NEDO-31441
DRF A00-02463
CLASS I
MAY 1987

SINGLE-LOOP OPERATION ANALYSIS FOR RIVER BEND STATION, UNIT 1

GENERAL ELECTRIC

NEDO-31441
DRF A00-02463
Class I
May 1987

SINGLE-LOOP OPERATION ANALYSIS
FOR
RIVER BEND STATION
UNIT 1

Approved: R. J. Brandon, Manager
Engineering Analysis Services

Approved: D. J. Robare, Manager
Plant Licensing Services

Approved: J. S. Charnley, Manager
Fuel Licensing

NUCLEAR ENERGY BUSINESS OPERATIONS • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of General Electric Company respecting information in this document are contained in the contract between Gulf States Utility (GSU) and General Electric Company, as identified in the purchase order for this report and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than GSU or for any purpose other than that for which it is intended, is not authorized; and with respect to any unauthorized use, General Electric Company makes no representation or warranty, and assumes no liability as to the completeness, accuracy or usefulness of the information contained in this document.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1. INTRODUCTION AND SUMMARY

The capability of operation at reduced power with a single recirculation loop, single-loop operation (SLO), is highly desirable in the event that a recirculation pump or other component maintenance renders one loop inoperative. To justify SLO, accidents and abnormal operational transients associated with power operations [as presented in the Final Safety Analysis Report (FSAR), Sections 6.2 and 6.3, and 15.0] were reviewed for the case with only one recirculation loop in operation. This report presents the results of the safety evaluation for the operation of the River Bend Station (RBS) with an inoperable single recirculation loop. This evaluation is performed on an equilibrium cycle basis and is applicable to initial and reload cycle P8x8R fuel operation. This evaluation is for continued operation in the operating domain currently defined in FSAR Figure 4.4.5 of Chapter 4, up to maximum power of 70.2% 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 SLO. No increase in rated MCPR operating limit and no change in the power-dependent and flow-dependent MCPR limit ($MCPR_p$ and $MCPR_f$) are required because all abnormal operational transients analyzed for SLO indicate that 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 equations given in the RBS Technical Specifications must be adjusted for one-loop 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 controller in the active loop should be in the manual mode for SLO.

The limiting maximum average planar linear heat generation rate (MAPLHGR) reduction factor for SLO is calculated to be 0.84.

The containment response for a design basis accident (DBA) recirculation line break with SLO is bounded by the rated power two-loop operation analysis presented in Section 6.2. This is valid for all SLO power/flow conditions.

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

It was determined that the fuel will operate within the fuel mechanical design bases during normal steady-state operation and anticipated operational occurrences under SLO.

A recirculation pump drive flow limit is imposed for SLO in order to meet acceptable vessel internal vibration criteria. Actual drive flow limit for SLO was determined at Kuo Sheng 1, the BWR6/218 prototype plant, and is about 33,000 gpm.

2. MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

Except for core total flow and TIP readings, 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, Figure 4.4-9. A 6% core flow measurement uncertainty has been established for SLO (compared to 2.5% for two-loop operation). As shown in Subsection 2.1, this value conservatively reflects the one standard deviation (1σ) accuracy of the core flow measurement system documented in Reference 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 2.2. This revision resulted in an SLO process computer effective TIP uncertainty of 6.8% for 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 increase in the required MCPR fuel cladding integrity safety limit.

2.1 CORE FLOW UNCERTAINTY

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 SLO, however, some inactive jet pumps will be backflowing (at active pump flow above approximately 35%). 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 SLO, the total core flow is derived by the following formula:

$$\text{Total Core Flow} = \begin{pmatrix}\text{Active Loop}\\ \text{Indicated Flow}\end{pmatrix} - C \begin{pmatrix}\text{Inactive Loop}\\ \text{Indicated Flow}\end{pmatrix}$$

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 two steps: (a) calibrating core support plate ΔP versus core flow during one-pump and two-pump operation along the 100% flow control line; and (b) calculating the correct value of C based on the core support plate ΔP and the loop flow indicator readings.

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 1. The analysis of one-pump core flow uncertainty is summarized as follows.

For SLO, the total core flow can be expressed as follows (see Figure 2-1):

$$W_C = W_A - W_I$$

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

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 preceding equation, the variance of the total flow uncertainty can be approximated by:

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

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

$\sigma_{W_{sys}}$, $\sigma_{W_{A_{rand}}}$, $\sigma_{W_{I_{rand}}}$ and σ_C 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:

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

$$= (5.0\%)^2$$

When the effect of a 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:

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

which is less than 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.

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 which was 59.3% of rated with a single recirculation pump in operation (core flow was 46.3% of rated). A rotationally symmetric control rod pattern existed during the test.

---

*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 RBS 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 1σ process computer total effective TIP uncertainty value for SLO of 6.8% for initial cores and 9.1% for reload cores.

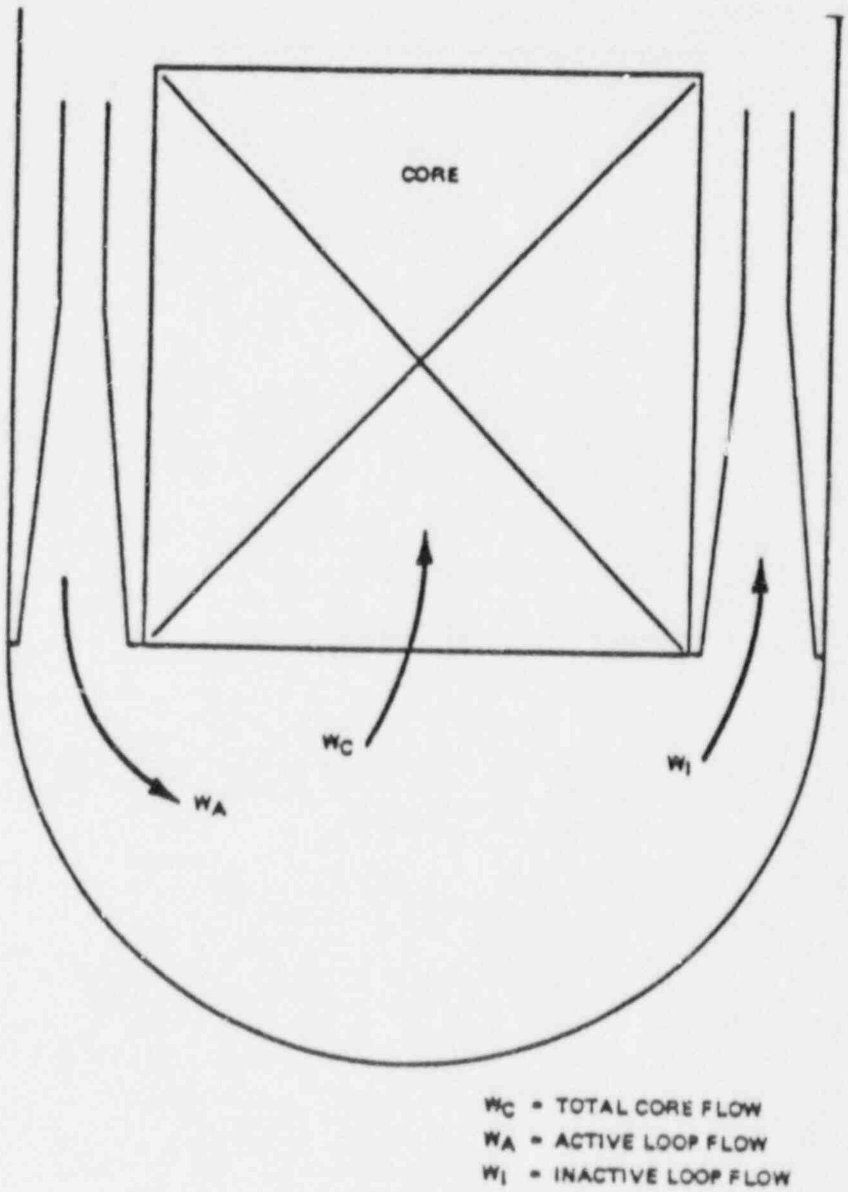


Figure 2-1. Single Recirculation Loop Operation Flows

3. MCPR OPERATING LIMIT

3.1 CORE-WIDE 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 operating 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, the results presented in Chapter 15 of the FSAR bound both the thermal and overpressure consequences of one-loop operation.

Figure 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, where the maximum initial power level is about 70%, 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 assuming both recirculation loop controllers fail. This condition produces the maximum possible power increase and, hence, maximum ΔCPR 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 lone 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 that 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 (Chapter 15.4.4) and is still applicable for single-loop operation.

From the preceding 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, the results of two of the most limiting transients analyzed for single-loop operation are presented. They are, respectively:

(1) Feedwater flow controller failure (maximum demand) (FWCF)

(2) Generator load rejection with bypass failure (LRBPF)

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

3.1.1 Feedwater Controller Failure - Maximum Demand

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

The analysis has been performed with the plant conditions tabulated in Table 3-1, except the initial vessel water level was at Level 4 for conservatism. By lowering the initial water level, more cold feedwater will be injected before Level 8 is reached, resulting in higher heat fluxes.

The simulated feedwater controller failure transient is shown in Figure 3-2.

Table 3-2 gives a summary of the transient analysis results. The calculated MCPR is 1.27, which is well above the safety limit MCPR of 1.07; therefore, no fuel failure due to boiling transition is predicted. The peak vessel pressure predicted is 1165 psig and is well below the ASME limit of 1375 psig.

3.1.2 Generator Load Rejection With Bypass Failure

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

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

The simulated generator load rejection with bypass failure is shown in Figure 3-3.

Table 3-2 summarizes the transient analysis results. The peak vessel pressure predicted is 1182 psig, which is well below the ASME limit of 1375 psig. The calculated MCPR is 1.34, which is considerably above the safety limit MCPR of 1.07.

3.1.3 Summary and Conclusions

The transient peak value and the critical power ratio (CPR) results are summarized in Table 3-2. This table indicates that for the transient events analyzed herein, the MCPRs for all transients are above the single-loop operation safety limit value of 1.07. It is concluded that the operating limit MCPRs established for two-pump operation are also applicable to single-loop operation conditions. For overpressure protection, Table 3-2 indicates that the peak pressures are below the ASME code value of 1375 psig. Hence, it is concluded that the pressure barrier integrity is maintained under single-loop operation conditions.

3.2 ROD WITHDRAWAL ERROR

The rod withdrawal error (RWE) transient for two-loop operation documented in Chapter 15 employs a statistical evaluation of the minimum critical power ratio (MCPR) 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 analysis covered all off-rated conditions 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 upcoming APRM rod block equation is required to maintain the two-loop rod block versus power relationship when in one-loop operation.

One-pump operation results in backflow through 10 of the 20 jet pumps, while the flow is being supplied into the lower plenum from the 10 active jet pumps. As a result of the backflow through the inactive jet pumps, the present rod block equation was conservatively modified for use during one-pump operation. This is because the direct active-loop flow measurement may not indicate actual flow above about 35% 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 (this value is expected to be 8% of rated);

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 previously discussed.

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 increase in the 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 that 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 3-1

INPUT PARAMETERS AND INITIAL CONDITIONS

1.	Thermal Power Level, MWt Analysis Value	2032
2.	Steam Flow, lb/sec Analysis Value	2286
3.	Core Flow, lb/hr	4.53×10^7
4.	Feedwater Flow Rate, lb/sec Analysis Value	2286
5.	Feedwater Temperature, °F	383
6.	Vessel Dome Pressure, psig	978
7.	Core Pressure, psig	983
8.	Turbine Bypass Capacity, % NBR	10
9.	Core Coolant Inlet Enthalpy, Btu/lb	509.5
10.	Turbine Inlet Pressure, psig	943
11.	Fuel Lattice	P8x8R
12.	Core Leakage Flow, %	12.7
13.	Required MCPR Operating Limit First Core	1.39
14.	MCPR Safety Limit for Incident of Moderate Frequency	
	First Core	1.07
	Reload Core	1.08
15.	Doppler Coefficient (-)/°F Analysis Data	a
16.	Void Coefficient (-)/% Rated Voids	a
17.	Scram Reactivity Analysis Data	a
18.	Control Rod Drive Speed Position versus Time	See FSAR Figure 15.0-3

[a] These values are calculated within the computer code (Reference 2).

Table 3-1 (Continued)

19.	Core Average Rated Void Fraction, %	45.7
20.	Jet Pump Ratio, M	2.47
21.	Safety/Relief Valve Capacity, % NBR at 1210 psig Manufacturer Quantity Installed	 109.4 Crosby 16
22.	Relief Function Delay, sec	0.40
23.	Relief Function Response Time Constant, sec	0.10
24.	Safety Function Delay, sec	0.0
25.	Safety Function Response Time Constant, sec	0.2
26.	Setpoints for Safety/Relief Valves Safety Function, psig Relief Function, psig	 1175,1185,1195,1205,1215 1145,1155,1165,1175
27.	Number of Valve Groupings Simulated Safety Function Relief Function	 5 4
28.	SRV Reclosure Setpoint - Both Modes (% of Setpoint) Maximum Safety Limit Minimum Operational Limit	 98 89
29.	High Flux Trip, % NBR Analysis Setpoint (122x1.042), % NBR	127.2
30.	High Pressure Scram Setpoint, psig	1095
31.	Vessel Level Trips, Feet Above Separator Skirt Bottom Level 8 - (L8), ft Level 4 - (L4), ft Level 3 - (L3), ft Level 2 - (L2), ft	 6.00 3.87 1.94 -(2.86)
32.	APRM Thermal Trip Setpoint, % NBR (114.x1.042)	118.8
33.	TPM Time Constant, sec	7.0

Table 3-1 (Continued)

34.	RPT Delay on Load Rejection or Turbine Trip, sec	0.14
35.	RPT Inertia Time Constant for Analysis, sec	5.0
36.	Total Steamline Volume, ft^3	3275
37.	Pressure Setpoint of Recirculation Pump Trip - psig (Nominal)	1135

Table 3-2

SUMMARY OF TRANSIENT PEAK VALUE AND CPR RESULTS

	LRBFF	FWCF
Initial Power/flow, % Rated	70.2/53.6	70.2/53.6
Peak Neutron Flux, % NBR	70.3	84.4
Peak Heat Flux, % Initial	100.3	107.4
Peak Dome Pressure, psig	1169	1153
Peak Vessel Bottom Pressure, psig	1182	1165
Required Two-Loop Initial MCPR Operating Limit at SLO Condition	1.39	1.39
ΔCPR	0.05	0.12
Transient MCPR	1.34	1.27
SLMCPR at SLO	1.07	1.07

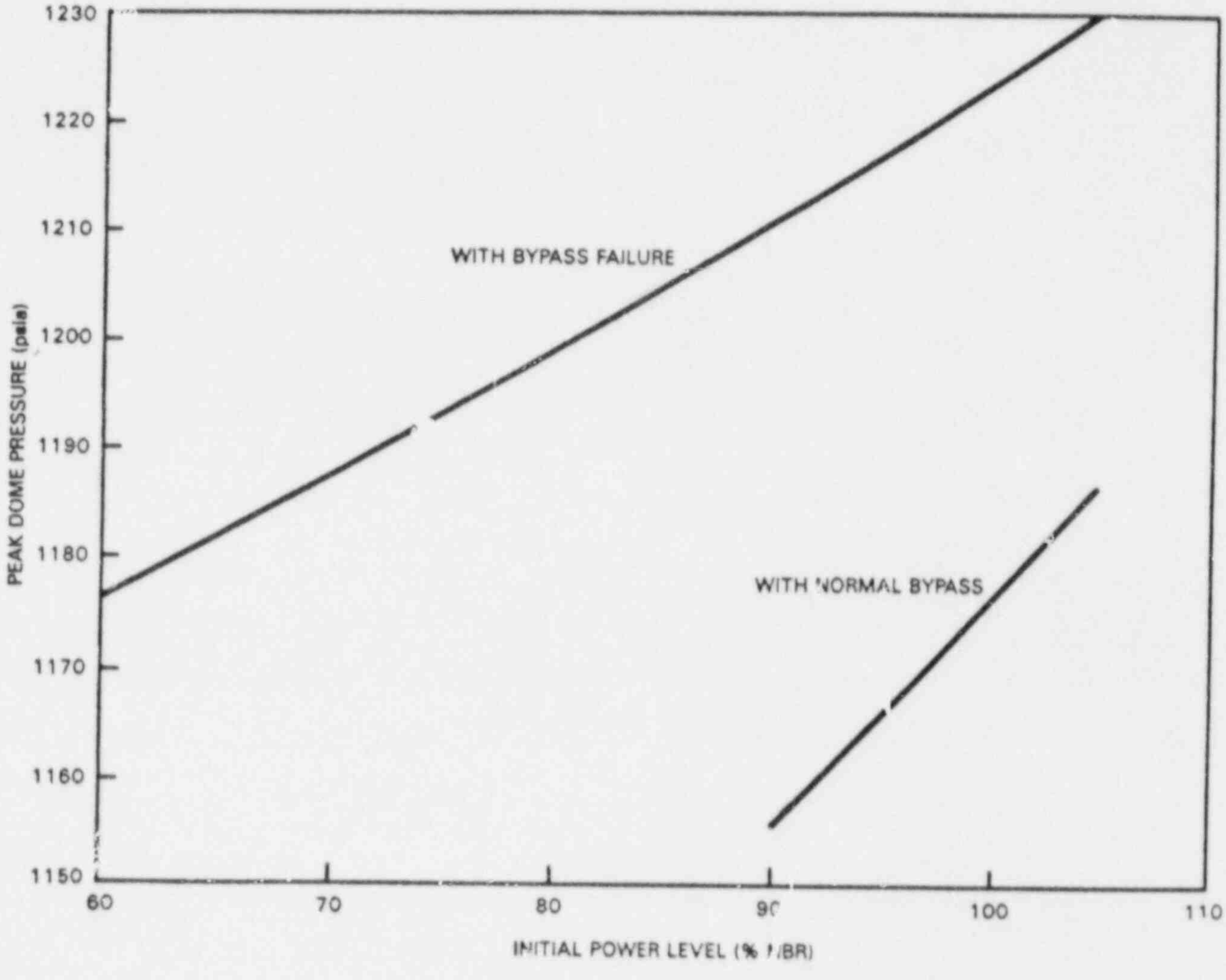


Figure 3-1. Peak Dome Pressure vs. Initial Power Level, Turbine Trip at EOEC

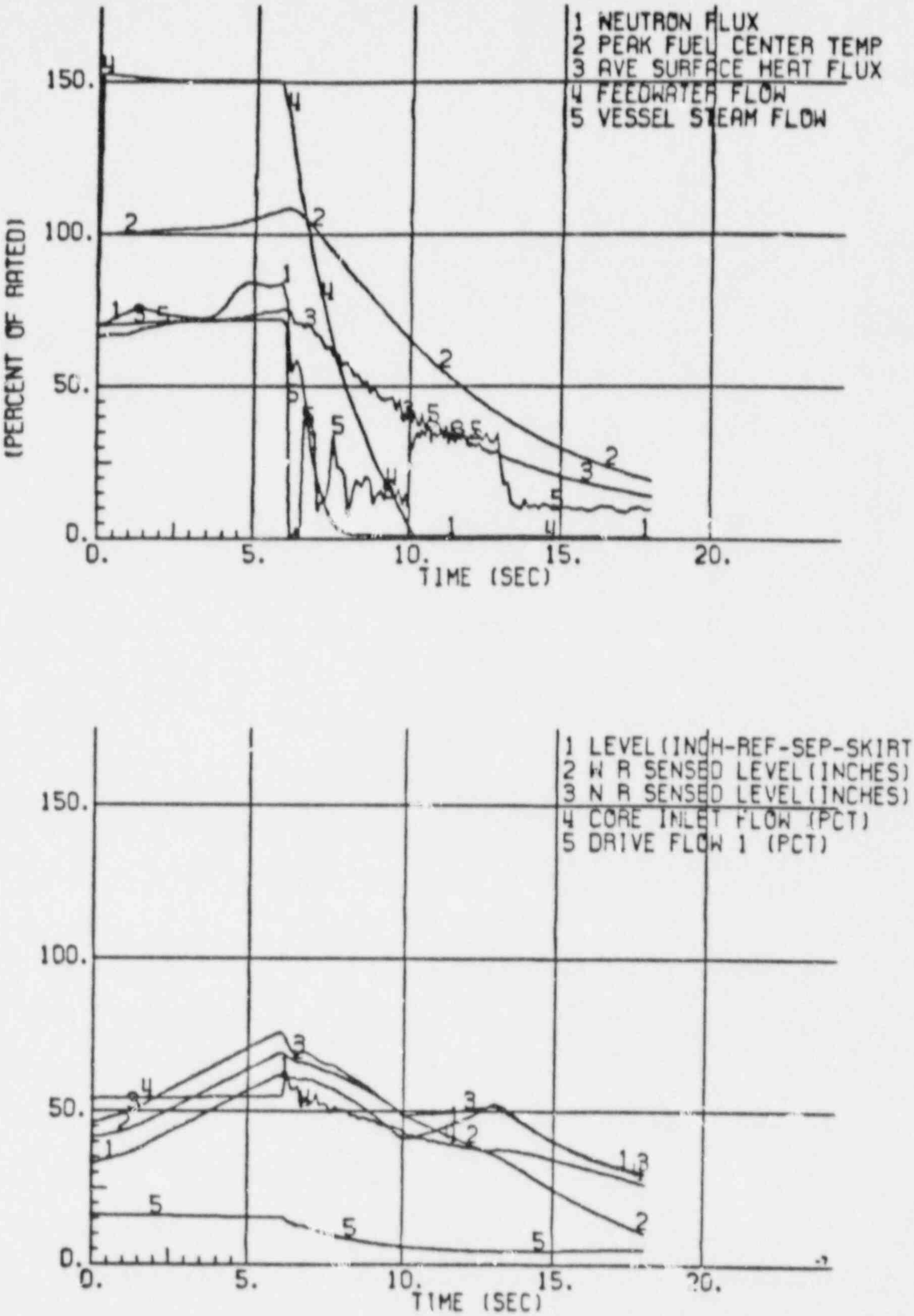


Figure 3-2. Feedwater Controller Failure Maximum Demand, 70.2% Power, 53.6% Flow

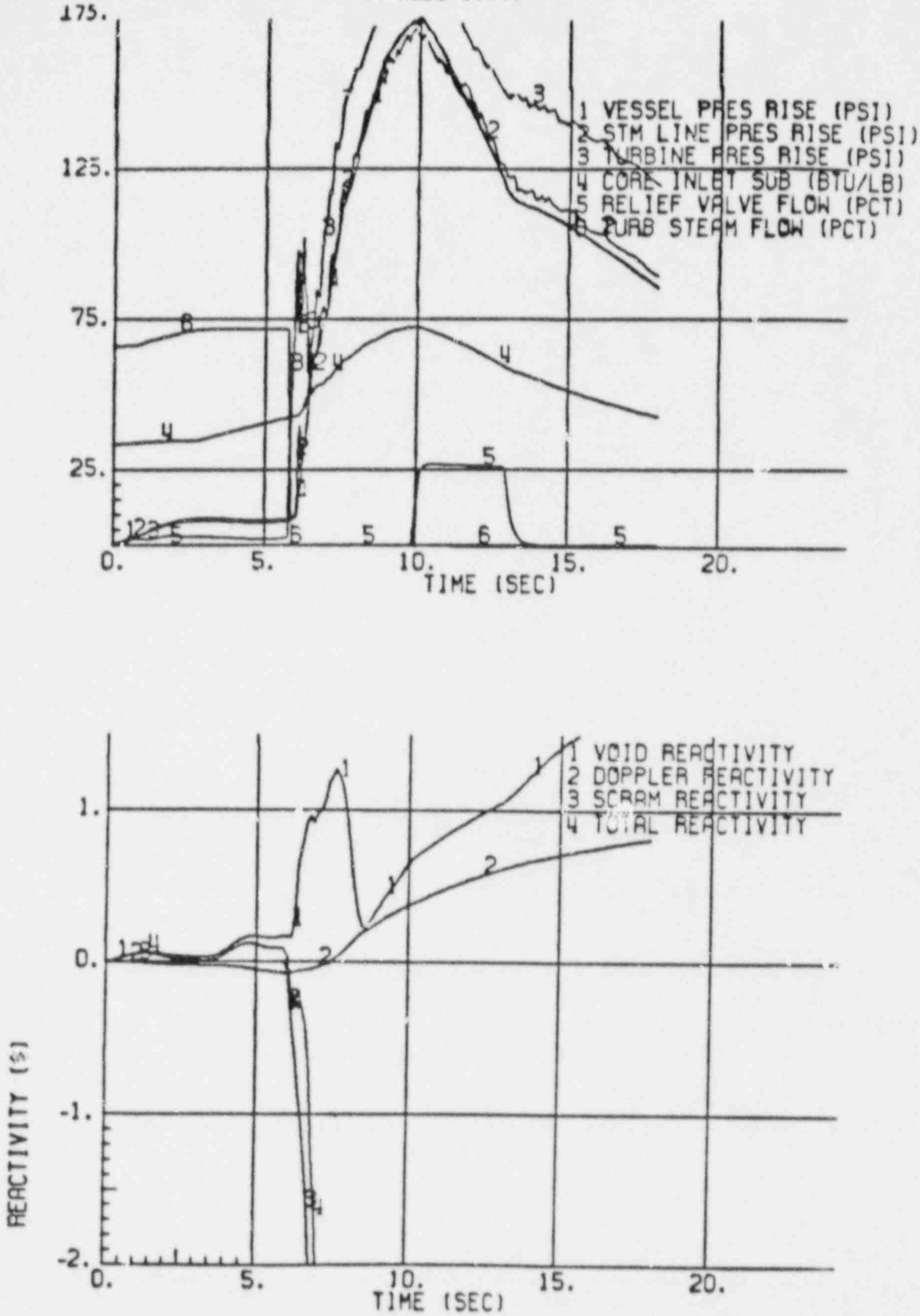


Figure 3-2. Feedwater Controller Failure - Maximum Demand, 70.2% Power, 53.6% Flow (Continued)

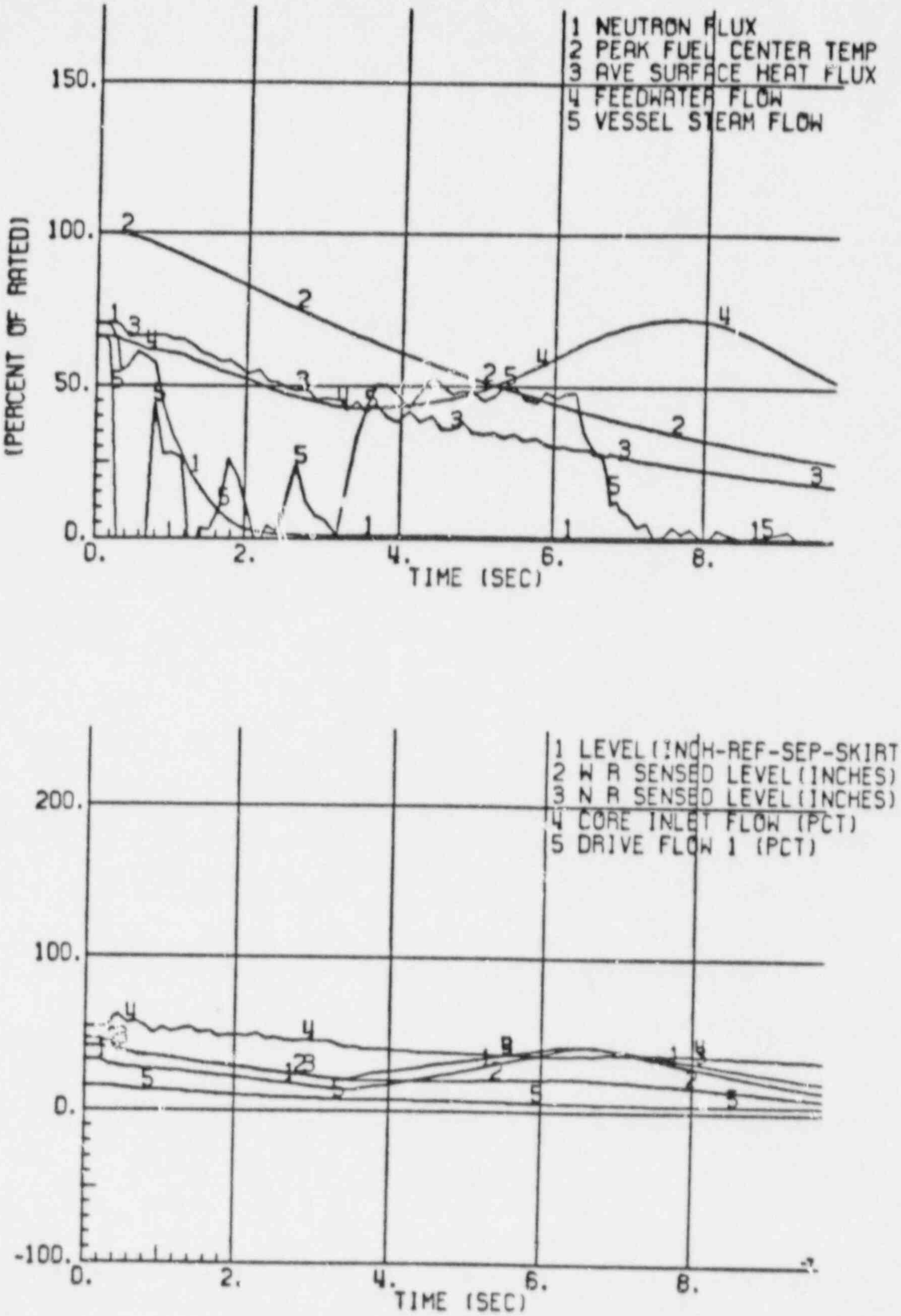
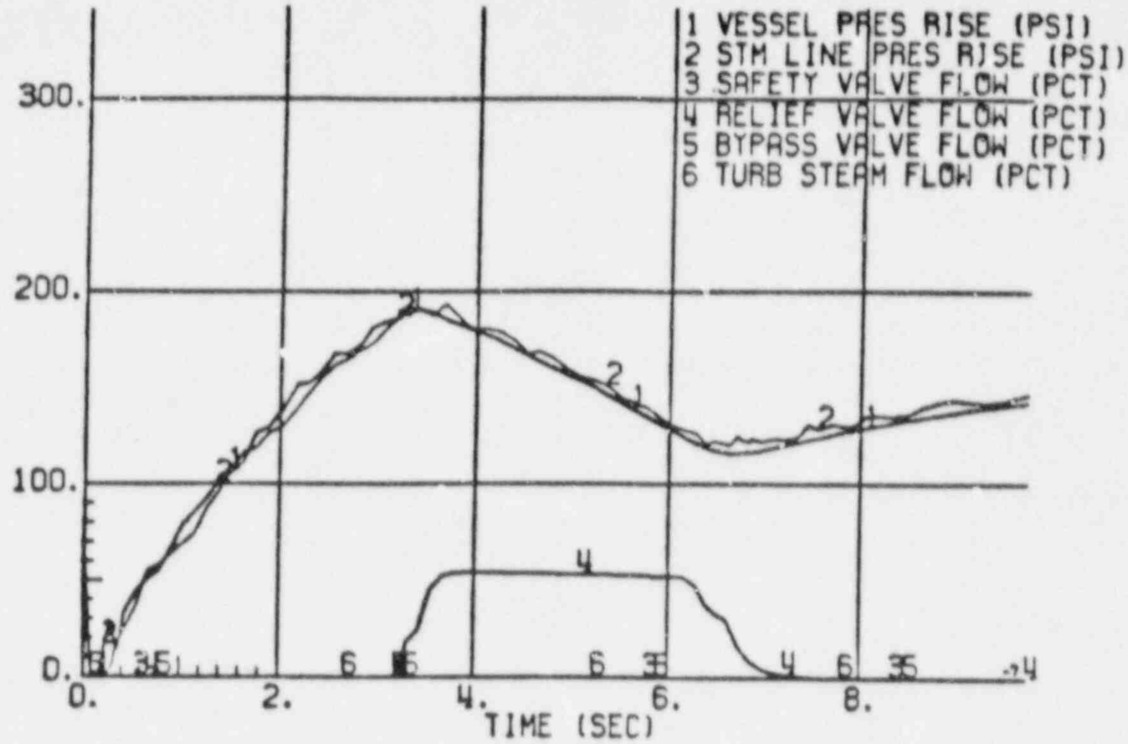


Figure 3-3. Load Rejection With Bypass Failure, 70.2% Power, 53.6% Flow



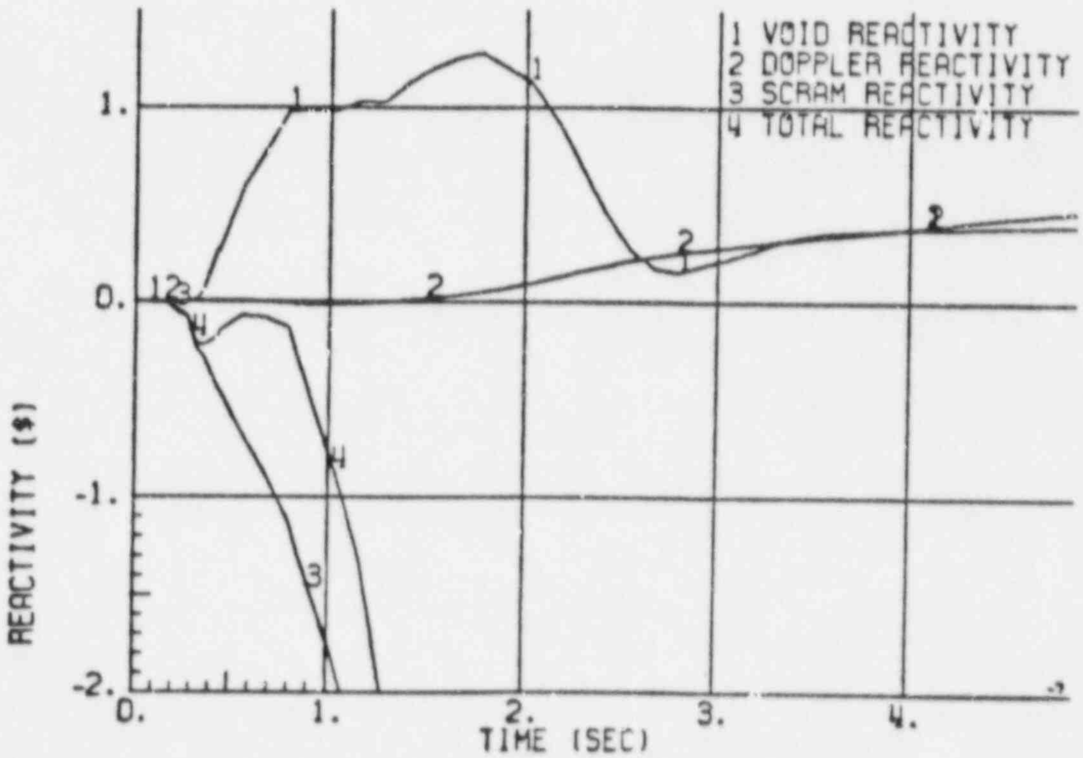


Figure 3-3. Load Rejection With Bypass Failure, 70.2% Power, 53.6% Flow (Continued)

4. STABILITY ANALYSIS

4.1 PHENOMENA

The least stable power/flow condition attainable under normal operating conditions (with both reactor coolant system recirculation loops in operation) occurs at minimum flow and at 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 delay ratio) occurs 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 8-3). 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 in stability tests at operating BWRs (Reference 8-4).

In addition to the preceding 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 8-3). 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; however, 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 result in an increase in system noise (jet pump, core flow and neutron flux noise).

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 8-5). The generic stability analysis has been performed covering all licensed GE BWR fuel designs including those fuels contained in the General Electric Standard Application for Reactor Fuel (GESTAR, Reference 8-6). The analysis demonstrates that specified acceptable fuel design limits are not exceeded.

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 analyses (Reference 8-5) 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; instead, they are addressed on a plant-specific basis. A Service Information Letter-380, Revision 1 (Reference 8-7) has been developed to inform plant operators on how to recognize and suppress unanticipated oscillations encountered during plant operation.

As a result of the preceding analysis and operator recommendations, the NRC staff has approved the generic stability analysis for application to single-loop operation (Reference 8-8), provided that the recommendations of SIL-380 have been incorporated into the Plant Technical Specifications. The River Bend Technical Specifications reflect these recommendations.

5. LOSS-OF-COOLANT ACCIDENT ANALYSIS

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

5.1 BREAK SPECTRUM ANALYSIS

SAFE/REFLOOD calculations were performed using assumptions given in Section II.A.7.3.1 of Reference 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 5-1.

The total uncovered time for two-loop operation is 184 seconds for the 100% DBA suction break. For single-loop operation, the total uncovered time is 183 seconds for the 100% DBA suction break. The 100% DBA suction break is the most limiting break for both the two-loop and single-loop operation.

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 in Reference 8-9, the one- and two-loop SAFE/REFLOOD results can be considered similar. The generic alternate procedure described in Section II.A.7.4 of Reference 8-9 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 P8X8R fuel is 0.84. Single-loop operation MAPLHGR values are derived by multiplying the current two-loop MAPLHGR values by the reduction factor (0.84). As discussed in Reference 8-9, single-loop MAPLHGR values are

conservative when calculated in this manner. This MAPLHGR multiplier is directly applicable to all P8X8R fuel in the initial core. For reload situations, the MAPLHGR must be assessed for each cycle to determine if it is still applicable. This is because the single-loop MAPLHGR multiplier was based on the calculated peak cladding temperature (PCT) from the two-loop analysis for the initial core fuel.

5.3 SMALL BREAK PEAK CLADDING TEMPERATURE

Section II.A.7.4.4.2 of Reference 8-9 discusses the low sensitivity of the calculated PCT to the assumptions used in the single-loop operation analysis and the duration of nucleate boiling. Since this slight increase ($\sim 50^{\circ}F$) in PCT is overwhelmingly offset by the decreased MAPLHGR (equivalent to 300 to $500^{\circ}F$ PCT) for single-loop operation, the calculated PCT values for small breaks will be well below the $1547^{\circ}F$ small break PCT value previously reported for RBS, and significantly below the $2200^{\circ}F$ 10CFR50.46 cladding temperature limit.

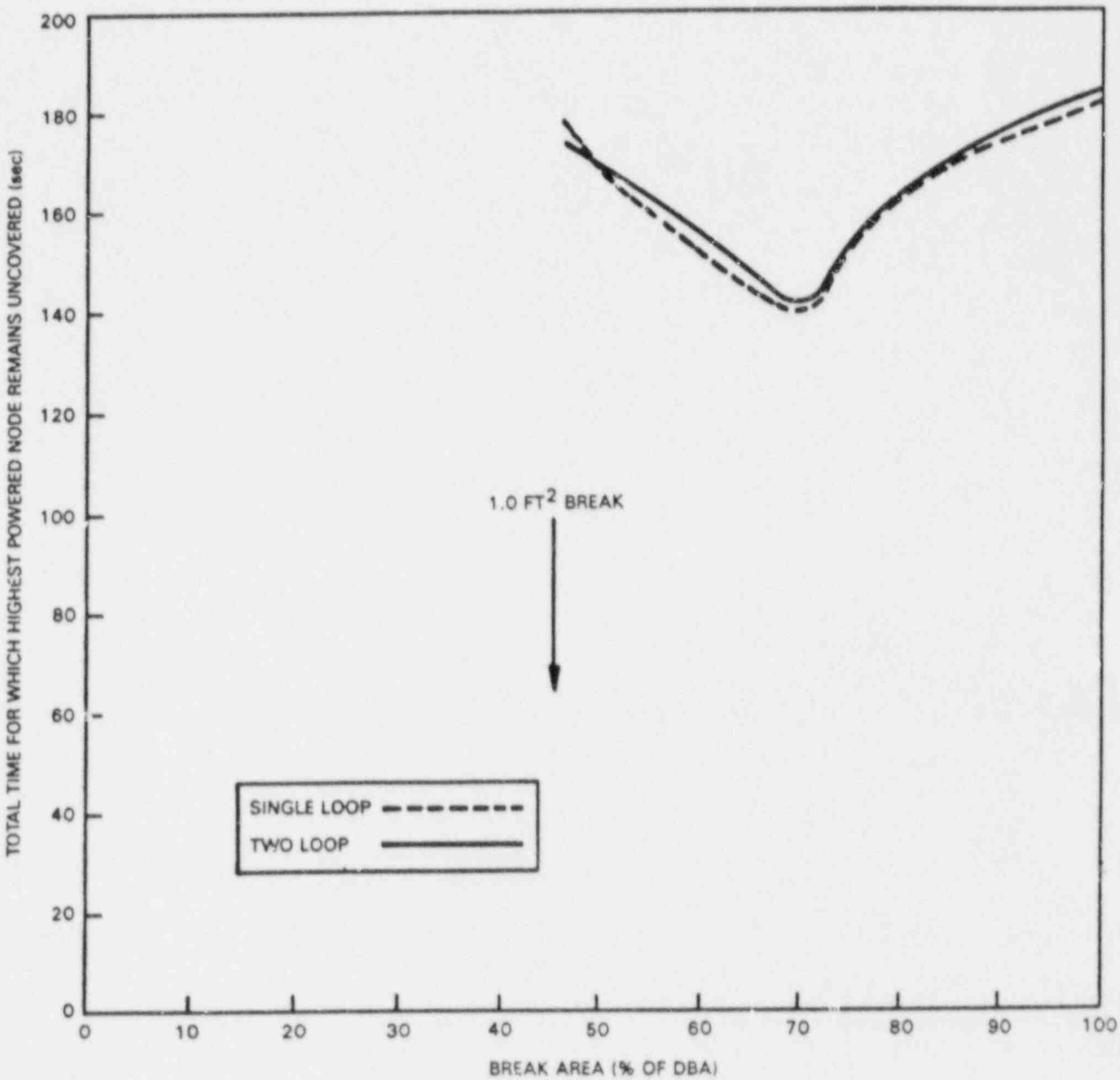


Figure 5-1. Uncovered Time vs. 3reak Area - Suction Break, LPCI Diesel Generator Failure

6. CONTAINMENT ANALYSIS

A single-loop operation containment analysis was performed for RBS. The peak wetwell pressure, peak drywell pressure, chugging loads, condensation oscillation and pool swell containment response were evaluated over the entire single-loop operation power/flow region.

The peak drywell-to-wetwell pressure during single-loop operation occurred under recirculation line break at the maximum vessel subcooling condition in the power/flow map. This peak differential pressure decreased by 0.15 psi compared to the value for the DBA main steam line break at the rated two-loop operation given in Section 6 of the FSAR. The chugging loads, condensation oscillation and pool swell loads evaluated at the worst power/flow condition during single-loop operation vary slightly over the peak values presented in Section 6. The analyses showed that there is enough design margin to handle these loads during SLO.

7. MISCELLANEOUS IMPACT EVALUATION

7.1 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) IMPACT EVALUATION

The principal difference between single-loop operation (SLO) and normal two-loop operation (TLO) affecting 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.

It is concluded that if an ATWS event were initiated at RBS from the SLO conditions, the results would be less severe than if it were initiated from rated conditions.

7.2 FUEL MECHANICAL PERFORMANCE

Evaluations were performed to determine the acceptability of RBS single-loop operation on P8X8R fuel rod and assembly thermal/mechanical performance. Component pressure differential and fuel rod overpower values were determined for anticipated operational occurrences initiated from SLO conditions. These values were found to be bounded by those applied in the fuel assembly design bases. In addition, fuel rod performance was determined to be within the design basis and limits specified in Reference 8-6.

7.3 VESSEL INTERNAL VIBRATION

A recirculation pump drive flow limit is imposed for SLO in order to meet acceptable vessel internal vibration criteria. An assessment has been made for the expected reactor vibration level during SLO for RBS for rated reactor water temperature and pressure. The maximum recirculation pump drive flow for RBS is about 33,000 gpm. This is a measured value from the Kuo Sheng-1 plant, which is the prototype plant for River Bend. Startup tests at the Kuo Sheng-1 plant showed all components, including the in-core guide tube during SLO, to have vibration levels within acceptance limits.

8. REFERENCES

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

8-2 "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", General Electric Company, October 1978 (NEDO-24154).

8-3 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.

8-4 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).

8-5 G. A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria", General Electric Company, October 1984 (NEDE-22277-P-1, Proprietary Information).

8-6 "General Electric Standard Application for Reactor Fuel", General Electric Company, May 1986 (NEDE-24011-P-A-8).

8-7 "BWR Core Thermal Hydraulic Stability", General Electric Company, February 10, 1984 (Service Information Letter-380, Revision 1).

8-8 Letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, Thermal Hydraulic Stability Amendment to GESTAR II," April 24, 1985.

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", General Electric Company, July 1978 (NEDO-20566-2, Revision 1).