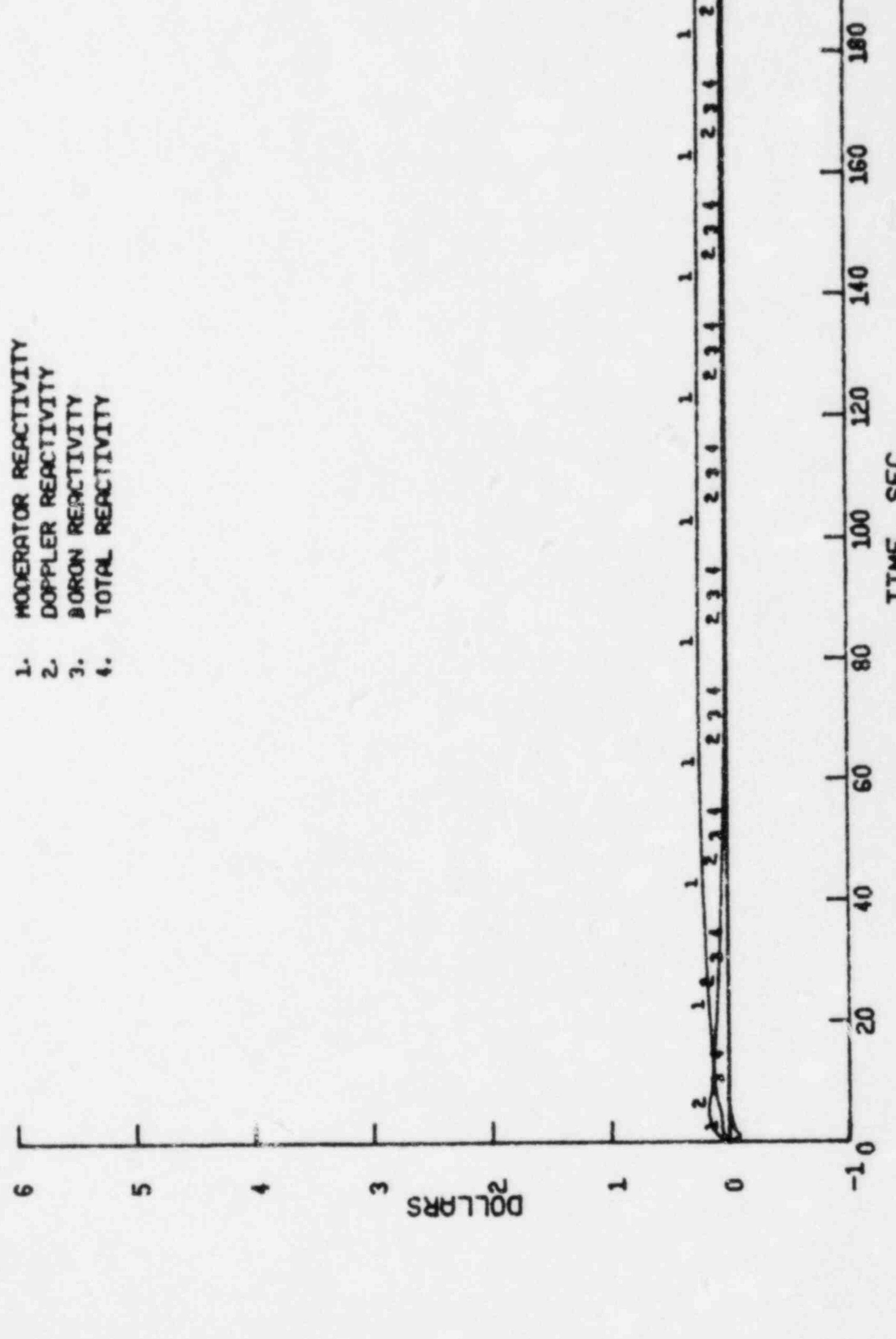

Figure 3.66 St. Lucie Unit 1 - Reactivity - CEA Drop

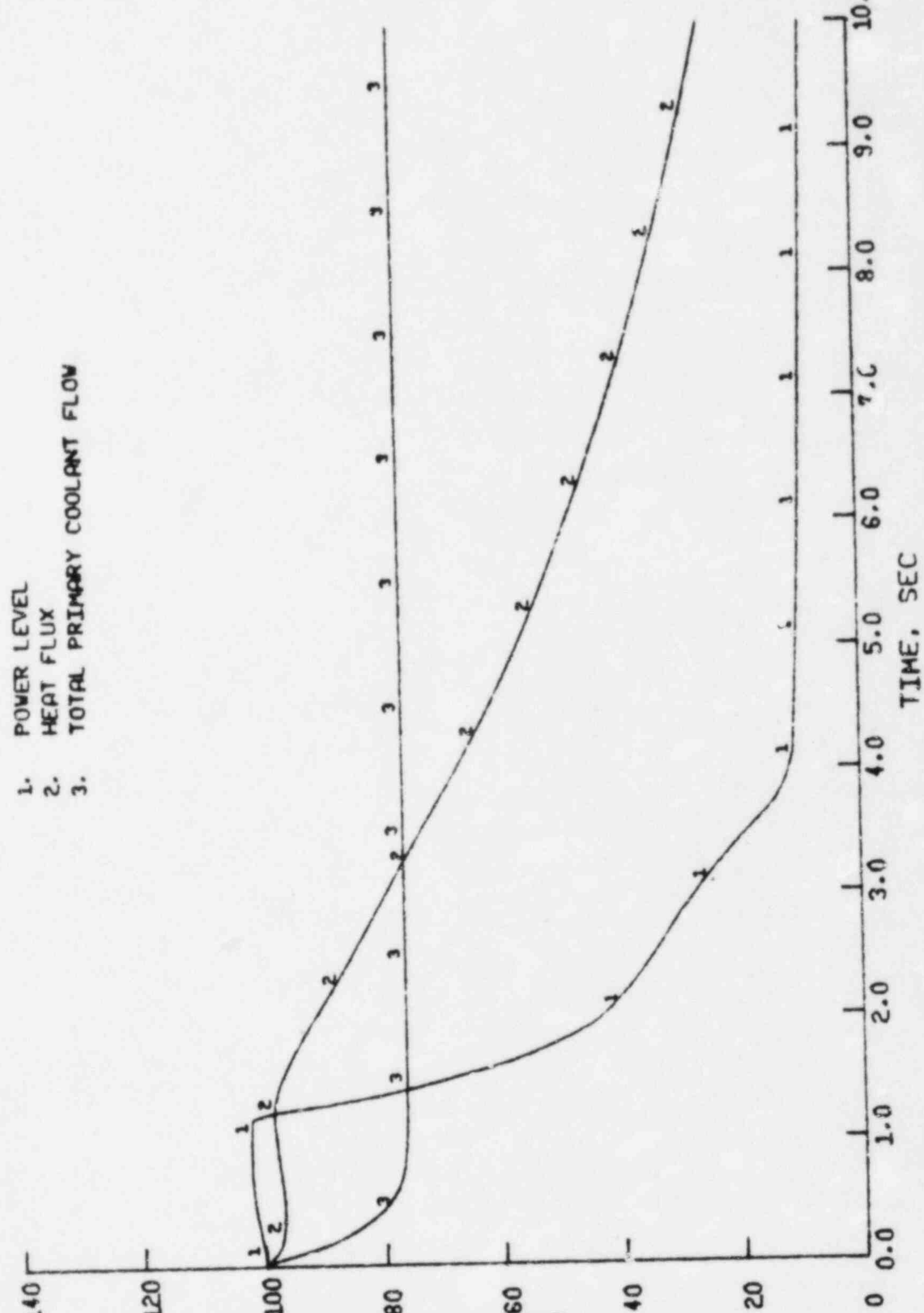

Figure 3.67 St. Lucie Unit 1 - Power, Heat Flux and Flow - Seized Rotor (DNB)

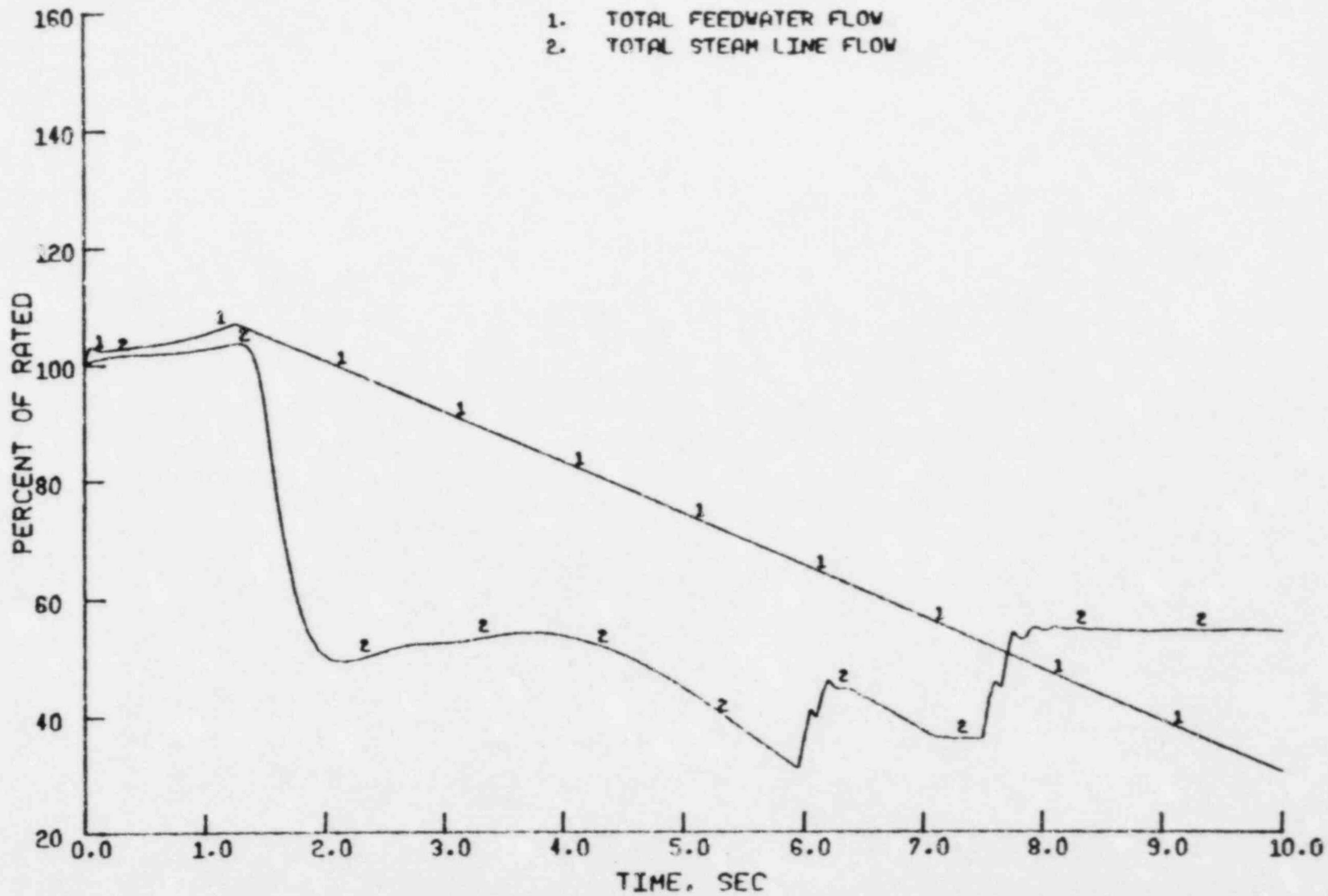

Figure 3.68 St. Lucie Unit 1 - S. G. Flows - Seized Rotor (DNB)

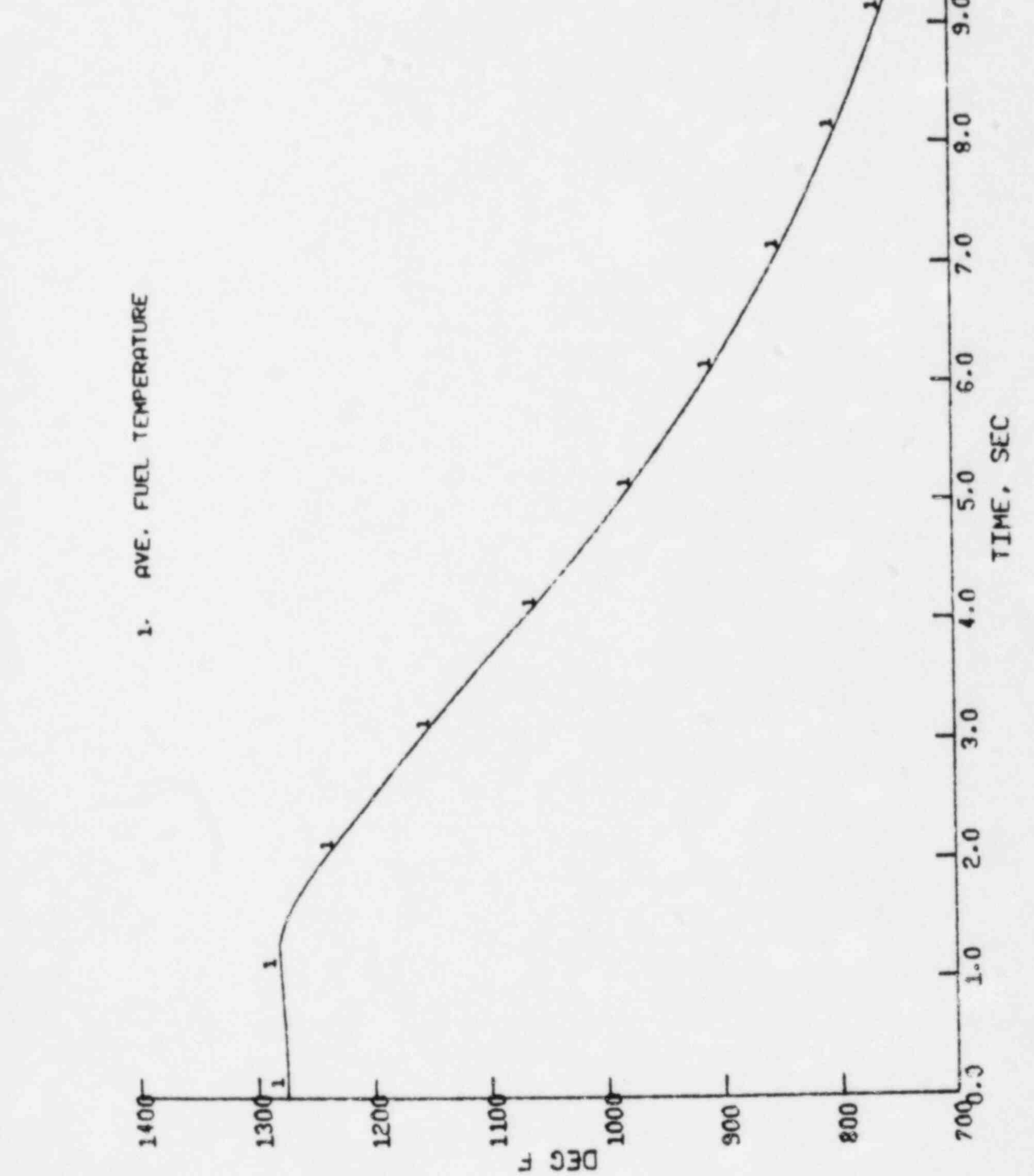

Figure 3.69 St. Lucie Unit 1 - Fuel Temperature - Seized Rotor (DNB)

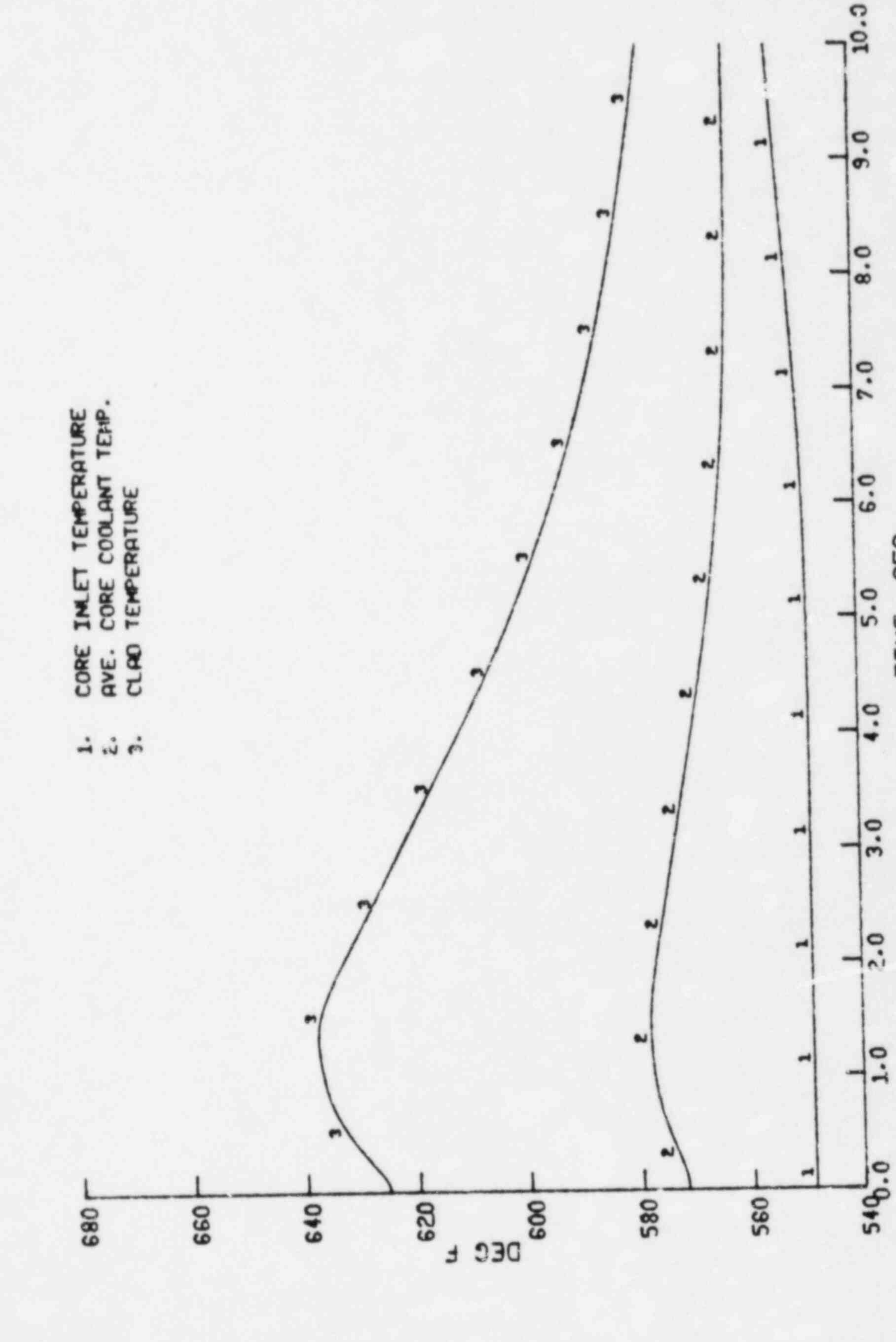

Figure 3.70 St. Lucie Unit 1 - Core Temperatures - Seized Rotor (DNB)

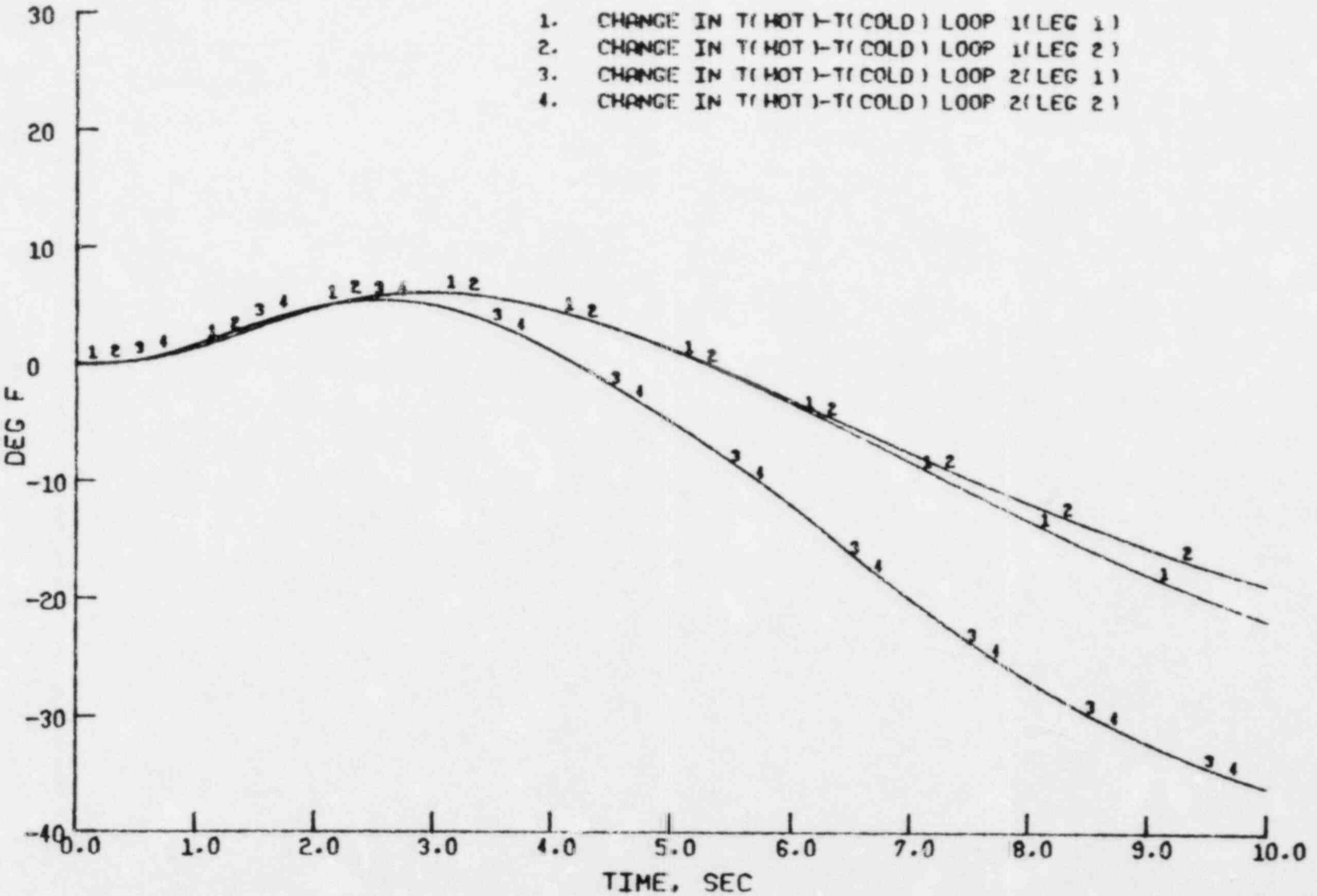

Figure 3.71 St. Lucie Unit 1 - Loop Temperature Differences - Seized Rotor (DNB)

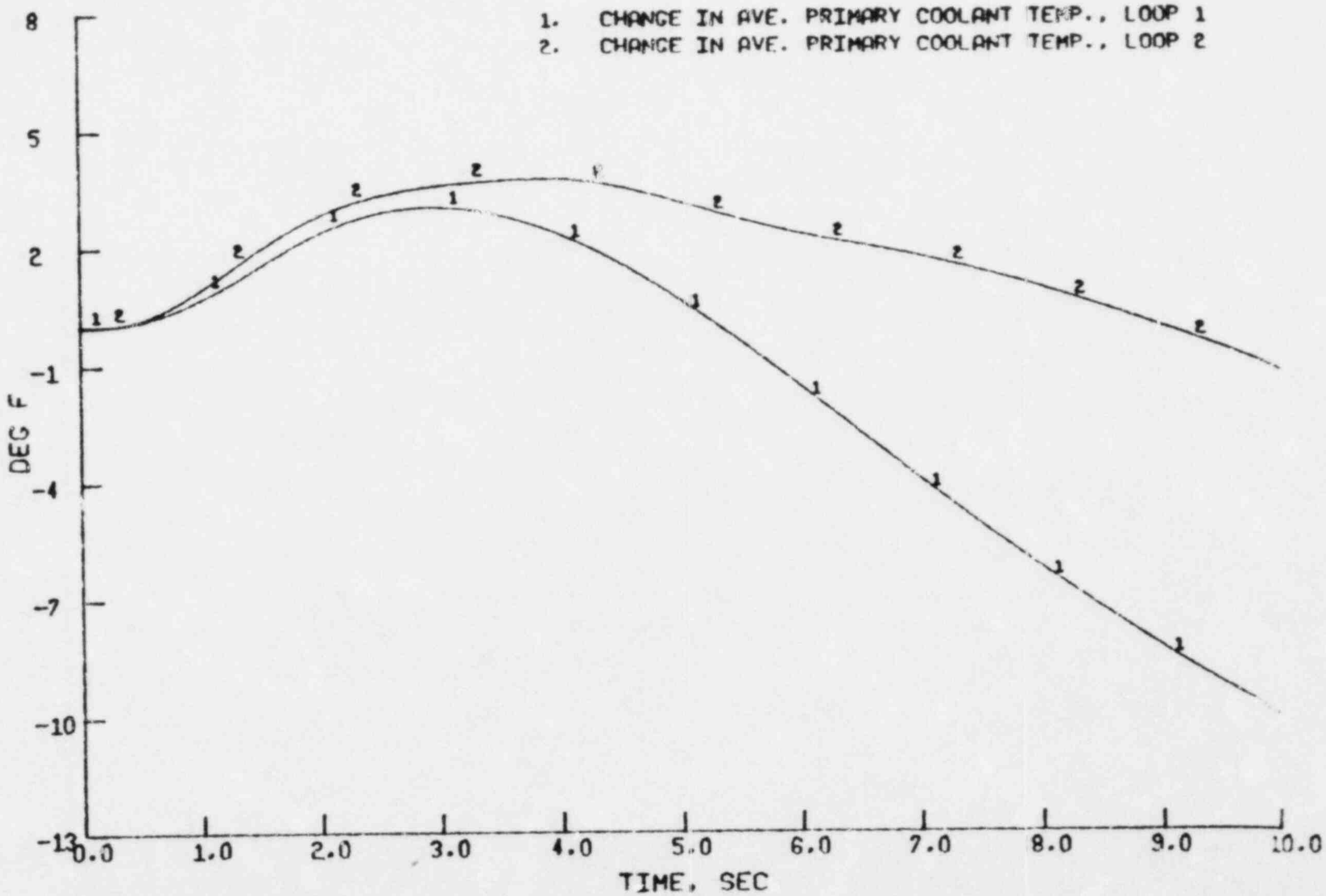

Figure 3.72 St. Lucie Unit 1 - Average Temperatures - Seized Rotor (DNB)

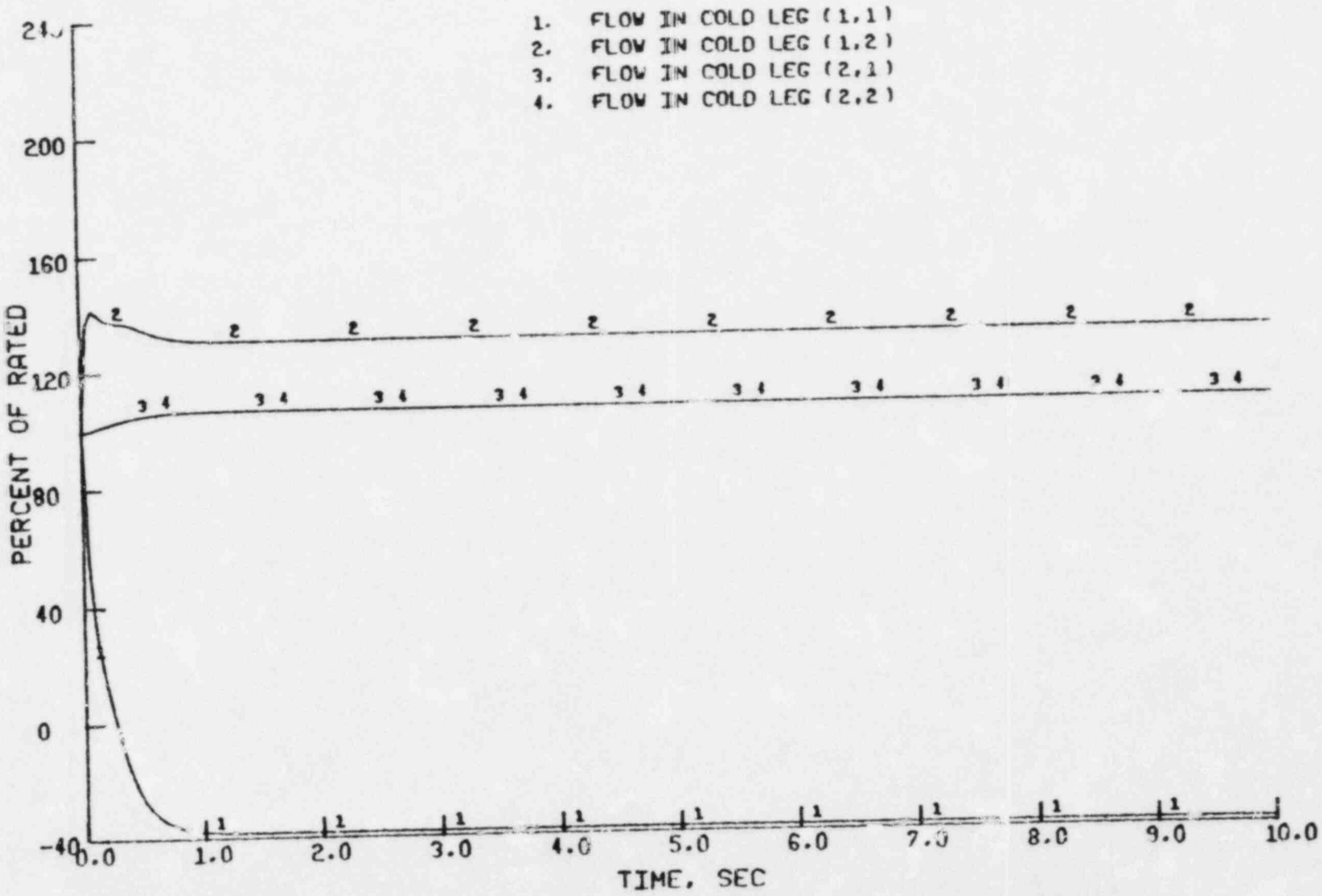

Figure 3.73 St. Lucie Unit 1 - Cold Leg Flows - Seized Rotor (DNB)

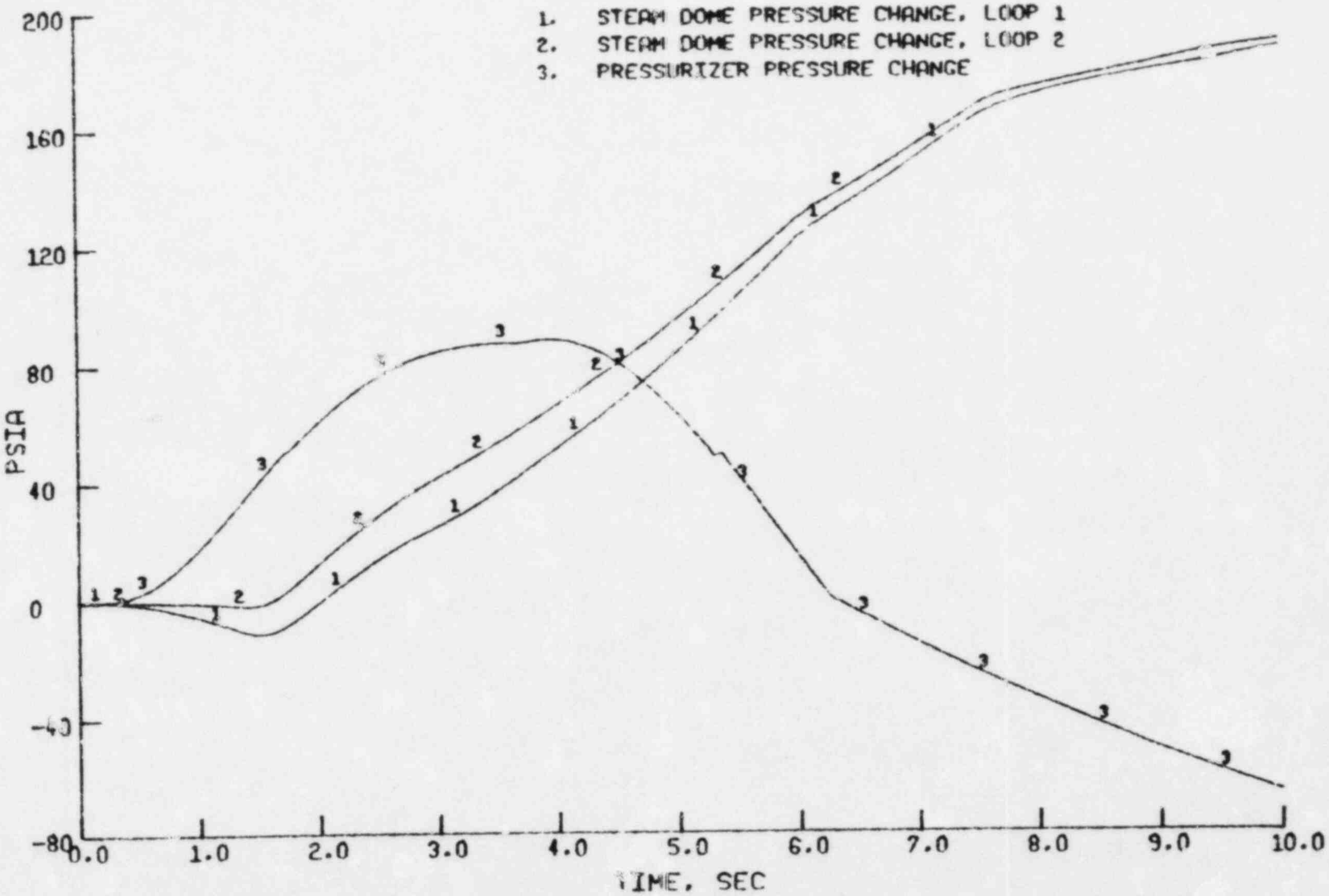

Figure 3.74 St. Lucie Unit 1 - Pressures - Seized Rotor (DNB)

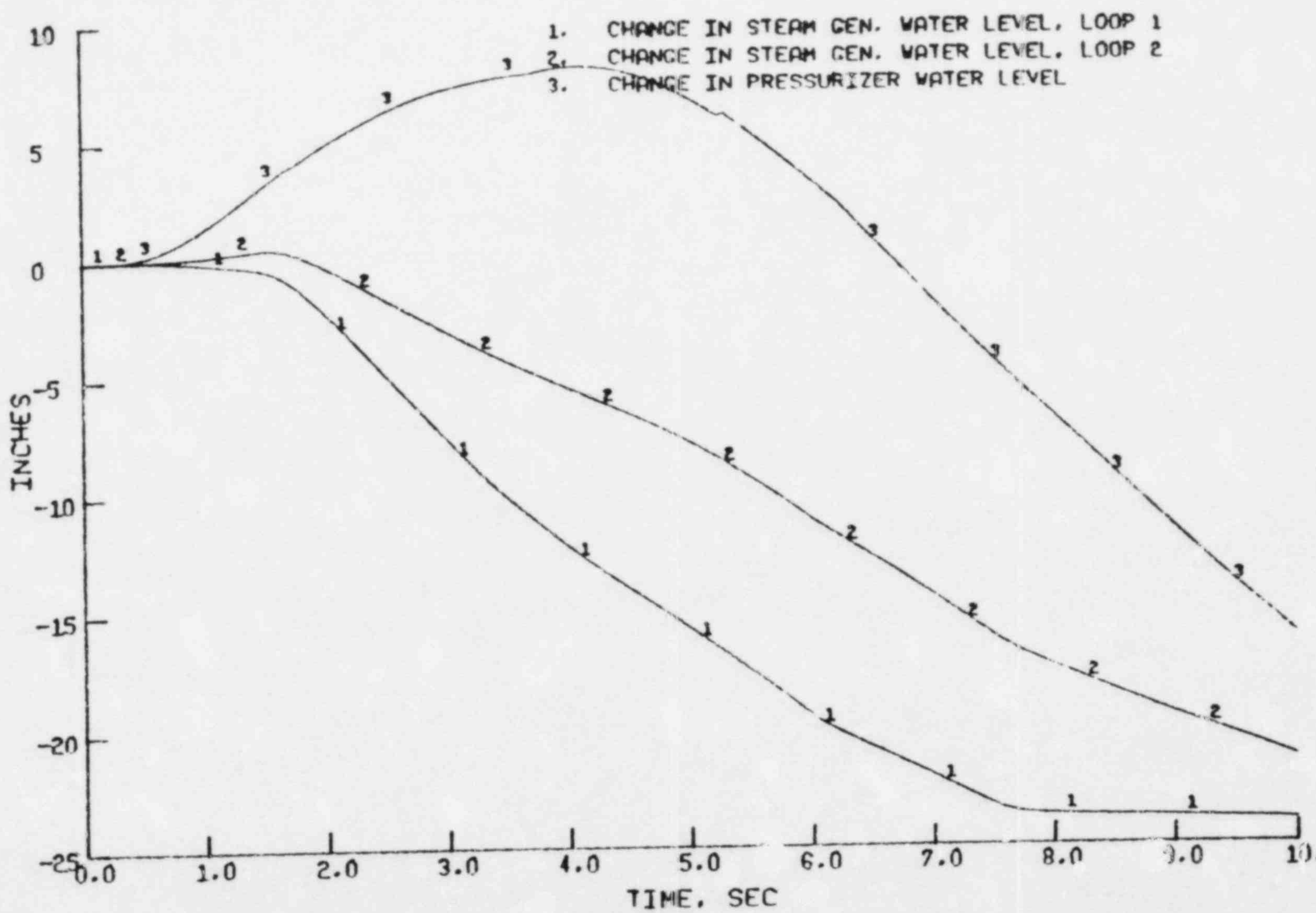

Figure 3.75 St. Lucie Unit 1 - Water Levels - Seized Rotor (DNB)

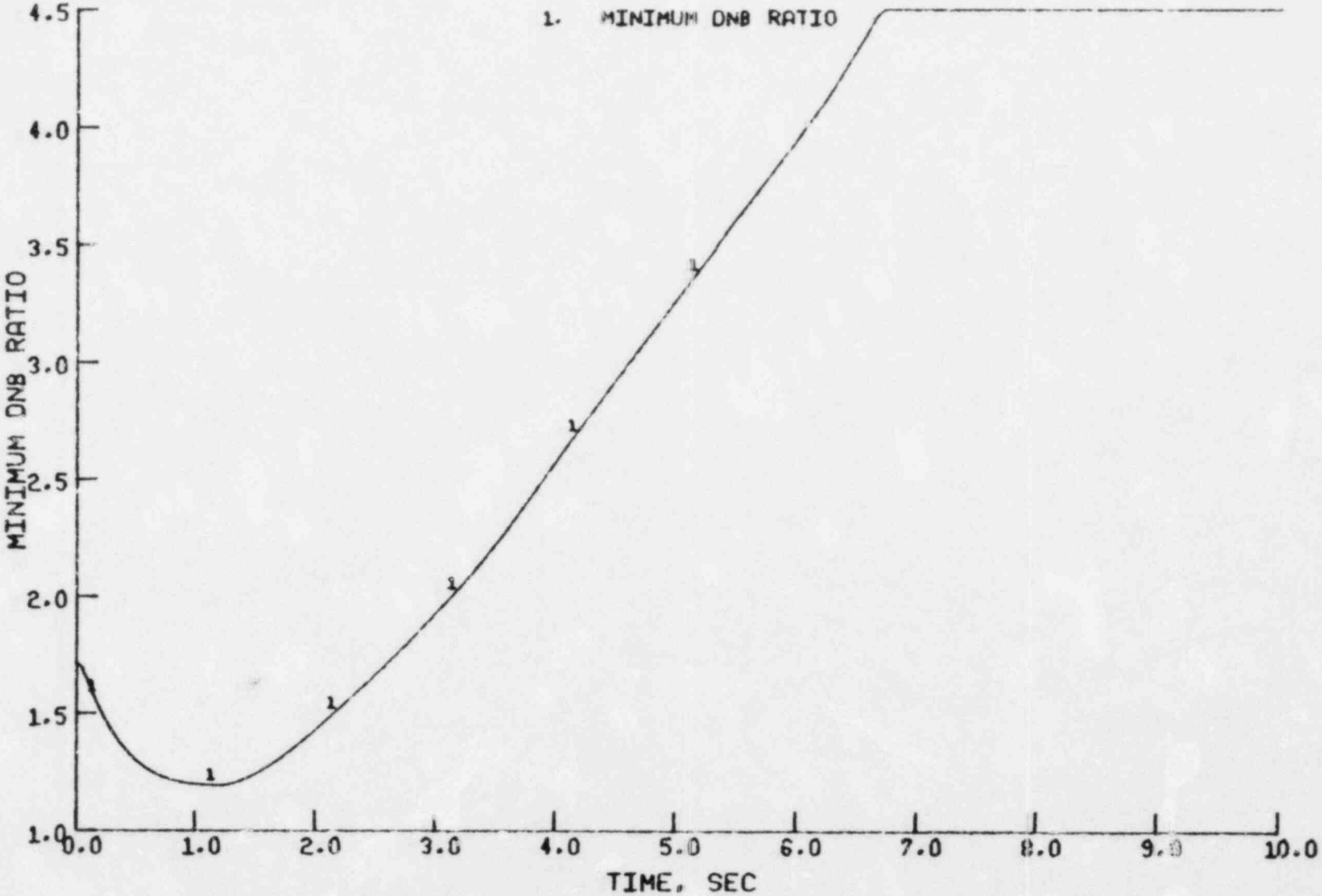

Figure 3.76 St. Lucie Unit 1 - DNBR - Seized Rotor (DNB)

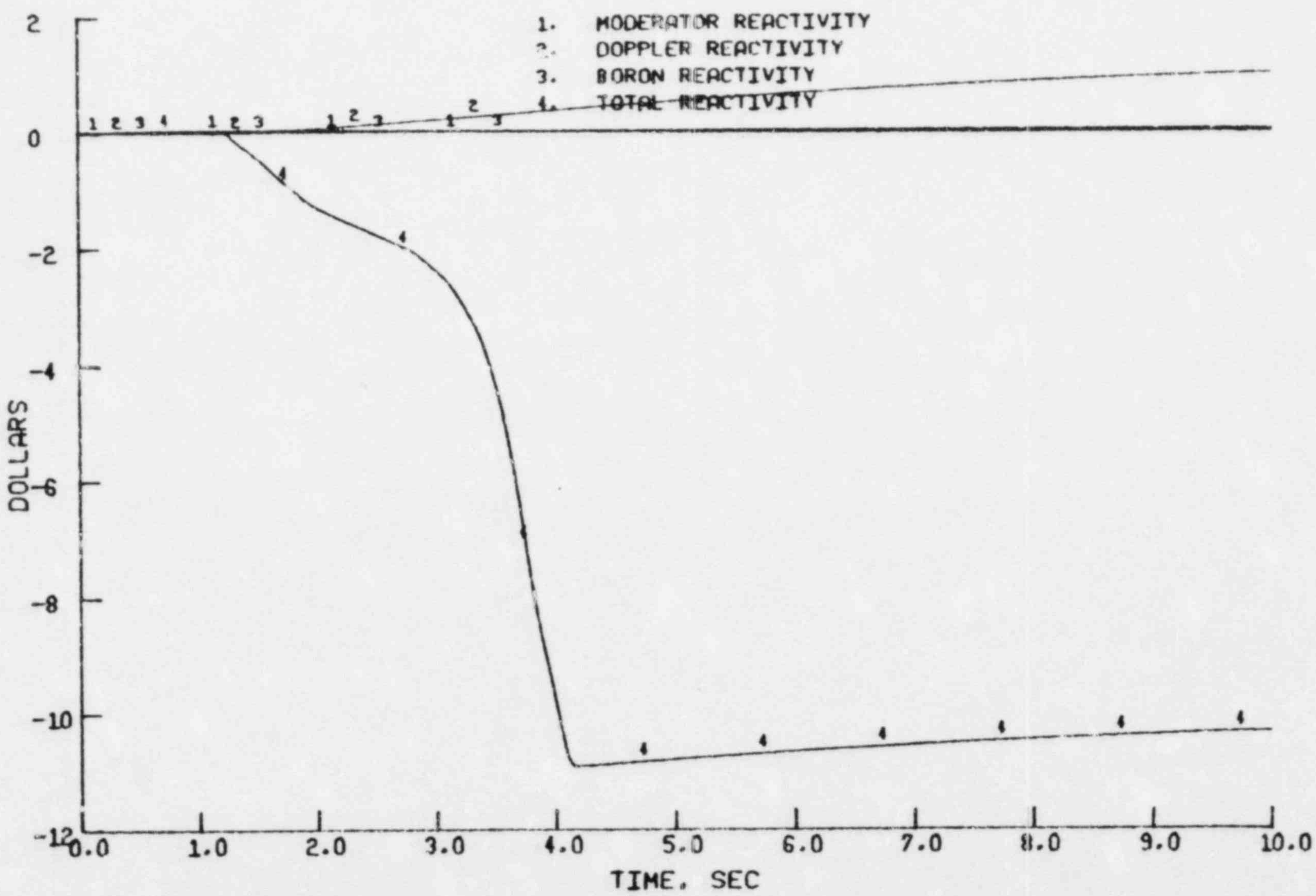

Figure 3.77 St. Lucie Unit 1 - Reactivity - Seized Rotor (DNB)

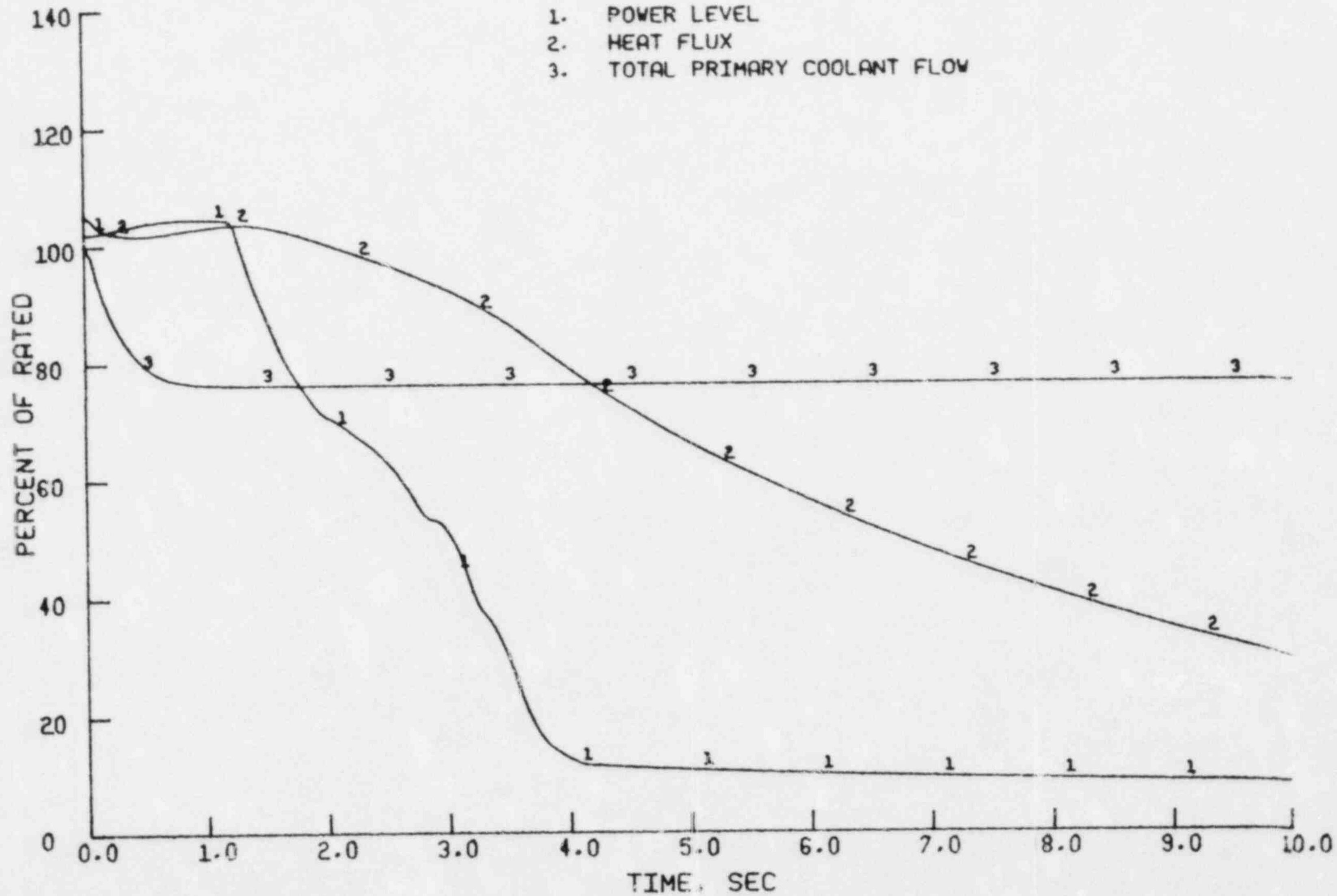

Figure 3.78 St. Lucie Unit 1 - Power, Heat Flux and Flow - Seized Rotor (Pressure)

STLUCND 23/12/82 21.32.51.

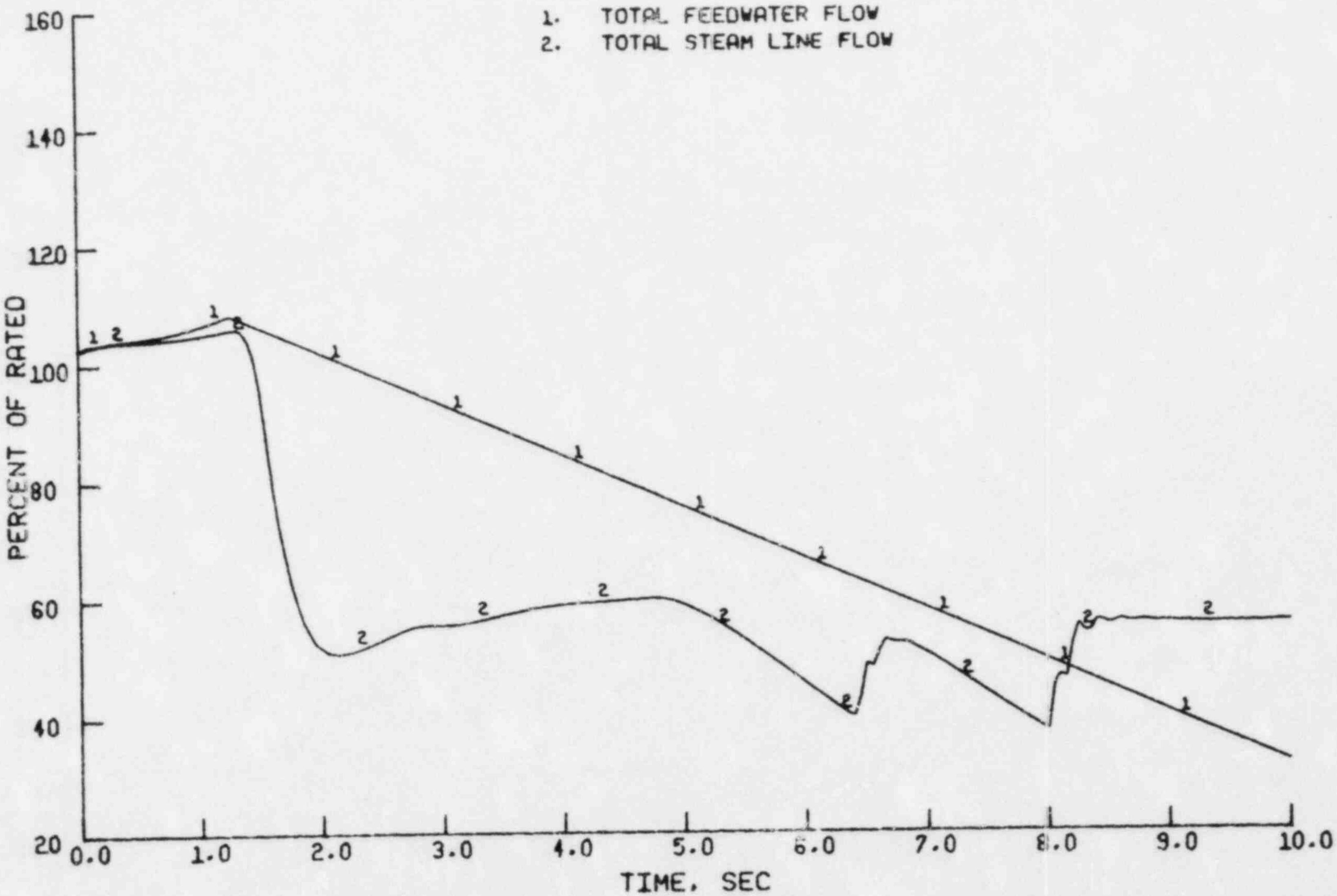

Figure 3.79 St. Lucie Unit 1 - S. G. Flows - Seized Rotor (Pressure)

STLUCNO 23/12/82 21.32.51.

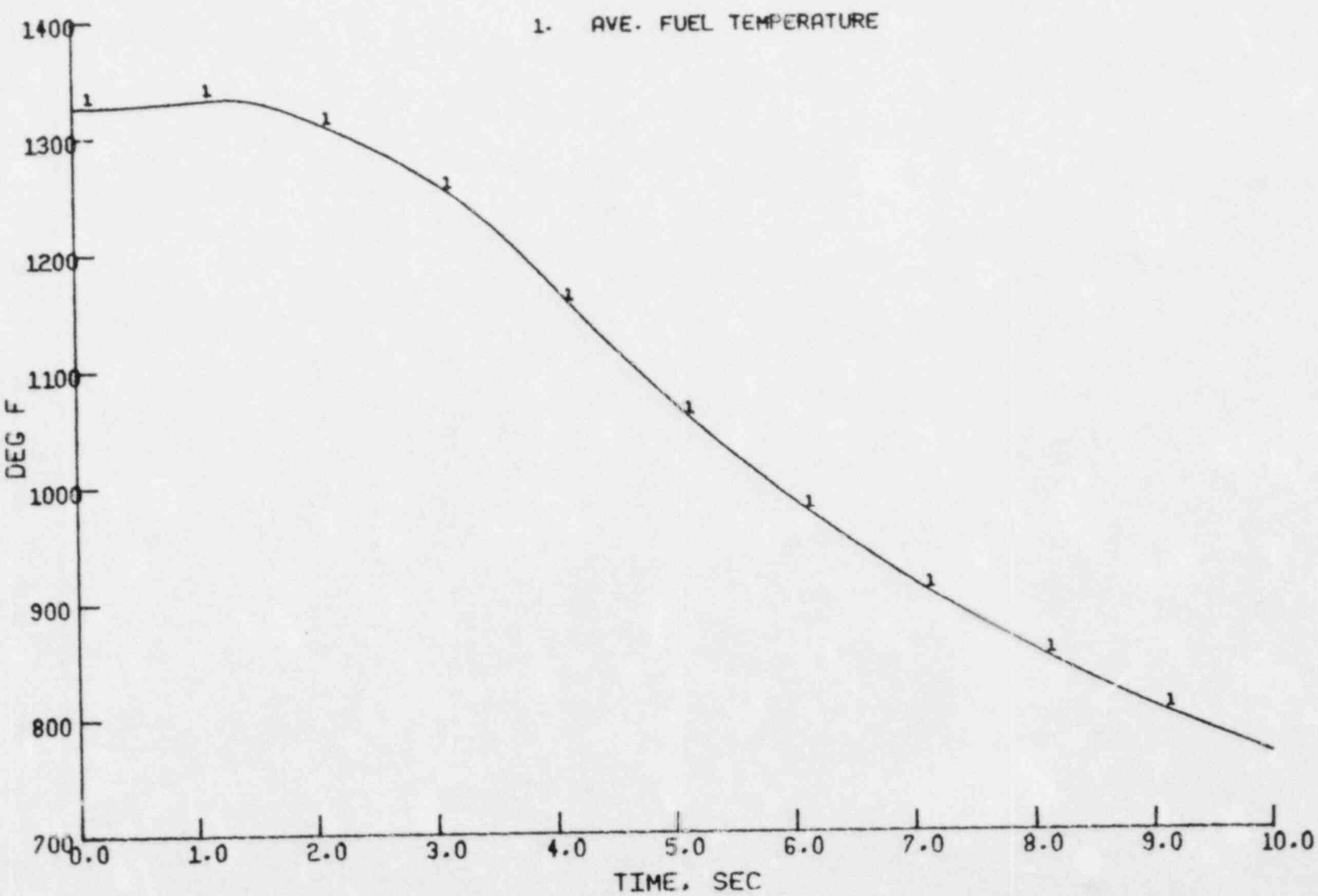

Figure 3.79 St. Lucie Unit 1 - Fuel Temperature - Seized Rotor (Pressure)

STLUCND 23/12/82 21.32.51.

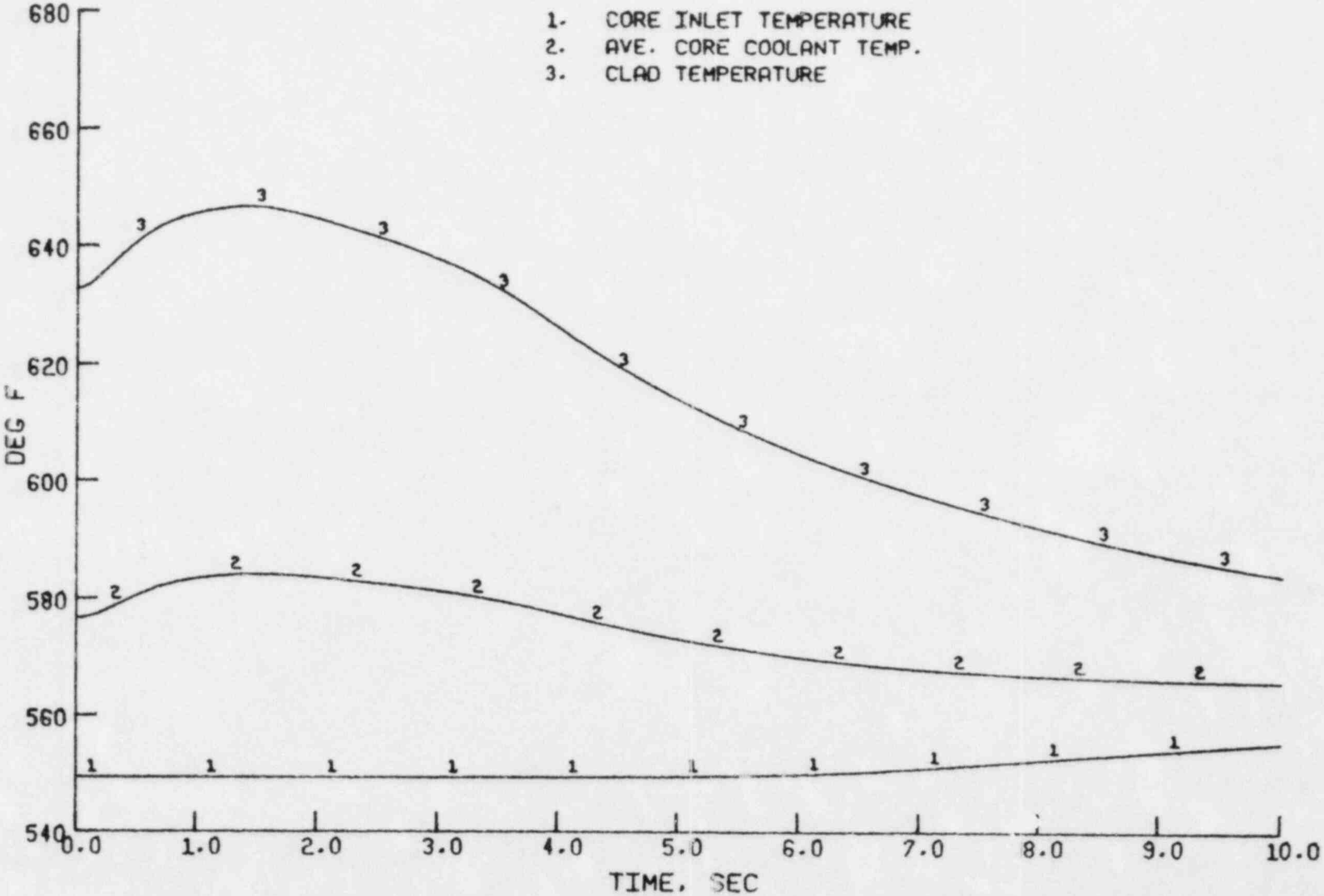

Figure 3.81 St. Lucie Unit 1 - Core Temperatures - Seized Rotor (Pressure)

STLUCND 23/12/82 21.32.51.

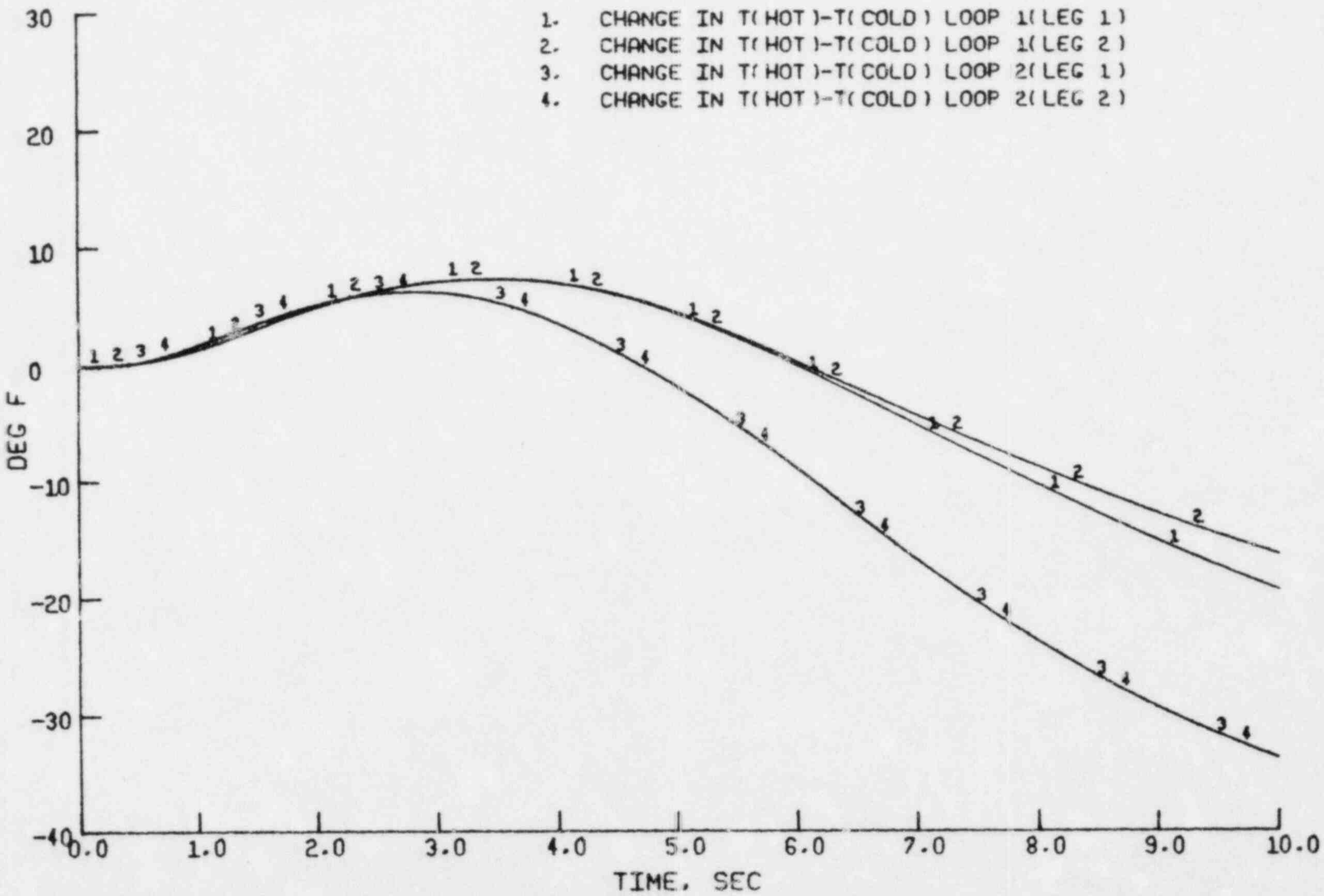

Figure 3.82 St. Lucie Unit 1 - Loop Temperature Differences - Seized Rotor (Pressure)

STLUCND 03/12/82 21.32.51.

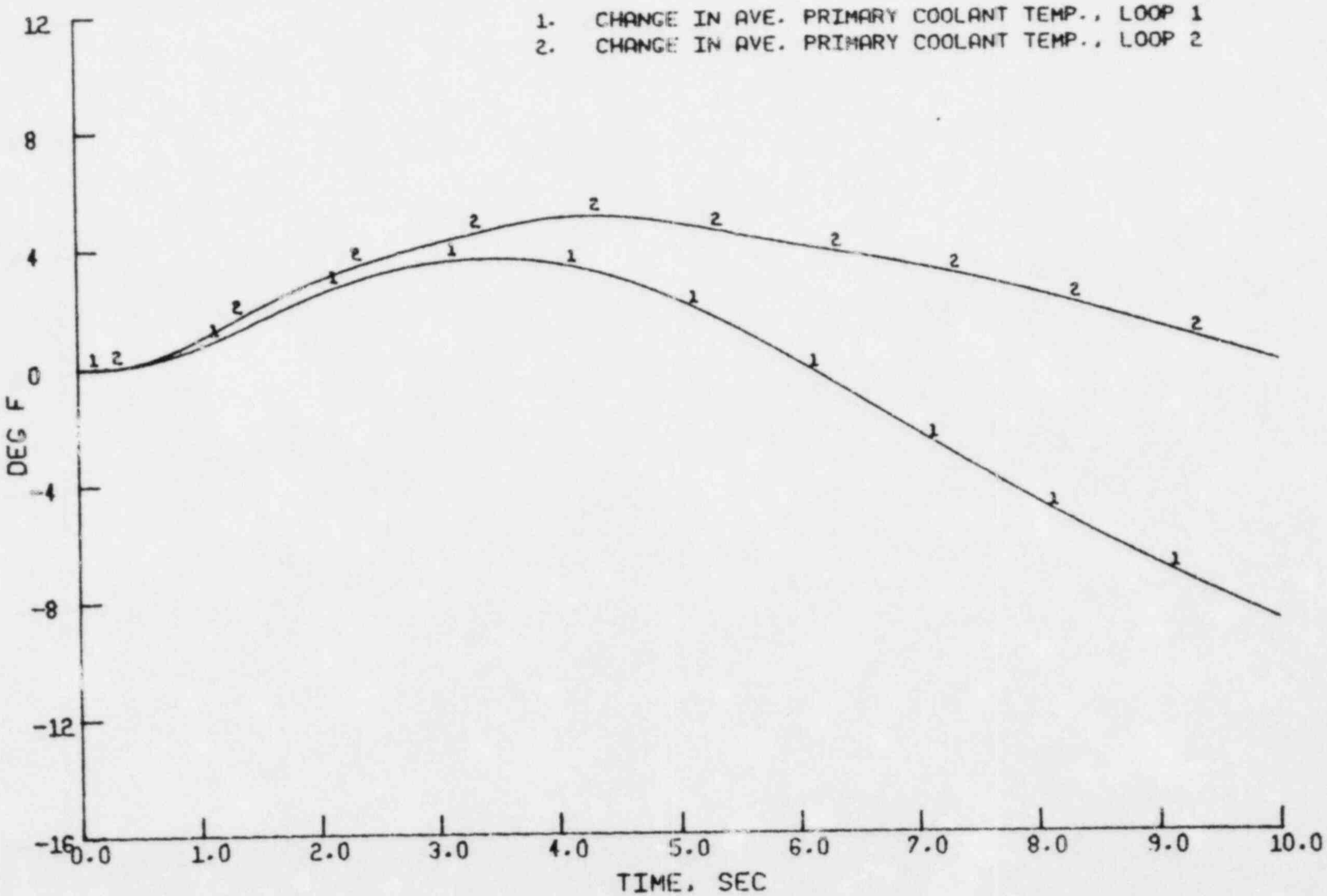

Figure 3.83 St. Lucie Unit 1 - Average Temperatures - Seized Rotor (Pressure)

STLUCND 23/12/82 21.32.51.

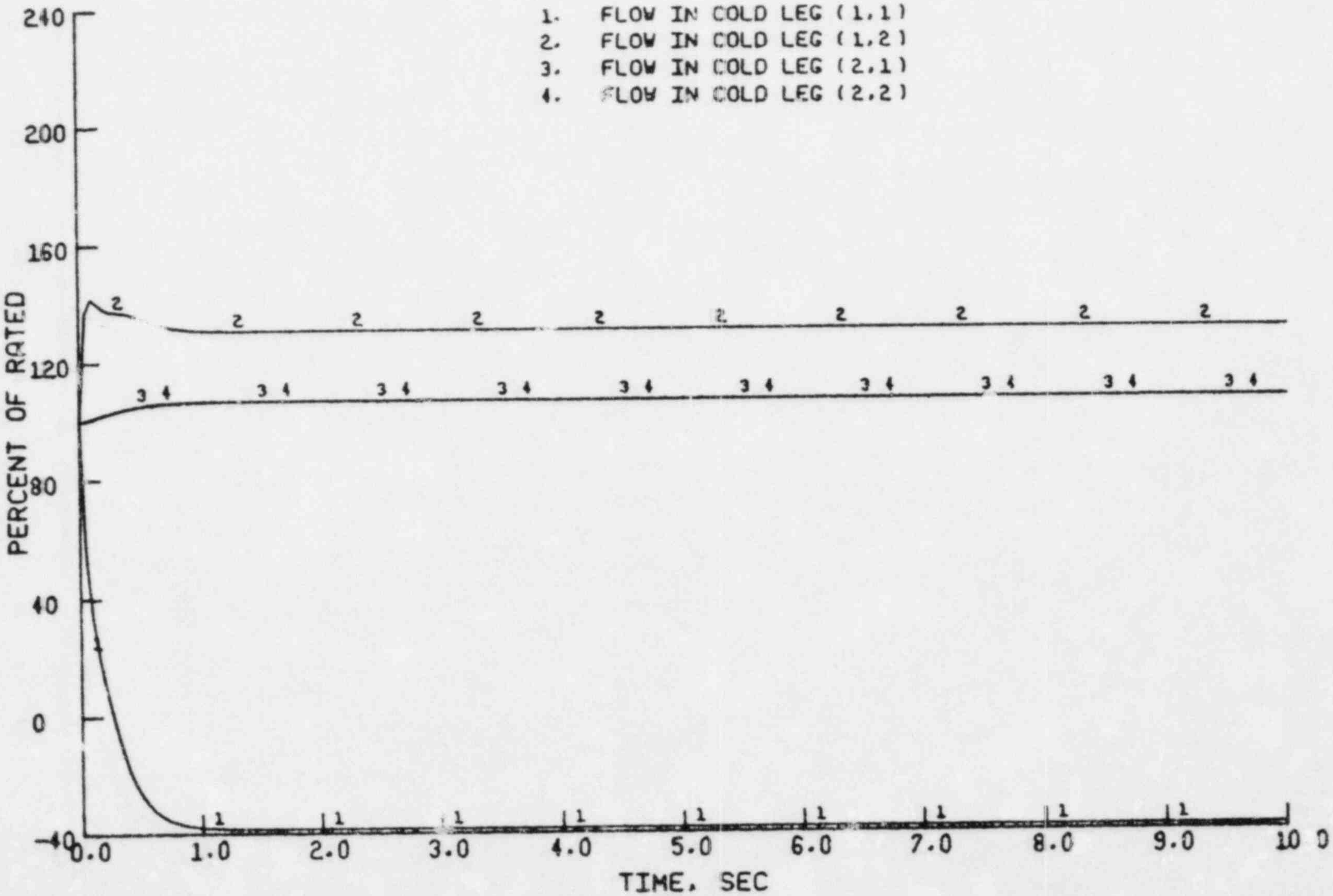

Figure 3.84 St. Lucie Unit 1 - Cold Leg Flows - Seized Rotor (Pressure)

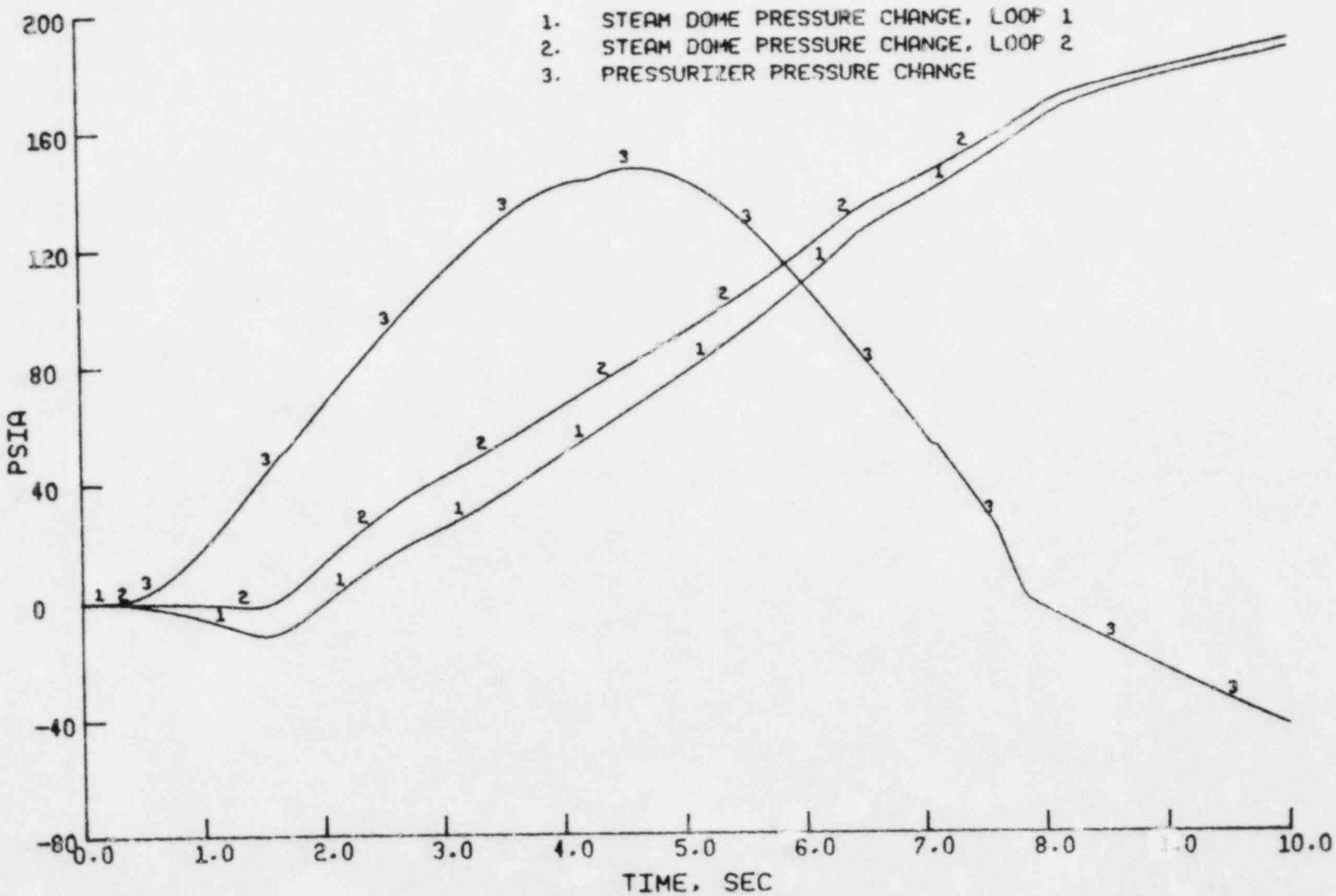

Figure 3.85 St. Lucie Unit 1 - Pressures - Seized Rotor (Pressure)

STLUCND 23/12/82 21.32.51.

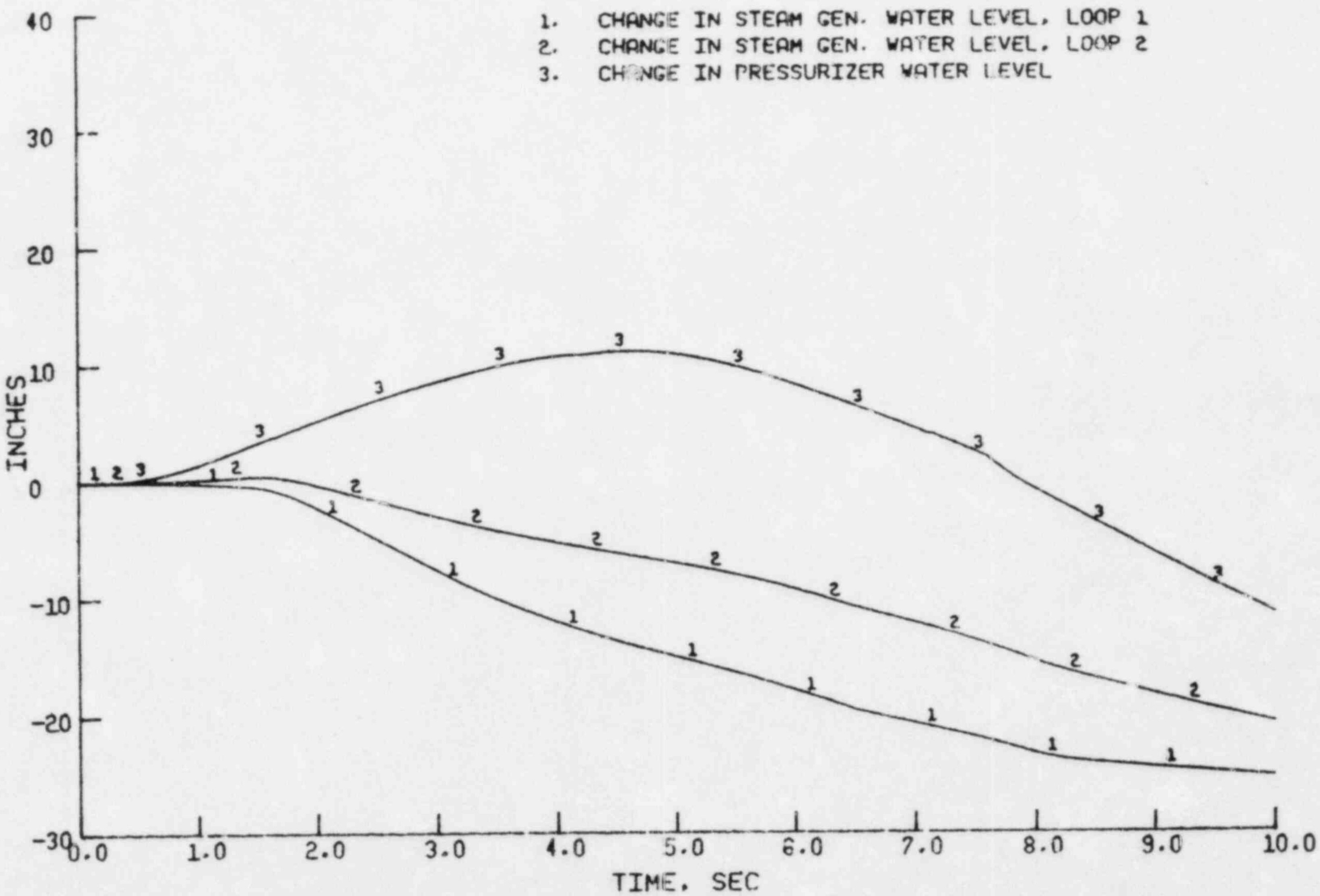

Figure 3.86 St. Lucie Unit 1 - Water Levels - Seized Rotor (Pressure)

STLUCND 23/10/82 21.32.51.

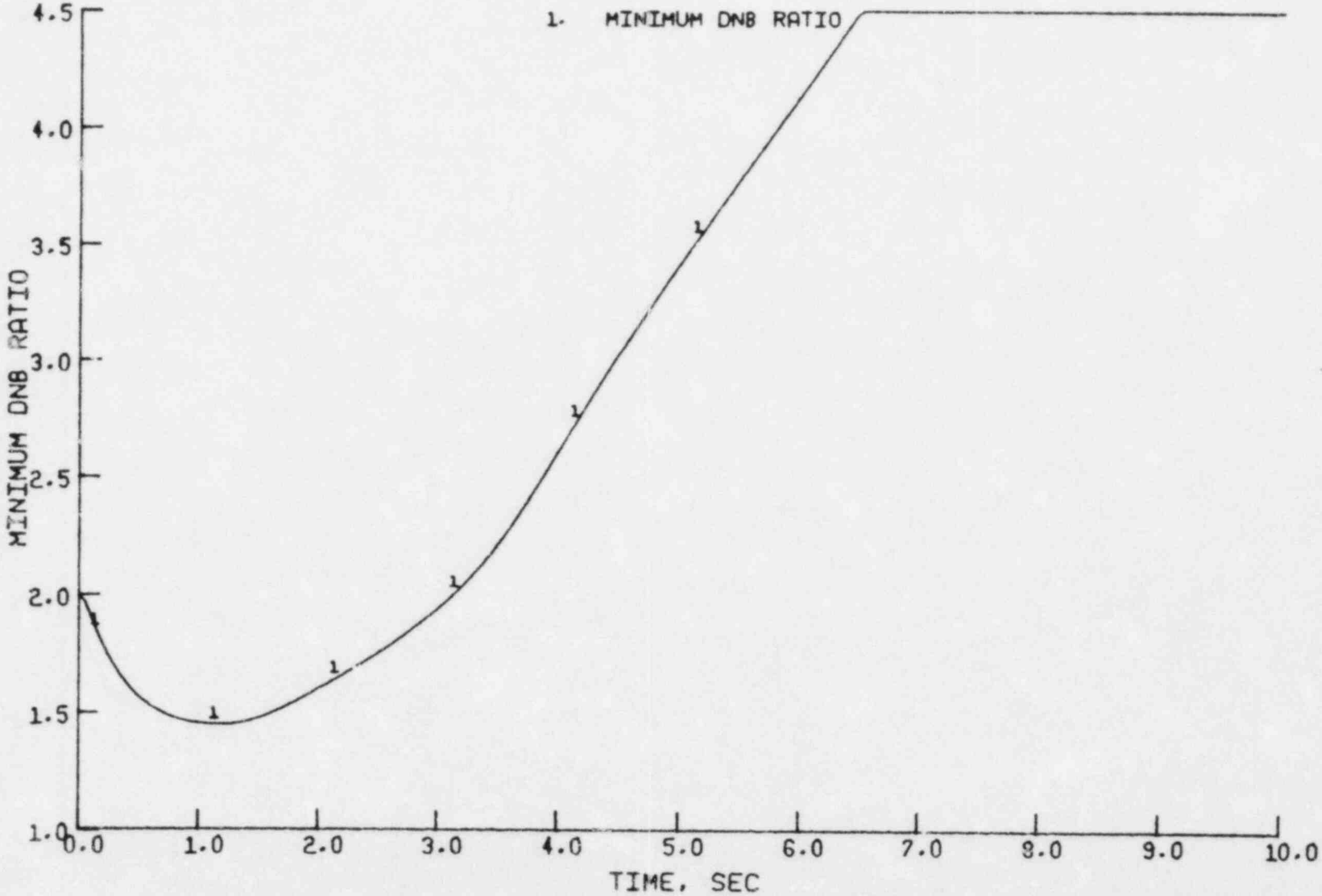

Figure 3.87 St. Lucie Unit 1 - DNBR - Seized Rotor (Pressure)

STLUCND 23/12/82 21.32.51.

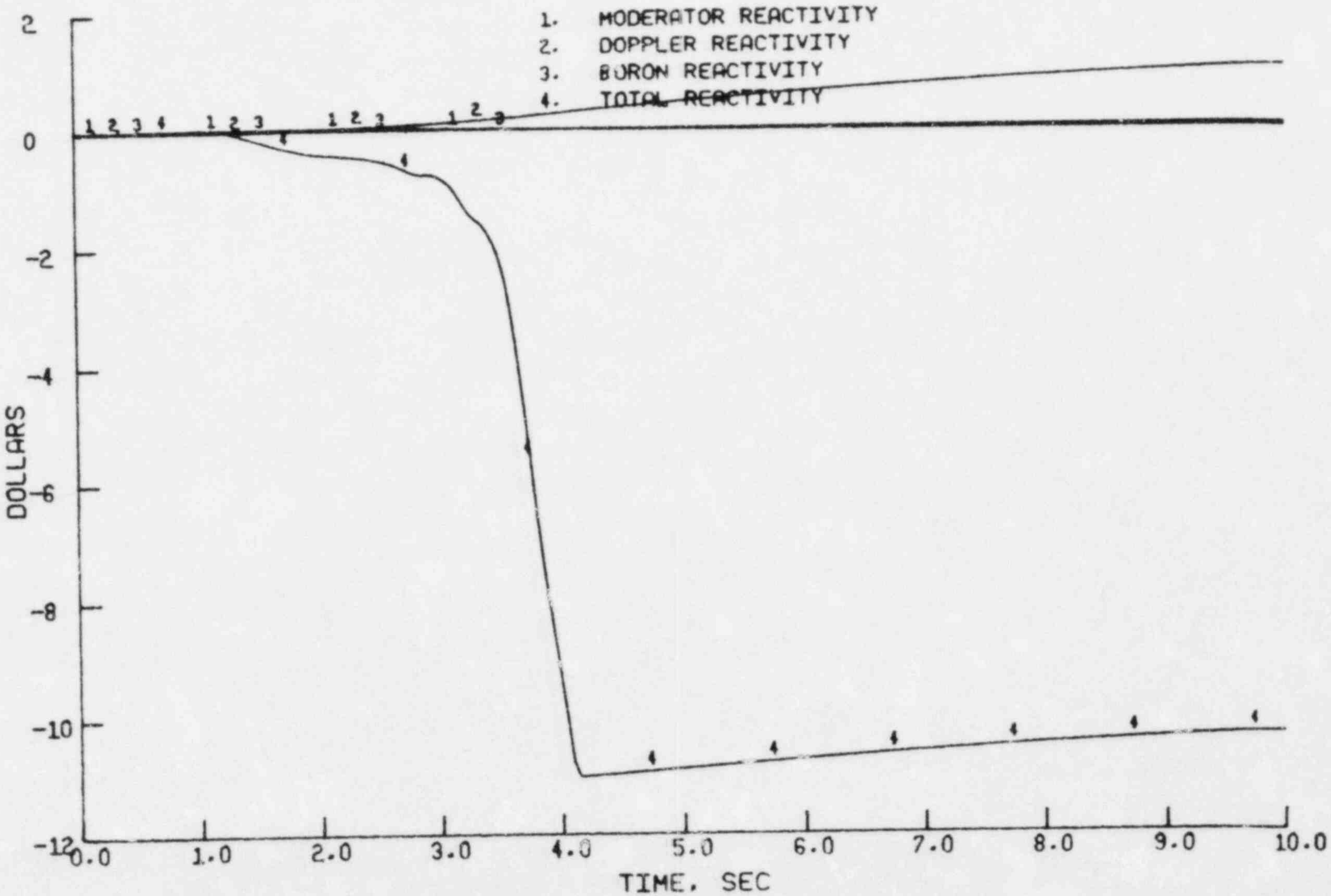

Figure 3.88 St. Lucie Unit 1 - React.vity - Seized Rotor (Pressure)

STLUCND 23/12/82 11.32.51

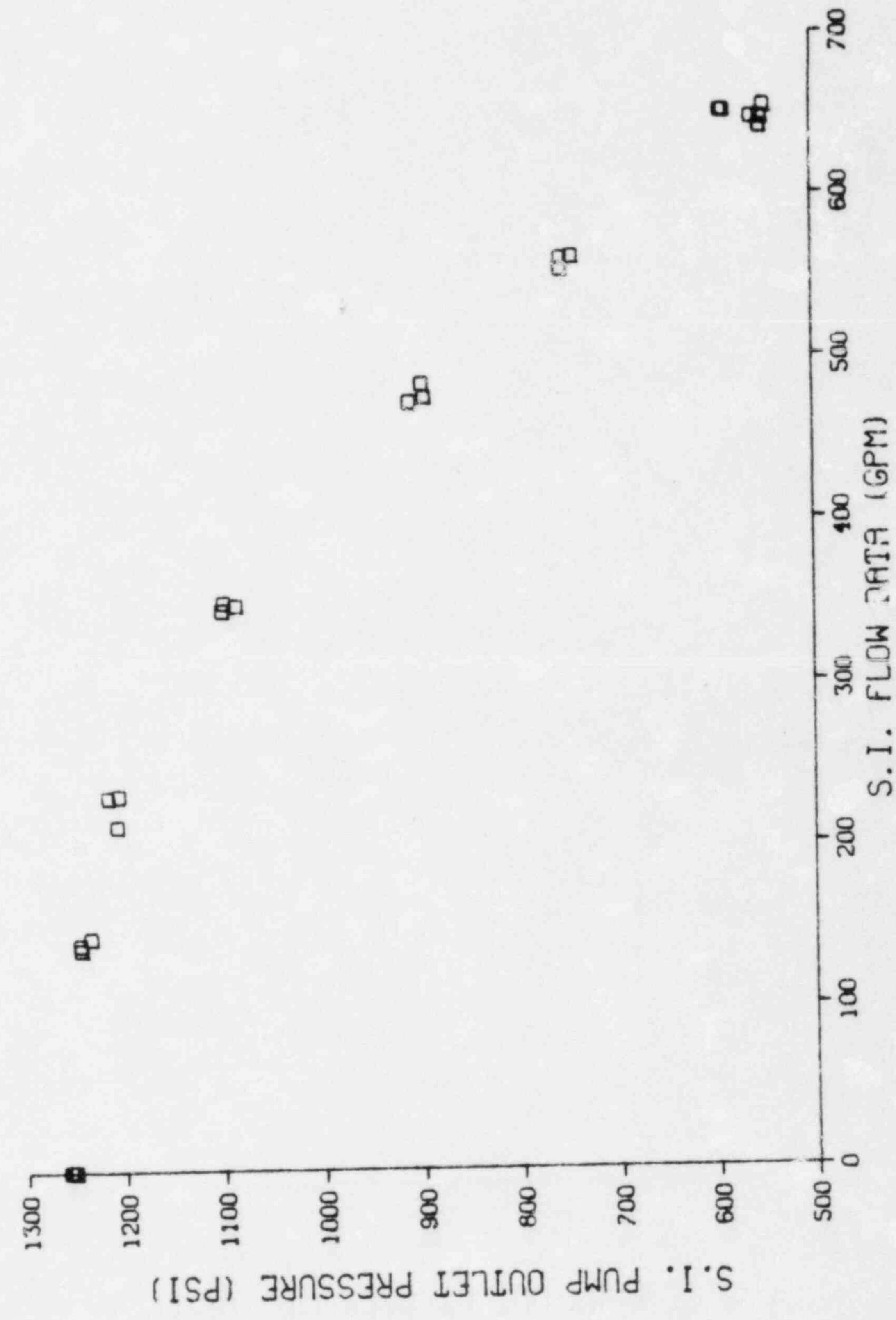

Figure 3.89 RCS Pressure Versus HPSI Flow For One Pump

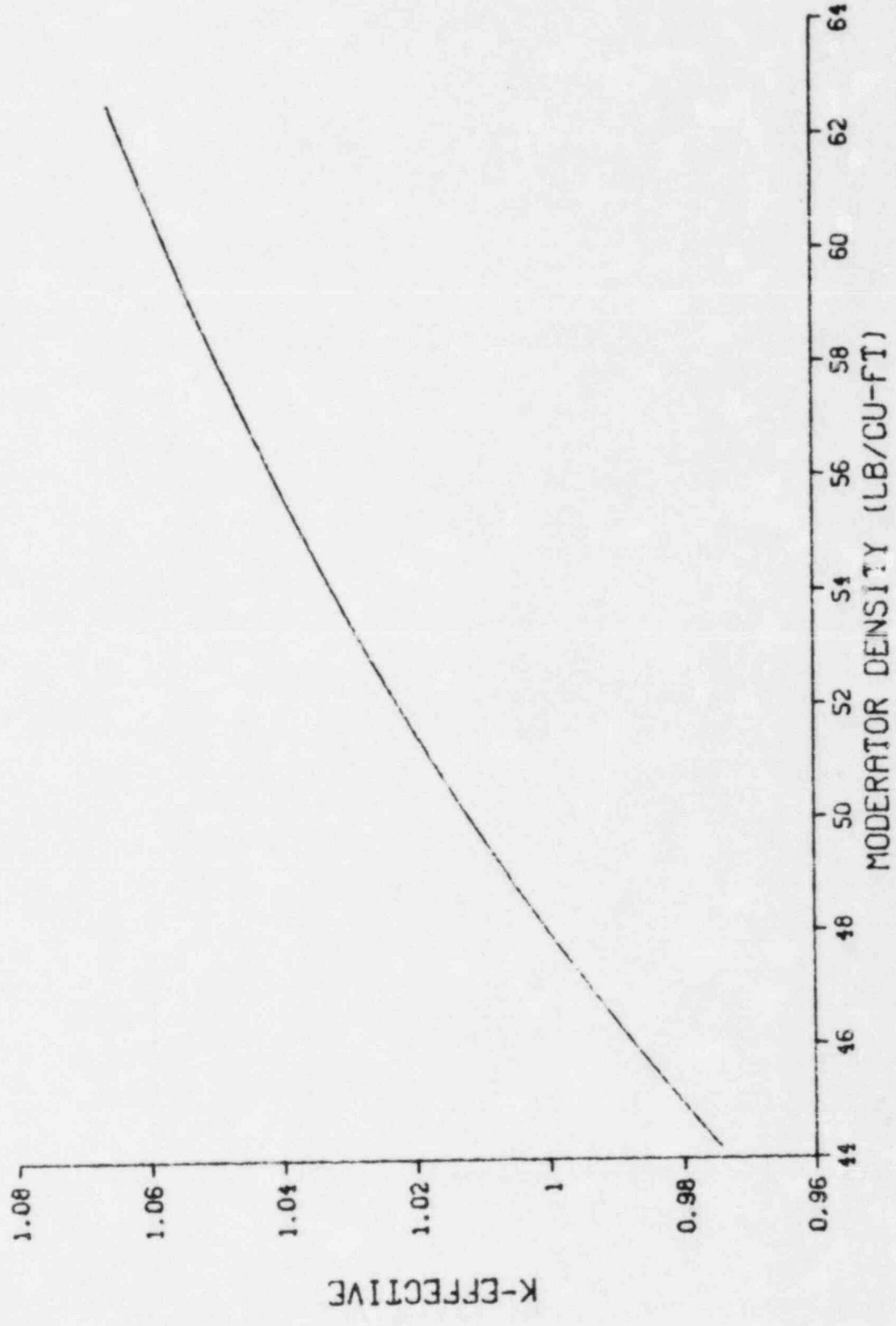

Figure 3.90 Moderator Temperature Feedback For Steamline Break Analysis

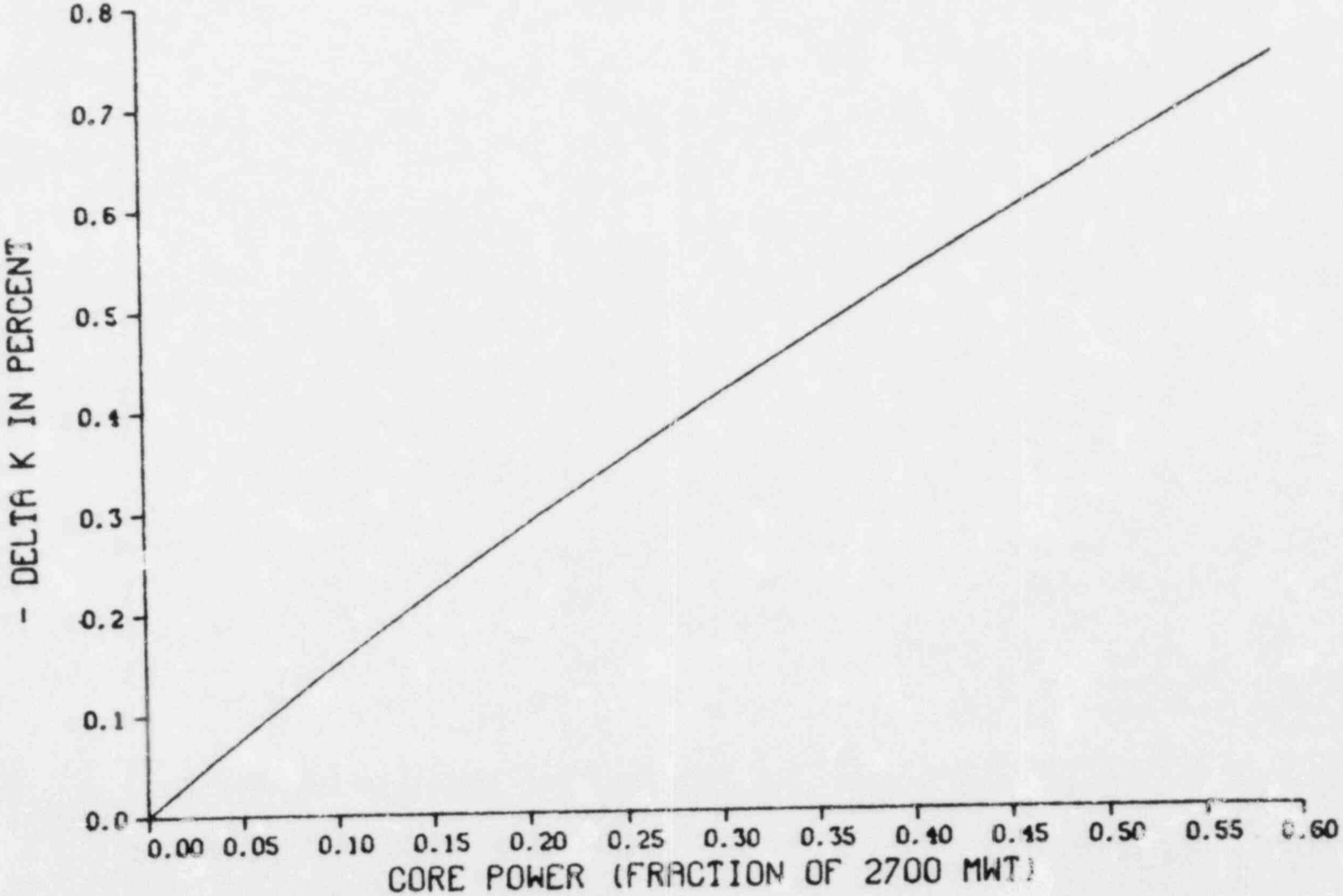

Figure 3.91 Doppler Feedback For Power Operation During A Steamline Break

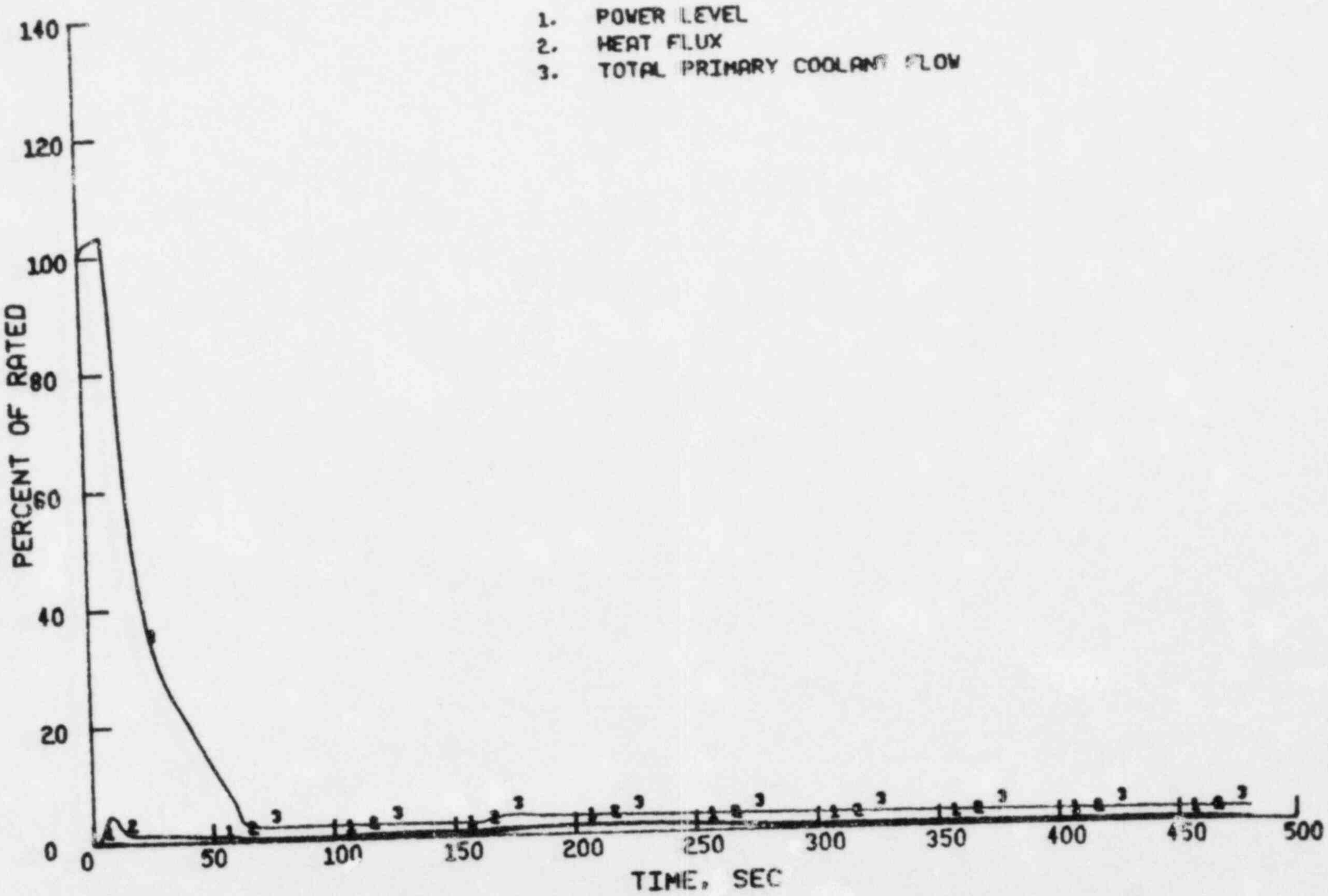

Figure 3.92 St. Lucie Unit 1 - Power, Heat Flux and Flow - Steam Line Rupture

STLUCIE 21.23.33.

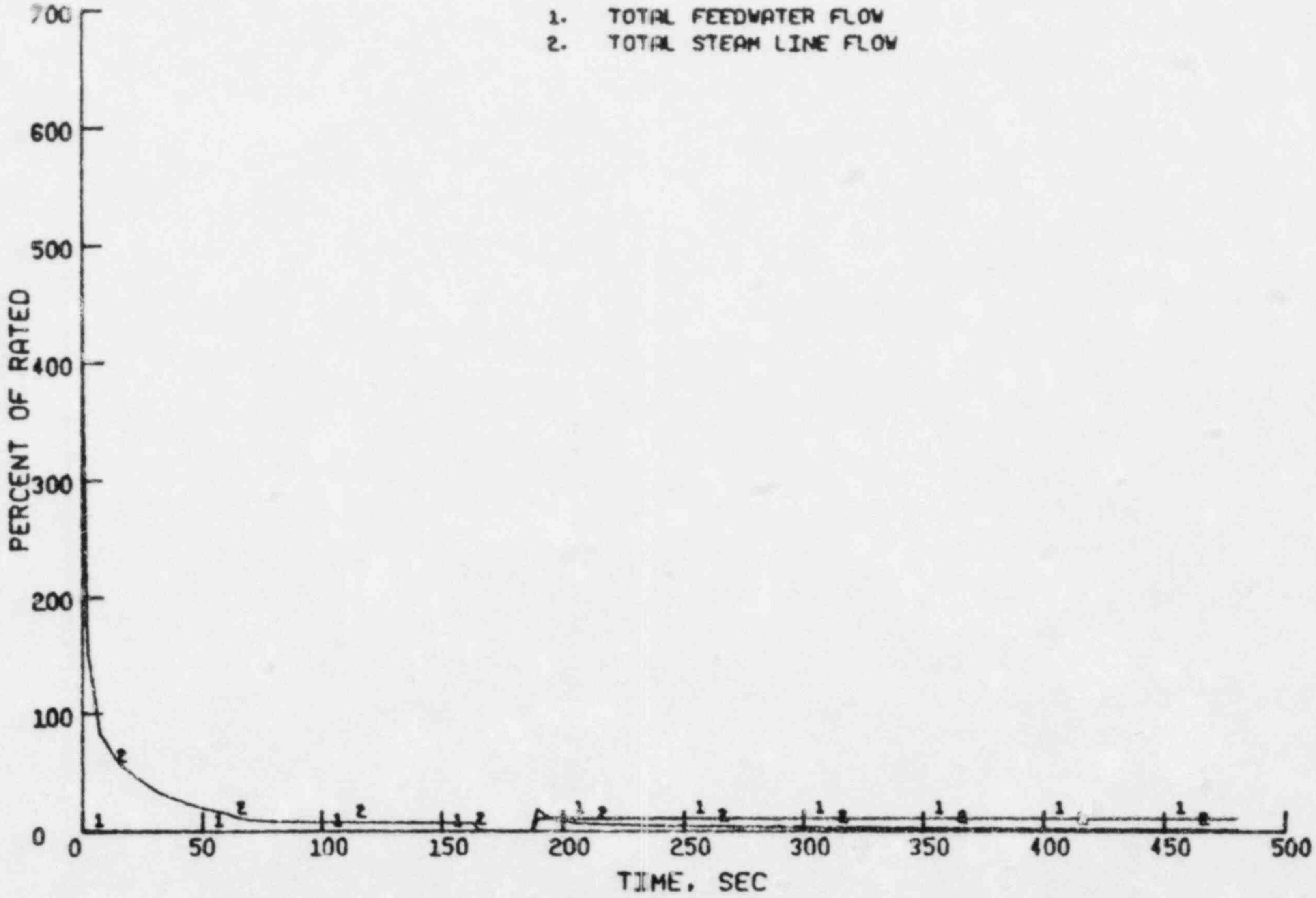

Figure 3.93 St. Lucie Unit 1 - S. G. Flows - Steam Line Rupture

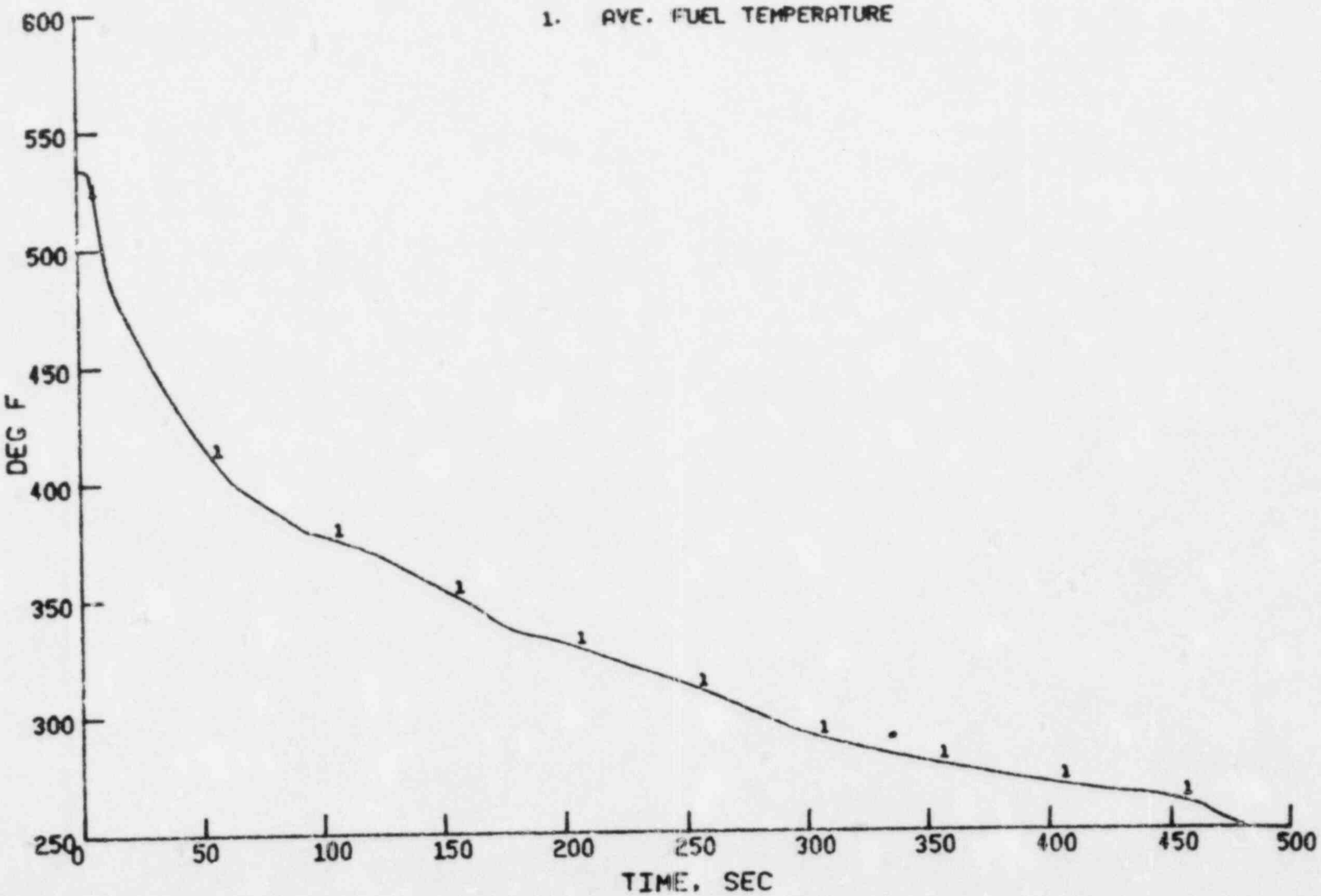

Figure 3.94 St. Lucie Unit 1 - Fuel Temperature - Steam Line Rupture

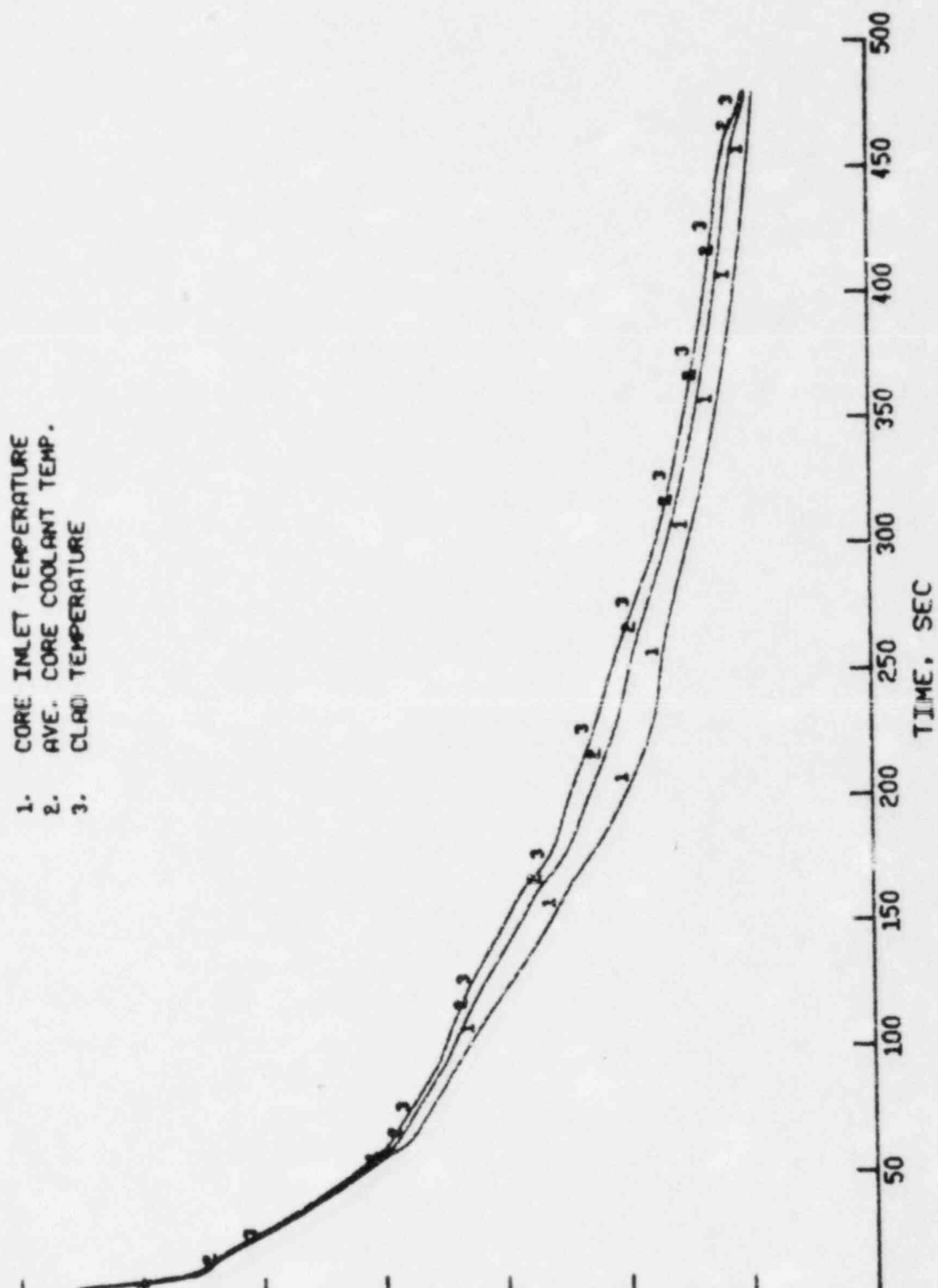

Figure 3.95 St. Lucie Unit 1 - Core Temperatures - Steam Line Rupture

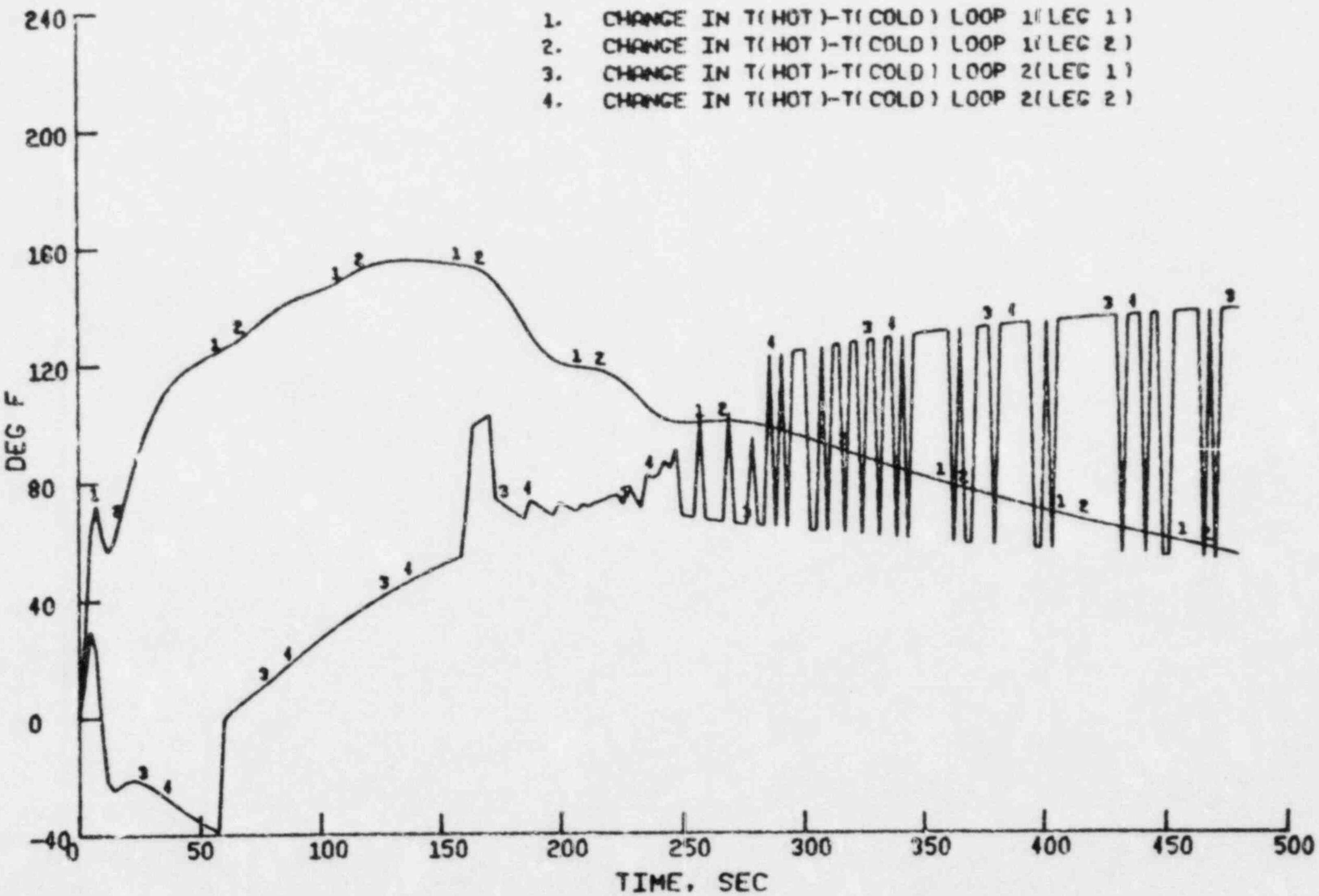

Figure 3.96 St. Lucie Unit 1 - Loop Temperature Differences - Steam Line Rupture

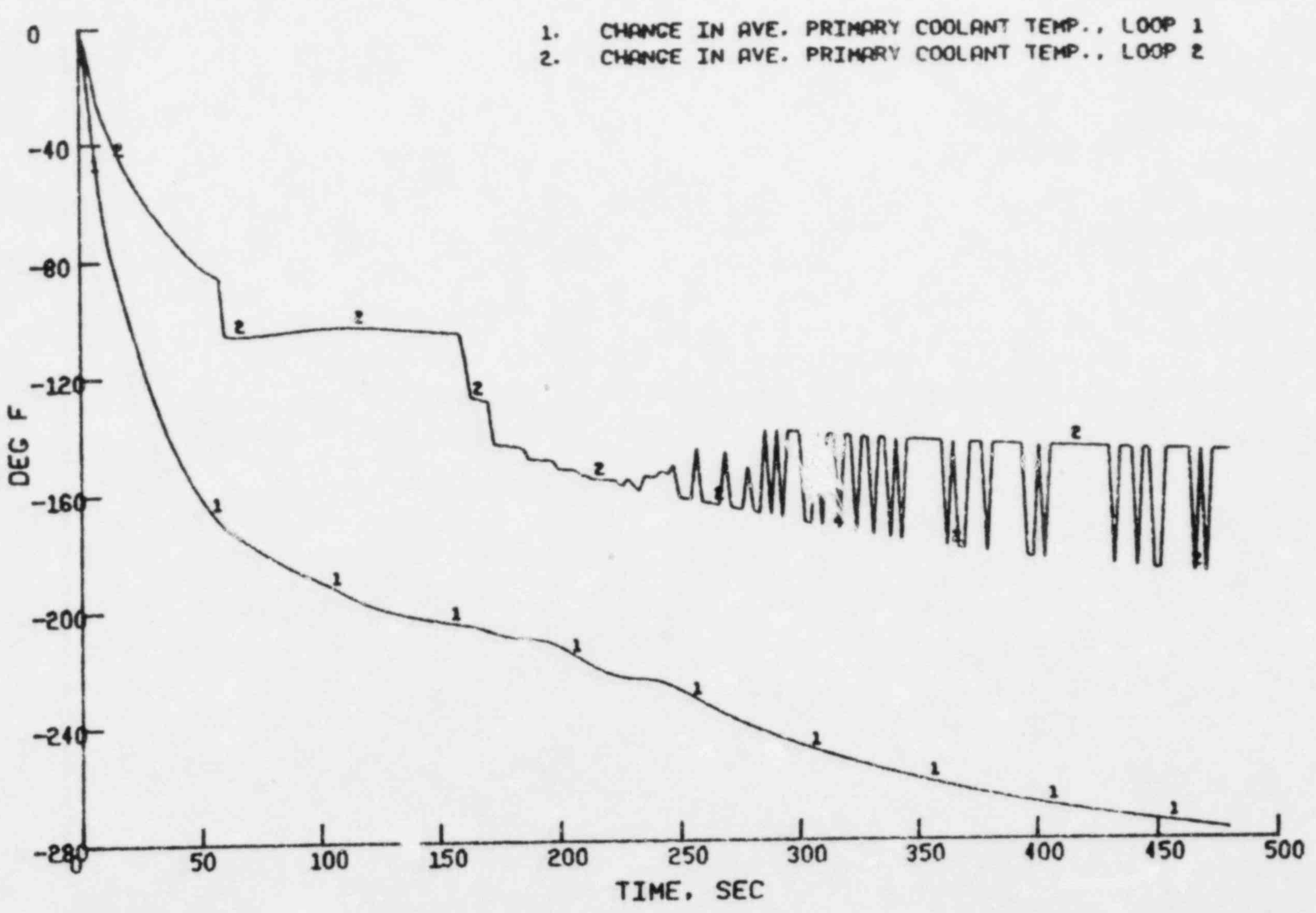

Figure 3.97 St. Lucie Unit 1 - Average Temperatures - Steam Line Rupture

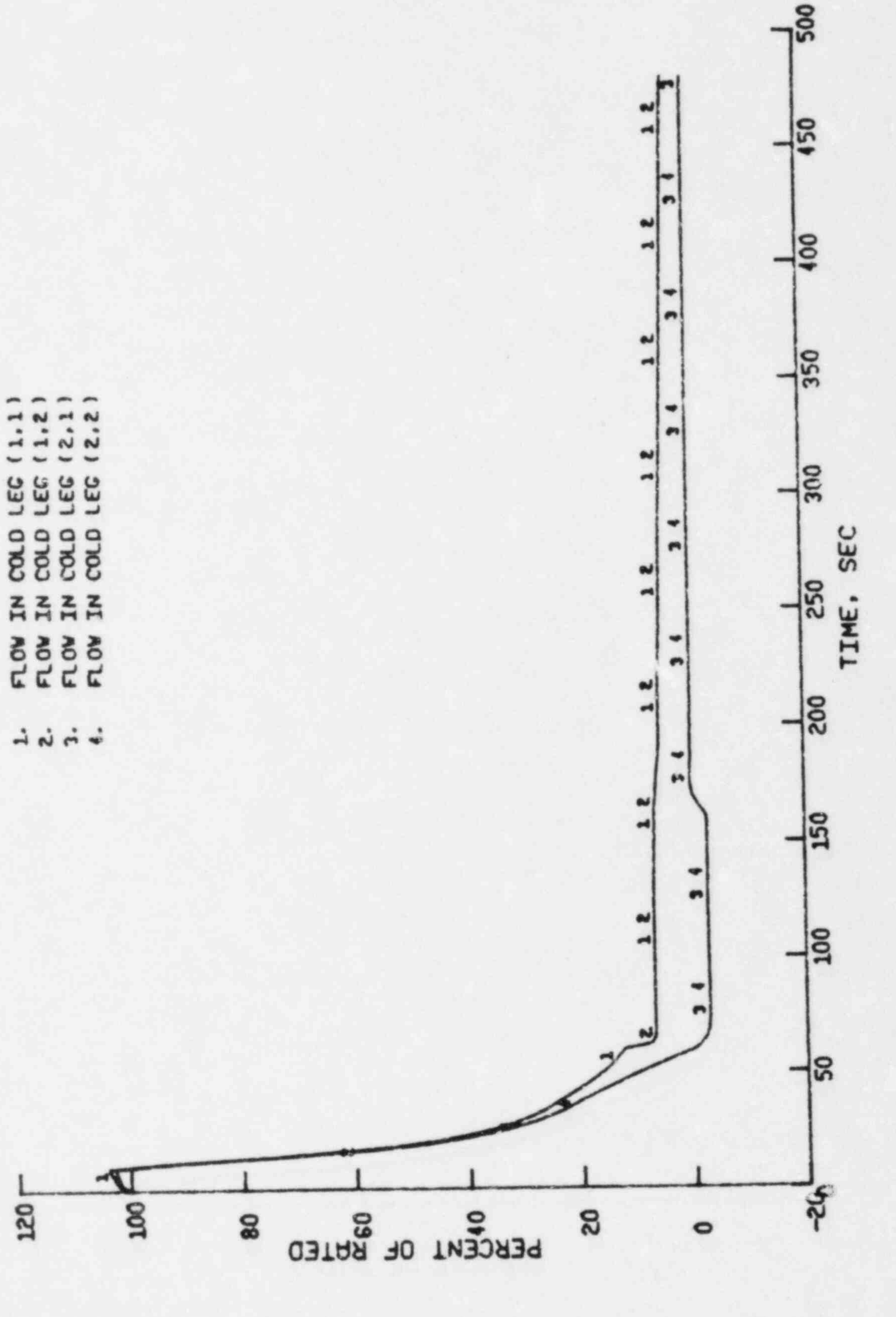

Figure 3.98 St. Lucie Unit 1 - Cold Leg Flows - Steam Line Rupture

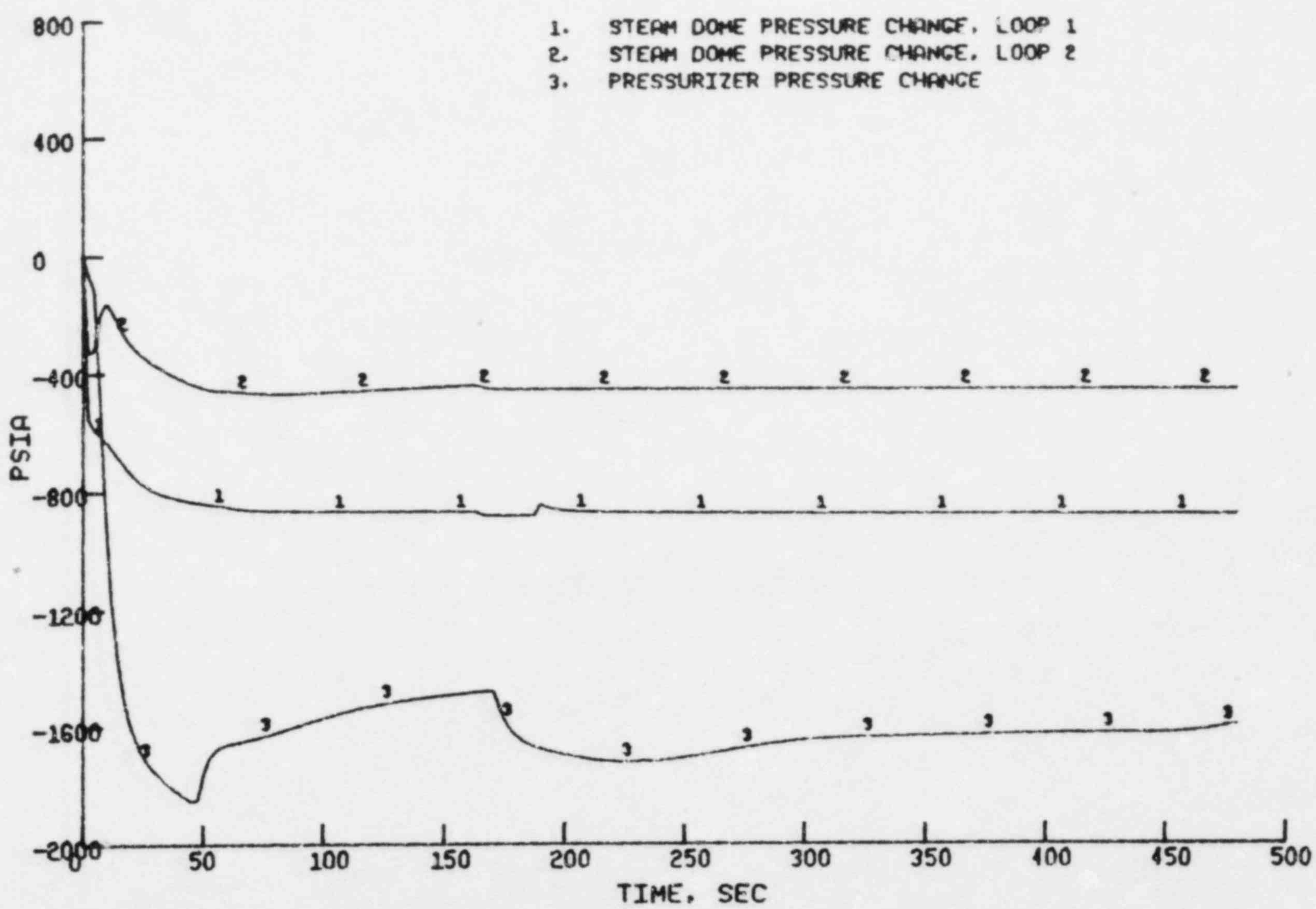

Figure 3.99 St. Lucie Unit 1 - Pressures - Steam Line Rupture

21.23.33.

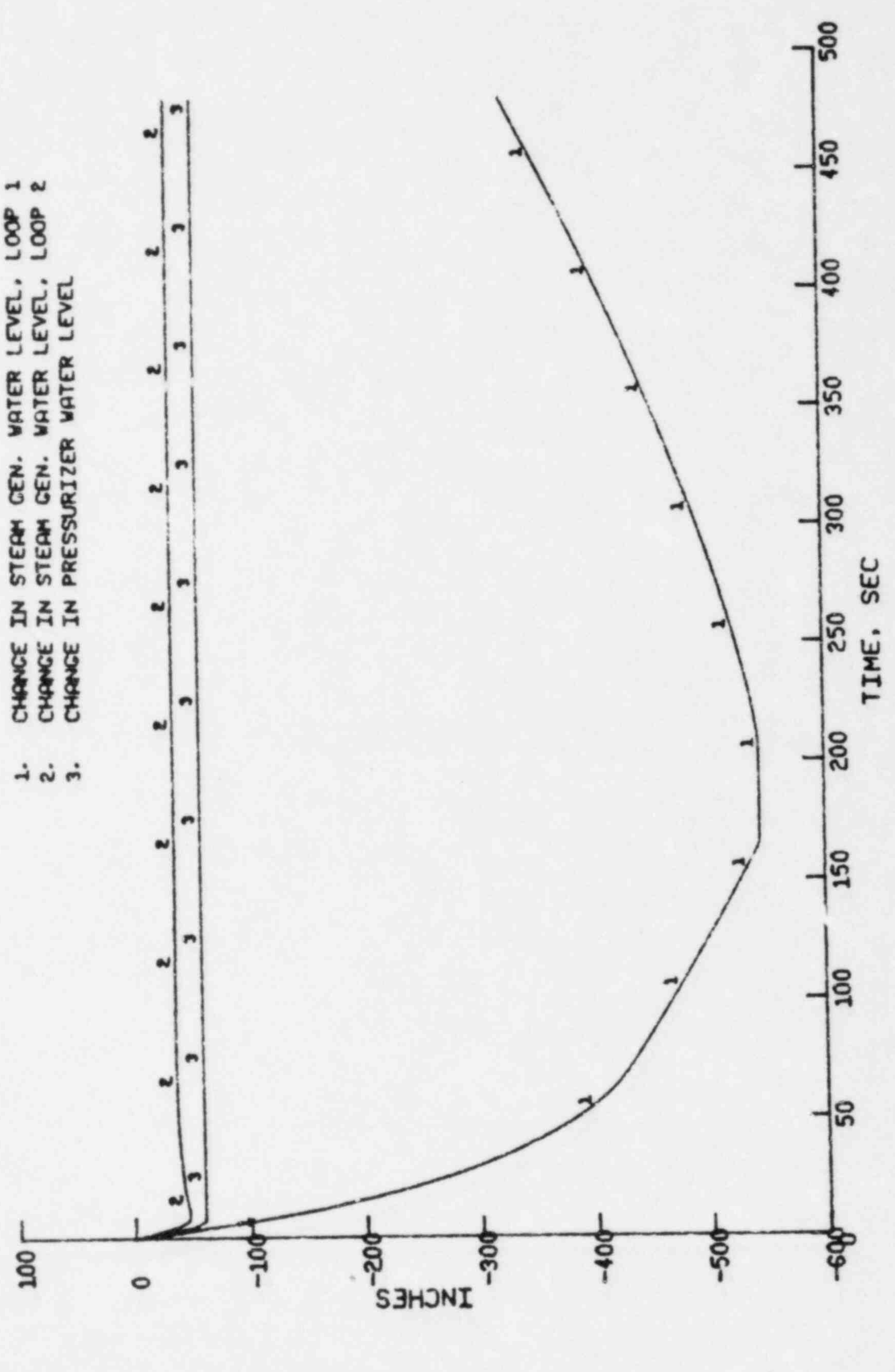

Figure 3.100 St. Lucie Unit 1 - Water Levels - Steam Line Rupture

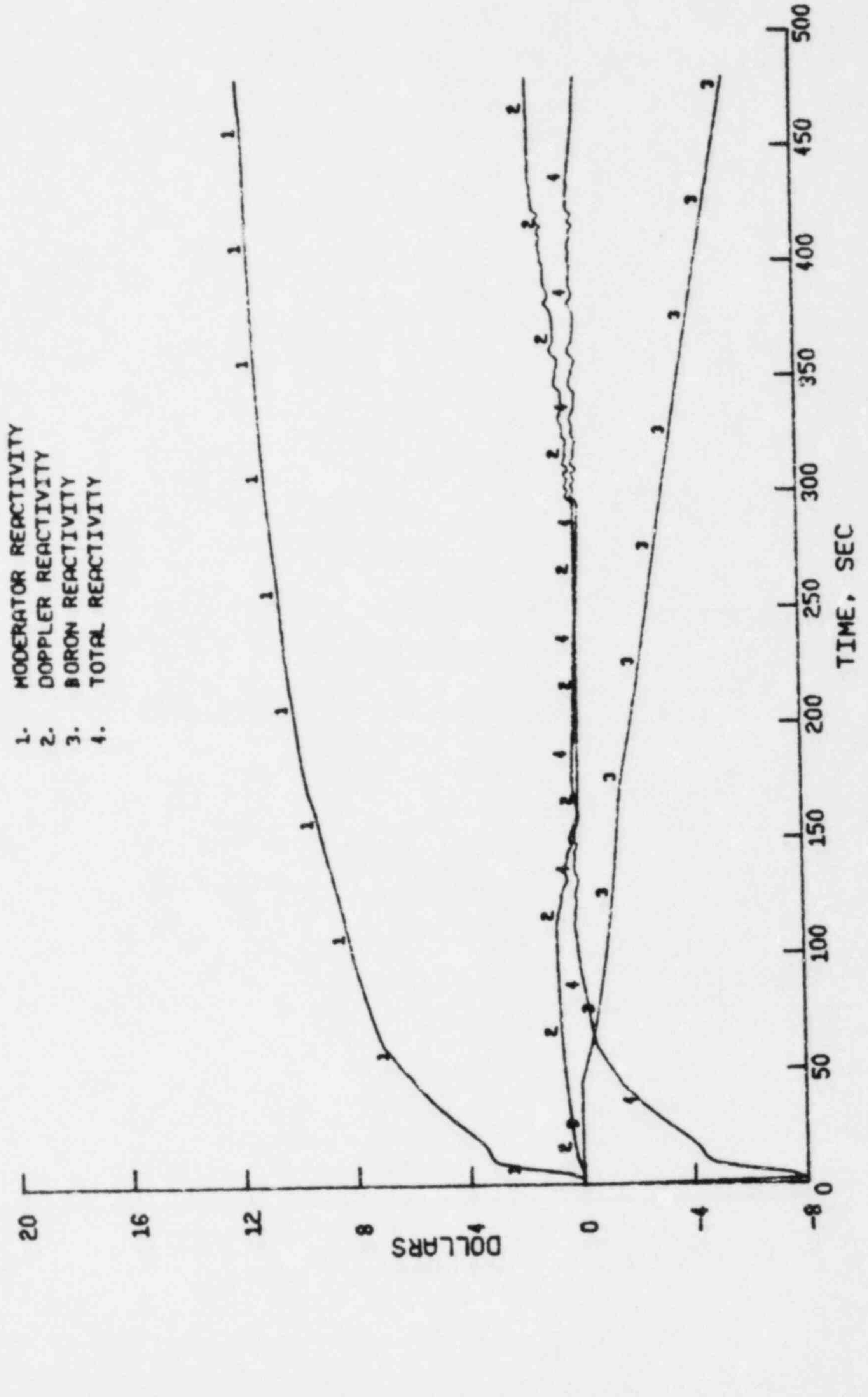

Figure 3.101 St. Lucie Unit 1 - Reactivity - Steam Line Rupture

4.0 DISCUSSION

The ENC transient analysis performed for St. Lucie Unit 1 nuclear power plant demonstrates adequate margin to fuel and vessel design limits for a mixed core of ENC/CE fuel under normal operation, anticipated transients and postulated accidents The transients analyzed in Section 3 were selected because they were shown in the stretch power submittal and the FSAR(7) to have less margin than the transients not analyzed.

The loss-of-load event was analyzed as an overpressurization transient and as such, bounds events such as the loss-of-feedwater or a loss-of-heat-sink in one steam generator. The action of the pressurizer safety valve in controlling the overpressurization is sufficient to demonstrate the acceptability of the plant for overpressurization transients.

The excess-load event was analyzed as the limiting cooldown AOO. The action of the variable high power trip in terminating the transient without a significant degradation in DNBR was sufficient to bound the results of an excess-feedwater transient.

The RCS depressurization transient represents the most pressure transient in the AOO category and was used to test the TM/LP bias. As a test of the TM/LP bias, it was found to be more limiting than the CEA-withdrawal event.

The loss-of-coolant flow event is a limiting AOO for flow reduction and bounds the loss of A.C. power. Further, it provided one of the two transients which was analyzed to set the LCO for DNB.

The CEA-withdrawal event provides a bounding analysis for reactivity insertion transients at full power. The chemical and volume control malfunctions all introduce smaller reactivity ramp rates than this event. This transient is not limiting for the TM/LP bias, since it trips on variable high power significantly before reaching the TM/LP trip setpoint. This occurs primarily because the LPD trip prevents analysis of this transient with a top-peaked core, which would have been far more limiting.

The CEA drop was analyzed for two reasons: (1) it is not protected by a trip; and (2) it was used to verify the LCO based on DNB. The transient simulation supports the existsing LCO.

The seized-rotor event was analyzed as both a DNB transient and as a pressure transient. It was found to be a limiting pressure transient, and it does produce an MDNBR which is essentially at the 95:95 limit for the XNB critical heat flux correlation. The expected pin damage is, however, significantly less than 1%, and thus meets the criterion for radiation release.

The asymmetric steam generator transients were not analyzed since the analyses for prior cycles demonstrated that the ASGPT provided for a reactor scram before the asymmetric reactor inlet flow condition, against which it was designed to protect, could occur. The limiting event, a loss-of-load to one steam generator, results in a trip signal within 2.5 seconds. A cooldown heatup transient initiated in the steam generators requires about 5 seconds to reach the core inlet. Allowing 1.4 seconds from the time the trip condition exists until the CEAs begin to fall leaves 1.1 seconds for the rods to fall before the asymmetry develops.

The steam generator tube rupture transient was not reanalyzed explicitly. The adequacy of the TM/LP trip to protect against a rapid depressurization transient was demonstrated for the bounding case, a failure of all pressurizer relief valves to open, which resulted in an acceptable MDNBR. The steam generator tube rupture event results in a less severe depressurization transient, and thus the TM/LP trip protects against fuel damage for the mixed core. Since this protection removes the fuel dependency of the analysis, it is therefore anticipated that the accident analysis would progress exactly as described in analyses for prior cycles.

In summary, the analysis presented in this report shows acceptable results for core thermal margin during AOOs or PAs for Cycle 6 at St. Lucie Unit 1.

5.0 REFERENCES

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7. Florida Power & Light Company, St. Lucie Plant, Unit No. 7, Updated Final Safety Analysis Report.

8. "St. Lucie Unit 1 Safety Analysis Report," XN-NF-82-81, Exxon Nuclear Company, Inc., Richland, Washington 99352, December 1982.

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