PNPP SINGLE LOOP OPERATION ANALYSIS

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APPENDIX 15.F

TABLE OF CONTENTS

DTS:rf:dlv/G07227*-2
9/23/85

TABLE OF CONTENTS (Continued)

DTS:rf:dlv/G07227*-3
9/23/85

LIST OF TABLES

DTS:rf:dlv/G07227*-4
9/23/85

LIST OF FIGURES

15.F RECIRCULATION SYSTEMS SINGLE-LOOP OPERA.'ION

This appendix justifies that PNPP can saf(ly operate with single recirculation loop up to 70% of rated thermal power. This appendix presents the results of this safety evaluation for the operation of the Perry Nuclear Power Plants (PNPP) with single recirculation loop operating. This evaluation is performed for PNPP on a equilibrium cycle basis with the current GE6 (8x8R) fuel design and is applicable to initial and reload cycles operation with normal feedwater heating and within the operating domain shown in Figure 4-4-2 of the FSAR.

15.F.1 INTRODUCTION AND SUMMARY

Single-loop operation (SLO) is 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 operation, as presented in Chapters 6 and 15, were reviewed with one recirculation pump in operation. The safety-limit MCPR is determined using the General Electri Thermal Analysis Basis (GETAB), a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. During SLO, the total core flow and TIP readings uncertainty input to the GETAB model is slightly increased resulting in a small increase in the safety limit MCPR.

The results of the evaluation are summarized below:

(a) Accounting for the uncertainties in the core total flow and Traversing In-Core Probe (TIP) readings, a 0.01 incremental increase is required in the Minimum Critical Power Ratio (MCPR) flow cladding integrity safety limit during single-loop operation.

(b) The recirculation flow rate dependent rod block and scram setpoint equation given in the Technical Specifications for two loop operation must be adjusted for one-pump operation.

(c) 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.

(d) The Maximum Average Planar Linear Heat Generation Rate curve for two-loop operation in the Technical Specification must be reduced (multiplication factor 0.84) for single-loop operation.

(e) 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.

(f) The consequences of single loop operation on the Anticipated Transient Without Scram (ATWS) analysis are bounded by those from two-loop operation.

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

15.F.2 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis (GETAB) which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. 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.F.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.F.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. An analysis was performed to show the net effect of these two revised uncertainties is a 0.01 incremental increase in the required MCPR fuel cladding integrity safety limit.

15.F.2.1 Core Flow Uncertainty

15.F.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 (when the active loop pump flow is 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 single-loop operation, the total core flow is derived by the following formula:

$$\text{Total Core Flow} = \text{Active Loop Indicated Flow} - C * \text{Inactive Loop Flow}$$

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.*

15.F.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.F.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.F.2-1):

$$W_C = W_A - W_I$$

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:

*The analytical expected value of the "C" coefficient for PNPP is ~0.83.

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

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 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:

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

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

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:

*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 PNPP is ~0.28.

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

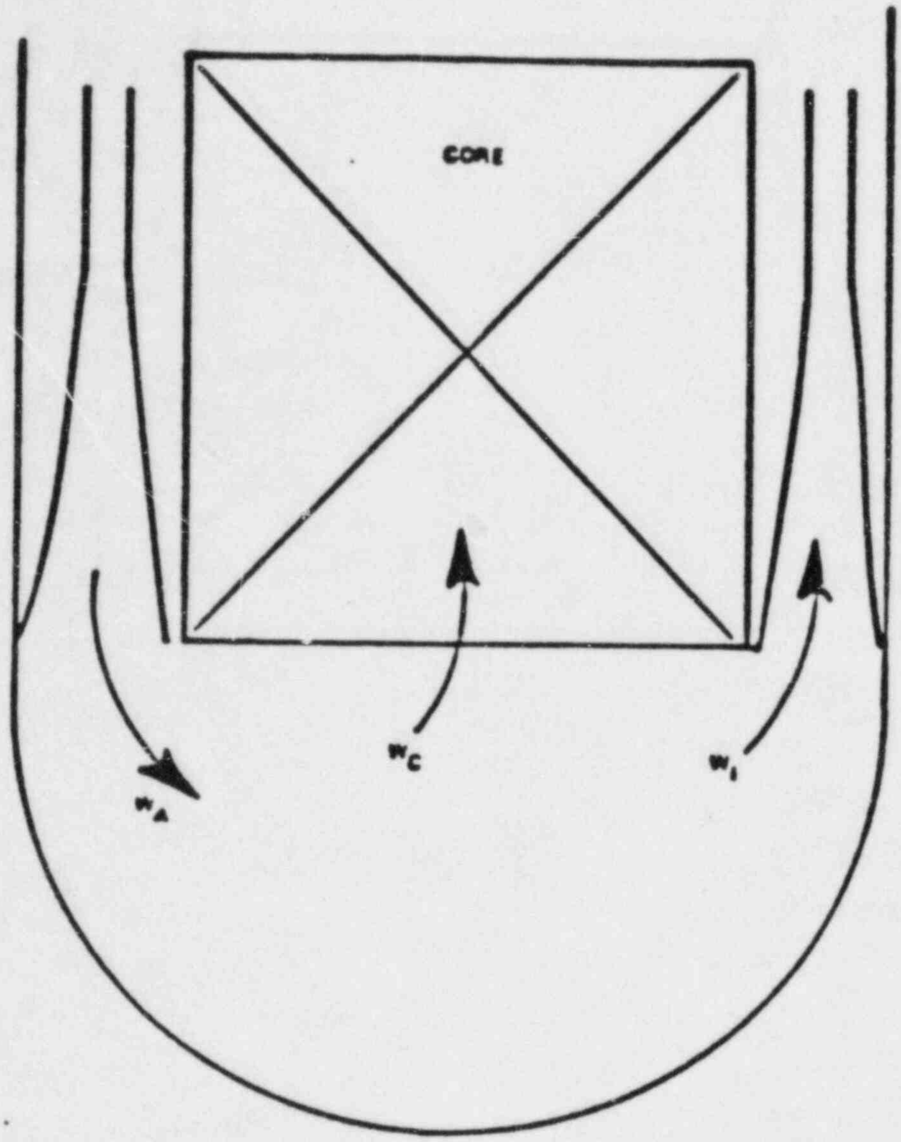
which is less than the 6% flow uncertainty assumed in the statistical analysis.

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

15.F.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 of 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.

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.



W_C = Total Core Flow
W_A = Active Loop Flow
W_I = Inactive Loop Flow

CLEVELAND ELECTRIC	ILLUSTRATION OF SINGLE RECIRCULATION LOOP OPERATION FLOWS	FIGURE 15.F.2-1

15.F.3 MCPR OPERATING LIMIT

15.F.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 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.F.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 (about 30% less during one loop operation).

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 one recirculation pump seizure accident has been reviewed for single loop operation. Results show that this accident poses no threats to thermal limits.

The worst flow increase transient results from recirculation flow controller failure, and the worst cold water injection transient results from the sudden loss of feedwater heating. When operating with only one recirculation loop, the flow and power increase associated with the flow controller failure with only one loop will be less than that associated with both loops. The latter event, sudden 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 heating 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 bounding 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 the full power two-loop analyses presented in the FSAR. The maximum power level that can be attained with one-loop operation is restricted by one pump operation flow capability, MCPR, and the overpressure limits established from a full-power analysis.

In the following sections, two of the most limiting transients are discussed for single-loop operation. They are:

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

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

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

15.F.3.1.1 Feedwater Controller Failure - Maximum Demand

15.F.3.1.1.1 Core and System Performance

Mathematical Model

The computer model described in Reference 15.F.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.F.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. The initial conditions used in the analysis are consistent with those in Table 15.0-1 except those which are related to SLO.

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 reference pressure of 1065 psig.

Results

The simulated feedwater controller transient is shown in Figure 15.F.3-2 for the case of 69.9% power 53.2% core flow. The high-water level turbine trip and feedwater pump trip are initiated at approximately 4.4 seconds. Scram occurs simultaneously from Level 8, and limits the peak neutron flux. The turbine bypass system opens to limit peak pressure in the steamline near the safety valves to 1046 psig and the pressure at the bottom of the vessel to about 1059 psig. MCPR is considerably above the safety limit.

15.F.3.1.2 Generator Load Rejection With Bypass Failure

15.F.3.1.2.1 Core and System Performance

Mathematical Model

The computer model described in Reference 15.F.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.F.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 recirculation 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.F.3-3.

Table 15.F.3-2 shows for the case of bypass failure, peak neutron flux reaches about 70.1% of rated and peak pressure at the valves reaches 1169 psig. The peak nuclear system pressure reaches 1180 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig. The calculated MCPR is well above the safety limit.

15.F.3.1.3 Summary and Conclusions

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

For pressurization events, lower initial pressure at lower power during single loop operation assures that the bounding MSIV closure flux scram event is bounded by the full power analysis. As expected, Table 15.F.3-2 also indicates the peak pressures from the two limiting transients are well below the ASME code value of 1375 psig. Hence, it is concluded the pressure barrier integrity is maintained under single-loop operation conditions.

15.F.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 Limiter (RWL) system. This analysis covered all off-rated conditions in the power/flow operating map with additional MCPR safety limit margin. Therefore, single-loop operation at reduced power level and with an increase of 0.01 MCPR safety limit is bounded by the technical specification for two loop operation.

The Average Power Range Monitor (APRM) rod block system provides additional alarms and rod blocks when power levels are grossly exceeded. Since the APRM rod block setpoints are recirculation drive flow biased, modification of the APRM rod block equation in the Technical Specification is required to maintain the two loop rod block versus power relationship when in one loop operation. This is because the direct active-loop flow measurement may not indicate actual core flow due to backflow through the inactive jet pump during single loop operation. Since the APRM scram trip settings are also recirculation drive flow biased, they are subject to the same modifications as the rod block settings.

TABLE 15.F.3-1

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

1. Thermal Power Level Analysis Value, MWt	2500 (69.9% Rated)
2. Steam Flow, lb/sec	2834
3. Core Flow, lb/hr	55.3×10^6 (53.2% Rated)
4. Feedwater Flow Rate, lb/sec	2834
5. Feedwater Temperature, °F	392
6. Vessel Dome Pressure, psig	970
7. Vessel Core Pressure, psig	974
8. Turbine Bypass Capacity, % NBR	35
9. Core Coolant Inlet Enthalpy, Btu/lb	507.4
10. Turbine Inlet Pressure, psig	934
11. Fuel Lattice	P8x8R
12. Core Leakage Flow, %	12.9
13. Required MCPR Operating Limit First Core	1.42[(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, %	46.1[(b)]

*These values are consistent with Table 15.0-1 except those which are related to 69.9% power and 53.2% flow.

TABLE 15.F.3-1 (Continued)

18.	Jet Pump Ratio, M	3.30
19.	Safety/Relief Valve Capacity, % NBR	
	1210 psig	111.4
	Manufacturer	DIKKER
	Quantity Installed	19
20.	Relief Function Delay, Seconds	0.4
21.	Relief Function Response, Seconds	0.1
22.	Analyses Inputs 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 (1.21 x 1.043), % NBR	127.2
25.	High Pressure Scram Setpoint, psig	1095
26.	Vessel Level Trips, Feet Above Separator Skirt Bottom	
	Level 8 - (L8), Feet	5.89
	Level 4 - (L4), Feet	4.04
	Level 3 - (L3), Feet	2.165
	Level 2 - (L2), Feet	(-)1.739
27.	APRM Thermal trip	
	Setpoint, % NBR @ 100% Core FLow	118.8
28.	RPT Delay, Seconds	0.14
29.	RPT Inertia Time Constant for Analysis, secs.	5
30.	Total steamline volume, ft^3	3850

(a) The operating limit corresponds to 1.18 at rated condition.

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

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

SINGLE-LOOP OPERATION

PARA-GRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLUX (% NBR)	MAXIMUM DOME PRESSURE (psig)	MAXIMUM VESSEL PRESSURE (psig)	MAXIMUM STEAMLINE PRESSURE (psig)	FREQUENCY* Category
		Initial Condition	69.9	970	982	982	N/A
15.F.3.1.1	15.F.3.2	Feedwater flow Controller Failure (Maximum Demand)	83.7	1047	1059	1046	a
15.F.3.1.2	15.F.3.3	Generator Load Rejection With Bypass Failure	70.1	1167	1180	1169	b

*a = Moderate frequency incident; b = infrequent;

TABLE 15.F.3-3

SUMMARY OF CRITICAL POWER RATIO RESULTS - SINGLE-LOOP OPERATION

	FWCF	LRNBP
Initial Operating Condition (% power/% flow)	69.9/53.2	69.9/53.2
Required Two Loop Initial MCPR Operating Limit at SLO Condition	1.42[a]	1.42[a]
ΔCPR	0.10	0.00[b]
Transient MCPR at SLO	1.32	1.42
SLMCPR at SLO*	1.07	1.07
Margin Above SLMCPR	0.25	0.35
Frequency Category	Moderate frequent incident	Infrequent incident

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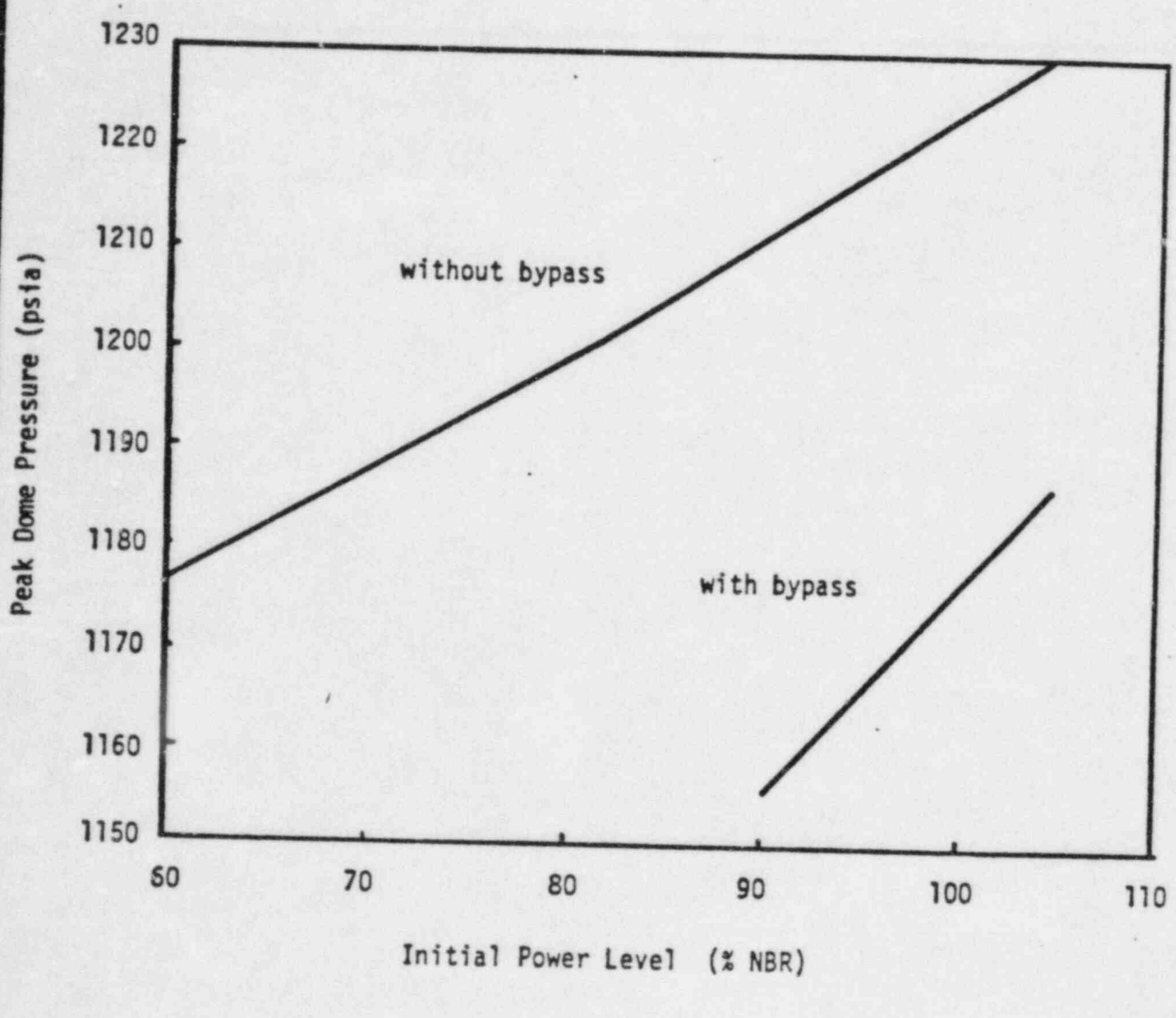
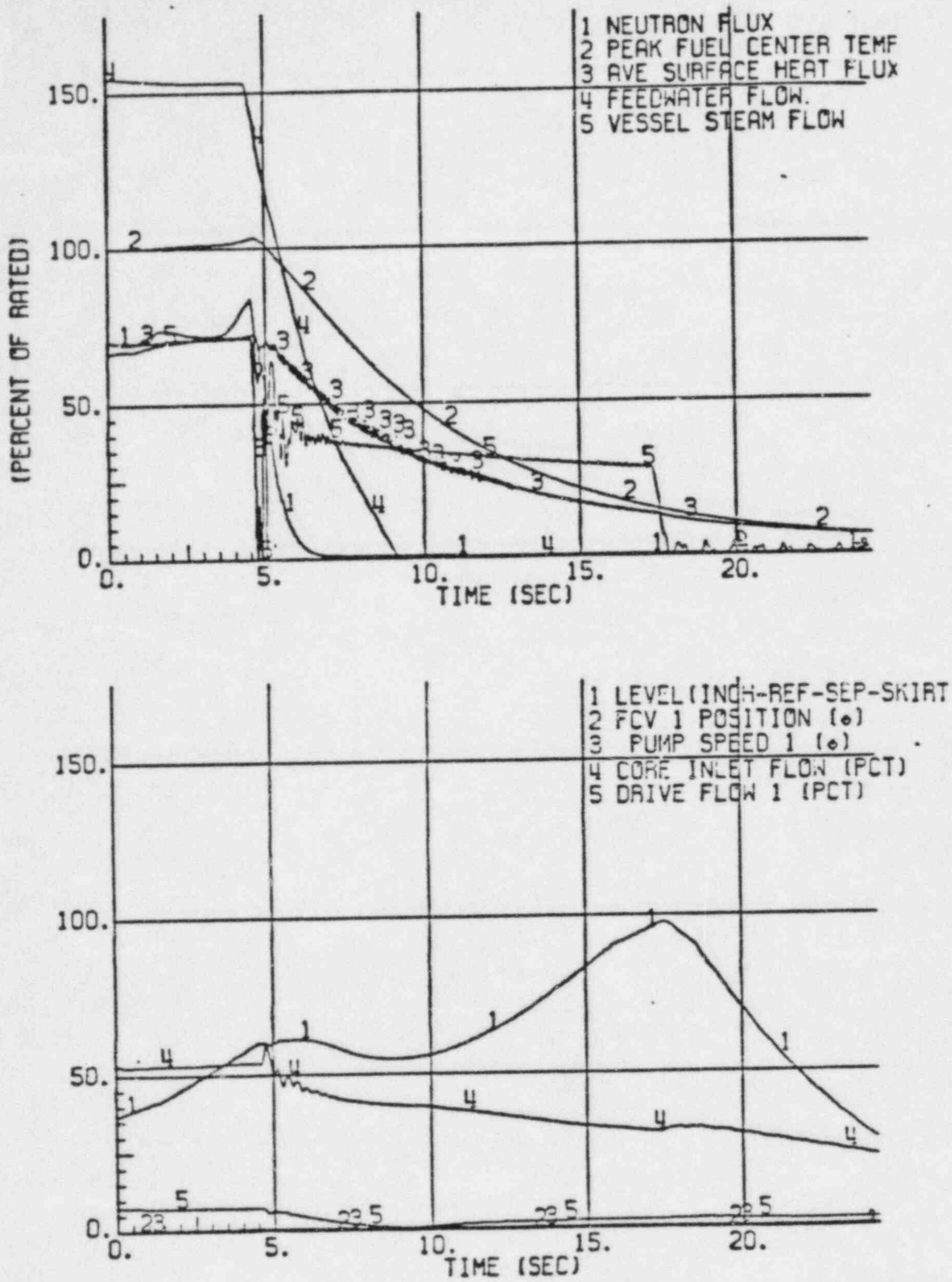
[a] This corresponds to MCPR of 1.18 at rated conditions. [b] ΔCPR is less than 0.002.

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

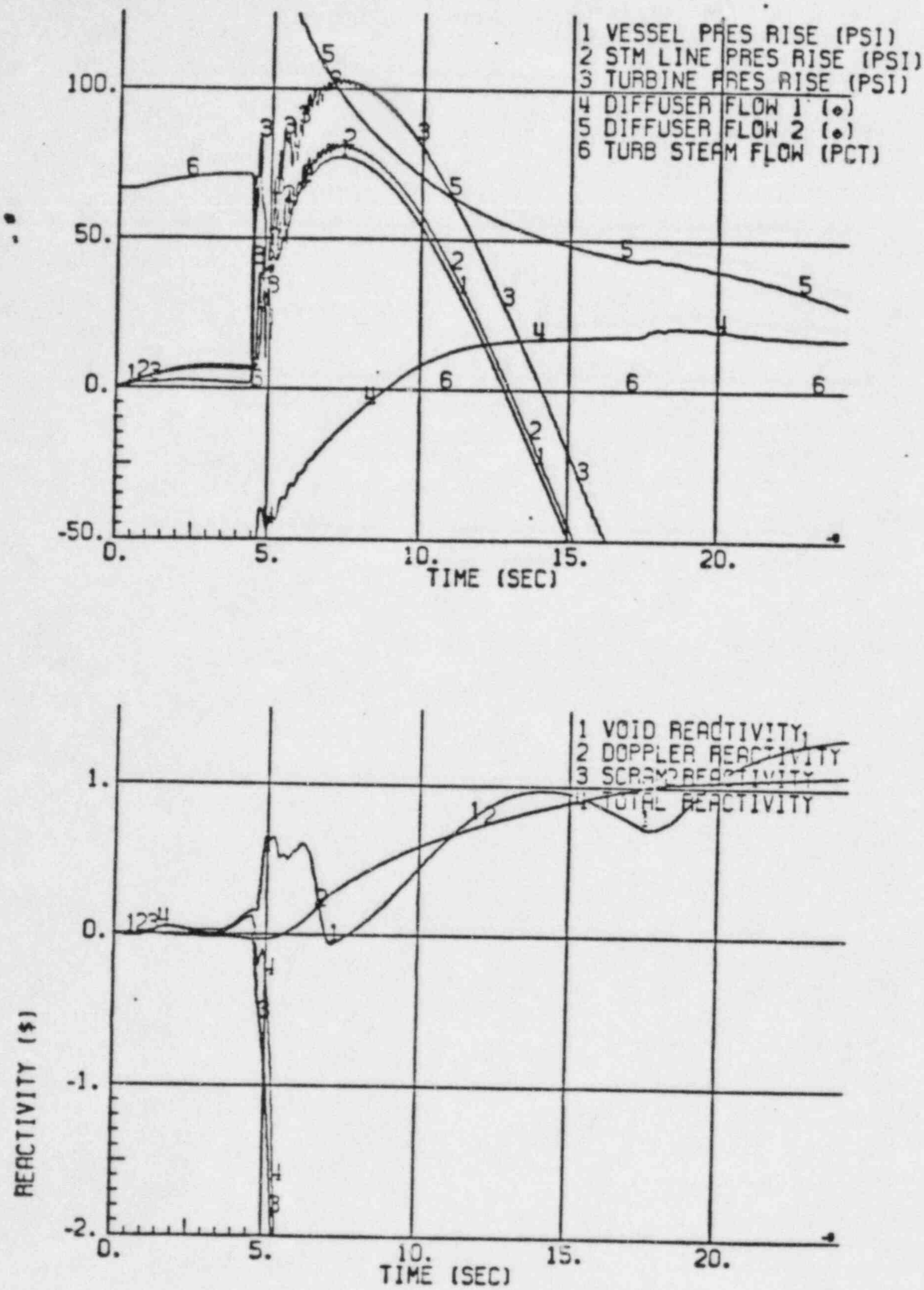


CLEVELAND ELECTRIC	Peak Dome Pressure versus Initial Power Level, Turbine Trip at EOEC	Figure 15.F.3-1

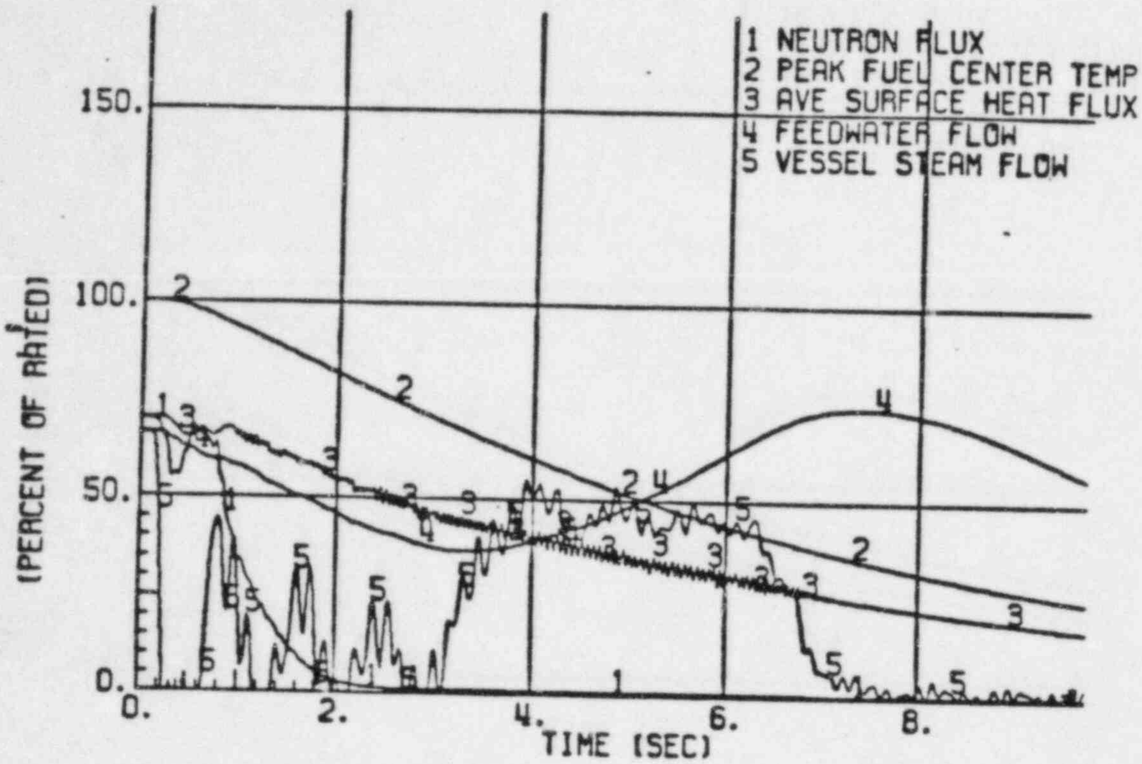


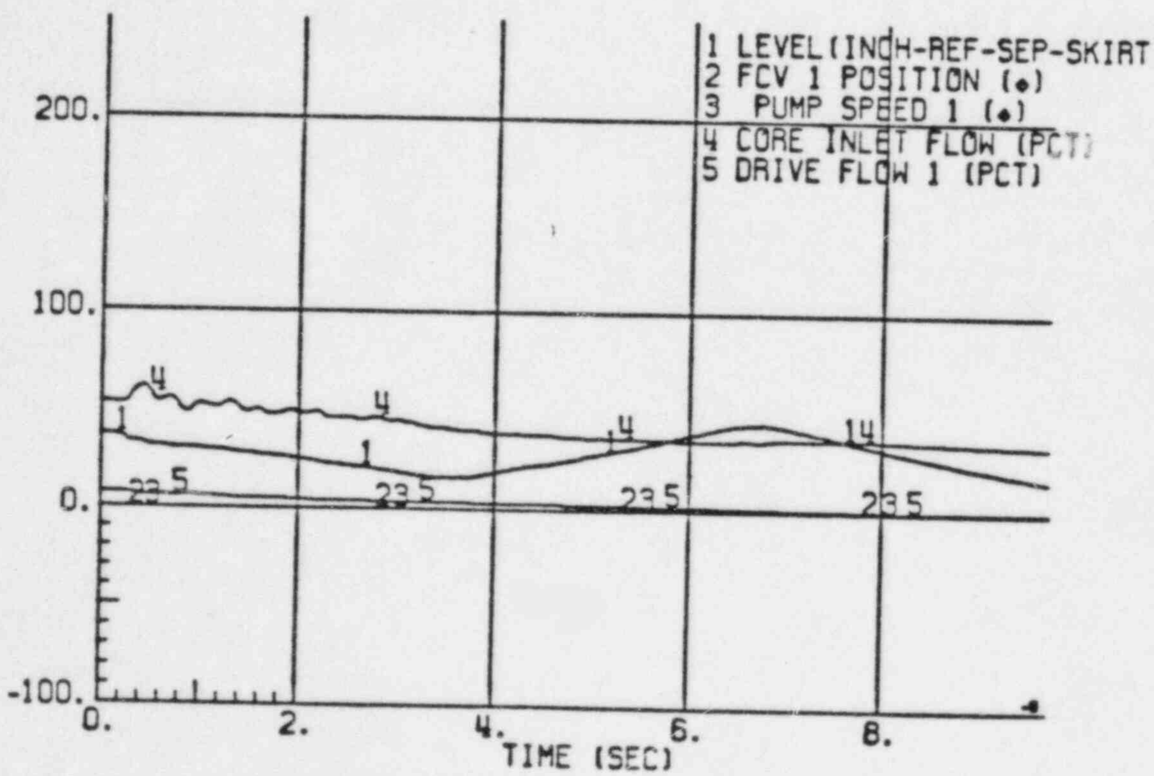
FWCF.MAX DEMA W/ HWL TRIPS

CLEVELAND ELECTRIC	FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND SINGLE-LOOP OPERATION	FIGURE 15.F.3-2

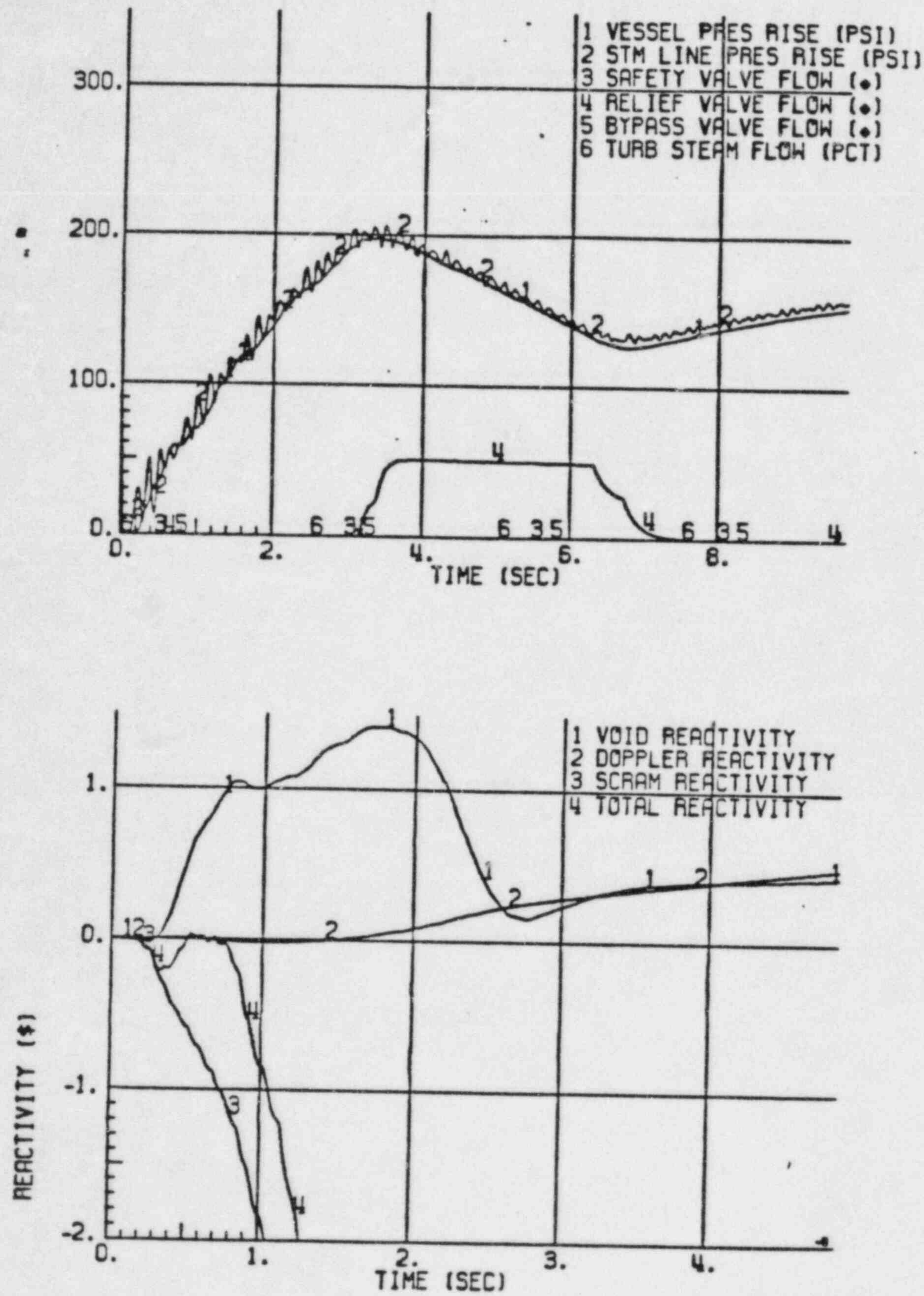


CLEVELAND ELECTRIC	FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND SINGLE-LOOP OPERATION	FIGURE 15.F.3-2 Cont'd.





CLEVELAND ELECTRIC	GENERATOR LOAD REJECTION, W/O BYPASS SINGLE-LOOP OPERATION	FIGURE 15.F.3-3



CLEVELAND ELECTRIC	GENERATOR LOAD REJECTION, W/O BYPASS SINGLE-LOOP OPERATION	FIGURE 15.F.3-3 Cont'd

15.F.4 FUEL INTEGRITY - STABILITY

Summary

The General Electric Company has established criteria to demonstrate compliance to requirements set forth in 10CFR50 Appendix A, General Design Criteria (GDC) 10 & 12. These stability compliance criteria consider potential limit cycle response within the limits of safety system or operator intervention and assure that for GE BWR fuel designs this operating mode does not result in specified acceptable fuel design limits being exceeded. Furthermore, the onset of power oscillations for which corrective actions are necessary is reliably and readily detected and suppressed by operator actions and/or automatic system functions. The stability compliance of those GE BWR fuel designs contained in the General Electric Standard Application for Reactor Fuel (GESTAR, Reference 15.F.8-5) is demonstrated on a generic basis in Reference 15.F.8-4. The NRC has reviewed and approved this in Reference 15.F.8-6. Therefore, a specific analysis for each cycle is not required, and Single Loop Operation is included as an acceptable condition.

Discussion

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 smaller 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 smaller stability margin as SLO core flow was increased, an evaluation was performed which phenomenologically accounts for single loop operation effects on stability (Reference 15.F.8-7). Additionally, 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 test data (Reference 15.F.8-7). 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). These are consistent with observed behavior in stability tests at operating BWRs (Reference 15.F.8-8).

The analyses and the evaluations in Reference 15.F.8-4 are performed on a generic basis and are directly applicable to single loop operation. The same reference has already demonstrated that, assuming the core is in an oscillatory mode without stability margin (limit cycle), specified acceptable fuel design limits are still not exceeded. The increase in MCPR safety limit of 0.01 is

well within the margin of the limit cycle analyses in Reference 15.F.8-4. In addition, a set of GE operating recommendations (Reference 15.F.8-9) which have been approved by the NRC, is to be utilized at PNPP during single loop operation.

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

If two recirculation loops are operating and a pipe break occurs in one of the two recirculation loops, the pump in the unbroken loop is assumed to immediately trip and begin to coast down. The decaying core flow due to the pump coastdown results in very effective heat transfer (nucleate boiling) during the initial phase of the blowdown. Typically, nucleate boiling will be sustained during the first 5 to 9 seconds after the accident, for the design basis accident (DBA).

If only one recirculation loop is operating, and the break occurs in the operating loop, continued core flow is provided only by natural circulation because the vessel is blowing down to the reactor containment through both sections of the broken loop. The core flow decreases more rapidly than in the two-loop operating case, and the departure from nucleate boiling for the high power node might occur 1 or 2 seconds after the postulated accident, resulting in more severe cladding heatup for the one-loop operating case.

In addition to changing the blowdown heat transfer characteristics, losing recirculation pump coastdown flow can also affect the system inventory and reflooding phenomena. Of particular interest are the changes in the high-power node uncovery and reflooding times, the system pressure and the time of rated core spray for different break sizes. One-loop operation results in small changes in high-power node uncovery times and times of rated spray. The effect on the reflooding times for various break sizes is also generally small.

An analysis of single recirculation loop operation using the models and assumptions documented in Reference 15.F.8-9 was performed for PNPP. Using this method, SAFE/REFLOOD computer code runs were made for a full spectrum of break sizes for only the recirculation suction line breaks (most limiting for PNPP). 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.

15.F.5.1 Break Spectrum Analysis

SAFE/REFLOOD calculations were performed using assumptions given in Section II.A.7.3.1 of Reference 15.F.8-10. The analytical model's hot node uncovered time (time between uncovery and reflooding) for single-loop operation is compared to that for two-loop operation in Figure 15.F.5-1.

The maximum uncovered time for two-loop operation is 176 seconds and occurs at 100% DBA suction break. This is the most limiting break for two-loop operation. For single-loop operation, the maximum uncovered time is 174 seconds and occurs also at 100% DBA suction break. This is the most limiting break for single-loop operation.

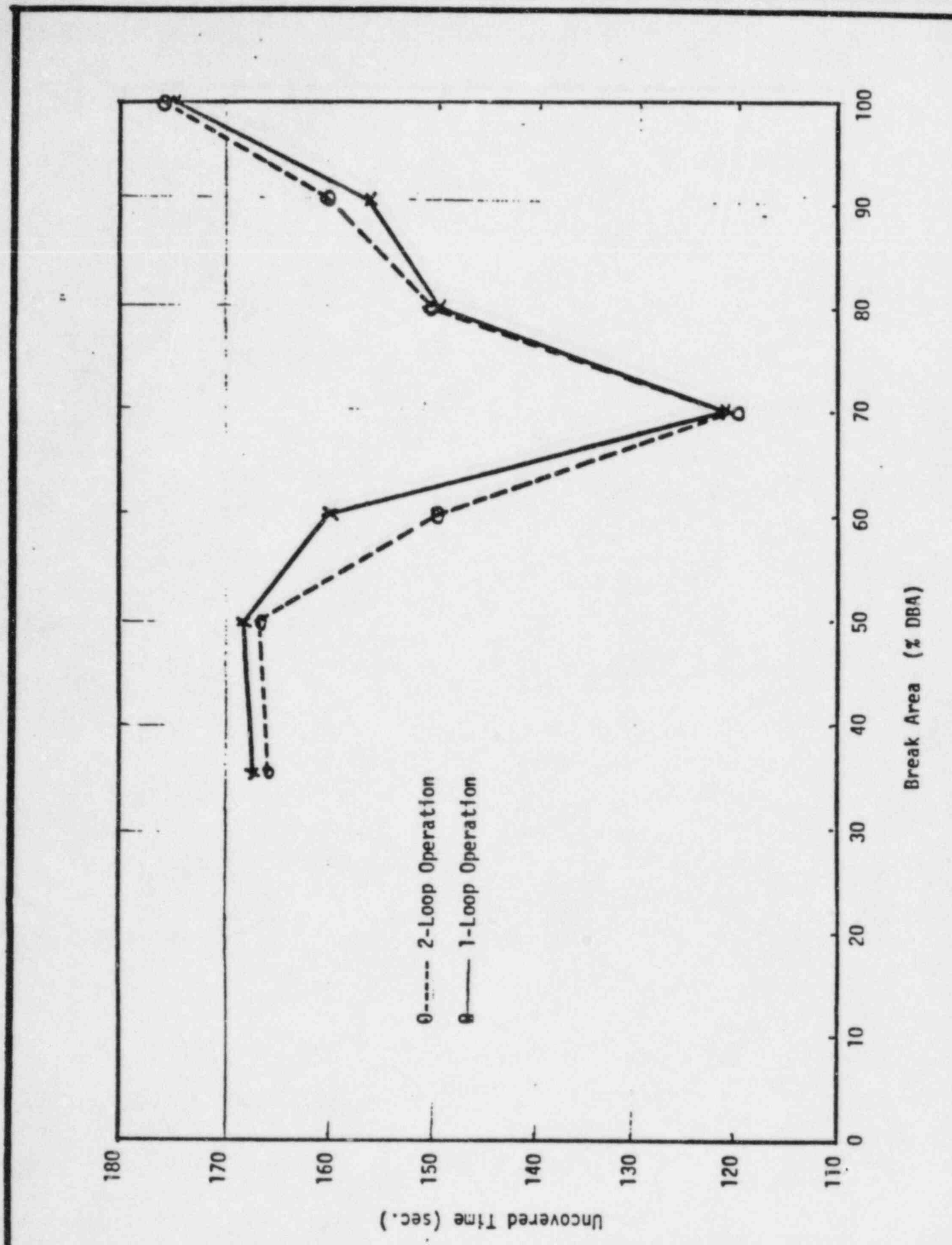
15.F.5.2 Single-Loop MAPLHGR

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 increase during core uncovery. Therefore, as noted in Reference 15.F.8-10, 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 0.84. One-loop operation MAPLHGR values are derived by multiplying the current two-loop MAPLHGR values by the reduction factor 0.84. As discussed in Reference 15.F.8-10, single recirculation loop MAPLHGR values are conservative when calculated in this manner.

15.F.5.3 Small Break Peak Cladding Temperature

Section II.A.7.4.4.2 of Reference 15.F.8-10 discusses the low sensitivity of the calculated peak cladding temperature (PCT) for the small break to the assumptions used in the one-pump operation analysis and the duration of nucleate boiling. This slight increase (~ 50°F) in small break PCT is over-

whelmingly offset by the single-loop 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 1345°F small break PCT value previously reported for PNPP, and significantly below the 2200°F 10CFR50.46 cladding temperature limit.



CLEVELAND ELECTRIC	Uncovered Time vs Break Area Suction Break, LPCS Failure	Figure 15.F.5-1

15.F.6 CONTAINMENT RESPONSE ANALYSIS

A single-loop operation containment analysis was performed for PNPP. 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 highest peak wetwell pressure during single-loop operation occurred at the maximum power/flow condition of 70% power/53% core flow. This peak wetwell pressure decreased by about fifteen percent compared to the rated two-loop operation pressure given in Section 6.2. The peak drywell pressure evaluated at the worst power/flow condition during single-loop operation was found to decrease by about 6 percent compared to the values given in Section 6.2. 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.F.7 MISCELLANEOUS IMPACT EVALUATION

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

The principal difference between single loop operation (SLO) and normal two loop operation 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 the two-loop ATWS analysis, the transient response is less severe and therefore bounded by the two-loop analyses. All ATWS acceptance criteria are met during SLO. Therefore, SLO is an acceptable mode of operation for ATWS considerations.

15.F.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 has been observed that due to the 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 show 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.F.7.3 Vessel Internals Vibration

A recirculation pump drive flow limit will be imposed for SLO. The highest drive flow tested during the normal startup test program at PNPP that shows 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 PNPP.

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 PNPP is about 42,000 gpm.

From the experiences from GE's BWR-6 jet pump development tests and Kuo Sheng 1 startup tests, the reactor internal components for the Perry plant would be expected to be within acceptance limits during single-loop operation with maximum flow as defined above.

15.F.8 REFERENCES

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

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

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

15.F.8-4 "Compliance of the General Electric Boiling Water, Reactor Fuel Designs to Stability Licensing Criteria", NEDE-22277-P-1, October 1984.

15.F.8-5 "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A, January 1982.

15.F.8-6 Acceptance for Referencing of Licensing Topical Report NEDE-24011 Revision 6, Amendment 8, "Thermal Hydraulic Stability Amendment to GESTAR II" NRC's C. O. Thomas to H. C. Pfefferlen, April 24, 1985.

(NOTE: Reference 15.F.8-4 above is the principal supporting document for GE's Topical Report NEDE-24011 Amendment 8)

15.F.8-7 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.F.8-8 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.F.8.9 "BWR Core Thermal Hydraulic Stability" SIL No. 380 Revision 1, February 10, 1984.

15.F.8-10 "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.