GGNS SINGLE LOOP OPERATION ANALYSIS

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Prepared for

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15.C RECIRCULATION SYSTEMS SINGLE-LOOP OPERATION

15.C.1 INTRODUCTION AND SUMMARY

Single-loop operation (SLO) at reduced power is highly desirable in the event recirculation pump or other component maintenance renders one loop inoperative. To justify single-loop operation, accidents and abnormal operational transients associated with power operations, as presented in Sections 6.2 and 6.3 and the main text of Chapter 15.0, were reviewed for the single-loop case with only one pump in operation. This appendix presents the results of this safety evaluation for the operation of the Grand Gulf Nuclear Stations (GGNS) with single recirculation loop inoperable. This evaluation is performed for GE-6 fueled GGNS on an initial cycle basis and is applicable to GE-6 fueled normal annual 12 month initial cycle operation. The conditions are those of continued operation in the operating domain currently defined in Figure 4.4.5 of Chapter 4 up a maximum power of 70.6% of rated.

Increased uncertainties in the core total flow and Traversing In-Core Probe (TIP) readings resulted in a 0.01 incremental increase in the Minimum Critical Power Ratio (MCPR) fuel cladding integrity safety limit during single-loop operation. No increase in rated MCPR operating limit and no change in the power dependent and flow dependent MCPR limit ($MCPR_f$ and $MCPR_p$) are required because all abnormal operational transients analyzed for single-loop operation indicated there is more than enough MCPR margin to compensate for this increase in MCPR safety limit. The recirculation flow rate dependent rod block and scram setpoint equation given in Chapter 16 (Technical Specifications) are adjusted for one-pump operation.

Thermal-hydraulic stability was evaluated for its adequacy with respect to General Design Criteria 12 (10CFR50, Appendix A). It is shown that SLO satisfies this stability criterion. It is further shown that the increase in neutron noise observed during SLO is independent of system stability margin.

To prevent potential control oscillations from occurring in the recirculation flow control system, the flow control should be in master manual for single-loop operation.

The limiting Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) reduction factor for single-loop operation is calculated to be 0.86.

The containment response for a Design Basis Accident (DBA) recirculation line break with single-loop operation is bounded by the rated power two-loop operation analysis presented in Section 6.2. This conclusion covers all single-loop operation power/flow conditions.

The impact of single loop operation on the Anticipated Transient Without Scram (ATWS) analysis was evaluated. It is found that all ATWS acceptance criteria are met during SLO.

The fuel thermal and mechanical duty for transient events occurring during SLO is found to be bounded by the fuel design bases. The Average Power Range Monitor (APRM) fluctuation should not exceed a flux amplitude of ±15% of rated and the core plate differential pressure fluctuation should not exceed 3.2 psi peak to peak to be consistent with the fuel rod and assembly design bases.

A recirculation pump drive flow limit will be imposed for SLO. The highest drive flow tested during the startup test program at GGNS that meets acceptable vessel internal vibration criteria will be the drive flow limit for SLO.

15.C.2 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

Except for core total flow and TIP reading, the uncertainties used in the statistical analysis to determine the MCPR fuel cladding integrity safety limit are not dependent on whether coolant flow is provided by one or two recirculation pumps. Uncertainties used in the two-loop operation analysis are documented in the FSAR. A 6% core flow measurement uncertainty has been established for single-loop operation (compared to 2.5% for two-loop operation). As shown below, this value conservatively reflects the one standard deviation (one sigma) accuracy of the core flow measurement system documented in Reference 15.C.8-1. The random noise component of the TIP reading uncertainty was revised for single recirculation loop operation to reflect the operating plant test results given in Subsection 15.C.2.2. This revision resulted in a single-loop operation process computer effective TIP uncertainty of 6.8% of initial cores and 9.1% for reload cores. Comparable two-loop process computer uncertainty values are 6.3% for initial cores and 8.7% for reload cores. The net effect of these two revised uncertainties is a 0.01 incremental increase in the required MCPR fuel cladding integrity safety limit.

15.C.2.1 Core Flow Uncertainty

15.C.2.1.1 Core Flow Measurement During Single-Loop Operation

The jet pump core flow measurement system is calibrated to measure core flow when both sets of jet pumps are in forward flow; total core flow is the sum of the indicated loop flows. For single-loop operation, however, some inactive jet pumps will be backflowing (at active pump flow above approximately 36%). Therefore, the measured flow in the backflowing jet pumps must be subtracted from the measured flow in the active loop to obtain the total core flow. In addition, the jet pump coefficient is different for reverse flow than for forward flow, and the measurement of reverse flow must be modified to account for this difference.

In single-loop operation, the total core flow is derived by the following formula:

$$\text{Total Core Flow} = \left(\begin{array}{c}\text{Active Loop}\\ \text{Indicated Flow}\end{array}\right) - C \left(\begin{array}{c}\text{Inactive Loop}\\ \text{Flow}\end{array}\right)$$

Where C (= 0.95) is defined as the ratio of "Inactive Loop True Flow" to "Inactive Loop Indicated Flow". "Loop Indicated Flow" is the flow measured by the jet pump "single-tap" loop flow summers and indicators, which are set to read forward flow correctly.

The 0.95 factor was the result of a conservative analysis to appropriately modify the single-tap flow coefficient for reverse flow.* If a more exact, less conservative core flow is required, special in-reactor calibration tests would have to be made. Such calibration tests would involve: calibrating core support plate ΔP versus core flow during one-pump and two-pump operation along with 100% flow control line and calculating the correct value of C based on the core support plate ΔP and the loop flow indicator readings.

15.C.2.1.2 Core Flow Uncertainty Analysis

The uncertainty analysis procedure used to establish the core flow uncertainty for one-pump operation is essentially the same as for two-pump operation, with some exceptions. The core flow uncertainty analysis is described in Reference 15.C.8-1. The analysis of one-pump core flow uncertainty is summarized below.

For single-loop operation, the total core flow can be expressed as follows (refer to Figure 15.C.2-1):

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*The analytical expected value of the "C" coefficient for GGNS is ~0.82.

$$W_C = W_A - W_I$$

where:

W_C = total core flow,

W_A = active loop flow, and

W_I = inactive loop (true) flow.

By applying the "propagation of errors" method to the above equation, the variance of the total flow uncertainty can be approximated by:

$$\sigma^2_{W_C} = \sigma^2_{W_{sys}} + \left(\frac{1}{1-a}\right)^2 \sigma^2_{W_{A_{rand}}} + \left(\frac{a}{1-a}\right)^2 \left(\sigma^2_{W_{I_{rand}}} + \sigma^2_C\right)$$

where:

σ_{W_C} = uncertainty of total core flow;

$\sigma_{W_{sys}}$ = uncertainty systematic to both loops;

$\sigma_{W_{A_{rand}}}$ = random uncertainty of active loop only;

$\sigma_{W_{I_{rand}}}$ = random uncertainty of inactive loop only;

σ_C = uncertainty of "C" coefficient; and

a = ratio of inactive loop flow (W_I) to active loop flow (W_A).

From an uncertainty analysis, the conservative, bounding values of σW_{sys}, σWA_{rand}, σWI_{rand} and σC are 1.6%, 2.6%, 3.5%, and 2.8%, respectively. Based on the above uncertainties and a bounding value of 0.36* for "a", the variance of the total flow uncertainty is approximately:

$$\sigma^2_{W_C} = (1.6)^2 \quad + \left(\frac{1}{1-0.36}\right)^2 \; (2.6)^2 \; + \left(\frac{0.36}{1-0.36}\right)^2 \left((3.5)^2 + (2.8)^2\right)$$

$$= (5.0)^2$$

When the effect of 4.1% core bypass flow split uncertainty at 12% (bounding case) bypass flow fraction is added to the total core flow uncertainty, the active coolant flow uncertainty is:

$$\sigma^2_{\substack{\text{active} \\ \text{coolant}}} = (5.0)^2 \quad + \left(\frac{0.12}{1-0.12}\right)^2 \quad (4.1)^2 = (5.1)^2$$

which is less than the 6% flow uncertainty assumed in the statistical analysis.

In summary, core flow during one-pump operation is measured in a conservative way and its uncertainty has been conservatively evaluated.

15.C.2.2 TIP READING UNCERTAINTY

To ascertain the TIP noise uncertainty for single recirculation loop operation, a test was performed at an operating BWR. The test was performed at a power level 59.3% of rated with a single recirculation pump in operation (core flow 46.3% of rated). A rotationally symmetric control rod pattern existed during the test.

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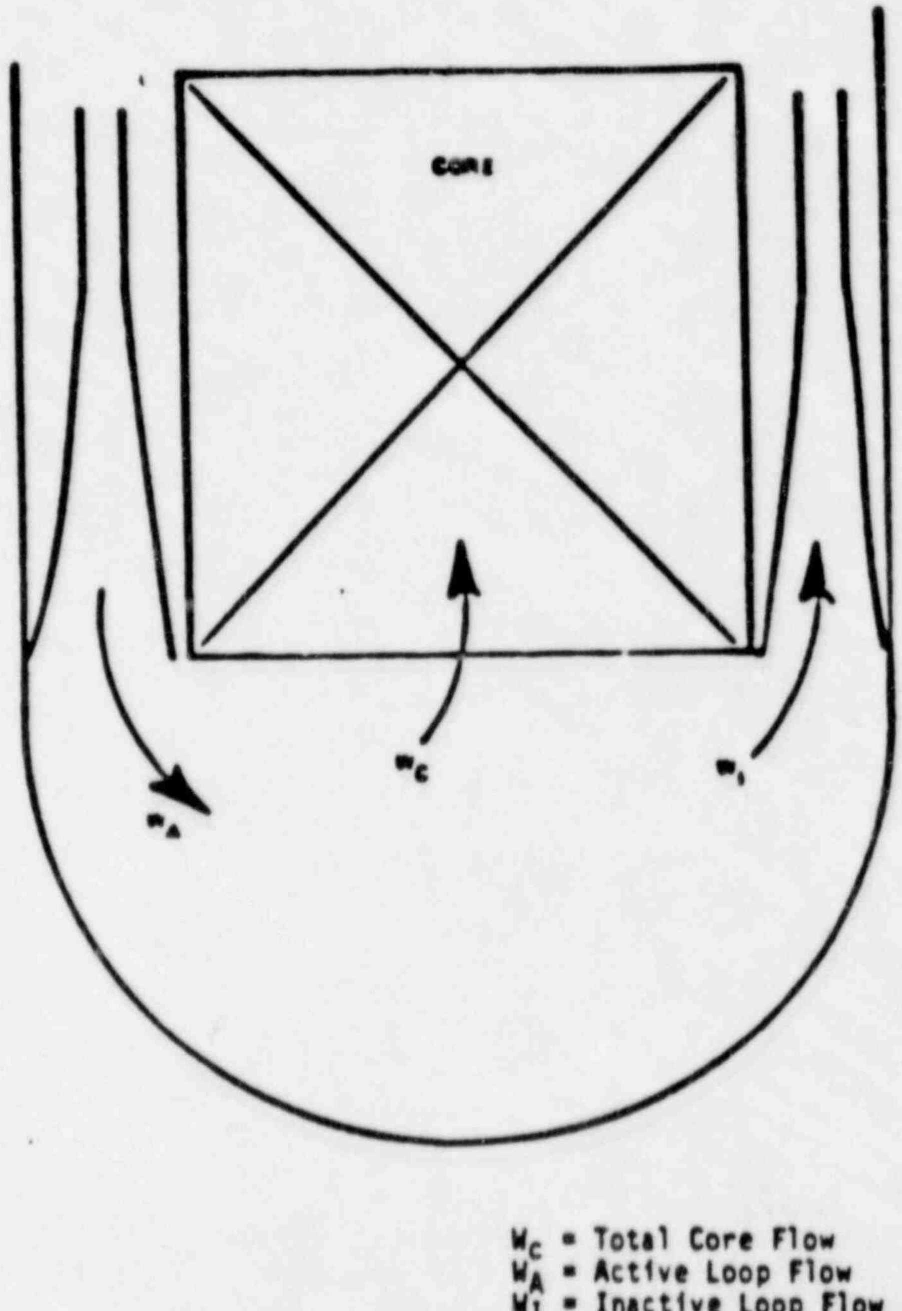
*This flow split ratio varies from about 0.13 to 0.36. The 0.36 value is a conservative bounding value. The analytical expected value of the flow split ratio for GGNS is ~ 0.28.

Five consecutive traverses were made with each of five TIP machines, giving a total of 25 traverses. Analysis of this data resulted in a nodal TIP noise of 2.85%. Use of this TIP noise value as a component of the process computer total uncertainty results in a one-sigma process computer total effective TIP uncertainty value for single-loop operation of 6.8% for initial cores and 9.1% for reload cores. The results of the analysis is directly applicable to GGNS because the data collected are typical random neutron, electronic and boiling noise during SLO for a BWR.



MISSISSIPPI POWER & LIGHT	ILLUSTRATION OF SINGLE RECIRCULATION LOOP OPERATION FLOWS	FIGURE 15.C.2-1

15.C.3 MCPR OPERATING LIMIT

15.C.3.1 ABNORMAL OPERATING TRANSIENTS

Operating with one recirculation loop results in a maximum power output which is about 30% below that which is attainable for two-pump operation. Therefore, the consequences of abnormal operation transients from one-loop operation will be considerably less severe than those analyzed from a two-loop operational mode. For pressurization, flow increase, flow decrease, and cold water injection transients, results presented in the FSAR bound both the thermal and overpressure consequences of one-loop operation.

Figure 15.C.3-1 shows the consequences of a typical pressurization transient (turbine trip) as a function of power level. As can be seen, the consequences of one-loop operation are considerably less because of the associated reduction in operating power level.

The consequences of flow decrease transients are also bounded by the full power analysis. A single pump trip from one-loop operation is less severe than a two-pump trip from full power because of the reduced initial power level.

The worst flow increase transient results from recirculation flow controller failure, and the worst cold water injection transient results from the loss of feedwater heater. For the former, the $MCPR_f$ curve is derived from a postulated event involving runout of both recirculation loops. This condition produces the maximum possible power increase and hence maximum ΔMCPR for transients initiated from less than rated power and flow. When operating with only one recirculation loop, the flow and power increase associated with this failure with only one loop will be less than that associated with both loops; therefore, the $MCPR_f$ curve derived with the two-pump assumption is conservative for single-loop operation. The latter event, loss of feedwater heating, is generally the most severe cold water increase event with respect to increase in core power. This event is caused by positive reactivity insertion from core inlet subcooling and it is relatively insensitive to initial power level. A generic statistical loss of feedwater heater analysis using different initial power levels and other core design parameters concluded one-pump operation with lower initial power level is conservatively bounded by the full power two-pump analysis. Inadvertent restart of the idle recirculation pump has been analyzed in the FSAR and is still applicable for single-loop operation.

From the above discussions, it is concluded that the transient consequence from one-loop operation is bounded by previously submitted full power analyses. The maximum power level that can be attained with one-loop operation is only restricted by the MCPR and overpressure limits established from a full-power analysis.

In the following sections, three of the most limiting transients of core flow increase, pressurization, and flow decrease events are analyzed for single-loop operation. They are, respectively:

a. feedwater flow controller failure (maximum demand), (FWCF)

b. generator load rejection with bypass failure, (LRNBP), and

c. one pump seizure accident. (PS)

The plant initial conditions are given in Table 15.C.3-1.

15.C.3.1.1 Feedwater Controller Failure - Maximum Demand

15.C.3.1.1.1 Core and System Performance

Mathematical Model

The computer model described in Reference 15.C.8-2 was used to simulate this event.

Input Parameters and Initial Conditions

The analysis has been performed with the plant conditions tabulated in Table 15.C.3-1, except the initial vessel water level is at level setpoint L4 for conservatism. By lowering the initial water level, more cold feedwater will be injected before Level 8 is reached resulting in higher heat fluxes.

End of cycle (all rods out) scram characteristics are assumed. The safety/relief valve action is conservatively assumed to occur with higher than nominal setpoints. The transient is simulated by programming an upper limit failure in the feedwater system such that 130% of rated feedwater flow occurs at the design pressure of 1065 psig.

Results

The simulated feedwater controller transient is shown in Figure 15.C.3-2 for the case of 70.6% power 54.1% core flow. The high-water level turbine trip and feedwater pump trip are initiated at approximately 4.2 seconds. Scram occurs simultaneously from Level 8, and limits the peak neutron flux. MCPR is considerably above the safety limit so no fuel failure due to boiling transition is predicted. The turbine bypass system opens to limit peak pressure in the steamline near the safety valves to 1045 psig and the pressure at the bottom of the vessel to about 1059 psig.

Consideration of Uncertainties

All systems used for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints, scram stroke time, etc.) Expected plant behavior is, therefore, expected to lead to a less severe transient.

15.C.3.1.1.2 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed; therefore, these barriers maintain integrity and function as designed.

15.C.3.1.1.3 Radiological Consequences

The consequences of this event do not result in any calculated fuel failures; however, radioactive steam is discharged to the suppression pool as a result of SRV activation.

15.C.3.1.2 Generator Load Rejection With Bypass Failure

15.C.3.1.2.1 Core and System Performance

Mathematical Model

The computer model described in Reference 15.C.8-2 was used to simulate this event.

Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.C.3-1.

The turbine electro-hydraulic control system (EHC) power/load imbalance device detects load rejection before a measurable speed change takes place.

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 second.

Auxiliary power is independent of any turbine generator overspeed effects and is continuously supplied at rated frequency, assuming automatic fast transfer to auxiliary power supplies.

The reactor is operating in the manual flow-control mode when load rejection occurs. Results do not significantly differ if the plant had been operating in the automatic flow-control mode.

Results

The simulated generator load rejection without bypass is shown in Figure 15.C.3-3.

Table 15.C.3-2 shows for the case of bypass failure, peak neutron flux reaches about 70.7% of rated and peak steamline pressure at the valves reaches 1167 psig. The peak nuclear system pressure reaches 1179 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig. The calculated MCPR is 1.41, which is well above the safety limit.

Consideration of Uncertainties

The full-stroke closure rate of the turbine control valve of 0.15 second is conservative. Typically, the actual closure rate is approximately 0.2 second. The less time it takes to close, the more severe the pressurization effect.

All systems used for protection in this event were assumed to have the poorest allowable response (e..g, relief setpoints, scram stroke time, etc.). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.C.3.1.2.2 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed and, therefore, these barriers maintain their integrity as designed.

15.C.3.1.2.3 Radiological Consequences

The consequences of this event do not result in any calculated fuel failures; however, radioactivity is nevertheless discharged to the suppression pool as a result of SRV activation.

15.C.3.1.3 Recirculation Pump Seizure Accident

15.C.3.1.3.1 Core and System Performance

Mathematical Model

The computer model described in Reference 15.C.8-3 was used to simulate this event.

Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.C.3-1. For the purpose of evaluating consequences to the fuel thermal limits, this transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of the active recirculation pump shaft while the reactor is operating at ~71% NB rated power under single-loop operation. Also, the reactor is assumed to be operating at thermally limiting conditions.

The void coefficient is adjusted to the most conservative value; that is, the least negative value in Table 15.C.3-1.

Results

Figure 15.C.3-4 presents the results of the accident. Core coolant flow drops rapidly, reaching a minimum value of 26% rated at about 1.3 seconds. The minimum CPR value during the transient is 1.24 and poses no threats

to thermal limits.

15.C.3.1.3.2 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel pressure vessel or containment are designed. Therefore, these barriers maintain integrity and function as designed.

15.C.3.1.3.3 Radiological Consequences

The consequences of this event do not result in any calculated fuel failures.

15.C.3.1.4 Summary and Conclusions

The transient peak value results are summarized in Table 15.C.3-3. The Critical Power Ratio (CPR) results are summarized in Table 15.C.3-3. This table indicates that for the transient events analyzed here, the MCPRs for all transients are above the single-loop operation safety limit value of 1.07. It is concluded the thermal margin safety limits established for two-pump operation are also applicable to single-loop operation conditions.

For pressurization, Table 15.C.3-2 indicates the peak pressures are below the ASME code value of 1375 psig. Hence, it is concluded the pressure barrier integrity is maintained under single-loop operation conditions.

15.C.3.2 ROD WITHDRAWAL ERROR

The rod withdrawal error (RWE) transient for two-loop operation documented in the main text of this chapter employs a statistical evaluation of the minimum critical power ratio (MCPR) and linear heat generation rate (LHGR) response to the withdrawal of ganged control rods for both rated and off-rated conditions. The required MCPR limit protection for the event is provided by the rod withdrawal limits (RWL) system. Since this analyses covered all off-rated condition in the power/flow operating map, single-loop operation is bounded by the current technical specification.

The Average Power Range Monitor (APRM) rod block system provides additional alarms and rod blocks when power levels are grossly exceeded. Modification of the APRM rod block equation (below) is required to maintain the two loop rod block versus power relationship when in one loop operation.

One-pump operation results in backflow through 12 of the 24 jet pumps while the flow is being supplied into the lower plenum from the 12 active jet pumps. Because of the backflow through the inactive jet pumps, the present rod block equation was conservatively modified for use during one-pump operation because the direct active-loop flow measurement may not indicate actual flow above about 36% core flow without correction.

A procedure has been established for correcting the APRM rod block equation to account for the discrepancy between actual flow and indicated flow in the active loop. This preserves the original relationship between APRM rod block and actual effective drive flow when operating with a single loop.

The two-pump rod block equation is:

$$RB = mW + RB_{100} - m(100)$$

The one-pump equation becomes:

$$RB = mW + RB_{100} - m(100) - m\Delta W$$

where

ΔW = difference between two-loop and single-loop effective drive flow at the same core flow.

RB = power at rod block in %;

m = flow reference slope

W = drive flow in % of rated.

RB_{100} = top level rod block at 100% flow.

If the rod block setpoint (RB_{100}) is changed, the equation must be recalculated using the new value.

The APRM scram trip settings are flow biased in the same manner as the APRM rod block setting. Therefore, the APRM scram trip settings are subject to the same procedural changes as the rod block settings discussed above.

15.C.3.3 OPERATING MCPR LIMIT

For single-loop operation, the operating MCPR limit remains unchanged from the normal two-loop operation limit. Although the increased uncertainties in core total flow and TIP readings resulted in a 0.01 incremental increase in MCPR fuel cladding integrity safety limit during single-loop operation (Section 15.C.2), the limiting transients have been analyzed to indicate that there is more than enough MCPR margin during single-loop operation to compensate for this increase in safety limit. For single loop operation at off-rated conditions, the steady-state operating MCPR limit is established by the $MCPR_p$ and $MCPR_f$ curves. This ensures the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational occurrence. The abnormal operating transients analyzed concluded that current power dependent $MCPR_p$ limits are bounding for single loop operation. Since the maximum core flow runout during single loop operation is only about 54% of rated, the current flow dependent $MCPR_f$ limits which are generated based on the flow runout up to rated core flow are also adequate to protect the flow runout events during single loop operation.

TABLE 15.C.3-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS FOR SINGLE-LOOP OPERATION

1. Thermal Power Level Analysis Value, MWt	2708 (70.6% Rated)
2. Steam Flow, lb/hr	11.06×10^6
3. Core Flow, lb/hr	60.9×10^6 (54.1% Rated)
4. Feedwater Flow Rate, lb/sec	3072
5. Feedwater Temperature, °F	386
6. Vessel Dome Pressure, psig	981
7. Vessel Core Pressure, psig	985
8. Turbine Bypass Capacity, % NBR	35
9. Core Coolant Inlet Enthalpy, Btu/lb	509.5
10. Turbine Inlet Pressure, psig	946
11. Fuel Lattice	8x8R
12. Core Leakage Flow, %	10.65
13. Required MCPR Operating Limit	1.41(a)
14. MCPR Safety Limit for incident of Moderate frequency	
First Core	1.07
Reload Core	1.08
15. Doppler Coefficient (-)¢/°F Analysis Data	0.132(b)
16. Void Coefficient (-)¢/% Rated Voids	
Analysis Data for Power Decrease Events	4.0(b)
Analysis Data for Power Increase Events	14.0(b)
17. Core Average Void Fraction, %	41.9(b)
18. Jet Pump Ratio, M	3.521

TABLE 15.C.3-1 (Continued)

19.	Safety/Relief Valve Capacity, % NBR	
	@1145 psig	102.4
	Manufacturer	DIKKER
	Quantity Installed	20
20.	Relief Function Delay, Seconds	0.4
21.	Relief Function Response, Seconds	0.1
22.	Setpoints for Safety/Relief Valves	
	Safety Function, psig	1175, 1185, 1195, 1205, 1215
	Relief Function, psig	1145, 1155, 1165, 1175
23.	Number of Valve Groupings Simulated	
	Safety Function, No.	5
	Relief Function, No.	4
24.	High Flux Trip, % NBR	
	Analysis Setpoint (1.22 x 1.042), % NBR	127.2
25.	High Pressure Scram Setpoint, psig	1095
26.	Vessel Level Trips, Feet Above Separator Skirt Bottom	
	Level 8 - (L8), Feet	5.88
	Level 4 - (L4), Feet	4.03
	Level 3 - (L3), Feet	2.16
	Level 2 - (L2), Feet	-2.182
27.	APRM Thermal trip	
	Setpoint, % NBR @ 100% Core FLow	118.8
28.	RPT Delay, Seconds	0.19
29.	RPT Inertia Time Constant for Analysis, secs.	5
30.	Total steamline volume, ft^3	4358

(a) Operation operating limit is given by $MCPR_f$ for a core flow of 54.1%.

(b) Parameters used in Reference 15.C.8-3 analysis only. Reference 15.C.8-2 values are calculated within the code for end of Cycle 1 condition. These are rated condition values.

TABLE 15.C.3-2
SUMMARY OF TRANSIENT PEAK VALUE RESULTS

SINGLE-LOOP OPERATION

PARAGRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX (% NBR)	MAXIMUM DOME PRESSURE (psig)	MAXIMUM VESSEL PRESSURE (psig)	MAXIMUM STEAMLINE PRESSURE (psig)	FREQUENCY* Category
		Initial Condition	70.6	981	998	974	N/A
15.C.3.1.1	15.C.3.2	Feedwater flow Controller Failure (Maximum Demand)	79.2	1045	1059	1045	a
15.C.3.1.2	15.C.3.3	Generator Load Rejection With Bypass Failure	70.7	1166	1179	1167	b
15.C.3.1.3	15.C.3.4	Seizure of Active Recirculation Pump	70.6	984	998	976	c

*a = Moderate frequency incident; b = infrequent; c = limiting faults

TABLE 15.C.3-3

SUMMARY OF CRITICAL POWER RATIO RESULTS - SINGLE-LOOP OPERATION

	FWCF	LRNBT	PS
Initial Operating Condition (% power/% flow)	70.6/54.1	70.6/54.1	70.6/54.1
Required Two Loop Initial MCPR Operating Limit at SLO Condition	1.41	1.41	1.41
ΔCPR	0.07[a]	[a] 0.00[b]	0.17
Transient MCPR at SLO	1.34	1.41	1.24
SLMCPR at SLO*	1.07	1.07	1.07
Margin Above SLMCPR**	0.27	0.34	0.17
Frequency Category	Moderate frequent incident	Infrequent incident	Limiting fault

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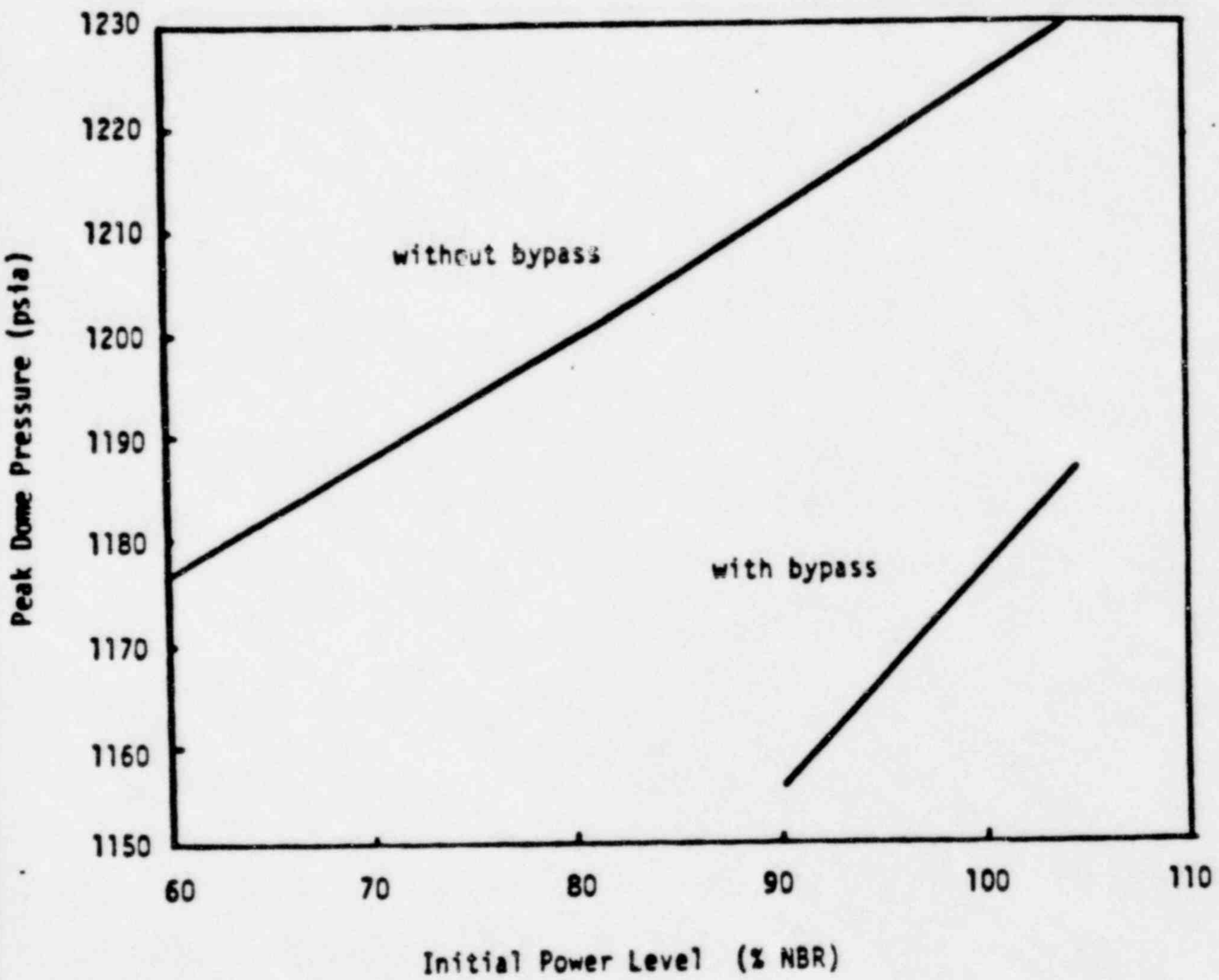
[a] value includes option A adder

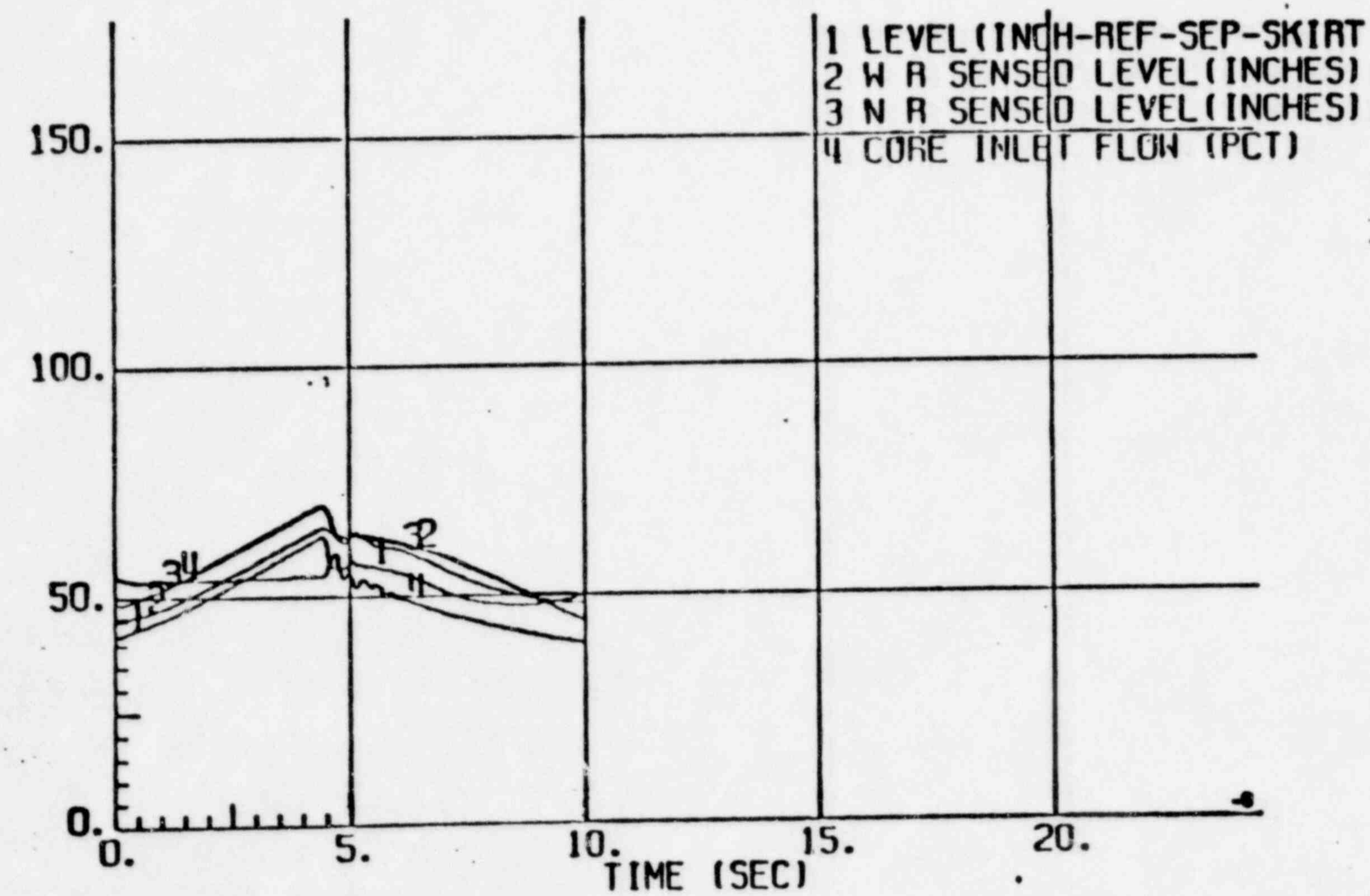
[b] ΔCPR is less than 0.002.

*Values shown for initial cycle. Add 0.01 for reload cycles.

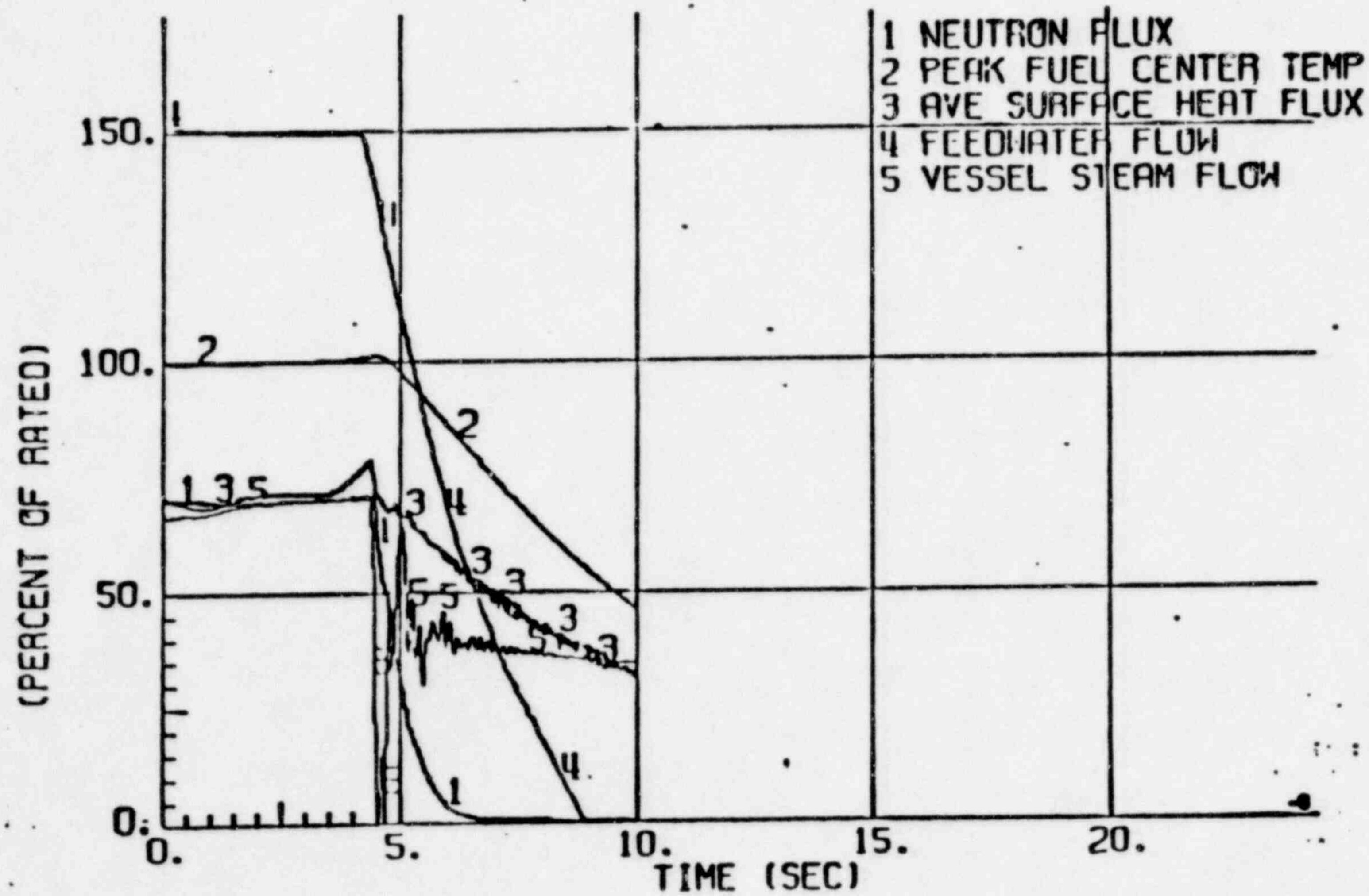
**Reduce margin by 0.01 for reload safety limit increase.



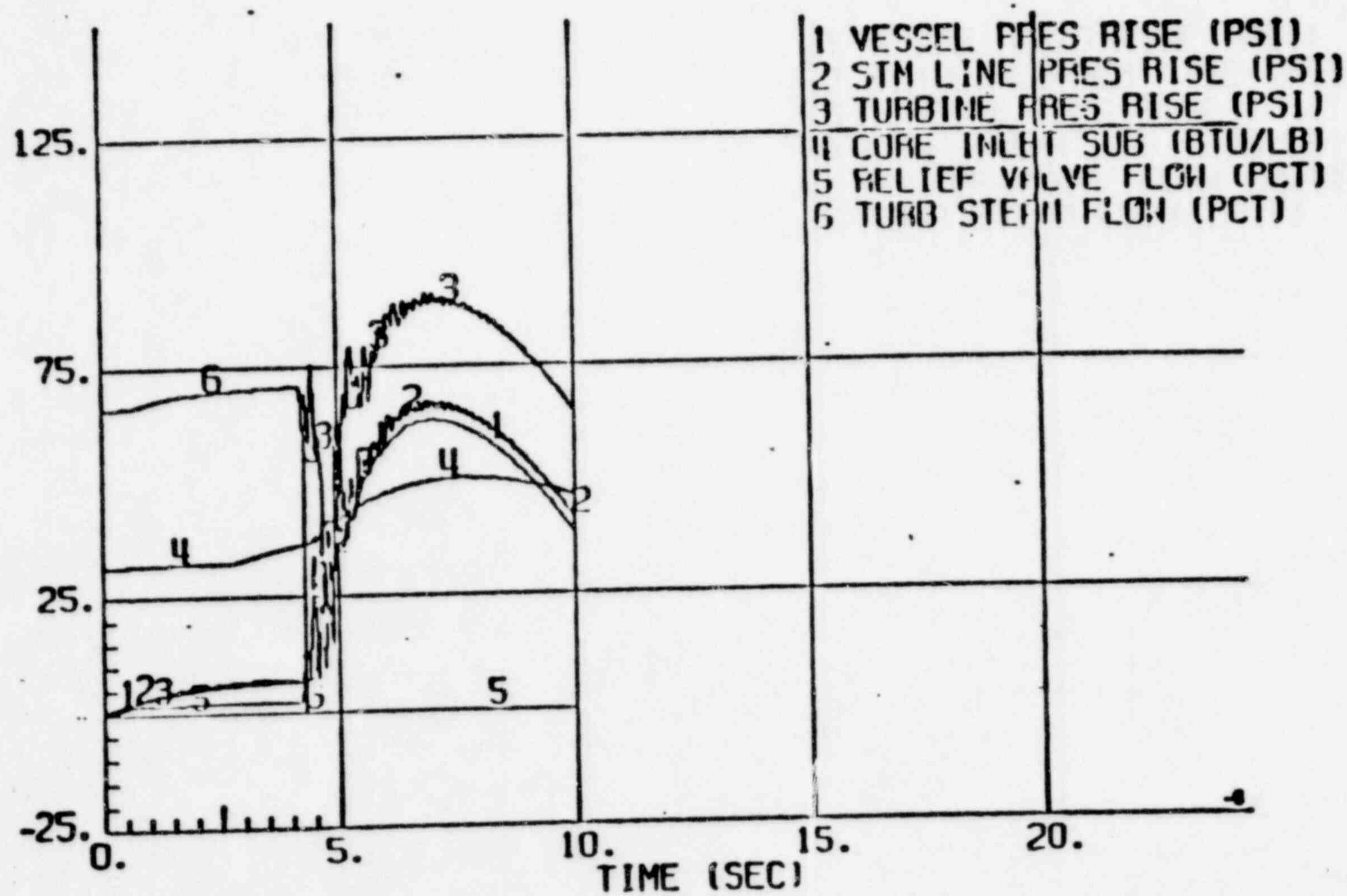
MISSISSIPPI POWER & LIGHT	Peak Dome Pressure versus Initial Power Level, Turbine Trip at EOEC	FIGURE 15.C.3-1



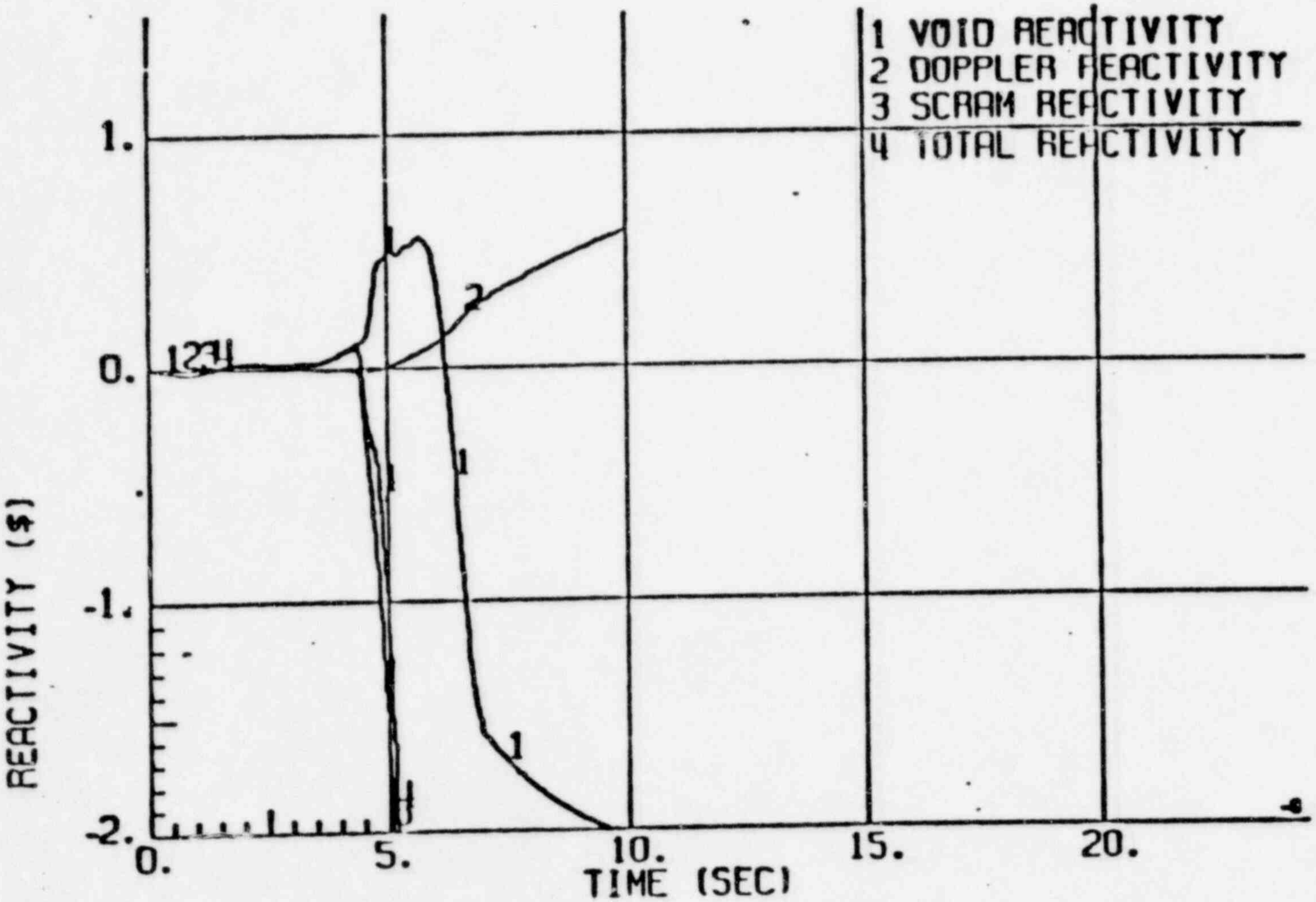
MISSISSIPPI POWER & LIGHT	FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-2



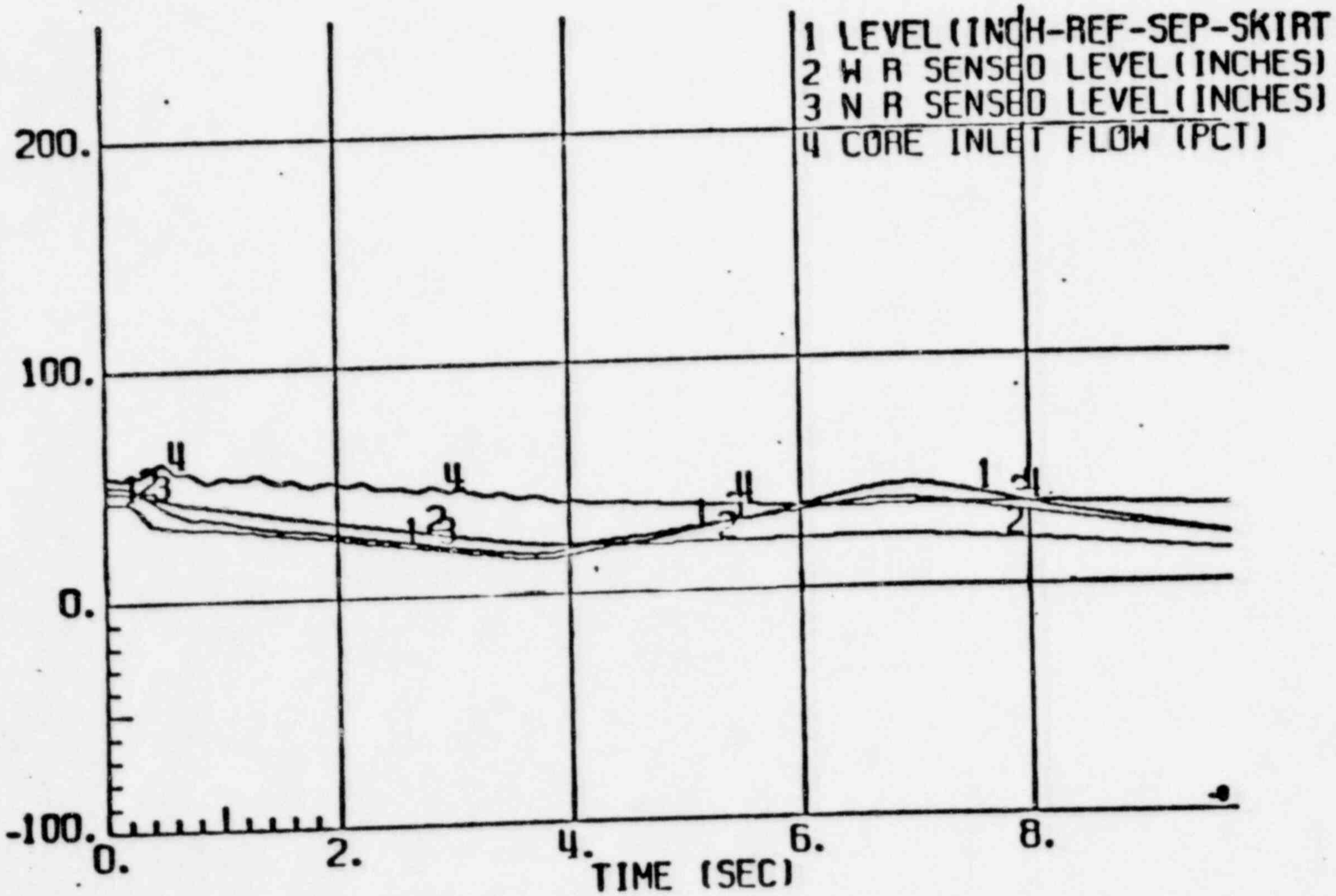
MISSISSIPPI POWER & LIGHT	FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-2



MISSISSIPPI POWER & LIGHT	FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-2



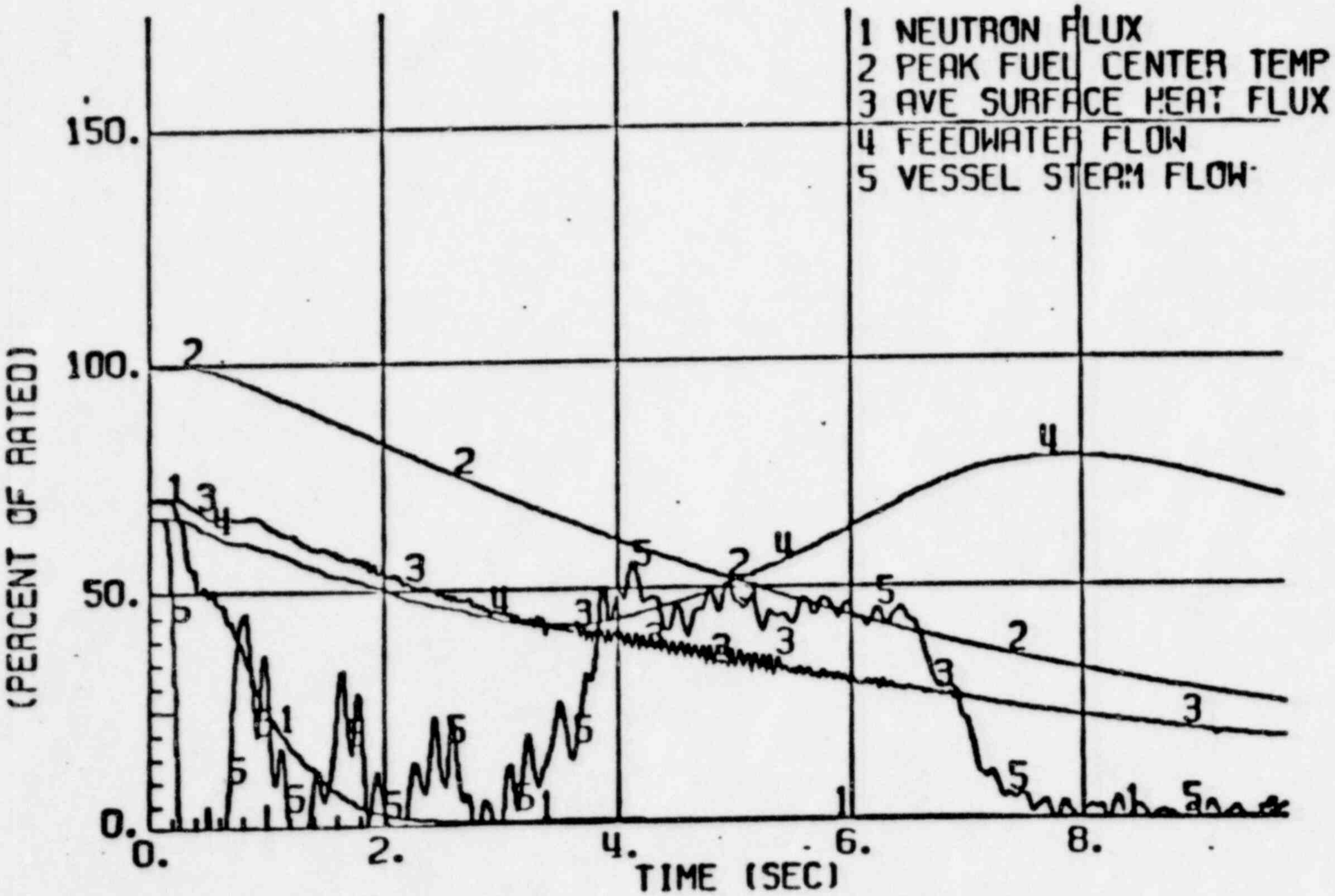
MISSISSIPPI POWER & LIGHT	FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-2



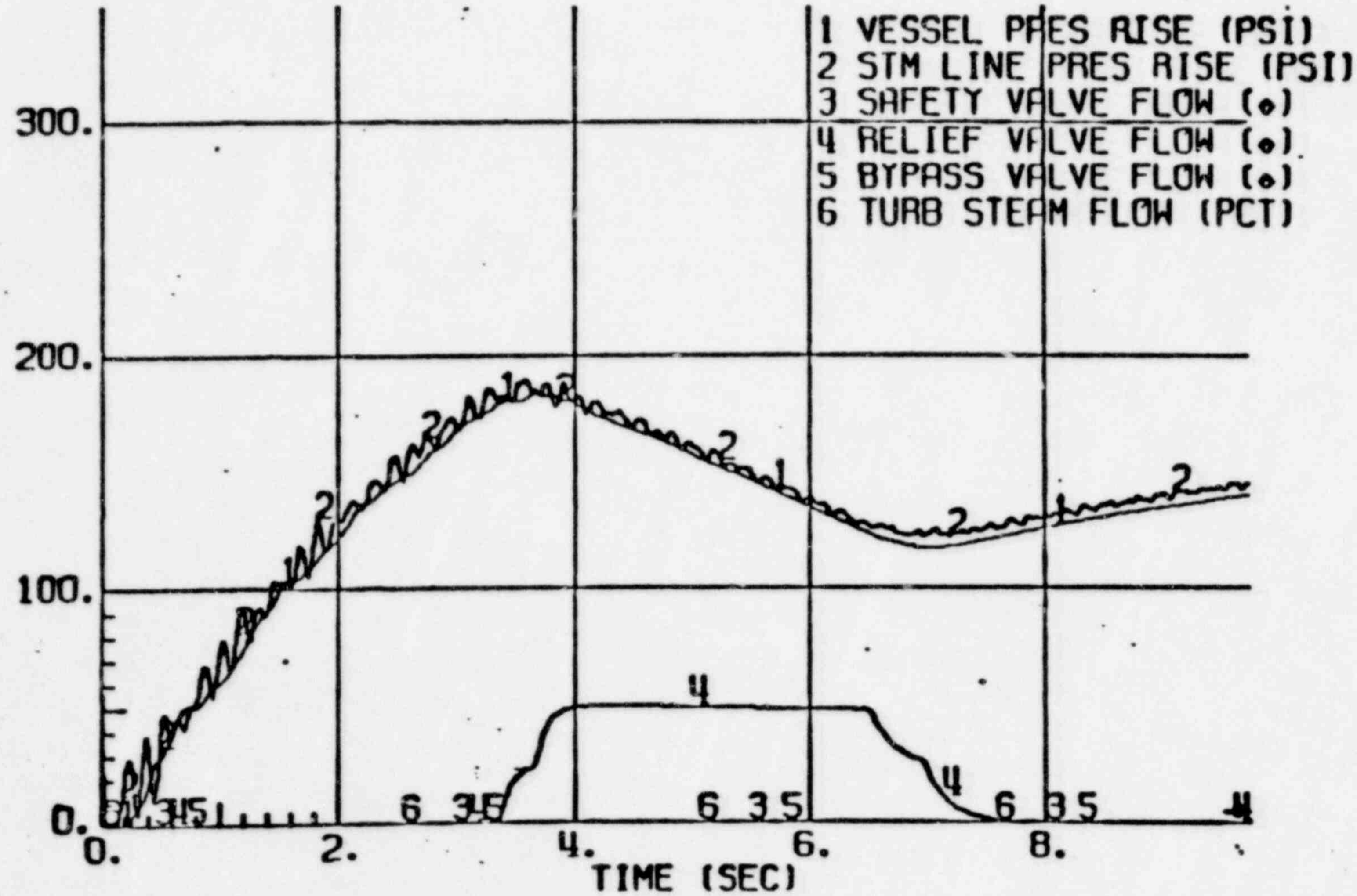
MISSISSIPPI POWER & LIGHT

GENERATOR LOAD REJECTION, BYPASS FAILURE, SINGLE LOOP OPERATION, 71%P/54%F

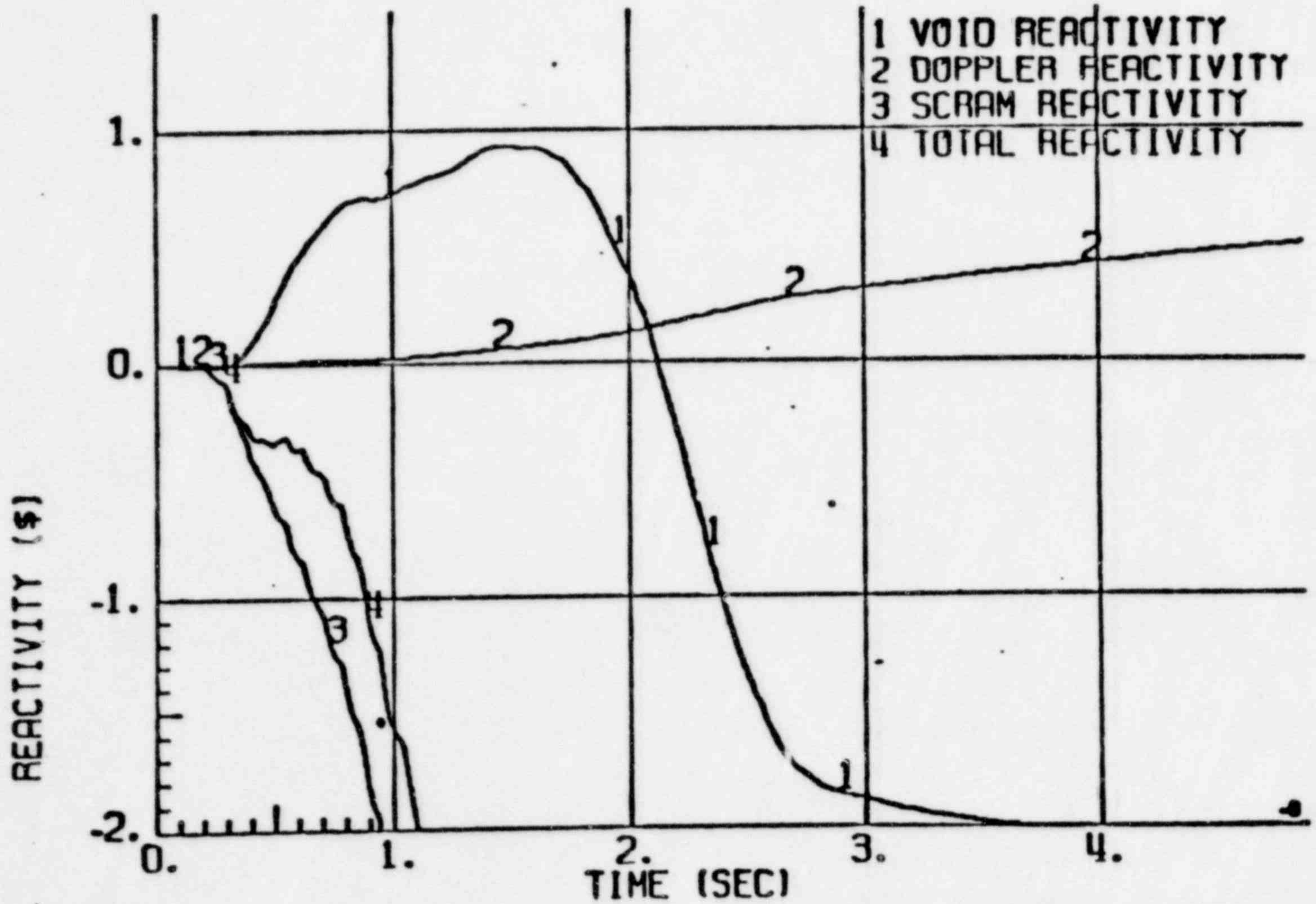
FIGURE 15.C.3-3



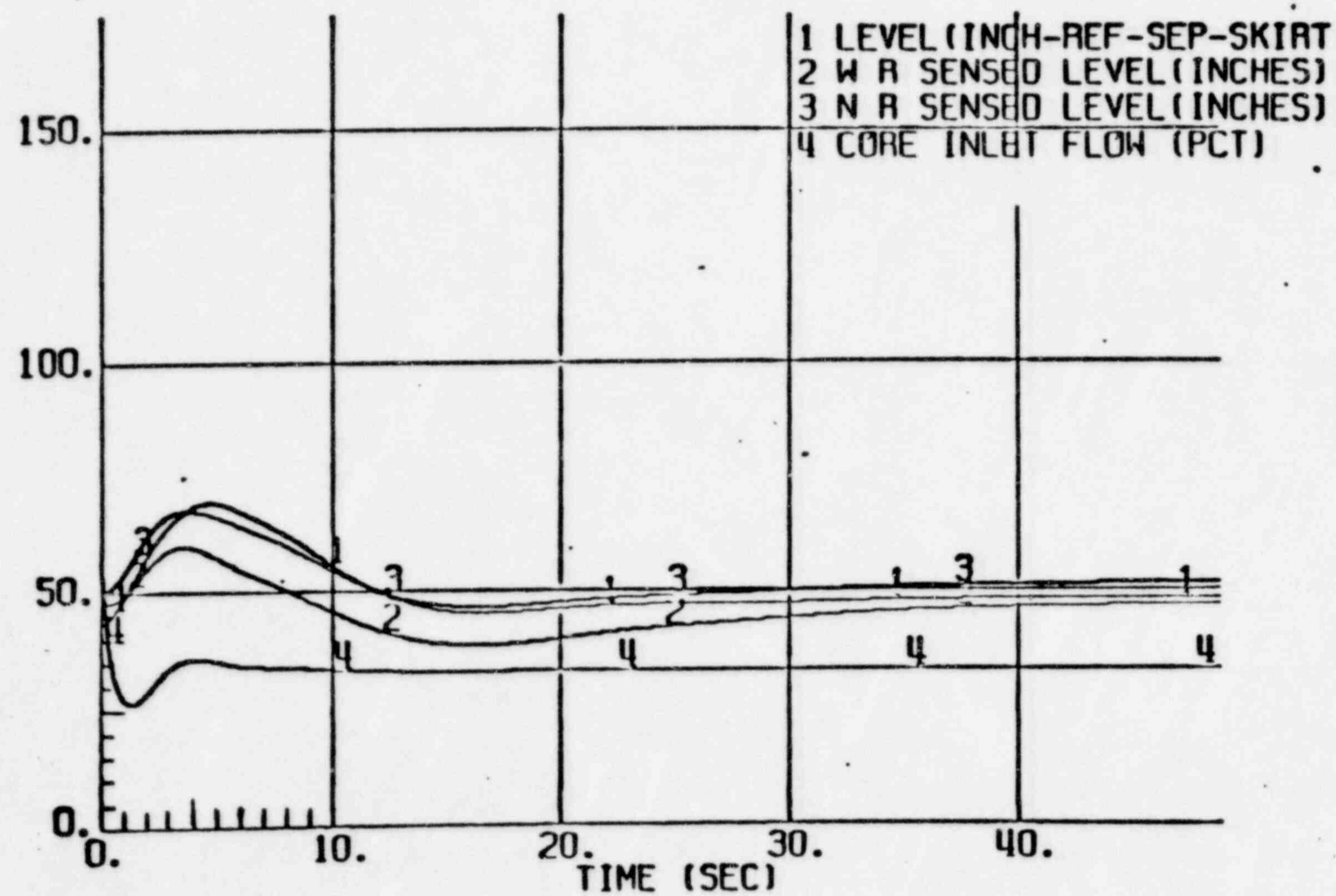
MISSISSIPPI POWER & LIGHT	GENERATOR LOAD REJECTION, BYPASS FAILURE, SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-3



MISSISSIPPI POWER & LIGHT	GENERATOR LOAD REJECTION, BYPASS FAILURE, SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-3



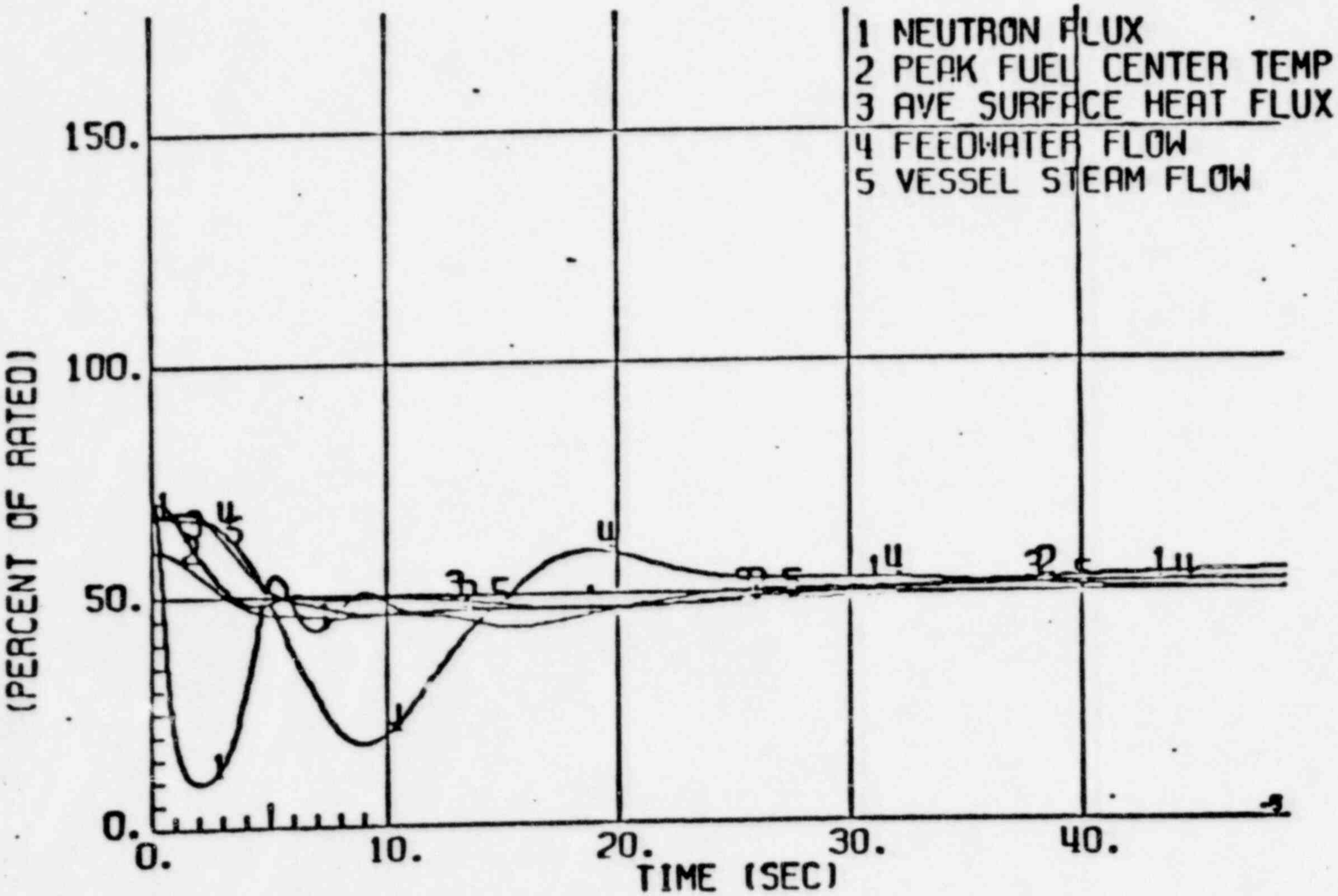
MISSISSIPPI POWER & LIGHT	GENERATOR LOAD REJECTION, BYPASS FAILURE, SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-3



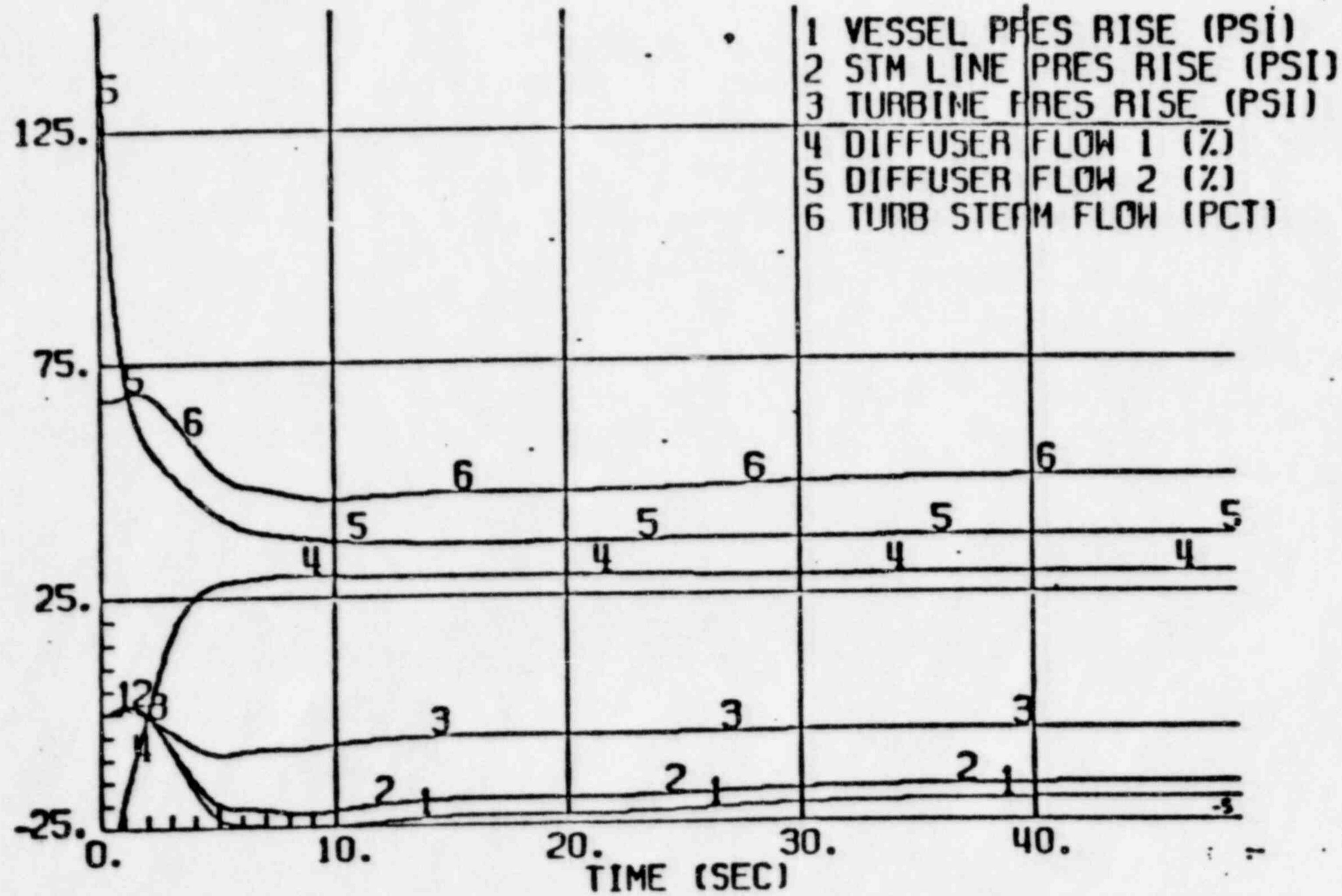
MISSISSIPPI POWER & LIGHT

PUMP SEIZURE SINGLE LOOP OPERATION, 71%P/54%F

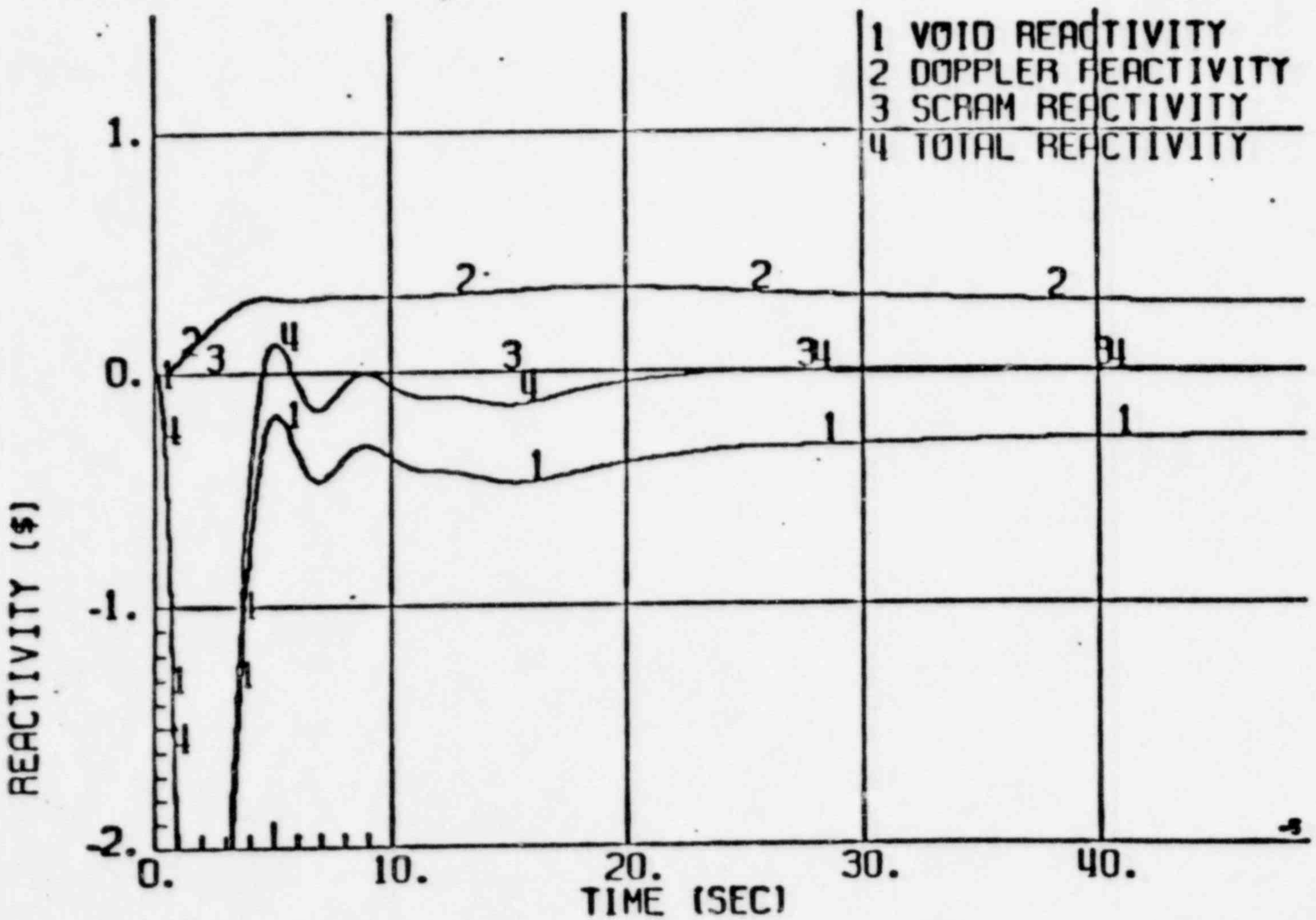
FIGURE 15.C.3-4



MISSISSIPPI POWER & LIGHT	PUMP SEIZURE SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-4



MISSISSIPPI POWER & LIGHT	PUMP SEIZURE SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-4



MISSISSIPPI POWER & LIGHT	PUMP SEIZURE SINGLE LOOP OPERATION, 71%P/54%F	FIGURE 15.C.3-4

15.C.4 STABILITY ANALYSIS

15.C.4.1 Phenomena

The least stable power/flow condition attainable under normal operating conditions (both reactor coolant system recirculation loops in operation) occurs at minimum flow and the highest achievable power level. For all operating conditions, the least stable power/flow condition may correspond to operation with one or both recirculation loops not in operation. The primary contributing factors to the stability performance with one or both recirculation loops not in service are the power/flow ratio and the recirculation loop characteristics. At natural circulation flow the highest power/flow ratio is achieved. At forced circulation with one recirculation loop not in operation, the reactor core stability may be influenced by the inactive recirculation loop. As core flow increases in SLO, the inactive loop forward flow decreases because the natural circulation driving head decreases with increasing core flow. The reduced flow in the inactive loop reduces the resistance that the recirculation loops impose on reactor core flow perturbations thereby adding a destabilizing effect. At the same time the increased core flow results in a lower power/flow ratio which is a stabilizing effect. These two countering effects may result in decreased stability margin (higher decay ratio) initially as core flow is increased (from minimum) in SLO and then an increase in stability margin (lower decay ratio) as core flow is increased further and reverse flow in the inactive loop is established.

As core flow is increased further during SLO and substantial reverse flow is established in the inactive loop an increase in jet pump flow, core flow and neutron noise is observed. A cross flow is established in the annular downcomer region near the jet pump suction entrance caused by the reverse flow of the inactive recirculation loop. This cross flow interacts with the jet pump suction flow of the active recirculation loop and increases the jet pump flow noise. This effect increases the total core flow noise which tends to drive the neutron flux noise.

To determine if the increased noise was being caused by reduced stability margin as SLO core flow was increased, an evaluation was performed which phenomenologically accounts for single loop operation effects on stability (Reference 15.C.8-4). The model predictions were initially compared to test data and showed very good agreement for both two loop and single loop test conditions. An evaluation was performed to determine the effect of reverse flow on stability during SLO. With increasing reverse flow, SLO exhibited slightly lower decay ratios than two loop operation. However, at low core flow conditions with no reverse flow, SLO was slightly less stable. This is consistent with observed behavior at stability tests at operating BWRs (Reference 15.C.8-5).

In addition to the above analyses, the cross flow established during reverse flow conditions was simulated analytically and shown to cause an increase in the individual and total jet pump flow noise, which is consistent with tests data (Reference 15.C.8-4). The results of these analyses and tests indicate that the stability characteristics are not significantly different from two loop operation. At low core flows, SLO may be slightly less stable than two loop operation but as core flow is increased and reverse flow is established the stability performance is similar. At even higher core flows with substantial reverse flow in the inactive recirculation loop, the effects of cross flow on the flow noise results in an increase in system noise (jet pump, core flow and neutron flux noise).

15.C.4.2 Compliance to Stability Criteria

Consistent with the philosophy applied to two loop operation, the stability compliance during single loop operation is demonstrated on a generic basis. Stability acceptance criteria have been established to demonstrate compliance with the requirements set forth in 10CFR50, Appendix A, General Design Criterion (GDC) 12 (Reference 15.C.8-6). A generic analyses which covers those fuels contained in the General Electric Standard Application for Reactor fuel (Reference 15.C.8-7) has been performed. The analyses demonstrates that in the event limit cycle neutron flux oscillations occur within the bounds of safety system intervention, specified acceptable fuel design limits are not exceeded. Since the reactor core is assumed to be in an oscillatory mode, the question of stability margin during SLO is not relevant from a safety standpoint (i.e. the analysis already assumes no stability margin).

The fuel performance during limit cycle oscillations is characteristically dependent on fuel design and certain fixed system features (high neutron flux scram setpoint, channel inlet orifice diameter, etc.). Therefore the acceptability of GE fuel designs independent of plant and cycle parameters has been established. Only those parameters unique to SLO which affect fuel performance need to be evaluated. The major consideration of SLO is the increased Minimum Critical Power Ratio (MCPR) safety limit caused by increased uncertainties in system parameters during SLO. However, the increase in MCPR safety limit (0.01) is well within the margin of the limit cycle analyses (Reference 15.C.8-6) and therefore it is demonstrated that stability compliance criteria are satisfied during single loop operation. Operationally, the effects of higher flow noise and neutron flux noise observed at high SLO core flows are evaluated to determine if acceptable vessel internal vibration levels are met and to determine the effects on fuel and channel fatigue. However, these are not considered in the compliance to stability criteria but are instead addressed on a plant specific basis. These evaluations are addressed in Section 15.C.7.

A Service Information Letter-380, Revision 1 (Reference 15.C.8-8) has been developed to inform plant operators how to recognize and suppress unanticipated oscillations when encountered during plant operation. Evaluation of additional SLO test data taken from an operating BWR in late 1983 has been completed. Results of which have been documented in revision 1 of the reference 15.C.8-6 report (NEDE-22277-P-1). These efforts combined with the analyses previously documented in References 15.C.8-4 and 15.C.8-6 provide justification that GGNS can operate at the highest achievable power with a single recirculation loop in operation.

15.C.5 LOSS-OF-COOLANT ACCIDENT ANALYSIS

An analysis of single recirculation loop operation using the models and assumptions documented in Reference 15.C.8-9 was performed for GGNS. Using this method, SAFE/REFLOOD computer code runs were made for a full spectrum of large break sizes for only the recirculation suction size breaks (most limiting for GGNS). Because the reflood minus uncovery time for the single-loop analysis is similar to the two-loop analysis, the maximum planar linear heat generation rate (MAPLHGR) curves were modified by derived reduction factors for use during one recirculation pump operation.

15.C.5.1 BREAK SPECTRUM ANALYSIS

SAFE/REFLOOD calculations were performed using assumptions given in Section II.A.7.3.1 of Reference 15.C.8-9. Hot node uncovered time (time between uncovery and reflood) for single-loop operation is compared to that for two-loop operation in Figure 15.C.5-1.

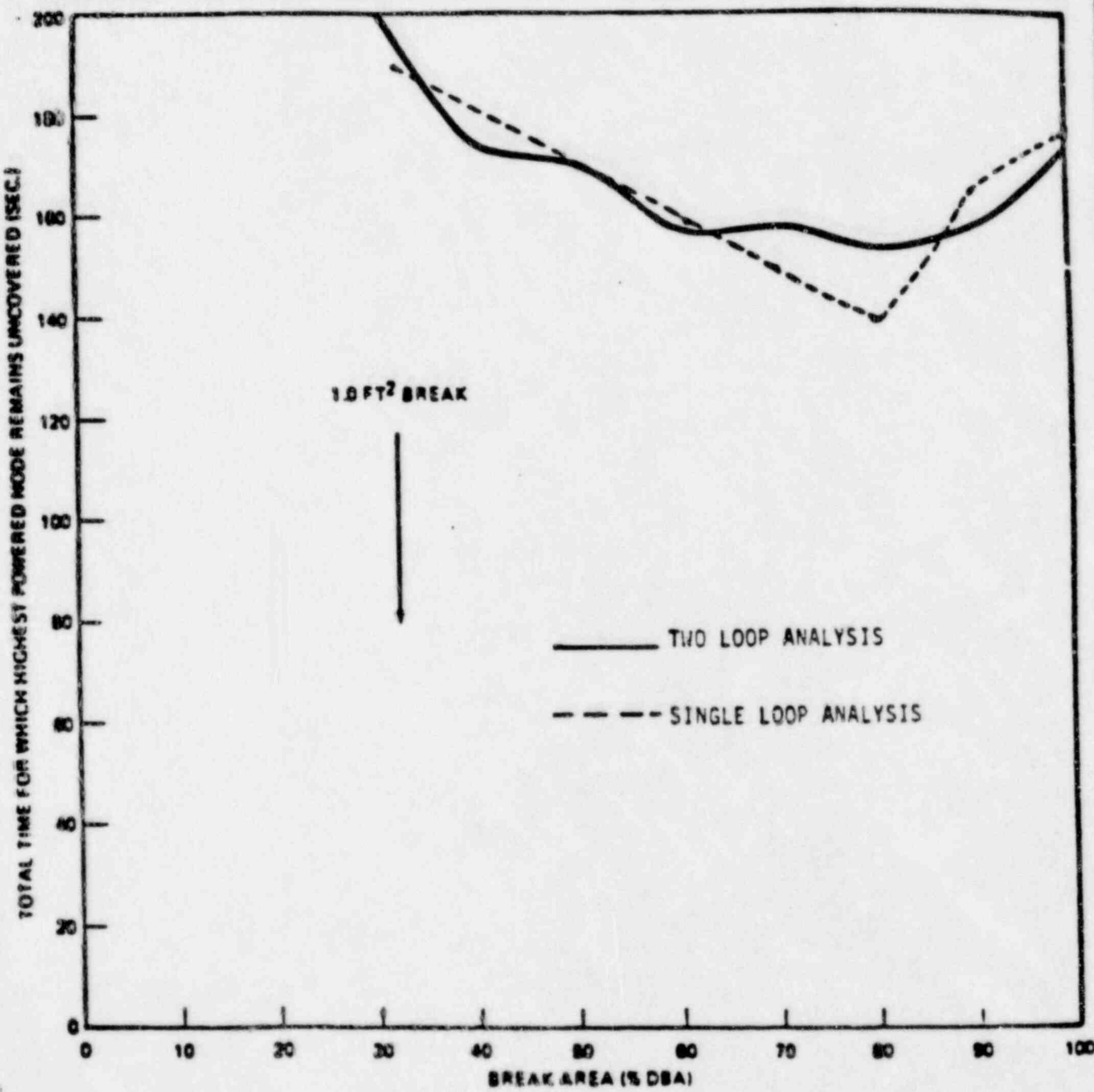
The total uncovered time for two-loop operation is 174 seconds for the 100% DBA suction break. This is the most limiting break for two-loop operation. For single-loop operation, the total uncovered time is 177 seconds and for the 100% DBA suction break. This is the most limiting break for single-loop operation. In both cases, the 1.0 ft^2 suction break has a longer total uncovered time but results in a less severe PCT response due to a later uncovery time.

15.C.5.2 SINGLE-LOOP MAPLHGR DETERMINATION

The small differences in uncovered time and reflood time for the limiting break size would result in a small change in the calculated peak cladding temperature. Therefore, as noted as Reference 15.C.8-9, the one and two-loop SAFE/REFLOOD results can be considered similar and the generic alternate procedure described in Section II.A.7.4. of this reference was used to calculate the MAPLHGR reduction factors for single-loop operation. The most limiting single-loop operation MAPLHGR reduction factor (i.e., yielding the lowest MAPLHGR) for GE6 8x8 retrofit-fuel is 0.86. One-loop operation MAPLHGR values are derived by multiplying the current two-loop MAPLHGR values by the reduction factor (0.86). As discussed in Reference 15.C.8-9, single recirculation loop MAPLHGR values are conservative when calculated in this manner.

15.C.5.3 SMALL BREAK PEAK CLADDING TEMPERATURE

Section II.A.7.4.4.2 of Reference 15.C.8-9 discusses the low sensitivity of the calculated peak cladding temperature (PCT) to the assumptions used in the one-pump operation analysis and the duration of nucleate boiling. As this slight increase (~ 50°F) in PCT is overwhelmingly offset by the decreased MAPLHGR (equivalent to 300°F to 500°F PCT) for one-pump operation, the calculated PCT values for small breaks will be well below the 1404°F small break PCT value previously reported for GGNS, and significantly below the 2200°F 10CFR50.46 cladding temperature limit.



MISSISSIPPI POWER & LIGHT	Uncovered Time vs Break Area Suction Break, LPCS Failure	FIGURE 15.C.5-1

15.C.6 CONTAINMENT ANALYSIS

A single-loop operation containment analysis was performed for GGNS based on a bounding analysis performed for a standard BWR6 plant. The peak wetwell pressure, peak drywell pressure, chugging loads, condensation oscillation and pool and swell containment load responses were estimated over the entire single-loop operation power/flow region.

The analysis shows peak drywell and wetwell pressures for the worst single loop operation condition of 34.5 psia and 21 psia, respectively. The corresponding differential peak drywell and wetwell pressures are 19.8 psig and 6.3 psig which is less than the 22 psig and 9.9 psig reported in Chapter 6. The chugging loads, condensation oscillation download and pool swell velocity evaluated at the worst power/flow condition during single-loop operation were also found to be bounded by the rated power analysis.

15.C.7 MISCELLANEOUS IMPACT EVALUATION

15.C.7.1 Anticipated Transient Without Scram (ATWS) Impact Evaluation

The principal difference between single loop operation (SLO) and normal two loop operation (TLO) affecting Anticipated Transient Without Scram (ATWS) performance is that of initial reactor conditions. Since the SLO initial power flow condition is less than the rated condition used for TLO ATWS analysis, the transient response is less severe and therefore bounded by the TLO analyses. All ATWS acceptance criteria are met during SLO. Therefore, SLO is an acceptable mode of operation for ATWS considerations.

15.C.7.2 Fuel Mechanical Performance

The thermal and mechanical duty for the transients analyzed have been evaluated and found to be bounded by the fuel design bases.

It is observed that due to the substantial reverse flow established during SLO both the Average Power Range Monitor (APRM) noise and core plate differential pressure noise are slightly increased. An analysis has been carried out to determine that the APRM fluctuation should not exceed a flux amplitude of ±15% of rated and the core plate differential pressure fluctuation should not exceed 3.2 psi peak to peak to be consistent with the fuel rod and assembly design bases.

15.C.7.3 Vessel Internal Vibration

A recirculation pump drive flow limit will be imposed for SLO. The highest drive flow tested during the startup test program at GGNS that show acceptable vessel internal vibration criteria will be the drive flow limit for SLO.

A preliminary assessment has been made for the expected reactor vibration level during SLO for GGNS.

Before providing the results of the assessment, it is prudent to define the term "maximum flow" during balanced 2-loop operation and single loop operation. Maximum flow for two-pump balanced operation is equal to rated volumetric core flow at normal reactor operating conditions. Maximum flow for single-pump operation is that flow obtained with the recirculation pump drive flow equal to that required for maximum flow during two-pump balanced operation. For rated reactor water temperature and pressure, this maximum flow for GGNS is about 44,600 gpm.

During the GE BWR-6 jet pump development tests at GE test facility HF^2, the reactor internal components were subjected to the maximum flows, as defined above, for both two-pump balanced and single-loop operating conditions. All components were found to be within acceptance limits with the exception of in-core guide tube during single-loop operation. Due to the non-prototypical configuration of the in-core guide tube supports at HF^2, it was decided that no design changes need to be made. Instead, the in-core guide tube was to be monitored for vibration response at the Kuo Sheng 1 plant. Startup tests at the Kuo Sheng 1 plant showed all components, including the in-core guide tube during single-loop operation, to have vibration levels within acceptance limits.

From the above, it can be inferred that the vibration levels of the reactor internal components for GGNS would be expected to be within acceptance limits during single-loop operation with maximum flow as defined above. However, since GGNS reactor internals have extensive instrumentation, final and definitive conclusions can be arrived after vibration data acquisition and data reduction are completed.

15.C.8 REFERENCES

15.C.8-1 "General Electric BWR Thermal Analysis Basis (GETAB); Data, Correlation, and Design Application", NEDO-10958-A, January 1977.

15.C.8-2 "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", NEDO-24154, October 1978.

15.C.8-3 R. B. Linford, "Analytical Methods of Plant Transients Evaluation for the General Electric Boiling Water Reactor", NEDO-10802, April 1973.

15.C.8-4 Letter, H. C. Pfefferlen (GE) to C. O. Thomas (NRC), "Submittal of Response to Stability Action Item from NRC Concerning Single-Loop Operation," September 1983.

15.C.8-5 S. F. Chen and R. O. Niemi, "Vermont Yankee Cycle 8 Stability and Recirculation Pump Trip Test Report", General Electric Company, August 1982 (NEDE-25445, Proprietary Information).

15.C.8-6 G. A. Watford, "Compliance of the General Electric Boiling Water, Reactor Fuel Designs to Stability Licensing Criteria", General Electric Company, December 1982 (NEDE-22277-P, Proprietary Information).

15.C.8-7 "General Electric Standard Application for Reload Fuel", General Electric Company, January 1982 (NEDE-24011-P-A-4).

15.C.8-8 "BWR Core Thermal Hydraulic Stability", General Electric Company, February 10, 1984 (Service Information Letter-380, Revision 1).

15.C.8 REFERENCES (Cont'd)

15.C.8-9 "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K Amendment No. 2 - One Recirculation Loop Out-of-Service", NEDO-20566-2 Revision 1, July 1978.